May 16, 2025 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Approved "-A" Version of Topical Report "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-A, Revision 5

REFERENCES:
1. NRC email to NuScale, "Final Safety Evaluation for NuScale Non-Loss-of-Coolant Accident Analysis Methodology Prop and non-Prop," dated May 15, 2025
2. NuScale letter to NRC, "NuScale Power, LLC Submittal of Topical Report "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416, Revision 5, dated March 26, 2025 (ML25085A357)

By referenced email dated May 15, 2025 (Reference 1), the NRC issued a final safety evaluation report documenting the NRC Staff conclusion that the NuScale topical report "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416, Revision 5, is acceptable for referencing in licensing applications for the NuScale small modular reactor design. Reference 1 requested that NuScale publish the approved version of TR-0516-49416 as soon as possible.

Enclosure 1 contains the proprietary version of the report entitled, "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-A, Revision 5. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirement of 10 CFR Part 810. Enclosure 2 contains the nonproprietary version of the report.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 16, 2025

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-A, Revision 5, Proprietary Version
Enclosure 2: "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-A, Revision 5, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-182333

Enclosure 1:

"Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-A, Revision 5, Proprietary Version

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Enclosure 2:

"Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-A, Revision 5, Nonproprietary Version

Contents

Section	Description
A	Letter from NRC to NuScale, “Final Safety Evaluation for NuScale Non-Loss-of-Coolant Accident Analysis Methodology Prop and non-Prop,” dated May 15, 2025
B	NuScale Topical Report: “Non-Loss-of-Coolant Accident Analysis Methodology,” TR-0516-49416-A, Revision 5
C	Letters from NuScale to the NRC, Responses to Requests for Additional Information on the NuScale Topical Report, “Non-Loss-of-Coolant Accident Analysis Methodology,” TR-0516-49416, Revision 5
D	Letters from NuScale to the NRC, Docketed Audit Responses on the NuScale Topical Report, “Non-Loss-of-Coolant Accident Analysis Methodology,” TR-0516-49416, Revision 5

Section A

From: Thomas Hayden <Thomas.Hayden@nrc.gov>
Sent: Thursday, May 15, 2025 10:25 AM
To: Regulatory Affairs
Cc: Griffith, Thomas; Bode, Amanda; Lynn, Kevin; Mahmoud -MJ- Jardaneh; Getachew Tesfaye
Subject: Final Safety Evaluation for NuScale Non-Loss-of-Coolant Accident Analysis Methodology Prop and non-Prop
Attachments: Final Safety Evaluation of the NuScale Power LLC Topical Report TR-0516-49416 Revision 5 Non-Loss-of-Coolant Accident Analysis Methodology - 051525.docx; Final Safety Evaluation of the NuScale Power LLC Topical Report TR-0516-49416 Revision 5 Non-Loss-of-Coolant Accident Analysis Methodology (Proprietary) - 051525.docx

By letter dated March 26, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML25085A358 (Proprietary) and ML25085A357 (non-Proprietary)), NuScale Power, LLC (NuScale), submitted Topical Report (TR) TR-0516-49416, Revision 5, "Non-Loss-of-Coolant Accident Analysis Methodology" to the U.S. Nuclear Regulatory Commission (NRC). The NRC staff has prepared a final safety evaluation for TR-0516-49416, Revision 5. The non-proprietary (ML25098A249) and proprietary (ML25098A248) final safety evaluations are enclosed. The NRC staff has found that TR-0516-49416, Revision 5, is acceptable for referencing in licensing applications for the NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the enclosed final safety evaluation.

The NRC staff requests that NuScale publish the accepted version of this TR as soon as possible following receipt of this electronic mail. The accepted version shall incorporate this electronic mail and the enclosed final safety evaluation after the title page. It must be well indexed such that information is readily located. Also, it must contain historical review information, including NRC requests for additional information and accepted responses. The accepted version of the TR shall include a "-A" (designated accepted) following the report identification number.

If the NRC's criteria or regulations change such that the NRC staff's conclusion in this electronic mail (that the TR is acceptable) is invalidated, NuScale and/or the applicant referencing the TR will be expected either to revise and resubmit its respective documentation or to submit justification for continued applicability of the TR without revision of the respective documentation.

If you have any questions or comments concerning this matter, I can be reached at (301) 415-2956 or via e-mail address at Thomas.Hayden@nrc.gov. The attached proprietary document is password protected. Password to follow in a separate email.

Docket Nos. 99902078, 05200050

Sincerely,

Tommy Hayden
Project Manager

Email: thomas.hayden@nrc.gov
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

SAFETY EVALUATION BY THE U.S. NUCLEAR REGULATORY COMMISSION

TOPICAL REPORT TR-0516-49416, REVISION 5

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

NUSCALE POWER, LLC

Table of Contents

1 INTRODUCTION 1

1.1 Background 1

1.2 Scope of the Submittal 1

1.3 Scope of the Review 4

2 REGULATORY BASIS FOR NON-LOCA EM REVIEW 4

2.1 Regulatory Requirements 4

2.2 Regulatory Guidance 5
2.2.1 Documentation 7
2.2.2 Evaluation Model 7
2.2.3 Accident Scenario Identification Process 7
2.2.4 Code Assessment 7
2.2.5 Uncertainty Analysis 7
2.2.6 Quality Assurance Plan 8

3 TECHNICAL EVALUATION 8

3.1 Introduction 8
3.1.1 Purpose 8
3.1.2 Scope 9

3.2 Background 9
3.2.1 Non-LOCA Evaluation Model Roadmap 10
3.2.2 Regulatory Requirements and Regulatory Guidance 10

3.3 Plant Design Overview 10
3.3.1 Description of NuScale Plant 10
3.3.2 Plant Operation 11
3.3.3 Decay Heat Removal System 11
3.3.4 Emergency Core Cooling System 12
3.3.5 Other Important Systems and Functions 12

3.4 Transient and Accident Analysis Overview 12
3.4.1 Design-Basis Events and Event Classification 12
3.4.2 Design Basis Event Acceptance Criteria 13
3.4.3 Non-LOCA Transient Analysis Process 15

3.5 NRELAP5 Applicability for Non-LOCA Transient Analysis 21
3.5.1 Non-LOCA Phenomena Identification and Ranking Table 21
3.5.2 Evaluation of Non-LOCA Phenomena Identification and Ranking Table High-Ranked Phenomena 25
3.5.3 NRELAP5 Validation and Assessments for Non-LOCA 25
3.5.4 Conclusions of NRELAP5 Applicability for Non-LOCA 42

3.6 NuScale NRELAP5 Plant Model 43
3.6.1 Thermal-Hydraulic Volumes and Heat Structures 43
3.6.2 Material Properties 49
3.6.3 Control and Protection Systems 49

3.7 Non-LOCA Analysis Methodology 49
3.7.1 General Aspects of Non-LOCA Methodology 49

3.7.2 Event-Specific Methodology 55
3.8 Representative Calculations 83
3.9 Quality Assurance 84
4 LIMITATIONS AND CONDITIONS 85
5 CONCLUSION 86
6 REFERENCES 88

1 INTRODUCTION

1.1 Background

On January 5, 2023, NuScale Power, LLC (NuScale), hereinafter referred to as "the applicant," submitted Topical Report (TR) TR-0516-49416-P, Revision 4, "Non-Loss-of- Coolant-Accident Analysis Methodology," to the U.S. Nuclear Regulatory Commission (NRC) for review and approval in support of the NuScale US460 Standard Design Approval (SDA) (ADAMS Accession Number (ML23005A305). The NRC accepted the TR for review on July 31, 2023 (ML23206A107). The completeness determination was updated to include a resource estimate on September 22, 2023 (ML23265A154). The completeness determination was revised on June 27, 2024, (ML24178A422) to update the schedule and resource estimates, to account for issuing requests for additional information (RAIs). The applicant submitted Revision 5 of the TR on March 26, 2025 (ML25085A356), which addressed NRC staff RAIs and issues developed during the staff's regulatory audit (ML24262A257).

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

The EM uses a modified version of the RELAP5 computer code, referred to as NRELAP5, and follows a graded approach outlined in Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," dated December 2005 (ML053500170). The TR addresses the high-ranked phenomena identified by the non-LOCA phenomena identification and ranking table (PIRT). TR-0516-49422-P-A, "Loss-of-Coolant Accident Evaluation Model" (LOCA TR), Revision 5 (Reference 1) addresses the high-ranked phenomena that are related to the performance of the NRELAP5 computer code but are not addressed in the non-LOCA TR.

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

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

1.2 Scope of the Submittal

The TR includes information on the following topics:

- The EM roadmap and relevant regulatory requirements.
- Key NPM design characteristics.

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

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

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

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

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

The TR is focused on the short-term non-LOCA transient progression, defined as the time frame during which the mixture level remains above the top of the riser and primary side natural circulation is maintained. The applicant's long-term cooling analysis methodology, including events that transition from decay heat removal system (DHRS) cooling to emergency core cooling system (ECCS) heat removal, is addressed in TR-124587-P, Revision 1, "Extended Passive Cooling and Reactivity Control Methodology," (Reference 2).

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

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

There are holes in the riser of the NuScale reactor designs. The purpose of these holes is to mitigate potential boron dilution in the downcomer during extended periods of DHRS operation in which the riser may uncover and steam condenses on the steam generator (SG) tubes. The riser holes were not designed to influence the short term DHRS cooldown during the non-LOCA phase. The TR states that the PIRT and test assessment discussion in TR Section 5 does not incorporate the small holes in the riser. NRC staff notes that the NRELAP5 example calculations in TR Section 8 also do not incorporate the small holes in the riser. However, the applicant's NRELAP5 models for non-LOCA event analyses include the riser holes and the model description in Section 6 of the TR reflects the riser hole flow **{{**

}}. The impact of riser holes on the PIRT and test assessment are discussed in Section 3.5 of this safety evaluation (SE), and the NRC staff's review of the sensitivity studies and example calculations are in Sections 3.7 and 3.8 of this SE, respectively. Broadly, neither the event-specific sensitivity studies nor the example calculations are representative of the NPM-20 design, including the presence of riser holes. Although the riser holes were not discussed in these sections of the TR, NRELAP5 models implementing the TR methodology must include riser holes. The applicant further states that during the short-term time frame considered in the non-LOCA EM, the mixture level remains above the top of the riser and primary side natural circulation is maintained. During its audit review (ML24262A257), the staff confirmed that the applicant's evaluation demonstrated that the riser holes remain covered by water during the non-LOCA events and the holes have an insignificant effect on steady-state parameters as well as short-term non-LOCA transient progressions and FOMs. The staff's sensitivity analyses also support this conclusion.

1.3 Scope of the Review

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

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

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

2 REGULATORY BASIS FOR NON-LOCA EM REVIEW

2.1 Regulatory Requirements

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

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

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

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

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

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

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

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

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

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

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

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

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

2.2 Regulatory Guidance

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

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

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

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

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

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

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

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

- The release of radioactive material shall not result in offsite doses in excess of the guidelines of [10 CFR 52.137(a)(2)(iv)] and 10 CFR Part 100. The acceptance criterion for IEs is a small fraction (10 percent) of [10 CFR 52.137(a)(2)(iv)] and 10 CFR Part 100.

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

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

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

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

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

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

Each of these areas is discussed below.

2.2.1 Documentation

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

2.2.2 Evaluation Model

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

2.2.3 Accident Scenario Identification Process

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

2.2.4 Code Assessment

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

2.2.5 Uncertainty Analysis

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

accident initial conditions, setpoint values, and boundary conditions.

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

2.2.6 Quality Assurance Plan

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

3 TECHNICAL EVALUATION

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

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

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

3.1 Introduction

3.1.1 Purpose

TR Section 1.1, "Purpose," describes the purpose of the TR and states that the NuScale non-LOCA EM follows a graded approach to the EMDAP. Significant overlap exists between the EM documented in the non-LOCA TR, the EM documented in the XPC TR (Reference 2), and the EM documented in the LOCA TR (Reference 1). The non-LOCA TR references the EMs in these TRs for the overlapping areas to avoid duplication of information. The NRC staff notes that a graded approach to the EMDAP, as discussed in RG 1.203, may be acceptable, provided that the modifications that form the EM are based on a previously approved EM. Therefore, any

future changes to the LOCA EM need to be assessed by the applicant for their potential impact on the non-LOCA EM. Any subsequent changes to the non-LOCA methodology will require NRC approval. This is listed as Condition #2 in Section 4of this SE.

3.1.2 Scope

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

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

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

During its review, the staff notes that although the TR does not reference a specific NPM design, the methodologies presented in the TR for analyses of the non-LOCA events are based on the design features, trip signals, and setpoints of the NPM-20. In addition, example and supporting analyses provided by the applicant during the audit review (ML24262A257) are based on the NPM-20 design. The applicant stated that the non-LOCA EM remains applicable to the approved NPM-160 design as well, however, the staff is imposing Limitation and Condition (L&C) #1, which discusses the requirements an applicant or licensee wishing to apply the EM to either the NPM-160 design or a future design needs to fulfill.

L&C No. 3 discusses that the non-LOCA EM is no longer applicable when the coolant in the RCS shrinks sufficiently to drop below the top of the riser.TR Section 5.1.3, "Phenomena Identification and Ranking Table Figures-of-Merit and Phenomenon Ranking," clarifies that this phenomenon could occur "well after reactor trip and engineered safety features have responded to the initiating event." The XPC EM (Reference 2) provides methodologies for analyses of the system performance when RCS coolant level shrinks sufficiently to go below the top of the riser.

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

3.2 Background

TR Section 2, "Background," describes the basic principles identified in RG 1.203 that are important in the development and assessment of an EM. This section also specifies that the EM uses the NRELAP5 computer code (identified as Version 1.7 in TR Section 11.0, "References"), which is a descendant of the Idaho National Laboratory (INL) RELAP5-3D© computer code. NuScale initially submitted Version 1.6 of NRELAP5 (ML23011A012) for the US460 Standard Design Approval Application (SDAA) as the systems analysis computer code for LOCA and non-LOCA Evaluation Methodologies. With respect to relevance to non-LOCA transients, the significant difference between RELAP5-3D and NRELAP5 is the helical coil hydraulic component (HLCOIL), which is included in NRELAP5 and used to model an NPM's

unique SGs; the NPM SGs are parts of the DHRS loops and the DHRS is of central importance to non-LOCA transients. Subsequently during the US460 review, NuScale submitted NRELAP5 Version 1.7 (ML24228A242) as the systems analysis computer code for the NuScale LOCA Evaluation Methodology, replacing NRELAP5 Version 1.6.

3.2.1 Non-LOCA Evaluation Model Roadmap

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

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

3.2.2 Regulatory Requirements and Regulatory Guidance

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

3.3 Plant Design Overview

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

3.3.1 Description of NuScale Plant

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

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

- Reduced reactor core size
- Natural circulation reactor coolant flow (i.e., no reactor coolant pumps)

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

- A safety-related ECCS system that does not require electrical power and does not use ECCS pumps

- A two-train safety-related two-phase natural circulation DHRS

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

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

The NRC staff reviewed the general description of the plant design in TR-0516-49416-P, Revision 5 and concludes it provides a sufficient description of the design to support the description of the methodology.

3.3.2 Plant Operation

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

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

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

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

3.3.3 Decay Heat Removal System

TR Section 3.3, "Decay Heat Removal System," states that the DHRS is a closed-loop, two-phase natural circulation cooling system. The DHRS consists of two trains, one attached to

each SG loop. Each train removes decay heat from the RPV and rejects it to the reactor pool via condensers submerged in the reactor pool. The staff notes that DHRS capability is assessed through the design review in a licensing application. The representation of decay heat for non-LOCA analysis is discussed in Section 3.6.1, "Thermal-Hydraulic Volumes and Heat Structures," of this SE.

3.3.4 Emergency Core Cooling System

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

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

The ECCS is not actuated in the short time frame of any of the non-LOCA events documented in the TR, presuming there is not a loss of EDAS power. Once, or if, the ECCS valves open, those events or event phases are analyzed according to the methodology in TR-0516-49422, "Loss-of-Coolant Evaluation Model," Revision 5 (Reference 1). The consequences of the non-LOCA events described in this TR are mitigated by the actuation and the operation of the DHRS for the short-term transient period evaluated with this methodology. The ECCS actuation, which includes the ECCS Supplemental Boron (ESB) feature, following the DHRS actuation is addressed in the XPC TR (Reference 2).

3.3.5 Other Important Systems and Functions

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

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

3.4 Transient and Accident Analysis Overview

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

3.4.1 Design-Basis Events and Event Classification

TR Section 4.1, "Design-Basis Events and Event Classification," provides the event categories. The categories according to the frequency of occurrence are: AOOs, IEs, and PAs. The

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

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

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

TR Section 4.1, "Design-Basis Events and Event Classification," notes that the non-LOCA EM analyses are performed for a single module. Some initiating events, such as a loss of AC power, may affect multiple modules. Since the only shared safety system among modules is the UHS, the applicant assumes a pool temperature that bounds possible interactions between modules. (Non-Public: ML25014A157, Public: ML25014A156) Section 3.6.1, "Thermal-Hydraulic Volumes and Heat Structures," of this SE further discusses the pool temperature assumption.

3.4.2 Design Basis Event Acceptance Criteria

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

AOOs

- Maximum RCS primary system pressure ≤ 110 percent of design pressure.
- Maximum main steam secondary system pressure ≤ 110 percent of design pressure.
- Minimum critical heat flux ratio (MCHFR) > 95/95 critical heat flux ratio (CHFR) limit.[1]
- Maximum fuel centerline temperature ≤ melting temperature (adjusted for burnup effects).[1]
- An AOO should not result in a significant loss of reactor containment

[1] 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 evaluation, a pre-screening process using NRELAP5 helps to determine the cases evaluated in the subchannel analysis.

barrier.[2]

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

IEs and PAs

- Maximum RCS primary system pressure ≤ 120 percent of design pressure.
- Maximum main steam secondary system pressure ≤ 120 percent of design pressure.
- Fuel cladding integrity: If MCHFR ≤ 95/95 CHFR limit, or if maximum fuel centerline temperature > melting temperature, fuel rod is assumed to be failed[1].
- Containment integrity: Margins to containment pressure and temperature limits are maintained.[2]

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

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

SRP Section 15.0.2 states that a complete uncertainty analysis is not needed if suitably conservative input parameters are used. TR Section 4.2, "Design Basis Event Acceptance Criteria," states that the methodology includes performing sensitivity calculations to determine that suitably conservative inputs that result in the minimum margins to acceptance criteria are chosen. Section 7.2 of the TR specifies a minimum set of sensitivity studies that must be performed when the TR is implemented in order to assess margin to acceptance criteria. However, when these results show that margins to acceptance criteria are challenged, additional sensitivity studies on additional parameters are performed in order to ensure that acceptance criteria are met. In this way, the methodology may demonstrate margin without extensive sensitivity studies to minimize the margin to those unchallenged acceptance criteria. The applicant further clarified (ML18270A469) during the review of the previous revision of this topical report (i.e., revision 3 of TR-0516-49416-P-A) 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 in TR Section 7, "Non-LOCA Analysis Methodology," which demonstrate the consistent behavior.

[2] Margin to this acceptance criterion is demonstrated by the peak containment pressure/temperature analysis results performed according to a separate analysis methodology. Maximum containment pressure and temperature analysis is outside the scope of this topical report.

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 SE, the NRC staff finds that the applicant’s approach provides for use of suitably conservative input parameters and is consistent with SRP Section 15.0.2**.**

3.4.3 Non-LOCA Transient Analysis Process

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

3.4.3.1 Develop Plant Base Model NRELAP5 Input

The NRELAP5 computer code, which is based on modifications to the RELAP5-3D (Version 4.1.3) computer code developed by Idaho National Laboratory, is the system thermal-hydraulics code that the applicant uses for its non-LOCA system transient analyses. The NRELAP5 code is being maintained within NuScale’s QAP. The NRELAP5 computer code, Version 1.6 (ML23011A012), was submitted for the NuScale LOCA Evaluation Methodology (LOCA EM), and other topical reports, including the non-LOCA EM, as the systems analysis computer code. Subsequently, NuScale submitted NRELAP5 Version 1.7 (ML24228A242) as the systems analysis computer code for the NuScale LOCA EM, replacing NRELAP5 Version 1.6. Of note in Version 1.7, NuScale modified some algorithms used to blend between process models, particularly the choking models, to return smoother physics transitions; also, a parametric card option was added to allow users to specify the magnitude of a numerical stability modifier applied at cell junctions. Generally, topics covered by the non-LOCA TR are unaffected by the difference(s) between Versions 1.6 and 1.7 of NRELAP5 (Public: ML25058A423, Non-Public: ML25058A424) . In addition to the FOMs for non-LOCA system transient analyses listed in Section 3.4.2, “Design Basis Event Acceptance Criteria,” of this SE (i.e., maximum RCS pressure and maximum secondary pressure), TR Section 4.3.1, “Develop Plant Base Model NRELAP5 Input,” states that the RCS water level response is also evaluated for non-LOCA events that result in an RCS inventory decrease.

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

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

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

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

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

The applicant provides methodologies in the TR that must be used for performing sensitivity studies on the impacts of the axial power shape on the primary and secondary system pressure, flow and fluid temperature responses, to include the parameters and acceptance criteria. The staff reviewed the methodologies presented in the TR and finds that the methodologies are appropriate for use to perform safety analyses of non-LOCA events because they will be able to capture the impacts of the axial power shape on primary and secondary system pressure, flow rate and fluid temperature responses.

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

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

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

The NRC staff discussed these topics with the applicant during its audits (ML24262A257). The applicant clarified that it assumed burnup corresponds to an average value for a typical UO_2 core ranging from 0 gigawatt-days per metric ton of uranium (GWd/MTU) at beginning of cycle (BOC) to about **{{ }}** at end of cycle (EOC) (ML24348A054 Public, ML24348A055 Non-Public). The NRC staff confirmed that these values are consistent with those in the NuScale NPM-20 but recognizes that they may change if the fuel design or operation strategy changes. As TR Section 4.3.1.2.2 requires use of representative time-in-cycle core average burnup, it requires users applying the non-LOCA TR to examine these cycle-average fuel burnup values to assure they are within the assumed range because burnup has a significant impact on fuel thermal conductivity. NuScale described through response to audit questions (ML24348A054 Public, ML24348A055 Non-Public) that **{{**

}}. The NRC staff audited the NPM-20 NRELAP5 non-LOCA model calculation package and confirmed that the maximum core-average fuel exposure was calculated in this way. The applicant also stated that the fuel thermal conductivity is consistent with the burnup-dependent value calculated by the fuel performance code. This additional information adequately clarified the docketed material, and the NRC staff concludes that the interface with the fuel design analysis, as described in the TR, is acceptable.

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

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

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

3.4.3.2 Adapt Plant Base Model NRELAP5 Input for Event Transient Analysis

TR Section 4.3.2, "Adapt Plant Base Model NRELAP5 Input for Event 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 SE.

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 SE.

3.4.3.4 Evaluation of the Results of Transient Analysis Calculations

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

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

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

The NRC staff notes that, in some circumstances, the criterion that AOOs should not generate a postulated accident without other faults occurring independently or result in consequential loss of function of the RCS or containment barriers (the non-escalation criterion) may be challenged if SSCs are subjected to circumstances for which they are not qualified. For example, pressure relief valves may not be qualified to relieve water, and failure of such a valve to reseat could result in a design-basis accident (small-break LOCA). However, the staff notes that qualification of safety-related components is typically reviewed separately during licensing applications for

specific plants and plant designs in order to ensure that the design meets 10 CFR 50.49 and 10 CFR 50 Appendix A GDC 1, 2, 3, and 4.

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

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

TR Section 4.3.5 states that the impact of pressure on critical heat flux (CHF) is evaluated by varying initial PZR pressure. The NRC staff notes that variation of other parameters as prescribed by event-specific tables in TR Section 7, such as operation of pressure control systems or evaluating a spectrum of initiating events, also aid in variation of RCS pressure. The staff reviewed the methodology provided in the TR for determining the effect of pressure on CHF and audited the applicant's example calculations examining the sensitivity of CHF to RCS pressure for NPM-20 operating conditions and finds that these requirements provide sufficient confidence that the methodology will be able to identify and capture the impact of pressure on the CHF.

The NRELAP5 system analysis methodology for determining the limiting CHF cases for downstream subchannel analysis is primarily dependent on the limiting condition and event initialization. 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 **{{**

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TR Figures 4-1. Figure 4-2, and Figure 4-3 **{{**

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The applicant also stated that **{{** **}}**

{{

}} will be analyzed in VIPRE-01, at minimum (ML24348A027**).**

The NRC staff audited engineering documentation supporting Figures 4-2 and 4-3 (ML24262A257) to determine **{{**

}} MCHFR calculated by VIPRE-01.

During the audit, NRC staff observed **{{**

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Based on the description of the process to identify limiting cases as described in the TR, and confirmed during the staff's audit, the staff finds that the NRELAP5 calculations and screening methodology requirements provide sufficient confidence that potential challenging cases necessitating subchannel analyses will be identified.

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

3.4.3.6 Identification of Cases for Accident Radiological Analysis

As discussed in TR Section 4.3.6, "Identification of Cases for Accident Radiological Analysis," NRELAP5 transient analysis results are provided as input to accident radiological analyses for

events that result in reactor coolant loss outside of the containment (e.g., failure of small lines carrying primary coolant outside containment and steam generator tube failure (SGTF)). The conservative bias directions for accident radiological analyses are:

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

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

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

The TR provides various interface information to the radiological consequence analysis, including time of the reactor trip, time of the reactor coolant release isolation, time-dependent mass release, and other time-dependent system parameters. The staff notes that, similar to traditional large PWRs, accident radiological consequences for the NPM tend to increase with increasing integrated mass release outside of containment prior to isolation of the source, with iodine spiking and the timing of events potentially affecting the radiological consequences. On this basis, the NRC staff finds that the approach for identification of cases for accident radiological analysis is appropriate for the NPM design, and it is therefore acceptable.

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

3.5 NRELAP5 Applicability for Non-LOCA Transient Analysis

3.5.1 Non-LOCA Phenomena Identification and Ranking Table

A panel of experts developed the non-LOCA PIRT described in TR Section 5.1, “Non-LOCA Phenomena Identification and Ranking Table and Evaluation of High-Ranked Phenomena,” based on the state-of-knowledge at the time of the PIRT development. The PIRT was originally developed based on the earlier NPM-160 design and applied to the NPM-20 design. The applicant compared these two designs and identified that the major differences between the two designs are: (1) NPM-20 has a higher power output (250 MWth), (2) increased reactor temperature, and (3) increased reactor pressure, and some geometric and dimensional changes such as minor reflector bypass flow area, additional riser holes. However, the only significant

geometry changes affecting the NPM-20 non-LOCA system transient behavior are those associated with the DHRS loop. In addition, a new ECCS supplemental boron system is added to address potential re-criticality concern under long term core cooling operation. The methodologies for analyzing long term cooling events are discussed in the XPC TR (Reference 2).

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

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

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

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

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

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

Phase	Phase Description	FOM
1 – Pre-trip transient	Begins with the event initiation and ends with the actuation of the MPS.	• CHF (may be challenged by cooldown and reactivity-initiated events) • Primary pressure (may be challenged by heatup and RCS inventory increase events)

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

The TR states that if the coolant mixture level is not maintained above the top of the riser, natural circulation may be interrupted, ending Phase 3, and that this is well after reactor trip and engineered safety features have responded to the initiating event. The EM is applicable for the short-term non-LOCA transient progression, and during this time frame, the mixture level remains above the top of the riser and primary side natural circulation is maintained. The reactivity control and extended passive cooling analysis methodology in the long-term, including events that transition from DHRS cooling to ECCS cooling, is addressed in the XPC TR (Reference 2).

The staff performed confirmatory analyses on the liquid level during non-LOCA transients and confirmed that the mixture level remains above the top of the riser and that primary side natural circulation is maintained during the non-LOCA transient time frame. Based on the staff's evaluation of the information provided by the applicant, the NRC staff agrees that the riser uncovery and interrupted natural circulation scenario will not be encountered in the short- term following those non-LOCA events within the scope of the TR. The XPC EM TR (Reference 2) addresses the time after which mixture level has dropped below the top of the riser.

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

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

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

The knowledge level rankings are defined as:

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

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

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

The staff reviewed the discussions presented in TR Subsections 5.1.4.1 to 5.1.4.1.57 and finds that the applicant has correctly identified all phenomena that have significant impacts on the safety of the NPM design, the causes, progressions, safety significance, and the corresponding ranking. The staff also finds that the applicant followed the guidance provided in RG 1.203, "Transient and Accident Analysis Methods" (ML053500170) and industry practice in identifying and ranking the phenomena and knowledge gaps. On these bases, the staff finds that the applicant has correctly identified and ranked the highly safety significant phenomena and the knowledge gaps.

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

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

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

}} are of particular interest, among others. These and other highly ranked phenomena were discussed extensively in the audit (ML19039A090) conducted in the review of the previous version of this TR. The discussions clarified how the phenomena were appropriately considered, ranked, and

addressed. For review of the applicability of the PIRT to the NPM-20 design, the staff audited (ML24262A257) relevant documentation.

The NRC staff finds that the applicant adequately identified highly ranked phenomena and provided the corresponding knowledge levels, systems/components in which the phenomena are applicable, bases for the rankings, and explained how the phenomena are addressed. This conclusion is based on the NRC staff's knowledge and understanding of the NuScale NPM-20 design, the NRC approved PIRT for the NPM-160 design, and information from other LWR PIRTs that have been previously developed and/or approved by the NRC staff.

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

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

3.5.3 NRELAP5 Validation and Assessments for Non-LOCA

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

3.5.3.1 KAIST

As discussed in TR Section 5.3.1, "KAIST," the applicant used high-pressure condensation data from experiments performed at the KAIST facility to assess NRELAP5 predictions of condensation inside, and heat transfer across, DHRS tubes. The staff has reviewed the applicability of the KAIST experiment data for validation of the NRELAP5 computer code for the NPM-160 design. The staff's conclusions are documented in Non-LOCA TR-0516-49416-P-A, Revision 3 (ML20191A281 Public, ML20191A285 Non-Public). The staff reviewed the conclusions and finds them to be applicable to the NPM-20 design. Based on its review, the staff finds that the KAIST tests verify the EM against the various phenomena identified in the PIRT. Since the PIRT for the NPM-160 was confirmed during the review of revision 3 of TR-0516-49416-P-A, and staff finds the PIRT to be applicable to the NPM-20 because no new phenomena were introduced between the NPM-160 and NPM-20 design in terms of operating range and geometry, the staff finds that the KAIST tests are applicable to the NPM-20. For this

reason, the staff decided not to include a detailed discussion of the assessment in this SE version.

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

3.5.3.2 NIST-1 Decay Heat Removal System Separate Effects Tests

As part of the non-LOCA EM validation, the applicant performed SETs at the NuScale Integral System Test-1 (NIST-1) test facility, which is described in Section 5.3.2.1, "NIST-1 Facility," of the TR. The staff previously reviewed the applicability of the NIST-1 experiment data for validation of the NRELAP5 computer code for the NPM-160 design. The staff's conclusions are documented in non-LOCA TR-P-A, Revision 3 (ML20191A281 Public, ML20191A285 Non-Public)). The staff reviewed the conclusions and finds them to be applicable to the NPM-20 design. For this reason, the staff decided not to include detailed discussion of the assessment in this SE. The staff reviewed these conclusions, and the information provided in the response to the RAI (ML19221B483), and finds they remain appropriate for the NPM-20 design because determining the time step size based on the material Courant limit is derived from the basic physical principle and there is no change in the NPM-20 dimensions compared to NPM-160.

The L&C specified in the SER for Revision 3 of the TR (ML20148M391) states that if the NPM design changes significantly from what the staff has reviewed (e.g., MPS logic changes that impact non-LOCA transient progressions, reduced margin to acceptance criteria), additional justification would be needed to confirm that the application of a DHRS heat transfer bias is not necessary. The staff notes that this L&C remains valid because the TR remains the same as in Revision 3 with respect to relevant integral and separate effects tests for NRELAP5 code validation. Therefore, the NRC staff retained a modified L&C in Section 4 of this SE that an applicant or licensee seeking to apply this methodology to a design other than the design represented in the NPM model, Revision 5 (or any NPM model update made pursuant to a change process specifically approved by the NRC for changes to the NPM model) must evaluate DHRS heat transfer biases to determine if the elimination of the biases within this methodology remains justified based on the margins to non-LOCA FOMs.

Sections 5.3.6 and 5.3.7 of TR Rev 4 are meant to provide additional benchmark comparisons to improve the validation basis for NRELAP5 simulation of SG and DHRS heat transfer as described in SER Sections 3.5.3.6 and 3.5.3.7.

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

3.5.3.2.1 NIST HP-03 Separate Effects Tests

TR Section 5.3.2.4, “HP-03 Test Description,” describes the NIST-1 HP-03 tests, which used a full-height DHRS heat exchanger to assess the ability of NRELAP5 to predict condensation within, and heat transfer across, the DHRS tubes. As discussed in the SER for non-LOCA TR-P-A, Revision 3 (ML20148M391), the staff found that the NIST-1 HP-03 SETs, together with the KAIST tests adequately covered part of the expected ranges of the DHRS operation because the NIST-1 HP-03 tests used DHRS pressure and temperature that are lower than that of the NPM-160 design. With respect to the NPM-20, although the NIST-1 HP-03 tests used DHRS pressure and temperature that are lower than that of the NPM-20 design, the staff finds that the tests are still valid for the lower ranges of the NPM-20 DHRS pressure and temperature during DHRS operation because of the rationale above; the staff recognizes that the DHRS temperature and pressure may be higher than the NIST HP-03 tests at actuation and then go down as the system cools.

For primary to secondary heat transfer, the applicant referenced the assessment of NRELAP5 predictions of the SIET TF-1 (for secondary side heat transfer) and SIET TF-2 (for primary to secondary side heat transfer) tests. The NRC staff’s evaluation of the NRELAP5 assessment against the SIET tests is documented in Section 3.5.3.5, “Steam Generator Modeling,” Section 3.5.3.6, “Heat Transfer Correlation Comparison” and Section 3.5.3.7, “NIST-2 Steam Generator – Decay Heat Removal Systems Integral Effects Tests,” of this SE. However, the assessments considering primary-to-secondary heat transfer were limited in scope. To account for this, the LTR provides event-specific SG heat transfer biases.

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. This bounding is accomplished by assuming temperatures that bound fixed Technical Specification allowed “low” or “high” pool temperatures, depending on the transient being considered. The staff requested (ML24305A002) the applicant to submit supplemental information to elaborate on their justification for using such a simplified pool model because it may not capture the effect of potentially elevated pool water temperature in very close proximity to the DHRS condenser tubes (i.e. boiling on the condenser tube outer surfaces). The staff expected that film boiling could potentially occur on the surface of DHRS condenser tubes because the upper part of the tubes outer surfaces are predicted to operate in the subcooled boiling or near nucleate boiling regime. Additionally, there are some uncertainties in the heat transfer correlations, and the code may be underrelaxing the temperature rise rate in the condenser tube walls such that instantaneous surface temperatures are potentially underpredicted, which could produce film boiling at the onset of DHRS operations, when the secondary pressures increase after isolation and steam temperatures are highest.

Upon secondary isolation and subsequent DHRS actuation, based on the extent of RCS pressurization—due to loss of the SG heat sinks—and the average primary system temperature reached during these transient excursions, the resulting secondary steam generator pressures occurring could reach 1550 psia, indicating that the steam temperatures at DHRS inlet could reach 600 degrees F. This scenario is very important for non-LOCA events, particularly if one train of the DHRS loop is disabled by the event, but in addition this scenario is important for small LOCAs where RCS depressurization may be very slow. If the RCS pressurization is not turned over by DHRS heat removal, the RSVs could lift to relieve pressure or the ECCS logic may open the RVVs. The staff notes that in version 5 of the NRELAP5 basemodel, the

applicant has added {{ }},
which will decrease fluid enthalpy at the DHRS condensers inlets.

The applicant provided supplemental information (ML25014A156 Public, ML25014A157 Non-Public) with details of DHRS startup from cold conditions, and an evaluation describing detail of the associated NRELAP5 DHRS modeling and heat transfer correlations involved, with some benchmarks using NIST integral DHRS test data. During normal plant operations, the DHRS is on standby and filled with feedwater up to the actuation valves, which are located above the pool. After secondary isolation and DHRS actuation, steam enters the top of the DHRS and the liquid in the DHRS loop redistributes between the DHRS and SG. Steam from the steam generators diverted to the DHRS piping begins to condense, and subcooled/saturated nucleate boiling on the outer condenser tube wall is efficient and capable of removing the energy conducted through the pipe wall from the steam condensation. {{

}}. The staff acknowledges the reality of this phenomena comparison based on NRELAP5 output but consider that these results are influenced by the overly simplistic modeling used. The staff also considers that the NIST DHRS testing does not adequately represent the geometry or the extremely high temperature and pressures of the NPM-20 that could result, at least initially, in a violent boiling excursion leading to film boiling, {{

}}.

The applicant provided a review of the NRELAP5 pool boiling correlation options selected in the input modeling which simulate natural convection heat transfer in which they justify adequacy of the 1-D reactor pool modeling approach. {{

}}. The applicant stated that the natural circulation flow along the outer tube surfaces is inherently included in the correlations themselves, which staff agrees are developed from a vast array of reasonably similar test conditions. For the temperature boundary condition evaluations, the applicant again points to the inherent developmental basis of the correlations to include thermal boundary layer effects. The staff, however, does not agree with this assertion because of the extremely large volume of the node attached to the DHRS tube surface. The applicant also acknowledged that an axial temperature change in the bulk fluid will develop, especially near the top of the DHRSs, where heat transfer rate is highest but concluded in their response that the correlations in NRELAP5 can adequately account for the natural convection heat transfer and boiling heat transfer on the DHRS tubes.

The applicant additionally provided more detail geometry descriptions of the DHRS tubing layout indicating that the effective tube pitch used is considerably larger than typical tube bundle condensers, improving the likelihood that saturated boiling can be maintained and avoid potential for higher heat fluxes that could cause film boiling. Also, the DHRS tube thickness is larger than that generally used, which reduces the potential for higher heat fluxes on the tube outer surface that could lead to film boiling.

The proper modeling, description, validation, and uncertainty analysis of the DHRS heat exchangers is essential to confirm that the LOCA EM and the Non-LOCA EM adequately

represent the NPM-20 responses in AOO and design basis accident events. To substantiate and expand upon the evaluations provided by the applicant, the staff performed several sensitivity studies with NRELAP5 and TRACE and the results confirmed that transition/film boiling did occur in short periods in some severe scenarios. Yet, it would be limited to the upper condenser tube region and would not be sustained to the point where it would cause any significant degradation in DHRS design performance capability. Therefore, the staff finds that the DHRS modeling and coupled pool nodalization is sufficient to model the overall decay heat removal responses and heat transfer capability. As discussed in Section 3.6.1, "Thermal-Hydraulic Volumes and Heat Structures," of this SE, the NRC staff finds this treatment acceptable.

TR Section 5.3.2.5.4, "HP-03 Summary," summarizes the comparison of the NRELAP5 code predictions to the HP-03 test series data. The staff's evaluation of the HP-03 setup and testing results is documented in the approved version of Revision 3 for this TR. The staff reviewed the test facility, compared with the reactor design, **{{**
}}.

3.5.3.2.2 NIST HP-04 Separate Effects Tests

TR Section 5.3.2.6, "HP-04 Test Description," describes the NIST-1 HP-04 test series performed to assess the ability of NRELAP5 to predict **{{**
}}. Like the HP-03 test setup, steam produced in the SG was routed to the simulated full-height DHRS, and the condensate line discharged to the environment. The HP- 04 test series consists of two runs at different DHRS pressures, as shown in TR Table 5-10, "NIST-1 HP04 test ranges."

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

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

The staff reviewed the NIST HP-04 test facility and results and finds it to be sufficiently similar to the NPM-20 design and the test results are appropriate for the NRELAP5 code validation for analyses of **{{** **}}** phenomena. The SER for Revision 3 of the non-LOCA EM (ML20148M391) provides the detailed staff evaluation of the descriptions and test results of the NIST HP-04 test facility.

3.5.3.3 NIST-1 Non-LOCA Integral Effects Tests

TR Section 5.3.3, "NIST-1 Non-LOCA Integral Test," discusses the NIST-1 facility non-LOCA IETs, which include NLT-02a, NLT-02b, and NLT-15p2. The objectives of these tests were, respectively: to measure the integral response to a loss of feedwater transient to the point of a

reactor trip; to examine DHRS-driven cooling following the initial DHRS actuation; and to measure the integral response to a loss of feedwater transient and subsequent DHRS cooling.

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

3.5.3.3.1 NIST-1 NLT-02a Test

TR Section 5.3.3.4, "NLT-2a Test Description," provides selected initial conditions and the sequence of events for the NLT-02a loss of feedwater test, and the test results are presented in TR Section 5.3.3.5, "NLT-2a Test Results." **{{ }}**. TR Section 5.3.3.5 compares the NRELAP5-calculated values for primary and secondary parameters against the test data for the first 150 seconds after feedwater flow interruption. Feedwater flow (TR Figure 5-41, "NLT-02a transient feedwater flow comparison"), core heater rod power (TR Figure 5-42, "NLT-02a transient core heater rod power comparison"), and steam line pressure (TR Figure 5-51) were boundary conditions for the NRELAP5 simulation. The applicant added results obtained with NRELAP5 Version 1.6 to the LTR. Primary pressure and core inlet temperature simulation results (TR Figures 5-43a, "NLT-02a transient pressurizer pressure comparison," 5-45a, "NLT-02a transient pressurizer level comparison," and 5-46a, "NLT-02a transient core inlet temperature") are within the data uncertainty bands and follow the trend of the data well, and therefore, the NRC staff agrees with the applicant that these parameters show a reasonable-to-excellent or excellent agreement with the test data. The applicant concluded that all other calculated parameters demonstrate reasonable agreement and, based on its review of the parameters, the NRC staff agrees with the applicant's conclusion.

The NRC staff noted as part of the previous review for revision 3 of TR-0516-49416-P-A that the riser mass flow rate (TR Figure 5-44a, "NLT-02a transient riser mass flow rate comparison") generally showed the least agreement of the parameters, as the prediction was outside the measurement uncertainty for the duration of the test. The NRC staff previously audited sensitivity studies, as described in the associated audit report (ML20036C849), that the applicant performed to assess **{{ }}**. The applicant also described (ML18270A466) modeling approaches associated with some level of uncertainty that could contribute to the overpredicted riser mass flow rate, such as **{{ }}**. While the NIST-1 tests are not scaled to NPM-20 conditions, these conclusions form a partial basis for NRELAP5 validation for the NPM-20 because no new phenomena were introduced in the NPM-20 DHRS design. NIST-2 tests provide NRELAP5 validation that is scaled to NPM-20 conditions.

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

3.5.3.3.2 NIST-1 NLT-02b Test

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

PZR level presented in TR Figure 5-55a, “NLT-02b phase 1 transient pressurizer level comparison”) was calculated using Version 1.6 of the NRELAP5 code **{{ }}**, which the applicant attributed to **{{ }}**. In addition, the NRC staff notes that similar trends are observed on the predicted DHRS power using NRELAP5 Version 1.6 as shown in Figure 5-65a and Figure 5-66a comparing to TR Figure 5-61, “NLT-02b phase 1 transient DHRS heat exchanger thermal power comparison,” is **{{ }}**. Based on the comparisons between the calculated and test results, the NRC staff concluded that the **{{ }}** DHRS condensate temperature (TR Figure 5-65a, “NLT-02b phase 1 transient DHRS condensate temperature comparison”) and **{{ }}** were sufficiently addressed by the applicant, as described in Section 3.5.3.2.1, “NIST HP-03 Separate Effects Tests,” of this SE. On these bases the staff determined that this conclusion remains valid for the NPM-20 design.

TR Section 5.3.3.8, “NLT-2b Phase 2 Test Results,” compares NLT-02b Phase 2 test results to NRELAP5 predictions. Phase 2 spans the period of **{{ }}**. The NRC staff agrees with the applicant that most key parameters show a reasonable-or-better agreement between the predictions and data for Phase 2. Like Phase 1, the Phase 2 CPV and condensate temperatures (TR Figures 5-87a, “NLT-02b Phase 2 transient cooling pool vessel region 5 temperature comparison (near bottom of DHRS heat exchanger),” to 5-88, “NLT-02b phase 2 transient cooling pool vessel Region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region),” and 5-82a, “NLT-02b Phase 2 transient decay heat removal system condensate temperature comparison,” respectively) **{{ }}**. The predicted DHRS condensate flow (TR Figure 5-83a, “NLT-02b phase 2 transient decay heat removal system condensate flow comparison”), and consequently, the DHRS power (TR Figure 5-79, “NLT-02b Phase 2 transient decay heat removal system heat exchanger thermal power comparison”), **{{ }}** but still exhibit a reasonable agreement with the data. The overall trends observed between NRELAP5 v1.4 and 1.6 remain similar. On these bases, as well as those provided in this SER concerning differences between NRELAP5 v1.6 and v1.7, the staff determined that this conclusion remains valid for the NPM-20 design.

TR Section 5.3.3.9, “NLT-2b Phase 3 Test Results,” describes NLT-02b Phase 3. The staff’s evaluation of NLT-2b Phase 3 Test is documented in the SER for non-LOCA-P-A, Revision 3 (ML20148M391). The staff reviewed the SER and finds that the conclusions made in the review of Revision 3 of this TR remain applicable to the NPM-20 design because between the minor

geometric changes to the DHRS loop and despite the substantial core thermal power increase for NPM-20 over NPM-160, the DHRS is a natural circulation flow driven by the heat input into it. Within reasonable limits, the condensate level in the condenser tubes adjusts to the applied steam pressure such that the DHRS will transfer heat to the reactor pool at a rate in proportion to the rate at which heat is supplied to it.

3.5.3.3.3 NIST-1 NLT-15p2 Test

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

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

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

}}.

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

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

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

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

}}. The applicant judged the NRELAP5 predictions for differential pressures across the DHRS condensate line (TR Figure 5-147, "NLT-15-p2, transient DHRS condensate line differential pressure") and steam line (TR Figure 5-148, "NLT-15-p2, transient DHRS steam line differential pressure") to be **{{**

}}. The NRC staff agrees with the applicant's assessment of these parameters because **{{**

}}

{{ }}.

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

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

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

3.5.3.3.4 NIST-1 Integral Effects Tests Summary

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

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

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

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

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

> For some plants and transients, code-to-code comparisons can be very helpful. In particular, if a new code or device is intended to have a limited application, the results may be compared to calculations using a previous code. However, the

previous code should be well-assessed to integral or plant data for the plant type and transient being considered for the new device. Differences in key input (such as system nodalization) should be explained so that favorable comparisons provide the right answers for the right reasons. Such benchmark calculations would not replace assessment of the new code.

The staff agrees with the applicant that the previous assessment on the adequacy of the point kinetic model for modeling NPM-160 is acceptable for the NPM-20 design because the power increase and possible larger flux gradient do not impact the system's response to reactivity change, which is a global parameter. The SER for non-LOCA TR, Revision 3 (ML20191A281 Public, ML20191A285 Non-Public) provides more detailed discussions on validation of the NRELAP5 computer code that the staff finds to be valid for the NPM-20 design.

3.5.3.5 Steam Generator Modeling

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

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

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

condition {{
}}. Secondary side laminar and turbulent heat transfer correlations for single-phase flow discussed in Section 6.7.2.1, “Helical Coil Single-Phase Heat Transfer,” of the LOCA TR {{
}}. As described in Section 6.7.2.2, “Helical Coil Two-Phase Subcooled and Saturated Flow Boiling Heat Transfer,” of the LOCA TR, two-phase subcooled and saturated boiling heat transfer are {{

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

}}. Therefore, this option is not approved for this revision of the non-LOCA EM as reflected in L&C #7.

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

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

}}.

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

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. During review of revision 3 of TR-0516-49416-P-A, the NRC staff reviewed the figures provided in Section 7.4.1 of the LOCA TR and the NRC staff agreed with the applicant’s assessment. In addition, the NRC staff’s confirmatory calculations using the TRACE code showed reasonable to excellent agreement with the TF-1 data and support the applicant’s TF-1 conclusion noted above. The staff compared the testing facility and the NPM-20 design and concludes that the test results and staff’s

conclusions remain applicable to the NPM-20 design because the NPM-20 is very similar to the NPM-160 in SG design in terms of operating condition and geometry.

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. In addition, integral effects testing of the DHRS to support the applicability of the EM was performed at the NIST-2 facility. Section 5.3.7 of the TR provides a summary of the test. While compensating errors during calculation of primary and secondary side heat balances that might mask errors in primary flow are possible, the NRC staff observed no such errors during its review of the revised TF-2 assessment data.

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

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The NRC staff agrees that the TF-2 test data-to-model comparisons presented in the LOCA TR are in reasonable to excellent agreement. However, due to the concerns and potential limitations noted above, the NRC staff could not confirm that the TF-2 tests fully represent NPM steady-state and non-LOCA transient conditions or that the SG heat transfer coefficient biases were appropriately conservative for non-LOCA events. To address the question of SG heat transfer biasing during review of revision 3 of TR-0516-49416-P-A, 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). Staff evaluation of these sensitivity analyses are in the safety evaluation report for revision 3 of TR-0516-49416-P-A.

Revision 4 of TR-0516-49416 was submitted concurrently with the SDAA for the US460 design. The NRC staff audited SG heat transfer sensitivity studies that were included as part of NPM-20 analysis of increase in heat removal, decrease in heat removal, reactivity anomaly, and increase in RCS inventory events. Staff observed that non-LOCA figures of merit were not sensitive to SG heat transfer in these calculations.

Based on its review of the sensitivity study information, as confirmed in audits, the NRC staff agrees that the FOMs for non-LOCA events are insensitive to reasonable variations in SG heat transfer for the design reflected in NPM model Revision 5. Therefore, the staff finds that the applicant’s sensitivity studies using NPM model Revision 5 are sufficient to support the lack of a SG heat transfer bias as part of the non-LOCA EM.

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

full power.

The NRC staff notes that the effect of SG heat transfer on normal operations (steady-state initial conditions) is addressed by the applicant's technical specifications (TS) in the NuScale SDAA FSAR, Chapter 16, which are based on the values supported by the safety analysis, specifically SDAA FSAR Table 15.0-6, "Module Initial Conditions Ranges for Design Basis Event Evaluation."

Based on the relative 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 necessary for the NPM-20 design reflected by NRELAP5 non-LOCA model Revision 5 (noting that input file version numbers were reset to 1 for the NPM-20 design). The NRC staff will require justification be provided for 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 appreciably decrease. The LTR provides event-specific SG heat transfer biases.

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 event-specific biases described above.

3.5.3.6 Heat Transfer Correlation Comparison

The staff audited (ML24348A010) the CFD calculations performed by the applicant and found that the applicant analyzed several cases for comparison to literature heat transfer correlations **{{ }}** as well as for parameter and mesh size sensitivity evaluation. The NRC staff also finds that the applicant's CFD calculation results indicate the **{{ }}** for crossflow over horizontal tubes is applicable to NuScale's unique helical coil steam generator.

In TR Section 5.3.6.3 the applicant states that NRELAP5 Version 1.7 provides the option of using **{{**

}} is in Section 3.5.3.5 of this SE.

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

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

The applicant performed tests for the performance of the DHRS under Non-LOCA events. To assess and validate the NRELAP5 code in predicting SG-DHRS heat transfer behavior, the applicant developed the NIST-2 testing facility. The NIST-2 facility is an upgrade to the NIST-1 facility, as discussed in Section 3.5.3.3 of this SE. The key upgrades were to increase the maximum allowable pressure for the main steam and DHRS piping to perform scaled separate and integral effects tests for the NRELAP5 code that is used for NPM-20 safety analyses. When compared to NIST-1, the NIST-2 facility test conditions are closer to the NPM-20 operational

conditions. Per the Non-LOCA PIRT discussion in Section 3.5.1 of this SE, several high ranked phenomena are identified to have particular relevance to DHRS operation in non-LOCA Phase 3 (stable natural circulation phase). The staff reviewed the description of the NIST-2 tests (ML24215A251) to evaluate the DHRS performance during LOCA and non-LOCA transients. The NIST-2 test assessment for LOCA events are primarily discussed in Section 4.7.5 of the LOCA TR SE (see Reference 3)). In this section, the NRC staff focused on the NIST-2 non-LOCA SG-DHRS performance during a limiting loss of feedwater (LOFW) transient because this event is related to the performance of the secondary side of the SG which also serves as the heat removal function of the DHRS. Besides Section 5.3.7 "NIST-2 SG-DHRS Integral Effects Tests" of the non-LOCA TR, staff also audited (ML24215A240) the engineering calculation packages to assess the applicability, including scalability and distortion, of the test facility and the testing results, and the applicant provided further information to address these scaling issues (ML25042A064 Non-Public, ML25042A063 Public).

It was important for the applicant to show that DHRS in NIST-2 is correctly scaled for the NPM-20 to support acceptable code validation. Documentation supporting the TR includes two reports for the scaling approach: EE-107403 Section 3.2 and ER-126485 Section 3.4.2. NRC staff audited these documents and noted that the approach correctly identifies nondimensional similarity (PI-) groups for natural convection, liquid mass balance and rate of change of system pressure (ML24348A029 Public, ML24348A030 Non-Public). However, **{{**

}}. The DHRS actuation period may incur distortion, but the actuation period is fairly short when compared to the long-term operation period. The NRC staff finds this acceptable because the short period distortion does not significantly contribute to the overall distortion.

Like the NIST-2 facility, the NPM-20 DHRS is a heat exchange circuit that is connected to the SG secondary side and a condenser which is immersed in the reactor pool. A DHRS actuation signal causes the DHRS actuation valve to begin a slow opening and the SG secondary side to isolate by closing the steam and feedwater isolation valves. The in-DHRS-loop steam is produced in the HCSG and condenses in the DHRS condensers. Condensate returns to the portion of the feedwater line that is isolated and reenters the HCSG. Applicability of NRELAP5 to model the NPM-20 DHRS is determined by modeling tests with active DHRS such as the SG-DHRS IET tests conducted in the non-LOCA NIST-2 test series. Condensation induced oscillations were observed in some of the tests. These oscillations are geometry and thermal power dependent. Nevertheless, the thermal energy is transferred out of the RPV and into the reactor pool through vaporization, flow, and condensation. The DHRS loop conditions will adjust to SG conditions. The heat transfer to the reactor pool is an important figure of merit for the DHRS. The hydraulic resistance and water level (height) in the DHRS condenser tubes and inside the helical coils are important factors influencing the natural circulation inside the DHRS loop.

The applicant's NIST-2 scaling and distortion analysis for the SG/DHRS loop is focused on demonstrating the similarity of the testing facility to the NPM-20 for the quasi-steady heat removal conditions during DHRS operation in non-LOCA phase 3. It would be inappropriate to generally apply all NIST-2 IET results for validation of NRELAP5 simulation of NPM-20 stability during the long-term cooling phase. The staff reviewed the NIST-2 tests and the sensitivity study provided (ML25042A064 Non-Public, ML25042A063 Public), with supporting calculations ; the staff finds that the DHRS flow oscillation observed in NIST-2 IET test RUN 2 does not

significantly affect the integral ability of the NIST-2 NPM-20 DHRS-surrogate test to remove stored energy and decay heat, nor does it significantly affect the margins to FOMs predicted by NRELAP5. It is important to note that the RUN 2 test where oscillations in condensate level were observed utilized the most restrictive steam line orifice of all the NIST-2 DHRS experiments, and, that this orifice is more restrictive than that included in the NPM-20 design.

In a boiling two-phase heat transfer leg, as the ratio of partial pressure drop in the vapor-only length to total pressure drop in the boiling leg increases, the susceptibility of the leg to two-phase flow oscillations (e.g. compressible volume interaction (aka “pressure drop oscillation”)) increases. It is expected that an orifice with sufficient hydraulic resistance would cause the NIST-2 DHRS experiment to oscillate. In the condensation leg of the two-phase DHRS loop in NIST-2, the condensate level oscillations were not large enough to be described as the two-phase instability known as “condensation chugging.” The staff notes that **{{**

}}. It is clear to the staff that while NRELAP5 can adequately capture the overall integrated heat transfer of NIST-2 DHRS experiments, it cannot replicate the two-phase instability in the loop or an associated oscillation in system heat transfer rate; the implication here is that NRELAP5 should not be relied on to predict whether two-phase flow instabilities with interruptions in heat transfer rate will occur in the as-built DHRS system. However, staff emphasizes that the evidence gives no reason to expect that similar potential oscillations would meaningfully degrade the integral DHR system performance. Structural damage or material fatigue evaluations are outside the scope of the non-LOCA evaluation model.

The NIST-2 SG-DHRS integral effects testing is used to qualify the NRELAP5-based non-LOCA EM, **{{**

}}. Three NIST-2 integral effects tests (Runs 1 to 3) are modeled with NRELAP5 to develop a validation for the non-LOCA EM. The staff audited (ML24262A257) the tests and the corresponding NRELAP5 simulations, focusing on the FOMs that are affected by DHRS performance during phase 3 of the non-LOCA events. The code-to-data differences due to test configuration differences or NRELAP5 modeling simplifications are identified (ML25042A064 Non-Public, ML25042A063 Public). The staff finds that in general the NRELAP5 simulation shows reasonable agreement to NIST-2 test results with respect to core power, loop inventory, levels, DHRS flow rate, and heat removal rate. **{{**

}}. The staff notes that the NRELAP5 simulations tend to **{{**

}}, as seen from Figures 5-226, 5-227, and 5-230 of the TR, which is conservative. From this comparison, the staff did not observe that significant flow instabilities occurred in the DHRS loop. **{{**

}}. The staff notices that Figures 5-236 thru 5-238 of the TR show that the code prediction of cooling pool vessel liquid level and temperatures deviate from test data with time. The applicant believed this difference is due to **{{**

}}. Based on the discussion above, the staff concludes the explanation of the code-to-data discrepancy of CPV parameters is acceptable.

In the TR and scaling analysis reports, the applicant extends the similarity and applicability of DHRS tests in non-LOCA SG-DHRS IET to the LOCA condition by stating that the phenomena

before ECCS actuation are similar to the test conditions in the NIST-2 non-LOCA tests. A few sensitivity study cases with small break LOCAs **{{**

}} were performed besides the base case of the LOFW with no break to confirm that there were no non-LOCA PIRT unidentified high ranked phenomena for DHRS in a small break LOCA. From the NRELAP5 simulation of NIST-2 LOFW cases that are shown on Figures 5-204 to 5-211, the staff observed consistent results for RCS temperature, PZR pressure, DHRS flow and heat transfer rates. **{{**

}}. Hence, staff finds that the NIST-2 SG-DHRS integral effects test performance can be applicable to small-break LOCA scenarios before ECCS actuation.

The staff noted however, that there is no data nor analysis for the phenomena after ECCS actuation and in the long-term cooling period where the riser uncovery occurs and SG coils are exposed to hot steam. To address the issue with lack of data or analyses, the applicant provided information (ML25042A064 Non-Public, ML25042A063 Public) to address the staff's concern. Before ECCS actuation in LOCA events, there is a short period that the RCS level may fall below the top of the riser and the steam generator tubes are exposed to water vapor. NuScale explains that the riser hole flows in NPM-20 provides convective heat transfer as the level first falls below the top of the riser. If more of the tubes are exposed, uppermost "group 4" riser hole flows stop and there is condensation heat transfer to heat up the secondary fluid. In other words, in this short period before ECCS actuation, steam production needed for the DHRS heat exchanger will not be impacted. This is demonstrated in several small break loss-of-coolant accidents described in audited calculations (ML24264A049). No significant degradation of DHRS heat transfer is found in this period. After ECCS actuation, the DHRS heat transfer is insignificant compared to ECCS heat transfer. Based on numerous Chapter 15 event analyses and associated sensitivity studies, the DHRS model is reasonably proven to be applicable in the LOCA analyses. Therefore, the staff approves the extension of DHRS EM applicability from non-LOCA to LOCA events.

The applicant uses a top-down scaling approach to define the PI groups and provided plots. **{{**

}}, NuScale designed several specific tests in the SG-DHRS test suite. In these tests, NuScale varied the steam section pressure drop by changing the orifice size. The results in the audited scaling analysis, the PI groups, show similar DHRS operation and heat transfer phenomena in different pressure drop ranges. Based on the discussion above, the staff concludes that the additional tests conducted are adequate to characterize the distortion and uncertainty and are therefore acceptable. Other PI groups have acceptable distortions as shown in Figures 4-15 to 4-17 (ML24215A235**)**.

The staff audited (ML24262A257) the IET data evaluation report on the comparison NRELAP

prediction of DHRS performance with data for a few transient runs in NIST-2. Some differences were observed. For example, **{{**

}}. Similar oscillations also occur in calculations the staff audited (ML24262A257) relative to the US460 SDAA NPM-20 simulation of some Chapter 15 transients. NuScale provided more analyses (ML25042A064 Non-Public, ML25042A063 Public) regarding the oscillatory behaviors. In the analyses, **{{**

}}. None of these sensitivity studies indicate the generation of steam entering the DHRS condenser is impacted. In other words, the direction of the steam generator steam flow is always positive, and the magnitude is minimally affected. Degradation of DHRS heat transfer is not seen in these analyses. The staff concluded that the oscillation of DHRS condensate flow is common in a passive isolated boiling-condensation loop, Mismatch of timing of the boiling and condensation processes could cause oscillation in the loop. The applicant states **{{**

}}, and thus the performance in the DHRS condenser in not affected. The staff considers the above rationales acceptable to address the oscillation issue.

In reviewing the NRELAP5 assessment, the influence of DHRS loop inventory on heat removal was identified by the applicant. Several studies were provided to address the staff's concern on the similarity between NIST-2 and NPM. **{{**

}}.

{{

}}. The staff issued an RAI (ML25042A064 Non-Public, ML25042A063 Public) to request the applicant to provide additional information **{{**

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{{

}}.

To avoid compensating errors in the assessment process, the staff requested additional scaling analysis to justify the distortion via PI group is insignificant compared to other high ranked phenomena in the transient. In response to the staff's request, NuScale provided **{{**

}}, the staff accepts the response.

The staff notes that the NPM-20 DHRS is a passive heat transfer system. It is designed to transfer heat from the RPV to the pool, which is the ultimate heat sink. The staff agrees that the proposed design meets this objective. However, the rate of heat transfer is a high ranked phenomenon that affects SG and RPV pressures and temperatures in various operating conditions. The staff concludes that distortions exist between the integral test facility (NIST-2) and the plant (NPM-20), and the uncertainty of NRELAP5 prediction of NPM-20 DHRS heat transfer was not adequately quantified (ML25014A269 Public, ML25014A271 Public, ML25014A272 Non-Public). Therefore, L&C #4 is issued for future application of the DHRS EM in analyzing non-LOCA transients.

The staff's evaluation of the NRELAP5 code for predicting the LOCA event is documented in the SE for the LOCA TR (Reference 3). The staff's evaluation of the NRELAP5 code for predicting the performance of the DHRS during extended cooling is documented in the SE for the XPC TR (Reference 4). Considering both the code and NPM design changes, NRELAP5 Version 1.7 is generally applicable for use in the non-LOCA EM, subject to L&C #7 of this SE.

3.5.4 Conclusions of NRELAP5 Applicability for Non-LOCA

TR Section 5.4, "Conclusions of NRELAP5 Applicability for Non-LOCA," summarizes the applicant's conclusions regarding the applicability of the NRELAP5 Version 1.7 computer code to the non-LOCA transient analyses. The applicant concluded that, based on the highly ranked non-LOCA phenomena and the various methods used to address them, NRELAP5 is applicable

to the non-LOCA analysis. Based on the evaluations in the preceding subsections, the NRC staff finds that the applicant has adequately addressed the phenomena important to non-LOCA events and has demonstrated that NRELAP5 is an acceptable tool for non-LOCA event analysis, subject to the limitations and conditions in Section 4 of this evaluation.

3.6 NuScale NRELAP5 Plant Model

TR Section 6, "NuScale NRELAP5 Plant Model," describes how NPM plant components and processes are modeled with NRELAP5 for non-LOCA transients. The descriptions cover modeling of the reactor primary and secondary (SG) systems, fuel, ECCS, DHRS, CNV, reactor pool, and protection and control systems. TR Section 6 also provides modeling of the reflector bypass region and modeling of the lower and upper riser holes.

NRC staff noted in multiple instances that plant descriptions are written to apply to the NPM-20 or NPM-160 design. Additionally, several figures showing model nodalization were not updated from Revision 3 to Revision 5 of the topical report, and still show features that are specific to the NPM-160 design, such as inclusion of three RVVs. However, during the review NRC staff noted several changes to the topical report methodology that are based on the NPM-20 design and raised questions regarding the applicability of the methodology to the NPM-160 design. Although the approved version of the topical report includes these figures representing the NPM-160 design and generic description in the topical report, Revision 5 of this methodology was reviewed and approved for application to the NPM-20 design, as specified in L&C # 1.

3.6.1 Thermal-Hydraulic Volumes and Heat Structures

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

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

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

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

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

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

The NRC staff noted that CHF is directly affected by natural circulation flow. Discussion of the

high importance phenomena **{{ }}** listed in TR Table 5-3, “High-ranked phenomena for non-LOCA events,” states that **{{ }}**. The NRC staff finds that the effect on CHF margin is conservative **{{ }}**.

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

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

{{ }}.

As discussed in Section 3.5.1, “Non-LOCA Phenomena Identification and Ranking Table,” of this SE, several non-LOCA highly ranked phenomena that are identified in Section 5.1.4, “Highly Ranked Phenomena,” of the non-LOCA TR, including **{{ }}**, are not reflected in the NRELAP5 non-LOCA EM representation of the NPM**.** Table 5-3, “High- ranked phenomena for non-LOCA events,” of the TR states that the phenomena are addressed by the subchannel analysis except for **{{ }}**. The NRC staff confirmed through audit during review of revision 3 of TR-0516-49416-P-A, as documented in the associated audit report ML19039A090, that those highly ranked phenomena are not relevant to the non-LOCA TR except for **{{ }}** and are instead applicable to other portions of the non-LOCA EM (e.g., the subchannel analysis), and, the staff position remains the same for the NPM-20.

Section 6.1.1, “Reactor Primary,” of Revision 3 of the TR, stated that **{{**

}}. The NRELAP5 modeling for the NPM-20 design, along with Revision 5 of the TR, have been modified to combine the lower and upper riser sections that were named at the start of this paragraph. As indicated above, several highly ranked phenomena identified by the applicant are related to multi-dimensional flows and complex flow behavior **{{**

}}.

During review of revision 3 of TR-0516-49416-P-A, as documented in the associated audit report (ML18234A537), the applicant described expected multi-dimensional flow and thermal behavior in **{{**

}}.

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

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

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

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

}}.

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 SE, 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 **{{**
}} (ML24348A048) with power distributed solely in the fuel pellet as specified by the applicant (ML24348A048 Public, ML24348A049 Non-Public), which the NRC staff finds to be acceptable, as discussed in Section 3.4.3.1, “Develop Plant Base Model NRELAP5 Input,” of this SE.

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 NPM-20, 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 NPM-20 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 NPM-20 and is therefore acceptable.

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

The TR also states that **{{**
}}. The applicant further states that **{{**

}}. The staff finds this acceptable because this modeling approach adequately bounds the physical phenomenon of the heat transfer during DHRS operation.

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

}}

{{

}}.

Staff interrogation of NPM-20 NRELAP5 simulation results, focusing on DHRS condenser surface temperature at the top nodes {{

}}, staff requested additional information (ML25014A157 Non-Public, ML25014A156 Public) to support the use of the highly simplified DHRS-to-pool heat transfer modelling. Based on its review of the applicant's sensitivity analyses and the staff's own modified simulations, the staff finds that the applicant's consideration of the effect of pool temperature and thermal stratification on the performance of the DHRS is acceptable; considering the similarities of DHRS design between NPM-160 and NPM-20, staff does not find a reason to re-evaluate the magnitude of scalar heat flux argument that was made for the finding on the topic of pool thermal stratification for the staff's review of revision 3 of TR-0516-49416-P-A.

Subsequent to the staff completing the NRELAP5 modeling reviews, NuScale adjusted its DHRS heat transfer modeling to credit {{

}}. NuScale made additional information on this DHRS modeling change available for audit (ML24264A049) and, additionally, NuScale noted that corrections were needed {{ }}. The basis for the corrections were reviewed and concurred upon by staff as part of the concurrent US460 SDAA review, since these corrections are not captured in Rev. 5 of the NRELAP5 basemodel. The NRC staff considers the DHRS modeling adjustments and corrections discussed in this paragraph to be acceptable revisions to the NPM-20 basemodel for the purposes of L/C 7 because they were reviewed and approved by NRC staff.

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

}}

{{ }}. For the US460 SDAA, staff noted NuScale's existing response to "eRAI No. 9466," (ML18128A341) which was submitted during the US600 DCA review. The staff evaluated the findings that were made during its review of TR-0516-49416-P-A, revision 3 and found that they remain valid for the NPM-20 design because the NRELAP5 non-LOCA model for NPM-20 still uses the same correlations, and the code is the same (in terms of relevant algorithm(s) for picking correlations).

In a sensitivity study for a representative loss of ac power event from the US600 DCA review, "eRAI No. 9466" (ML18128A341), the applicant compared the results using the **{{**

}}. There was no difference in the peak RPV pressure, and only a very small variation in the peak SG peak pressure **{{**

}}.

The NRC staff reviewed the sensitivity studies performed for and discussed in the safety evaluation report for TR-0516-49416-P-A, revision 3, and finds that the use of the **{{** **}}** is acceptable, since the **{{** **}}** incorporates the **{{** **}}** and sensitivities for a representative non-LOCA event demonstrate that there is little difference in the peak primary pressure, peak secondary pressure, and transient progression when using the **{{** **}}**. While this decision was made with respect to revision 3 of the non-LOCA EM TR, the staff has no reason to re-evaluate this finding for the review of this revision.

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

}}. The applicant did not describe certain NPM-20 specific features that do not affect the progression of non-LOCA events prior to establishment of long-term cooling, such as the ESB feature of the ECCS (ML24348A050 Public, ML24348A051 Non-Public). The ECCS does not actuate in the timeframe covered by the non-LOCA methodology and ECCS modeling is outside the scope of the non-LOCA evaluation model. Once, or if, ECCS valves open, those events or event phases are analyzed according to the methodology in the LOCA LTR (Reference 1). Long-term behavior following ECCS actuation is analyzed as described in the XPC LTR (Reference 2). NRC staff's assessment of ECCS modeling is discussed in safety evaluation reports on the XPC and LOCA LTRs.

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

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

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

}}. Since the sensitivity of transients to changing pool temperature can be accounted for by running multiple cases, of a given transient, with different reactor pool constant temperature, the NRC staff finds this simplified pool heat up model acceptable. On these bases, the NRC staff finds that the description of the NuScale NRELAP5 Plant Model provided in Revision 5 of TR Section 6.1, “Thermal-Hydraulic Volumes and Heat Structures,” is a sufficient description of the model.

3.6.2 Material Properties

TR Section 6.2, “Material Properties,” discusses the thermal conductivity and volumetric heat capacity associated several materials used in the heat structures. The TR states that the material properties will be amended as the NPM design evolves; the staff notes that, at the time of this review, the non-LOCA basemodel (r5) contains one additional material (316 steel) than listed in the current (Revision 5) of non-LOCA EM. The NRC staff finds this acceptable as the method specifies that the material properties used in the model will reflect the operating plant, and the details in this section also reflect the NPM-20 design. Staff also notes **{{**

}}.

3.6.3 Control and Protection Systems

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

3.7 Non-LOCA Analysis Methodology

3.7.1 General Aspects of Non-LOCA Methodology

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

parameters; the biasing of other analysis parameters; and typical MPS signals and associated analytical limits and time delays.

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

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

}}. During review of Revision 3 of TR-0516-49416-P, the NRC staff found the null transient process used to establish biased NRELAP5 stable, steady-state, initial conditions for non-LOCA transient analyses reasonable and acceptable based on standard industry practice and as confirmed by audit discussions on the bias application methodology (ML19039A090). The staff finds that these conclusions remain valid for Revision 5 of this TR because the methodology for identifying the relevant parameters of the initial conditions remains the same for the NPM-20 design.

TR Section 7.1.2, “Treatment of Plant Controls,” discusses the treatment of normal, non-safety related plant control systems (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 SE.

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

TR Sections 7.1.3.1, “Background,” through 7.1.3.3, “Electrical Systems with Important Loads,” discuss the electric power requirements and supply duration. The applicant states that EDAS provides uninterrupted DC power for 72 hours to essential loads while shedding low importance or non-essential loads. Following loss of normal AC power, EDAS batteries power the ECCS

valves for 24 hours, then these valves open: RVVs first, then, RRVs when RPV-to-CNT pressure difference matches the IAB release threshold. ECCS actuation due to load shedding may be pre-empted by a signal that actuates ECCS 8 hours after any reactor trip in the NPM-20 design. The applicant states that actuating ECCS after 8 or 24 hours is not relevant to the short-term FOM(s) addressed by this report (MCHFR and RCS and SG maximum pressures). The NRC staff agrees and notes that the EM for an inadvertent opening of an ECCS valve is addressed in the LOCA TR, (Reference 1) and discharge of reactor coolant after 24 hours is addressed in the XPC TR (Reference 2).

TR Section 7.1.3.4, "Timing for Loss of Power," discusses the timing for the loss-of-power. The loss of normal AC power is assumed to occur either coincident with the initiation of the event or coincident with turbine trip. The basis for selecting these two times is that the loss of AC power could be the event initiator or be caused as a result of the event. The applicant also notes that the random loss of non-safety related electrical systems are not assumed for the NuScale non-LOCA EM, but the failure of the DC power (normal DC power system (EDNS) and augmented DC power system (EDAS)) are related to the loss of AC power or at the time of the initiating event. The specific electric power assumptions are reviewed as part of a design-specific application of this methodology, such as the US460 SDA. This is reflected in L&C #6 in Section 4, 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 precedents and therefore is acceptable. Components that change state or position in a fluid system are considered active components and are subject to the single failure criteria. The staff notes that a failure of the MPS which causes the opening of ECCS valves is not considered within the scope of the non-LOCA methodology and instead is analyzed using the methodology provided in the LOCA TR (Reference 1). TR Section 3.4 provides a description of the IAB design.TR Section 7.1.4.2, "Consideration of Single Failures," describes the various means used to identify the potential active single failures. Passive electrical failures are also considered consistent with the SECY-94-084 SRM. TR Section 7.1.4.4, "Single Failures to Evaluate," identifies the single active failures considered for the NuScale non-LOCA EM analyses and identifies a passive electrical single failure in the MPS as the failure to signal one ECCS RRV and one RVV to open upon demand. The evaluation of the appropriate active and passive single failures is performed on an event-specific basis as part of the application of this methodology to a specific design, such as the NuScale SDA Chapter 15 review.

The IAB valve is a first-of-a-kind, safety-significant, active component integral to the NuScale ECCS. To meet the requirements for the ECCS in 10 CFR Part 50, an applicant must show that it has evaluated the single failure criterion (SFC). The SFC is defined in 10 CFR Part 50, Appendix K and derived from the definition of single failure in 10 CFR Part 50, Appendix A. During its review of the NPM-160 design, the NRC staff noted that although the applicant assumed a single failure of a main ECCS valve to open, the applicant did not apply the SFC to the IAB valve regarding the valve's function to close. Because the NPM-20 design incorporates IAB valves, although only on the RRVs and with modified release thresholds, the staff determined the following information regarding the decision on the application of the SFC to the IAB valves for the NPM-160 design also applies to the IABs present in the NPM-20 design.

For the NPM-160 design, NuScale disagreed with the NRC staff's application of the SFC to the

IAB valve, which led the NRC staff to request the Commission's direction to resolve this issue, SECY-19-0036, "Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves." [3] In SECY-19-0036, the NRC staff summarized the NRC's historical practice for applying the SFC. Specifically, the NRC staff summarized SECY-77-439, [4] in which it informed the Commission of how the NRC staff then generally applied the SFC, and, SECY-94-084, [5] in which the NRC staff requested the Commission's direction on the application of the SFC in specified fact- or application-specific circumstances. In view of this historical practice, the NRC staff in SECY-19-0036, requested the Commission's direction on the application of the SFC to the IAB valve's function to close.

In response to the paper, The Commission directed the NRC staff in SRM-SECY-19-0036, "Staff Requirements - SECY-19-0036 - Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," [6] to "review Chapter 15 of the NuScale Design Certification Application without assuming a single active failure of the inadvertent actuation block valve to close." The Commission further stated that "[t]his approach is consistent with the Commission's safety goal policy and associated core damage and large release frequency goals and existing Commission direction on the use of risk-informed decision-making, as articulated in the 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities and the White Paper on Risk-Informed and Performance-Based Regulation (in SRM SECY- 98-0144 and Yellow Announcement 99-019)."

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

Specific non-LOCA event limiting single failures are evaluated as part of a design-specific application of this methodology, such as the NuScale SDA. This is reflected in L&C #6, in Section 4 of this SE.

TR Section 7.1.5, "Bounding Reactivity Parameters," discusses the use of bounding reactivity parameters in non-LOCA analyses. Section 7.1.5.1, "Moderator Temperature Coefficient," discusses the moderator temperature coefficient (MTC) and provides example values. Section 7.1.5.2, "Doppler Temperature Coefficient," discusses the Doppler temperature coefficient and provides example values. The use of low and high multipliers on the decay heat contribution and inclusion or exclusion of the actinide contribution is discussed in TR Section 7.1.5.3, "Decay Heat Contribution." TR Figure 7-1, "Example of decay heat

[3] See SECY-19-0036, "Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," (April 11, 2019) (Agencywide Documents Access and Management System (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19060A081).
[4] See SECY-77-439, "Single Failure Criterion," (August 17, 1977) (ML060260236).
[5] SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs (March 28, 1994) (ADAMS Accession No. ML003708068), and associated SRM (June 30, 1994) (ML003708098).
[6] See SRM-SECY-19-0036, "SECY-19-0036 Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," (July 2, 2019) (ML19183A408).

comparisons," shows that use of the multipliers and inclusion or exclusion of actinide contribution conservatively bounds the best- estimate decay heat calculated using the ORIGEN code for a generic equilibrium cycle. TR Section 7.1.5.3 states that a review of the applicable core physics parameters will be performed for each cycle to confirm that the multipliers remain bounding. The NRC staff finds the applicant's use of the 1973 ANS decay heat standard with appropriate multipliers and actinide contribution confirmed on a cycle-by-cycle basis acceptable because it ensures that the values used in the analyses remain bounding.

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

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

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

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

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

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

}} (ML24348A070 Public, ML24348A071 Non-Public).

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

Section 7.1.6.3, "Analytical Limits and Response Times," of the TR discusses analytical limits

and response times modeled in the NuScale non-LOCA transient analyses. Table 7-3 of the TR provides examples of analytical limits and actuation delays. While many of these functions are comparable to protection system actuation functions in traditional large PWRs, some functions, such as high or low steam superheat, are specific to the NPM design. The staff reviewed the parameters and finds that the information is sufficient from the perspective of an evaluation of the methodology. However, the staff notes that these are examples and the actual values of these parameters are directly associated with a specific NPM design. The methodology requires use of actuation setpoints and delays that are associated with the specific NPM design when the methodology is applied. Changes in the signals, setpoints, and delays used for both reactor trip and ESFAS actuations may alter event-specific transient progressions and limiting bias directions compared to what is shown in the report. This requirement is reflected in L&C #8.

The NRC staff also notes that this topical report was submitted concurrently with the SDAA for the NPM-20 design. Calculations implementing the non-LOCA methodology were audited during this review, as discussed in the following sections. During its review, the staff noted and the applicant confirmed (ML24348A033) that Table 7-3 values are not comparable with the designed operating conditions of the NPM-20. Deviations between the signals, setpoints, and delays used in Table 7-3 should be considered when applying this methodology to designs with different RTS and ESFAS configurations.

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

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

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

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

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

Section 7.1.7 of the TR also indicates that operator action is not credited in the non-LOCA evaluation model. The determination for the need of operator actions to mitigate specific non-LOCA events is to be evaluated as part of a design-specific application of this methodology, such as the NuScale SDA. This is reflected in L&C #6 in Section 4 of this evaluation.

The NRC staff finds that use of the non-safety-related feedwater regulating and check valves acceptable as a backup to safety-related components because it is consistent with NUREG-0138, Issue 1. The use of the secondary MSIV is an extension of NUREG-0138, Issue 1 as it deals with maintaining primary side inventory. The NRC staff finds this acceptable subject to L&C #5 in Section 4 of this evaluation, which requires an applicant or licensee using this

methodology and seeking to credit the non-safety-related MSIV in the analysis of a SGTF event to receive specific approval to credit the non-safety-related MSIV through the design review.

3.7.2 Event-Specific Methodology

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

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

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

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

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

}}.

The applicant provided sample sensitivity studies for each event as discussed below. The NRC staff notes that these sensitivity studies were not performed based on the NPM-20 design,

rather they were provided to illustrate how future TR users would perform sensitivity studies when the methodology is implemented. As the US460 SDAA (utilizing the NPM-20) was submitted for NRC review concurrently with revision 4 of TR-0516-49416, the NRC staff considered SDAA calculations for non-LOCA events to be representative of the non-LOCA evaluation model and noted oscillatory behavior following DHRS actuation that is consistent with the applicant's description.

DHRS is credited to achieve transition to a safe and stable condition, but DHRS performance can be challenged in certain ranges of SG-DHRS loop inventory. NRC staff requested that NuScale provide a bounding SG-DHRS loop inventory analysis s to ensure DHRS has sufficient heat removal capacity. NuScale's response stated that events where DHRS loop inventory can be negatively impacted are analyzed to ensure that adequate DHRS heat removal capacity can be demonstrated. NuScale analyzed (ML25035A247 Public, ML25035A248 Non-Public) limiting events that could potentially affect SG-DHRS loop inventory, e.g. the increase of feedwater flow transient, which could potentially result in steam generator overfill, and main steam line break events. The non-LOCA LTR methodology for the increase in feedwater flow transient requires analysis of single failures, isolation times, feedwater pump responses, and other initial conditions that maximize SG-DHRS inventory and challenge DHRS performance. Steam line break and feedwater line break events require analysis of break sizes and locations which can drain one train of DHRS. Additionally, as described in TR Section 7.2, demonstration of stable or downward trends in RCS and DHRS pressures is an acceptance criterion for all non-LOCA events. Because the methodology requires demonstration of adequate DHRS performance for all non-LOCA events, and because it includes methods to identify limiting DHRS inventory for events that NuScale identified as limiting for the NPM-20 design, the NRC staff finds this acceptable.

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

3.7.2.1 Decrease in Feedwater Temperature

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

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

The methodology assumes that the initial feedwater temperature starts at the full-power feedwater temperature and decreases to the coldest temperature in the secondary side. The TR states that sensitivity studies are performed to identify the limiting condition, including the transition between RPS trip setpoints. 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 power-related reactor trips are generated based on an effective power which accounts for the decalibration of the ex-core neutron detectors as downcomer density changes. This method effectively increases the high power analytical limit, and delays generation of the high power rate trip. The adjustment to the effective power is to be based on **{{**

}}. The applicant provided sensitivity studies that show how use of this effective power conservatively delays the high power rate trip. The studies also show that use of a larger adjustment results in a longer delay of the rate trip, provided that the adjustment itself does not generate a power rate trip. To ensure this does not occur, the TR requires **{{**

}}. While the specific adjustment 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 adjustment to account for the decalibration effect on credited power-related reactor trips necessary to ensure a conservative calculation of the 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 states that peak primary and secondary pressures are bounded by undercooling events discussed in other parts of the TR, and therefore, sensitivities to maximize these parameters are not analyzed as part of the decrease in feedwater temperature event. The NRC staff agrees that primary and secondary peak pressures are bounded by other events and that MCHFR is the principal FOM for the decrease in feedwater temperature event.

The NRC staff reviewed the initial conditions, biases, and conservatisms in TR Table 7-7, "Initial conditions, biases, and conservatisms – decrease in feedwater temperature," including the PCS operating function assumptions. For the parameters that are not varied as part of each application of the methodology (i.e., parameters whose bias directions are specified in the TR), the NRC staff confirmed during its review of revision 3 of TR-0516-49416-P-A that the bias directions are appropriately conservative or otherwise appropriate. For example, the initial bias directions for reactor power and initial RCS average temperature are conservative because these biases are consistent with known directions of conservatism for MCHFR (ML19067A256). NRC staff confirmed that these bias directions remain appropriate for the NPM-20 design. In addition, the EOC MTC bias provides the largest reactivity change during cooling and minimizes MCHFR. Some parameters **{{** **}}** are set to a nominal initial value, which is acceptable because a conservative bias direction does not exist for these parameters.

TR Table 7-7 indicates that the initial PZR water level is varied **{{**
}}. The TR methodology also requires variation of the initial RCS pressure and operation of the PZR spray **{{**
}}.

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

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

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

The applicant presented sample results from sensitivity studies in TR Tables 7-8, "Representative fuel exposure study," through 7-11, "Representative boundary condition type / single active failure studies," to demonstrate the effects of fuel exposure, initial fuel temperature, boundary condition type, and the single active failure of an MSIV to isolate. While the results are not applicable to the NPM-20 design, sensitivity studies on boundary condition type and single active failure illustrate how parameters will be varied to perform the sensitivity studies specified in TR Table 7-7, "Initial conditions, biases, and conservatisms." (Public: ML25009A002, Non-public: ML25009A003)

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

}}. While these results are not applicable to the NPM-20 design, they illustrate how the methodology requires variation of the rate of feedwater temperature change in order to identify transitions between trip signals, as limiting MCHFR may occur at these points (Public: ML25009A002, Non-public: ML25009A003).

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

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

3.7.2.2 Increase in Feedwater Flow

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

The event is initiated by a malfunction that increases the feedwater flow rate. Like the decrease in feedwater temperature event, the overcooling of the RCS decreases the core inlet temperature and increases the core power. Reactor trip may result from various MPS signals (e.g., high power, low steam line superheat, high PZR pressure, or high steam line pressure).

Secondary system isolation ends the overcooling event, and DHRS provides decay heat removal. Additionally, increase in feedwater flow transients increase secondary side inventory. After DHRS actuation, a larger secondary side inventory may challenge DHRS performance because a higher collapsed liquid level in the DHRS condensers reduces the surface area available for steam condensation.

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

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

}}. The methodology specifies that sensitivity studies to determine **{{**

}}, as well as limiting bias directions for certain parameters, should be performed.

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

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

TR Table 7-15 provides initial conditions, biases, and conservatisms used in cases that challenge DHRS performance for increase in feedwater temperature. The NRC staff reviewed the initial condition biases and the rationale provided that these conditions will maximize secondary side inventory prior to DHRS actuation. Additionally, NRC staff audited “Increase in feedwater flow” calculations provided with the US460 SDAA, which was submitted concurrently with this topical report. The case resulting in the highest SG level assumes a single failure that prevents closure of a FWIV. In this case, flow is terminated when pressure on the secondary side exceeds the shutoff head of the feedwater pumps. The methodology requires sensitivity studies **{{**

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}}. The methodology requires extensive sensitivity studies on other parameters, including a range of initial power levels, to confirm that the selection of initial condition biases maximizes SG level. Therefore, the NRC staff finds the initial conditions, biases, and conservatisms for these cases acceptable**.**

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

}} (Public: ML25009A002, Non-public: ML25009A003). These sensitivity studies will be repeated when the methodology is implemented, as described in TR Table 7-14.

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

3.7.2.3 Increase in Steam Flow

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

An increase in main steam flow causes an increase in heat transfer from the primary to the secondary, a decrease in RCS temperature, an increase in core power and heat flux, and a decrease in RCS and SG pressures. The decreasing RCS temperature causes the rod control system to withdraw the regulating bank. Reactor trip may occur on high power, high riser temperature, or low steam pressure signals. Based on the information presented in Table 7-17 indicating that fuel centerline temperature is not challenged by this event, the NRC staff considers MCHFR to be the primary acceptance criterion of interest for this event. and a challenge to fuel centerline temperature would be bounded by the rod withdrawal event. Should this criteria not be bounded when applying this method, then the initial conditions, biases and conservatisms provided in Table 7-19 may no longer be sufficient.

The TR states that the limiting MCHFR typically 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 TR describes how the effect of changing downcomer temperature on ex-core detector signals is accounted for when assessing the high power trip and high power rate trip. The method is the same as that described in Section 7.2.1, “Decrease in Feedwater Temperature,” of the TR. While the effect of downcomer coolant temperature changes on the ex-core detector response is not event-specific, modifications to these setpoints may only be valid for certain event progressions. However, because the event timing and trip signals are similar between the

increase in steam flow and decrease in feedwater temperature events, the NRC staff finds that application of these trip modifications to the increase in steam flow event is also acceptable. 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 sensitivity study results presented in Tables 7-20, “Representative steam flow study – nominal steam generator heat transfer,” and 7-21, “Representative steam flow study – steam generator heat transfer biased low” of the TR illustrate how a user of the methodology could identify the limiting steam flow increase. While the results are not applicable to the NPM-20 design, they would indicate that **{{**
}} (Public: ML25009A002, Non-public: ML25009A003)**.** These sensitivity studies will be repeated when the methodology is implemented per TR Table 7-19.

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

3.7.2.4 Steam System Piping Failure Inside or Outside of Containment

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

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

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

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

density from the cooldown is accounted for by evaluating the high power trip and high power rate trip based on an effective power level. The method is the same as that described in Section 7.2.1, “Decrease in Feedwater Temperature,” of the TR. While the effect of downcomer coolant temperature changes on the ex-core detector response is not event-specific, modifications to these setpoints may only be valid for certain event progressions. However, because the event timing and trip signals are similar between the steam line break and decrease in feedwater temperature events, the NRC staff finds that application of these trip modifications to the steam line break event is also acceptable.

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

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

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

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

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

TR Table 7-25, “Representative steam line break study,” presents results for example sensitivity studies for the steam line break in terms of MCHFR. While the results are not applicable to the NPM-20 design, the example studies illustrate how a user of the methodology should identify the parameter biases and assumptions that provide a bounding transient simulation. (Public: ML25009A002, Non-public: ML25009A003)

Based on the information submitted by the applicant, as confirmed by the NRC staff's review, and audit (ML24262A257), the NRC staff finds that the applicant's methodology for this event is acceptable because it 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 Containment Vacuum," discusses the containment flooding or loss of containment vacuum event, which is unique to the NPM.

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

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

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

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

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

3.7.2.6 Turbine Trip / Loss of External Load

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

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

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

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

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

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

additional sensitivity studies would be performed on parameters beyond those listed as “varied” in Table 7-32, “Initial conditions, biases, and conservatisms - turbine trip / loss of external load,” although the studies may involve simultaneous variation of additional parameters to those listed as “varied.” (Public: ML25009A002, Non-public: ML25009A003)

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

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

3.7.2.7 Loss of Condenser Vacuum

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

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

The NRC staff audited the applicant’s analyses for the turbine trip / loss of external load events and the loss of condenser vacuum event that were submitted with the US460 SDAA, as these calculations were performed using the TR EM and were submitted concurrently with this TR. The NRC staff considered the initial conditions regarding assumed loss of feedwater flow at event initiation. The loss of condenser vacuum event results in a loss of feedwater flow at event initiation, but the turbine trip / loss of external load does not. However, the applicant also assumes that normal AC power is lost at event initiation for the turbine trip / loss of external load events, which results in the same initiation of loss of feedwater flow occurring at event initiation. For these reasons, the NRC staff finds that the scenarios analyzed as part of Section 7.2.6 address the loss of condenser vacuum event. In addition, the staff agrees that the relevant acceptance criteria, SAF, and LOP scenarios from Table 7-30 are also applicable to the loss of

condenser vacuum event.

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

TR Section 7.2.7.3, “Biases, conservatisms, and sensitivity studies,” states that the biases and conservatisms presented in Table 7-32 for the turbine trip / loss of external load events are applicable to the loss of condenser vacuum event. Revision 5 of the TR does not include a separate table for the initial conditions, biases, and conservatisms for the LOCV transient. The staff finds this approach to be acceptable because the LOCV event initiation, progression, and consequences are similar to the turbine trip / loss of external load event as discussed above.

L&C #6, described in Section 4 of this SE, requires an applicant or licensee seeking to apply this methodology to a design to receive separate approval through that design review for the event-specific electrical power assumptions. This review should confirm that NRC staff findings concerning the similarity of the turbine trip / loss of external load event and the loss of condenser vacuum event remain valid.

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

3.7.2.8 Main Steam Isolation Valve(s) Closure

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

The NRC staff reviewed the applicant’s methodology for this event to determine whether it specified appropriate biases for the applicable parameters and whether the methodology would ensure conservative results when implemented. Table 7-40, “Initial conditions, biases, and conservatisms – main steam isolation valve closure,” lists the initial conditions, biases and conservatisms. Initial RCS flow rate and PZR level are additional parameters varied for the MSIV closure event over those for the turbine trip/loss of external load and LOCV events, which is appropriate given the difference in proximity of the MSIVs to the NPM compared to the turbine stop/closure valves. For the initial conditions whose bias directions are specified, the NRC staff confirmed that the bias directions are limiting for this event . 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. NRC staff notes that the turbine throttle valves are downstream of the MSIVs, and operation of the turbine throttle valves will not influence system response when all MISVs close at event initiation. Additionally, the EM for MSIV closure event considers loss of power scenarios in which turbine throttle control and feedwater pumps will be disabled. Because of this and the other conservatisms in the MSIV closure EM discussed above, the NRC staff finds that these control system related assumptions for the MSIV closure event are acceptable.

The applicant provided results of example sensitivity studies in Table 7-41, “Representative sensitivity studies – main steam isolation valve closure.” The results are not applicable to the NPM-20 design, but the example sensitivity studies illustrate how a user of the methodology could vary parameters in order to identify the appropriate limiting initial conditions (ML24348A072). 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, indicating that only a subset of sensitivity studies are presented. The NRC staff finds the representative sensitivity studies to be acceptable to demonstrate the general process to determine limiting biases and assumptions because they illustrate that maximum RCS pressure is similar between cases in which the RSV actuation limits the RCS pressurization (Public: ML25009A002, Non-public: ML25009A003).

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

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

3.7.2.9 Loss of Nonemergency AC Power

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

The loss of normal nonemergency AC power means a loss of power from either the high voltage (EHVS), medium voltage (EMVS) or low voltage (ELVS) AC electrical distribution system. A loss of AC power results in the turbine generator tripping and a loss of pumps on the secondary side, causing an increase in RCS and SG pressure. The NRC staff finds that the applicant has correctly identified primary and secondary pressures as the acceptance criteria of interest for

this event as provided in Table 7-42, “Acceptance criteria, single active failure, loss of power scenarios – loss of normal AC power.” The applicant states that a reactor trip and DHRS actuation (which result from the MPS response to loss of AC power to the battery chargers or other trip signals, depending on the scenario and relevant actuation delays) end the transient and transition the NPM to a safe, stable condition.

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

The TR states that of the various loss of normal power scenarios, the limiting scenario typically does not result in immediate reactor trip or full CRA insertion at event initiation. The NRC staff finds that the analysis assumption of no reactor trip on loss of AC power will tend to be more limiting because this will result in more rapid heating and pressurization of the primary side.

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

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

TR Table 7-45, “Representative sensitivity studies – loss of normal AC power,” provides the results of sensitivity studies for the loss of nonemergency AC power performed on the NPM-160 design (Public: ML25009A002, Non-public: ML25009A003). During review of Revision 3 of TR-0516-49416-P, the NRC staff reviewed supplementary information provided by the applicant (ML18184A589), including the times to reach analytical limits and actuate RTS, DHRS, and CNV isolation, to understand the trends and behavior in TR Table 7-45 and their implications on the biases for the parameters in TR Table 7-44, “Initial conditions, biases, and conservatisms – loss of normal AC power.” Due to the modeled relief capacity of the RSV, the peak primary pressure is nearly invariant for a wide range of differing bias conditions. The one case in which the primary pressure is lower than the rest of the cases results when the combined effect of the initial condition biases delays the RCS pressure rise such that reactor trip occurs before the RSV lift setpoint is reached. Although not shown in TR Table 7-45, nominal biasing of parameters also does not result in reaching the RSV lift setpoint. During review of Chapter 15 calculations supporting the NPM-20 design, the NRC staff also observed that peak RCS pressure was nearly invariant for a wide range of differing initial conditions that approach the RCS pressure limit and result in RSV lift.

Biasing the initial RCS average temperature high tends to result in higher peak SG pressures.

Furthermore, a higher initial SG pressure also tends to result in a higher peak SG pressure, which is due to increasing the initial SG inventory.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review, the NRC staff finds that the applicant's methodology for this event is consistent with DSRS Section 15.2.6 and acceptable.

3.7.2.10 Loss of Normal Feedwater Flow

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

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

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

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

TR Table 7-49, "Representative sensitivity studies – loss of normal feedwater flow," presents results of the sensitivity studies for a loss of normal feedwater flow that were performed in the NPM-160 design (Public: ML25009A002, Non-public: ML25009A003). The NRC staff notes that these results are limited in scope, **{{**

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

The NRC staff gathered additional information from the DCA and SDA Chapter 15.2.7 presented results, which are the relevant sections for the loss of feedwater flow event, The staff reviewed the previous TR sensitivity analyses to ascertain the limiting bias directions for an

application of the loss of feedwater flow event methodology. in addition, the NRC staff reviewed information in the NRC staff's previous audits, as documented in the associated audit report (ML19039A090). Based on the review of the FSAR Section 15.2.7 presented results, the sensitivity results in Revision 5 of the TR, and the previous audit report, the NRC staff confirmed some of the behavior and trends observed in the sensitivity studies supporting the initial conditions, biases, and conservatisms listed in TR Table 7-48.

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

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

3.7.2.11 Inadvertent Decay Heat Removal System Actuation

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

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

Scenario 1: Inadvertent opening of a single DHRS valve at full power conditions or reduced power conditions.

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

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

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

signals are reached and secondary pressure peaks following DHRS actuation.

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

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

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

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

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

3.7.2.12 Feedwater System Pipe Break Inside or Outside of Containment

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

A feedwater line break can occur inside or outside of containment. A feedwater line break inside containment results in a loss of containment vacuum and a high containment pressure signal that actuates a reactor trip, isolates the secondary system and CVCS, and opens the DHRS valves. The SG, DHRS piping, and DHRS condenser for the faulted SG drain through the break into the containment. The non-faulted SG and DHRS loop provide cooling to the RCS via heat transfer to the reactor pool. The response of smaller feedwater line breaks inside containment is similar to the larger feedwater line breaks except that other MPS setpoints, such

as high PZR pressure, may be reached before high containment pressure.

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

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

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

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

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

The applicant states that sensitivity studies are performed for licensing applications to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). (Public: ML25009A002, Non-public: ML25009A003)

TR Table 7-57, "Representative sensitivity studies – feedwater line break," provides the results of example sensitivity studies. While the results are not applicable to the NPM-20 design, the studies help to illustrate how the non-LOCA EM could be applied to identify the limiting conditions for the feedwater line break event. The NRC staff finds that performing the sensitivity studies by varying the feedwater break size, location, single active failures (per TR Section 7.1.4.1), loss of power assumptions (per TR Section 7.1.3.1), and parameters identified in Table 7-56, "Initial conditions, biases, and conservatisms – feedwater line break," to identify the limiting response(s) for the acceptance criteria challenged by the event provides a bounding transient simulation. Based on the information submitted by the applicant, as confirmed by the NRC staff's review, the NRC staff finds that the applicant's methodology for this event will ensure conservative results when implemented.

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

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

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

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

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

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

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

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

TR Table 7-60, “Initial conditions, biases, and conservatisms – uncontrolled control rod bank withdrawal from subcritical or low power startup conditions,” lists the initial conditions, biases, and conservatisms for the uncontrolled control rod bank withdrawal from subcritical or low power startup conditions. The NRC staff confirmed 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 the NRC staff finds is acceptable given that the parameters typically vary as a function of power below a certain power level. In addition, the NRC staff does not expect the parameters set to nominal values to significantly impact MCHFR or fuel centerline temperature due to the low initial power level. The NRC staff also notes that BOC conditions, including the most positive MTC, are appropriate for this event because they minimize negative reactivity feedback as moderator temperatures increase.

TR Table 7-61, "Representative sensitivity studies – uncontrolled control rod bank withdrawal from subcritical or low power startup conditions," provides the results of example sensitivity studies. While the results are not applicable to the NPM-20 design, the example studies help to illustrate how the non-LOCA EM could be applied to identify the limiting conditions for the subcritical or low power control rod withdrawal cases (Public: ML25009A002, Non-public: ML25009A003).

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

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

3.7.2.14 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

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

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

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

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

}}. The NRC staff notes that PZR spray and PZR 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 the method acceptable as it varies PZR pressure control and PZR level control to ensure the most conservative situation is analyzed. Operation of PZR pressure control or PZR level control is conservative if **{{**

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TR Table 7-65, “Representative sensitivity studies – uncontrolled control rod bank withdrawal at power,” provides the results of example sensitivity studies. While the results are not applicable to the NPM-20 design, they help to illustrate how the non-LOCA EM could be applied to identify the limiting conditions for the control rod withdrawal cases at power (Public: ML25009A002, Non-public: ML25009A003). During its audits, as documented in the audit associated report (ML24262A257), the NRC staff examined calculation documents implementing the topical report methodology in support of the NPM-20 SDAA. The NRC staff confirmed that limiting fuel centerline temperature cases correspond to **{{**

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

}} (Public: ML25009A002, Non-public: ML25009A003) . As such, the NRC staff 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,”:

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

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

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Dropping one control rod assembly adds negative reactivity, reducing the core power. The rod control system would normally attempt to restore the power level but cannot react quickly enough to preclude a reactor trip on high power rate. In some cases a high power-rate trip does not occur, and the reactor eventually returns to the initial power level. The phase of the transient in which the reactor returns to the initial power level is effectively a CRA bank withdrawal with a single CRA fully inserted, which may increase power peaking. However, the total reactivity insertion will be on the order of the negative reactivity insertion of the dropped CRA, and the combination of initial power level and total reactivity insertion are reduced compared to the transients discussed in SER Section 3.7.2.14 as high dropped CRA worth and high initial power cases tend to result in high power-rate trips. Based on the limited total reactivity insertion and potentially higher power peaking, the applicant evaluates rod drops that do not result in immediate rate trip to determine whether they are bounded by single CRA withdrawals.

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

For the condition in which one or more control rod assemblies do not move for a control rod bank demand, referred to as a control rod assembly misalignment, the applicant 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. The NRC staff finds this to be

acceptable because it is consistent with SRP Section 15.4.3. While TR Table 7-66 includes high-level statements about loss of power scenarios, the specific electric power assumptions are reviewed as part of design-specific applications of this methodology, such as the NPM-20 SDA. This is reflected in L&C #6 of Section 4 of this evaluation.

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

TR Table 7-69, “Representative sensitivity studies – control rod misoperation, single control rod assembly withdrawal,” shows the results of sensitivity studies for the single control rod assembly withdrawal event (Public: ML25009A002, Non-public: ML25009A003). While the results of these sensitivity studies are not applicable to the NPM-20 design, they help to illustrate how parameters will be varied to identify a set of potentially limiting single CRA withdrawal cases for subchannel analysis.

TR Table 7-71, “Representative sensitivity studies – control rod misoperation, dropped control rod assemblies,” presents the results of example sensitivity studies. While these results are not applicable to the NPM-20 design, they illustrate how a user of the methodology would identify the limiting bias directions for the rod drop event. These sensitivity studies would be performed for rod drops that are not screened from transient analysis (Public: ML24348A012, Non-Public: ML24348A013). Based on the information provided as part of the NRC staff audit ML24262A257, the NRC staff confirmed that the methodology ensures CRA drops that do not result in an immediate rate trip are bounded by single CRA withdrawal with a method that conservatively accounts for power overshoot and variations in the reactivity insertion rate for bank withdrawal due to automatic control system response compared to the reactivity insertion rates modeled in the single CRA withdrawal analysis. The NRC staff finds that the sensitivity studies provided for audit with the SDA review for the NPM-20 design for the control rod misoperation events adequately demonstrate the process that may be used to screen non-limiting cases from transient evaluation to ensure that CRA drop consequences are bounded by the single CRA withdrawal transient.

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

}}. For the second group, the applicant provides a methodology to confirm that single CRA withdrawal events yield a more limiting MCHFR and LHGR are bounded by analyzed single CRA withdrawal events based on **{{**

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

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

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

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

3.7.2.16 Inadvertent Decrease in Boron Concentration

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

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

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

staff finds the use of these two models for Mode 1 acceptable because they show the two extremes of the reactivity insertion rates.

In Mode 1 at hot full power, the methodology states that the NRELAP5 results for the case with the same initial power and the same reactivity insertion rate is used to determine the time of reactor trip and isolation of the dilution source via closure of the DWS isolation valves. Calculations are performed with the perfect mixing model to determine the remaining available shutdown margin and the time shutdown margin would be lost if the dilution source was not terminated. The NRELAP5 results for the uncontrolled control rod bank withdrawal event can be used to assess the total reactivity insertion in order to calculate remaining shutdown margin as an alternative to performing NRELAP5 analysis for specific boron dilution reactivity insertion rates. For cases where automatic letdown is disabled, the shutdown margin remaining at the time of reactor trip and DWS isolation can be determined from the inventory addition needed to induce a high pressurizer level trip without NRELAP5 analysis (ML25056A424 Non-Public, ML25056A424 Public).

In the Mode 1 hot zero power case, the applicant stated that the NRELAP5 results corresponding to the reactivity insertion rates from both the perfect mixing model and the wave front model are used for the time of reactor trip and isolation of the dilution source via closure of DWS isolation valves. Calculations are performed using the wave front model to determine the remaining available shutdown margin and the time shutdown margin would be lost if the dilution source was not terminated. The NRELAP5 results for the uncontrolled control rod bank withdrawal from subcritical or low power startup conditions event can be used to assess the total reactivity insertion in order to calculate remaining shutdown margin as an alternative to performing NRELAP5 analysis for specific boron dilution reactivity insertion rates. For cases where automatic letdown is disabled, the shutdown margin remaining at the time of reactor trip and DWS isolation can be determined from the inventory addition needed to induce a high pressurizer level trip without NRELAP5 analysis (ML25056A424 Non-Public, ML25056A424 Public).

During Mode 2 (Hot Shutdown) and Mode 3 (Safe Shutdown), the inadvertent decrease in boron concentration case in the NPM depends upon the RCS flow rate. The low RCS flow rate MPS signal is credited to isolate DWS if the RCS flow rate is less than a particular setpoint such as 1.7 ft^3/s (763 gpm). If the RCS flow rate is greater than or equal to that setpoint, the high count-rate signal is credited to isolate the DWS. Calculations are performed using the wave front model to determine the remaining available shutdown margin and the time shutdown margin would be lost if the dilution source was not terminated. The NRC staff finds the use of the wave front model for Modes 2 and 3 acceptable because the delays associated with the wave front model combined with reliance on a count rate trip produce a conservatively large total reactivity insertion, and shutdown margin degradation at the time the DWS isolation valves close. For cases where automatic letdown is disabled, the shutdown margin remaining at the time of DWS or CVCS isolation can be determined from the inventory addition needed to induce a high pressurizer level trip (ML25056A424 Non-Public, ML25056A424 Public).

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, loss of shutdown margin is assessed by comparing the volume of potential sources of diluted or unborated water to the volume of water that could lead to loss of shutdown margin if allowed to enter the pool (ML24062A009). The TR describes two methods of calculating the volume of water that could lead to loss of shutdown margin. For dilution scenarios that can be modeled assuming a constant pool volume, the perfect mixing model (TR

equation 7-1) with an arbitrary flow rate is used, then the arbitrary flow rate is used to calculate a dilution volume rather than a dilution time. The NRC staff notes that this is effectively an algebraic rearrangement of TR equation 7-1 such that selection of the arbitrary flow rate does not affect the calculation result. For dilution scenarios that increase the pool volume, the dilution volume is obtained by calculating the change in boron concentration that would lead to loss of an initial assumed shutdown margin assuming a constant boron worth.

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 TR also includes results for examples of such sensitivity studies performed for the NPM-160 design in TR Tables 7-75, "Representative results – inadvertent decrease in boron concentration in Mode 1 at hot full power," through 7-79, "Representative results – inadvertent decrease in boron concentration in Mode 3."

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

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

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

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

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

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

The NRC staff reviewed the initial conditions, biases, and conservatisms in TR Table 7-82,

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

TR Table 7-83, "Representative sensitivity studies – reactor coolant system inventory increase," provides the results of the example sensitivity studies for the CVCS malfunction that increases inventory. While the results are not applicable to the NPM-20 design, they demonstrate the type of methodology that would be followed to identify the limiting biases for a licensing-basis calculation. The example sensitivity studies indicate that the interplay of the parameter biases may be an important consideration for analyses of this event (Public: ML25009A002, Non-public: ML25009A003). 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 acceptable because it will ensure conservative results when implemented.

3.7.2.18 Failure of Small Lines Outside Containment

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

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

Failure of a spray line or high point vent line outboard of the containment isolation valves is typically less limiting in terms of mass and energy release than a break in a makeup line or letdown line.

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

TR Table 7-85, “Acceptance criteria – breaks in small lines carrying primary coolant outside containment,” discusses the non-LOCA FOMs relative to the failure of a small line outside containment event. The NRC staff finds 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” acceptable because the event postulates that RCS inventory is lost outside containment, and the event does not challenge other non-LOCA acceptance criteria. TR Section 7.2.18.1 notes that bounding values, rather than system transient analysis results, may be used as input to radiological consequence analysis as described in TR Section 4.3.6 and discussed in Section 3.4.3.6 of this SE.

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

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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 results of the applicant’s example sensitivity studies. While the results are not applicable to the NPM-20 design, the studies illustrate how sensitivity studies will be performed to identify the limiting conditions when the methodology is applied. In this example, the largest integrated mass release occurs for the 100 percent break of the letdown line full power plus the heat balance uncertainty with biased-high RCS average temperature and assuming a 100-percent break in the makeup line at the time of reactor trip. The maximum iodine spiking time case also assumes biased-high initial RCS average temperature (Public: ML25009A002, Non-public: ML25009A003).

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

responses are identified.

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

3.7.2.19 Steam Generator Tube Failure

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

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

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

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

TR Table 7-92, "Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure" shows example sensitivity studies performed for the SGTF event. While the results are not applicable to the NPM-20 design, these sensitivity studies illustrate the methodology for identifying conditions that result in limiting integrated mass release and iodine spiking time (Public: ML25009A002, Non-public: ML25009A003).

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

3.8 Representative Calculations

The TR Section 8.0, “Representative Calculations,” includes the results of several representative transient calculations as a demonstration of the analysis methodology based upon the logical framework established in earlier sections. Furthermore, the TR states that the results are limited to demonstrating the application of the non-LOCA methodology to the NPM and the applicant is not seeking approval of these demonstrative calculations. Therefore, the staff did not review Section 8.0 of the TR since these representative calculations are for illustration purposes and based on design features of NPM-160. In addition, since the NPM-20 SDAA was under staff review in parallel with this new revision, and since per L&C #1 the staff is limiting the application of this method to the NPM-20, the staff leveraged the review of those calculations performed for Chapter 15 of the SDAA in determining the that the non-LOCA method can be applied and produces adequate and expected results. As such, this SE does not include evaluation of the sample calculations in Section 8.0 of the TR.

3.9 Quality Assurance

In TR Section 9, “Quality Assurance,” the applicant describes how the NuScale QA TR and the implementing QAP are used to control the activities supporting this TR. The TR states that the QAP complies with the requirements of 10 CFR Part 50, Appendix B and is implemented using the guidance of ASME NQA-1 2008 and NQA-1a-2009 Addenda (Reference 5).

The SRP requires that the EM be maintained under a QAP that meets the requirements of 10 CFR Part 50, Appendix B. The TR references the NuScale QAP which states that NuScale intends to comply with the NRC requirements. Compliance with QA requirements is described in “NuScale Topical Report: Quality Assurance Program Description,” MN-122626-A. The NRC staff reviewed the Quality Assurance Program Description and documented its approval in its SE for that Topical Report (ML25031A372). Further, the NRC staff inspected NuScale’s design control process and code development procedures, and these inspections are documented in the inspection report dated April 12, 2024 (ML24099A129).

Subsequent to the inspection, the staff requested NuScale to confirm that information presented, or conclusions stated within the non-LOCA EM TR are drawn from engineering documents subject to design verification in accordance with the NuScale QAPD, Section 2.3.1, “Design Verification.” This request was based on the NRC staff’s review of the categorization of some of the engineering documents and calculations underlying portions of the non-LOCA EM TR. As part of the NRC staff’s audit, NuScale made available to staff a list of all engineering documents supporting such information or conclusions in the non-LOCA EM TR where NuScale’s classification of these documents did not require design verification in accordance with QAPD, Section 2.3.1. Based, in part, on that list, the NRC staff identified a set of documents for NuScale to confirm the level of design verification that had been performed. NuScale responded (ML25045A239 Non-Public, ML25045A238 Public) and stated that it had confirmed that all but two of the documents had met applicable verification requirements of Appendix B to 10 CFR Part 50 and ASME NQA-1. The NuScale response stated that the confirmation was made through a review of applicable procedural instructions and authentication for each of the records, including the assigned roles of the signatories. For those two documents, one was confirmed to have been subject to software integrity level 3 (safety-related) software controls that require appropriate verification, and the other document was generated under a vendor’s Appendix B to 10 CFR Part 50 quality program and accepted by NuScale. Based on the information in the response to RAI-10297 R1, Question NonLOCA.LTR-55 (ML25045A239 Non-Public, ML25045A238 Public), provided by the applicant, the staff finds that QA controls consistent with RG 1.203 have been implemented for the non-LOCA EM.

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4 LIMITATIONS AND CONDITIONS

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

1. Use of the non-LOCA EM documented in the non-LOCA TR, Revision 5, is limited to evaluations of the NPM-20 design. An applicant or licensee seeking approval to use the non-LOCA EM TR, Revision 5 for a design other than the NPM-20, such as the NPM-160, or another future NPM design, is required to demonstrate the applicability of the non-LOCA EM to the specific NPM design. The use of this methodology for a specific NPM design other than the NPM-20 requires NRC staff review and approval of the applicant's or licensee's determination of applicability. Changes made to an NPM-20 through established change processes are addressed in L&C #7.

2. The staff's approval is limited to the use of the non-LOCA EM with the -A version to TR-0516-49422-P, "Loss-of-Coolant Accident Evaluation Model," Revision 5 (Reference 1). Any future changes or revisions to the -A version to TR-0516-49422-P "Loss-of-Coolant Accident Evaluation Model," Revision 5 (Reference 1), must be assessed by the applicant for their potential impact on the non-LOCA EM. Any subsequent changes to the non-LOCA EM require NRC approval.

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

4. The uncertainty in the model of the DHRS heat transfer has not been quantified and is not approved. An applicant or licensee seeking to apply this methodology to the NPM-20 must evaluate DHRS heat transfer biases to determine if the treatment of uncertainty is justified based on margins to non-LOCA FOMs affected by DHRS performance. Subsequent changes made to the NPM-20 also require this evaluation, except for changes to SSC physical/process input parameters only via established change control processes (such as 10 CFR 50.59) not otherwise requiring NRC approval. Future changes to the non-LOCA LTR methodology must include an evaluation of the uncertainty in DHRS heat transfer if the changes appreciably reduce the margin to the non-LOCA FOMs affected by DHRS performance. An applicant or licensee seeking to apply this methodology to a design other than the NPM-20 must evaluate whether the NRELAP5 prediction of DHRS heat transfer remains applicable or if additional biases are warranted, considering margins to non-LOCA FOMs affected by DHRS performance.

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

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

7. Unless changes are made pursuant to a change control process specifically approved by the NRC staff for changes to NRELAP5 and the NPM model, use of NRELAP5 is limited to Version 1.7 (v1.7) in conjunction with NPM-20 basemodel Revision 5 (or later NRELAP5 versions and/or new NPM-20 basemodel revisions if the changes are demonstrated to produce either essentially the same or conservative results and are consistent with the approved methodology, or if the revision is to the basemodel and due to a change made to SSC physical/process input parameters only made via established change control processes (such as 10 CFR 50.59)). Option 134 of the NRELAP5 code, i.e. use of combination of ESDU and Dittus-Boelter correlations is not approved for use with this revision.

   When NRELAP5 v1.7 and NPM-20 basemodel Revision 5 are referenced in other EMs, those applications for use of NRELAP5 v1.7 and NPM-20 basemodel Revision 5 within another EM require separate approvals to ensure the models and assumptions are defined appropriately for the analyzed FOMs. Use of the NRELAP5 v1.7 and NPM-20 basemodel Revision 5 are therefore approved only for the events listed in Table 4-1 of this TR.

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

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

5 CONCLUSION

The NRC staff reviewed TR-0516-49416-P, revision 5, and the applicant's responses to staff RAIs and audited supporting documentation. As a result of this review, in accordance with the applicable NRC regulations documented in Section 2, "Regulatory Criteria," of this SE, and the NRC staff finds that the use of the NRELAP5 code with the non-LOCA analysis methodology described in the TR is appropriate for the non- LOCA safety analyses of the NuScale NPM design.

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

The NRC staff concludes that the non-LOCA methodology, as documented in TR Revision 5, is acceptable for analysis of the non-LOCA events in the NPM-20 design subject to the limitations and conditions stated in Section 4 of this SE.

6 REFERENCES

1. NuScale Power, LLC, “Loss-of-Coolant Accident Evaluation Model,” TR-0516-49422-P, Revision 5, March 27, 2025, ADAMS Accession No. ML25086A182 Public and ML25086A183 Non-Public.

2. NuScale Power, LLC, “Extended Passive Cooling and Reactivity Control Methodology,” TR-124587-P, Revision 1, March 28, 2025, ADAMS Accession No. ML25087A228 Public and ML25087A229 Non-Public.

3. NRC, Final Safety Evaluation of the NuScale Power, LLC Topical Report TR-0516-49422-P, Revision 5 “Loss-of-Coolant Accident Evaluation Model,” ADAMS Package Accession No. ML25098A245 Non-Public and ML25098A246 Public.

4. NRC, Final Safety Evaluation of the NuScale Power, LLC Topical Report TR-124587-P, Revision 1 “Extended Passive Cooling and Reactivity Control Methodology,” ADAMS Package Accession No. ML25098A229 Non-Public and ML25098A228 Public.

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

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10. **{{**

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Section B

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Non-Loss-of-Coolant Accident Analysis Methodology

May 2025

Revision 5

Docket: 52-050

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Table of Contents

Abstract **1**
Executive Summary **2**
1.0 Introduction **6**
1.1 Purpose 6
1.2 Scope 6
1.3 Abbreviations 8
2.0 Background **13**
2.1 Non-LOCA Evaluation Model Roadmap 13
2.2 Regulatory Requirements 18
3.0 Plant Design Overview **21**
3.1 Description of NuScale Plant 21
3.2 Plant Operation 23
3.3 Decay Heat Removal System 24
3.4 Emergency Core Cooling System 24
3.5 Other Important Systems and Functions 25
4.0 Transient and Accident Analysis Overview **28**
4.1 Design-Basis Events and Event Classification 28
4.2 Design Basis Event Acceptance Criteria 31
4.3 Non-LOCA Transient Analysis Process 36
4.3.1 Develop a Plant Base Model NRELAP5 Input 36
4.3.2 Adapt Plant Base Model NRELAP5 Input for Event Transient Analysis 41
4.3.3 Perform NRELAP5 Steady State and Transient System Analysis Calculations 41
4.3.4 Evaluate Results of Transient Analysis Calculations 41
4.3.5 Identification of Cases for Subchannel Analysis and Extraction of Boundary Condition Data 42
4.3.6 Identification of Cases for Accident Radiological Analysis 46
5.0 NRELAP5 Applicability for Non-LOCA Transient Analysis **49**
5.1 Non-LOCA Phenomena Identification and Ranking Table and Evaluation of High-Ranked Phenomena 49
5.1.1 Phenomena Identification and Ranking Table Process 49
5.1.2 Non-LOCA Event Scenarios and Phases 50

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Table of Contents

5.1.3 Phenomena Identification and Ranking Table Figures-of-Merit and Phenomenon Ranking . . . 51
5.1.4 Highly Ranked Phenomena . . . 53
5.2 Evaluation of Non-LOCA Phenomena Identification and Ranking Table High-Ranked Phenomena . . . 114
5.3 NRELAP5 Validation and Assessments for Non-LOCA . . . 114
5.3.1 KAIST . . . 116
5.3.2 NIST-1 Decay Heat Removal System Separate Effects Tests . . . 121
5.3.3 NIST-1 Non-LOCA Integral Test . . . 193
5.3.4 Code to Code Benchmark for Integral Assessment of Reactivity Event Response . . . 317
5.3.5 Steam Generator Modeling . . . 339
5.3.6 Heat Transfer Correlation Comparison . . . 349
5.3.7 NIST-2 Steam Generator - Decay Heat Removal System Integral Effects Tests . . . 357
5.4 Conclusions of NRELAP5 Applicability for Non-LOCA . . . 472
6.0 NuScale NRELAP5 Plant Model . . . 477
6.1 Thermal-Hydraulic Volumes and Heat Structures . . . 477
6.1.1 Reactor Primary . . . 481
6.1.2 Core Kinetics . . . 491
6.1.3 Fuel Rod Design Input . . . 491
6.1.4 Secondary System . . . 492
6.1.5 Decay Heat Removal System . . . 496
6.1.6 Emergency Core Cooling System . . . 498
6.1.7 Containment Vessel . . . 498
6.1.8 Reactor Cooling Pool . . . 501
6.2 Material Properties . . . 501
6.3 Control Systems . . . 501
6.3.1 Module Control System (Nonsafety-related) . . . 502
6.3.2 Module Protection System (Safety-related) . . . 504
7.0 Non-LOCA Analysis Methodology . . . 507
7.1 General . . . 507
7.1.1 Achieving Steady State Conditions . . . 507

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Table of Contents

7.1.2 Treatment of Plant Controls 510
7.1.3 Loss of Power Conditions. 512
7.1.4 Single Failures 515
7.1.5 Bounding Reactivity Parameters 518
7.1.6 Biasing of Other Parameters 521
7.1.7 Credit for Nonsafety-related Components or Operator Actions 527
7.2 Event Specific Methodology. 528
7.2.1 Decrease in Feedwater Temperature. 530
7.2.2 Increase in Feedwater Flow 539
7.2.3 Increase in Steam Flow 549
7.2.4 Steam System Piping Failure Inside or Outside of Containment 556
7.2.5 Containment Flooding / Loss of Containment Vacuum 563
7.2.6 Turbine Trip / Loss of External Load 569
7.2.7 Loss of Condenser Vacuum. 574
7.2.8 Main Steam Isolation Valve(s) Closure 575
7.2.9 Loss of Nonemergency AC Power 580
7.2.10 Loss of Normal Feedwater Flow. 586
7.2.11 Inadvertent Decay Heat Removal System Actuation 590
7.2.12 Feedwater System Pipe Break Inside or Outside Containment 596
7.2.13 Uncontrolled Control Rod Assembly Bank Withdrawal from Subcritical or Low Power Startup Conditions 601
7.2.14 Uncontrolled Control Rod Assembly Bank Withdrawal at Power 608
7.2.15 Control Rod Misoperation. 613
7.2.16 Inadvertent Decrease in Boron Concentration 628
7.2.17 Chemical and Volume Control System Malfunction that Increases Reactor Coolant System Inventory 638
7.2.18 Failure of Small Lines Outside Containment 642
7.2.19 Steam Generator Tube Failure. 651
8.0 Representative Calculations 660
8.1 Cooldown and/or Depressurization of the Reactor Coolant System 660
8.1.1 Decrease in Feedwater Temperature. 660
8.1.2 Increase in Steam Flow 668

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Table of Contents

8.1.3 Main Steam Line Break 676
8.2 Heatup and/or Pressurization of the Reactor Coolant System. 683
8.2.1 Loss of Normal Feedwater Flow. 683
8.2.2 Loss of Normal AC Power 697
8.2.3 Feedwater Line Break 702
8.3 Reactivity Anomaly. 709
8.3.1 Uncontrolled Control Rod Assembly Bank Withdrawal from Subcritical or Low Power Startup Conditions 709
8.3.2 Control Rod Misoperation. 716
8.4 Increase in Reactor Coolant System Inventory 723
8.4.1 Chemical and Volume Control System Malfunction that Increases Reactor Coolant System Inventory 723
8.5 Decrease in Reactor Coolant System Inventory 732
8.5.1 Small Line Break Outside of Containment 732
8.5.2 Steam Generator Tube Failure. 740
9.0 Quality Assurance 749
10.0 Summary and Conclusions. 750
11.0 References 752

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List of Tables

Table 1-1 Abbreviations 8
Table 1-2 Definitions. 11
Table 2-1 Evaluation model development and assessment process steps and associated application in the non-LOCA evaluation model 15
Table 4-1 Design basis events for which the non-LOCA system transient analysis is performed, event category, and event classification 30
Table 4-2 Acceptance criteria for anticipated operational occurrences 33
Table 4-3 Acceptance criteria for infrequent events 33
Table 4-4 Acceptance criteria for postulated accidents 35
Table 5-1 Importance rankings. 52
Table 5-2 Knowledge levels 53
Table 5-3 High-ranked phenomena for non-LOCA events 54
Table 5-4 Comparison between NuScale Power Module decay heat removal system and KAIST test section dimensions 118
Table 5-5 Comparison between NuScale Power Module decay heat removal system and KAIST range of operations 118
Table 5-6 Comparison between NuScale Power Module decay heat removal system and KAIST NRELAP5 model nodalization 118
Table 5-7 NIST-1 decay heat removal system separate effects tests for NRELAP5 code validation 135
Table 5-8 Comparison between NPM and NIST-1 decay heat removal NRELAP5 nodalizations 137
Table 5-9 NIST-1 HP-03 test cases 139
Table 5-10 NIST-1 HP04 test ranges 177
Table 5-11 NIST-1 integral effects tests for NRELAP5 code validation 193
Table 5-12 NLT-02a sequence of events 201
Table 5-13 NLT-02b sequence of events 214
Table 5-14 NLT-15p2 sequence of events 286
Table 5-15 Non-LOCA transients helical coil steam generator operating range vs. NRELAP5 validated range 341
Table 5-16 Comparison of correlation to CFD results - CFD model 1 351
Table 5-17 Comparison of correlation to CFD results - CFD model 2 353
Table 5-18 Not Used 354
Table 5-19 π groups in dimensionless control volume balance equations for decay heat removal system 364
Table 5-20 NIST-2 integral effects SG/DHRS testing for NRELAP5 code validation 391

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List of Tables

Table 5-21 NIST-2 Run 1 sequence of events . . . 392
Table 5-22 NIST-2 Run 2 sequence of events . . . 422
Table 5-23 NIST-2 Run 3 sequence of events . . . 448
Table 6-1 Typical reactor coolant system regions and associated NRELAP5 components . . . 481
Table 7-1 Typical list of initial conditions considered . . . 509
Table 7-2 Example of normalized trip worth vs. time after trip . . . 521
Table 7-3 Examples of analytical limits and actuation delays (reactor trip system and engineered safety features actuation system) . . . 527
Table 7-4 Regulatory acceptance criteria . . . 528
Table 7-5 Acceptance criteria, single active failure, loss of power scenarios - decrease in feedwater temperature . . . 532
Table 7-6 Acceptance criteria - decrease in feedwater temperature . . . 532
Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature . . . 533
Table 7-8 Representative fuel exposure study . . . 536
Table 7-9 Representative fuel temperature study . . . 536
Table 7-10 Representative feedwater temperature transient study . . . 537
Table 7-11 Representative boundary condition type / single active failure studies . . . 537
Table 7-12 Acceptance criteria, single active failure, loss of power scenarios - increase in feedwater flow . . . 541
Table 7-13 Acceptance criteria - increase in feedwater flow . . . 541
Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow . . . 542
Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow . . . 546
Table 7-16 Representative increase in feedwater flow study - high and low SG performance with maximum power and minimum RCS flow . . . 549
Table 7-17 Acceptance criteria, single active failure, loss of power scenarios - increase in steam flow . . . 551
Table 7-18 Acceptance criteria - increase in steam flow . . . 551
Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow . . . 553
Table 7-20 Representative steam flow study - nominal steam generator heat transfer . . . 556
Table 7-21 Representative steam flow study - steam generator heat transfer biased low . . . 556

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List of Tables

Table 7-22 Acceptance criteria, single active failure, loss of power scenarios - steam line break 558
Table 7-23 Acceptance criteria - steam line break 558
Table 7-24 Initial conditions, biases, and conservatisms - steam line break 559
Table 7-25 Representative steam line break study 562
Table 7-26 Acceptance criteria, single active failure, loss of power scenarios - containment flooding / loss of containment vacuum 564
Table 7-27 Acceptance criteria - containment flooding / loss of containment vacuum 565
Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum 566
Table 7-29 Representative sensitivity studies - containment flooding / loss of containment vacuum 569
Table 7-30 Acceptance criteria, single active failure, loss of power scenarios - turbine trip / loss of external load 570
Table 7-31 Acceptance criteria - turbine trip / loss of external load 570
Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load. 571
Table 7-33 Representative sensitivity studies - turbine trip / loss of external load 574
Table 7-34 Not Used 575
Table 7-35 Not Used 575
Table 7-36 Not Used 575
Table 7-37 Not Used 575
Table 7-38 Acceptance criteria, single active failure, loss of power scenarios - main steam isolation valve closure 576
Table 7-39 Acceptance criteria - main steam isolation valve closure. 576
Table 7-40 Initial conditions, biases, and conservatisms - main steam isolation valve closure 577
Table 7-41 Representative sensitivity studies - main steam isolation valve closure 580
Table 7-42 Acceptance criteria, single active failure, loss of power scenarios - loss of normal AC power 581
Table 7-43 Acceptance criteria - loss of normal AC power 582
Table 7-44 Initial conditions, biases, and conservatisms - loss of normal AC power 582
Table 7-45 Representative sensitivity studies - loss of normal AC power 585

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List of Tables

Table 7-46 Acceptance criteria, single active failure, loss of power scenarios - loss of normal feedwater flow 586
Table 7-47 Acceptance criteria - loss of normal feedwater flow 586
Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow 587
Table 7-49 Representative sensitivity studies - loss of normal feedwater flow 590
Table 7-50 Acceptance criteria, single active failure, loss of power scenarios - inadvertent decay heat removal system actuation 591
Table 7-51 Acceptance criteria - inadvertent decay heat removal system actuation 592
Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation 592
Table 7-53 Representative sensitivity studies - inadvertent decay heat removal system actuation 595
Table 7-54 Acceptance criteria, single active failure, loss of power scenarios - feedwater line break 597
Table 7-55 Acceptance criteria - feedwater line break 597
Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break 598
Table 7-57 Representative sensitivity studies - feedwater line break 601
Table 7-58 Acceptance criteria, single active failure, loss of power scenarios - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions 603
Table 7-59 Acceptance criteria - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions 604
Table 7-60 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions 604
Table 7-61 Representative sensitivity studies - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions 607
Table 7-62 Acceptance criteria, single active failure, loss of power scenarios - uncontrolled control rod bank withdrawal at power 609
Table 7-63 Acceptance criteria - uncontrolled control rod bank withdrawal at power 609
Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power 610
Table 7-65 Representative sensitivity studies - uncontrolled control rod bank withdrawal at power 613
Table 7-66 Acceptance criteria, single active failure, loss of power scenarios - control rod misoperation 617
Table 7-67 Acceptance criteria - control rod misoperation 621

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List of Tables

Table 7-68 Initial conditions, biases, and conservatisms - control rod misoperation, single control rod assembly withdrawal 622

Table 7-69 Representative sensitivity studies - control rod misoperation, single control rod assembly withdrawal 624

Table 7-70 Initial conditions, biases, and conservatisms - control rod misoperation, dropped control rod assemblies 626

Table 7-71 Representative sensitivity studies - control rod misoperation, dropped control rod assemblies 628

Table 7-72 Acceptance criteria, single active failure, loss of power scenarios - inadvertent decrease in boron concentration 633

Table 7-73 Acceptance criteria - inadvertent decrease in boron concentration 633

Table 7-74 Initial conditions, biases, and conservatisms - inadvertent decrease in boron concentration 634

Table 7-75 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot full power 636

Table 7-76 Representative results - inadvertent decrease in boron concentration in Mode 1 at 25 percent rated thermal power 636

Table 7-77 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot zero power. 636

Table 7-78 Representative results - inadvertent decrease in boron concentration in Mode 2 637

Table 7-79 Representative results - inadvertent decrease in boron concentration in Mode 3 637

Table 7-80 Acceptance criteria, single active failure, loss of power scenarios - reactor coolant system inventory increase 638

Table 7-81 Acceptance criteria - reactor coolant system inventory increase 639

Table 7-82 Initial conditions, biases, and conservatisms - reactor coolant system inventory increase 639

Table 7-83 Representative sensitivity studies - reactor coolant system inventory increase 642

Table 7-84 Acceptance criteria, single active failure, loss of power scenarios - breaks in small lines carrying primary coolant outside containment 645

Table 7-85 Acceptance criteria - breaks in small lines carrying primary coolant outside containment 646

Table 7-86 Initial conditions, biases, and conservatisms - breaks in small lines carrying primary coolant outside containment 647

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List of Tables

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 650

Table 7-88 Representative break, time in life, power, flow, and temperature sensitivity study for iodine spiking time - breaks in small lines carrying primary coolant outside containment. 651

Table 7-89 Acceptance criteria, single active failure, loss of power scenarios - steam generator tube failure 653

Table 7-90 Acceptance criteria - steam generator tube failure. 654

Table 7-91 Initial conditions, biases, and conservatisms - steam generator tube failure 655

Table 7-92 Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure 658

Table 8-1 Decrease in feedwater temperature sequence of events. 662

Table 8-2 Increase in steam flow sequence of events 670

Table 8-3 Main steam line break sequence of events 678

Table 8-4 Loss of normal feedwater flow sequence of events - reactor coolant system pressure case 685

Table 8-5 Loss of normal feedwater flow sequence of events - secondary pressure case 692

Table 8-6 Sequence of events for loss of AC power 698

Table 8-7 Sequence of events for feedwater line break outside containment 704

Table 8-8 Withdrawal of a control rod assembly bank from a low power startup condition sequence of events. 711

Table 8-9 Single rod withdrawal sequence of events - MCHFR case 717

Table 8-10 Increase in reactor coolant system inventory sequence of events. 724

Table 8-11 Sequence of events for small line breaks carrying primary coolant outside containment 734

Table 8-12 Sequence of events for steam generator tube failure 742

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List of Figures

Figure 2-1 Evaluation model development and assessment process 14
Figure 3-1 NuScale Power Module schematic. 22
Figure 4-1 {{ }}$^{2(a),(c)}$ 44
Figure 4-2 {{ }}$^{2(a),(c)}$ 45
Figure 4-3 {{ }}$^{2(a),(c)}$ 46
Figure 5-1 Schematic of KAIST test facility 117
Figure 5-2 Measured vs predicted heat transfer coefficient. 120
Figure 5-2a Measured vs predicted heat transfer coefficient. 120
Figure 5-3 Schematic of NIST-1 integral test facility 122
Figure 5-4 NIST-1 test facility configuration. 123
Figure 5-5 Photograph of the NIST-1 facility 125
Figure 5-6 Reactor pressure vessel thermal-hydraulic regions 127
Figure 5-7 Lower core flow plate 129
Figure 5-8 Full-height decay heat removal condenser 131
Figure 5-9 Scaled decay heat removal heat condensers 132
Figure 5-10 NIST-1 noding diagram for full-height decay heat removal system separate effect tests 138
Figure 5-11 NIST-1 HP-03-01 decay heat removal system enthalpy flow rate code-to-data comparison 142
Figure 5-11a NIST-1 HP-03-01 decay heat removal system power code-to-data comparison. 143
Figure 5-12 NIST-1 HP-03-01 decay heat removal system level code-to-data comparison. 144
Figure 5-12a NIST-1 HP-03-01 decay heat removal system level code-to-data comparison. 145
Figure 5-13 NIST-1 HP-03-01 decay heat removal system internal fluid temperature code-to-data comparison 146
Figure 5-13a NIST-1 HP-03-01 decay heat removal system internal fluid temperature code-to-data comparison 147
Figure 5-14 NIST-1 HP-03-01 cooling pool vessel temperature code-to-data comparison (1 of 2) 148
Figure 5-14a NIST-1 HP-03-01 cooling pool vessel temperature code-to-data comparison (2 of 2) 149

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List of Figures

Figure 5-14b NIST-1 HP-03-01 cooling pool vessel temperature code-to-data comparison . . . 150
Figure 5-15 NIST-1 HP-03-01 cooling pool vessel level code-to-data comparison . . . 151
Figure 5-15a NIST-1 HP-03-01 cooling pool vessel level code-to-data comparison . . . 152
Figure 5-16 NIST-1 HP-03-02c decay heat removal system enthalpy flow rate code-to-data comparison . . . 154
Figure 5-16a NIST-1 HP-03-02c decay heat removal system power code-to-data comparison . . . 155
Figure 5-17 NIST-1 HP-03-02c decay heat removal system level code-to-data comparison . . . 156
Figure 5-17a NIST-1 HP-03-02c decay heat removal system level code-to-data comparison . . . 157
Figure 5-18 NIST-1 HP-03-02c decay heat removal system internal fluid temperature code-to-data comparison . . . 158
Figure 5-18a NIST-1 HP-03-02c decay heat removal system internal fluid temperature code-to-data comparison . . . 159
Figure 5-19 NIST-1 HP-03-02c cooling pool vessel temperature code-to-data comparison (1 of 2) . . . 160
Figure 5-19a NIST-1 HP-03-02c cooling pool vessel temperature code-to-data comparison (2 of 2) . . . 161
Figure 5-19b NIST-1 HP-03-02c cooling pool vessel temperature code-to-data comparison . . . 162
Figure 5-20 NIST-1 HP-03-02c cooling pool vessel level code-to-data comparison . . . 163
Figure 5-20a NIST-1 HP-03-02c cooling pool vessel level code-to-data comparison . . . 164
Figure 5-21 NIST-1 HP-03-03-P1 decay heat removal system enthalpy flow rate code-to-data comparison . . . 166
Figure 5-21a NIST-1 HP-03-03-Part1 decay heat removal system power code-to-data comparison . . . 167
Figure 5-22 NIST-1 HP-03-03-Part1 decay heat removal system level code-to-data comparison . . . 168
Figure 5-22a NIST-1 HP-03-03-Part1 decay heat removal system level code-to-data comparison . . . 169
Figure 5-23 NIST-1 HP-03-03-Part1 decay heat removal system internal fluid temperature code-to-data comparison . . . 170
Figure 5-23a NIST-1 HP-03-03-Part1 decay heat removal system internal fluid temperature code-to-data comparison . . . 171
Figure 5-24 NIST-1 HP-03-03-Part1 cooling pool vessel temperature code-to-data comparison (1 of 2) . . . 172

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List of Figures

Figure 5-24a NIST-1 HP-03-03-Part1 cooling pool vessel temperature code-to-data comparison (2 of 2) . . . 173
Figure 5-24b NIST-1 HP-03-03-Part1 cooling pool vessel temperature code-to-data comparison . . . 174
Figure 5-25 NIST-1 HP-03-03-Part1 cooling pool vessel level code-to-data comparison . . . 175
Figure 5-25a NIST-1 HP-03-03-Part1 cooling pool vessel level code-to-data comparison . . . 176
Figure 5-26 NIST-1 HP-04-02 decay heat removal system energy transfer rate . . . 180
Figure 5-26a NIST-1 HP-04-02 decay heat removal system energy transfer rate . . . 180
Figure 5-27 NIST-1 HP-04-02 decay heat removal system condensate temperature . . . 181
Figure 5-27a NIST-1 HP-04-02 decay heat removal system condensate temperature . . . 181
Figure 5-28 NIST-1 HP-04-02 decay heat removal system internal collapsed level . . . 182
Figure 5-28a NIST-1 HP-04-02 decay heat removal system internal collapsed level . . . 182
Figure 5-29 NIST-1 HP-04-02 cooling pool vessel level . . . 183
Figure 5-29a NIST-1 HP-04-02 cooling pool vessel level . . . 183
Figure 5-30 NIST-1 HP-04-02 mid cooling pool vessel fluid temperatures . . . 184
Figure 5-30a NIST-1 HP-04-02 mid cooling pool vessel fluid temperatures . . . 184
Figure 5-31 NIST-1 HP-04-02 upper cooling pool vessel fluid temperatures . . . 185
Figure 5-31a NIST-1 HP-04-02 upper cooling pool vessel fluid temperatures . . . 185
Figure 5-32 NIST-1 HP-04-03 decay heat removal system energy transfer rate . . . 187
Figure 5-32a NIST-1 HP-04-03 decay heat removal system energy transfer rate . . . 187
Figure 5-33 NIST-1 HP-04-03 decay heat removal system condensate temperature . . . 188
Figure 5-33a NIST-1 HP-04-03 decay heat removal system condensate temperature . . . 188
Figure 5-34 NIST-1 HP-04-03 decay heat removal system internal collapsed level comparison . . . 189
Figure 5-34a NIST-1 HP-04-03 decay heat removal system internal collapsed level comparison . . . 189
Figure 5-35 NIST-1 HP-04-03 cooling pool vessel level comparison . . . 190
Figure 5-35a NIST-1 HP-04-03 cooling pool vessel level comparison . . . 190
Figure 5-36 NIST-1 HP-04-03 mid cooling pool vessel axial temperatures . . . 191
Figure 5-36a NIST-1 HP-04-03 mid cooling pool vessel axial temperatures . . . 191
Figure 5-37 NIST-1 HP-04-03 upper cooling pool vessel axial temperatures . . . 192
Figure 5-37a NIST-1 HP-04-03 upper cooling pool vessel axial temperatures . . . 192
Figure 5-38 NRELAP5 noding diagram for the NIST-1 facility . . . 197

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List of Figures

Figure 5-39 NRELAP5 NIST-1 model secondary system nodalization layout (NLT-2b) . . . 198
Figure 5-39a NRELAP5 NIST-1 model secondary system nodalization layout (NLT-15p2) . . . 199
Figure 5-40 NIST-1 feedwater split headers at the steam generator tube coils connection (at the time of NLT-2b test) . . . 200
Figure 5-41 NLT-02a transient feedwater flow comparison . . . 203
Figure 5-42 NLT-02a transient core heater rod power comparison . . . 203
Figure 5-43 NLT-02a transient pressurizer pressure comparison . . . 204
Figure 5-43a NLT-02a transient pressurizer pressure comparison . . . 204
Figure 5-44 NLT-02a transient riser mass flow rate comparison . . . 205
Figure 5-44a NLT-02a transient riser mass flow rate comparison . . . 205
Figure 5-45 NLT-02a transient pressurizer level comparison . . . 206
Figure 5-45a NLT-02a transient pressurizer level comparison . . . 206
Figure 5-46 NLT-02a transient core inlet temperature . . . 207
Figure 5-46a NLT-02a transient core inlet temperature . . . 207
Figure 5-47 NLT-02a transient combined middle and outer steam generator tube coil differential pressure comparison . . . 208
Figure 5-48 NLT-02a transient inner steam generator tube coil differential pressure comparison . . . 208
Figure 5-49 NLT-02a transient core exit fluid temperature comparison . . . 209
Figure 5-49a NLT-02a transient core exit fluid temperature comparison . . . 209
Figure 5-49b NLT-02a transient riser inlet fluid temperature comparison . . . 210
Figure 5-50 NLT-02a transient pressurizer heater rod power comparison . . . 210
Figure 5-51 NLT-02a transient steam line pressure comparison . . . 211
Figure 5-52 NLT-02a transient steam line mass flow rate comparison . . . 212
Figure 5-52a NLT-02a transient steam line mass flow rate comparison . . . 212
Figure 5-53 NLT-02b phase 1 transient core power comparison . . . 218
Figure 5-54 NLT-02b phase 1 transient pressurizer pressure comparison . . . 219
Figure 5-54a NLT-02b phase 1 transient pressurizer pressure comparison . . . 219
Figure 5-55 NLT-02b phase 1 transient pressurizer level comparison . . . 220
Figure 5-55a NLT-02b phase 1 transient pressurizer level comparison . . . 220
Figure 5-56 Not Used . . . 220
Figure 5-57 NLT-02b phase 1 transient core inlet and outlet temperature comparison . . . 221
Figure 5-57a NLT-02b phase 1 transient core inlet and outlet temperature comparison . . . 221

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List of Figures

Figure 5-58 Not Used . . . 222
Figure 5-59 NLT-02b phase 1 transient steam generator steam pressure comparison . . . 222
Figure 5-59a NLT-02b phase 1 transient steam generator steam pressure comparison . . . 222
Figure 5-60 NLT-02b phase 1 transient steam generator thermal power comparison 223
Figure 5-61 NLT-02b phase 1 transient decay heat removal system heat exchanger thermal power comparison . . . 223
Figure 5-62 NLT-02b phase 1 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization . . . 224
Figure 5-62a NLT-02b phase 1 integrated compensation flow for SG and DHRS HX level equalization . . . 224
Figure 5-63 NLT-02b phase 1 transient steam generator level comparison . . . 225
Figure 5-64 NLT-02b phase 1 transient decay heat removal system heat exchanger level comparison . . . 226
Figure 5-64a NLT-02b phase 1 transient decay heat removal system heat exchanger level comparison . . . 226
Figure 5-65 NLT-02b phase 1 transient decay heat removal system condensate temperature comparison . . . 227
Figure 5-65a NLT-02b phase 1 transient decay heat removal system condensate temperature comparison . . . 227
Figure 5-66 NLT-02b phase 1 transient decay heat removal system condensate flow comparison . . . 228
Figure 5-66a NLT-02b phase 1 transient decay heat removal system condensate flow comparison . . . 228
Figure 5-67 NLT-02b phase 1 transient steam generator outlet temperature comparison . . . 229
Figure 5-68 NLT-02b phase 1 transient cooling pool vessel level comparison . . . 229
Figure 5-69 NLT-02b phase 1 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger) . . . 230
Figure 5-69a NLT-02b phase 1 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger) . . . 230
Figure 5-69b NLT-02b phase 1 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger) . . . 231
Figure 5-70 NLT-02b phase 1 transient cooling pool vessel region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region) . . . 231
Figure 5-71 NLT-02b phase 2 transient core power comparison . . . 234
Figure 5-72 NLT-02b phase 2 transient pressurizer pressure comparison . . . 235

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List of Figures

Figure 5-72a NLT-02b phase 2 transient pressurizer pressure comparison 235
Figure 5-73 NLT-02b phase 2 transient pressurizer level comparison 236
Figure 5-73a NLT-02b phase 2 transient pressurizer level comparison 236
Figure 5-74 Not Used 236
Figure 5-75 NLT-02b phase 2 transient core inlet and outlet temperature comparison . . . 237
Figure 5-75a NLT-02b phase 2 transient core inlet and outlet temperature comparison . . . 237
Figure 5-76 Not Used 237
Figure 5-77 NLT-02b phase 2 transient steam generator steam pressure comparison . . . 238
Figure 5-77a NLT-02b phase 2 transient steam generator steam pressure comparison . . . 238
Figure 5-78 NLT-02b phase 2 transient steam generator thermal power comparison 239
Figure 5-79 NLT-02b phase 2 transient decay heat removal system heat exchanger thermal power comparison 239
Figure 5-80 NLT-02b phase 2 transient steam generator level comparison 240
Figure 5-81 NLT-02b phase 2 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization 241
Figure 5-81a NLT-02b phase 2 integrated compensation flow for SG and DHRS HX level equalization 241
Figure 5-82 NLT-02b phase 2 transient decay heat removal system condensate temperature comparison 242
Figure 5-82a NLT-02b phase 2 transient decay heat removal system condensate temperature comparison 242
Figure 5-83 NLT-02b phase 2 transient decay heat removal system condensate flow comparison 243
Figure 5-83a NLT-02b phase 2 transient decay heat removal system condensate flow comparison 243
Figure 5-84 NLT-02b phase 2 transient steam generator outlet temperature comparison 244
Figure 5-85 NLT-02b phase 2 transient decay heat removal system heat exchanger level comparison 245
Figure 5-85a NLT-02b phase 2 transient decay heat removal system heat exchanger level comparison 245
Figure 5-86 NLT-02b phase 2 transient cooling pool vessel level comparison 246
Figure 5-87 NLT-02b phase 2 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger) 247
Figure 5-87a NLT-02b phase 2 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger) . . . 247

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List of Figures

Figure 5-87b NLT-02b phase 2 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger) . . . 248
Figure 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) . . . 248
Figure 5-89 NLT-02b phase 3 transient core power comparison . . . 252
Figure 5-90 NLT-02b phase 3 transient pressurizer pressure comparison . . . 253
Figure 5-90a NLT-02b phase 3 transient pressurizer pressure comparison . . . 253
Figure 5-91 NLT-02b phase 3 transient pressurizer level comparison . . . 254
Figure 5-91a NLT-02b phase 3 transient pressurizer level comparison . . . 254
Figure 5-92 Not Used . . . 255
Figure 5-93 NLT-02b phase 3 transient core inlet and outlet temperature comparison . . . 255
Figure 5-93a NLT-02b phase 3 transient core inlet and outlet temperature comparison . . . 255
Figure 5-94 Not Used . . . 256
Figure 5-95 NLT-02b phase 3 transient steam generator steam pressure comparison . . . 256
Figure 5-95a NLT-02b phase 3 transient steam generator steam pressure comparison . . . 256
Figure 5-96 NLT-02b phase 3 transient steam generator thermal power comparison . . . 257
Figure 5-97 NLT-02b phase 3 transient decay heat removal system heat exchanger thermal power comparison . . . 257
Figure 5-98 NLT-02b phase 3 transient steam generator level comparison . . . 258
Figure 5-99 NLT-02b phase 3 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization . . . 259
Figure 5-99a NLT-02b phase 3 integrated compensation flow for SG and DHRS HX level equalization . . . 259
Figure 5-100 NLT-02b phase 3 transient decay heat removal system condensate temperature comparison . . . 260
Figure 5-100a NLT-02b phase 3 transient decay heat removal system condensate temperature comparison . . . 260
Figure 5-101 NLT-02b phase 3 transient decay heat removal system condensate flow comparison . . . 261
Figure 5-101a NLT-02b phase 3 transient decay heat removal system condensate flow comparison . . . 261
Figure 5-102 NLT-02b phase 3 transient steam generator outlet temperature comparison . . . 262
Figure 5-103 NLT-02b phase 3 transient decay heat removal system heat exchanger level comparison . . . 263

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List of Figures

Figure 5-103a NLT-02b phase 3 transient decay heat removal system heat exchanger level comparison . . . 263
Figure 5-104 NLT-02b phase 3 transient cooling pool vessel level comparison . . . 264
Figure 5-105 NLT-02b phase 3 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger) . . . 265
Figure 5-105a NLT-02b phase 3 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger) . . . 265
Figure 5-105b NLT-02b phase 3 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger) . . . 266
Figure 5-106 NLT-02b phase 3 transient cooling pool vessel region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region) . . . 266
Figure 5-107 NLT-02b phase 4 transient core power comparison . . . 269
Figure 5-108 NLT-02b phase 4 transient pressurizer pressure comparison . . . 270
Figure 5-108a NLT-02b phase 4 transient pressurizer pressure comparison . . . 270
Figure 5-109 NLT-02b phase 4 transient pressurizer level comparison . . . 271
Figure 5-109a NLT-02b phase 4 transient pressurizer level comparison . . . 271
Figure 5-110 NLT-02b phase 4 transient reactor pressure vessel level . . . 272
Figure 5-111 NLT-02b phase 4 transient reactor pressure vessel upper plenum temperature comparison . . . 272
Figure 5-112 NLT-02b phase 4 transient core inlet and outlet temperature comparison . . . 273
Figure 5-112a NLT-02b phase 4 transient core inlet and outlet temperature comparison . . . 273
Figure 5-113 Not Used . . . 274
Figure 5-114 NLT-02b phase 4 transient steam generator steam pressure comparison . . . 274
Figure 5-114a NLT-02b phase 4 transient steam generator steam pressure comparison . . . 274
Figure 5-115 NLT-02b phase 4 transient steam generator thermal power comparison . . . 275
Figure 5-116 NLT-02b phase 4 transient decay heat removal system heat exchanger thermal power comparison . . . 275
Figure 5-117 NLT-02b phase 4 transient steam generator level comparison . . . 276
Figure 5-118 NLT-02b phase 4 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization . . . 277
Figure 5-118a NLT-02b phase 4 integrated compensation flow for SG and DHRS HX level equalization . . . 277
Figure 5-119 NLT-02b phase 4 transient decay heat removal system condensate temperature comparison . . . 278

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List of Figures

Figure 5-119a NLT-02b phase 4 transient decay heat removal system condensate temperature comparison . . . 278
Figure 5-120 NLT-02b phase 4 transient decay heat removal system condensate flow comparison . . . 279
Figure 5-120a NLT-02b phase 4 transient decay heat removal system condensate flow comparison . . . 279
Figure 5-121 NLT-02b phase 4 transient steam generator outlet temperature comparison . . . 280
Figure 5-122 NLT-02b phase 4 transient decay heat removal system heat exchanger level comparison . . . 281
Figure 5-122a NLT-02b phase 4 transient decay heat removal system heat exchanger level comparison . . . 281
Figure 5-123 NLT-02b phase 4 transient cooling pool vessel level comparison . . . 282
Figure 5-124 NLT-02b phase 4 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger) . . . 282
Figure 5-124a NLT-02b phase 4 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger) . . . 283
Figure 5-124b NLT-02b phase 4 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger) . . . 283
Figure 5-125 NLT-02b phase 4 transient cooling pool vessel region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region) . . . 284
Figure 5-126 NLT-15p2, transient core power . . . 293
Figure 5-127 NLT-15p2, transient RPV pressure short term . . . 294
Figure 5-128 NLT-15p2, transient RPV pressure . . . 295
Figure 5-128a NLT-15p2, transient RPV pressure . . . 295
Figure 5-129 NLT-15p2, transient pressurizer level . . . 296
Figure 5-129a NLT-15p2, transient pressurizer level . . . 296
Figure 5-130 NLT-15p2, transient RPV level . . . 297
Figure 5-130a NLT-15p2, transient RPV level . . . 297
Figure 5-131 NLT-15p2, transient riser mass flow rate . . . 298
Figure 5-132 NLT-15p2, transient core inlet temperature . . . 298
Figure 5-133 NLT-15p2, transient riser inlet temperature . . . 299
Figure 5-133a NLT-15p2, transient core inlet and riser inlet temperatures . . . 299
Figure 5-134 NLT-15p2, transient upper plenum temperature . . . 300

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List of Figures

Figure 5-134a NLT-15p2, transient upper plenum temperature 300
Figure 5-135 NLT-15p2, transient secondary side pressure - 0 to 500 seconds 301
Figure 5-136 NLT-15p2, transient DHRS loop flow - 0 to 500 seconds 301
Figure 5-137 NLT-15p2, transient measured steam line temperatures - 0 to 500 seconds 302
Figure 5-138 NLT-15p2, transient DHRS HX level - 0 to 500 seconds 302
Figure 5-139 NLT-15p2, transient secondary side pressure 303
Figure 5-139a NLT-15p2, transient secondary side pressure 303
Figure 5-140 NLT-15p2, transient DHRS HX inlet temperature 304
Figure 5-140a NLT-15p2, transient DHRS HX inlet temperature 304
Figure 5-141 NLT-15p2, transient DHRS HX outlet temperature 305
Figure 5-142 NLT-15p2, transient DHRS loop flow - short term 305
Figure 5-143 NLT-15p2, transient DHRS loop flow rate - long term 306
Figure 5-143a NLT-15p2, transient DHRS loop flow rate - long term 306
Figure 5-144 NLT-15p2, transient DHRS HX level 307
Figure 5-144a NLT-15p2, transient DHRS HX level 307
Figure 5-145 NLT-15p2, transient steam generator tube coil level - long term 308
Figure 5-146 NLT-15p2, transient steam generator tube coil level - short term 308
Figure 5-147 NLT-15p2, transient DHRS condensate line differential pressure 309
Figure 5-148 NLT-15p2, transient DHRS steam line differential pressure 309
Figure 5-149 NLT-15p2, transient steam generator tube coil power removal 310
Figure 5-149a NLT-15p2, transient steam generator tube coil power removal 310
Figure 5-150 NLT-15p2, transient DHRS power removal 311
Figure 5-150a NLT-15p2, transient DHRS power removal 311
Figure 5-151 NLT-15p2, transient cooling pool vessel level 312
Figure 5-152 NLT-15p2, transient cooling pool vessel fluid temperature at level 3 (below decay heat removal heat exchanger) 312
Figure 5-153 NLT-15p2, transient cooling pool vessel fluid temperature at level 5 (near bottom of DHRS heat exchanger) 313
Figure 5-154 NLT-15p2, transient cooling pool vessel fluid temperature at level 6 (near midpoint of DHRS heat exchanger) 313
Figure 5-155 NLT-15p2, transient cooling pool vessel fluid temperature at level 7 (top to just above DHRS heat exchanger region) 314
Figure 5-156 NLT-15p2, transient cooling pool vessel fluid temperature at level 8 (above DHRS heat exchanger region) 314

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List of Figures

Figure 5-157 NLT-15p2, transient cooling pool vessel fluid temperature at level 9 (above DHRS heat exchanger region) . . . 315
Figure 5-158 Core power (fast uncontrolled rod withdrawal) . . . 321
Figure 5-159 Total reactivity (fast uncontrolled rod withdrawal) . . . 321
Figure 5-160 Pressurizer pressure (fast uncontrolled rod withdrawal) . . . 322
Figure 5-161 Pressurizer level (fast uncontrolled rod withdrawal) . . . 322
Figure 5-162 Core flow (fast uncontrolled rod withdrawal) . . . 323
Figure 5-163 Core inlet temperatures (fast uncontrolled rod withdrawal) . . . 323
Figure 5-164 Core outlet temperatures (fast uncontrolled rod withdrawal) . . . 324
Figure 5-165 Core power (slow uncontrolled rod withdrawal) . . . 326
Figure 5-166 Total reactivity (slow uncontrolled rod withdrawal) . . . 326
Figure 5-167 Pressurizer pressure (slow uncontrolled rod withdrawal) . . . 327
Figure 5-168 Pressurizer level (slow uncontrolled rod withdrawal) . . . 327
Figure 5-169 Core flow (slow uncontrolled rod withdrawal) . . . 328
Figure 5-170 Core inlet temperature (slow uncontrolled rod withdrawal) . . . 328
Figure 5-171 Core outlet temperature (slow uncontrolled rod withdrawal) . . . 329
Figure 5-172 Core power (power reduction) . . . 330
Figure 5-173 Total reactivity (power reduction) . . . 331
Figure 5-174 Pressurizer pressure (power reduction) . . . 331
Figure 5-175 Pressurizer level (power reduction) . . . 332
Figure 5-176 Core flow (power reduction) . . . 332
Figure 5-177 Core inlet temperature (power reduction) . . . 333
Figure 5-178 Core outlet temperature (power reduction) . . . 333
Figure 5-179 Core power (dropped control rod) . . . 335
Figure 5-180 Total reactivity (dropped control rod) . . . 335
Figure 5-181 Pressurizer pressure (dropped control rod) . . . 336
Figure 5-182 Pressurizer level (dropped control rod) . . . 336
Figure 5-183 Core flow (dropped control rod) . . . 337
Figure 5-184 Core inlet temperature (dropped control rod) . . . 337
Figure 5-185 Core outlet temperature (dropped control rod) . . . 338
Figure 5-186 Coil 1 representative pressure drop for {{ }}$^{2(a),(c)}$ nodes (left) and {{ }}$^{2(a),(c)}$ nodes (right) . . . 343
Figure 5-187 Coil 1 representative fluid temperature for {{ }}$^{2(a),(c)}$ nodes (left) and {{ }}$^{2(a),(c)}$ nodes (right) . . . 343

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List of Figures

Figure 5-188 Coil 1 representative wall temperature for {{ }}$^{2(a),(c)}$ nodes (left) and {{ }}$^{2(a),(c)}$ nodes (right) 344

Figure 5-189 Decrease in feedwater temperature nodalization sensitivity steam generator secondary side inlet pressure. 344

Figure 5-190 Decrease in feedwater temperature nodalization sensitivity steam generator secondary side outlet pressure 345

Figure 5-191 Decrease in feedwater temperature nodalization sensitivity reactor coolant system flow rate 346

Figure 5-192 Decrease in feedwater temperature nodalization sensitivity reactor coolant system lower plenum pressure. 347

Figure 5-193 Decrease in feedwater temperature nodalization sensitivity reactor coolant system core inlet temperature 348

Figure 5-194 Decrease in feedwater temperature nodalization sensitivity reactor power . . . 348

Figure 5-195 Comparison of heat transfer correlation options for steam generator primary side heat transfer rate (linear time scale). 355

Figure 5-196 Comparison of heat transfer correlation options for steam generator primary side heat transfer rate (logarithmic time scale) 356

Figure 5-197 NIST-2 decay heat removal system configuration 359

Figure 5-198 Normalized time-averaged π groups for decay heat removal system liquid mass balance in {{ }}$^{2(a),(c),ECI}$ 366

Figure 5-199 Significances vs. distortions of π groups for decay heat removal system liquid mass balance in {{ }}$^{2(a),(c),ECI}$ 367

Figure 5-200 Normalized time-averaged π groups for decay heat removal system energy balance in {{ }}$^{2(a),(c),ECI}$ 368

Figure 5-201 Significances vs. distortions of π groups for decay heat removal system energy balance in {{ }}$^{2(a),(c),ECI}$ 369

Figure 5-202 Normalized time-averaged π groups for decay heat removal system momentum balance in {{ }}$^{2(a),(c),ECI}$ 370

Figure 5-203 Significances vs. distortions of π groups for decay heat removal system momentum balance in {{ }}$^{2(a),(c),ECI}$ 371

Figure 5-204 Reactor coolant system average temperature transient response for loss of feedwater break scenarios. 375

Figure 5-205 Reactor coolant system saturation temperature transient response for loss of feedwater break scenarios 376

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List of Figures

Figure 5-206 Reactor coolant system flow transient response for loss of feedwater break scenarios 377
Figure 5-207 Reactor coolant system pressure transient response for loss of feedwater break scenarios 378
Figure 5-208 Secondary side pressure transient response for loss of feedwater break scenarios 379
Figure 5-209 Decay heat removal system flow transient response for loss of feedwater break scenarios 380
Figure 5-210 Decay heat removal system power transient response for loss of feedwater break scenarios 381
Figure 5-211 Decay heat removal system power compared to reactor coolant system temperature for loss of feedwater break scenarios 382
Figure 5-212 NIST-2 modified separate effects main steam line part 1 model 383
Figure 5-213 NIST-2 modified separate effects main steam line part 2 model 384
Figure 5-214 NIST-2 modified bypass steam line to containment vessel model 385
Figure 5-215 NIST-2 new integral effects decay heat removal system steam line model 386
Figure 5-216 NIST-2 modified separate effects decay heat removal system steam line model 387
Figure 5-217 NIST-2 new decay heat removal system heat exchanger model 388
Figure 5-218 NIST-2 modified decay heat removal system condensate return line model 389
Figure 5-219 NIST-2 modified feedwater system model 390
Figure 5-220 NIST-2 Run 1 core power comparison 397
Figure 5-221 NIST-2 Run 1 steam generator and decay heat removal system active loop inventory comparison 398
Figure 5-222 NIST-2 Run 1 steam generator and decay heat removal system heat exchanger power comparison 399
Figure 5-223 NIST-2 Run 1 steam generator and decay heat removal system heat exchanger power comparison - short-term 400
Figure 5-224 NIST-2 Run 1 steam generator and decay heat removal system loop flow comparison 401
Figure 5-224a NIST-2 Run 1 decay heat removal condensate line flow comparison (without time average) 402
Figure 5-225 NIST-2 Run 1 steam generator and decay heat removal system loop flow comparison - short-term 403
Figure 5-225a NIST-2 Run 1 decay heat removal condensate line flow comparison (without time average) - short-term 404

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List of Figures

Figure 5-226 NIST-2 Run 1 primary and secondary pressure comparison 405
Figure 5-227 NIST-2 Run 1 steam generator and decay heat removal system level comparison 406
Figure 5-228 NIST-2 Run 1 steam drum level comparison 407
Figure 5-229 NIST-2 Run 1 decay heat removal system heat exchanger inlet and outlet temperatures comparison 408
Figure 5-230 NIST-2 Run 1 steam generator secondary side inlet and outlet temperatures comparison 409
Figure 5-231 NIST-2 Run 1 steam generator secondary side saturation temperature comparison 410
Figure 5-232 NIST-2 Run 1 steam generator primary side inlet and outlet temperatures comparison 411
Figure 5-233 NIST-2 Run 1 pressurizer level comparison 412
Figure 5-234 NIST-2 Run 1 reactor pressure vessel level comparison 413
Figure 5-235 NIST-2 Run 1 primary flow rate comparison 414
Figure 5-236 NIST-2 Run 1 cooling pool vessel level comparison 415
Figure 5-237 NIST-2 Run 1 cooling pool vessel level 5 temperature comparison 416
Figure 5-238 NIST-2 Run 1 cooling pool vessel level 6 temperature comparison 417
Figure 5-239 NIST-2 Run 1 decay heat removal system steam line differential pressure comparison 418
Figure 5-240 NIST-2 Run 1 decay heat removal system steam line orifice differential pressure comparison 419
Figure 5-241 NIST-2 Run 1 decay heat removal system heat exchanger tube lower middle section fluid temperatures - data only 420
Figure 5-242 NIST-2 Run 1 decay heat removal system heat exchanger tube bottom fluid temperatures - data only 421
Figure 5-243 NIST-2 Run 2 core power comparison 426
Figure 5-244 NIST-2 Run 2 steam generator and decay heat removal system active loop inventory comparison 427
Figure 5-245 NIST-2 Run 2 steam generator and decay heat removal system heat exchanger power comparison 428
Figure 5-246 NIST-2 Run 2 steam generator and decay heat removal system heat exchanger power comparison - short-term 429
Figure 5-247 NIST-2 Run 2 steam generator and decay heat removal system loop flow comparison 430
Figure 5-248 NIST-2 Run 2 steam generator and decay heat removal system loop flow comparison - short-term 431

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List of Figures

Figure 5-249 NIST-2 Run 2 decay heat removal system condensate line flow comparison (without time average) 432
Figure 5-250 NIST-2 Run 2 decay heat removal system condensate line differential pressure comparison 433
Figure 5-251 NIST-2 Run 2 primary and secondary pressure comparison 434
Figure 5-252 NIST-2 Run 2 steam generator and decay heat removal system level comparison 435
Figure 5-253 NIST-2 Run 2 decay heat removal system heat exchanger inlet and outlet temperatures comparison 436
Figure 5-254 NIST-2 Run 2 steam generator secondary side inlet and outlet temperatures comparison 437
Figure 5-255 NIST-2 Run 2 steam generator secondary side saturation temperature comparison 438
Figure 5-256 NIST-2 Run 2 steam generator primary side inlet and outlet temperatures comparison 439
Figure 5-257 NIST-2 Run 2 pressurizer level comparison 440
Figure 5-258 NIST-2 Run 2 reactor pressure vessel level comparison 441
Figure 5-259 NIST-2 Run 2 primary flow rate comparison 442
Figure 5-260 NIST-2 Run 2 cooling pool vessel level comparison 443
Figure 5-261 NIST-2 Run 2 cooling pool vessel level 5 temperature comparison 444
Figure 5-262 NIST-2 Run 2 cooling pool vessel level 6 temperature comparison 445
Figure 5-263 NIST-2 Run 2 decay heat removal system steam line differential pressure comparison 446
Figure 5-264 NIST-2 Run 2 decay heat removal system steam line orifice differential pressure comparison 447
Figure 5-265 NIST-2 Run 3 core power comparison 451
Figure 5-266 NIST-2 Run 3 steam generator and decay heat removal system active loop inventory comparison 452
Figure 5-267 NIST-2 Run 3 steam generator and decay heat removal system heat exchanger power comparison 453
Figure 5-268 NIST-2 Run 3 steam generator and decay heat removal system heat exchanger power comparison - short-term 454
Figure 5-269 NIST-2 Run 3 steam generator and decay heat removal system loop flow comparison 455
Figure 5-270 NIST-2 Run 3 steam generator and decay heat removal system loop flow comparison - short-term 456

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List of Figures

Figure 5-271 NIST-2 Run 3 decay heat removal system condensate line flow comparison (without time average) 457
Figure 5-272 NIST-2 Run 3 primary and secondary pressure comparison 458
Figure 5-273 NIST-2 Run 3 steam generator and decay heat removal system level comparison 459
Figure 5-274 NIST-2 Run 3 decay heat removal system heat exchanger inlet and outlet temperatures comparison 460
Figure 5-275 NIST-2 Run 3 steam generator secondary side inlet and outlet temperatures comparison 461
Figure 5-276 NIST-2 Run 3 steam generator secondary side saturation temperature comparison 462
Figure 5-277 NIST-2 Run 3 steam generator primary side inlet and outlet temperatures comparison 463
Figure 5-278 NIST-2 Run 3 pressurizer level comparison 464
Figure 5-279 NIST-2 Run 3 reactor pressure vessel level comparison 465
Figure 5-280 NIST-2 Run 3 primary flow rate comparison 466
Figure 5-281 NIST-2 Run 3 cooling pool vessel level comparison 467
Figure 5-282 NIST-2 Run 3 cooling pool vessel level 5 temperature comparison 468
Figure 5-283 NIST-2 Run 3 cooling pool vessel level 6 temperature comparison 469
Figure 5-284 NIST-2 Run 3 decay heat removal system steam line differential pressure comparison 470
Figure 5-285 NIST-2 Run 3 decay heat removal system steam line orifice differential pressure comparison 471
Figure 6-1 NuScale Power Module (typical) 478
Figure 6-2 Typical primary and secondary side model (heat structures and component cell details excluded) 479
Figure 6-3 Typical NRELAP5 plant module volume regions 480
Figure 6-4 Typical reactor pressure vessel downcomer model 482
Figure 6-5 Typical core and lower plenum model 484
Figure 6-6 Reflector / core bypass without fuel assemblies (for illustration only) 485
Figure 6-7 Lower riser region, immediately above the core (for illustration only) 486
Figure 6-8 Typical reactor pressure vessel core and lower riser model 487
Figure 6-9 Typical reactor pressure vessel upper riser model 488
Figure 6-10 Typical reactor pressure vessel pressurizer model 490
Figure 6-11 Typical steam generator model 494
Figure 6-12 Typical main steam system model 495

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List of Figures

Figure 6-13 Typical decay heat removal system division 1 model 497
Figure 6-14 Not used. 497
Figure 6-15 Not used. 498
Figure 6-16 Typical containment and reactor pool model 500
Figure 7-1 Example of decay heat comparisons 520
Figure 7-2 Example of setpoint relationships. 526
Figure 7-3 Example power reduction vs. initial power and worth for dropped rod event at {{ }}$^{2(a),(c)}$ 618
Figure 7-4 Example components of local power vs. time for dropped rod event. 619
Figure 7-5 Example comparison of parameters associated with non-tripped rod drop event cases to single rod withdrawal event analysis values 620
Figure 8-1 Temperature of feedwater during the representative decrease in feedwater temperature event 663
Figure 8-2 Power response for the representative decrease in feedwater temperature event. 664
Figure 8-3 Core outlet temperature for the representative decrease in feedwater temperature event 664
Figure 8-4 Steam generator 2 pressure response for the representative decrease in feedwater temperature event 665
Figure 8-5 Reactor pressure vessel pressure response for the representative decrease in feedwater temperature event 665
Figure 8-6 Pressurizer level for the representative decrease in feedwater temperature event. 666
Figure 8-7 Reactor coolant system flow rate for the representative decrease in feedwater temperature event 666
Figure 8-8 Core inlet temperature for the representative decrease in feedwater temperature event 667
Figure 8-9 Net reactivity for the representative decrease in feedwater temperature event. 667
Figure 8-10 Main steam transient flow rate during the representative increase in steam flow event 671
Figure 8-11 Steam generator 2 pressure response for the representative increase in steam flow event 671
Figure 8-12 Steam generator 2 secondary side flow for the representative increase in steam flow event 672
Figure 8-13 Core inlet temperature for the representative increase in steam flow event. 672

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List of Figures

Figure 8-14 Power response for the representative increase in steam flow event 673
Figure 8-15 Core outlet temperature for the representative increase in steam flow event. 673
Figure 8-16 Reactor pressure vessel pressure response for the representative increase in steam flow event. 674
Figure 8-17 Pressurizer level for the representative increase in steam flow event 674
Figure 8-18 Reactor coolant system flow rate for the representative increase in steam flow event 675
Figure 8-19 Net reactivity for the representative increase in steam flow event 675
Figure 8-20 Steam generators 1 (unaffected) and 2 (affected) secondary flow rates for the representative main steam line break event. 679
Figure 8-21 Core inlet temperature for the representative main steam line break event. 679
Figure 8-22 Net reactivity for the representative main steam line break event 680
Figure 8-23 Power response for the representative main steam line break event 680
Figure 8-24 Steam generators 1 (unaffected) and 2 (affected) pressure response for the representative main steam line break event. 681
Figure 8-25 Pressurizer level for the representative main steam line break event 681
Figure 8-26 Reactor pressure vessel pressure response for the representative main steam line break event 682
Figure 8-27 Reactor coolant system flow rate for the representative main steam line break event. 682
Figure 8-28 Core outlet temperature for the representative main steam line break event. 683
Figure 8-29 Reactor pressure vessel pressure response for the representative loss of normal feedwater flow event - reactor coolant system pressure case 686
Figure 8-30 Pressurizer level for the representative loss of normal feedwater flow event - reactor coolant system pressure case 686
Figure 8-31 Steam generator 2 pressure response for the representative loss of normal feedwater flow event - reactor coolant system pressure case 687
Figure 8-32 Power response for the representative loss of normal feedwater flow event - reactor coolant system pressure case 687
Figure 8-33 Reactor coolant system flow rate for the representative loss of normal feedwater flow event - reactor coolant system pressure case 688
Figure 8-34 Core inlet temperature for the representative loss of normal feedwater flow event - reactor coolant system pressure case 688
Figure 8-35 Core outlet temperature for the representative loss of normal feedwater flow event - reactor coolant system pressure case 689

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List of Figures

Figure 8-36 Net reactivity for the representative loss of normal feedwater flow event - reactor coolant system pressure case 689
Figure 8-37 Steam generator 2 secondary flow for the representative loss of normal feedwater flow event - reactor coolant system pressure case 690
Figure 8-38 Core inlet temperature for the representative loss of normal feedwater flow event - secondary pressure case 692
Figure 8-39 Core outlet temperature for the representative loss of normal feedwater flow event - secondary pressure case 693
Figure 8-40 Reactor pressure vessel pressure response for the representative loss of normal feedwater flow event - secondary pressure case 693
Figure 8-41 Pressurizer level for the representative loss of normal feedwater flow event - secondary pressure case 694
Figure 8-42 Steam generator 2 pressure response for the representative loss of normal feedwater flow event - secondary pressure case 694
Figure 8-43 Power response for the representative loss of normal feedwater flow event - secondary pressure cas 695
Figure 8-44 Reactor coolant system flow rate for the representative loss of normal feedwater flow event - secondary pressure case 695
Figure 8-45 Net reactivity for the representative loss of normal feedwater flow event - secondary pressure case 696
Figure 8-46 Steam generator 2 secondary flow for the representative loss of normal feedwater flow event - secondary pressure case 696
Figure 8-47 Primary temperature response for the representative loss of AC power event. 699
Figure 8-48 System pressure response for the representative loss of AC power event . . . 699
Figure 8-49 Reactor pressure vessel core power response for the representative loss of AC power event 700
Figure 8-50 Decay heat removal system response for the representative loss of AC power event 700
Figure 8-51 RSV flow response for the representative loss of AC power event 701
Figure 8-52 Reactor coolant system flow response for the representative loss of AC power event 701
Figure 8-53 Pressurizer level response for the representative loss of AC power event . . . 702
Figure 8-54 Feedwater line break flow response for the representative feedwater line break event. 705
Figure 8-55 Primary temperature response for the representative feedwater line break event. 706

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List of Figures

Figure 8-56 System pressure response for the representative feedwater line break event. . . . 706
Figure 8-57 Reactor pressure vessel core power response for the representative feedwater line break event . . . 707
Figure 8-58 Reactor safety valve flow response for the representative feedwater line break event. . . . 707
Figure 8-59 Decay heat removal system response for the representative feedwater line break event. . . . 708
Figure 8-60 Pressurizer level response for the representative feedwater line break event. . . . 708
Figure 8-61 Reactor coolant system flow response for the representative feedwater line break event. . . . 709
Figure 8-62 Pressurizer pressure response for the bank withdrawal from a low power startup condition . . . 712
Figure 8-63 Reactor pressure vessel and steam generator pressure responses for the bank withdrawal from a low power startup condition . . . 712
Figure 8-64 Power response for the bank withdrawal from a low power startup condition . . . 713
Figure 8-65 Core inlet temperature for the bank withdrawal from a low power startup condition . . . 713
Figure 8-66 Core inlet density for the bank withdrawal from a low power startup condition . . . 714
Figure 8-67 Core outlet temperature for the bank withdrawal from a low power startup condition . . . 714
Figure 8-68 Reactor coolant system flow rate for the bank withdrawal from a low power startup condition . . . 715
Figure 8-69 Net reactivity for the bank withdrawal from a low power startup condition 715
Figure 8-70 Power response for the representative single rod withdrawal MCHFR case . . . 718
Figure 8-71 Reactor pressure vessel pressure response for the representative single rod withdrawal MCHFR case . . . 719
Figure 8-72 Pressurizer level for the representative single rod withdrawal MCHFR case . . . 719
Figure 8-73 Core outlet temperature for the representative single rod withdrawal MCHFR case . . . 720
Figure 8-74 Steam generator 2 pressure response for the representative single rod withdrawal MCHFR case . . . 720

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List of Figures

Figure 8-75 Reactor coolant system flow rate for the representative single rod withdrawal MCHFR case . . . 721
Figure 8-76 Steam generator 2 secondary flow for the representative single rod withdrawal MCHFR case . . . 721
Figure 8-77 Core inlet temperature for the representative single rod withdrawal MCHFR case . . . 722
Figure 8-78 Net reactivity for the representative single rod withdrawal MCHFR case . . . 722
Figure 8-79 Makeup flow for increase in reactor coolant system inventory . . . 725
Figure 8-80 Recirculation pump flow for increase in reactor coolant system inventory . . . 726
Figure 8-81 Letdown flow for increase in reactor coolant system inventory . . . 726
Figure 8-82 CVCS recirculation flow rate into the reactor pressure vessel for increase in reactor coolant system inventory . . . 727
Figure 8-83 CVCS recirculation flow rate out of the reactor pressure vessel for increase in reactor coolant system inventory . . . 727
Figure 8-84 Pressure at the bottom of the pressurizer for increase in reactor coolant system inventory . . . 728
Figure 8-85 Pressurizer level for increase in reactor coolant system inventory . . . 728
Figure 8-86 Reactor power for increase in reactor coolant system inventory . . . 729
Figure 8-87 Reactor coolant system flow for increase in reactor coolant system inventory . . . 729
Figure 8-88 Decay heat removal system flow rate for increase in reactor coolant system inventory . . . 730
Figure 8-89 Steam generator pressure for increase in reactor coolant system inventory . . . 730
Figure 8-90 Core inlet and exit coolant liquid temperature for increase in reactor coolant system inventory . . . 731
Figure 8-91 Total reactivity for increase in reactor coolant system inventory . . . 731
Figure 8-92 Instantaneous break flow response (0 to 350 sec) for the representative small break outside containment event . . . 735
Figure 8-93 Pressurizer level response for the representative small break outside containment event . . . 735
Figure 8-94 Reactor pressure vessel pressure response (0 to 350 sec) for the representative small break outside containment event . . . 736
Figure 8-95 Steam generator pressure responses for the representative small break outside containment event . . . 736
Figure 8-96 Core power response for the representative small break outside containment event . . . 737

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List of Figures

Figure 8-97 Integrated break flow response (0 to 350 sec) for the representative small break outside containment event . . . 737
Figure 8-98 Reactor coolant system flow rate response for the representative small break outside containment event . . . 738
Figure 8-99 Core outlet temperature response for the representative small break outside containment event . . . 738
Figure 8-100 Reactor pressure vessel pressure response (0 to 3000 sec) for the representative small break outside containment event . . . 739
Figure 8-101 Net reactivity response for the representative small break outside containment event . . . 739
Figure 8-102 Level above top of core response for the representative small break outside containment event . . . 740
Figure 8-103 Pressurizer level response for the representative steam generator tube failure event . . . 743
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) . . . 743
Figure 8-105 Core power response for the representative steam generator tube failure event. . . . 744
Figure 8-106 Instantaneous break flow response for the representative steam generator tube failure event . . . 744
Figure 8-107 Steam generator level response for the representative steam generator tube failure event . . . 745
Figure 8-108 Integrated break mass release to steam generator before isolation (0 to 500 sec) for the representative steam generator tube failure event. . . . 745
Figure 8-109 Reactor coolant system flow rate response for the representative steam generator tube failure event . . . 746
Figure 8-110 Core inlet and exit temperature responses for the representative steam generator tube failure event . . . 746
Figure 8-111 Reactor pressure vessel and steam generator responses (0 to 6000 sec) for the representative steam generator tube failure event . . . 747
Figure 8-112 Net reactivity response for the representative steam generator tube failure event. . . . 747
Figure 8-113 Level above top of core response for the representative steam generator tube failure event . . . 748

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Abstract

The purpose of this report is to present the NuScale evaluation model (EM) used to evaluate a NuScale Power Module (NPM) short-term system transient response to non-loss-of-coolant accident (non-LOCA) events. The non-LOCA evaluation model is developed following a graded approach to the guidance provided in Regulatory Guide (RG) 1.203 for the evaluation model development and assessment process (EMDAP). This report summarizes the NuScale plant designs, non-LOCA initiating events, classification of the events and acceptance criteria. The scope of the non-LOCA system transient analysis is described in this report, as well as the interfaces to other safety analysis methodologies. The non-LOCA events cover several different event types based on the main effect on the reactor coolant system (RCS). A comprehensive, integrated phenomena identification and ranking table (PIRT) was developed for the range of non-LOCA event types and phases of the event progression. The high-ranked phenomena of the PIRT and how they are assessed are summarized in this report. The NRELAP5 code is the system thermal-hydraulic code used for non-LOCA system transient analysis. Applicability of NRELAP5 for non-LOCA system transient analysis is assessed based on the high ranked phenomena identified in the PIRT. This report describes the selection of appropriately conservative input when applying this EM to perform non-LOCA system transient analyses. The non-LOCA methodology ensures that system transient calculations are executed for sufficient duration to demonstrate that the initiating event is mitigated and stable cooling is established. For non-LOCA initiating events that actuate the decay heat removal system, the EM is applicable for the short-term transient progression; during this time frame the mixture level remains above the top of the riser and primary side natural circulation is maintained. Representative calculations of non-LOCA events demonstrate margins to the primary and secondary pressure acceptance criteria. Other quantitative acceptance criteria, such as minimum critical heat flux ratio and radiological dose limits applicable for the non-LOCA events, are evaluated in downstream subchannel and accident radiological analyses, which are documented in separate reports.

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Executive Summary

The purpose of this report is to present the NuScale evaluation model (EM) used to evaluate a NuScale Power Module (NPM) short-term system transient response to non-loss-of-coolant accident (LOCA) events. This report summarizes the NuScale plant designs, non-LOCA initiating events, classification of the events and acceptance criteria. The scope of the non-LOCA system transient analysis and interfaces to other safety analysis methodologies is described. Development of the non-LOCA phenomena identification and ranking table is described; the high-ranked phenomena and how they are assessed are summarized. The NRELAP5 code is the system thermal-hydraulic code used for non-LOCA system transient analysis. Applicability of NRELAP5 for non-LOCA system transient analysis is assessed. Parameters considered in the system transient analyses to specify appropriately conservative input in application of the EM are discussed. Representative transient calculations for different types of non-LOCA events are presented. These representative calculations demonstrate the application of the method only and are not intended for NRC approval of a NuScale plant design.

An NPM is a small, light water integral pressurized water reactor (PWR) consisting of a nuclear core, two helical coil steam generators (SGs), and a pressurizer, all contained within a single reactor vessel. The reactor vessel is located within a small, compact steel containment vessel. An NPM is designed to operate efficiently at full power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps. The helical coil SGs, which produce superheated steam, are located in the annular space between the RCS hot leg riser and the reactor vessel inside diameter wall. The relative locations of the thermal centers in the core and the SGs promote buoyancy driven natural circulation flow. Power conversion occurs via a secondary system that includes the steam turbine-generator, the main condenser, and the plant components necessary to provide feedwater. Each NPM has dedicated chemical and volume control system (CVCS), emergency core cooling system (ECCS), and decay heat removal system (DHRS). The CVCS is used to regulate the primary side inventory via makeup and letdown to maintain pressurizer level and boron concentration during normal operation. Pressurizer heaters and spray control primary side pressure. The DHRS is a normally isolated, closed-loop, two-phase natural circulation cooling system; two trains of decay heat removal equipment are provided, one connected to each SG secondary side loop. These major features are common to NPM designs.

A NuScale power plant consists of one or more NPMs, each partially immersed in its own bay of the common reactor pool. The reactor pool serves as the ultimate heat sink (UHS) and is located in a Seismic Category I building designed to withstand postulated adverse natural conditions and aircraft impact. The NuScale design instrumentation and control architecture includes the safety-related module protection system, and nonsafety-related module and plant control systems. These major features are common to NuScale power plant designs.

The non-LOCA system transient evaluation model for analysis of an NPM system transient response to non-LOCA events is developed following a graded approach to the guidance provided in Regulatory Guide (RG) 1.203 for the evaluation model development and assessment process (EMDAP). The EMDAP as defined in RG 1.203 provides a structured process to establish the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems. Six basic principles are identified in RG 1.203 as important in the process of developing and assessing an EM. Four of the principles (using 20 steps as

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identified in the EMDAP process) are addressed in this report. The remaining principles related to establishing an appropriate quality assurance program and providing comprehensive, accurate, up-to-date documentation are addressed outside of this report as part of “NuScale Power, LLC Quality Assurance Program Description,” MN-122626 (Reference 3).

As part of defining the requirements of the non-LOCA transient system analysis methodology, the events to which the methodology applies and the specific event acceptance criteria applicable to the events are identified. The NPM designs were evaluated to assure that a sufficiently broad spectrum of transients, accidents, and initiating events have been included in the scope of design basis analyses presented in Final Safety Analysis Report (FSAR) Chapter 15. An NPM is a natural circulation integral PWR. Many of the events analyzed for operating plants and in recent design certification applications are applicable to the NuScale designs. NuScale-specific events reflect unique aspects of the NuScale designs such as the DHRS and normal operation of the containment at vacuum conditions. The design-basis events for which non-LOCA system transient analysis are performed are categorized into one of 5 categories:

1. Increase in heat removal from the RCS
2. Decrease in heat removal by the secondary system
3. Reactivity and power distribution anomalies
4. Increase in reactor coolant inventory
5. Decrease in reactor coolant inventory

Each event is classified as an anticipated operational occurrence (AOO), infrequent event (IE), or accident. The specific event acceptance criteria are derived from the regulatory requirements and guidance. The non-LOCA transient system analysis is part of several stages of analysis performed to confirm a plant design meets applicable acceptance criteria for a limiting set of AOOs, IEs, and accidents. The interfaces of the non-LOCA transient system analysis with downstream subchannel analysis and accident radiological analysis methodologies are identified. The subchannel analysis codes and methods, and the accident radiological source term and dose analyses are covered in separate methodologies and assessments. For non-LOCA initiating events that actuate the DHRS, the EM is applicable for the short-term transient progression; during this time frame the mixture level remains above the top of the riser and primary side natural circulation is maintained.

As part of developing the non-LOCA evaluation model, a phenomena identification and ranking table (PIRT) was developed. The non-LOCA events cover several different event types based on the main effect on the RCS. A comprehensive, integrated PIRT was performed for the range of non-LOCA event types and phases of the event progression. The PIRT panel considered an NPM design to identify systems, components, and subcomponents of the design for which phenomena were assessed. Phenomena were identified and ranked considering their level of importance relative to identified figures of merit for the different non-LOCA event types and phases of the transient progression; a knowledge ranking was established for each of the phenomena.

NRELAP5 is NuScale’s system thermal-hydraulics code used to simulate an NPM system response during both the non-LOCA and LOCA short-term transient event progression. The

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NRELAP5 code was derived from the Idaho National Laboratory (INL) RELAP5-3D© computer code. RELAP5-3D©, version 4.1.3 was used as the baseline development platform for the NRELAP5 code. RELAP5-3D© was procured and commercial grade dedication was performed by NuScale. Subsequently, features were added and changes made to address unique aspects of an NPM design and licensing methodology. NRELAP5 is a non-homogenous, non-equilibrium two-fluid thermal hydraulic systems analysis code capable of performing non-LOCA system transient analyses for an NPM. The NRELAP5 code has a heat conduction and heat transfer package that is similar in capability to other thermal-hydraulic codes in its class (such as TRAC, RETRAN or TRACE). It includes the trips and logic control systems that enable simulation of the plant protection and control system logic for analysis of a non-LOCA event in an NPM. The NRELAP5 code is described in the NuScale LOCA Evaluation Model licensing topical report. The NRELAP5 code has been assessed against several separate effects and integral effects tests as part of the code development and development of the NuScale LOCA evaluation model to demonstrate the capability to simulate an NPM response to LOCA events.

Phenomena identified as high-ranked for the non-LOCA transients were evaluated with respect to the high-ranked phenomena identified and assessed as part of the NuScale LOCA evaluation model development. Additional validation of NRELAP5 against separate effects testing, integral effects testing, and code to code benchmarking, were performed as necessary to justify applicability of the NRELAP5 code for non-LOCA system thermal-hydraulic analysis. High-ranked phenomena for non-LOCA events that were not assessed as part of the NuScale LOCA evaluation model development were therefore addressed in different ways:

1. additional NRELAP5 code assessment performed against separate effects or integral effects test data
2. code-to-code benchmark performed between NRELAP5 and independent system thermal-hydraulics code
3. phenomena addressed as part of the downstream subchannel analysis
4. phenomena addressed by specifying appropriately conservative input to the system transient analysis

In particular, separate and integral effects testing were performed at the NIST-1 and NIST-2 facilities to support applicability of the NRELAP5 code for non-LOCA system transient analysis. Separate effects testing of the DHRS was performed. Integral effects testing of an NPM response to a decrease in secondary side heat transfer, and integral effects testing of DHRS operation were performed. A code-to-code benchmark was performed to compare the NRELAP5 and RETRAN-3D responses to a range of reactivity insertion conditions in an NPM. Computational fluid dynamics was used to independently assess the heat transfer correlation used in NRELAP5 for the helical SGs of the NuScale designs.

The general non-LOCA transient analysis process is described in this report. The general methodology for conservatively biasing initial and boundary conditions for event analysis is presented. Each initiating event is then considered to identify the acceptance criteria that may be challenged during the event. For each non-LOCA event, a description of the event is provided including biases and conservatisms applied, sensitivity studies performed, single active failures and loss of power scenarios that challenge the event acceptance criteria. For each transient event, the acceptance criteria where margin to the limit may be challenged are identified. For

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these acceptance criteria, sensitivity calculations are performed to confirm that appropriately conservative inputs are specified and to determine conditions that result in minimum margin. For other acceptance criteria where margin to the limit is not challenged, representative results from the overall scope of sensitivity calculations performed are sufficient to demonstrate that margin to the acceptance criterion is maintained. For non-LOCA initiating events that actuate the DHRS, the EM is applicable for the short-term transient progression; during this time frame the mixture level remains above the top of the riser and primary side natural circulation is maintained.

For selected non-LOCA events, representative system transient results are provided to demonstrate the application of the evaluation model for an NPM. System transient calculations are executed for sufficient duration to demonstrate that the initiating event is mitigated and stable cooling is established. Results of representative calculations show that the maximum primary and secondary pressure acceptance criteria are not significantly challenged in an NPM design. Margin to other quantitative acceptance criteria for minimum critical heat flux ratio, fuel centerline temperature, and radiological dose limits applicable for the non-LOCA events are demonstrated as part of downstream subchannel or accident radiological analyses that are described in separate reports. In addition, long-term cooling analysis methodology is presented in a separate report.

NuScale requests U.S. Nuclear Regulatory Commission (NRC) approval to use the EM described in this report for analyses of NPM design basis non-LOCA events that require system transient analysis. The specific scope of the non-LOCA events for which the EM applies is delineated in Section 1.2. NuScale is not seeking NRC approval of the representative calculations that are described in this report.

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1.0 Introduction

1.1 Purpose

The purpose of this report is to present the NuScale evaluation model (EM) used to evaluate a NuScale Power Module (NPM) system transient response to non-loss-of-coolant accident (non-LOCA) events with the NRELAP5 code. This report summarizes the NuScale plant designs and identifies the potential non-LOCA initiating events for an NPM analyzed by this EM. The classification of these non-LOCA events and relevant acceptance criteria that are prescribed in the NRC standard review plan (SRP) and the NuScale design specific review standard (DSRS) are discussed in this report.

The purpose of the non-LOCA evaluation model is to model an NPM response to a non-LOCA design basis event. The non-LOCA system transient evaluation model is developed following a graded approach to the guidance provided in Regulatory Guide (RG) 1.203 (Reference 1). The non-LOCA phenomena identification and ranking table (PIRT) is described, including a summary of the high-ranked phenomena and how they are assessed. The applicability of NRELAP5 for non-LOCA system transient analysis is assessed. The scope of the non-LOCA system transient analysis is described in this report, as well as interfaces to other safety analysis methodologies. This report describes the selection of appropriately conservative input when applying this EM to perform non-LOCA system transient analyses. Representative transient calculations from application of the EM for the range of non-LOCA events are presented.

1.2 Scope

NuScale requests U.S. Nuclear Regulatory Commission (NRC) approval to use the EM described in this report for analyses of NPM design basis non-LOCA events that require system transient analysis. Representative analysis results are provided in Section 8.0 of this report to illustrate results from application of the EM. These representative cases are not necessarily based on final NuScale NPM design inputs, and NRC approval of the representative results is not requested. The scope of this report includes the applicability and acceptability of this methodology to evaluate the primary and secondary system pressure acceptance criteria found in Chapter 15 of the NuScale DSRS and the SRP. This report also describes how the non-LOCA evaluation model interfaces with other analyses that evaluate acceptance criteria that are not evaluated by the non-LOCA evaluation model.

The scope of the NuScale non-LOCA system transient analysis EM is summarized below:

- The non-LOCA evaluation model uses the NRELAP5 code to perform system transient analysis of the NPM design basis events listed in Table 4-1. The general and event-specific analysis methodologies of the EM are presented in Section 7.0. Sensitivity studies justifying the selected biasing direction are presented in Section 7.2 as part of the event-specific analysis methodology of the EM. The NRELAP5 code is described in the LOCA Evaluation Model licensing topical report (Reference 2).

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- The non-LOCA evaluation model is applicable to a nuclear power plant that follows the general description of the NuScale plant designs in Section 3.0. The applicability of the EM is based on the non-LOCA phenomena identification and ranking table and assessment of the high-ranked phenomena that are treated as part of the system transient analysis.
- The non-LOCA evaluation model does not address the evaluation of specified acceptable fuel design limits (SAFDLs), which are evaluated in a downstream subchannel analysis. The subchannel analysis codes and methods are covered in separate methodologies and assessments (Reference 6, supplemented by Reference 28). However, the interface of the non-LOCA system transient analysis with the downstream subchannel analysis is part of the non-LOCA evaluation model.
- The non-LOCA evaluation model does not address the evaluation of the accident radiological source term and dose. The accident radiological source term and dose analyses are covered in separate methodologies and assessments (Reference 8). However, the interface of the non-LOCA system transient analysis with downstream radiological analysis is part of the non-LOCA evaluation model.
- The EM is applicable for the short-term non-LOCA transient progression; the non-LOCA transient analysis short-term duration and analysis process are discussed further in Section 4.2 and Section 4.3. During this time frame the mixture level remains above the top of the riser and primary side natural circulation is maintained. The reactivity control and extended passive cooling analysis methodology in the long-term, including events that transition from DHRS cooling to ECCS cooling, is addressed in separate methodologies and assessments (Reference 26).
- Riser holes are included in the NPM designs to mitigate potential boron dilution impacts of long term DHRS cooling and riser uncovery. The PIRT and test assessments in Section 5.0 did not incorporate the riser holes; however, the PIRT and assessments documented in Section 5.0 remain valid, since the design feature of the riser holes results in negligible differences in the prediction of RCS parameters for the short-term non-LOCA transient progression as confirmed by sensitivity studies discussed in Section 5.0 and Section 7.0. The effect of the riser holes during extended passive cooling is addressed in separate methodologies and assessments (Reference 26).
- Control rod ejection accident analysis is addressed by a separate methodology (Reference 21) and is not part of the non-LOCA evaluation model.
- Loss of coolant accident analysis, including analysis of an inadvertent opening of one or more valves on the RPV, is addressed by a separate methodology (Reference 2) and is not part of the non-LOCA evaluation model.
- Analysis of the peak containment pressure and temperature response is addressed by a separate methodology (Reference 2) and is not part of the non-LOCA evaluation model.
- Evaluation of a return to power assuming the worst case stuck rod, if applicable to a NuScale design, is addressed in the final safety analysis report (FSAR) and is not part of the non-LOCA evaluation model.

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1.3 Abbreviations

Table 1-1 Abbreviations

Term	Definition
AC	alternating current
ANS	American Nuclear Society
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BOC	beginning of cycle
CFR	Code of Federal Regulations
CHF	critical heat flux
CHFR	critical heat flux ratio
CNV	containment vessel
CPV	cooling pool vessel
CRA	control rod assembly
CVC	chemical and volume control
CVCS	chemical and volume control system
DACS	data acquisition and control system
DC	direct current
DCA	Design Certification Application
DFWT	decrease in feedwater temperature
DHRS	decay heat removal system
DNB	departure from nucleate boiling
DSRS	Design Specific Review Standard
DTC	Doppler temperature coefficient
ECCS	emergency core cooling system
EDAS	augmented DC power system
EDNS	normal DC power system
EDSS	highly reliable DC power system
EHVS	high voltage AC electrical distribution system
ELVS	low voltage AC electrical distribution system
EM	evaluation model
EMDAP	evaluation model development and assessment process
EMVS	medium voltage AC electrical distribution system
EOC	end of cycle
ESFAS	engineered safety features actuation system
FOM	figure of merit
FSAR	Final Safety Analysis Report
FWIV	feedwater isolation valve
FWRV	feedwater regulating valve
GDC	General Design Criteria
HTP	heat transfer plate
HX	heat exchanger
IAB	inadvertent actuation block
ID	inside diameter
IE	infrequent event
IET	integral effects test
INL	Idaho National Laboratory

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Table 1-1 Abbreviations (Continued)

Term	Definition
KAIST	Korea Advanced Institute of Science and Technology
L/D	length to diameter ratio
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LOP	loss of power
MASLWR	multi-application small light water reactor
MCHFR	minimum critical heat flux ratio
MCS	module control system
MPS	module protection system
MSIBV	main steam isolation bypass valve
MSIV	main steam isolation valve
MSS	main steam system
MTC	moderator temperature coefficient
NIST	NuScale Integral System Test Facility
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
NRELAP5	NuScale version of RELAP5-3D©
OD	outside diameter
OSU	Oregon State University
P/L	pressure/level
PCS	plant control system
P/D	pitch to diameter ratio
PIRT	phenomena identification and ranking table
PWR	pressurized water reactor
PZR	pressurizer
QAP	Quality Assurance Program
RCS	reactor coolant system
RG	Regulatory Guide
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RSV	reactor safety valve
RTP	rated thermal power
RTS	reactor trip system
RVV	reactor vent valve
SAF	single active failure
SAFDL	specified acceptable fuel design limit
SDAA	Standard Design Approval Application
SET	separate effects test
SG	steam generator
SGTF	steam generator tube failure
SMR	small modular reactor
SRP	standard review plan
SSC	structures, systems, and components
TMDPJUN	time dependent junction
TMDPVOL	time dependent volume

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Table 1-1 Abbreviations (Continued)

Term	Definition
UCRW	uncontrolled rod withdrawal
UCRWS	uncontrolled rod withdrawal from subcritical (or low power)
UHS	ultimate heat sink
VAC	volts alternating current

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Table 1-2 Definitions

Term	Definition
Analytical limit	Limit of a measured or calculated variable established by the safety analysis to ensure that a safety limit is not exceeded.
Anticipated operational occurrences (AOOs)	Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.
Design basis accidents	A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures and components necessary to ensure public health and safety.
Design basis events	Postulated events used in the design to establish the acceptable performance requirements for the structures, systems and components.
Design bases	Information that identifies the specific functions to be performed by a system, structure or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted “state of the art” practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effect of a postulated accident for which a structure, system or component must meet its functional goals.
Excellent agreement	One of the acceptance criteria defined in RG 1.203. “Excellent” agreement applies when the code exhibits no deficiencies in modeling a given behavior. Major and minor phenomena and trends are correctly predicted. The calculated results are judged to agree closely with the data. The calculation, with few exceptions, lies within the specified or inferred uncertainty bands of the data. The code may be used with confidence in similar applications.
Infrequent events (IEs)	Events that are not classified as AOOs or as accidents, and are not expected to occur during the design life of the plant. For IEs, the acceptance criteria are specified such that some fuel damage may occur but the radiological acceptance criteria are stricter than those imposed for accidents. These include events that may be historically considered as AOOs for operating plants, but due to aspects of the NPM designs, the events are not expected to occur during the design life of the plant.
Insufficient agreement	One of the acceptance criteria defined in RG 1.203. “Insufficient” agreement applies when the code exhibits major deficiencies. The code provides an unacceptable prediction of the test data because major trends are not predicted correctly. Most calculated values lie outside the specified or inferred uncertainty bands of the data.
Loss-of-coolant accident	Those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

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Table 1-2 Definitions (Continued)

Term	Definition
Minimal agreement	One of the acceptance criteria defined in RG 1.203. “Minimal” agreement applies when the code exhibits significant deficiencies. Overall, the code provides a prediction that is only conditionally acceptable. Some major trends or phenomena are not predicted correctly, and some calculated values lie considerably outside the specified or inferred uncertainty bands of the data. Incorrect conclusions about trends and phenomena may be reached if the code were used in similar applications, and an appropriate warning needs to be issued to users. Selected code models or facility model noding need to be reviewed, modified and assessed before the code can be used with confidence in similar applications.
Non-LOCA transient	Reactor coolant system transients described in the standard review plan Sections 15.1, 15.2, 15.4, and 15.5, and other comparable transients that may be unique to the NuScale systems. Other sections in the standard review plan are specific to events with reactor coolant pumps, LOCA, radiological analysis, anticipated transient without scram, or boiling water reactors, and are outside of scope.
Postulated accidents	A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures and components necessary to ensure public health and safety.
“Reasonable” agreement	One of the acceptance criteria defined in RG 1.203. “Reasonable” agreement applies when the code exhibits minor deficiencies. Overall, the code provides an acceptable prediction. All major trends and phenomena are correctly predicted. Differences between calculation and data are greater than deemed necessary for excellent agreement. The calculation frequently lies outside but near the specified or inferred uncertainty bands of the data. However, the correct conclusions about trends and phenomena would be reached if the code were used in similar applications.
Safety-related structures, system and components	Those structures, systems and components that are relied upon to remain functional during and following design-basis events to assure: (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1).

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2.0 Background

This topical report provides a description of the NuScale non-LOCA system transient analysis EM. The non-LOCA system transient analysis EM is developed following a graded application of the guidelines in the evaluation model development and assessment process (EMDAP) of RG 1.203.

Six basic principles are identified in RG 1.203 as important in the process of developing and assessing an EM. Four of the principles (using 20 steps as identified in the EMDAP process) are addressed in this report. The remaining principles related to establishing an appropriate quality assurance program and providing comprehensive, accurate, up-to-date documentation are addressed outside of this report as part of “NuScale Power, LLC Quality Assurance Program Description,” MN-122626 (Reference 3).

This EM utilizes the NRELAP5 code that was developed from the Idaho National Laboratory (INL) RELAP5-3D© computer code. The NRELAP5 code is described in Reference 2. Applicability of the NRELAP5 code for application in non-LOCA system transient analysis is discussed in this report.

2.1 Non-LOCA Evaluation Model Roadmap

Analyses are performed to demonstrate that a nuclear power plant can meet applicable NRC regulatory acceptance criteria for a limiting set of AOOs, IEs, and accidents.

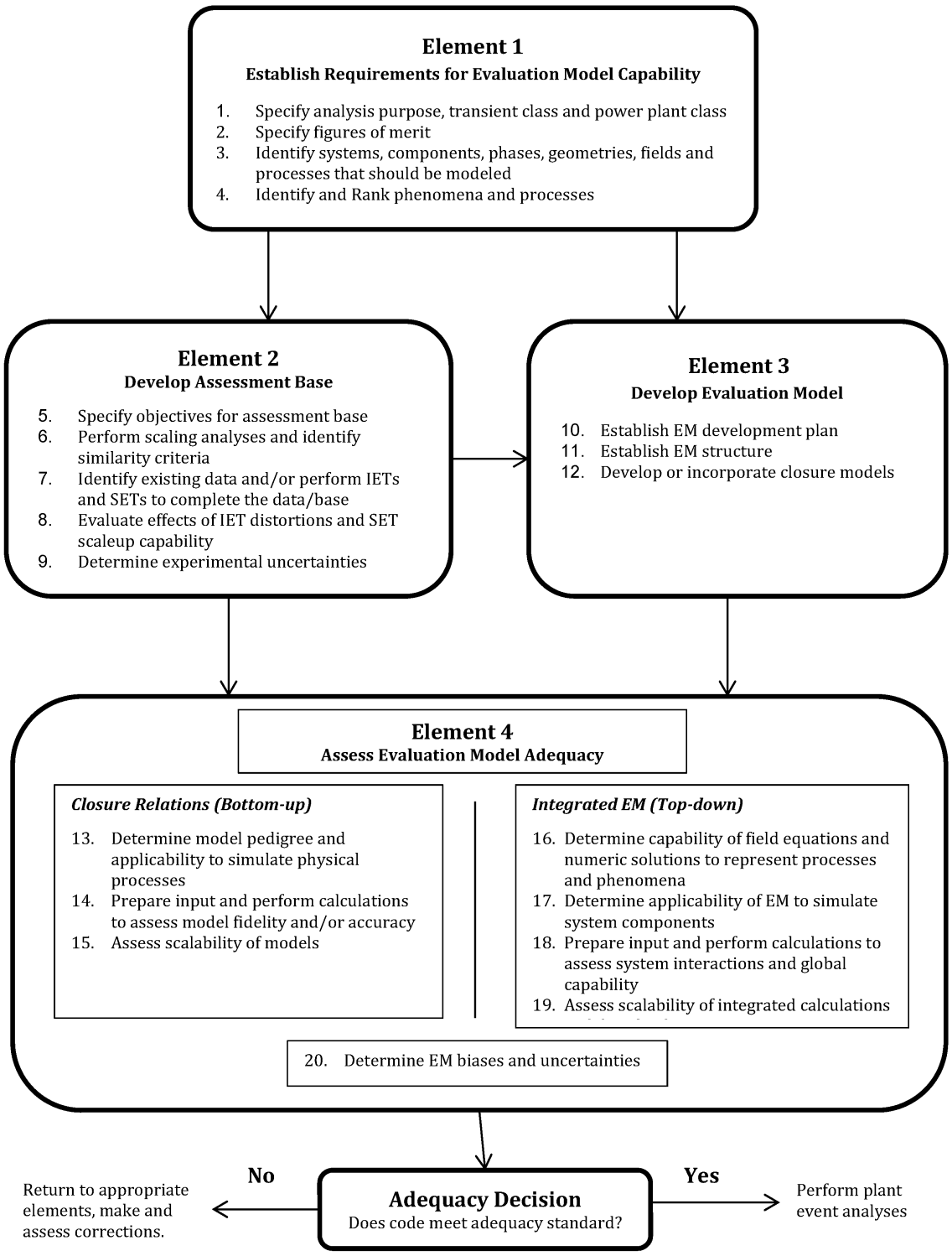
The EMDAP as defined in RG 1.203 provides a structured process to establish the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems. The EM described here is developed for simulating an NPM system transient response to non-LOCA events.

NRELAP5 is NuScale’s system thermal-hydraulics code used to simulate an NPM system response during both the non-LOCA and LOCA short-term transient event progression. The NuScale LOCA EM (Reference 2) was developed following the EMDAP guidelines of RG 1.203. As described in Section 5.0, phenomena identified as high-ranked for the non-LOCA transients were evaluated with respect to the high-ranked phenomena identified as part of the NuScale LOCA evaluation model development. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the non-LOCA plant transient calculations (Section 7.0), a graded approach to the EMDAP is applied for development of the non-LOCA system transient EM.

Figure 2-1 shows various elements of EMDAP as defined in RG 1.203. The elements of the EMDAP and sections of this report that relate to the elements and steps of the EMDAP are summarized in Table 2-1.

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Figure 2-1 Evaluation model development and assessment process



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Table 2-1 Evaluation model development and assessment process steps and associated application in the non-LOCA evaluation model

EMDAP Step	Description	EM Section
Element 1, Establish Requirements for Evaluation Model Capability		
1	Specify analysis purpose, transient class and power plant class.	The purpose of the non-LOCA system transient analysis methodology is described in Section 1.1. Section 2.0 briefly describes the background of the process followed to develop the non-LOCA system transient analysis methodology and the principal software used. Section 3.0 provides an overview of an NPM and a description of the plant operation. This overview includes the safety systems, the system logic, and operational phases that could occur in a NuScale SMR design. The high level regulatory requirements that the methodology is designed to comply with are described in Section 2.2. In Section 4.1 the non-LOCA initiating events and the classification of the events for an NPM are discussed. The acceptance criteria for the events are identified in Section 4.2. As identified in Section 4.2, margin to some of these acceptance criteria are demonstrated based on the results of the non-LOCA system transient analysis; other acceptance criteria are met as part of downstream analyses such as subchannel and radiological analyses. Downstream analyses are outside the scope of this topical report as discussed in Section 1.2. The non-LOCA transient analysis process, including interfaces with other safety analysis methodologies, is described in Section 4.3.
2	Specify figures of merit (FOMs).	Section 5.1 discusses the FOMs that are used for the development of the NPM non-LOCA PIRT.
3	Identify systems, components, phases, geometries, fields, and processes that should be modeled.	Systems, components, phases and processes are identified as a part of the non-LOCA PIRT discussed in Section 5.1.
4	Identify and rank phenomena and processes.	Section 5.1 describes the NPM non-LOCA PIRT.
Element 2, Develop Assessment Base		
5	Specify objectives for assessment base.	Section 5.2 describes the high ranked phenomena identified from the PIRT process and how the phenomena are addressed by NRELAP5 assessment or other approach. Many of the high ranked phenomena were assessed against experimental data as part of the LOCA evaluation model development (Reference 2); additional assessments were identified as described in Section 5.2.

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Table 2-1 Evaluation model development and assessment process steps and associated application in the non-LOCA evaluation model (Continued)

EMDAP Step	Description	EM Section
6	Perform scaling analysis and identify similarity criteria.	A scaling analysis of the LOCA and emergency core cooling system (ECCS) has been performed for the NPM centered on the NuScale Integral Systems Test-1 (NIST-1) facility. The results of the scaling analysis are discussed in Reference 2. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the non-LOCA plant transient calculations, these assessments are considered adequate for the non-LOCA system transient EM.
7	Identify existing data and perform integral effects test (IETs) and separate effects tests (SETs) to complete database.	Reference 2 and Section 5.3 of this report provide the results of the NRELAP5 validation against the SETs and IETs.
8	Evaluate effects of IET distortions and SET scaleup capability.	In Reference 2, a bottom-up assessment of NRELAP5 closure models and correlations is presented; this assessment addresses the fidelity of the models and correlations to the appropriate fundamental or SET data. In Reference 2, a top-down assessment of the NRELAP5 governing equations and numerics is presented. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the non-LOCA plant transient calculations, these assessments are considered adequate for the non-LOCA system transient EM.
9	Determine experimental uncertainties.	Reference 2 and Section 5.3 of this report cover experimental uncertainties for NRELAP5 assessments against the SETs and IETs.
Element 3, Develop Evaluation Model		
10	Establish EM development plan.	The NRELAP5 development plan includes programming standards and procedures, quality assurance procedures, and configuration control, which are summarized in Reference 2.
11	Establish EM structure.	Reference 2 provides a summary of NRELAP5 models and correlations. The non-LOCA transient analysis process, including interfaces with other safety analysis methodologies, is described in Section 4.3. For non-LOCA system transient analysis, the typical plant model is described in Section 6.0. The non-LOCA analysis methodology is described in Section 7.0.
12	Develop or incorporate closure models.	Reference 2 provides a summary of NRELAP5 models and correlations. A full description of the closure models and the associated equations used in the non-LOCA evaluation model is provided in the NRELAP5 theory and users manuals.

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Table 2-1 Evaluation model development and assessment process steps and associated application in the non-LOCA evaluation model (Continued)

EMDAP Step	Description	EM Section
Element 4, Assess Evaluation Model Adequacy Closure Relations (Bottom-up)		
13	Determine model pedigree and applicability to simulate physical processes.	Reference 2 includes a bottom-up assessment of important NRELAP5 models and correlations essential to simulate high-ranked PIRT phenomena for LOCA events, including discussion of model pedigree and applicability. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the non-LOCA plant transient calculations, these assessments are considered adequate for the non-LOCA system transient EM.
14	Prepare input and perform calculations to assess model fidelity and accuracy.	Reference 2 and Section 5.3 summarize the results of comparison of NRELAP5 against the selected SETs and IETs including evaluation of code fidelity and accuracy.
15	Assess scalability of models.	Reference 2 includes discussion on scalability of dominant NRELAP5 models and correlations that are essential to simulate high-ranked PIRT phenomena for LOCA events. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the non-LOCA plant transient calculations, these assessments are considered adequate for the non-LOCA system transient EM.
Element 4, Assess Evaluation Model Adequacy Integrated EM (Top-down)		
16	Determine capability of field equations and numeric solutions to represent processes and phenomena.	NRELAP5 field equations and the numeric solution scheme are discussed in Reference 2 and evaluated for their applicability to NPM LOCA. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the non-LOCA plant transient calculations, these assessments are considered adequate for the non-LOCA system transient EM.
17	Determine applicability of EM to simulate system components.	The applicability of the EM to simulate the NPM systems and components is demonstrated by assessment of NRELAP5 against NuScale design-specific SETs and IETs.
18	Prepare input and perform calculations to assess system interactions and global capability.	Reference 2 and Section 5.3 summarize the results of an assessment of NRELAP5 against IET data.
19	Assess scalability of integrated calculations and data for distortions.	Reference 2 provides an evaluation of scaling distortions between the NIST-1 LOCA IET data and the NPM design. The scalability of the EM to represent NPM LOCA phenomena and processes is presented therein. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the non-LOCA plant transient calculations, these assessments are considered adequate for the non-LOCA system transient EM.
20	Determine EM biases and uncertainties.	For the non-LOCA system transient analyses, suitably conservative input is specified in the plant calculations as described in Section 7.0, considering the effects on the appropriate acceptance criteria.

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2.2 Regulatory Requirements

The following General Design Criteria (GDC) of 10 CFR Part 50 Appendix A (Reference 4) are relevant to the non-LOCA transient analyses:

- GDC 5, as it relates to demonstrating that sharing of structures, systems, and components (SSC) does not significantly impact the ability of the SSC to perform their safety function.
- GDC 10, as it relates to demonstrating that SAFDLs are not exceeded during AOOs.
- GDC 15, as it relates to demonstrating that the reactor coolant system pressure boundary is not breached during AOOs.
- GDC 17, as it relates to providing electric power systems to permit functioning of SSC to assure that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs, and that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The NuScale designs support an exemption from GDC 17 and therefore GDC 17 is not relevant to this methodology.
- GDC 20, as it relates to demonstrating that the automatic operation of systems by the reactor protection system ensures that a plant does not exceed SAFDLs during AOOs.
- GDC 25, as it relates to demonstrating that the protection system design assures that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal (not ejection or dropout) of control rods.
- GDC 26, as it relates to demonstrating that the control rods are capable of reliably controlling reactivity changes to assure that SAFDLs are not exceeded during AOOs, with appropriate margin for malfunctions such as stuck rods.
- GDC 27, as it relates to demonstrating that the reactivity control systems are capable of reliably controlling reactivity changes to assure capability to cool the core is maintained under postulated accident conditions, with appropriate margin for stuck rods. Some NuScale designs support an exemption from GDC 27, and instead implement a NuScale-specific Principal Design Criterion (PDC) 27, as documented in the FSAR for those designs. As relevant to this methodology, NuScale PDC 27 is equivalent to GDC 27 (i.e., differences are not within the scope of this evaluation methodology).
- GDC 28, as it relates to demonstrating that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor impair significantly the capability to cool the core.
- GDC 31, as it relates to demonstrating that the probability of a rapidly propagating fracture of the reactor coolant pressure boundary is minimized when the pressure boundary is stressed under postulated accident conditions.
- GDC 34, as it relates to demonstrating that residual heat is removed from the reactor core at a rate such that SAFDLs and the design conditions of the reactor coolant pressure boundary are not exceeded, assuming a single failure and considering

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offsite power availability. The NuScale designs support an exemption from GDC 34, and instead implement a NuScale-specific Principal Design Criterion (PDC) 34, as documented in the FSAR for the designs. As relevant to this methodology, NuScale PDC 34 is equivalent to GDC 34 (i.e., differences are not within the scope of this evaluation methodology).

Regulatory guidance documents relevant to the non-LOCA transient system analysis EM development include:

- Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” December 2005.
- NuScale Design Specific Review Standard Sections:
  - 15.0, Revision 0, “Introduction - Transient and Accident Analyses” June 2016.
  - 15.1.1-15.1.4, Revision 0, “Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System,” June 2016.
  - 15.1.5, Revision 0, “Steam System Piping Failures Inside and Outside of Containment,” June 2016.
  - 15.1.6, Revision 0, “Loss of Containment Vacuum,” June 2016.
  - 15.2.1-15.2.5, Revision 0, “Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed),” June 2016.
  - 15.2.6, Revision 0, “Loss of Nonemergency AC Power to the Station Auxiliaries,” June 2016.
  - 15.2.7, Revision 0, “Loss of Normal Feedwater Flow,” June 2016.
  - 15.2.8, Revision 0, “Feedwater System Pipe Breaks Inside and Outside Containment (PWR),” June 2016.
  - 15.5.1-15.5.2, Revision 0, “Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory,” June 2016.
- NUREG-0800 Sections:
  - 15.0.2, Revision 0, “Review of Transient and Accident Analysis Methods,” March 2007.
  - 15.4.1, Revision 3, “Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition,” March 2007.
  - 15.4.2, Revision 3, “Uncontrolled Control Rod Assembly Withdrawal at Power,” March 2007.
  - 15.4.3, Revision 3, “Control Rod Misoperation (System Malfunction or Operator Error),” March 2007.

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- 15.4.4-15.4.5, Revision 2, “Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate,” March 2007.
- 15.4.6, Revision 2, “Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR),” March 2007.

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3.0 Plant Design Overview

An overview is provided of the major features common to NPMs and NuScale Power Plants. The non-LOCA EM is generally applicable to NPM and NuScale Power Plant designs that are consistent with the major features and plant operation described in the following sections. Applicability of the non-LOCA EM to a specific NPM design is based on assessment of the design as described in Section 5.0.

3.1 Description of NuScale Plant

An NPM is a small, light water cooled, pressurized water reactor (PWR) consisting of a nuclear core, two helical coil SGs, and a pressurizer, all contained within a single containment vessel (CNV) (Figure 3-1). Power conversion occurs via a standard secondary system that includes the steam turbine-generator, the main condenser, and the plant components necessary to provide feedwater.

Each NPM is covered by a reinforced concrete biological shield and enclosed in a Reactor Building, and has a dedicated chemical and volume control system (CVCS), ECCS, and DHRS.

An NPM is designed to operate efficiently at full power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps.

Unique features of an NPM include:

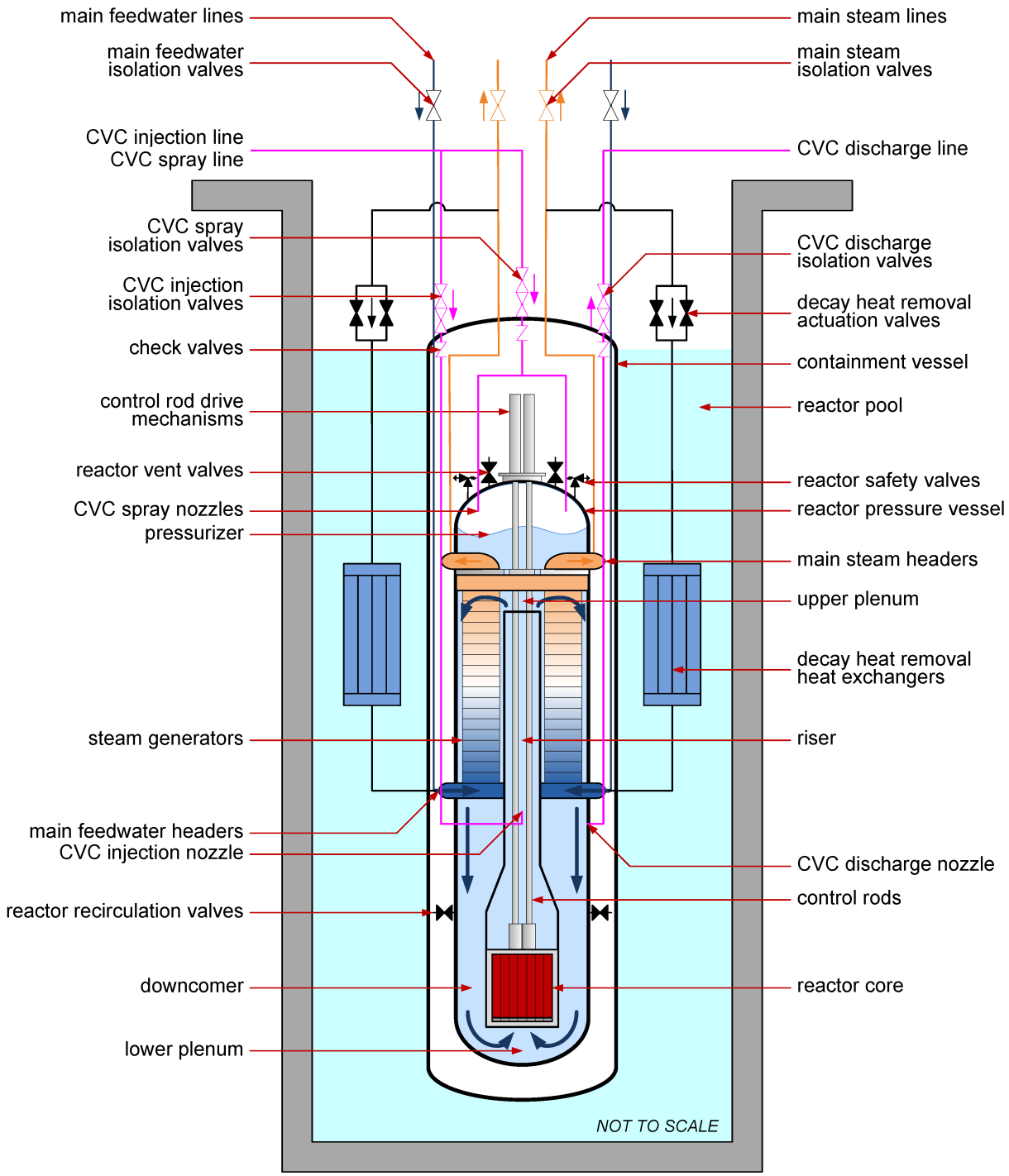
- a reduced core size relative to operating PWRs,
- natural circulation reactor coolant flow (i.e., no reactor coolant pumps),
- integrated SGs and pressurizer inside the reactor pressure vessel (RPV) (i.e., there is no piping connecting the SGs or pressurizer with the reactor),
- simplified passive safety systems that do not rely on ECCS pumps, accumulators, tanks, or connected piping,
- a high-pressure steel containment, and
- containment partially immersed in a water-filled pool providing an effective passive heat sink for emergency cooling and decay heat removal.

NuScale has achieved a substantial improvement in safety over existing plants through simplicity of design, reliance on passive safety systems, and small fuel inventory.

A NuScale Power Plant consists of one or more NPMs, each in its own bay of the common reactor pool. The pool is designed for leakage control. Each bay has a reinforced concrete cover that serves as a biological shield. The cover also serves to prevent deposition of foreign materials onto an NPM. The reactor pool is located in a Seismic Category I building designed to withstand postulated adverse natural conditions and aircraft impact.

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Figure 3-1 NuScale Power Module schematic



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3.2 Plant Operation

During nominal full power operation, the control rods are retracted to within their insertion limits and borated water is used as the primary coolant, which is driven by natural circulation. The CVCS is used to regulate the primary side inventory via manually operated makeup and either automatic or manual letdown, depending on the design, to maintain pressurizer level and boron concentration to maintain criticality. Pressurizer heaters and spray control primary side pressure. The helical coil SGs transfer the heat from the primary side to the feedwater. The DHRS heat exchangers are isolated during normal operation. The containment is evacuated to provide an insulated barrier between the reactor pressure vessel and containment. Each NPM is partially immersed in the reactor pool within the Reactor Building, which serves as the ultimate heat sink (UHS) and is open to atmospheric pressure. The pool cooling equipment is designed to maintain an average bulk pool temperature such that plant personnel can work in the Reactor Building.

The NuScale instrumentation and control architecture primarily consists of the following systems:

- module control system (MCS)
- plant control system (PCS)
- module protection system (MPS)
- plant protection system

The MCS and PCS provide control and monitoring of the non-safety nuclear steam supply system (e.g., steam bypass to condensers, pressurizer heaters and sprays, and feedwater control), balance of plant systems (e.g., turbine control); rod control and position indication, and plant-wide, non-safety control and indication.

The MPS is composed primarily of the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS). The MPS protection functions are limited to automated safety responses to off-normal conditions. The MPS functional response to an initiating event is a reactor trip, isolation (as necessary) of feedwater, MSS, CVCS (including demineralized water system isolation to mitigate boron dilution), and containment, followed by an integrated safety actuation of one or more of the passive safety systems (DHRS and ECCS).

The RTS consists of four independent separation groups with independent measurement channels to monitor plant parameters that can generate a reactor trip. Each measurement channel trips when the parameter exceeds a predetermined setpoint. The RTS coincident logic is designed so that no single failure can prevent a reactor trip when required, and no failure in a single measurement channel can generate an unnecessary reactor trip.

The ESFAS consists of four independent separation groups with independent measurement channels to monitor plant parameters that can activate the operation of the engineered safety features. Each measurement channel trips when the parameter

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exceeds a predetermined setpoint. The ESFAS coincident logic is designed so that no single failure can prevent a safeguards actuation when required, and no failure in a single measurement channel can generate an unnecessary safeguards actuation.

Transients requiring decay heat removal are addressed by the DHRS, which provides cooling through one or both of the SGs. For a steam generator tube failure (SGTF), main steam line break, and feedwater line break, the affected SG is isolated and the DHRS provides cooling through the intact SG (depending on the break location DHRS may be operational in both SGs). Manual operation of the nonsafety-related CVCS can also be used to offset decreases in RCS inventory. If the CVCS is inadequate to address the inventory decrease, containment isolation occurs and the DHRS is actuated. If RCS inventory loss to containment persists, ECCS is actuated.

Module-specific systems and functions that operate to mitigate the effects of postulated non-LOCA events (and credited in the safety analysis) include the ECCS, DHRS, CVCS and demineralized water system isolation, MPS, RTS, containment isolation and PZR heater isolation. The only safety system shared between modules is the UHS. The non-LOCA safety analyses consider ranges of UHS conditions and heat transfer such that non-LOCA analysis of a single module bounds possible UHS interactions between modules.

3.3 Decay Heat Removal System

The DHRS is a closed-loop, two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each SG loop. Each train is designed to remove decay heat and cool the RCS. Each train has a passive condenser immersed in the reactor pool. Upon receipt of an actuation signal, the main steam isolation valves (MSIVs) and the feedwater isolation valves close, and the decay heat removal actuation valves open, allowing heat removal via the SGs. The decay heat removal actuation valves would open upon the loss of power, thus enabling reliable long term cooling. For successful operation, liquid water enters the SG through the feedwater line and is boiled by heat from the RCS. The vapor exits the SG through the steam line and is directed to the DHRS condenser where it condenses back to liquid to return to the SG. Thus, the loop transfers heat from the RCS to the DHRS fluid using the SG and then from the DHRS to the reactor pool water.

3.4 Emergency Core Cooling System

The ECCS consists of two or three independent reactor vent valves (RVVs), depending on NPM design, and two independent reactor recirculation valves (RRVs). The ECCS is initiated by simultaneously actuating the RVVs on the top of the RPV in the pressurizer region and the RRVs on the side of the RPV in the downcomer region. Opening the ECCS valves allows a natural circulation path to be established - water is vaporized in the core, leaves as steam through the RVVs, condenses and collects in the containment, and returns to the downcomer region inside the RPV through the RRVs. During normal operation, each ECCS valve is held closed by the hydraulic pressure across the valve main disc. Included in the RRV design is an inadvertent actuation block (IAB) consisting of a spring loaded arming valve in the vent port path from the main disc chamber to the

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vent line. If the differential pressure across this arming valve is greater than a threshold value, the arming valve closes, which prevents the main disc chamber from discharging through the vent line, blocking the RRV from opening. The RRV does not open until the arming valve differential pressure decreases below the release pressure. The IAB is also used in the RVVs for some NPM designs.

Provided the IAB device setpoint is reached, if applicable, the RVV and RRV components fail to the open (safe) position upon the loss of power, thus enabling reliable long-term cooling without operator actions, alternating current (AC) or direct current (DC) power, or make-up water. Successful operation of the ECCS requires isolation of the containment, such that the coolant inventory of the RCS is preserved.

3.5 Other Important Systems and Functions

Other systems and functions that are important in mitigating plant response during a postulated non-LOCA event are discussed below.

Reactor Coolant System

The reactor coolant system (RCS) consists of the RPV, reactor core, riser, upper plenum, SGs (shell side), downcomer, lower plenum, and pressurizer (PZR). The arrangement of the RCS and the relative locations of the thermal centers in the core and the SGs promote buoyancy driven natural circulation flow.

The RPV consists of a steel cylinder with an inside diameter of approximately 10 ft and an overall height of approximately 60 ft and is designed for a normal operating pressure of approximately 1850-2000 psia. Nozzles on the upper head provide connections for reactor safety valves (RSVs) and RVVs.

The core configuration for an NPM consists of 37 fuel assemblies and 16 control rod assemblies (CRAs). The fuel assembly design is modeled from a standard 17x17 PWR fuel assembly with 24 guide tube locations for control rod fingers and a central instrument tube. The assembly is nominally half the height of standard plant fuel and is supported by five spacer grids. The U-235 enrichment is below the current U.S. manufacturer limit of 4.95 weight percent.

Each NPM uses two once-through helical coil SGs for steam production. The SGs, which produce superheated steam, are located in the annular space between the RCS hot leg riser and the reactor vessel inside diameter wall. Each SG is designed to remove 50 percent of the rated core thermal power. Flow paths in the upper riser are provided to mitigate potential boron dilution impacts during long-term DHRS cooling and riser uncovery. An NPM design may include flow paths in the lower riser to mitigate potential boron dilution impacts near the core inlet during long-term ECCS cooling.

The PZR provides the primary means for controlling RCS pressure. PZR heaters and spray maintain a constant reactor coolant pressure during operation. A steel PZR baffle plate, integral with the SG tube sheets and the RPV, acts as a thermal barrier and allows for surge flow between the PZR and the RCS.

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Feedwater System

Feedwater from the condenser is pumped by condensate pumps to the condensate polishing equipment, where impurities are removed. Downstream of the polishing equipment, variable speed feedwater pumps supply flow to the feedwater heaters before the feedwater regulating valves control feed to the SGs. In unit operation, preheated feedwater is pumped into the tube side of the SGs where it boils. Upon receipt of a DHRS actuation signal the feedwater isolation valves close.

Main Steam System

Superheated steam produced in the SGs flows to a dedicated steam turbine. A generator, driven by the turbine, generates electric power that is delivered to the utility grid through a step-up transformer. A turbine steam bypass valve is provided that allows the reactor to remain in operation in the event of a turbine trip. Upon receipt of a DHRS actuation signal the MSIVs close (the steam system contains backup isolation valves in the event that an MSIV fails to isolate).

Chemical and Volume Control System

The primary functions of the CVCS are to purify reactor coolant, adjust the boron concentration in the reactor coolant, and supply spray flow to the pressurizer. Makeup and letdown operation can also be used to adjust the RCS inventory as needed. Equipment within the CVCS also allows for chemical addition to the reactor coolant, and heats the reactor coolant during startup. The CVCS includes demineralized water system isolation valves to mitigate boron dilution. When used for reactor coolant heating, the CVCS heats the RCS to the hot standby startup temperature, and develops natural circulation through the core sufficient to maintain the required RCS flow prior to nuclear heat addition.

Containment Vessel

The major safety functions of the CNV are to contain the release of radioactivity following postulated accidents, protect the RPV and its contents from external hazards, and to provide heat rejection to the reactor pool following ECCS actuation.

Following an actuation of the ECCS, heat removal through the CNV rapidly reduces the containment pressure and temperature and maintains them at less than design conditions for extended periods of time. Steam is condensed on the inside surface of the CNV, which is passively cooled by conduction and convection heat transfer to the reactor pool water.

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Reactor Pool

The reactor pool is a large pool located below the plant ground level in the Reactor Building. During normal plant operations, heat is removed from the pool through a cooling system and ultimately rejected into the atmosphere through a cooling tower or other external heat sink. In an event where AC power is lost, heat is removed from an NPM by allowing the pool to heat up and boil. Water inventory in the reactor pool is maintained at a level that is sufficient to provide at least three days of DHRS operation.

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4.0 Transient and Accident Analysis Overview

As part of defining the requirements of the non-LOCA transient system analysis methodology, the events to which the methodology applies and the specific event acceptance criteria applicable to the events are identified. The specific event acceptance criteria are derived from the regulatory requirements and guidance discussed in Section 2.2. The non-LOCA transient system analysis is part of several stages of analyses performed to confirm a plant design meets applicable acceptance criteria for a limiting set of AOOs, IEs, and accidents. The methodology includes the interfaces of the non-LOCA transient system analysis with other analysis methodologies, and identifies where margin to the event acceptance criteria is demonstrated.

4.1 Design-Basis Events and Event Classification

The NuScale design-basis events for which the non-LOCA system transient analysis is performed, the event category, and the event classification are listed in Table 4-1.

A broad spectrum of transients, accidents, and initiating events are considered in the scope of design basis analyses presented in Chapter 15 of a NuScale plant FSAR. The design basis events are identified based on:

- review of the NuScale plant systems to identify failures that would result in a design basis initiating event
- review of initiating events considered in the NuScale probabilistic risk assessment analyses to identify design basis initiating events. The probabilistic risk assessment initiating events included consideration of:
  - the master logic diagram of failure mechanisms that may result in core damage
  - industry generic data sources reviewed to identify initiating events
  - advanced reactor probabilistic risk assessments
- review of similarities and differences of the plant design to previously evaluated NuScale plant designs

As described in Section 3.0, an NPM is a natural circulation PWR with SGs that are integral to the reactor vessel. Many of the events analyzed for operating plants and in recent design certification applications are applicable to a NuScale design. NuScale-specific events reflect unique aspects of a NuScale design such as the DHRS and vacuum conditions of containment during normal operation.

The design basis events are categorized by type and expected frequency of occurrence. Limiting cases in each group are quantitatively analyzed and specific acceptance criteria for each postulated initiating event are applied. The design basis events that require non-LOCA system transient analysis are categorized into one of five categories:

1. increase in heat removal from the RCS

   In a NuScale design, this increase may be due to increased heat removal by the secondary system or due to increased heat removal to the containment.

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2. decrease in heat removal by the secondary system
3. reactivity and power distribution anomalies
4. increase in reactor coolant inventory
5. decrease in reactor coolant inventory

The design-basis events are classified into one of three event categories:

1. AOOs - These are events that are expected to occur one or more times in the design life of the plant, conservatively quantified as events with a frequency of occurrence of $1x10^{-2}$ per module year or greater. An event that is not expected to occur in the design life of the plant may also be categorized as an AOO for conservatism.
2. Infrequent event - These are events that are not expected to occur during the design life of the plant. For IEs, the acceptance criteria are specified such that some fuel damage may occur but the radiological acceptance criteria are stricter than those imposed for accidents.
3. Postulated Accidents - These are design-basis events that are not expected to occur during the design life of the plant.

Historical precedent is used for event classification where the event is initiated by abnormal system conditions that are similar to those experienced in currently operating plants and certified designs. Event frequencies from the probabilistic risk assessment are considered for events that are unique to an NPM design or where an NPM design is such that the event frequency is expected to differ from currently operating plants and certified designs.

The non-LOCA EM is applicable for the initiating events listed in Table 4-1. Because the control rod ejection accident analysis, LOCA analysis, and analysis of an inadvertent opening of one or more valves on the RPV initiating event are addressed by different methodologies, these initiating events are not included in Table 4-1.

The non-LOCA system transient analyses are performed for a single module. In a NuScale design, the only shared safety-related system relied upon for event mitigation in the design basis system transient event analysis is the reactor pool portion of the UHS. Some initiating events may affect only a single module; others may affect multiple modules. In the non-LOCA system transient analysis calculations, the initial temperature of the reactor pool is bounded as described in Section 7.0. This approach bounds various module responses that may occur due to an initiating event that affects more than one module in a plant.

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Table 4-1 Design basis events for which the non-LOCA system transient analysis is performed, event category, and event classification

Initiating Event	Event Classification
Increase in Heat Removal from the Reactor Coolant System	
Decrease in feedwater temperature	AOO
Increase in feedwater flow	AOO
Increase in steam flow	AOO
Inadvertent opening of SG relief or safety valve	AOO
Steam system piping failure inside or outside of containment	Postulated accident
Containment flooding/loss of containment vacuum	AOO
Inadvertent DHRS actuation(1)	AOO
Decrease in Heat Removal by the Secondary System	
Loss of external load	AOO
Turbine trip	AOO
Loss of condenser vacuum	AOO
Main steam isolation valve closure	AOO
Loss of nonemergency AC power to station auxiliaries	AOO
Loss of normal feedwater flow	AOO
Inadvertent DHRS actuation(1)	AOO
Feedwater system pipe break inside or outside of containment	Postulated accident
Reactivity and Power Distribution Anomalies	
Uncontrolled control rod assembly bank withdrawal from a subcritical or low power startup condition	AOO
Uncontrolled control rod assembly bank withdrawal at power	AOO
Control rod misoperation(2) Single control rod assembly drop Control rod bank drop Single control rod assembly withdrawal	AOO
Inadvertent decrease in boron concentration in the reactor coolant	AOO
Inadvertent System Operation that Increases Reactor Coolant Inventory	
Chemical and volume control system malfunction that increases reactor coolant system inventory	AOO
Decrease in Reactor Coolant System Inventory	
Failure of small lines carrying primary coolant outside containment	Infrequent event
Steam generator tube failure	Postulated accident

1. Depending on the operating power at which an inadvertent DHRS actuation occurs, it may initially result in either an increase or decrease in heat removal from the RCS.
2. Control rod misoperation includes several types of events; those that require system transient analysis are identified.

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4.2 Design Basis Event Acceptance Criteria

Safety analyses are performed to demonstrate that a nuclear power plant can meet applicable acceptance criteria for a limiting set of AOOs, IEs, and accidents. If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of a plant should be such that events produce about the same level of risk. The acceptance criteria indicated by the GDC for nuclear power plants (Reference 4) reflect the risk of an event. Relatively frequent events such as AOOs are prohibited from resulting in serious consequences, but relatively rare events (postulated accidents) are allowed to produce more severe consequences.

Design basis events for an NPM are categorized as AOOs, IEs, or postulated accidents. Table 4-2, Table 4-3, and Table 4-4 summarize the acceptance criteria applied for AOOs, IEs, and postulated accidents, respectively.

The applicable acceptance criteria identified for each event are based on the event classification as identified in Table 4-1. For a limited number of events, a more conservative acceptance criterion may be applied than required based on the event classification.

For many non-LOCA transient events, the specific acceptance criterion is not challenged during the event progression. For example, events that result in an increase in heat removal from the RCS may have a maximum RCS pressure higher than the initial operating pressure, but does not challenge the margin to the maximum RCS pressure acceptance criterion. In contrast, events that result in a decrease in heat removal from the RCS may result in an RCS pressurization that could challenge the maximum RCS pressure acceptance criterion. In Section 7.2, the acceptance criteria of interest for each non-LOCA event are identified. The acceptance criteria of interest are those where margin to the limit may be challenged during the event progression.

In the event-specific transient analysis, sensitivity calculations are performed to ensure that the event meets acceptance criteria that may be challenged. These sensitivity calculations are performed to confirm that appropriately conservative inputs are specified to identify the case that results in minimum margin to the acceptance criterion of interest. For other acceptance criteria where margin to the limit is not challenged, representative results from the overall scope of sensitivity calculations performed are sufficient to demonstrate that margin to the acceptance criterion is maintained.

A prime example of an acceptance criterion where an NPM design has significant margin is the maximum secondary system pressure. Unlike in typical PWR designs, in a NuScale design, the design pressure of the SG secondary side up to the second containment isolation valves is equal to the RCS design pressure. This feature supports the design and operation of the SG and DHRS. In a non-LOCA event that results in DHRS actuation, typically the maximum secondary side pressure occurs in the first minutes of the transient progression, following DHRS actuation. After DHRS is actuated, the fluid in the DHRS flows into the SG. Heat is transferred from the RCS primary system to the SG, where the DHRS loop inventory boils in the SG tubes. The steam flow is then condensed in the DHRS condensers and the energy is transferred to the reactor pool UHS. The maximum

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pressure in the SG secondary side is limited to the saturation pressure at the temperature of the RCS fluid on the SG primary side. Therefore, the maximum secondary pressure is affected by the secondary side inventory and the primary side conditions at the time of DHRS actuation, and is less sensitive to a specific initiating event. The SG design pressure is significantly higher than pressures expected during DHRS operation. The margin to the SG design pressure is physically limited, based on the primary side conditions. The representative transient results in Section 8.0 demonstrate that significant margin to the maximum SG pressure acceptance criterion is maintained for all types of events. Therefore, extensive sensitivity calculations to maximize secondary side pressure are not necessary for the non-LOCA transient analysis calculations. {{

}}2(a),(c)

Another example is RCS pressure. In the NPM, operation of an RSV at its lift setpoint is sufficient to mitigate an increasing pressure with minimal overshoot. This design feature ensures that the maximum RCS pressure does not significantly exceed the RSV lift setpoint. With appropriate selection of the RSV lift setpoint, including consideration of uncertainty, the maximum RCS pressure will be well below the acceptance criterion. Event analyses where pressure increases during the transient apply biases and include sensitivities to reach conditions where RSV lift may occur. If RSV lift occurs, modification of other biases can generate many cases with similar maximum RCS pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases with RSV lift. If RSV lift does not occur for an event, then RCS pressure will not challenge the acceptance criterion and extensive sensitivity studies are not required to investigate differences in RCS pressure below the RSV lift setpoint. {{

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Table 4-2 Acceptance criteria for anticipated operational occurrences

Parameter	Acceptance Criterion	How Acceptance Criterion is Satisfied
Maximum reactor coolant primary system pressure	≤ 110% of design pressure	Margin to this acceptance criterion is demonstrated by the non-LOCA system transient analysis results.
Maximum main steam secondary system pressure	≤ 110% of design pressure	Margin to this acceptance criterion for the steam system piping up to the second containment isolation valve is demonstrated by the non-LOCA system transient analysis results.
Minimum critical heat flux ratio	> 95/95 critical heat flux ratio (CHFR) Limit	Margin to this acceptance criterion is demonstrated by the subchannel analysis results. Subchannel analysis is outside scope of this topical report.
Maximum fuel centerline temperature	≤ melting temperature (adjusted for burnup effects)	Margin to this acceptance criterion is demonstrated by the subchannel analysis results. Subchannel analysis is outside scope of this topical report.
An AOO should not result in a significant loss of reactor containment barrier	Margins to containment pressure and temperature limits are maintained.	Margin to this acceptance criterion is demonstrated by the peak containment pressure/temperature analysis results performed according to a separate analysis methodology. Maximum containment pressure and temperature analysis is outside scope of this topical report.
An AOO should not generate a postulated accident without other faults occurring independently	A postulated accident is not generated by the AOO.	This acceptance criterion is satisfied by demonstrating that the other acceptance criteria are met.

Table 4-3 Acceptance criteria for infrequent events

Parameter	Acceptance Criterion	How Acceptance Criterion is Satisfied
Maximum reactor coolant primary system pressure	≤ 120% of design pressure	Margin to this acceptance criterion is demonstrated by the non-LOCA system transient analysis results.

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Table 4-3 Acceptance criteria for infrequent events (Continued)

Parameter	Acceptance Criterion	How Acceptance Criterion is Satisfied
Maximum main steam secondary system pressure	≤ 120% of design pressure	Margin to this acceptance criterion for the steam system piping up to the second containment isolation valve is demonstrated by the non-LOCA system transient analysis results.
Fuel cladding integrity	If the minimum CHFR is less than or equal to the 95/95 CHFR limit, the fuel rod is assumed to be failed. If the maximum fuel centerline temperature exceeds the melting temperature, the fuel rod is assumed to be failed.	Margin to this acceptance criterion is demonstrated by the subchannel analysis results. Subchannel analysis is outside scope of this topical report.
Containment integrity	Margins to containment pressure and temperature limits are maintained.	Margin to this acceptance criterion is demonstrated by the peak containment pressure/temperature analysis results performed according to a separate analysis methodology. Maximum containment pressure and temperature analysis is outside scope of this topical report.
Release of radioactive material	Calculated offsite doses are less than 10% of the 10 CFR 52.47(a)(2)(iv) (Reference 5) or 52.137(a)(2)(iv) (Reference 27) reference values.	Margin to this acceptance criterion is demonstrated by the accident radiological analysis results. Accident radiological analysis is outside scope of this topical report.

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Table 4-4 Acceptance criteria for postulated accidents

Parameter	Acceptance Criterion	How Acceptance Criterion is Satisfied
Maximum reactor coolant primary system pressure	≤ 120% of design pressure	Margin to this acceptance criterion is demonstrated by the non-LOCA system transient analysis results.
Maximum main steam secondary system pressure	≤ 120% of design pressure	Margin to this acceptance criterion for the steam system piping up to the second containment isolation valve is demonstrated by the non-LOCA system transient analysis results.
Fuel cladding integrity	If the minimum CHFR is less than or equal to the 95/95 CHFR limit, the fuel rod is assumed to be failed. If the maximum fuel centerline temperature exceeds the melting temperature, the fuel rod is assumed to be failed.	Margin to this acceptance criterion is demonstrated by the subchannel analysis results. Subchannel analysis is outside scope of this topical report.
Containment integrity	Margins to containment pressure and temperature limits are maintained.	Margin to this acceptance criterion is demonstrated by the peak containment pressure/temperature analysis results performed according to a separate analysis methodology. Maximum containment pressure and temperature analysis is outside scope of this topical report.
Release of radioactive material	Release does not result in offsite doses in excess of the guidelines of 10 CFR 52.47(a)(2)(iv) (Reference 5) or 52.137(a)(2)(iv) (Reference 27).	Margin to this acceptance criterion is demonstrated by the accident radiological analysis results. Accident radiological analysis is outside scope of this topical report.

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4.3 Non-LOCA Transient Analysis Process

The main steps of the non-LOCA system transient analysis process for a specific transient event are:

1. Develop a plant base model NRELAP5 input.
2. Adapt NRELAP5 base model as necessary for specific event analysis, and desired initial conditions.
3. Perform steady state and transient system analysis calculations with NRELAP5.
4. Evaluate results of transient analysis calculations:
   a. Confirm margin to maximum RCS pressure acceptance criterion
   b. Confirm margin to maximum SG pressure acceptance criterion
   c. Confirm appropriate transient run execution time
5. Identify which cases provide input for downstream subchannel analysis and extract boundary condition data.
6. Identify which cases provide input for downstream accident radiological analysis and extract boundary condition data.

The non-LOCA system transient analysis is performed and documented in accordance with NuScale's QAP (Reference 3).

The main steps of the non-LOCA system transient analysis are discussed in the following subsections.

4.3.1 Develop a Plant Base Model NRELAP5 Input

NRELAP5 is NuScale's system thermal-hydraulics code used to simulate an NPM system response during non-LOCA short-term transient event progression, for events that require system transient analysis.

The NRELAP5 code was developed based on the Idaho National Laboratory (INL) RELAP5-3D© computer code. RELAP5-3D©, version 4.1.3 was used as the baseline development platform for the NRELAP5 code. RELAP5-3D© was procured and commercial grade dedication was performed by NuScale, and subsequently features were added and changes made to address unique aspects of an NPM design and licensing methodology.

The NRELAP5 code includes models for characterization of hydrodynamics, heat transfer between structures and fluids, modeling of fuel, point reactor kinetics models, and control systems. NRELAP5 utilizes a two-fluid, non-equilibrium, non-homogenous fluid model to simulate system thermal-hydraulic responses. The NRELAP5 code is described in Reference 2. As discussed in Reference 2 and in Section 9.0, the NRELAP5 code has been developed and is maintained within NuScale's QAP.

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The applicability of NRELAP5 for non-LOCA transient analysis is discussed in Section 5.0.

As discussed in Section 4.2, the non-LOCA figures of merit evaluated with NRELAP5 are as follows:

- Primary side pressure
- Secondary side pressure (up to the second containment isolation valve)

In addition, the RCS level response is evaluated with NRELAP5 for non-LOCA events that result in decrease in RCS inventory (SGTF and small line breaks outside of containment) to demonstrate that the inventory decrease is isolated in a timely manner and the DHRS provides effective decay heat removal for these events.

The basis of an NRELAP5 plant base model is described in Section 6.0. An NRELAP5 plant base model is developed for a specific NPM design.

Interfaces with the core design and fuel rod performance design that provide input to the transient analyses are described in the following subsections.

4.3.1.1 Interface with Core Design (Input to the Transient Analysis)

Core design analysis performed in accordance with a methodology approved for a NuScale design provides input to the system transient analysis. The NuScale transient analysis methodology using NRELAP5 can be applied to a typical light water reactor fuel assembly design, and does not require that a specific code or suite of codes be used for the steady state core design analysis. The non-LOCA evaluation model methodology for specifying the input for the reactor kinetics model, axial power shape, and energy deposition factor are described in the following subsections.

4.3.1.1.1 Reactor Kinetics Model

The system transient analyses are performed assuming either beginning of cycle (BOC) or end of cycle (EOC) conditions based on direction of conservatism for reactivity feedback for the transient.

The total core power during a non-LOCA transient is the combination of the fission power and the decay heat. For the non-LOCA transients the fission power response is modeled using the separable point reactor kinetics model in NRELAP5. The NRELAP5 point reactor kinetics model computes both the immediate (prompt and delayed) fission power and the power from decay of fission products.

In NRELAP5, input needed for the separable point kinetics model includes:

- Effective delayed neutron precursor yield of group i (β_i)

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- Effective delayed neutron decay constant of group i (λ_i)
- Prompt neutron generation time (Λ)
- Reactivity feedback
- Reactivity changes from control rod movement (normal controls), or scram
- Decay heat model input

In the non-LOCA evaluation model, the input to the point kinetics model is specified to give a conservatively high power response prior to actuation of reactor scram. The power response is biased by the input accounting for the reactivity feedback effects of moderator temperature, fuel temperature, and the normal control rod movement. After a reactor scram signal, the negative reactivity associated with insertion of the control and safety banks is conservatively modeled as described in Section 7.1.5.

The moderator temperature coefficient (MTC) is a measure of the relative change in reactivity associated with a change in moderator (coolant) temperature. The Doppler temperature coefficient (DTC) is a measure of the relative change in the reactivity as the fuel temperature changes. In the non-LOCA evaluation model, reactivity feedback effects from moderator temperature changes and fuel temperature changes are conservatively bounded as described in Section 7.1.5. Negative reactivity insertion due to void generation impacts on moderator density is conservatively neglected.

As described in Section 6.0 and Section 7.1.2, the rod control system and associated control logic are incorporated into the NRELAP5 model to allow simulation of reactivity changes (negative or positive) associated with normal control rod movement in response to postulated transients. In cooldown events, the normal rod control function attempts to increase average temperature by withdrawing control rods and therefore adding positive reactivity. For transients where operation of the normal rod control function provides more adverse consequences of the transient, the control rod movement is modeled to increase the power response during the transient. A conservative power response is ensured by using maximum rod worth and maximum rod speed for the control rod movement. A conservative power response is also ensured when using nominal rod speed if the 'insert only' mode is neglected as described in Section 6.3.1.3.

The decay heat power contribution is conservatively bounded high or low, as appropriate for the specific transient, as described in Section 7.1.5.

Appropriate input based on the core design is used for other parameters needed as input to the point reactor kinetics model.

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4.3.1.1.2 Axial Power Shape

For the system transient analysis, a single channel core model is used, as described in Section 6.0. A nominal center-peaked average axial power shape is input for the single core channel for consistency with development of the reactivity feedback coefficients determined in the core design analyses.

Uncertainties associated with the axial power shape and axial and radial power peaking factors that can affect the MCHFR and peak centerline fuel temperature are accounted for in the downstream subchannel analyses as described in Reference 6, supplemented by Reference 28. The design features of an NPM preclude challenge to the primary and secondary pressure acceptance criteria as discussed in Section 7.0 and as shown in the representative results in Section 8.0. 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, use of a nominal center-peaked average axial power shape input is appropriate for the system transient analyses.

4.3.1.1.3 Energy Deposition Factor

The energy deposition factor is the portion of the energy generated in the core that is directly deposited in the fuel. A bounding high energy deposition factor that results in all energy being deposited in the fuel is used in the non-LOCA analyses. For cooldown or reactivity insertion events that cause total core power changes, increasing the core energy deposited in the fuel slows the thermal-hydraulic response of the system to changing power levels. The effect of the energy deposition factor on the primary system pressure response is generally insignificant except in very fast reactivity events such as control rod ejection where Doppler reactivity feedback is important; control rod ejection analysis is outside the scope of this EM. Sensitivity studies on this parameter confirm that its effect on the system response is not significant with respect to demonstrating margin to acceptance criteria for NPM events addressed by the non-LOCA evaluation model due to the MPS design and selection of the analytical limits to actuate reactor trip.

4.3.1.2 Interface with Fuel Rod Performance Design (Input to the Transient Analysis)

Fuel rod design analysis performed using a fuel performance code approved for a NuScale design provides input to the system transient analysis. The NuScale transient analysis methodology using NRELAP5 can be applied to a typical light water reactor fuel assembly design, and does not require that a specific code or suite of codes be used for the fuel performance analysis. The fuel assembly geometry, required material properties, and the fuel performance data needed for appropriate steady state initialization of the model are specified as input to the transient system analysis.

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The NuScale transient analysis requires input for the fuel rod and assembly geometry, fuel rod thermo-mechanical properties, and fuel rod performance information to assure adequate modeling of the initial fuel rod stored energy and transient response.

4.3.1.2.1 Fuel Rod and Fuel Assembly Geometry

Nominal fuel rod and fuel assembly geometry information forms the basis for the NRELAP5 inputs required to describe the core heat structures, core coolant channels, and bypass channels. Pressure losses due to fuel assembly grid spacers and wall friction are accounted for. The NRELAP5 model is described in Section 6.0.

4.3.1.2.2 Fuel Rod Material Properties

Fuel rod material properties appropriate for the fuel assembly design are specified by user input in the NRELAP5 model. Fuel rod material properties needed include:

- fuel pellet thermal conductivity
- fuel pellet specific heat
- fuel pellet density
- cladding thermal conductivity
- cladding specific heat
- cladding density

Nominal fuel rod material properties as a function of temperature are specified.

For the fuel pellet properties, UO_2 properties are used to reflect the core average response. Since the fuel pellet thermal conductivity is a function of burnup and fuel temperature, and the fuel pellet thermal conductivity degrades with burnup, a representative time-in-cycle core average burnup is used to calculate the fuel thermal conductivity as a function of temperature.

4.3.1.2.3 Fuel Rod Performance Data

The fuel rod gap conductance, specific heat and density in the NRELAP5 fuel rod heat structure are used to set the initial core average fuel temperature. Bounding values for fuel rod gap conductance are selected to provide conservatively high or low core average fuel temperature for the time-in-life of interest for the calculation.

The conservative core average fuel temperatures are confirmed on a cycle-specific basis.

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4.3.2 Adapt Plant Base Model NRELAP5 Input for Event Transient Analysis

The basis of an NRELAP5 plant base model is described in Section 6.0. An NRELAP5 plant base model is adapted as necessary to perform the specific event analysis. Section 7.0 describes the non-LOCA analysis methodology, including conservative biasing of initial and boundary conditions, single failures, and loss of power scenarios for the event analyses.

4.3.3 Perform NRELAP5 Steady State and Transient System Analysis Calculations

For each analysis, one or more steady state initialization calculations are developed. After acceptable steady state conditions are obtained, the transient calculations are performed. Section 7.0 describes the non-LOCA analysis methodology including the steady state initialization and performance of null transients prior to the transient calculation.

4.3.4 Evaluate Results of Transient Analysis Calculations

The results of the transient analysis calculations are assessed to confirm that the system transient response is acceptable.

4.3.4.1 Maximum Pressure Acceptance Criteria

Based on the results of the transient analysis calculations performed for an event, margin to the maximum RCS pressure acceptance criterion and margin to the maximum SG secondary pressure acceptance criterion are confirmed.

4.3.4.2 Short-Term Transient Duration

For each transient calculation, the following parameters are reviewed to assure that the transient calculation is executed for an appropriate duration to confirm that design basis event acceptance criteria are met.

- MPS actuations expected in direct response to the initiating event, for mitigation of the design basis event, have occurred
- if reactor trip occurs during the transient, the nuclear heat source 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

These conditions are demonstrated for a reasonable time, typically a few hundred seconds, following the last safety system actuation expected to occur in the short term transient progression, to demonstrate that core cooling is established and conditions that result in minimum margin to the acceptance criteria have occurred.

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

For NPM non-LOCA events, VIPRE-01 is used for performance of subchannel analysis calculations. Reference 6, supplemented by Reference 28, describes the subchannel analysis methodology.

In the transient subchannel analysis, the following acceptance criteria are assessed:

- MCHFR
- maximum fuel centerline temperature

For non-LOCA events that require subchannel analysis, the NRELAP5 transient analyses provide the following input from the system transient calculation to the downstream subchannel analysis:

- reactor power as a function of time
- core exit pressure as a function of time
- core inlet temperature as a function of time
- total system flow rate as a function of time

In the NRELAP5 system transient analysis, cases for downstream subchannel analysis are identified based on the conservative bias directions for the boundary condition input and considering an NPM natural circulation design. The conservative bias directions are discussed below, followed by a description of the methodology for identifying cases for downstream subchannel analysis.

As identified in Reference 6, supplemented by Reference 28, for the system transient parameters provided by NRELAP5, the conservative bias directions to minimize the CHFR are as follows:

- maximum reactor power (higher power increases the actual heat flux)
- maximum core inlet temperature (higher temperature reduces energy addition needed to raise coolant to saturated conditions)
- minimum system flow rate (minimum flow is conservative as there is less coolant flow in the reactor core available for heat transfer)

The effect of pressure on critical heat flux (CHF) is established by varying initial pressurizer pressure to determine the appropriate direction for biasing for events that are challenging for margin to CHF.

In a NuScale design, for a given reactor module operating condition, reactor power, core inlet temperature and system flow rate are tightly coupled. As described in Section 7.1, ranges in these parameters are considered as part of biasing the system transient analysis steady state initial conditions. The NRELAP5 system analysis methodology for determining the limiting CHF cases for downstream subchannel analysis is primarily dependent on the limiting initialization. The CHF cases are

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evaluated at the minimum flow initialization. Other initial conditions are forced to the limiting initialization for a given transient progression to ensure the maximum power and core inlet fluid temperature are reached prior to reactor trip system actuation. For example, in the case of a heatup event, the RCS increases in temperature, causing a pressurizer insurge and subsequent increase in pressure. The limiting CHF scenario is the transient progression that results in the highest core outlet temperature at the time of reactor trip on high pressure, which is generally the faster heatups where the pressurizer initialization is biased to delay the high pressure trip.

For some transients, a spectrum of cases is analyzed from the limiting initialization.
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After the system transient analysis calculations are performed and assessed, for events that require subchannel analysis, a number of cases are identified as limiting for MCHFR. For the limiting cases selected, the required system transient parameters are tabulated as a function of time for input to the downstream subchannel analysis calculations, to calculate margin to CHF.

The system transient parameters are provided for subchannel analysis for sufficient time for the subchannel analyses to demonstrate that the MCHFR has occurred, typically 10-15 seconds following reactor trip.

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Figure 4-1 {{

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Figure 4-2 {{

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Figure 4-3 {{ }}2(a),(c)

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4.3.6 Identification of Cases for Accident Radiological Analysis

Radiological acceptance criteria are assessed in the accident radiological analyses. Reference 8 describes the NuScale accident radiological source term analysis methodology. This section only describes radiological analyses cases that involve nuclear steam supply system transients (i.e., primary and secondary coolant system transients) and does not encompass any radiological analyses involving spent fuel movement or postulated failures in the radioactive waste system.

The transient analyses can be used to provide input to the accident radiological analyses for events that result in RCS fluid loss outside of containment such as a break of small RCS piping outside of containment or SGTF. For these events, one or more transient analysis cases are identified to provide conservative input to the

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accident radiological analysis. The conservative bias directions for the transient analysis input to the accident radiological analysis are:

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

  Basis: For a constant radionuclide concentration in the RCS, the greater the mass released outside of containment, the more severe the radiological consequences.

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

  Basis: Reference 8 describes how iodine spiking is accounted for in the accident radiological analyses. During a transient progression, changes in reactor power, RCS average temperature, or RCS pressure could result in iodine spiking, changing the radionuclide concentration in the RCS. Consistent with Standard Review Plan Section 15.6.2 (Reference 9), for accident radiological calculations assuming a coincident iodine spike (iodine spike occurring during the event), the iodine spiking is assumed to begin at the time of reactor trip as the result of the reactor shutdown or depressurization of the primary system. In some cases, particularly for smaller breaks in RCS piping, the time between reactor trip and isolation of the break flow could be extended compared to larger break sizes. The increased time between reactor trip and isolation increases the time of mass release when the RCS radionuclide inventory reflects iodine spiking.

For each transient analysis case identified, the input provided for accident radiological analysis includes:

- Time of reactor trip if it is calculated to occur
- Isolation time, at which point release of RCS fluid outside of containment is stopped
- RCS fluid mass release outside of containment as a function of time
- System transient response parameters such as
  - Reactor power as a function of time
  - RCS average temperature as a function of time
  - RCS pressure as a function of time
  - Secondary side feedwater and steam flow rates as a function of time

As an alternative to transient analysis, the accident radiological analysis can use bounding values for both mass release and isolation times. An example bounding approach is provided as follows. If the MPS of an NPM design includes reactor trip and isolation setpoints based on pressurizer level, these setpoints effectively limit the RCS inventory that can be released to the difference between a high pressurizer level and the pressurizer level associated with the isolation, and limit the time between reactor trip actuation and completion of isolation valve closure. A bounding mass release based on the MPS pressurizer level setpoints should also include the additional mass released between the time the setpoint is reached and the time

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isolation is complete, as well as account for pressurizer level uncertainty. Using a bounding mass release and isolation time, the accident radiological analysis can be completed independent of system transient analyses; however, the system transient analyses can still be used to confirm the assumptions are bounding. This alternative approach results in more limiting accident radiological analysis results and is therefore typically used when the accident radiological analysis results are expected to have margin.

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5.0 NRELAP5 Applicability for Non-LOCA Transient Analysis

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

5.1.1 Phenomena Identification and Ranking Table Process

The PIRT process is a systematic way of gathering information from experts on a specific subject and ranking the importance of the information to meet some decision-making objective. It has been applied to many nuclear technology issues to help guide research and develop activities to satisfy regulatory requirements.

The purpose of the NuScale non-LOCA PIRT is to provide an assessment of the relative importance of phenomena and processes that may occur in a NuScale Power Module (NPM) during non-LOCA events in relation to specified figures of merit (FOMs). This assessment is part of the process prescribed by Regulatory Guide 1.203 (Reference 1).

The non-LOCA PIRT was developed by a panel of experts for an NPM and was built upon the state-of-knowledge at the time of its development. Non-LOCA events can be divided into several different event types based on the main effect on the reactor coolant system, as described in Section 5.1.2. A comprehensive, integrated PIRT was performed for the range of non-LOCA event types and phases of the event progression. The PIRT panel considered an NPM design to identify systems, components, and subcomponents of its design for which phenomena were assessed. The panel then followed the PIRT process to identify and rank phenomena considering the level of importance for each phenomena relative to identified FOMs for the different non-LOCA event types and phases of the transient progression. The panel also established a knowledge ranking for each of the phenomena.

Following the development of the PIRT, additional insights from testing, code validation, plant calculations and analysis were established. The following discussions of the PIRT phenomena, importance, knowledge rankings, and how the phenomena are addressed reflect these developments as appropriate.

The PIRT was originally developed based on the specific NPM design in development at that time. Subsequent design work has resulted in other NPM designs within the confines of the general NPM description in Section 3.0. An assessment of the applicability of the PIRT to other NPM designs was performed. The assessment included comparisons between NPM designs of: the normal operating conditions at full power; design limits; and the geometric parameters. The results of the assessment identified no new phenomena that are applicable during the non-LOCA event phases. The assessment concluded that the non-LOCA PIRT remains applicable to NPM designs that are consistent with the description in Section 3.0. Further insights from the assessment are included in the discussions of the PIRT in the following sections.

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5.1.2 Non-LOCA Event Scenarios and Phases

As described in Section 4.1, the non-LOCA events for an NPM are divided into five event categories based on the type of effect on the RCS. These categories are summarized below.

1. increase in heat removal from the reactor coolant system
2. decrease in heat removal by the secondary system
3. reactivity and power distribution anomalies
4. increase in reactor coolant inventory
5. decrease in reactor coolant inventory

For the non-LOCA PIRT, the panel evaluated five design-basis non-LOCA events, one representative of each category:

- Main steam line break inside containment: Representative of events that result in an increase in heat removal from the RCS (cooldown/depressurization events)
- Feedwater line break inside containment: Representative of events that result in a decrease in heat removal from the RCS (heatup/pressurization events)
- CRA withdrawal: Representative of events that result in a reactivity increase
- CVCS malfunction: Representative of events that result in an increase in RCS inventory
- SGTF: Representative of events that result in a decrease in RCS inventory

The PIRT panel divided the short-term, non-LOCA event progression into three phases, generic for each transient type:

Phase 1 - Pre-trip transient

Phase 1 begins with the event initiation. The RCS power, pressure and flow rates increase or decrease from the normal power conditions, depending on the event type. The DHRS is inactive. This phase ends with actuation of the MPS response to the off-normal NPM conditions.

Phase 2 - Post-trip transition

Phase 2 begins with MPS actuation of reactor trip and, for most non-LOCA events, actuation of the DHRS in response to faulted secondary heat removal conditions. In cases where DHRS is actuated, the DHRS actuation valves open and the normal secondary side flow paths from the feedwater system and to the main steam system (MSS) are isolated. The secondary side DHRS loop pressurizes as decay and residual heat are transferred from the primary, causing the DHRS loop inventory to boil off in the steam generator and condense in the DHRS condenser. The reactor coolant system power and flow rates transition towards decay heat levels.

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Phase 3 - Stable natural circulation

During phase 3, stable primary side natural circulation conditions exist with reactor coolant system power and flow rates reflecting decay power levels. In cases where DHRS is actuated, stable natural circulation in the DHRS loop is established. Secondary side flow rates and pressures in the DHRS loop decrease as the primary side pressure and temperature decrease.

Section 4.3 describes criteria for the short-term non-LOCA transient end conditions.

5.1.3 Phenomena Identification and Ranking Table Figures-of-Merit and Phenomenon Ranking

The PIRT panel identified figures-of-merit for each phase of the non-LOCA transient. The FOMs reflect the non-LOCA event acceptance criteria (Section 4.2) and important factors relative to an NPM design.

Phase 1 - Pre-trip transient

- CHF - Demonstrating that margin to CHF is maintained for AOOs, or identifying the number of fuel rods that exceed CHF for IEs or postulated accidents is a primary non-LOCA acceptance criterion. The non-LOCA event analyses confirm that the plant system design is such that margin to CHF is maintained until the module protection system actuates to mitigate the event.

  In an NPM design, margin to CHF may be challenged during cooldown events or reactivity insertion events.
- Primary pressure - The maximum primary system pressure is one of the non-LOCA acceptance criteria to demonstrate acceptable RCS performance. The non-LOCA event analyses confirm that the plant system design is such that margin to the maximum primary system pressure limits is maintained until the module protection system actuates to mitigate the event.

  In an NPM design, margin to the maximum primary system pressure limits may be challenged during heatup events or events that increase RCS inventory.

Phase 2 - Post-trip transition

- CHF - See above.
- Primary pressure - See above.
- Secondary pressure - The maximum secondary system pressure is one of the non-LOCA acceptance criteria to demonstrate acceptable RCS performance.

  In an NPM design, as the DHRS is actuated in Phase 2, the maximum secondary side pressure in the SG can occur.
- Containment pressure - The maximum containment pressure is an indicator of containment integrity.

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In an NPM design, vapor released into containment is condensed on the containment wall and heat is transferred through the containment wall directly to the reactor pool portion of the UHS. Non-LOCA events could result in mass and energy release into containment. The containment design is intended to effectively transfer heat to the UHS so that containment integrity is maintained.

Phase 3 - Stable natural circulation

- CHF - See above.
- Coolant mixture level - The RCS primary mixture level represents the boundary between a two-phase or single-phase liquid region below the level and a single-phase vapor region above the level. This level indicates whether the primary side natural circulation flow path is maintained.

  In an NPM design, depending on the initial and boundary condition assumptions such as initial RCS level and temperature, decay heat, and number of DHRS trains operating, the DHRS heat removal may provide sufficient cooling of the RCS that the increased primary side liquid density results in coolant volume shrinkage sufficient to decrease the RCS water level to below the top of the riser, resulting in interrupted natural circulation. If interruption of natural circulation occurs, it is the end of Phase 3, and it occurs well after reactor trip and engineered safety features have responded to the initiating event.
- Subcriticality - Maintaining subcriticality following reactor trip limits the nuclear fuel heat source to decay heat levels. If the soluble boron concentration in the core is significantly reduced following reactor trip and the MPS response to the initiating event, then maintaining subcriticality could be adversely affected. While RCS cooldown following reactor trip also affects the net reactivity, the non-LOCA analysis methodology is applicable to DHRS cooling shutdown evaluations only when the mixture level is above the top of the riser and primary side natural circulation flow is maintained.

  In an NPM design, the boron in the primary system during Phase 3 is limited to the soluble boron at the RCS critical boron concentration from normal operating conditions. During Phase 3 of the non-LOCA event progression, primary side natural circulation flow is maintained. The addition of supplemental boron by the ECCS, if applicable to the design, is outside the non-LOCA Phase 3 scope and is assessed separately (Reference 26).

Each phenomenon identified in the PIRT was assigned an importance ranking and knowledge level ranking. Table 5-1 and Table 5-2 describe the importance rankings and the knowledge level rankings considered by the PIRT panel.

Table 5-1 Importance rankings

Importance Ranking	Definition
High (H)	Significant influence on FOM
Medium (M)	Moderate influence on FOM
Low (L)	Small influence on FOM
Inactive (I)	Phenomenon not present or negligible

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Table 5-2 Knowledge levels

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

5.1.4 Highly Ranked Phenomena

The following subsections summarize the phenomena listed in Table 5-3 that were ranked high importance by the PIRT panel in at least one of the three phases of the non-LOCA short-term transient response scenarios. The knowledge level assigned by the PIRT panel, and the systems and components where the phenomenon was ranked as high importance are also included.

The non-LOCA PIRT is a comprehensive, integrated PIRT covering the range of non-LOCA event types and the phases of event progression. The NPM systems, components, and relevant phenomena are considered in detail.

As discussed in Section 1.1, NRELAP5 is the system thermal-hydraulics code used to simulate an NPM system response during the non-LOCA short-term transient event progression. The NRELAP5 assessments performed as part of the development of Reference 2 demonstrate the capability of the code to simulate an NPM response to LOCA events.

The high-ranked phenomena identified by the PIRT process for the non-LOCA transients were evaluated with respect to the high-ranked phenomena identified by the PIRT process for the LOCA scenarios, and the code assessments performed as part of the development of Reference 2. A gap analysis was performed to identify high-ranked phenomena for non-LOCA transients that were not assessed as part of the development of Reference 2.

High-ranked phenomena for non-LOCA events which were not assessed as part of the development of Reference 2 were addressed in a variety of ways that included:

1. Additional NRELAP5 code assessments performed against separate effects or integral effects test data.
2. Code-to-code benchmark performed between NRELAP5 and an independent system thermal-hydraulics code.
3. The phenomenon was addressed as part of the downstream subchannel analysis.
4. The phenomenon was addressed by specifying appropriately conservative input to the system transient analysis.

The following subsections also describe the means by which each high-ranked phenomenon for non-LOCA events was assessed. In this instance, the NRELAP5 assessments performed as part of the development of Reference 2 were combined

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with one or more of the means described above to qualify NRELAP5 for the high rank non-LOCA PIRT phenomena. Specific details related to the qualification of NRELAP5 include:

1. The NRELAP5 assessment against Korea Advanced Institute of Science and Technology (KAIST) high pressure condensation data, as presented in Reference 2, is described in Section 5.3.1 for convenience.
2. Separate effects testing of the full-length DHRS was performed at the NIST-1 facility and the NRELAP5 assessment is summarized in Section 5.3.2.
3. An integral effects test of an NPM response to a decrease in heat transfer from the secondary side, and integral effects test of DHRS operation was performed at the NIST-1 facility, and is summarized in Section 5.3.3.
4. A code-to-code benchmark was performed to assess the NRELAP5 prediction of an NPM response to reactivity insertion events as described in Section 5.3.4.
5. The NRELAP5 assessments against the SIET data (TF-1 and TF-2) and other legacy experiments are summarized in Reference 2. Due to the importance of heat transfer through the SG as part of the DHRS loop for non-LOCA transients, further review of these assessments is documented in Section 5.3.5.
6. A computational fluid dynamics model was used to evaluate heat transfer between primary reactor coolant and the SG helical coil tubes. Results of the simulation were compared to the ESDU model implemented in NRELAP5, as described in Section 5.3.6.
7. An integral effects test of an NPM response to a decrease in heat transfer from the secondary side, including DHRS operation, was performed at the NIST-2 facility, as is summarized in Section 5.3.7.

Table 5-3 High-ranked phenomena for non-LOCA events

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Table 5-3 High-ranked phenomena for non-LOCA events (Continued)

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

The information in this section is moved to Section 5.1.4. Evaluation results of high ranked phenomena and text discussing how they are addressed are incorporated into the discussion of each of the phenomenon listed in Section 5.1.4. This section is retained to maintain accuracy in other documents that may have referenced material from this document in sections located after this section.

5.3 NRELAP5 Validation and Assessments for Non-LOCA

In Section 5.1 and Section 5.2, the high-ranked phenomena from the non-LOCA PIRT are summarized and how they are assessed in the non-LOCA evaluation model is identified. As identified in Table 5-3, additional assessments were performed to support the qualification of the NRELAP5 code for some of the non-LOCA high-ranked phenomena. This section summarizes the additional assessments performed. This section also discusses a limited number of assessments performed as part of the LOCA evaluation model development that demonstrate qualification of NRELAP5 for prediction of heat transfer from the RCS to the SG to the DHRS.

Heat transfer from the RCS to reactor pool via the SG and the DHRS is important for prediction of non-LOCA transient progression after DHRS actuation, to demonstrate heat removal from the core. The NRELAP5 prediction of heat transfer in the DHRS is assessed with the KAIST high-pressure condensation experiments (Section 5.3.1), and with separate effects testing of the full-length DHRS at the NIST-1 facility (Section 5.3.2). The KAIST data was assessed as part of the LOCA evaluation model development (Reference 2) and key results are summarized in this report. The NRELAP5 code validation for the helical coil SG was assessed as part of the LOCA evaluation model (Reference 2), with testing performed at the SIET facility and other legacy experiments. The operating ranges expected during the non-LOCA transients are assessed to demonstrate the adequacy of the validation for the range of conditions expected in the non-LOCA transients (Section 5.3.5).

Integral effects testing was performed at the NIST-1 test facility to support the NRELAP5 validation for non-LOCA events (Section 5.3.3). The testing assessed primary side pressurization and heat-up due to a decrease in heat transfer from the secondary side, and primary and secondary side conditions during operation of the scaled integral DHRS.

In addition to the validation against test data, a code-to-code benchmark against the RETRAN-3D code (Reference 12) was performed to assess the NRELAP5 prediction of an NPM power and integral primary system response to reactivity insertion events (Section 5.3.4).

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Computational fluid dynamics was used to independently assess the heat transfer correlation incorporated into NRELAP5 for the helical SGs of NuScale designs (Section 5.3.6).

Integral effects testing was performed at the NIST-2 test facility to support the NRELAP5 validation for non-LOCA events (Section 5.3.7). The testing assessed SG and DHRS performance during a decrease in heat transfer from the secondary side. Analysis of NPM transient response extends the applicability of the DHRS performance to small break LOCA sequences before ECCS actuation.

Agreement between code predictions and data or for the code to code comparison is assessed according to the criteria described in RG 1.203 (Reference 1):

Validation Requirements:

"Excellent Agreement" applies when the code exhibits no deficiencies in modeling a given behavior. Major and minor phenomena and trends are correctly predicted. The calculated results are judged to agree closely with data.

"Reasonable Agreement" applies when the code exhibits minor deficiencies. Overall, the code provides an acceptable prediction. All major trends and phenomena are predicted correctly. Differences between calculated values and data are greater than are deemed necessary for excellent agreement.

"Minimal Agreement" applies when the code exhibits significant deficiencies. Overall, the code provides a prediction that is not acceptable. Some major trends or phenomena are not predicted correctly, and some calculated values lie considerably outside the specified or inferred uncertainty bands of the data.

"Insufficient Agreement" applies when the code exhibits major deficiencies. The code provides an unacceptable prediction of the test data because major trends are not predicted correctly. Most calculated values lie outside the specified or inferred uncertainty bands of the data.

Note that the results of assessments previously performed using the version of NRELAP5 associated with Reference 25 (or earlier) are retained throughout Section 5.3. The assessments were re-performed using the Reference 29 version of NRELAP5. A limited selection of the results of the updated assessments are provided. The heat transfer correlation assessments using NRELAP5 provided in Section 5.3.6.3 and the NIST-2 assessments provided in Section 5.3.7 were performed with the Reference 29 version of NRELAP5. The Reference 31 version of NRELAP5 added features and made corrections (relative to the Reference 29 version) that do not impact the non-LOCA EM. Testing associated with the release of the Reference 31 version of NRELAP5 confirmed no impact on the non-LOCA EM. Therefore, updated assessment results using the Reference 31 version of the non-LOCA EM are not included in Section 5.3.

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5.3.1 KAIST

The NRELAP5 prediction of high pressure condensation experimental data produced at the KAIST experimental facility was assessed as part of the LOCA EM development (Reference 2). The phenomenon addressed with the KAIST experimental cases provides test data for condensation inside tubes and heat transfer across tubes for the DHRS. As part of demonstrating qualification of NRELAP5 for calculation of DHRS heat transfer rates, the assessment of NRELAP5 against the KAIST tests is summarized. The data from the KAIST experiment are qualified for use by applying non-mandatory guidance provided by NQA-1 2008/2009 Addendum (Reference 20).

5.3.1.1 Facility Description and Range of Experimental Data Assessed

A schematic of the KAIST experimental facility is shown in Figure 5-1. The maximum design pressure and temperature of the test facility were 7.5 MPa (1088 psia) and 300 degrees C (572 degrees F), respectively.

The major components of the test facility include: SG, which supplied steam (maximum power 200 kW, test section tube, the cooling pool (cools the test section), steam line (transports steam from SG to the test section inlet), condensate drain line, lower plenum (or condensate collection tank) and air supply system. The test section was immersed in the cooling pool and was cooled by boiling and single-phase convective heat transfer on the outside surface of the test section (Reference 24).

The test section was a vertical tube with an inside diameter of 4.62 cm and an effective heat transfer length of 1.8 m. The thickness of the tube wall was 2.3 mm. To reduce the entrance effect, the top 0.5 m length of the test section was insulated. The test section was submerged in the cooling pool (1.2 m x 1.2 m x 2.5 m). A steam line with an inside diameter 2.34 cm was connected from the top of the SG to the top of the test section. The condensate from the test section was drained to the lower plenum (or condensate collection tank) by gravity and then pumped back to the SG.

Table 5-4 summarizes the KAIST and NPM decay heat removal system tube geometry. Table 5-5 summarizes the operational range covered by the KAIST experiment and in the NPM decay heat removal system. Table 5-6 shows a comparison between the NRELAP5 discretization between the KAIST and NPM decay heat removal system condenser tubes. Based on Table 5-4, Table 5-5, and Table 5-6 it is concluded that the geometry and operating range comparisons are acceptable for validation. {{

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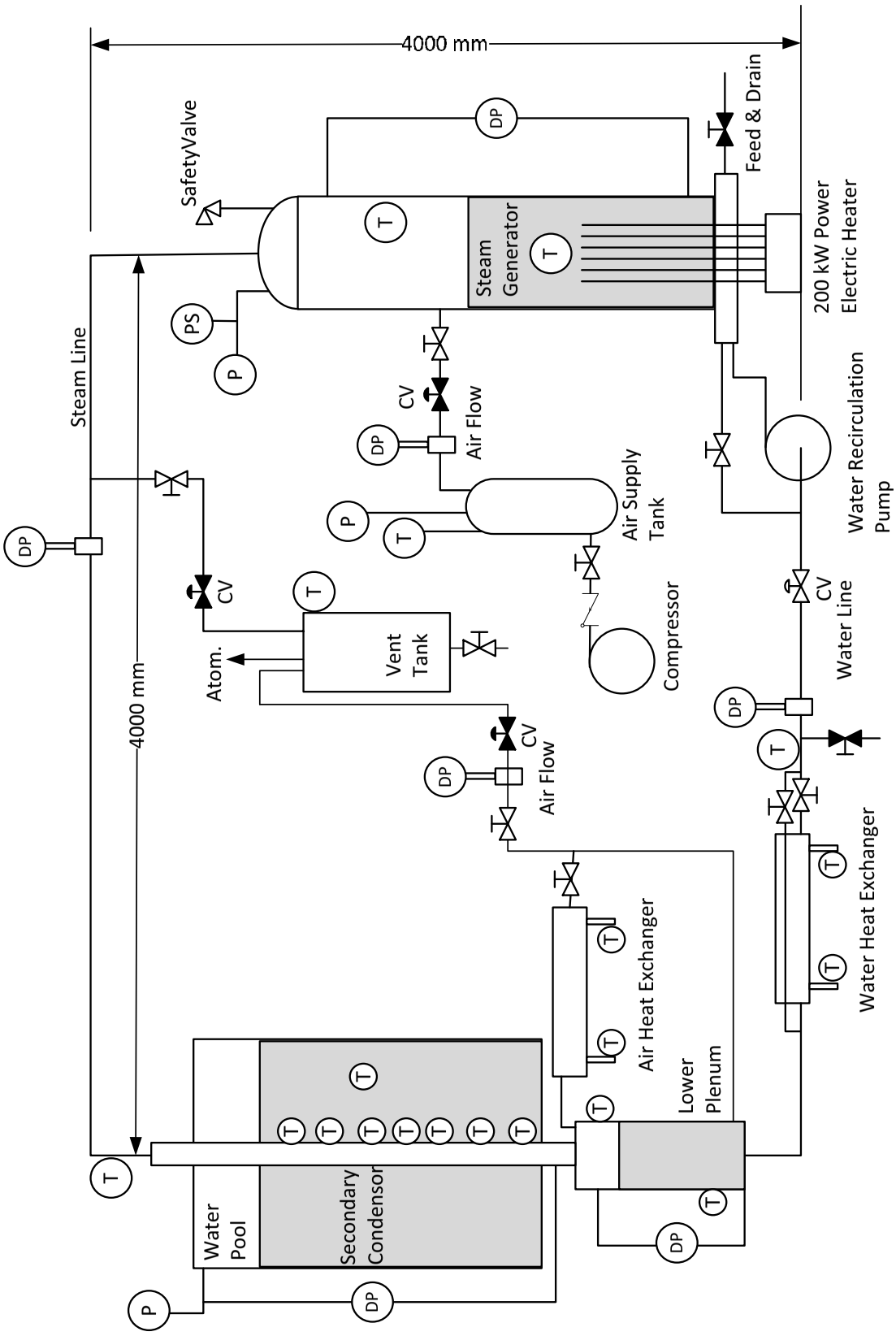


Figure 5-1 Schematic of KAIST test facility

Table 5-4 Comparison between NuScale Power Module decay heat removal system and KAIST test section dimensions

Parameter	KAIST Test Section	NPM DHRS
Geometry	Single circular cross-section tube	For a single DHRS condenser tube
ID, cm (in)	4.62 (1.818)	2.79 (1.097)
OD, cm (in)	5.08 (1.91)	3.34 (1.315)
Active length, m (ft)	1.8 (5.9)	2.648-2.874 (8.7-9.4 ft)
Wall thickness, mm (in)	2.3 (0.091)	2.8 (0.109)
Material	Stainless steel	Stainless steel

Table 5-5 Comparison between NuScale Power Module decay heat removal system and KAIST range of operations

Parameter	KAIST	NPM DHRS(1)
SG pressure, psia (MPa)	155.3 to 1059.5 (1.071 to 7.305)	{{ }}2(a),(c)
Steam flow rates lbm/hr (kg/s)	79.37 to 793.7 (0.01 to 0.1)	{{ }}2(a),(c)
SG steam temperature F (C)	362.12 to 552.2 (183.4 to 289)	{{ }}2(a),(c)

1. Approximate range expected for DHRS operation during the non-LOCA short-term response
2. Flow rate per tube

Table 5-6 Comparison between NuScale Power Module decay heat removal system and KAIST NRELAP5 model nodalization

Parameter	KAIST Condenser Tube NRELAP5 Nodalization	NPM DHRS Condenser Tube NRELAP5 Nodalization
L/D	3.788 to 6.494	{{ }}2(a),(c)
Total number of nodes	8	{{ }}2(a),(c)

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5.3.1.2 Experimental Procedure

Conditions into the condenser tubes and pool were specified. The experiments were started by purging all non-condensable gas (i.e., air) from the test loop. This purge was done by supplying steam to the test loop and venting it to the atmosphere through the vent valve located below the test section. After all non-condensable gas was purged, the vent valve was closed and the test section was allowed to fill with the condensate by keeping the condensate drain valve closed. After the test section was completely filled, the SG pressure was increased to the test pressure. As soon as the test pressure was reached, the condensate drain valve was opened and the condensate recirculation pump was started. A constant water level in the lower plenum was maintained by control of the recirculation pump flow rate. Data acquisition was started after the process had reached a steady state.

5.3.1.3 Assessment Results

The comparison results of condensed liquid flows, heat transfer coefficients, and inner wall temperatures show reasonable to excellent agreement between the calculated NRELAP5 and the KAIST measured experimental data. This agreement is a result of implementation of the 2009 extended Shah correlation in NRELAP5, which is intended to improve the predicted high pressure condensation response.

Figure 5-2 presents a summary of the measured versus calculated assessments of the KAIST steam condensation experiments. The overall majority of the assessments lie within the experimental uncertainty (28 percent for heat transfer coefficient). Figure 5-2a presents the same comparison as Figure 5-2, but with the Reference 29 version of NRELAP5.

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Figure 5-2 Measured vs predicted heat transfer coefficient

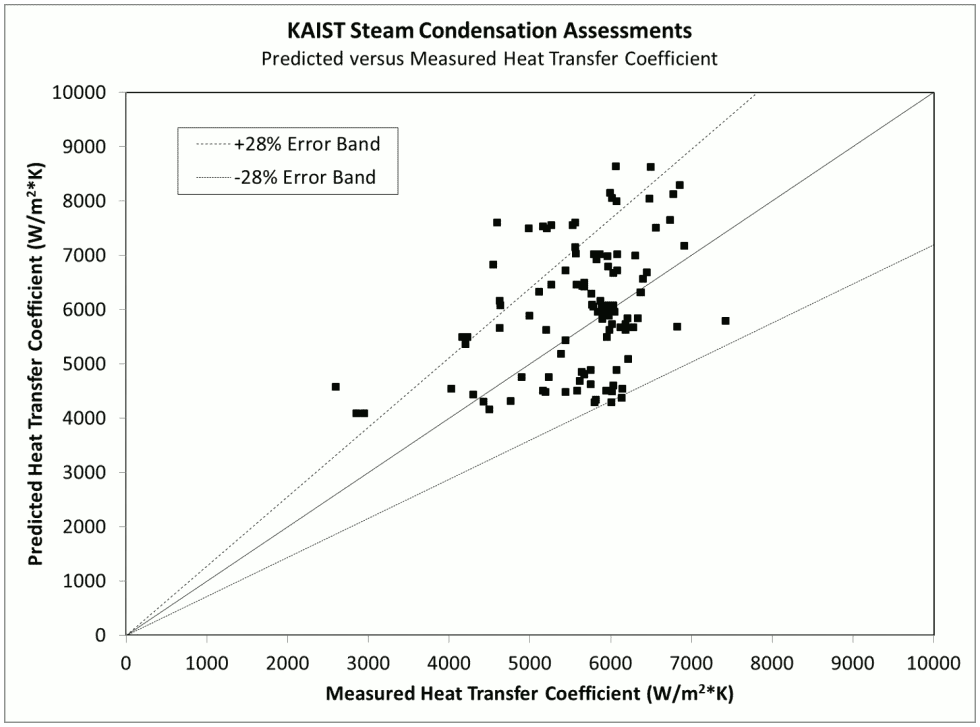
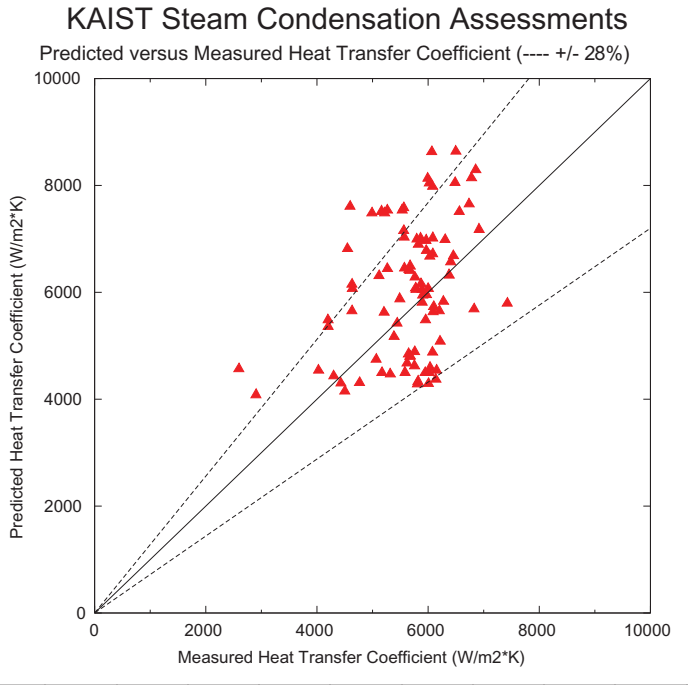


Figure 5-2a Measured vs predicted heat transfer coefficient



5.3.2 NIST-1 Decay Heat Removal System Separate Effects Tests

5.3.2.1 NIST-1 Facility

An NPM design relies on natural circulation flow to remove energy produced in the core. Energy is transferred to the secondary side as the primary coolant flows down over the SG tube coils. During off-normal transients, an NPM design relies on natural circulation driven by steam condensation within the CNV or the DHRS to cool the RPV. Energy within the CNV is transferred through the CNV walls into the surrounding UHS, represented by the reactor cooling pool. An NPM containment vessel is designed to accept and promote steam condensation at pressures varying from vacuum to maximum design pressures.

Due to the unique nature of an NPM design the number of IET facilities is limited. A scaled facility of an NPM was constructed at Oregon State University (OSU), referred to as the NuScale Integral System Test-1 (NIST-1) facility, to assist in validation of the NRELAP5 system thermal-hydraulic code. The NIST-1 facility is a scaled facility of an NPM and consists of the major components in an NPM. These components include: an RPV, a helical coil SG system with a DHRS, a CNV, and a cooling pool vessel (CPV).

The NIST-1 facility was originally conceived at OSU in 2000 as a proof-of-concept testing platform for development of small modular reactor (SMR) technology. During this period it was referred to as the multi-application small light water reactor (MASLWR) facility (Reference 11).

Although the NuScale design was based on MASLWR, the concept has evolved considerably since NuScale's inception in 2008. At the time that NuScale was formed, the facility was renamed the NIST facility. The NIST facility is a scaled, non-nuclear reactor that uses electric heater rods to analogously represent the heat produced from fission. It is designed to produce experimental data in support of verification and validation of thermal-hydraulic codes.

In 2014 and 2015, the original NIST facility was modified by NuScale to necessitate accurate simulation and to bring the facility in-line with the NuScale plant design configuration at that time. A scaling analysis was employed for design of the NIST test facility to ensure that the facility design is capable of capturing important plant phenomena with minimal distortions. Following the upgrade, the NIST facility became the NIST-1 facility.

Updates to the NIST facility included in NIST-1 are:

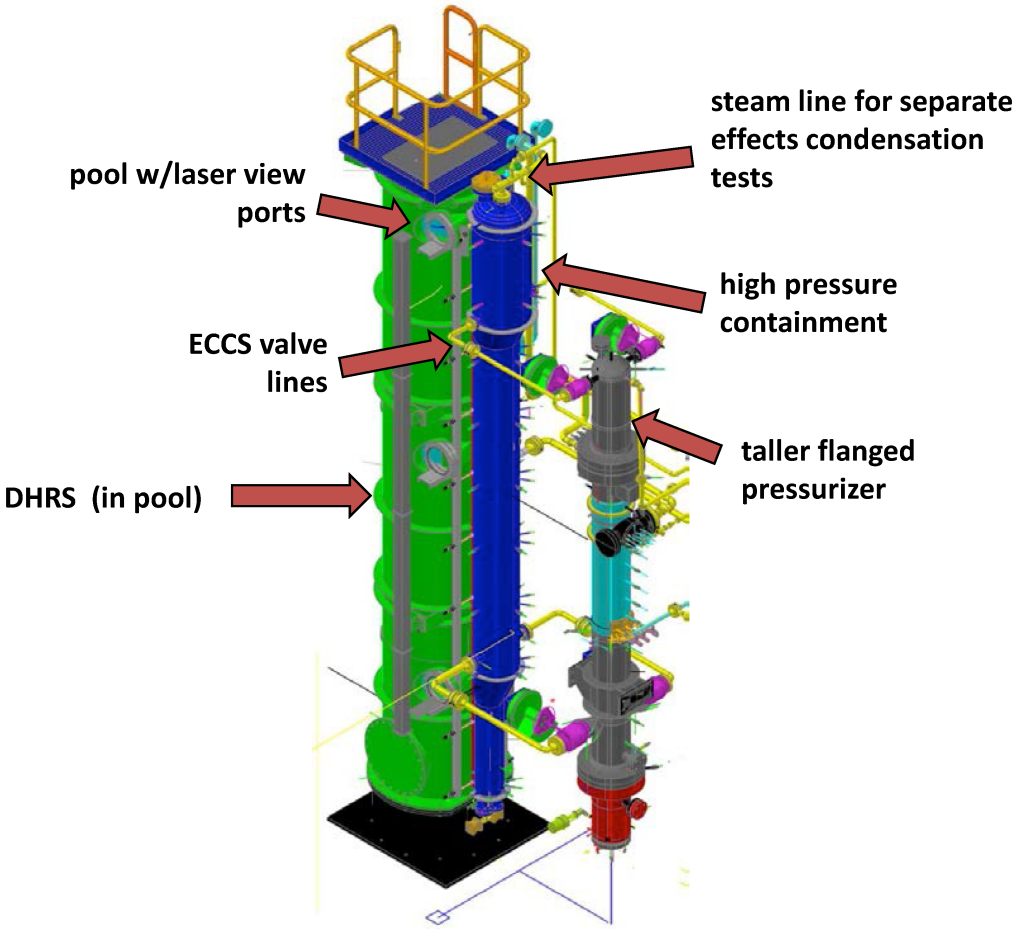
- installation of a DHRS
- scaling of the RVVs and RRVs to the plant design
- replacement of the existing CNV heat transfer plate (HTP), ECCS, and CPV; increased containment pressure rating from about 300 psi to approximately 1,000 psi

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- replacement and installation of new instrumentation
- upgrade of the data acquisition and control system (DACS) and rewiring the instrumentation with DACS
- modification of portions of integral reactor vessel

The updated NIST-1 facility provides a well-scaled representation of a NuScale reactor design that minimizes distortions and provides the measurements necessary for safety code and reactor design validation. A schematic of the NIST-1 facility is shown in Figure 5-3.

Figure 5-3 Schematic of NIST-1 integral test facility



The configuration of the NIST-1 facility is shown in Figure 5-4.

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Figure 5-4 NIST-1 test facility configuration

{{

}}2(a),(c)

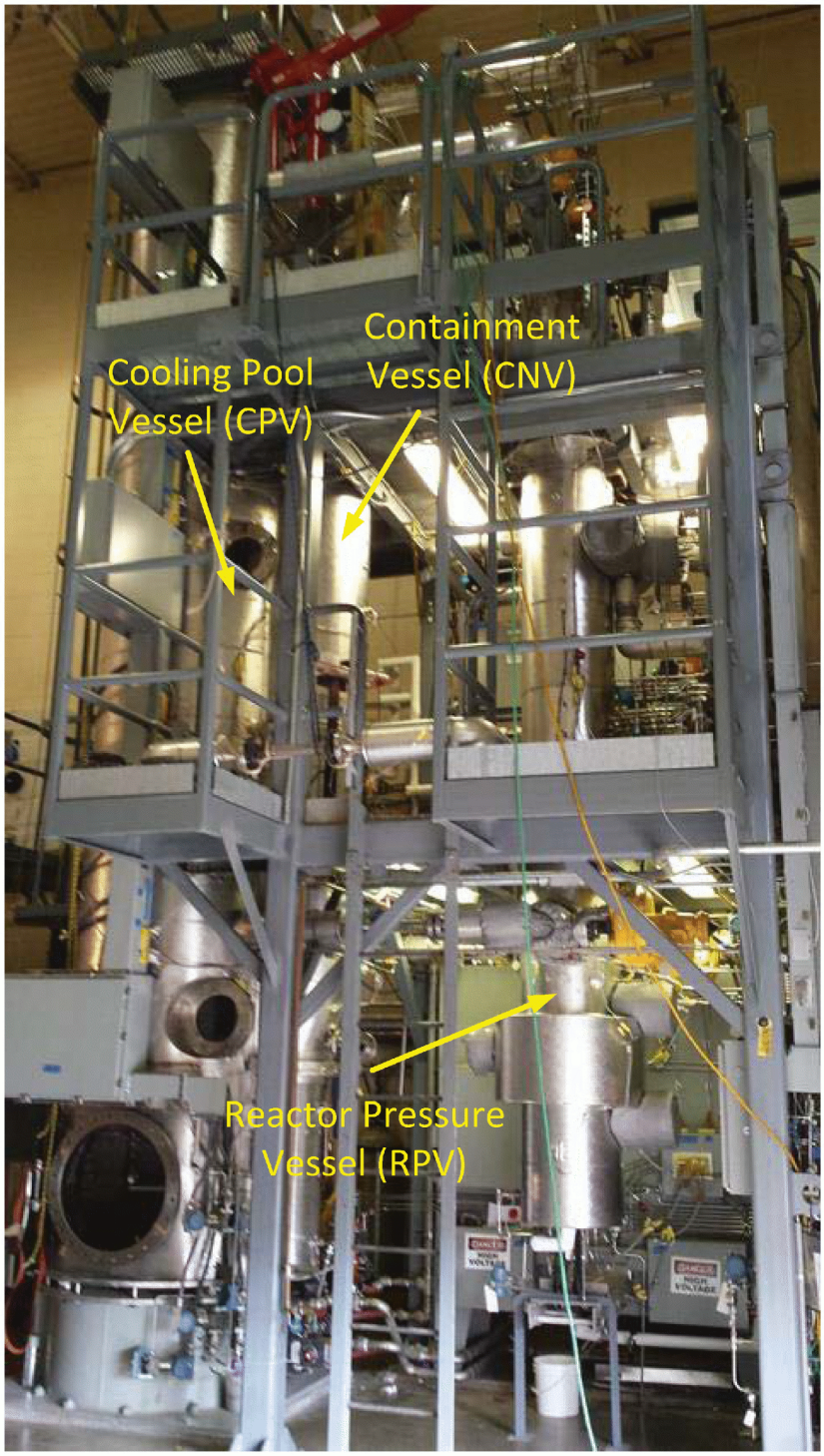
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The NIST-1 facility models the NPM at 1:3.3 length scale, 1:227.5 volume scale, and 1:1 time scale. There are three vessels in the NIST-1 facility: the RPV, CNV, and CPV as shown in Figure 5-3. A photograph of the NIST-1 facility is shown in Figure 5-5. Unlike an actual NuScale plant, the RPV and CNV are not concentric and the CNV is not immersed in the cooling pool. Rather the RPV and CNV are connected by piping that contains valves that perform the functions of the RRVs, RVVs and breaks as shown in Figure 5-3. The CNV is connected to the CPV through an HTP that is scaled to allow energy transfer to the pool in the same proportion as in an NPM.

Natural circulation flow in the primary circuit is driven by heat input in the core region and heat removal to the SG tubes. Fluid heated in the core region flows upward through the RCS hot leg riser, and then downward around the outside of the SG tubes, the cold leg and the downcomer. The flow then returns to the core through the lower plenum. The core is comprised of a {{ }}$^{2(a),(c),ECI}$ electric heater rod bundle with a maximum power of {{ }}$^{2(a),(c),ECI}$ a power level scaled to simulate decay heat. System pressure is controlled by the pressurizer component, which contains heater rods to bring the pressurizer fluid up to saturation temperature.

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Figure 5-5 Photograph of the NIST-1 facility



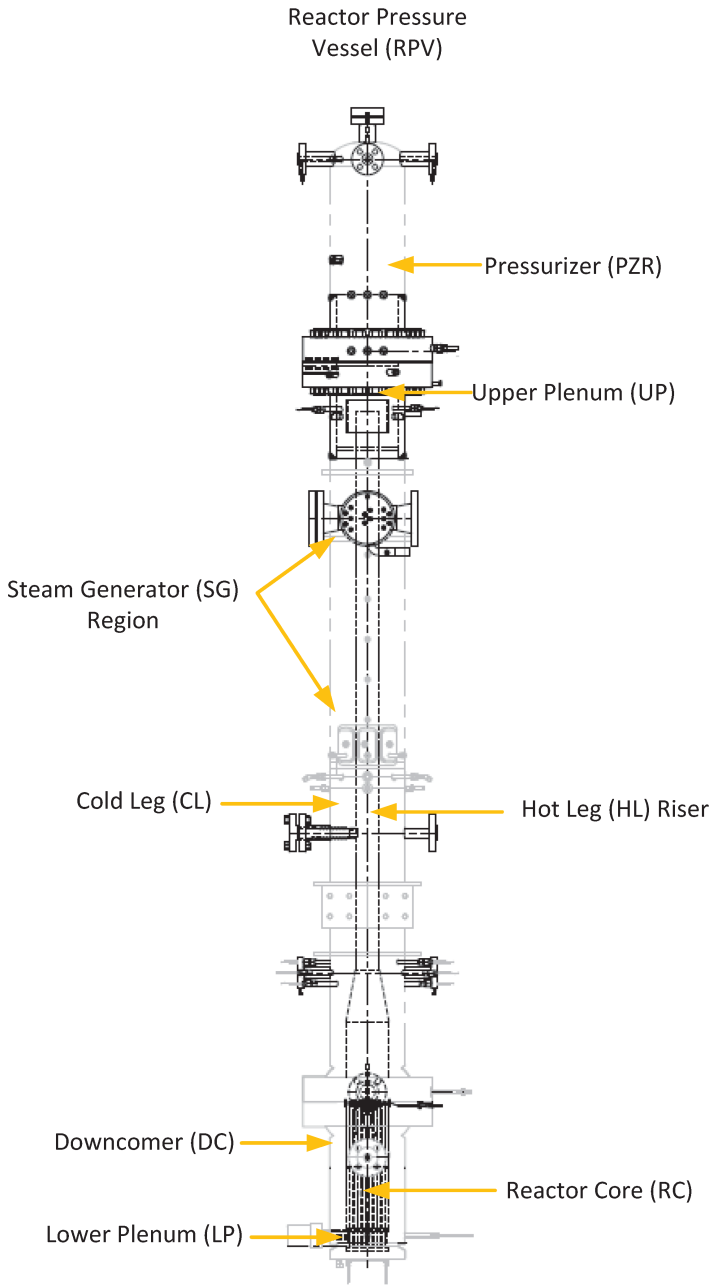
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Reactor Pressure Vessel

Major internal components in the RPV are the core, RCS hot leg riser, pressurizer, and SG bundle.

Figure 5-6 shows a view of the RPV thermal-hydraulic regions with the pressurizer at the top, separated from the lower part of the RPV by a perforated pressurizer baffle plate. The upper plenum occupies the region below the pressurizer baffle plate and above the RCS hot leg riser that extends down to the top of the core. The upper annulus between the RCS hot leg riser and the RPV shell contains the helical coil SG tubes and is labeled as the SG region. The lower part of the annulus immediately below the SG tubes is the cold leg. The lower annulus at the core elevation is the downcomer, which is separated from the core by the core shell. The lower plenum occupies the bottom of the RPV and hydraulically connects the downcomer and the core. The RPV shells and flanges are covered by {{ }}$^{2(a),(c),ECI}$

Figure 5-6 Reactor pressure vessel thermal-hydraulic regions



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The RPV houses the core, which is modeled by a {{

}}2(a),(c),ECI

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Figure 5-7 Lower core flow plate

{{

}}2(a),(c),ECI

Reactor Coolant System Hot Leg Riser

After leaving the core, the flow enters the chimney of the RCS hot leg riser. The RCS hot leg riser extends from above the core shroud to the upper plenum, creating a riser and downcomer configuration to enable natural circulation. As shown in Figure 5-6, the RCS hot leg riser consists of a lower shell, a conical transition, a middle shell containing the flowmeter for the primary circuit, and an upper shell. Flow exits the riser into the upper plenum, which is the space between the RCS hot leg riser outlet and the bottom of the pressurizer baffle plate.

Upper Plenum

After leaving the top of the RCS hot leg riser, the flow enters the upper plenum and is directed radially outward to flow down in the annulus between the riser and the RPV shell. The pressurizer baffle plate separates the upper plenum from the pressurizer. Hydraulic communication between the pressurizer and the RPV occurs via {{ }}2(a),(c) holes in the pressurizer baffle plate that are grouped in

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eight clusters of three circles radially located {{ }}$^{2(a),(c),ECI}$ from the center of the plate.

Pressurizer

The pressurizer is located above the upper plenum and is in thermal hydraulic communication with the upper plenum via the pressurizer baffle plate holes. The pressurizer maintains primary system static pressure during normal steady-state and transient conditions through the use of three heater elements. Each element has {{ }}$^{2(a),(c),ECI}$ of power and is modulated by the facility control system to maintain system static pressure.

Cold Leg Downcomer

After leaving the upper plenum, the flow continues downward through the SG section and into the cold leg downcomer region. The cold leg downcomer is the annular space bounded by the RPV shell inner diameter and the RCS hot leg riser outer diameter. When fluid reaches the RCS hot leg riser conical transition shell, the flow area is reduced. Flow exits the cold leg downcomer into the lower plenum before it recirculates back into the core.

Steam Generators

The SG is a helical coil, once-through HX consisting of three vertical, parallel banks of tubes (an inner, middle, and outer coil) that wrap counter to each other in the annular space between the RCS hot leg riser and the RPV shell inner surface. In the NIST-1 facility, the primary coolant is circulated around the outside of the SG tubes. Feedwater supplied from the feedwater storage tank is pumped through the SG coils by a regenerative turbine pump. This pump utilizes a variable speed controller, which allows for precise control of the feedwater mass flow rate. Pressure in the secondary side is regulated by a pneumatically operated variable position valve located in the steam line portion of the flow loop.

In the feed line, feedwater is pre-heated in an in-line heater before going into the different inlet headers. Feedwater enters the coil bundle at the bottom of the SG and is fully boiled in the tubes, resulting in steam that is superheated at the SG outlet. The boil off length is a function of core power, core exit temperature, main feed pump flow rate, and secondary side pressure. Every coil exhausts the superheated steam into a common steam bustle (header), at which point, depending on the test, piping either directs the steam to atmosphere (i.e., the stack), the CNV, or the DHRS inlet header.

Lower Plenum

The lower plenum is the region bounded between the tubesheet and the lower core flow plate. The lower plenum provides the connection between the downcomer and the core, thus completing the RPV flow loop.

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Decay Heat Removal System

The NIST-1 facility has three possible configurations with the DHRS. A full-height DHRS is used for separate effects testing. The full-height DHRS has a total of eight tubes, distributed between three rows. Figure 5-8 shows the full-height DHRS. Two decay heat removal HXs are scaled for testing integral DHRS effects in the NIST-1 facility. Because an NPM has two decay heat removal trains available for use, one scaled DHR HX has one tube for simulating one NPM decay heat removal train and the second scaled HDR HX has two tubes representing two NPM decay heat removal trains. Figure 5-9 shows the scaled decay heat removal HXs.

Figure 5-8 Full-height decay heat removal condenser

{{

}}2(a),(c),ECI

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Figure 5-9 Scaled decay heat removal heat condensers

{{

}}2(a),(c),ECI

Containment and Cooling Pool Vessel

The CNV, representing the cavity volume between the RPV outer surface and the containment inside surface, is conjoined to the CPV and thermally separated by a scaled HTP. For an NPM, the RPV is located inside containment. However, with the NIST-1 facility, to maintain both volume and surface area scaling similitude, as well as allow proper instrumentation, the RPV is thermal hydraulically separated from the CNV. The heat transfer plate (HTP) models the scaled condensation heat transfer surface between the CNV and CPV. Fluid in the CPV, which is at ambient pressures, models the scaled volume in which an NPM containment vessel is partially immersed.

The CPV has a set of four ports allowing for the installation of one of three decay heat removal HXs. The baseline configuration is with a full height decay heat removal, which is used for SETs. Two decay heat removal HXs are scaled for testing integral DHRS effects in the NIST-1 facility. Because an NPM has two decay heat removal trains available for use, to simulate either one or both trains, one scaled decay heat removal has one tube for simulating one NPM decay heat removal train and the second scaled decay heat removal has two tubes representing two NPM decay heat removal trains.

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Emergency Core Cooling System and Chemical and Volume Control System Lines

Eight lines connect the RPV to the CNV. Five of these lines belong to the facility ECCS, whereas the other three are part of the CVCS. As part of the ECCS, there are two independent reactor vent lines near the top of the pressurizer section, and two reactor recirculation lines in the lower downcomer of the RPV. The fifth ECCS line is a SET line that also models reactor recirculation. For the CVCS, two lines penetrate the vessel near the bottom of the SG. One of these lines penetrates both the vessel wall and the hot leg riser, simulating the make-up line into the hot leg. The other CVCS line connects to the cold leg and thus penetrates only the RPV wall. This line represents the facility CVCS discharge break line. A third CVCS line between the RPV and CNV is located at the very top of the pressurizer and functions as an analogy for the pressurizer spray line. Each line that is installed has a pneumatic isolation valve that is actuated through the test facility control system. Any lines that are not installed use a blank flange for isolation.

The facility baseline configuration consists {{

}}2(a),(c),ECI

NIST-1 Facility Instrumentation and Control

Instrumentation is used throughout the facility to measure the thermal hydraulic behavior during steady-state and transient operations. The following information is obtained and recorded by the DACS:

- {{

}}2(a),(c),ECI

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- {{

}}2(a),(c),ECI

The ECCS, CVCS and SET lines are heavily instrumented to obtain data on break flow and ECCS blowdown and recirculation flows. The components and instruments include:

- an isolation valve.
- {{

}}2(a),(c),ECI

The facility control system generates signals per the system logic, including valve and relay control signals, heater and pump control signals, etc. The following systems can be regulated by the test facility control system:

- core heaters (including decay power modeling)
- pressurizer heaters
- feedwater pump
- coolant charging pump
- containment heaters (used to maintain an adiabatic boundary condition on all walls of containment except for the prescribed condensation surface of the HTP)
- strip heat power in the ECCS, CVCS, DHRS, and steam lines (used to minimize heat loss in the lines connecting the RPV and CNV)
- CNV vacuum pump
- various pneumatic and solenoid valves for flow regulation or transient initiation

The NIST-1 facility is used to assess the operation of an NPM under normal operating conditions and to assess the passive safety system responses during transient conditions. The data generated and collected by the facility DACS is

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used to assist in validation of the NRELAP5 system thermal-hydraulic code for NPM analysis.

5.3.2.2 Decay Heat Removal System Separate Effects Test Matrix

The NIST-1 facility was used to perform separate effects testing with the full-height DHRS. Table 5-7 describes DHRS separate effects testing performed at the NIST-1 facility; these test assessments are presented in the following subsections.

Table 5-7 NIST-1 decay heat removal system separate effects tests for NRELAP5 code validation

{{

}}2(a),(c),ECI

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5.3.2.3 NRELAP5 Model Description

For the NIST-HP-03 and NIST-HP-04 test assessments, a separate effects model of the NIST-1 facility is used. This model contains hydrodynamic components and heat structures for:

- {{

}}2(a),(c)

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{{

}}2(a),(c)

Table 5-8 Comparison between NPM and NIST-1 decay heat removal NRELAP5 nodalizations

{{

}}2(a),(c)

Figure 5-10 NIST-1 noding diagram for full-height decay heat removal system separate effect tests

{{

}}2(a),(c)

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5.3.2.4 HP-03 Test Description

The NIST-1 HP-03 test (HP-03) was used to assess the capability of NRELAP5 to predict convective condensation within the DHRS and heat transfer to the reactor pool across the DHRS tubes.

For HP-03, the heated NIST-1 primary system was used to produce steam within the SG tubes. The steam from the SG tubes was transferred to the steam line and routed to the full-height DHRS steam header, which was located in the upper CPV. Steam was condensed in the condenser tubes, entered the DHRS condensate header, and flowed through the condensate line. The condensate control valve in the condensate line maintained DHRS pressure by controlling the rate at which condensate discharged to the atmosphere.

During HP-03, superheated steam was delivered to the DHRS steam header at a range of flow rates and pressures. The incoming steam was allowed to condense within the condenser tubes and a pseudo steady state liquid level (DHRS level) was established.

Table 5-9 summarizes the initial conditions of the HP-03 cases.

Table 5-9 NIST-1 HP-03 test cases

{{

}}2(a),(c),ECI

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5.3.2.5 HP-03 Results

The HP-03 data trends were well predicted by NRELAP5 with reasonable to excellent agreement {{ }}$^{2(a),(c),ECI}$ Detailed results are presented herein for three HP-03 cases, specifically HP-03-01, HP-03-02c, and HP-03-03-Part1.

For the higher DHRS inlet mass flow rate cases (i.e., HP-03-01c-Part1, Part2, and Part3), NRELAP5 code-to-data comparisons show [A.] reasonable-to-excellent or excellent agreement for DHRS power, [B.] reasonable, reasonable-to-excellent, or excellent agreement for other parameters of interest, and [C.] reasonable agreement for DHRS level. {{

}}$^{2(a),(c)}$

5.3.2.5.1 HP-03-01 Run

This section compares the NRELAP5 simulation results with the measured data for HP-03-01, which was run at a DHRS pressure of {{ }}$^{2(a),(c),ECI}$

Figure 5-11 presents the flow enthalpy into the inlet and outlet headers of the DHRS heat exchanger (an alternate view of DHRS heat removal rate, or DHRS power). The mass flow rate, pressure, and temperature at the inlet are specified as boundary conditions. As there is a very large difference in magnitude between the inlet and outlet enthalpy flow rates, the NRELAP5 outlet enthalpy flow rate and data outlet enthalpy flow rate do not need to be perfectly matched in order for NRELAP5 DHRS power to show excellent code-to-data agreement.

Figure 5-11a presents the DHRS power. Code-to-data agreement is excellent. This agreement indicates that NRELAP5 appropriately calculates the DHRS-to-CPV heat transfer.

Figure 5-12 presents the NRELAP5 calculated DHRS heat exchanger level. Code-to-data agreement is reasonable. {{ }}$^{2(a),(b),(c)}$

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{{

}}2(a),(b),(c) Figure 5-12a presents the same comparison as Figure 5-12, but with the Reference 29 version of NRELAP5.

Figure 5-13 presents the DHRS condenser tube internal fluid temperature. {{

}}2(a),(b),(c) Code-to-data agreement is reasonable. Figure 5-13a presents the same comparison as Figure 5-13, but with the Reference 29 version of NRELAP5.

Figure 5-14 and Figure 5-14a present the CPV temperature. Code-to-data agreement is excellent. While the NRELAP5 simulation is not run for an extended period of time, the CPV temperature is well-matched at the end of the comparison period. The high temperatures in the upper CPV indicate that boiling was occurring. Figure 5-14b presents the same comparison as Figure 5-14 and the bottom elevation of Figure 5-14a, but with the Reference 29 version of NRELAP5.

Figure 5-15 presents the CPV level. {{

}}2(a),(c) code-to-data agreement is excellent. The CPV level is observed to drop during testing, which indicates that boiling was occurring. Figure 5-15a presents the same comparison as Figure 5-15, but with the Reference 29 version of NRELAP5.

{{

}}2(a),(c)

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Figure 5-11 NIST-1 HP-03-01 decay heat removal system enthalpy flow rate code-to-data comparison

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}}2(a),(b),(c),ECI

Figure 5-11a NIST-1 HP-03-01 decay heat removal system power code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-12 NIST-1 HP-03-01 decay heat removal system level code-to-data comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-12a NIST-1 HP-03-01 decay heat removal system level code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-13 NIST-1 HP-03-01 decay heat removal system internal fluid temperature code-to-data comparison

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}}2(a),(b),(c),ECI

Figure 5-13a NIST-1 HP-03-01 decay heat removal system internal fluid temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-14 NIST-1 HP-03-01 cooling pool vessel temperature code-to-data comparison (1 of 2)

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}}2(a),(b),(c),ECI

Figure 5-14a NIST-1 HP-03-01 cooling pool vessel temperature code-to-data comparison (2 of 2)

{{

}}2(a),(b),(c),ECI

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Figure 5-14b NIST-1 HP-03-01 cooling pool vessel temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-15 NIST-1 HP-03-01 cooling pool vessel level code-to-data comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-15a NIST-1 HP-03-01 cooling pool vessel level code-to-data comparison

{{

}}2(a),(b),(c),ECI

5.3.2.5.2 HP-03-02c Run

This section compares the NRELAP5 simulation results with the measured data HP-03-02c, which was run at a DHRS pressure of {{

}}2(a),(c),ECI

Figure 5-16 presents the DHRS inlet and outlet enthalpy flow rate. {{

}}2(a),(b),(c) As there is a very large difference in magnitude between the inlet and outlet enthalpy flow rates, the NRELAP5

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outlet enthalpy flow rate and data outlet enthalpy flow rate do not need to be perfectly matched in order for NRELAP5 DHRS power to show excellent code-to-data agreement.

Figure 5-16a presents the DHRS power. Code-to-data agreement is excellent. This agreement indicates that NRELAP5 appropriately calculates the DHRS-to-CPV heat transfer.

Figure 5-17 presents the DHRS level. Code-to-data agreement is reasonable. Figure 5-17a presents the same comparison as Figure 5-17, but with the Reference 29 version of NRELAP5.

Figure 5-18 presents the DHRS condenser tube internal fluid temperature. {{

}}$^{2(a),(b),(c)}$ Code-to-data agreement is reasonable. Figure 5-18a presents the same comparison as Figure 5-18, but with the Reference 29 version of NRELAP5.

Figure 5-19 and Figure 5-19a present the CPV temperature. Overall code-to-data agreement is judged to be reasonable, as most trends are captured {{

}}$^{2(a),(b),(c)}$ Figure 5-19b presents the same comparison as Figure 5-19 and the bottom elevation of Figure 5-19a, but with the Reference 29 version of NRELAP5.

Figure 5-20 presents the CPV level. {{

}}$^{2(a),(c)}$ code-to-data agreement is excellent. The CPV level is observed to drop during testing, which indicates that boiling was occurring. Figure 5-20a presents the same comparison as Figure 5-20, but with the Reference 29 version of NRELAP5.

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Figure 5-16 NIST-1 HP-03-02c decay heat removal system enthalpy flow rate code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-16a NIST-1 HP-03-02c decay heat removal system power code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-17 NIST-1 HP-03-02c decay heat removal system level code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-17a NIST-1 HP-03-02c decay heat removal system level code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-18 NIST-1 HP-03-02c decay heat removal system internal fluid temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-18a NIST-1 HP-03-02c decay heat removal system internal fluid temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-19 NIST-1 HP-03-02c cooling pool vessel temperature code-to-data comparison (1 of 2)

{{

}}2(a),(b),(c),ECI

Figure 5-19a NIST-1 HP-03-02c cooling pool vessel temperature code-to-data comparison (2 of 2)

{{

}}2(a),(b),(c),ECI

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Figure 5-19b NIST-1 HP-03-02c cooling pool vessel temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-20 NIST-1 HP-03-02c cooling pool vessel level code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-20a NIST-1 HP-03-02c cooling pool vessel level code-to-data comparison

{{

}}2(a),(b),(c),ECI

5.3.2.5.3 HP-03-03 Part 1 Run

This section compares the NRELAP5 simulation results with the measured data for the HP-03-03-Part 1 test, which was run at a DHRS pressure of {{ }}2(a),(c),ECI

Figure 5-21 presents the DHRS inlet and outlet enthalpy flow rate. Note that the inlet temperature, pressure, and mass flow rate are specified as boundary conditions. {{

}}2(a),(b),(c) As there is a very large difference in magnitude between the inlet and outlet enthalpy flow rates, the NRELAP5

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outlet enthalpy flow rate and data outlet enthalpy flow rate do not need to be perfectly matched in order for NRELAP5 DHRS power to show excellent code-to-data agreement.

Figure 5-21a presents the DHRS power. Code-to-data agreement is reasonable-to-excellent. This agreement indicates that NRELAP5 appropriately calculates the DHRS-to-CPV heat transfer. {{

}}2(a),(c)

Figure 5-22 presents the DHRS level. Code-to-data agreement is reasonable-to-excellent. Figure 5-22a presents the same comparison as Figure 5-22, but with the Reference 29 version of NRELAP5.

Figure 5-23 presents the DHRS condenser tube internal fluid temperature. {{

}}2(a),(b),(c) Code-to-data agreement is reasonable. Figure 5-23a presents the same comparison as Figure 5-23, but with the Reference 29 version of NRELAP5.

Figure 5-24 and Figure 5-24a present the CPV temperature. Overall code-to-data agreement is reasonable-to-excellent. Most trends that appear in the test data are captured by NRELAP5. Figure 5-24b presents the same comparison as Figure 5-24 and the bottom elevation of Figure 5-24a, but with the Reference 29 version of NRELAP5.

Figure 5-25 presents the CPV level. {{

}}2(a),(b),(c)

Figure 5-25a presents the same comparison as Figure 5-25, but with the Reference 29 version of NRELAP5.

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Figure 5-21 NIST-1 HP-03-03-P1 decay heat removal system enthalpy flow rate code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-21a NIST-1 HP-03-03-Part1 decay heat removal system power code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-22 NIST-1 HP-03-03-Part1 decay heat removal system level code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-22a NIST-1 HP-03-03-Part1 decay heat removal system level code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-23 NIST-1 HP-03-03-Part1 decay heat removal system internal fluid temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-23a NIST-1 HP-03-03-Part1 decay heat removal system internal fluid temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-24 NIST-1 HP-03-03-Part1 cooling pool vessel temperature code-to-data comparison (1 of 2)

{{

}}2(a),(b),(c),ECI

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Figure 5-24a NIST-1 HP-03-03-Part1 cooling pool vessel temperature code-to-data comparison (2 of 2)

{{

}}2(a),(b),(c),ECI

Figure 5-24b NIST-1 HP-03-03-Part1 cooling pool vessel temperature code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-25 NIST-1 HP-03-03-Part1 cooling pool vessel level code-to-data comparison

{{

}}2(a),(b),(c),ECI

Figure 5-25a NIST-1 HP-03-03-Part1 cooling pool vessel level code-to-data comparison

{{

}}2(a),(b),(c),ECI

5.3.2.5.4 HP-03 Summary

The comparisons between the HP-03 data and HP-03 NRELAP5 simulations show that NRELAP5 has the capability to accurately predict the DHRS-to-CPV heat transfer.

For DHRS power, code-to-data comparisons show reasonable-to-excellent or excellent agreement.

For the identified additional parameters of interest, including DHRS level and CPV level, code-to-data comparisons show reasonable, reasonable-to-excellent, or excellent agreement.

The re-performance of the comparisons using the Reference 29 version of NRELAP5 shows the same level of agreement as for the original comparisons.

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5.3.2.6 HP-04 Test Description

HP-04 was used to assess NRELAP5 capability to predict {{

}}2(a),(c)

For the HP-04 test, the heated primary system was used to provide transfer heat to the SG and create steam. The steam from the SG tubes was transferred to the steam line and routed to the full-height DHRS steam header, located in the CPV. Steam is condensed in the condenser tubes and enters the DHRS condensate plenum and condensate line. For this test the condensate line directed the DHRS discharge to the environment.

{{

}}2(a),(c) These conditions were achieved at two pressure conditions. Table 5-10 presents the test conditions considered for the assessment of NRELAP5.

Table 5-10 NIST-1 HP04 test ranges

{{

}}2(a),(c),ECI

5.3.2.7 HP-04 Test Results

The NIST-1 HP-04 test data was compared to NRELAP5 predictions designed to simulate the test conditions and test procedures in effect during the experiment. HP-04 test data trends were well predicted by NRELAP5 with reasonable to excellent agreement for DHRS heat exchanger {{

}}2(a),(c),ECI based on an enthalpy balance across the tubes.

The comparison of the calculated DHRS and cooling pool water levels show reasonable to excellent agreement with the data. The NIST-1 CPV heat-up response was not fully captured in the code to data comparison.

The comparisons between the experimental data and the code calculated values show that NRELAP5 has the capability to accurately predict the energy transfer across the DHRS heat exchanger tubes to the CPV fluid, although the CPV

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heat-up response is not fully captured. The heat transfer across the tubes is influenced by the condensate level inside the tubes and the cooling pool level.

These results are discussed in more detail in the following subsections for NIST-1 generated data at DHRS pressures of {{ }}$^{2(a),(c),ECI}$

5.3.2.7.1 HP-04-02 Run

This section compares the NRELAP5 simulation results with the measured data for the HP-04-02 run {{ }}$^{2(a),(c),ECI}$ In figures, the data measurement uncertainty is shown in dotted lines.

Code-to-data comparisons of the DHRS heat removal rate produced reasonable agreement, as shown by the prediction of the enthalpy change over the DHRS in Figure 5-26. {{

}}$^{2(a),(c)}$ Figure 5-26a presents the same comparison as Figure 5-26, but with the Reference 29 version of NRELAP5.

The NIST-1 profiles for the DHRS condensate outlet temperature are not fully captured in the code-to-data comparisons. {{

}}$^{2(a),(c)}$ Figure 5-27a presents the same comparison as Figure 5-27, but with the Reference 29 version of NRELAP5.

Code-to-data comparisons of DHRS heat exchange tube level produced reasonable agreement, as shown in Figure 5-28. {{

}}$^{2(a),(c)}$ Figure 5-28a presents the same comparison as Figure 5-28, but with the Reference 29 version of NRELAP5.

Although the code-to-data comparisons of CPV level produced reasonable-to-excellent agreement, {{

}}$^{2(a),(b),(c),ECI}$

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{{

}}2(a),(b),(c),ECI Figure 5-29a, Figure 5-30a, and Figure 5-31a present the same comparisons as Figure 5-29, Figure 5-30, and Figure 5-31, respectively, but with the Reference 29 version of NRELAP5.

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Figure 5-26 NIST-1 HP-04-02 decay heat removal system energy transfer rate

{{

}}2(a),(b),(c),ECI

Figure 5-26a NIST-1 HP-04-02 decay heat removal system energy transfer rate

{{

}}2(a),(b),(c),ECI

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Figure 5-27 NIST-1 HP-04-02 decay heat removal system condensate temperature

{{

}}2(a),(b),(c),ECI

Figure 5-27a NIST-1 HP-04-02 decay heat removal system condensate temperature

{{

}}2(a),(b),(c),ECI

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Figure 5-28 NIST-1 HP-04-02 decay heat removal system internal collapsed level

{{

}}2(a),(b),(c),ECI

Figure 5-28a NIST-1 HP-04-02 decay heat removal system internal collapsed level

{{

}}2(a),(b),(c),ECI

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Figure 5-29 NIST-1 HP-04-02 cooling pool vessel level

{{

}}2(a),(b),(c),ECI

Figure 5-29a NIST-1 HP-04-02 cooling pool vessel level

{{

}}2(a),(b),(c),ECI

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Figure 5-30 NIST-1 HP-04-02 mid cooling pool vessel fluid temperatures

{{

}}2(a),(b),(c),ECI

Figure 5-30a NIST-1 HP-04-02 mid cooling pool vessel fluid temperatures

{{

}}2(a),(b),(c),ECI

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Figure 5-31 NIST-1 HP-04-02 upper cooling pool vessel fluid temperatures

{{

}}2(a),(b),(c),ECI

Figure 5-31a NIST-1 HP-04-02 upper cooling pool vessel fluid temperatures

{{

}}2(a),(b),(c),ECI

5.3.2.7.2 HP-04-03 Run

This section compares the NRELAP5 simulation results with the measured data for the HP-04-03 run {{ }}2(a),(c),ECI In figures, the data measurement uncertainty is shown in dotted lines.

Code-to-data comparison of the DHRS heat removal rate produced reasonable to excellent agreement, as shown by the prediction of the enthalpy change over the DHRS in Figure 5-32. {{

}}2(a),(c) Figure 5-32a presents the same comparison as Figure 5-32, but with the Reference 29 version of NRELAP5.

The NIST-1 profiles for DHRS condensate outlet temperatures are not fully captured in the code-to-data comparisons. {{

}}2(a),(c) Figure 5-33a presents the same comparison as Figure 5-33, but with the Reference 29 version of NRELAP5.

Code-to-data comparison of DHRS heat exchanger tube level shows reasonable agreement, as shown in Figure 5-34. {{

}}2(a),(c) Figure 5-34a presents the same comparison as Figure 5-34, but with the Reference 29 version of NRELAP5.

Although the code-to-data comparisons of CPV level produced reasonable-to-excellent agreement, {{

}}2(a),(b),(c),ECI Figure 5-35a, Figure 5-36a, and Figure 5-37a present the same comparisons as Figure 5-35, Figure 5-36, and Figure 5-37, respectively, but with the Reference 29 version of NRELAP5.

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Figure 5-32 NIST-1 HP-04-03 decay heat removal system energy transfer rate

{{

}}2(a),(b),(c),ECI

Figure 5-32a NIST-1 HP-04-03 decay heat removal system energy transfer rate

{{

}}2(a),(b),(c),ECI

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Figure 5-33 NIST-1 HP-04-03 decay heat removal system condensate temperature

{{

}}2(a),(b),(c),ECI

Figure 5-33a NIST-1 HP-04-03 decay heat removal system condensate temperature

{{

}}2(a),(b),(c),ECI

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Figure 5-34 NIST-1 HP-04-03 decay heat removal system internal collapsed level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-34a NIST-1 HP-04-03 decay heat removal system internal collapsed level comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-35 NIST-1 HP-04-03 cooling pool vessel level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-35a NIST-1 HP-04-03 cooling pool vessel level comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-36 NIST-1 HP-04-03 mid cooling pool vessel axial temperatures

{{

}}2(a),(b),(c),ECI

Figure 5-36a NIST-1 HP-04-03 mid cooling pool vessel axial temperatures

{{

}}2(a),(b),(c),ECI

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Figure 5-37 NIST-1 HP-04-03 upper cooling pool vessel axial temperatures

{{

}}2(a),(b),(c),ECI

Figure 5-37a NIST-1 HP-04-03 upper cooling pool vessel axial temperatures

{{

}}2(a),(b),(c),ECI

5.3.2.7.3 HP-04 Summary

Although the CPV fluid heat-up profiles for the NIST-1 HP-04 test data were not fully captured in the NRELAP5 simulations of HP-04, the code-to-data comparisons of DHRS heat removal rate were well-matched. The re-performance of the comparisons using the Reference 29 version of

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NRELAP5 shows the same level of agreement as for the original comparisons.

This assessment concludes that NRELAP5 simulations of an NPM decay heat removal system should be expected to adequately predict the DHRS heat removal rates related to a large envelope of reactor pool liquid conditions.

5.3.3 NIST-1 Non-LOCA Integral Test

5.3.3.1 NIST-1 Facility

The NIST-1 facility is described in Section 5.3.2.1. The NLT-2 tests were conducted in two sessions; first NLT-2a was conducted and a few weeks later NLT-2b was conducted. Test NLT-02a was a loss of feed water transient with a subsequent pressurization and depressurization of the RPV. In Test NLT-02b the DHRS was activated from a low reactor power level to demonstrate long term cooling capability over several hours.

5.3.3.2 Non-LOCA Integral Effects Test Matrix

The NIST-1 facility was used to perform integral effects testing with the scaled two-tube DHRS heat exchanger. Table 5-11 describes DHRS integral effects testing performed at the NIST-1 facility; these test assessments are presented in the following subsections.

Table 5-11 NIST-1 integral effects tests for NRELAP5 code validation

{{

}}2(a),(c),ECI

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Table 5-11 NIST-1 integral effects tests for NRELAP5 code validation (Continued)

{{

}}2(a),(c),ECI

5.3.3.3 NRELAP5 Model Description

The NRELAP5 model of the NIST-1 facility is constructed to include the major flow paths in the facility, including the major systems; the RPV primary and secondary, the CNV, and the CPV, as well as the ECCS. {{

}}2(a),(c) A nodalization diagram is shown in Figure 5-38.

{{

}}2(a),(c)

{{

}}2(a),(c)

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{{

}}2(a),(c)

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Figure 5-38 NRELAP5 noding diagram for the NIST-1 facility

{{

}}2(a),(c),ECI

Figure 5-39 NRELAP5 NIST-1 model secondary system nodalization layout (NLT-2b)

{{

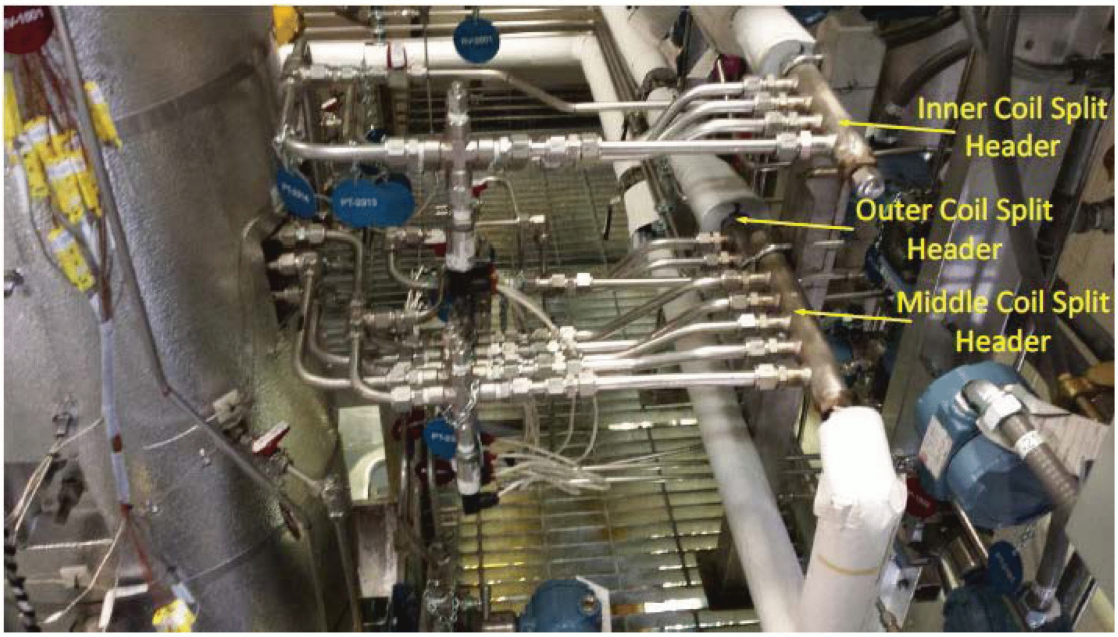
}}2(a),(c),ECI

Figure 5-39a NRELAP5 NIST-1 model secondary system nodalization layout (NLT-15p2)

{{

}}2(a),(c),ECI

Figure 5-40 NIST-1 feedwater split headers at the steam generator tube coils connection (at the time of NLT-2b test)



5.3.3.4 NLT-2a Test Description

The objective of the NLT-02 test was to measure the scaled integral system response to a decrease in heat removal from the secondary side following a loss of feedwater event, to the point of reactor trip. In an NPM design, response to a decrease in heat removal includes a reactor trip based on PZR pressure, core outlet temperature, or PZR level, as well as the activation of the DHRS heat exchanger, which permits circulation between the DHRS and the SG coils.
{{

}}2(a),(c)

The NIST-1 facility had the following configuration and initial conditions at the start of the NLT-02a test:

1. {{

}}2(a),(b),(c),ECI

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5. {{

}}2(a),(b),(c),ECI

The sequence of events for the NLT-02a test is shown in Table 5-12.

Table 5-12 NLT-02a sequence of events

{{

}}2(a),(b),(c),ECI

5.3.3.5 NLT-2a Test Results

The transient was initiated with the termination of feedwater flow. Feedwater flow was a boundary condition in the NRELAP5 simulation. Figure 5-41 compares the calculated feedwater flow with the data. Note that in figures, the data measurement uncertainty is shown in dotted lines.

The core heater rod power was another boundary condition applied to the simulation. Figure 5-42 compares core heater rod power used in the simulation with the measured core heater rod power that is supplied to the core heater rods.

Figure 5-43 compares the PZR pressure. The calculated pressure shows reasonable to excellent agreement with the measured data. {{

}}2(a),(c) Figure 5-43a presents the same comparison as Figure 5-43, but with the Reference 29 version of NRELAP5.

Figure 5-44 compares the calculated riser flow rate with the measured data. The predicted response shows reasonable agreement with the data. {{

}}2(a),(c) Figure 5-44a presents the same comparison as Figure 5-44, but with the Reference 29 version of NRELAP5.

The comparison of the measured and calculated PZR level is shown in Figure 5-45. The predicted level is considered in reasonable to excellent

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agreement with the data; the calculation is within the uncertainty bands of the data {{

}}[2(a),(c)] Figure 5-45a presents the same comparison as Figure 5-45, but with the Reference 29 version of NRELAP5.

The predicted core inlet temperature is compared to data in Figure 5-46. {{

}}[2(a),(c)] Figure 5-46a presents the same comparison as Figure 5-46, but with the Reference 29 version of NRELAP5.

The core exit temperature comparison is shown in Figure 5-49. The initial temperature is underpredicted in the steady state but is within the uncertainty band when the core power is shut off. The trend of the predicted temperature is slightly different than the measured value between 50-125 seconds, but it remains within the uncertainty bands of the data. Figure 5-49a presents the same comparison as Figure 5-49, but with the Reference 29 version of NRELAP5. The predicted riser inlet fluid temperature, shown in Figure 5-49b, is in reasonable agreement with the measured temperature.

A comparison of the calculated and measured PZR heater rod power is shown in Figure 5-50. The PZR heater rods are used to keep the RPV primary side at a target pressure. If the pressure declines, the PZR heater rod power increases. Conversely, if the PZR pressure increases, the PZR heater rod power declines. {{

}}[2(a),(c)]

The code-to-data comparison of the steam line pressure is provided in Figure 5-51. {{

}}[2(a),(c)]

A comparison of the steam line mass flow rate is shown in Figure 5-52. The predicted steam line mass flow rate shows reasonable agreement with the data, considering {{

}}[2(a),(c).] Figure 5-52a presents the same comparison as Figure 5-52, but with the Reference 29 version of NRELAP5.

In conclusion, it is demonstrated that NRELAP5 is capable of predicting the data trends of non-LOCA events, such as a loss of feedwater, with a high degree of confidence.

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Figure 5-41 NLT-02a transient feedwater flow comparison

{{

}}2(a),(b),(c),ECI

Figure 5-42 NLT-02a transient core heater rod power comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-43 NLT-02a transient pressurizer pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-43a NLT-02a transient pressurizer pressure comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-44 NLT-02a transient riser mass flow rate comparison

{{

}}2(a),(b),(c),ECI

Figure 5-44a NLT-02a transient riser mass flow rate comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-45 NLT-02a transient pressurizer level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-45a NLT-02a transient pressurizer level comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-46 NLT-02a transient core inlet temperature

{{

}}2(a),(b),(c),ECI

Figure 5-46a NLT-02a transient core inlet temperature

{{

}}2(a),(b),(c),ECI

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Figure 5-47 NLT-02a transient combined middle and outer steam generator tube coil differential pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-48 NLT-02a transient inner steam generator tube coil differential pressure comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-49 NLT-02a transient core exit fluid temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-49a NLT-02a transient core exit fluid temperature comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-49b NLT-02a transient riser inlet fluid temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-50 NLT-02a transient pressurizer heater rod power comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-51 NLT-02a transient steam line pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-52 NLT-02a transient steam line mass flow rate comparison

{{

}}2(a),(b),(c),ECI

Figure 5-52a NLT-02a transient steam line mass flow rate comparison

{{

}}2(a),(b),(c),ECI

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5.3.3.6 NLT-2b Test Description

The objectives of the NLT-02b test were to observe the integral response of the DHRS and RPV after initial DHRS activation, under quasi-steady conditions as the cooling pool was allowed to heat-up, and finally in a period of DHRS-driven cooling and depressurization after the core power transitioned to decay heat mode. The principal parameters of interest during initial DHRS activation are DHRS condensate mass flow rate, DHRS pressure, and temperatures throughout the SG secondary and DHRS loop. After the initial pressure peak due to DHRS activation subsides, additional parameters of interest are the primary flow rate, primary level, primary pressure and temperatures, and CPV temperatures adjacent to the DHRS heat exchanger.

{{

}}2(a),(c)

The NIST-1 facility had the following configuration and initial conditions at the start of the NLT-02b test:

1. {{

}}2(a),(c)

The sequence of events for the NLT-02b test is shown in Table 5-13.

{{

}}2(a),(c)

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Table 5-13 NLT-02b sequence of events

{{

}}2(a),(b),(c)

5.3.3.7 NLT-2b Phase 1 Test Results

Code-to-data comparisons of key parameters for the first phase of the NLT-2b test are shown in Figure 5-53 through Figure 5-70. Note that in figures, the data measurement uncertainty is shown in dotted lines. This phase consisted of termination of the feedwater flow and turning off the core heater rod power prior to the opening of the DHRS steam line and condensate line valves to begin the transition to DHRS cooling. {{

}}2(a),(b),(c)

In general, the NRELAP5 simulation showed reasonable agreement with the measured data for this time period.

The measured core heater rod power is used as a boundary condition in the simulation. A comparison of the core heater rod power is shown in Figure 5-53. {{

}}2(a),(c)

Comparison of the PZR pressure response is provided in Figure 5-54. {{

}}2(a),(b),(c),ECI

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{{

}}2(a),(b),(c),ECI Figure 5-55a presents the same comparison as Figure 5-55, but with the Reference 29 version of NRELAP5.

Code-to-data comparisons for the core inlet and exit temperatures are shown in Figure 5-57. {{

}}2(a),(b),(c)

Comparison of the measured and calculated steam line pressure is shown in Figure 5-59. {{

}}2(a),(b),(c)

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{{

}}2(a),(b),(c)

The initial liquid mass inventory indicated by the steady state SG tube level and DHRS HX tube level showed reasonable agreement with the data in the simulation. {{

}}2(a),(b),(c)

The feedwater mass flow rate was terminated prior to the opening of the DHRS loop steam and condensate valves.

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Figure 5-63 and Figure 5-64 show code-to-data comparisons of the SG and DHRS heat exchanger tube liquid level, respectively. {{

}}2(a),(c) Figure 5-64a presents the same comparison as Figure 5-64, but with the Reference 29 version of NRELAP5.

Figure 5-65 provides a comparison of the measured and calculated DHRS condensate fluid temperature. Although the trends of the data are calculated, the comparison is considered minimal, {{

}}2(a),(c) Figure 5-65a presents the same comparison as Figure 5-65, but with data from a different location and using the Reference 29 version of NRELAP5.

Figure 5-66 shows that DHRS condensate flow is slightly underpredicted. The predicted flow rate starts out slightly higher than the data, but the trend is correct, and the magnitude is reasonable compared to the data during the quasi steady state phase. The flow plot in combination with the DHRS heat exchanger level (Figure 5-64) and SG level (Figure 5-63), with inspection of the pressure drops in the steam lines and condensate lines, provides confidence that the total DHRS loop resistance is appropriately accounted for. Figure 5-66a presents the same comparison as Figure 5-66, but with the Reference 29 version of NRELAP5.

Figure 5-67 shows {{

}}2(a),(b),(c)

Figure 5-68 compares predicted CPV level during phase 1 of the transient. {{

}}2(a),(b),(c)

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{{

}}2(a),(b),(c)

Figure 5-53 NLT-02b phase 1 transient core power comparison

{{

}}2(a),(b),(c),ECI

Figure 5-54 NLT-02b phase 1 transient pressurizer pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-54a NLT-02b phase 1 transient pressurizer pressure comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-55 NLT-02b phase 1 transient pressurizer level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-55a NLT-02b phase 1 transient pressurizer level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-56 Not Used

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Figure 5-57 NLT-02b phase 1 transient core inlet and outlet temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-57a NLT-02b phase 1 transient core inlet and outlet temperature comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-58 Not Used

Figure 5-59 NLT-02b phase 1 transient steam generator steam pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-59a NLT-02b phase 1 transient steam generator steam pressure comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-60 NLT-02b phase 1 transient steam generator thermal power comparison

{{

}}2(a),(b),(c),ECI

Figure 5-61 NLT-02b phase 1 transient decay heat removal system heat exchanger thermal power comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-62 NLT-02b phase 1 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization

{{

}}2(a),(b),(c),ECI

Figure 5-62a NLT-02b phase 1 integrated compensation flow for SG and DHRS HX level equalization

{{

}}2(a),(b),(c),ECI

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Figure 5-63 NLT-02b phase 1 transient steam generator level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-64 NLT-02b phase 1 transient decay heat removal system heat exchanger level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-64a NLT-02b phase 1 transient decay heat removal system heat exchanger level comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-65 NLT-02b phase 1 transient decay heat removal system condensate temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-65a NLT-02b phase 1 transient decay heat removal system condensate temperature comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-66 NLT-02b phase 1 transient decay heat removal system condensate flow comparison

{{

}}2(a),(b),(c),ECI

Figure 5-66a NLT-02b phase 1 transient decay heat removal system condensate flow comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-67 NLT-02b phase 1 transient steam generator outlet temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-68 NLT-02b phase 1 transient cooling pool vessel level comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-69 NLT-02b phase 1 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger)

{{

}}2(a),(b),(c),ECI

Figure 5-69a NLT-02b phase 1 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger)

{{

}}2(a),(b),(c),ECI

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Figure 5-69b NLT-02b phase 1 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger)

{{

}}2(a),(b),(c),ECI

Figure 5-70 NLT-02b phase 1 transient cooling pool vessel region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region)

{{

}}2(a),(b),(c),ECI

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5.3.3.8 NLT-2b Phase 2 Test Results

Code-to-data comparisons of key parameters for the second phase of the NLT-2b test are shown in Figure 5-71 through Figure 5-88. Note that in figures, the data measurement uncertainty is shown in dotted lines. {{

}}2(a),(c)

In general, the NRELAP5 simulation showed reasonable to excellent agreement with the measured data for this time period, except for the CPV temperatures adjacent to and above the DHRS heat exchanger. {{

}}2(a),(c)

It is noted that in the simulation, the fluid temperatures at and above the DHRS heat exchanger in the CPV warm up to saturation. {{

}}2(a),(c)

Figure 5-71 shows the core power comparison. The core power from the test was input directly into NRELAP5.

Figure 5-72 shows that the PZR pressure {{

}}2(a),(c) Figure 5-72a presents the same comparison as Figure 5-72, but with the Reference 29 version of NRELAP5.

Figure 5-73 shows that the PZR level {{

}}2(a),(c)

Figure 5-73a presents the same comparison as Figure 5-73, but with the Reference 29 version of NRELAP5.

Figure 5-75 shows the core inlet and outlet temperature comparisons for phase 2. The agreement is reasonable to excellent. This agreement indicates that the total energy added to the RPV fluid is reasonably calculated by the model. The

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reasonably predicted temperature indicates the RPV fluid volume change due to density change should also be reasonably predicted. Figure 5-75a presents the same comparison as Figure 5-75, but with the Reference 29 version of NRELAP5.

As discussed in Section 5.3.3.7 for Phase 1, the RPV loop flow rate is not compared.

Figure 5-77 compares the SG pressure data to the NRELAP5 prediction. The SG pressure {{

}}2(a),(c)

Figure 5-77a presents the same comparison as Figure 5-77, but with the Reference 29 version of NRELAP5.

Comparisons of the SG power and the DHRS HX thermal power are shown on Figure 5-78 and Figure 5-79, respectively. The thermal powers for both SG and DHRS track the data reasonably well {{ }}2(a),(c)

The SG level (Figure 5-80) and DHRS heat exchanger level (Figure 5-85) compare reasonably well to the data. {{

}}2(a),(c) Figure 5-85a presents the same comparison as Figure 5-85, but with the Reference 29 version of NRELAP5.

The DHRS condensate temperature (Figure 5-82) follows the same trends as the data, but similar to phase 1 the code prediction is about 50°F higher than the data. Figure 5-82a presents the same comparison as Figure 5-82, but with data from a different location and using the Reference 29 version of NRELAP5.

Figure 5-83 compares the DHRS condensate flow. The DHRS flow {{

}}2(a),(c)

Figure 5-83a presents the same comparison as Figure 5-83, but with the Reference 29 version of NRELAP5.

The SG tube exit temperatures are compared in Figure 5-84. Similar to phase 1, {{ }}2(a),(c) For reference, the fluid temperature in the steam bustle is also shown in the figure. The steam temperature increases with core power {{

}}2(a),(c)

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Figure 5-86 compares the measured and calculated CPV liquid level. The comparison shows reasonable agreement when recognizing that the NRELAP5 level increase is due to thermal expansion, where the NRELAP5 liquid temperatures in the upper CPV reach saturation temperature, well above the data temperature (Figure 5-88). The CPV fluid temperatures were reset at the start of the restart run to match the data after stirring the CPV, hence the good level match at the start of phase 2 (Figure 5-87, Figure 5-87a, Figure 5-87b, and Figure 5-88). {{

}}2(a),(c)

Figure 5-71 NLT-02b phase 2 transient core power comparison

{{

}}2(a),(b),(c),ECI

Figure 5-72 NLT-02b phase 2 transient pressurizer pressure comparison

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}}2(a),(b),(c),ECI

Figure 5-72a NLT-02b phase 2 transient pressurizer pressure comparison

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}}2(a),(b),(c),ECI

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Figure 5-73 NLT-02b phase 2 transient pressurizer level comparison

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}}2(a),(b),(c),ECI

Figure 5-73a NLT-02b phase 2 transient pressurizer level comparison

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}}2(a),(b),(c),ECI

Figure 5-74 Not Used

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Figure 5-75 NLT-02b phase 2 transient core inlet and outlet temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-75a NLT-02b phase 2 transient core inlet and outlet temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-76 Not Used

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Figure 5-77 NLT-02b phase 2 transient steam generator steam pressure comparison

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}}2(a),(b),(c),ECI

Figure 5-77a NLT-02b phase 2 transient steam generator steam pressure comparison

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}}2(a),(b),(c),ECI

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Figure 5-78 NLT-02b phase 2 transient steam generator thermal power comparison

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}}2(a),(b),(c),ECI

Figure 5-79 NLT-02b phase 2 transient decay heat removal system heat exchanger thermal power comparison

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}}2(a),(b),(c),ECI

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Figure 5-80 NLT-02b phase 2 transient steam generator level comparison

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}}2(a),(b),(c),ECI

Figure 5-81 NLT-02b phase 2 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization

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}}2(a),(b),(c),ECI

Figure 5-81a NLT-02b phase 2 integrated compensation flow for SG and DHRS HX level equalization

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}}2(a),(b),(c),ECI

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Figure 5-82 NLT-02b phase 2 transient decay heat removal system condensate temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-82a NLT-02b phase 2 transient decay heat removal system condensate temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-83 NLT-02b phase 2 transient decay heat removal system condensate flow comparison

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}}2(a),(b),(c),ECI

Figure 5-83a NLT-02b phase 2 transient decay heat removal system condensate flow comparison

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}}2(a),(b),(c),ECI

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Figure 5-84 NLT-02b phase 2 transient steam generator outlet temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-85 NLT-02b phase 2 transient decay heat removal system heat exchanger level comparison

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}}2(a),(b),(c),ECI

Figure 5-85a NLT-02b phase 2 transient decay heat removal system heat exchanger level comparison

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}}2(a),(b),(c),ECI

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Figure 5-86 NLT-02b phase 2 transient cooling pool vessel level comparison

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}}2(a),(b),(c),ECI

Figure 5-87 NLT-02b phase 2 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

Figure 5-87a NLT-02b phase 2 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

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Figure 5-87b NLT-02b phase 2 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

Figure 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)

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}}2(a),(b),(c),ECI

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5.3.3.9 NLT-2b Phase 3 Test Results

Code-to-data comparisons of key parameters for the third phase of the NLT-2b test are shown in Figure 5-89 through Figure 5-106. Note that in figures, the data measurement uncertainty is shown in dotted lines.

{{

}}2(a),(c)

In general, the NRELAP5 simulation showed reasonable to excellent agreement with the measured data for this time period.

A comparison of the SG and DHRS heat exchanger tube levels are shown in Figure 5-98 and Figure 5-103, respectively. The SG level calculated response exhibits excellent agreement with the measured data, and the calculated DHRS HX tube level shows reasonable agreement. During this phase, as seen in Figure 5-99, {{

}}2(a),(c) Figure 5-99a shows that there is {{

}}2(a),(c) Figure 5-103a presents the same comparison as Figure 5-103, but with the Reference 29 version of NRELAP5.

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Figure 5-96 and Figure 5-97 compare the heat removal in the SG and DHRS heat exchanger tubes, respectively. {{ }}2(a),(c) Figure 5-95 shows that the SG pressure is reasonably predicted in this phase. Figure 5-95a presents the same comparison as Figure 5-95, but with the Reference 29 version of NRELAP5.

The PZR pressure is compared in Figure 5-90. {{ }}2(a),(c) Figure 5-90a presents the same comparison as Figure 5-90, but with the Reference 29 version of NRELAP5.

The calculated PZR level is compared to data in Figure 5-91. {{ }}2(a),(c) Figure 5-91a presents the same comparison as Figure 5-91, but with the Reference 29 version of NRELAP5.

The core inlet and outlet temperatures are compared in Figure 5-93. Reasonable agreement is observed in these figures, {{ }}2(a),(c) Figure 5-93a presents the same comparison as Figure 5-93, but with the Reference 29 version of NRELAP5.

As discussed in Section 5.3.3.7 for Phase 1, the RPV loop flow rate is not compared.

Figure 5-101 compares the measured and calculated DHRS condensate flow. {{ }}2(a),(c) Figure 5-101a presents the same comparison as Figure 5-101, but with the Reference 29 version of NRELAP5.

SG tube exit temperatures are compared in Figure 5-102. The predicted temperature is {{ }}2(a),(c)

The condensate fluid temperature is compared in Figure 5-100. {{ }}2(a),(c)

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{{

}}2(a),(c)

Figure 5-104 compares the measured and calculated CPV liquid level. The comparison shows reasonable agreement with the data out to the time of the drain and refill of the CPV {{

}}2(a),(c)

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Figure 5-89 NLT-02b phase 3 transient core power comparison

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}}2(a),(b),(c),ECI

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Figure 5-90 NLT-02b phase 3 transient pressurizer pressure comparison

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}}2(a),(b),(c),ECI

Figure 5-90a NLT-02b phase 3 transient pressurizer pressure comparison

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}}2(a),(b),(c),ECI

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Figure 5-91 NLT-02b phase 3 transient pressurizer level comparison

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}}2(a),(b),(c),ECI

Figure 5-91a NLT-02b phase 3 transient pressurizer level comparison

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}}2(a),(b),(c),ECI

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Figure 5-92 Not Used

Figure 5-93 NLT-02b phase 3 transient core inlet and outlet temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-93a NLT-02b phase 3 transient core inlet and outlet temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-94 Not Used

Figure 5-95 NLT-02b phase 3 transient steam generator steam pressure comparison

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}}2(a),(b),(c),ECI

Figure 5-95a NLT-02b phase 3 transient steam generator steam pressure comparison

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}}2(a),(b),(c),ECI

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Figure 5-96 NLT-02b phase 3 transient steam generator thermal power comparison

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}}2(a),(b),(c),ECI

Figure 5-97 NLT-02b phase 3 transient decay heat removal system heat exchanger thermal power comparison

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}}2(a),(b),(c),ECI

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Figure 5-98 NLT-02b phase 3 transient steam generator level comparison

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}}2(a),(b),(c),ECI

Figure 5-99 NLT-02b phase 3 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization

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}}2(a),(b),(c),ECI

Figure 5-99a NLT-02b phase 3 integrated compensation flow for SG and DHRS HX level equalization

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}}2(a),(b),(c),ECI

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Figure 5-100 NLT-02b phase 3 transient decay heat removal system condensate temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-100a NLT-02b phase 3 transient decay heat removal system condensate temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-101 NLT-02b phase 3 transient decay heat removal system condensate flow comparison

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}}2(a),(b),(c),ECI

Figure 5-101a NLT-02b phase 3 transient decay heat removal system condensate flow comparison

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}}2(a),(b),(c),ECI

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Figure 5-102 NLT-02b phase 3 transient steam generator outlet temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-103 NLT-02b phase 3 transient decay heat removal system heat exchanger level comparison

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}}2(a),(b),(c),ECI

Figure 5-103a NLT-02b phase 3 transient decay heat removal system heat exchanger level comparison

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}}2(a),(b),(c),ECI

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Figure 5-104 NLT-02b phase 3 transient cooling pool vessel level comparison

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}}2(a),(b),(c),ECI

Figure 5-105 NLT-02b phase 3 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

Figure 5-105a NLT-02b phase 3 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

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Figure 5-105b NLT-02b phase 3 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

Figure 5-106 NLT-02b phase 3 transient cooling pool vessel region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region)

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}}2(a),(b),(c),ECI

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5.3.3.10 NLT-2b Phase 4 Test Results

Code-to-data comparisons of key parameters for the fourth phase of the NLT-2b test are shown in Figure 5-107 through Figure 5-125. Note that in figures, the data measurement uncertainty is shown in dotted lines.

{{

}}2(a),(c) the fluid volume had shrunk sufficiently to uncover the top of the riser pipe (located at the bottom of the upper plenum). Consistent with the scope of the short-term non-LOCA transient EM, the results presented here focus on the time after the decay heat mode was actuated until top of the riser uncovered.

The simulation gave reasonable results for this phase out to the point of riser uncovery.

Measured and calculated core heater rod power is compared in Figure 5-107. The core heater rod power is a boundary condition in the simulation.

Energy continued to be removed through the SG tube coil {{

}}2(a),(c) The energy removal resulted in a cooldown of the primary system. The cooldown caused a shrinkage of the RPV fluid volume and corresponding decrease in RPV pressure (Figure 5-108). Figure 5-108a presents the same comparison as Figure 5-108, but with the Reference 29 version of NRELAP5. The level in the PZR declined due to the system cooldown as shown in Figure 5-109. The rate of PZR level decline in the calculation is in reasonable agreement with the measured data. {{

}}2(a),(c) Figure 5-109a presents the same comparison as Figure 5-109, but with the Reference 29 version of NRELAP5.

As discussed in Section 5.3.3.7 for Phase 1, the RPV loop flow rate is not compared.

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A comparison of the measured and calculated upper plenum fluid temperature and the core inlet and outlet temperatures are provided in Figure 5-111 and Figure 5-112, respectively. {{

}}2(a),(c) Figure 5-112a presents the same comparison as Figure 5-112, but with the Reference 29 version of NRELAP5.

The SG pressure is compared in Figure 5-114. {{

}}2(a),(c) Figure 5-114a presents the same comparison as Figure 5-114, but with the Reference 29 version of NRELAP5.

Figure 5-115 and Figure 5-116 show that both the SG and DHRS power {{

}}2(a),(c)

Figure 5-118 and Figure 5-118a show {{

}}2(a),(c)

Figure 5-117 and Figure 5-122 show comparisons on the SG tube level and DHRS HX tube level, respectively. {{

}}2(a),(c) Figure 5-122a presents the same comparison as Figure 5-122, but with the Reference 29 version of NRELAP5.

Figure 5-119 compares the measured and calculated DHRS condensate temperature. {{

}}2(a),(c) Figure 5-119a presents the same comparison as Figure 5-119, but with data from a different location and using the Reference 29 version of NRELAP5.

Figure 5-120 compares the measured and calculated DHRS condensate flow. Figure 5-120a presents the same comparison as Figure 5-120, but with the Reference 29 version of NRELAP5.

A comparison of the SG outlet temperature is provided in Figure 5-121. The calculated outlet temperature shows {{

}}2(a),(c)

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The measured and calculated CPV level comparison is shown in Figure 5-123. After the drain and fill process, {{
}}2(a),(c), the calculated level shows reasonable agreement with the data.

Measured and calculated CPV fluid temperatures within the elevation of the DHRS heat exchanger are shown in Figure 5-124, Figure 5-124a, Figure 5-124b, and Figure 5-125. {{

}}2(a),(c)

Figure 5-107 NLT-02b phase 4 transient core power comparison

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}}2(a),(b),(c),ECI

Figure 5-108 NLT-02b phase 4 transient pressurizer pressure comparison

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}}2(a),(b),(c),ECI

Figure 5-108a NLT-02b phase 4 transient pressurizer pressure comparison

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}}2(a),(b),(c),ECI

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Figure 5-109 NLT-02b phase 4 transient pressurizer level comparison

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}}2(a),(b),(c),ECI

Figure 5-109a NLT-02b phase 4 transient pressurizer level comparison

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}}2(a),(b),(c),ECI

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Figure 5-110 NLT-02b phase 4 transient reactor pressure vessel level

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}}2(a),(b),(c),ECI

Figure 5-111 NLT-02b phase 4 transient reactor pressure vessel upper plenum temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-112 NLT-02b phase 4 transient core inlet and outlet temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-112a NLT-02b phase 4 transient core inlet and outlet temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-113 Not Used

Figure 5-114 NLT-02b phase 4 transient steam generator steam pressure comparison

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}}2(a),(b),(c),ECI

Figure 5-114a NLT-02b phase 4 transient steam generator steam pressure comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-115 NLT-02b phase 4 transient steam generator thermal power comparison

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}}2(a),(b),(c),ECI

Figure 5-116 NLT-02b phase 4 transient decay heat removal system heat exchanger thermal power comparison

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}}2(a),(b),(c),ECI

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Figure 5-117 NLT-02b phase 4 transient steam generator level comparison

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}}2(a),(b),(c),ECI

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Figure 5-118 NLT-02b phase 4 calculated compensation flow for steam generator and decay heat removal system heat exchanger level equalization

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}}2(a),(b),(c),ECI

Figure 5-118a NLT-02b phase 4 integrated compensation flow for SG and DHRS HX level equalization

{{

}}2(a),(b),(c),ECI

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Figure 5-119 NLT-02b phase 4 transient decay heat removal system condensate temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-119a NLT-02b phase 4 transient decay heat removal system condensate temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-120 NLT-02b phase 4 transient decay heat removal system condensate flow comparison

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}}2(a),(b),(c),ECI

Figure 5-120a NLT-02b phase 4 transient decay heat removal system condensate flow comparison

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}}2(a),(b),(c),ECI

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Figure 5-121 NLT-02b phase 4 transient steam generator outlet temperature comparison

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}}2(a),(b),(c),ECI

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Figure 5-122 NLT-02b phase 4 transient decay heat removal system heat exchanger level comparison

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}}2(a),(b),(c),ECI

Figure 5-122a NLT-02b phase 4 transient decay heat removal system heat exchanger level comparison

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}}2(a),(b),(c),ECI

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Figure 5-123 NLT-02b phase 4 transient cooling pool vessel level comparison

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}}2(a),(b),(c),ECI

Figure 5-124 NLT-02b phase 4 transient cooling pool vessel region 4 temperature comparison (below decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

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Figure 5-124a NLT-02b phase 4 transient cooling pool vessel region 5 temperature comparison (near bottom of decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

Figure 5-124b NLT-02b phase 4 transient cooling pool vessel region 6 temperature comparison (near mid-point of decay heat removal system heat exchanger)

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}}2(a),(b),(c),ECI

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Figure 5-125 NLT-02b phase 4 transient cooling pool vessel region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region)

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}}2(a),(b),(c),ECI

5.3.3.11 NLT-2 Summary

The comparisons between the NIST-1 NLT-2a integral experimental data and the code calculated values showed that NRELAP5 has the capability to accurately predict primary heat up and pressurization due to a loss of feedwater.

NLT-02b is a transient test with quasi-steady core heater rod power during an extended DHRS recirculation mode and with core heater rod decay power during a DHRS-driven decay power mode. During the DHRS recirculation mode, the transient response of pressure, level, and temperature in both the RPV and the DHRS were predicted by NRELAP5 with reasonable agreement.

NRELAP5 assessment of the NLT-2b phases 1 through 4 demonstrates the ability of the code to predict the heat transfer from primary side to the SG and from the DHRS to CPV. After completion of core power maneuvering and after full start of the DHRS loop, NRELAP5 calculated pressurizer pressure, core inlet and outlet temperature, primary flow rate, SG pressure, energy transfer from the primary to secondary, energy transfer from DHRS to CPV, SG level, DHRS level, and CPV level results are all within reasonable to excellent agreement.

The pressurizer level for NLT-2b phase 1 was {{

}}2(a),(b),(c)

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{{

}}2(a),(b),(c) However, even with these discrepancies, the results show that the important parameters of total energy transfer from the primary side to the SG and from the DHRS to CPV are well predicted.

For NLT-2b phases 1 through 4, the calculated condensate line temperature is in minimal agreement with data, with the temperature being over predicted by NRELAP5. However, even with the discrepancy, the important parameter of heat transfer from primary side to the SG and from the DHRS to CPV are well predicted.

For NLT-2b phases 1 through 4, the CPV fluid heat-up profiles were not fully captured in the NRELAP5 simulations. {{

}}2(a),(c)

For NLT-2b phases 1 through 4, code-to-data comparisons show converged reasonable agreement of DHRS heat removal rate, which is considered to be the most important parameter in this assessment, demonstrating that NRELAP5 is capable of capturing total energy removal rate of the DHRS to the cooling pool.

The re-performance of the comparisons using the Reference 29 version of NRELAP5 shows the same level of agreement as for the original comparisons.

5.3.3.12 NLT-15-p2 Test Description

The objective of the NLT-15-p2 test was to measure the scaled integral system response to a decrease in heat removal from the secondary side following a loss of feedwater event, followed by core decay heat removal via the DHRS. During the test, {{

}}2(a),(c)

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4. {{

}}2(a),(c)

The sequence of events for the NLT-15p2 test is shown in Table 5-14

Table 5-14 NLT-15p2 sequence of events

{{

}}2(a),(c)

5.3.3.13 NLT-15 p2 Test Results

{{

}}2(a),(c)

Comparison of NRELAP5 calculated parameters to the measured data are presented below. Note that in the figures, the data uncertainty is shown in dot-dashed lines. A comparison of the primary side behavior is discussed first. This comparison is followed by a discussion of the startup of the DHRS (first 500 seconds). A comparison of the secondary side is then presented, and finally, a discussion of the CPV behavior concludes this section.

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Primary side response

Figure 5-126 shows the core heater rod power. As expected, the predicted power is an overlay of the measured power since core power is a boundary condition for the simulation.

Pressurizer pressure for short term is shown in Figure 5-127. {{

}}2(a),(c) The calculated pressure in the short term, compared to the measured data, is considered reasonable.

The long term pressurizer pressure comparison is shown in Figure 5-128.
{{

}}2(a),(c) Figure 5-128a presents the same comparison as Figure 5-128, but with the Reference 29 version of NRELAP5.

{{

}}2(a),(c) Long term, the pressurizer level response shows excellent agreement with the measured data. Figure 5-129a presents the same comparison as Figure 5-129, but with the Reference 29 version of NRELAP5.

Once DHRS loop flow is established, heat removal through the steam generator tube coil cools the RPV fluid. The liquid volume in the primary loop shrinks as the RPV system cools. After the pressurizer empties, a decline in the RPV level below the pressurizer baffle plate continues as shown in Figure 5-130. This figure compares the calculated and measured RPV level. As observed, the NRELAP5

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comparison of the RPV level shows excellent agreement with the measured data, and thus gives credence that NRELAP5 correctly calculates the change in volume due to fluid cooldown. {{

}}2(a),(c) Figure 5-130a presents the same comparison as Figure 5-130, but with the Reference 29 version of NRELAP5.

The RPV riser flow rate {{

}}2(a),(c)

Figure 5-132 through Figure 5-134 compare calculated RPV loop temperatures to the measured data for the core inlet, the riser inlet, and the upper plenum. Reasonable to excellent agreement is observed in the fluid temperatures. {{

}}2(a),(c) Figure 5-133a presents the same comparisons as Figure 5-132 and Figure 5-133, but with the Reference 29 version of NRELAP5. Similarly, Figure 5-134a presents the same comparison as Figure 5-134, but with the Reference 29 version of NRELAP5.

DHRS Startup

The sequence of events shows {{

}}2(a),(c)

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{{

}}2(a),(c)

The measured DHRS HX level is shown in Figure 5-138. {{

}}2(a),(c)

{{

}}2(a),(c) the response of NRELAP5 is judged to be reasonable.

DHRS and Steam Generator Long Term Behavior

A comparison of the short term and long term steam line pressure response is given in Figure 5-135 and in Figure 5-139, respectively. {{

}}2(a),(c) Figure 5-139a presents the same comparison as Figure 5-139, but with the Reference 29 version of NRELAP5.

The comparison of the DHRS HX inlet and outlet temperatures are shown in Figure 5-140 and Figure 5-141, respectively. {{

}}2(a),(c) The overall response of the predicted inlet temperature is deemed reasonable. Figure 5-140a presents the same comparison as Figure 5-140, but with the Reference 29 version of NRELAP5.

The DHRS HX outlet temperature shown in Figure 5-141 {{

}}2(a),(c)

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Figure 5-142 and Figure 5-143 compares the short term and long term DHRS loop mass flow rate, respectively. {{

}}2(a),(c) Overall, the predicted DHRS flow rate follows the trends of the data and is deemed reasonable. Figure 5-143a presents the same comparison as Figure 5-143, but with the Reference 29 version of NRELAP5.

Comparisons of the level in the DHRS HX and in the steam generator tube coil are shown in Figure 5-144 and Figure 5-145. In general, the predicted DHRS HX level compares reasonably with the measured data. {{

}}2(a),(c)
NRELAP5 shows reasonable agreement with the data trends. Figure 5-144a presents the same comparison as Figure 5-144, but with the Reference 29 version of NRELAP5.

Differential pressure comparisons across the condensate line and across the DHRS steam line are given in Figure 5-147 and Figure 5-148. {{

}}2(a),(c)

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{{

}}2(a),(c) The calculation of the heat removal through the steam generator tube coil is deemed reasonable. Figure 5-149a presents the same comparison as Figure 5-149, but with the Reference 29 version of NRELAP5.

Figure 5-150 compares the measured heat removal rate through the DHRS HX tubes with the predicted results. {{

}}2(a),(c) Again, overall, the trend of the data is represented and is deemed reasonable. Figure 5-150a presents the same comparison as Figure 5-150, but with the Reference 29 version of NRELAP5.

CPV Behavior

A comparison of the CPV liquid level and the liquid temperatures are provided here.

Figure 5-151 compares the measured CPV level with the calculated level. {{

}}2(a),(c) The prediction of the CPV level is deemed reasonable.

Figure 5-152 through Figure 5-157 compares the measured and calculated CPV fluid temperature throughout the CPV. {{

}}2(a),(c)

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}}2(a),(c)

Figure 5-126 NLT-15p2, transient core power

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}}2(a),(b),(c),ECI

Figure 5-127 NLT-15p2, transient RPV pressure short term

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}}2(a),(b),(c),ECI

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Figure 5-128 NLT-15p2, transient RPV pressure

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}}2(a),(b),(c),ECI

Figure 5-128a NLT-15p2, transient RPV pressure

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}}2(a),(b),(c),ECI

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Figure 5-129 NLT-15p2, transient pressurizer level

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}}2(a),(b),(c),ECI

Figure 5-129a NLT-15p2, transient pressurizer level

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}}2(a),(b),(c),ECI

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Figure 5-130 NLT-15p2, transient RPV level

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}}2(a),(b),(c),ECI

Figure 5-130a NLT-15p2, transient RPV level

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}}2(a),(b),(c),ECI

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Figure 5-131 NLT-15p2, transient riser mass flow rate

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}}2(a),(b),(c),ECI

Figure 5-132 NLT-15p2, transient core inlet temperature

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}}2(a),(b),(c),ECI

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Figure 5-133 NLT-15p2, transient riser inlet temperature

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}}2(a),(b),(c),ECI

Figure 5-133a NLT-15p2, transient core inlet and riser inlet temperatures

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}}2(a),(b),(c),ECI

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Figure 5-134 NLT-15p2, transient upper plenum temperature

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}}2(a),(b),(c),ECI

Figure 5-134a NLT-15p2, transient upper plenum temperature

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}}2(a),(b),(c),ECI

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Figure 5-135 NLT-15p2, transient secondary side pressure - 0 to 500 seconds

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}}2(a),(b),(c),ECI

Figure 5-136 NLT-15p2, transient DHRS loop flow - 0 to 500 seconds

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}}2(a),(b),(c),ECI

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Figure 5-137 NLT-15p2, transient measured steam line temperatures - 0 to 500 seconds

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}}2(a),(b),(c),ECI

Figure 5-138 NLT-15p2, transient DHRS HX level - 0 to 500 seconds

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}}2(a),(b),(c),ECI

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Figure 5-139 NLT-15p2, transient secondary side pressure

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}}2(a),(b),(c),ECI

Figure 5-139a NLT-15p2, transient secondary side pressure

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}}2(a),(b),(c),ECI

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Figure 5-140 NLT-15p2, transient DHRS HX inlet temperature

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}}2(a),(b),(c),ECI

Figure 5-140a NLT-15p2, transient DHRS HX inlet temperature

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}}2(a),(b),(c),ECI

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Figure 5-141 NLT-15p2, transient DHRS HX outlet temperature

{{

}}2(a),(b),(c),ECI

Figure 5-142 NLT-15p2, transient DHRS loop flow - short term

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}}2(a),(b),(c),ECI

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Figure 5-143 NLT-15p2, transient DHRS loop flow rate - long term

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}}2(a),(b),(c),ECI

Figure 5-143a NLT-15p2, transient DHRS loop flow rate - long term

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}}2(a),(b),(c),ECI

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Figure 5-144 NLT-15p2, transient DHRS HX level

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}}2(a),(b),(c),ECI

Figure 5-144a NLT-15p2, transient DHRS HX level

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}}2(a),(b),(c),ECI

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Figure 5-145 NLT-15p2, transient steam generator tube coil level - long term

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}}2(a),(b),(c),ECI

Figure 5-146 NLT-15p2, transient steam generator tube coil level - short term

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}}2(a),(b),(c),ECI

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Figure 5-147 NLT-15p2, transient DHRS condensate line differential pressure

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}}2(a),(b),(c),ECI

Figure 5-148 NLT-15p2, transient DHRS steam line differential pressure

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}}2(a),(b),(c),ECI

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Figure 5-149 NLT-15p2, transient steam generator tube coil power removal

{{

}}2(a),(b),(c),ECI

Figure 5-149a NLT-15p2, transient steam generator tube coil power removal

{{

}}2(a),(b),(c),ECI

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Figure 5-150 NLT-15p2, transient DHRS power removal

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}}2(a),(b),(c),ECI

Figure 5-150a NLT-15p2, transient DHRS power removal

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}}2(a),(b),(c),ECI

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Figure 5-151 NLT-15p2, transient cooling pool vessel level

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}}2(a),(b),(c),ECI

Figure 5-152 NLT-15p2, transient cooling pool vessel fluid temperature at level 3 (below decay heat removal heat exchanger)

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}}2(a),(b),(c),ECI

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Figure 5-153 NLT-15p2, transient cooling pool vessel fluid temperature at level 5 (near bottom of DHRS heat exchanger)

{{

}}2(a),(b),(c),ECI

Figure 5-154 NLT-15p2, transient cooling pool vessel fluid temperature at level 6 (near midpoint of DHRS heat exchanger)

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}}2(a),(b),(c),ECI

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Figure 5-155 NLT-15p2, transient cooling pool vessel fluid temperature at level 7 (top to just above DHRS heat exchanger region)

{{

}}2(a),(b),(c),ECI

Figure 5-156 NLT-15p2, transient cooling pool vessel fluid temperature at level 8 (above DHRS heat exchanger region)

{{

}}2(a),(b),(c),ECI

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Figure 5-157 NLT-15p2, transient cooling pool vessel fluid temperature at level 9 (above DHRS heat exchanger region)

{{

}}2(a),(b),(c),ECI

5.3.3.14 NLT-15p2 Summary

Test NLT-15p2 is a loss of feedwater scenario followed by core decay heat removal via the DHRS system. {{

}}2(a),(c)

The NRELAP5 simulation of this test showed reasonable to excellent agreement with the initial pressure and loop temperature response in the RPV. After isolation of the secondary side and during the initial stages of DHRS heat removal operation, the RPV pressure rose due to heat addition from the core decay power and lack of complete removal through the steam generator tube coil. A rise in the pressurizer liquid level was also observed. Both the rise in the RPV pressure and the rise in the pressurizer level were well predicted by NRELAP5.

After DHRS heat removal flow was established, the RPV pressure turned over and declined throughout the remainder of the test. NRELAP5 also predicted this pressure behavior in general.

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Both the pressurizer level response and the RPV level response showed excellent agreement with the measured response, thus showing that NRELAP5 can accurately predict change in fluid volume due to cooldown.

The predicted RPV loop temperatures showed reasonable to excellent agreement with the measured data demonstrating that heat removal through the steam generator tube coil was reasonably predicted.

Prediction of the secondary side behavior showed reasonable agreement with the measured data. All major trends of the data were modeled correctly by NRELAP5. {{

}}2(a),(c) However, NRELAP5 reasonably calculated the trends of the data.

{{

}}2(a),(c)

The calculated fluid temperature inside the DHRS heat exchanger tubes generally predicted the trends of the measured data. {{

}}2(a),(c)

The calculated differential pressure in the DHRS condensate line and steam line {{

}}2(a),(c)

Overall, NRELAP5 is capable of producing reasonable results in simulating events such as those observed in the NLT-15p2 test.

The re-performance of the comparisons using the Reference 29 version of NRELAP5 shows the same level of agreement as for the original comparisons.

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5.3.4 Code to Code Benchmark for Integral Assessment of Reactivity Event Response

5.3.4.1 Background

The NRELAP5 code is used in the non-LOCA evaluation model to perform non-LOCA system transient analysis for an NPM. A series of code-to-code benchmarking comparison calculations was performed with the RETRAN-3D code to validate the performance of NRELAP5's point kinetics model during the reactivity transient events and to supplement the validation of the integral primary system thermal-hydraulic response to reactivity transients.

RETRAN-3D was developed to perform transient thermal-hydraulic analysis of light water reactors. RETRAN-3D was developed as an evolution to the RETRAN codes that have been sponsored by EPRI since 1975 and used for licensing basis analyses of commercial light water reactors in the U.S.

The RETRAN-3D code is based on the one-dimensional homogeneous equilibrium model, in comparison to the two-fluid, non-equilibrium non-homogeneous fluid model utilized in the NRELAP5 code.

The NRC reviewed RETRAN-3D and issued a Safety Evaluation Report indicating that RETRAN-3D is an acceptable tool for performing PWR licensing analyses for a series of categories of anticipated transients, infrequent incidents and accident analyses for PWRs, including the reactivity transient events (Reference 12). Compared to operating PWRs, an NPM natural circulation primary system flow and helical coil SG design are unique features. For NRELAP5, a new helical coil SG component was added to the NRELAP5 code as described in Reference 2; primary and secondary side fluid flow and heat transfer over the SG were validated against testing performed at the SIET-TF1 and SIET-TF2 facilities, as described in Section 5.3.5. RETRAN-3D does not include specific models for helical coil SG heat transfer and wall friction; as described in Section 5.3.4.2, for the purposes of the benchmark calculations, a modeling simplification was made such that the RETRAN-3D and NRELAP5 primary side heat transfer coefficients were consistent under steady state conditions. Therefore, the scope of the benchmark calculation is focused on the reactivity response and the integral primary side response during a reactivity transient.

The scope of the code-to-code benchmarking includes comparisons of reactor power, reactivity, primary side flow, pressure and temperatures for a set of benchmarking reactivity transients. These stylized benchmark cases are designed so as to be representative of the NuScale reactivity transients, but simplified to minimize impact of model differences in the secondary side or control system responses that are not the focus of the benchmark comparisons.

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5.3.4.2 Approach for RETRAN-3D Benchmark

To perform the code-to-code benchmark calculations, an NPM was modeled using both NRELAP5 and RETRAN-3D. The NRELAP5 base model used for code-to-code benchmarking was consistent with the base model described in Section 6.0 of this report.

The RETRAN-3D base model was developed based on the NRELAP5 base model. The NRELAP5 model provided the geometric information, trips, control systems and reactor kinetics to convert to RETRAN-3D input cards.

The NRELAP5 model volume, junction and heat conductor (heat structure) input were used as the initial basis for the RETRAN-3D model input and nodalization. {{

}}2(a),(c)

For NRELAP5, a new helical coil SG component was added to the NRELAP5 code as described in Reference 2 to model the helical coil SG heat transfer and wall friction in an NPM design. In RETRAN-3D there is no helical coil model, {{

}}2(a),(c)

Differences in the predicted pressurizer pressure and level were observed in the benchmark calculation results due to differences between the pressurizer modeling in the NRELAP5 and RETRAN-3D calculations. {{

}}2(a),(c) The RETRAN-3D thermal-hydraulics computer code used in this study, the version with a safety evaluation report (SER) from the NRC, uses a basic three-equation homogeneous equilibrium model. When two-phase, the constraint on this representation is equilibrium (saturation) between the phases of water. When fluid surges into or out of the pressurizer, steam can be compressed or water can flash, which is why RETRAN-3D has a two-region non-equilibrium pressurizer model that allows non-equilibrium between the phases of water.

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RETRAN-3D also includes a volume change term in the phasic energy equations for the two-region component. The RETRAN-3D two region model allows the user limited control at the interface. Additional interfacial modeling is available in RETRAN-3D when using the sub-node option in the code. Comparisons between the NRELAP5 code and two region models (Reference 13, for example) can show larger pressures in a two-region component than in NRELAP5 for in-surge transients. This difference is due to some degree because of work terms for volume change and using enthalpy in the formulation. RETRAN-3D uses enthalpy at the junctions for the energy equations whereas NRELAP5 uses internal energy. Enthalpy contains internal energy in addition to the flow work term, which NRELAP5 does not include in its formulation, except for special application models for blow down.

There are also differences in pressurizer spray, the letdown and charging models, along with insurge for the two codes. Contrary to the NRELAP5 model that uses a standard junction to model spray, the RETRAN-3D model uses a special spray junction in the pressurizer spray system that condenses sufficient steam from the vapor region to bring the spray flow to saturation. This model involves removal of mass and energy from the vapor region of the pressurizer and depositing the mass of the spray and condensed vapor directly into the liquid region of the pressurizer without a time delay, which leads to higher pressurizer pressure.

After the RETRAN-3D base deck was developed, a steady-state calculation was performed where adequately consistent results between the two code calculations were obtained. Then, four transient calculations were performed. These stylized transients were selected to cover a range of reactivity insertion rates and RCS response that are observed in NPM reactivity event calculations. The calculations performed were:

1. Uncontrolled rod withdrawal from full power steady-state, using a higher reactivity insertion rate to result in a high power MPS actuation signal (fast uncontrolled rod withdrawal [UCRW]).
2. Uncontrolled rod withdrawal from full power steady-state, using a lower reactivity insertion rate to result in a high pressurizer pressure MPS actuation signal (slow UCRW).
3. Power reduction from full power to 50 percent of rated power.
4. Dropped control rod assembly from 50 percent rated power steady state conditions, using a dropped rod worth that is relatively large, but insufficient to cause reactor trip.

{{

}}2(a),(c)

For the reactivity transients, the focus of the benchmark comparison is on the primary side response before scram, driven by the reactivity insertion and power

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change, and the short term responses after scram, driven by the scram worth. The longer term and secondary side response is not examined in detail due to the simplified modeling of the helical coil SG in the RETRAN-3D model, and the high pressure condensation conditions after DHRS actuation that are not typical of operating pressurized water reactors. Therefore the key parameters of interest for the reactivity benchmarking transients are reactor power, reactivity, pressurizer pressure and level, core flow, and core temperatures. These parameters are plotted and compared for each transient.

When performing code-to-code comparisons, agreement must be assessed in some manner. The method used herein is based on RG 1.203 (Reference 1) as described in Section 5.3. The acceptance criterion applied herein for the benchmarking calculations is that at least “reasonable agreement” must be observed.

5.3.4.3 Fast Uncontrolled Rod Withdrawal from Full Power

In this case, the uncontrolled rod withdrawal occurs at time zero from the full power steady state condition and a high reactivity insertion rate (13.36 pcm/sec) is assumed. This reactivity insertion rate causes a reactor scram on high power. The DHRS is not actuated on high power and therefore normal secondary side cooling continues. The automatic turbine trip following reactor trip is not credited. The steam flow varies depending on the SG pressure. {{

}}2(a),(c)

Figure 5-158 through Figure 5-164 show excellent agreement between the two models. At the end of the full power steady state initialization, the core flow, inlet temperatures and outlet temperatures are slightly different between the two models. These differences are carried over to the transient, which can be observed at time zero on Figure 5-162 through Figure 5-164. Due to these steady-state differences and their effects to the reactivity feedback, the core power increase rates are slightly different when the UCRW reactivity insertion is modeled. The RETRAN-3D model predicts a slightly earlier time to reach the high core power trip analytical limit (Figure 5-158 and Figure 5-159). The core power and reactivity curves are very close to each other between the two models during the transient. The pressurizer pressure and level match well during the short time after scram, and then the NRELAP5 model predicts a slightly higher pressurizer pressure and level (Figure 5-160 and Figure 5-161).

The core flow and outlet temperature match fairly well between the RETRAN-3D model and the NRELAP5 model, after considering the differences from the steady state condition at the transient initiation (Figure 5-162 and Figure 5-164). The timing of the core inlet temperature drop is slightly off (Figure 5-163), indicating the heat transfer to the secondary side is slightly different between the two models.

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Figure 5-158 Core power (fast uncontrolled rod withdrawal)

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}}2(a),(b),(c)

Figure 5-159 Total reactivity (fast uncontrolled rod withdrawal)

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}}2(a),(b),(c)

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Figure 5-160 Pressurizer pressure (fast uncontrolled rod withdrawal)

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}}2(a),(b),(c)

Figure 5-161 Pressurizer level (fast uncontrolled rod withdrawal)

{{

}}2(a),(b),(c)

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Figure 5-162 Core flow (fast uncontrolled rod withdrawal)

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}}2(a),(b),(c)

Figure 5-163 Core inlet temperatures (fast uncontrolled rod withdrawal)

{{

}}2(a),(b),(c)

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Figure 5-164 Core outlet temperatures (fast uncontrolled rod withdrawal)

{{

}}2(a),(b),(c)

5.3.4.4 Slow Uncontrolled Rod Withdrawal from Full Power

In this case, the uncontrolled rod withdrawal occurs at time zero from the full power steady state condition, and a lower reactivity insertion rate (5.344 pcm/sec) is assumed. This reactivity insertion results in RCS heatup and reactor scram from a high RCS pressure rather than a high power condition. During the slow UCRW transient, upon reactor trip on high RCS pressure, DHRS is actuated and therefore normal steam flow and feedwater flow are isolated.

In addition to the reduced reactivity insertion rate, pressurizer spray and CVCS were isolated on transient initiation to maximize the pressurization rate. The UCRW reactivity insertion is initiated at time zero.

Figure 5-165 through Figure 5-171 show reasonable to excellent agreement between the two codes. As explained in Section 5.3.4.3, at the end of the full power steady state initialization, the core flow, inlet and outlet temperatures are slightly different between the two models. These differences are carried over to the transient, which can be observed at time zero on Figure 5-169 through Figure 5-171. Due to these steady-state differences and their effects to the reactivity feedback, the rates of core power increase are slightly different when the UCRW reactivity insertion is modeled. The RETRAN-3D model predicts a slightly faster power increase rate than NRELAP5 (Figure 5-165). Due to the faster power increase and different pressurizer models (discussion in

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Section 5.3.4.2), RETRAN-3D predicts higher increase rates on the pressurizer pressure and level after transient initiation (Figure 5-167 and Figure 5-168), and therefore a slightly earlier timing of the reactor trip on high pressurizer pressure.

Following the reactor scram, the core power and reactivity curves are very close to each other between the two models during the transient (Figure 5-165 and Figure 5-166). Pressurizer pressure and level continue to increase and reach their peak values before decreasing. NRELAP5 predicts a slightly higher peak pressurizer pressure and level (Figure 5-167 and Figure 5-168). As the pressure and level decrease, RETRAN-3D has a faster initial depressurization resulting in a lower pressurizer pressure. Then the relative rate of decrease changes and the RETRAN-3D pressure is higher than the NRELAP5 pressure. {{

}}2(a),(b),(c) The pressurizer levels predicted by the two codes show similar behavior. The discrepancy is attributed to different heat removal rates from the pressurizer between the NRELAP5 and RETRAN-3D pressurizer models, due to differences in the pressurizer modeling as discussed in Section 5.3.4.2.

The core flow and outlet temperature match fairly well between the RETRAN-3D model and the NRELAP5 model, after considering the differences from the steady state condition at the transient initiation (Figure 5-169 and Figure 5-171). The core inlet temperatures shown the same trend and the biggest difference is approximately 2 percent (Figure 5-170). The core inlet temperature difference could be attributed to the core flow differences.

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Figure 5-165 Core power (slow uncontrolled rod withdrawal)

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}}2(a),(b),(c)

Figure 5-166 Total reactivity (slow uncontrolled rod withdrawal)

{{

}}2(a),(b),(c)

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Figure 5-167 Pressurizer pressure (slow uncontrolled rod withdrawal)

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}}2(a),(b),(c)

Figure 5-168 Pressurizer level (slow uncontrolled rod withdrawal)

{{

}}2(a),(b),(c)

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Figure 5-169 Core flow (slow uncontrolled rod withdrawal)

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}}2(a),(b),(c)

Figure 5-170 Core inlet temperature (slow uncontrolled rod withdrawal)

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}}2(a),(b),(c)

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Figure 5-171 Core outlet temperature (slow uncontrolled rod withdrawal)

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}}2(a),(b),(c)

5.3.4.5 Power Reduction

In this case, the benchmark calculations start at full power conditions, followed by a reduction to 50 percent power {{

}}2(a),(c) It is noted that the dropped rod transient benchmark calculation (Section 5.3.4.6) is then performed as a continuation from the 50 percent power steady state condition.

{{

}}2(a),(b)

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Figure 5-172 through Figure 5-178 show reasonable to excellent agreement between the two codes. The core power and reactivity curves are almost identical (Figure 5-172 and Figure 5-173).

At the time of the power reduction, the RETRAN-3D pressurizer level drops below the pressurizer level of NRELAP5. {{

}}2(a),(c),(b) During the power reduction, the RETRAN-3D predicted pressurizer pressure is higher than NRELAP5. The biggest difference of approximately 3.5 percent is attributed to different heat removal rates from the pressurizer between the NRELAP5 and RETRAN-3D pressurizer models, due to differences in the pressurizer modeling as discussed in Section 5.3.4.2.

As explained in Section 5.3.4.3, at the end of the full power steady state initialization, the core flow, core inlet and core outlet temperatures are slightly different between the two models. These differences are carried over to the transient calculation, which can be observed at time zero on Figure 5-176 through Figure 5-178. During the transient power reduction, the core flow, core inlet and core outlet temperatures have a consistent small difference between the RETRAN-3D and NRELAP5 predictions.

Figure 5-172 Core power (power reduction)

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}}2(a),(b),(c)

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Figure 5-173 Total reactivity (power reduction)

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}}2(a),(b),(c)

Figure 5-174 Pressurizer pressure (power reduction)

{{

}}2(a),(b),(c)

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Figure 5-175 Pressurizer level (power reduction)

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}}2(a),(b),(c)

Figure 5-176 Core flow (power reduction)

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}}2(a),(b),(c)

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Figure 5-177 Core inlet temperature (power reduction)

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}}2(a),(b),(c)

Figure 5-178 Core outlet temperature (power reduction)

{{

}}2(a),(b),(c)

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5.3.4.6 Dropped Control Rod

The dropped control rod case is initiated from the 50 percent power steady state condition at time zero. {{

}}2(a),(c)

Figure 5-179 through Figure 5-185 show reasonable to excellent agreement between the two codes. The core power drops from 80 MWth to {{ }}2(a),(b),(c) due to reactivity feedback effects (Figure 5-179 and Figure 5-180). The core power and reactivity curves are very close.

The pressurizer pressure and pressurizer level are close when they decrease following the dropped rod negative reactivity insertion. When the pressurizer pressure and level start to increase following the power recovery, NRELAP5 predicts slightly higher pressure and level compared to RETRAN-3D. {{

}}2(a),(b),(c)

As explained in Section 5.3.4.5, when the transient reaches steady state at the end of the power reduction, the core flow, core inlet and core outlet temperatures are slightly different between the two models. These differences are carried over to the transient calculation, which can be observed at time zero on Figure 5-183 through Figure 5-185. During the transient, the core flow, core inlet and core outlet temperatures have a consistent small difference between the RETRAN-3D and NRELAP5 predictions.

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Figure 5-179 Core power (dropped control rod)

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}}2(a),(b),(c)

Figure 5-180 Total reactivity (dropped control rod)

{{

}}2(a),(b),(c)

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Figure 5-181 Pressurizer pressure (dropped control rod)

{{

}}2(a),(b),(c)

Figure 5-182 Pressurizer level (dropped control rod)

{{

}}2(a),(b),(c)

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Figure 5-183 Core flow (dropped control rod)

{{

}}2(a),(b),(c)

Figure 5-184 Core inlet temperature (dropped control rod)

{{

}}2(a),(b),(c)

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Figure 5-185 Core outlet temperature (dropped control rod)

{{

}}2(a),(b),(c)

5.3.4.7 Conclusions from Benchmark

Four different transients were performed for code-to-code benchmarking between NRELAP5 and RETRAN-3D: Reactivity insertion reflecting a fast UCRW from full power conditions, reactivity insertion reflecting a slow UCRW from full power conditions, negative reactivity insertion to reduce power from 100 percent to 50 percent power, and negative reactivity insertion simulating a dropped rod from 50 percent power. The results from all four of the transients showed that the comparison between the power and the total reactivity were consistently excellent, in that the calculation results of the two codes lined up nearly identically with one another. This agreement is important because the main purpose of the benchmark is to compare the point kinetics model response between NRELAP5 and RETRAN-3D.

There was also an overall pattern in the core inlet and outlet temperatures of a small temperature difference from the steady state conditions that continued throughout the calculation; however, the overall phenomena matched well.

The pressurizer pressure and the pressurizer level were the two characteristics of the system that did not provide as close of a comparison as the other four characteristics, but still reasonably close. The discrepancy is attributed to differences between the NRELAP5 and RETRAN-3D pressurizer models, as discussed in Section 5.3.4.2.

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Overall, NRELAP5 and RETRAN-3D have reasonable to excellent agreement on the parameters of interest in the reactivity feedback and primary system response in the benchmark calculations. The power and reactivity responses predicted by the NRELAP5 point kinetics model during the transients show excellent agreement with the RETRAN-3D prediction.

5.3.5 Steam Generator Modeling

5.3.5.1 Background

NuScale's LOCA Topical Report (Reference 2) Section 7.4 discusses the validation of NRELAP5 for helical coil SG (HCSG) modeling. The validation was mainly against SIET TF-1 and TF-2 test data. It was concluded that NRELAP5 showed reasonable to excellent agreement with test data.

The validation is further investigated in this report to ensure the unique characteristics of the non-LOCA transients (comparing to LOCA) are identified and evaluated. Specifically, this investigation ensures the operating ranges expected during the non-LOCA transients are adequately covered by the validated ranges. The evaluation was originally performed for the NPM design at the time. Subsequent design work resulted in other NPM designs within the confines of the general NPM description in Section 3.0. Additional bottom-up evaluation confirmed that the Reference 29 NRELAP5 key physical models for SG and DHRS heat removal during non-LOCA events remain applicable.

5.3.5.2 Helical Coil Steam Generator Modeling

In the LOCA topical report (Reference 2), NRELAP5 shows reasonable to excellent agreement with test data on the HCSG primary and secondary side, based on comparisons against the SIET TF-2 and SIET TF-1 test data.

5.3.5.3 Helical Coil Steam Generator Operating Ranges vs. Validated Ranges

In the LOCA topical report (Reference 2), NRELAP5 shows reasonable to excellent agreement with test data on the helical coil SG secondary side.
{{

}}2(a),(c) Further evaluation is provided herein to ensure NRELAP5 is validated for DHRS operation.

Based on the typical helical coil SG secondary pressure, temperature and flow rate during DHRS operation, {{

}}2(a),(b),(c)

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{{

}}2(a),(b),(c)

Table 5-15 summarizes the helical coil SG operating range for non-LOCA transients vs. the validated range in NRELAP5. The majority of the helical coil SG secondary side operating range is covered by the validated range of NRELAP5.
{{

}}2(a),(c) Therefore, the operating range of the helical coil SG primary side is sufficiently covered by the validated range of NRELAP5.

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Table 5-15 Non-LOCA transients helical coil steam generator operating range vs. NRELAP5 validated range

{{

}}2(a),(b),(c)

5.3.5.4 Helical Coil Steam Generator Nodalization Sensitivity

{{

}}2(a),(c)

{{

}}2(a),(c)

Based on these studies, modeling the helical coil SG with {{

}}2(a),(c) nodes is expected to produce reasonably accurate results for the non-LOCA transients. Considering these studies, steam generator modeling requirements are summarized in Section 6.1.1 and Section 6.1.4.2 for the primary and the secondary, respectively.

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Figure 5-186 Coil 1 representative pressure drop for {{ }}$^{2(a),(c)}$ nodes (left) and {{ }}$^{2(a),(c)}$ nodes (right)

{{

}}$^{2(a),(b),(c),ECI}$

Figure 5-187 Coil 1 representative fluid temperature for {{ }}$^{2(a),(c)}$ nodes (left) and {{ }}$^{2(a),(c)}$ nodes (right)

{{

}}$^{2(a),(b),(c),ECI}$

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Figure 5-188 Coil 1 representative wall temperature for {{ }}$^{2(a),(c)}$ nodes (left) and {{ }}$^{2(a),(c)}$ nodes (right)

{{

}}2(a),(b),(c),ECI

Figure 5-189 Decrease in feedwater temperature nodalization sensitivity steam generator secondary side inlet pressure

{{

}}2(a),(b),(c)

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Figure 5-190 Decrease in feedwater temperature nodalization sensitivity steam generator secondary side outlet pressure

{{

}}2(a),(b),(c)

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Figure 5-191 Decrease in feedwater temperature nodalization sensitivity reactor coolant system flow rate

{{

}}2(a),(b),(c)

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Figure 5-192 Decrease in feedwater temperature nodalization sensitivity reactor coolant system lower plenum pressure

{{

}}2(a),(b),(c)

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Figure 5-193 Decrease in feedwater temperature nodalization sensitivity reactor coolant system core inlet temperature

{{

}}2(a),(b),(c)

Figure 5-194 Decrease in feedwater temperature nodalization sensitivity reactor power

{{

}}2(a),(b),(c)

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5.3.6 Heat Transfer Correlation Comparison

5.3.6.1 Background

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}}2(a),(c)

5.3.6.2 Computational Fluid Dynamics Calculations

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}}2(a),(c),ECI

5.3.6.2.1 Model 1 Analysis and Results

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}}2(a),(c),ECI

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{{

}}2(a),(c),ECI

{{

}}2(a),(c),ECI

Table 5-16 Comparison of correlation to CFD results - CFD model 1

{{

}}2(a),(c),ECI

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5.3.6.2.2 Model 2 Analysis and Results

{{

}}2(a),(c),ECI

Table 5-17 Comparison of correlation to CFD results - CFD model 2

{{

}}2(a),(c),ECI

5.3.6.2.3 Model 3 Analysis and Results

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}}2(a),(c),ECI

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{{

}}2(a),(c),ECI

Table 5-18 Not Used

5.3.6.2.4 Computational Fluid Dynamics Conclusions

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}}2(a),(c),ECI

5.3.6.3 Comparison of Heat Transfer Correlations within NRELAP5

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure 5-195 Comparison of heat transfer correlation options for steam generator primary side heat transfer rate (linear time scale)

{{

}}2(a) (c)

Figure 5-196 Comparison of heat transfer correlation options for steam generator primary side heat transfer rate (logarithmic time scale)

{{

}}2(a),(c)

5.3.6.4 Heat Transfer Correlation Conclusion

{{

}}2(a),(c)

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

5.3.7.1 NIST-2 Facility

Section 5.3.2.1 describes the NIST-1 facility. In 2018 work began on the NIST-1 facility to modify the secondary flow piping, the DHRS, and minor components of other systems. Specific modifications include:

- Increased main steam system pressure capacity
- Increased steam generator and DHRS pressure capacity
- Replacement of most secondary side piping and valves
- Installation of a steam-flow control valve with improved controllability
- Installed new instruments or increased existing instrument ranges

The DHRS is a system of piping that interfaces with the secondary flow piping, allowing closed loop DHRS testing or once through DHRS testing. The DHRS is divided into five pipe sections; the DHRS bypass steam line, the DHRS steam line, the DHRS scaled height two tube heat exchanger, the DHRS condensate line, and the DHRS vent line. The DHRS piping configuration is shown in Figure 5-197 and a description of each pipe section follows.

- {{

}}2(a),(c),ECI

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After this upgrade, the facility name was changed to NIST-2. Except for the changes described above, the NIST-2 facility is consistent with the descriptions and figures shown in Section 5.3.2.1.

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Figure 5-197 NIST-2 decay heat removal system configuration

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5.3.7.2 NIST-2 Steam Generator - Decay Heat Removal System Integral Effects Test Objectives and Scaling

5.3.7.2.1 Test Objectives

Initial assessment and validation of NRELAP5 for prediction of SG-DHRS phenomena during non-LOCA event sequences was provided in Section 5.3.1, Section 5.3.2, Section 5.3.3, and Section 5.3.5. NuScale identified that additional testing and analysis was needed to reduce the uncertainty associated with heat removal by the DHRS during non-LOCA event sequences.

Section 3.0 provides general discussion of NPM designs, including the DHRS. Changes between NPM designs have resulted in changes to the DHRS geometry, such as the DHRS steam piping configuration, and detailed condenser design. The PIRT in Section 5.1 identifies several high ranked

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phenomena associated with DHRS operation in Phase 3. NuScale identified that additional testing and analysis was needed to understand the impact of potential DHRS design changes on the relevant high ranked phenomena.

The NIST-2 SG-DHRS integral effects testing is designed to enhance the DHRS validation database to address the above considerations. The NIST-2 facility permits test conditions that are significantly closer to the prototypical NPM operation conditions than was achieved in prior testing.

5.3.7.2.2 Scaling Analysis

As part of the assessment of EM adequacy, a top-down scaling analysis was performed. The scaling analysis compared dimensionless π groups for the governing field equations of liquid mass, total energy, and total momentum balances in the DHRS loop. Comparisons were made between {{

}}2(a),(c),ECI

The SG/DHRS loop is a boiling/condensation-driven natural circulation system. {{

}}2(a),(c),ECI

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The instantaneous π group balance equations for the total liquid mass in the SG and DHRS loop are given by Equations 5-1 and 5-2, respectively. {{

}}2(a),(c),ECI

The instantaneous π group balance equation for the total energy balance is given in Equation 5-3. {{

}}2(a),(c),ECI

The instantaneous π group balance equation for the total two-phase mixture fluid momentum conservation in the SG/DHRS loop is given in Equation 5-4. {{

}}2(a),(c),ECI

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For each π group, distortion factors were defined as shown in Equation 5:

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}}2(a),(c),ECI

Time-dependent and normalized time-averaged π groups were compared. Normalized, time-averaged results of the scaling and distortion analyses are shown in the following plots. The normalized time-averaged π groups and distortion factors are shown for the NPM-20 design, the ISD-20 model, and NIST-2. Table 5-19 provides the π group definitions and figure legend information.

The normalized time-averaged π groups for the liquid mass balance in the SG and DHRS heat exchanger are compared in Figure 5-198 while the significances and distortions of the π groups are compared in Figure 5-199. {{

}}2(a),(c),ECI

Normalized time-averaged π groups for the energy balance are compared in Figure 5-200. Significances and distortions of the π groups are compared in Figure 5-201. {{

}}2(a),(c),ECI

Normalized time-averaged π groups for the momentum balance are compared in Figure 5-202. Significances and distortions of the π groups are compared in

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Figure 5-203. {{

}}2(a),(c),ECI

The top-down scaling and distortion analysis demonstrates similarity in dimensionless π groups, and supports the conclusion that the NRELAP5 mathematical models (field equations) are applicable for NPM integral effects non-LOCA analysis with SG/DHRS heat removal.

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Table 5-19 π groups in dimensionless control volume balance equations for decay heat removal system

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}}2(a),(c),ECI

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Table 5-19 π groups in dimensionless control volume balance equations for decay heat removal system (Continued)

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}}2(a),(c),ECI

Figure 5-198 Normalized time-averaged π groups for decay heat removal system liquid mass balance in {{ }}2(a),(c),ECI

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}}2(a),(c),ECI

Figure 5-199 Significances vs. distortions of π groups for decay heat removal system liquid mass balance in {{ }}2(a),(c),ECI

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}}2(a),(c),ECI

Figure 5-200 Normalized time-averaged π groups for decay heat removal system energy balance in {{ }}2(a),(c),ECI

{{

}}2(a),(c),ECI

Figure 5-201 Significances vs. distortions of π groups for decay heat removal system energy balance in {{ }}2(a),(c),ECI

{{

}}2(a),(c),ECI

Figure 5-202 Normalized time-averaged π groups for decay heat removal system momentum balance in {{ }}2(a),(c),ECI

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}}2(a),(c),ECI

Figure 5-203 Significances vs. distortions of π groups for decay heat removal system momentum balance in {{ }}2(a),(c),ECI

{{

}}2(a),(c),ECI

5.3.7.3 Applicability of Decay Heat Removal System Operation to Small Break Loss-of-Coolant Accident Sequences

The assessment and validation of NRELAP5 for prediction of SG-DHRS phenomena provided in Section 5.3.1, Section 5.3.2, and Section 5.3.3 was focused on application during non-LOCA event sequences. DHRS actuation is also expected to occur during LOCA event sequences. For smaller LOCA break sizes, there may be an extended period where DHRS is in operation but ECCS is not yet actuated. In a small break LOCA sequence where there is such an extended period, the phenomena associated with DHRS performance are applicable.

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}}2(a),(c)

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}}2(a),(c)

Based on these assessments, the additional NIST-2 testing and analysis is applicable to justify the heat removal by the DHRS during the small break LOCA conditions prior to ECCS valve opening.

5.3.7.3.1 Effect of Break Size and Location on Decay Heat Removal System Performance

As will be discussed further in Section 5.3.7.5, the NIST-2 testing is representative of an LOFW transient. Therefore, an analysis of the SG-DHRS performance in the LOFW transient and small break LOCA was performed to confirm that there were no unique high-ranked phenomena for DHRS performance in a small break LOCA as compared to an LOFW transient and that the SG-DHRS performance is similar to the performance in the small

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break LOCA scenario. A summary of the analysis and results is presented below.

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}}2(a),(c)

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}}[2(a),(c)] Therefore, NRELAP5 validation and applicability to simulate DHRS heat removal during non-LOCA events extends to simulate DHRS heat removal during LOCA events, particularly small LOCA events with an extended timeframe of liquid flow over the riser and DHRS heat removal prior to ECCS actuation.

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Figure 5-204 Reactor coolant system average temperature transient response for loss of feedwater break scenarios

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}}2(a),(c)

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Figure 5-205 Reactor coolant system saturation temperature transient response for loss of feedwater break scenarios

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}}2(a),(c)

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Figure 5-206 Reactor coolant system flow transient response for loss of feedwater break scenarios

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}}2(a),(c)

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Figure 5-207 Reactor coolant system pressure transient response for loss of feedwater break scenarios

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}}2(a),(c)

© Copyright 2025 by NuScale Power, LLC

Figure 5-208 Secondary side pressure transient response for loss of feedwater break scenarios

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}}2(a),(c)

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Figure 5-209 Decay heat removal system flow transient response for loss of feedwater break scenarios

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}}2(a),(c)

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Figure 5-210 Decay heat removal system power transient response for loss of feedwater break scenarios

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}}2(a),(c)

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Figure 5-211 Decay heat removal system power compared to reactor coolant system temperature for loss of feedwater break scenarios

{{

}}2(a),(c)

5.3.7.4 NIST-2 NRELAP5 Model Description

The NRELAP5 model of the NIST-1 facility is described in Section 5.3.3.3. The differences between the NRELAP5 models for NIST-2 and NIST-1 model are the result of incorporating the facility changes described in Section 5.3.7.1. The main changes are in the secondary side and the DHRS. Figure 5-212 through Figure 5-219 show the NRELAP5 component types and numbers for the secondary side and DHRS for NIST-2.

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Figure 5-212 NIST-2 modified separate effects main steam line part 1 model

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}}2(a),(c),ECI

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Figure 5-213 NIST-2 modified separate effects main steam line part 2 model

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}}2(a),(c),ECI

Figure 5-214 NIST-2 modified bypass steam line to containment vessel model

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}}2(a),(c),ECI

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Figure 5-215 NIST-2 new integral effects decay heat removal system steam line model

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}}2(a),(c),ECI

Figure 5-216 NIST-2 modified separate effects decay heat removal system steam line model

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}}2(a),(c),ECI

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Figure 5-217 NIST-2 new decay heat removal system heat exchanger model

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}}2(a),(c),ECI

Figure 5-218 NIST-2 modified decay heat removal system condensate return line model

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}}2(a),(c),ECI

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Figure 5-219 NIST-2 modified feedwater system model

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}}2(a),(c),ECI

5.3.7.5 NIST-2 Steam Generator - Decay Heat Removal System Integral Effects Test Matrix

Based on the acceptance criteria described in Section 4.2, figures of merit in the non-LOCA events include CHFR, RCS pressure, SG pressure, and CNV pressure. These figures of merit are affected by the DHRS heat removal performance during Phase 3. In addition, Section 5.3.7.3 concludes that DHRS heat removal performance is also applicable during small break LOCA events prior to ECCS performance (corresponding to Phase 3). The high-ranked phenomena identified in the NPM non-LOCA PIRT focusing on the DHRS and its interfacing systems (i.e., RCS and reactor pool) during Phase 3 can be summarized as follows.

- {{

}}2(a),(c)

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- {{

}}2(a),(c) Therefore, the SG and DHRS heat exchanger are the primary focus for the NIST-2 DHRS testing. Table 5-20 identifies SG/DHRS testing performed at the NIST-2 facility and assessed with NRELAP5 as part of the non-LOCA EM development.

Table 5-20 NIST-2 integral effects SG/DHRS testing for NRELAP5 code validation

Test Identifier	DHRS Steam Line Orifice Resistance	Initial DHRS Inventory	Test Description
Run 1	Base	Base	This is the base configuration for the testing.
Run 2	Larger	Base	Smaller orifice size to provide larger steam line resistance.
Run 3	Smaller	Base	Larger orifice size to provide smaller steam line resistance.

5.3.7.6 Run 1 Test Description

The NIST-2 facility had the following initial conditions at the start of the Run 1 test:

{{

}}2(a),(b),(c),ECI

{{

}}2(a),(b),(c),ECI

The NIST-2 valve configuration at the start of the Run 1 test is shown in Figure 5-197. {{

}}2(a),(b),(c),ECI

The sequence of events for the Run 1 test is shown in Table 5-21.

Table 5-21 NIST-2 Run 1 sequence of events

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}}(a),(b),(c),ECI

5.3.7.7 Run 1 Test Results

Figure 5-220 through Figure 5-240 show the results of the code-to-data comparisons for Run 1.

Figure 5-220 shows the core power. Figure 5-221 shows the SG/DHRS active loop inventory. {{

}}2(a),(c)

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{{

}}2(a),(c)

Figure 5-222 and Figure 5-223 show the DHRS heat exchanger and SG power comparison for the long-term and short-term respectively. Figure 5-224 and Figure 5-225 show the SG-DHRS loop flow comparison for the long-term and short-term respectively. {{

}}2(a),(c) Figure 5-224a and Figure 5-225a show comparison of NRELAP5 results with the unaveraged flow meter data and NRELAP5 predicts the test data with reasonable to excellent agreement. {{

}}2(a),(c) as shown in Figure 5-222 and Figure 5-223, the NRELAP5 calculation has reasonable to excellent agreement with the test data.

Figure 5-226 shows the code-to-data comparison for the PZR pressure and the secondary side pressure near the steam drum. Before the PZR heater is tripped, both data and code maintain the PZR pressure at approximately {{

}}2(a),(c)

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{{ }}[2(a),(c)] Overall the secondary side pressure has excellent agreement with the test data.

Figure 5-227 shows the code-to-data comparison for the DHRS heat exchanger and SG levels. {{

}}[2(a),(c)] Nevertheless, the overall predicted DHRS heat exchanger and SG levels have reasonable to excellent agreement with the test data.

Figure 5-228 shows the code-to-data comparison for the steam drum collapsed liquid level. {{

}}[2(a),(c)]

Figure 5-229 shows the code-to-data comparison for the DHRS heat exchanger inlet and outlet temperatures. Figure 5-230 shows the code-to-data comparison for the SG secondary side inlet and outlet temperatures. The SG secondary side outlet temperature and DHRS heat exchanger inlet temperatures are both close to the loop saturation temperature (Figure 5-231). The code predictions of these two temperatures both have reasonable to excellent agreement with the data, similar to the secondary side pressure.

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From Figure 5-229 and Figure 5-230, the predicted DHRS heat exchanger outlet temperature and SG secondary side inlet temperature are {{

}}2(a),(c)

Figure 5-232 shows the code-to-data comparison of the RCS primary temperatures at two locations: at the SG primary side's inlet (upper plenum) and outlet (mid downcomer). {{

}}2(a),(c)

Figure 5-233 shows the code-to-data comparison of the PZR level. Figure 5-234 shows the code-to-data comparison of the RPV level. {{

}}2(a),(c)

Figure 5-235 shows the code-to-data comparison of the RCS primary flow. The prediction has excellent agreement with the data during the transient. {{

}}2(a),(c)

Figure 5-236 shows the code-to-data comparison of the CPV level. The prediction has excellent agreement with data {{

}}2(a),(c)

Figure 5-239 shows the code-to-data comparison of the DHRS steam line differential pressure. Excellent agreement is observed. Figure 5-240 shows the comparison of the DHRS steam line orifice differential pressure. Reasonable agreement is observed. {{ }}2(a),(c)

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{{

}}2(a),(c) Code-to-data comparisons of other differential pressures show that most along the SG-DHRS loop have excellent agreement.

{{

}}2(a),(b),(c)

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Figure 5-220 NIST-2 Run 1 core power comparison

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}}2(a),(b),(c),ECI

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Figure 5-221 NIST-2 Run 1 steam generator and decay heat removal system active loop inventory comparison

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}}2(a),(b),(c),ECI

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Figure 5-222 NIST-2 Run 1 steam generator and decay heat removal system heat exchanger power comparison

{{

}}2(a),(b),(c),ECI

Figure 5-223 NIST-2 Run 1 steam generator and decay heat removal system heat exchanger power comparison - short-term

{{

}}2(a),(b),(c),ECI

Figure 5-224 NIST-2 Run 1 steam generator and decay heat removal system loop flow comparison

{{

}}2(a),(b),(c),ECI

Figure 5-224a NIST-2 Run 1 decay heat removal condensate line flow comparison (without time average)

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}}2(a),(b),(c),ECI

Figure 5-225 NIST-2 Run 1 steam generator and decay heat removal system loop flow comparison - short-term

{{

}}2(a),(b),(c),ECI

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Figure 5-225a NIST-2 Run 1 decay heat removal condensate line flow comparison (without time average) - short-term

{{

}}2(a),(b),(c),ECI

Figure 5-226 NIST-2 Run 1 primary and secondary pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-227 NIST-2 Run 1 steam generator and decay heat removal system level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-228 NIST-2 Run 1 steam drum level comparison

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}}2(a),(b),(c),ECI

Figure 5-229 NIST-2 Run 1 decay heat removal system heat exchanger inlet and outlet temperatures comparison

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}}2(a),(b),(c),ECI

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Figure 5-230 NIST-2 Run 1 steam generator secondary side inlet and outlet temperatures comparison

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}}2(a),(b),(c),ECI

Figure 5-231 NIST-2 Run 1 steam generator secondary side saturation temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-232 NIST-2 Run 1 steam generator primary side inlet and outlet temperatures comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-233 NIST-2 Run 1 pressurizer level comparison

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}}2(a),(b),(c),ECI

Figure 5-234 NIST-2 Run 1 reactor pressure vessel level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-235 NIST-2 Run 1 primary flow rate comparison

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}}2(a),(b),(c),ECI

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Figure 5-236 NIST-2 Run 1 cooling pool vessel level comparison

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}}2(a),(b),(c),ECI

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Figure 5-237 NIST-2 Run 1 cooling pool vessel level 5 temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-238 NIST-2 Run 1 cooling pool vessel level 6 temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-239 NIST-2 Run 1 decay heat removal system steam line differential pressure comparison

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}}2(a),(b),(c),ECI

Figure 5-240 NIST-2 Run 1 decay heat removal system steam line orifice differential pressure comparison

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}}2(a),(b),(c),ECI

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Figure 5-241 NIST-2 Run 1 decay heat removal system heat exchanger tube lower middle section fluid temperatures - data only

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}}2(a),(b),(c),ECI

Figure 5-242 NIST-2 Run 1 decay heat removal system heat exchanger tube bottom fluid temperatures - data only

{{

}}2(a),(b),(c),ECI

5.3.7.8 Run 2 Test Description

The NIST-2 facility had the following initial conditions at the start of the Run 2 test:

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}}2(a),(b),(c),ECI

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{{

}}2(a),(b),(c),ECI

The valve configuration and execution of the Run 2 test was the same as described for Run 1 in Section 5.3.7.6. The sequence of events of the Run 2 test is shown in Table 5-22.

Table 5-22 NIST-2 Run 2 sequence of events

{{

}}(a),(b),(c),ECI

5.3.7.9 Run 2 Test Results

Figure 5-243 through Figure 5-264 show the code-to-data comparison of the key parameters for Run 2.

Figure 5-243 shows the core power. Figure 5-244 shows the SG/DHRS active loop inventory. {{

}}2(a),(c) The active loop inventory is discussed in greater detail for Run 1 in Section 5.3.7.7.

Figure 5-245 and Figure 5-246 show the DHRS heat exchanger and SG power comparison for the long-term and short-term, respectively. Figure 5-247 and Figure 5-248 show the SG/DHRS loop flow comparison for the long-term and short-term, respectively. As described in Section 5.3.7.7 for Run 1, {{

}}2(a),(c) The results are consistent with those observed in Run 1 in Section 5.3.7.7.

{{

}}2(a),(c)

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{{ }}2(a),(c)

Figure 5-249 compares the DHRS loop flow data without time average with the NRELAP5 results. Both are DHRS condensate line flow rates. {{

}}2(a),(c)

Figure 5-250 shows the differential pressure across the DHRS condensate line flow meter, {{

}}2(a),(c)

Figure 5-251 shows the code-to-data comparison for the pressurizer pressure and secondary side pressure near the steam drum. {{

}}2(a),(c)

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{{

}}2(a),(c) Overall the secondary side pressure has reasonable to excellent agreement with the test data.

Figure 5-252 shows the code-to-data comparison for the DHRS heat exchanger level and SG level. {{

}}2(a),(c) Overall, the predicted DHRS heat exchanger and SG levels have reasonable to excellent agreement with the test data.

Figure 5-253 shows the code-to-data comparison for the DHRS heat exchanger inlet and outlet temperatures. Figure 5-254 shows the code-to-data comparison for the SG secondary side inlet and outlet temperatures. {{

}}2(a),(c) The code predictions of these two temperatures both have reasonable to excellent agreement with the data, similar to the secondary side pressure.

From Figure 5-253 and Figure 5-254, {{

}}2(a),(c)

Figure 5-256 shows the code-to-data comparison of the RCS primary temperatures. {{

}}2(a),(c)

Figure 5-257 shows the code-to-data comparison of the PZR level. Figure 5-258 shows the code-to-data comparison of the RPV level. {{

}}2(a),(c)

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Figure 5-259 shows the code-to-data comparison of the RCS primary flow. The prediction has excellent agreement with the data during the transient. {{

}}2(a),(c)

Figure 5-260 shows the code-to-data comparison of the CPV level. The prediction has excellent agreement with data {{

}}2(a),(c)

Figure 5-263 shows the code-to-data comparison of the DHRS steam line differential pressure. Figure 5-264 shows the comparison of the DHRS steam line orifice differential pressure. Reasonable to excellent agreements are shown {{

}}2(a),(c)

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Figure 5-243 NIST-2 Run 2 core power comparison

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}}2(a),(b),(c),ECI

Figure 5-244 NIST-2 Run 2 steam generator and decay heat removal system active loop inventory comparison

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}}2(a),(b),(c),ECI

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Figure 5-245 NIST-2 Run 2 steam generator and decay heat removal system heat exchanger power comparison

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}}2(a),(b),(c),ECI

Figure 5-246 NIST-2 Run 2 steam generator and decay heat removal system heat exchanger power comparison - short-term

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}}2(a),(b),(c),ECI

Figure 5-247 NIST-2 Run 2 steam generator and decay heat removal system loop flow comparison

{{

}}2(a),(b),(c),ECI

Figure 5-248 NIST-2 Run 2 steam generator and decay heat removal system loop flow comparison - short-term

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}}2(a),(b),(c),ECI

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Figure 5-249 NIST-2 Run 2 decay heat removal system condensate line flow comparison (without time average)

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}}2(a),(b),(c),ECI

Figure 5-250 NIST-2 Run 2 decay heat removal system condensate line differential pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-251 NIST-2 Run 2 primary and secondary pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-252 NIST-2 Run 2 steam generator and decay heat removal system level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-253 NIST-2 Run 2 decay heat removal system heat exchanger inlet and outlet temperatures comparison

{{

}}2(a),(b),(c),ECI

Figure 5-254 NIST-2 Run 2 steam generator secondary side inlet and outlet temperatures comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-255 NIST-2 Run 2 steam generator secondary side saturation temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-256 NIST-2 Run 2 steam generator primary side inlet and outlet temperatures comparison

{{

}}2(a),(b),(c),ECI

Figure 5-257 NIST-2 Run 2 pressurizer level comparison

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}}2(a),(b),(c),ECI

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Figure 5-258 NIST-2 Run 2 reactor pressure vessel level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-259 NIST-2 Run 2 primary flow rate comparison

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}}2(a),(b),(c),ECI

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Figure 5-260 NIST-2 Run 2 cooling pool vessel level comparison

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}}2(a),(b),(c),ECI

Figure 5-261 NIST-2 Run 2 cooling pool vessel level 5 temperature comparison

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}}2(a),(b),(c),ECI

Figure 5-262 NIST-2 Run 2 cooling pool vessel level 6 temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-263 NIST-2 Run 2 decay heat removal system steam line differential pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-264 NIST-2 Run 2 decay heat removal system steam line orifice differential pressure comparison

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}}2(a),(b),(c),ECI

5.3.7.10 Run 3 Test Description

The NIST-2 facility had the following initial conditions at the start of the Run 3 test:

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}}2(a),(b),(c),ECI

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}}2(a),(b),(c),ECI

The valve configuration and execution of the Run 3 test was the same as described for Run 1 in Section 5.3.7.6. The sequence of events of the Run 3 test is shown in Table 5-23.

Table 5-23 NIST-2 Run 3 sequence of events

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}}(a),(b),(c),ECI

5.3.7.11 Run 3 Test Results

Figure 5-265 through Figure 5-285 show the code-to-data comparison of the key parameters for Run 3.

Figure 5-265 shows the core power. Figure 5-266 shows the SG/DHRS active loop inventory. {{

}}2(a),(c) The active loop inventory is discussed in greater detail for Run 1 in Section 5.3.7.7.

Figure 5-267 and Figure 5-268 show the DHRS heat exchanger and SG power comparison for the long-term and short-term, respectively. Figure 5-269 and Figure 5-270 show the SG/DHRS loop flow comparison for the long-term and short-term, respectively. As described in Section 5.3.7.7 for Run 1, {{

}}2(a),(c) The results are consistent with those observed in Run 1 in Section 5.3.7.7.

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{{

}}2(a),(c) Figure 5-271 compares the DHRS condensate line flow data without time average with the NRELAP5 results. {{

}}2(a),(c)

Figure 5-272 shows the code-to-data comparison for the pressurizer pressure and secondary side pressure near the steam drum. {{

}}2(a),(c) The predicted secondary side pressure shows excellent agreement with the test data for the entire transient, including the peak pressure.

Figure 5-273 shows the code-to-data comparison for the DHRS heat exchanger level and SG level. {{

}}2(a),(c) Overall the predicted DHRS heat exchanger and SG levels have reasonable to excellent agreement with the test data.

Figure 5-274 shows the code-to-data comparison for the DHRS heat exchanger inlet and outlet temperatures. Figure 5-275 shows the code-to-data comparison for the SG secondary side inlet and outlet temperatures. {{

}}2(a),(c) The code predictions of these two temperatures both have reasonable to excellent agreement with the data, similar to the secondary side pressure.

From Figure 5-274 and Figure 5-275, {{

}}2(a),(c)

Figure 5-277 shows the code-to-data comparison of the RCS primary temperatures. {{

}}2(a),(c)

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{{

}}2(a),(c)

Figure 5-278 shows the code-to-data comparison of the PZR level. Figure 5-279 shows the code-to-data comparison of the RPV level. {{

}}2(a),(c) The predictions have reasonable agreement with test data.

Figure 5-280 shows the code-to-data comparison of the RCS primary flow. The prediction has excellent agreement with the data during the transient. {{

}}2(a),(c)

Figure 5-281 shows the code-to-data comparison of the CPV level. The prediction has excellent agreement with data {{

}}2(a),(c)

Figure 5-284 shows the code-to-data comparison of the DHRS steam line differential pressure. Figure 5-285 shows the comparison of the DHRS steam line orifice differential pressure. Predicted differential pressures have reasonable to excellent agreement with test data.

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Figure 5-265 NIST-2 Run 3 core power comparison

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}}2(a),(b),(c),ECI

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Figure 5-266 NIST-2 Run 3 steam generator and decay heat removal system active loop inventory comparison

{{

}}2(a),(b),(c),ECI

Figure 5-267 NIST-2 Run 3 steam generator and decay heat removal system heat exchanger power comparison

{{

}}2(a),(b),(c),ECI

Figure 5-268 NIST-2 Run 3 steam generator and decay heat removal system heat exchanger power comparison - short-term

{{

}}2(a),(b),(c),ECI

Figure 5-269 NIST-2 Run 3 steam generator and decay heat removal system loop flow comparison

{{

}}2(a),(b),(c),ECI

Figure 5-270 NIST-2 Run 3 steam generator and decay heat removal system loop flow comparison - short-term

{{

}}2(a),(b),(c),ECI

Figure 5-271 NIST-2 Run 3 decay heat removal system condensate line flow comparison (without time average)

{{

}}2(a),(b),(c),ECI

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Figure 5-272 NIST-2 Run 3 primary and secondary pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-273 NIST-2 Run 3 steam generator and decay heat removal system level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-274 NIST-2 Run 3 decay heat removal system heat exchanger inlet and outlet temperatures comparison

{{

}}2(a),(b),(c),ECI

Figure 5-275 NIST-2 Run 3 steam generator secondary side inlet and outlet temperatures comparison

{{

}}2(a),(b),(c),ECI

Figure 5-276 NIST-2 Run 3 steam generator secondary side saturation temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-277 NIST-2 Run 3 steam generator primary side inlet and outlet temperatures comparison

{{

}}2(a),(b),(c),ECI

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Figure 5-278 NIST-2 Run 3 pressurizer level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-279 NIST-2 Run 3 reactor pressure vessel level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-280 NIST-2 Run 3 primary flow rate comparison

{{

}}2(a),(b),(c),ECI

Figure 5-281 NIST-2 Run 3 cooling pool vessel level comparison

{{

}}2(a),(b),(c),ECI

Figure 5-282 NIST-2 Run 3 cooling pool vessel level 5 temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-283 NIST-2 Run 3 cooling pool vessel level 6 temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 5-284 NIST-2 Run 3 decay heat removal system steam line differential pressure comparison

{{

}}2(a),(b),(c),ECI

Figure 5-285 NIST-2 Run 3 decay heat removal system steam line orifice differential pressure comparison

{{

}}2(a),(b),(c),ECI

5.3.7.12 NIST-2 Decay Heat Removal System Integral Effects Testing Summary

Based on the overall comparison between NRELAP5 and the NIST-2 test data for the simulated LOFW transients, there is reasonable to excellent agreement between code prediction and data. Predicted SG-DHRS loop flow and DHRS heat exchanger and SG power match the test data very closely during quasi steady state. Predicted secondary side pressure and temperatures, as well as DHRS heat exchanger and SG levels, have reasonable to excellent agreement with test data. SG-DHRS loop differential pressures are also predicted with reasonable to excellent agreement to test data. The comparisons demonstrate that NRELAP5 has the capability to accurately predict SG-DHRS performance during transients.

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5.3.7.13 Impact of NPM Riser Holes on NIST Test Assessments

The NPM design may include holes in the riser that allow cross-flow between the riser and downcomer. {{

}}2(a),(c) The overall performance of the NRELAP5 code and model, in terms of its level of code-to-data agreement, did not change from the previous assessment.

The NIST-2 assessments described in Section 5.3.7 {{

}}2(a),(c)

The above results provide confidence in the ability of NRELAP5 to simulate natural circulation flow and temperature distribution in the presence of cross-flow holes in the riser while also demonstrating that such riser holes do not significantly impact the NIST test assessments.

5.4 Conclusions of NRELAP5 Applicability for Non-LOCA

The high-ranked phenomena identified by the PIRT process for NPM non-LOCA transients were evaluated with respect to the high-ranked phenomena identified by the PIRT process for NPM LOCA scenarios, as well as the NRELAP5 assessments performed as part of the NuScale LOCA evaluation model development. A gap analysis was performed to identify high-ranked phenomena for non-LOCA transients that are not assessed as part of the NuScale LOCA evaluation model development.

High-ranked phenomena for non-LOCA events that are not assessed as part of the NuScale LOCA evaluation model development were addressed in different ways:

1. Additional NRELAP5 code assessment performed against separate effects or integral effects test data
2. Code-to-code benchmark performed between NRELAP5 and independent system thermal-hydraulics code
3. Phenomenon is addressed as part of the downstream subchannel analysis
4. Phenomenon is addressed by specifying appropriately conservative input to the system transient analysis

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In addition to the NRELAP5 assessments performed as part of the LOCA evaluation model development, further assessments were performed to demonstrate NRELAP5 qualification for high rank non-LOCA PIRT phenomena.

Assessment of NRELAP5 against KAIST data demonstrates the code's ability to model heat transfer within tubes and appropriately model the key thermal-hydraulic phenomena associated with condensation within the DHRS heat exchanges tubes.

The comparisons between the NIST-1 HP-03 experimental data and the code calculated values showed that NRELAP5 has the capability to accurately predict the energy transfer across the DHRS heat exchanger tubes to the CPV fluid resulting in reasonable to excellent agreement in capturing cooling pool heat up during these tests in which the cooling pool remains subcooled.

The comparisons between the NIST-1 HP-04 experimental data and the code calculated values showed that NRELAP5 has the capability to accurately predict the energy transfer across the DHRS heat exchanger tubes to the CPV fluid. The HP-04 test duration was longer than that of HP-03. In HP-04, saturated conditions in the pool were reached and thermal stratification developed. Although the CPV fluid heat-up profiles for the HP-04 test data were not fully reproduced in the NRELAP5 simulations of HP-04, the code-to-data comparisons of DHRS heat removal rate were well-matched, demonstrating that NRELAP5 is capable of predicting the total energy removal rate of the DHRS to the cooling pool.

The comparisons between the NIST-1 NLT-2a integral experimental data and the code calculated values showed that NRELAP5 has the capability to accurately predict primary coolant system heat up and pressurization due to a loss of feedwater flow.

NRELAP5 assessment of the NIST-1 NLT-2b phases 1 through 4 demonstrates the ability of the code to predict the heat transfer from primary side to the SG and from the DHRS to CPV. After completion of core power maneuvering and after full start of the DHRS loop, NRELAP5 calculated pressurizer pressure, core inlet and outlet temperature, SG pressure, energy transfer from the primary to secondary, energy transfer from DHRS to CPV, SG level, DHRS level, and CPV level results are all within reasonable to excellent agreement with NLT-2b test data.

The pressurizer level for NLT-2b phase 1 was {{

}}$^{2(a),(b),(c)}$ However, even with these discrepancies, the results show that the important parameter of total energy transfer from the primary side to the SG and from the DHRS to CPV are well predicted.

For NLT-2b phases 1 through 4, the calculated condensate temperature is in minimal agreement with data, with the temperature being over predicted by NRELAP5. However,

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even with this discrepancy, the important parameter of heat transfer from primary side to the SG and from the DHRS to CPV are well predicted.

For NLT-2b phases 1 through 4, although the CPV fluid heat-up profiles were not fully captured in the NRELAP5 simulations, the code-to-data comparisons of DHRS heat removal rate were well-matched, demonstrating that NRELAP5 is capable of capturing the important parameter of total energy removal rate of the DHRS to the cooling pool and that the integral response is insensitive to the CPV temperature profile. {{

}}2(a),(c)

NRELAP5 assessment of the NIST-1 NLT-15p2 test demonstrates the ability of the code to predict the heat transfer from primary side to the SG and from the DHRS to CPV. The NRELAP5 calculated pressurizer and RPV level, core inlet and outlet temperature, primary flow rate, steam generator pressure, energy transfer from the primary to secondary, energy transfer from DHRS to CPV, SG level, DHRS heat exchanger level, and CPV level results are predicted reasonably compared to the NLT-15p2 test data. The NRELAP5 calculation of the initial pressurizer pressure response to decrease in secondary side heat transfer due to loss of feedwater, and the RPV pressure turn over after core power decreased to decay heat levels and DHRS heat removal flow was established, was well-predicted.

{{

}}2(a),(c)

The calculated fluid temperature inside the DHRS heat exchanger tubes predicted the trends of the measured data. {{

}}2(a),(c)

The calculated differential pressure across the DHRS steam line {{

}}2(a),(c)

The overall comparison between NRELAP5 and the NIST-2 test data for the simulated LOFW transient, shows reasonable to excellent agreement between code prediction and data. Predicted SG-DHRS loop flow and DHRS heat exchanger and SG power match the

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test data very closely during quasi steady state. Predicted secondary side pressure and temperatures, as well as DHRS heat exchanger and SG levels, have reasonable to excellent agreement with test data. SG-DHRS loop differential pressures are also predicted with reasonable to excellent agreement to test data. The comparisons demonstrate that NRELAP5 has the capability to accurately predict SG-DHRS performance during transients.

Four different transients were performed for code-to-code benchmarking between NRELAP5 and RETRAN-3D: Reactivity insertion representative of a fast UCRW from full power conditions, reactivity insertion representative of a slow UCRW from full power conditions, negative reactivity insertion to reduce power from 100 percent to 50 percent power, and negative reactivity insertion simulating a dropped rod from 50 percent power. The results from all four of the transients showed that the comparison between the power and the total reactivity were consistently excellent, in that the calculation results of the two codes were nearly identically with one another.

NuScale's LOCA Topical Report (Reference 2) Section 7.4 discusses the validation of NRELAP5 for helical coil SG modeling. The validation was mainly against SIET TF-1 and TF-2 test data. The operating range of the helical coil SG primary and secondary side is demonstrated to be sufficiently covered by the validated range of NRELAP5. NRELAP5 showed reasonable to excellent agreement with test data for all phenomena at conditions important for the non-LOCA analysis. Computational fluid dynamics was also used to validate the model incorporated in NRELAP5 for the primary side heat transfer in the helical SGs.

A nodalization sensitivity of the steam generator for a main steam line break scenario was performed comparing the effect of modeling the SG {{

}}2(a),(c)

DHRS actuation is also expected to occur during LOCA event sequences. For smaller LOCA break sizes, there may be an extended period where DHRS is in operation but ECCS is not yet actuated. In a small break LOCA sequence where there is such an extended period, the phenomena associated with DHRS performance are applicable. The DHRS heat removal before ECCS actuation is important to determine {{

}}2(a),(c) The effect of these different ranges of conditions has been examined and the impact on DHRS performance is reasonable considering underlying physics of the integral system behavior. NRELAP5 sensitivity calculations show that the discharge of RCS mass and energy during a small break LOCA does not

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degrade the performance of DHRS from the transient initiation until ECCS actuation near the top of the riser. No new important phenomena are identified due to DHRS operation in conjunction with a small liquid or vapor break, before ECCS actuation. Therefore, conclusions of NRELAP5 applicability to simulate non-LOCA SG/DHRS performance can be extended to SG/DHRS performance during LOCA conditions, particularly small LOCA conditions with an extended phase of liquid flow over the top of the riser and DHRS heat removal prior to ECCS actuation.

A bottom-up code applicability evaluation for the non-LOCA EM focused on NRELAP5 model ranges for SG/DHRS boiling/condensation wall heat transfer phenomena. {{

}}2(a),(c) It is concluded that the models and correlations implemented in NRELAP5 are applicable to simulate NPM SG/DHRS heat transfer.

A top-down scaling evaluation was performed to evaluate PI groups for high-ranked SG/DHRS phenomena in the non-LOCA PIRT and compare the relative importance between the prototype (NPM) and model (NIST-2 test facility). {{

}}2(a),(c) The top-down scaling analysis demonstrates the consistency in the dominant PI groups for SG/DHRS operation between the NPM and NIST-2 while phenomena/processes that are less important may be more distorted between the prototype and model. It is concluded that NRELAP5 is applicable for NPM non-LOCA integral analysis.

Considering the high-ranked phenomena identified from the PIRT process, the NRELAP5 code along with an NPM system model is applicable for calculation of an NPM system response for the non-LOCA short-term transient event progression as part of this EM based on separate effects and integral effects testing, code-to-code benchmarking, bottom-up and top-down evaluations of SG/DHRS performance, and appropriate conservative input for initial and boundary conditions.

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6.0 NuScale NRELAP5 Plant Model

This section discusses the NuScale NRELAP5 non-LOCA plant transient model. A summary overview of the plant components and features simulated by the NRELAP5 model is provided, including the reactor primary and secondary (SG) systems, core fuel rods and kinetics, ECCS and DHRS, containment and reactor pool, and trips and controls.

The NRELAP5 plant model is developed to support the non-LOCA analysis methodology described in Section 7.0. The model is developed following the NRELAP5 code manual user guidelines, supplemented by NuScale-specific modeling guidelines. The guidelines describe how to model a NuScale Plant Module using the NRELAP5 code, and include directions on how to select code options, nodalizing the system, and selecting heat transfer correlations.

The NuScale NRELAP5 non-LOCA plant transient model described in this section is considered typical and representative of the methodology. The base model for a specific NPM design reflects the parameters and features of that design. For example, an NPM design with fewer RVVs uses fewer NRELAP5 components to model the RVVs. Such NPM design differences do not affect the method used to develop the base model.

6.1 Thermal-Hydraulic Volumes and Heat Structures

The NRELAP5 plant model contains multiple hydraulic components, heat structures and junctions. The model simulates the majority of a typical NPM (Figure 6-1) including the RPV and internals, the containment, and the reactor cooling pool. {{

}}2(a),(c) Both DHRS trains are included in the model, along with the ECCS consisting of the RVVs and RRVs. Control system components include variable and logical trips, control blocks and general tables. Figure 6-2 shows a typical nodalization diagram for the primary and secondary systems and is meant to convey the overall model structure rather than show nodalization details of any particular component.

Figure 6-3 shows a cut-away of the typical NPM reactor coolant system and CNV with the key nodalization regions included in the NRELAP5 model. The circled numbers in the figure represent RCS fluid regions and the numbers in squares represent containment regions. Table 6-1 lists the typical RCS regions and the associated NRELAP5 components.

The NRELAP5 model of an NPM serves as the standalone baseline model for non-LOCA safety analysis, as well as various aspects of plant design support. The information presented herein describes a typical base model as it is configured for non-LOCA analysis.

{{

}}2(a),(c)

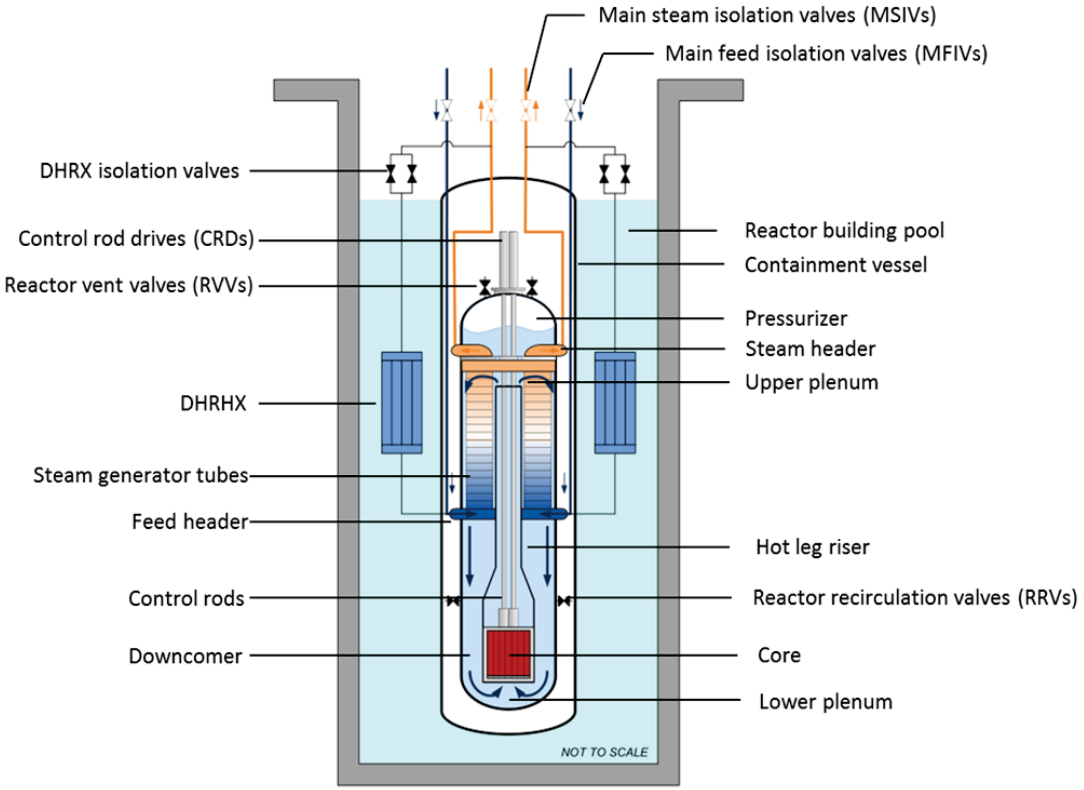
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{{

}}2(a),(c)

Where precise noding is described in Section 6.1, the specified level of detail is considered the minimum level of detail required for the component of interest. Should additional detail be needed in the future, the relevant benchmarks, sensitivity studies, and transient analyses will be reviewed for continued applicability. If necessary, the relevant benchmarks, sensitivity studies, and transient analyses will be revised to demonstrate the higher level of detail for the component of interest is applicable to an NPM.

Figure 6-1 NuScale Power Module (typical)



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Figure 6-2 Typical primary and secondary side model (heat structures and component cell details excluded)

{{

}}2(a),(c)

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Figure 6-3 Typical NRELAP5 plant module volume regions

{{

}}2(a),(c)

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Table 6-1 Typical reactor coolant system regions and associated NRELAP5 components

{{

}}2(a),(c)

6.1.1 Reactor Primary

Reactor Pressure Vessel Downcomer

{{

}}2(a),(c)

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Figure 6-4 Typical reactor pressure vessel downcomer model

{{

}}2(a),(c)

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Steam Generator Primary

The SG is a helical coiled, once-through HX, with the primary system on the shell side and the secondary system on the tube side. The primary system water is cooled as it flows over the outer surfaces of the SG tubes before passing over the feedwater plena that are located above the elevation of the conical transition riser fairing in the RPV. On the secondary side, feedwater enters the bottom of the SG tubes via the feedwater plena and is heated as it flows upward, with superheated steam exiting the tops of the tubes. Two independent sets of interwoven SG tube banks occupy the SG region, each having independent feedwater and steam plena. If the tube banks experience different secondary side conditions, the primary coolant does not experience any corresponding asymmetries because of the interwoven design of the helical coiled tubes.

The heat transfer to the SG tubes occurs in the upper downcomer. The heat transfer and pressure drop resulting from the presence of the SG tubes was assessed with data from SIET TF-2 (Section 5.3.5). This assessment used special heat transfer model options to determine the methodology for accurately predicting the pressure drop in the region. {{

}}2(a),(c)

Core and Lower Plenum

{{

}}2(a),(c)

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{{

}}2(a),(c)

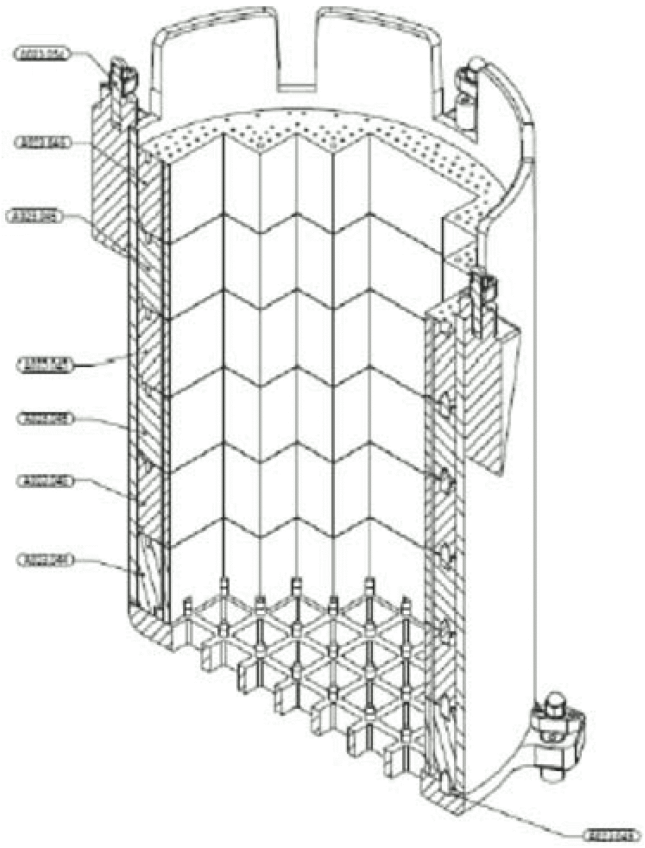
Figure 6-5 Typical core and lower plenum model

{{

}}2(a),(c)

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Figure 6-6 Reflector / core bypass without fuel assemblies (for illustration only)



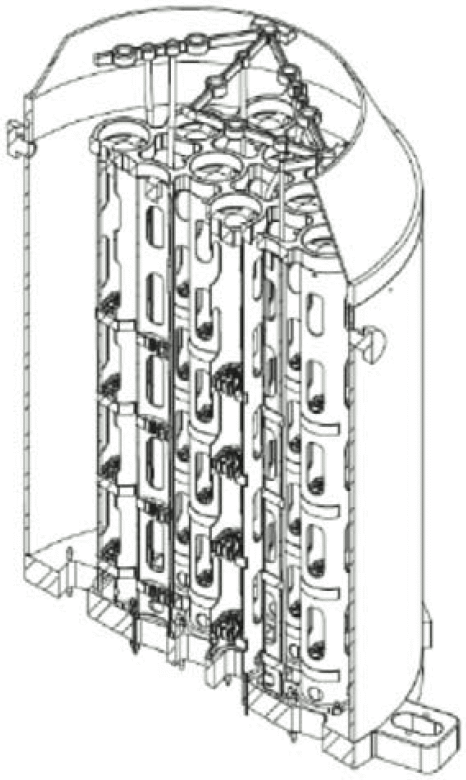
Lower Riser

{{

}}2(a),(c)

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Figure 6-7 Lower riser region, immediately above the core (for illustration only)



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Figure 6-8 Typical reactor pressure vessel core and lower riser model

{{

}}2(a),(c)

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Upper Riser

{{

}}2(a),(c) Normal flow in the riser is single-phase subcooled water. Transients that involve RPV depressurization or inventory loss can result in flashing and two-phase flow in the riser region.

{{

}}2(a),(c)

Figure 6-9 Typical reactor pressure vessel upper riser model

{{

}}2(a),(c)

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Pressurizer

{{

}}2(a),(c)

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Figure 6-10 Typical reactor pressure vessel pressurizer model

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}}2(a),(c)

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6.1.2 Core Kinetics

The separable point kinetics model is used to calculate reactivity feedback to the core power from the moderator, the fuel, and decay heat. The various point kinetic parameters are input based on the fuel burnup (for example, new core, BOC, or EOC) and control rod insertion amounts assumed for the analysis. NRELAP5 assumes an infinite core operating time at the initialized power when determining the decay heat power. The fission product decay type is specified as 'gamma-ac' with the 'ans73' model, which calculates decay heat in accordance with the 1973 ANS standard while adding the contribution from actinides. A fission product yield factor of 1.0 is specified in the base models, which can be changed to suit the scenario being analyzed.

Control variable inputs to the core kinetics model are used to simulate control rod movement and reactivity feedback from a variety of parameters including, but not limited to fuel temperature, moderator temperature, and moderator density. Trips are used to enable or disable each reactivity feedback mechanism as needed. The fuel and moderator feedback controllers utilize volume weighting of the fuel and moderator temperatures to specify bounding reactivity feedback input; volume weighting is used for consistency with how the reactivity feedback design limits are confirmed in the core design.

A scram table is used to simulate control rod insertion following reactor trip. Appropriately conservative scram curves are developed based on the core time-in-life, the initial power level, the location of the control bank, and other relevant factors to preserve the minimum shutdown margin.

6.1.3 Fuel Rod Design Input

{{

}}2(a),(c) Fuel performance data (Section 4.3.1.2) is incorporated into the NRELAP5 non-LOCA model via material thermal property tables for the UO_2 fuel region, the gas gap, and the cladding. Because the density of UO_2 changes with burnup, the thermal property tables can be revised as needed to match time in cycle.

As discussed in Section 4.3.1.1, the core power distribution is based on a nominal average axial power shape with power distributed solely in the fuel pellet. {{

}}2(a),(c)

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{{

}}2(a),(c) The gap thermal conductivity is calculated at bounding values burnup to ensure that the fuel volume average temperature is appropriately bounded compared to fuel performance design data. This bounding method also accounts for gap closure over the fuel cycle.

6.1.4 Secondary System

6.1.4.1 Feedwater System

In an NPM design two feedwater lines penetrate the CNV immediately downstream of the FWIVs. Each feedwater line splits into two lines before connecting to the SGs. {{

}}2(a),(c)

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6.1.4.2 Steam Generator Secondary

The NRELAP5 specific helical coil SG component ('hlcoil') is used to simulate the helical coil SG that is characteristic of an NPM (Reference 2 describes the helical coil component). The typical steam generator nodalization is shown in Figure 6-11. The primary coolant flows through the SG shell side while the feedwater and steam flow through the tube side. The tube and shell side of the SG elevation nodalization schemes are one-to-one. The SG nodes are uniform, or may be finer towards inlet and coarser towards the exit if necessary to capture the phase change process in the lower region of the tubes. For non-LOCA transient calculations, finer nodalization is not needed near the tube exits due to the state of the fluid being single phase vapor. Section 5.3.5.4 summarizes SG nodalization sensitivity calculations. Based on these studies, modeling the helical coil SG with {{ }}$^{2(a),(c)}$ nodes is expected to produce reasonably accurate results for the non-LOCA transients. Should additional detail be needed in the future, the relevant benchmarks, sensitivity studies, and transient analyses will be reviewed for continued applicability and updated as necessary to demonstrate the higher level of detail for the component is applicable to an NPM.

{{

}}$^{2(a),(c)}$

For an NPM design that includes an SG tube inlet flow restrictor, {{

}}$^{2(a),(c)}$

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Figure 6-11 Typical steam generator model

{{

}}2(a),(c)

6.1.4.3 Main Steam System

{{

}}2(a),(c) The typical MSS nodalization is shown in Figure 6-12.

{{

}}2(a),(c) DHRS modeling is discussed in Section 6.1.5.

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure 6-12 Typical main steam system model

{{

}}2(a),(c)

6.1.5 Decay Heat Removal System

An NPM incorporates two separate DHRSs that are treated individually in the NRELAP5 model. {{

}}[2(a),(c)] Figure 6-13 shows the typical nodalization for DHRS loop 1 {{

}}[2(a),(c)] Loop 2 is modeled similarly. While each DHRS line in an NPM features two parallel actuation valves, {{

}}[2(a),(c)]

The number of hydrodynamic volumes in the DHRS condenser piping and HX regions are based on results from NRELAP5 assessments using data from the NIST-1 facility (Section 5.3.2). {{

}}[2(a),(c)]

In addition to the DHRS condenser piping and HX regions, heat removal occurs in the DHRS steam line piping below the level of the reactor cooling pool and from the DHRS condenser headers. {{

}}[2(a),(c)]

In an actual NPM, the DHRS condensers are located in the reactor cooling pool. {{

}}[2(a),(c)] The long-term use of DHRS is addressed separately in Reference 26.

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Figure 6-13 Typical decay heat removal system division 1 model

{{

}}2(a),(c)

Figure 6-14 Not used.

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6.1.6 Emergency Core Cooling System

The ECCS hydrodynamic components consist of two reactor recirculation valves (RRVs) and, depending on the NPM design, either two or three reactor vent valves (RVVs). {{

}}2(a),(c)

Figure 6-15 Not used.

6.1.7 Containment Vessel

{{

}}2(a),(c)

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{{

}}2(a),(c)

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Figure 6-16 Typical containment and reactor pool model

{{

}}2(a),(c)

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6.1.8 Reactor Cooling Pool

{{

}}2(a),(c)

6.2 Material Properties

Thermal properties (thermal conductivity and volumetric heat capacity) are specified by user input in the NRELAP5 non-LOCA base models for the following materials typically used in the heat structures. These material properties may be amended or revised as the NPM design evolves:

1. fuel cladding (AREVA's M5® cladding)
2. inconel 690 (SG tubes)
3. uranium dioxide (UO_2)
4. stainless steel (SA-240 304L)
5. fuel-to-cladding gas gap (initially pressurized helium at BOC; mixture of fission product gases and helium after irradiation)
6. carbon steel (SA-508)
7. martensitic stainless steel (SA-336, F6NM)
8. austenitic stainless steel (SA-965, FXM-19)
9. {{

}}2(a),(c)

6.3 Control Systems

With its combination of trips, control functions, and user-defined tables, NRELAP5 provides flexibility to accurately simulate plant control and protection system responses during both steady-state and transient operation. The NRELAP5 non-LOCA base models contains logic for "normal controls" that simulate normal operational plant response, as well as user-convenience controls that make it easier to initialize the model for particular

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transients and easier to interpret the transient results. It also contains trip and control logic that accurately simulates the MPS, i.e., the safety-related trips that protect the reactor core and fission product boundaries.

6.3.1 Module Control System (Nonsafety-related)

The MCS as implemented in the NRELAP5 non-LOCA model provides control systems that may be used for model initialization and for simulating normal operating transients. These control systems can be used to initialize the model at new steady state conditions or to evaluate the nominal MCS response to a transient initiator. The MCS model allows the simulation of a variety of prototypic control scheme responses to transient conditions to provide nominal responses of these control schemes. The model also includes the capability to disable these control schemes to allow for conservative modeling applications as necessary.

6.3.1.1 Pressurizer Pressure Control (Nonsafety-related)

The RPV pressure is controlled via the use of pressurizer heaters and spray to maintain the pressurizer steam pressure at a target value. When the pressure drops, the power to the PZR heaters increases, and if pressure increases, a portion of the CVCS recirculation flow is diverted to the pressurizer spray nozzles to collapse the steam space via condensation and reduce pressure.

6.3.1.2 Chemical Volume Control System Control (Nonsafety-related)

The control systems implemented in the NRELAP5 non-LOCA model include simplified mechanisms for controlling CVCS recirculation flow, injection temperature and RCS inventory control (makeup and letdown). {{

}}[2(a),(c)] The water level in the pressurizer is controlled to a programmed setpoint by operation of the CVCS makeup pumps and the letdown control valve. Operation of CVCS letdown may either occur automatically or with operator permission, depending on the design. Operator permission is required for CVCS makeup to be initiated.

For non-LOCA transient analyses where loss of inventory or inventory shrinkage is expected to indicate makeup is needed, the process by which the operator approves makeup is considered. Since approval to initiate makeup is a manual action by the operator, the non-LOCA transient analyses do not credit this action. For spurious inventory addition events where letdown could reduce the event consequences, credit for this action is not taken for the non-LOCA transient analyses.

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6.3.1.3 Reactor Coolant System Temperature Control (Nonsafety-related)

The MCS model controls RCS average temperature by changing reactivity in the core to increase or decrease core power. This reactivity change is accomplished with a control rod controller and a boron controller. The control rod controller uses design data to model a calculated rate of reactivity insertion due to maximum or nominal rod movement rates. {{

}}2(a),(c) Neither controller accounts for all the actual core physics including the effect of xenon or other decay products or poisons that could be expected with control rod repositioning.

The average coolant temperature is controlled by adjusting core power, which is accomplished by moving the control rods or changing the boron concentration of the reactor coolant. The choice of which method is based on the desired rate of change for core power. The control rods are moved to achieve faster power changes to meet the target average coolant temperature; slower power changes are accomplished by changing the boron concentration of the reactor coolant. At full power, the rod control system is set to 'insert only' mode to prevent automatic withdrawal of the control rods during a transient. If this design feature is neglected in the model, the rod control system will allow control rods to be withdrawn even if reactor power is at or above full power, resulting in a conservative power response to a transient.

6.3.1.4 Steam Pressure Control (Nonsafety-related)

In an NPM design, the turbine throttle and bypass valves are used to control steam pressure at the programmed values, {{

}}2(a),(c)

6.3.1.5 Feedwater and Turbine Load Control (Nonsafety-related)

An NPM prototypic control scheme design for the feedwater system is based on turbine load demand. The feedwater pumps are variable speed and can provide variable flow for module operations over a wide range of power without adjustments to the feedwater regulating valve. {{

}}2(a),(c)

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{{

}}2(a),(c)

6.3.1.6 Containment Pressure Control (Nonsafety-related)

The containment pressure is established at sub-atmospheric conditions via operation of the containment evacuation system. The impact of this system continuing to operate is considered for the non-LOCA transient analyses.

6.3.2 Module Protection System (Safety-related)

6.3.2.1 Analytical Limits and Delays

The MPS implemented in the NRELAP5 base models is intended for the purposes of performing safety analysis transient simulations. As such, the logic and actuation points are based on the NPM safety analysis analytical limits. Fixed delay times are specified considering different sensor response times. {{

}}2(a),(c) In addition to the sensor delays, a given safety signal is subject to instrumentation string delays, an MPS processing delay, and an actuation delay. The NRELAP5 non-LOCA model incorporates the methodology assumption that a bounding total for these additional delays is applied as a signal delay in addition to the individual sensor delay.

The MPS actuation signals in an NPM design are typically based on the following types of parameters:

- power range power
- power range power rate
- intermediate range log power rate
- source range count rate
- source range log power rate

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- RCS flow
- RCS temperature
- pressurizer pressure
- pressurizer level
- RPV riser level
- containment pressure
- containment water level
- main steam pressure
- main steam superheat

These parameters can be monitored in NRELAP5 to credit MPS actuations in the safety analysis. A single parameter may be used for multiple MPS actuations. For example, a high pressurizer pressure signal could result in reactor trip, containment isolation, and DHRS actuation. The MPS actuation signals may vary by NPM design and may be incorporated into the NRELAP5 base model for the design.The control logic for other MPS signals may be added if needed for a particular event analysis or as necessary to maintain consistency with the MPS design.

6.3.2.2 NRELAP5 Modeling

The NRELAP5 components used to model the MPS include variable and logical trips that are used to sense the MPS trip limits, apply signal delays and generate actuation signals. The MPS instrumentation and sensors are typically modelled by comparing the volume or junction parameter at the location of the sensor and comparing it to the associated analytical limit, rather than explicitly modeling the sensor.

The reactor trip system (RTS) logic is fairly simple, as each RTS signal is computed and compared against the trip setpoint on each timestep. The trips are organized together in a series of “or” logical trips. Should one trip limit be reached, the cascade of logical trips immediately reach the final RTS logical trip, which then causes the RPS actuation signal to become true after a fixed delay that conservatively accounts for signal processing and rod latch mechanism delays. Other subsystems such as containment isolation, DHRS actuation, ECCS actuation, etc. are modeled similarly.

{{

}}2(a),(c)

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The pressurizer level signal is generated by modeling the collapsed liquid level, {{

}}2(a),(c)

The ECCS actuation logic includes the ability for the user to set additional electrical or mechanical conditions that are external to the NRELAP5 hydrodynamic model. {{

}}2(a),(c) The effects of external conditions can be added to any of the control system models as needed for a given transient scenario.

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7.0 Non-LOCA Analysis Methodology

7.1 General

The discussion in the following sub-sections is applicable for all transients unless the event-specific methodology states otherwise in Section 7.2.

7.1.1 Achieving Steady State Conditions

This section identifies the initial and boundary conditions considered for biasing in non-LOCA analyses, including prioritization during the initialization process. While the majority of parameters identified herein are initial conditions relevant to the steady state, other parameters are considered as bounding input for an NPM response during the transient progression.

7.1.1.1 Background

Establishing the appropriate initial conditions is paramount to obtaining an appropriate plant response for the transient of interest. To this end, it is important to ensure the NRELAP5 model achieves a valid steady state prior to initiating the transient.

The effect of various initial conditions on the response to a specific acceptance criterion is assessed for each non-LOCA transient. Several means are available to perform the assessment. For example, the assessment may consist of a combination of:

- Qualitative engineering assessment in the calculation to identify why an initial condition bias is limiting or non-limiting
- Quantitative assessment by execution of appropriate sensitivity calculations in the calculation
- Reference to applicable regulatory precedent that identifies why a particular initial condition bias is not limiting for a particular transient or type of transient, or why a nominal condition is appropriate

7.1.1.2 Identification of Relevant Parameters

A list of initial conditions to be considered for biasing was developed for the non-LOCA analyses, including the target value for the initial plant condition and acceptable tolerance to the target value.

Other parameters needed to obtain a steady state include: certain fuel-related and core-related inputs; measurement uncertainties for various safety-related processes; plant operational limits assumed in the safety analyses; and the DHRS initial conditions.

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Table 7-1 identifies a typical list of initial conditions considered for non-LOCA analyses. This list of initial conditions includes both direct inputs and calculated results. Most of these parameters are directly input to NRELAP5. The calculated results are identified as a “target” because the analyst uses this parameter as a target during the initialization process.

Under best-estimate primary flow conditions and various reactor powers, it has been shown that there is relatively little variation in liquid inventory in the steam generator. The inventory does not change appreciably when considering variations in primary coolant temperature and primary flow rates. Consequently, steam generator inventory is not a target during the initialization process.
{{

}}2(a),(c)

For reactor powers greater than or equal to 20 percent full power (FP) and best-estimate primary flow conditions, the steam superheat increases with power for the defined main feedwater conditions. {{

}}2(a),(c)

The removal of any restrictions regarding steam generator inventory and steam superheat allows use of the parameters in Table 7-1 for initialization without over-specifying or under-specifying the problem.

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Table 7-1 Typical list of initial conditions considered

Region	Parameter
Core	Power
	Calorimetric power uncertainty
	Volume-weighted core average fuel temperature (target)
	Moderator temperature coefficient
	Doppler temperature coefficient
	Effective delayed neutron fraction
	Ratio of effective delayed neutron fraction to prompt neutron lifetime
	^{238}U neutron capture to fission ratio
	Energy deposition factor
	Average core axial power shape
	Scram reactivity (shutdown margin)
	Control rod bank differential worth
	Decay heat and decay heat uncertainty
	Boron concentration
RCS	Primary system mass flow rate (target)
	RCS average fluid temperature (target)
	Pressurizer pressure
	Pressurizer level
	Total core bypass flow rate
	Pressurizer spray bypass flow rate
	Pressurizer heater power
	Heat losses to containment from the RPV and from piping inside containment
Steam generator secondary, feedwater system, and MSS	Feedwater mass flow rate
	Feedwater temperature
	Steam chest pressure (turbine header pressure)
	Steam generator tube plugging, fouling factor
Containment	Pressure
	Heat losses from containment to reactor pool and reactor building
	Temperature
Reactor pool	Temperature
	Level
DHRS	Liquid volume

7.1.1.3 Prioritization of Initial Conditions

An important design feature of NPMs is the natural circulation of primary reactor coolant. This design feature may also limit the ability of the NRELAP5 non-LOCA transient model to achieve a given set of initial conditions. In particular, changes in core power or parameters that affect the SG secondary side heat removal rate alter the reactor coolant flow rate and density distribution in the primary coolant. The initial conditions are prioritized based on the greatest impact to the transient of interest.

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Initial conditions may be treated in the NRELAP5 non-LOCA transient model in different ways. For instance, the initial conditions can be specified directly as input or calculated by the code. Parameters identified as important initial conditions for a particular transient (whether directly input or indirectly determined by the code) are checked as part of the steady state balance to confirm that either: 1) the parameter is within the allowable tolerance to the target value based on design references; or, 2) the parameter conservatively bounds the target design value and is adequately steady (within acceptable tolerances). If the initial condition is calculated by the code but not identified as important for a transient, the parameter may or may not be directly checked as part of the steady state initialization process against design values.

7.1.1.4 Typical Initialization Process

Prior to the initialization process, certain parameters critical to establishing the correct steady state for the event of interest are identified. Once the parameters of interest achieve steady state target values within acceptable tolerances, on the basis of engineering judgement, {{ }}$^{2(a),(c)}$ A steady state solution is achieved when the change in the target value during the loop transits is within the variance band described for each parameter.

A typical list of critical parameters for initializing the primary and secondary systems is provided below.

- reactor power: tolerance (or bounded) and variance
- fuel temperature: tolerance (or bounded) and variance
- RCS temperature: tolerance (or bounded) and variance
- PZR pressure: tolerance (or bounded) and variance
- PZR level: tolerance (or bounded) and variance
- RCS flow: tolerance (or bounded) and variance
- steam pressure: tolerance (or bounded) and variance
- feedwater flow rate: variance
- feedwater temperature: tolerance (or bounded) and variance

Null transients are used to ensure the re-initialization cases achieve the desired steady state conditions. {{ }}$^{2(a),(c)}$ In addition, feedback from the point kinetics model is active for the null transients.

7.1.2 Treatment of Plant Controls

Control systems are necessary to maneuver an NPM within the power range in accordance with normal operating transients. However, not all of these control

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systems are relevant to the non-LOCA transient analyses because the action of the control system does not alter the event consequences. This section identifies the normal, nonsafety-related plant controls considered in non-LOCA transient analyses.

The effect of various plant controls on the response to a specific acceptance criterion is assessed for each non-LOCA transient. Several means are available to perform the assessment. For example, the assessment may consist of a combination of:

- Providing a qualitative engineering assessment that identifies why operation of a PCS is limiting or non-limiting.
- Performing a quantitative assessment via appropriate sensitivity cases.
- Referencing to applicable regulatory precedent that identifies why a particular normal plant control is not limiting for a specific transient or type of transient.

When considering operation of the various plant controls, the approach is based on the event consequences for a given acceptance criterion. Specifically, if operation of the control system leads to a less severe plant response, then the actions of the control system are not simulated for the transient of interest. Conversely, if operation of the control system causes the event consequences to be more severe, the PCS is assumed to operate as designed. If operation of the control system has minimal or no impact on the event consequences, then control system operation may either be modeled or not modeled. {{

}}2(a),(c)

The normal PCSs to be considered in design basis event analysis are:

Pressurizer Pressure Control

Pressurizer pressure is controlled via operation of the pressurizer spray and the pressurizer heaters.

Pressurizer Water Level Control

The water level in the pressurizer is controlled to a programmed setpoint by operation of the CVCS makeup pumps and the letdown control valve. Operation of CVCS letdown may either occur automatically or with operator permission, depending on the design. Operator permission is required for CVCS makeup to be initiated.

For non-LOCA transient analyses where loss of inventory or inventory shrinkage is expected to indicate makeup is needed, the process by which the operator approves makeup is considered. Since approval to initiate makeup is a manual action by the operator, the non-LOCA transient analyses do not credit this action. For spurious inventory addition events where letdown could reduce the event consequences, credit for this action is not taken for the non-LOCA transient analyses.

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Core Average Coolant Temperature Control

The average coolant temperature is controlled by moving the control rods or changing the boron concentration of the reactor coolant. This selection is based on the desired rate of change for core power. Hence, the control rods are moved to achieve faster power changes to meet the target average coolant temperature; slower power changes are accommodated by changing the boron concentration of the reactor coolant. At full power, the rod control system is set to ‘insert only’ mode to prevent automatic withdrawal of the control rods during a transient. Neglecting this feature, if assumed, is conservative.

Steam Pressure Control

Steam pressure is controlled to the desired value using the turbine throttle valves or the turbine bypass valves. The effect of these valves to a change in steam pressure is considered for the non-LOCA transient analyses.

Turbine Load Control

The mass flow rate and pressure provided by the feedwater pump is used to meet the desired turbine load, which reflects the power generation rate. The impact of the feedwater pumps continuing to operate until the feedwater line is isolated is considered for the non-LOCA transient analyses when DHRS is actuated.

Containment pressure control

The containment pressure is established at sub-atmospheric conditions via operation of the containment evacuation system. The impact of this system continuing to operate is considered for the non-LOCA transient analyses.

7.1.3 Loss of Power Conditions

This section defines the term “loss of normal power” as applied to an NPM; describes the various power supplies (AC and DC); and, explains how the loss of these power supplies is treated by the non-LOCA transient analyses.

7.1.3.1 Background

Chapter 15 of the SRP (Reference 15) does not ordinarily consider a loss of offsite power for events that require a malfunction of an active system for which power must be available; however, exceptions are made for some reactivity initiated events. The role of offsite power is less defined for an NPM plant than for traditional plants for several reasons, but the use of natural circulation for normal operation and safety systems is the fundamental reason. Consequently, these design features limit the impact of a power loss to an NPM plant compared to a traditional plant design that relies on forced circulation.

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The term ‘loss of normal power’ is used herein because the normal power source for an NPM plant is not the offsite grid, and there are no onsite safety-related power generating sources (AC or DC). Although NPM plant designs do not include a safety-related electrical power generation source, the designs include two different DC power systems, both with a battery backup. This battery backup allows the DC power system that supplies MPS to provide uninterrupted DC power for 72 hours of MPS operation. Because the MPS also provides power to keep the ECCS valves closed, if a loss of normal AC power occurs, the MPS sheds the load to the ECCS valves after 24 hours. The load shedding allows the ECCS valves without IAB to open at that time; ECCS valves with IAB will open when the pressure difference between the RPV and the containment is below the IAB threshold.

The effect of various power availability scenarios on the response to a specific acceptance criterion is assessed for each non-LOCA transient. This assessment can be performed using several methods. For example, the assessment may consist of a combination of:

- providing a qualitative engineering assessment that identifies why operation of a power availability scenario is limiting or non-limiting
- performing a quantitative assessment via appropriate sensitivity cases
- referencing to applicable regulatory precedent that identifies why a particular power availability scenario is not limiting for a specific transient or type of transient

When considering various power availability scenarios, the approach is based on the event consequences for a given acceptance criterion. Because nonsafety-related AC power sources are not credited for the non-LOCA event analyses, in all scenarios where AC power is lost, the ECCS valves open at some point during the transient. For non-LOCA transients that do not demand ECCS actuation during the short-term event progression, i.e., during the time frame in which the peak pressures, MCHFR, or peak fuel centerline temperature occur, the time at which the ECCS valves open is beyond the scope of this methodology document. Analysis of ECCS valve opening events is addressed using the Reference 2 methodology.

7.1.3.2 Consideration of Loss of Power to All Transients

The subsections of SRP Chapter 15 (Reference 15) applicable to an NPM, and the Chapter 15 DSRS sections, were reviewed to identify which subsections explicitly address a loss of power or GDC 17 (Reference 4). This review concluded that given the differences between the design of an NPM and typical PWRs, particularly with respect to the safety system design and response, the effect of loss of power on each non-LOCA transient should be considered.

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7.1.3.3 Electrical Systems with Important Loads

A review of the electrical systems applicable to the NPM designs was undertaken to identify which electrical systems have loads important to the non-LOCA transient analyses. Consideration was given to both direct and indirect impact to the transient analyses. A direct impact to the transient analyses is one that alters the primary or secondary side conditions if the electrical system is lost. An indirect impact to the transient analyses is one that affects the ability to monitor the system or affects a connecting system. The following typical electrical systems have loads important to the non-LOCA transient analyses.

- EHVS - High voltage (13.8 kV) AC electrical distribution system
- EMVS - Medium voltage (4.16 kV) AC electrical distribution system
- ELVS - Low voltage (480 V and 120 V) AC electrical distribution system
- EDNS - Normal DC power system
- EDSS - Highly reliable DC power system
- EDAS - Augmented DC power system

The NPM designs utilize either an EDAS or EDSS, depending on the design. The EDNS and the EDAS/EDSS power systems are both designed with battery backups to allow a continuous power supply in the event the power supply to the chargers is not available. The duration of the battery backup is based on the relative importance of the loads. Because the loads supplied by the EDNS are of lower importance (i.e., all loads are non-essential) the batteries are typically sized to supply these loads for 40 minutes. In contrast, for the higher importance (essential) loads supplied by the EDAS/EDSS, with the exception of the ECCS valves, the batteries are sized to supply these loads for greater than 24 hours. The ECCS valves are unique because the MPS acts to shed the load for these valves at 24 hours after the loss of AC power to the EDAS/EDSS battery chargers. The loss of normal AC power to the EDAS/EDSS chargers also causes the MPS to initiate a reactor trip, actuate DHRS, and close the containment isolation valves.

Based on the electrical system design for an NPM, a loss of normal AC power means a loss of power from the ELVS. Such a condition could be due to: 1) a failure within the ELVS; or, 2) a loss of power from the EHVS or EMVS. Since a failure of the EHVS or EMVS results in the same response from the electrical systems for non-LOCA transient analyses, these failures are not considered separately. With respect to the battery backups for the DC power supply systems, the availability of the battery backups is considered because the EDNS and EDAS/EDSS are not safety-related systems.

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7.1.3.4 Timing for Loss of Power

The review of the subsections of SRP or DSRS Chapter 15 discussed in Section 7.1.3.2 indicates the SRP/DSRS is not prescriptive regarding when the loss of power occurs. An example is provided in DSRS Subsection 15.1.5, which states:

> *Assumptions as to the loss of offsite power (LOOP) and the time of loss should be made to study their effects on the consequences of the accident. A LOOP could occur simultaneously with the pipe break or during the accident, or offsite power may not be lost.*

For the non-LOCA transient analyses applicable to an NPM, the loss of normal AC power is considered at two different times during the event progression:

- coincident with event initiation; or,
- coincident with turbine trip.

The basis for considering a loss of power coincident with event initiation is the loss of normal AC power could be caused by the event initiator or as a result of the event. For example, the mass and energy exiting a ruptured steam pipe could damage equipment that results in a loss of normal AC power.

The basis for considering loss of power coincident with turbine trip is because the loss of power is a consequence of the event. Specifically, the loss of AC power occurs as a result of a disruption to the electrical grid following a turbine trip. Because the electrical output for an NPM is much smaller (less than 100 MWe) than the electrical output for a typical large plant (approxiately 1000 MWe), the loss of an NPM is not anticipated to significantly disrupt the electrical grid. Regardless, the possibility of causing such a disruption is assessed as part of the non-LOCA transient analyses.

The random loss of a nonsafety-related system during an event is not typically required as part of the evaluation, and is not being postulated for the NPMs. If the EDNS or EDAS/EDSS design demonstrates that its failure is not linked to failure of the normal AC power supply, then the timing of a loss of EDNS or EDAS/EDSS is considered independently of the loss of normal AC timing described above. Assessment of non-LOCA events with and without the EDNS or EDAS/EDSS is consistent with their design as nonsafety-related.

7.1.4 Single Failures

This section discusses active single failures in fluid systems and single failures in electrical systems for consideration in non-LOCA transient analyses, as well as the relevance of fluid system passive single failures.

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7.1.4.1 Background

The effect of various single active failures on the response to a specific acceptance criterion is assessed for each non-LOCA transient. Several means are available to perform the assessment. For example, the assessment may consist of a combination of:

- Providing a qualitative engineering assessment that identifies why a single failure is limiting or non-limiting.
- Performing a quantitative assessment via appropriate sensitivity cases.
- Referencing to applicable regulatory precedent that identifies why a particular single failure is not limiting for a specific transient or type of transient.

When considering various single active failures, the approach is based on the event consequences for a given acceptance criterion. Only safety-related components are considered for possible single active failures.

A nonsafety-related component whose action benefits the transient consequences is typically assumed inactive, unless the component is demonstrated to not be affected by the initiating event or the component is acting as backup protection. This treatment of nonsafety-related components is consistent with RG 1.206 (Reference 14 Section C.I.15.6.2), which states:

> *Only safety-related systems or components should be used to mitigate transient or accident conditions. However, analyses may assume that nonsafety-related systems or components are operable for the following cases:*
>
> > *(1) when a detectable and nonconsequential random and independent failure must occur in order to disable the system*
> >
> > *(2) when nonsafety-related components are used as backup protection*
>
> *...*
>
> *For any nonsafety-related systems or components credited in the design-basis analyses for mitigating the event consequences, the applicant must provide proper justification. The application must take into account nonsafety-related systems or components that may adversely affect transient or accident analyses.*

Safety-related SSC are defined in 10 CFR 50.2 (Reference 16), as those SSC:

> *... that are relied upon to remain functional during and following design basis events to assure:*
>
> > *1) The integrity of the reactor coolant pressure boundary;*

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2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11, as applicable.

A limited number of safety-related SSC interact with the reactor coolant system, secondary system, or containment. In most instances these safety-related SSC are considered using the definition presented in ANSI/ANS-58.9-2002 R2015 (Reference 17), i.e., an active failure is a malfunction of a component that relies on mechanical movement to complete its intended function upon demand. Passive failures of fluid systems are not considered for the non-LOCA transient analyses during the short term (up to 24 hours).

7.1.4.2 Consideration of Single Failures

An NPM considers single active failures of fluid systems or single failures of electrical systems that could affect the non-LOCA transient analyses. Various means are used to identify these failures, including: 1) the application of design criteria and industry standards to the design of the equipment; 2) the arrangement and type of valves incorporated into the design; and, 3) the level of redundancy in equipment functionality incorporated into the design.

7.1.4.3 Consideration of Passive Single Failures

An NPM does not consider passive single failures of fluid systems during the short term (up to 24 hours) non-LOCA transient analyses; however, passive single failures of electrical systems are considered. This approach is consistent with the regulatory position in SECY-94-084 (Reference 18) and SRP Chapter 15 (Reference 15).

7.1.4.4 Single Failures to Evaluate

An NPM considers the worst active single failure in conjunction with the initiating event and with loss of power for the short term non-LOCA transient analyses. During the short term, passive single failures of fluid systems are not considered; however, passive single failures of electrical systems are considered.

The following active single failures are considered for non-LOCA transient analyses.

- safety-related main steam isolation valve - failure to close when demanded
- safety-related main steam isolation bypass valve - failure to close when demanded
- safety-related feedwater isolation valve - failure to close when demanded

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- safety-related feedwater check valve - failure to close when demanded
- MPS - failure of a single channel to trip when demanded (e.g., for asymmetric reactivity events where trips based on asymmetric core response are credited for mitigation)
- ECCS reactor recirculation valve - failure to open when demanded (mechanical failure prevents opening on low ΔP)
- ECCS reactor vent valve - failure to open when demanded (mechanical failure prevents opening on low ΔP)

Depending on the design, the following single failure may be active or passive electrical, but is considered for non-LOCA transient analyses.

- MPS - failure to signal one ECCS reactor recirculation valve and one RVV to open when demanded (each valve still opens on low ΔP)

A failure of the MPS which causes spurious opening of EECS valves is not considered for non-LOCA transient analyses. The ECCS valves are either prevented from opening by IAB or their opening is analyzed using the Reference 2 methodology.

7.1.5 Bounding Reactivity Parameters

This section describes the bounding reactivity parameters used for the non-LOCA transient analyses. A brief discussion of each parameter is provided in the following sections.

- Moderator Temperature Coefficient (Section 7.1.5.1)
- Doppler Temperature Coefficient (Section 7.1.5.2)
- Decay heat contribution (Section 7.1.5.3)
- Scram worth (Section 7.1.5.4)
- Reactivity versus time for scram insertion (Section 7.1.5.5)

7.1.5.1 Moderator Temperature Coefficient

Bounding values for MTC are generally selected to represent the most positive value at BOC and the most negative value at EOC. For most non-LOCA transient analyses, the coefficient is conservatively applied as a constant value for the full range of expected moderator temperatures. If the MTC at zero power is positive, this value could be used to bound all power levels. Alternately, the positive MTC could be applied over a limited range of power levels with a different, yet bounding, MTC value for the remaining power levels.

The following MTC values are examples for a representative NPM. A review of the applicable core physics parameters is performed each cycle to confirm the bounding nature of the values utilized for the non-LOCA transient analyses.

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- Most Positive at power > 25 percent RTP = 0 pcm / degrees F
- Most Positive at power ≤ 25 percent RTP = +6 pcm / degrees F
- Most Negative at hot full power = - 43.0 pcm / degrees F
- Most Negative at hot zero power = - 15.0 pcm / degrees F

7.1.5.2 Doppler Temperature Coefficient

Bounding values for DTC are generally selected to represent the least negative value at BOC and the most negative value at EOC. For most non-LOCA transient analyses, the coefficient is conservatively applied as a constant value for the full range of expected fuel temperatures.

The following DTC values are examples for a representative NPM. A review of the applicable core physics parameters is performed each cycle to confirm the bounding nature of the values utilized for the non-LOCA transient analyses.

- Minimum (least negative) = - 1.40 pcm/°F
- Maximum (most negative) = - 2.25 pcm/°F

7.1.5.3 Decay Heat Contribution

Energy production in the core comes from fission and fission product decay. This latter component is the decay heat contribution. During the initialization process the specified kinetics inputs are used to determine the balance between fission power and fission product decay. Changing the decay heat contribution, therefore, changes the fission power because the total core power is not altered.

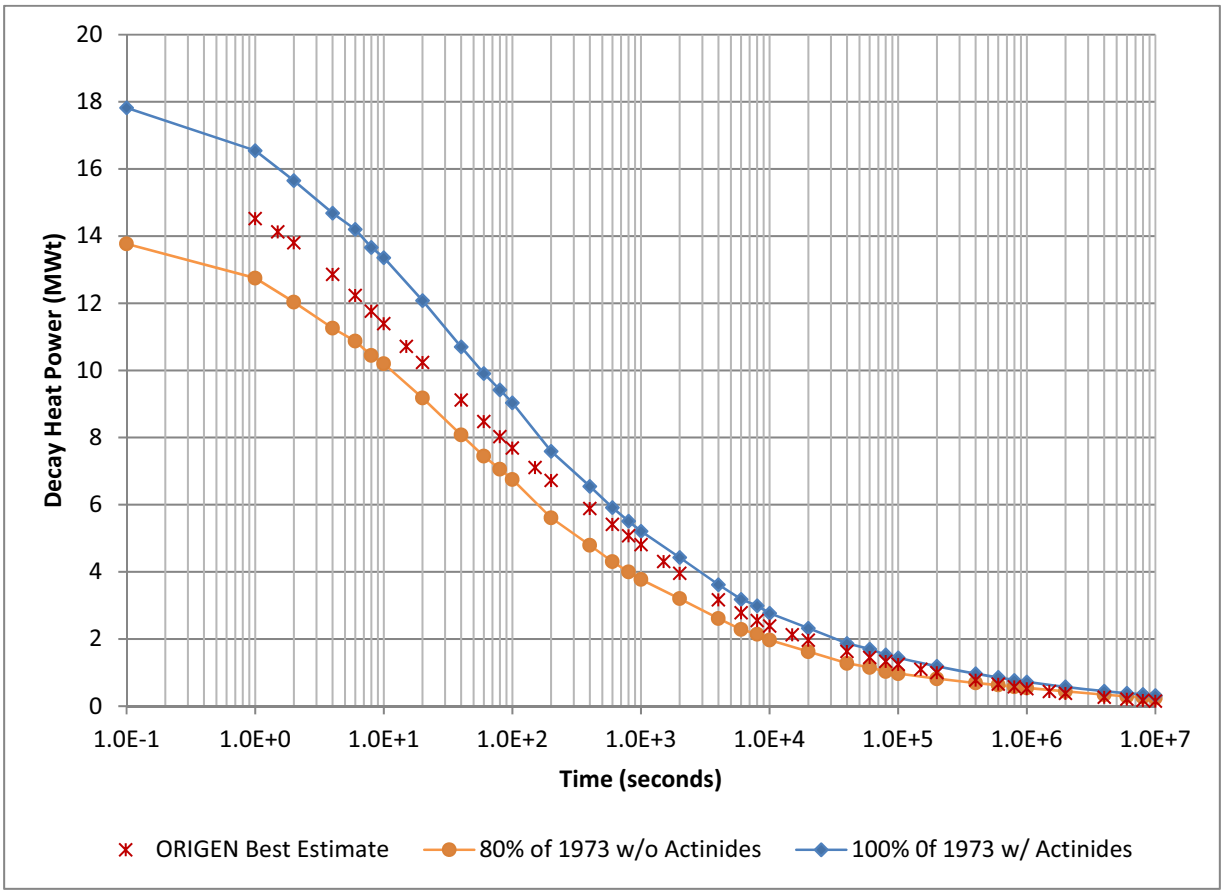
Bounding values for decay heat are designated to represent the high contribution and the low contribution. Once specified, the decay heat contribution is utilized for the duration of the event of interest. For the non-LOCA transient analyses the decay heat contribution is based on the 1973 ANS decay heat standard, which is varied by utilizing different decay heat multipliers and specifying whether or not to include the actinide contribution.

The following decay heat contribution values are examples for a representative NPM. Figure 7-1 provides an example of the decay heat as a function of time for an equilibrium cycle. A review of the applicable core physics parameters is performed each cycle to confirm the bounding nature of the values utilized for the non-LOCA transient analyses.

- Low = use multiplier of 0.8 while excluding the actinide contribution
- High = use multiplier of 1.0 while including the actinide contribution

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Figure 7-1 Example of decay heat comparisons



7.1.5.4 Scram Worth

After a reactor trip, the insertion of the control and safety banks imposes negative reactivity on the core. The reactivity available for insertion varies based on the time-in-life and the location of the control bank prior to trip. The inserted reactivity is sufficient to not exceed SAFDLs under conditions of normal operation, including AOOs; and, to limit fuel damage during accidents to assure the capability to cool the core is maintained.

As applied to the non-LOCA transient analyses, the scram worth is the magnitude of the reactivity needed to take the initial core power to the desired shutdown conditions at zero power (for example, 420 degrees F) while, at a minimum, accounting for the moderator defect, the Doppler defect, the maximum worth stuck rod, and the minimum shutdown margin.

7.1.5.5 Reactivity Versus Time for Scram Insertion

The manner in which the scram worth is inserted into the core is derived as a bounding depiction of the reactivity insertion characteristics associated with a

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reactor trip from hot full power. An example of a normalized trip worth as a function of time, for which the control rods are fully inserted 2.278 seconds after being released by the control rod drive mechanisms, is presented in Table 7-2.

Table 7-2 Example of normalized trip worth vs. time after trip

Time After Trip (sec)	Normalized Trip Worth
0.0	0.0
0.428	0.011
0.616	0.044
0.766	0.099
0.900	0.176
1.022	0.276
1.138	0.397
1.220	0.502
1.250	0.540
1.458	0.706
1.952	0.893
2.278	1.0

7.1.6 Biasing of Other Parameters

This section describes the biasing of non-reactivity parameters used for the non-LOCA transient analyses. A brief discussion of each parameter is provided in the following sections.

- Initial Conditions (Section 7.1.6.1)
- Valve Characteristics (Section 7.1.6.2)
- Analytical Limits and Response Times (Section 7.1.6.3)
- Riser Holes (Section 7.1.6.4)

7.1.6.1 Initial Conditions

The initial conditions assumed for the non-LOCA transient analyses are the most adverse with respect to the acceptance criterion of interest. These conditions are normally consistent with steady state operation, allowing for calibration and instrument errors and steady state fluctuations. Recognizing that the initial conditions do not contribute equally to the severity of the event consequences, alternate approaches may be used to set these conditions. For instance, bounding values may be used for certain parameters to provide a more restrictive response for a specific acceptance criterion. Alternately, nominal conditions may be used if the event consequences are insensitive to a specific initial condition. A general description of the biasing of initial conditions, as applied to the non-LOCA transient analyses for an NPM, is provided below.

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Initial Core Power

The initial core power is biased high by an amount equal to the heat balance uncertainty. As an example, consider full power operation with a heat balance uncertainty of ±2 percent RTP. For this situation, the indicated core power is less than the actual core power by 2 percent RTP, i.e., the actual core power is 102 percent RTP while the indicated power is 100 percent RTP.

Initial Reactor Coolant System Average Temperature

At operating conditions the initial RCS average temperature is biased to either end of the range centered at the nominal value after consideration for any control system deadband and system/sensor measurement uncertainty.

As an example, consider a control system deadband and system/sensor measurement uncertainty of ± 10 degrees F for RCS average temperature. For this situation, the initial RCS average temperature is set to either -10 degrees F or +10 degrees F relative to the nominal temperature for the core power level of interest.

Initial Pressurizer Pressure

The initial pressurizer pressure is biased to either end of the range centered at the nominal value after consideration for any control system deadband and system/sensor measurement uncertainty.

As an example, consider a control system deadband and system/sensor measurement uncertainty of ± 70 psi for pressurizer pressure. For this situation, the initial pressurizer pressure is set to either -70 psi or +70 psi relative to the nominal pressure for the core power level of interest.

Initial Pressurizer Level

The initial pressurizer level is biased to either end of the range centered at the nominal value after consideration for any control system deadband and system/sensor measurement uncertainty.

As an example, consider a control system deadband and system/sensor measurement uncertainty of ± 8 percent for pressurizer level. For this situation, the initial pressurizer level is set to either -8 percent or +8 percent relative to the nominal level for the core power level of interest.

Initial Containment Pressure

The initial containment pressure is biased to either end of the range expected for normal operation.

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As an example, consider a normal operational range of 0.037 psia to 2.0 psia for containment pressure. For this situation, the initial containment pressure is set to either 0.037 psia or 2.0 psia.

Initial Steam Generator Pressure

The initial SG pressure is biased to either end of the range centered at the nominal value after consideration for any control system deadband and system/sensor measurement uncertainty.

As an example, consider a control system deadband and system/sensor measurement uncertainty of ± 35 psi for SG pressure. For this situation, the initial SG pressure is set to either -35 psi or +35 psi relative to the nominal pressure for the core power level of interest.

Initial Feedwater Temperature

The initial feedwater temperature is biased to either end of the range centered at the nominal value after consideration for any control system deadband and system/sensor measurement uncertainty.

As an example, consider a control system deadband and system/sensor measurement uncertainty of ± 10 degrees F for feedwater temperature. For this situation, the initial feedwater temperature is set to either -10 degrees F or +10 degrees F relative to the nominal temperature for the core power level of interest.

Initial Reactor Coolant System Flow Rate

The initial RCS flow rate is biased to either end of the range expected for normal operation.

As an example, consider a normal operational range of 535 kg/s to 690 kg/s for RCS flow rate at 100 percent RTP. For this situation, the initial RCS flow rate is set to either 535 kg/s or 690 kg/s.

Initial Volume Weighted Core Average Fuel Temperature

The initial volume weighted core average fuel temperature is biased to either end of the range expected assuming limiting power histories, power shapes, and core burnups.

As an example, consider a BOC range of 960 degrees F to 1065 degrees F for volume weighted core average fuel temperature. For this situation, the initial fuel temperature is set to either 960 degrees F or 1065 degrees F.

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Initial Reactor Pool Temperature

The initial reactor pool temperature is biased to either end of the range expected for normal operation.

As an example, consider a normal operational range of 40 degrees F to 200 degrees F for reactor pool temperature. For this situation, the initial reactor pool temperature is set to either 40 degrees F or 200 degrees F.

7.1.6.2 Valve Characteristics

The valve characteristics assumed for the non-LOCA transient analyses may differ depending on the type of valve. These differences are based on providing the most conservative characteristics for the acceptance criterion of interest.

A general description of the biasing of valve characteristics, as applied to the non-LOCA transient analyses for an NPM, is provided below.

Pressure Relief Valves

The characteristics applied to the pressure relief valves for the non-LOCA transient analyses include: 1) the nominal lift setpoint; 2) the lift tolerance; 3) the set pressure drift; 4) the accumulation; 5) the blowdown; 6) the blowdown tolerance; and, 7) the stroke time. In general, the characteristics are applied to delay opening a specific valve for as long as possible and to minimize the time the valve is open.

Isolation Valves

As an example, the following characteristics are applied to the isolation valves for the non-LOCA transient analyses. In general, the characteristics are applied to delay closing a specific valve for as long as possible.

- primary MSIV closing stroke time = 5 seconds
- secondary MSIV closing stroke time = 30 seconds
- FWIV closing stroke time = 5 seconds
- feedwater regulating valve closing stroke time = 30 seconds
- CVCS containment isolation valves closing stroke time = 5 seconds

Other Valves

As an example, the following characteristics are applied to the DHRS valves for the non-LOCA transient analyses. In general, the characteristics are applied to delay opening for as long as possible.

- DHRS open stroke time = 30 seconds

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As an example, the following characteristics are applied to the feedwater check valves for the non-LOCA transient analyses. In general, the characteristics are applied to provide the most adverse characteristics for the acceptance criterion of interest.

- feedwater safety-related check valve closing stroke time = 1.0 second when reverse flow is more limiting; otherwise, a high reverse loss coefficient is employed
- feedwater nonsafety-related check valve closing stroke time = 1.0 second when used to stop flow following a single failure of the feedwater safety-related check valve

As an example, the following characteristics are applied to the turbine valves for the non-LOCA transient analyses. In general, the characteristics are applied to close as fast as possible.

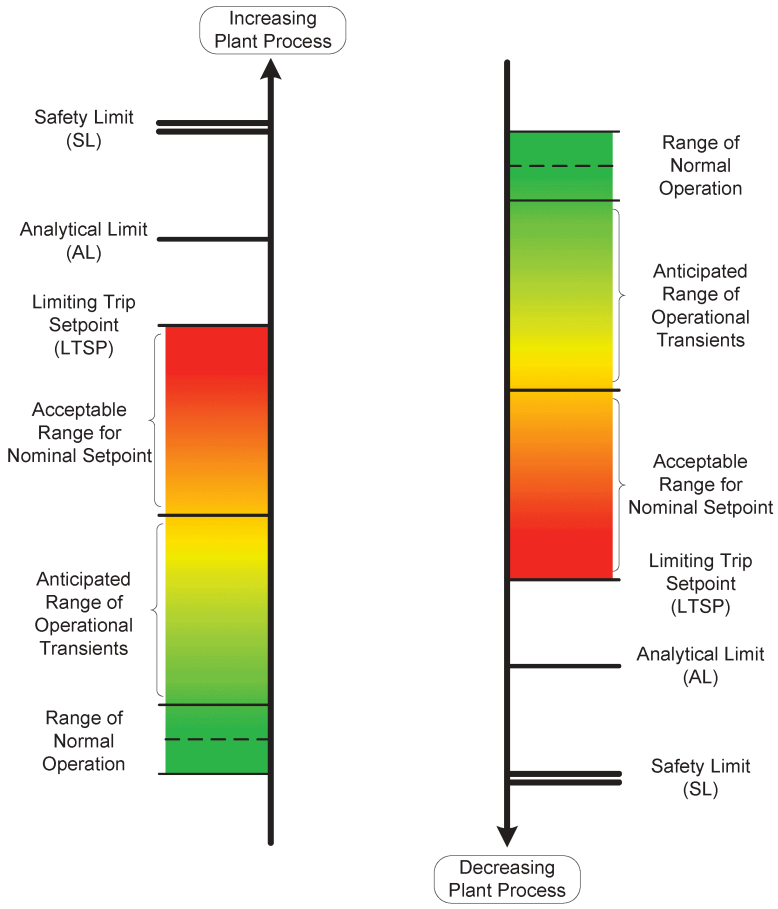
- turbine stop valve closing stroke time = 0.1 second
- turbine control valve closing stroke time = 0.15 second

7.1.6.3 Analytical Limits and Response Times

The analytical limits are the limits of measured or calculated variables (such as pressures, temperatures, and levels) established by the safety analysis to assure that a safety limit is not exceeded. In the non-LOCA transient analysis, when an analytical limit is calculated to be reached, a reactor trip or engineered safeguards actuation is signaled, and then an appropriate response time is accounted for. The analytical limits and response times assumed for the non-LOCA transient analyses are selected to provide sufficient operating margin to prevent spurious actuation, yet still provide adequate protection for the acceptance criterion of interest. In general, the response times are applied to delay action while bounding the expected instrumentation delays. As an example, a diagram of the different types of limits and setpoints created to account for margin, allowances, and uncertainties is provided in Figure 7-2.

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Figure 7-2 Example of setpoint relationships



The RTS response time is defined as the elapsed time from when the process reaches the analytical limit until the control rods are free to fall. Thus, the response time includes the control rod drive mechanism delatch time.

The ESFAS response time definition is similar, but does not include the control rod drive mechanism delatch time. Specifically, the ESFAS response time is defined as the elapsed time from when the process reaches the analytical limit until the actuation signal is received at the component (e.g. valve solenoid).

Examples of the analytical limits and actuation delays are summarized in Table 7-3. The values in Table 7-3 are example values used to aid the subsequent methodology discussion. Application of the methodology to a specific NPM design requires the use of analytical limits and actuation delays associated with that NPM design.

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Table 7-3 Examples of analytical limits and actuation delays (reactor trip system and engineered safety features actuation system)

Signal	Analytical Limit	Actuation Delay
High power	25% RTP (power < 15% RTP)	2.0 seconds
	120% RTP (power ≥ 15% RTP)	2.0 seconds
High count rate	5.0E+5 counts/second	2.0 seconds
Startup rate	3 decades/minute	31 seconds (source range)
	3 decades/minute	2 seconds (intermediate range)
High power rate	±15% RTP/minute	2.0 seconds
High RCS riser temperature	610°F	8.0 seconds
High containment pressure	9.5 psia	2.0 seconds
High pressurizer pressure	2000 psia	2.0 seconds
High pressurizer level	80%	3.0 seconds
Low pressurizer pressure	1720 psia	2.0 seconds
Low low pressurizer pressure	1600 psia	2.0 seconds
Low pressurizer level	35%	3.0 seconds
Low low pressurizer level	20%	3.0 seconds
Low steam pressure	300 psia	2.0 seconds
Low low steam pressure	100 psia	2.0 seconds
High steam pressure	800 psia	2.0 seconds
High steam superheat	150°F	8.0 seconds
Low steam superheat	0°F	8.0 seconds
Low RCS flow	1.7 ft^3/second	6.0 seconds
Low low RCS flow	0.0 ft^3/second	6.0 seconds
Low RCS level	350-390 inches	3.0 seconds
High CNV water level	220-260 inches	3.0 seconds
Low AC voltage	0 VAC	60.0 seconds

7.1.6.4 Riser Holes

Flow paths in the lower riser and upper riser are modeled based on their nominal dimensions and form losses per the NPM design. Tolerances specified on riser hole dimensions and elevations in the design ensure the uncertainty in riser hole area is small. Riser holes have minimal impact on the short-term non-LOCA transient progression. Sensitivity studies {{

}}$^{2(a),(c)}$ were performed. The sensitivity study results confirmed that {{ }}$^{2(a),(c)}$ the impact on figures of merit relevant to short-term non-LOCA was negligible.

7.1.7 Credit for Nonsafety-related Components or Operator Actions

There are three occasions where nonsafety-related equipment is credited for event mitigation by the non-LOCA transient analyses. Listed below is the equipment associated with these occurrences.

1. The nonsafety-related secondary MSIV serves as the backup isolation device to the safety-related primary MSIV for isolation of the MSS piping penetrating the

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containment. Similarly, the nonsafety-related primary main steam isolation bypass valve (MSIBV) serves as the backup isolation device to the safety-related primary MSIBV. (Section 7.1.4)

2. The nonsafety-related feedwater regulating valve serves as the backup isolation device to the safety-related FWIV for isolation of the feedwater system piping penetrating the containment. (Section 7.1.4)
3. The nonsafety-related feedwater check valve serves as the backup isolation device to the safety-related feedwater check valve for isolation of the DHRS when reverse flow is experienced during a break in the feedwater system piping. (Section 7.1.4)

For an NPM, there are no occasions where operator action is credited for event mitigation by the non-LOCA transient analyses.

7.2 Event Specific Methodology

The non-LOCA event simulations are performed using conservative methodologies. Pertinent event-specific methodologies, as well as inputs and results for a representative NPM for non-LOCA event simulations are presented herein, and compared with the regulatory acceptance criteria listed in Table 7-4. Section 4.1 contains additional discussion of Chapter 15 design basis events and acceptance criteria.

All acceptance criteria are considered for each event, and the criteria with the potential for being challenged are identified and evaluated in further detail (i.e., overcooling events do not challenge the acceptance criterion for primary side pressure, but may challenge the CHFR acceptance criteria). An event-specific parameter that is relevant to the acceptance criterion may be described as “challenging” in the event-specific summary, however, it is recognized that the parameter may not present the worst challenge for any event.

Table 7-4 Regulatory acceptance criteria

Description	AOO Criteria	IE Criteria	Accident Criteria
RCS pressure	≤ 110% of design	≤ 120% of design	≤ 120% of design
SG pressure	≤ 110% of design	≤ 120% of design	≤ 120% of design
CHFR(1)	> Limit	Note (3)	Note (3)
Maximum fuel centerline temperature(1)	≤ Limit	Note (3)	Note (3)
Containment integrity(2)	< Limits	< Limits	< Limits
Escalation of an AOO to an accident (AOO) or consequential loss of system functionality (IE or accident)?	No	No	No

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Table 7-4 Regulatory acceptance criteria (Continued)

Description	AOO Criteria	IE Criteria	Accident Criteria
Dose(1)	Normal operations	< Limit	< Limit

1. This criterion is confirmed as part of a separate follow-on analysis.
2. Containment integrity is evaluated by a separate analysis methodology.
3. If the minimum CHFR is less than or equal to the CHFR analysis limit, or if the maximum fuel centerline temperature exceeds the melting temperature, the fuel rod is assumed to be failed. If fuel failure is calculated, this is accounted for in the downstream radiological dose analysis.

For each event analyzed following the non-LOCA evaluation model, a description of the event progression, significant inputs and results, and representative results of sensitivity studies are presented in the following sections. Sensitivity studies are performed to identify plant conditions that result in bounding transient analyses. Studies that identify acceptance criteria challenges or bounding transient forcing functions are discussed as well. Other sensitivity studies that determine bounding inputs for RCS flow, fuel parameters, etc. may not necessarily be discussed for every event. The selection of parameters to be studied is focused on the acceptance criteria challenged by the event. For events where a parameter has minimal or no impact on the consequences, the bias or conservatism identified for the parameter is that typically applied; use of an alternate assumption is also acceptable. The sensitivity study results provided herein are for a representative NPM (NPM-160). Sensitivity studies performed as part of the implementation of the methodology are identified in the following sections. In addition, Section 7.2.1.3 provides example sensitivity studies for one specific event that are then used to generally define and demonstrate the methodology for the events in Section 7.2.

Initial RCS flow is typically biased to the low condition in all event simulations because this is bounding for MCHFR. {{

}}2(a),(c)

Steam generator tube plugging is considered for each event in the "Initial conditions, biases, and conservatisms" tables. The term, "Biased to the low condition" indicates no tube plugging is assumed. “Biased to the high condition” indicates {{ }}2(a),(c) steam generator tube plugging.

In an NPM design, the rod control system is set to “insert only” mode at full power to prevent automatic withdrawal of the control bank at full power. Although this plant feature exists, the feature is not credited during events where control rod withdrawal results in a bounding result.

An NPM utilizes a nonsafety-related turbine bypass system sized to handle full steam flow rate at 100 percent RTP. As such, the turbine bypass valves open following a turbine trip to control the RCS temperature without steam relief to the atmosphere. Since the turbine bypass system enhances heat removal by the secondary system, these actions are not typically credited for the non-LOCA transient analyses.

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}}2(a),(c)

Separate analyses for subchannel CHF, fuel centerline temperature, and containment pressure calculations are performed using the appropriate licensed NuScale methodologies. Extended cooldown via the DHRS is considered as part of the system design.

7.2.1 Decrease in Feedwater Temperature

The methodology used to simulate a postulated decrease in feedwater temperature for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.1.1 General Event Description and Analysis Methodology

The event is initiated by a feedwater system malfunction that causes a decrease in feedwater temperature, resulting in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity due to moderator feedback, which raises reactor power. Decreasing average RCS temperature also prompts the control rod controller to withdraw the regulating bank from the core if automatic control is enabled. Rising reactor power typically causes RTS actuation on a high power or high power rate signal, and DHRS actuation and isolation of the secondary system occur post-reactor trip on other MPS signals.

Closure of the FWIVs isolates the SGs from the colder feedwater, ending the overcooling event. Core decay heat drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient terminates with the NPM in a safe, stable condition. Table 7-5 lists the relevant acceptance criteria, SAF, and LOP scenarios.

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The limiting MCHFR typically occurs when the event is initiated from full power conditions. {{

}}2(a),(c)

The increase in downcomer density from the cooldown causes a decrease in neutron flux at the excore neutron detectors associated with power-related reactor trip signals in the NPM. Therefore, the power used to generate power-related reactor trips accounts for this decalibration effect if the power-related reactor trips are credited in the analysis. For the high power reactor trip, the high power analytical limit is effectively increased to account for the decalibration effect, for example from 120 percent (Table 7-3) to 125 percent RTP. For the high power rate reactor trip, accounting for the decalibration effect delays actuation of the trip due to the time for the excore detectors to register the power increase. {{

}}2(a),(c) Neglecting reactor trip on a power-related reactor trip signal is a conservative alternative to accounting for the decalibration effect on that power-related reactor trip signal in the analysis.

To maximize the overall feedwater temperature change, the feedwater temperature transient starts at the initial (full power) feedwater temperature biased to the high condition, and terminates at the coldest temperature in the secondary, which is saturation temperature at condenser vacuum conditions. A sensitivity study on feedwater temperature cooldown rate is performed to identify the rate that results in limiting conditions.

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}}2(a),(c)

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Additional sensitivity studies are performed on other parameters, as identified in Table 7-7, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation, covered by a separate methodology.

Table 7-5 Acceptance criteria, single active failure, loss of power scenarios - decrease in feedwater temperature

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	CHF is the challenged acceptance criterion for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
No single failure	The challenging cases typically occur when all equipment operates as designed.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event.

7.2.1.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-6.

Table 7-6 Acceptance criteria - decrease in feedwater temperature

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of the event, sensitivities that maximize primary pressure are not analyzed. Peak primary pressure resulting from a decrease in feedwater temperature is bounded by other AOO events.
Secondary pressure	Pressure in the portion of the secondary system between the FWIVs and MSIVs increases rapidly post-DHRS actuation. However, due to the depressurizing nature of this cooldown event, sensitivities that maximize peak secondary pressure are not analyzed. Peak secondary pressure resulting from a decrease in feedwater temperature is bounded by other AOO events.
CHFR	Due to the increase in reactor power and subsequent reduction of MCHFR, this acceptance criterion is challenged for the decrease in feedwater temperature event. Consequently, sensitivity cases are performed to support the follow-on MCHFR evaluation.

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Table 7-6 Acceptance criteria - decrease in feedwater temperature (Continued)

Acceptance Criteria	Discussion
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.1.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-7 are considered in identifying a bounding transient simulation for MCHFR.

Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Nominal.	
Moderator Temperature Coefficient (MTC)	Biased to EOC conditions.	}}2(a),(c)

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Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature (Continued)

Parameter	Bias / Conservatism	Basis {{
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Minimum feedwater temperature	Biased to the low condition.	
Feedwater temperature cooldown rate	Spectrum.	
RCS Temperature Control Automatic rod control Boron concentration	 Varied. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied. Nominal.	}}2(a),(c)

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Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature (Continued)

Parameter	Bias / Conservatism	Basis
PZR Level Control Charging Letdown	Not credited. Disabled.	{{ }}2(a),(c)
Steam Pressure Control Turbine throttle valves Turbine bypass valves	Varied Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	
CNV Pressure Control CNV evacuation system	Enabled.	

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

For example, sensitivity studies are performed to consider the effects of fuel-related parameters (initial fuel temperature, time in life). Representative results for these studies are presented in Table 7-8 and Table 7-9. {{ }}2(a),(c)

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{{

}}2(a),(c)

The next example sensitivity study, Table 7-10, identifies the feedwater temperature decrease rate that minimizes MCHFR. {{

}}2(a),(c)

The last example sensitivity study for this event evaluates the effects of steam system boundary condition type on MCHFR and assesses the effects of a single failure of an MSIV to isolate. Representative results are presented in Table 7-11. The boundary condition in Table 7-11 is a constant steam flow, which allows the steam pressure to move in response to the transient. The previous study in Table 7-10 models a constant steam pressure boundary condition that allows flow to vary while maintaining pressure. {{

}}2(a),(c)

Table 7-8 Representative fuel exposure study

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Table 7-9 Representative fuel temperature study

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}}2(a),(c)

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Table 7-9 Representative fuel temperature study (Continued)

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Table 7-10 Representative feedwater temperature transient study

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Table 7-11 Representative boundary condition type / single active failure studies

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}}2(a),(c)

The results of these representative sensitivity studies are provided as examples for this event. However, the sensitivity studies also demonstrate the methodology used generally for the events in Section 7.2.

- Table 7-8 demonstrates how to “vary” a parameter that has a range of possible initial values. {{

}}2(a),(c)

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}}2(a),(c)

- Table 7-9 also demonstrates how to “vary” a parameter that has a range of possible initial values. {{

}}2(a),(c)

- Table 7-10 demonstrates how to consider a “spectrum” of values for a parameter. {{

}}2(a),(c)

- Table 7-11 demonstrates how to “vary” a control system parameter that has two basic states {{

}}2(a),(c)

- Based on the results of sensitivity studies for individual parameters, engineering judgment is used to determine whether a figure of merit is

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sensitive to a specific parameter. The parameters that show sensitivity in the figure of merit are then combined to identify limiting cases for the figure of merit. {{

}}2(a),(c) No specific quantification is provided to define the threshold for sensitive vs. insensitive. Instead, a two-tiered approach is used based on the available margin for the figure of merit as described in Section 4.2. When the margin to an acceptance criterion is less than the thresholds defined in Section 4.2, a reduction in margin due to change in a parameter should not be considered negligible.

7.2.2 Increase in Feedwater Flow

The methodology used to simulate a postulated increase in feedwater flow for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.2.1 General Event Description

A feedwater system malfunction that causes an increase in feedwater flow results in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity due to moderator feedback, which raises reactor power. Decreasing average RCS temperature also prompts the control rod controller to withdraw the regulating bank from the core if automatic control is enabled. Rising reactor power can cause RTS actuation on the high power or high pressurizer pressure signal. The feedwater flow increase can also cause RTS actuation on low steam line superheat or high steam line pressure. DHRS also actuates on one of the several signals that cause RTS.

Closure of the FWIVs following DHRS actuation isolates the SGs from the feedwater source, ending the overcooling event. Core decay heat drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient calculation is terminated with the NPM in a safe, stable condition. Table 7-12 lists the relevant acceptance criteria, SAF, and LOP scenarios.

A rapid (step) increase in feedwater flow is simulated. The limiting MCHFR typically occurs when the event is initiated from full power conditions, {{

}}2(a),(c)

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}}2(a),(c)

The increase in downcomer density from the cooldown causes a decrease in neutron flux at the excore neutron detectors associated with power-related reactor trip signals in the NPM. Therefore, the power used to generate power-related reactor trips accounts for this decalibration effect if the power-related reactor trips are credited in the analysis. For the high power reactor trip, the high power analytical limit is effectively increased to account for the decalibration effect, for example from 120 percent (Table 7-3) to 125 percent RTP. For the high power rate reactor trip, accounting for the decalibration effect delays actuation of the trip due to the time for the excore detectors to register the power increase. {{

}}2(a),(c) Neglecting reactor trip on a power-related reactor trip signal is a conservative alternative to accounting for the decalibration effect on that power-related reactor trip signal in the analysis.

{{

}}2(a),(c). Additional sensitivity studies are performed on other parameters, as identified in Table 7-14, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

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Table 7-12 Acceptance criteria, single active failure, loss of power scenarios - increase in feedwater flow

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	CHF is challenged for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
SG overfill	An increase in feedwater flow could cause the SG to overfill.
No single failure	Except when considering SG overfill, the challenging cases typically occur when all equipment operates as designed.
Failure of one FWIV to close	FWIV single failure is typically limiting for SG overfill.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event and the potential for SG overfill.

7.2.2.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-13.

Table 7-13 Acceptance criteria - increase in feedwater flow

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of the event, sensitivities that maximize primary pressure are not analyzed. Peak primary pressure resulting from an increase in feedwater flow is bounded by other AOO events.
Secondary pressure	A rapid initial secondary pressure increase occurs prior to RTS actuation. Pressure in the secondary side continues to increase to the peak following DHRS actuation. This second pressure increase is expected behavior following DHRS actuation and is not a direct consequence of the increase in feedwater flow event itself.

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Table 7-13 Acceptance criteria - increase in feedwater flow (Continued)

Acceptance Criteria	Discussion
CHFR	Due to the increase in reactor power and subsequent reduction of MCHFR, this acceptance criterion is challenged for the increase in feedwater flow event. Consequently, sensitivity cases are performed to support the follow-on MCHFR evaluation.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.
SG overfill	An increase in feedwater flow can lead to overfilling the SG. If the SG is overfilled, DHRS performance is negatively impacted. This criterion is satisfied by demonstrating the SG does not overfill and DHRS cooling is adequate, consistent with the discussion in the acceptance criterion for escalation of an AOO to an accident.

7.2.2.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-14 are considered in identifying a bounding transient simulation for criteria other than SG overfill.

Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	

}}2(a),(c)

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Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial PZR level	Varied	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Nominal.	
MTC	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the low condition.	
Initial SG pressure(1)	Varied	
SG heat transfer	Varied.	
RSV lift setpoint	Nominal	
SG tube plugging	Varied.	

}}2(a),(c)

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Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
RCS Temperature Control		
Automatic rod control	Enabled.	
Boron concentration	Not credited.	
PZR Pressure Control		
PZR spray	Varied.	
PZR heaters	Nominal.	
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control		
Turbine throttle valves	Enabled to control to constant steam pressure.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control		

{{

}}2(a),(c)

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Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
Feedwater pump speed	Spectrum.	
CNV Pressure Control		
CNV evacuation system	Enabled.	

{{ }}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

For example, two sensitivity studies are performed to identify the most challenging secondary biases for the feedwater flow transient for MCHFR. {{

}}2(a),(c) Representative results for these studies are presented in Table 7-16. These results demonstrate an example of {{ }}2(a),(c) as described in the general methodology discussion in Section 7.2.1.3.

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The increase in feedwater flow also creates the potential for overfilling the SG. The assumptions in Table 7-14 related to identifying the limiting MCHFR do not necessarily result in the maximum SG level. The biases and conservatisms indicated in Table 7-15, and the single failure assumption in Table 7-12, are considered in identifying a bounding transient simulation for SG overfill. Sensitivity studies are performed to identify the case(s) with a potentially limiting SG level.

Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow

Parameter	Bias / Conservatism	Basis
Initial reactor power	Varied.	{{ }}2(a),(c)
Initial RCS average temperature	Varied.	{{ }}2(a),(c)
Initial RCS flow rate	Varied.	{{ }}2(a),(c)
Initial PZR pressure	Varied.	{{ }}2(a),(c)
Initial PZR level	Varied.	{{ }}2(a),(c)
Initial feedwater temperature	Varied.	{{ }}2(a),(c)
Initial fuel temperature	Biased to the high condition.	{{ }}2(a),(c)
MTC	Varied.	{{ }}2(a),(c)
Kinetics	Varied.	{{ }}2(a),(c)
Decay heat	Biased to the high condition.	{{ }}2(a),(c)
Initial SG pressure(1)	Biased to the low condition.	{{ }}2(a),(c)

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Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
SG heat transfer	Varied.	{{ }}2(a),(c)
RSV lift setpoint	Nominal.	{{ }}2(a),(c)
SG tube plugging	Varied.	{{ }}2(a),(c)
RCS Temperature Control Automatic rod control Boron concentration	 Varied. Not credited.	{{ }}2(a),(c)
PZR Pressure Control PZR spray PZR heaters	 Disabled. Nominal.	{{ }}2(a),(c)
PZR Level Control Charging Letdown	 Not credited. Disabled.	{{ }}2(a),(c)

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Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
Steam Pressure Control Turbine throttle valves	Varied.	{{
Turbine bypass valves	Varied.	}}2(a),(c)
Feedwater and Turbine Load Control Feedwater pump speed	N/A.	{{ }}2(a),(c)
CNV Pressure Control CNV evacuation system	Enabled.	{{ }}2(a),(c)

1. {{ }}2(a),(c)

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Table 7-16 Representative increase in feedwater flow study - high and low SG performance with maximum power and minimum RCS flow

{{

}}2(a),(c)

7.2.3 Increase in Steam Flow

The methodology used to simulate a postulated increase in steam flow for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.3.1 General Event Description

In an NPM design, two main steam lines exit the CNV, and combine to form a common main steam line that connects to the high pressure turbine. The turbine bypass and main steam safety valves are located in the common main steam line, which is downstream of the MSIVs. A spurious opening of the turbine bypass valve or a main steam safety valve would cause an increase in steam flow, which results in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity due to moderator feedback, which raises reactor power. Decreasing average RCS temperature also prompts the control rod controller to withdraw the regulating bank from the core if automatic control is enabled (rod withdrawal at 100 percent power is inhibited, however, it is conservatively allowed in the analysis). Rising reactor power typically causes RTS actuation on a high power or high power rate signal. Additionally, a large increase in steam flow would rapidly depressurize the secondary system and could cause

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RTS and isolation of the secondary system on the low steam pressure or low steam superheat signal.

Isolation of the secondary system, if it has not already occurred, and actuation of DHRS occur post-trip on other MPS signals. Closure of the FWIVs and MSIVs isolates the steam generators, which ends the overcooling event. Core decay heat drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient calculation is terminated with the NPM in a safe, stable condition. Table 7-17 lists the relevant acceptance criteria, SAF, and LOP scenarios.

The limiting MCHFR typically occurs when the event is initiated from full power conditions, and the increase in steam flow is such that the immediate RTS on low steam pressure is avoided and RTS is actuated sometime later (after the minimum CHFR conditions develop).

The increase in downcomer density from the cooldown causes a decrease in neutron flux at the excore neutron detectors associated with power-related reactor trip signals in the NPM. Therefore, the power used to generate power-related reactor trips accounts for this decalibration effect if the power-related reactor trips are credited in the analysis. For the high power reactor trip, the high power analytical limit is effectively increased to account for the decalibration effect, for example from 120 percent (Table 7-3) to 125 percent RTP. For the high power rate reactor trip, accounting for the decalibration effect delays actuation of the trip due to the time for the excore detectors to register the power increase. {{

}}$^{2(a),(c)}$ Neglecting reactor trip on a power-related reactor trip signal is a conservative alternative to accounting for the decalibration effect on that power-related reactor trip signal in the analysis.

The increase in steam flow event starts at the initial (full power) steam flow. Sensitivity studies are performed on the degree of steam flow increase, {{ }}$^{2(a),(c)}$

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{{ }}[2(a),(c)] to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

Table 7-17 Acceptance criteria, single active failure, loss of power scenarios - increase in steam flow

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	CHF is challenged for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
No single failure	Note that a single active failure of a FWIV to close would occur after RTS and DHRS actuation, subsequent to when the MCHFR occurs. Consequently, the MCHFR occurs before the single active failure of an FWIV to close could affect the transient. Otherwise, the challenging cases typically occur when all equipment operates as designed.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event.

7.2.3.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-18.

Table 7-18 Acceptance criteria - increase in steam flow

Acceptance Criteria	Discussion
Primary pressure	Primary pressure initially drops as inventory shrinks due to increased heat removal. As reactor power increases and as the PZR heaters respond, an increase (typically less than 100 psi) in primary pressure is observed.

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Table 7-18 Acceptance criteria - increase in steam flow (Continued)

Acceptance Criteria	Discussion
Secondary pressure	Peak secondary pressure does not change significantly during the initial phase of the transient due to the relatively small increase in steam flow rate analyzed in the challenging case. It is only after DHRS actuation and closure of the FWIVs and MSIVs that secondary pressure begins to increase rapidly. Steam generator pressure increase resulting from MS isolation is expected and is not a direct consequence of the increase in steam flow event itself.
CHFR	Due to the increase in reactor power and subsequent reduction of MCHFR, this acceptance criterion is challenged for the increase in steam flow event. Consequently, sensitivity cases are performed to support the follow-on MCHFR evaluation.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.3.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-19 are considered in identifying a bounding transient simulation for MCHFR.

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Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Biased to the high condition.	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Steam flow increase	Spectrum.	}}2(a),(c)

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Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow (Continued)

Parameter	Bias / Conservatism	Basis
RCS Temperature Control		{{
Automatic rod control	Varied.	
Boron concentration	Not credited.	
PZR Pressure Control		
PZR spray	Varied.	
PZR heaters	Nominal.	
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control		
Turbine throttle valves	N/A.	
Turbine bypass valves	N/A.	}}2(a),(c)

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Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow (Continued)

Parameter	Bias / Conservatism	Basis
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	{{ }}2(a),(c)
CNV Pressure Control CNV evacuation system	Enabled.	

1. {{ }}2(a),(c)
2. {{

}}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

For example, two sensitivity studies are performed to identify the most challenging increase in steam flow for MCHFR. {{

}}2(a),(c) Representative results for these studies are presented in Table 7-20 and Table 7-21. These results provide an example of varying a parameter {{ }}2(a),(c) as described in the general methodology discussion in Section 7.2.1.3.

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Table 7-20 Representative steam flow study - nominal steam generator heat transfer

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}}2(a),(c)

Table 7-21 Representative steam flow study - steam generator heat transfer biased low

{{

}}2(a),(c)

7.2.4 Steam System Piping Failure Inside or Outside of Containment

The methodology used to simulate a postulated steam system piping failure for an NPM, and an evaluation of the acceptance criteria listed in Table 7-4, are presented below. Since both split breaks (relatively higher event frequency) and double-ended guillotine breaks (relatively lower event frequency) are analyzed, the more restrictive AOO criteria for system pressures, critical heat flux ratio, and fuel centerline melt applicable to breaks with higher event frequency are used in the evaluation. Radiological dose consequences are assessed as part of the downstream accident radiological dose analysis, documented in a separate report, and compared against the appropriate acceptance criteria.

7.2.4.1 General Event Description

The steam line break event ranges from small breaks to double ended ruptures of a main steam line causing an increase in steam flow and an over cooling of the RCS. This event can occur inside or outside the containment vessel (CNV). A

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break inside CNV causes a rapid pressurization of the CNV resulting in a reactor trip and CNV isolation with a DHRS actuation. This break location is non-limiting for pressure and MCHFR but potentially challenging to the DHRS as one loop is disabled with the break inside the CNV. A steam line break outside of the CNV causes an increase in steam flow event that causes either a low SG pressure trip or a high core power trip due to the reactor power response from the decreased RCS temperature. The break flow is stopped by the MSIVs closing and depressurization of the steam system piping. Smaller breaks cause a slower loss of secondary pressure due to the increased steam demand that could cause a high core power trip. These smaller breaks can result in a significant delay in detection time, making the small break cases potentially challenging for MCHFR. Reactor trip and transition to stable DHRS flow eventually terminate the transients and bring the NPM to a safe, stable condition.

The increase in downcomer density from the cooldown causes a decrease in neutron flux at the excore neutron detectors associated with power-related reactor trip signals in the NPM. Therefore, the power used to generate power-related reactor trips accounts for this decalibration effect if the power-related reactor trips are credited in the analysis. For the high power reactor trip, the high power analytical limit is effectively increased to account for the decalibration effect, for example from 120 percent (Table 7-3) to 125 percent RTP. For the high power rate reactor trip, accounting for the decalibration effect delays actuation of the trip due to the time for the excore detectors to register the power increase. {{

}}$^{2(a),(c)}$ Neglecting reactor trip on a power-related reactor trip signal is a conservative alternative to accounting for the decalibration effect on that power-related reactor trip signal in the analysis.

{{

}}$^{2(a),(c)}$

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{{

}}2(a),(c)

The relevant acceptance criteria, SAF, and LOP scenarios are listed in Table 7-22.

Table 7-22 Acceptance criteria, single active failure, loss of power scenarios - steam line break

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	Critical heat flux is potentially challenged for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
Radiological consequences	A postulated break in the main steam line is evaluated for radiological consequences.
Failure of one MSIV to close on the train with break	MSIV single failure typically has no effect on MCHFR since DHRS actuation is not before the time when MCHFR occurs. MSIV single failure is typically limiting for mass releases for break locations outside the CNV if mass release is calculated for use in the downstream radiological analysis.
Failure of one FWIV to close on the train with the break	FWIV single failure typically has no effect on MCHFR since DHRS actuation is not before the time when MCHFR occurs. FWIV single failure is typically limiting for mass releases for break locations inside the CNV if mass release is calculated for use in the downstream radiological analysis.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event.

7.2.4.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-23.

Table 7-23 Acceptance criteria - steam line break

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of this cooldown event, primary pressure remains below the acceptance criterion for peak primary pressure.

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Table 7-23 Acceptance criteria - steam line break (Continued)

Acceptance Criteria	Discussion
Secondary pressure	Due to the depressurizing nature of this cooldown event, secondary pressure remains below the acceptance criterion for peak secondary pressure.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis. Sensitivity cases are performed to support the follow-on MCHFR evaluation.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation to a more serious accident or consequential loss of functionality	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.4.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-24, and a spectrum of break locations and sizes, are considered in identifying a bounding transient simulation for MCHFR and mass release.

In the radiological analysis, primary-to-secondary leakage is assumed, which allows primary coolant to be released with the SG break flow. The transient SG mass release can be calculated for use as an input to the downstream radiological analysis. Alternatively, bounding assumptions for primary coolant release can be used in the radiological analysis to eliminate the need for calculating SG mass release. If such bounding assumptions are used, transient analysis to maximize SG mass release is not required.

Table 7-24 Initial conditions, biases, and conservatisms - steam line break

Parameter	Bias / Conservatism	Basis(1) {{
Initial reactor power	Biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	}}2(a),(c)

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Table 7-24 Initial conditions, biases, and conservatisms - steam line break (Continued)

Parameter	Bias / Conservatism	Basis(1) {{
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased low for BOC conditions and high for EOC conditions.	
MTC	Both EOC and BOC conditions.	
Kinetics	Both EOC and BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(2)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Enabled. (MCHFR) Not credited.	
PZR Pressure Control PZR spray	 Varied.	}}2(a),(c)

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Table 7-24 Initial conditions, biases, and conservatisms - steam line break (Continued)

Parameter	Bias / Conservatism	Basis[1]
PZR heaters	Nominal.	{{
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control		
Turbine throttle valves	Varied.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control		
feedwater pump speed	Disabled.	
CNV Pressure Control		
CNV evacuation system	Enabled.	}}2(a),(c)

{{

}}2(a),(c)

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Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response and challenging mass releases for this overcooling event.

Representative sensitivity study results for MCHFR are presented in Table 7-25. In the cases modeling breaks outside of containment, the MSIV on the train with the break is assumed to fail to close, resulting in the complete emptying of the affected SG. In cases modeling breaks inside of containment, the FWIV on the train with the break is assumed to fail to close. Break size is the faction (percent) of the pipe cross-sectional area. {{

}}2(a),(c) The sensitivity study demonstrates using a spectrum analysis {{

}}2(a),(c) consistent with the general methodology discussion in Section 7.2.1.3.

Table 7-25 Representative steam line break study

{{

}}2(a),(c)

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Table 7-25 Representative steam line break study (Continued)

{{

}}2(a),(c)

7.2.5 Containment Flooding / Loss of Containment Vacuum

The methodology used to simulate a postulated containment flooding / loss of containment vacuum for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below. This AOO is unique to the NPM designs.

7.2.5.1 General Event Description and Methodology

The “loss of containment vacuum” terminology refers specifically to vapor / air or minimal water leakage into the containment vessel (CNV) that does not result in water build-up inside the CNV. The containment evacuation (CE) system is used to maintain a vacuum in the CNV during normal operation. If the leakage rate exceeds the CE system’s capacity, a high containment pressure (> 9.5 psia in the example Table 7-3) signal can occur, which trips the reactor and isolate all containment penetrations.

The “containment flooding” terminology refers to liquid build-up in the CNV that can potentially cause loss of CNV vacuum and result in containment isolation and reactor trip.

The potential CNV flooding sources considered are piping failures inside the CNV. Depending on the CNV operating pressure, some of these lines inside the CNV carry fluid at a lower temperature than the CNV saturation temperature. If one of these low temperature lines fails, flashing inside the CNV does not occur instantly, so the high containment pressure analytical limit is not immediately reached. Therefore these low temperature line failures can potentially create different challenges than failures in other lines carrying higher temperature fluid such as main steam and feedwater lines.

The CNV flooding sources due to a piping failure inside containment include: feedwater (FW), main steam (MS), chemical and volume control system (CVCS), high point vent (HPV), and reactor component cooling water (RCCW). Only a RCCW piping failure inside the CNV is considered as a flooding source for the CNV flooding / loss of CNV vacuum event. Other break locations are evaluated as separate initiating events.

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During this event, heat removal from the RCS increases due to loss of CNV vacuum or CNV flooding from RCCW. As a result, the CNV flooding/loss of CNV vacuum event does not introduce more challenging conditions for RCS pressure and secondary side pressure compared to other AOOs such as the overheating events.

Due to the overcooling effect of the event, an increase in reactor power and subsequent reduction of MCHFR may occur. A series of CNV flooding sensitivities are conducted to assess the effects on MCHFR from RCCW flow, RCCW temperature, containment pressure, primary pressure, and pool temperature.

The overcooling effect of this event typically does not result in power increases large enough to result in power-related reactor trips in the analysis. However, power-related trips, if credited in the analysis, are adjusted to account for the decalibration effect of increased downcomer density or increased containment fluid density, as applicable, in the same manner as other cooldown events.

{{

}}2(a),(c) The purpose is to show that there is little observable difference between the two cases with the change in reactor power and the additional heat loss is minimal. Any loss of CNV vacuum cases that cause high CNV pressure trip should be bounded by the results obtained from the CNV flooding cases that result in reactor trip.

The relevant acceptance criteria, single active failure, and loss of power scenarios are listed in Table 7-26.

Table 7-26 Acceptance criteria, single active failure, loss of power scenarios - containment flooding / loss of containment vacuum

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	MCHFR is potentially challenged during this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
No single failure	The challenging cases typically occur when all equipment operates as designed.
No loss of power	A loss of AC power at event initiation is typically non-limiting because reactor trip and containment isolation mitigate the event, while a loss of AC power coincident with turbine trip typically does not alter the MCHFR because the time of reactor trip is not changed.

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7.2.5.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-27.

Table 7-27 Acceptance criteria - containment flooding / loss of containment vacuum

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of the event, sensitivities that maximize primary pressure are not analyzed. Peak primary pressure resulting from CNV flooding/loss of CNV vacuum is bounded by other AOO events.
Secondary pressure	Due to the primary system depressurization of this cooldown event, secondary pressure remains below the acceptance criterion for peak secondary pressure.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.5.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-28 are considered in order to identify the bounding transient simulation for MCHFR for the CNV flooding/loss of CNV vacuum event.

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Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied	
Initial PZR level	Nominal	
Initial feedwater temperature	Nominal	
Initial fuel temperature	Nominal	
MTC	Biased to the EOC condition.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Initial containment pressure	Varied	
Initial pool temperature	Varied	
RCCW leak flow	Varied	
RCCW temperature	Varied	}}2(a),(c)

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Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum (Continued)

Parameter	Bias / Conservatism	Basis
RCS Temperature Control		{{
Automatic rod control	Enabled.	
Boron concentration	Not credited.	
PZR Pressure Control		
PZR spray	Varied.	
PZR heaters	Nominal.	
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	}}2(a),(c)

Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum (Continued)

Parameter	Bias / Conservatism	Basis
Steam Pressure Control Turbine throttle valves Turbine bypass valves	Enabled. Disabled.	{{ }}2(a),(c)
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled. (CNV flooding) N/A. (CNV vacuum loss)	

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

As an example, the initial reactor power, pressurizer pressure, RCS average coolant temperature, RCS flow rate, containment pressure, pool temperature, RCCW leak flow rate, RCCW temperature are varied for a representative NPM for the CNV flooding/loss of CNV vacuum event as indicated in Table 7-29. The sensitivity studies indicate that the CNV flooding cases are more challenging to MCHFR than loss of CNV vacuum. The sensitivity study results also indicate that for a variety of initial RCS conditions, reactor pool conditions, and condition of liquid or air ingress to containment, loss of containment vacuum or containment flooding results in a slow overcooling transient that is non-limiting with respect to MCHFR compared to other cooldown event AOOs.

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Table 7-29 Representative sensitivity studies - containment flooding / loss of containment vacuum

{{

}}2(a),(c)

7.2.6 Turbine Trip / Loss of External Load

The methodology used to simulate a postulated turbine trip/loss of external load (LOEL) for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.6.1 General Event Description

Turbine trip initiates with a turbine stop valve closure while loss of external load initiates with a turbine control valve closure. Otherwise, these transients are essentially equivalent and result in the sudden removal of the secondary side heat sink, overpressurization of the secondary system, and overheating of the RCS. Rising system pressures typically result in RTS actuation on the high PZR or steam pressure signal. Reactor trip and transition to stable DHRS flow terminates

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the transient with the NPM in a safe, stable condition. Table 7-30 lists the relevant acceptance criteria, SAF, and LOP scenarios.

The limiting pressure responses typically occur when the event is initiated from full power conditions, and the initial conditions are biased in the conservative directions. Sensitivity studies on initial primary temperature and primary/secondary pressures are performed to identify the conditions that maximize peak primary and secondary pressures.

Table 7-30 Acceptance criteria, single active failure, loss of power scenarios - turbine trip / loss of external load

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Primary and secondary pressures are challenged during this overheating event.
Failure of one FWIV to close	Typically challenging for secondary side pressurization cases (negligible effect for primary side pressurization cases, which assume loss of AC power at event initiation and therefore feedwater is lost).
Loss of AC power at transient initiation No loss of power	Typically maximizes primary pressure (feedwater is lost). Typically maximizes secondary pressure.

7.2.6.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-31.

Table 7-31 Acceptance criteria - turbine trip / loss of external load

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation and continued FW operation, in addition to the actual turbine trip or loss of external load. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.

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Table 7-31 Acceptance criteria - turbine trip / loss of external load (Continued)

Acceptance Criteria	Discussion
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.6.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-32 are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased to the high condition to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Biased to the high condition.	
Initial feedwater temperature	Nominal.	}}2(a),(c)

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Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load (Continued)

Parameter	Bias / Conservatism	Basis
Initial fuel temperature	Biased to the high condition	{{
MTC	Consistent with BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
Steam generator heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Disabled. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Disabled. Disabled.	}}2(a),(c)

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Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load (Continued)

Parameter	Bias / Conservatism	Basis {{
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

For example, representative sensitivity study results are presented in Table 7-33. The results indicate that maximum RPV pressure occurs when PZR pressure is biased to the high condition, average temperature is biased to the low condition, and a loss of power leads to a loss of the secondary side heat sink (since feedwater is lost, the failure of an FWIV to close would have no effect on the event). The results also confirm that many cases have similar maximum RPV pressures and that there is significant margin to the acceptance criterion (2310 psia for this representative NPM). With respect to secondary side SG pressure, maximum SG pressure occurs when average temperature and SG pressure are biased to the high condition, and all power sources are available - the failure of an FWIV to close increases the peak SG pressure by a very small amount. The results also confirm that many cases have similar maximum secondary side pressures and remain well below the secondary design pressure (2100 psia for this representative NPM).

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Table 7-33 Representative sensitivity studies - turbine trip / loss of external load

{{

}}2(a),(c)

7.2.7 Loss of Condenser Vacuum

The methodology used to simulate a postulated loss of condenser vacuum for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4 are presented below.

7.2.7.1 General Event Description

Loss of condenser vacuum initiates with a turbine stop valve closure. Also, a loss of condenser vacuum is postulated to lead to a loss of feedwater flow. Turbine trip and loss of feedwater result in the sudden removal of the secondary side heat sink, pressurization of the secondary system, and overheating of the RCS. Rising system pressures typically result in a rapid RTS actuation on either high PZR or steam pressure. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition.

The loss of condenser vacuum event is essentially equivalent to the turbine trip / loss of external load events discussed in Section 7.2.6. The main difference is that the loss of condenser vacuum event includes a loss of feedwater flow at event initiation. However, because the turbine trip / loss of external load events consider a loss of normal AC power at event initiation, those events also model a loss of feedwater flow at event initiation. As a result, the scenarios analyzed as part of Section 7.2.6 address the loss of condenser vacuum event. Therefore, the relevant acceptance criteria, SAF, and LOP scenarios from Table 7-30 are also applicable to the loss of condenser vacuum event.

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Table 7-34 Not Used

7.2.7.2 Acceptance Criteria

The evaluation of the most challenging case(s) relative to the acceptance criteria for the turbine trip / loss of external load events presented in Table 7-31 is applicable to the loss of condenser vacuum event.

Table 7-35 Not Used

7.2.7.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-32 for the turbine trip / loss of external load events are applicable to the loss of condenser vacuum event.

Table 7-36 Not Used

Table 7-37 Not Used

7.2.8 Main Steam Isolation Valve(s) Closure

The methodology used to simulate a postulated main steam isolation valve closure for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4 are presented below.

7.2.8.1 General Event Description

A closure of one or both main steam isolation valves results in a pressurization of the secondary system and overheating of the RCS. Rising pressures typically result in a rapid RTS actuation on the high pressurizer pressure or high steam pressure signals. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition. The relevant acceptance criteria, single active failure, and loss of power scenarios are listed in Table 7-38.

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The MSIV closure event can occur when one or both MSIVs close unexpectedly. The limiting pressure responses typically occur when the event is initiated from full power conditions, and the initial conditions are biased in the conservative directions.

Sensitivity studies on number of MSIVs closing, initial primary temperature and primary/secondary pressures are performed to identify the conditions that maximize peak primary and secondary pressures. Additional sensitivity studies are performed on other parameters, as identified in Table 7-40, to identify the case(s) with the potentially limiting peak primary and secondary pressures.

Table 7-38 Acceptance criteria, single active failure, loss of power scenarios - main steam isolation valve closure

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Primary and secondary pressures are challenged during this overheating event.
No single failure Failure of one FWIV to close	Typically challenging for primary pressure. Typically challenging for steam generator pressure.
Loss of AC Power at transient initiation No loss of power	Typically maximizes primary pressure. Typically maximizes secondary pressure.

7.2.8.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-39.

Table 7-39 Acceptance criteria - main steam isolation valve closure

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation and continued FW operation, in addition to the actual main steam isolation valve closure. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.

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Table 7-39 Acceptance criteria - main steam isolation valve closure (Continued)

Acceptance Criteria	Discussion
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.8.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-40 are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Table 7-40 Initial conditions, biases, and conservatisms - main steam isolation valve closure

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the high condition	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
Steam generator heat transfer	Nominal.	}}2(a),(c)

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Table 7-40 Initial conditions, biases, and conservatisms - main steam isolation valve closure (Continued)

Parameter	Bias / Conservatism	Basis
		{{
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Disabled. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Enabled.	}}2(a),(c)

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Table 7-40 Initial conditions, biases, and conservatisms - main steam isolation valve closure (Continued)

Parameter	Bias / Conservatism	Basis
CNV Pressure Control CNV evacuation system	Enabled.	{{ }}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

For example, representative sensitivity study results are presented in Table 7-41. The results presented consider the closure of two MSIVs, which isolates both steam generators, and no single active failure. Maximum pressures are similar for cases in which the RSV actuation limits the RPV pressurization and remain well below the acceptance criterion (2310 psia for this representative NPM). For example, the maximum calculated pressure for the peak RPV pressure study is approximately equal for two additional cases, namely, decreased steam generator heat transfer and decreased feedwater temperature. Additionally, the results of the sensitivity studies indicate that maximum RPV pressure occurs when PZR pressure is biased to the high condition, RCS flow is at the nominal value (if maximum RCS pressure is below the RSV lift pressure), and a loss of AC power leads to a loss of the secondary side heat sink (loss of AC power occurs coincident with reactor trip). With respect to secondary side SG pressure, maximum SG pressure occurs when average temperature is biased to the high condition, RCS flow is biased to the low condition, and all power sources are available. The results also confirm that many cases have similar maximum secondary side pressures and remain well below the secondary design pressure (2100 psia for this representative NPM).

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Table 7-41 Representative sensitivity studies - main steam isolation valve closure

{{

}}2(a),(c)

7.2.9 Loss of Nonemergency AC Power

The methodology used to simulate a postulated loss of nonemergency (normal) AC power for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.9.1 General Event Description

The low voltage AC electrical distribution system (ELVS) supplies AC power to plant motors, heaters, packaged equipment, and battery chargers. Loss of normal AC power to the station auxiliaries can result from electrical grid-related failures, failures in plant or switchyard equipment, or external weather events. The nonsafety-related EDNS and EDAS/EDSS may remain available via battery operation; the primary loads for these systems include the module control system (MCS) and the MPS. Loss of the EDNS and/or EDAS/EDSS batteries with the loss of normal AC power is considered as described in Section 7.1.3. A loss of AC power results in a pressurization of the secondary system and overheating of the RCS. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition. Table 7-42 lists the relevant acceptance criteria, SAF, and LOP scenarios.

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The electrical system design may vary by NPM design. Review of the electrical system is performed to determine the impact on plant equipment from the loss of power. In general, the following typically occur from the loss of power (AC or DC or both):

- turbine trip occurs
- the feedwater pumps and CVCS pumps stop
- the PZR heaters turn off
- control rods begin to drop due to loss of power to control rod drive mechanisms (CRDMs)
- the MPS actuates reactor trip, DHRS, and various system isolations within 60 seconds after event initiation (if not already actuated during that time), due to loss of AC power to the EDAS/EDSS battery chargers
- the MPS actuates ECCS within 24 hours after event initiation, due to loss of AC power to the EDAS/EDSS battery chargers

Of the various scenarios, the limiting one is typically the scenario that does not result in an immediate reactor trip or full control rod insertion at event initiation. Typically this scenario is the loss of the ELVS at event initiation, with EDNS and EDAS/EDSS available. Cases considering the loss of EDSS battery backup coincident with the initiating event are considered as described in Section 7.1.3. Consequently, the limiting pressure responses typically occur when the event is initiated from full power conditions, reactor trip is delayed until MPS actuation, and the initial conditions are biased in the conservative directions. Sensitivity studies on initial primary temperature and primary/secondary pressures are performed to identify the conditions that maximize peak primary and secondary pressures. Additional sensitivity studies are performed on other parameters, as identified in Table 7-44, to identify the case(s) with the potentially limiting peak primary and secondary pressures.

Table 7-42 Acceptance criteria, single active failure, loss of power scenarios - loss of normal AC power

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Primary and secondary pressures are challenged during this overheating event.
No single failure	The challenging cases for primary pressure typically occur when all equipment is operational. Since feedwater is lost at transient initiation, peak secondary pressures are insensitive to the single failure of an FWIV to isolate.
Loss of AC power at transient initiation	Initiating event.

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7.2.9.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-43.

Table 7-43 Acceptance criteria - loss of normal AC power

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation, in addition to the actual loss of normal AC power. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.9.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-44 are considered in identifying the bounding transient simulation for primary and secondary pressure.

Table 7-44 Initial conditions, biases, and conservatisms - loss of normal AC power

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Varied.	}}2(a),(c)

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Table 7-44 Initial conditions, biases, and conservatisms - loss of normal AC power (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial PZR pressure	Varied.	
Initial PZR level	Biased to the high condition.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the high condition	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Enabled. Disabled.	
PZR Level Control		}}2(a),(c)

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Table 7-44 Initial conditions, biases, and conservatisms - loss of normal AC power (Continued)

Parameter	Bias / Conservatism	Basis {{
Charging Letdown	Not credited. Enabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Enabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)
2. Loss of normal AC power initiating event results in a loss of system function by loss of power to the system thereby making the system control not relevant to the event.

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

For example, representative sensitivity studies are performed to identify the highest primary and secondary side pressures and presented in Table 7-45. The

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results of the sensitivity studies indicate that the lowest-setpoint RSV operates to mitigate peak primary side pressurization {{

}}[2(a),(c)] The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

Table 7-45 Representative sensitivity studies - loss of normal AC power

{{

}}[2(a),(c)]

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7.2.10 Loss of Normal Feedwater Flow

The methodology used to simulate a postulated loss of normal feedwater flow for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.10.1 General Event Description

A postulated fault results in a partial or complete loss of feedwater flow, and the water in the steam generators boils off. The loss of steam generators as a heat sink leads to a rise in the RCS temperature and pressure until the reactor typically trips due to high PZR pressure. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition.

The relevant acceptance criteria, SAF, and LOP scenarios are listed in Table 7-46.

Table 7-46 Acceptance criteria, single active failure, loss of power scenarios - loss of normal feedwater flow

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	System pressures are challenged during this overheating event.
No single failure	The challenging cases typically occur when all equipment is operational.
Loss of AC power at turbine trip	Typically maximizes system pressures.

7.2.10.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-47.

Table 7-47 Acceptance criteria - loss of normal feedwater flow

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation, in addition to the actual loss of feedwater. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.

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Table 7-47 Acceptance criteria - loss of normal feedwater flow (Continued)

Acceptance Criteria	Discussion
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.10.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-48 are considered in identifying a bounding transient simulation for primary and steam generator pressure.

Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased to the high condition.	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	}}2(a),(c)

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Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
SG heat transfer	Nominal.	{{
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Feedwater flow decrease	Spectrum.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Disabled. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves	 Enabled.	}}2(a),(c)

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Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
Turbine bypass valves	Disabled.	{{
Feedwater and Turbine Load Control feedwater pump speed	N/A.	
CNV Pressure Control CNV evacuation system	Disabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest primary and secondary pressures. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

For example, representative results of a sensitivity study for a spectrum of feedwater flow rate decreases are presented in Table 7-49. The results of the sensitivity study indicates that peak primary pressure occurs during a total loss of feedwater. Because the pressurization associated with the total loss of feedwater is sufficient to actuate the lowest-setpoint RSV, {{

}}2(a),(c) well below the acceptance criterion of 2310 for this representative NPM). With respect to maximum secondary side SG pressure, maximum SG pressure occurs during a partial loss of feedwater, as the mismatch between primary heat production and secondary heat sink causes the RCS temperature to increase, and eventually actuates the RTS on high RCS riser temperature. The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

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Table 7-49 Representative sensitivity studies - loss of normal feedwater flow

{{

}}2(a),(c)

7.2.11 Inadvertent Decay Heat Removal System Actuation

The methodology used to simulate a postulated inadvertent DHRS actuation for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below. This event is unique to the NPM designs.

7.2.11.1 General Event Description

Inadvertent actuation of the DHRS may result from either an unexpected DHRS valve actuation or a spurious DHRS actuation signal. Due to the DHRS design and configuration, several scenarios exist for consideration. The specific scenarios considered are discussed below. The sequence of events and the MPS signals that are reached first depend on the scenario. Reactor trip and transition to stable DHRS flow terminates the transients with the NPM in a safe, stable condition. The relevant acceptance criteria, SAF, and LOP scenarios are listed in Table 7-50.

Scenario 1: The unexpected opening of a single DHRS valve can occur at full power or reduced power conditions. At low power conditions, a portion of the DHRS liquid inventory drains into the feedwater line, which momentarily increases feedwater flow and causes overcooling. This overcooling event is not considered further because it is bounded by other, more limiting overcooling events (i.e., increase in feedwater flow). The most challenging conditions for this heatup scenario occur at full power with initial conditions biased in the conservative directions. Since feedwater flow tends to increase in response to the reduced steam enthalpy and turbine load, limiting the feedwater response maximizes the heatup.

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Scenario 2: An inadvertent actuation signal isolates one steam generator, and initiates one DHRS train. This scenario is typically bounded by Scenario 3 discussed below.

Scenario 3: An inadvertent actuation signal isolates both steam generators, and initiates both DHRS trains. This scenario represents a complete loss of normal heat removal from the RCS. The limiting conditions for this heatup scenario occur at full power with initial conditions biased in the conservative directions.

Scenario 4: An inadvertent isolation of one steam generator. The valves associated with one SG are closed, but the associated DHRS train is not actuated. The heatup caused by the isolation of one SG causes an increase in primary pressure that results in reactor trip and RSV opening. The actuation of DHRS occurs later in the transient after MPS signals are reached. Secondary pressure peaks after DHRS actuation.

Scenario 5: An inadvertent isolation of both steam generators. The valves associated with both SGs are closed, but neither DHRS train is actuated. The response is similar to Scenario 4, except the increase in primary pressure is more rapid and results in earlier reactor trip and RSV opening. The actuation of DHRS occurs later in the transient after MPS signals are reached. Secondary pressure peaks after DHRS actuation.

Each scenario is considered to determine the limiting cases for the acceptance criteria. Sensitivity studies on initial primary and secondary conditions are performed as identified in Table 7-52 to identify the conditions that maximize peak system pressures.

Table 7-50 Acceptance criteria, single active failure, loss of power scenarios - inadvertent decay heat removal system actuation

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Challenged during the heatup scenarios of this event.
No single failure	The challenging primary pressurization cases typically occur when all equipment is operational.
Failure of one FWIV to close	Potentially challenging for secondary pressure, but typically bounded by other heatup events.
No loss of power	Maximizes system pressures.

7.2.11.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-51.

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Table 7-51 Acceptance criteria - inadvertent decay heat removal system actuation

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation. The DHRS heat removal is limited by the DHR condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.11.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-52 are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	}}2(a),(c)

Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the high condition	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
Steam generator heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray	 Varied.	}}2(a),(c)

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Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation (Continued)

Parameter	Bias / Conservatism	Basis
PZR heaters	Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Disabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

For example, representative sensitivity studies performed to identify the highest primary and secondary side pressure, varying primary conditions, steam

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generator heat transfer and pressure, are presented in Table 7-53. Unless otherwise noted, cases use BOC kinetics. The results of the sensitivity studies indicate that peak primary pressure occurs when both trains of DHRS inadvertently actuate (i.e., total loss of normal heat sink). The lowest-setpoint RSV operates to mitigate peak pressurization well below the acceptance criterion of 2310 for this representative NPM). {{

}}2(a),(c) The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

Table 7-53 Representative sensitivity studies - inadvertent decay heat removal system actuation

{{

}}2(a),(c)

7.2.12 Feedwater System Pipe Break Inside or Outside Containment

The methodology used to simulate a postulated feedwater system pipe break for an NPM, and an evaluation of the acceptance criteria listed in Table 7-4, are presented below. Since both split breaks (relatively higher event frequency) and double-ended guillotine breaks (relatively lower event frequency) are analyzed, the more restrictive AOO criteria for system pressures, critical heat flux ratio, and fuel centerline melt applicable to breaks with higher event frequency are used in the evaluation.

7.2.12.1 General Event Description

A feedwater line break can occur inside or outside of containment, and can range in size from a small split crack to a double ended rupture. A large feedwater line break inside containment results in a loss of containment vacuum and a high containment pressure MPS signal that actuates a reactor trip, isolates the secondary system and CVCS, and opens the DHRS valves. The steam generator, DHRS piping, and DHRS condenser on the affected side drain through the break. The non-affected steam generator system and DHRS loop provide cooling to the RCS via heat transfer to the reactor pool. The response of smaller feedwater line breaks inside containment is similar except that other MPS setpoints, such as high PZR pressure, may be reached before high containment pressure.

A feedwater line break outside containment causes a loss of feedwater flow to the steam generators and a heatup of the RCS. Larger breaks result in rapid heatup events that pressurize the RCS beyond the high PZR pressure analytical limit. Smaller breaks cause a more gradual heatup, loss of secondary pressure, and reactor trip and DHRS actuation may occur on other MPS signals. The DHRS provides cooling to the RCS via heat transfer to the reactor pool. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition.

{{

}}2(a),(c)

Table 7-54 lists the relevant acceptance criteria, SAF, and LOP scenarios.

Sensitivity studies on primary and secondary conditions, and break size/location are performed to identify the conditions that maximize peak primary and secondary pressures.

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Table 7-54 Acceptance criteria, single active failure, loss of power scenarios - feedwater line break

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	System pressures are challenged during this overheating event.
Failure of one FWIV to close on the train with the break	The challenging cases typically occur when FWIV single failure is assumed.
Loss of AC power at transient initiation	Typically maximizes system pressures.

7.2.12.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-55.

Table 7-55 Acceptance criteria - feedwater line break

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation, in addition to the actual FWLB. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation to a more serious accident or consequential loss of functionality	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

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7.2.12.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-56, and a spectrum of break locations and sizes, are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased to the high condition.	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	}}2(a),(c)

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Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break (Continued)

Parameter	Bias / Conservatism	Basis
SG tube plugging	Nominal.	
Break size / location	Spectrum.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Disabled.	
CNV Pressure Control		}}2(a),(c)

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Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break (Continued)

Parameter	Bias / Conservatism	Basis
		{{
CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

For example, representative sensitivity studies performed to identify the highest primary and secondary side pressures, varying primary conditions, break size and location, and steam generator heat transfer and pressure, are presented in Table 7-57. Unless otherwise noted, breaks are located at the bioshield, RCS flows are initialized at the minimum value, and losses of normal AC power occur at event initiation. The results of the sensitivity studies indicate that peak primary pressures occur during a break coincident with a loss of AC power; under these conditions the lowest-setpoint RSV operates to mitigate peak pressurization well below the acceptance criterion of 2310 for this representative NPM. Initial condition biasing contributions are secondary compared to the pressure response to the total loss of heat sink and multiple cases have similar peak pressures. With respect to maximum secondary side SG pressure, maximum SG pressure occurs during a 10 percent split break just outside of containment coincident with a loss of AC power when initial SG pressure is biased to the high condition. The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

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Table 7-57 Representative sensitivity studies - feedwater line break

{{

}}2(a),(c)

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

The methodology used to simulate a postulated uncontrolled CRA bank withdrawal from subcritical or low power startup conditions for an NPM, and an evaluation of the acceptance criteria listed for an AOO in Table 7-4, are presented below.

The range of initial power levels associated with low power startup conditions for an NPM is based on the low setting for the high power signal (below 15 percent RTP for the example in Table 7-3). When core power reaches the low setting level, a hold

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point is established to alter the high power setting. Thus, low power startup conditions exist until reactor power reaches the low setting level.

7.2.13.1 General Event Description and Methodology

The limiting event consequences to an uncontrolled CRA bank withdrawal from subcritical or low power startup conditions typically (for most PWR designs) occur for cases with very low initial power levels (~1 Watt). The primary reason for this behavior is the flux rate signals associated with the source range and intermediate range are typically either not safety related or not of sufficient quantity to adequately address single failures. An NPM, however, incorporates a safety related signal for each of these channels from each core quadrant into the MPS. Consequently, the limiting event consequences for an NPM typically occur for cases with higher initial power levels.

A spectrum of constant reactivity insertion rates is evaluated, {{ }}[2(a),(c)] These reactivity insertion rates encompass the credible range resulting from a single control bank withdrawal. If necessary, this range is supplemented to include the reactivity insertion rates associated with an inadvertent decrease in boron concentration event (Section 7.2.16).

Two event scenarios with different protection schemes are evaluated to determine which scenario produces the limiting event consequences. The first scenario arises when the high count rate signal is available because the intermediate range channel does not have an established signal. In this instance, the high power rate signal is not active (below 15 percent RTP in the Table 7-3 example), so core protection is provided by the high count rate signal and the startup rate (source range) signal. {{

}}[2(a),(c)]

The second scenario arises when the high count rate signal is not available because the intermediate range channel has an established signal. In this instance, the high power rate signal is not active (below 15 percent RTP in the Table 7-3 example), thus core protection is provided by the high power signal and the startup rate (intermediate range) signal. The event scenario with the highest core power typically corresponds to the initial power level and reactivity insertion rate that cause the high power signal (low setting) and the startup rate (intermediate range) signal to occur at nearly the same time. This scenario is typically limiting because it represents the maximum rate of power change at the maximum core power. If the initial power is increased, the reactor trips on the high power signal but at a slower rate of power increase. Similarly, if the reactivity

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insertion rate is increased, the reactor trips on the startup rate but at a lower core power.

Before initiating an approach to critical, the reactor coolant is heated to the minimum temperature for criticality. The heating of the reactor coolant is performed by the Module Heatup System (MHS) via the CVCS.

At least one feedwater pump is operating at these RCS temperatures. Since feedwater flow is provided to both SGs, it may continue to provide decay heat removal following an uncontrolled CRA bank withdrawal from subcritical or low power startup conditions. If normal feedwater flow is not available, then depending on the point at which the event occurs in the startup process, the flooded containment or the DHRS provides decay heat removal. The peak power and duration of the power spike are not sufficient to cause a significant temperature or pressure increase. Hence, the maximum power and minimum CHFR occur shortly after reactor trip while the RCS pressure and MS pressure do not challenge the relevant acceptance criteria.

Sensitivity studies are performed as identified in Table 7-60 to identify the case(s) with a potentially limiting MCHFR or fuel centerline temperature. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

Table 7-58 lists the relevant acceptance criteria, SAF, and LOP scenarios.

Table 7-58 Acceptance criteria, single active failure, loss of power scenarios - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR and maximum fuel centerline temperature.	MCHFR and maximum fuel centerline temperature are challenged during this reactivity anomaly event.
No single failure.	The challenging cases typically occur when all equipment is operational.
No loss of power.	The challenging cases typically occur when AC power is available for the event duration.

7.2.13.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-59.

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Table 7-59 Acceptance criteria - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions

Acceptance Criteria	Discussion
Primary pressure	This criterion is not an acceptance criterion listed in Section 15.4.1 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for primary pressure compared to other AOOs.
Secondary pressure	This criterion is not an acceptance criterion listed in Section 15.4.1 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for secondary pressure compared to other AOOs.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS (if actuated) pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.13.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-60 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature.

Table 7-60 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Nominal.	}}2(a),(c)

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Table 7-60 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions (Continued)

Parameter	Bias / Conservatism	Basis
		{{
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Nominal.	
Initial PZR level	Nominal.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Biased to BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Reactivity insertion rate	Spectrum.	
RCS Temperature Control Automatic rod control Boron concentration	 N/A. Not credited.	}}2(a),(c)

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Table 7-60 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions (Continued)

Parameter	Bias / Conservatism	Basis
PZR Pressure Control PZR spray PZR heaters	 Enabled. Enabled.	{{ }}2(a),(c)
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 N/A. Enabled.	
Feedwater and Turbine Load Control feedwater pump speed	 N/A.	
CNV Pressure Control CNV evacuation system	 Enabled.	

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases for lowest MCHFR and highest fuel centerline temperature for this reactivity event.

For example, representative sensitivity studies for the initial core power and reactivity insertion rate are presented in Table 7-61. The results demonstrate the lack of challenging MCHFR values predicted for this event by NRELAP5, which is further supported by the values predicted with the approved subchannel methodology. {{ }}2(a),(c)

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{{

}}2(a),(c)

Table 7-61 Representative sensitivity studies - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions

{{

}}2(a),(c)

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7.2.14 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

The methodology used to simulate a postulated uncontrolled control rod assembly (CRA) bank withdrawal at power for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.14.1 General Event Description and Methodology

As stated in Section 7.2.13, low power startup conditions exist for an NPM until reactor power reaches the low setting level (15 percent RTP for the example in Table 7-3). Accordingly, the uncontrolled CRA bank withdrawal at power event extends from the low setting level to HFP.

The withdrawal of the control bank causes a reactivity insertion that increases reactor power and leads to a rise in coolant temperature, pressurizer level, and RCS pressure. Reactivity feedback from the rising fuel temperature partially counteracts the reactivity insertion, slowing the power increase, which continues until the system trips on high power, high power rate, high pressurizer pressure, or high RCS temperatures. The maximum power and minimum MCHFR occur just after the resulting scram, while the peak primary pressure occurs some time later, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established by the end of the transient.

Following reactor trip and subsequent turbine trip, the turbine bypass to the condenser opens to control the RCS temperature. However, the actions of the turbine bypass system are not credited, so as to minimize heat removal by the secondary side. Although turbine load is an input to the feedwater controller, no changes are made to this controller because the RCS responses are not sufficient to affect feedwater control.

The limiting MCHFR typically occurs for a reactivity insertion rate that results in reactor trip on core power, pressurizer pressure, and RCS riser temperature signals at approximately the same time. These conditions typically arise for events initiated with lower reactivity insertion rates because higher reactivity insertion rates cause the MPS to trip much earlier on high power rate. The earlier reactor trip reduces the energy added to the reactor coolant, thereby producing a higher MCHFR. The range of reactivity insertion rates considered is sufficient to identify the point of transition {{ }}$^{2(a),(c)}$ Sensitivity studies are performed on a variety of parameters, as identified in Table 7-64, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

The maximum fuel centerline temperature typically occurs when core power exceeds its analytical limit. This condition typically arises for events initiated from full power with the highest reactivity insertion rate as determined from the resulting bank worth and control rod step speed.

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A spectrum of constant reactivity insertion rates is evaluated. These reactivity insertion rates encompass the credible range resulting from a single control bank withdrawal. If necessary, this range is supplemented to cover the reactivity insertion rates associated with an inadvertent decrease in boron concentration event (Section 7.2.16).

Table 7-62 lists the relevant acceptance criteria, SAF, and LOP scenarios.

Table 7-62 Acceptance criteria, single active failure, loss of power scenarios - uncontrolled control rod bank withdrawal at power

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR and maximum fuel centerline temperature	MCHFR and maximum fuel centerline temperature are challenged during this reactivity anomaly event.
No single failure	The challenging cases typically occur when all equipment is operational.
No loss of power	The challenging cases typically occur when AC power is available for the event duration.

7.2.14.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-63.

Table 7-63 Acceptance criteria - uncontrolled control rod bank withdrawal at power

Acceptance Criteria	Discussion
Primary pressure	This criterion is not an acceptance criterion listed in Section 15.4.2 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for primary pressure compared to other AOOs.
Secondary pressure	This criterion is not an acceptance criterion listed in Section 15.4.2 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for secondary pressure compared to other AOOs.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.

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Table 7-63 Acceptance criteria - uncontrolled control rod bank withdrawal at power

Acceptance Criteria	Discussion
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.14.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-64 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature.

Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied. RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied. Biased to the high condition.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	}}2(a),(c)

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Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial fuel temperature	Biased to the low condition.	
MTC	Biased to BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Reactivity insertion rate	Spectrum. Maximum	
RCS Temperature Control Automatic rod control Boron concentration	 N/A. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied. Nominal.	
PZR Level Control Charging	 Not credited.	}}2(a),(c)

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Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power (Continued)

Parameter	Bias / Conservatism	Basis
Letdown	Varied.	{{ }}2(a),(c)
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Enabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases for lowest MCHFR for this reactivity event.

For example, representative sensitivity studies for initial core power, reactivity insertion rate, pressurizer pressure, pressurizer level, reactor coolant average temperature, pressurizer spray flow, pressurizer heater status, letdown status, and loss of power are presented in Table 7-65. The results demonstrate the two

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most limiting MCHFRs occur at initial power levels of 75 percent RTP and above, with AC power available for the event duration, the RCS average temperature biased low, and pressurizer spray modeled to delay reactor trip on high pressurizer pressure.

Table 7-65 Representative sensitivity studies - uncontrolled control rod bank withdrawal at power

{{

}}2(a),(c)

7.2.15 Control Rod Misoperation

The methodology used to simulate a postulated control rod misoperation for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

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7.2.15.1 General Event Description and Methodology

The rod control system is used to move (insert or withdraw) the control rod assemblies (CRAs) in response to an operator action or an automatic control. Since these transients are initiated by a malfunction in the rod control system, a variety of reactivity related conditions can result. Specific reactivity conditions for an NPM include: 1) withdrawing a single CRA; 2) dropping one or more CRAs; or, 3) leaving one or more CRAs behind when inserting or withdrawing a control bank. The consequences for each of these reactivity conditions are discussed below.

Table 7-66 lists the relevant acceptance criteria, single active failure, and loss of power scenarios.

Withdrawal of a Single CRA

The withdrawal of a single CRA causes a reactivity insertion that increases reactor power and leads to a rise in coolant temperature, pressurizer level, and RCS pressure. Feedback from the rising fuel temperature is not sufficient to counteract the reactivity insertion, so the power increases until the system trips on high power, high power rate, high pressurizer pressure, or high RCS temperatures. The maximum power and minimum MCHFR occur just after the resulting scram, while the peak primary pressure occurs some time later, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established at the end of the transient.

The limiting MCHFR typically occurs for a reactivity insertion rate that results in reactor trip on core power, pressurizer pressure, and RCS temperature signals at approximately the same time. These conditions arise for events initiated from partial power with lower reactivity insertion rates because higher reactivity insertion rates cause the MPS to trip much earlier on high power rate. The earlier reactor trip reduces the energy added to the reactor coolant, thereby producing a higher MCHFR. The asymmetry associated with the core power response causes the ex-core detectors to respond differently for each quadrant. Consequently, the range of reactivity insertion rates considered is sufficient to identify the point of transition {{ }}[2(a),(c)] (with power related trips using the lowest reading ex-core detector based on the minimum after to before event initiation ratio of the radial peaking factors for the outer row of fuel assemblies). Sensitivity studies are performed on a variety of parameters, as indicated in Table 7-68, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

In contrast to the uncontrolled CRA bank withdrawal at power event (Section 7.2.14), the highest reactivity insertion rate as determined from the resulting rod worth and control rod step speed is significantly lower for the withdrawal of a single CRA event.

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Note that instead of using case-specific values for rod worth and parameters associated with the asymmetry, more conservative bounding input values can be used. The use of bounding input values provides a conservative analysis simplification for the withdrawal of a single CRA event. In addition, the bounding input values can then be used to compare this event to the dropping of one or more CRAs event described below.

Dropping One or More CRAs

Based on the minimum worth at any time during the cycle for a given core power, i.e., with the control bank positioned at the PDIL, dropping a single CRA causes a reactivity insertion that decreases reactor power. Feedback from the decreasing fuel temperature and the actions of the rod control system to restore power are generally not sufficient to counteract the reactivity insertion, so the power decreases until the system trips on high power rate. For event scenarios without a return to power, the maximum core power, peak primary pressure, and MCHFR occur at event initiation. The peak secondary system pressure occurs some time after the scram, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established at the end of the transient.

The potential for a return to power exists only for events initiated from less than RTP because the reduced worth of the dropped rod gives the rod control system time to act. The corresponding MCHFR for a dropped rod event with a return to power is typically greater than the MCHFR for events initiated from HFP. Hence, the limiting MCHFR cases typically occur at HFP conditions.

Following reactor trip and subsequent turbine trip, the turbine bypass to the condenser opens to control the RCS temperature. However, the actions of the turbine bypass system are not credited, so as to minimize heat removal by the secondary side. Although turbine load is an input to the feedwater controller, no changes are made to this controller because the RCS responses are not sufficient to affect feedwater control.

The asymmetry associated with the core power response causes the ex-core detectors to respond differently for each quadrant. The power input to the high power rate signal uses the highest reading ex-core detector, multiplying the core average power by the maximum after drop to before drop ratio of the radial peaking factors for the outer row of fuel assemblies. Sensitivity studies are performed, as identified in Table 7-70, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

The maximum fuel centerline temperature typically occurs at event initiation for those event scenarios with an immediate reactor trip. If the event scenario has a return to power, the maximum fuel centerline temperature is typically bounded by the fuel centerline temperature at HFP because the associated power peak is less than full power.

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As an alternative to performing a system transient analysis, the MCHFR and linear heat generation rate of the dropped rod event can be confirmed to be bounded by other events. Most rod drops result in a reactor trip on high power rate because of the immediate decrease in power from the dropped rod. Figure 7-3 shows the decrease in power from the dropped rod {{ }}2(a),(c) as a function of rod worth for a variety of initial power levels. The figure is based on a representative NPM core design and also overlays a representative high power rate trip. {{

}}2(a),(c) Sensitivity studies were performed to confirm this behavior using several representative NPM core designs, a range of initial power levels, and a variety of rod worths. {{

}}2(a),(c) The alternative bounding method for the rod drop screens rod drop cases into two groups: those that result in reactor trip within a short period and those that do not.

For the larger group of cases that result in the early reactor trip {{ }}2(a),(c), the dropped rod, and subsequent reactor trip, cause a decrease in global power. The dropped rod does cause an asymmetry and results in an increase in local power peaking (primarily a radial increase with a small axial increase). The subsequent reactor trip eliminates the asymmetry and the associated local power peaking. Figure 7-4 shows the impact on the local power from the decrease in global power and the increase in local peaking for a representative NPM rod drop case that results in early reactor trip. {{

}}2(a),(c) The relevant acceptance criteria in Table 7-66 for the first group of the rod drop event can be assessed by the limiting steady-state subchannel analysis.

For the smaller group of cases that do not result in the early reactor trip {{

}}2(a),(c) However, because power overshoot may occur during the event, these cases are bounded instead by comparison to the withdrawal of a single CRA. A core design code consistent with the discussion in Section 4.3.1.1, such as SIMULATE5 (Reference 23), is used to calculate {{

}}2(a),(c

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{{

}}2(a),(c) the conditions associated with the dropped rod cases are bounded by the values used in the analysis of the withdrawal of a single CRA. Therefore, the limiting MCHFR and linear heat generation rate for the dropped rod cases that do not result in early reactor trip are bounded by the results of the single rod withdrawal analysis. This comparison of the rod drop analysis cases to the single rod withdrawal analysis values is performed on a cycle-specific basis. A rod drop analysis case with a parameter that falls outside the values used in the single rod withdrawal analysis, {{

}}2(a),(c), would require use of the system transient approach described earlier in this section. When the cycle-specific comparison is bounding, the relevant acceptance criteria in Table 7-66 for the second group of the rod drop event can be assessed by the limiting single rod withdrawal subchannel analysis.

Misalignment of One or More CRAs

The misalignment of CRAs occurs as a result of one or more CRAs being left behind when inserting or withdrawing the control bank. These conditions are not evaluated with NRELAP5 as part of the non-LOCA event methodology because this event is not a transient. Instead, the MCHFR is determined as part of a detailed subchannel evaluation.

Table 7-66 Acceptance criteria, single active failure, loss of power scenarios - control rod misoperation

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR and maximum fuel centerline temperature	MCHFR and maximum fuel centerline temperature are challenged during this reactivity anomaly event.
Single failure of an ex-core flux detector	This single failure is typically limiting because it delays time of reactor trip by requiring actuation based on responses of the least affected core quadrants.

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Table 7-66 Acceptance criteria, single active failure, loss of power scenarios - control rod misoperation (Continued)

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
No loss of power	A loss of AC power at event initiation is typically non-limiting (early reactor trip), while a loss of AC power coincident with reactor trip typically does not alter the MCHFR or fuel centerline temperature because the time of reactor trip is not changed.

Figure 7-3 Example power reduction vs. initial power and worth for dropped rod event at {{ }}2(a),(c)

{{

}}2(a),(c)

Figure 7-4 Example components of local power vs. time for dropped rod event

{{

}}2(a),(c)

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Figure 7-5 Example comparison of parameters associated with non-tripped rod drop event cases to single rod withdrawal event analysis values

{{

}}2(a),(c)

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7.2.15.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-67.

Table 7-67 Acceptance criteria - control rod misoperation

Acceptance Criteria	Discussion
Primary pressure	This criterion is not an acceptance criterion listed in Section 15.4.3 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for primary pressure compared to other AOOs.
Secondary pressure	This criterion is not an acceptance criterion listed in Section 15.4.3 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for secondary pressure compared to other AOOs.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates, and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.15.3 Biases, Conservatisms, and Sensitivity Studies

Withdrawal of a Single CRA

The biases and conservatisms presented in Table 7-68 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature for the withdrawal of a single CRA event.

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Table 7-68 Initial conditions, biases, and conservatisms - control rod misoperation, single control rod assembly withdrawal

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the low condition.	
MTC	Biased to BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Reactivity insertion rate	Spectrum.	
RCS Temperature Control Automatic rod control	N/A.	}}2(a),(c)

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Table 7-68 Initial conditions, biases, and conservatisms - control rod misoperation, single control rod assembly withdrawal (Continued)

Parameter	Bias / Conservatism	Basis
		{{
Boron concentration	Not credited.	
PZR Pressure Control		
PZR spray	Varied.	
PZR heaters	Varied.	
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control		
Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control		
feedwater pump speed	Enabled.	
CNV Pressure Control		
CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently,

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sensitivity studies are performed to identify cases for lowest MCHFR for this reactivity event.

For example, representative sensitivity studies for initial core power, reactivity insertion rate, pressurizer pressure, pressurizer level, reactor coolant average temperature, pressurizer spray flow, pressurizer heater status, letdown status, and loss of power are presented in Table 7-69 for the withdrawal of a single CRA event. The results demonstrate the MCHFR occurs at an initial power level of 75 percent RTP, with AC power available for the event duration, the RCS average temperature biased low, the pressurizer pressure biased low, the pressurizer level biased low, and spray modeled to delay reactor trip on high pressurizer pressure.

Table 7-69 Representative sensitivity studies - control rod misoperation, single control rod assembly withdrawal

{{

}}2(a),(c)

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Table 7-69 Representative sensitivity studies - control rod misoperation, single control rod assembly withdrawal (Continued)

{{

}}2(a),(c)

Dropping One or More CRAs

The biases and conservatisms presented in Table 7-70 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature for the dropped CRA(s) event. If the alternative bounding approach is used, where cases that result in early trip are bounded by steady-state condtions and cases that do not result in early trip are bounded by the single rod withdrawal, the biases and conservatisms in Table 7-70 associated with system transient analysis are not applicable. However, the alternative bounding approach considers combinations of initial core power, dropped CRA worth, core time-in-life, axial offset, flow, and temperature to screen the cases into groups and compare the non-tripped cases to single rod withdrawal analysis limits.

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Table 7-70 Initial conditions, biases, and conservatisms - control rod misoperation, dropped control rod assemblies

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Nominal.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Varied.	
Kinetics	Varied.	
Decay heat	Biased to the high condition.	
Initial SG pressure	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Dropped CRA worth	Minimum	
RCS Temperature Control Automatic rod control	Enabled.	}}2(a),(c)

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Table 7-70 Initial conditions, biases, and conservatisms - control rod misoperation, dropped control rod assemblies (Continued)

Parameter	Bias / Conservatism	Basis {{
Boron concentration	Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Nominal. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Enabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases for lowest MCHFR for this reactivity event.

For example, representative sensitivity studies for the initial core power, dropped CRA worth, and core time-in-life are presented in Table 7-71 for the dropped CRA(s) event. The results demonstrate the MCHFR occurs at full power conditions.

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Table 7-71 Representative sensitivity studies - control rod misoperation, dropped control rod assemblies

{{

}}2(a),(c)

7.2.16 Inadvertent Decrease in Boron Concentration

The methodology used to simulate an inadvertent decrease in boron concentration for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.16.1 General Event Description and Methodology

The boric acid blend system incorporated into the NuScale plant design permits the operator to control the boron concentration of the reactor coolant via the charging fluid chemistry. While the NuScale plant design incorporates both automatic and manual controls, strict administrative procedures govern the process for adjusting the boron concentration of the reactor coolant. These administrative procedures establish limits on the rate and duration of the dilution.

The primary means of causing an inadvertent decrease in boron concentration is failure of the blend system, either by controller or mechanical failure, or operator error. The event is terminated by isolating the source for the diluted water, i.e., by closing the demineralized water system (DWS) isolation valves or closing the CVCS isolation valves.

For Mode 1 plant operating conditions, the perfect mixing model and the wave front model are both evaluated. The perfect mixing model is evaluated for Mode 1 operating conditions because it provides a slower reactivity insertion rate, delaying detection, potentially allowing further loss of shutdown margin. The wave front model is physically conservative because it assumes the maximum amount of reactivity as the diluted slug of water sweeps through the core. This model does not assume any axial blending to ensure that this reactivity insertion rate is maximized. For Modes 2 and 3 where boron dilution is allowed and limited mixing

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exists, a wave front model is used. These mixing models are generally performed as a hand calculation, but may be automated via a spreadsheet or other process.

The following mathematical expression is used to determine the time required to erode the shutdown margin with the perfect mixing model. Equation 7-1 shows that the reactivity insertion rate depends on the dilution rate and the total RCS mass.

$$t_{dil} = -\frac{1}{K}\ln\left(\frac{C_i}{C_f}\right) \qquad \text{Equation 7-1}$$

where:

t_{dil} = time required to dilute from the initial boron concentration to the final boron concentration, s

K = $-(Q_{in}\rho_{in})/(60V_r\rho_r)$

Q_{in} = dilution flow rate of unborated water, gpm

ρ_{in} = dilution water density, lbm/ft^3

V_r = effective water volume of the RCS, gal

ρ_r = density of the water in the RCS, lbm/ft^3

C_i = initial boron concentration (maximum critical boron concentration including uncertainties), ppm

C_f = final boron concentration (boron concentration at which shutdown margin is lost), ppm

The following mathematical expressions are used to determine the number of wave fronts and the time required to erode the shutdown margin with the wave front model. In this model, the boron concentration of the RCS is reduced in discrete steps at each time, *t*, corresponding to the time the wave front passes through the core. The reactivity insertion rate is determined from the reactivity step change calculated as the product of the change in boron concentration and boron worth, divided by the core transport time. The core transport time is calculated as the total mass in the core divided by the RCS flow rate. As shown in Equation 7-2, the change in boron concentration is also inversely proportional to the RCS flow rate, therefore the ratio of total reactivity step change and core

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transport time makes the initial reactivity insertion rate independent of the RCS flow rate.

$$C_N = C_i\left[\frac{W_{NC}}{(W_D + W_{NC})}\right]^N \qquad \text{Equation 7-2}$$

$$\mathrm{t} = \frac{M_{RCSI}}{(W_D + W_{NC})} + (N-1)\frac{M_{RCS}}{(W_D + W_{NC})} \qquad \text{Equation 7-3}$$

where:

C_N = the Nth front boron concentration, ppm

C_i = initial boron concentration, ppm

W_D = dilution mass flow rate, lbm/s

W_{NC} = natural circulation mass flow rate, lbm/s

M_{RCS} = RCS fluid mass minus the pressurizer, lbm

M_{RCSI} = initial pass RCS fluid mass (mass between the CVCS injection point to core inlet), lbm

N = number of times the wave front passes through the core

Mode 1 (Operations) HFP to 25 percent RTP

In this mode of plant operation, an inadvertent decrease in boron concentration causes a reactivity insertion that increases reactor power, which leads to a rise in coolant temperature, pressurizer level, and RCS pressure. The highest reactivity insertion rate will occur with the maximum dilution flow rate (e.g., 50 gpm for 2 CVCS pumps) with unborated water. However, operation with two CVCS pumps is typically limited to certain conditions by administrative controls. For example, two pump operation may only be allowed at certain power levels or when critical boron concentration is below a certain threshold. As a result, the single pump dilution flow rate (e.g., 25 gpm) is also evaluated. Similarly, the operation of letdown in an automatic mode could prevent detection and mitigation of an inadvertent dilution event by high pressurizer level. Therefore, automatic letdown is also typically limited to certain conditions by administrative controls.

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The reactivity insertion rates associated with these configurations are determined using both the perfect mixing model (Equation 7-1) and the wave front model (Equation 7-2 and Equation 7-3). The times of reactor trip and isolation of the dilution source via closure of the DWS isolation valves are obtained from the NRELAP5 results for these reactivity insertion rates. The calculations performed with the perfect mixing model are also used to determine the shutdown margin available after isolation of the DWS, and the time at which the shutdown margin would be lost if the dilution source is not terminated. The system responses for all other acceptance criteria, such as MCHFR and peak RCS pressure, are comparable to the uncontrolled control rod bank withdrawal at power event (Section 7.2.14) by comparison of the reactivity insertion rates. The spectrum of NRELAP5 analyses for the uncontrolled control rod bank withdrawal at power event (Section 7.2.14) can also be used to determine the limiting total reactivity insertion prior to reactor trip and DWS isolation to determine remaining shutdown margin as an alternative to performing NRELAP5 analysis for specific boron dilution reactivity insertion rates to determine shutdown margin. For cases where automatic letdown is disabled and high pressurizer level trip is credited, the shutdown margin remaining at the time of reactor trip and DWS isolation can be determined from the volume change without NRELAP analysis.

Mode 1 (Operations) HZP

The reactivity insertion rates associated with these configurations are determined using both the perfect mixing model (Equation 7-1) and the wave front model (Equation 7-2 and Equation 7-3). The time of reactor trip and isolation of the dilution source via closure of DWS isolation valves is obtained from the NRELAP5 results for these reactivity insertion rates. The calculations performed with the wave front model are also used to determine the shutdown margin available after isolation of the DWS, and the time at which the shutdown margin would be lost if the dilution source is not terminated. The system responses for all other acceptance criteria, such as MCHFR and peak RCS pressure, are comparable to the uncontrolled CRA bank withdrawal from subcritical or low power startup conditions event (Section 7.2.13) by comparison of the reactivity insertion rates. The spectrum of NRELAP5 analyses for the uncontrolled control rod bank withdrawal from subcritical or low power startup conditions event (Section 7.2.13) can also be used to determine the limiting total reactivity insertion prior to reactor trip and DWS isolation to determine remaining shutdown margin as an alternative to performing NRELAP5 analysis for specific boron dilution reactivity insertion rates to determine shutdown margin. For cases where automatic letdown is disabled and high pressurizer level trip is credited, the shutdown margin remaining at the time of reactor trip and DWS isolation can be determined from the volume change without NRELAP analysis.

Mode 2 (Hot Shutdown) and Mode 3 (Safe Shutdown)

In these modes of plant operation, the MPS protection logic ensures the DWS is isolated when the RCS flow rate is less than the low flow setpoint (1.7 ft^3/s in the

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example Table 7-3). This protection scheme precludes the possibility for an inadvertent decrease in boron concentration.

When the RCS flow rate is greater than or equal to the low flow setpoint (1.7 ft^3/s in the example Table 7-3), the reactivity insertion from the maximum dilution flow rate (e.g., 25 gpm for 1 CVCS pump) with unborated water causes an increase in reactor power (neutron population). The increase in neutron flux is detected by the MPS count rate protection signal and used to close the DWS isolation valves. The calculations performed with the wave front model (Equation 7-2 and Equation 7-3) determine the shutdown margin available after isolation of the DWS, and the time at which the shutdown margin would be lost if the dilution source is not terminated. For cases where automatic letdown is disabled and high pressurizer level isolation is credited, the shutdown margin remaining at the time of DWS or CVCS isolation can be determined from the volume change.

Mode 4 (Transition)

In this mode of plant operation, all CVCS connections to an NPM are disconnected, isolated, or locked out. Thus, the possibility of a design-basis inadvertent decrease in boron concentration is precluded.

Mode 5 (Refueling)

In this mode of plant operation, the Technical Specifications require the pool boron concentration to be sufficient to have appropriate shutdown margin. For some NPM designs, the Technical Specifications also require the pool level to be maintained within a narrow range. Surveillance of the boron concentration, and level if applicable, of the pool is performed at appropriate intervals and is expected to prompt operator actions in accordance with Technical Specifications during an inadvertent dilution of the pool; such operator actions are not credited.

Two methods can be used for assessing the dilution volume that, if allowed to enter the pool, could cause a loss of shutdown margin in Mode 5. If the pool volume is treated as constant, then the perfect mixing model described above for Mode 1 can be used to calculate the time to dilution for an arbitrary flow rate. The dilution time and arbitrary flow rate are then used to determine the dilution volume. Alternatively, if pool volume is treated as increasing during the event, the initial boron mass is calculated using the assumed initial pool volume and concentration. The boron concentration at which shutdown margin is lost is determined from the initial shutdown margin and boron worth. The total water volume having the initial boron mass and the boron concentration at which shutdown margin is lost is calculated. The assumed initial volume is subtracted from the total water volume to determine the dilution volume that would result in loss of shutdown margin. The dilution volume determined from either of these two methods is then compared to the volumes of possible dilution sources.

The relevant acceptance criteria, SAF, and LOP scenarios are listed in Table 7-72.

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Table 7-72 Acceptance criteria, single active failure, loss of power scenarios - inadvertent decrease in boron concentration

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	MCHFR is challenged during this reactivity anomaly event. (Reactivity insertion rates are insufficient to challenge fuel centerline temperature.)
No single failure	The challenging cases typically occur when all equipment is operational.
No loss of power	The challenging cases typically occur when AC power is available to the CVCS equipment for the event duration.

7.2.16.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-73.

Table 7-73 Acceptance criteria - inadvertent decrease in boron concentration

Acceptance Criteria	Discussion
Primary pressure	This criterion is not evaluated because an NPM does not introduce more challenging conditions for primary pressure compared to other AOOs.
Secondary pressure	This criterion is not evaluated because an NPM does not introduce more challenging conditions for secondary pressure compared to other AOOs.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis; or by demonstrating that a loss of shutdown margin does not occur.
Maximum fuel centerline temperature	This criterion is not evaluated because the reactivity insertion rates are insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS (if actuated) pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

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7.2.16.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-74 are considered in order to identify the bounding conditions for loss of shutdown margin.

Table 7-74 Initial conditions, biases, and conservatisms - inadvertent decrease in boron concentration

Parameter	Bias / Conservatism	Basis
Initial reactor power	Varied.	{{ }}2(a),(c)
Initial RCS average temperature	Biased to the high condition.	{{ }}2(a),(c)
Initial RCS flow rate	Biased to the low condition.	{{ }}2(a),(c)
Initial PZR pressure	Nominal.	{{ }}2(a),(c)
Initial PZR level	Varied.	{{ }}2(a),(c)
Initial feedwater temperature	Excluded.	Not part of mixing model.
Initial fuel temperature	Excluded.	Not part of mixing model.
MTC	Excluded.	Not part of mixing model.
Kinetics	Biased to BOC conditions.	{{ }}2(a),(c)
Decay heat	Excluded.	Not part of mixing model.
Initial SG pressure	Excluded.	Not part of mixing model.
SG heat transfer	Excluded.	Not part of mixing model.
RSV lift setpoint	Excluded.	Not part of mixing model.
SG tube plugging	Excluded.	Does not alter active RCS volume.
Shutdown margin	Biased to the low condition.	{{ }}2(a),(c)
Initial boron concentration	Biased to the high condition	{{ }}2(a),(c)
Boron worth	Biased to the high condition.	{{ }}2(a),(c)

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Table 7-74 Initial conditions, biases, and conservatisms - inadvertent decrease in boron concentration (Continued)

Parameter	Bias / Conservatism	Basis
Active RCS volume	Biased to the low condition.	{{ }}2(a),(c)
Makeup flow rate	Varied.	{{ }}2(a),(c)
Makeup temperature	Biased to the low condition.	{{ }}2(a),(c)
RCS Temperature Control		
Automatic rod control	Excluded.	Not part of mixing model.
Boron concentration	Not credited.	{{ }}2(a),(c)
PZR Pressure Control		
PZR spray (normal)	Excluded.	Not part of mixing model.
(bypass)	Excluded.	Not part of mixing model.
PZR heaters (non-prop.)	Excluded.	Not part of mixing model.
(prop.)	Excluded.	Not part of mixing model.
PZR Level Control		
Charging	Enabled.	{{ }}2(a),(c)
Letdown	Varied.	{{ }}2(a),(c)
Steam Pressure Control		
Turbine throttle valves	Excluded.	Not part of mixing model.
Turbine bypass valves	Excluded.	Not part of mixing model.
Feedwater and Turbine Load Control		
feedwater pump speed	Excluded.	Not part of mixing model.

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Table 7-74 Initial conditions, biases, and conservatisms - inadvertent decrease in boron concentration (Continued)

Parameter	Bias / Conservatism	Basis
CNV Pressure Control CNV evacuation system	Excluded.	Not part of mixing model.

{{

}}2(a),(c)

Table 7-75 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot full power

Mode 1 (HFP)	1 CVCS Pump	2 CVCS Pumps
Dilution Flow Rate (gpm)	25	50
Initial boron concentration (ppm)	1400	1400
Final boron concentration (ppm)	1196	1196
Perfect mixing reactivity insertion rate (pcm/s)	0.50	1.00
Reactor trip / DWS isolation time (s)	161	78
DWS isolation valves closed (s)	166	83
Shutdown margin at time of isolation (pcm)	Greater than 1944	Greater than 1912
Shutdown margin lost assuming no isolation (s)	4092 (68.2 min)	2046 (34.1 min)

Table 7-76 Representative results - inadvertent decrease in boron concentration in Mode 1 at 25 percent rated thermal power

Mode 1 (25 percent RTP)	1 CVCS Pump
Dilution flow rate (gpm)	25
Initial boron concentration (ppm)	1800
Final boron concentration (ppm)	1596
Perfect mixing reactivity insertion rate (pcm/s)	0.65
Wave front reactivity insertion rate (pcm/s)	19.1
Wave front initial reactivity step (pcm)	91
Shutdown margin lost assuming no isolation (s)	3120 (52 min)

Table 7-77 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot zero power

Mode 1 (HZP, 1 MWt)	1 CVCS Pump
Dilution Flow Rate (gpm)	25
Initial boron concentration (ppm)	1800
Final boron concentration (ppm)	1596
Initial reactivity step (pcm)	685
Wave front reactivity insertion rate (pcm/s)	17.3

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Table 7-77 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot zero power (Continued)

Mode 1 (HZP, 1 MWt)	1 CVCS Pump
Duration of reactivity insertion rate for each wave (s)	39.5
Initial wave front reaches core inlet (s)	900
Initial wave front reaches core exit (s)	939.5
Reactor trip / DWS isolation time(1) (s) (wave front at core exit + UCRWS trip time)	Approximately 2028
DWS isolation valves closed (s)	Approximately 2033
Shutdown margin at time of isolation (pcm)	Greater than 697
Shutdown margin lost assuming no isolation (s) (critical wave at core inlet)	4200 (70 min)

1. Conservatively estimated to occur upon arrival of the subsequent wave

Table 7-78 Representative results - inadvertent decrease in boron concentration in Mode 2

Mode 2 (Coolant Flow ≥ 1.7 ft^3/s)	1 CVCS Pump
Dilution Flow Rate (gpm)	25
Initial boron concentration (ppm)	785.5
Final boron concentration (ppm)	600
Initial reactivity step (pcm)	329
Wave front reactivity insertion rate (pcm/s)	8.32
Duration of reactivity insertion rate for each wave (s)	39.5
Reactor trip/DWS isolation wave at core inlet (s)	5280 (88 min)
DWS isolation valves closed (s)	5475
Shutdown margin at time of isolation (pcm)	Greater than 517
Shutdown margin lost assuming no isolation (s) (critical wave at core inlet)	7440 (124 min)

Table 7-79 Representative results - inadvertent decrease in boron concentration in Mode 3

Mode 3 (Coolant Flow = 1.7 ft^3/s)	1 CVCS Pump
Dilution flow rate (gpm)	25
Initial boron concentration (ppm)	813
Final boron concentration (ppm)	650
Initial reactivity step (pcm)	378
Wave front reactivity insertion rate (pcm/s)	9.55
Duration of reactivity insertion rate for each wave (s)	39.6
Reactor trip / DWS isolation wave at core inlet (s)	4200 (70 min)
DWS isolation valves closed (s)	4395
Shutdown margin at time of isolation (pcm)	Greater than 613
Shutdown margin lost assuming no isolation (s) (critical wave at core inlet)	6360 (106 min)

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

The methodology used to simulate a postulated CVCS malfunction that increases RCS inventory for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.17.1 General Event Description

The NuScale reactor does not have a pumped ECCS, therefore, the unplanned increase in RCS inventory event is caused by a malfunction of the CVCS makeup pumps or pressurizer level control system. If borated water at the same concentration of the primary system is added to the RCS, the addition of large amounts of water to the primary system typically generates a reactor trip on high pressurizer (PZR) water level or high PZR pressure. Table 7-80 lists the relevant acceptance criteria, SAF, and LOP scenarios.

The malfunction is assumed to isolate letdown and actuate both makeup pumps (each flowing at their maximum capacity), causing an unplanned increase in RCS inventory. The limiting pressure response occurs when the event is initiated from full power conditions, and the initial conditions are biased in the conservative directions. The increase in RCS inventory event is typically terminated by CVCS isolation on high PZR level. (The CVCS containment isolation valves are dual safety related valves.)

Sensitivity studies to identify the challenging conditions are performed, as identified in Table 7-82, to identify the case(s) with the potentially limiting system pressures. Cases that include the effects of CVCS recirculation, which increases the flow into the RCS, are also analyzed.

Table 7-80 Acceptance criteria, single active failure, loss of power scenarios - reactor coolant system inventory increase

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	System pressures are challenged during this mass addition event.
No single failure	The CVCS is isolated via dual safety-related isolation valves. If one of the isolation valves were to fail, the other CVCS isolation valve would provide system isolation.
No loss of power	Continued operation of the CVCS typically maximizes system pressures.

7.2.17.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-81.

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Table 7-81 Acceptance criteria - reactor coolant system inventory increase

Acceptance Criteria	Discussion
Primary pressure	Primary pressure rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Secondary pressure increases rapidly to the peak value upon turbine trip, then decreases during the cool down with DHRS.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.17.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-82 are considered in identifying the bounding transient simulation for primary pressure.

Table 7-82 Initial conditions, biases, and conservatisms - reactor coolant system inventory increase

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the low condition	
MTC	Consistent with EOC kinetics.	}}2(a),(c)

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Table 7-82 Initial conditions, biases, and conservatisms - reactor coolant system inventory increase (Continued)

Parameter	Bias / Conservatism	Basis {{
Kinetics	Biased to EOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal.	
Steam generator heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Makeup temperature	Varied	
RCS Temperature Control Automatic rod control Boron concentration	 Enabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied. Nominal.	
PZR Level Control Charging Letdown	 N/A. Disabled.	
Steam Pressure Control		}}2(a),(c)

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Table 7-82 Initial conditions, biases, and conservatisms - reactor coolant system inventory increase (Continued)

Parameter	Bias / Conservatism	Basis
Turbine throttle valves	Enabled.	{{
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control		
Feedwater pump speed	Enabled.	
CNV Pressure Control		
CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)
2. These inputs, in conjunction with least negative Doppler temperature coefficient, are selected to maximize the power response (if any) induced by the addition of colder CVCS water. However, since this event is driven by mass addition, reactivity effects prior to RTS actuation (if any) are small when compared to the pressurization associated with the increase in primary inventory.

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event, including cases with the highest pressure responses for this inventory increase event.

For example, sensitivity studies performed to identify the highest primary and secondary side pressures, varying primary conditions, makeup temperature, and spray availability are presented in Table 7-83. The results indicate that peak primary pressure occurs when initial primary temperature and pressure are biased to the low condition, and makeup temperature is biased to the high condition as the RTS actuates on high pressurizer pressure and the lowest-setpoint RSV operates to mitigate peak pressurization well below the acceptance criterion of 2310 for this representative NPM. With respect to maximum secondary side SG pressure, maximum SG pressure occurs when spray is used to delay the RTS actuation until high pressurizer level is reached.

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Table 7-83 Representative sensitivity studies - reactor coolant system inventory increase

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}}2(a),(c)

7.2.18 Failure of Small Lines Outside Containment

The methodology used to simulate a postulated failure of a small line connected to the primary coolant system outside of containment for an NPM, and an evaluation of the acceptance criteria for an infrequent event listed in Table 7-4, are presented below.

A postulated break in a small line carrying primary coolant is typically only evaluated for radiological consequences. Neither the plant design nor the use of natural circulation flow for an NPM introduces a more challenging condition for other acceptance criteria. Evaluation of postulated small line breaks within the context of a

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10 CFR 50 Appendix K regulatory LOCA analysis is covered separately in the LOCA-EM topical report.

7.2.18.1 General Event Description and Methodology

The event is initiated by a break in a line connected to the primary coolant system outside containment that causes a decrease in pressurizer pressure and level. The rate of decrease for both parameters depends on the break location and size. The subsequent decrease in RCS pressure provides little core reactivity from moderator feedback, so the reactor power remains relatively constant until reactor trip. In the absence of a loss of power at event initiation, the decreasing pressurizer pressure and level causes RTS actuation on the low pressurizer pressure signal or the low pressurizer level signal.

Regardless of the presence of a reactor trip, the sustained loss of reactor coolant from the break causes a continuous decrease in pressurizer pressure and level. Eventually an MPS low pressurizer pressure or level signal is generated. Depending on design, one or both of these MPS signals leads to a reactor trip (if not previously generated), DHRS actuation, and CVCS isolation. Closure of the FWIVs and MSIVs following DHRS actuation isolates the steam generators, while CVCS isolation terminates the loss of reactor coolant through the break (assumed to be in the CVCS system since this is the only primary coolant system pipe line that penetrates the containment). Core decay heat drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient terminates with the NPM in a safe, stable condition.

Several lines within the CVCS qualify as candidates for evaluation. These lines include makeup lines (CVCS injection); letdown lines (CVCS discharge); pressurizer spray lines; and, the degassing (high point vent) line. However, depending on the design, the degassing line may be eliminated from evaluation based on similarity with the spray lines if: 1) the spray lines and degassing line are the same size; 2) both line types connect to the top of the pressurizer; and, 3) both line types penetrate the containment head. From a system response perspective, the response to a break in a spray line is nearly the same as the response to a break in the degassing line. Thus, a break in the degassing line may not be required to be explicitly modeled.

When determining the magnitude of mass released for a break in a CVCS line outside containment the timing of CVCS isolation is critical, as CVCS isolation stops the release of mass from the RPV into the reactor building. Maximizing the mass released from the RPV and the duration of the iodine spike maximizes the radiological consequences. Section 4.3.6 describes an example approach for performing the radiological consequence analysis using bounding inputs for the mass release and isolation time. If a bounding approach is used, then system transient results for mass release and isolation time are not required for the downstream radiological analysis. System transient analysis can be used to confirm the radiological analysis assumptions are bounding, if necessary.

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The nature of a spray line or degassing line break differs from that for a break in either the makeup line or letdown line because vapor is expelled from the break instead of liquid. As a result, steam production in the pressurizer is a key phenomenon for a spray line or degassing line break, but much less important for makeup or letdown line breaks. A brief description of each break location is presented below.

Spray and Degassing Line Breaks

Calculations are performed to assess the responses to a spectrum of break sizes. Specifically, each break is modelled as either a double-ended guillotine rupture, or a smaller size break, of the piping. {{

}}2(a),(c) A break with an area of 100 percent of the line causes a rapid depressurization that quickly trips the reactor and isolates the break. As such, a break of this size is not typically limiting. While the rate of depressurization declines proportionally as the break area is reduced, a break area within the steam production capability of the pressurizer heaters allows the pressurizer pressure to be maintained until the reactor trips on a low pressurizer level signal. Due to reactor coolant volume shrinkage following reactor trip, the rate of decrease for both the pressurizer pressure and level is nearly independent of break flow, and any reduction in pressure reduces the mass released through the break. Regardless of the break size, a break in these lines is typically less limiting from a dose perspective than a break in either a makeup line or a letdown line because vapor is released instead of liquid, with no significant difference in event duration or iodine spiking time, thus the total mass released is smaller.

Makeup and Letdown Line Breaks

The evaluation of the makeup and letdown line breaks is divided into two phases:

1. Before Reactor Trip

   This phase lasts until the reactor trips. During this phase, the total break mass release depends primarily on the liquid density at the location of the break. The break is modelled as either a double-ended guillotine rupture, or a smaller size break, of the line piping. {{

   }}2(a),(c)

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2. After Reactor Trip

   This phase lasts from the time of reactor trip until isolation of the CVCS. During this phase, the pressurizer level decreases independently of the break flow due to the mismatch between heat generation in the core and heat removal via the steam generators. Since the rate of decrease for both the pressurizer pressure and level is nearly independent of break flow, and any reduction in pressure reduces the mass released through the break, the mass released is maximized by increasing the break area to include both lines prior to CVCS isolation. In contrast, the spiking time is maximized when the break is restricted to a single location.

Table 7-84 lists the relevant acceptance criteria, single active failure, and loss of power scenarios.

Table 7-84 Acceptance criteria, single active failure, loss of power scenarios - breaks in small lines carrying primary coolant outside containment

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Radiological consequences	A postulated break in a small line carrying primary coolant is evaluated for radiological consequences.
No single failure	The isolation valves on the makeup, letdown, and spray lines are safety grade and redundant. Therefore, failure of a single valve does not prevent isolation or significantly increase the radiological consequences.
Loss of AC at event initiation	The loss of heat transfer to the secondary system associated with a loss of AC power at event initiation typically results in the most challenging integrated mass release and spiking time.

7.2.18.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-85.

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Table 7-85 Acceptance criteria - breaks in small lines carrying primary coolant outside containment

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of the small line breaks carrying primary coolant event, sensitivities that maximize primary pressure are not analyzed. The peak primary pressure experienced during this event does not vary significantly from the pressure at event initiation, unless a pressure increase occurs due to an assumed loss of power. In all instances, the peak primary pressure remains below the design pressure, thereby providing margin to the acceptance criterion. If a system transient analysis is not required as input for the radiological consequence analysis, maximum primary pressure can be selected as the relevant acceptance criterion to address with the system transient analysis.
Secondary pressure	Secondary pressure increases rapidly post-DHRS actuation. However, due to the depressurizing nature of this event, sensitivities that maximize peak secondary pressure are not analyzed. In all instances the peak secondary pressure experienced during this event remains below the design pressure, thereby providing margin to the acceptance criterion.
Fuel cladding integrity	Fuel failure directly relates to dose by dictating the isotopic concentration of the reactor coolant being released. Consequently, the MCHFR and fuel centerline temperature criteria are relevant to determining the appropriate source term for the downstream radiological analysis. The depressurization following a break in a small line carrying primary coolant is slow enough to preclude an increase in core power or significant reduction in core flow prior to reactor trip. After considering the NuScale CHF trend with pressure, and the ranges bounded by the steady-state subchannel analysis, the CHFR is not challenged during the transient prior to reactor trip. Therefore, the fuel cladding integrity acceptance criteria are not challenged and event-specific follow-on MCHFR and fuel centerline temperature evaluations are not necessary.
Containment integrity	Containment integrity is evaluated in a separate analysis methodology.

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Table 7-85 Acceptance criteria - breaks in small lines carrying primary coolant outside containment (Continued)

Acceptance Criteria	Discussion
Consequential loss of system functionality	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, the water level in the RPV remains above the top of the core throughout the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient.
Radiological consequences	The radiological consequences acceptance criteria are evaluated by downstream radiological analysis using the mass release calculated in the non-LOCA transient analysis or are evaluated with bounding assumptions as described in Section 4.3.6.

7.2.18.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-86 are considered in identifying a bounding transient simulation for dose.

Table 7-86 Initial conditions, biases, and conservatisms - breaks in small lines carrying primary coolant outside containment

Parameter	Bias / Conservatism	Basis(1) {{
Initial reactor power	Varied.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Biased to the high condition	
Initial PZR level	Biased to the high condition.	
Initial feedwater temperature	Varied.	}}2(a),(c)

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Table 7-86 Initial conditions, biases, and conservatisms - breaks in small lines carrying primary coolant outside containment (Continued)

Parameter	Bias / Conservatism	Basis(1) {{
Initial fuel temperature	Varied.	
MTC	Varied.	
Kinetics	Nominal.	
Decay heat	Biased to the high condition.	
Initial SG pressure(2)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
CVCS volume outside containment	Biased to the high condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Disabled. Nominal.	
PZR Level Control Charging	 Not credited.	}}2(a),(c)

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Table 7-86 Initial conditions, biases, and conservatisms - breaks in small lines carrying primary coolant outside containment (Continued)

Parameter	Bias / Conservatism	Basis(1)
		{{
Letdown	Disabled.	
Steam Pressure Control Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{

}}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event.

For example, representative sensitivity studies for break location and size to determine impact on mass release and iodine spiking time are presented in Table 7-87 and Table 7-88, respectively. The Table 7-87 results indicate the mass release is maximized with a break in the letdown line of 100 percent area, AC power lost at event initiation, RCS average temperature biased high, and a break in the makeup line of 100 percent area coincident with reactor trip. The Table 7-88 results indicate the iodine spiking duration (elapsed time from reactor trip to CVCS isolation) is maximized with a break in the makeup line, AC power lost at event initiation, RCS average temperature biased high, and no break in the letdown line coincident with reactor trip.

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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

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}}2(a),(c)

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Table 7-88 Representative break, time in life, power, flow, and temperature sensitivity study for iodine spiking time - breaks in small lines carrying primary coolant outside containment

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}}2(a),(c)

7.2.19 Steam Generator Tube Failure

The methodology used to simulate a postulated failure of a steam generator tube for an NPM, and an evaluation of the acceptance criteria for an accident listed in Table 7-4, are presented below.

A postulated failure of a steam generator tube is typically only evaluated for radiological consequences. Neither the steam generator design nor the use of natural circulation flow for an NPM introduce a more challenging condition for other acceptance criteria.

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7.2.19.1 General Event Description and Methodology

The event is initiated by the failure of a steam generator tube that causes a decrease in pressurizer pressure and level. The rate of decrease for both parameters depends on the break location and size. The subsequent decrease in RCS pressure provides minimal core reactivity feedback from moderator feedback, so the reactor power remains relatively constant until reactor trip. In the absence of a loss of power at event initiation, the decreasing pressurizer pressure and level causes RTS actuation on the low pressurizer pressure signal or the low pressurizer level signal.

Regardless of the presence of a reactor trip, the sustained loss of reactor coolant from the failed tube causes a continuous decrease in pressurizer pressure and level. Eventually an MPS low pressurizer pressure or level signal is generated. Depending on the design, one or both of these MPS signals actuates DHRS and closure of the FWIVs and MSIVs isolates the steam generator, which also terminates the loss of reactor coolant from the failed tube to the environment. Core decay heat drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient ends with the NPM in a safe, stable condition.

When determining the magnitude of mass released for a steam generator tube failure (SGTF) event the timing of SG isolation is critical, since SG isolation terminates the release of mass from the RPV to other plant areas. Maximizing the mass released to the environment and the duration of the iodine spike (elapsed time from reactor trip to steam generator isolation) maximizes the radiological consequences. In cases where SG isolation occurs on the same signal as RTS, spiking time is restricted to only the time required for valves to close. Section 4.3.6 describes an example approach for performing the radiological consequence analysis using bounding inputs for the mass release and isolation time. If a bounding approach is used, then system transient results for mass release and isolation time are not required for the downstream radiological analysis. System transient analysis can be used to confirm the radiological analysis assumptions are bounding, if necessary.

Calculations are performed to assess the responses to a spectrum of break sizes. Specifically, each break is modelled as either a double-ended guillotine rupture, or a smaller size break, of the steam generator tube. {{

}}[2(a),(c)] The results of sensitivity studies on break type and location indicate a rupture of the steam generator tube at the top of the steam generator typically provides the greatest integrated mass released and the longest spiking time. While the rate of depressurization declines proportionally as the break area is reduced, the rate of decrease after reactor trip for both the pressurizer pressure and level is nearly independent of break flow, and any reduction in pressure reduces the mass released through the break.

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An SGTF event initiated from HFP leads to a higher break flow, and thus a higher integrated mass released, because the pressure difference between the primary and secondary system is larger for longer. Similarly, an SGTF event initiated from HFP typically leads to a longer spiking time because the stored energy of the core is greater.

Table 7-89 lists the relevant acceptance criteria, single active failure, and loss of power scenarios.

Table 7-89 Acceptance criteria, single active failure, loss of power scenarios - steam generator tube failure

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Radiological consequences	A postulated failure of a steam generator tube is evaluated for radiological consequences.
Failure of primary MSIV to close on affected SG	The primary MSIV is typically assumed to fail to isolate to maximize the radiological consequences. Delaying isolation until closure of the secondary MSIV leads to additional mass released to the environment and a longer iodine spike duration.
No loss of power	A loss of AC power at event initiation or coincident with turbine trip typically results in less integrated mass released and a shorter iodine spiking duration.

7.2.19.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-90.

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Table 7-90 Acceptance criteria - steam generator tube failure

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of the steam generator tube failure event, sensitivities that maximize primary pressure are not analyzed. However, sensitivities were performed to demonstrate that, with the exception of those cases that involve a loss of power at event initiation, the peak primary pressure experienced during this event does not vary significantly from the pressure at event initiation. In all instances (i.e., with and without a loss of AC power at event initiation) the peak primary pressure remains below the design pressure, thereby providing margin to the acceptance criterion. If a system transient analysis is not required as input for the radiological consequence analysis, maximum primary pressure can be selected as a relevant acceptance criterion to address with the system transient analysis.
Secondary pressure	Pressure in the portion of the secondary system between the FWIVs and MSIVs increases rapidly post-DHRS actuation for both SGs, and the failed SG tube allows the pressure in the affected SG to approach RCS pressure conditions. In all instances, i.e., with and without a loss of AC power at event initiation, the peak secondary system pressure remains below the design pressure, thereby providing margin to the acceptance criterion. If a system transient analysis is not required as input for the radiological consequence analysis, maximum secondary pressure can be selected as a relevant acceptance criterion to address with the system transient analysis.
Fuel cladding integrity	Fuel failure directly relates to dose by dictating the isotopic concentration of the reactor coolant being released. Consequently, the MCHFR and fuel centerline temperature criteria are relevant to determining the appropriate source term for the downstream radiological analysis. The SGTF is a slow depressurization event that does not result in an increased core power or significantly reduced core flow prior to reactor trip. After considering the NuScale CHF trend with pressure and the ranges bounded by steady-state subchannel analysis, the CHFR is not challenged during the transient prior to reactor trip. Therefore, the fuel cladding integrity acceptance criteria are not challenged and event-specific follow-on MCHFR and fuel centerline temperature evaluations are not necessary.

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Table 7-90 Acceptance criteria - steam generator tube failure (Continued)

Acceptance Criteria	Discussion
Containment integrity	Containment integrity is evaluated in a separate analysis methodology.
Consequential loss of system functionality	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, the water level in the RPV remains above the top of the core throughout the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient.
Radiological consequences	The radiological consequences acceptance criteria are evaluated by downstream radiological analysis using the mass release calculated in the non-LOCA transient analysis or are evaluated with bounding assumptions as described in Section 4.3.6.

7.2.19.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-91 are considered in identifying a bounding transient simulation for dose.

Table 7-91 Initial conditions, biases, and conservatisms - steam generator tube failure

Parameter	Bias / Conservatism	Basis (1) {{
Initial reactor power	RTP biased to the high condition.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Biased to the high condition	
Initial PZR level	Biased to the high condition.	}}2(a),(c)

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Table 7-91 Initial conditions, biases, and conservatisms - steam generator tube failure (Continued)

Parameter	Bias / Conservatism	Basis (1) {{
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased to the high condition.	
MTC	Varied.	
Kinetics	Varied.	
Decay heat	Biased to the high condition.	
Initial SG pressure(2)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Varied.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	}}2(a),(c)

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Table 7-91 Initial conditions, biases, and conservatisms - steam generator tube failure (Continued)

Parameter	Bias / Conservatism	Basis (1)
PZR Pressure Control		{{
PZR spray	Disabled.	
PZR heaters	Nominal.	
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control		
Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control		
feedwater pump speed	Enabled.	
CNV Pressure Control		
CNV evacuation system	Enabled.	}}2(a),(c)

{{

}}2(a),(c)

Sensitivity studies are performed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event.

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For example, representative sensitivity studies for break characteristics, core time-in-life, SG pressure, SG tube plugging, loss of power, single failure, feedwater temperature and reactor coolant temperature for a steam generator tube failure with respect to mass release and iodine spiking duration are presented in Table 7-92. The results indicate the mass release and spiking time are maximized with a double-ended guillotine break at the top of the SG tube, AC power available for the event duration, and a single active failure of the primary MSIV on the affected SG to close.

Table 7-92 Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure

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}}2(a),(c)

Table 7-92 Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure (Continued)

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}}2(a),(c)

8.0 Representative Calculations

The methodology of Chapter 7 is utilized in conjunction with the NRELAP5 model of Chapter 6 to provide representative transient results. The calculations are for a representative NPM design, consistent with the design characteristics in Chapter 3. Riser holes are not modeled in the representative calculations. The transients noted below were selected to demonstrate the application of the NuScale Non-LOCA methodology for analysis of the representative plant responses to a wide range of postulated equipment failures and malfunctions.

1. Cooldown and/or Depressurization of the RCS (Section 8.1)
2. Heatup and/or Pressurization of the RCS (Section 8.2)
3. Reactivity Anomaly (Section 8.3)
4. Increase in RCS Inventory (Section 8.4)
5. Decrease in RCS Inventory (Section 8.5)

The information included for each representative transient includes: an event description; the results for the acceptance criteria of interest; and, conclusions regarding the acceptance criteria of interest. These results are presented to demonstrate the application of the non-LOCA methodology to a representative NPM. Fuel rod and core physics parameter inputs for the representative transients were developed using COPERNIC (Reference 22) and SIMULATE5 (Reference 23) respectively.

8.1 Cooldown and/or Depressurization of the Reactor Coolant System

8.1.1 Decrease in Feedwater Temperature

The purpose of this section is to present the thermal-hydraulic response of a representative NPM for a decrease in feedwater temperature event. This event is evaluated for MCHFR.

8.1.1.1 Event Description

The general decrease in feedwater temperature (DFWT) event description can be found in Section 7.2.1.1. Based on Section 7.2.1.1, MCHFR is the only acceptance criterion that may be potentially challenged during the DFWT event. No single failure is applied since the challenging cases occur when all equipment operates as designed. No loss of power is applied since all loss of power scenarios terminate feedwater or trip the reactor, thus reducing the overcooling event. Chosen from a series of MCHFR sensitivity cases, the representative DFWT case presented here represents a case that could challenge MCHFR, based on the NRELAP5 MCHFR pre-screening. This case features the following conditions:

- Conservative initial condition biasing (as shown in Table 7-7) is applied in order to maximize the consequences of the overcooling event in terms of MCHFR. This representative case is initialized at 102 percent reactor power.

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RCS average temperature is biased at high condition (555 degrees F). RCS flow rated is biased to the low condition (535 kg/s). Pressurizer pressure is biased to the high condition (1920 psia). Pressurizer level is biased to the high condition (53 percent). Initial feedwater temperature is biased to the high condition (307.5 degrees F).

- Transient feedwater temperature decreases linearly from 307.5 degrees F at the rate of 1.18 degrees F per second which minimize the MCHFR based on the MCHFR pre-screening process.
- Feedwater flow is maintained at a constant volumetric flow rate. This prevents the controller from reducing feedwater flow in response to decreasing feedwater temperature which maximizes RCS overcooling.
- Turbine trip on reactor trip and subsequent operation of the turbine bypass system is not credited in this case, rather the turbine boundary is conservatively held constant at the pre-reactor trip condition in order to maximize the overcooling of the RCS.
- EOC reactivity coefficients are applied which maximizes the reactor power response.
- A low fuel temperature bias (applied by increasing gap conductance) is applied in this case although a nominal temperature is acceptable.
- No operator action was credited in the representative case. Normal control system such as PZR spray, heater, letdown controls and automatic rod control are modeled based on the control status shown in Table 7-7.

8.1.1.2 Analysis Results

The following describes the event sequence of the representative DFWT event. Table 8-1 summarizes the sequence of events. Figure 8-1 through Figure 8-9 show some key parameters during the representative DFWT event.

The DFWT event begins at time zero. The hydraulic feedwater source boundary condition is linearly changed from a constant temperature at 307.5 degrees F at a rate of 1.18 degrees F per second (Figure 8-1). Since the feedwater flow control is held at a constant volumetric pump rate, decreasing feedwater temperature means feedwater density and mass flow rate increases. The RCS response to the overcooling event begins once the cold feedwater front propagates through the secondary system piping and reaches the SG.

As heat removal from the RCS through the SG increases above its steady state value, downcomer temperature begins to decrease. The drop in average RCS temperature prompts the control rod controller to begin pulling the regulating bank out of the core at ~28 seconds. The addition of positive reactivity causes reactor power to increase. This increase in power is significant enough that moderator feedback stays slightly negative in response to the core heatup. Power and RCS riser temperature continue to rise until the high RCS riser temperature analytical limit (610 degrees F) is reached at ~139 seconds (Figure 8-2 and Figure 8-3).

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Because there is a total 8 second delay between the high RCS riser temperature signal and reactor scram, power continues to rise until reaching the high power limit (125 percent RTP) at ~145 seconds; there is a 2 second delay between the high power signal and reactor scram. Both the high power signal and the high RCS riser temperature signal initiate reactor scram at ~147 seconds, which terminates the power excursion. Power peaks at 200.6 MWth before reactor scram.

The high RCS riser temperature signal also initiates DHRS actuation at the same time as reactor scrams. This closes the FWIVs and MSIVs, which isolate the SG from the remaining secondary system and ends the overcooling transient. Steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the decrease in feedwater temperature event itself. SG pressure starts to decrease once the DHRS cooling is established (Figure 8-4). RCS pressure and level have an overall decreasing trend as inventory shrinks due to increased heat removal during the overcooling transient. Once the PZR level is lower than the low PZR level analytical limit (35 percent) at ~725 seconds, PZR heaters are disabled and RCS pressure continues to decrease after that (Figure 8-5 and Figure 8-6).

After reactor trip and actuation of DHRS, RCS flow decreases rapidly and becomes stagnant and slightly reversed at ~180 seconds (Figure 8-7). Following that, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2 and shown in Figure 8-3, Figure 8-7, and Figure 8-8); therefore the calculation is continued to verify that the module transitions into passive and stable DHRS cooling. At ~30 minutes, RCS flow has stabilized, and the RCS temperature and pressure are steadily decreasing as the DHRS transfers decay heat from the RPV to the reactor pool (Figure 8-3, Figure 8-5, Figure 8-7, and Figure 8-8). It is concluded that by 30 minutes the overcooling transient has been terminated and that stable DHRS cooling has been achieved. Subcritical margin is verified as net reactivity remains less than 0.0 dollars at the time stable DHRS cooling has been achieved (Figure 8-9). No operator action was credited to mitigate this event.

Table 8-1 Decrease in feedwater temperature sequence of events

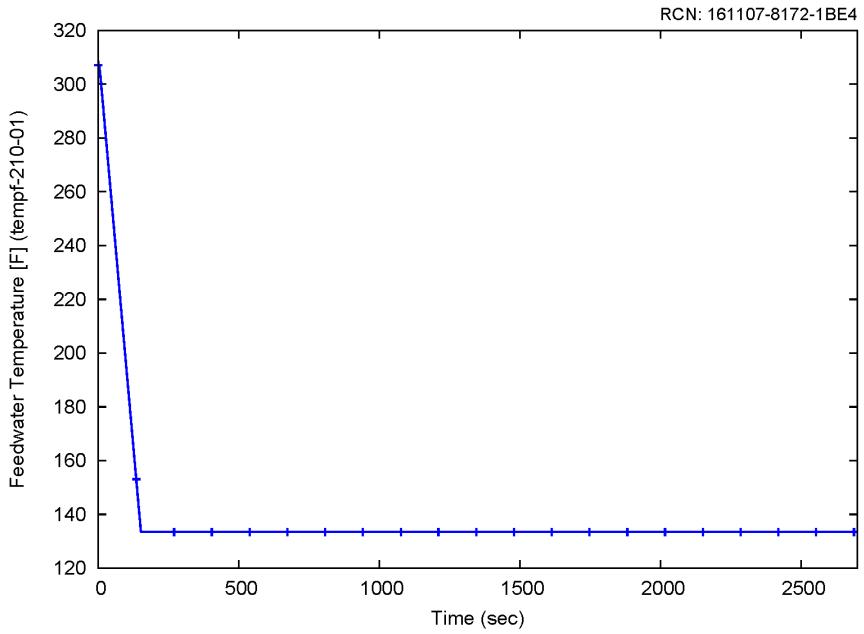
Event	Time (sec)
Malfunction that initiates the decrease in feedwater temperature event. Feedwater temperature is linearly decreased from 307.5°F at a rate of 1.18°F per second.	0
Cold water front reaches core inlet. Regulating bank begins to withdraw in response to a decrease in average RCS temperature. Reactor power begins to rise.	28
High RCS riser temperature limit is reached (610°F). Control rod insertion begins after 8 second delay.	139
Peak RPV pressure is reached (1951 psia).	145
High reactor power limit is reached (125% RTP). Control rod insertion begins after 2 second delay.	145
Peak reactor power is reached (200.6 MW).	147

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Table 8-1 Decrease in feedwater temperature sequence of events (Continued)

Event	Time (sec)
RTS actuation on both high power and high RCS riser temperature signals, control rods begin to insert into the core.	147
DHRS actuation on the high RCS riser temperature signal. DHRS actuation valves open immediately.	147
Limiting MCHFR is reached (2.591 as calculated by NRELAP5).	148
FWIVs and MSIVs are fully closed. Feedwater flow stops.	152
RCS flow is stagnant and slightly reversed	~ 180
Peak SG pressure is reached (1463 psia).	193
Low PZR level limit is reached (35%). PZR heaters are disabled after 3 second delay.	725
Low low PZR level is reached (20%). Containment and CVCS isolation begins after 3 second delay.	1767
Establishment of stable RCS flow. Pressure and temperature are steadily decreasing.	1800
End of calculation. Stable DHRS cooling has been established. Net reactivity remains < $0.0.	2700

Figure 8-1 Temperature of feedwater during the representative decrease in feedwater temperature event



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Figure 8-2 Power response for the representative decrease in feedwater temperature event

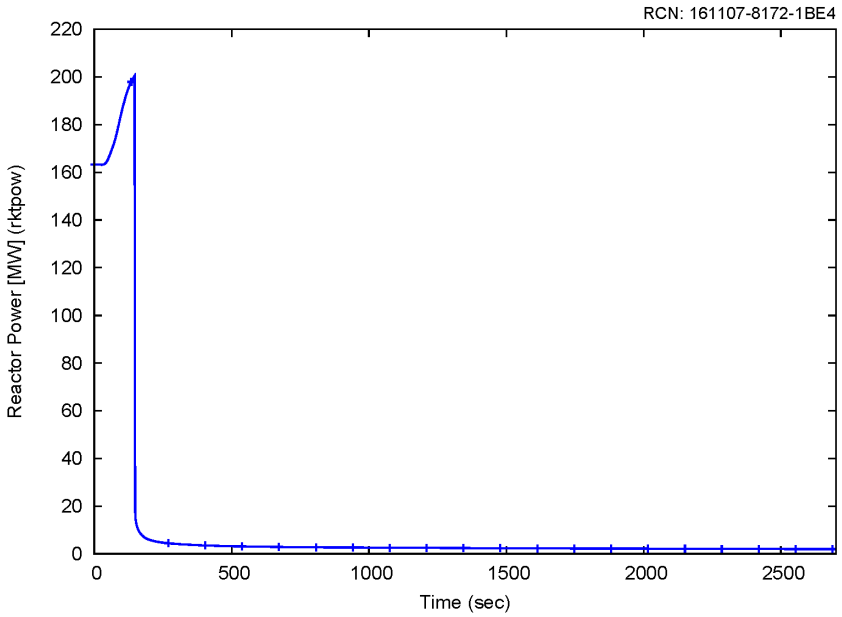
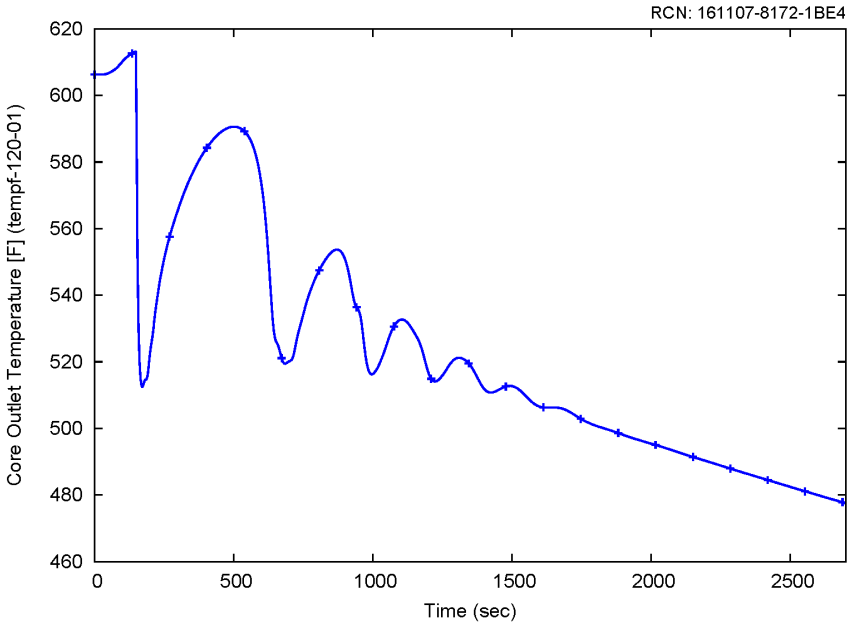


Figure 8-3 Core outlet temperature for the representative decrease in feedwater temperature event



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Figure 8-4 Steam generator 2 pressure response for the representative decrease in feedwater temperature event

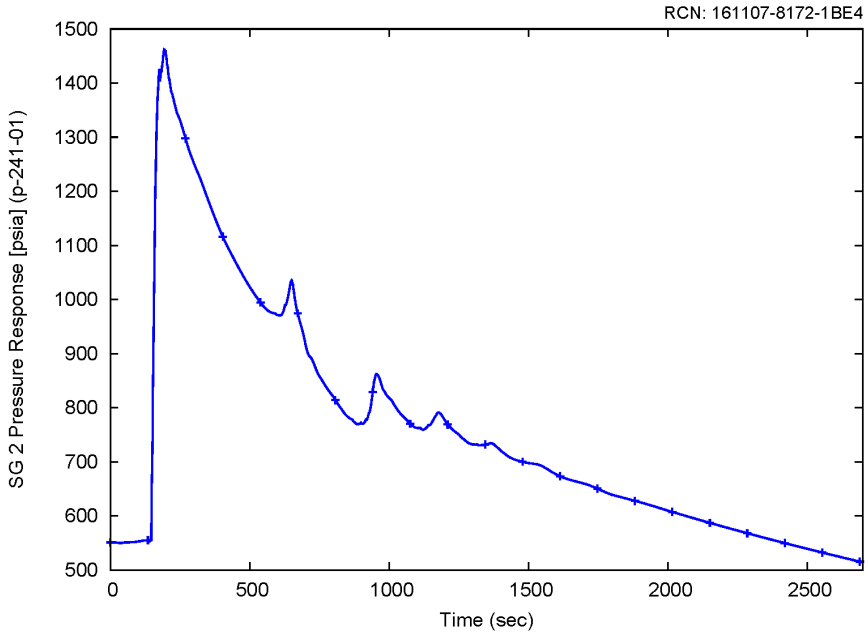
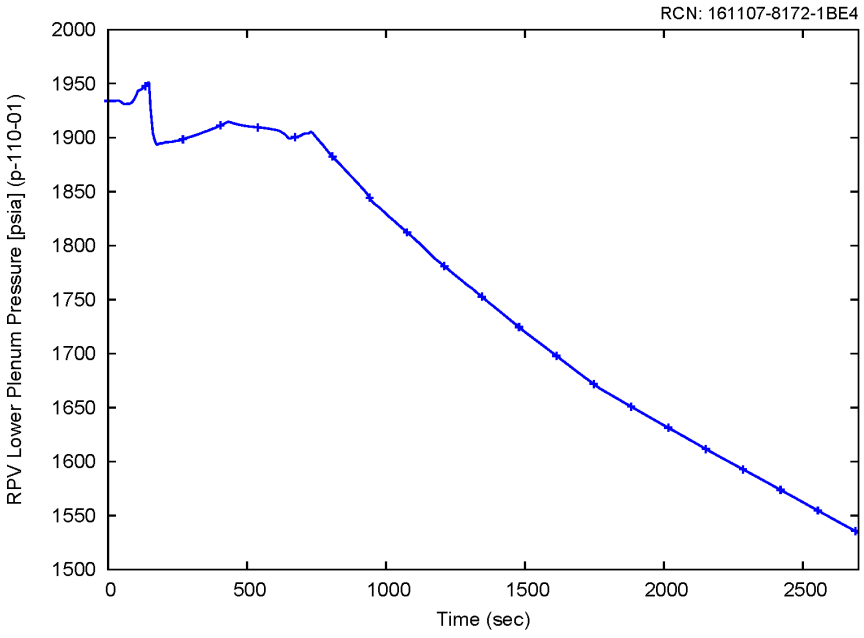


Figure 8-5 Reactor pressure vessel pressure response for the representative decrease in feedwater temperature event



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Figure 8-6 Pressurizer level for the representative decrease in feedwater temperature event

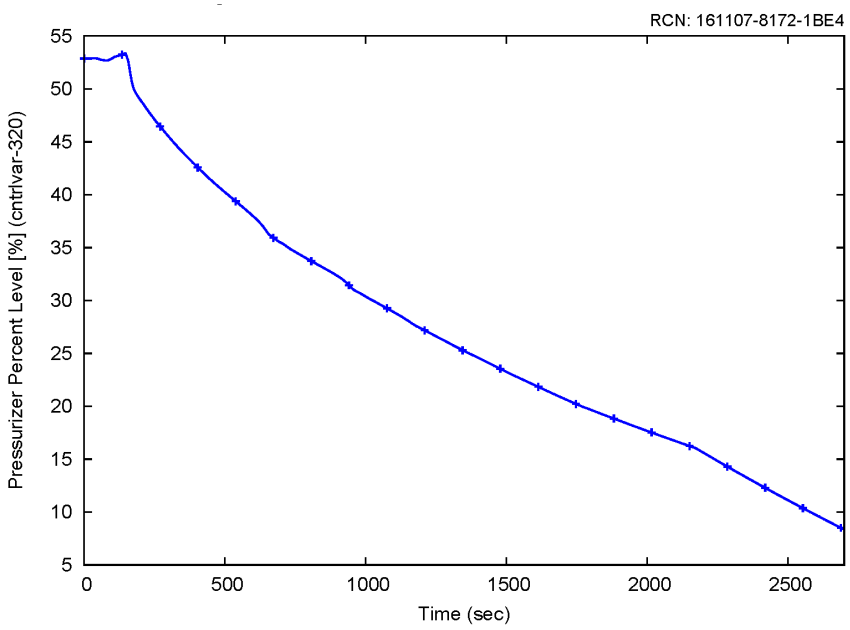
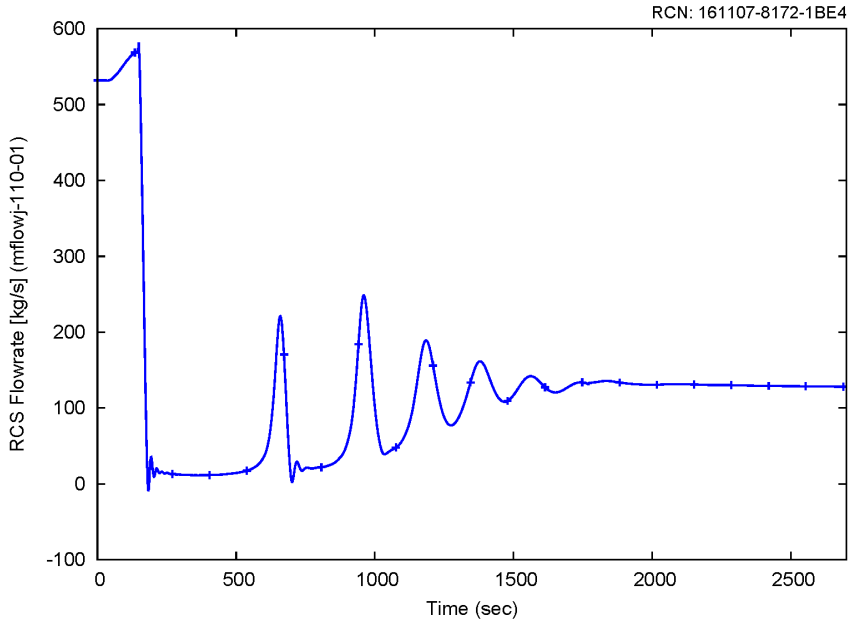


Figure 8-7 Reactor coolant system flow rate for the representative decrease in feedwater temperature event



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Figure 8-8 Core inlet temperature for the representative decrease in feedwater temperature event

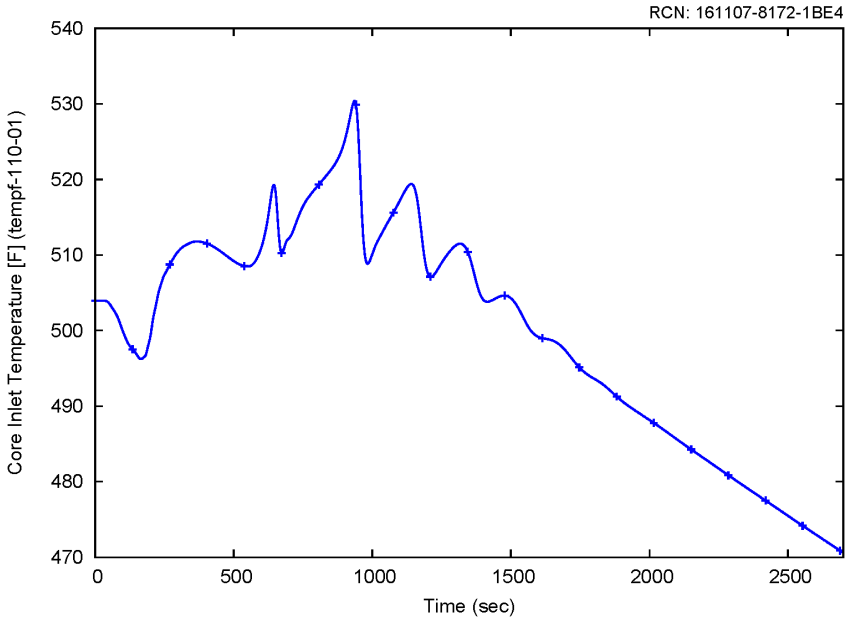
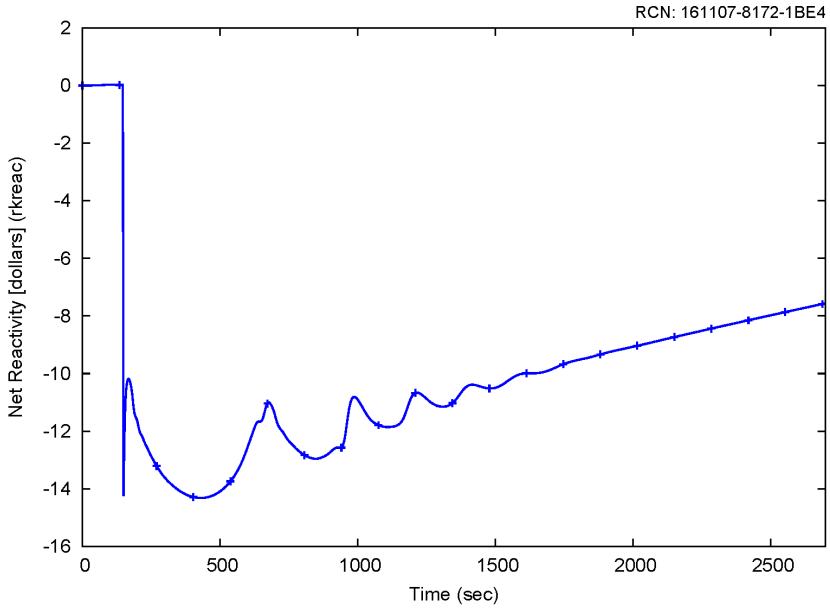


Figure 8-9 Net reactivity for the representative decrease in feedwater temperature event



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8.1.1.3 Conclusion

A representative case that could challenge MCHFR was identified for a decrease in feedwater temperature event. The results of this case, as presented in Section 8.1.1.2, are subsequently used as input to an MCHFR evaluation using the NuScale subchannel analysis methodology.

8.1.2 Increase in Steam Flow

The purpose of this section is to present the thermal-hydraulic response of a representative NPM for an increase in steam flow event. This event is evaluated for MCHFR.

8.1.2.1 Event Description

The general increase in steam flow event description can be found from Section 7.2.3.1. Based on Section 7.2.3.1, MCHFR is the only acceptance criterion that may be potentially challenged during the increase in steam flow event. No single failure is applied since the challenging cases occur when all equipment operates as designed. No loss of power is applied since all loss of power scenarios terminate feedwater or trip the reactor, thus reducing the overcooling event. Chosen from a series of MCHFR sensitivity cases, the representative increase in steam flow case presented here represents a case that could challenge MCHFR, based on the NRELAP5 MCHFR pre-screening. This case features the following conditions:

- Conservative initial condition biasing (as shown in Table 7-19) is applied in order to maximize the consequences of the overcooling event in terms of MCHFR. This representative case is initialized at 102 percent reactor power. RCS average temperature is biased at high condition (555 degrees F). RCS flow rated is biased to the low condition (535 kg/s). Pressurizer pressure is biased to the high condition (1920 psia). Pressurizer level is biased to the high condition (53 percent).
- SG heat transfer is decreased 30 percent by applying a heat transfer coefficient multiplier of 0.7 in the steady state initialization model. As identified in Section 7.2.3.3, this biasing has insignificant impact on the overall limiting MCHFR conditions for the transient.
- Steam flow is increased 14.45 percent instantly at the beginning of the event. A time-dependent junction that controls steam mass flow rate is used to model the turbine.
- During the increase in steam flow event, the feedwater pump speed remains constant and the pump curve allows a 1.0 lbm/s increase in feedwater flow for every 1 psi decrease in SG pressure. This maximizes the overcooling event by increasing the available source of secondary coolant.
- EOC reactivity coefficients are applied which maximizes the reactor power response.

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- A low fuel temperature bias (applied by increasing gap conductance) is applied in this case although a nominal temperature is acceptable.
- No operator action was credited in the representative case. Normal control system such as PZR spray, heater, letdown controls and automatic rod control are modeled based on the control status shown in Table 7-19.

8.1.2.2 Analysis Results

The following describes the event sequence of the representative increase in steam flow event. Table 8-2 summarizes the sequence of events. Figure 8-10 through Figure 8-19 show some key parameters during the representative increase in steam flow event.

The increase in steam flow event begins at time zero, as can be seen in Figure 8-10. The main steam flow rate at the secondary system hydraulic exit boundary is increased by 14.45 percent. Figure 8-11 shows that decreasing pressure in the secondary system also causes an increase in feedwater pump flow due to the pump curve, as shown in Figure 8-12. The RCS response to the overcooling event begins once steam flow through the steam generators starts to increase.

As heat removal from the RCS through the SG increases above its steady state value, downcomer temperature begins to decrease (Figure 8-13). The drop in average RCS temperature prompts the CR controller to pull the regulating bank out of the core at ~5 seconds. The addition of positive reactivity causes reactor power to increase, as seen in Figure 8-14. Power continues to rise until peaking at ~58 seconds. Peak power is calculated to be 199.97 MW, which is slightly lower than the high power analytical limit (200 MW or 125 percent RTP - increased from the analytical limit of 120 percent to conservatively account for the decalibration of the excore neutron detectors as downcomer density increases in response to an overcooling event). Therefore the representative increase in steam flow event is not tripped on high power. Soon after, the limiting MCHFR is reached at ~62 seconds. Figure 8-15 shows the RCS riser temperature, which continues to rise until the high RCS riser temperature limit (610 degrees F) is reached at ~68 seconds. There is a total of 8 second delay between the high RCS riser temperature signal and RTS/DHRS actuation. Once RTS and DHRS actuate, the subsequent reactor scram and steam generator isolation terminate the overcooling event at ~76 seconds.

DHRS actuation closes the FWIVs and MSIVs, which isolate the SG from the remaining secondary system and ends the overcooling transient. Steam generator pressure increase resulting from MS isolation is expected and is not a direct consequence of the increase in steam flow event itself. SG pressure starts to decrease once the DHRS cooling is established (Figure 8-11). RCS pressure and level have an overall decreasing trend as inventory shrinks due to increased heat removal during the overcooling transient. Once the PZR level is lower than the low

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PZR level analytical limit (35 percent) at ~670 seconds, PZR heaters are disabled and RCS pressure continues to decrease after that (Figure 8-16 and Figure 8-17).

After reactor trip and actuation of DHRS, RCS flow decreases rapidly and becomes stagnant and slightly reversed at ~112 seconds (Figure 8-18). Following that, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2 and shown in Figure 8-13, Figure 8-15, and Figure 8-18); therefore the calculation is continued to verify that the module transitions into passive and stable DHRS cooling. By 40 minutes, RCS flow has stabilized, and the RCS temperature and pressure are steadily decreasing as the DHRS transfers decay heat from the RCS to the reactor pool (Figure 8-13, Figure 8-15, Figure 8-16 and Figure 8-18). It is concluded that by 40 minutes the transient has been terminated and that stable DHRS cooling has been achieved. Subcritical margin is verified as net reactivity remains less than 0.0 dollars at the time stable DHRS cooling has been achieved (Figure 8-19). No operator action was credited to mitigate this event.

Table 8-2 Increase in steam flow sequence of events

Event	Time (sec)
An instant increase of 14.45% is applied to the steam flow to model the spurious opening of the turbine bypass valve or the main steam safety valve	0
Cold water front reaches core inlet. Regulating bank begins to withdraw in response to a decrease in average RCS temperature. Reactor power begins to rise.	5
Peak reactor power is reached (199.97 MW).	58
Limiting MCHFR is reached (3.570 as calculated by NRELAP5).	62
High RCS riser temperature limit is reached (610°F).	68
Actuation of the RTS. Control rods begin to insert into the core.	76
Actuation of the DHRS. The DHRS actuation valves open immediately. The FWIVs and MSIVs begin closing.	76
Peak RPV pressure is reached (1991 psia).	77
FWIVs and MSIVs are fully closed. Steam generator isolation from the remaining secondary system terminates the overcooling transient.	81
RCS flow is stagnant and slightly reversed.	112
Peak steam generator pressure is reached (1248 psia).	128
Low PZR level limit is reached (35%). PZR heaters are disabled after 3 second delay.	670
Low low PZR level is reached (20%). Containment and CVCS isolation begins after a 5 second delay.	1620
End of calculation. Stable DHRS cooling has been established. Net reactivity remains < $0.0.	2400

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Figure 8-10 Main steam transient flow rate during the representative increase in steam flow event

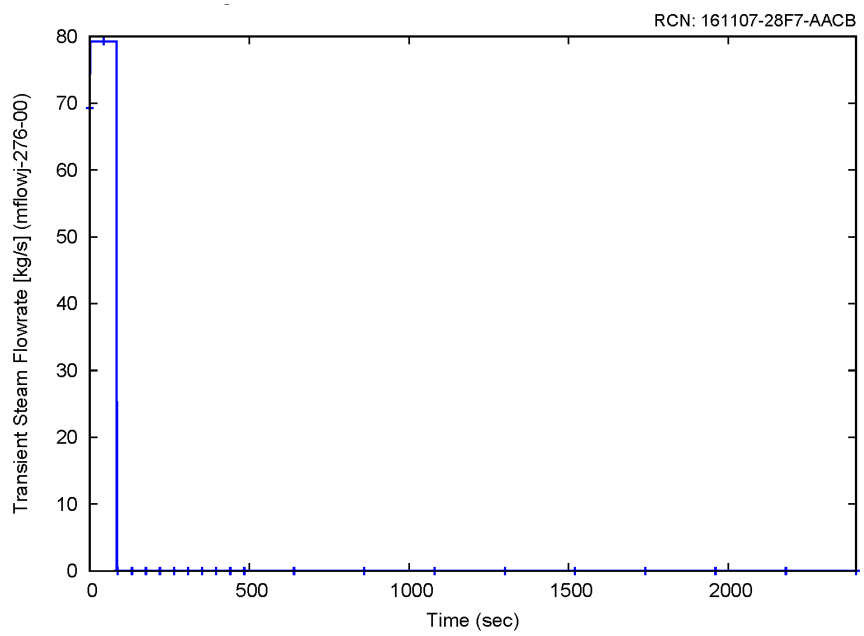
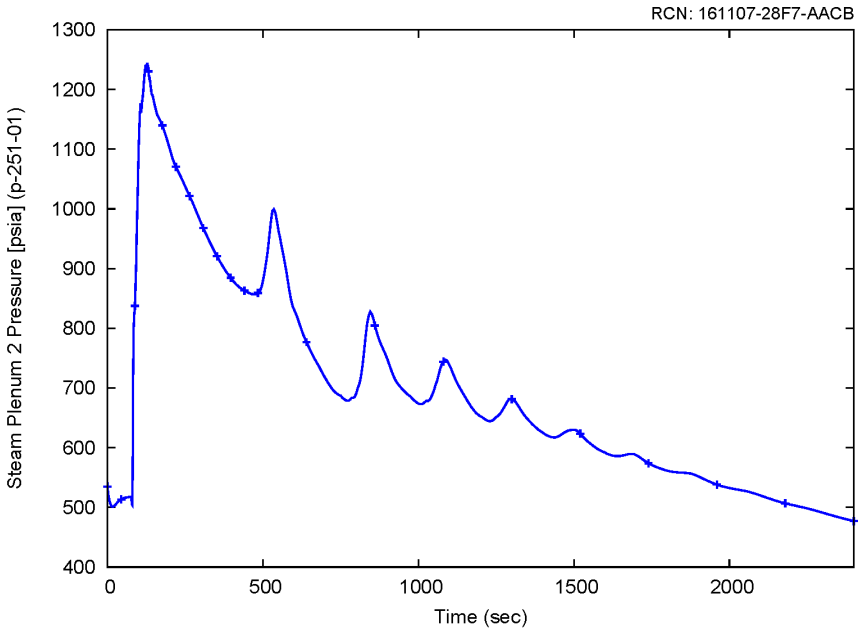


Figure 8-11 Steam generator 2 pressure response for the representative increase in steam flow event



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Figure 8-12 Steam generator 2 secondary side flow for the representative increase in steam flow event

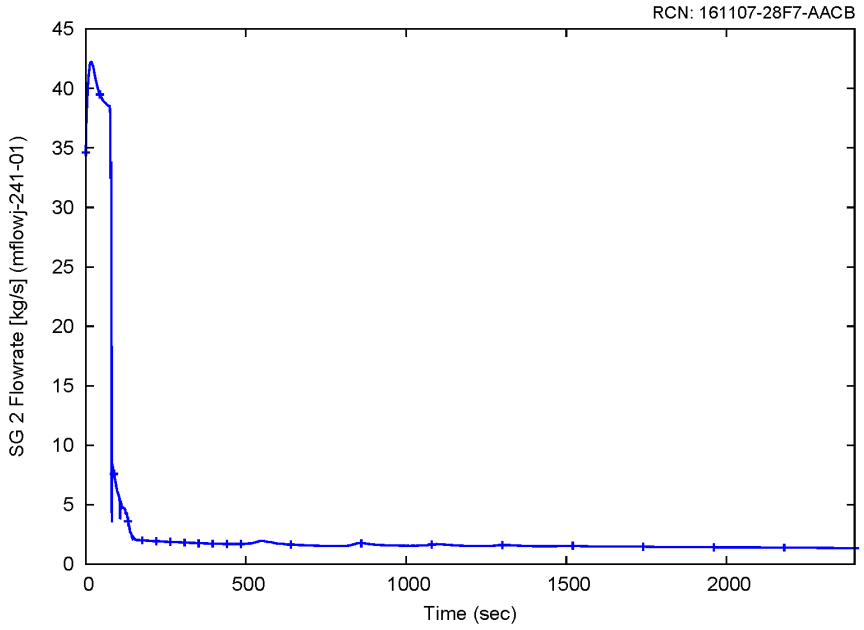
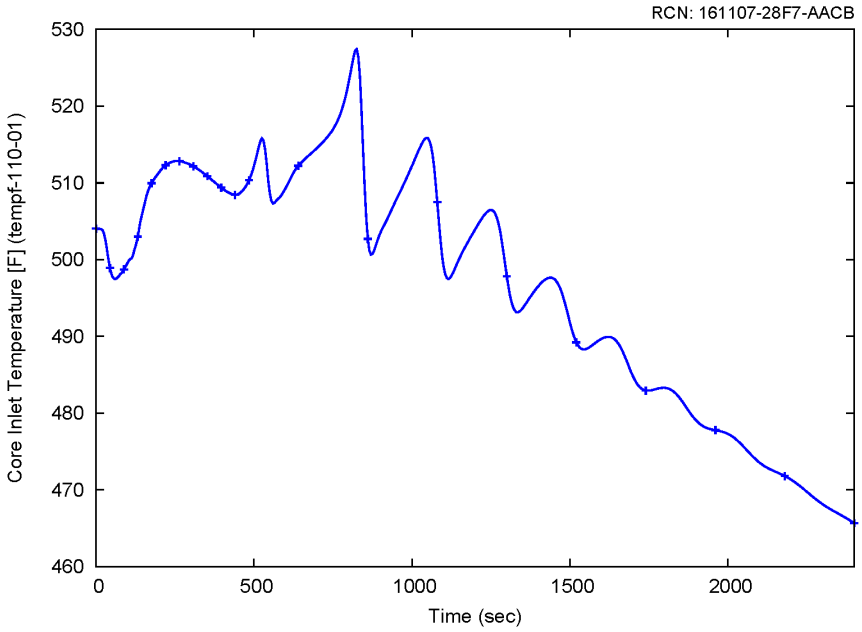


Figure 8-13 Core inlet temperature for the representative increase in steam flow event



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Figure 8-14 Power response for the representative increase in steam flow event

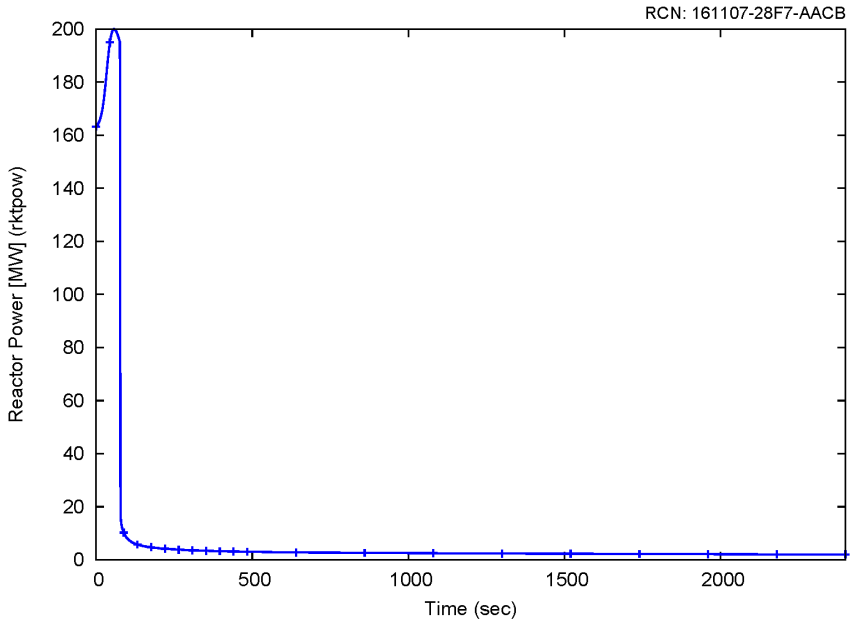
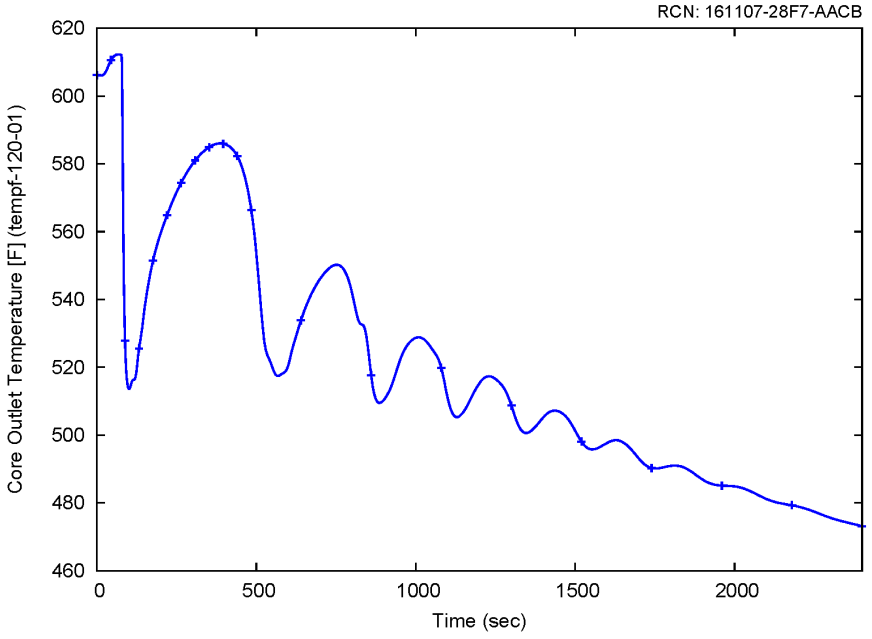


Figure 8-15 Core outlet temperature for the representative increase in steam flow event



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Figure 8-16 Reactor pressure vessel pressure response for the representative increase in steam flow event

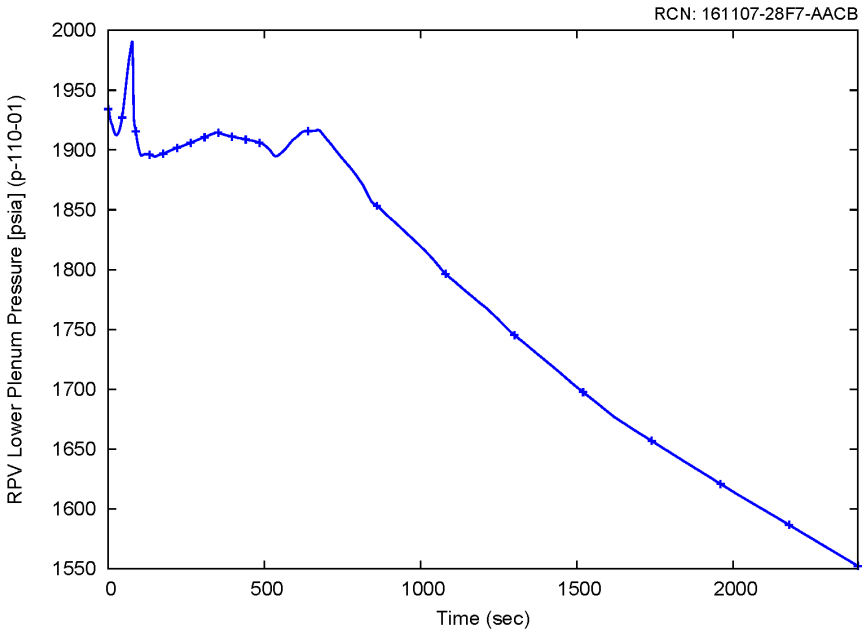
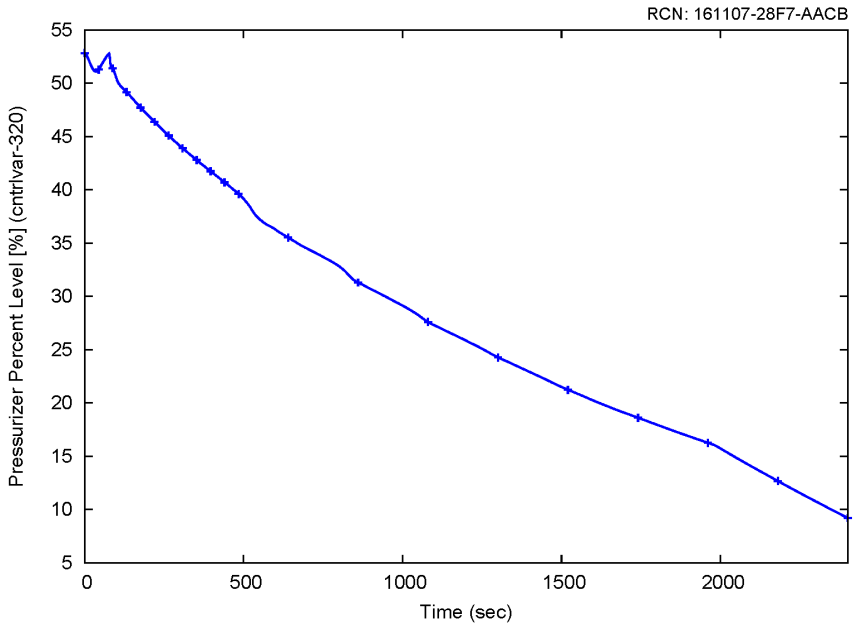


Figure 8-17 Pressurizer level for the representative increase in steam flow event



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Figure 8-18 Reactor coolant system flow rate for the representative increase in steam flow event

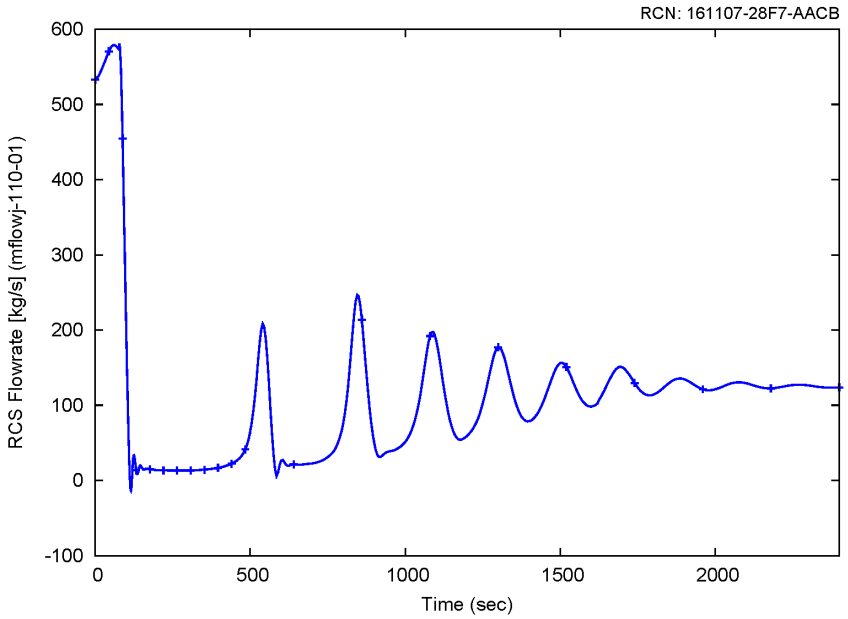
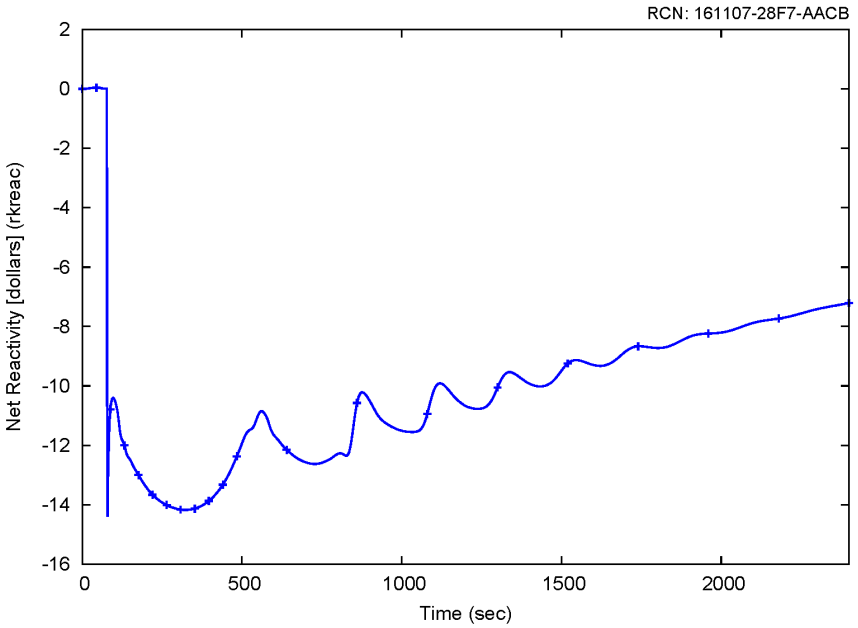


Figure 8-19 Net reactivity for the representative increase in steam flow event



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8.1.2.3 Conclusion

A representative case that could challenge MCHFR was identified for an increase in steam flow event. The results of this case, as presented in Section 8.1.2.2, are subsequently used as input to an MCHFR evaluation using the NuScale subchannel analysis methodology.

8.1.3 Main Steam Line Break

The purpose of this section is to present the thermal-hydraulic response of a representative NPM for a main steam line break event. This event is evaluated for MCHFR, and mass releases are determined for input to downstream accident radiological dose analysis. A representative case evaluated for MCHFR is presented.

8.1.3.1 Event Description

The general description for main steam line break event can be found from Section 7.2.4.1. Based on Section 7.2.4.1, MCHFR is the only acceptance criterion that may be potentially challenged during the main steam line break event. The MCHFR case assumes a failed MSIV on the affected SG train; however, the timing of MCHFR is well before the MSIV would have closed so it is concluded there is no limiting failure for the MCHFR case. No loss of power is applied since all loss of power scenarios terminate feedwater or trip the reactor, thus reducing the overcooling event. Chosen from a series of MCHFR sensitivity cases, the representative main steam line break case presented here represents a case that could challenge MCHFR, based on the NRELAP5 MCHFR pre-screening. This case features the following conditions:

- Conservative initial condition biasing (as shown in Table 7-24) is applied in order to maximize the consequences of the overcooling event in terms of MCHFR. The case is initialized at 102 percent reactor power with conservatively high RCS temperature and pressure. RCS average temperature is biased at high condition (555 degrees F). RCS flow rated is biased to the low condition (535 kg/s). Pressurizer pressure is biased to the high condition (1920 psia). Pressurizer level is biased to the high condition (58 percent). Feedwater temperature is biased to the high condition (310 degrees F). Minimum RCS design flow is assumed.
- SG heat transfer is increased 30 percent by applying a heat transfer coefficient multiplier of 1.3 in the steady state initialization model. As identified in Section 7.2.4.3, this biasing has insignificant impact on the overall limiting MCHFR conditions, spiking time or mass released. There is no tube plugging in the SGs for this representative calculation.
- The feedwater controller is based on FW pressure error rather than turbine load demand. This allows for the implementation of the feedwater pump flow response to the pressure loss events. In the steam line piping failure transient, the details of how the pumps speed controller will respond are ignored and

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bounded by a conservative pump curve to maximize the flow response due to the drop in FW pressure.

- Steam pressure control is a flow based controller rather than a back pressure control. This is a better physical representation of the turbine response to a drop in steam pressure. During the transients the turbine is treated as a constant steam flow sink without automated runback.
- A low fuel temperature bias (applied via increase gap conductance) is applied which minimizes negative Doppler feedback following an increase in power.
- EOC reactivity coefficients are applied which maximizes the reactor power response.
- No operator action was credited in the representative case. Normal control system such as PZR spray, heater, letdown controls and automatic rod control are modeled based on the control status shown in Table 7-24.

8.1.3.2 Analysis Results - MCHFR Case

As discussed in Section 7.2.4.1, in an NPM design the smaller breaks can result in a delayed detection time compared to larger breaks and be more challenging for MCHFR. The MCHFR case is identified as a small (3.3 percent of the pipe cross section area) split break in the MS piping just outside of containment.

The following describes the event sequence of the MCHFR case for the steam line break event. Table 8-3 summarizes the sequence of events. Figure 8-20 through Figure 8-28 show some key parameters during the event.

The main steam line break event begins at time zero. Steam flow increases following the initiation of the break, as seen in Figure 8-20. As heat removal from the RCS through the SG increases above its steady state value, RCS temperature begins to decrease (Figure 8-21). The drop in average RCS temperature prompts the CR controller to pull the regulating bank out of the core. The addition of positive reactivity causes reactor power to increase (Figure 8-22). At ~47 seconds, the high power analytical limit (200 MWth or 125 percent RTP) is reached (Figure 8-23). At ~49 seconds, power is peaked at 202.8 MWth when reactor starts to scram. Approximately at the same time, the limiting MCHFR calculated by NRELAP5 is reached.

SG pressure decreases during the initial phase of the transient due to the break. As reactor power increases the SG pressure starts to increase and reaches the analytical limit of 800 psia at ~59 seconds (Figure 8-24). This is the time when DHRS is actuated. DHRS actuation closes the FWIVs and MSIVs, which isolate the SGs and the break from the remaining secondary system and ends the overcooling transient. Steam generator pressure increase resulting from MS isolation is expected. SG pressure starts to decrease once the DHRS cooling is established. Because of the single failure of the MSIV on the affected SG (failure to close), the affected SG and the associated DHRS train are depleted through

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the break after DHRS actuation. The unaffected SG and the associated DHRS train remove the decay heat (Figure 8-20 and Figure 8-24).

RCS level has an overall decreasing trend as inventory shrinks due to increased heat removal due to steam line break (Figure 8-25). The PZR level is still above the low PZR level analytical limit (35 percent) by the end of the transient so the PZR heaters can still function.

The RCS pressure initially drops as inventory shrinks due to increased heat removal. As reactor power increases and as the PZR heaters respond, an increase in RPV pressure to ~1997 psia is observed. When the reactor is tripped, the RCS pressure has a sudden drop and then gradually returns to its setpoint, using pressurizer heaters (Figure 8-26).

After reactor trip and actuation of DHRS, RCS flow decreases rapidly and becomes stagnant and slightly reversed at ~85 seconds (Figure 8-27). Following that, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2 and shown in Figure 8-20, Figure 8-21, Figure 8-27, and Figure 8-28); therefore the calculation is continued to verify that the module transitions into passive and stable DHRS cooling. By 30 minutes, RCS flow has stabilized, and the RCS temperature and pressure are steadily decreasing as the DHRS transfers decay heat from the RCS to the reactor pool (Figure 8-21, Figure 8-26, Figure 8-27, and Figure 8-28). It is concluded that by 30 minutes the transient has been terminated and that stable DHRS cooling has been achieved. Subcritical margin is verified as net reactivity remains less than 0.0 dollars at the time stable DHRS cooling has been achieved (Figure 8-22). No operator action was credited to mitigate this event.

Table 8-3 Main steam line break sequence of events

Event	Time (sec)
Initiation of a split main steam line break (3.3% of the pipe cross sectional area) on SG 2	0
High power (125% RTP) analytical limit is reached.	47
RTS is actuated and control rods begin to insert into the core.	49
Peak power is 202.8 MW.	49
MCHFR is reached (3.682 as calculated by NRELAP5)	49
Control rods fully inserted.	51
Peak RCS pressure is reached (~1997 psia).	52
High SG pressure analytical limit (800 psia) is reached. DHRS is actuated.(1)	59
RCS flow is stagnant and slightly reversed.	~85
Peak main steam system pressure (~1207 psia) is reached.	133
End of calculation. Stable DHRS cooling has been established. Net reactivity remains < $0.0.	1800

(1) In this example transient calculation the pressurizer heater trip on high SG pressure was not modeled. The calculation shows the heaters maintain the RCS pressure. The overall event progression is not affected because reactor trip and DHRS actuation have occurred.

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Figure 8-20 Steam generators 1 (unaffected) and 2 (affected) secondary flow rates for the representative main steam line break event

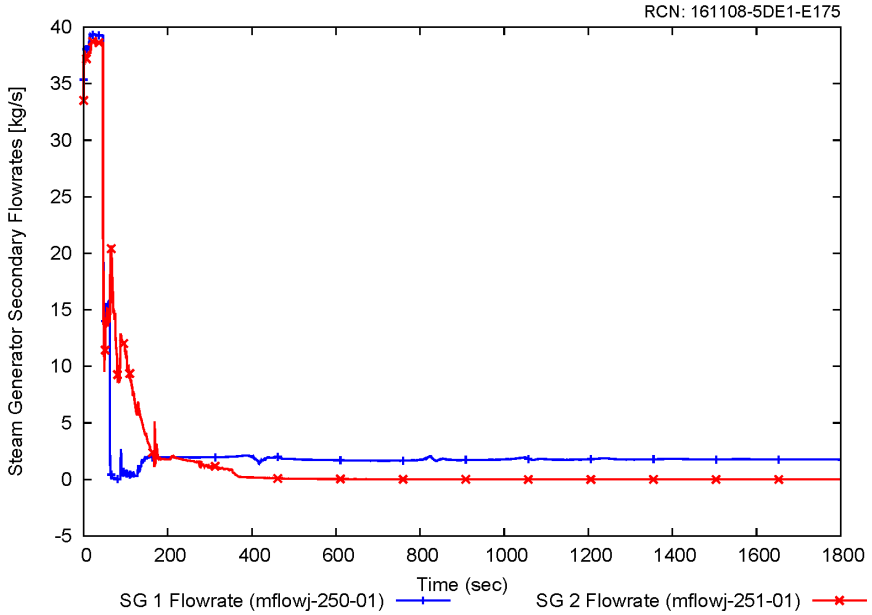
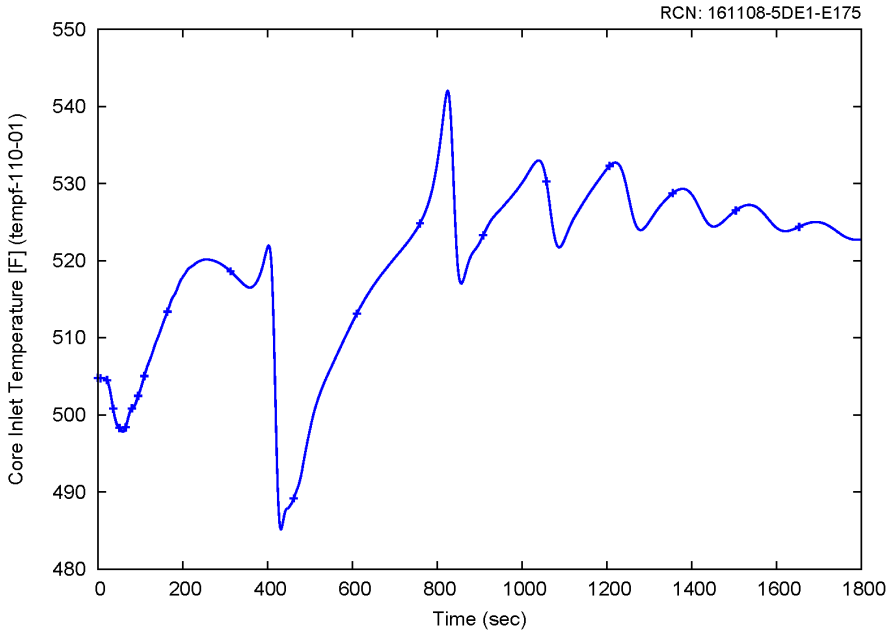


Figure 8-21 Core inlet temperature for the representative main steam line break event



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Figure 8-22 Net reactivity for the representative main steam line break event

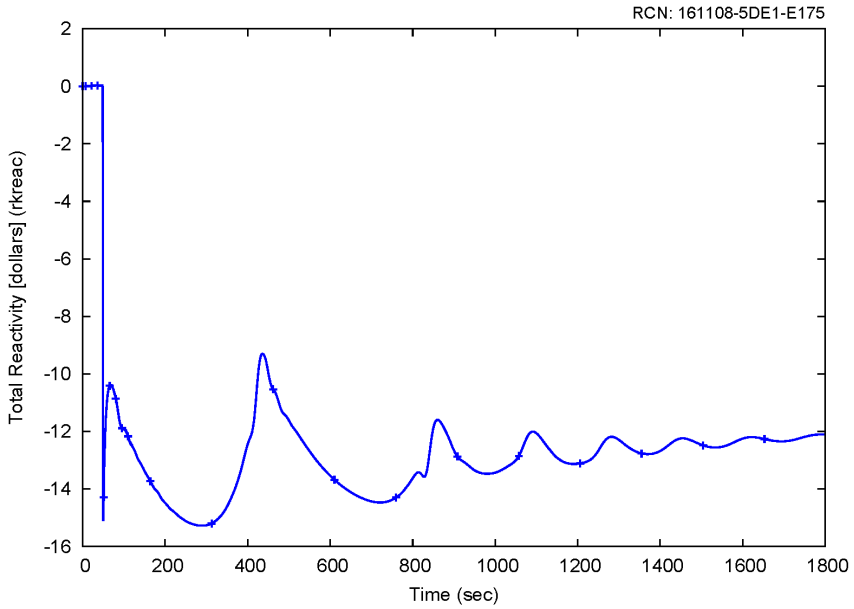
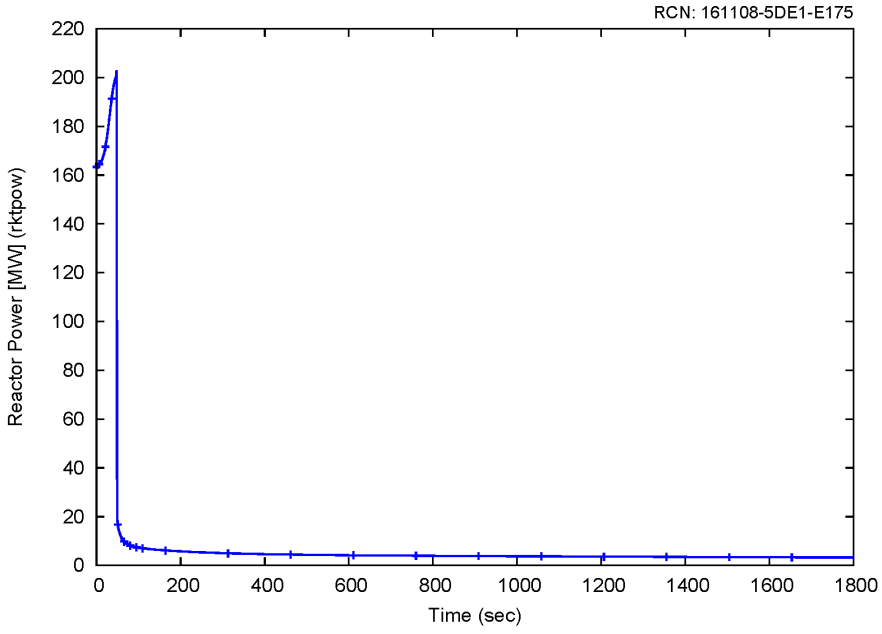


Figure 8-23 Power response for the representative main steam line break event



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Figure 8-24 Steam generators 1 (unaffected) and 2 (affected) pressure response for the representative main steam line break event

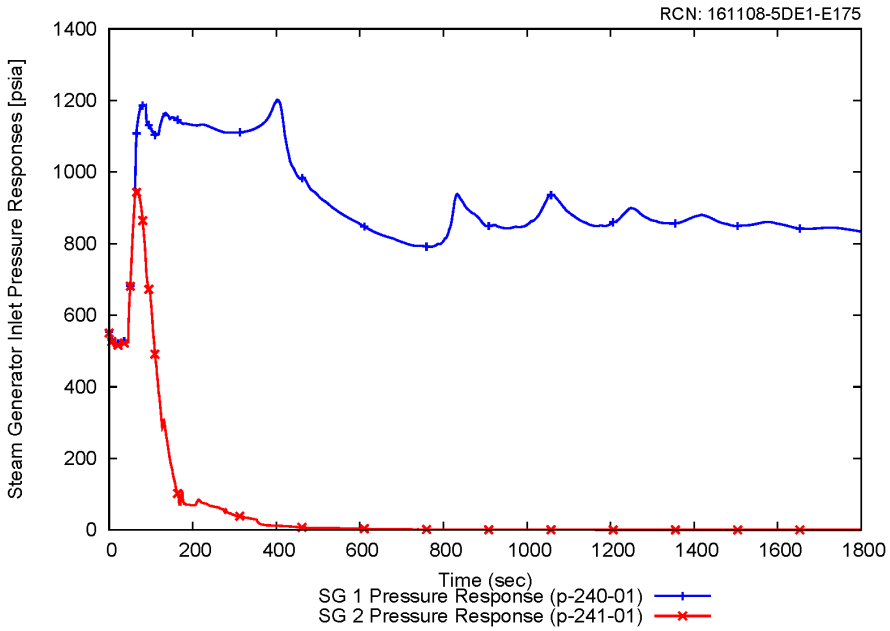
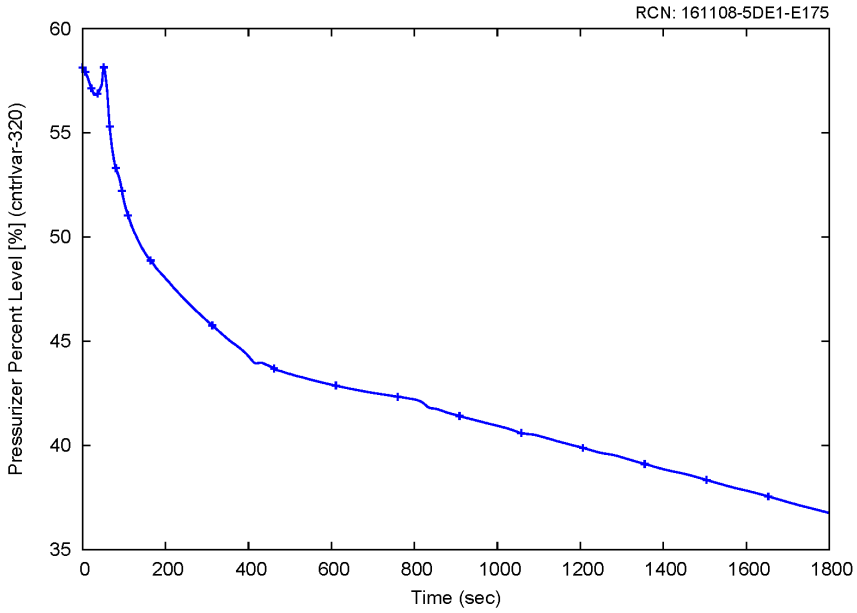


Figure 8-25 Pressurizer level for the representative main steam line break event



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Figure 8-26 Reactor pressure vessel pressure response for the representative main steam line break event

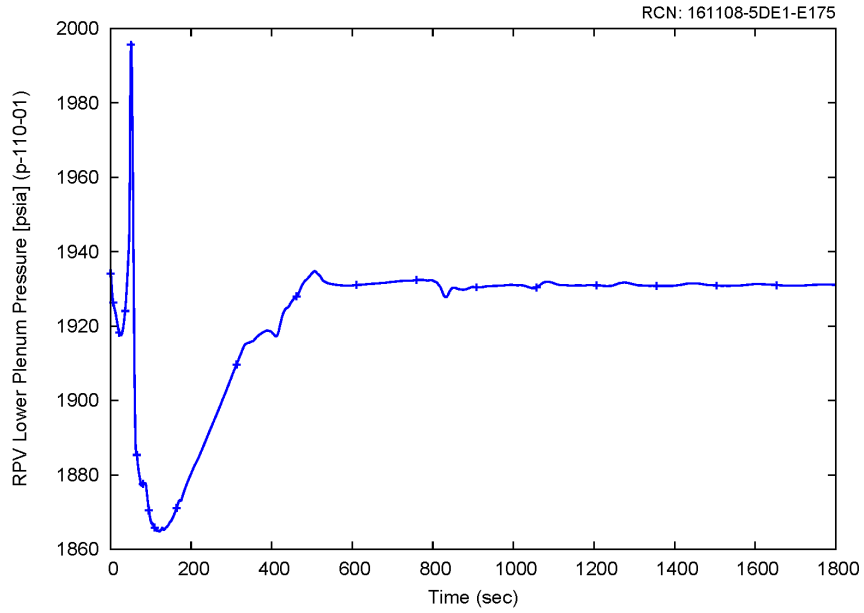
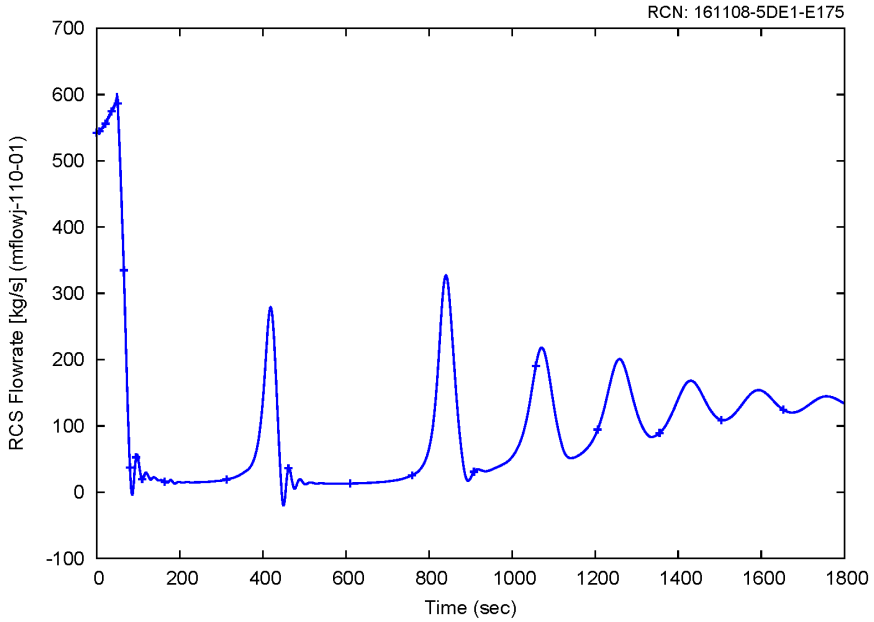
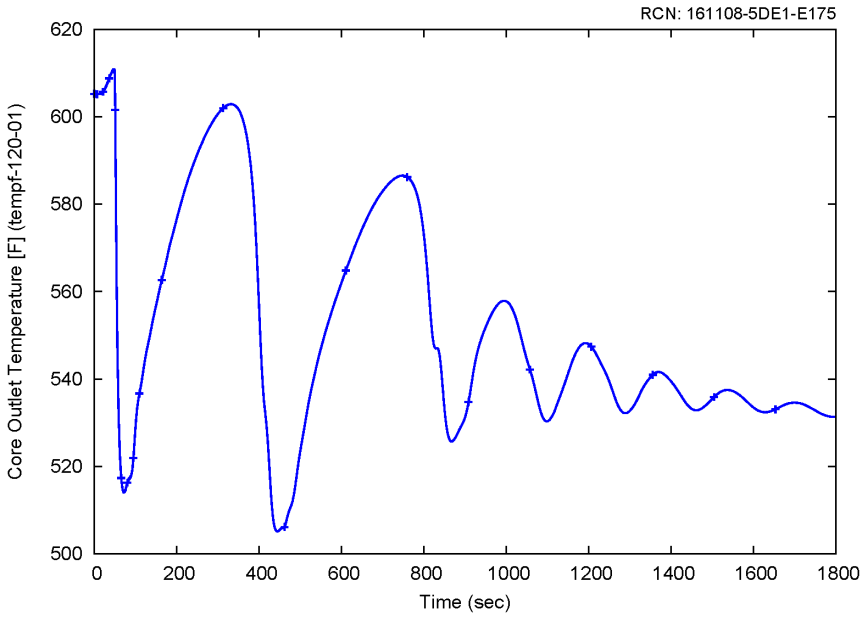


Figure 8-27 Reactor coolant system flow rate for the representative main steam line break event



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Figure 8-28 Core outlet temperature for the representative main steam line break event



8.1.3.3 Conclusion

A representative case that could challenge MCHFR was identified for a main steam line break event. The results of this case, as presented in Section 8.1.3.2, are subsequently used as input to an MCHFR evaluation using the NuScale subchannel analysis methodology.

8.2 Heatup and/or Pressurization of the Reactor Coolant System

8.2.1 Loss of Normal Feedwater Flow

The purpose of this section is to present the thermal-hydraulic response of a representative NPM for loss of normal feedwater flow. This event is evaluated for primary pressure and secondary pressure. Different initial condition biases and conservatisms are used for the RCS pressure case and the secondary pressure case.

8.2.1.1 Event Description - Reactor Coolant System Pressure Case

The general loss of normal feedwater flow event description can be found in Section 7.2.10. Chosen from a series of RCS pressure sensitivity cases, the sample loss of normal feedwater flow case here represents a case that could challenge the RCS pressure acceptance criterion. No single failure is applied since the challenging cases occur when all equipment operates as designed.

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Normal AC power is lost at turbine trip since this maximizes the system pressure responses. This case features the following conditions:

- The initial power is 102 percent. RCS average temperature is biased at the high condition (555 degrees F). RCS flow rated is biased to the low condition (535 kg/s). Pressurizer pressure is biased to the low condition (1780 psia). Pressurizer level is biased to the high condition (58 percent). Initial feedwater temperature is biased to the high condition (312.5 degrees F). Initial SG pressure is biased to the high condition (535 psia).
- SG heat transfer is increased 30 percent by applying a heat transfer coefficient multiplier of 1.3 on both the primary and secondary sides of the steam generator tubes in the steady state initialization model. As identified in Section 7.2.10.3, this biasing does not significantly affect margin to the RCS pressure acceptance criteria. There is no SG tube plugging for this representative calculation.
- BOC reactivity coefficients are used since they are bounding for overheating events.
- No operator action was credited in the representative case. Normal control system such as PZR spray, heater, letdown controls and automatic rod control are modeled based on the control status shown in Table 7-48.

8.2.1.2 Analysis Results - RCS Pressure Case

The following describes the event sequence of a representative case for the loss of normal feedwater flow event that could challenge the primary pressure. Table 8-4 summarizes the sequence of events. Figure 8-29 through Figure 8-37 show some key parameters during the representative loss of normal feedwater flow event.

The feedwater flow is completely lost at time zero of the transient. Due to the overheating after loss of feedwater, pressurizer pressure and level start to increase (Figure 8-29 and Figure 8-30). At 17.6 seconds after transient initiation, the pressurizer pressure reaches the reactor trip analytical limit (Figure 8-29). At 18.6 seconds, the turbine trips (on the reactor trip signal), normal power is lost-causing MSIV closure-and steam generator pressures begin to increase (Figure 8-31). At 19.6 seconds, scram rod insertion begins, which causes the reactor power to decrease (Figure 8-32), resulting in a RCS flow decrease (Figure 8-33). After reaching high pressurizer pressure analytical limit, the DHRS is also actuated at 19.6 seconds, but conservative modeling of the DHRS actuation valve opening delays initiation of flow in the DHRS until 49.3 seconds.

At 20.0 seconds, minimum MCHFR is reached (4.37 as calculated by NRELAP5).

At 24.2 seconds, the pressurizer dome pressure reaches the 2137.3 psia Reactor Safety Valve 1 biased setpoint, Reactor Safety Valve 1 begins to stroke open. At 24.6 seconds, the pressure at the bottom of the reactor vessel reaches a maximum value of 2156.1 psia and begins to decrease (Figure 8-29). That

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maximum RCS pressure value is well below 110 percent of the RCS design pressure (2310 psia). At 33.1 seconds, Reactor Safety Valve 1 closes. For the remainder of the transient, pressurizer pressure and level continue to decrease (Figure 8-29 and Figure 8-30).

At 49.3 seconds, flows through the DHRS actuation valves and into the steam generator inlet plena begin. The steam generator pressure starts to increases. At ~322 seconds, steam generator pressure reaches the maximum of 1243.4 psia and begins to decrease (Figure 8-31).

After reactor trip and actuation of DHRS, RCS flow decreases rapidly and becomes almost stagnant at ~50 seconds (Figure 8-33). Following that, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2). The oscillations gradually diminish as the stable natural circulation flow is established in the RCS (Figure 8-31, Figure 8-33, and Figure 8-34).

By 2500 seconds, DHRS operation and RCS flow are stable (Figure 8-33, Figure 8-34, and Figure 8-37); RCS pressure, temperatures, and steam generator pressure are trending downward (Figure 8-29, Figure 8-30, Figure 8-34 and Figure 8-35); the subcritical margin remains large at the end of the transient calculation (Figure 8-36).

Table 8-4 Loss of normal feedwater flow sequence of events - reactor coolant system pressure case

Event	Time (sec)
Total loss of feedwater	0.0 s
Pressurizer pressure analytical limit (2000 psia) is reached.	17.6 s
Turbine trips, normal AC power is assumed to be lost	18.6 s
Scram rod insertion begins, and DHRS is actuated	19.6 s
MCHFR is reached (4.37 as calculated by NRELAP5)	20.0 s
Peak RCS pressure is reached (2156.1 psia)	24.6 s
RCS flow is stagnant	~50 s
SG pressures reach maximum (1243.4 psia) and begin to decrease	322.2 s
End of calculation. Stable DHRS cooling has been established. Net reactivity remains < $0.0.	2500.0 s

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Figure 8-29 Reactor pressure vessel pressure response for the representative loss of normal feedwater flow event - reactor coolant system pressure case

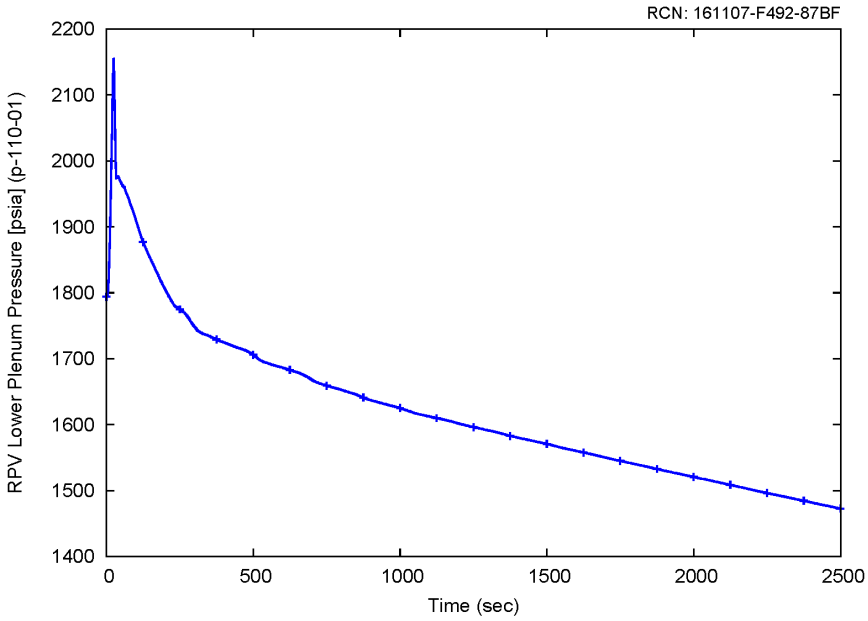
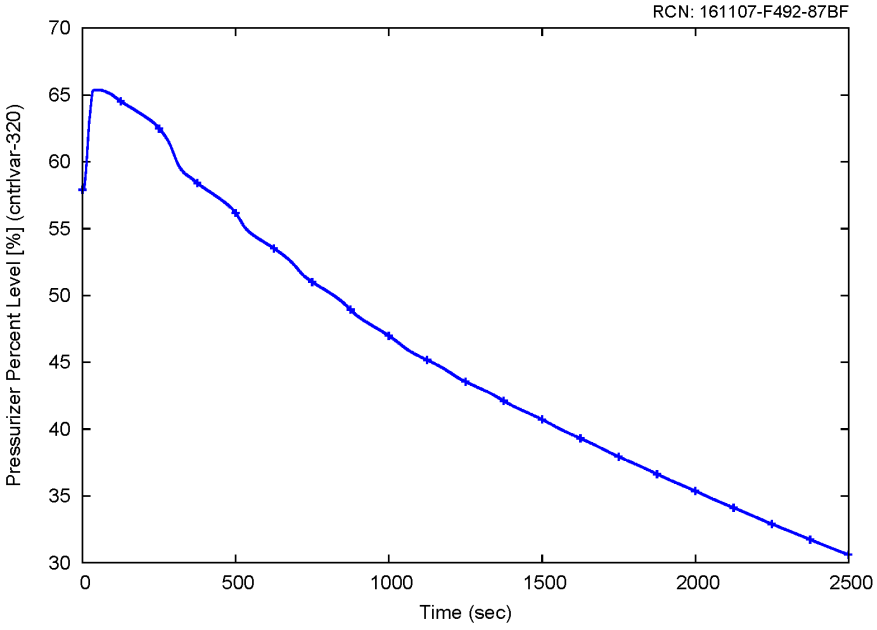


Figure 8-30 Pressurizer level for the representative loss of normal feedwater flow event - reactor coolant system pressure case



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Figure 8-31 Steam generator 2 pressure response for the representative loss of normal feedwater flow event - reactor coolant system pressure case

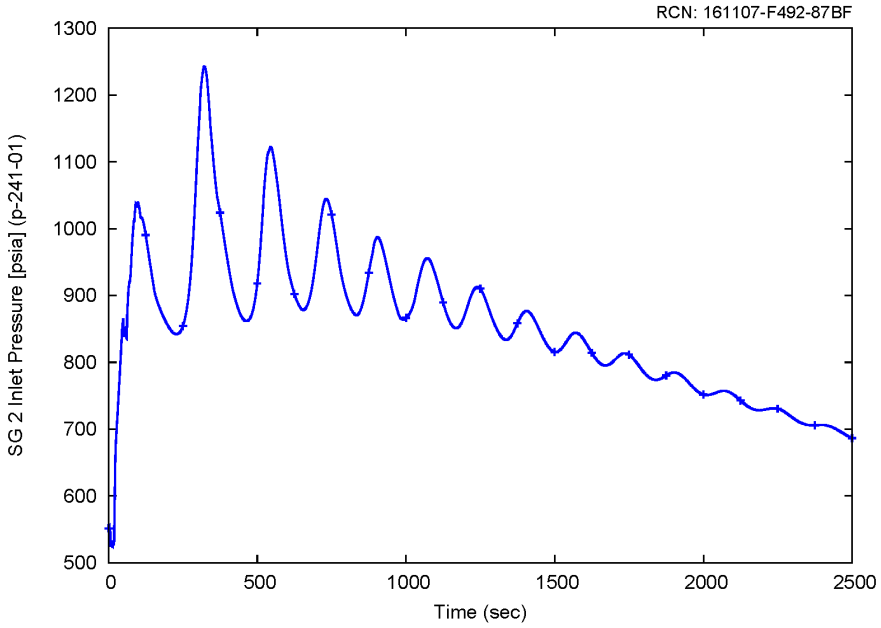
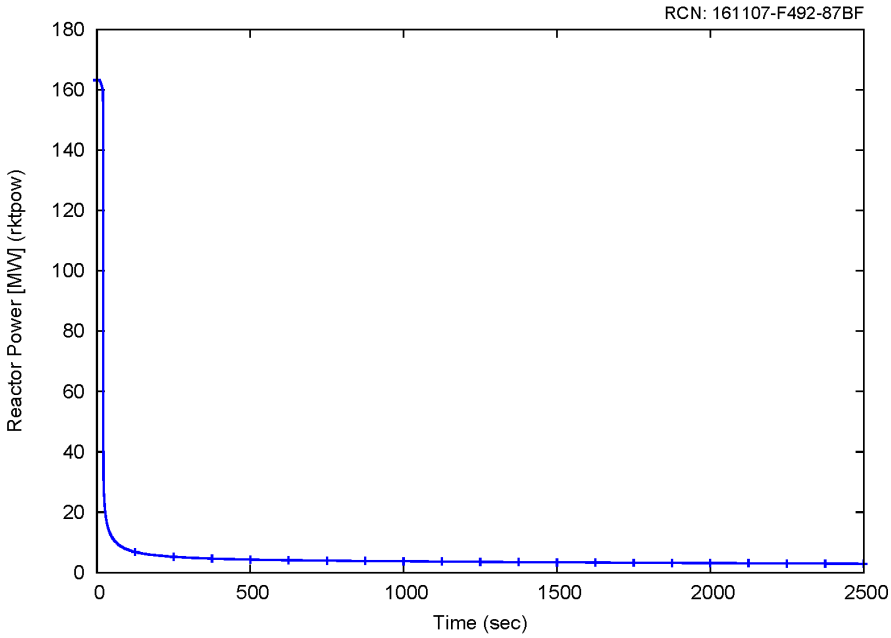


Figure 8-32 Power response for the representative loss of normal feedwater flow event - reactor coolant system pressure case



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Figure 8-33 Reactor coolant system flow rate for the representative loss of normal feedwater flow event - reactor coolant system pressure case

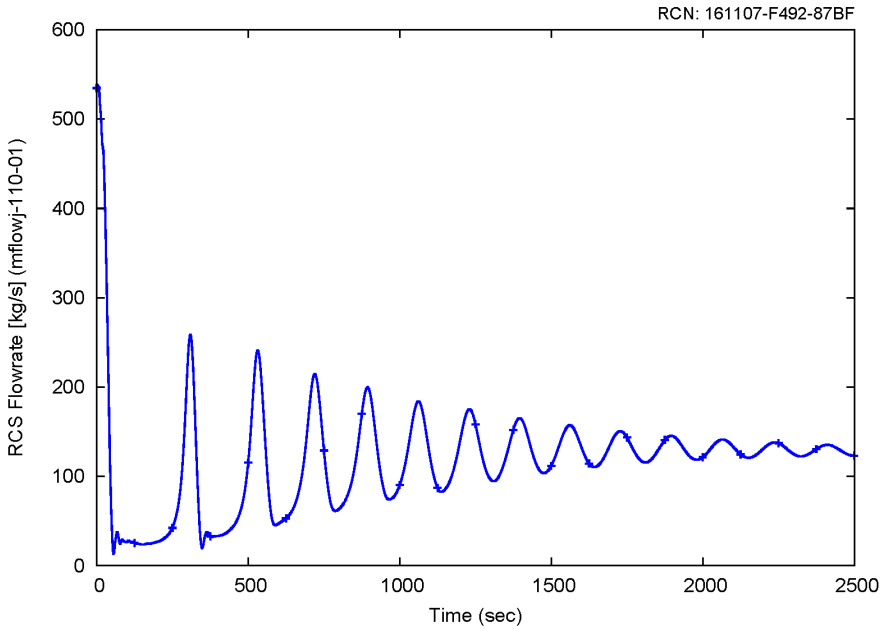
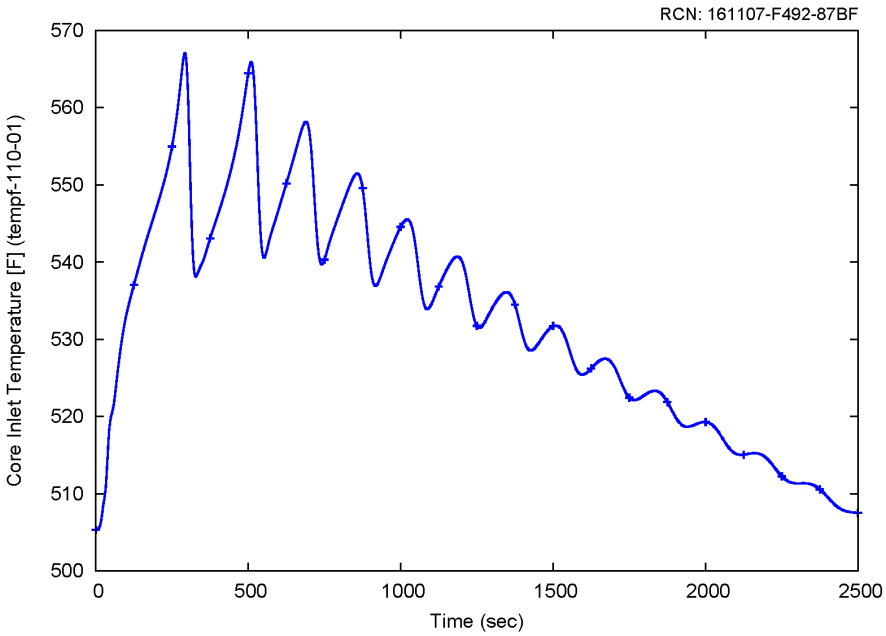


Figure 8-34 Core inlet temperature for the representative loss of normal feedwater flow event - reactor coolant system pressure case



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Figure 8-35 Core outlet temperature for the representative loss of normal feedwater flow event - reactor coolant system pressure case

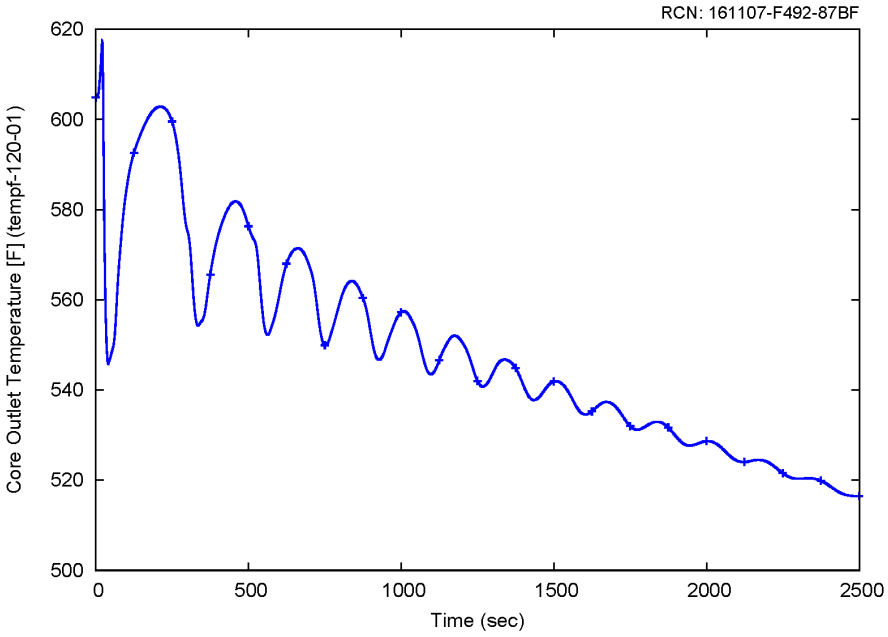
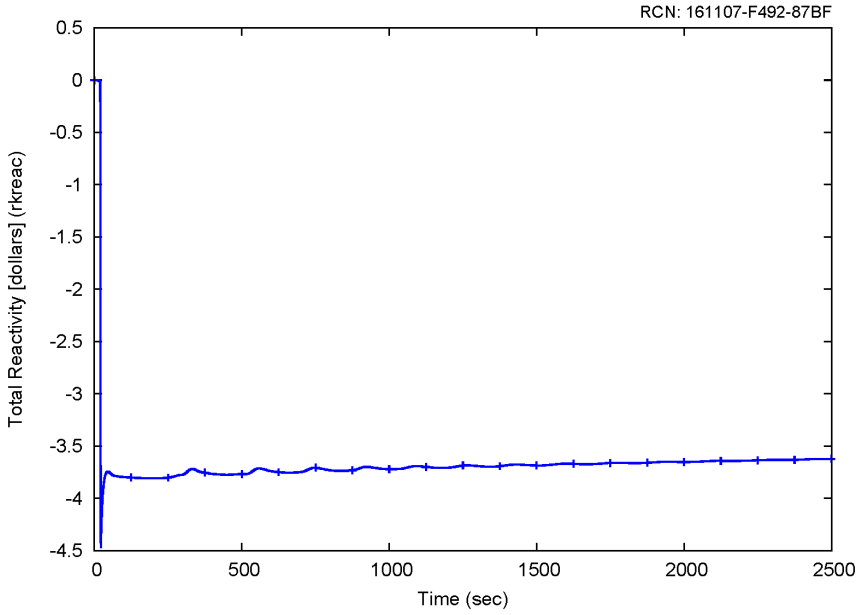
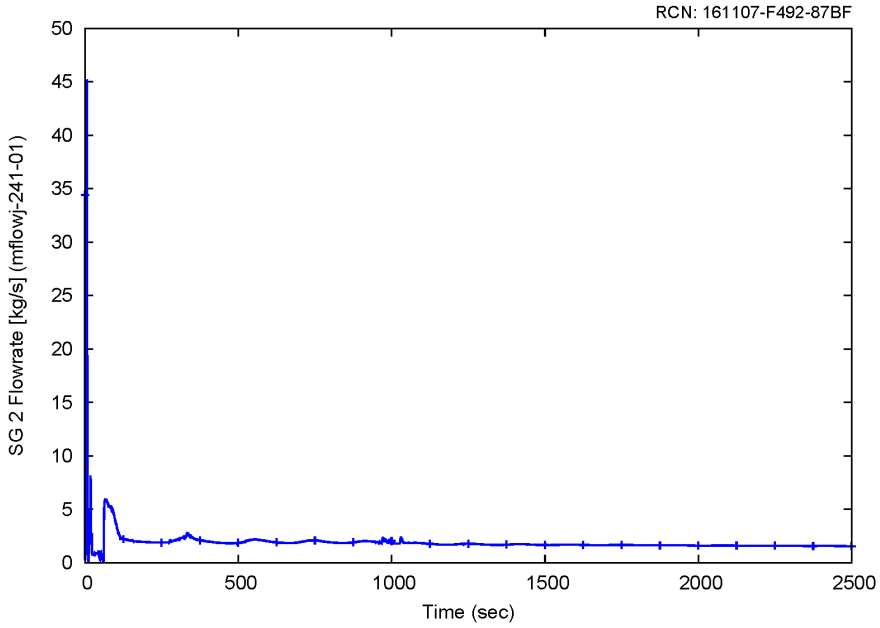


Figure 8-36 Net reactivity for the representative loss of normal feedwater flow event - reactor coolant system pressure case



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Figure 8-37 Steam generator 2 secondary flow for the representative loss of normal feedwater flow event - reactor coolant system pressure case



8.2.1.3 Conclusion - RCS Pressure Case

A representative case that could challenge the RCS pressure acceptance criterion was identified. The results of this case, as presented in Section 8.2.1.2, demonstrate the RCS pressure acceptance criterion is met.

8.2.1.4 Event Description - Secondary Pressure Case

The general loss of normal feedwater flow event description can be found in Section 7.2.10. Chosen from a series of secondary pressure sensitivity cases, the sample loss of normal feedwater flow case here represents a case that could challenge the secondary pressure acceptance criterion. No single failure is applied since the challenging cases occur when all equipment operates as designed. Normal AC power is lost at turbine trip since this maximizes the system pressure responses. This case features the same conditions as for the RCS pressure case (shown in Section 8.2.1.1), except that the feedwater flow is not totally lost at the beginning of the event. Instead, only a partial feedwater flow is lost (2.3 percent).

8.2.1.5 Analysis Results - Secondary Pressure Case

The following describes the event sequence of a representative case for the loss of normal feedwater flow event that could challenge the secondary pressure. Table 8-5 summarizes the sequence of events. Figure 8-38 through Figure 8-46

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show some key parameters during the representative loss of normal feedwater flow event.

The transient for this case is initiated by a fault that is postulated to result in a partial (2.3 percent) loss of feedwater flow. Automatic rod control is conservatively disabled and therefore control rods are not modeled to insert to reduce power. The RCS heats up slowly (Figure 8-38 and Figure 8-39), and the RCS pressure and pressurizer level start to increase (Figure 8-40 and Figure 8-41).

At 638.3 seconds after transient initiation, the RCS riser leg temperature reaches the reactor trip analytical limit. At 645.3 seconds, it is assumed that the turbine trips (on the reactor trip signal) and normal power AC power is lost. The loss of AC power causes MSIV closure, and the steam generator pressures and RCS pressure begin to increase (Figure 8-40 and Figure 8-42). At 646.3 seconds, scram rod insertion begins, which causes the reactor power to decrease (Figure 8-43), resulting in a RCS flow decrease (Figure 8-44). After reaching the high RCS riser leg temperature analytical limit, the DHRS is also actuated at 646.3 seconds, but conservative modeling of the DHRS actuation valve opening delays initiation of DHRS flow until 676.0 seconds.

At 658.9 seconds, the RCS pressure reaches a maximum (1939.4 psia) and begins to decrease (Figure 8-40); the maximum value in this case is less than that for the RCS pressure case and the RSVs do not open. For the remainder of the transient, the RCS pressure and pressurizer level continue to decrease (Figure 8-40 and Figure 8-41).

At 676.0 seconds, flow through the DHRS actuation valves and into the steam generator inlet plena begins (Figure 8-46). The steam generator pressure starts to increase. At ~716.3 seconds, steam generator pressure reaches the maximum of 1421.6 psia and begins to decrease (Figure 8-42).

After reactor trip and actuation of DHRS, RCS flow decreases rapidly and briefly becomes almost stagnant at ~700 seconds (Figure 8-44). Following that, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2). The oscillations gradually diminish as the stable natural circulation flow is established in the RCS (Figure 8-38 and Figure 8-42 through Figure 8-44).

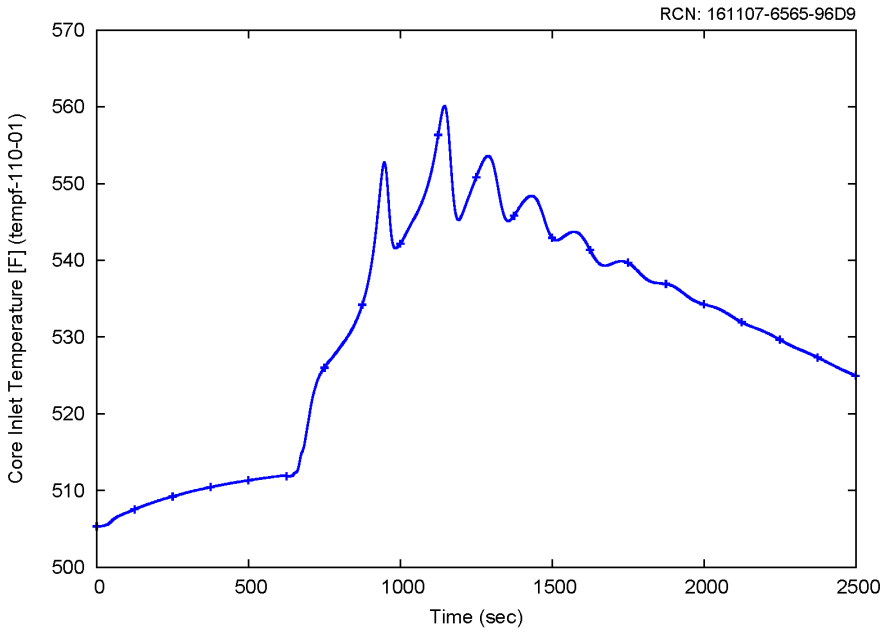
By 2500 seconds, DHRS operation and RCS flow are stable (Figure 8-38 through Figure 8-44); RCS pressure, temperatures, and steam generator pressure are trending downward (Figure 8-38, Figure 8-39, Figure 8-40, and Figure 8-42); the subcritical margin remains large (Figure 8-45).

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Table 8-5 Loss of normal feedwater flow sequence of events - secondary pressure case

Event	Time (sec)
Feedwater flow begins 0.1 s rampdown to 97.7% of initial value	0.0 s
RCS riser leg temperature analytical limit (610°F) is reached	638.3 s
Turbine trips, normal AC power is assumed to be lost	645.3 s
Scram rod insertion begins, and DHRS is actuated	646.3 s
Peak RCS pressure is reached (1939.4 psia)	658.9 s
RCS flow briefly becomes almost stagnant	~700 s
SG pressures reach maximum (1421.6 psia) and begin to decrease	716.3 s
End of calculation. Stable DHRS cooling has been established. Net reactivity remains < $0.0.	2500.0 s

Figure 8-38 Core inlet temperature for the representative loss of normal feedwater flow event - secondary pressure case



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Figure 8-39 Core outlet temperature for the representative loss of normal feedwater flow event - secondary pressure case

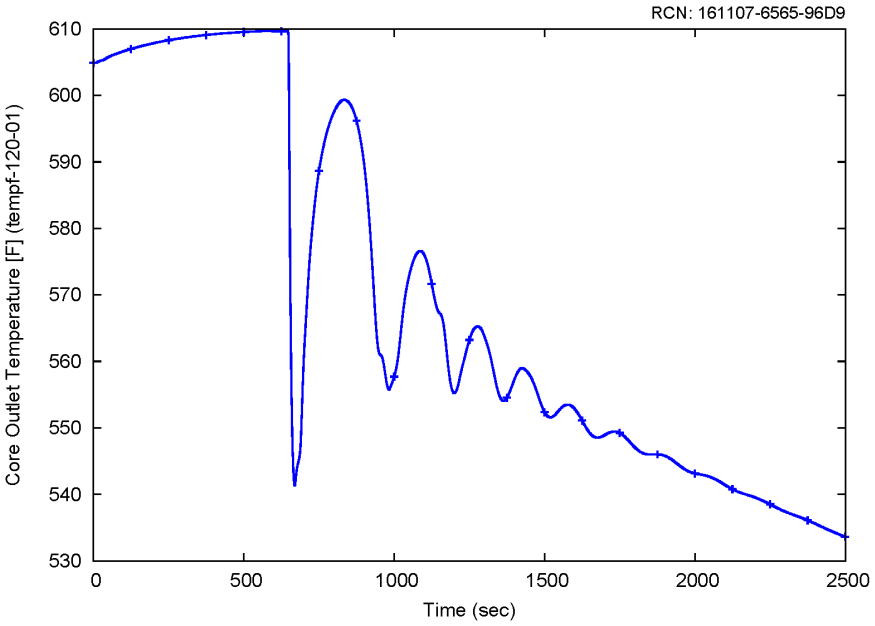
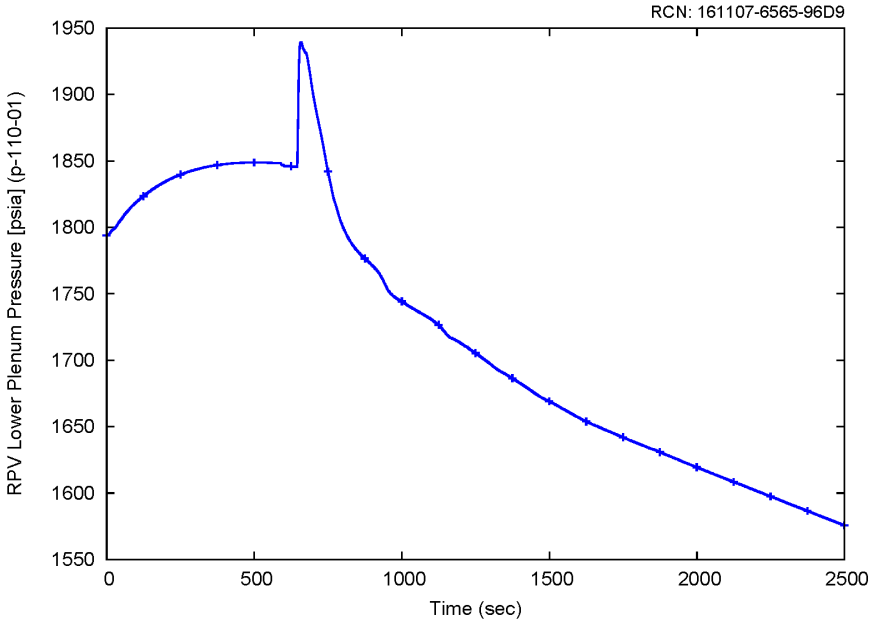


Figure 8-40 Reactor pressure vessel pressure response for the representative loss of normal feedwater flow event - secondary pressure case



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Figure 8-41 Pressurizer level for the representative loss of normal feedwater flow event - secondary pressure case

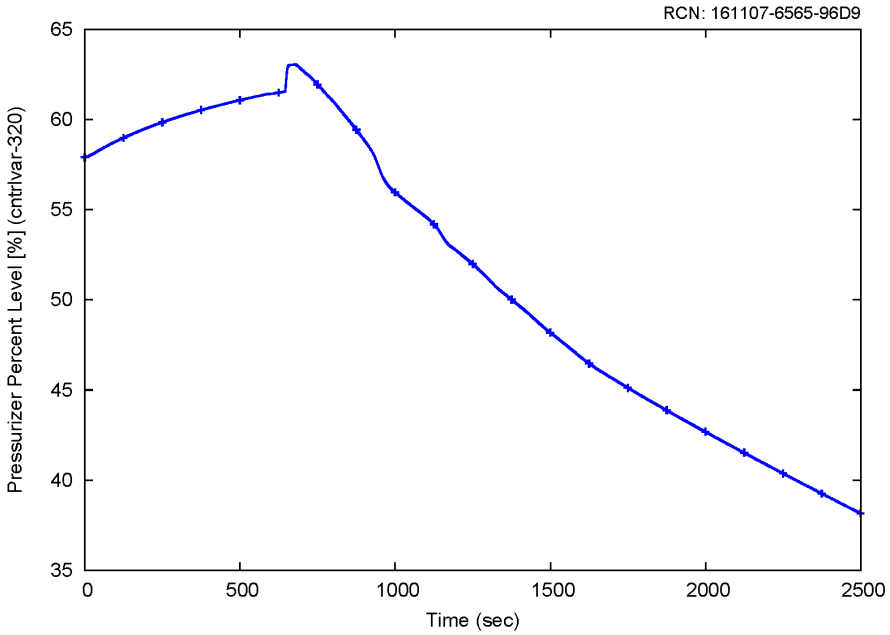
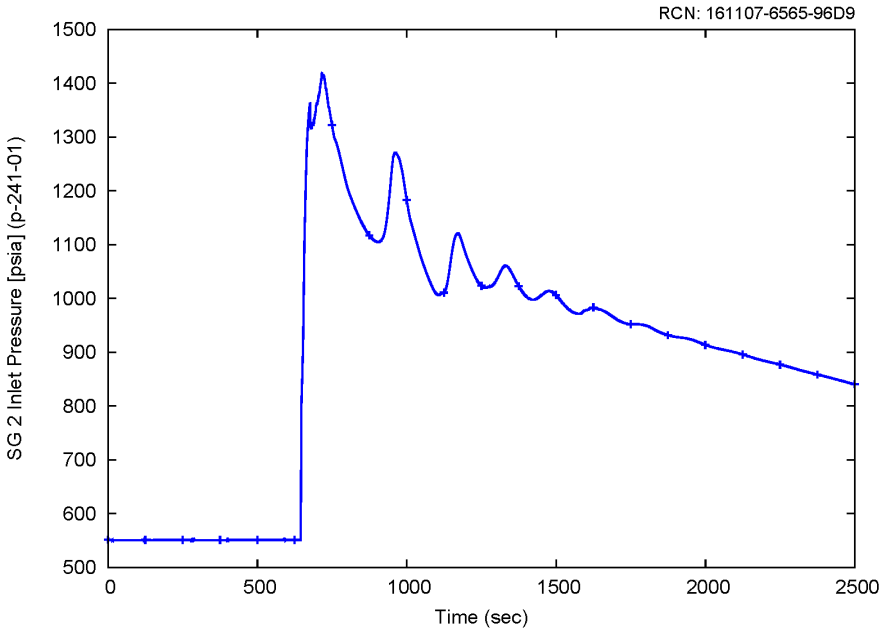


Figure 8-42 Steam generator 2 pressure response for the representative loss of normal feedwater flow event - secondary pressure case



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Figure 8-43 Power response for the representative loss of normal feedwater flow event - secondary pressure cas

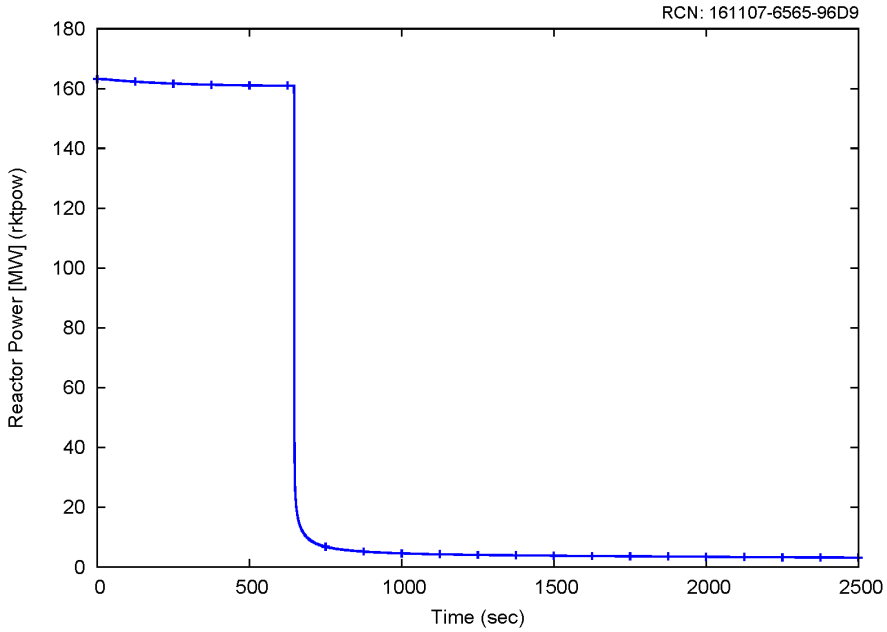
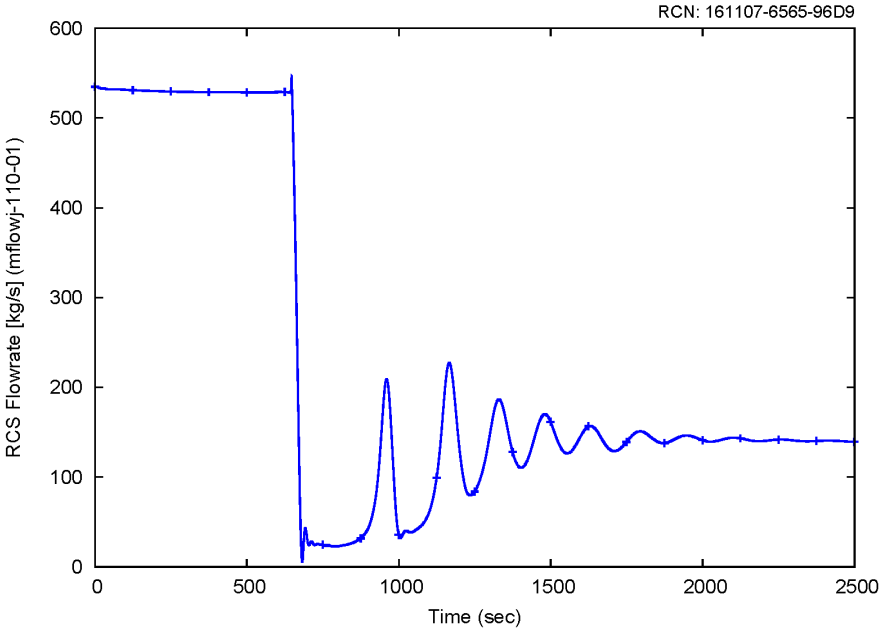


Figure 8-44 Reactor coolant system flow rate for the representative loss of normal feedwater flow event - secondary pressure case



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Figure 8-45 Net reactivity for the representative loss of normal feedwater flow event - secondary pressure case

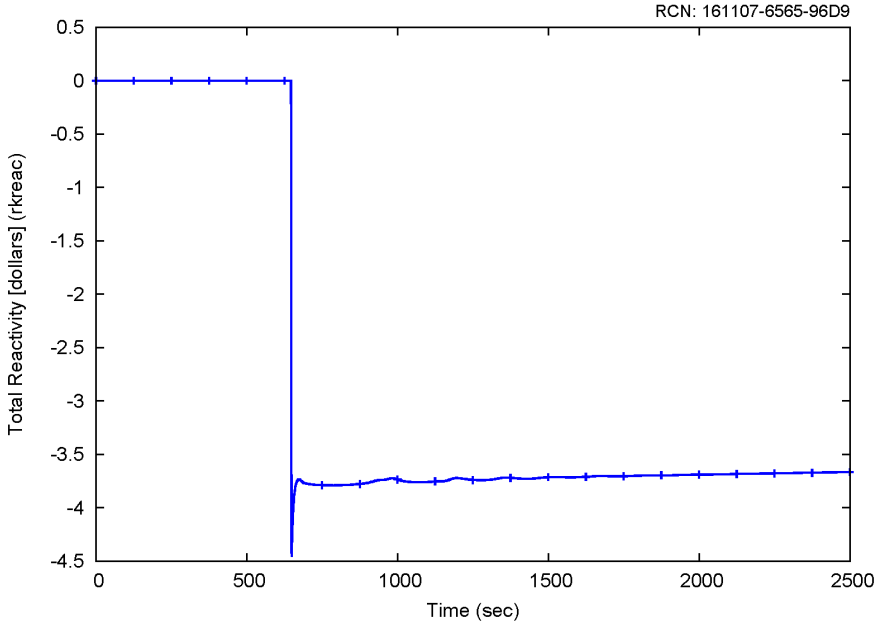
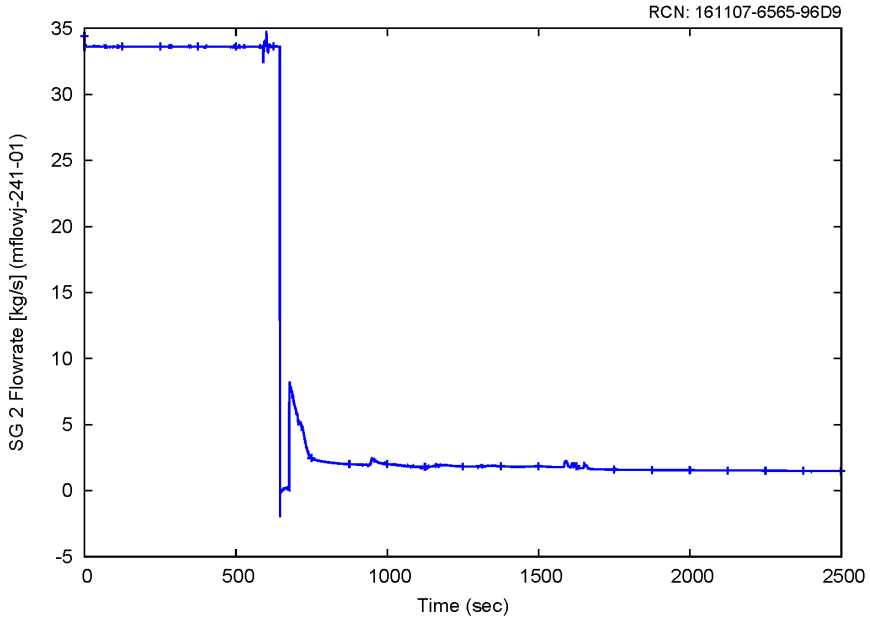


Figure 8-46 Steam generator 2 secondary flow for the representative loss of normal feedwater flow event - secondary pressure case



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8.2.1.6 Conclusion - Secondary Pressure Case

A representative case that could challenge secondary pressure acceptance criteria was identified. The results of this case, as presented in Section 8.2.1.5 demonstrate the secondary pressure acceptance criterion is met.

8.2.2 Loss of Normal AC Power

The purpose of this section is to present the thermal-hydraulic response of a representative NPM for loss of normal AC power. This event is evaluated for primary pressure and secondary pressure. A representative case evaluated for primary pressure is presented.

8.2.2.1 Event Description

The general event description for a loss of normal AC power is found in Section 7.2.9.1. Based on Section 7.2.9.1, the acceptance criterion for primary pressure is potentially challenged by the loss of normal AC power. The challenging case occurs when all equipment is operational.

The representative case presented here represents a case that could challenge the primary pressure response (as shown in Table 7-44). Case features include the following conditions:

- Conservative initial condition biasing is applied in order to maximize the consequences of this event. This representative case is initialized at 102 percent reactor power.
- The RSV set pressure is biased to the high condition (2137.25 psia, which includes a 3 percent drift allowance) to maximize primary pressure response.
- The reactor pool temperature is biased to the maximum bounding value (200 degrees F) to provide a bounding high temperature for events/cases that require heat removal using the DHRS.
- The initial pressurizer pressure is biased to the maximum value (1920 psia) for the primary side pressurization case.
- The initial pressurizer level is biased to the maximum value (58 percent) to maximize the primary side pressure response.
- BOC reactivity coefficients are applied to maximize the primary side pressurization.

8.2.2.2 Analysis Results

The following describes the event sequence of the loss of normal AC power event. Table 8-6 summarizes the sequence of events. Figure 8-47 through Figure 8-53 show some key parameters during the loss of normal AC power event.

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The loss of normal AC power occurs at zero seconds, which causes an immediate turbine trip. In this case, it is assumed that the DC power system (EDSS for the representative NPM) batteries are available. The control rods do not immediately insert on loss of AC power. Rather, a delay to reactor trip, DHRS actuation, and containment isolation are assumed to maximize RCS pressure. During this delay period, primary and secondary pressure increase due to the turbine trip. Primary temperatures (Figure 8-47) and system pressures (Figure 8-48, curves for SG1 and SG2 overlap) initially increase due to the mismatch between primary side heat production and secondary side heat sink. The analytical limit for high pressurizer pressure is reached at approximately 6 seconds, actuating reactor trip (Figure 8-49). DHRS valves begin to open at approximately 8 seconds (Figure 8-50). At approximately 12 seconds one of the two RSVs lifts (Figure 8-51). A peak pressure of 2155 psia is reached in the RPV.

Subsequent to reactor trip and RSV lift, system pressures and temperatures decrease, and system shrinkage reduces the PZR level (Figure 8-53).

After reactor trip and actuation of DHRS, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2); therefore the calculation is continued to verify that the module transitions into passive and stable DHRS cooling. Figure 8-47, Figure 8-50, and Figure 8-52 show that at ~15 minutes, RCS flow has stabilized, and the RCS temperatures are steadily decreasing as the DHRS is transferring decay heat from the RPV to the reactor pool. It is concluded that by 30 minutes the transient has been terminated and that stable DHRS cooling has been achieved. No operator action was credited to mitigate this event.

Table 8-6 Sequence of events for loss of AC power

Event	Time (sec)
Loss of AC power occurs	0.0
Turbine trip occurs	0.0
CVCS isolation occurs	0.0
High PZR pressure analytical limit is reached (2000 psia).	6
RTS actuation on high pressurizer pressure signal.	8
DHRS actuation on the high pressurizer pressure signal. DHRS actuation valves begin to open. FWIVs and MSIVs begin to close.	8
RSV1 opens	12
Peak RPV pressure is reached (2155 psia)	12
MSIVs are fully closed.	13
FWIVs are fully closed.	13
DHRS actuation valves are fully open.	38
Peak steam generator pressure is reached (1250 psia).	80
Establishment of stable RCS flow. Pressure and temperature are steadily decreasing.	1500
End of calculation. Stable DHRS cooling has been established.	1800

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Figure 8-47 Primary temperature response for the representative loss of AC power event

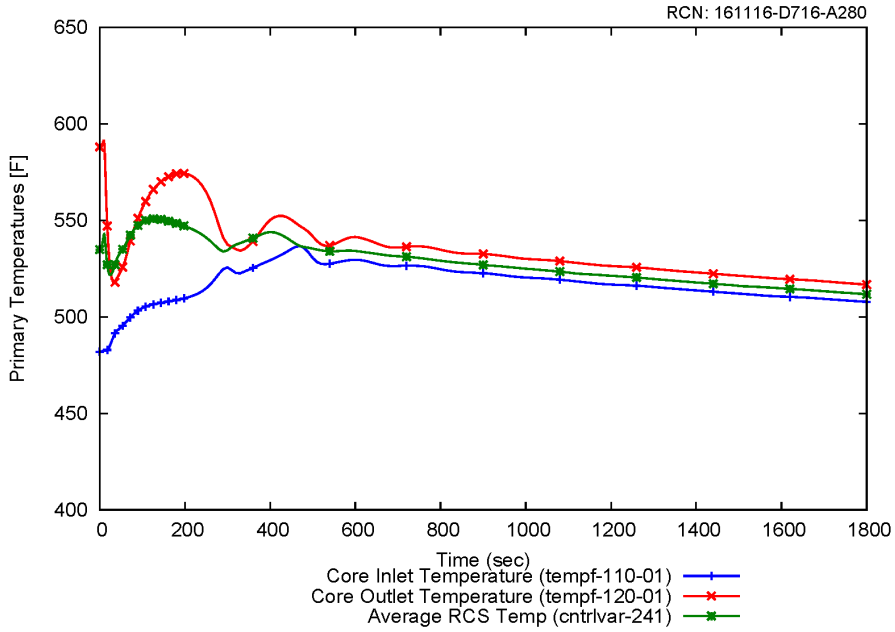
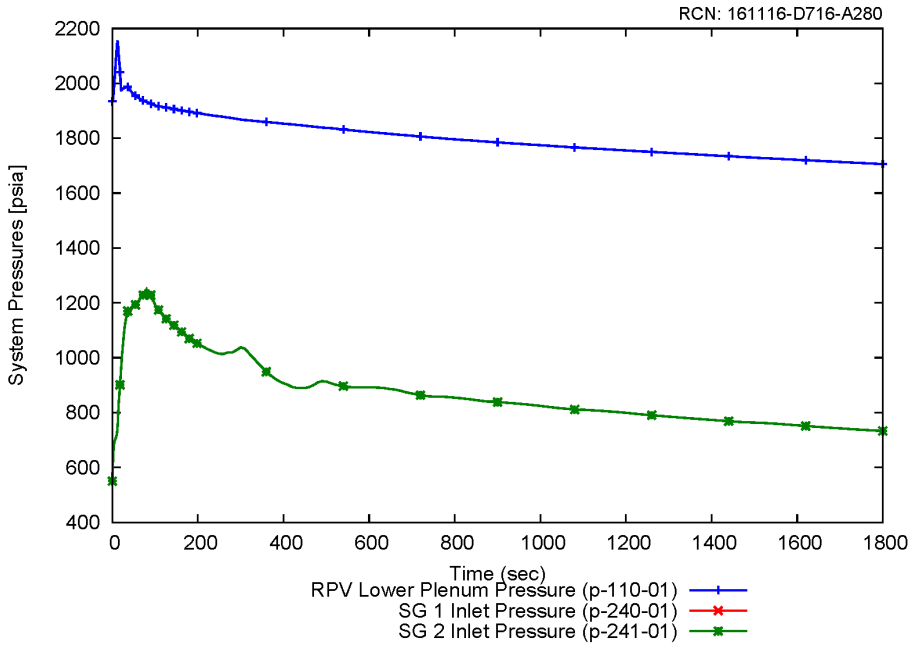


Figure 8-48 System pressure response for the representative loss of AC power event



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Figure 8-49 Reactor pressure vessel core power response for the representative loss of AC power event

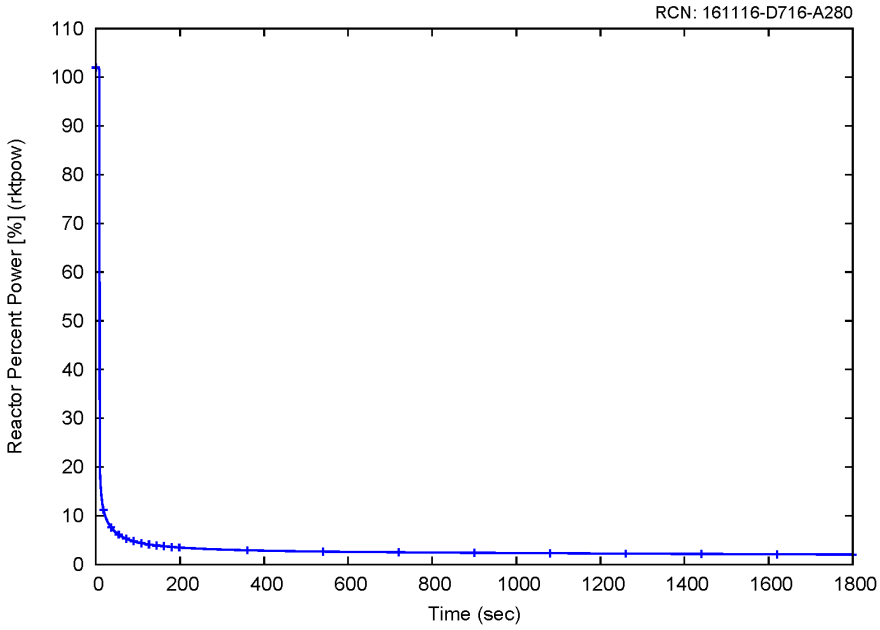
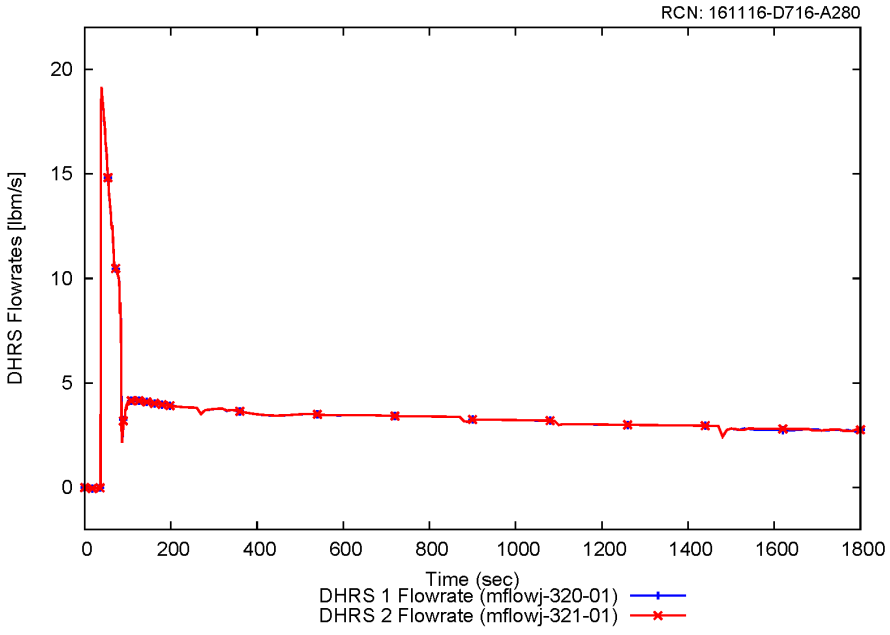


Figure 8-50 Decay heat removal system response for the representative loss of AC power event



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Figure 8-51 RSV flow response for the representative loss of AC power event

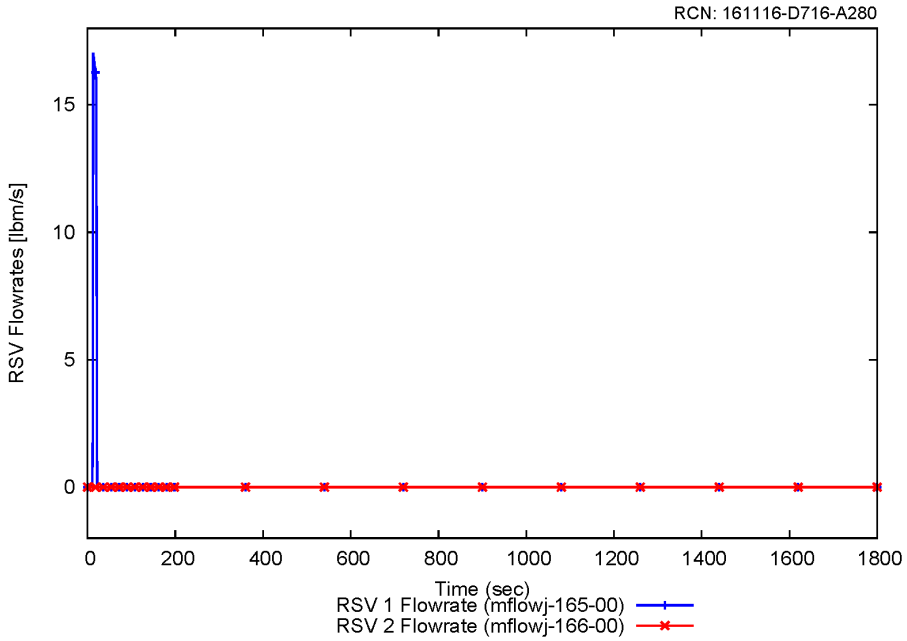
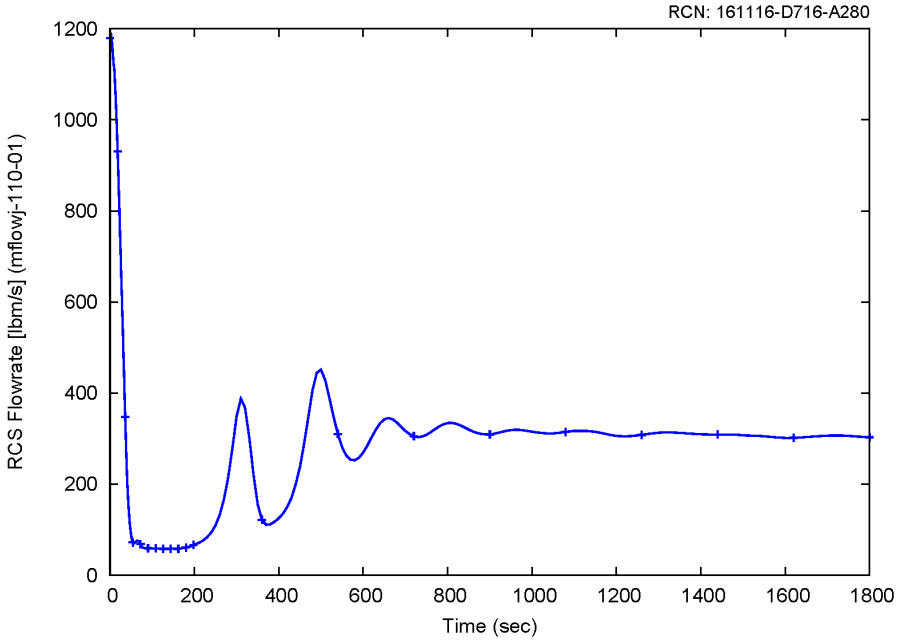
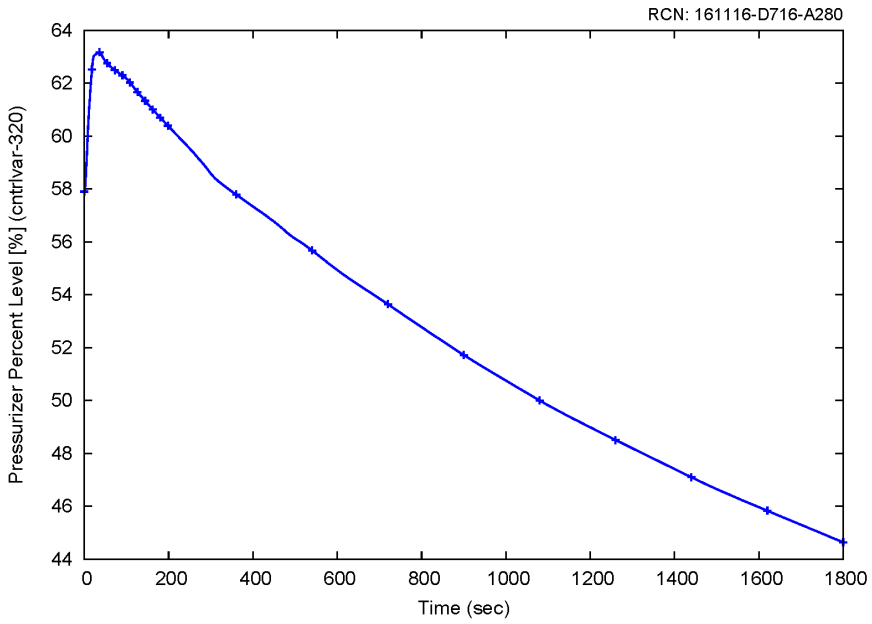


Figure 8-52 Reactor coolant system flow response for the representative loss of AC power event



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Figure 8-53 Pressurizer level response for the representative loss of AC power event



8.2.2.3 Conclusion

A representative case that could challenge primary pressure acceptance criterion was identified. The results of this case, as presented in Section 8.2.2.2, demonstrate that the primary pressure acceptance criterion is met.

8.2.3 Feedwater Line Break

The purpose of this section is to present the thermal-hydraulic response of a representative NPM to a feedwater system piping failure. The challenging case occurs when AC power is lost at event initiation and all equipment is operational. This event is evaluated for primary pressure and secondary pressure. A representative case evaluated for primary pressure is presented.

8.2.3.1 Event Description

The general event description for a feedwater system piping failure is found in Section 7.2.12.1. Based on Section 7.2.12.1, the acceptance criterion for primary pressure is potentially challenged by the feedwater system piping failure.

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The representative case presented here represents a case that could challenge the primary pressure response (as shown in Table 7-56). Case features include the following conditions:

- Conservative initial condition biasing is applied in order to maximize the consequences of this event. This representative case is initialized at 102 percent reactor power.
- The initial primary temperatures are biased to the maximum value (~545 degrees F) to produce a (slightly) higher energy system.
- The RSV set pressure is biased to the high condition (2137.25 psia, which includes a 3 percent drift allowance) to maximize primary pressure response.
- The initial feedwater temperature is biased to the maximum value (~305 degrees F) to produce a (slightly) higher energy system.
- The reactor pool temperature is biased to the maximum bounding value (~200 degrees F) to provide a bounding high temperature for events/cases that require heat removal using the DHRS.
- The initial pressurizer pressure is biased to the maximum value (1920 psia) for the primary side pressurization case.
- The initial pressurizer level is biased to the maximum value (58 percent) to maximize the primary side pressure response.
- BOC reactivity coefficients are applied to maximize the primary side pressurization.
- Automatic rod control is disabled.

8.2.3.2 Analysis Results

The following describes the event sequence of the feedwater line break event. Table 8-7 summarizes the sequence of events. Figure 8-54 through Figure 8-61 show some key parameters during the feedwater line break event.

A feedwater line break outside of containment occurs at zero seconds coincident with a loss of AC power, causing an immediate turbine trip and feedwater pump trip (Figure 8-54). In this case, it is assumed that the DC power system (EDSS for the representative NPM) batteries are available. To maximize RCS pressure, in this case it is assumed that after normal AC power is lost and MPS senses the loss of power to the EDSS battery chargers, there is a delay before MPS actuates reactor trip, DHRS and containment isolation. Since the reactor does not immediately trip, primary temperatures (Figure 8-55, curves for inlet temperature and RCS temperature overlap) and system pressures (Figure 8-56, curves for SG1 and SG2 overlap) initially increase due to the mismatch between primary side heat production and secondary side heat sink. The analytical limit for PZR pressure is reached at approximately 6 seconds. RTS actuation (Figure 8-57) occurs at approximately 8 seconds and RSV lift (Figure 8-58) occurs at approximately 11 seconds. A peak pressure of 2158 psia is reached in the RPV.

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The DHRS valves are fully open at approximately 36 seconds (Figure 8-59, curves for DHRS 1 and DHRS 2 overlap).

Subsequent to reactor trip and RSV lift, system pressures and temperatures decrease, and system shrinkage reduces the PZR level (Figure 8-60).

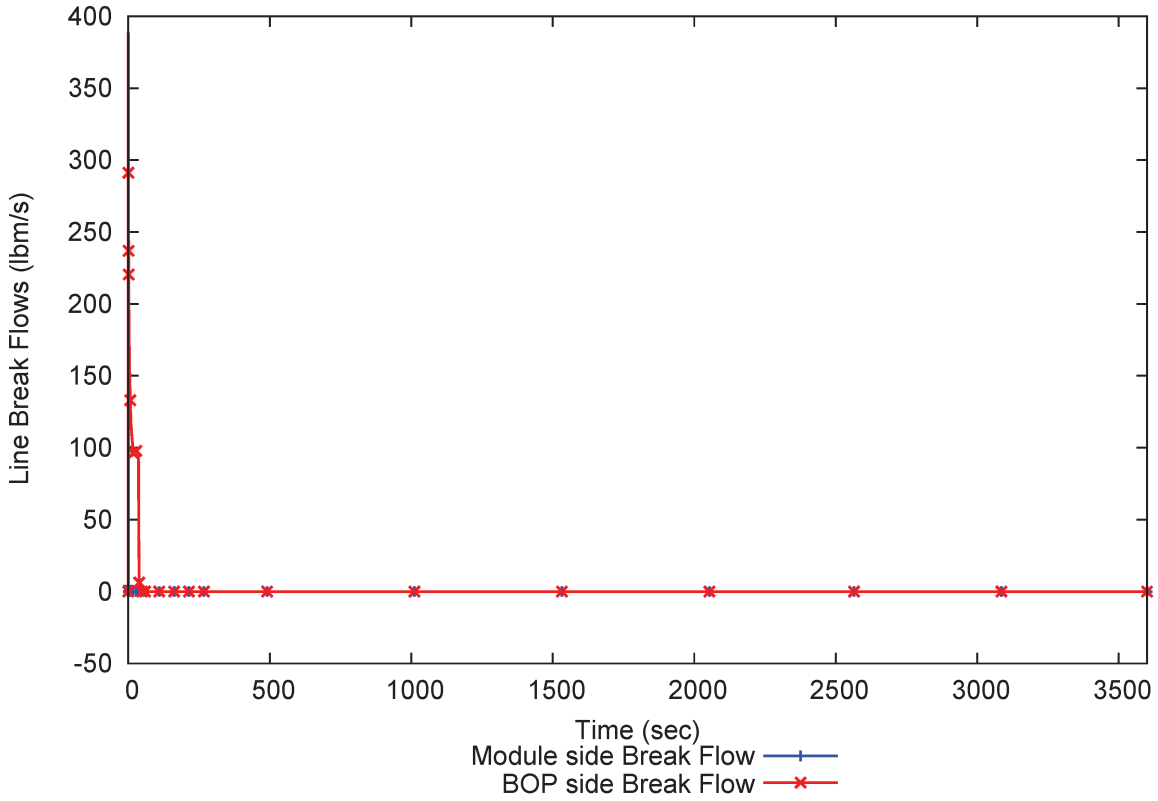
After reactor trip and actuation of DHRS, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2); therefore the calculation is continued to verify that the module transitions into passive and stable DHRS cooling. Figure 8-55, Figure 8-59, and Figure 8-61 show that at ~30 minutes, RCS flow has stabilized, and the RCS temperatures are steadily decreasing as the DHRS is transferring decay heat from the RPV to the reactor pool. It is concluded that by ~40 minutes the transient has been terminated and that stable DHRS cooling has been achieved. In this case, the EDSS batteries hold the ECCS valves closed for the duration of the transient calculation. No operator action was credited to mitigate this event.

Table 8-7 Sequence of events for feedwater line break outside containment

Event	Time (sec)
A double ended guillotine break in the SG 2 feedwater line occurs under the bioshield (just outside of containment).	0
AC power is lost resulting in turbine trip and FW pump trip	0
High PZR pressure analytical limit is reached (2000 psia)	6
RTS and DHRS ESFAS actuated	8
MSIVs close	8
Control rods fully inserted	10
RSV lift point is reached (2137 psia)	11
Peak RCS pressure reached (2158 psia)	12
FWIVs close (check valves already seated)	15
RSV reseats	22
DHRS actuation valves open	36
Peak pressure reached in SG 1 (1299 psia)	77
End of calculation. Stable DHRS cooling has been established.	3600

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Figure 8-54 Feedwater line break flow response for the representative feedwater line break event



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Figure 8-55 Primary temperature response for the representative feedwater line break event

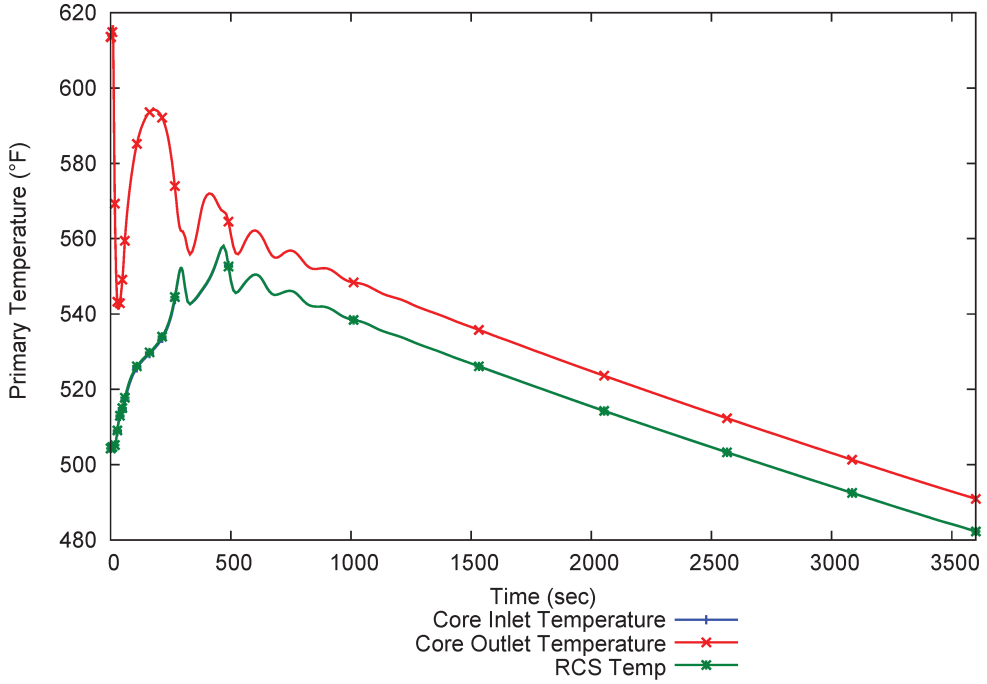
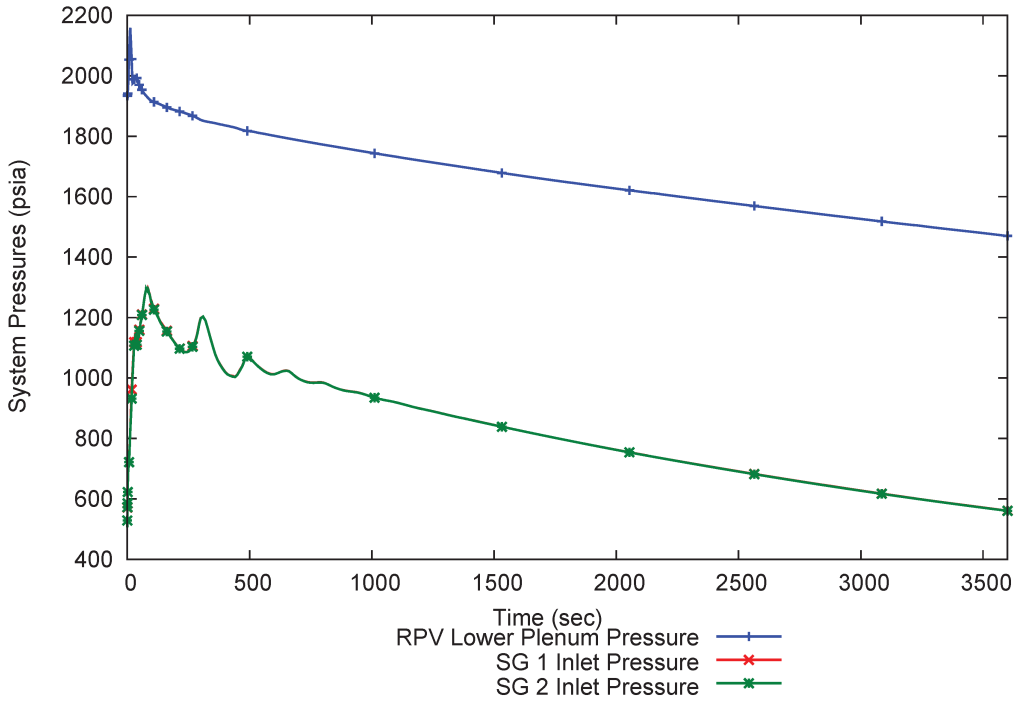


Figure 8-56 System pressure response for the representative feedwater line break event



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Figure 8-57 Reactor pressure vessel core power response for the representative feedwater line break event

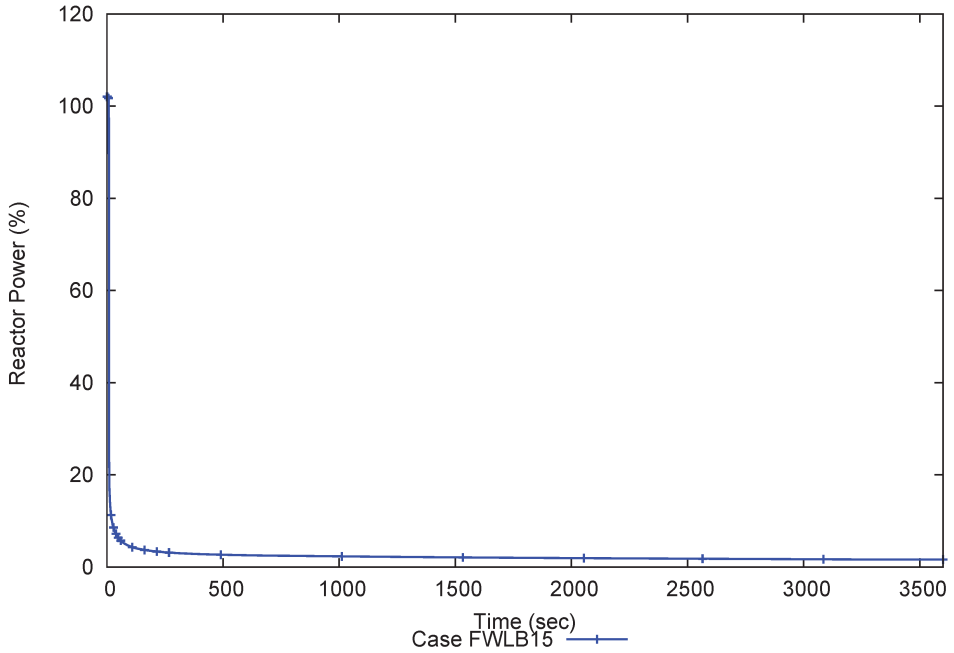
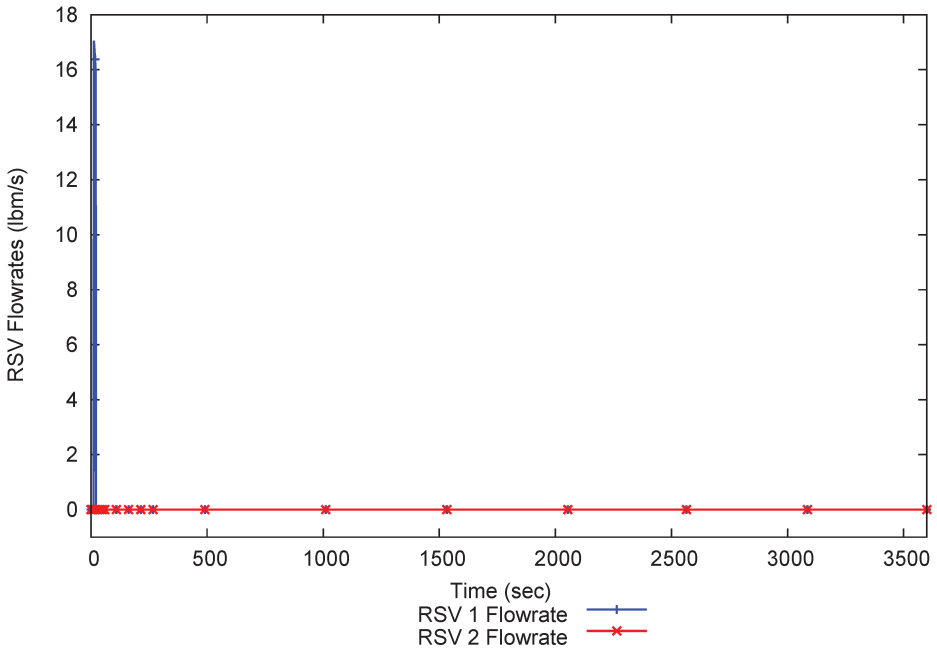


Figure 8-58 Reactor safety valve flow response for the representative feedwater line break event



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Figure 8-59 Decay heat removal system response for the representative feedwater line break event

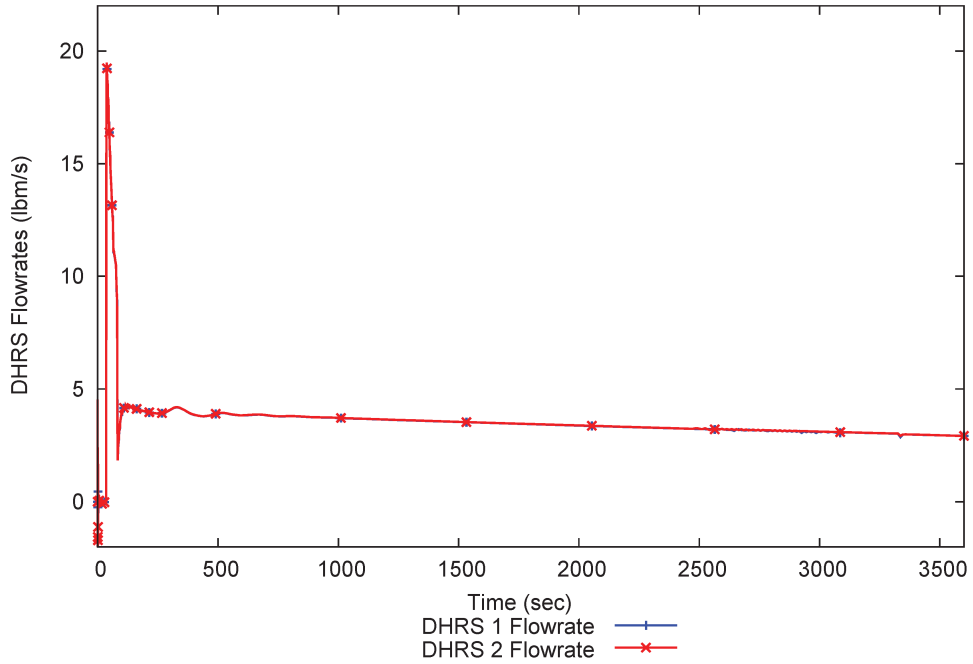
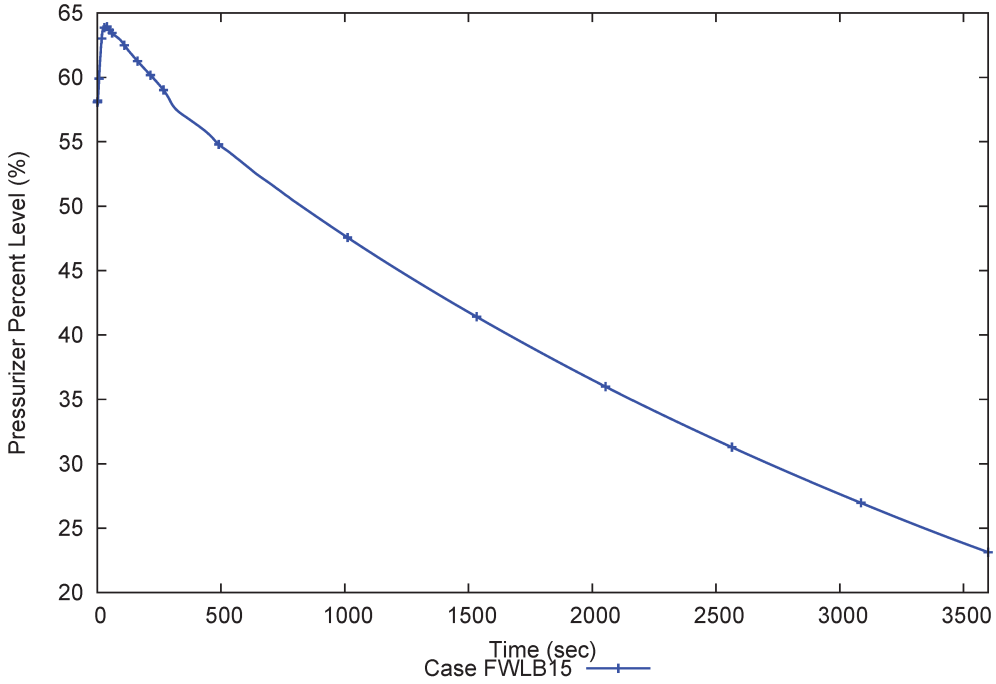
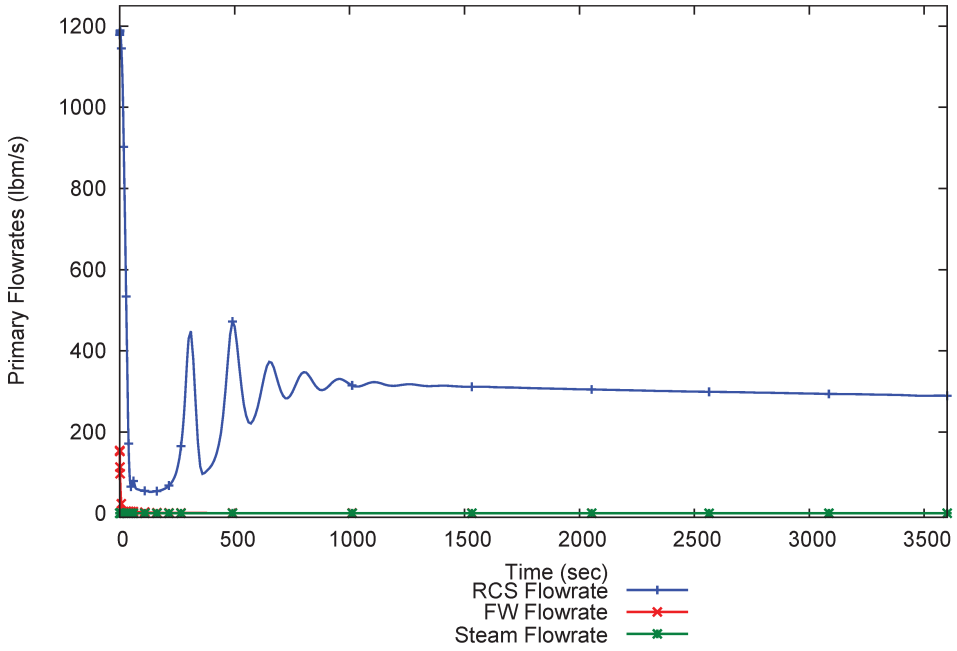


Figure 8-60 Pressurizer level response for the representative feedwater line break event



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Figure 8-61 Reactor coolant system flow response for the representative feedwater line break event



8.2.3.3 Conclusion

A representative case that could challenge the RCS pressure acceptance criteria was identified. The results of this case, as presented in Section 8.2.3.2, demonstrate the RCS pressure acceptance criterion is met.

8.3 Reactivity Anomaly

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

The purpose of this section is to present the thermal-hydraulic and core neutronic responses of a representative NPM for an uncontrolled control rod assembly bank withdrawal from subcritical or low power startup conditions event. This event is evaluated for both MCHFR and maximum fuel centerline temperature.

8.3.1.1 Event Description

The event description for a bank withdrawal from low power startup conditions can be found in Section 7.2.13. Based on Section 7.2.13, the acceptance criteria for MCHFR and maximum fuel centerline temperature are potentially challenged. For the representative NPM, the results of the sensitivity studies indicate the event scenario with the lowest MCHFR may differ from the event scenario with the

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highest fuel centerline temperature; but a single active failure cannot make the event consequences worse.

The representative case presented here corresponds to one of the more limiting peak power cases, and therefore a case that may challenge MCHFR. This case features the following conditions for a bank withdrawal from low power startup conditions:

- The initial core power is 15 percent RTP, which is the maximum power level for this event classification.
- The high count rate signal is inactive.
- The RCS core inlet temperature is the minimum allowed for criticality (420 degrees F).
- Heat removal is through the steam generators with coolant provided by the operating feedwater pump.
- The reactivity insertion rate is 35 pcm/s.
- No loss of normal AC power.
- No operator action was credited in the representative case.

8.3.1.2 Analysis Results

The following describes the event sequence of the withdrawal of a control rod assembly bank from a low power startup condition. Table 8-8 summarizes the sequence of events. Figure 8-62 through Figure 8-69 show some key parameters for this case.

The bank withdrawal occurs at time zero. The addition of positive reactivity (Figure 8-69) causes reactor power to increase (Figure 8-64). Since the high power rate signal is not active (below 15 percent RTP), core protection is provided by the high power signal and the startup rate (intermediate range) signal. In this instance, the analytical limit for the startup rate (intermediate range) is reached (3 DPM (decades per minute)) at 4 seconds. The control rods are free to fall after two seconds (Figure 8-69), while the DWS isolation valves begin to close. This latter action occurs as a precaution to the reactivity addition being caused by a dilution of the reactor coolant boron concentration. The peak core power (42.0 percent RTP) occurs at 7 seconds.

The quick action of the MPS to trip the reactor precludes the addition of significant energy into the reactor coolant. For instance, the pressurizer pressure increases by less than 10 psia (Figure 8-62), while the increase in SG pressures is comparable (Figure 8-63). The delay between energy production in the fuel and heat addition to the reactor coolant causes the core outlet temperature (Figure 8-67) and RCS flow rate (Figure 8-68) to remain relatively constant until increasing shortly before reactor trip. Both parameters peak at ~10 seconds before decreasing rapidly as the core heat flux reduces. The core inlet temperature (Figure 8-65) and density (Figure 8-66) remain relatively constant

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because the transient is terminated prior to completion of one loop transit. The MCHFR predicted by NRELAP5 (greater than 25) occurs at 7 seconds.

At high initial power levels, the calculation would normally be continued to verify that the module transitions into passive and stable DHRS cooling. However, the system responses at this operating condition are negligible compared to the heat removal capability of the plant (normal feedwater, containment flooding, etc.). Since there is little additional energy to remove, core cooling is ensured and the evaluation can be quickly terminated. As noted above, the RCS flow peaked and is starting to decrease (Figure 8-68), while the core outlet temperature (Figure 8-67) and pressurizer pressure (Figure 8-62) are steady or decreasing. No operator action was credited to mitigate this event.

Table 8-8 Withdrawal of a control rod assembly bank from a low power startup condition sequence of events

Event	Time (sec)
Malfunction that initiates the withdrawal of a CRA bank	0
High startup rate (intermediate range) analytical limit (3 DPM) reached	4
RTS actuation (control rods are free to fall)	6
DWS isolation (valves begin to close)	6
Maximum core power occurs (42.0% RTP)	7
Lowest MCHFR occurs (> 25 as calculated by NRELAP5)	7
Maximum RCS pressure occurs (1873 psia)	9
DWS isolation valves closed	11
Transient terminated	20

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Figure 8-62 Pressurizer pressure response for the bank withdrawal from a low power startup condition

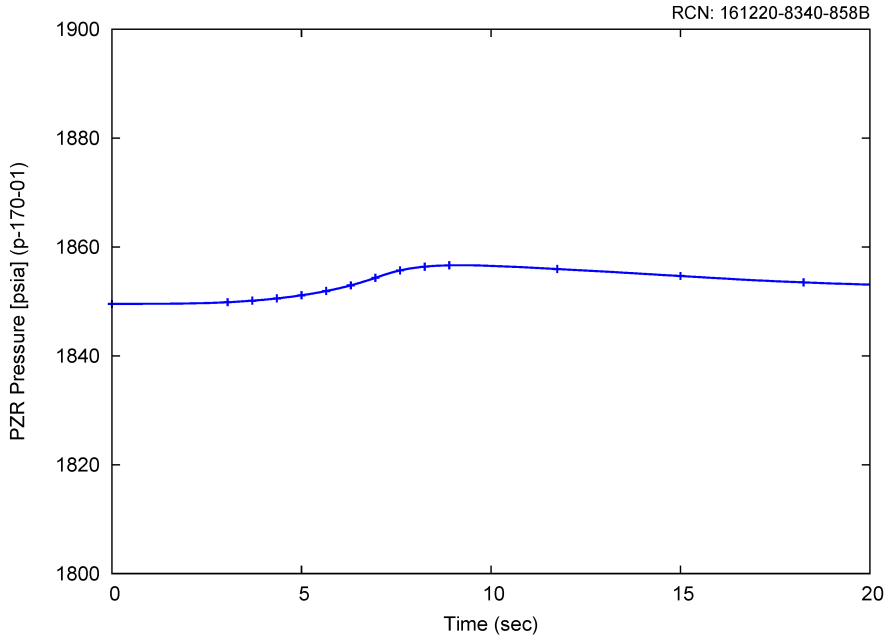
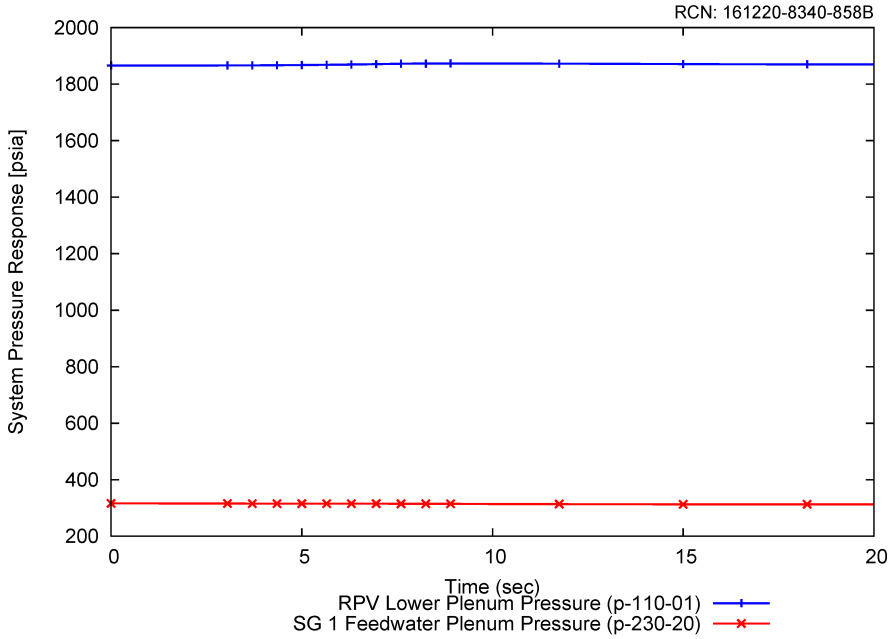


Figure 8-63 Reactor pressure vessel and steam generator pressure responses for the bank withdrawal from a low power startup condition



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Figure 8-64 Power response for the bank withdrawal from a low power startup condition

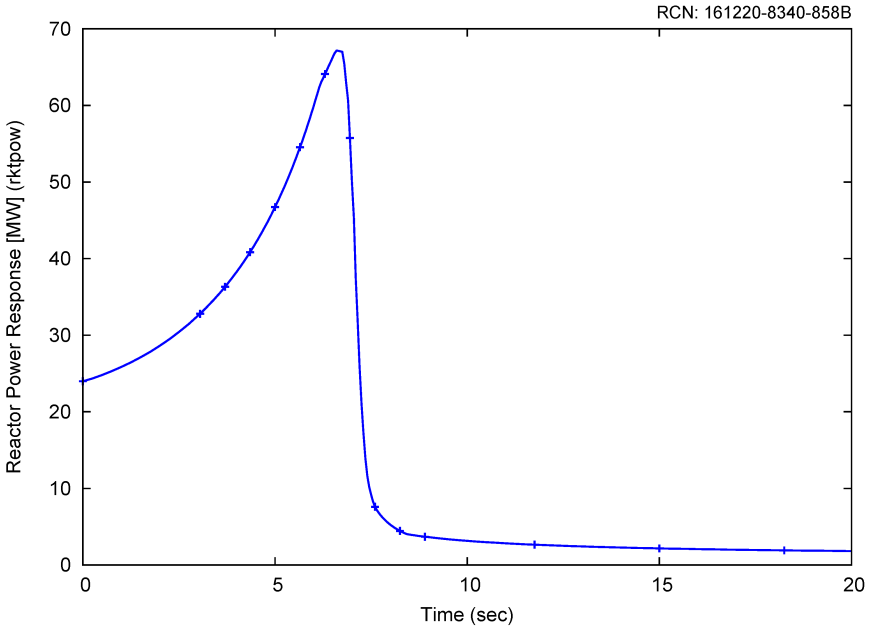
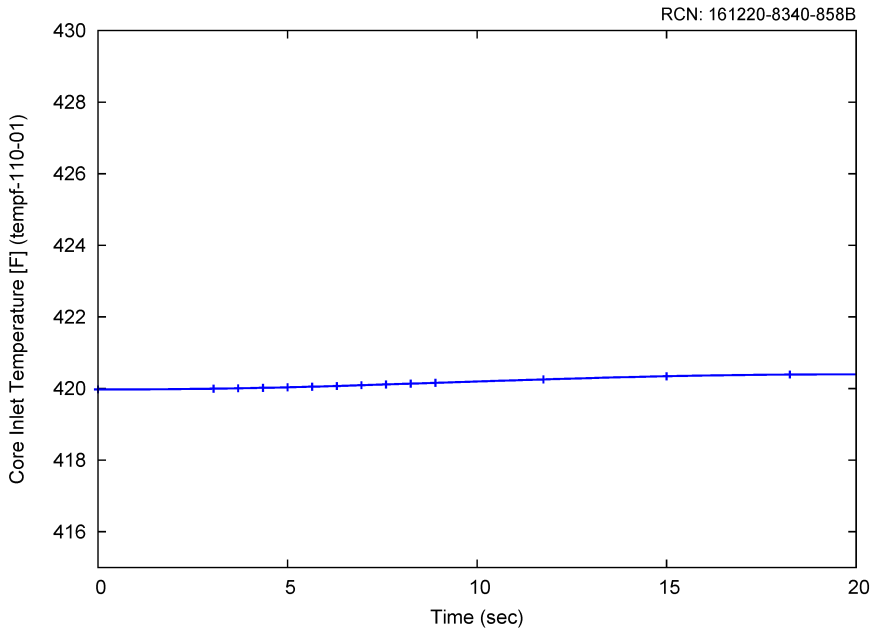


Figure 8-65 Core inlet temperature for the bank withdrawal from a low power startup condition



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Figure 8-66 Core inlet density for the bank withdrawal from a low power startup condition

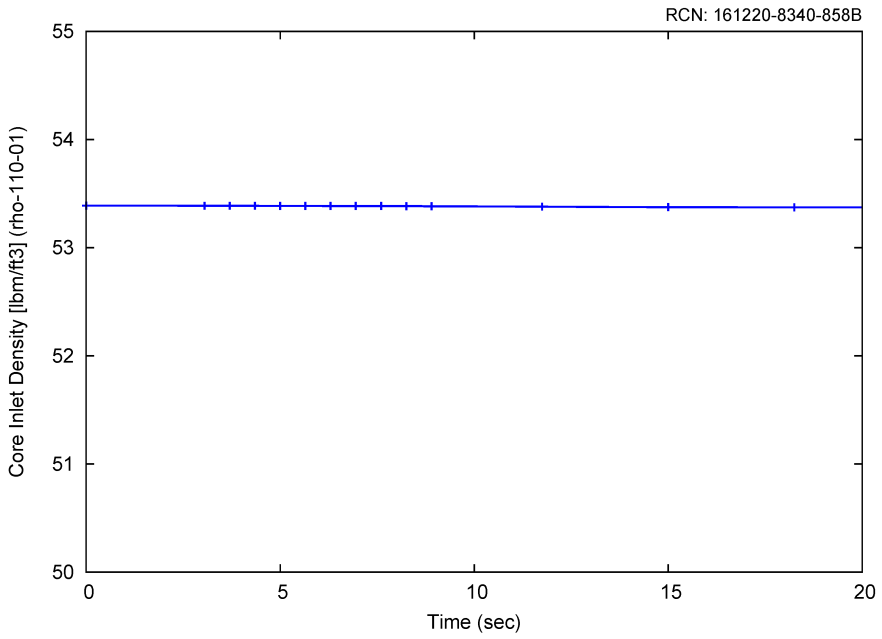
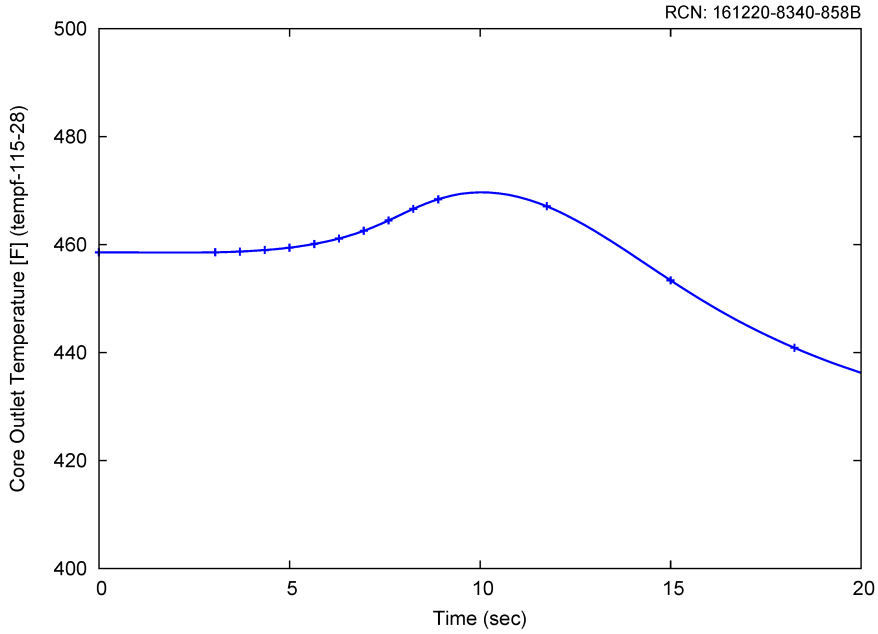


Figure 8-67 Core outlet temperature for the bank withdrawal from a low power startup condition



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Figure 8-68 Reactor coolant system flow rate for the bank withdrawal from a low power startup condition

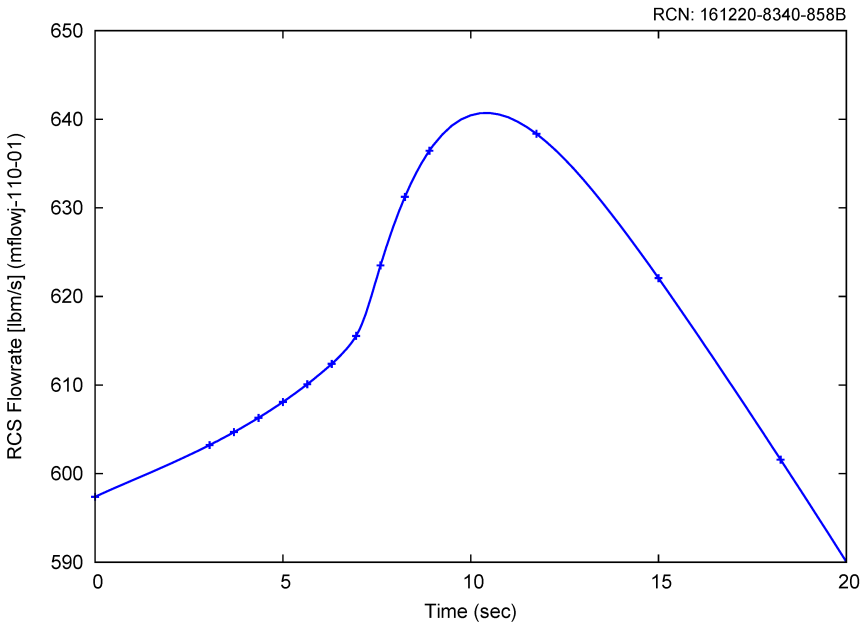
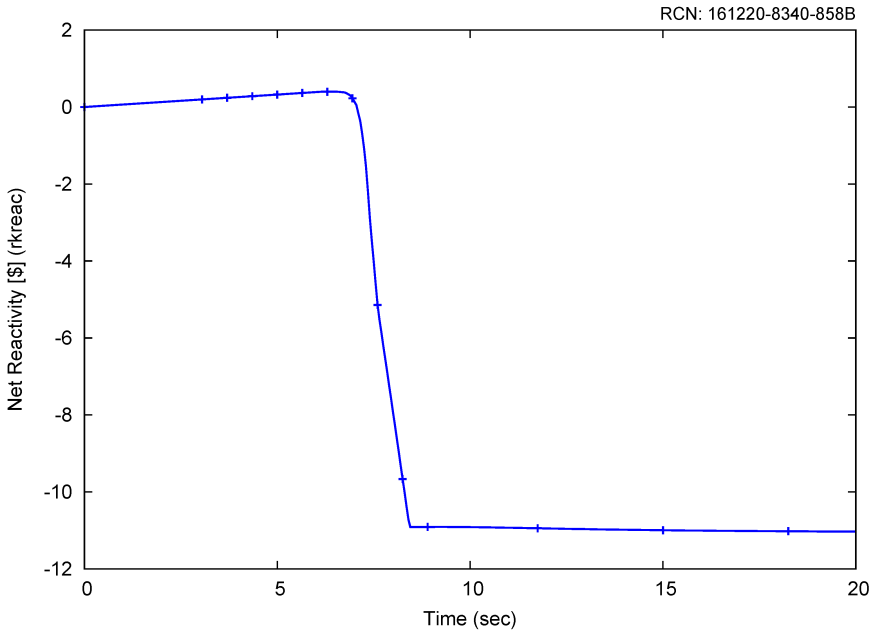


Figure 8-69 Net reactivity for the bank withdrawal from a low power startup condition



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8.3.1.3 Conclusion

A representative challenging case regarding MCHFR was identified for a bank withdrawal from a low power startup condition. The results of this case, as presented in Section 8.3.1.2, are subsequently used as input to an MCHFR evaluation using the NuScale subchannel analysis methodology.

8.3.2 Control Rod Misoperation

The purpose of this section is to present the thermal-hydraulic and core neutronic responses of a representative NPM for control rod misoperation events. Based on Section 7.2.15.1, the control rod misoperation analysis consists of two major transients: single rod withdrawal and rod drop. A representative single rod withdrawal transient is presented in this section. For the single rod withdrawal transient, MCHFR is the acceptance criterion that may be potentially challenged during the transient.

8.3.2.1 Single Rod Withdrawal MCHFR Case - Event Description

The general description for the control rod misoperation event can be found from Section 7.2.15.1. Chosen from a series of MCHFR sensitivity cases, the representative single rod withdrawal case presented here represents a case that could challenge MCHFR, based on the NRELAP5 MCHFR pre-screening. No single failure is applied since the challenging cases occur when all equipment operates as designed. No loss of power is applied since loss of power scenarios trips the reactor and does not make the MCHFR worse. This case features the following conditions:

- The initial power is 75 percent.
- Conservative initial condition biasing (as shown in Table 7-68) is applied in order to maximize the consequences of the single rod withdrawal event in terms of MCHFR. RCS flow rate is biased at low conditions for MCHFR. Average RCS temperature is biased at low condition (535 degrees F) for maximum delay to the high RCS riser temperature trip.
- To maximize the RCS pressure at time of trip without causing an earlier trip on pressurizer pressure, letdown is active, the pressurizer heater is off, and the spray flow is set to 97.6 percent of the CVCS recirculation flow. Additionally, the pressurizer pressure and level were given biases of -70 psi and -3 percent, respectively.
- The reactivity insertion rate is 2.3 pcm/s, which is the maximum that does not result in an earlier trip on high power rate.
- Automatic rod control is disabled since it will counteract the reactivity insertion due to the rod withdrawal at the beginning of the event.
- BOC reactivity coefficients are applied, which is conservative for overpower events.

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8.3.2.2 Single Rod Withdrawal MCHFR Case - Analysis Results

The following describes the event sequence of the single rod withdrawal MCHFR case. Table 8-9 summarizes the sequence of events. Figure 8-70 through Figure 8-78 show some key parameters for the single rod withdrawal MCHFR case.

The single rod withdrawal event begins at time zero. The addition of positive reactivity causes reactor power, pressurizer pressure and level to increase (Figure 8-70, Figure 8-71, and Figure 8-72). To bound the limiting single failure of an ex-core flux detector, the lowest reading ex-core detector is used to determine if the high power trip analytical limit of 120 percent is reached. Therefore the actual reactor power reaches a maximum of ~ 198 MWth, slightly exceeding the 120 percent limit, at ~ 147 s; this power level is not sufficient for the lowest reading ex-core detector to indicate that the 120 percent analytical limit is reached. As the power increases, the RCS riser temperature increases; in this case, the high RCS riser temperature limit of 610 degrees F is reached at ~139 seconds (Figure 8-73). There is a total 8 second delay between the high RCS riser temperature signal and RTS/DHRS actuation. RTS and DHRS are actuated at ~147 seconds. The limiting MCHFR is also reached at ~147 seconds.

DHRS actuation closes the FWIVs and MSIVs. Steam generator pressure increases as a result of the MS isolation. SG pressure starts to decrease once the DHRS cooling is established (Figure 8-74, SG2 shown but SG1 response is identical). Pressurizer pressure reaches its peak value, less than 2100 psia, around 157 seconds and starts to decrease. Pressurizer level has a similar transient response (Figure 8-71 and Figure 8-72).

After reactor trip and actuation of DHRS, RCS flow decreases rapidly and becomes stagnant and slightly reversed at ~184 seconds (Figure 8-75). Following that, flow oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2 and shown by Figure 8-73, Figure 8-75, and Figure 8-77). By 40 minutes, RCS flow and DHRS flow have stabilized (Figure 8-75 and Figure 8-76), and RCS temperature and pressure are steadily decreasing as the DHRS transfers decay heat from the RCS to the reactor pool (Figure 8-71, Figure 8-73, and Figure 8-77). It is concluded that by 40 minutes the transient has been terminated and that stable DHRS cooling has been achieved. Subcritical margin is verified as net reactivity remains less than 0.0 dollars at the time stable DHRS cooling has been achieved (Figure 8-78). No operator action was credited to mitigate this event.

Table 8-9 Single rod withdrawal sequence of events - MCHFR case

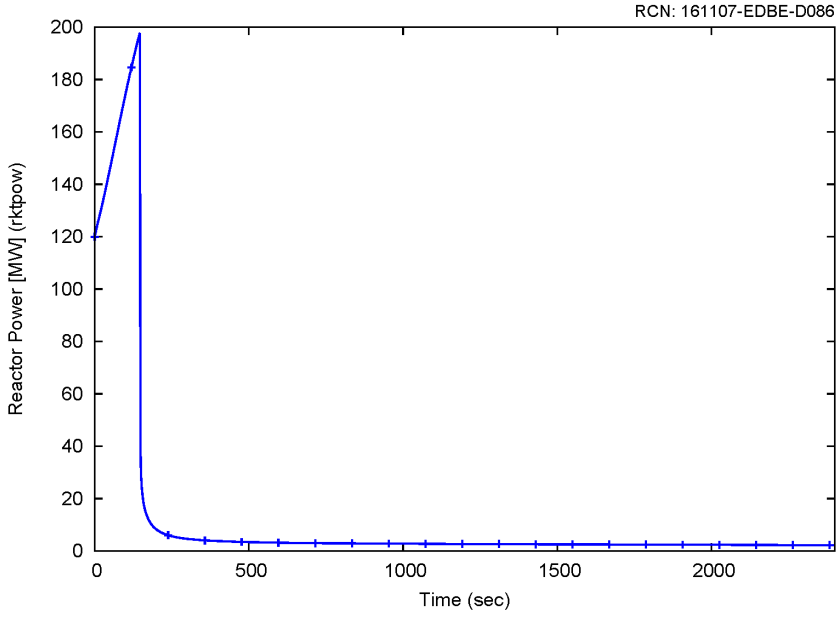
Event	Time (sec)
Malfunction that initiates the withdrawal of a single CRA.	0
High RCS riser temperature analytical limit (610°F) is reached	139
High pressurizer pressure analytical limit (2000 psia) is reached	145

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Table 8-9 Single rod withdrawal sequence of events - MCHFR case (Continued)

Event	Time (sec)
RTS is actuated and control rod insertion begins	147
DHRS is actuated. The DHRS actuation valves begin opening. The FWIVs and MSIVs begin closing.	147
Lowest MCHFR is reached (3.107 as calculated by NRELAP5)	147
RCS flow is stagnant and slightly reversed.	~184
End of calculation. Stable DHRS cooling has been established. Net reactivity remains < $0.0.	2400

Figure 8-70 Power response for the representative single rod withdrawal MCHFR case



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Figure 8-71 Reactor pressure vessel pressure response for the representative single rod withdrawal MCHFR case

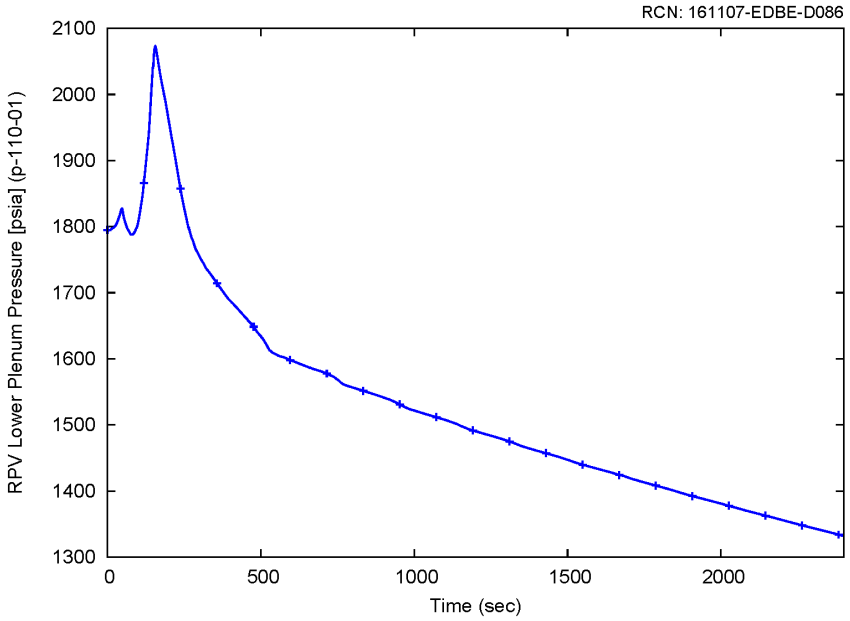
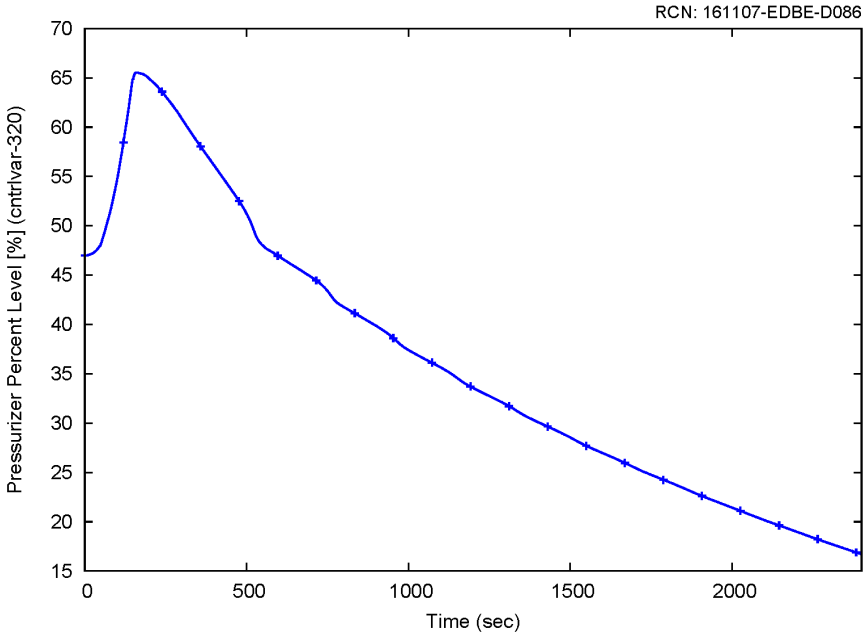


Figure 8-72 Pressurizer level for the representative single rod withdrawal MCHFR case



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Figure 8-73 Core outlet temperature for the representative single rod withdrawal MCHFR case

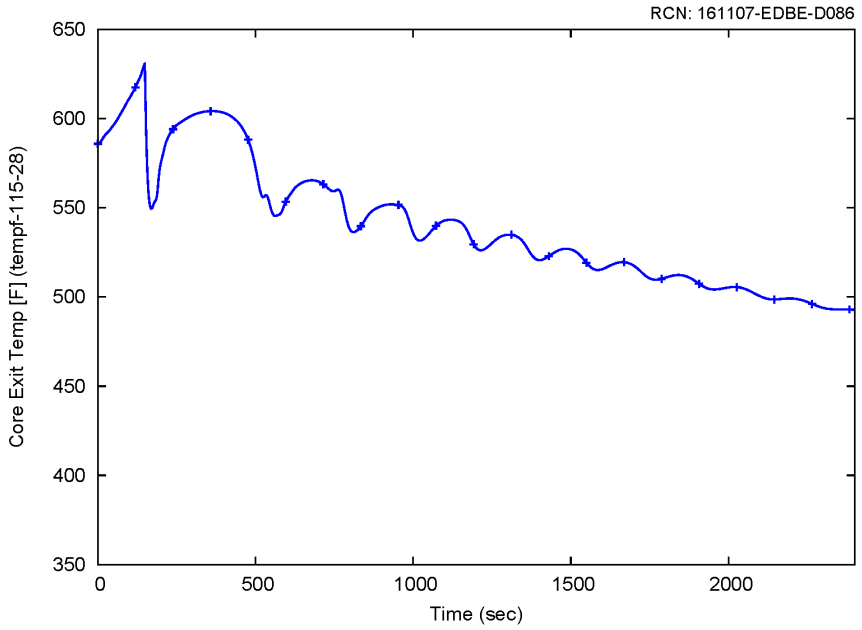
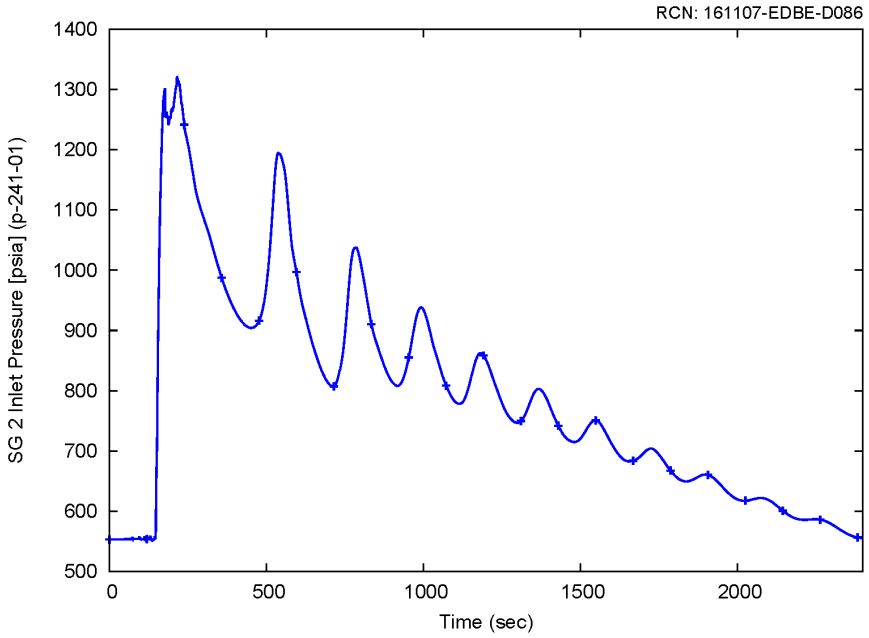


Figure 8-74 Steam generator 2 pressure response for the representative single rod withdrawal MCHFR case



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Figure 8-75 Reactor coolant system flow rate for the representative single rod withdrawal MCHFR case

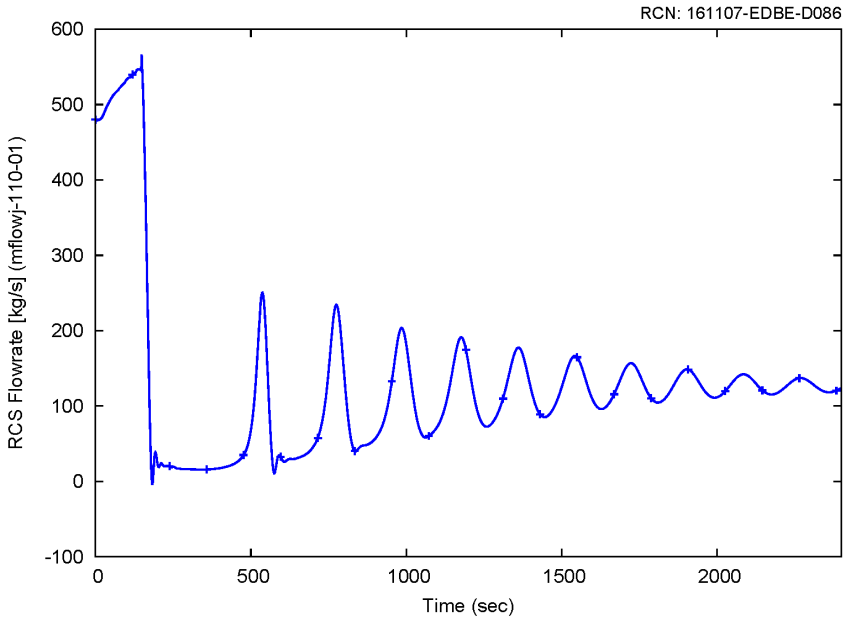
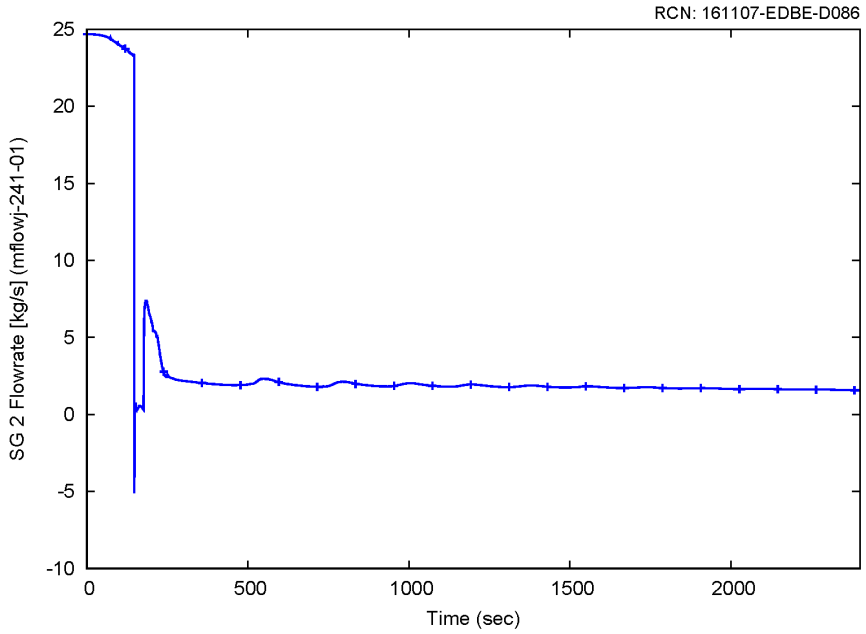


Figure 8-76 Steam generator 2 secondary flow for the representative single rod withdrawal MCHFR case



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Figure 8-77 Core inlet temperature for the representative single rod withdrawal MCHFR case

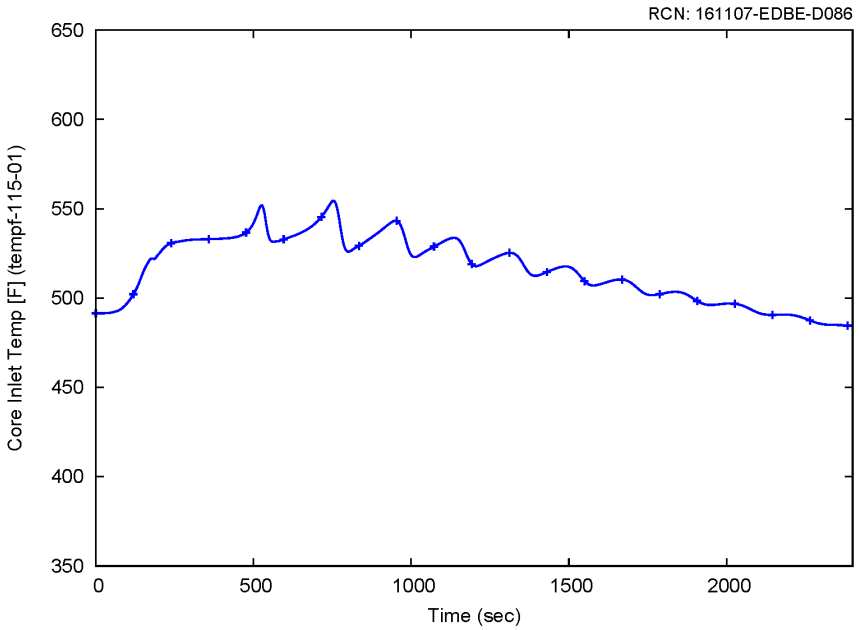
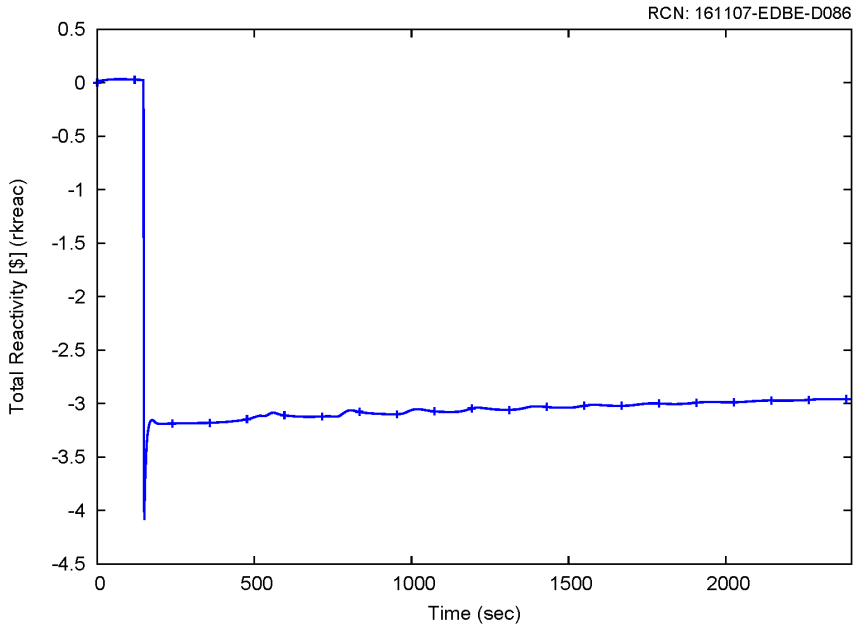


Figure 8-78 Net reactivity for the representative single rod withdrawal MCHFR case



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8.3.2.3 Single Rod Withdrawal MCHFR Case - Conclusion

A representative case that could challenge MCHFR was identified for the single rod withdrawal event. The results of this case, as presented in Section 8.3.2.2, are subsequently used as input to an MCHFR evaluation using the NuScale subchannel analysis methodology.

8.4 Increase in Reactor Coolant System Inventory

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

The purpose of this section is to present the thermal-hydraulic response of a representative NPM for a CVCS malfunction that increases the RCS inventory. This event is evaluated for primary and secondary pressure. A representative case evaluated for primary pressure is presented.

8.4.1.1 Event Description

The general event description associated with a malfunction of the CVCS that increases RCS inventory is provided in Section 7.2.17. Based on Section 7.2.17, the RCS primary pressure is the acceptance criterion that may be challenged during the CVCS malfunction event. A representative case that could challenge the RCS primary pressure is presented here. This case features the following conditions:

- Conservative initial condition biasing is applied to maximize the consequences of the RCS inventory increase event. This case is initialized at 102 percent reactor power, low pressurizer pressure and high pressurizer level. Minimum RCS design flow is assumed. Low RCS average temperature is assumed.
- Net CVCS mass flow rate of 5.4 lbm/s (40 gpm) into the RPV (this includes CVCS recirculation flow rate into and out of the RPV)
- High CVCS makeup temperature of 150 degrees F
- Pressurizer spray and letdown assumed unavailable

8.4.1.2 Analysis Results

The following describes the event sequence of the representative CVCS malfunction that increases RCS inventory event. Table 8-10 summarizes the sequence of events. Figure 8-79 through Figure 8-91 show some key parameters during the representative increase in RCS inventory event.

For this event, constant makeup flow is injected into the RCS until the CVCS is isolated (Figure 8-79). The CVCS also recirculates flow (Figure 8-80); no letdown flow is modeled (Figure 8-81). Therefore, there is a net increase of flow into the RCS (Figure 8-82 and Figure 8-83).

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Upon initiation of makeup, pressurizer level and RCS pressure were observed to rise. Pressurizer pressure and level are shown in Figure 8-84 and Figure 8-85, respectively.

RCS pressure reached the high pressurizer pressure analytical limit, actuating reactor trip (Figure 8-86) and DHRS (Figure 8-88 and Figure 8-89, curves for SG1 and SG2 overlap). In this case, the normal control system is assumed to trip the turbine on reactor trip, which reduced secondary side heat removal, increasing the rate of RCS pressurization. The inventory addition continued following reactor trip because RCS flow remained above 0 lbm/s (Figure 8-87) and therefore CVCS remained un-isolated.

After reactor trip and actuation of DHRS, RCS flow oscillations were observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2). Oscillations can be seen in Figure 8-87, Figure 8-90 and Figure 8-91.

After reactor trip, the pressurizer pressure and level began to decrease for a short period of time but the makeup flow continued and the pressurizer level and primary pressure eventually began to increase again. Once the RSV 1 lift pressure was reached, RCS pressure was reduced as vapor from the pressurizer was vented into the containment vessel. A peak pressure of 2155 psia was reached in the RPV. Inventory increase continued for approximately 100 seconds further before the high pressurizer level analytical limit was reached. After reaching the high pressurizer level of 80 percent, CVCS was isolated and the event ended. Pressurizer level never exceeds 80 percent, thus ensuring a steam bubble in the pressurizer throughout this transient.

The calculation is continued to verify that the module transitions into passive and stable DHRS cooling. Figure 8-84, Figure 8-87, Figure 8-88, Figure 8-89, and Figure 8-90 show that at ~ 1 hour, RCS and DHRS flows have stabilized, primary and secondary pressures are decreasing, and the RCS temperatures are steadily decreasing as the DHRS is transferring decay heat from the RPV to the reactor pool. Shutdown margin is maintained at the end of the transient calculation (Figure 8-91). It is concluded that by ~ 1 hour the transient has been terminated and that stable DHRS cooling has been achieved.

Table 8-10 Increase in reactor coolant system inventory sequence of events

Event	Time (sec)
CVCS malfunction initiates an excess CVCS mass flow rate of 5.4 lbm/s into the RPV	0.0
Analytical limit for high pressurizer pressure (2000 psia) is reached	510.9
Turbine stop valve closed	511.8
RTS[1] and DHRS actuation on high pressurizer pressure analytical limit	512.9
Primary MSIVs fully closed	518.8
RSV 1 actuates	3370.9

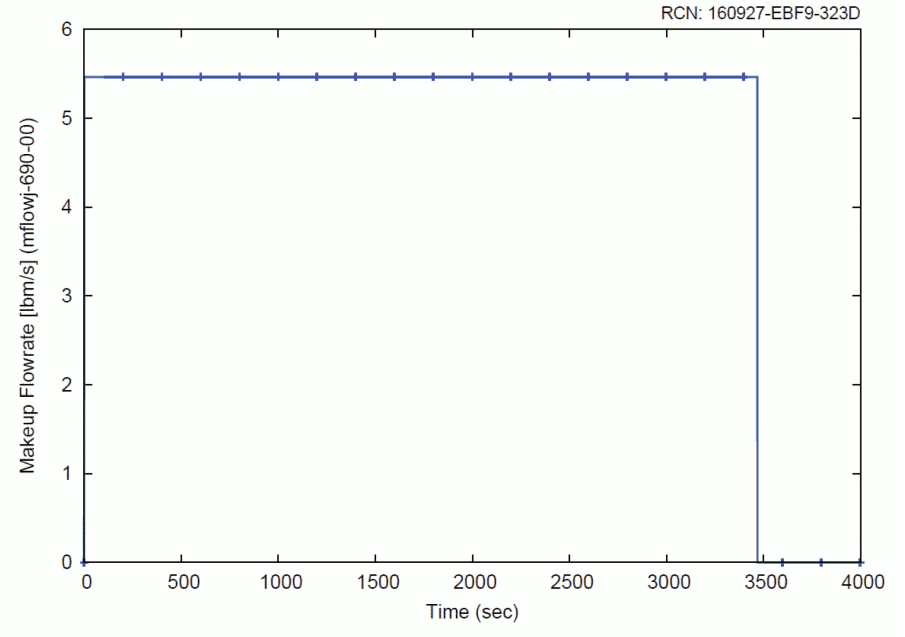
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Table 8-10 Increase in reactor coolant system inventory sequence of events (Continued)

Event	Time (sec)
Peak RCS pressure is reached (2155 psia)	3371.0
Analytical limit for high pressurizer level (80%) is reached	3465.9
CVCS isolation actuation on high pressurizer level analytical limit	3468.9
CVCS isolation valves closed on high pressurizer level analytical limit	3473.9
End of calculation. Stable DHRS cooling has been established.	4000

1. Control rods are assumed to be fully inserted 2.278 seconds following RTS actuation.

Figure 8-79 Makeup flow for increase in reactor coolant system inventory



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Figure 8-80 Recirculation pump flow for increase in reactor coolant system inventory

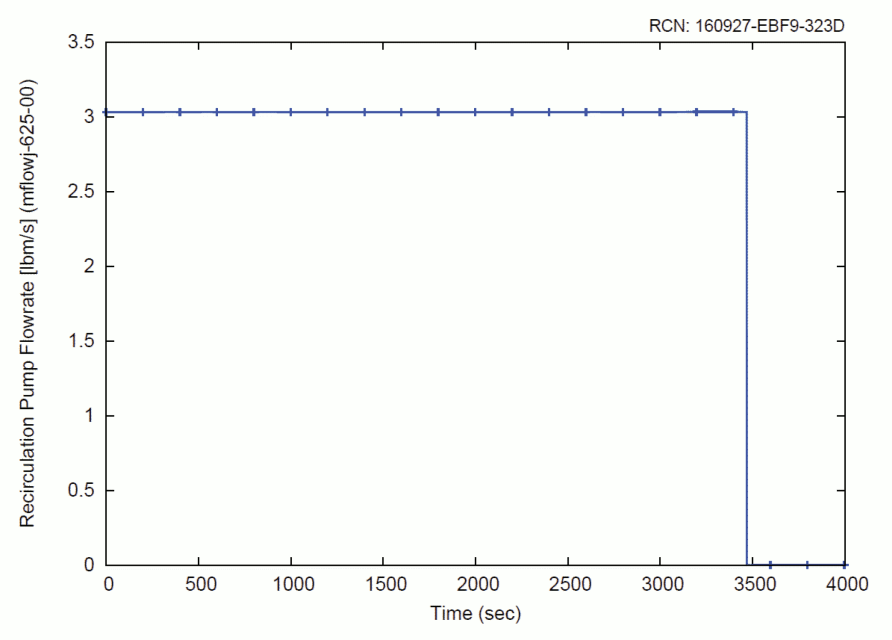
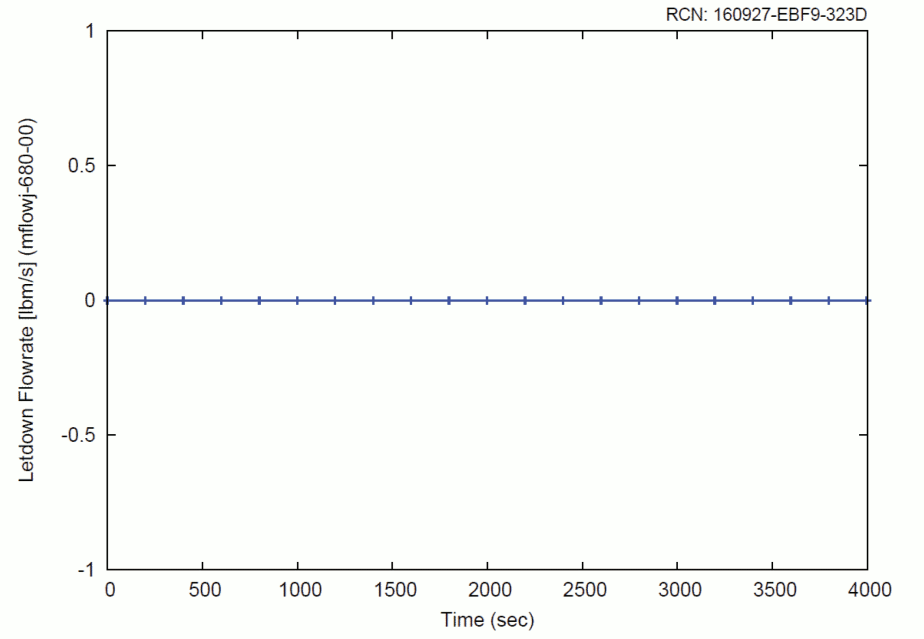


Figure 8-81 Letdown flow for increase in reactor coolant system inventory



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Figure 8-82 CVCS recirculation flow rate into the reactor pressure vessel for increase in reactor coolant system inventory

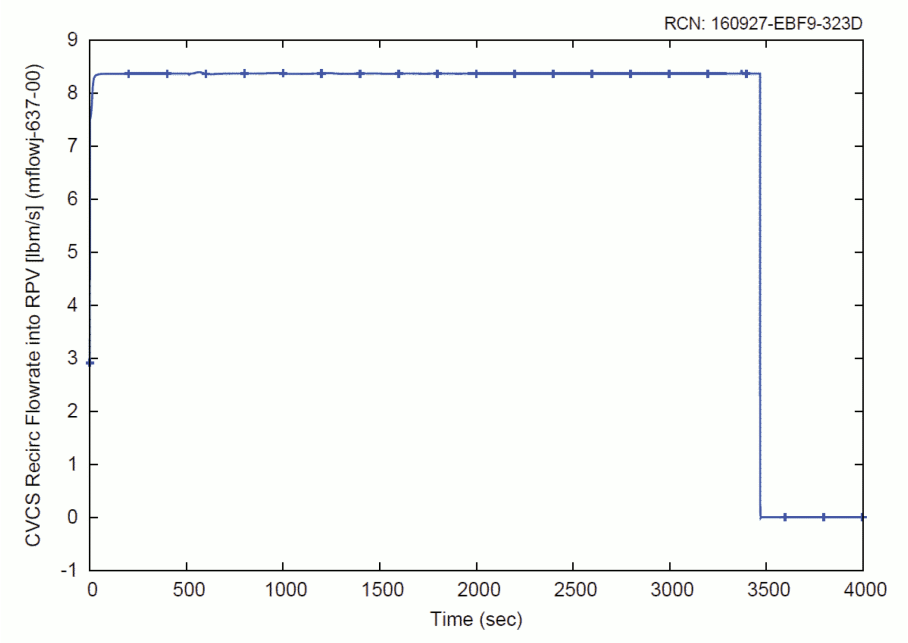
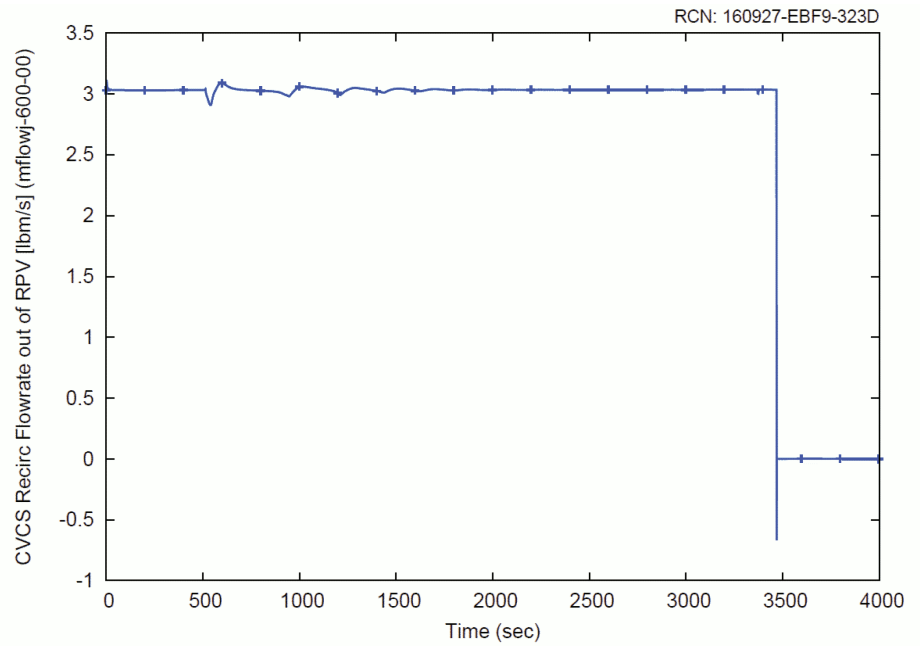


Figure 8-83 CVCS recirculation flow rate out of the reactor pressure vessel for increase in reactor coolant system inventory



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Figure 8-84 Pressure at the bottom of the pressurizer for increase in reactor coolant system inventory

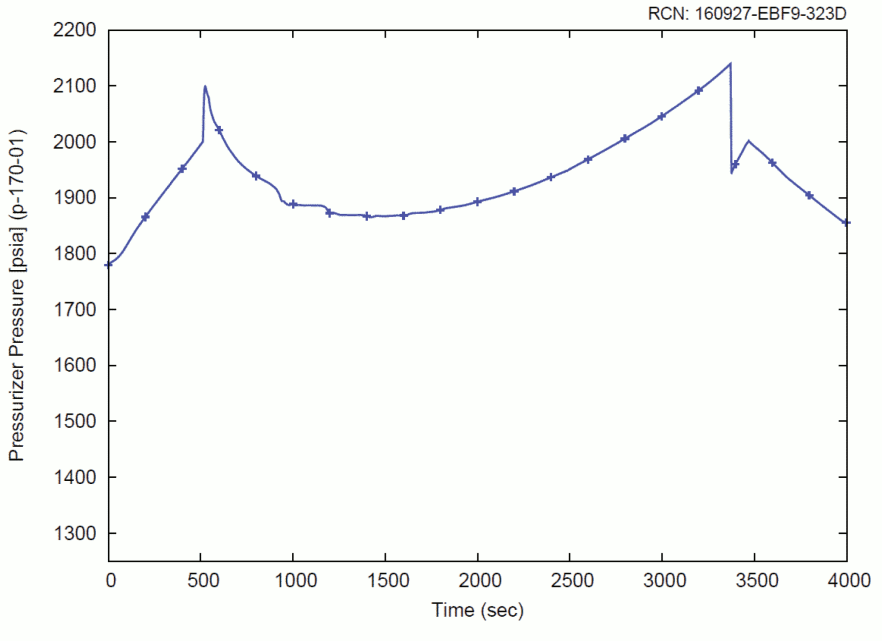
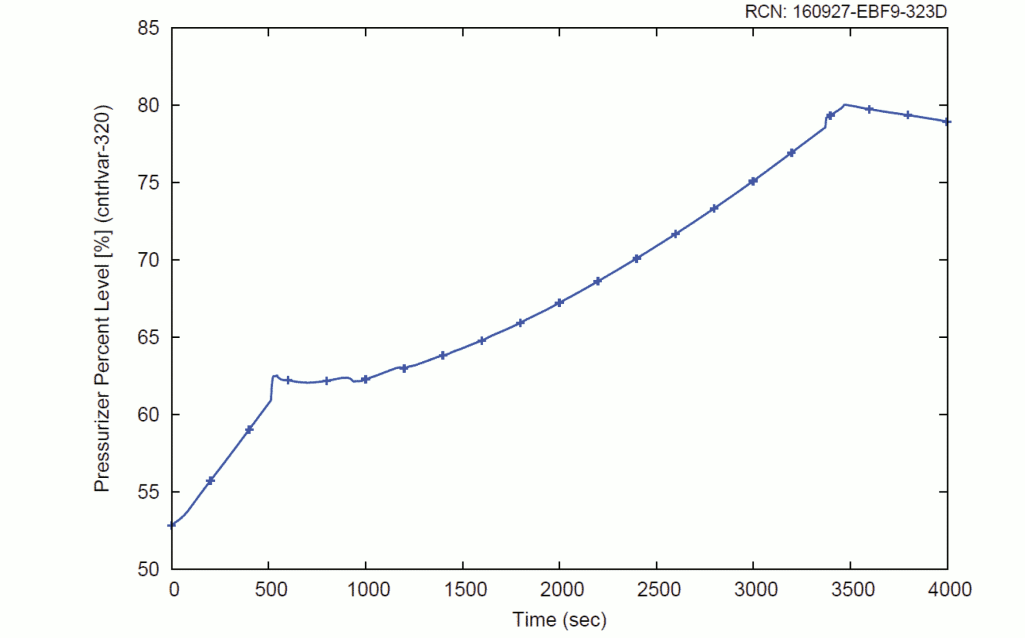


Figure 8-85 Pressurizer level for increase in reactor coolant system inventory



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Figure 8-86 Reactor power for increase in reactor coolant system inventory

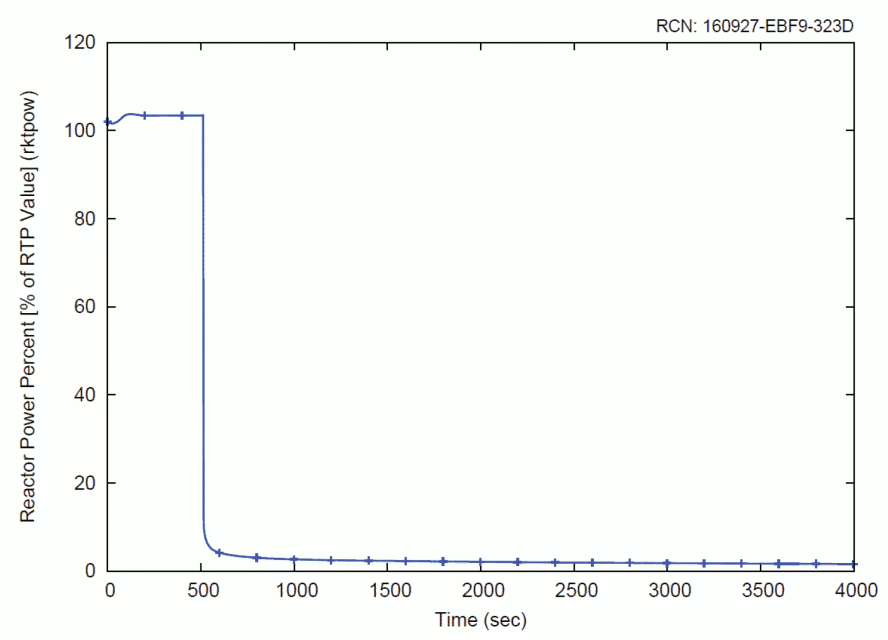
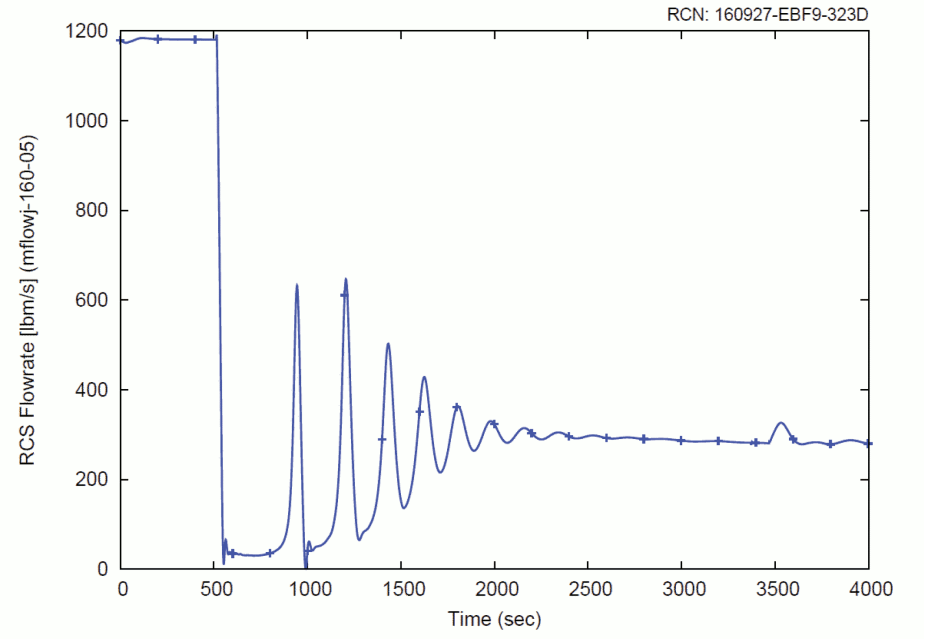


Figure 8-87 Reactor coolant system flow for increase in reactor coolant system inventory



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Figure 8-88 Decay heat removal system flow rate for increase in reactor coolant system inventory

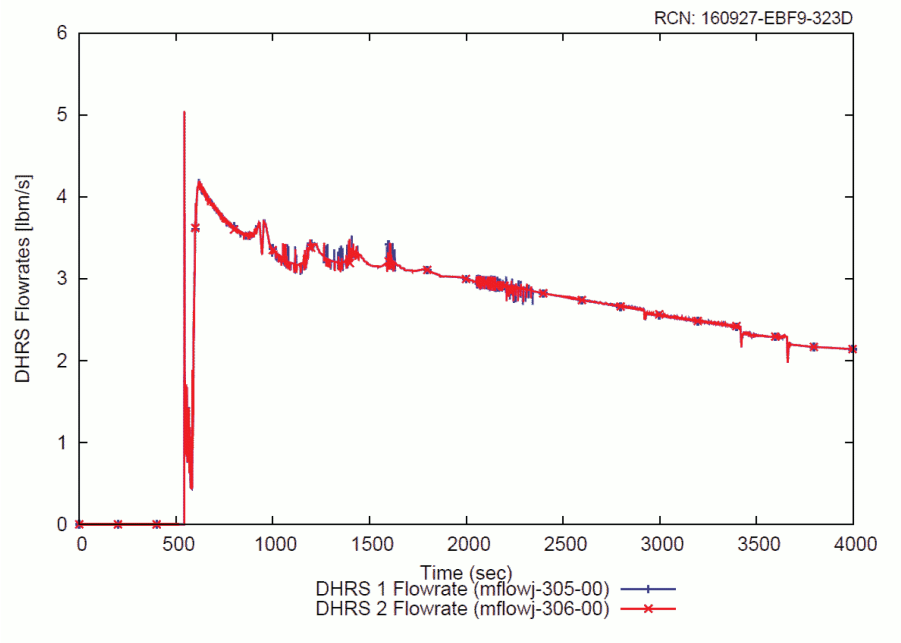
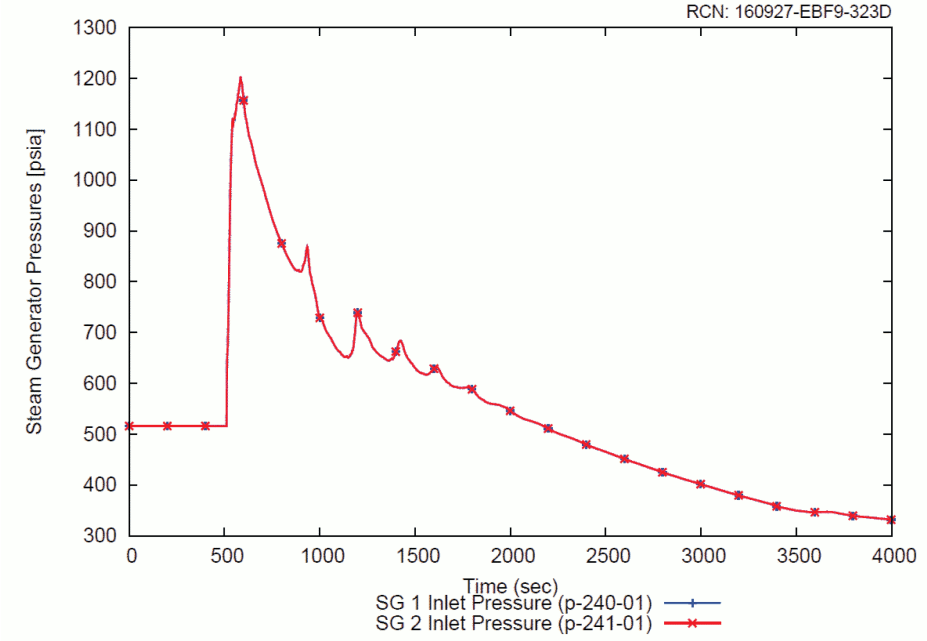


Figure 8-89 Steam generator pressure for increase in reactor coolant system inventory



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Figure 8-90 Core inlet and exit coolant liquid temperature for increase in reactor coolant system inventory

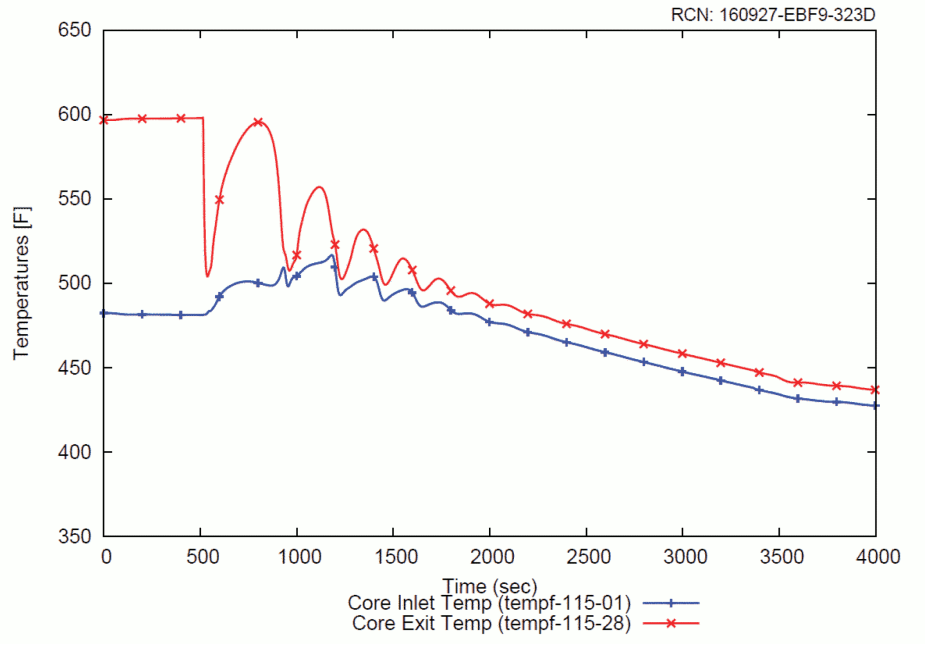
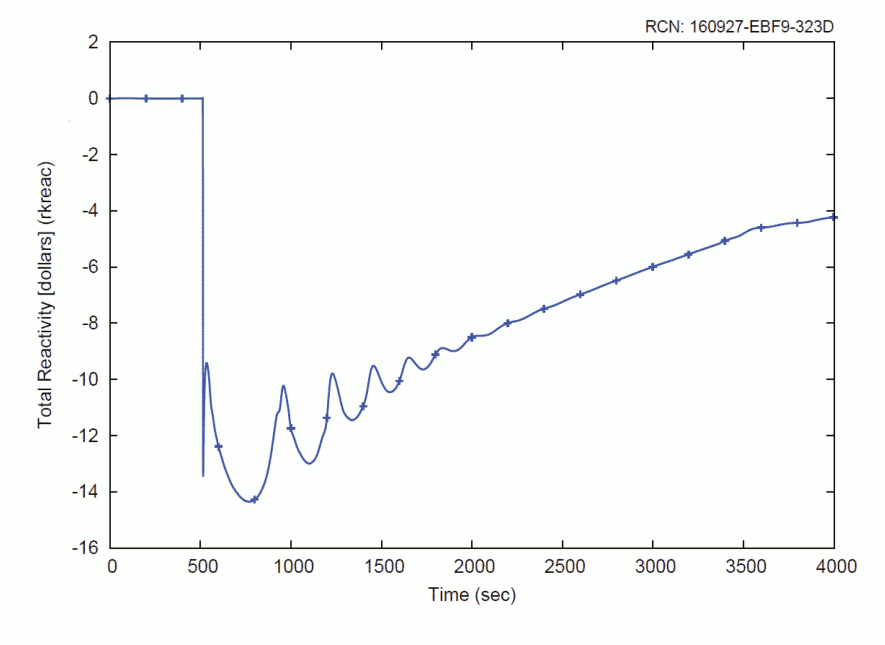


Figure 8-91 Total reactivity for increase in reactor coolant system inventory



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8.4.1.3 Conclusion

A representative case that could challenge the RCS pressure acceptance criterion was identified. The results of this case, as presented in Section 8.4.1.2, demonstrate the RCS pressure acceptance criterion is met.

8.5 Decrease in Reactor Coolant System Inventory

8.5.1 Small Line Break Outside of Containment

The purpose of this section is to present the thermal-hydraulic and core neutronic responses of a representative NPM for a postulated break in a small line carrying primary coolant. For the reasons discussed in Section 7.2.18, the lines evaluated are the CVCS charging lines, the CVCS letdown lines, and the CVCS pressurizer spray lines. This event is evaluated for offsite and onsite radiological dose consequences. The transient analysis results are input to downstream radiological consequences evaluations and downstream subchannel evaluations. This analysis is not intended to be used for 10 CFR 50 Appendix K compliance for a LOCA as the separate LOCA-EM topical report encompasses this subject.

8.5.1.1 Event Description

The event description for a break in a small line carrying reactor coolant can be found in from Section 7.2.18. Based on Section 7.2.18, the acceptance criteria for radiological consequences are potentially challenged by the break in a small line carrying reactor coolant event; but a single active failure cannot make the event consequences worse.

Radiological Consequences

The representative case presented here corresponds to one of the more limiting integrated mass released cases, and therefore a case with higher dose consequences. This case features the following conditions for a DEG break of 100 percent area in the letdown line:

- Conservative initial condition biasing (as shown in Table 7-86) is applied in order to maximize the consequences of this event. This representative case is initialized at 102 percent reactor power.
- The initial RCS average temperature is biased to the maximum value (555 degrees F) to maximize the mass released through the break.
- The initial feedwater temperature is biased to the minimum value (~290 degrees F) in this case.
- The initial pressurizer pressure is biased to the maximum value (1920 psia) to delay actuation of the low pressurizer pressure RTS and ESFAS protection signals.

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- The initial pressurizer level is biased to the maximum value (68 percent) to delay actuation of the low pressurizer level RTS and ESFAS protection signals.
- EOC reactivity coefficients are applied to obtain the mass released through the break associated with a specific time in core life.
- A loss of normal AC Power at event initiation is utilized, consistent with Table 7-87, to maximize the mass released through the break.
- No operator action was credited in the representative case.

8.5.1.2 Analysis Results

The following describes the event sequence of the representative small break outside CNV event. Table 8-11 summarizes the sequence of events. Figure 8-92 through Figure 8-102 show some key parameters during the representative small break outside CNV event.

A letdown line break of 100 percent area occurs at time zero with a coincident loss of AC power. Critical flow conditions are quickly established at the break location causing the break flow rate to stabilize (Figure 8-92). The reactor coolant lost through the break causes an immediate decrease in pressurizer level (Figure 8-93) and RPV pressure (Figure 8-94). In contrast, the SG pressure immediately increases (Figure 8-95) as a result of the turbine trip induced by the loss of AC power. The reduced secondary heat sink associated with tripping the feedwater pumps on loss of AC power is sufficient to cause the pressurizer level and RPV pressure to begin to increase. The exit pressures of the SGs reach the analytical limit for high steam line pressure (800 psia) at 11.7 seconds. Another two seconds is needed before the control rods are free to fall (Figure 8-96). Coincident with reactor trip at 13.7 seconds, the DHRS valves begin to open; the FWIVs begin to close; the MSIVs begin to close; and, the makeup line break opens fully (Figure 8-92). Opening the break in the makeup line causes the total break flow rate to increase sharply (Figure 8-92), then decrease and stabilize as critical flow conditions are established through the makeup line. Closing the FWIVs and MSIVs isolates the SGs from the remaining secondary system.

The system shrinkage associated with the reactor trip works in conjunction with the increased break flow to cause the pressurizer level and RPV pressure to decrease. The analytical limit for low pressurizer level (35 percent) is reached at 86.3 seconds, and the pressurizer heaters are de-energized two seconds later. The pressurizer pressure continues to decrease until the analytical limit for low pressurizer pressure (1600 psia) is reached at 89.9 seconds. Containment isolation is initiated after a delay of 2 seconds. Closing the CVCS isolation valves terminates the break flow (Figure 8-92). The maximum integrated break flow of 12,940 lbm is reached at 97 seconds (Figure 8-97).

After reactor trip and actuation of DHRS, oscillations are observed due to temperature and density differences between the riser and downcomer (as

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discussed in Section 7.2); therefore the calculation is continued to verify that the module transitions into passive and stable DHRS cooling. At ~25 minutes, RCS flow (Figure 8-98) has stabilized, and the RCS temperature (Figure 8-99) and pressure (Figure 8-100) are steadily decreasing as the DHRS transfers decay heat from the RPV to the reactor pool. The net reactivity (Figure 8-101) becomes negative shortly after reactor trip and remains negative during the transition to stable DHRS cooling. Lastly, the RPV level (Figure 8-102) remains well above the top of the core throughout this transient. It is concluded that, by 25 minutes, the transient has been terminated and that stable DHRS cooling has been achieved. No operator action was credited to mitigate this event.

Table 8-11 Sequence of events for small line breaks carrying primary coolant outside containment

Event	Time (sec)
Letdown line break (DEG, 100% area) occurs with coincident loss of normal AC power; TSVs begin to close.	0
Peak reactor power is reached (163.6 MW).	0
TSVs are fully closed.	0.1
Limiting MCHFR is reached (5.009 as calculated by NRELAP5).	1.0
High steam line 1 pressure limit is reached (800 psia).	11.7
High steam line 2 pressure limit is reached (800 psia).	11.7
RTS actuation on high steam line 1 pressure signal, control rods are inserted into the core.	13.7
CVCS makeup line break (DEG, 100% area) occurs.	13.7
DHRS actuation on the high steam line 1 pressure signal. DHRS actuation valves begin to open. FWIVs and MSIVs begin to close.	13.7
Peak RPV pressure is reached (1983 psia).	13.9
FWIVs and MSIVs are fully closed.	18.7
DHRS actuation valves are fully open.	43.7
Low PZR level limit is reached (35%). PZR heaters are disabled after 2 second delay.	86.3
Low PZR pressure is reached (1600 psia). Containment and CVCS isolation begins after 2 second delay.	89.9
CVCS isolation valves are fully closed.	96.9
Maximum integrated RCS break flow (11,940 lbm) occurs.	97
Establishment of stable RCS flow. Pressure and temperature are steadily decreasing.	1500
End of calculation. Stable DHRS cooling has been established.	3000

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Figure 8-92 Instantaneous break flow response (0 to 350 sec) for the representative small break outside containment event

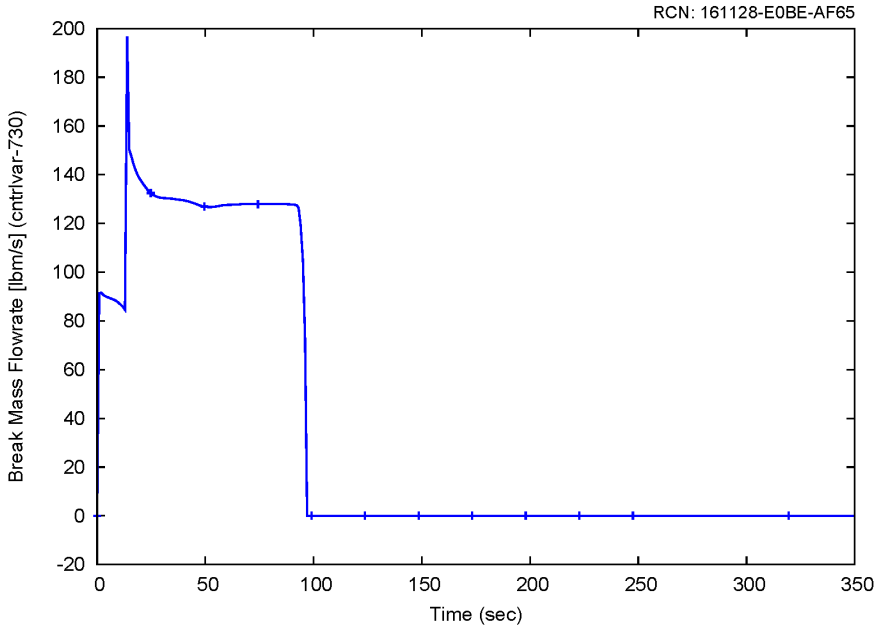
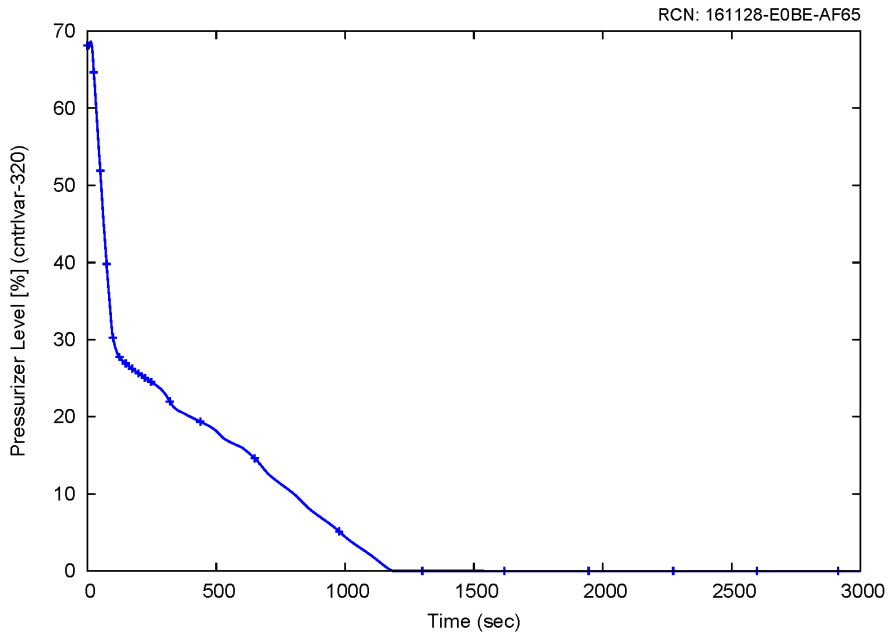


Figure 8-93 Pressurizer level response for the representative small break outside containment event



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Figure 8-94 Reactor pressure vessel pressure response (0 to 350 sec) for the representative small break outside containment event

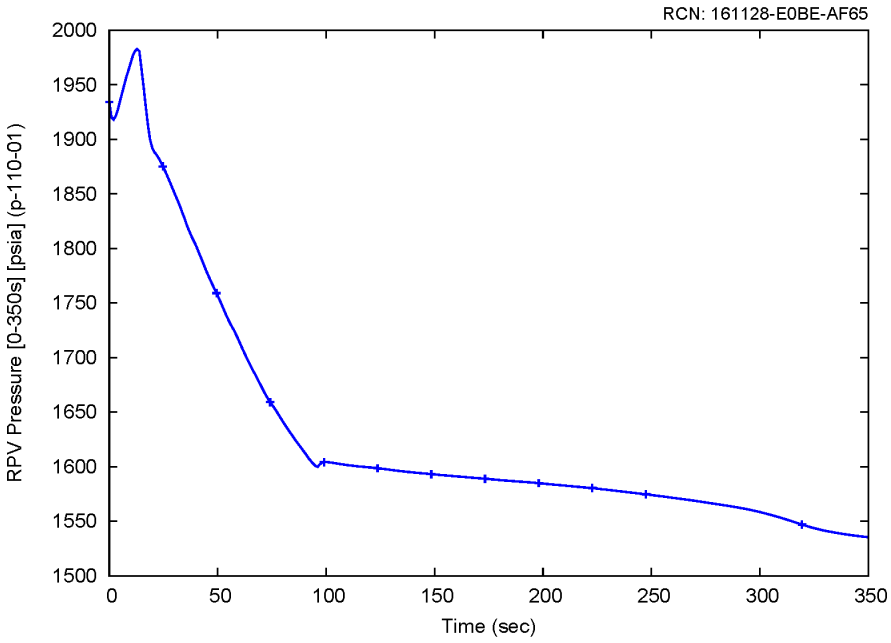
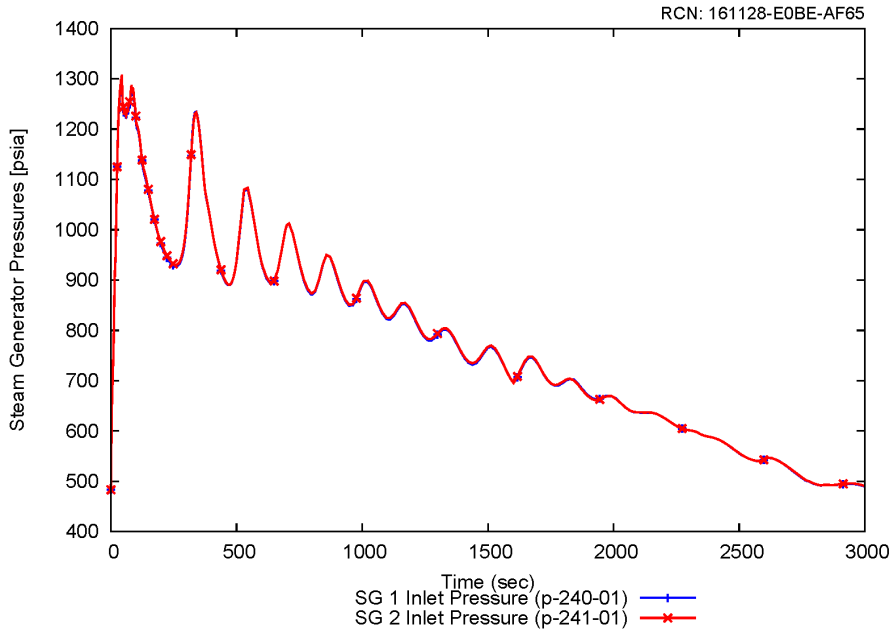


Figure 8-95 Steam generator pressure responses for the representative small break outside containment event



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Figure 8-96 Core power response for the representative small break outside containment event

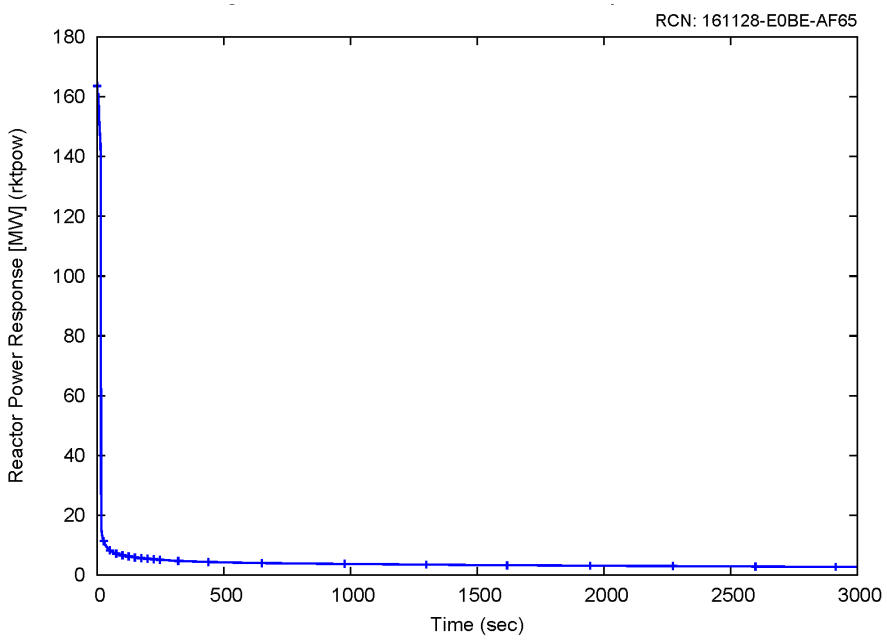
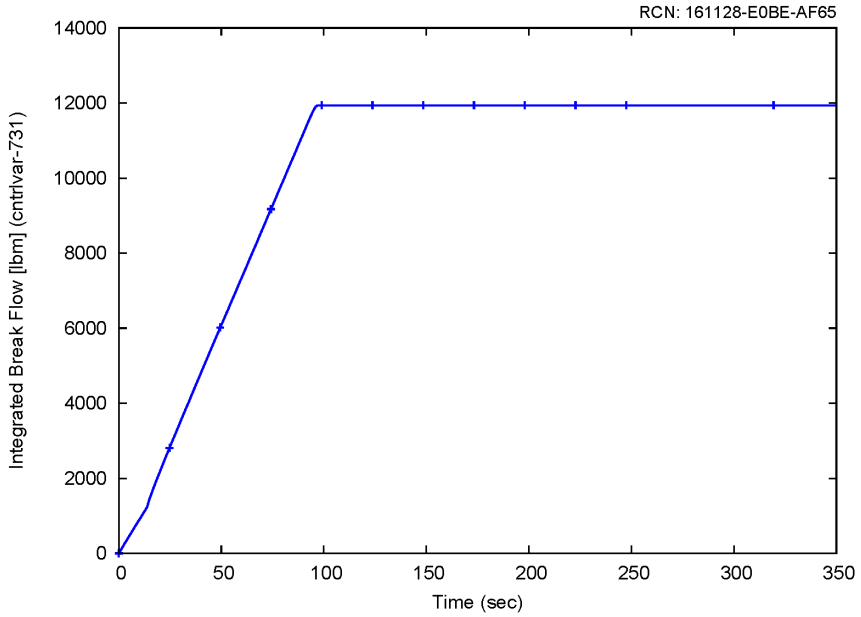


Figure 8-97 Integrated break flow response (0 to 350 sec) for the representative small break outside containment event



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Figure 8-98 Reactor coolant system flow rate response for the representative small break outside containment event

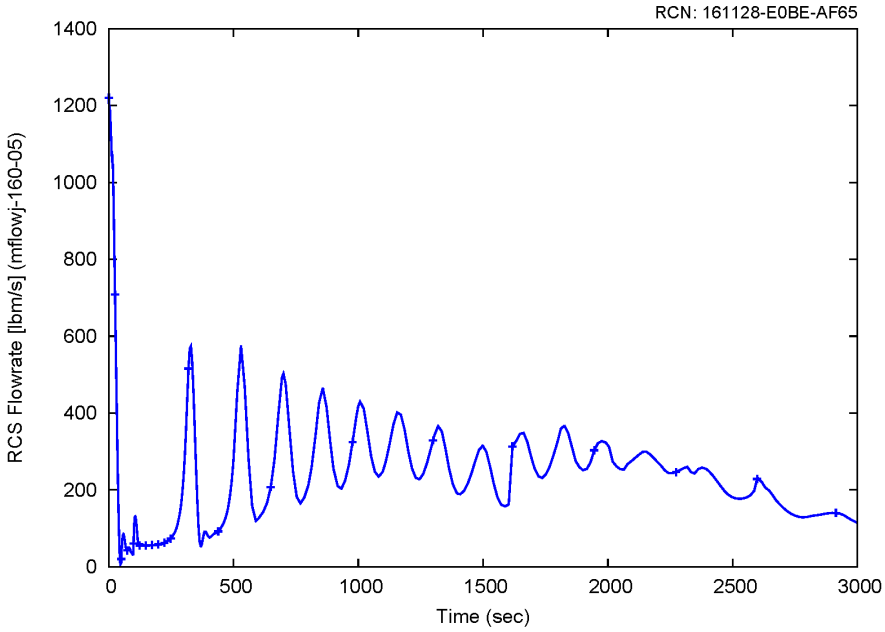
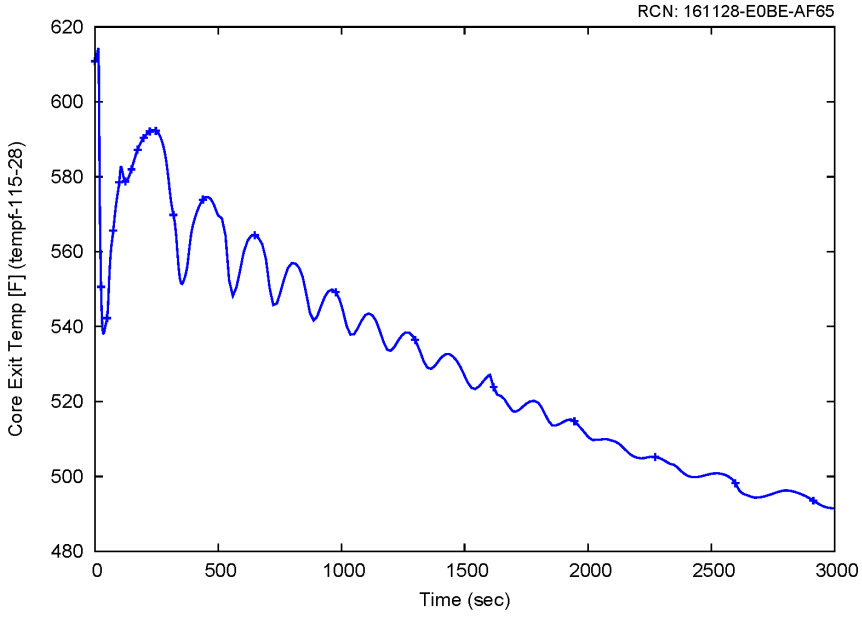


Figure 8-99 Core outlet temperature response for the representative small break outside containment event



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Figure 8-100 Reactor pressure vessel pressure response (0 to 3000 sec) for the representative small break outside containment event

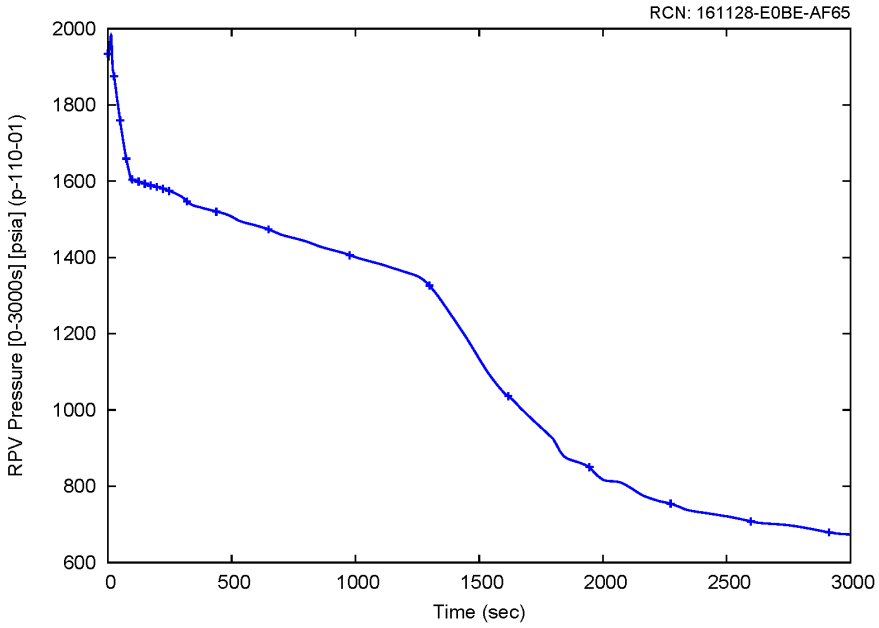
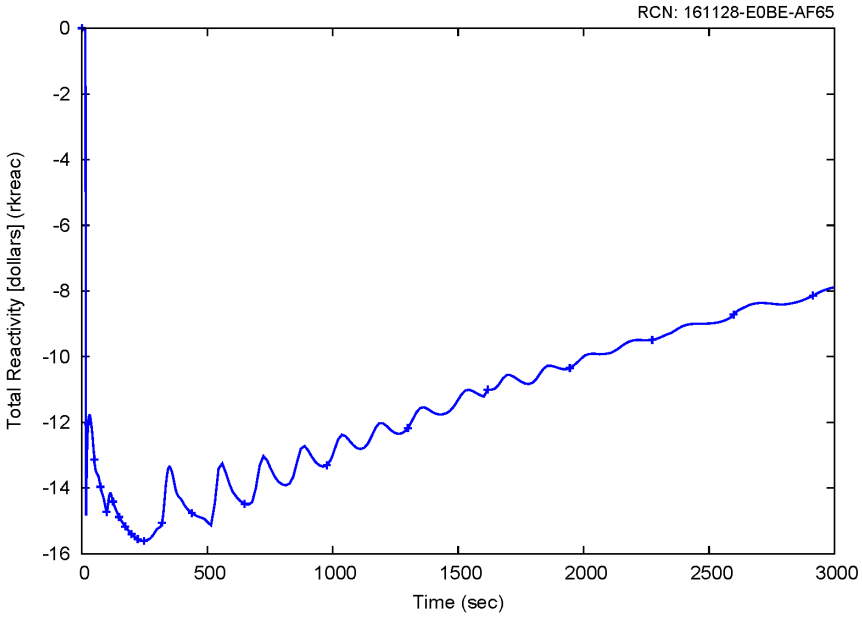
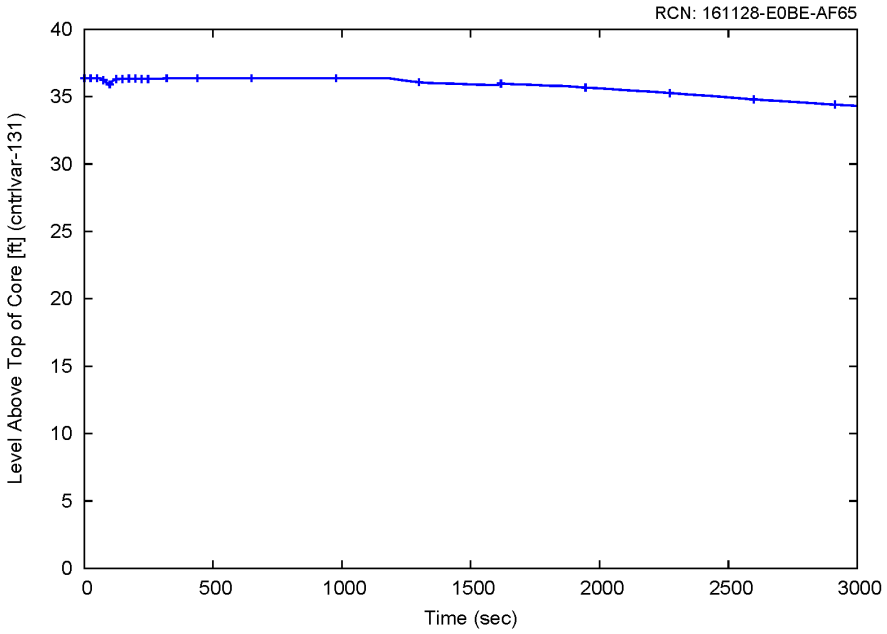


Figure 8-101 Net reactivity response for the representative small break outside containment event



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Figure 8-102 Level above top of core response for the representative small break outside containment event



8.5.1.3 Conclusion

A representative challenging case regarding integrated mass released was identified for a break in a small line carrying reactor coolant. The results of this case, as presented in Section 8.5.1.2, are subsequently used as input to a dose evaluation using the NuScale radiological consequences methodology to demonstrate the respective acceptance criteria are met.

8.5.2 Steam Generator Tube Failure

The purpose of this section is to present the thermal-hydraulic and core neutronic responses of a representative NPM for a steam generator tube failure event. This event is evaluated for offsite and onsite doses, which includes an assessment of the fuel cladding integrity. Thus, the transient analysis results are provided for input to downstream radiological consequences evaluations.

8.5.2.1 Event Description

The event description for a failure of the steam generator tube can be found in Section 7.2.19. Based on Section 7.2.19, the acceptance criteria for radiological consequences are potentially challenged by a steam generator tube failure event. For the representative NPM, the results of the sensitivity studies indicate the event scenario with the highest integrated mass released to the environment may differ from the event scenario with the longest iodine spiking duration. The limiting

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radiological dose consequences are associated with the event scenario having the highest integrated mass released to the environment.

Radiological Consequences - Mass Released

The representative case presented here corresponds to one of the more limiting integrated mass released to the environment cases, i.e., a case with higher dose consequences. This case features the following conditions for the failure of a steam generator tube:

- Conservative initial condition biasing (as shown in Table 7-91) is applied in order to maximize the consequences of this event. This representative case is initialized at 102 percent reactor power.
- Double-ended guillotine break of 100 percent area at top of a single steam generator tube.
- Single active failure of primary MSIV to close on affected SG.
- No loss of normal AC Power.
- No operator action was credited in the representative case.

8.5.2.2 Analysis Results

The following describes the event sequence of the representative steam generator tube failure event. Table 8-12 summarizes the sequence of events. Figure 8-103 through Figure 8-113 show some key parameters during the representative steam generator tube failure event.

The tube failure occurs at time zero. The reactor coolant lost through the break causes an immediate decrease in pressurizer level (Figure 8-103) and RPV pressure (Figure 8-104). In contrast, the SG pressure remains constant (Figure 8-104) as the flow from the tube failure is carried to the turbine. The pressurizer heater power increases in an attempt to compensate for the pressure reduction. The flow through the failed tube causes the pressurizer level to reach the analytical limit for low pressurizer level (35 percent) at 146.0 seconds. The pressurizer heaters are de-energized one second later, while the control rods are free to fall after two seconds (Figure 8-105). The system shrinkage associated with the reactor trip works in conjunction with the break flow to increase the rate of RPV depressurization. The pressurizer pressure continues to decrease until the analytical limit for low pressurizer pressure (1600 psia) is reached at 168.4 seconds. Two seconds later the DHRS valves begin to open; the FWIVs begin to close; and the MSIVs begin to close. The SGs are isolated from the remaining secondary system when the FWIVs and MSIVs close five seconds later. However, the SAF of the primary MSIV on the affected SG to close delays isolation of that SG until the secondary MSIV closes 30 seconds after the DHRS actuation. Isolating the SGs terminates the releases to the environment, but does not terminate flow from the failed tube (Figure 8-106). Following isolation of the SGs, the water level for the affected SG increases and ultimately goes off-scale high, while the level for the intact SG is maintained at the nominal post-trip level

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(Figure 8-107). The maximum integrated break flow to the environment (8477 lbm) is reached at 200.5 seconds (Figure 8-108).

After reactor trip and actuation of DHRS, oscillations are observed due to temperature and density differences between the riser and downcomer (as discussed in Section 7.2); therefore the calculation is continued to verify that the module transitions into passive and stable DHRS cooling. At 30 minutes, RCS flow (Figure 8-109) has stabilized, and the primary system temperature (Figure 8-110) and pressure (Figure 8-111) are steadily decreasing as the DHRS transfers decay heat from the RPV to the reactor pool. The net reactivity (Figure 8-112) becomes negative shortly after reactor trip and remains negative during the transition to stable DHRS cooling. The RPV level (Figure 8-113) remains well above the top of the core for the entire transient. Since stable DHRS cooling has been achieved and minimal flow exists through the failed tube, the transient is terminated at 60 minutes. No operator action was credited to mitigate this event.

Table 8-12 Sequence of events for steam generator tube failure

Event	Time (sec)
Steam generator tube fails in SG1.	0
Peak RPV pressure is reached (1931 psia).	0.5
Peak reactor power is reached (165.4 MW).	3.5
Limiting MCHFR is reached (5.297 as calculated by NRELAP5).	21.0
Low PZR level limit is reached (35%). PZR heaters are disabled after 1 second delay.	146.0
RTS actuation on low PZR level signal, control rods are inserted into the core.	148.0
Low PZR pressure is reached (1600 psia). Containment isolation and DHRS actuation begins after 2 seconds delay.	168.4
DHRS actuation on the low PZR pressure signal. DHRS actuation valves begin to open. FWIVs and MSIVs begin to close. Primary MSIV for faulted SG remains open.	170.4
FWIVs and MSIV (intact SG) are fully closed.	175.4
DHRS actuation valves are fully open.	200.4
Secondary MSIV (faulted SG) is fully closed.	200.5
Maximum integrated break flow to SG (8477 lbm) occurs.	200.5
Peak steam generator pressure is reached (1382 psia).	305.5
Establishment of stable RCS flow. Pressure and temperature are steadily decreasing.	1800
End of calculation. Stable DHRS cooling has been established and pressure difference across failed tube minimized.	6000

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Figure 8-103 Pressurizer level response for the representative steam generator tube failure event

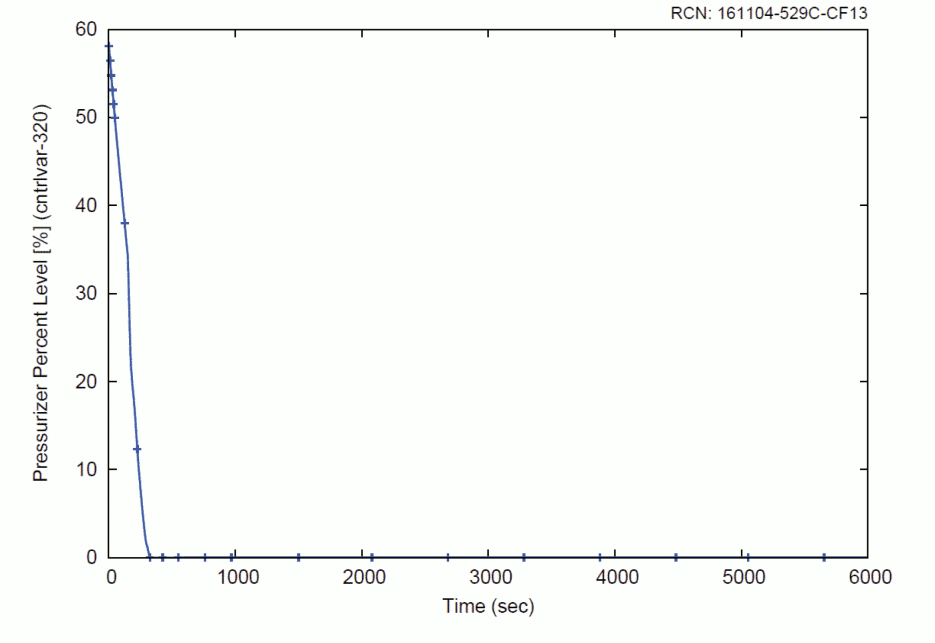
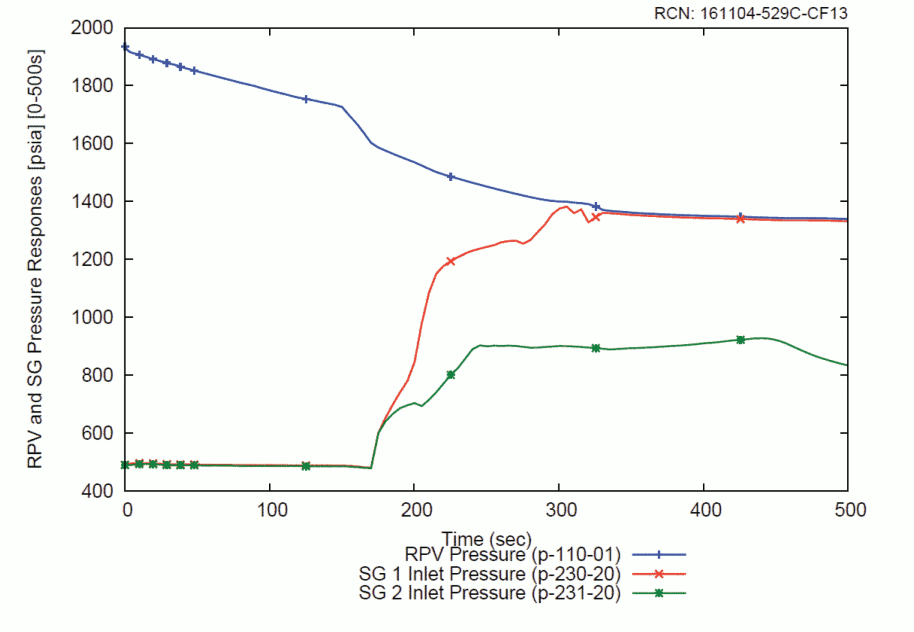


Figure 8-104 Reactor pressure vessel and steam generator pressure responses (0 to 500 sec) for the representative steam generator tube failure event (tube failure occurs in SG1)



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Figure 8-105 Core power response for the representative steam generator tube failure event

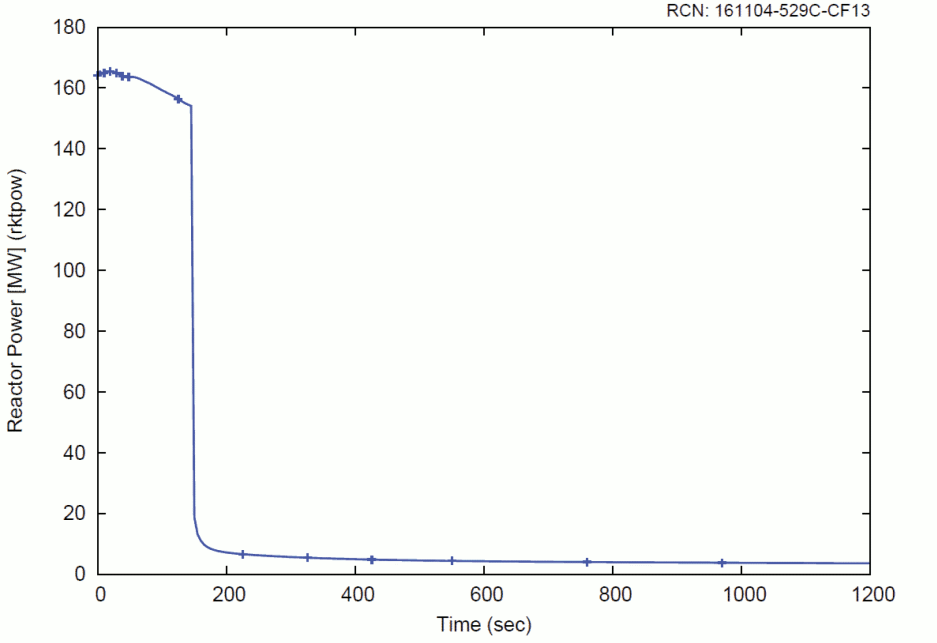
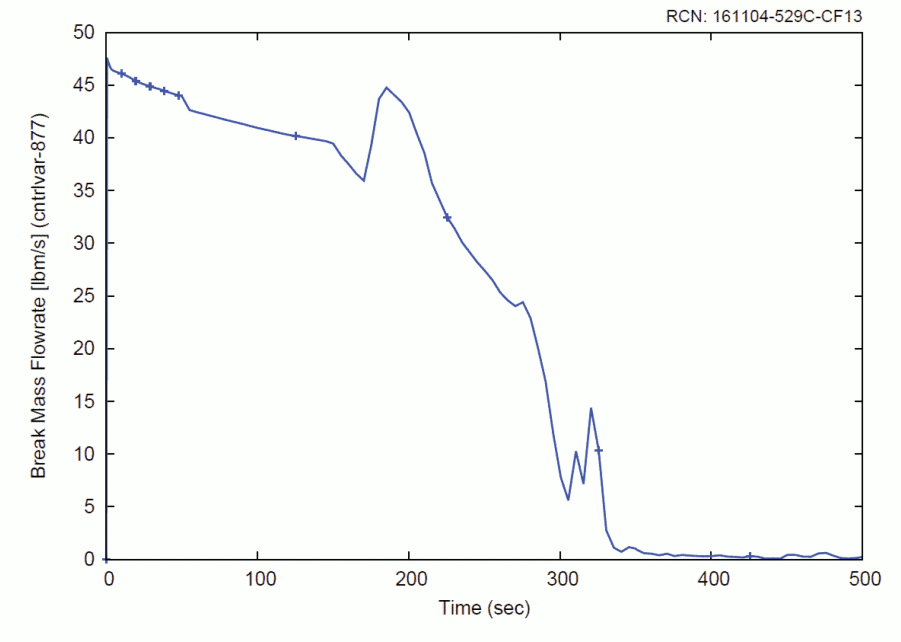


Figure 8-106 Instantaneous break flow response for the representative steam generator tube failure event



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Figure 8-107 Steam generator level response for the representative steam generator tube failure event

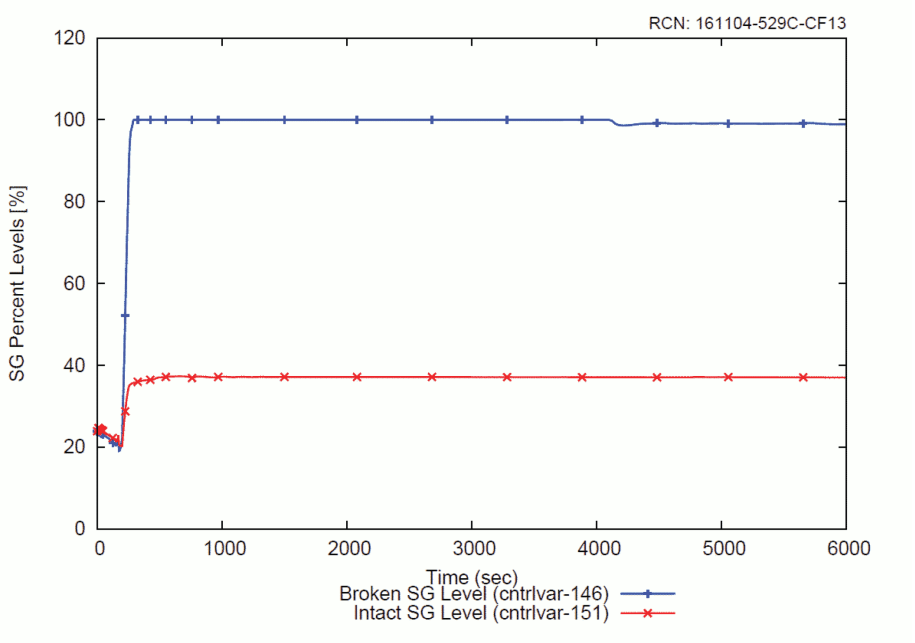
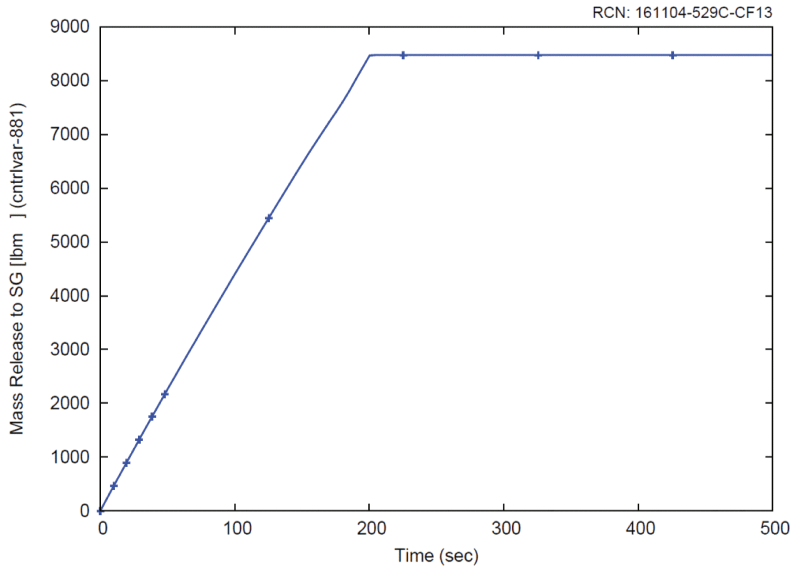


Figure 8-108 Integrated break mass release to steam generator before isolation (0 to 500 sec) for the representative steam generator tube failure event



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Figure 8-109 Reactor coolant system flow rate response for the representative steam generator tube failure event

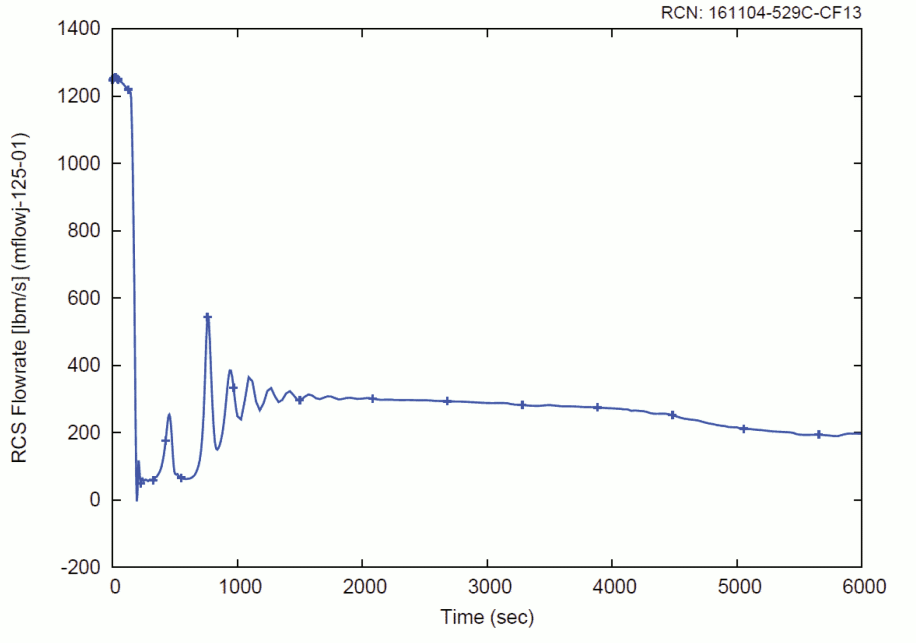
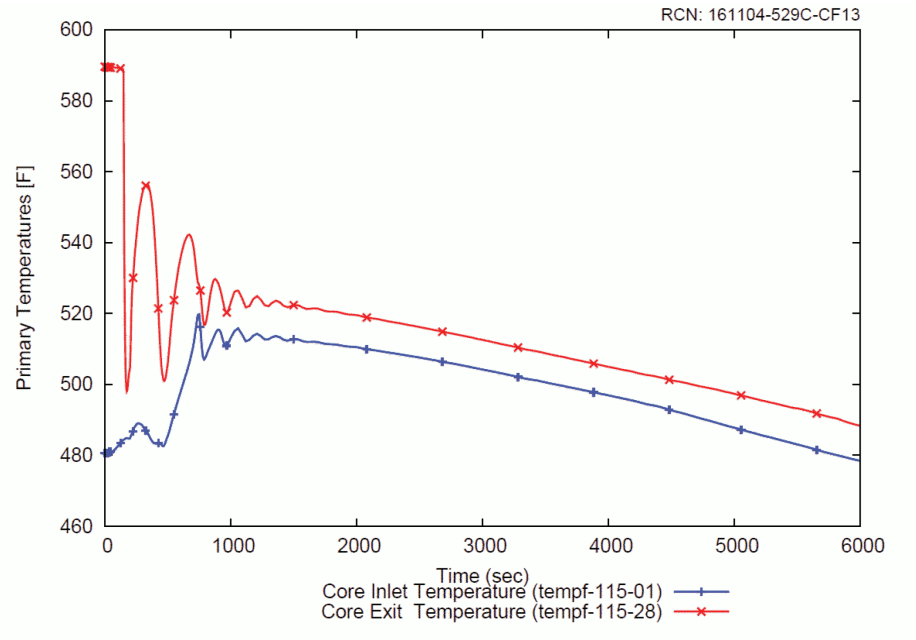


Figure 8-110 Core inlet and exit temperature responses for the representative steam generator tube failure event



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Figure 8-111 Reactor pressure vessel and steam generator responses (0 to 6000 sec) for the representative steam generator tube failure event

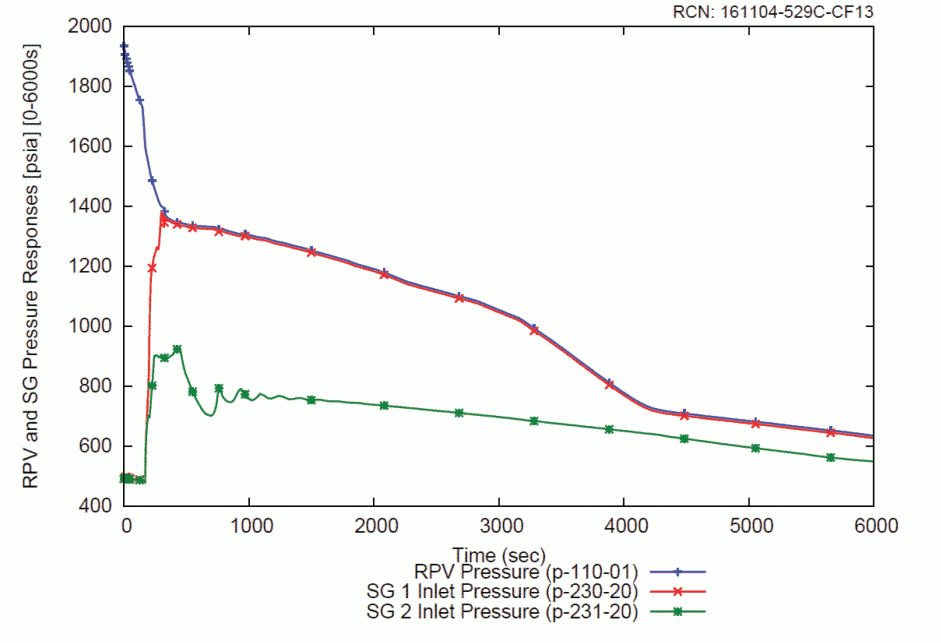
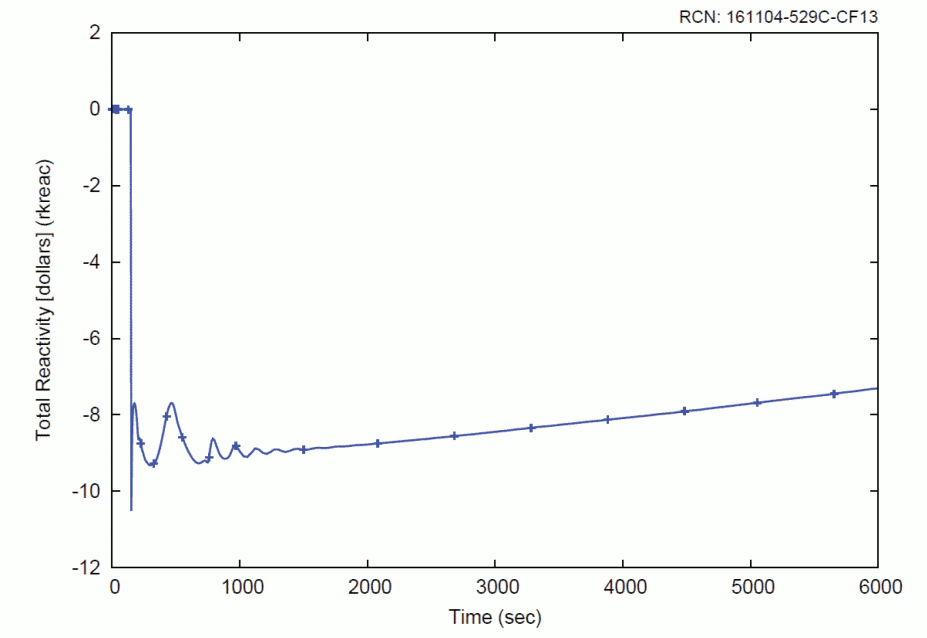
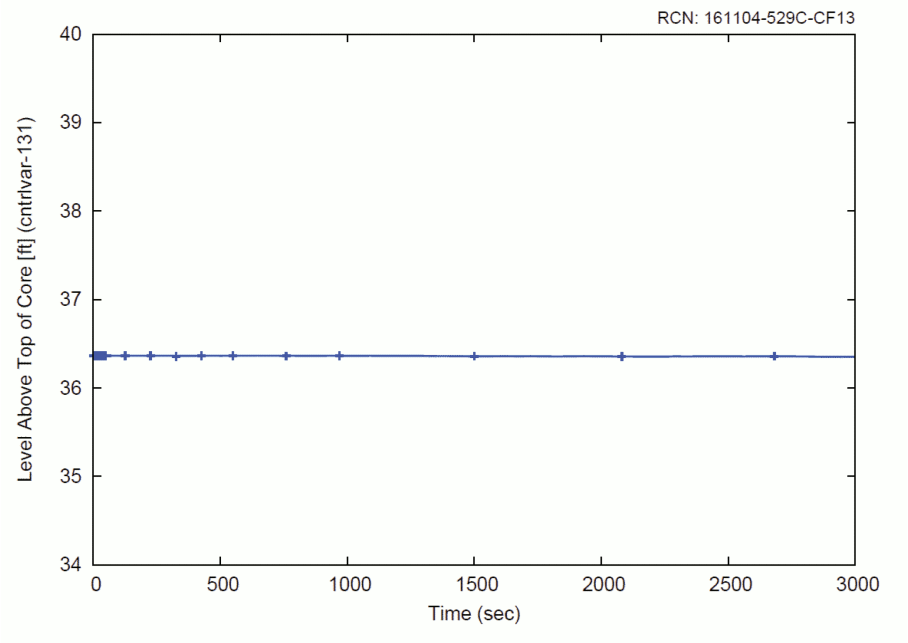


Figure 8-112 Net reactivity response for the representative steam generator tube failure event



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Figure 8-113 Level above top of core response for the representative steam generator tube failure event



8.5.2.3 Conclusion

A representative challenging case regarding integrated mass released was identified for the failure of a steam generator tube. The results of this case, as presented in Section 8.5.2.2, are subsequently used as input to a dose evaluation using the NuScale radiological consequences methodology to demonstrate the acceptance criteria are met.

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9.0 Quality Assurance

The NuScale Power, LLC Quality Assurance Program Description, MN-122626 (Reference 3), complies with the requirements of 10 CFR 50 Appendix B (Reference 19) and Quality Assurance Requirements for Nuclear Facility Applications, ASME NQA-1 2008 and NQA-1a-2009 Addenda (Reference 20).

As described in Reference 2, the NRELAP5 code was developed following the requirements of NuScale's QAP.

The non-LOCA system transient analysis is performed and documented in accordance with NuScale's QAP.

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10.0 Summary and Conclusions

The NuScale evaluation model used to evaluate an NPM system short term transient thermal-hydraulic response to non-LOCA events is presented in this report. The non-LOCA system transient evaluation model is developed following a graded approach in accordance with the guidance provided in RG 1.203.

The NPM plant designs for which this evaluation model is applicable is described in this report. An NPM is a natural circulation pressurized water reactor (PWR) with a reactor core, two helical coil SGs, and a pressurizer integral to the reactor vessel. Many of the events analyzed for operating PWRs and in recent design certification applications are applicable to the NuScale designs. NuScale-specific events reflect unique aspects of the NuScale designs such as the DHRS and normal operation of the containment at vacuum conditions. The NPM designs are evaluated in detail to assure that a sufficiently broad spectrum of transients, accidents, and initiating events have been included in the scope of design basis analyses presented in FSAR Chapter 15. The design-basis events are categorized by type and expected frequency of occurrence so that limiting cases in each group may be quantitatively analyzed and specific acceptance criteria applicable to each postulated initiating event are applied. The NPM design basis events for which the non-LOCA evaluation model is applicable are identified.

NRELAP5 is NuScale's system thermal-hydraulics code used to simulate an NPM system response during both the non-LOCA and LOCA short-term transient event progression. The NRELAP5 code is described in the separate NuScale LOCA evaluation model topical report. Applicability of the NRELAP5 code for non-LOCA system transient analysis is presented in this report. The NRELAP5 code is applicable for calculation of an NPM thermal-hydraulic system response for the non-LOCA short-term transient event progression as part of this EM. This conclusion is based on the high-ranked phenomena identified from the non-LOCA and LOCA PIRT processes, separate effects and integral effects testing, code to code benchmarking, and appropriately conservative input for initial and boundary conditions.

The non-LOCA transient analysis process is described in this report. The methodology for conservatively biasing initial and boundary conditions for event analysis is presented. Then, each initiating event is considered to identify the acceptance criteria that may be challenged during the event. For each non-LOCA event, a description of the event is provided including biases and conservatisms applied, sensitivity studies performed, single active failures and loss of power scenarios that challenge the event acceptance criteria. For each transient event, the acceptance criteria where margin to the limit may be challenged are identified. For these acceptance criteria, sensitivity calculations are performed as appropriate to confirm that suitably conservative inputs are specified and to determine conditions that result in minimum margin. For other acceptance criteria where margin to the limit is not challenged, representative results from the overall scope of sensitivity calculations performed demonstrate that margin to the acceptance criterion is maintained. For non-LOCA initiating events that actuate the decay heat removal system, the EM is applicable for the short-term transient progression; during this time frame the mixture level remains above the top of the riser and primary side natural circulation is maintained.

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For selected non-LOCA events, representative system transient results are provided to demonstrate application of the evaluation model for an NPM. System transient calculations are executed for sufficient duration to demonstrate that the initiating event is mitigated and stable cooling is established. Results of representative calculations show that the maximum primary system and secondary system pressure acceptance criteria are not challenged in an NPM design. The representative results indicate that the primary system pressure is limited by the reactor safety valve and peak values are less than 2200 psia, compared to acceptance criteria for the representative NPM of 2310 psia or 2520 psia, depending on the event classification; in the representative results the maximum secondary side pressure is less than 1600 psia, compared to acceptance criteria for the representative NPM of 2310 psia or 2520 psia, depending on the event classification. Margin to other quantitative acceptance criteria for MCHFR, fuel centerline temperature, and radiological dose limits applicable for the non-LOCA events are demonstrated as part of separate downstream subchannel or accident radiological analyses, presented in separate reports, which are not part of the scope of this topical report.

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11.0 References

1. U.S. Nuclear Regulatory Commission, “Transient and Accident Analysis Methods,” Regulatory Guide 1.203, December 2005.

2. NuScale Power, LLC, “Loss-of-Coolant Accident Evaluation Model,” TR-0516-49422-P-A, Revision 5.

3. NuScale Power, LLC, “NuScale Power, LLC Quality Assurance Program Description,” MN-122626-A, Revision 2.

4. U.S. Code of Federal Regulations, “General Design Criteria for Nuclear Power Plants,” Appendix A, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50 Appendix A).

5. U.S. Code of Federal Regulations, “Contents of applications; technical information,” Section 52.47, Part 52 Chapter I, Title 10, “Energy,” (10 CFR 52.47).

6. NuScale Power, LLC, “Subchannel Analysis Methodology,” TR-0915-17564-P-A, Revision 2.

7. Not Used

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9. U.S. Nuclear Regulatory Commission, “Standard Review Plan, Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment,” NUREG-0800, Chapter 15, Section 15.6.2, Rev. 2, July 1981.

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13. Moghanaki, S. K. and Rahgoshay, M., “Pressurizer Modeling: Using Different Thermodynamic Models and Comparing Results with RELAP Code Results,” Applied Mechanics and Materials (Volumes 423-426), pp. 1444-1448, 2013.

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15. U.S. Nuclear Regulatory Commission, “Standard Review Plan, Transient and Accident Analyses,” NUREG-0800, Chapter 15, Revision 3, March 2007 and various Chapter 15 subsections.

16. U.S. Code of Federal Regulations, “Definitions,” Section 50.2, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50.2).

17. American National Standards Institute / American Nuclear Society, “Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems,” ANSI/ANS-58.9-2002 R2015.

18. U.S. Nuclear Regulatory Commission, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” SECY-94-084, March 1994.

19. U.S. Code of Federal Regulations, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Appendix B, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50 Appendix B).

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21. NuScale Power, LLC, “Rod Ejection Accident Methodology,” TR-0716-50350-P-A, Revision 3.

22. NuScale Power, LLC, “Applicability of AREVA Fuel Methodology for the NuScale Design,” TR-0116-20825-P-A, Revision 1.

23. NuScale Power, LLC, “Nuclear Analysis Codes and Methods Qualification,” TR-0616-48793-P-A, Revision 1.

24. Kim, S.J., “Turbulent film condensation of high pressure steam in a vertical tube of passive secondary consensation system,” PhD thesis, Korea Advanced Institute of Science and Technology, 2000.

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26. NuScale Power, LLC, “Extended Passive Cooling and Reactivity Control Methodology,” TR-124587-P-A, Revision 1.

27. U.S. Code of Federal Regulations, “Contents of applications; technical information,” Section 52.137, Part 52 Chapter I, Title 10, “Energy,” (10 CFR 52.137).

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30. Not Used

31. SwUM-0304-17023, Revision 13, NRELAP5 Version 1.7 Theory Manual.

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Section C

RAI Number	Question Number	NuScale Letter Number
10357	15.1.1-8, NonLOCA.LTR-53	RAIO-176409 dated 12/13/2024 (ML24348A130)
10297	NonLOCA.LTR-1	RAIO-177428 dated 12/19/2024 (ML24354A252)
10297	NonLOCA.LTR-60	RAIO-177719 dated 12/19/2024 (ML24354A131)
10297	NonLOCA.LTR-31, 32, 46, 56, 65	RAIO-177581 dated 01/09/2025 (ML25009A002)
10297	NonLOCA.LTR-3, 18, 27	RAIO-178267 dated 01/14/2025 (ML25014A156)
10297	NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69	RAIO-179388 dated 02/11/2025 (ML25042A063)
10297	NonLOCA.LTR-55	RAIO-179400 dated 02/14/2025 (ML25045A238)
10297	NonLOCA.LTR-50	RAIO-179797 dated 02/27/2025 (ML25058A423)

December 13, 2024 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 037 (RAI-10357 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 037 (RAI-10357 R1)," dated October 17, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10357 R1:

- 15.1.1-8, NonLOCA.LTR-53

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 037 (RAI-10357 R1, Question 15.1.1-8, NonLOCA.LTR-53). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 13, 2024.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Stacy Joseph, Senior Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10357 R1, Question 15.1.1-8, NonLOCA.LTR-53, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10357 R1, Question 15.1.1-8, NonLOCA.LTR-53, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-176410

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10357 R1, Question 15.1.1-8, NonLOCA.LTR-53, Nonproprietary Version

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10357
Date of RAI Issue: 10/17/2024

NRC Question No.: 15.1.1-8, NonLOCA.LTR-53

Issue

Table 7-7 of the Non-LOCA LTR states that an applicant implementing the methodology will perform sensitivity studies on initial feedwater (FW) temperature and initial steam generator (SG) pressure in order to identify the limiting MCHFR for the decrease in feedwater temperature event. The applicant provided for NRC staff audit (ML23067A300) the sensitivity studies to support the standard design approval application for the NPM-20. However, the staff noted that the initial feedwater temperature and initial steam generator pressure were varied simultaneously in these studies. The staff also noted that these studies varied bypass of the high power rate trip and operation of the CRA regulating bank. Because these four parameters are varied simultaneously, the impact of any individual change on MCHFR cannot be determined. As such, the NRC staff cannot conclude that initial SG pressure and FW temperature were varied to identify the limiting MCHFR consistent with the Non-LOCA LTR evaluation model.

During the staff's audit, the applicant stated that examining biased-high initial feedwater temperature {{ }}$^{2(a),(c)}$ that is evaluated in the decrease in feedwater temperature event. The staff considers evaluation of cases with biased-high initial feedwater temperature to be a necessary step to ensure that the limiting MCHFR is identified, but it is not clear whether {{ }}$^{2(a),(c)}$ will minimize margin to acceptance criteria.

During the audit discussion, the applicant also stated that maintaining consistent steam pressure allows comparison of the cooldown effect driven by the initiating event (i.e., decreasing feedwater temperature). However, NRC staff does not consider that studying the cooldown effect with consistent initial SG pressure precludes variation of initial SG pressure in a separate set of cases in order to assure that the limiting MCHFR is identified. The staff also noted additional calculations that deviate from the methodologies defined in the Non-LOCA LTR.

The audited calculation that supports the single CRA withdrawal evaluation, uses a {{ }}[2(a),(c)] . However, Table 7-68 of the Non-LOCA LTR specifies that a biased-low initial fuel temperature should be used. The audited calculation does not specify that assuming a {{ }}[2(a),(c)] deviates from the LTR methodology nor does it provide an acceptable justification for deviation from the proposed NPM-20 licensing basis. The audited calculation also states that using a biased-low initial fuel temperature is more conservative.

Information Requested

1. Provide results of sensitivity studies that vary initial feedwater temperature independently of other parameters to demonstrate that the limiting MCHFR is identified for the decrease in feedwater temperature event.
2. Provide results of sensitivity studies that vary initial steam generator pressure independently of other parameters to demonstrate that the limiting MCHFR is identified for the decrease in feedwater temperature event.
3. Provide revised calculations for the single CRA withdrawal event with initial fuel temperature assumptions that bias initial fuel temperature consistent with the Non-LOCA LTR methodology, or provide appropriate justification for use of the {{ }}[2(a),(c)] initial fuel temperature and modify the FSAR to describe the deviation from the LTR methodology.

NuScale Response:

Parts 1 and 2

EC-104467, Revision 0, “NPM-20 Decrease in Feedwater Temperature Transient Analysis” (previously provided in the Chapter 15 electronic reading room (eRR) for NRC audit), evaluates biased initial feedwater (FW) temperature and initial steam generator (SG) pressure conditions to determine minimum critical heat flux ratio (MCHFR) conditions in accordance with Table 7-7 of TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology.” The stated NRC question is that initial FW temperature and initial SG pressure in EC-104467, Revision 0, are not varied individually, but varied in conjunction with other parameters that could make it difficult to determine their individual effect on MCHFR.

EC-104467, Revision 1, “NPM-20 Decrease in Feedwater Temperature Transient Analysis,” has been issued to address the NRC question and is provided in the eRR with this response. The following sections of EC-104467, Revision 1, are relevant to address the NRC question:

{{

}}2(a),(c)

In conclusion, EC-104467, Revision 1, contains the requisite sensitivity studies for initial FW temperature and initial SG pressure in accordance with Table 7-7 of TR-0516-49416-P, Revision 4. The parameters are varied both individually and along with combinations of other parameters and the results are used to identify the limiting case for MCHFR.

Part 3

Table 7-68 of TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” identifies the initial conditions, biases, and conservatisms associated with the single control rod assembly withdrawal event. For the initial fuel temperature, Table 7-68 indicates that a low bias is used.

EC-0000-8579, Revision 1, “Control Rod Misoperation Transient Analysis,” (previously provided in the eRR for NRC audit) implements the parameter assumptions from Table 7-68 of TR-0516-49416-P, Revision 4. {{

}}2(a),(c)

NuScale has reviewed the inputs and outputs associated with EC-0000-8579 and confirms that the initial fuel temperature is consistent with bias in the low direction. No calculation revisions are necessary due to this question as the calculation correctly implements the bias identified by Table 7-68 of TR-0516-49416-P.

Impact on US460 SDAA:

There are no impacts to the US460 SDAA as a direct result of this response. However, the calculation revision described in this response impacts FSAR Section 15.1.1. The revisions to FSAR Section 15.1.1 to incorporate the calculation results will be included in FSAR Revision 2. One impact of note is that the revised calculation modified the treatment of reactivity feedback. In the prior revision, the most limiting time in cycle (i.e., end of cycle) was assumed, resulting in the use of the most negative Doppler temperature coefficient (DTC) and most negative moderator temperature coefficient (MTC). In the current revision, the most positive DTC and most negative MTC are used. This combination of DTC and MTC, although not representative of an actual time in cycle, is conservative for the power response. FSAR Section 15.1.1.3.2 is revised as follows:

- The most positive Doppler temperature coefficient (DTC) and most negative moderator temperature coefficient (MTC) are used to provide a limiting power response. ~~The most limiting end of cycle (EOC) reactivity feedback is used to provide a limiting power response.~~

Impact on Topical Report:

There are no impacts to Topical Report TR-0516-49416, Non-Loss-of-Coolant Accident Analysis Methodology, as a result of this response.

Enclosure 3:

Affidavit of Mark W. Shaver, AF-176410

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10357 R1, Question 15.1.1-8, NonLOCA.LTR-53) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10357 R1, Question 15.1.1-8, NonLOCA.LTR-53. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 13, 2024.

Mark W. Shaver

December 19, 2024 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 032 (RAI-10297 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 032 (RAI-10297 R1)," dated October 31, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10297 R1:

- NonLOCA.LTR-1

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 032 (RAI-10297 R1, Question NonLOCA.LTR-1). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirement of 10 CFR § 810. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 19, 2024.

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-1, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-1, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-177435

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10297 R1,
Question NonLOCA.LTR-1, Nonproprietary Version

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10297
Date of RAI Issue: 10/31/2024

NRC Question No.: NonLOCA.LTR-1

Issue
The Non-LOCA methodology does not provide an adequate basis for model validation vs. test data {{

}}$^{2(a),(c)}$. Riser holes have an impact on thermal hydraulic conditions, for example but not limited to, initial conditions, natural circulation and RCS response; these impacts and any others need to be addressed.

Information Requested
a) Provide justification via sensitivity studies demonstrating that NRELAP5 can adequately calculate the actual expected NPM response with respect to the integrated topical report model. The total integral response of the Non-LOCA LTR model (from beginning of the event until the end of the period of interest) to various design basis events and conditions should be validated, {{

}}$^{2(a),(c)}$. The potential impact on the RCS response and conditions for the integrated model on the figures of merit calculated by the topical report model and downstream activities should be shown analytically as described below:

i. Provide sensitivity analyses and evaluations of the impact of riser holes on the integral effects tests responses used for Evaluation Model validation for the Non- LOCA LTR, including the referenced integral effects test responses. The various impacts of riser holes on the integral test validation response should be captured in the impacts on the integrated NRELAP5 evaluation model results for Non- LOCA analyses.

ii. Provide sensitivities and evaluations for the impact of riser holes on the total integral response of the LTR model (Non-LOCA) to various design basis events and conditions that

show that the test validation response to riser holes is captured. The impact on the RCS response and conditions for the integrated model on the figures of merit calculated by the Non-LOCA LTR model should be shown analytically.

b) Revise the LTR to include the above information.

NuScale Response:

Executive Summary

NuScale previously provided information in the response to audit question A-NonLOCA.LTR-1 that demonstrated that {{ }}$^{2(a),(c)}$ riser holes on a NuScale Power Module (NPM) during both steady-state conditions and the short-term transient response to non-loss-of-coolant accident (non-LOCA) events, with focus on post-trip decay heat removal system (DHRS) cooling conditions. NuScale concluded that no further studies of riser holes were warranted {{ }}$^{2(a),(c)}$ on non-LOCA events. However, in response to this request for additional information (RAI) specifically requesting additional studies, NuScale provides the results of multiple sensitivity studies focused on the impact of riser holes on the short-term transient response. The sensitivity studies include {{ }}$^{2(a),(c)}$ In the sensitivity studies, the riser hole areas are modified {{ }}$^{2(a),(c),ECI}$ The results show that modifying the riser hole areas in NRELAP5 results in {{ }}$^{2(a),(c)}$ The impacts of the modifications on figures of merit, {{ }}$^{2(a),(c)}$ Therefore, it is concluded that modeling nominal riser hole areas is reasonable {{ }}$^{2(a),(c)}$ The conclusion extends to modeling nominal form losses for the flow paths, because the form losses and flow area together characterize the flow path pressure drop. The provided sensitivities are consistent with those requested in this RAI. In addition, TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” is revised as requested in this RAI.

Relevant Information from Previous Audit Question Response

NuScale previously provided information regarding riser holes in the response to audit question A-NonLOCA.LTR-1, available in the electronic reading room (eRR) for TR-0516-49416-P. The

response to the audit question concluded {{

}}2(a),(c) Relevant information from the previous audit question response is listed here to provide background and for comparison to new sensitivity results described later in this response. {{

}}2(a),(c),ECI

The previous audit response concluded that NRELAP5 reasonably calculates the small impact of riser holes on an NPM during both steady-state conditions and the short-term transient response to LOCA and non-LOCA events.

Additional Riser Hole Geometry Considerations

As described in the previous response to audit question A-NonLOCA.LTR-1 and summarized above, {{

}}$^{2(a),(c),ECI}$ the largest allowable increase in lower riser hole flow area is a factor of {{ }}$^{2(a),(c),ECI}$ relative to nominal. Similarly, the largest allowable decrease in lower riser hole flow area is a factor of {{ }}$^{2(a),(c),ECI}$ relative to nominal.

As described in the previous response to audit question A-NonLOCA.LTR-1 and summarized above, {{

}}$^{2(a),(c),ECI}$ the largest allowable increase in upper riser hole flow area is a factor of {{ }}$^{2(a),(c),ECI}$ relative to nominal. Similarly, the largest allowable decrease in lower riser hole flow area is a factor of {{ }}$^{2(a),(c),ECI}$ relative to nominal.

{{

}}$^{2(a),(c),ECI}$ the above estimates of possible increases or decreases in riser hole area are considered bounding.

New Sensitivity Studies

Based on the prior discussion, NuScale concluded that no further studies of riser holes were warranted {{ }}$^{2(a),(c)}$ However, to support the response to this RAI question specifically requesting further studies, additional sensitivity studies are performed. Sensitivity studies for the NPM response are performed for the following Final Safety Analysis Report (FSAR) Chapter 15 events: {{

}}$^{2(a),(c)}$ In addition, sensitivity studies are performed

for {{ }}2(a),(c) test assessments. The sensitivity studies, and their results, are provided in the following sections.

Riser Hole Sensitivity Study for {{ }}2(a),(c)

A base case {{ }}2(a),(c) is selected to perform a riser hole sensitivity study. {{

}}2(a),(c) Based on the riser hole geometry considerations described above, these sensitivity cases are {{ }}2(a),(c),ECI

The results of the sensitivity study are shown in Table 1. {{

}}2(a),(c)

Table 1: {{ }}[2(a),(c)] Riser Hole Sensitivity Study

{{

}}[2(a),(c)]

Transient plots of relevant parameters for the sensitivity study are shown in Figure 1 through Figure 14. {{

}}[2(a),(c)]

{{

}}2(a),(c)

Overall, the results of the riser hole sensitivity study {{
}}[2(a),(c)] demonstrate the following: {{

}}[2(a),(c)]

Figure 1: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 2: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 3: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 4: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 4a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 4b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 4c: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 5: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 6: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 7: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 8: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 9: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 10: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 11: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 12: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 13: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 14: {{

}}2(a),(c)

{{

}}2(a),(c)

Riser Hole Sensitivity Study for {{
}}2(a),(c)

{{

}}2(a),(c)

A base group of cases {{
}}2(a),(c) is selected to perform a riser hole sensitivity study. {{

}}2(a),(c) Based on the riser hole geometry considerations described above, these sensitivity cases are {{ }}2(a),(c),ECI

The results of the sensitivity study are shown in Table 2. {{

}}2(a),(c)

Table 2: {{ }}[2(a),(c)] Riser Hole Sensitivity Study

{{

}}[2(a),(c)]

{{

}}2(a),(c)

{{

}}2(a),(c) Transient plots of relevant parameters for the sensitivity study are shown in Figure 15 through Figure 23. The figures focus on the short-term transient response {{

}}2(a),(c)

{{

}}2(a),(c) {{

}}2(a),(c)

Overall, the results of the riser hole sensitivity study {{

}}2(a),(c) demonstrate the following: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 15: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 16: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 17: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 18: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 19: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 20: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 21: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 22: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 23: {{

}}2(a),(c)

{{

}}2(a),(c)

Riser Hole Sensitivity Study for Loss-of-Coolant Accident Event

Analyses of LOCA events are outside of the scope of TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology” (i.e., the non-LOCA topical report); the analyses are addressed in TR-0516-49422-P, Revision 3, “Loss-of-Coolant Accident Evaluation Model” (i.e., the LOCA topical report). However, similar audit questions regarding riser holes were asked of both topical reports (audit questions A-NonLOCA.LTR-1 and A-LOCA.LTR-2). For this reason, additional sensitivity studies for riser holes are performed for {{ }}2(a),(c) LOCA events and provided in this response.

The LOCA analyses consider multiple break locations and a spectrum of break sizes. As a result, a group of cases with differing locations or differing sizes is run for a consistent set of other initial conditions and bias parameters. Base groups of LOCA cases are selected to perform a riser hole sensitivity study. {{

}}2(a),(c)

Based on the riser hole geometry considerations described above, these sensitivity cases are {{ }}2(a),(c),ECI

The results of the sensitivity study are shown in Table 3 {{

}}2(a),(c

{{

}}[2(a),(c)]

Table 4 shows the results of the same sensitivity study {{

}}[2(a),(c)] The results in Table 4 are similar to those in Table 3. Similar conclusions can be drawn. {{

}}[2(a),(c)]

Table 5 shows the results of the same sensitivity study {{

}}[2(a),(c)] The results in Table 5 are similar to those in Table 3 and Table 4. {{

}}[2(a),(c)]

An additional set of sensitivity study results are provided in Table 6 {{

}}[2(a),(c)] The Table 6 results are also similar to those in Table 3 through Table 5.

Table 3: {{ }}[2(a),(c)] Loss-of-Coolant Accident Riser Hole Sensitivity Study

{{

}}[2(a),(c)]

Table 4: {{ }}[2(a),(c)] Loss-of-Coolant Accident Riser Hole Sensitivity Study

{{

}}[2(a),(c)]

Table 5: {{ }}[2(a),(c)] Loss-of-Coolant Accident Riser Hole Sensitivity Study

{{

}}[2(a),(c)]

Table 6: {{ }}[2(a),(c)] Loss-of-Coolant Accident {{ }}[2(a),(c)] Riser Hole Sensitivity Study

{{

}}[2(a),(c)]

Transient plots of relevant parameters for the sensitivity study are shown in Figure 24 through Figure 32 {{

}}[2(a),(c)]

{{

}}2(a),(c)

Similar transient plots are provided in Figure 33 through Figure 35 {{

}}2(a),(c)

Overall, the results of the riser hole sensitivity study for the LOCA {{
}}2(a),(c) demonstrate the following: {{

}}2(a),(c)

Figure 24: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 25: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 26: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 27: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 28: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 29: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 30: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 31: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 32: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 33: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 34: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 35: {{

}}2(a),(c)

{{

}}2(a),(c)

Riser Hole Sensitivity Studies for Additional Chapter 15 Events

In addition to the riser hole sensitivity studies for the Chapter 15 events described above, {{

}}2(a),(c)

Riser Hole Sensitivity Study for Test Assessments

As described previously, {{

}}2(a),(c) The following {{ }}2(a),(c) test assessments are selected for sensitivity studies {{

}}2(a),(c)

{{

}}2(a),(c)

For each of the above test assessments, {{

}}2(a),(c)

Figure 36 to Figure 40 provide the results of the sensitivity {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 41 to Figure 45 provide the results of the sensitivity {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 46 to Figure 50 provide the results of the sensitivity {{

}}2(a),(c)

Figure 51 to Figure 53 provide the results of the sensitivity {{

}}2(a),(c)

{{

}}2(a),(c)

Overall, the results of the sensitivity studies for test assessments demonstrate the following: {{

}}2(a),(c)

Figure 36: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 37: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 38: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 39: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 40: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 41a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 41b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 42a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 42b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 43a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 43b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 44a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 44b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 45a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 45b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 46a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 46b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 47a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 47b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 48a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 48b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 49a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 49b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 50a: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 50b: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 51: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 52: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 53: {{

}}2(a),(c)

{{

}}2(a),(c)

Conclusion

The information provided in the previous response to audit question A-NonLOCA.LTR-1 and this response demonstrates that NRELAP5 reasonably calculates the small impact of riser holes on an NPM during both steady-state conditions and the short-term transient response to non-LOCA events. TR-0516-49416-P is revised to identify that sensitivity studies were performed for test assessments and NPM design-basis events and that the results show that riser holes do not have a significant impact on non-LOCA transient progression and figures of merit. The conclusions from the sensitivity studies in this response are consistent with the prior NuScale conclusion that additional studies were not warranted.

Impact on Topical Report:

Topical Report TR-0516-49416, Non-Loss-of-Coolant Accident Analysis Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

- The non-LOCA evaluation model is applicable to a nuclear power plant that follows the general description of the NuScale plant designs in Section 3.0. The applicability of the EM is based on the non-LOCA phenomena identification and ranking table and assessment of the high-ranked phenomena that are treated as part of the system transient analysis.
- The non-LOCA evaluation model does not address the evaluation of specified acceptable fuel design limits (SAFDLs), which are evaluated in a downstream subchannel analysis. The subchannel analysis codes and methods are covered in separate methodologies and assessments (Reference 6, supplemented by Reference 28). However, the interface of the non-LOCA system transient analysis with the downstream subchannel analysis is part of the non-LOCA evaluation model.
- The non-LOCA evaluation model does not address the evaluation of the accident radiological source term and dose. The accident radiological source term and dose analyses are covered in separate methodologies and assessments (Reference 8). However, the interface of the non-LOCA system transient analysis with downstream radiological analysis is part of the non-LOCA evaluation model.
- The EM is applicable for the short-term non-LOCA transient progression; the non-LOCA transient analysis short-term duration and analysis process are discussed further in Section 4.2 and Section 4.3. During this time frame the mixture level remains above the top of the riser and primary side natural circulation is maintained. The reactivity control and extended passive cooling analysis methodology in the long-term, including events that transition from DHRS cooling to ECCS cooling, is addressed in separate methodologies and assessments (Reference 26).

RAI 10297 Question NonLOCA.LTR-1

- ~~The plant design overview description in Section 3.0, plant model description in Section 6.0, input and biasing discussion in Section 7.0, and the example calculations in Section 8.0 do not include the design feature of riser holes (upper or lower).~~ Riser holes are included in the NPM designs to mitigate potential boron dilution impacts of long term DHRS cooling and riser uncovery. ~~The riser holes do not impact the short-term DHRS cooldown during the non-LOCA phase.~~ The PIRT and test assessments in Section 5.0 did not incorporate the riser holes; however, the PIRT and assessments documented in Section 5.0 remain valid, since the design feature of the riser holes results in negligible differences in the prediction of RCS parameters for the short-term non-LOCA transient progression as confirmed by sensitivity studies discussed in Section 5.0 and Section 7.0. The effect of the riser holes during extended passive cooling is addressed in separate methodologies and assessments (Reference 26).
- Control rod ejection accident analysis is addressed by a separate methodology (Reference 21) and is not part of the non-LOCA evaluation model.
- Loss of coolant accident analysis, including analysis of an inadvertent opening of one or more valves on the RPV, is addressed by a separate methodology (Reference 2) and is not part of the non-LOCA evaluation model.
- Analysis of the peak containment pressure and temperature response is addressed by a separate methodology (Reference 2) and is not part of the non-LOCA evaluation model.

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of a spring loaded arming valve in the vent port path from the main disc chamber to the vent line. If the differential pressure across this arming valve is greater than a threshold value, the arming valve closes, which prevents the main disc chamber from discharging through the vent line, blocking the RRV from opening. The RRV does not open until the arming valve differential pressure decreases below the release pressure. The IAB is also used in the RVVs for some NPM designs.

Provided the IAB device setpoint is reached, if applicable, the RVV and RRV components fail to the open (safe) position upon the loss of power, thus enabling reliable long-term cooling without operator actions, alternating current (AC) or direct current (DC) power, or make-up water. Successful operation of the ECCS requires isolation of the containment, such that the coolant inventory of the RCS is preserved.

3.5 Other Important Systems and Functions

Other systems and functions that are important in mitigating plant response during a postulated non-LOCA event are discussed below.

Reactor Coolant System

The reactor coolant system (RCS) consists of the RPV, reactor core, riser, upper plenum, SGs (shell side), downcomer, lower plenum, and pressurizer (PZR). The arrangement of the RCS and the relative locations of the thermal centers in the core and the SGs promote buoyancy driven natural circulation flow.

The RPV consists of a steel cylinder with an inside diameter of approximately 10 ft and an overall height of approximately 60 ft and is designed for a normal operating pressure of approximately 1850-2000 psia. Nozzles on the upper head provide connections for reactor safety valves (RSVs) and RVVs.

The core configuration for an NPM consists of 37 fuel assemblies and 16 control rod assemblies (CRAs). The fuel assembly design is modeled from a standard 17x17 PWR fuel assembly with 24 guide tube locations for control rod fingers and a central instrument tube. The assembly is nominally half the height of standard plant fuel and is supported by five spacer grids. The U-235 enrichment is below the current U.S. manufacturer limit of 4.95 weight percent.

RAI 10297 Question NonLOCA.LTR-1

Each NPM uses two once-through helical coil SGs for steam production. The SGs, which produce superheated steam, are located in the annular space between the RCS hot leg riser and the reactor vessel inside diameter wall. Each SG is designed to remove 50 percent of the rated core thermal power. Flow paths in the upper riser are provided to mitigate potential boron dilution impacts during long-term DHRS cooling and riser uncovery. An NPM design may include flow paths in the lower riser to mitigate potential boron dilution impacts near the core inlet during long-term ECCS cooling.

The PZR provides the primary means for controlling RCS pressure. PZR heaters and spray maintain a constant reactor coolant pressure during operation. A steel PZR baffle

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RAI 10297 Question NonLOCA.LTR-1

5.3.7.13 Impact of NPM Riser Holes on NIST Test Assessments

The NPM design may include holes in the riser that allow cross-flow between the riser and downcomer. {{

}}2(a),(c) The overall performance of the NRELAP5 code and model, in terms of its level of code-to-data agreement, did not change from the previous assessment.

The NIST-2 assessments described in Section 5.3.7 {{

}}2(a),(c)

The above results provide confidence in the ability of NRELAP5 to simulate natural circulation flow and temperature distribution in the presence of cross-flow holes in the riser while also demonstrating that such riser holes do not significantly impact the NIST test assessments.

5.4 Conclusions of NRELAP5 Applicability for Non-LOCA

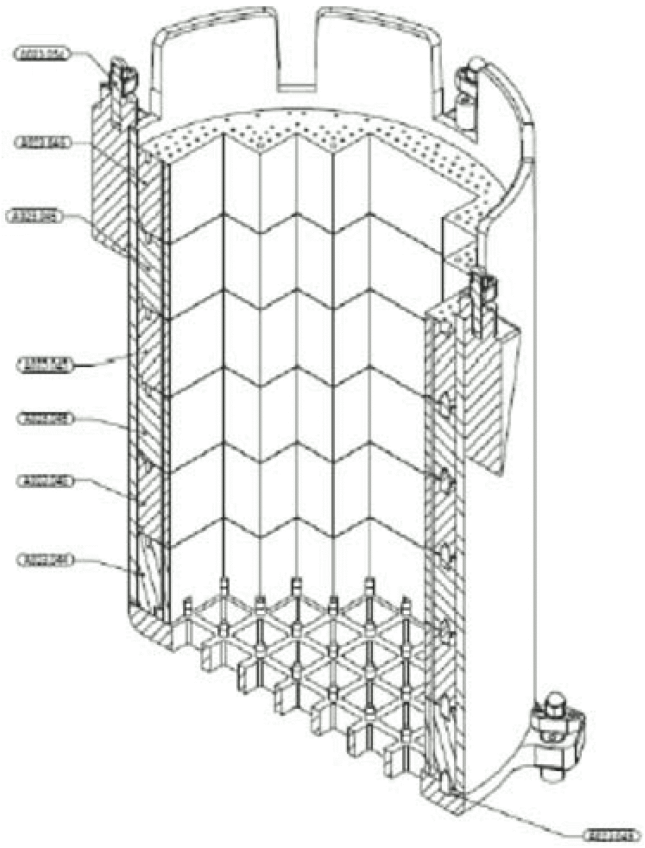
The high-ranked phenomena identified by the PIRT process for NPM non-LOCA transients were evaluated with respect to the high-ranked phenomena identified by the PIRT process for NPM LOCA scenarios, as well as the NRELAP5 assessments performed as part of the NuScale LOCA evaluation model development. A gap analysis was performed to identify high-ranked phenomena for non-LOCA transients that are not assessed as part of the NuScale LOCA evaluation model development.

High-ranked phenomena for non-LOCA events that are not assessed as part of the NuScale LOCA evaluation model development were addressed in different ways:

1. Additional NRELAP5 code assessment performed against separate effects or integral effects test data
2. Code-to-code benchmark performed between NRELAP5 and independent system thermal-hydraulics code
3. Phenomenon is addressed as part of the downstream subchannel analysis

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Figure 6-6 Reflector / core bypass without fuel assemblies (for illustration only)



Lower Riser

Audit Question A-NonLOCA.LTR-63
RAI 10297 Question NonLOCA.LTR-1

{{

}}2(a),(c)

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Upper Riser

Audit Question A-NonLOCA.LTR-63
RAI 10297 Question NonLOCA.LTR-1

{{

}}2(a),(c) Normal flow in the riser is single-phase subcooled water. Transients that involve RPV depressurization or inventory loss can result in flashing and two-phase flow in the riser region.

{{

}}2(a),(c)

Figure 6-9 Typical reactor pressure vessel upper riser model

{{

}}2(a),(c)

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reactor trip from hot full power. An example of a normalized trip worth as a function of time, for which the control rods are fully inserted 2.278 seconds after being released by the control rod drive mechanisms, is presented in Table 7-2.

Table 7-2 Example of normalized trip worth vs. time after trip

Time After Trip (sec)	Normalized Trip Worth
0.0	0.0
0.428	0.011
0.616	0.044
0.766	0.099
0.900	0.176
1.022	0.276
1.138	0.397
1.220	0.502
1.250	0.540
1.458	0.706
1.952	0.893
2.278	1.0

7.1.6 Biasing of Other Parameters

This section describes the biasing of non-reactivity parameters used for the non-LOCA transient analyses. A brief discussion of each parameter is provided in the following sections.

- Initial Conditions (Section 7.1.6.1)
- Valve Characteristics (Section 7.1.6.2)
- Analytical Limits and Response Times (Section 7.1.6.3)

RAI 10297 Question NonLOCA.LTR-1

- Riser Holes (Section 7.1.6.4)

7.1.6.1 Initial Conditions

The initial conditions assumed for the non-LOCA transient analyses are the most adverse with respect to the acceptance criterion of interest. These conditions are normally consistent with steady state operation, allowing for calibration and instrument errors and steady state fluctuations. Recognizing that the initial conditions do not contribute equally to the severity of the event consequences, alternate approaches may be used to set these conditions. For instance, bounding values may be used for certain parameters to provide a more restrictive response for a specific acceptance criterion. Alternately, nominal conditions may be used if the event consequences are insensitive to a specific initial condition. A general description of the biasing of initial conditions, as applied to the non-LOCA transient analyses for an NPM, is provided below.

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Table 7-3 Examples of analytical limits and actuation delays (reactor trip system and engineered safety features actuation system)

Signal	Analytical Limit	Actuation Delay
High power	25% RTP (power < 15% RTP)	2.0 seconds
	120% RTP (power ≥ 15% RTP)	2.0 seconds
High count rate	5.0E+5 counts/second	2.0 seconds
Startup rate	3 decades/minute	31 seconds (source range)
	3 decades/minute	2 seconds (intermediate range)
High power rate	±15% RTP/minute	2.0 seconds
High RCS riser temperature	610°F	8.0 seconds
High containment pressure	9.5 psia	2.0 seconds
High pressurizer pressure	2000 psia	2.0 seconds
High pressurizer level	80%	3.0 seconds
Low pressurizer pressure	1720 psia	2.0 seconds
Low low pressurizer pressure	1600 psia	2.0 seconds
Low pressurizer level	35%	3.0 seconds
Low low pressurizer level	20%	3.0 seconds
Low steam pressure	300 psia	2.0 seconds
Low low steam pressure	100 psia	2.0 seconds
High steam pressure	800 psia	2.0 seconds
High steam superheat	150°F	8.0 seconds
Low steam superheat	0°F	8.0 seconds
Low RCS flow	1.7 ft^3/second	6.0 seconds
Low low RCS flow	0.0 ft^3/second	6.0 seconds
Low RCS level	350-390 inches	3.0 seconds
High CNV water level	220-260 inches	3.0 seconds
Low AC voltage	0 VAC	60.0 seconds

RAI 10297 Question NonLOCA.LTR-1

7.1.6.4 Riser Holes

Flow paths in the lower riser and upper riser are modeled based on their nominal dimensions and form losses per the NPM design. Tolerances specified on riser hole dimensions and elevations in the design ensure the uncertainty in riser hole area is small. Riser holes have minimal impact on the short-term non-LOCA transient progression. Sensitivity studies {{ }}$^{2(a),(c)}$ were performed. The sensitivity study results confirmed that {{ }}$^{2(a),(c)}$ the impact on figures of merit relevant to short-term non-LOCA was negligible.

7.1.7 Credit for Nonsafety-related Components or Operator Actions

There are three occasions where nonsafety-related equipment is credited for event mitigation by the non-LOCA transient analyses. Listed below is the equipment associated with these occurrences.

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8.0 Representative Calculations

RAI 10297 Question NonLOCA.LTR-1

The methodology of Chapter 7 is utilized in conjunction with the NRELAP5 model of Chapter 6 to provide representative transient results. The calculations are for a representative NPM design, consistent with the design characteristics in Chapter 3. Riser holes are not modeled in the representative calculations. The transients noted below were selected to demonstrate the application of the NuScale Non-LOCA methodology for analysis of the representative plant responses to a wide range of postulated equipment failures and malfunctions.

1. Cooldown and/or Depressurization of the RCS (Section 8.1)
2. Heatup and/or Pressurization of the RCS (Section 8.2)
3. Reactivity Anomaly (Section 8.3)
4. Increase in RCS Inventory (Section 8.4)
5. Decrease in RCS Inventory (Section 8.5)

The information included for each representative transient includes: an event description; the results for the acceptance criteria of interest; and, conclusions regarding the acceptance criteria of interest. These results are presented to demonstrate the application of the non-LOCA methodology to a representative NPM. Fuel rod and core physics parameter inputs for the representative transients were developed using COPERNIC (Reference 22) and SIMULATE5 (Reference 23) respectively.

8.1 Cooldown and/or Depressurization of the Reactor Coolant System

8.1.1 Decrease in Feedwater Temperature

The purpose of this section is to present the thermal-hydraulic response of a representative NPM for a decrease in feedwater temperature event. This event is evaluated for MCHFR.

8.1.1.1 Event Description

The general decrease in feedwater temperature (DFWT) event description can be found in Section 7.2.1.1. Based on Section 7.2.1.1, MCHFR is the only acceptance criterion that may be potentially challenged during the DFWT event. No single failure is applied since the challenging cases occur when all equipment operates as designed. No loss of power is applied since all loss of power scenarios terminate feedwater or trip the reactor, thus reducing the overcooling event. Chosen from a series of MCHFR sensitivity cases, the representative DFWT case presented here represents a case that could challenge MCHFR, based on the NRELAP5 MCHFR pre-screening. This case features the following conditions:

- Conservative initial condition biasing (as shown in Table 7-7) is applied in order to maximize the consequences of the overcooling event in terms of

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Enclosure 3:

Affidavit of Mark W. Shaver, AF-177435

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NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10297 R1, Question NonLOCA.LTR-1) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10297 R1, Question NonLOCA.LTR-1. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 19, 2024.

Mark W. Shaver

Mark W. Shaver

December 19, 2024 Docket No. 52-050

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 032 (RAI-10297 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 032 (RAI-10297 R1)," dated October 31, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10297 R1:

- NonLOCA.LTR-60

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 032 (RAI-10297 R1, Question NonLOCA.LTR-60). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 19, 2024.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-60, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-60, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-177721

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-60, Nonproprietary Version

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10297
Date of RAI Issue: 10/31/2024

NRC Question No.: NonLOCA.LTR-60

Issue

Section 7.1.2, “Treatment of Plant Controls,” of TR-0516-49416-P, “Non-Loss-of-Coolant Accident Analysis Methodology,” describes the approach for determining whether operation of non-safety-related control systems should be modeled. Specifically, the LTR states that operation of the control system will be modeled if it leads to a more severe event response and will not be modeled if operation leads to a less severe event response. Topical report section 7.2.2.3, “Biases, Conservatisms, and Sensitivity Studies,” identifies that as the increase in feedwater flow event creates the potential for overfilling the steam generator, additional sensitivity studies are performed to identify the case(s) with a potentially limiting steam generator level. {{

}}$^{2(a),(c)}$. As described in Section 5.4.3.3.4 of the final safety analysis report (FSAR) provided with the NPM-20 standard design approval application, high DHRS inventory may inhibit DHRS performance if there is inadequate surface area in contact with the two-phase mixture for boiling and condensation to be effective.

In response to questions raised during the audit, NuScale provided the biases, conservatisms, and sensitivity studies that will be evaluated to identify the highest steam generator level resulting from a potential increase in feedwater flow event. {{

}}$^{2(a),(c)}$.

NRC staff audited NuScale’s calculation of the increase in feedwater flow event supporting the standard design approval application for the NPM-20 design. {{

}}$^{2(a),(c)}$

{{

}}2(a),(c) .

{{

}}2(a),(c) . While NRC staff notes that the calculation methodology contains some conservatisms which help ensure a bounding steam generator level is identified {{

}}2(a),(c) it is unclear to NRC staff if these conservatisms compensate for the potential nonconservatism from assuming {{

}}2(a),(c) .

Information Requested

Demonstrate that each of these assumptions result in the limiting steam generator level or limiting DHRS capacity for the increase in feedwater flow event, or revise the Non-LOCA LTR with a revised treatment of {{ }}2(a),(c) . If assuming {{ }}2(a),(c) is more limiting for the NPM-20 design, revise the Non-LOCA LTR as well and make corresponding changes to FSAR Chapters 5, 15, or others as appropriate.

NuScale Response:

TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” is revised as indicated in the markup attached to this response. Because the markup includes

changes on top of those previously provided in the response to related audit question A-NonLOCA.LTR-60, a summary of the changes is provided for convenience as follows:

- Section 7.2 is revised to clarify that the turbine bypass system response to a turbine trip is not typically credited. This is an editorial change {{ }}2(a),(c)
- Table 7-15 is revised to indicate that the effect of steam pressure control on steam generator level and decay heat removal system performance is considered {{ }}2(a),(c)

The increase in feedwater flow event analysis has been revised to implement the changes to TR-0516-49416-P described above. The revised analysis, EC-0000-8328, Revision 1, “Increase in Feedwater Flow Transient Analysis,” is made available for NRC audit in the electronic reading room (eRR) for Final Safety Analysis Report (FSAR) Chapter 15. The EC-0000-8328, Revision 1, analysis shows that {{

}}2(a),(c)

Impact on Topical Report:

Topical Report TR-0516-49416, Non-Loss-of-Coolant Accident Analysis Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

RAI 10297 Question NonLOCA.LTR-60

An NPM utilizes a nonsafety-related turbine bypass system sized to handle full steam flow rate at 100 percent RTP. As such, the turbine bypass valves open following a turbine trip to control the RCS temperature without steam relief to the atmosphere. Since the turbine bypass system enhances heat removal by the secondary system, these actions are not typically credited for the non-LOCA transient analyses.

{{

}}2(a),(c)

Separate analyses for subchannel CHF, fuel centerline temperature, and containment pressure calculations are performed using the appropriate licensed NuScale methodologies. Extended cooldown via the DHRS is considered as part of the system design.

7.2.1 Decrease in Feedwater Temperature

The methodology used to simulate a postulated decrease in feedwater temperature for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.1.1 General Event Description and Analysis Methodology

The event is initiated by a feedwater system malfunction that causes a decrease in feedwater temperature, resulting in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity due to moderator feedback, which raises reactor power. Decreasing average RCS temperature also prompts the control rod controller to withdraw the regulating bank from the core if automatic control is enabled. Rising reactor power typically causes RTS actuation on a high power or high power rate signal, and DHRS actuation and isolation of the secondary system occur post-reactor trip on other MPS signals.

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Audit Question A-NonLOCA.LTR-60

Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow Not Used

Parameter	Bias / Conservatism	Basis
Initial reactor power	Varied.	{{ }}2(a),(c)
Initial RCS average temperature	Varied.	{{ }}2(a),(c)
Initial RCS flow rate	Varied.	{{ }}2(a),(c)
Initial PZR pressure	Varied.	{{ }}2(a),(c)
Initial PZR level	Varied.	{{ }}2(a),(c)
Initial feedwater temperature	Varied.	{{ }}2(a),(c)
Initial fuel temperature	Biased to the high condition.	{{ }}2(a),(c)
MTC	Varied.	{{ }}2(a),(c)
Kinetics	Varied.	{{ }}2(a),(c)
Decay heat	Biased to the high condition.	{{ }}2(a),(c)
Initial SG pressure(1)	Biased to the low condition.	{{ }}2(a),(c)
SG heat transfer	Varied.	{{ }}2(a),(c)
RSV lift setpoint	Nominal.	{{ }}2(a),(c)
SG tube plugging	Varied.	{{ }}2(a),(c)

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Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow (Continued) ~~Not Used~~

Parameter	Bias / Conservatism	Basis
Steam Pressure Control Turbine throttle valves	Varied.	{{ }}2(a),(c)
Turbine bypass valves	Varied.	
Feedwater and Turbine Load Control Feedwater pump speed	N/A.	{{ }}2(a),(c)
CNV Pressure Control CNV evacuation system	Enabled.	{{ }}2(a),(c)
1. {{		}}2(a),(c)

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Enclosure 3:

Affidavit of Mark W. Shaver, AF-177721

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NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10297 R1, Question NonLOCA.LTR-60) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10297 R1, Question NonLOCA.LTR-60. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 19, 2024.



Mark W. Shaver

January 09, 2025 Docket No. 52-050

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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 032 (RAI-10297 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 032 (RAI-10297 R1)," dated October 31, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10297 R1:

- NonLOCA.LTR-31, 32, 46, 56, 65

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 032 (RAI-10297 R1, Question NonLOCA.LTR-31, 32, 46, 56, 65). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on Janaury 09, 2025.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
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Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-31, 32, 46, 56, 65, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-31, 32, 46, 56, 65, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-177583

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-31, 32, 46, 56, 65, Nonproprietary Version

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10297
Date of RAI Issue: 10/31/2024

NRC Question No.: NonLOCA.LTR-31, 32, 46, 56, 65

Issue
The methodologies presented in the Non-LOCA LTR rely on sensitivity studies for various parameters in various events as identified in Table 7-7, Table 7-14, Table 7-19, Table 7-24, Table 7-28, Table 7-32, Table 7-36, Table 7-40, Table 7-44, Table 7-48, Table 7-52, Table 7-56, Table 7-60, Table 7-64, Table 7-68, Table 7-70, 7-74, Table 7-82, Table 7-86, and Table 7-91. However, the LTR only indicates what parameters require sensitivity studies without providing methodologies for these sensitivity studies. Additionally, the aforementioned tables also indicate bias directions for some parameters that result in limiting margins to figures of merit but does not explain how the biasing methodology considers the full range of operating conditions for each transient to determine the minimum margin parameters acceptance criterion (e.g. primary pressure).

• Determination of the limiting values of the input parameters

The Non-LOCA LTR does not provide a complete methodology to determine the limiting values of these parameters to assure the analyses will appropriately capture limiting initial conditions that would be challenging to acceptance criteria. It is unclear how these sensitivity studies are to be performed as part of the methodology. While the audited documents (ML23067A300) provided sample calculations that showed how the sensitivity studies are performed for these audited cases, the methodology used in the calculation may not be appropriate for all parameters as indicated in some event-specific methodology descriptions in the Non-LOCA LTR (e.g. Non-LOCA LTR Section 7.2.14.1).

• Determination of the initial points for which sensitivity studies are to be performed

Many events require consideration of a spectrum of event initiators. The purpose of sensitivity studies is to find the gradient of the function, which indicates the direction and magnitude the function would respond to a small perturbation. The results of the sensitivity analyses depend

on the initial points at which the analyses are performed. It is important to identify those points to appropriately assess sensitivities to the given parameters. However, the Non-LOCA LTR does not describe how the points along this spectrum are selected for analysis, or how other parameters are varied with these points. {{

}}[2(a),(c)] but is not clear how these points will be identified and whether this process will be repeated when other parameters are varied.

The previous revision of this LTR contained example sensitivity studies and accompanying discussion that demonstrated how sensitivity studies were to be performed, but these example sensitivity studies and discussion were removed in the current revision. As such, the Non-LOCA LTR does not adequately describe the methodology for ensuring that limiting cases are identified.

• When sensitivity studies should be performed

During a regulatory audit (ML23067A300), NRC staff requested clarification of various statements in Section 7.2 of the Non-LOCA LTR that sensitivity studies are performed “as needed” to identify the limiting responses for acceptance criteria challenged by this event.

{{

}}[2(a),(c)] However, the Non-LOCA LTR does not specify when sensitivity studies are needed, and instead leaves it to the discretion of the analysts.

• Generalized conclusion RCS pressure is insensitive to SG heat transfer

During its review, the staff also notes that for several specific events (e.g., turbine trip and main steam isolation valve (MSIV) closure) the initial conditions, bias, and conservatisms tables, Table 7-32 and Table 7-40, denote the steam generator heat transfer is set as “normal”. The staff requested the applicant to provide information to support that these settings are appropriate. During the audit (ML23067A300), the applicant stated {{

}}[2(a),(c)]

• Information on what additional sensitivity analyses are needed

The methodology presented in the Non-LOCA LTR states that additional sensitivity studies are performed on parameters, as necessary, for multiple events, such as a decrease in feedwater temperature, increase in feedwater flow, etc., to identify the case(s) with a potentially limiting MCHFR. For example, for the turbine trip / loss of external load event, Section 7.2.6.1 of the Non-LOCA LTR states: “Sensitivity studies on initial primary temperature and primary/secondary pressures are performed to identify the conditions that maximize peak primary and secondary pressures. Additional sensitivity studies are performed on other parameters, as necessary, to identify the case(s) with the potentially limiting peak primary and secondary pressures.” The Non-LOCA LTR, however, does not provide information on how to do the additional sensitivity studies, what sensitivities will be examined in these additional studies, and what the acceptance criterion is for these sensitivity studies, i.e., when a sensitivity study is sufficient to identify the limiting cases or how the analyst would determine the limiting case has been correctly identified.

• Supporting information on various biasing schemes

In all of the event-specific tables showing initial conditions, biases, and conservatisms, the staff notes that biasing the initial conditions for the normal initial full power case will not always result in a conservative change in the acceptance criterion parameters, due to the potential non-linear response from a change in a parameter (e.g., the coupled neutronic and thermal-hydraulic responses) and event progression, particularly when assuming different initial conditions in the base or reference case for an event. For example, the relative power peaking within an assembly may skew to significantly higher values at lower power levels than the one at full power, which can challenge the acceptance criteria in an AOO and can be more limiting than at full power. {{

}}[2(a),(c)] In determining a conservative bias, those cases which do have a maximum relative change in acceptance criteria need to be considered if they become the limiting case in determining the minimum margin to acceptance criteria. The applied biases also need to account for the potential non-linear response of a parameter where the most conservative case could be missed. Analyses to demonstrate that the biases used are appropriate and conservative are needed.

• Applicability of sensitivity studies performed for Non-LOCA LTR Revision 3

During the audit (ML23067A300), NuScale indicated that there are no substantial changes in Non-LOCA event progressions or the important phenomena between the NPM-160 and NPM-20 and that the use of biases in Revision 3, or update of biases based on NPM-160 insights, is

reasonable. However, the LTR does not include justification for the applicability of the methodologies to the NPM-20 design, which includes significant changes in the design features of the NPM-20 reactor.

Information Requested

a) Modify the LTR to provide methodologies for performing sensitivity studies for the parameters that are identified as “varied” in Table 7-7, Table 7-14, Table 7-19, Table 7-24, Table 7-28, Table 7-32, Table 7-36, Table 7-40, Table 7-44, Table 7-48, Table 7-52, Table 7-56, Table 7-60, Table 7-64, Table 7-68, Table 7-70, 7-74, Table 7-82, Table 7-86, and Table 7-91 as well as the methodology for biasing parameters in these tables. The information should include:
i. How to perform the sensitivity studies required by the LTR methodology
ii. How to determine the initial conditions at which the parameters are to be varied
iii. The acceptance criteria
iv. How biases are assured to be conservative at the other statepoints examined in sensitivity studies.

b) Modify the LTR to provide methodologies for performing sensitivity studies for the parameters that are referred to as “as needed” or “as necessary” in Section 7 of the LTR. The information should include:
i. Under what conditions the sensitivity studies need to be performed
ii. How to perform the sensitivity studies required by the LTR methodology
iii. How to determine the initial conditions at which the parameters are to be varied
iv. The acceptance criteria
v. How biases are assured to be conservative at the other statepoints examined in sensitivity studies.

c) Modify the LTR to provide methodologies for performing the “additional sensitivity studies” as described in Section 7 of the LTR. The information should include:
i. Under what conditions additional sensitivity studies need to be performed
ii. How to perform the sensitivity studies required by the LTR methodology
iii. How to determine the initial conditions at which the parameters are to be varied
iv. The acceptance criteriav.
v. Biases are assured to be conservative at the other statepoints examined in sensitivity studies.

d) Revise the LTR to:
i. Clearly identify which evaluation methodology (EM), as approved in revision 3 of the LTR,

remains applicable to which event(s) as listed in Section 7 in revision 4 of the LTR
ii. Provide technical justifications supporting the EM applicability conclusion made for each transient event. The design changes from NPM-160 to NPM-20 must be addressed in the technical justifications.

NuScale Response:

Parts a, b, and c

To address these three parts of the request additional information (RAI), NuScale provides a revised markup to TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology.” The markup also addresses the NRC concerns from the original audit questions (i.e., A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, and A-NonLOCA.LTR-65) associated with this RAI. The markup is attached to this response. Because the markup includes changes on top of those previously provided in the original response to some of these audit questions, a summary of the changes is provided for convenience as follows:

- Deleted use of "as necessary" or “as needed” discussion of sensitivity studies. The phrases are either just deleted or else replaced with a reference to the table that identifies the sensitivity studies to perform.
- Added “minimum margin” criteria for reactor coolant system (RCS) pressure, steam generator (SG) pressure, and minimum critical heat flux ratio (MCHFR) figures of merit. A result that goes below the minimum margin then requires additional sensitivity studies be performed to ensure the limiting case is identified.
- Restored the example sensitivity study results (both tables and associated text discussion) from the previously approved topical report version (TR-0516-49416-P-A, Revision 3).
- In the first section with representative sensitivity results (i.e., Section 7.2.1.3), added discussion for how the representative sensitivity studies define elements of the method. The general explanation in Section 7.2.13 is then applied to the remainder of the events. Examples include:
    - how to vary parameters,
    - how to determine if results of sensitivity studies are sensitive or insensitive, and
    - how to consider a spectrum of parameters (usually tied to event initiator) rather than a simple variation.

Note that the markup originally provided is revised based on NRC feedback to provide the following clarifications.

- The role of the reactor safety valve lifts in determining limiting reactor coolant system pressures is clarified.
- The discussion of sensitivity studies that are “repeated” is clarified.
- The criteria and examples for performing sensitivities studies to more than one parameter at a time is clarified.
- The role of engineering judgment in reviewing sensitivity study results is clarified.

In addition, NuScale identified some further changes to TR-0516-49416-P, Revision 4, to ensure conservatism of the event-specific methodology in Section 7.2. The changes are listed below. These changes are shown in markups that are appended to the end of the other markups (i.e., these new changes are shown in the last seven pages of the attached markups).

- The initial pressurizer level for decrease in feedwater temperature (Table 7-7), increase in feedwater flow (Table 7-14), and increase in steam flow (Table 7-19) events is modified from biased high to varied. The basis {{

  }}$^{2(a),(c)}$ The resultant methodology now includes more sensitivity cases than previously and, therefore, this change is conservative.
- The steam pressure control for the increase in steam flow (Table 7-19) event is clarified {{ }}$^{2(a),(c)}$ may be used to model the increase in steam flow initiating event. This is not a change in methodology, but simply a clarification of how the increase in steam flow can be modeled.
- The turbine throttle valve aspect of steam pressure control for the steam line break (Table 7-24) event is modified from enabled to varied. The basis {{

  }}$^{2(a),(c)}$ Therefore, sensitivity studies to determine the effect of the turbine throttle valve treatment should be performed to determine the treatment that

results in the more limiting MCHFR. The resultant methodology now includes more sensitivity cases than previously and therefore, this change is conservative.

Part d

The RAI states that “the LTR does not include justification for the applicability of the methodologies to the NuScale Power Module (NPM)-20 design, which includes significant changes in the design features of the NPM-20 reactor.” The RAI does not identify any specific design changes of concern to the non-loss-of-coolant accident (non-LOCA) event progressions. NuScale disagrees with the implied assertion that the design features of the NPM-20 reactor are “significant”’ with respect to the high-ranked phenomena for non-LOCA events that affect the non-LOCA event progressions. Part d of the request in this focuses specifically on the event-specific methodologies described in Section 7.2 and therefore these are discussed specifically in this response. However, technical justification for the event-specific methodologies described in Section 7.2 inherently rely on the overall EM development and applicability assessment provided in Sections 3, 4, 5, and 6 of TR-0516-49416-P.

NuScale followed the evaluation model development and assessment process (EMDAP) described in Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” to develop the previously approved non-LOCA evaluation model (EM), as described in TR-0516-49416-P-A, Revision 3. NuScale used the EMDAP to extend the EM for the NPM-20 design. There are four elements of the EMDAP:

1. Establish requirements for EM capabilities
   The analysis purpose, transient class, figures of merit, systems, components, phases, geometries, fields and processes to be modeled are unchanged. A key part of this element is to identify and rank phenomena and processes. NuScale evaluated the design changes from the NPM-160 to NPM-20. The NPM changes in initial conditions affect the ranges of some system transient process parameters. These changes were evaluated as part of demonstrating EM adequacy (element 4). NuScale determined that the design changes did not result in any new or different high-ranked phenomena for the scope of the non-LOCA event progression from event initiation through establishment of stable decay heat removal system (DHRS) cooling. Therefore, in development of TR-0516-49416-P, Revision 4, NuScale made minor changes to update Section 3, Plant Design Overview. Discussion of RCS pressure biasing to minimize critical heat flux margin was modified in Section 4, Transient and Accident Analysis Overview, to be applicable to both the NPM-160 and NPM-20 designs. Other parts of Section 4, and the

Section 5.1 identification of high-ranked phenomena remain applicable to the NPM-160 and NPM-20 designs.

2. Assessment base development
   The assessment basis of the previously approved TR-0516-49416-P-A, Revision 3, remains relevant to demonstrate applicability of the NRELAP5 code to predict the non-LOCA high-ranked phenomena and event progressions; therefore these sections of the topical report were maintained. NuScale chose to perform additional non-LOCA DHRS integral effects tests (IETs) at the NIST-2 facility to extend the NRELAP5 validation basis. These tests were scaled to representative NPM-20 conditions as described in TR-0516-49416-P, Revision 4, Section 5.3.7.2.2.
3. Develop evaluation model
   The structure of the previously approved EM is unchanged and remains applicable for both the NPM-160 and NPM-20 designs (i.e., use of NRELAP5 to predict the system thermal-hydraulic response). Therefore, in development of TR-0516-49416-P, Revision 4, NuScale made minor changes to update the Section 6 NRELAP5 plant model description to clarify critical aspects of the model while emphasizing that the base model for a specific NPM design reflects the parameters and features of that design. The previously approved NRELAP5 closure models remain applicable for the NPM-20 design. NuScale chose to develop {{ }}$^{2(a),(c)}$ for steam generator primary side heat transfer and provided justification of the model applicability as described in Section 5.3.6. This closure model provides consistent results to the previously approved model (i.e., {{ }}$^{2(a),(c)}$) and therefore is applicable to both the NPM-160 and NPM-20 designs.
4. Assess evaluation model applicability
   NuScale performed bottom-up and top-down assessments of the EM applicability to the NPM-20 design. As summarized in the TR-0516-49416-P, Revision 4, Section 5.4, many of the high-ranked phenomena for non-LOCA transients are also addressed as part of the LOCA evaluation model. Based on gap analyses, the non-LOCA EM applicability focuses on the steam generator and DHRS heat transfer phenomena. Bottom-up assessment concluded that the dominant constitutive models for boiling/condensation wall heat transfer phenomena in the steam generator and DHRS were developed over wide ranges of conditions and apply to the NPM-20 condition ranges. Top-down scaling analyses demonstrated the consistency in the dominant PI groups for steam generator and DHRS operation between the NPM-2 and NIST-2, thereby supporting the conclusion that the use of NRELAP5 in the EM is applicable for NPM-20 non-LOCA integral analysis.

Therefore, contrary to the statement in the RAI, NuScale followed the EMDAP to justify applicability of the EM to the NPM-20 design and the technical bases for the justification are provided throughout TR-0516-49416-P, Revision 4.

When preparing the event-specific methodology in Section 7.2 of TR-0516-49416-P, Revision 4, the starting point was TR-0516-49416-P-A, Revision 3. In general, the content in Section 7.2 of Revision 4 is similar to Revision 3 and therefore the technical basis is similar to that previously reviewed and approved by the NRC. However, for each event in Section 7.2 of Revision 4, the event-specific methodology has been validated as appropriate for the NPM associated with the Standard Design Approval Application (SDAA) of the US460 design (i.e., the NPM-20) as follows:

- The methodology used in Revision 3 was considered a starting point.
- Nuscale performed analyses supporting other design work between DCA and SDAA. Prior to developing the SDAA for the US460, other potential designs were investigated. The preliminary analyses associated with this design work validated the continued applicability of the Revision 3 methodology or otherwise identified required changes. The changes were incorporated in Revision 4.
- NuScale performed preliminary analyses supporting the SDAA. The preliminary analyses validated the continued applicability of the Revision 3 methodology or otherwise identified required changes. The changes were incorporated in Revision 4.
- NuScale performed the final analysis supporting the Final Safety Analysis Report (FSAR) Revision 0. The final analyses validated the continued applicability of the Revision 3 methodology or otherwise identified required changes. The changes were incorporated in Revision 4.
- NuScale updated the event-specific analyses to address NRC audit questions and RAIs received during the SDAA review, incorporate design changes made during the SDAA review, and resolve issues identified through the NuScale Corrective Action Program. The updated analyses are to be incorporated in FSAR Revision 2. The updated analyses validated the applicability of the methodology identified in Revision 4 or otherwise identified required changes. The changes were made as markups to Revision 4.

Because of the process used above, the event-specific methodology in Section 7.2 of Revision 4, including markups, is applicable to the NPM-20 of the US460 design. For this reason, item i of part d of the NRC request is not applicable. Regarding item ii of part d, the technical justification is provided throughout TR-0516-49416, as described, and is confirmed by the event-specific analyses for NPM-20 supporting the FSAR. These analyses demonstrate

conservative results for NPM-20 using the methodology in Section 7.2 of Revision 4, including markups. The NRC was provided the event-specific analyses supporting the FSAR for audit as part of the SDAA review. The fact that NuScale has made updates to Section 7.2 in Revision 4 and in the markups is evidence that the methodology has been revised to be applicable to, and consistent with, the NPM-20 of the US460 design.

Impact on Topical Report:

Topical Report TR-0516-49416, Non-Loss-of-Coolant Accident Analysis Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

NRELAP5 code was derived from the Idaho National Laboratory (INL) RELAP5-3D© computer code. RELAP5-3D©, version 4.1.3 was used as the baseline development platform for the NRELAP5 code. RELAP5-3D© was procured and commercial grade dedication was performed by NuScale. Subsequently, features were added and changes made to address unique aspects of an NPM design and licensing methodology. NRELAP5 is a non-homogenous, non-equilibrium two-fluid thermal hydraulic systems analysis code capable of performing non-LOCA system transient analyses for an NPM. The NRELAP5 code has a heat conduction and heat transfer package that is similar in capability to other thermal-hydraulic codes in its class (such as TRAC, RETRAN or TRACE). It includes the trips and logic control systems that enable simulation of the plant protection and control system logic for analysis of a non-LOCA event in an NPM. The NRELAP5 code is described in the NuScale LOCA Evaluation Model licensing topical report. The NRELAP5 code has been assessed against several separate effects and integral effects tests as part of the code development and development of the NuScale LOCA evaluation model to demonstrate the capability to simulate an NPM response to LOCA events.

Phenomena identified as high-ranked for the non-LOCA transients were evaluated with respect to the high-ranked phenomena identified and assessed as part of the NuScale LOCA evaluation model development. Additional validation of NRELAP5 against separate effects testing, integral effects testing, and code to code benchmarking, were performed as necessary to justify applicability of the NRELAP5 code for non-LOCA system thermal-hydraulic analysis. High-ranked phenomena for non-LOCA events that were not assessed as part of the NuScale LOCA evaluation model development were therefore addressed in different ways:

1. additional NRELAP5 code assessment performed against separate effects or integral effects test data
2. code-to-code benchmark performed between NRELAP5 and independent system thermal-hydraulics code
3. phenomena addressed as part of the downstream subchannel analysis
4. phenomena addressed by specifying appropriately conservative input to the system transient analysis

In particular, separate and integral effects testing were performed at the NIST-1 and NIST-2 facilities to support applicability of the NRELAP5 code for non-LOCA system transient analysis. Separate effects testing of the DHRS was performed. Integral effects testing of an NPM response to a decrease in secondary side heat transfer, and integral effects testing of DHRS operation were performed. A code-to-code benchmark was performed to compare the NRELAP5 and RETRAN-3D responses to a range of reactivity insertion conditions in an NPM. Computational fluid dynamics was used to independently assess the heat transfer correlation used in NRELAP5 for the helical SGs of the NuScale designs.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The general non-LOCA transient analysis process is described in this report. The general methodology for conservatively biasing initial and boundary conditions for event analysis is presented. Each initiating event is then considered to identify the acceptance criteria that may be challenged during the event. For each non-LOCA event, a description of the event is provided including biases and conservatisms applied, sensitivity studies performed, single active failures and loss of power scenarios that challenge the event acceptance criteria. For each transient

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event, the acceptance criteria where margin to the limit may be challenged are identified. For these acceptance criteria, sensitivity calculations are performed ~~as necessary~~ to confirm that appropriately conservative inputs are specified and to determine conditions that result in minimum margin. For other acceptance criteria where margin to the limit is not challenged, representative results from the overall scope of sensitivity calculations performed are sufficient to demonstrate that margin to the acceptance criterion is maintained. For non-LOCA initiating events that actuate the DHRS, the EM is applicable for the short-term transient progression; during this time frame the mixture level remains above the top of the riser and primary side natural circulation is maintained.

For selected non-LOCA events, representative system transient results are provided to demonstrate the application of the evaluation model for an NPM. System transient calculations are executed for sufficient duration to demonstrate that the initiating event is mitigated and stable cooling is established. Results of representative calculations show that the maximum primary and secondary pressure acceptance criteria are not significantly challenged in an NPM design. Margin to other quantitative acceptance criteria for minimum critical heat flux ratio, fuel centerline temperature, and radiological dose limits applicable for the non-LOCA events are demonstrated as part of downstream subchannel or accident radiological analyses that are described in separate reports. In addition, long-term cooling analysis methodology is presented in a separate report.

NuScale requests U.S. Nuclear Regulatory Commission (NRC) approval to use the EM described in this report for analyses of NPM design basis non-LOCA events that require system transient analysis. The specific scope of the non-LOCA events for which the EM applies is delineated in Section 1.2. NuScale is not seeking NRC approval of the representative calculations that are described in this report.

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4.2 Design Basis Event Acceptance Criteria

Safety analyses are performed to demonstrate that a nuclear power plant can meet applicable acceptance criteria for a limiting set of AOOs, IEs, and accidents. If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of a plant should be such that events produce about the same level of risk. The acceptance criteria indicated by the GDC for nuclear power plants (Reference 4) reflect the risk of an event. Relatively frequent events such as AOOs are prohibited from resulting in serious consequences, but relatively rare events (postulated accidents) are allowed to produce more severe consequences.

Design basis events for an NPM are categorized as AOOs, IEs, or postulated accidents. Table 4-2, Table 4-3, and Table 4-4 summarize the acceptance criteria applied for AOOs, IEs, and postulated accidents, respectively.

The applicable acceptance criteria identified for each event are based on the event classification as identified in Table 4-1. For a limited number of events, a more conservative acceptance criterion may be applied than required based on the event classification.

For many non-LOCA transient events, the specific acceptance criterion is not challenged during the event progression. For example, events that result in an increase in heat removal from the RCS may have a maximum RCS pressure higher than the initial operating pressure, but does not challenge the margin to the maximum RCS pressure acceptance criterion. In contrast, events that result in a decrease in heat removal from the RCS may result in an RCS pressurization that could challenge the maximum RCS pressure acceptance criterion. In Section 7.2, the acceptance criteria of interest for each non-LOCA event are identified. The acceptance criteria of interest are those where margin to the limit may be challenged during the event progression.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

In the event-specific transient analysis, sensitivity calculations are performed ~~as necessary~~ to ensure that the event meets acceptance criteria that may be challenged. These sensitivity calculations are performed to confirm that appropriately conservative inputs are specified to identify the case that results in minimum margin to the acceptance criterion of interest. For other acceptance criteria where margin to the limit is not challenged, representative results from the overall scope of sensitivity calculations performed are sufficient to demonstrate that margin to the acceptance criterion is maintained.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

A prime example of an acceptance criterion where an NPM design has significant margin is the maximum secondary system pressure. Unlike in typical PWR designs, in a NuScale design, the design pressure of the SG secondary side up to the second containment isolation valves is equal to the RCS design pressure. This feature supports the design and operation of the SG and DHRS. In a non-LOCA event that results in DHRS actuation, typically the maximum secondary side pressure occurs in the first minutes of the transient progression, following DHRS actuation. After DHRS is actuated, the fluid in the DHRS flows into the SG. Heat is transferred from the RCS primary system to the SG, where the

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DHRS loop inventory boils in the SG tubes. The steam flow is then condensed in the DHRS condensers and the energy is transferred to the reactor pool UHS. The maximum pressure in the SG secondary side is limited to the saturation pressure at the temperature of the RCS fluid on the SG primary side. Therefore, the maximum secondary pressure is affected by the secondary side inventory and the primary side conditions at the time of DHRS actuation, and is less sensitive to a specific initiating event. The SG design pressure is significantly higher than pressures expected during DHRS operation. The margin to the SG design pressure is physically limited, based on the primary side conditions. The representative transient results in Section 8.0 demonstrate that significant margin to the maximum SG pressure acceptance criterion is maintained for all types of events. Therefore, extensive sensitivity calculations to maximize secondary side pressure are not necessary for the non-LOCA transient analysis calculations. {{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Another example is RCS pressure. In the NPM, operation of an RSV at its lift setpoint is sufficient to mitigate an increasing pressure with minimal overshoot. This design feature ensures that the maximum RCS pressure does not significantly exceed the RSV lift setpoint. With appropriate selection of the RSV lift setpoint, including consideration of uncertainty, the maximum RCS pressure will be well below the acceptance criterion. Event analyses where pressure increases during the transient apply biases and include sensitivities to reach conditions where RSV lift may occur. If RSV lift occurs, modification of other biases can generate many cases with similar maximum RCS pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases with RSV lift. If RSV lift does not occur for an event, then RCS pressure will not challenge the acceptance criterion and extensive sensitivity studies are not required to investigate differences in RCS pressure below the RSV lift setpoint. {{

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

}}2(a),(c)

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4.3.1.1.2 Axial Power Shape

For the system transient analysis, a single channel core model is used, as described in Section 6.0. A nominal center-peaked average axial power shape is input for the single core channel for consistency with development of the reactivity feedback coefficients determined in the core design analyses.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Uncertainties associated with the axial power shape and axial and radial power peaking factors that can affect the ~~minimum critical heat flux ratio (~~MCHFR~~)~~ and peak centerline fuel temperature are accounted for in the downstream subchannel analyses as described in Reference 6, supplemented by Reference 28. The design features of an NPM preclude challenge to the primary and secondary pressure acceptance criteria as discussed in Section 7.0 and as shown in the representative results in Section 8.0. 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, use of a nominal center-peaked average axial power shape input is appropriate for the system transient analyses.

4.3.1.1.3 Energy Deposition Factor

The energy deposition factor is the portion of the energy generated in the core that is directly deposited in the fuel. A bounding high energy deposition factor that results in all energy being deposited in the fuel is used in the non-LOCA analyses. For cooldown or reactivity insertion events that cause total core power changes, increasing the core energy deposited in the fuel slows the thermal-hydraulic response of the system to changing power levels. The effect of the energy deposition factor on the primary system pressure response is generally insignificant except in very fast reactivity events such as control rod ejection where Doppler reactivity feedback is important; control rod ejection analysis is outside the scope of this EM. Sensitivity studies on this parameter confirm that its effect on the system response is not significant with respect to demonstrating margin to acceptance criteria for NPM events addressed by the non-LOCA evaluation model due to the MPS design and selection of the analytical limits to actuate reactor trip.

4.3.1.2 Interface with Fuel Rod Performance Design (Input to the Transient Analysis)

Fuel rod design analysis performed using a fuel performance code approved for a NuScale design provides input to the system transient analysis. The NuScale transient analysis methodology using NRELAP5 can be applied to a typical light water reactor fuel assembly design, and does not require that a specific code or suite of codes be used for the fuel performance analysis. The fuel assembly geometry, required material properties, and the fuel performance data needed for

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Table 7-4 Regulatory acceptance criteria (Continued)

Description	AOO Criteria	IE Criteria	Accident Criteria
Containment integrity(2)	< Limits	< Limits	< Limits
Escalation of an AOO to an accident (AOO) or consequential loss of system functionality (IE or accident)?	No	No	No
Dose(1)	Normal operations	< Limit	< Limit

1. This criterion is confirmed as part of a separate follow-on analysis.
2. Containment integrity is evaluated by a separate analysis methodology.
3. If the minimum CHFR is less than or equal to the CHFR analysis limit, or if the maximum fuel centerline temperature exceeds the melting temperature, the fuel rod is assumed to be failed. If fuel failure is calculated, this is accounted for in the downstream radiological dose analysis.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For each event analyzed following the non-LOCA evaluation model, a description of the event progression, significant inputs and results, and representative results of sensitivity studies are presented in the following sections. Sensitivity studies are performed to identify plant conditions that result in bounding transient analyses. Studies that identify acceptance criteria challenges or bounding transient forcing functions are discussed as well. Other sensitivity studies that determine bounding inputs for RCS flow, fuel parameters, etc. may not necessarily be discussed for every event. The selection of parameters to be studied is focused on the acceptance criteria challenged by the event. For events where a parameter has minimal or no impact on the consequences, the bias or conservatism identified for the parameter is that typically applied; use of an alternate assumption is also acceptable. The sensitivity study results provided herein are for a representative NPM (NPM-160). Sensitivity studies ~~are repeated as necessary for different NPM designs~~performed as part of the implementation of the methodology are identified in the following sections. In addition, Section 7.2.1.3 provides example sensitivity studies for one specific event that are then used to generally define and demonstrate the methodology for the events in Section 7.2.

Initial RCS flow is typically biased to the low condition in all event simulations because this is bounding for MCHFR. {{

}}2(a),(c)

Steam generator tube plugging is considered for each event in the "Initial conditions, biases, and conservatisms" tables. The term, "Biased to the low condition" indicates no tube plugging is assumed. “Biased to the high condition” indicates {{ }}2(a),(c) steam generator tube plugging.

In an NPM design, the rod control system is set to “insert only” mode at full power to prevent automatic withdrawal of the control bank at full power. Although this plant feature exists, the feature is not credited during events where control rod withdrawal results in a bounding result.

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conservative alternative to accounting for the decalibration effect on that power-related reactor trip signal in the analysis.

To maximize the overall feedwater temperature change, the feedwater temperature transient starts at the initial (full power) feedwater temperature biased to the high condition, and terminates at the coldest temperature in the secondary, which is saturation temperature at condenser vacuum conditions. A sensitivity study on feedwater temperature cooldown rate is performed to identify the rate that results in limiting conditions.

{{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Additional sensitivity studies are performed on other parameters, as identified in Table 7-7~~necessary~~, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation, covered by a separate methodology.

Table 7-5 Acceptance criteria, single active failure, loss of power scenarios - decrease in feedwater temperature

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	CHF is the challenged acceptance criterion for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
No single failure	The challenging cases typically occur when all equipment operates as designed.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event.

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7.2.1.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-7 are considered in identifying a bounding transient simulation for MCHFR.53

Audit Question A-NonLOCA.LTR-28
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.~~Biased to the high condition~~	
Initial PZR level	Varied.~~Biased to the high condition.~~	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Nominal.	
Moderator Temperature Coefficient (MTC)	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	}}2(a),(c)

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Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature (Continued)

Parameter	Bias / Conservatism	Basis {{
SG tube plugging	Biased to the low condition.	
Minimum feedwater temperature	Biased to the low condition.	
Feedwater temperature cooldown rate	Spectrum.~~Varied~~	
RCS Temperature Control Automatic rod control Boron concentration	 Varied. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied.~~Disabled.~~ Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	}}2(a),(c)

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Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature (Continued)

Parameter	Bias / Conservatism	Basis
Steam Pressure Control Turbine throttle valves Turbine bypass valves	Varied Disabled.	{{
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

For example, sensitivity studies are performed to consider the effects of fuel-related parameters (initial fuel temperature, time in life). Representative results for these studies are presented in Table 7-8 and Table 7-9. {{

}}2(a),(c)

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The next example sensitivity study, Table 7-10, identifies the feedwater temperature decrease rate that minimizes MCHFR. {{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The last example sensitivity study for this event evaluates the effects of steam system boundary condition type on MCHFR and assesses the effects of a single failure of an MSIV to isolate. Representative results are presented in Table 7-11. The boundary condition in Table 7-11 is a constant steam flow, which allows the steam pressure to move in response to the transient. The previous study in Table 7-10 models a constant steam pressure boundary condition that allows flow to vary while maintaining pressure. {{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

~~The results of these studies are provided to demonstrate the process by which impact of the the bias of a parameter can be determined. For sensitivity studies that show a response similar to Table 7-8, the limiting bias should be selected. For sensitivity studies that show a response similar to Table 7-9, there is not a bias direction that is limiting and the parameter can be selected based on other analysis considerations. Other events in Section 7.2 use the same general sensitivity study process.~~

Table 7-8 Representative fuel exposure study

{{

}}2(a),(c)

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Table 7-9 Representative fuel temperature study

{{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-10 Representative feedwater temperature transient study~~Not Used~~

{{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-11 Representative boundary condition type / single active failure studies~~Not Used~~

{{

}}2(a),(c)

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The results of these representative sensitivity studies are provided as examples for this event. However, the sensitivity studies also demonstrate the methodology used generally for the events in Section 7.2.

- Table 7-8 demonstrates how to "vary" a parameter that has a range of possible initial values. {{

}}2(a),(c)

- Table 7-9 also demonstrates how to "vary" a parameter that has a range of possible initial values. {{

}}2(a),(c)

- Table 7-10 demonstrates how to consider a "spectrum" of values for a parameter. {{

}}2(a),(c)

- Table 7-11 demonstrates how to "vary" a control system parameter that has two basic states {{

}}2(a),(c)

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{{

}}2(a),(c)

- Based on the results of sensitivity studies for individual parameters, engineering judgment is used to determine whether a figure of merit is sensitive to a specific parameter. The parameters that show sensitivity in the figure of merit are then combined to identify limiting cases for the figure of merit. {{

}}2(a),(c) No specific quantification is provided to define the threshold for sensitive vs. insensitive. Instead, a two-tiered approach is used based on the available margin for the figure of merit as described in Section 4.2. When the margin to an acceptance criterion is less than the thresholds defined in Section 4.2, a reduction in margin due to change in a parameter should not be considered negligible.

7.2.2 Increase in Feedwater Flow

The methodology used to simulate a postulated increase in feedwater flow for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.2.1 General Event Description

A feedwater system malfunction that causes an increase in feedwater flow results in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity due to moderator feedback, which raises reactor power. Decreasing average RCS temperature also prompts the control rod controller to withdraw the regulating bank from the core if automatic control is enabled. Rising reactor power can cause RTS actuation on the high power or high pressurizer pressure signal. The feedwater flow increase can also cause RTS actuation on low steam line superheat or high steam line pressure. DHRS also actuates on one of the several signals that cause RTS.

Closure of the FWIVs following DHRS actuation isolates the SGs from the feedwater source, ending the overcooling event. Core decay heat drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient calculation is

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

{{

}}[2(a),(c)]. Additional sensitivity studies are performed on other parameters, as ~~necessary~~identified in Table 7-14, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

Audit Question A-NonLOCA.LTR-60

Table 7-12 Acceptance criteria, single active failure, loss of power scenarios - increase in feedwater flow

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	CHF is challenged for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
SG overfill	An increase in feedwater flow could cause the SG to overfill.
No single failure	~~T~~Except when considering SG overfill, the challenging cases typically occur when all equipment operates as designed.
Failure of one FWIV to close	FWIV single failure is typically limiting for SG overfill.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event and the potential for SG overfill.

7.2.2.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-13.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied. ~~Biased to the high condition.~~	
Initial PZR level	Varied ~~Biased to the high condition.~~	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Nominal.	
MTC	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the low condition.	

}}2(a),(c)

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Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
Steam Pressure Control Turbine throttle valves	Enabled to control to constant steam pressure.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control		
~~f~~Feedwater pump speed	Spectrum.~~N/A.~~	
CNV Pressure Control		
CNV evacuation system	Enabled.	

{{ }}2(a),(c)

1. {{ }}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

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For example, two sensitivity studies are performed to identify the most challenging secondary biases for the feedwater flow transient for MCHFR. {{ }}2(a),(c)

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{{

}}2(a),(c) Representative results for these studies are presented in Table 7-16. These results demonstrate an example of {{

}}2(a),(c) as described in the general methodology discussion in Section 7.2.1.3.

Audit Question A-NonLOCA.LTR-60
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The increase in feedwater flow also creates the potential for overfilling the SG. The assumptions in ~~Table 7-12 and~~ Table 7-14 related to identifying the limiting MCHFR do not necessarily result in the maximum SG level. The biases and conservatisms indicated in Table 7-15, and the single failure assumption in Table 7-12, are considered in identifying a bounding transient simulation for SG overfill. ~~For example, a single failure of a feedwater isolation valve to close is expected to maximize SG level. Additional s~~Sensitivity studies are performed ~~on the Table 7-14 parameters, as necessary,~~ to identify the case(s) with a potentially limiting SG level.

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RAI 10297 Question NonLOCA.LTR-60, RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow ~~Not Used~~

Parameter	Bias / Conservatism	Basis
Initial reactor power	Varied.	{{ }}2(a).(c)
Initial RCS average temperature	Varied.	{{ }}2(a).(c)
Initial RCS flow rate	Varied.	{{ }}2(a).(c)
Initial PZR pressure	Varied.	{{ }}2(a).(c)
Initial PZR level	Varied.	{{ }}2(a).(c)
Initial feedwater temperature	Varied.	{{ }}2(a).(c)
Initial fuel temperature	Biased to the high condition.	{{ }}2(a).(c)
MTC	Varied.	{{ }}2(a).(c)
Kinetics	Varied.	{{ }}2(a).(c)
Decay heat	Biased to the high condition.	{{ }}2(a).(c)
Initial SG pressure(1)	Biased to the low condition.	{{ }}2(a).(c)
SG heat transfer	Varied.	{{ }}2(a).(c)
RSV lift setpoint	Nominal.	{{ }}2(a).(c)

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Table 7-15 Initial conditions, biases, and conservatisms for SG overfill - increase in feedwater flow (Continued) ~~Not Used~~

Parameter	Bias / Conservatism	Basis
SG tube plugging	Varied.	{{ }}2(a),(c)
RCS Temperature Control Automatic rod control Boron concentration	Varied. Not credited.	{{ }}2(a),(c)
PZR Pressure Control PZR spray PZR heaters	Disabled. Nominal.	{{ }}2(a),(c)
PZR Level Control Charging Letdown	Not credited. Disabled.	{{ }}2(a),(c)

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Table 7-16 Representative increase in feedwater flow study - high and low SG performance with maximum power and minimum RCS flow~~Not Used~~

{{

}}2(a),(c)

7.2.3 Increase in Steam Flow

The methodology used to simulate a postulated increase in steam flow for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.3.1 General Event Description

In an NPM design, two main steam lines exit the CNV, and combine to form a common main steam line that connects to the high pressure turbine. The turbine bypass and main steam safety valves are located in the common main steam line, which is downstream of the MSIVs. A spurious opening of the turbine bypass valve or a main steam safety valve would cause an increase in steam flow, which results in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity due to moderator feedback, which raises reactor power. Decreasing average RCS temperature also prompts the control rod controller to withdraw the regulating bank from the core if automatic control is enabled (rod withdrawal at 100 percent power is inhibited, however, it is conservatively allowed in the analysis). Rising reactor power typically causes RTS actuation on a high power or high power rate signal. Additionally, a large increase in steam flow would rapidly depressurize the secondary system and could cause

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}}2(a),(c) Neglecting reactor trip on a power-related reactor trip signal is a conservative alternative to accounting for the decalibration effect on that power-related reactor trip signal in the analysis.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The increase in steam flow event starts at the initial (full power) steam flow. Sensitivity studies are performed on the degree of steam flow increase.

{{

}}2(a),(c) to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

Table 7-17 Acceptance criteria, single active failure, loss of power scenarios - increase in steam flow

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	CHF is challenged for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
No single failure	Note that a single active failure of a FWIV to close would occur after RTS and DHRS actuation, subsequent to when the MCHFR occurs. Consequently, the MCHFR occurs before the single active failure of an FWIV to close could affect the transient. Otherwise, the challenging cases typically occur when all equipment operates as designed.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event.

7.2.3.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-18.

Table 7-18 Acceptance criteria - increase in steam flow

Acceptance Criteria	Discussion
Primary pressure	Primary pressure initially drops as inventory shrinks due to increased heat removal. As reactor power increases and as the PZR heaters respond, an increase (typically less than 100 psi) in primary pressure is observed.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.~~Biased to the high condition.~~	
Initial PZR level	Varied.~~Biased to the high condition.~~	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Biased to the high condition.	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Steam flow increase	Spectrum.~~Varied~~	}}2(a),(c)

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Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow (Continued)

Parameter	Bias / Conservatism	Basis
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	{{ }}2(a),(c)
CNV Pressure Control CNV evacuation system	Enabled.	

1. {{ }}2(a),(c)
2. {{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, two sensitivity studies are performed to identify the most challenging increase in steam flow for MCHFR. {{

}}2(a),(c) Representative results for these studies are presented in Table 7-20 and Table 7-21. These results provide an example of varying a parameter {{ }}2(a),(c) as described in the general methodology discussion in Section 7.2.1.3.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-20 Representative steam flow study - nominal steam generator heat transfer~~Not Used~~

{{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-21 Representative steam flow study - steam generator heat transfer biased low~~Not Used~~

{{

}}2(a),(c)

7.2.4 Steam System Piping Failure Inside or Outside of Containment

The methodology used to simulate a postulated steam system piping failure for an NPM, and an evaluation of the acceptance criteria listed in Table 7-4, are presented below. Since both split breaks (relatively higher event frequency) and double-ended guillotine breaks (relatively lower event frequency) are analyzed, the more restrictive AOO criteria for system pressures, critical heat flux ratio, and fuel centerline melt applicable to breaks with higher event frequency are used in the evaluation. Radiological dose consequences are assessed as part of the downstream accident radiological dose analysis, documented in a separate report, and compared against the appropriate acceptance criteria.

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7.2.4.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-23.

Table 7-23 Acceptance criteria - steam line break

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of this cooldown event, primary pressure remains below the acceptance criterion for peak primary pressure.
Secondary pressure	Due to the depressurizing nature of this cooldown event, secondary pressure remains below the acceptance criterion for peak secondary pressure.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis. Sensitivity cases are performed to support the follow-on MCHFR evaluation.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation to a more serious accident or consequential loss of functionality	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.4.3 Biases, Conservatisms, and Sensitivity Studies

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The biases and conservatisms indicated in Table 7-24, and a spectrum of break locations and sizes, are considered in identifying a bounding transient simulation for MCHFR and mass release.

In the radiological analysis, primary-to-secondary leakage is assumed, which allows primary coolant to be released with the SG break flow. The transient SG mass release can be calculated for use as an input to the downstream radiological analysis. Alternatively, bounding assumptions for primary coolant release can be used in the radiological analysis to eliminate the need for calculating SG mass release. If such bounding assumptions are used, transient analysis to maximize SG mass release is not required.

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Table 7-24 Initial conditions, biases, and conservatisms - steam line break (Continued)

Parameter	Bias / Conservatism	Basis[1]
Steam Pressure Control Turbine throttle valves	~~Enabled~~Varied.	{{
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

{{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response and challenging mass releases for this overcooling event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Representative sensitivity study results for MCHFR are presented in Table 7-25. In the cases modeling breaks outside of containment, the MSIV on the train with the break is assumed to fail to close, resulting in the complete emptying of the

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affected SG. In cases modeling breaks inside of containment, the FWIV on the train with the break is assumed to fail to close. Break size is the faction (percent) of the pipe cross-sectional area. {{

}}2(a),(c) The sensitivity study demonstrates using a spectrum analysis {{

}}2(a),(c) consistent with the general methodology discussion in Section 7.2.1.3.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-25 Representative steam line break study~~Not Used~~

{{

}}2(a),(c)

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Table 7-27 Acceptance criteria - containment flooding / loss of containment vacuum

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of the event, sensitivities that maximize primary pressure are not analyzed. Peak primary pressure resulting from CNV flooding/loss of CNV vacuum is bounded by other AOO events.
Secondary pressure	Due to the primary system depressurization of this cooldown event, secondary pressure remains below the acceptance criterion for peak secondary pressure.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.5.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-28 are considered in order to identify the bounding transient simulation for MCHFR for the CNV flooding/loss of CNV vacuum event.

Audit Question A-NonLOCA.LTR-28
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied	}}2(a),(c)

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Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial PZR level	Nominal	
Initial feedwater temperature	Nominal	
Initial fuel temperature	Nominal	
MTC	Biased to the EOC condition.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Initial containment pressure	Varied	
Initial pool temperature	Varied	
RCCW leak flow	Varied	
RCCW temperature	Varied	
RCS Temperature Control Automatic rod control Boron concentration	 Enabled. Not credited.	 }}2(a),(c)

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Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum (Continued)

Parameter	Bias / Conservatism	Basis
PZR Pressure Control		{{
PZR spray	Varied.~~Disabled.~~	
PZR heaters	~~Enabled~~Nominal.	
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control		
Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	}}2(a),(c)

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Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum (Continued)

Parameter	Bias / Conservatism	Basis
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	{{ }}2(a),(c)
CNV Pressure Control CNV evacuation system	Enabled. (CNV flooding) N/A. (CNV vacuum loss)	

1. {{ }}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

As an example, the initial reactor power, pressurizer pressure, RCS average coolant temperature, RCS flow rate, containment pressure, pool temperature, RCCW leak flow rate, RCCW temperature are~~were~~ varied for a representative NPM for the CNV flooding/loss of CNV vacuum event as indicated in Table 7-29. The sensitivity studies indicate~~d~~ that the CNV flooding cases are more challenging to MCHFR than loss of CNV vacuum. The sensitivity study results also indicate~~d~~ that for a variety of initial RCS conditions, reactor pool conditions, and condition of liquid or air ingress to containment, loss of containment vacuum or containment flooding results in a slow overcooling transient that is non-limiting with respect to MCHFR compared to other cooldown event AOOs.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-29 Representative sensitivity studies - containment flooding / loss of containment vacuum~~Not Used~~

{{

}}2(a),(c)

7.2.6 Turbine Trip / Loss of External Load

The methodology used to simulate a postulated turbine trip/loss of external load (LOEL) for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.6.1 General Event Description

Turbine trip initiates with a turbine stop valve closure while loss of external load initiates with a turbine control valve closure. Otherwise, these transients are essentially equivalent and result in the sudden removal of the secondary side heat sink, overpressurization of the secondary system, and overheating of the RCS. Rising system pressures typically result in RTS actuation on the high PZR or steam pressure signal. Reactor trip and transition to stable DHRS flow terminates

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the transient with the NPM in a safe, stable condition. Table 7-30 lists the relevant acceptance criteria, SAF, and LOP scenarios.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The limiting pressure responses typically occur when the event is initiated from full power conditions, and the initial conditions are biased in the conservative directions. Sensitivity studies on initial primary temperature and primary/secondary pressures are performed to identify the conditions that maximize peak primary and secondary pressures. ~~Additional sensitivity studies are performed on other parameters, as necessary, to identify the case(s) with the potentially limiting peak primary and secondary pressures.~~

Table 7-30 Acceptance criteria, single active failure, loss of power scenarios - turbine trip / loss of external load

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Primary and secondary pressures are challenged during this overheating event.
Failure of one FWIV to close	Typically challenging for secondary side pressurization cases (negligible effect for primary side pressurization cases, which assume loss of AC power at event initiation and therefore feedwater is lost).
Loss of AC power at transient initiation No loss of power	Typically maximizes primary pressure (feedwater is lost). Typically maximizes secondary pressure.

7.2.6.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-31.

Table 7-31 Acceptance criteria - turbine trip / loss of external load

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation and continued FW operation, in addition to the actual turbine trip or loss of external load. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.

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Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load (Continued)

Parameter	Bias / Conservatism	Basis {{
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Disabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Disabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity study results are presented in Table 7-33. The results indicate that maximum RPV pressure occurs when PZR pressure is biased to the high condition, average temperature is biased to the low condition, and a loss of power leads to a loss of the secondary side heat sink (since feedwater is lost, the failure of an FWIV to close would have no effect on the event). The results also confirm that many cases have similar maximum RPV pressures and that there is significant margin to the acceptance criterion (2310 psia for this representative NPM). With respect to secondary side SG pressure, maximum SG pressure occurs when average temperature and SG pressure are biased to the high condition, and all power sources are available - the failure of an FWIV to close increases the peak SG pressure by a very small amount. The results also confirm that many cases have similar maximum secondary side pressures and remain well below the secondary design pressure (2100 psia for this representative NPM).

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-33 Representative sensitivity studies - turbine trip / loss of external load~~Not Used~~

{{

}}2(a),(c)

7.2.7 Loss of Condenser Vacuum

The methodology used to simulate a postulated loss of condenser vacuum for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4 are presented below.

7.2.7.1 General Event Description

Loss of condenser vacuum initiates with a turbine stop valve closure. Also, a loss of condenser vacuum is postulated to lead to a loss of feedwater flow. Turbine trip and loss of feedwater result in the sudden removal of the secondary side heat sink, pressurization of the secondary system, and overheating of the RCS. Rising system pressures typically result in a rapid RTS actuation on either high PZR or steam pressure. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition.

The loss of condenser vacuum event is essentially equivalent to the turbine trip / loss of external load events discussed in Section 7.2.6. The main difference is that the loss of condenser vacuum event includes a loss of feedwater flow at event initiation. However, because the turbine trip / loss of external load events consider a loss of normal AC power at event initiation, those events also model a loss of feedwater flow at event initiation. As a result, the scenarios analyzed as part of Section 7.2.6 address the loss of condenser vacuum event. Therefore, the relevant acceptance criteria, SAF, and LOP scenarios from Table 7-30 are also applicable to the loss of condenser vacuum event.

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The MSIV closure event can occur when one or both MSIVs close unexpectedly. The limiting pressure responses typically occur when the event is initiated from full power conditions, and the initial conditions are biased in the conservative directions.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies on number of MSIVs closing, initial primary temperature and primary/secondary pressures are performed to identify the conditions that maximize peak primary and secondary pressures. Additional sensitivity studies are performed on other parameters, as identified in Table 7-40~~necessary~~, to identify the case(s) with the potentially limiting peak primary and secondary pressures.

Table 7-38 Acceptance criteria, single active failure, loss of power scenarios - main steam isolation valve closure

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Primary and secondary pressures are challenged during this overheating event.
No single failure Failure of one FWIV to close	Typically challenging for primary pressure. Typically challenging for steam generator pressure.
Loss of AC Power at transient initiation No loss of power	Typically maximizes primary pressure. Typically maximizes secondary pressure.

7.2.8.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-39.

Table 7-39 Acceptance criteria - main steam isolation valve closure

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation and continued FW operation, in addition to the actual main steam isolation valve closure. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.

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Table 7-40 Initial conditions, biases, and conservatisms - main steam isolation valve closure (Continued)

Parameter	Bias / Conservatism	Basis
		{{
feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity study results are presented in Table 7-41. The results presented consider the closure of two MSIVs, which isolates both steam generators, and no single active failure. Maximum pressures are similar for cases in which the RSV actuation limits the RPV pressurization and remain well below the acceptance criterion (2310 psia for this representative NPM). For example, the maximum calculated pressure for the peak RPV pressure study is approximately equal for two additional cases, namely, decreased steam generator heat transfer and decreased feedwater temperature. Additionally, the results of the sensitivity studies indicate that maximum RPV pressure occurs when PZR pressure is biased to the high condition, RCS flow is at the nominal value (if maximum RCS pressure is below the RSV lift pressure), and a loss of AC power leads to a loss of the secondary side heat sink (loss of AC power occurs coincident with reactor trip). With respect to secondary side SG pressure, maximum SG pressure occurs when average temperature is biased to the high condition, RCS flow is biased to the low condition, and all power sources are available. The results also confirm that many cases have similar maximum secondary side pressures and remain well below the secondary design pressure (2100 psia for this representative NPM).

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-41 Representative sensitivity studies - main steam isolation valve closure~~Not Used~~

{{

}}2(a),(c)

7.2.9 Loss of Nonemergency AC Power

The methodology used to simulate a postulated loss of nonemergency (normal) AC power for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.9.1 General Event Description

The low voltage AC electrical distribution system (ELVS) supplies AC power to plant motors, heaters, packaged equipment, and battery chargers. Loss of normal AC power to the station auxiliaries can result from electrical grid-related failures, failures in plant or switchyard equipment, or external weather events. The nonsafety-related EDNS and EDAS/EDSS may remain available via battery operation; the primary loads for these systems include the module control system (MCS) and the MPS. Loss of the EDNS and/or EDAS/EDSS batteries with the loss of normal AC power is considered as described in Section 7.1.3. A loss of AC power results in a pressurization of the secondary system and overheating of the RCS. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition. Table 7-42 lists the relevant acceptance criteria, SAF, and LOP scenarios.

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The electrical system design may vary by NPM design. Review of the electrical system is performed to determine the impact on plant equipment from the loss of power. In general, the following typically occur from the loss of power (AC or DC or both):

- turbine trip occurs
- the feedwater pumps and CVCS pumps stop
- the PZR heaters turn off
- control rods begin to drop due to loss of power to control rod drive mechanisms (CRDMs)
- the MPS actuates reactor trip, DHRS, and various system isolations within 60 seconds after event initiation (if not already actuated during that time), due to loss of AC power to the EDAS/EDSS battery chargers
- the MPS actuates ECCS within 24 hours after event initiation, due to loss of AC power to the EDAS/EDSS battery chargers

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Of the various scenarios, the limiting one is typically the scenario that does not result in an immediate reactor trip or full control rod insertion at event initiation. Typically this scenario is the loss of the ELVS at event initiation, with EDNS and EDAS/EDSS available. Cases considering the loss of EDSS battery backup coincident with the initiating event are considered as described in Section 7.1.3. Consequently, the limiting pressure responses typically occur when the event is initiated from full power conditions, reactor trip is delayed until MPS actuation, and the initial conditions are biased in the conservative directions. Sensitivity studies on initial primary temperature and primary/secondary pressures are performed to identify the conditions that maximize peak primary and secondary pressures. Additional sensitivity studies are performed on other parameters, as ~~necessary~~identified in Table 7-44, to identify the case(s) with the potentially limiting peak primary and secondary pressures.

Table 7-42 Acceptance criteria, single active failure, loss of power scenarios - loss of normal AC power

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Primary and secondary pressures are challenged during this overheating event.
No single failure	The challenging cases for primary pressure typically occur when all equipment is operational. Since feedwater is lost at transient initiation, peak secondary pressures are insensitive to the single failure of an FWIV to isolate.
Loss of AC power at transient initiation	Initiating event.

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Table 7-44 Initial conditions, biases, and conservatisms - loss of normal AC power (Continued)

Parameter	Bias / Conservatism	Basis
Charging Letdown	Not credited. Enabled.	{{
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Enabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)
2. Loss of normal AC power initiating event results in a loss of system function by loss of power to the system thereby making the system control not relevant to the event.

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies are performed to identify the highest primary and secondary side pressures and presented in Table 7-45. The

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results of the sensitivity studies indicate that the lowest-setpoint RSV operates to mitigate peak primary side pressurization {{

}}2(a),(c) The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-45 Representative sensitivity studies - loss of normal AC power~~Not Used~~

{{

}}2(a),(c)

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Table 7-47 Acceptance criteria - loss of normal feedwater flow (Continued)

Acceptance Criteria	Discussion
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.10.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-48 are considered in identifying a bounding transient simulation for primary and steam generator pressure.

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased to the high condition.	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal.	}}2(a),(c)

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Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
RSV lift setpoint	Biased to the high condition.	{{
SG tube plugging	Biased to the low condition.	
Feedwater flow decrease	~~Varied~~Spectrum.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Disabled. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 N/A.	}}2(a),(c)

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Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis
CNV Pressure Control CNV evacuation system	Disabled.	{{ }}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest primary and secondary pressures~~, varying the magnitude of the feedwater flow rate decrease (other parameters are biased as indicated in Table 7-48)~~. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative results of a sensitivity study for a spectrum of feedwater flow rate decreases are presented in Table 7-49. The results of the sensitivity study indicates that peak primary pressure occurs during a total loss of feedwater. Because the pressurization associated with the total loss of feedwater is sufficient to actuate the lowest-setpoint RSV, {{ }}2(a),(c) well below the acceptance criterion of 2310 for this representative NPM). With respect to maximum secondary side SG pressure, maximum SG pressure occurs during a partial loss of feedwater, as the mismatch between primary heat production and secondary heat sink causes the RCS temperature to increase, and eventually actuates the RTS on high RCS riser temperature. The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-49 Representative sensitivity studies - loss of normal feedwater flow~~Not Used~~

{{

}}2(a),(c)

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Table 7-49 Representative sensitivity studies - loss of normal feedwater flow~~Not Used~~

{{

}}2(a),(c)

7.2.11 Inadvertent Decay Heat Removal System Actuation

The methodology used to simulate a postulated inadvertent DHRS actuation for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below. This event is unique to the NPM designs.

7.2.11.1 General Event Description

Inadvertent actuation of the DHRS may result from either an unexpected DHRS valve actuation or a spurious DHRS actuation signal. Due to the DHRS design and configuration, several scenarios exist for consideration. The specific scenarios considered are discussed below. The sequence of events and the MPS signals that are reached first depend on the scenario. Reactor trip and transition to stable DHRS flow terminates the transients with the NPM in a safe, stable condition. The relevant acceptance criteria, SAF, and LOP scenarios are listed in Table 7-50.

Scenario 1: The unexpected opening of a single DHRS valve can occur at full power or reduced power conditions. At low power conditions, a portion of the DHRS liquid inventory drains into the feedwater line, which momentarily increases feedwater flow and causes overcooling. This overcooling event is not considered further because it is bounded by other, more limiting overcooling events (i.e., increase in feedwater flow). The most challenging conditions for this heatup scenario occur at full power with initial conditions biased in the conservative directions. Since feedwater flow tends to increase in response to the reduced steam enthalpy and turbine load, limiting the feedwater response maximizes the heatup.

Scenario 2: An inadvertent actuation signal isolates one steam generator, and initiates one DHRS train. This scenario is typically bounded by Scenario 3 discussed below.

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Scenario 3: An inadvertent actuation signal isolates both steam generators, and initiates both DHRS trains. This scenario represents a complete loss of normal heat removal from the RCS. The limiting conditions for this heatup scenario occur at full power with initial conditions biased in the conservative directions.

Scenario 4: An inadvertent isolation of one steam generator. The valves associated with one SG are closed, but the associated DHRS train is not actuated. The heatup caused by the isolation of one SG causes an increase in primary pressure that results in reactor trip and RSV opening. The actuation of DHRS occurs later in the transient after MPS signals are reached. Secondary pressure peaks after DHRS actuation.

Scenario 5: An inadvertent isolation of both steam generators. The valves associated with both SGs are closed, but neither DHRS train is actuated. The response is similar to Scenario 4, except the increase in primary pressure is more rapid and results in earlier reactor trip and RSV opening. The actuation of DHRS occurs later in the transient after MPS signals are reached. Secondary pressure peaks after DHRS actuation.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Each scenario is considered to determine the limiting cases for the acceptance criteria. Sensitivity studies on initial primary and secondary conditions are performed as identified in Table 7-52~~needed~~ to identify the conditions that maximize peak system pressures.

Table 7-50 Acceptance criteria, single active failure, loss of power scenarios - inadvertent decay heat removal system actuation

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	Challenged during the heatup scenarios of this event.
No single failure	The challenging primary pressurization cases typically occur when all equipment is operational.
Failure of one FWIV to close	Potentially challenging for secondary pressure, but typically bounded by other heatup events.
No loss of power	Maximizes system pressures.

7.2.11.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-51.

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Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation (Continued)

Parameter	Bias / Conservatism	Basis
PZR heaters	Nominal.	{{
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Disabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies performed to identify the highest primary and secondary side pressure, varying primary conditions, steam generator heat transfer and pressure, are presented in Table 7-53. Unless otherwise noted, cases use BOC kinetics. The results of the sensitivity studies indicate that peak primary pressure occurs when both trains of DHRS inadvertently actuate (i.e., total loss of normal heat sink). The lowest-setpoint RSV operates to mitigate peak pressurization well below the acceptance criterion of 2310 for this representative NPM). {{

}}2(a),(c) The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-53 Representative sensitivity studies - inadvertent decay heat removal system actuation~~Not Used~~

{{

}}2(a),(c)

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Table 7-53 Representative sensitivity studies - inadvertent decay heat removal system actuation~~Not Used~~ (Continued)

{{

}}2(a),(c)

7.2.12 Feedwater System Pipe Break Inside or Outside Containment

The methodology used to simulate a postulated feedwater system pipe break for an NPM, and an evaluation of the acceptance criteria listed in Table 7-4, are presented below. Since both split breaks (relatively higher event frequency) and double-ended guillotine breaks (relatively lower event frequency) are analyzed, the more restrictive AOO criteria for system pressures, critical heat flux ratio, and fuel centerline melt applicable to breaks with higher event frequency are used in the evaluation.

7.2.12.1 General Event Description

A feedwater line break can occur inside or outside of containment, and can range in size from a small split crack to a double ended rupture. A large feedwater line break inside containment results in a loss of containment vacuum and a high containment pressure MPS signal that actuates a reactor trip, isolates the secondary system and CVCS, and opens the DHRS valves. The steam generator, DHRS piping, and DHRS condenser on the affected side drain through the break. The non-affected steam generator system and DHRS loop provide cooling to the RCS via heat transfer to the reactor pool. The response of smaller feedwater line breaks inside containment is similar except that other MPS setpoints, such as high PZR pressure, may be reached before high containment pressure.

A feedwater line break outside containment causes a loss of feedwater flow to the steam generators and a heatup of the RCS. Larger breaks result in rapid heatup events that pressurize the RCS beyond the high PZR pressure analytical limit. Smaller breaks cause a more gradual heatup, loss of secondary pressure, and

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Table 7-55 Acceptance criteria - feedwater line break (Continued)

Acceptance Criteria	Discussion
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation to a more serious accident or consequential loss of functionality	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.12.3 Biases, Conservatisms, and Sensitivity Studies

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The biases and conservatisms presented in Table 7-56, and a spectrum of break locations and sizes, are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	}}2(a),(c)

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Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased to the high condition.	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Nominal.	
Break size / location	~~Varied~~Spectrum.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	}}2(a),(c)

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Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break (Continued)

Parameter	Bias / Conservatism	Basis {{
Steam Pressure Control Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies performed to identify the highest primary and secondary side pressures, varying primary conditions, break size and location, and steam generator heat transfer and pressure, are presented in Table 7-57. Unless otherwise noted, breaks are located at the bioshield, RCS flows are initialized at the minimum value, and losses of normal AC power occur at event initiation. The results of the sensitivity studies indicate that peak primary pressures occur during a break coincident with a loss of AC power; under these conditions the lowest-setpoint RSV operates to mitigate peak pressurization well below the acceptance criterion of 2310 for this representative NPM. Initial condition biasing contributions are secondary compared to the pressure response to the total loss of heat sink and multiple cases have similar peak pressures. With

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respect to maximum secondary side SG pressure, maximum SG pressure occurs during a 10 percent split break just outside of containment coincident with a loss of AC power when initial SG pressure is biased to the high condition. The peak secondary side pressures remain well below the design pressure (2100 for this representative NPM).

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-57 Representative sensitivity studies - feedwater line break~~Not Used~~

{{

}}2(a),(c)

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

The methodology used to simulate a postulated uncontrolled CRA bank withdrawal from subcritical or low power startup conditions for an NPM, and an evaluation of the acceptance criteria listed for an AOO in Table 7-4, are presented below.

The range of initial power levels associated with low power startup conditions for an NPM is based on the low setting for the high power signal (below 15 percent RTP for the example in Table 7-3). When core power reaches the low setting level, a hold point is established to alter the high power setting. Thus, low power startup conditions exist until reactor power reaches the low setting level.

7.2.13.1 General Event Description and Methodology

The limiting event consequences to an uncontrolled CRA bank withdrawal from subcritical or low power startup conditions typically (for most PWR designs) occur for cases with very low initial power levels (~1 Watt). The primary reason for this behavior is the flux rate signals associated with the source range and intermediate range are typically either not safety related or not of sufficient quantity to adequately address single failures. An NPM, however, incorporates a safety related signal for each of these channels from each core quadrant into the MPS. Consequently, the limiting event consequences for an NPM typically occur for cases with higher initial power levels.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

A spectrum of constant reactivity insertion rates is evaluated, {{ }}2(a),(c). These reactivity insertion rates encompass the credible range resulting from a single control bank withdrawal. If necessary, this range is supplemented to include the reactivity insertion rates associated with an inadvertent decrease in boron concentration event (Section 7.2.16).

Two event scenarios with different protection schemes are evaluated to determine which scenario produces the limiting event consequences. The first scenario arises when the high count rate signal is available because the intermediate range channel does not have an established signal. In this instance, the high power rate signal is not active (below 15 percent RTP in the Table 7-3 example), so core protection is provided by the high count rate signal and the startup rate (source range) signal. {{

}}2(a),(c)

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The second scenario arises when the high count rate signal is not available because the intermediate range channel has an established signal. In this instance, the high power rate signal is not active (below 15 percent RTP in the Table 7-3 example), thus core protection is provided by the high power signal and the startup rate (intermediate range) signal. The event scenario with the highest core power typically corresponds to the initial power level and reactivity insertion rate that cause the high power signal (low setting) and the startup rate (intermediate range) signal to occur at nearly the same time. This scenario is typically limiting because it represents the maximum rate of power change at the maximum core power. If the initial power is increased, the reactor trips on the high power signal but at a slower rate of power increase. Similarly, if the reactivity insertion rate is increased, the reactor trips on the startup rate but at a lower core power.

Before initiating an approach to critical, the reactor coolant is heated to the minimum temperature for criticality. The heating of the reactor coolant is performed by the Module Heatup System (MHS) via the CVCS.

At least one feedwater pump is operating at these RCS temperatures. Since feedwater flow is provided to both SGs, it may continue to provide decay heat removal following an uncontrolled CRA bank withdrawal from subcritical or low power startup conditions. If normal feedwater flow is not available, then depending on the point at which the event occurs in the startup process, the flooded containment or the DHRS provides decay heat removal. The peak power and duration of the power spike are not sufficient to cause a significant temperature or pressure increase. Hence, the maximum power and minimum CHFR occur shortly after reactor trip while the RCS pressure and MS pressure do not challenge the relevant acceptance criteria.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed as identified in Table 7-60~~on a variety of parameters, as necessary,~~ to identify the case(s) with a potentially limiting MCHFR or fuel centerline temperature. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

Table 7-58 lists the relevant acceptance criteria, SAF, and LOP scenarios.

Table 7-58 Acceptance criteria, single active failure, loss of power scenarios - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR and maximum fuel centerline temperature.	MCHFR and maximum fuel centerline temperature are challenged during this reactivity anomaly event.

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7.2.13.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-60 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-60 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Nominal.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Nominal.	
Initial PZR level	Nominal.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Biased to BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	}}2(a),(c)

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Table 7-60 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions (Continued)

Parameter	Bias / Conservatism	Basis
		{{
Reactivity insertion rate	~~Varied~~Spectrum.	
RCS Temperature Control Automatic rod control Boron concentration	 N/A. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Enabled. Enabled.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 N/A. Enabled.	
Feedwater and Turbine Load Control feedwater pump speed	 N/A.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, ~~a~~ sensitivity ~~study is generally~~studies are performed to identify cases for lowest MCHFR and highest fuel centerline temperature for this reactivity event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

~~Sensitivity study results typically~~For example, representative sensitivity studies for the initial core power and reactivity insertion rate are presented in Table 7-61. The results demonstrate the lack of challenging MCHFR values predicted for this event by NRELAP5, which is further supported by the values predicted with the approved subchannel methodology. {{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-61 Representative sensitivity studies - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions~~Not Used~~

{{

}}2(a),(c)

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Table 7-61 Representative sensitivity studies - uncontrolled control rod bank withdrawal from subcritical or low power startup conditions~~Not Used~~ (Continued)

{{

}}2(a),(c)

7.2.14 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

The methodology used to simulate a postulated uncontrolled control rod assembly (CRA) bank withdrawal at power for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.14.1 General Event Description and Methodology

As stated in Section 7.2.13, low power startup conditions exist for an NPM until reactor power reaches the low setting level (15 percent RTP for the example in Table 7-3). Accordingly, the uncontrolled CRA bank withdrawal at power event extends from the low setting level to HFP.

The withdrawal of the control bank causes a reactivity insertion that increases reactor power and leads to a rise in coolant temperature, pressurizer level, and RCS pressure. Reactivity feedback from the rising fuel temperature partially counteracts the reactivity insertion, slowing the power increase, which continues until the system trips on high power, high power rate, high pressurizer pressure, or high RCS temperatures. The maximum power and minimum MCHFR occur just after the resulting scram, while the peak primary pressure occurs some time later, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established by the end of the transient.

Following reactor trip and subsequent turbine trip, the turbine bypass to the condenser opens to control the RCS temperature. However, the actions of the turbine bypass system are not credited, so as to minimize heat removal by the secondary side. Although turbine load is an input to the feedwater controller, no changes are made to this controller because the RCS responses are not sufficient to affect feedwater control.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The limiting MCHFR typically occurs for a reactivity insertion rate that results in reactor trip on core power, pressurizer pressure, and RCS riser temperature signals at approximately the same time. These conditions typically arise for events initiated with lower reactivity insertion rates because higher reactivity insertion rates cause the MPS to trip much earlier on high power rate. The earlier reactor trip reduces the energy added to the reactor coolant, thereby producing a higher MCHFR. The range of reactivity insertion rates considered is sufficient to identify the point of transition {{ }}2(a),(c)
Sensitivity studies are performed on a variety of parameters, as identified in Table 7-64~~necessary~~, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

The maximum fuel centerline temperature typically occurs when core power exceeds its analytical limit. This condition typically arises for events initiated from full power with the highest reactivity insertion rate as determined from the resulting bank worth and control rod step speed.

A spectrum of constant reactivity insertion rates is evaluated. These reactivity insertion rates encompass the credible range resulting from a single control bank withdrawal. If necessary, this range is supplemented to cover the reactivity insertion rates associated with an inadvertent decrease in boron concentration event (Section 7.2.16).

Table 7-62 lists the relevant acceptance criteria, SAF, and LOP scenarios.

Table 7-62 Acceptance criteria, single active failure, loss of power scenarios - uncontrolled control rod bank withdrawal at power

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR and maximum fuel centerline temperature	MCHFR and maximum fuel centerline temperature are challenged during this reactivity anomaly event.
No single failure	The challenging cases typically occur when all equipment is operational.
No loss of power	The challenging cases typically occur when AC power is available for the event duration.

7.2.14.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-63.

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Table 7-63 Acceptance criteria - uncontrolled control rod bank withdrawal at power

Acceptance Criteria	Discussion
Primary pressure	This criterion is not an acceptance criterion listed in Section 15.4.2 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for primary pressure compared to other AOOs.
Secondary pressure	This criterion is not an acceptance criterion listed in Section 15.4.2 of the SRP (Reference 15). The analysis shows an NPM does not introduce more challenging conditions for secondary pressure compared to other AOOs.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.14.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-64 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature.

Audit Question A-NonLOCA.LTR-59
RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied. RTP biased upwards to account for measurement uncertainty.	}}2(a),(c)

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Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial RCS average temperature	Varied. Biased to the high condition.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the low condition.	
MTC	Biased to BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Reactivity insertion rate	Spectrum.~~Varied~~ Maximum	}}2(a),(c)

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Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power (Continued)

Parameter	Bias / Conservatism	Basis
CNV Pressure Control CNV evacuation system	Enabled.	{{ }}2(a),(c)

1. {{ }}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, ~~a~~ sensitivity ~~study is generally~~studies are performed to identify cases for lowest MCHFR for this reactivity event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies for initial core power, reactivity insertion rate, pressurizer pressure, pressurizer level, reactor coolant average temperature, pressurizer spray flow, pressurizer heater status, letdown status, and loss of power are presented in Table 7-65. The results demonstrate the two most limiting MCHFRs occur at initial power levels of 75 percent RTP and above, with AC power available for the event duration, the RCS average temperature biased low, and pressurizer spray modeled to delay reactor trip on high pressurizer pressure.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-65 Representative sensitivity studies - uncontrolled control rod bank withdrawal at power~~Not Used~~

{{

}}2(a),(c)

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Table 7-65 Representative sensitivity studies - uncontrolled control rod bank withdrawal at power~~Not Used~~ (Continued)

{{

}}2(a),(c)

7.2.15 Control Rod Misoperation

The methodology used to simulate a postulated control rod misoperation for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.15.1 General Event Description and Methodology

The rod control system is used to move (insert or withdraw) the control rod assemblies (CRAs) in response to an operator action or an automatic control. Since these transients are initiated by a malfunction in the rod control system, a variety of reactivity related conditions can result. Specific reactivity conditions for an NPM include: 1) withdrawing a single CRA; 2) dropping one or more CRAs; or, 3) leaving one or more CRAs behind when inserting or withdrawing a control bank. The consequences for each of these reactivity conditions are discussed below.

Table 7-66 lists the relevant acceptance criteria, single active failure, and loss of power scenarios.

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Withdrawal of a Single CRA

The withdrawal of a single CRA causes a reactivity insertion that increases reactor power and leads to a rise in coolant temperature, pressurizer level, and RCS pressure. Feedback from the rising fuel temperature is not sufficient to counteract the reactivity insertion, so the power increases until the system trips on high power, high power rate, high pressurizer pressure, or high RCS temperatures. The maximum power and minimum MCHFR occur just after the resulting scram, while the peak primary pressure occurs some time later, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established at the end of the transient.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The limiting MCHFR typically occurs for a reactivity insertion rate that results in reactor trip on core power, pressurizer pressure, and RCS temperature signals at approximately the same time. These conditions arise for events initiated from partial power with lower reactivity insertion rates because higher reactivity insertion rates cause the MPS to trip much earlier on high power rate. The earlier reactor trip reduces the energy added to the reactor coolant, thereby producing a higher MCHFR. The asymmetry associated with the core power response causes the ex-core detectors to respond differently for each quadrant. Consequently, the range of reactivity insertion rates considered is sufficient to identify the point of transition {{ }}2(a),(c) ~~to the high power rate signal~~ (with power related trips using the lowest reading ex-core detector based on the minimum after to before event initiation ratio of the radial peaking factors for the outer row of fuel assemblies). Sensitivity studies are performed on a variety of parameters, as ~~necessary~~indicated in Table 7-68, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

In contrast to the uncontrolled CRA bank withdrawal at power event (Section 7.2.14), the highest reactivity insertion rate as determined from the resulting rod worth and control rod step speed is significantly lower for the withdrawal of a single CRA event.

Audit Question A-NonLOCA.LTR-5

Note that instead of using case-specific values for rod worth and parameters associated with the asymmetry, more conservative bounding input values can be used. The use of bounding input values provides a conservative analysis simplification for the withdrawal of a single CRA event. In addition, the bounding input values can then be used to compare this event to the dropping of one or more CRAs event described below.

Dropping One or More CRAs

Based on the minimum worth at any time during the cycle for a given core power, i.e., with the control bank positioned at the PDIL, dropping a single CRA causes a reactivity insertion that decreases reactor power. Feedback from the decreasing fuel temperature and the actions of the rod control system to restore power are

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generally not sufficient to counteract the reactivity insertion, so the power decreases until the system trips on high power rate. For event scenarios without a return to power, the maximum core power, peak primary pressure, and MCHFR occur at event initiation. The peak secondary system pressure occurs some time after the scram, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established at the end of the transient.

The potential for a return to power exists only for events initiated from less than RTP because the reduced worth of the dropped rod gives the rod control system time to act. The corresponding MCHFR for a dropped rod event with a return to power is typically greater than the MCHFR for events initiated from HFP. Hence, the limiting MCHFR cases typically occur at HFP conditions.

Following reactor trip and subsequent turbine trip, the turbine bypass to the condenser opens to control the RCS temperature. However, the actions of the turbine bypass system are not credited, so as to minimize heat removal by the secondary side. Although turbine load is an input to the feedwater controller, no changes are made to this controller because the RCS responses are not sufficient to affect feedwater control.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

The asymmetry associated with the core power response causes the ex-core detectors to respond differently for each quadrant. The power input to the high power rate signal uses the highest reading ex-core detector, multiplying the core average power by the maximum after drop to before drop ratio of the radial peaking factors for the outer row of fuel assemblies. Sensitivity studies are performed, as identified in Table 7-70 ~~on a variety of parameters, as necessary~~, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

The maximum fuel centerline temperature typically occurs at event initiation for those event scenarios with an immediate reactor trip. If the event scenario has a return to power, the maximum fuel centerline temperature is typically bounded by the fuel centerline temperature at HFP because the associated power peak is less than full power.

Audit Question A-NonLOCA.LTR-5

As an alternative to performing a system transient analysis, the MCHFR and linear heat generation rate of the dropped rod event can be confirmed to be bounded by other events. Most rod drops result in a reactor trip on high power rate because of the immediate decrease in power from the dropped rod. Figure 7-3 shows the decrease in power from the dropped rod {{ }}$^{2(a),(c)}$ as a function of rod worth for a variety of initial power levels. The figure is based on a representative NPM core design and also overlays a representative high power rate trip. {{

}}$^{2(a),(c)}$

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-68 Initial conditions, biases, and conservatisms - control rod misoperation, single control rod assembly withdrawal

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the low condition.	
MTC	Biased to BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Reactivity insertion rate	Spectrum.~~Varied.~~	
RCS Temperature Control Automatic rod control	N/A.	}}2(a),(c)

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Table 7-68 Initial conditions, biases, and conservatisms - control rod misoperation, single control rod assembly withdrawal (Continued)

Parameter	Bias / Conservatism	Basis
		{{
Boron concentration	Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied. Varied.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Enabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, ~~a~~ sensitivity ~~study is generally~~studies are performed to identify cases for lowest MCHFR for this reactivity event.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies for initial core power, reactivity insertion rate, pressurizer pressure, pressurizer level, reactor coolant average temperature, pressurizer spray flow, pressurizer heater status, letdown status, and loss of power are presented in Table 7-69 for the withdrawal of a single CRA event. The results demonstrate the MCHFR occurs at an initial power level of 75 percent RTP, with AC power available for the event duration, the RCS average temperature biased low, the pressurizer pressure biased low, the pressurizer level biased low, and spray modeled to delay reactor trip on high pressurizer pressure.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-69 Representative sensitivity studies - control rod misoperation, single control rod assembly withdrawal~~Not Used~~

{{

}}2(a),(c)

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Table 7-69 Representative sensitivity studies - control rod misoperation, single control rod assembly withdrawal~~Not Used~~ (Continued)

{{

}}2(a),(c)

Dropping One or More CRAs

The biases and conservatisms presented in Table 7-70 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature for the dropped CRA(s) event. If the alternative bounding approach is used, where cases that result in early trip are bounded by steady-state condtions and cases that do not result in early trip are bounded by the single rod withdrawal, the biases and conservatisms in Table 7-70 associated with system transient analysis are not applicable. However, the alternative bounding approach considers combinations of initial core power, dropped CRA worth, core time-in-life, axial offset, flow, and temperature to screen the cases into groups and compare the non-tripped cases to single rod withdrawal analysis limits.

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Table 7-70 Initial conditions, biases, and conservatisms - control rod misoperation, dropped control rod assemblies (Continued)

Parameter	Bias / Conservatism	Basis {{
Boron concentration	Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Nominal. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Enabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, ~~a~~ sensitivity ~~study is generally~~studies are performed to identify cases for lowest MCHFR for this reactivity event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies for the initial core power, dropped CRA worth, and core time-in-life are presented in Table 7-71 for the dropped CRA(s) event. The results demonstrate the MCHFR occurs at full power conditions.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-71 Representative sensitivity studies - control rod misoperation, dropped control rod assemblies~~Not Used~~

{{

}}2(a),(c)

7.2.16 Inadvertent Decrease in Boron Concentration

The methodology used to simulate an inadvertent decrease in boron concentration for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.16.1 General Event Description and Methodology

The boric acid blend system incorporated into the NuScale plant design permits the operator to control the boron concentration of the reactor coolant via the charging fluid chemistry. While the NuScale plant design incorporates both automatic and manual controls, strict administrative procedures govern the process for adjusting the boron concentration of the reactor coolant. These administrative procedures establish limits on the rate and duration of the dilution.

The primary means of causing an inadvertent decrease in boron concentration is failure of the blend system, either by controller or mechanical failure, or operator error. The event is terminated by isolating the source for the diluted water, i.e., by closing the demineralized water system (DWS) isolation valves.

For Mode 1 plant operating conditions, the perfect mixing model and the wave front model are both evaluated. The perfect mixing model is evaluated for Mode 1 operating conditions because it provides a slower reactivity insertion rate, delaying detection, potentially allowing further loss of shutdown margin. The wave front model is physically conservative because it assumes the maximum amount of reactivity as the diluted slug of water sweeps through the core. This model does not assume any axial blending to ensure that this reactivity insertion rate is maximized. For all other operating modes where boron dilution is allowed and limited mixing exists, a wave front model is used. These mixing models are

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

{{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-75 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot full power~~Not Used~~

Mode 1 (HFP)	1 CVCS Pump	2 CVCS Pumps
Dilution Flow Rate (gpm)	25	50
Initial boron concentration (ppm)	1400	1400
Final boron concentration (ppm)	1196	1196
Perfect mixing reactivity insertion rate (pcm/s)	0.50	1.00
Reactor trip / DWS isolation time (s)	161	78
DWS isolation valves closed (s)	166	83
Shutdown margin at time of isolation (pcm)	Greater than 1944	Greater than 1912
Shutdown margin lost assuming no isolation (s)	4092 (68.2 min)	2046 (34.1 min)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-76 Representative results - inadvertent decrease in boron concentration in Mode 1 at 25 percent rated thermal power~~Not Used~~

Mode 1 (25 percent RTP)	1 CVCS Pump
Dilution flow rate (gpm)	25
Initial boron concentration (ppm)	1800
Final boron concentration (ppm)	1596
Perfect mixing reactivity insertion rate (pcm/s)	0.65
Wave front reactivity insertion rate (pcm/s)	19.1
Wave front initial reactivity step (pcm)	91
Shutdown margin lost assuming no isolation (s)	3120 (52 min)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-77 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot zero power~~Not Used~~

Mode 1 (HZP, 1 MWt)	1 CVCS Pump
Dilution Flow Rate (gpm)	25
Initial boron concentration (ppm)	1800
Final boron concentration (ppm)	1596
Initial reactivity step (pcm)	685
Wave front reactivity insertion rate (pcm/s)	17.3
Duration of reactivity insertion rate for each wave (s)	39.5
Initial wave front reaches core inlet (s)	900
Initial wave front reaches core exit (s)	939.5
Reactor trip / DWS isolation time(1) (s) (wave front at core exit + UCRWS trip time)	Approximately 2028
DWS isolation valves closed (s)	Approximately 2033

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Table 7-77 Representative results - inadvertent decrease in boron concentration in Mode 1 at hot zero power~~Not Used~~ (Continued)

Mode 1 (HZP, 1 MWt)	1 CVCS Pump
Shutdown margin at time of isolation (pcm)	Greater than 697
Shutdown margin lost assuming no isolation (s) (critical wave at core inlet)	4200 (70 min)

1. Conservatively estimated to occur upon arrival of the subsequent wave

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-78 Representative results - inadvertent decrease in boron concentration in Mode 2~~Not Used~~

Mode 2 (Coolant Flow ≥ 1.7 ft^3/s)	1 CVCS Pump
Dilution Flow Rate (gpm)	25
Initial boron concentration (ppm)	785.5
Final boron concentration (ppm)	600
Initial reactivity step (pcm)	329
Wave front reactivity insertion rate (pcm/s)	8.32
Duration of reactivity insertion rate for each wave (s)	39.5
Reactor trip/DWS isolation wave at core inlet (s)	5280 (88 min)
DWS isolation valves closed (s)	5475
Shutdown margin at time of isolation (pcm)	Greater than 517
Shutdown margin lost assuming no isolation (s) (critical wave at core inlet)	7440 (124 min)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-79 Representative results - inadvertent decrease in boron concentration in Mode 3~~Not Used~~

Mode 3 (Coolant Flow = 1.7 ft^3/s)	1 CVCS Pump
Dilution flow rate (gpm)	25
Initial boron concentration (ppm)	813
Final boron concentration (ppm)	650
Initial reactivity step (pcm)	378
Wave front reactivity insertion rate (pcm/s)	9.55
Duration of reactivity insertion rate for each wave (s)	39.6
Reactor trip / DWS isolation wave at core inlet (s)	4200 (70 min)
DWS isolation valves closed (s)	4395
Shutdown margin at time of isolation (pcm)	Greater than 613
Shutdown margin lost assuming no isolation (s) (critical wave at core inlet)	6360 (106 min)

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

The methodology used to simulate a postulated CVCS malfunction that increases RCS inventory for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

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7.2.17.1 General Event Description

The NuScale reactor does not have a pumped ECCS, therefore, the unplanned increase in RCS inventory event is caused by a malfunction of the CVCS makeup pumps or pressurizer level control system. If borated water at the same concentration of the primary system is added to the RCS, the addition of large amounts of water to the primary system typically generates a reactor trip on high pressurizer (PZR) water level or high PZR pressure. Table 7-80 lists the relevant acceptance criteria, SAF, and LOP scenarios.

The malfunction is assumed to isolate letdown and actuate both makeup pumps (each flowing at their maximum capacity), causing an unplanned increase in RCS inventory. The limiting pressure response occurs when the event is initiated from full power conditions, and the initial conditions are biased in the conservative directions. The increase in RCS inventory event is typically terminated by CVCS isolation on high PZR level. (The CVCS containment isolation valves are dual safety related valves.)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies to identify the challenging conditions are performed, as ~~necessary~~identified in Table 7-82, to identify the case(s) with the potentially limiting system pressures. Cases that include the effects of CVCS recirculation, which increases the flow into the RCS, are also analyzed.

Table 7-80 Acceptance criteria, single active failure, loss of power scenarios - reactor coolant system inventory increase

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
Primary pressure, secondary pressure	System pressures are challenged during this mass addition event.
No single failure	The CVCS is isolated via dual safety-related isolation valves. If one of the isolation valves were to fail, the other CVCS isolation valve would provide system isolation.
No loss of power	Continued operation of the CVCS typically maximizes system pressures.

7.2.17.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-81.

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Table 7-82 Initial conditions, biases, and conservatisms - reactor coolant system inventory increase (Continued)

Parameter	Bias / Conservatism	Basis
Turbine bypass valves	Disabled.	{{
Feedwater and Turbine Load Control Feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)
2. These inputs, in conjunction with least negative Doppler temperature coefficient, are selected to maximize the power response (if any) induced by the addition of colder CVCS water. However, since this event is driven by mass addition, reactivity effects prior to RTS actuation (if any) are small when compared to the pressurization associated with the increase in primary inventory.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event, including cases with the highest pressure responses for this inventory increase event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, sensitivity studies performed to identify the highest primary and secondary side pressures, varying primary conditions, makeup temperature, and spray availability are presented in Table 7-83. The results indicate that peak primary pressure occurs when initial primary temperature and pressure are biased to the low condition, and makeup temperature is biased to the high condition as the RTS actuates on high pressurizer pressure and the lowest-setpoint RSV operates to mitigate peak pressurization well below the acceptance criterion of 2310 for this representative NPM. With respect to maximum secondary side SG pressure, maximum SG pressure occurs when spray is used to delay the RTS actuation until high pressurizer level is reached.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-83 Representative sensitivity studies - reactor coolant system inventory increase~~Not Used~~

{{

}}2(a),(c)

7.2.18 Failure of Small Lines Outside Containment

The methodology used to simulate a postulated failure of a small line connected to the primary coolant system outside of containment for an NPM, and an evaluation of the acceptance criteria for an infrequent event listed in Table 7-4, are presented below.

A postulated break in a small line carrying primary coolant is typically only evaluated for radiological consequences. Neither the plant design nor the use of natural circulation flow for an NPM introduces a more challenging condition for other acceptance criteria. Evaluation of postulated small line breaks within the context of a

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Table 7-86 Initial conditions, biases, and conservatisms - breaks in small lines carrying primary coolant outside containment (Continued)

Parameter	Bias / Conservatism	Basis(1)
		{{
Letdown	Disabled.	
Steam Pressure Control Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{

2. }}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies for break location and size to determine impact on mass release and iodine spiking time are presented in Table 7-87 and Table 7-88, respectively. The Table 7-87 results indicate the mass release is maximized with a break in the letdown line of 100 percent area, AC power lost at event initiation, RCS average temperature biased high, and a break in the makeup line of 100 percent area coincident with reactor trip. The Table 7-88 results indicate the iodine spiking duration (elapsed time from reactor trip to CVCS isolation) is maximized with a break in the makeup line, AC power lost at event initiation, RCS average temperature biased high, and no break in the letdown line coincident with reactor trip.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

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~~Not Used~~

{{

}}2(a),(c)

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-88 Representative break, time in life, power, flow, and temperature sensitivity study for iodine spiking time - breaks in small lines carrying primary coolant outside containment~~Not Used~~

{{

}}2(a),(c)

7.2.19 Steam Generator Tube Failure

The methodology used to simulate a postulated failure of a steam generator tube for an NPM, and an evaluation of the acceptance criteria for an accident listed in Table 7-4, are presented below.

A postulated failure of a steam generator tube is typically only evaluated for radiological consequences. Neither the steam generator design nor the use of natural circulation flow for an NPM introduce a more challenging condition for other acceptance criteria.

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Table 7-91 Initial conditions, biases, and conservatisms - steam generator tube failure (Continued)

Parameter	Bias / Conservatism	Basis (1) {{
PZR Pressure Control PZR spray PZR heaters	Disabled. Nominal.	
PZR Level Control Charging Letdown	Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

{{ 1.

2. }}2(a),(c)

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Sensitivity studies are performed ~~as needed~~ to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event.

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RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

For example, representative sensitivity studies for break characteristics, core time-in-life, SG pressure, SG tube plugging, loss of power, single failure, feedwater temperature and reactor coolant temperature for a steam generator tube failure with respect to mass release and iodine spiking duration are presented in Table 7-92. The results indicate the mass release and spiking time are maximized with a double-ended guillotine break at the top of the SG tube, AC power available for the event duration, and a single active failure of the primary MSIV on the affected SG to close.

RAI 10297 Question NonLOCA.LTR-31, 32, 46, 56, 65

Table 7-92 Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure~~Not Used~~

{{

}}2(a),(c)

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Table 7-92 Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure~~Not Used~~ (Continued)

{{

}}2(a),(c)

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7.2.1.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-7 are considered in identifying a bounding transient simulation for MCHFR.53

Audit Question A-NonLOCA.LTR-28
RAI Questions A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.~~Biased to the high condition~~	
Initial PZR level	Varied.~~Biased to the high condition.~~	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Nominal.	
Moderator Temperature Coefficient (MTC)	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	}}2(a),(c)

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Audit Questions A-NonLOCA.LTR-28, A-NonLOCA.LTR-65
RAI Questions, A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.~~Biased to the high condition.~~	
Initial PZR level	Varied~~Biased to the high condition.~~	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Nominal.	
MTC	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the low condition.	

}}2(a),(c)

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Audit Question A-NonLOCA.LTR-28
RAI Questions A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.~~Biased to the high condition.~~	
Initial PZR level	Varied.~~Biased to the high condition.~~	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Biased to the high condition.	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Steam flow increase	Varied	}}2(a),(c)

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Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow (Continued)

Parameter	Bias / Conservatism	Basis
RCS Temperature Control		{{
Automatic rod control	Varied.	
Boron concentration	Not credited.	
PZR Pressure Control		
PZR spray	Varied.~~Disabled.~~	
PZR heaters	Nominal.	
PZR Level Control		
Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control		
Turbine throttle valves	N/A.	
Turbine bypass valves	N/A.	}}2(a),(c)

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Table 7-24 Initial conditions, biases, and conservatisms - steam line break (Continued)

Parameter	Bias / Conservatism	Basis[1] {{
RCS Temperature Control Automatic rod control	Enabled. (MCHFR)	
Boron concentration	Not credited.	
PZR Pressure Control PZR spray	Varied.~~Disabled.~~	
PZR heaters	~~Enabled~~Nominal.	
PZR Level Control Charging	Not credited.	
Letdown	Disabled.	}}2(a),(c)

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Table 7-24 Initial conditions, biases, and conservatisms - steam line break (Continued)

Parameter	Bias / Conservatism	Basis[1]
Steam Pressure Control Turbine throttle valves	~~Enabled~~Varied.	{{
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

{{

}}2(a),(c)

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response and challenging mass releases for this overcooling event.

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Enclosure 3:

Affidavit of Mark W. Shaver, AF-177583

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10297 R1, Question NonLOCA.LTR-31, 32, 46, 56, 65) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10297 R1, Question NonLOCA.LTR-31, 32, 46, 56, 65. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 09, 2025.



Mark W. Shaver

January 14, 2025 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 032 (RAI-10297 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 032 (RAI-10297 R1)," dated October 31, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10297 R1:

- NonLOCA.LTR-3, 18, 27

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 032 (RAI-10297 R1, Question NonLOCA.LTR-3, 18, 27). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirement of 10 CFR § 810. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 14, 2025.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-3, 18, 27, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-3, 18, 27, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-178268

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-3, 18, 27, Nonproprietary Version

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Response to Request for Additional Information
Docket: 052000050

RAI No.: 10297

Date of RAI Issue: 10/31/2024

NRC Question No.: NonLOCA.LTR-3, 18, 27

Issue

The NPM-20 reactor pool is modeled by NuScale {{

}}$^{2(a),(c)}$

{{

}}2(a),(c) .

In addition, the staff notes that FSAR Chapter 5 states: “A turbine trip at full power without bypass capability is the most severe AOO and is the bounding event used in the determination of RSV capacity and the RPV overpressure analyses. Sizing of the RCS and the PZR steam space avoids an RSV lift during normal operational transients that produce the highest RPV pressure at full power conditions, with system and core parameters within normal operating range. In the event of a safety valve lift, the size of the PZR steam space is sufficient to preclude liquid discharge. The analytical model used for the analysis of the overpressure protection system and the basis for its validity is in the NuScale Topical Reports "Non-Loss-of-Coolant Accident Analysis Methodology" and "Loss-of-Coolant Accident Evaluation Model".” {{

}}2(a),(c) .

Information Requested

a) The Non-LOCA DHRS base model and its variants are not only used in Non-LOCA event simulations but also in LOCA analysis for FSAR Section 15.6.5 and Chapter 5 normal shutdown analysis. Provide evaluations and bases information that address the concerns, as described above, {{

}}2(a),(c)

{{

}}2(a),(c) , NuScale is requested to describe, from the perspective of condenser-to-pool heat transfer, the startup of a DHRS loop from cold (reactor pool temperature) conditions to the point of peak heat transfer rate, with focus on the various condenser-to-pool heat transfer modes and wall heat transfer correlations that will be involved with the startup of DHRS.

b) Provide an evaluation of the DHRS performance under a scenario {{

}}2(a),(c)

NuScale Response:

Executive Summary

The response describes performance of the decay heat removal system (DHRS) during startup from cold condition to the point of peak heat transfer rate. NRELAP5 predictions and NIST-2 data demonstrate that high fluid temperatures inside the condenser tubes rapidly transfer energy to the tube walls. {{

}}2(a),(c) For NuScale Power Module (NPM) conditions, the heat flux at the top of the condenser remains below the CHF, therefore film boiling conditions do not occur. Margin to CHF near the top of the DHRS condenser is justified based on {{

}}2(a),(c)

Evaluations conclude that the correlations in NRELAP5, {{

}}2(a),(c)

{{

}}2(a),(c) Sensitivity calculations demonstrate that safety analysis margins are not sensitive to {{

}}2(a),(c) While reactor safety valve (RSV) lift is calculated in non-LOCA Chapter 15 analyses, calculation results demonstrate that the RSV lift is associated with the initiating event, not conditions of reduced DHRS heat transfer capacity.

Evaluations and bases information are provided that address the concerns described in the RAI, {{

}}2(a),(c)

Overview

This response provides the requested information in the following sections (i.e., the numbering in the list identifies the numbered section in the response where the requested information is found):

1. RAI part (a) – Description of DHRS startup from cold conditions to point of peak heat transfer rate, from the perspective of condenser-to-pool heat transfer, focused on various condenser-to-pool heat transfer modes and wall heat transfer correlations associated with startup of DHRS. This discussion provides foundation and context for the subsequent evaluations and bases information and therefore is provided first in the response.
2. RAI part (a) – Evaluation and bases information that address the described NRC concerns associated with the DHRS modeling in NRELAP5. This section addresses the following NRC concerns: {{

}}2(a),(c)

3. RAI part (a) – Evaluation and bases information to address the described NRC concerns about {{

}}2(a),(c)

4. RAI part (a) – Discussion of RSV operation during two-train DHRS operation.
5. RAI part (b) – Evaluation of DHRS performance under a scenario with {{

}}2(a),(c)

6. Response to NRC feedback on material presented during clarification call.

1. **Decay Heat Removal System Startup**

During normal power operation, the DHRS is at standby conditions with the actuation valves closed. The DHRS condensate line, condenser, and steam piping up to the actuation valves are liquid-solid. The actuation valves are located above the pool water level. The liquid inside the DHRS is at equilibrium with surrounding pool fluid or pool atmosphere temperature conditions. Vertical condenser tubes are connected in parallel banks to horizontal header pipes; {{

}}2(a),(c)

Figure 1 through Figure 7 below provide NRELAP5 calculation results to illustrate key aspects of the DHRS startup and establishment of condenser-to-pool heat transfer. {{

}}2(a),(c)

When DHRS actuates, the containment isolation valves on the feedwater and main steam piping close, and the DHRS actuation valves open. Heat transfer from the RCS across the SG tubes generates steam in the SG tubes and pressure in the isolated secondary side loop increases due to the vapor generation, as shown in Figure 1. Liquid in the isolated secondary side loop redistributes between the DHRS and SG.

{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

Pool boiling curves show that as surface temperature increases, surface heat flux increases, until the heat flux reaches the CHF. {{

}}2(a),(c) However, heat fluxes in the DHRS condenser remain below pool boiling CHF values.

The behavior described above and shown in the NRELAP5 calculations for NPM conditions is consistent with measured data from NIST-2 non-LOCA integral effects testing. Figure 8 provides a schematic of relevant instrumentation locations. Figure 9 and Figure 10 show {{

}}2(a),(b),(c),ECI

{{

}}2(a),(b),(c),ECI

{{

}}2(a),(b),(c),ECI For NPM conditions, the heat flux at the top of the condenser remains below the CHF and therefore the startup evolution is as described above, and film boiling conditions do not occur. The pool boiling CHF appropriate for conditions around the DHRS condenser is discussed further in Section 3 of this response.

Table 1: Legend Description for Figure 9 and Figure 10

{{

}}2(a),(b),(c)

Figure 1: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 2: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 3: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 4: {{

}}2(a),(c)

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}}2(a),(c)

Figure 5: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 6: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 7: {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 8: Schematic of NIST-2 Instrument Locations on the Scaled Decay Heat Removal Heat Exchanger

{{

}}2(a),(b),(c)

Figure 9: NIST-2 Non-LOCA Run 3 Decay Heat Removal System Heat Exchanger Tube Fluid Temperatures Compared with NRELAP5 Simulation Results, Short-Term

{{

}}2(a),(b),(c),ECI

Figure 10: NIST-2 Non-LOCA Run 3 Decay Heat Removal System Heat Exchanger Tube Fluid Temperatures Compared with NRELAP5 Simulation Results, Long-Term

{{

}}2(a),(b),(c),ECI

2. **NRELAP5 Pool Modeling Effect on Decay Heat Removal System Heat Transfer Rate and Safety Analysis Event Margins**

The RAI identifies concerns that the NRELAP5 {{

}}2(a),(c)

{{

}}2(a),(c)

Therefore, it is concluded that the correlations in NRELAP5, {{ }}[2(a),(c)] can adequately account for natural convection heat transfer and boiling heat transfer on the DHRS tubes. {{ }}[2(a),(c)] Sensitivity calculations demonstrate that safety analysis margins are not sensitive to {{ }}[2(a),(c)]

3. **Evaluation of Margin to Film Boiling Conditions**

The discussion and calculations in Section 1 of this response demonstrate that during DHRS startup and initial operation, high fluid temperatures inside the condenser tubes rapidly transfer energy to the tube walls. {{

}}$^{2(a),(c)}$ around the DHRS condenser, to demonstrate margin to film boiling conditions. This section of the response provides the following information:

(a) Discussion of regulatory precedent
(b) Justification of similarity between US600 plant DHRS conditions and US460 plant DHRS conditions
(c) {{

}}$^{2(a),(c)}$
(e) Summary of evidence demonstrating margin to film boiling conditions

3(a) Regulatory Precedent

The key postulated NRC concern in the RAI is that if the wall heat flux exceeds the CHF, and film boiling conditions occurred, then DHRS heat transfer capacity could be reduced due to vapor blanketing at the top of the condenser. This concern was previously postulated during the NRC review of the 160 MWt NPM design. The concern was addressed in response to RAI 9374

Question 15.00.02-24 The RAI response demonstrated {{

}}[2(a),(c)] Similar conditions occur in the NPM-20 design, as discussed further in Section 3(b).

NuScale also notes that the AP1000 safety evaluation report in NUREG-1793 Section 21.5.4 identifies that concerns about film boiling degrading heat transfer on the AP1000 passive residual heat removal heat exchanger (PRHR HX) were "*resolved on the basis of the applicant's analyses of the margin of the PRHR HX heat flux to the critical heat flux limit ,and the fact that vapor blanketing was not observed in the APEX, SPES, and ROSA integral-effects test facilities*." Therefore, in this RAI response NuScale provides justification of margin between maximum DHRS heat flux and CHF for pool boiling conditions around the DHRS.

3(b) Justification of Similarity between NPM-160 and NPM-20 Decay Heat Removal System Conditions

The following points justify similarity in {{

}}[2(a),(c)] The comparison addresses the DHRS condenser tube geometry and accounts for differences in plant primary fluid operating conditions and the system design pressure.

- In both designs the DHRS condenser tubes are {{ }}[2(a),(c)] Therefore, the wall thickness and material are the same between the two designs.
- In both designs the condensers are located in a pool of subcooled water with nominal temperature around 100 degrees F.
- The design pressure of the secondary side, including the DHRS, increased from 2100 psia in the NPM-160 design to 2200 psia in the NPM-20 design. The saturation temperature at 2200 psia is 650 degrees F, compared to the saturation temperature of 643 degrees F at 2100 psia. However, this difference is not considered significant because {{

}}[2(a),(c)]

- Table 2 summarizes the maximum SG pressure for safety analysis heatup events, as presented in the US460 SDAA Final Safety Analysis Report (FSAR) and the US600 Design Certification Application (DCA) FSAR. The heatup events were selected as representative of the limiting secondary side pressures that could occur due to DHRS operation in response to a design-basis initiating event. It is noted that the SG tube failure accident results in the maximum secondary side pressure across the safety analysis events, but this is due to the initiating event where the failed SG equalizes pressure with the reactor coolant system. Table 2 shows that for both designs, maximum secondary side pressure for the range of events is around 1400-1600 psia. This demonstrates that for both designs the maximum saturated steam temperature is around 605 degrees F.
    - The similarity in the maximum secondary pressure is consistent with the similar total temperature gradient from the RCS primary to the pool between the two designs. A characteristic of NPM designs is that after reactor trip and DHRS actuation, the RCS hot and cold temperatures converge near the average temperature. This is shown in Figure 11, for {{

      }}[2(a),(c)] In the NPM-160 design, the RCS average temperature range is 535-555 degrees F. In the 250 MWt NPM-20 design the RCS average temperature is 535-545 degrees F. Similarity in the NPM-160 and NPM-20 RCS temperatures after reactor trip, during DHRS startup, can be discerned by comparison of a range of different transients.

Table 2: Comparison of Maximum Steam Generator Pressure between NPM-160 DCA and NPM-20 SDAA

Chapter 15 Initiating Event	Maximum Steam Generator Pressure (psia)	
	SDAA FSAR Revision 0	DCA FSAR Revision 5
Turbine trip	1426	1545
Main steam isolation valve closure	1597	1512
Loss of normal alternating current power	1459	1415
Loss of normal feedwater	1548	1528
Feedwater line break	1424	1389
Inadvertent operation of DHRS	1490	1592

Figure 11: {{

}}2(a),(c)

{{

}}2(a),(c)

3(c) Alternate Evaluation of Decay Heat Removal System Tube Heat Flux and Wall Temperature Gradient

NuScale performed {{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure 12: Schematic of Heat Transfer Thermal Resistance through Decay Heat Removal System to Reactor Pool

{{

}}2(a),(c)

Figure 13: Steady-State Temperature Distribution, {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 14: Steady-State Temperature Distribution, {{

}}2(a),(c)

{{

}}2(a),(c)

3(d) Critical Heat Flux around Decay Heat Removal System Condenser Tubes

As discussed in Section 3(a) of this response, for the NPM-160 design, NuScale demonstrated {{

}}2(a),(c) This comparison remains valid for the NPM-20 design as discussed in Section 3(b) of this response.

The CHF {{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}[2(a),(c)] the CHF remains above the maximum DHRS tube heat flux. Therefore, this evaluation confirms that appropriate parameters were previously used to evaluate the CHF limit and supports the continued conclusion that film boiling is not expected to occur around the DHRS condenser.

Reference 1: NUREG/CR-4567, “Prediction of Pool Void Fraction by New Drift Flux Correlation,” Isao Kataoka and Mamoru Ishii.

Table 3: Griffith-Zuber Pool Boiling Critical Heat Flux (kW/m^2)

{{

}}2(a),(c)

Table 4: Drift Flux Estimate of Void Fraction near Top of Decay Heat Removal System Condenser

{{

}}2(a),(c)

3(e) Summary

This section of the RAI response provides evaluation and technical bases demonstrating that the maximum heat flux near the top of the DHRS condensers remains below the pool boiling CHF. {{

}}2(a),(c)

NuScale previously evaluated margin between the maximum DHRS condenser tube heat flux and the pool boiling CHF as part of the DCA review. {{

}}2(a),(c) The similarity is justified based on comparison of DHRS condenser tube geometry and accounting for differences in plant primary fluid operating conditions and the system design pressure.

{{

}}2(a),(c) Therefore, this evaluation confirms that appropriate parameters were previously used to evaluate the CHF limit and supports the continued conclusion that film boiling is not expected to occur around the DHRS condenser.

These evaluations provide technical bases to demonstrate that high vapor temperatures inside the DHRS condensers will not cause departure from nucleate boiling and therefore film boiling conditions will not occur. These results are consistent with the NRELAP5 sensitivity calculation results discussed in Section 2 of this response.

4. **Reactor Safety Valve Operation during Two-Train Decay Heat Removal System Operation**

Although RSV operation is not an acceptance criteria in the non-LOCA topical report, the RAI requests {{

}}2(a),(c)

Section 3 of this response provides evaluations and technical basis information that departure from nucleate boiling does not occur for the maximum heat fluxes near the top of the DHRS condenser tubes. Section 2 of this response provides additional justification that the correlations in NRELAP5, {{ }}2(a),(c) can adequately account for natural convection heat transfer and boiling heat transfer on the DHRS tubes. {{

}}2(a),(c) Sensitivity calculations demonstrate that safety analysis margins are not sensitive to {{ }}2(a),(c)

The DHRS heat transfer capacity {{

}}2(a),(c) The increase in feedwater flow initiating events biased to limit DHRS heat transfer capacity demonstrate that the maximum pressurizer pressure is approximately 20 psi higher than the initial condition and RSV lift is not calculated to occur.

In other non-LOCA initiating events, particularly heatup events presented in FSAR Section 15.2, RSV lift is typically calculated to occur around the time of reactor trip and DHRS actuation, because the initiating event reduces heat transfer to the SG. The RSV lift under these conditions is not related to DHRS heat transfer capacity (i.e., is not related to DHRS modeling biases or uncertainties). The FSAR Section 15.5 results show that RSV lift is calculated to occur during two-train DHRS operation. The RCS pressure increase and RSV lifts in this event progression are due to the inventory addition of the initiating event that continues after reactor trip and DHRS actuation on high pressurizer pressure, until the inventory addition source is isolated by the high pressurizer level signal. The FSAR results demonstrate that the RCS average temperature generally decreases after reactor trip and DHRS actuation for two-train operation, demonstrating that the calculated RSV lifts are due to inventory addition and do not indicate unacceptable DHRS heat transfer capacity.

5. **Evaluation of DHRS Performance under Scenario with Rapid Condenser Wall Temperature Increase**

Based on the results shown in Section 1 of this response, rapid increase of the outside tube wall during DHRS startup {{

}}$^{2(a),(c)}$ no additional evaluation of DHRS performance under a scenario of rapid condenser wall heatup is needed.

6. **Response to NRC Feedback on Presentation Material**

NuScale presented material to the NRC as part of clarification calls associated with this RAI. The presentation material was subsequently made available for audit. The NRC provided feedback on the presentation material. The NuScale response to the NRC feedback is provided in this section.

(1) The NRC feedback requested a comparison of DHRS conditions between the SDAA design and the DCA design. The information is provided in Section 3 of the response.
(2) The NRC feedback requested an explanation for {{

}}$^{2(a),(c)}$
(3) The NRC feedback requested a comparison of DHRS conditions between the NPM and NIST-2. The information is provided in Section 1 of the response.
(4) The NRC feedback requests explanation of how steady-state heat transfer calculations are related to the RAI request about DHRS startup transient. The information about the DHRS startup transient is provided in Section 1 of the response. The information about how steady-state heat transfer calculations, including how it is useful for understanding the DHRS startup transient, is provided in Section 3 of the response.
(5) The NRC feedback requested explanation of pool-to-DHRS nodalization changes (i.e., SDAA vs DCA). NuScale reviewed the pool nodalization associated with DHRS for the SDAA and DCA. No significant changes have been made.

{{

}}2(a),(c) This change has no impact on the information provided in this RAI response. However, to ensure consistency and clarity, TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model,” is revised as shown in the attached markups {{

}}2(a),(c)

Impact on US460 SDAA:

Topical Report TR-0516-49422, Loss-of-Coolant Accident Evaluation Model, has been revised as described in the response above and as shown in the markup provided in this response.

RAI NonLOCA.LTR-3, 18, 27

Figure 5-2 Noding Diagram of NRELAP5 Loss-of-Coolant Input Model for the NPM-20

{{

}}2(a),(c)

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5.1.4 Containment Vessel and Reactor Pool

RAI NonLOCA.LTR-3, 18, 27

{{

RAI NonLOCA.LTR-3, 18, 27

}}$^{2(a),(c)}$

RAI NonLOCA.LTR-3, 18, 27

The reactor pool is the ultimate heat sink in the NPM design. The reactor pool volume corresponding to an individual NPM is represented by a {{

}}$^{2(a),(c)}$ A wide range of initial reactor pool temperatures is exercised to show the effect of the pool conditions on the LOCA behavior in Section 9.6.5.

{{

}}$^{2(a),(c)}$

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Enclosure 3:

Affidavit of Mark W. Shaver, AF-178268

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10297 R1, Question NonLOCA.LTR-3, 18, 27) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10297 R1, Question NonLOCA.LTR-3, 18, 27. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 14, 2025.



Mark W. Shaver

February 11, 2025 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 032 (RAI-10297 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 032 (RAI-10297 R1)," dated October 31, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10297 R1:

- NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 032 (RAI-10297 R1, Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 11, 2025.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-179389

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69, Nonproprietary Version

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10297
Date of RAI Issue: 10/31/2024

NRC Question No.: NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69

Issues

The methodology in NuScale topical report "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P, Revision 3 (ML23008A001) (LOCA LTR) and the Non-LOCA LTR does not contain sufficient scaling analysis updates with the NIST-2 testing facility to capture higher power operations of the NPM-20 RCS, containment, and cooling pool. Between the LOCA LTR and Non-LOCALTR there is missing justification to extend the similarity of the NPM-160/NIST-1 to the NPM-20/NIST-2 in terms of the distortion analysis of important phenomena. This scaling analysis is discussed in the justification for applicability of the NRELAP5 code but not provided in the submission adequately.

• Code validation for predicting DHRS performance

The Non-LOCA LTR presents a validation of the NRELAP5 code using the NIST-2 test data. The code validation is extended to applications of the code for LOCA events {{

}}$^{2(a),(c)}$.

• Similarities (or distortion) of NIST-2 tests for code validation

Distortion analysis is missing in the Non-LOCA LTR for validation of the NRELAP5 code based on the NIST-2 test for analyses of Non-LOCA and LOCA events. A distortion analysis must be performed to identify the necessary correction factor in the code and models because comparison between the NIST-2 test facility and the NPM-20 design shows significant dissimilarities.

Integral and separate effects tests are used to justify the development and applicability of an evaluation model to a given design. Since test facilities are not typically full-scale, distortions exist that can affect local and global elements of the analysis when compared to the full-scale plant. Therefore, a scaling analysis needs to be performed that identifies important non-dimensional parameters related to geometry and key phenomena and scaling distortions and their impact on the code assessment must be identified and evaluated.

• Applicability of the NIST-2 tests for validating code for analyzing events

In the Non-LOCA LTR, the applicability of the NRELAP5 DHRS modeling capability is extended to the LOCA LTR {{

}}$^{2(a),(c)}$.

• DHRS condensate flow oscillations and nodalization

{{

}}$^{2(a),(c)}$

{{

}}2(a),(c) .

• Prediction of the impact of coolant inventory on DHRS performance

During audit discussions, NuScale presented information that showed performance degradation in certain DHRS loop inventory ranges. The SG-DHRS loop inventory is determined when the SG-DHRS loop is isolated for DHRS operation. In various Chapter 15 events, the loop isolation timing varies depending on the event progression and plant response.

In reviewing the NRELAP assessment, NuScale identified the influence of DHRS loop inventory on heat removal. NuScale made available for staff audit (ML23067A300) several studies to address the staff's concern regarding the similarity between NIST-2 and the NPM-20. {{

}}2(a),(c) .

• NRELAP5 modeling of secondary-side mass inventory

{{

}}2(a),(c) .

Information Requested

a) Provide NRELAP5 DHRS model validation for LOCA events {{

}}2(a),(c) .

b) {{

}}2(a),(c) . Provide the bounding uncertainty of the DHRS methodology in LOCA until long-term cooling is established with or without ECCS operation.

c) {{

}}2(a),(c) .

d) {{

}}2(a),(c) .

e) Explain the impact of {{ }}[2(a),(c)] the scaling analysis in the Non-LOCA tests and the scalability to NPM-20.

f) {{ }}[2(a),(c)]. If not, provide a most-limiting SG-DHRS loop inventory analysis for applicable Chapter 15 events to ensure sufficient DHRS heat removal capacity.

NuScale Response:

Executive Summary

A response is provided to each part of the request for additional information (RAI) in the sections below. Overall, the response identifies that the decay heat removal system (DHRS) remains effective during loss-of-coolant accident (LOCA) events, including during uncovery of steam generator (SG) tubes prior to or after emergency core cooling system (ECCS) actuation and up until the point at which long-term cooling is established. There is no evidence to suggest that there is significant uncertainty in the NRELAP5 predictions of DHRS performance. Even if there were uncertainty, sensitivity study results provided in this response (and others previously provided) demonstrate {{ }}[2(a),(c)] has no impact on figures of merit (FOM) relevant to LOCA. Sensitivity study results also show that {{ }}[2(a),(c)] has no impact on LOCA event FOM margins. NRELAP5 predictions of DHRS performance are reasonable when compared to NuScale Integral System Test Facility (NIST)-2 test data. {{ }}[2(a),(c)] The NRC also expressed concern with a modeling approach used in NRELAP5 {{ }}[2(a),(c)] The response provides sensitivity studies that demonstrate that {{ }}[2(a),(c)] there is no significant impact on overall DHRS performance. {{ }}[2(a),(c)]

{{

}}2(a),(c) These sensitivity studies provide further evidence that DHRS performance is effective and can be reasonably modeled by NRELAP5.

As a further reassurance that DHRS will be effective during NuScale Power Module (NPM)-20 operation, the Final Safety Analysis Report (FSAR) Table 14.2-41 includes a DHRS system level test that is required to be performed for the first NPM tested. The test will verify that DHRS removes heat from the reactor coolant system (RCS). The cooldown rate of the test will be compared to the code of record (i.e., NRELAP5). The test will validate both the effectiveness of DHRS in actual performance as well as the ability of the NRELAP5 code to predict performance.

Part a

This part of the RAI requests validation of the NRELAP5 DHRS model for LOCA events
{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

The above information was added to TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model,” as shown in the markups provided with the response to RAI 10296 question LOCA.LTR-4, 45, 47.

Part b

This part of the RAI requests a bounding uncertainty of the DHRS methodology in LOCA until long-term cooling is established with or without ECCS operation. {{

}}2(a),(c) .

NuScale disagrees with the premise that there is a lack of information regarding DHRS performance during LOCA. As described in the response to part a above, DHRS remains highly effective during LOCA conditions.

To support the previous NRC approval of TR-0516-49416-P-A, Revision 3, “Non-Loss-of-Coolant Accident Analysis Methodology,” NuScale performed sensitivity analyses considering the impact of a 30 percent uncertainty in DHRS performance. Consistent with the prior approach to addressing NRC concerns regarding DHRS uncertainty during non-LOCAs, NuScale has similarly performed sensitivity analyses considering an even larger uncertainty in DHRS performance as described below.

NuScale has previously provided a sensitivity study considering DHRS uncertainty in the response to RAI 10359 question 6.2.1-1. {{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c) NuScale has not identified any information that would suggest the uncertainty in DHRS performance is that large; NuScale believes the uncertainty to be minimal. Therefore, explicit quantification of the uncertainty is not required.

Part c

This part of the RAI requests several different items. For clarity, each item of the request is listed in indented, italicized text, followed by the NuScale response.

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}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c

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}}2(a),(c)

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}}2(a),(c

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}}2(a),(c)

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}}2(a),(c

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}}2(a),(c)

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}}2(a),(c

As a further reassurance that DHRS will be effective during NPM-20 operation, the FSAR Table 14.2-41 includes a DHRS system level test that is required to be performed for the first NPM tested. The test will verify that DHRS removes heat from the RCS. The cooldown rate of the test will be compared to the code of record (i.e., NRELAP5). The test will validate both the effectiveness of DHRS in actual performance as well as the ability of the NRELAP5 code to predict performance.

Part d

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}}2(a),(c)

{{

}}2(a),(c)

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}}2(a),(c)

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}}2(a),(b),(c)

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}}2(a),(c)

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}}2(a),(b),(c)

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}}2(a),(b),(c)

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}}2(a),(b),(c)

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}}2(a),(c)

{{

}}2(a),(c)

Figure 1: NIST-2 Run 1 Steam Generator and Decay Heat Removal System Active Loop Inventory Comparison

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}}2(a),(b),(c)

Figure 2: NIST-2 Run 1 Primary Pressure Comparison

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}}2(a),(b),(c)

Figure 3: NIST-2 Run 1 Secondary Pressure Comparison

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}}2(a),(b),(c)

Figure 4: NIST-2 Run 1 Primary Flow Rate Comparison

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}}2(a),(b),(c)

Figure 5: NIST-2 Run 1 Condensate Flow Rate Comparison

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}}2(a),(b),(c)

Figure 6: NIST-2 Run 1 Decay Heat Removal System Heat Exchanger Inlet and Outlet Temperature Comparison

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}}2(a),(b),(c)

Figure 7: NIST-2 Run 1 Steam Generator Secondary Side Inlet and Outlet Temperature Comparison

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}}2(a),(b),(c)

Figure 8: NIST-2 Run 1 Steam Generator Primary Side Inlet and Outlet Temperature Comparison

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}}2(a),(b),(c)

Figure 9: NIST-2 Run 1 Decay Heat Removal System Level Comparison

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}}2(a),(b),(c)

Figure 10: NIST-2 Run 1 Steam Generator Level Comparison

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}}2(a),(b),(c)

Figure 11: NIST-2 Run 1 Decay Heat Removal System Heat Exchanger Power Comparison

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}}2(a),(b),(c)

Figure 12: NIST-2 Run 2 Steam Generator and Decay Heat Removal System Active Loop Inventory Comparison

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}}2(a),(b),(c)

Figure 13: NIST-2 Run 2 Primary Pressure Comparison

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}}2(a),(b),(c)

Figure 14: NIST-2 Run 2 Secondary Pressure Comparison

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}}2(a),(b),(c)

Figure 15: NIST-2 Run 2 Primary Flow Rate Comparison

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}}2(a),(b),(c)

Figure 16: NIST-2 Run 2 Condensate Flow Rate Comparison (with Running Average)

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}}2(a),(b),(c)

Figure 17: NIST-2 Run 2 Decay Heat Removal System Heat Exchanger Inlet and Outlet Temperature Comparison

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}}2(a),(b),(c)

Figure 18: NIST-2 Run 2 Steam Generator Secondary Side Inlet and Outlet Temperature Comparison

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}}2(a),(b),(c)

Figure 19: NIST-2 Run 2 Steam Generator Secondary Side Inlet and Outlet Temperature Comparison (Alternate Feedwater Location)

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}}2(a),(b),(c)

Figure 20: NIST-2 Run 2 Steam Generator Primary Side Inlet and Outlet Temperature Comparison

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}}2(a),(b),(c)

Figure 21: NIST-2 Run 2 Decay Heat Removal System Level Comparison

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}}2(a),(b),(c)

Figure 22: NIST-2 Run 2 Steam Generator Level Comparison

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}}2(a),(b),(c)

Figure 23: NIST-2 Run 2 Decay Heat Removal System Heat Exchanger Power Comparison (with Running Average)

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}}2(a),(b),(c)

Figure 24: NIST-2 Run 2 Steam Line Orifice Differential Pressure

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}}2(a),(b),(c)

Part e

{{

}}2(a),(c)

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}}2(a),(c)

{{

}}2(a),(c)

Table 1: NIST-2 Non-LOCA Run 1 Integral Energy Comparison

{{

}}2(a),(b),(c)

Figure 25: NIST-2 Mass Sensitivity Run 1 Base Case Energy Rate Comparison

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}}2(a),(b),(c)

Figure 26: NIST-2 Mass Sensitivity Run 1 Base Case Integral Energy Comparison

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}}2(a),(b),(c)

{{

}}2(a),(c)

Table 2: Mass Sensitivity Time-Averaged Energy Pseudo-Pi Group (Equation 1)

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}}2(a),(b),(c)

Table 3: Mass Sensitivity Time-Averaged Mass Pseudo-Pi Group (Equation 2)

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}}2(a),(b),(c)

Figure 27: DHRS Mass Boundary Energy Pi Group

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}}2(a),(b),(c)

Figure 28: Original Post-Test DHRS Energy Pi Groups

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}}2(a),(b),(c)

Figure 29: DHRS Mass Boundary Mass Pi Group

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}}2(a),(b),(c)

Figure 30: Original Post-Test DHRS Mass Pi Group

{{

}}2(a),(b),(c)

Part f

{{

}}2(a),(c)

The analysis of the increase in feedwater flow event in FSAR Section 15.1.2 includes consideration of the impact of high SG-DHRS inventory on DHRS performance. The analysis of the main steam line break event in FSAR Section 15.1.5 includes consideration of the impact on DHRS performance from a loss of one train of DHRS due to the break location and high SG-DHRS inventory in the intact train of DHRS. Given the discussion in parts d and e of this response, NuScale concludes that the FSAR Section 15.1.2 and FSAR Section 15.1.5 analyses are not impacted by the particular active inventory modeling approach used in the NIST-2 test assessments. Therefore, the FSAR Section 15.1.2 and FSAR Section 15.1.5 analyses remain valid. The analyses demonstrate that the DHRS adequately removes decay heat during limiting high SG-DHRS inventory conditions.

Unrelated to this RAI, the FSAR Section 15.1.2 analysis was revised as described in the response to RAI 10297 Question NonLOCA.LTR-60. The supporting document, EC-0000-8328, Revision 1, “Increase in Feedwater Flow Transient Analysis,” was made available for the NRC staff as part of the response to that RAI question.

Additional Information

The current NRELAP5 model for DHRS includes consideration of the heat transfer that occurs in the portion of the DHRS steam line piping that is below the level of the ultimate heat sink as well as the DHRS condenser headers. The modeling of heat transfer in these components is not described in TR-0516-49416, Revision 4. Therefore, TR-0516-49416, Revision 4, is revised as indicated in the attached markups to identify that these components may be considered in the heat transfer and identify the heat transfer boundary condition types used in NRELAP5 when modeling the heat transfer from these components.

Impact on Topical Report:

Topical Report TR-0516-49416, Non-Loss-of-Coolant Accident Analysis Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

6.1.5 Decay Heat Removal System

Audit Question A-NonLOCA.LTR-63
RAI 10297 Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69

An NPM incorporates two separate DHRSs that are treated individually in the NRELAP5 model. {{

}}2(a),(c) Figure 6-13 shows the typical nodalization for DHRS loop 1 {{

}}2(a),(c). Loop 2 is modeled similarly. While each DHRS line in an NPM features two parallel actuation valves, {{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-63

The number of hydrodynamic volumes in the DHRS condenser piping and HX regions are based on results from NRELAP5 assessments using data from the NIST-1 facility (Section 5.3.2). {{

}}2(a),(c)

RAI 10297 Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69

In addition to the DHRS condenser piping and HX regions, heat removal occurs in the DHRS steam line piping below the level of the reactor cooling pool and from the DHRS condenser headers. {{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-63

In an actual NPM, the DHRS ~~heat exchanger is~~condensers are located in the reactor cooling pool. {{

}}2(a),(c) The long-term use of DHRS is addressed separately in Reference 26.

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Enclosure 3:

Affidavit of Mark W. Shaver, AF-179389

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www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10297 R1, Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10297 R1, Question NonLOCA.LTR-3, 9, 18, 19, 20, 21, 69. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 11, 2025.



Mark W. Shaver

February 14, 2025 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 032 (RAI-10297 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 032 (RAI-10297 R1)," dated October 31, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10297 R1:

- NonLOCA.LTR-55

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 032 (RAI-10297 R1, Question NonLOCA.LTR-55). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 14, 2025.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-55, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-55, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-179402

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-55, Nonproprietary Version

NuScale Power, LLC
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Response to Request for Additional Information
Docket: 052000050

RAI No.: 10297
Date of RAI Issue: 10/31/2024

NRC Question No.: NonLOCA.LTR-55

Issue

As a result of the NRC staff's Quality Assurance Program (QAP) implementation inspection in February 2024, the NRC staff understands that the Non-LOCA LTR is controlled as a licensing product within NuScale, and the document itself is not considered to be the Non-LOCA Evaluation Model (EM). It is the NRC staff's understanding that information presented in the Non-LOCA LTR is derived from engineering documents. As such, the NRC staff needs to determine if there is information presented or conclusions stated in the Non-LOCA LTR, which is drawn from engineering documents that are not subject to design verification consistent with 10 CFR 50 Appendix B in accordance with Part II, "Quality Assurance Program Description Details," Section 2.3.1, "Design Verification" of Topical Report MN-122626, Revision 1-A, "NuScale Power, LLC Quality Assurance Program Description" (ML24033A318) (hereafter referred to as QAPD LTR). This includes information subject to Part III, "Nonsafety-Related Structures, Systems, and Components (SSC) Quality Control," Section 3.1.3, "Design Control" in the QAPD LTR, regardless of whether the verification process has been conducted in accordance with this section of the QAPD LTR, since the Section 3.1.3 verification is not required to conform to 10 CFR 50 Appendix B.

Information Requested

a) Provide a list of any such documents that have not undergone design verification as described above. For any of these documents not already available in eRR for staff audit, provide those additional documents available for audit. NuScale may choose to provide any relevant information regarding which portions of the LTR or conclusions those documents support.

b) In addition, provide a listing of any section of the Non-LOCA LTR that are either not considered part of the EM or not directly supported by engineering documents that are part of the EM.

NuScale Response:

The NRC staff provided a list of NuScale internal documents during the review of TR-0516-49422-P, “Loss-of-Coolant Accident (LOCA) Methodology”, TR-124587-P, “Extended Passive Cooling (XPC) and Reactivity Control Methodology”, and TR-0516-49416-P, “Non-Loss-of-Coolant Accident Analysis Methodology”, that the NRC staff’s technical reviewers were concerned did not meet pertinent Quality Assurance requirements. NuScale reviewed each of the 29 unique documents identified with a “QA flag” indicator, noting that the original list included one document three times (previous discussions referred to 31 total documents):

- For 26 of these documents, applicable 10 CFR 50, Appendix B / ASME NQA-1 requirements associated with verification have been met.
    - Procedural instructions, including verification of implementation, in place at the time each document was prepared and approved were evaluated against the requisite criteria for verification, regardless of the classification on the document.
- Final review of one document, EE-T080-13757, Rev. 2, “NuScale Integral System Test (NIST-1) Facility Scaling Analysis”, is complete and the status is detailed in Note 2 of Table 1.
- Two documents are governed by other processes and or procedures, not the design control process.
    - One document, SDR-0619-7331, Rev. 1, NIST-2 Facility Description Report, was generated under a vendor’s 10 CFR 50, Appendix B quality program and accepted by NuScale.
    - The second document, SwTR-0304-17153, Rev. 9, NRELAP5 Software Test Report, is governed by the NuScale software program, which identifies software integrity level (SIL) 3 software as safety-related and requires appropriate verification in those implementing procedures. The NRELAP5 code is classified as SIL3.

An evaluation has been initiated within NuScale’s Corrective Action Program to assess adequacy of processes and procedures that govern the development and maintenance of

evaluation models. The evaluation will generate specific, relevant actions to address any shortcomings identified, with completion dates assigned and tracked through to closure.

Table 1: NuScale Internal Documents the NRC Staff Provided

{{				
				}}2(a),(c)

{{				
				}}2(a),(c)

{{				

{{				

}}2(a),(c)

Impact on Topical Report:

There are no impacts to Topical Report TR-0516-49416, Non-Loss-of-Coolant Accident Analysis Methodology, as a result of this response.

Enclosure 3:

Affidavit of Mark W. Shaver, AF-179402

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NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10297 R1, Question NonLOCA.LTR-55) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10297 R1, Question NonLOCA.LTR-55. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

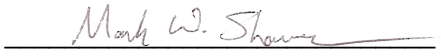
(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 14, 2025.



Mark W. Shaver

February 27, 2025 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 032 (RAI-10297 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, "Request for Additional Information No. 032 (RAI-10297 R1)," dated October 31, 2024

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The enclosure to this letter contains the NuScale response to the following RAI question from NRC RAI-10297 R1:

- NonLOCA.LTR-50

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 032 (RAI-10297 R1, Question NonLOCA.LTR-50). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 27, 2025.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-50, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-50, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-179798

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10297 R1, Question NonLOCA.LTR-50, Nonproprietary Version

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10297
Date of RAI Issue: 10/31/2024

NRC Question No.: NonLOCA.LTR-50

Issue

By letter dated January 6, 2023 (ML23011A012), NuScale submitted NRELAP5 files via DVD to support review of the Non-LOCA LTR and other LTRs that support the SDAA. NuScale informed NRC in February 2024, that a code error was found. Subsequently, NuScale informed NRC that a new code version, NRELAP5 v1.7, was being released and would be used for the SDAA and supporting topical reports. By letter dated August 6, 2024 (ML24228A242), NuScale submitted {{

}}$^{2(a),(c)}$.

Information Requested

In order for the staff to complete its technical review of the analytical approach and base its findings on submitted information relevant to the version of the code utilized in developing the EM, NuScale is requested to:
a) Re-submit any of the remaining files impacted by this code version change. All files affected by the revision to v1.7, {{ }}$^{2(a),(c)}$, need to be resubmitted to provide a complete submittal package for all four of these LTRs. {{

}}$^{2(a),(c)}$. Note that similar requests are being made for all four LTRs, and responses can reference each other.

b) The markup to Non-LOCA LTR draft revision 5 that the staff audited states "testing associated with the release of the Reference 31 version of NRELAP5 confirmed no impact on the Non-LOCA EM." Make available for staff audit the documentation of the completed testing.

c) Make available for audit an evaluation of the new version of the NRELAP5 code with all code qualification and design basis event analyses where the choked flow is involved to demonstrate impact on the current calculations. Provide LTR markups with updates to describe choked flow modeling changes and add or replace all evaluations in the LTR that are impacted.

NuScale Response:

Part a

NuScale previously submitted {{ }}$^{2(a),(c)}$ NRELAP5 version 1.7 (v1.7) in the following letter:

> NuScale letter to NRC, LO-172700, “NuScale Power, LLC Submittal of Supplemental Information in Support of Review of TR-131981-P, ‘Methodology for the Determination of the Onset of Density Wave Oscillations (DWO),’ Revision 1 Audit Question A-DWO.LTR-127,” dated August 6, 2024.

In addition, NuScale previously submitted supporting information for NRELAP5 version 1.7 (v1.7) {{ }}$^{2(a),(c)}$ in the following letter:

> NuScale letter to NRC, LO-175108, “NuScale Power, LLC Submittal of Supplemental Information in Support of NRC Review of NuScale Topical Reports Associated with the Standard Design Approval Application,” dated October 17, 2024.

The October 17, 2024 letter contained the files shown in Table 1 below.

Table 1: Files Contained in October 17, 2024 NuScale Letter to NRC

{{

}}2(a),(c)

In response to a clarification call with the NRC on November 25, 2024, NuScale submitted NRELAP5 input files in the following letter:

> NuScale letter to NRC, LO-176112, “NuScale Power, LLC Submittal of Supplemental Information in Support of NRC Review of NuScale Topical Reports Associated with the Standard Design Approval Application,” dated December 2, 2024.

The December 2, 2024 letter contained the files shown in Table 2 below.

Collectively, the submittals provide the information requested by this part.

Table 2: Files Contained in the December 2, 2024 NuScale Letter to NRC

{{

}}2(a),(c)

Part b

The document supporting the quoted statement is SwTR-0304-17153, Revision 9, “NRELAP5 Software Test Report.” The document SwTR-0304-17153 was previously provided to the NRC for audit as requested by this part. SwTR-0304-17153 also includes an attachment that is very large (i.e., several thousand pages). As discussed during a clarification call with the NRC on August 15, 2024, only the relevant portions of the attachment to SwTR-0304-17153 were provided to the NRC for audit. In the response to audit question A-NonLOCA.LTR-50, NuScale provided a roadmap to identify the information in the attachment that was most relevant.

{{

}}2(a),(c) Based on discussions with the NRC during a clarification call associated with the advanced draft RAI on September 21, 2024, the NRC clarified that additional comparisons between NRELAP5 version 1.6 (v1.6) and v1.7 for the NPM-20 design-basis events were needed to answer this question. Specifically, the NRC expressed interest in NRELAP5 v1.6 and v1.7 comparisons for the following non-loss-of-coolant

accident (non-LOCA) NPM-20 analyses: {{

}}2(a),(c)

NuScale performed comparisons of NRELAP5 v1.6 to v1.7 in EE-175345, Revision 0, "Comparison Study of NRELAP5 version 1.6 and 1.7." The document EE-175345 is provided for NRC audit with this response. The relevant sections of EE-175345 for the non-LOCA events identified above are indicated in the list below. {{

}}2(a),(c)

{{

}}2(a),(c)

The comparisons previously provided in SwTR-0304-17153 and the supplemental comparisons provided in EE-175345 with this response are adequate to demonstrate that NRELAP5 v1.7 is performing as expected and NPM non-LOCA transient results are not affected by the version change.

Part c

NuScale reviewed TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology” regarding choked flow modeling. {{

}}2(a),(c) Therefore, the NRELAP5 v1.7 changes do not impact the choked flow model discussions in TR-0516-49416-P, Revision 4. The only impacts to TR-0516-49416-P, Revision 4, because of the NRELAP5 v1.7 changes are those previously identified in the markups provided with the responses (original and supplement) to audit question A-NonLOCA.LTR-50.

Impact on US460 SDAA:

There are no impacts to Topical Report TR-0516-49416, Non-Loss-of-Coolant Accident Analysis Methodology, as a result of this response. A minor clarification to Topical Report TR-0516-49422, Loss-of-Coolant Accident Evaluation Model, is made based on NRC feedback, and is shown in the attached pages. Note this response references a proprietary version of the topical report that is marked as containing export controlled information (ECI). However, the extracted pages of the topical report that are attached to this response do not contain ECI as submitted herein. Notwithstanding, any proprietary information included in the response and the attachment hereto shall be withheld per 10 CFR 2.390.

5.1.2 Reactor Coolant System

The RCS model is composed of the LP, reactor core, riser (lower, transition, and upper sections), riser plenum, downcomer (upper section containing the helical coil SGs and lower section), and pressurizer.

5.1.2.1 Lower Plenum

{{

}}2(a),(c)

5.1.2.2 Reactor Core

5.1.2.2.1 General Model

The reactor core assembly is modeled with {{

}}2(a),(c)

111 U.S. Nuclear Regulatory Commission, "Design-Specific Review Standard for NuScale SMR Design," Chapter 6, Section 6.2.1.1.A, Rev. 0, June 2016.

112 U.S. Nuclear Regulatory Commission, "Design-Specific Review Standard for NuScale SMR Design," Chapter 6, Section 6.2.1.3, Rev. 0, June 2016.

113 U.S. Nuclear Regulatory Commission, "Design-Specific Review Standard for NuScale SMR Design," Chapter 6, Section 6.2.1.4, Rev. 0, June 2016.

114 NuScale Power, LLC, Topical Report "Non-LOCA Transient Analysis Methodology," TR 0516-49416-P, Rev. 3.

115 NuScale Power, LLC, Topical Report, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012-P-A, Revision 1.

116 Tong, L.S., Currin H.B., et. al., "Influence of axially non-uniform heat flux on DNB," AIChE Chemical Engineering Progress Symposium Series, Vol. 62, No. 64.

117 U.S. Nuclear Regulatory Commission, "Applying Statistics," NUREG-1475, Rev. 1.

118 NuScale Power, LLC, “Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2,” TR-108601-P-A, Revision 4.

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Enclosure 3:

Affidavit of Mark W. Shaver, AF-179798

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10297 R1, Question NonLOCA.LTR-50) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response to NRC Request for Additional Information RAI 10297 R1, Question NonLOCA.LTR-50. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

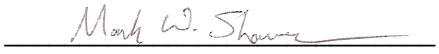
(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 27, 2025.



Mark W. Shaver

Section D

Audit Question Number	NuScale Letter Number
A-NonLOCA.LTR-4	LO-169995 dated 08/02/2024 (ML24215A229)
A-NonLOCA.LTR-6	LO-169995 dated 08/02/2024 (ML24215A230)
A-NonLOCA.LTR-7	LO-169995 dated 08/02/2024 (ML24215A235)
A-NonLOCA.LTR-10	LO-169995 dated 08/02/2024 (ML24215A237)
A-NonLOCA.LTR-12	LO-169995 dated 08/02/2024 (ML24215A239)
A-NonLOCA.LTR-13	LO-169995 dated 08/02/2024 (ML24215A241)
A-NonLOCA.LTR-14	LO-169995 dated 08/02/2024 (ML24215A243)
A-NonLOCA.LTR-15	LO-169995 dated 08/02/2024 (ML24215A250)
A-NonLOCA.LTR-16	LO-169995 dated 08/02/2024 (ML24215A252)
A-NonLOCA.LTR-17	LO-169995 dated 08/02/2024 (ML24215A253)
A-NonLOCA.LTR-2	LO-176318 dated 12/13/2024 (ML24348A010)
A-NonLOCA.LTR-5	LO-176318 dated 12/13/2024 (ML24348A012)
A-NonLOCA.LTR-5S	LO-176318 dated 12/13/2024 (ML24348A014)
A-NonLOCA.LTR-5S2	LO-176318 dated 12/13/2024 (ML24348A018)
A-NonLOCA.LTR-8	LO-176318 dated 12/13/2024 (ML24348A020)
A-NonLOCA.LTR-8S	LO-176318 dated 12/13/2024 (ML24348A022)
A-NonLOCA.LTR-22	LO-176318 dated 12/13/2024 (ML24348A024)
A-NonLOCA.LTR-23	LO-176318 dated 12/13/2024 (ML24348A027)
A-NonLOCA.LTR-24	LO-176318 dated 12/13/2024 (ML24348A029)
A-NonLOCA.LTR-25	LO-176318 dated 12/13/2024 (ML24348A031)
A-NonLOCA.LTR-26	LO-176318 dated 12/13/2024 (ML24348A033)
A-NonLOCA.LTR-28	LO-176318 dated 12/13/2024 (ML24348A038)
A-NonLOCA.LTR-28S	LO-176318 dated 12/13/2024 (ML24348A040)
A-NonLOCA.LTR-29	LO-176318 dated 12/13/2024 (ML24348A042)
A-NonLOCA.LTR-30	LO-176318 dated 12/13/2024 (ML24348A043)
A-NonLOCA.LTR-33	LO-176318 dated 12/13/2024 (ML24348A045)
A-NonLOCA.LTR-34	LO-176318 dated 12/13/2024 (ML24348A048)

Audit Question Number	NuScale Letter Number
A-NonLOCA.LTR-35	LO-176318 dated 12/13/2024 (ML24348A050)
A-NonLOCA.LTR-36	LO-176318 dated 12/13/2024 (ML24348A052)
A-NonLOCA.LTR-37	LO-176318 dated 12/13/2024 (ML24348A054)
A-NonLOCA.LTR-38	LO-176318 dated 12/13/2024 (ML24348A057)
A-NonLOCA.LTR-39	LO-176318 dated 12/13/2024 (ML24348A058)
A-NonLOCA.LTR-40	LO-176318 dated 12/13/2024 (ML24348A059)
A-NonLOCA.LTR-41	LO-176318 dated 12/13/2024 (ML24348A061)
A-NonLOCA.LTR-43	LO-176318 dated 12/13/2024 (ML24348A062)
A-NonLOCA.LTR-44	LO-176318 dated 12/13/2024 (ML24348A063)
A-NonLOCA.LTR-44S	LO-176318 dated 12/13/2024 (ML24348A067)
A-NonLOCA.LTR-45	LO-176318 dated 12/13/2024 (ML24348A069)
A-NonLOCA.LTR-49	LO-176318 dated 12/13/2024 (ML24348A070)
A-NonLOCA.LTR-54	LO-176318 dated 12/13/2024 (ML24348A072)
A-NonLOCA.LTR-57	LO-176318 dated 12/13/2024 (ML24348A073)
A-NonLOCA.LTR-58	LO-176318 dated 12/13/2024 (ML24348A075)
A-NonLOCA.LTR-59	LO-176318 dated 12/13/2024 (ML24348A077)
A-NonLOCA.LTR-59S	LO-176318 dated 12/13/2024 (ML24348A079)
A-NonLOCA.LTR-59S2	LO-176318 dated 12/13/2024 (ML24348A081)
A-NonLOCA.LTR-61	LO-176318 dated 12/13/2024 (ML24348A084)
A-NonLOCA.LTR-62	LO-176318 dated 12/13/2024 (ML24348A085)
A-NonLOCA.LTR-68	LO-176318 dated 12/13/2024 (ML24348A087)
A-NonLOCA.LTR-46	LO-178078 dated 01/14/2025 (ML25014A269)
A-NonLOCA.LTR-46S	LO-178078 dated 01/14/2025 (ML25014A271)
A-NonLOCA.LTR-56	LO-178078 dated 01/14/2025 (ML25014A273)
A-NonLOCA.LTR-63	LO-178078 dated 01/14/2025 (ML25014A275)
A-NonLOCA.LTR-66	LO-178078 dated 01/14/2025 (ML25014A277)
A-NonLOCA.LTR-20	LO-179124 dated 02/04/2025 (ML25035A247)

August 2, 2024 Docket No. 052-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of the NRC's Request for Docketing of Resolved Audit Responses

REFERENCE: NRC Letter to NuScale Power entitled, "Audit Status – The Staff Review of the NuScale Power, LLC Standard Design Approval Application – NuScale US460," dated May 24, 2024 (ML24138A200)

The purpose of this letter is to provide NuScale's response to the NRC's request to docket certain audit responses, noted in the reference above. The responses to the individual audit items are provided in the attachments.

This letter contains NuScale's responses to the following audit questions from the NRC:

Chapter 3

1. Information related to FSAR Section 3.3, "Wind and Tornado Loadings."
   Six Items:
   - A-3.3-1 (nonproprietary)
   - A-3.3.1.1-1 (nonproprietary)
   - A-3.3-2 (nonproprietary)
   - A-3.3-3 (nonproprietary)
   - A-3.3-4 (nonproprietary)
   - A-3.3-7 (nonproprietary)
2. Information related to FSAR Section 3.4, "Water Level (Flood) Design."
   Eight Items:
   - A-3.4.1-1 (nonproprietary)
   - A-3.4.1-2 (proprietary and nonproprietary) (contains Sensitive Unclassified Nonsafeguards Information)
   - A-3.4.1-7 (nonproprietary)
   - A-3.4.2-4 (nonproprietary)
   - A-3.4.2-5 (nonproprietary)
   - A-3.4.2-6 (nonproprietary)
   - A-3.4.2-9 (nonproprietary)
   - A-3.4.2-10 (proprietary and nonproprietary)

3. Information related to FSAR Section 3.5, “Missile Protection.”
   Three Items:
   - A-3.5.3-4 (nonproprietary)
   - A-3.5.3-6 (nonproprietary)
   - A-3.5.3-7 (nonproprietary)

4. Information related to FSAR Section 3.6, “Protection against Dynamic Effects Associated with Postulated Rupture of Piping.”
   Two Items:
   - A-3.6.1-2 (proprietary and nonproprietary)
   - A-3.6.2-1 (proprietary and nonproprietary)

   Not attached: A-3.6.2-2 (disregard, does not exist)

5. Information related to FSAR Section 3.7, “Seismic Design.”
   Eight Items:
   - A-3.7.2-14 (nonproprietary)
   - A-3.7.2-23 (nonproprietary)
   - A-3.7.2-24 (proprietary and nonproprietary)
   - A-3.7.2-25 (proprietary and nonproprietary)
   - A-3.7.2-27 (proprietary and nonproprietary)
   - A-3.7.2-28 (nonproprietary)
   - A-3.7.2-29 (nonproprietary)
   - A-3.7.3-1 (nonproprietary)

6. Information related to FSAR Section 3.8, “Design of Category I Structures.”
   Three Items:
   - A-3.8.5-9 (proprietary and nonproprietary)
   - A-3.8.5-10 (proprietary and nonproprietary)
   - A-3.8.5-12 (nonproprietary)

7. Information related to FSAR Section 3.9, “Mechanical Systems and Components.”
   Fourteen Items:
   - A-3.9.2-1 (proprietary and nonproprietary)
   - A-3.9.2-5 (proprietary and nonproprietary)
   - A-3.9.2-7 (proprietary and nonproprietary)
   - A-3.9.2-8 (proprietary and nonproprietary)
   - A-3.9.2-13 (nonproprietary)
   - A-3.9.2-18 (proprietary and nonproprietary)
   - A-3.9.2-19 (proprietary and nonproprietary)
   - A-3.9.2-21 (proprietary and nonproprietary)
   - A-3.9.2-22 (proprietary and nonproprietary)
   - A-3.9.2-25 (proprietary and nonproprietary)
   - A-3.9.2-33 (nonproprietary)
   - A-3.9.4-6 (proprietary and nonproprietary)
   - A-3.9.4-7 (nonproprietary)
   - A-3.9.6-15 (nonproprietary)

8. Information related to FSAR Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment.”
   Seven Items:
   - A-3.11-2 (nonproprietary)
   - A-3.11.2.3-1 (nonproprietary)
   - A-3.11-3 (nonproprietary)
   - A-3.11-4 (nonproprietary)
   - A-3.11-5 (proprietary and nonproprietary)
   - A-3.11-6 (nonproprietary)
   - A-3.11.12-1 (nonproprietary)
9. Information related to FSAR Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports.”
   One Item:
   - A-3.12.5.9-2 (proprietary and nonproprietary)
10. Information related to FSAR Section 3C.7, “Documentation.”
    One Item:
    - A-3C.7-1 (nonproprietary)

Chapter 4

1. Information related to FSAR Section 4.3, “Nuclear Design.”
   Three Items:
   - A-4.3-3 (proprietary and nonproprietary)
   - A-4.3-4 (nonproprietary)
   - A-4.3-11 (nonproprietary)
2. Information related to FSAR Section 4.5.1, “Control Rod Drive System Structural Materials.”
   Two Items:
   - A-4.5.1-3 (nonproprietary)
   - A-4.5.1-4 (proprietary and nonproprietary)
3. Information related to FSAR Section 4.5.2, “Reactor Internals and Core Support Structure Materials.”
   One Item:
   - A-4.5.2-3 (nonproprietary)

   Not attached: A-4.5.2-1 (per subsequent audit response)
4. Information related to FSAR Section 4.6, Nuclear Design ”Functional Design of Control Rod Drive System.”
   One Item:
   - A-4.6-3 (nonproprietary)

5. Information related to Licensing Technical Report, “Fluence Calculation Methodology and Results.”
   Four Items:
   - A-4.Fluence.TeR-4 (proprietary and nonproprietary)
   - A-4.Fluence.TeR-5 (proprietary and nonproprietary)
   - A-4.Fluence.TeR-6 (proprietary and nonproprietary)
   - A-4.Fluence.TeR-7 (proprietary and nonproprietary)

Chapter 5

1. Information related to FSAR Section 5.1.3.4, “Reactor Vessel Internals.”
   One Item:
   - A-5.1.3.4-1 (nonproprietary)
2. Information related to FSAR Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection.”
   One Item:
   - A-5.2.5-2 (proprietary and nonproprietary)
3. Information related to FSAR Section 5.2.3.4.2, “Cleaning and Contamination Protection Procedures.”
   One Item:
   - A-5.2.3.4.2-1 (nonproprietary)
4. Information related to FSAR Section 5.3.1.5, “Fracture Toughness.”
   One Item:
   - A-5.3.1.5-1 (nonproprietary)
5. Information related to FSAR Section 5.3.2.3, “Pressurized Thermal Shock.”
   One Item:
   - A-5.3.2.3-1 (nonproprietary)
6. Information related to FSAR Section 5.3.2.4, “Upper Shelf Energy.”
   One Item:
   - A-5.3.2.4-1 (nonproprietary)
7. Information related to FSAR Section 5.4, “Reactor Coolant System Component and Subsystem Design.”
   Four Items:
   - A-5.4-1 (proprietary and nonproprietary)
   - A-5.4-3 (nonproprietary)
   - A-5.4-6 (nonproprietary)
   - A-5.4-8 (nonproprietary)
8. Information related to FSAR Section 5.4.1, “Steam Generators.”
   One Item:
   - A-5.4.1-1 (nonproprietary)

9. Information related to FSAR Section 5.4.1.2, Steam Generators – “System Design.”
   Three Items:
   - A-5.4.1.2-4 (nonproprietary)
   - A-5.4.1.2-5 (nonproprietary)
   - A-5.4.1.2-7 (nonproprietary)
10. Information related to FSAR Section 5.4.1.3, Steam Generators – “Performance Evaluation.”
    One Item:
    - A-5.4.1.3-1 (nonproprietary)
11. Information related to FSAR Section 5.4.1.4, Steam Generators – “Tests and Inspections.”
    Two Items:
    - A-5.4.1.4-1 (nonproprietary)
    - A-5.4.1.4-2 (nonproprietary)
12. Information related to FSAR Section 5.4.3, “Decay Heat Removal System.”
    Two Items:
    - A-5.4.3-1 (proprietary and nonproprietary)
    - A-5.4.3-2 (nonproprietary)
13. Information related to Licensing Technical Report, “Pressure and Temperature Limits Methodology.”
    Six Items:
    - A-5.PTLR-6 (proprietary and nonproprietary)
    - A-5.PTLR-8 (nonproprietary)
    - A-5.PTLR-9 (proprietary and nonproprietary)
    - A-5.PTLR-10 (nonproprietary)
    - A-5.PTLR-14 (nonproprietary)
    - A-5.PTLR-17 (nonproprietary)

Chapter 6

1. Information related to FSAR Section 6.1, “Engineered Safety Feature Materials.”
   Two Items:
   - A-6.1.1-1 (proprietary and nonproprietary)
   - A-6.1.1-8 (nonproprietary)
2. Information related to FSAR Section 6.2, “Containment Systems.”
   One Item:
   - A-6.2-5 (proprietary and nonproprietary)

Chapter 8

1. Information related to FSAR Section 8.3, “Onsite Power Systems.”
   Eight Items:
   - A-EDAS audit meeting Action Item 1 (nonproprietary)
   - A-EDAS audit meeting Action Item 2 (proprietary and nonproprietary)
   - A-EDAS audit meeting Action Item 3 (nonproprietary)
   - A-EDAS audit meeting Action Item 4 (nonproprietary)
   - A-EDAS audit meeting Action Item 5 (nonproprietary)
   - A-EDAS audit meeting Action Item 6 (nonproprietary)
   - A-EDAS audit meeting Action Item 9 (nonproprietary)
   - A-EDAS audit meeting Action Item 14 (nonproprietary)

Chapter 14

1. Information related to FSAR Section 14.2, “Initial Plant Test Program.”
   One Item:
   - A-14.2.12-1 (nonproprietary)
2. Information related to FSAR Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria.”
   One Item:
   - A-14.3.2-1 (nonproprietary)
3. Information related to Part 8, Section 8, “Radioactive Waste Building.”
   Two Items:
   - A-Part 8-2.7-1 (proprietary and nonproprietary)
   - A-Part 8-3.12-5 (nonproprietary)

Chapter 15

1. Information related to FSAR Section 15.0.2, “Review of Transient and Accident Analysis Methods.”
   One Item:
   - A-15.0.5-2 (proprietary and nonproprietary)
2. Information related to FSAR Section 15.2.7, “Loss of Normal Feedwater Flow.”
   One Item:
   - A-15.2.7-1 (proprietary and nonproprietary)
3. Information related to FSAR Section 15.4.3, “Control Rod Misoperation (System Malfunction or Operator Error).”
   Two Items:
   - A-15.4.3-1 (proprietary and nonproprietary)
   - A-15.4.3-2 (proprietary and nonproprietary)

Chapter 16

1. Information related to Generic Technical Specification 4.3, “Fuel Storage.”
   One Item:
   A-16.4.3.1-1 (nonproprietary)
2. Information related to Generic Technical Specification 3.4, “Reactor Coolant System.”
   Three Items:
   A-16.3.4.7-1 (nonproprietary)
   A-16.3.4-1 (nonproprietary)
   A-16.3.4.3-1 (nonproprietary)
3. Information related to FSAR Chapter 16, “Technical Specifications.”
   Three Items:
   A-16-3 (nonproprietary)
   A-16-5 (nonproprietary)
   A-16-7 (nonproprietary)
4. Information related to Generic Technical Specifications, Volume 1, Specifications, 1.1, “Definitions,” and Generic Technical Specifications, Volume 2, Bases.
   Two Items:
   A-16-4 (nonproprietary)
   A-16-10 (nonproprietary)
5. Information related to TR-101310-NP, “US460 Standard Design Approval Technical Specifications Development,” Revision 0, Section 3.0.
   Two Items:
   A-16-6 (nonproprietary)
   A-16-9 (nonproprietary)
6. Information related to Generic Technical Specification 5.5.4, “Steam Generator (SG) Program.”
   One Item:
   A-16.5.5.04-1 (nonproprietary)
7. Information related to Generic Technical Specification 3.0, “Limiting Condition for Operation (LCO) Applicability.”
   Two Items:
   A-16.3.0-1 (nonproprietary)
   A-16.3.0-2 (nonproprietary)
8. Information related to Generic Technical Specification 1.1, “Definitions.”
   Two Items:
   A-16.1.1-2 (nonproprietary)
   A-16.1.1-4 (nonproprietary)
9. Information related to Generic Technical Specification 3.5.3, “Ultimate Heat Sink.”
   One Item:
   A-16.3.5.3-1 (nonproprietary)

10. Information related to Generic Technical Specification 3.3.1, “Module Protection System (MPS) Instrumentation.”
    Four Items:
    - A-16.3.3.1-1 (proprietary and nonproprietary)
    - A-16.3.3.1-2 (nonproprietary)
    - A-16.3.3.1-5 (nonproprietary)
    - A-16.3.3.1-6 (nonproprietary)
11. Information related to Generic Technical Specification 3.5.1, “Emergency Core Cooling System (ECCS).”
    Two Items:
    - A-16.3.5.1-1 (nonproprietary)
    - A-16.3.5.1-2 (nonproprietary)
12. Information related to Generic Technical Specification 3.3, “Instrumentation.”
    One Item:
    - A-16.3.3-1 (nonproprietary)
13. Information related to Generic Technical Specification 5.6.4, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR).”
    Two Items:
    - A-16.5.6.4-1 (nonproprietary)
    - A-16.5.6.4-2 (nonproprietary)
14. Information related to Generic Technical Specification 5.5.6, “Explosive Gas and Storage Tank Radioactivity Monitoring Program.”
    One Item:
    - A-16.5.5.06-1 (nonproprietary)
15. Information related to Generic Technical Specification 3.1.3, “Moderator Temperature Coefficient (MTC).”
    One Item:
    - A-16.3.1.3-1 (nonproprietary)
16. Information related to Generic Technical Specification 3.1.9, “Boron Dilution Control.”
    Three Items:
    - A-16.3.1.9-1 (nonproprietary)
    - A-16.3.1.9-2 (nonproprietary)
    - A-16.3.1.9-3 (nonproprietary)

Chapter 17

1. Information related to FSAR Section 17.4, “Reliability Assurance Program.”
   One Item:
   - A-17.4-11 (nonproprietary)

Chapter 19

1. Information related to FSAR Section 19.1, “Probabilistic Risk Assessment.”
   Eight Items:
   - A-19.1-23 (proprietary and nonproprietary)
   - A-19.1-30 (proprietary and nonproprietary)
   - A-19.1-39 (nonproprietary)
   - A-19.1-41 (nonproprietary)
   - A-19.1-46 (proprietary and nonproprietary)
   - A-19.1-51 (proprietary and nonproprietary)
   - A-19.1-56 (proprietary and nonproprietary)
   - A-19.1-60 (nonproprietary)

2. Information related to FSAR Section 19.2, “Severe Accident Evaluation.”
   One Item:
   - A-19.2-27 (proprietary and nonproprietary)

Non-LOCA

1. Information related to NuScale Topical Report: "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416, Revision 0.
   Ten Items:
   - A-NonLOCA.LTR-4 (nonproprietary)
   - A-NonLOCA.LTR-6 (proprietary and nonproprietary)
   - A-NonLOCA.LTR-7 (proprietary and nonproprietary)
   - A-NonLOCA.LTR-10 (proprietary and nonproprietary)
   - A-NonLOCA.LTR-12 (proprietary and nonproprietary)
   - A-NonLOCA.LTR-13 (proprietary and nonproprietary)
   - A-NonLOCA.LTR-14 (proprietary and nonproprietary)
   - A-NonLOCA.LTR-15 (proprietary and nonproprietary)
   - A-NonLOCA.LTR-16 (nonproprietary)
   - A-NonLOCA.LTR-17 (nonproprietary)

   Not attached: A-NonLOCA.LTR-25 (per subsequent update)

DWO

1. Information related to TR-131981, “Methodology for the Determination of the Onset of Density Wave Oscillations (DWO),” Revision 1.
   Four Items:
   - A-DWO.LTR-33 (proprietary and nonproprietary)
   - A-DWO.LTR-46 (proprietary and nonproprietary)
   - A-DWO.LTR-58 (proprietary and nonproprietary)
   - A-DWO.LTR-64 (proprietary and nonproprietary)

   Not attached: A-DWO.LTR-76 (per subsequent update)

The audit responses specified above are attached to the NRC Electronic Information Exchange submittal. For proprietary audit responses, both proprietary and nonproprietary versions are provided. An attachment includes Sensitive Unclassified Nonsafeguards Information. NuScale requests that the proprietary versions and the Sensitive Unclassified Nonsafeguards Information be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit supports this request. Some responses have also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Thomas Griffith at 541-452-7813 or tgriffith@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 2, 2024.

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Stacy Joseph, Senior Project Manager, NRC

Enclosure 1: Affidavit of Mark W. Shaver, AF-169996

Enclosure 1:

Affidavit of Mark W. Shaver, AF-169996

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NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying response reveals distinguishing aspects about the process by which NuScale developed the Standard Design Approval Application for the US460 design.

NuScale has performed significant research and evaluation to develop a basis for this process and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response entitled NuScale Power, LLC Submittal of the NRC's Request for Docketing of Resolved Audit Responses. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon

the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 08/02/2024.

Mark W. Shaver

Mark W. Shaver

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-4

Receipt Date: 04/24/2023

Question:

The inadvertent decrease in boron concentration calculational method utilizes two boron dilution models: perfect dilution and wave front dilution. Non-LOCA LTR (TR-0516-49416, Revision 4) Section 7.2.16.1 states: "The wave front model is physically conservative because it assumes the maximum amount of reactivity as the diluted slug of water sweeps through the core. This model does not assume any axial blending to ensure that this reactivity insertion rate is maximized. For all other operating modes where boron dilution is allowed and limited mixing exists, a wave front model is used. These mixing models are generally performed as a hand calculation but may be automated via a spreadsheet or other process."

However, there is not a discussion in SDAA Section 15.4.6, or in the Non-LOCA LTR on the validity of these models and their range of applicability. Given natural circulation depends on the differential coolant density and temperature how have these equations been validated for the SDA design which includes an increase in power and flow?

Response:

As described in Section 15.4.6.3.1, two calculational techniques are used to analyze the boron dilution event and provide conservative boron dilution assumptions for the evaluation of both reactivity insertion and loss of shutdown margin. The first technique evaluates the boron dilution by assuming an instantaneous perfect (complete) mixing model and is described further in Section 15.4.6.3.1.1. The second technique evaluates the boron dilution by assuming a slug flow or dilution front (wave front) mixing model and is described further in 15.4.6.3.1.2. Both of these sections provide the mathematical equations (Eqs. 15.4-1 through 15.4-4) and define the input parameters used in the two techniques. These equations are derived from basic first principles. The equations are identical to those used in the design certification application (DCA)

previously approved by the NRC. The NRC safety evaluation report (SER) for the boron dilution event does not identify any limitations associated with these basic equations. The differences between the DCA design (i.e., the US600) and the SDAA design (i.e., the US460) were reviewed to determine if any would affect the applicability of these basic equations. The only difference identified for potential impact was the additional riser holes present in the US460 design. The additional riser holes allow for additional cross-flow between the riser and downcomer which would promote mixing. The use of the different two calculational techniques, which are bounding of the mixing extremes, ensures that there is no impact from the riser holes. Therefore, there are no differences between the US600 and US460 designs that affect the applicability of the equations in the two calculational techniques.

The non-LOCA topical report, TR-0516-49416, also provides mathematical equations (Eqs. 7-1 through 7-3) associated with the two calculation techniques. The equations are the same as those presented in the FSAR, although rearranged in some cases and with minor differences in nomenclature. The non-LOCA topical report was previously approved by the NRC and the associated SER does not identify any limitations associated with these basic equations. The equations in the current revision (Rev. 4) are identical to those in the previously approved revision (Rev. 3). As stated before, there are no differences between the US600 and US460 designs that affect the applicability of these basic equations.

The power differences between the US600 and US460 do affect operating temperatures and natural circulation flow as discussed in this audit question. These differences affect the input used when solving Eqs. 15.4-1 through 15.4-4. For the results reported in SDAA Section 15.4.6, the appropriate value for each input parameter has been selected based on the US460 design.

Regarding when each calculational technique should be used, the non-LOCA topical report Section 7.2.16.1 identifies that both techniques are used in Mode 1. The non-LOCA topical report Section 7.2.16.1 further states that the perfect mixing model provides a slower reactivity insertion rate, delaying detection, potentially allowing further loss of shutdown margin, while the wave front model is physically conservative because it assumes the maximum amount of reactivity as the diluted slug of water sweeps through the core. In Mode 1 at 25% power and above, both techniques are used to calculate the reactivity insertion rates, but isolation times and shutdown margin remaining at the time of isolation are calculated with the perfect mixing model. In Mode 1 at hot zero power, both techniques are used to calculate the reactivity insertion rates, but isolation times and shutdown margin remaining at the time of isolation are calculated with the wave front model. In Modes 2 and 3, the wave front model is used to determine isolation times and shutdown margin remaining at the time of isolation. The SER for

the previously approved non-LOCA topical report revision (Rev. 3) accepted the use of the two techniques for the various mode and power conditions as described.

When comparing the US600 and US460 designs, the temperatures associated with the mode definitions have changed. In addition, the temperature and natural circulation flows at each of the power levels within Mode 1 have changed. However, these differences between the designs are small and do not affect which technique should be used for the various mode and power conditions. The higher flow rates at each power level for 25% power and above for the US460 design compared to the US600 would promote more mixing; therefore, use of the perfect mixing model remains valid. For Mode 1 hot zero power and Modes 2 and 3, the flow rate is comparably low in both designs; therefore, use of the wave front model is valid. For these reasons, the use of the calculational techniques for the various mode and power conditions in the current revision of the non-LOCA topical report (Rev. 4) are identical to those in the previously approved revision (Rev. 3).

The SDAA results in Tables 15.4-13 and 15.4-14 for Mode 1 hot full power and hot zero power, respectively, are reported consistent with the use of two techniques as described in the non-LOCA topical report and summarized above. Similarly, Tables 15.4-15 and 15.4-16 for Modes 2 and 3, respectively, are reported consistent with the use of just the one technique as described in the non-LOCA topical report and summarized above.

In conclusion, the mathematical equations used in the two calculational techniques for the boron dilution analysis are not impacted by design differences between the US600 and the US460. The inputs used to solve the equations are design specific and the appropriate inputs are used for the US460 design.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-6

Receipt Date: 07/24/2023

Question:

While reviewing {{ }}$^{2(a),(c)}$ the NRC staff noticed {{ }}$^{2(a),(c)}$ the NRELAP Version 1.6 Theory Manual (Revision 10) may have an error and appears to be inconsistent {{

}}$^{2(a),(c)}$ The staff can support a clarification call to discuss {{ }}$^{2(a),(c)}$

Response:

Reference 29 of the Non-LOCA topical report, TR-0516-49416-P, Revision 4, is for the NRELAP5 Version 1.6 Theory Manual (SwUM-0304-17023, Revision 10). {{

}}$^{2(a),(c)}$

{{

}}2(a),(c)

No changes to the non-LOCA topical report or SDAA FSAR are required. The inconsistency in the NRELAP5 theory manual has been entered into the NuScale Corrective Action Program (CAP). The revision of the NRELAP5 theory manual to correct the inconsistency will be performed under the NuScale CAP.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-7

Receipt Date: 08/21/2023

Question:

In the non-LOCA TR EM and RSI-17 response, ER-126485 was provided in the eRR as a reference. DHRS scaling analysis was described at a high level. In order to understand the assumptions of derivations, make available for staff audit the calculation notes containing the detailed derivations of the DHRS PI groups and calculations of the final PI group numbers presented in the LTR and the analysis report.

Response:

ER-126485 Rev. 0, previously provided in the electronic reading room (eRR), documents scaling analysis performed for the decay heat removal system (DHRS) testing at the NIST-2 facility. Additional information regarding the DHRS testing at the NIST-2 facility, including scaling analysis, is found in {{

}}$^{2(a),(c),ECI}$

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-10

Receipt Date: 09/18/2023

Question:

RSI-15 (Page 458/723 RSI responses), states “Therefore, the oscillations are expected to have minimal impact on the overall SG/DHRS heat transfer performance. This conclusion of minimal impact is consistent with the results of the NIST-2 SG/DHRS testing. The testing results for Run 2 show that measured heat transfer is reasonably predicted by NRELAP5 {{

}}2(a),(c)

Response:

As identified in this audit question, the response to RSI-15, states “Therefore, the oscillations are expected to have minimal impact on the overall SG/DHRS heat transfer performance. This conclusion of minimal impact is consistent with the results of the NIST-2 SG/DHRS testing. The testing results for Run 2 show that measured heat transfer is reasonably predicted by NRELAP5 {{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c) The non-LOCA topical report is updated to include revised assessment figures that more accurately demonstrate the ability of NRELAP5 to predict the SG/DHRS heat transfer in the NIST-2 Runs 1, 2, and 3 tests. In developing the non-LOCA NIST-2 test matrix, Run 1 was scaled for {{

}}2(a),(c) the NPM design. The smaller and larger orifices installed in Runs 2 and 3, respectively, provide a parametric set of test conditions for the purpose of code validation.
{{

}}2(a),(c) In conclusion, the figures in the non-LOCA topical report are corrected as shown in the mark-ups and further justification discussion in the topical report is not required.

{{

}}2(a),(b),(c),ECI

Figure 1: Run 2 NRELAP5 to Data Comparison
Decay Heat Removal System Condensate Flow Rate

{{

}}2(a),(b),(c),ECI

Markups of the affected changes, as described in the response, are provided below:

{{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-10

Figure 5-222 and Figure 5-223 show the DHRS heat exchanger and SG power comparison for the long-term and short-term respectively. Figure 5-224 and Figure 5-225 show the SG-DHRS loop flow comparison for the long-term and short-term respectively. {{

}}2(a),(c) Figure 5-224a and Figure 5-225a show comparison of NRELAP5 results with the unaveraged flow meter data and NRELAP5 predicts the test data with reasonable to excellent agreement. {{

}}2(a),(c) ~~Overall~~as shown in Figure 5-222 and Figure 5-223, the NRELAP5 calculation has reasonable to excellent agreement with the test data.

Figure 5-226 shows the code-to-data comparison for the PZR pressure and the secondary side pressure near the steam drum. Before the PZR heater is tripped, both data and code maintain the PZR pressure at approximately {{

}}2(a),(c)

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Figure 5-229 shows the code-to-data comparison for the DHRS heat exchanger inlet and outlet temperatures. Figure 5-230 shows the code-to-data comparison for the SG secondary side inlet and outlet temperatures. The SG secondary side outlet temperature and DHRS heat exchanger inlet temperatures are both close to the loop saturation temperature (Figure 5-231). The code predictions of these two temperatures both have reasonable to excellent agreement with the data, similar to the secondary side pressure.

From Figure 5-229 and Figure 5-230, the predicted DHRS heat exchanger outlet temperature and SG secondary side inlet temperature are {{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-10

Figure 5-232 shows the code-to-data comparison of the RCS primary temperatures at two locations: at the SG primary side's inlet (upper plenum) and outlet (mid downcomer). {{

}}2(a),(c)

Figure 5-233 shows the code-to-data comparison of the PZR level. Figure 5-234 shows the code-to-data comparison of the RPV level. {{

}}2(a),(c)

Figure 5-235 shows the code-to-data comparison of the RCS primary flow. The prediction has excellent agreement with the data during the transient. {{

}}2(a),(c)

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Audit Question A-NonLOCA.LTR-10

Figure 5-222 NIST-2 Run 1 steam generator and decay heat removal system heat exchanger power comparison

{{

}}2(a),(b),(c),ECI

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Audit Question A-NonLOCA.LTR-10

Figure 5-223 NIST-2 Run 1 steam generator and decay heat removal system heat exchanger power comparison - short-term

{{

}}2(a),(b),(c),ECI

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Audit Question A-NonLOCA.LTR-10

Figure 5-224a NIST-2 Run 1 decay heat removal condensate line flow comparison (without time average)

{{

}}2(a),(b),(c),ECI

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Audit Question A-NonLOCA.LTR-10

Figure 5-225a NIST-2 Run 1 decay heat removal condensate line flow comparison (without time average) - short-term

{{

}}2(a),(b),(c),ECI

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- {{

}}2(a).(b).(c).ECI

The valve configuration and execution of the Run 2 test was the same as described for Run 1 in Section 5.3.7.6. The sequence of events of the Run 2 test is shown in Table 5-20.

RSI 15

Table 5-20 NIST-2 Run 2 sequence of events

	{{		
		}}(a).(b).(c).ECI	

RSI 15

5.3.7.9 Run 2 Test Results

RSI 15

Figure 5-243 through Figure 5-264 show the code-to-data comparison of the key parameters for Run 2.

RSI 15

Figure 5-243 shows the core power. Figure 5-244 shows the SG/DHRS active loop inventory. {{

}}2(a).(c) The active loop inventory is discussed in greater detail for Run 1 in Section 5.3.7.7.

Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-245 and Figure 5-246 show the DHRS heat exchanger and SG power comparison for the long-term and short-term, respectively. Figure 5-247 and Figure 5-248 show the SG/DHRS loop flow comparison for the long-term and short-term, respectively. As described in Section 5.3.7.7 for Run 1, {{

}}2(a).(c)

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{{ }}2(a),(c) The results are consistent with those observed in Run 1 in Section 5.3.7.7.

Audit Question A-NonLOCA.LTR-10
RSI 15

{{

}}2(a),(c)

Figure 5-249 compares the DHRS loop flow data without time average with the NRELAP5 results. Both are DHRS condensate line flow rates. {{

RSI 15

}}2(a),(c)

Figure 5-250 shows the differential pressure across the DHRS condensate line flow meter. {{

RSI 15

Audit Question A-NonLOCA.LTR-10
RSI 15

}}2(a),(c)

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{{

}}2(a),(c)

RSI 15

Figure 5-251 shows the code-to-data comparison for the pressurizer pressure and secondary side pressure near the steam drum. {{

}}2(a),(c) Overall the secondary side pressure has reasonable to excellent agreement with the test data.

RSI 15

Figure 5-252 shows the code-to-data comparison for the DHRS heat exchanger level and SG level. {{

}}2(a),(c) Overall, the predicted DHRS heat exchanger and SG levels have reasonable to excellent agreement with the test data.

RSI 15

Figure 5-253 shows the code-to-data comparison for the DHRS heat exchanger inlet and outlet temperatures. Figure 5-254 shows the code-to-data comparison for the SG secondary side inlet and outlet temperatures. {{

}}2(a),(c) The code predictions of these two temperatures both have reasonable to excellent agreement with the data, similar to the secondary side pressure.

RSI 15

From Figure 5-253 and Figure 5-254, {{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-256 shows the code-to-data comparison of the RCS primary temperatures. {{

}}2(a),(c)

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{{

}}2(a),(c)

RSI 15

Figure 5-257 shows the code-to-data comparison of the PZR level. Figure 5-258 shows the code-to-data comparison of the RPV level. {{

}}2(a),(c)

RSI 15

Figure 5-259 shows the code-to-data comparison of the RCS primary flow. The prediction has excellent agreement with the data during the transient. {{

}}2(a),(c)

RSI 15

Figure 5-260 shows the code-to-data comparison of the CPV level. The prediction has excellent agreement with data {{

}}2(a),(c)

RSI 15

Figure 5-263 shows the code-to-data comparison of the DHRS steam line differential pressure. Figure 5-264 shows the comparison of the DHRS steam line orifice differential pressure. Reasonable to excellent agreements are shown {{

}}2(a),(c)

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Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-245 NIST-2 Run 2 steam generator and decay heat removal system heat exchanger power comparison

{{

}}2(a),(b),(c),ECI

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Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-246 NIST-2 Run 2 steam generator and decay heat removal system heat exchanger power comparison - short-term

{{

}}2(a),(b),(c),ECI

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- {{

}}2(a).(b).(c).ECI

RSI 15

The valve configuration and execution of the Run 3 test was the same as described for Run 1 in Section 5.3.7.6. The sequence of events of the Run 3 test is shown in Table 5-21.

RSI 15

Table 5-21 NIST-2 Run 3 sequence of events

	{{		
		}}(a).(b).(c).ECI	

RSI 15

5.3.7.11 Run 3 Test Results

RSI 15

Figure 5-265 through Figure 5-285 show the code-to-data comparison of the key parameters for Run 3.

RSI 15

Figure 5-265 shows the core power. Figure 5-266 shows the SG/DHRS active loop inventory. {{

}}2(a).(c) The active loop inventory is discussed in greater detail for Run 1 in Section 5.3.7.7.

Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-267 and Figure 5-268 show the DHRS heat exchanger and SG power comparison for the long-term and short-term, respectively. Figure 5-269 and Figure 5-270 show the SG/DHRS loop flow comparison for the long-term and short-term, respectively. As described in Section 5.3.7.7 for Run 1, {{

}}2(a).(c)

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{{ }}2(a),(c) The results are consistent with those observed in Run 1 in Section 5.3.7.7.

Audit Question A-NonLOCA.LTR-10
RSI 15

{{

}}2(a),(c) Figure 5-271 compares the DHRS condensate line flow data without time average with the NRELAP5 results. {{

}}2(a),(c)

RSI 15

Figure 5-272 shows the code-to-data comparison for the pressurizer pressure and secondary side pressure near the steam drum. {{

}}2(a),(c) The predicted secondary side pressure shows excellent agreement with the test data for the entire transient, including the peak pressure.

RSI 15

Figure 5-273 shows the code-to-data comparison for the DHRS heat exchanger level and SG level. {{

}}2(a),(c) Overall the predicted DHRS heat exchanger and SG levels have reasonable to excellent agreement with the test data.

RSI 15

Figure 5-274 shows the code-to-data comparison for the DHRS heat exchanger inlet and outlet temperatures. Figure 5-275 shows the code-to-data comparison for the SG secondary side inlet and outlet temperatures. {{

}}2(a),(c) The code predictions of these two temperatures both have reasonable to excellent agreement with the data, similar to the secondary side pressure.

RSI 15

From Figure 5-274 and Figure 5-275, {{

}}2(a),(c)

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{{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-277 shows the code-to-data comparison of the RCS primary temperatures. {{

}}2(a),(c)

RSI 15

Figure 5-278 shows the code-to-data comparison of the PZR level. Figure 5-279 shows the code-to-data comparison of the RPV level. {{ }}2(a),(c) The predictions have reasonable agreement with test data.

RSI 15

Figure 5-280 shows the code-to-data comparison of the RCS primary flow. The prediction has excellent agreement with the data during the transient. {{

}}2(a),(c)

RSI 15

Figure 5-281 shows the code-to-data comparison of the CPV level. The prediction has excellent agreement with data {{

}}2(a),(c)

RSI 15

Figure 5-284 shows the code-to-data comparison of the DHRS steam line differential pressure. Figure 5-285 shows the comparison of the DHRS steam line orifice differential pressure. Predicted differential pressures have reasonable to excellent agreement with test data.

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Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-267 NIST-2 Run 3 steam generator and decay heat removal system heat exchanger power comparison

{{

}}2(a),(b),(c),ECI

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Audit Question A-NonLOCA.LTR-10
RSI 15

Figure 5-268 NIST-2 Run 3 steam generator and decay heat removal system heat exchanger power comparison - short-term

{{

}}2(a),(b),(c),ECI

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-12

Receipt Date: 11/20/2023

Question:

Provide the following documents in eRR related to NIST-2 testing:
{{

}}2(a),(c),ECI

Response:

The audit question requests the following six items, identified by the question number and text as being related to NIST-2 testing for the non-LOCA topical report, be placed in the electronic reading room (eRR):

{{

}}2(a),(c),ECI

For items 1 through 4, the documents are assessments for testing that was performed to support the validation of NRELAP5 for the LOCA topical report (i.e., not the non-LOCA topical report). The tests were performed at the NIST-1 facility (i.e., they are not NIST-2 tests). In addition, the validation supported the LOCA topical report revision (i.e., TR-0516-49422-P-A, Rev. 2) that has already been approved by the NRC during the review of the design certification application (DCA) for the US600. The validation is still included as part of the LOCA topical report revision currently under review by the NRC (i.e., Rev. 3) for the standard design approval application (SDAA) for the US460. Therefore, items 1 through 4 are not provided in the non-LOCA eRR for NRC audit at this time.

Items 5 and 6 are the same document; EC-127301 is the NRELAP5 assessment against the NIST-2 non-LOCA {{ }}$^{2(a),(c)}$ The revision of EC-127301 requested in this audit question (i.e., Rev. 0), is the revision that was available at the time the SDAA was submitted. However, Rev. 0 contains information for only Run 1 of the NIST-2 non-LOCA testing. Other test runs, such as Runs 2 and 3 that have been included in markups to the non-LOCA topical report, are documented in later revisions of EC-127301. Therefore, the current revision of EC-127301 (i.e., Rev. 2) is provided in the non-LOCA eRR for NRC audit instead of the requested revision (i.e., Rev. 0).

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-13

Receipt Date: 12/04/2023

Question:

During review of the loss of feedwater event (described in EC-116269 R0), {{

}}$^{2(a),(c),ECI}$ flow is terminated much earlier and long before FWIV closure. Terminating the feedwater flow completely before the FWIV closes in the case results in DHRS inventory that is more optimum for RCS cooling than if feedwater flow were to continue {{

}}$^{2(a),(c),ECI}$

This is more important for transients where only 1 train of DHRS is available. The DHRS performance is closely tied to loop inventory, and it is not clear if NuScale has accounted for this deterioration in DHRS performance.

Please evaluate and address all applicable LOCA and non-LOCA analyses to confirm that adequate modeling has been applied for DHRS performance in this regard.

Response:

EC-116269, “NPM-20 Loss of Normal Feedwater Transient Analysis,” Revision 0, was provided to the NRC in the electronic reading room (eRR) as part of the response to audit question A-15.2.7-2. EC-116269 provides the NRELAP5 analysis supporting final safety analysis report (FSAR) Section 15.2.7 for the US460 standard design approval application (SDAA). The initiating event being modeled is a loss of feedwater flow. {{

}}2(a),(c),ECI

The NRC staff observation that feedwater flow is terminated much earlier than feedwater isolation valves (FWIVs) close is correct. However, the reason feedwater flow is terminated early is that termination of feedwater flow is the modeled initiating event. {{

}}2(a),(c),ECI

Therefore, feedwater flow is zero before the secondary system valves begin to change positions.

The NRC observation that potential continuing feedwater flow until FWIVs close will change the secondary side inventory and therefore impact DHRS performance is correct. However, the

purpose {{ }}2(a),(c),ECI is to maximize RCS pressure, not to study the effects of DHRS. None of the cases associated with EC-116269 are informative for studying the effects of high secondary inventory on DHRS behavior because of the loss of feedwater flow as the initiating event; other Chapter 15 initiating events are more appropriate for investigating this concern.

The increase in feedwater flow event in FSAR Section 15.1.2 is an event where secondary side inventory increases. Due to the potential impact of high inventory on DHRS performance, the increase in feedwater flow analysis explicitly addresses high inventory cases. As described in FSAR Section 15.1.2.3.3:

> "Steam generator overfill cases are evaluated with initial conditions including primary flow and RCS temperature biased to maximize SG level. Boundary conditions such as pool temperature are biased to minimize DHRS heat removal, high decay heat is assumed, the maximum FW pump curve is modeled, the maximum containment isolation valve closure time is assumed, and single failure of one FWIV to close is assumed. The SG level calculated during an increase in FW flow demonstrates the SG does not overfill as shown in Figure 15.1-18. Evaluation of cases biased for SG overfill demonstrate adequate DHRS heat removal capability is available under these conditions as shown in Figure 15.1-17."

The analysis supporting FSAR Section 15.1.2 is found in EC-0000-8328, "NPM-20 Increase in Feedwater Flow Analysis," Revision 0. EC-0000-8328 was provided to the NRC in the eRR as part of the response to audit question A-15.1.1-1. {{

}}2(a),(c),ECI

In addition to considering the impact of inventory on DHRS performance for events like the increase in feedwater flow event, FSAR Chapter 15 also includes events that disable an entire train of DHRS. The DHRS is designed to be able to perform its function even in the event of a single failure. As a result, the single failure assumptions in most Chapter 15 analyses can impact DHRS performance (e.g., loop inventory is altered by assuming single failure of an FWIV that delays loop isolation until the feedwater regulating valve closes), but do not disable an entire train. However, in the event of a steam system piping failure in FSAR Section 15.1.5 or a feedwater system pipe break in FSAR Section 15.2.8, the initiating event is a break that, depending on the location, can disable an entire DHRS train. For example, FSAR Section 15.1.5.2 states:

> "The DHRS is a safety-related system credited to actuate and mitigate the effects of this transient. The redundancy and passive nature of the DHRS ensure the system performs its intended function despite a single failure. The operation of a DHRS train is challenged when an SLB is located on the steam line downstream of the MSIV, but upstream of the secondary MSIV, with a single failure of the MSIV, or when an SLB occurs inside containment. These scenarios prevent one train of the DHRS from functioning by preventing isolation of one SG train."

Similar to the increase in feedwater flow event, the steam system piping failure calculation methodology uses a maximum feedwater flow rate {{

}}2(a),(c),ECI

NuScale agrees that DHRS performance is related to loop inventory and has therefore accounted for this effect in the analysis of relevant initiating events. Similarly, NuScale agrees that operation of DHRS with only a single train is more limiting for decay heat removal than when two trains are available. Although a single failure alone does not disable an entire train of DHRS, the safety analysis in Chapter 15 properly considers initiating events that can directly, or in combination with a single failure, result in failure of an entire train of DHRS. NuScale confirms that adequate modeling has been applied in the LOCA and non-LOCA transients for DHRS performance in this regard.

Additionally, consistent with the non-LOCA methodology previously approved by the NRC during the design certification application (DCA), TR-0516-49416-P-A Revision 3, no penalties, biases, or uncertainties are applied to the DHRS heat transfer. The sensitivity studies performed during the DCA review that considered biases on the heat transfer demonstrated no impact on the figures of merit. The associated safety evaluation report (SER) included a condition and limitation to confirm the sensitivity study results if the design changed in the future. In lieu of additional sensitivity studies, the current non-LOCA methodology topical report includes additional DHRS testing scaled to the SDAA design (i.e., addresses the design changes from DCA). The testing results demonstrate that NRELAP5 can predict DHRS heat transfer for the SDAA design with reasonable to excellent agreement, and therefore without the need to apply a bias or uncertainty in the NPM plant calculations.

{{

}}2(a),(c),ECI

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-14

Receipt Date: 12/04/2023

Question:

Provide TR markups specifying whether RETRAN-3D calculations described in TR section 5.3.4 use point kinetics or spatial kinetics. TR section 5.3.4 states that a code-to-code benchmark was performed to compare the NRELAP5 and RETRAN-3D responses to a range of reactivity insertion conditions in an NPM. Without specifying whether RETRAN-3D calculations used point kinetics or spatial kinetics, it is unclear whether the code-to-code benchmark provides evidence that point kinetics models are appropriate for these transients in the NPM-20. Because of the increased power in the NPM-20, the spatial dynamic nuclear response (multi-group spectrum effects, particularly near the core periphery and core locations of large neutron flux gradients) to the various reactivity events (e.g., control rod mis-operation event), may prevent a point kinetics model from determining the changing power and core kinetics. If the comparison is based on a RETRAN-3D point kinetics model, provide evidence that power and reactivity response are adequately modeled for transients described in TR Section 5.3.4.

Response:

The purpose of the non-LOCA topical report Section 5.3.4 is to benchmark the NRELAP5 point kinetics model to RETRAN-3D during reactivity transient events. A point kinetics model is used in both codes as already stated in Section 5.3.4.7 of the topical report: “This is important because the main purpose of the benchmark is to compare the point kinetics model response between NRELAP5 and RETRAN-3D.” Therefore, no topical report markups are required.

The code-to-code benchmarking was previously reviewed by the NRC as indicated in Section 3.5.3.4 of the safety evaluation report (SER) for the non-LOCA topical report, TR-0516-49416-P-A, Revision 3. {{ }}$^{2(a),(c)}$ the benchmark is not intended to be a validation of the NuScale Power Module (NPM) response itself, but one of many independent forms of evidence of the adequacy

of the point kinetics model in NRELAP5. Therefore, the benchmarking analysis was not re-performed for the NPM-20 design. As stated in the SER, “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.” For these reasons, non-LOCA topical report Section 5.3.4, and its subsections, were not revised in the Revision 4 submitted to support the standard design approval application (SDAA) and currently under review.

In addition, the point kinetics model used in NRELAP5 uses kinetics input calculated from the more detailed SIMULATE5 code as described in Chapter 4 of the final safety analysis report (FSAR). The kinetics inputs are then conservatively applied in NRELAP5 to ensure the overall conservatism of the NRELAP5 calculated power and reactivity responses. For more severe reactivity insertion events (i.e., rod ejections), the non-LOCA methodology is not used. In general, the limiting cases for reactivity events are driven by the module protection system (MPS) limits rather than by the worst-case combination of kinetics conditions.

The increased power in the NPM-20 does not significantly change the power distributions. In an NPM, like existing PWRs, core power gradients exists in all assemblies in the core, most acutely for assemblies on the periphery. The gradients are due to geometric buckling (i.e., radial leakage). Such gradients occur regardless of core design, though lower leakage cores will tend to exhibit lower power gradients than higher leakage cores. In the overall Chapter 15 safety analysis methodology, including the non-LOCA methodology, each unique core design must be specifically analyzed and compared to the bounding limits developed for the NPM. Thus, bounding evaluations of all power gradients, which inherently includes the corresponding depletion history, are evaluated in each application of this method. Because SIMULATE5 has been shown to reliability predict critical conditions, power distributions, and depletion, it reliably predicts the spatial dynamic nuclear response, including from important phenomena such as power and depletion gradients. Because the NPM-20 design does not significantly change the power distributions, the NRELAP5 validation discussed in non-LOCA topical report Section 5.3.4 applies to the NPM-20 as well.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-15

Receipt Date: 12/26/2023

Question:

This audit item is related to the following audit items: A-15.0.2-1, NonLOCA.LTR-12.
{{

}}2(a),(b),(c),ECI

Response:

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}}2(a),(b),(c),ECI

{{

}}2(a),(b),(c),ECI

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-16

Receipt Date: 01/15/2024

Question:

Please provide the evaluation used to determine that the PIRT developed for previous NPM is applicable and appropriate for the current submitted NPM.

Response:

ER-126485, Rev. 0, “NRELAP5 Applicability to NPM-20 Non-LOCA Analysis,” Section 4.1 provides the evaluation that the PIRT developed for the previous NPM design (i.e., NPM-160) is applicable to the current NPM design (i.e., NPM-20). ER-126485 has previously been provided in the non-LOCA area of the electronic reading room (eRR) in response to other audit questions. As stated in Section 4.1.2 of ER-126485, the PIRT developed for NPM-160 is not required to be updated for NPM-20 because there are no new additional phenomena identified for NPM-20.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-17

Receipt Date: 02/26/2024

Question:

Please make available in eRR: Non-LOCA Transient Analysis Methodology Report ER-0000-2379, Rev 1.

Response:

ER-0000-2379, Rev. 1, “Non-LOCA Transient Analysis Methodology Report,” is provided in the electronic reading room (eRR). Note that this document was issued in 2018 prior to the development of the Standard Design Approval Application (SDAA) for the US460 design.

No changes to the SDAA are necessary.

December 13, 2024 Docket No. 052-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of the NRC's Request for Docketing of Resolved Audit Responses

REFERENCE: NRC Letter to NuScale Power entitled, "Audit Status – The Staff Review of the NuScale Power, LLC Standard Design Approval Application – NuScale US460," Dated October 24, 2024 (ML24269A008)

The purpose of this letter is to provide NuScale's response to the NRC's request to docket certain audit responses, noted in the reference above. The responses to the individual audit items are provided in the attachments.

This letter contains NuScale's responses to the following audit questions from the NRC:

Chapter 15

Information related to FSAR Section 15.6.5 "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary."
One Item:

A-15.6.5-1 (Proprietary and Nonproprietary)

Non-LOCA

Information related to NuScale Topical Report: "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416, Revision 0.
Thirty-one Items:

A-NonLOCA.LTR-2 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-5 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-5S (Proprietary and Nonproprietary)
A-NonLOCA.LTR-5S2 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-8 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-8S (Proprietary and Nonproprietary)
A-NonLOCA.LTR-22 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-23 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-24 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-25 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-26 (Nonproprietary)
A-NonLOCA.LTR-28 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-28S (Proprietary and Nonproprietary)

A-NonLOCA.LTR-29 (Nonproprietary)
A-NonLOCA.LTR-30 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-33 (Nonproprietary)
A-NonLOCA.LTR-34 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-35 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-36 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-37 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-38 (Nonproprietary)
A-NonLOCA.LTR-39 (Nonproprietary)
A-NonLOCA.LTR-40 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-41 (Nonproprietary)
A-NonLOCA.LTR-43 (Nonproprietary)
A-NonLOCA.LTR-44 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-44S (Proprietary and Nonproprietary)
A-NonLOCA.LTR-45 (Nonproprietary)
A-NonLOCA.LTR-49 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-54 (Nonproprietary)
A-NonLOCA.LTR-57 (Nonproprietary)
A-NonLOCA.LTR-58 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-59 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-59S (Proprietary and Nonproprietary)
A-NonLOCA.LTR-59S2 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-61 (Nonproprietary)
A-NonLOCA.LTR-62 (Proprietary and Nonproprietary)
A-NonLOCA.LTR-68 (Nonproprietary)

The audit items specified above have been attached to the NRC EIE submittal. The responses that are labeled as proprietary include both proprietary and nonproprietary versions. The enclosed affidavit supports this request. Some responses have also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Thomas Griffith at 541-452-7813 or tgriffith@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 13, 2024.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Stacy Jospeh, Senior Project Manager, NRC
Ricky Vivanco, Project Manager, NRC

Enclosure 1: Affidavit of Mark W. Shaver, AF-176319

Enclosure 1:

Affidavit of Mark W. Shaver, AF-176319

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NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying response reveals distinguishing aspects about the process by which NuScale developed the Standard Design Approval Application for the US460 design.

NuScale has performed significant research and evaluation to develop a basis for this process and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed response entitled NuScale Power, LLC Submittal of the NRC's Request for Docketing of Resolved Audit Responses. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 13, 2024.

Mark W. Shaver

Mark W. Shaver

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-2

Receipt Date: 03/27/2023

Question:

The non-LOCA LTR is missing sufficient detail regarding the CFD simulations performed for primary side heat transfer in Section 5.3.6. Examples of information that will be needed for the review include boundary conditions, mesh sensitivity and parameters, initial conditions, and all other physical models used as inputs aside from the heat transfer models that are provided.

Response:

The requested detailed information regarding the CFD simulations performed for Section 5.3.6 is found in the supporting calculation EC-109948, Revision 0. The calculation EC-109948 has been provided for review in the Electronic Reading Room.

Section 5.3.6 of the topical report is revised to include additional information summarized from EC-109948.

Markups of the affected changes, as described in the response, are provided below:

5.3.6 Heat Transfer Correlation Comparison

5.3.6.1 Background

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}}2(a),(c)

5.3.6.2 Computational Fluid Dynamics Calculations

{{

A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

}}2(a),(c),ECI

A-NonLOCA.LTR-2

5.3.6.2.1 Model 1 Analysis and Results

A-NonLOCA.LTR-2

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}}2(a),(c),ECI

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A-NonLOCA.LTR-2

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}}2(a),(c),ECI

A-NonLOCA.LTR-2

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A-NonLOCA.LTR-2

}}2(a),(c),ECI

A-NonLOCA.LTR-2

Table 5-16 Comparison of correlation to CFD results - CFD model 1

	{{					
					}}2(a),(c),ECI	

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Table 5-16 Comparison of correlation to CFD results - CFD model 1 (Continued)

	{{					
					}}2(a),(c),ECI	

A-NonLOCA.LTR-2

5.3.6.2.2 Model 2 Analysis and Results

A-NonLOCA.LTR-2

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A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

}}2(a),(c),ECI

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A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

}}2(a).(c).ECI

A-NonLOCA.LTR-2

Table 5-17 Comparison of correlation to CFD results - CFD model 2

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		-				
					}}2(a).(c).ECI	

A-NonLOCA.LTR-2

5.3.6.2.3 Model 3 Analysis and Results

A-NonLOCA.LTR-2

{{

A-NonLOCA.LTR-2

}}2(a).(c).ECI

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A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

}}2(a),(c),ECI

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A-NonLOCA.LTR-2

Table 5-18 ~~Comparison of correlation to CFD results - CFD model 2~~

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~~}}2(a),(c)~~

A-NonLOCA.LTR-2

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~~}}2(a),(c)~~

A-NonLOCA.LTR-2

5.3.6.2.4 Computational Fluid Dynamics Conclusions

A-NonLOCA.LTR-2

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}}2(a),(c),ECI

5.3.6.3 Comparison of Heat Transfer Correlations within NRELAP5

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}}2(a),(c)

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-5

Receipt Date: 04/24/2023

Question:

The methodology of screening rod drop cases needs additional description. Section 7.2.15.1 states “For the smaller group of cases that do not result in the early reactor trip {{

}}$^{2(a),(c)}$ However, because power overshoot may occur during the event, these cases are bounded instead by comparison to the withdrawal of a single CRA. {{

}}$^{2(a),(c)}$ Describe what specific parameters in the rod drop and single rod withdrawal cases are being compared, and how the single rod withdrawal analytical limits are derived from the cases analyzed. {{

}}$^{2(a),(c)}$ Please place proposed markups to the LTR in the eRR.

Response:

{{

}}$^{2(a),(c)}$

{{

}}2(a),(c)

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}}2(a),(c)

Section 7.2.15.1 of the non-LOCA topical report, TR-0516-49416, is revised to provide clarifications consistent with the above response.

Markups of the affected changes, as described in the response, are provided below:

peaking factors for the outer row of fuel assemblies. Sensitivity studies are performed on a variety of parameters, as necessary, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

The maximum fuel centerline temperature typically occurs at event initiation for those event scenarios with an immediate reactor trip. If the event scenario has a return to power, the maximum fuel centerline temperature is typically bounded by the fuel centerline temperature at HFP because the associated power peak is less than full power.

Audit Question A-NonLOCA.LTR-5

As an alternative to performing a system transient analysis, the MCHFR and linear heat generation rate of the dropped rod event can be confirmed to be bounded by other events. Most rod drops result in a reactor trip on high power rate because of the immediate decrease in power from the dropped rod. Figure 7-3 shows the decrease in power from the dropped rod {{ }}$^{2(a),(c)}$ as a function of rod worth for a variety of initial power levels. The figure is based on a representative NPM core design and also overlays a representative high power rate trip. {{

}}$^{2(a),(c)}$ Sensitivity studies were performed to confirm this behavior using several representative NPM core designs, a range of initial power levels, and a variety of rod worths. {{

}}$^{2(a),(c)}$ The alternative bounding method for the rod drop ~~uses this result to~~ screens rod drop cases into two groups: those that result in reactor trip within a short period and those that do not.

Audit Question A-NonLOCA.LTR-5

For the larger group of cases that result in the early reactor trip {{ }}$^{2(a),(c)}$, the dropped rod, and subsequent reactor trip, cause a decrease in global power. The dropped rod does cause an asymmetry and results in an increase in local power peaking (primarily a radial increase with a small axial increase). The subsequent reactor trip eliminates the asymmetry and the associated local power peaking. Figure 7-4 shows the impact on the local power from the decrease in global power and the increase in local peaking for a representative NPM rod drop case that results in early reactor trip. {{

}}$^{2(a),(c)}$ The relevant

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acceptance criteria in Table 7-66 for the first group of the rod drop event can be assessed by the limiting steady-state subchannel analysis.

Audit Question A-NonLOCA.LTR-5

For the smaller group of cases that do not result in the early reactor trip {{

}}2(a),(c)

However, because power overshoot may occur during the event, these cases are bounded instead by comparison to the withdrawal of a single CRA. {{

}}2(a),(c) the conditions associated with the dropped rod cases are bounded by the limits used in the analysis of the withdrawal of a single CRA. Therefore, the limiting MCHFR and linear heat generation rate for the dropped rod cases that do not result in early reactor trip are bounded by the results of the single rod withdrawal analysis. This comparison of the rod drop analysis cases to the single rod withdrawal analysis limits is performed on a ~~design~~cycle-specific basis. A point that falls outside the single rod withdrawal analysis limits, {{

}}2(a),(c), would require use of the system transient approach described earlier in this section. When the ~~design~~cycle-specific comparison is bounding, the relevant acceptance criteria in Table 7-66 for the second group of the rod drop event can be assessed by the limiting single rod withdrawal subchannel analysis.

Misalignment of One or More CRAs

The misalignment of CRAs occurs as a result of one or more CRAs being left behind when inserting or withdrawing the control bank. These conditions are not evaluated with NRELAP5 as part of the non-LOCA event methodology because this event is not a transient. Instead, the MCHFR is determined as part of a detailed subchannel evaluation.

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-5S

Receipt Date: 02/23/2024

Question:

Audit question A-NonLOCA.LTR-5 was received by NuScale on April 24, 2023. NuScale provided a response on May 3, 2023. On February 23, 2024, NuScale received the following written feedback from the NRC:

The response to A-NonLOCA.LTR-5 does not describe how single rod withdrawal limits are selected to be bounding for a NPM design, fully describe the specific parameters in the rod drop and single rod withdrawal analysis limits that are being compared, or demonstrate the sufficiency of the set of parameters used to screen select rod drop cases (that do not result in an immediate reactor trip) from non-LOCA transient analysis.

To help address the original audit question, provide the staff EC-103274, Rev. 0, “Rod Drop Methodology Justification,” EE-0000-8185, Rev. 0, “Single Rod Withdrawal Methodology Improvement Scoping,” and calculations documents supporting Figures 7-3 through 7-5 of TR-0516-49416, Rev. 4 in the eRR.

Response:

This response is a supplement to the May 3, 2023 response; the May 3, 2023 response remains unchanged in the electronic reading room (eRR).

The May 3, 2003 response does describe how single rod withdrawal limits are selected to be bounding for a NuScale Power Module (NPM) design: {{

}}2(a),(c)

{{

}}2(a),(c)

The May 3, 2023 response does fully describe the specific parameters in the rod drop and single rod withdrawal analysis limits that are being compared: {{

}}2(a),(c)

Note that the comparison of rod drop to single rod withdrawal is only performed for the rod drop cases that do not immediately cause a reactor trip on negative power rate (i.e., a small population of the rod drop nuclear analysis cases).

The May 3, 2023 response does demonstrate the sufficiency of the set of parameters used to screen select rod drop cases:

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}}2(a),(c)

The February 23, 2024 NRC feedback requested EC-103274, Rev. 0, “Rod Drop Methodology Justification.” EC-103274 is provided in the eRR. Figures 7-3 through 7-5 of TR-0516-49416, Rev. 4, are taken directly from EC-103274 Figures 4-3, 4-13, and 4-20, respectively.

The February 23, 2024 NRC feedback also requested EE-0000-8185, Rev. 0, “Single Rod Withdrawal Methodology Improvement Scoping.” EE-0000-8185 is provided in the eRR. Another useful reference for additional information specific to the development of the rod drop methodology is EE-0000-7916, Rev. 0, “Rod Drop Methodology Improvement Scoping.” EE-0000-7916 started the investigation of rod drop methodology changes that ultimately led to the changes to the method described in TR-0516-49416, Rev. 4, and supported by EC-103274. EE-0000-7916 is listed as a reference in EC-103274 (i.e., Reference 1.5.7). EE-000-7916 is provided in the eRR. Note that both EE-0000-8185 and EE-0000-7916 are scoping documents issued in 2020 that predate that development of the Standard Design Approval Application (SDAA).

Finally, additional details of the nuclear analysis are provided in ER-A021-3589, Rev. 4, “Nuclear Analysis Methodology.” Sections 6.3.4 and A.10 discuss the single rod withdrawal event and Sections 6.3.6 and A.12 discuss the rod drop event. ER-A021-3589 was previously provided in the eRR for Chapter 4.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-5S2

Receipt Date: 04/29/2024

Question:

Audit question A-NonLOCA.LTR-5 was received by NuScale on April 24, 2023. NuScale provided a response on May 3, 2023. On February 23, 2024, NuScale received written feedback from the NRC. NuScale provided a supplemental response on February 27, 2024 to address the feedback. On April 29, 2024, NuScale received the following additional written feedback from the NRC: {{

}}2(a),(c)

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}}2(a),(c)

Response:

This response is a supplement to both the May 3, 2023 original response and the February 27, 2024 supplemental response; the May 3, 2023 and February 27, 2024 responses remain unchanged in the electronic reading room (eRR).

To address the detailed NRC second round of feedback, each aspect of the feedback is addressed by providing the text of the feedback in indented, italicized text, followed by the NuScale response in un-indented, regular text. {{

}}2(a),(c)

{{

}}2(a),(c) Using SIMULATE5 is appropriate for these static calculations and consistent with Section 4.3.1.1 of TR-0516-49416, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology”, that states “Core design analysis performed in accordance with a methodology approved for a NuScale design provides input to the system transient analysis.” The rod drop methodology discussion in Section 7.2.15.1 of TR-0516-49416 is revised as shown in the attached markup to reference Section 4.3.1.1 to use an approved core design method and describe how to calculate the parameters for the rod drop.

{{

}}2(a),(c)

If rod drop is bounded by single rod withdrawal, then there is no specific transient calculation using NRELAP5 and VIPRE-01 for the rod drop event. If rod drop is not bounded, then the previously approved rod drop methodology in the topical report is used; that methodology does include NRELAP5 and VIPRE-01 transient calculations. For the Standard Design Approval Application (SDAA), the rod drop cases are confirmed to be bounded by single rod withdrawal and therefore Final Safety Analysis Report (FSAR) Section 15.4.3.5.1 identifies that no specific transient analysis with NRELAP5 and VIPRE-01 is needed for rod drop. However, FSAR Section 15.4.3.5.1 and Section 15.4.3.5.2 are revised as shown in the attached markup to clarify

that the topical report only provides the method for how to determine if rod drop is bounded by other events; the actual confirmation is performed in the analysis that implements the methodology. In this case, the analysis results confirm the rod drop is bounded as indicated in FSAR Section 15.4.3.5.3.

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}}2(a),(c)

{{

}}2(a),(c)

Note that the discussion in Section 7.2.15.1 of TR-0516-49416 does not make a generic conclusion about the size of the power overshoot. Instead, Section 7.2.15.1 of TR-0516-49416 identifies that comparison to the single rod withdrawal analysis is made to account for the possibility of power overshoot. Therefore, TR-0516-49416 does not require further revision or clarification.

{{

}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

Note that the feedback also identified that certain previous descriptions "*should be docketed or incorporated into the TR*." The audit has an existing process where the NRC can request that a response to an audit question be docketed. If the NRC believes that information in a NuScale response should be docketed, the NRC should use the existing process to make that request.

{{

}}2(a),(c)

Markups of the affected changes, as described in the response, are provided below:

Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power (Continued)

Parameter	Bias / Conservatism	Basis
		{{
Steam Pressure Control Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, a sensitivity study is generally performed to identify cases for lowest MCHFR for this reactivity event.

Table 7-65 Not Used

7.2.15 Control Rod Misoperation

The methodology used to simulate a postulated control rod misoperation for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.15.1 General Event Description and Methodology

The rod control system is used to move (insert or withdraw) the control rod assemblies (CRAs) in response to an operator action or an automatic control. Since these transients are initiated by a malfunction in the rod control system, a

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variety of reactivity related conditions can result. Specific reactivity conditions for an NPM include: 1) withdrawing a single CRA; 2) dropping one or more CRAs; or, 3) leaving one or more CRAs behind when inserting or withdrawing a control bank. The consequences for each of these reactivity conditions are discussed below.

Table 7-66 lists the relevant acceptance criteria, single active failure, and loss of power scenarios.

Withdrawal of a Single CRA

The withdrawal of a single CRA causes a reactivity insertion that increases reactor power and leads to a rise in coolant temperature, pressurizer level, and RCS pressure. Feedback from the rising fuel temperature is not sufficient to counteract the reactivity insertion, so the power increases until the system trips on high power, high power rate, high pressurizer pressure, or high RCS temperatures. The maximum power and minimum MCHFR occur just after the resulting scram, while the peak primary pressure occurs some time later, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established at the end of the transient.

The limiting MCHFR typically occurs for a reactivity insertion rate that results in reactor trip on core power, pressurizer pressure, and RCS temperature signals at approximately the same time. These conditions arise for events initiated from partial power with lower reactivity insertion rates because higher reactivity insertion rates cause the MPS to trip much earlier on high power rate. The earlier reactor trip reduces the energy added to the reactor coolant, thereby producing a higher MCHFR. The asymmetry associated with the core power response causes the ex-core detectors to respond differently for each quadrant. Consequently, the range of reactivity insertion rates considered is sufficient to identify the point of transition to the high power rate signal (using the lowest reading ex-core detector based on the minimum after to before event initiation ratio of the radial peaking factors for the outer row of fuel assemblies). Sensitivity studies are performed on a variety of parameters, as necessary, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

In contrast to the uncontrolled CRA bank withdrawal at power event (Section 7.2.14), the highest reactivity insertion rate as determined from the resulting rod worth and control rod step speed is significantly lower for the withdrawal of a single CRA event.

Audit Question A-NonLOCA.LTR-5

Note that instead of using case-specific values for rod worth and parameters associated with the asymmetry, more conservative bounding input values can be used. The use of bounding input values provides a conservative analysis simplification for the withdrawal of a single CRA event. In addition, the bounding

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input values can then be used to compare this event to the dropping of one or more CRAs event described below.

Dropping One or More CRAs

Based on the minimum worth at any time during the cycle for a given core power, i.e., with the control bank positioned at the PDIL, dropping a single CRA causes a reactivity insertion that decreases reactor power. Feedback from the decreasing fuel temperature and the actions of the rod control system to restore power are generally not sufficient to counteract the reactivity insertion, so the power decreases until the system trips on high power rate. For event scenarios without a return to power, the maximum core power, peak primary pressure, and MCHFR occur at event initiation. The peak secondary system pressure occurs some time after the scram, as the DHRS begins to function and remove heat through the steam generators. Finally, stable DHRS cooling is established at the end of the transient.

The potential for a return to power exists only for events initiated from less than RTP because the reduced worth of the dropped rod gives the rod control system time to act. The corresponding MCHFR for a dropped rod event with a return to power is typically greater than the MCHFR for events initiated from HFP. Hence, the limiting MCHFR cases typically occur at HFP conditions.

Following reactor trip and subsequent turbine trip, the turbine bypass to the condenser opens to control the RCS temperature. However, the actions of the turbine bypass system are not credited, so as to minimize heat removal by the secondary side. Although turbine load is an input to the feedwater controller, no changes are made to this controller because the RCS responses are not sufficient to affect feedwater control.

The asymmetry associated with the core power response causes the ex-core detectors to respond differently for each quadrant. The power input to the high power rate signal uses the highest reading ex-core detector, multiplying the core average power by the maximum after drop to before drop ratio of the radial peaking factors for the outer row of fuel assemblies. Sensitivity studies are performed on a variety of parameters, as necessary, to identify the case(s) with a potentially limiting MCHFR. The NRELAP5 MCHFR pre-screening process is employed to identify the cases sent for a detailed subchannel evaluation.

The maximum fuel centerline temperature typically occurs at event initiation for those event scenarios with an immediate reactor trip. If the event scenario has a return to power, the maximum fuel centerline temperature is typically bounded by the fuel centerline temperature at HFP because the associated power peak is less than full power.

Audit Question A-NonLOCA.LTR-5

As an alternative to performing a system transient analysis, the MCHFR and linear heat generation rate of the dropped rod event can be confirmed to be

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bounded by other events. Most rod drops result in a reactor trip on high power rate because of the immediate decrease in power from the dropped rod. Figure 7-3 shows the decrease in power from the dropped rod {{ }}2(a),(c) as a function of rod worth for a variety of initial power levels. The figure is based on a representative NPM core design and also overlays a representative high power rate trip. {{

}}2(a),(c) Sensitivity studies were performed to confirm this behavior using several representative NPM core designs, a range of initial power levels, and a variety of rod worths. {{

}}2(a),(c) The alternative bounding method for the rod drop ~~uses this result to~~ screens rod drop cases into two groups: those that result in reactor trip within a short period and those that do not.

Audit Question A-NonLOCA.LTR-5

For the larger group of cases that result in the early reactor trip {{ }}2(a),(c), the dropped rod, and subsequent reactor trip, cause a decrease in global power. The dropped rod does cause an asymmetry and results in an increase in local power peaking (primarily a radial increase with a small axial increase). The subsequent reactor trip eliminates the asymmetry and the associated local power peaking. Figure 7-4 shows the impact on the local power from the decrease in global power and the increase in local peaking for a representative NPM rod drop case that results in early reactor trip. {{

}}2(a),(c) The relevant acceptance criteria in Table 7-66 for the first group of the rod drop event can be assessed by the limiting steady-state subchannel analysis.

Audit Question A-NonLOCA.LTR-5

For the smaller group of cases that do not result in the early reactor trip {{

}}2(a),(c)
However, because power overshoot may occur during the event, these cases are bounded instead by comparison to the withdrawal of a single CRA. ~~{{~~ A core design code consistent with the discussion in Section 4.3.1.1, such as SIMULATE5 (Reference 23), is used to calculate {{

}}2(a),(c)

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{{

}}2(a),(c) the conditions associated with the dropped rod cases are bounded by the values~~limits~~ used in the analysis of the withdrawal of a single CRA. Therefore, the limiting MCHFR and linear heat generation rate for the dropped rod cases that do not result in early reactor trip are bounded by the results of the single rod withdrawal analysis. This comparison of the rod drop analysis cases to the single rod withdrawal analysis values is performed on a ~~design~~cycle-specific basis. A rod drop analysis case with a parameter~~point~~ that falls outside the values used in the single rod withdrawal analysis~~ limits~~, {{ }}2(a),(c), would require use of the system transient approach described earlier in this section. When the ~~design~~cycle-specific comparison is bounding, the relevant acceptance criteria in Table 7-66 for the second group of the rod drop event can be assessed by the limiting single rod withdrawal subchannel analysis.

Misalignment of One or More CRAs

The misalignment of CRAs occurs as a result of one or more CRAs being left behind when inserting or withdrawing the control bank. These conditions are not evaluated with NRELAP5 as part of the non-LOCA event methodology because this event is not a transient. Instead, the MCHFR is determined as part of a detailed subchannel evaluation.

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Audit Question A-NonLOCA.LTR-5

Figure 7-5 Example comparison of parameters associated with non-tripped rod drop event~~s~~ cases to single rod withdrawal event analysis values~~limits~~

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}}2(a),(c)

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from the rising fuel and moderator temperatures partially counteracts the reactivity insertion, slowing the power increase. For CRM cases with higher reactivity insertion rates, the MPS trips the reactor on high power or high power rate. These cases are non-limiting because the reactor is tripped before the maximum amount of reactivity can be inserted. The limiting combination of reactivity insertion and reactivity feedback produces the maximum possible power increase without reaching the high power or high power rate limits. The power increase is terminated by MPS actuation of reactor trip when the high RCS hot temperature limit, high pressurizer pressure limit, or high RCS average temperature limit is reached.

The MPS high pressurizer pressure signal trips the reactor and actuates the DHRS for the limiting MCHFR case. The most limiting MCHFR (Figure 15.4-16) occurs at the moment before the power begins to decrease. The MCHFR remains above the CHF analysis limit and no fuel centerline melting is predicted for the withdrawal of a single CRA. The LHGR calculated for the single CRA withdrawal is below the calculated limits for cladding strain and fuel centerline melting. The maximum RCS pressure occurs after MCHFR occurs and is followed by decreasing RCS temperature and pressure. The limiting LHGR occurs in a single CRA withdrawal case where the initial power and limiting reactivity insertion rate is the same as that for the MCHFR case, and does not cause a high power rate trip like cases at lower initial powers. The pressurizer spray is assumed to function normally, which delays the trip on high pressure. This case maximizes power in combination with asymmetric peaking, resulting in a limiting LHGR. The LHGR remains below the design limit, so no fuel centerline melting is predicted for the single CRA withdrawal.

15.4.3.5 Control Rod Assembly Drop Thermal Hydraulic and Subchannel Analysis

15.4.3.5.1 Evaluation Models

Audit Question A-NonLOCA.LTR-5

There is no NPM thermal hydraulic analysis required for the CRA drop if it is determined to be bounded. The Non-LOCA Analysis Methodology Topical Report (Reference 15.4-3) ~~identifies that~~provides the methodology to determine if the CRA drop is bounded by either steady-state conditions at event initiation or by the single CRA withdrawal in Section 15.4.3.4. SIMULATE5 is used to calculate CRA worth and power distributions. A discussion of SIMULATE5 is provided in Section 4.3.

The subchannel core CHF analysis of the events bounding the CRA drop are performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.3.5.2 Input Parameters and Initial Conditions

Audit Question A-NonLOCA.LTR-5

A spectrum of possible CRA drop scenarios is analyzed to determine if~~ensure~~ the scenarios are bounded by either steady-state conditions at event initiation or by the single CRA withdrawal in Section 15.4.3.4. The following initial conditions and assumptions ensure the results have sufficient conservatism.

- The initial power levels of 20 percent, 50 percent, 75 percent, and 100 percent of nominal power are analyzed.
- Scenarios are analyzed for drop of a single CRA within a group and for drop of a whole group.
- The BOC, MOC, and EOC core conditions are analyzed.
- Other initial conditions varied in the analyzed cases include initial CRA position, axial offset, flow, and temperature.
- For the initial CRA drop, a representative drop time is used for all cases. For cases where a reactor trip occurs, conservative reactor trip characteristics are used, including a maximum time delay and holding the most reactive rod out of the core.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with RG 1.105.

15.4.3.5.3 Results

Following a CRA drop, there is a rapid decrease in core reactivity and power. The high power rate limit is reached in the first few seconds of the transient. The MPS sends a reactor trip signal, terminating the event. At lower powers, the power decrease is less pronounced, and the reactor does not trip. In these cases, the regulating CRA bank brings the reactor back to the initial power after an initial power overshoot.

Of the analyzed cases, most are confirmed to reach the high power rate limit by 1.5 seconds. Although local peaking increases due to the asymmetry caused by the CRA drop, total power decreases throughout the event due to the initial CRA drop and subsequent reactor trip. For these cases, the decrease in total power is greater than the increase in local peaking at all times such that local power decreases throughout the event. With the transient decrease in local power, the MCHFR and LHGR are limiting at the initial conditions and therefore bounded by the limiting steady-state subchannel analysis.

The cases that do not reach the high power rate limit by 1.5 seconds are characterized by low initial power and low CRA worth. Because of the lower initial power, these cases are also likely bounded by the limiting steady-state subchannel analysis. However, because power overshoot may occur during the event, these cases are instead compared to the single CRA withdrawal event. The single CRA withdrawal limits for CRA worth, radial peaking, and

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-8

Receipt Date: 08/21/2023

Question:

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}}2(a),(c),ECI

Response:

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}}2(a),(c),ECI

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}}2(a),(c),ECI

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-8S

Receipt Date: 03/29/2024

Question:

Audit question A-NonLOCA.LTR-8 was received by NuScale on August 21, 2023. NuScale provided a response on September 20, 2023. On March 29, 2024, NuScale received the following written feedback from the NRC:

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}}2(a),(c),ECI

Response:

This response is a supplement to the September 20, 2023 response to address the feedback from March 29, 2024; the September 20, 2023 response remains unchanged in the electronic reading room (eRR).

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}}2(a),(c),ECI

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}}2(a),(c),ECI

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-22

Receipt Date: 04/01/2024

Question:

Explain why the pressure in the primary system is not listed in the PIRT table as part of the Non-LOCA transient parameters in Non-LOCA. The system response and major parameters such as heat transfer coefficient vary as the pressure changes in the primary side during Non-LOCA transients. Section 4.3.5 of the TR clearly indicates that the effect of pressure on critical heat flux (CHF) must be established to determine the appropriate direction for biasing. However, Table 5-3 does not include primary side pressure. Please provide proposed markups to the TR.

Response:

The non-loss-of-coolant accident (non-LOCA) topical report (TR-0516-49416-P Revision 4) Table 5-3 summarizes the high-ranked phenomena for non-LOCA events as part of the phenomena identification and ranking table (PIRT). Primary system pressure is not listed in the non-LOCA PIRT because pressure is not a stand-alone phenomenon. Instead, primary pressure is a characteristic of the system, and it is a figure of merit in Phase 1 and Phase 2 of the non-LOCA transients, as identified in Section 5.1.3 of the non-LOCA topical report. The phenomena in the PIRT impact the primary pressure. Primary pressure is also considered in the range of conditions to demonstrate non-LOCA evaluation model (EM) applicability.

As described in Section 2.1 of the non-LOCA topical report, the high-ranked non-LOCA phenomena were evaluated with respect to the high-ranked phenomena identified as part of the NuScale LOCA evaluation model development. {{

}}$^{2(a),(c)}$ The non-LOCA EM validation basis and applicability assessment focuses on phenomena associated with the steam generator and

decay heat removal system that are more important for non-LOCA transients, compared to large LOCA pipe breaks. For prediction of conditions in the primary system, including primary pressure, the non-LOCA EM uses the NRELAP5 validation basis and applicability assessment presented in the LOCA topical report (TR-0516-49422-P Revision 3) as well as the validation basis presented in the non-LOCA EM. {{

}}2(a),(c)

In addition, as described in the response to audit question A-NonLOCA.LTR-16, the PIRT developed for the NuScale Power Module (NPM)-160 associated with the US600 design was not required to be updated for the NPM-20 associated with the US460 design. Therefore, the list of high-ranked phenomena in Table 5-3 of the non-LOCA topical report has not changed from the list included in the previous version of the non-LOCA topical report that was reviewed and approved by the NRC (TR-0516-49416-P-A Revision 3). Primary pressure is not included in Table 5-3 in TR-0516-49416-P-A Revision 3.

For the reasons provided above, it is not necessary to include primary system pressure in Table 5-3 of the non-LOCA topical report. Primary system pressure is appropriately modeled, calculated, and addressed in non-LOCA analyses.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-23

Receipt Date: 04/01/2024

Question:

Provide a methodology and acceptance criteria for identifying the limiting cases for MCHFR. The TR states: "After the system transient analysis calculations are performed and assessed, for events that require subchannel analysis, a number of cases are identified as limiting for MCHFR." However, it is not clear how the cases are identified and what are the criteria for selecting the cases as limiting for MCHFR. Please provide proposed markups to the TR.

Response:

The version of the non-LOCA topical report previously approved by the NRC (TR-0516-49416-P-A, Revision 3) contains the same quotation in Section 4.3.5 and also did not provide further details regarding selection of cases from NRELAP5 to analyze in the downstream subchannel (VIPRE-01) analyses. As stated in the safety evaluation report (SER), Section 3.4.3.5, "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." In other words, a more detailed explanation than that already provided in Section 4.3.5 of the topical report was not required during the NRC's previous approval of the non-LOCA topical report. However, some additional information is provided below.

In other audit requests, the NRC has requested the NRELAP5 and VIPRE-01 calculation documents for multiple Chapter 15 events. By comparison of the NRELAP5 and VIPRE-01 calculations documents for a specific Chapter 15 event, it can be determined the extent of cases selected for analysis of minimum critical heat flux ratio (MCHFR) in VIPRE-01. Examples are provided below.

- Audit question A-15.2.7-3 requested the calculations for the loss of feedwater flow (LOFW) analysis in Final Safety Analysis Report (FSAR) Section 15.2.7. In response, NuScale provided EC-116269 Revision 0 (for NRELAP5) and EC-117457 Revision 2 (for VIPRE-01). {{

}}2(a),(c) The overall limiting MCHFR from VIPRE-01 is 2.03 compared to the analysis limit of 1.45 (i.e., approximately 40 percent margin).
- Audit question A-15.4.2-1 requested the calculations for the uncontrolled control rod assembly bank withdrawal (UCBW) analysis in FSAR Section 15.4.2. In response, NuScale provided EC-0000-8672 Revision 0 (for NRELAP5) and EC-102745 Revision 2 (for VIPRE-01). {{

}}2(a),(c) The overall limiting MCHFR from VIPRE-01 is 1.58 compared to the analysis limit of 1.45 (i.e., approximately 10 percent margin).

{{

}}2(a),(c)

Markups of the affected changes, as described in the response, are provided below:

limiting initialization for a given transient progression to ensure the maximum power and core inlet fluid temperature are reached prior to reactor trip system actuation. For example, in the case of a heatup event, the RCS increases in temperature, causing a pressurizer insurge and subsequent increase in pressure. The limiting CHF scenario is the transient progression that results in the highest core outlet temperature at the time of reactor trip on high pressure, which is generally the faster heatups where the pressurizer initialization is biased to delay the high pressure trip.

For some transients, a spectrum of cases is analyzed from the limiting initialization. {{

Audit Question A-NonLOCA.LTR-23

}}2(a),(c)

After the system transient analysis calculations are performed and assessed, for events that require subchannel analysis, a number of cases are identified as limiting for MCHFR. For the limiting cases selected, the required system transient parameters are tabulated as a function of time for input to the downstream subchannel analysis calculations, to calculate margin to CHF.

The system transient parameters are provided for subchannel analysis for sufficient time for the subchannel analyses to demonstrate that the MCHFR has occurred, typically 10-15 seconds following reactor trip.

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-24

Receipt Date: 04/01/2024

Question:

In its response to audit question A-NonLOCA.LTR-7, the applicant indicates that the DHRS scaling analysis is provided in EE-107403. {{

}}$^{2(a),(c),ECI}$ Please make available for audit {{

}}$^{2(a),(c),ECI}$ for the staff to compare and determine the similarity between the test facility (NIST-2) and NPM-20.

Response:

As stated in the response to audit question A-NonLOCA.LTR-7, scaling analysis performed for decay heat removal system (DHRS) testing at the NIST-2 facility is provided in ER-126485, Revision 0, “NRELAP5 Applicability to NPM-20 Non-LOCA Analysis.” ER-126485 was previously provided in the electronic reading room (eRR) for the non-LOCA topical report.
{{

}}$^{2(a),(c),ECI}$ This scaling analysis {{

}}$^{2(a),(c),ECI}$ is summarized in Section 5.3.7.2.2 of the non-LOCA topical report, as described in response to RSI-17 and provided in markups to the non-LOCA topical report in response to RSI-15 (submitted by NuScale letter LO-145424 dated July 14, 2023 and available in ADAMS at ML23195A091).

In audit question A-NonLOCA.LTR-7, the NRC requested additional information to better understand the scaling analysis in ER-126485. {{

}}$^{2(a),(c),ECI}$ EE-107403 was

made available in the eRR for the non-LOCA topical report as part of the response to audit question A-NonLOCA.LTR-7.

EE-107403 and ER-126485 together provide the scaling analysis {{ }}$^{2(a),(c),ECI)}$ results that NuScale has performed for the non-LOCA testing performed at the NIST-2 facility. These two documents remain available in the eRR for the non-LOCA topical report.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-25

Receipt Date: 04/15/2024

Question:

{{

}}2(a),(c)

Provide markup to the non-LOCA topical report.

Response:

{{

}}2(a),(c)

- {{

}}2(a),(c)

- Applicability analyses have been performed, as described in the response to audit question A-NonLOCA.LTR-3, comparing the NuScale Power Module (NPM) for the US460 (i.e., NPM-20) to the NPM for the US600 (i.e., NPM-160) and confirm the EM is applicable to the US460.
- The phenomena identification and ranking table (PIRT) performed supporting the EM has been reviewed and it has been confirmed that there are no new additional phenomena applicable for the US460, as described in the response to audit question A-NonLOCA.LTR-16.

In addition, Section 5.1.4 of the topical report identifies that NRELAP5 assessments against SIET data (TF-1 and TF-2) were used to validate heat transfer through the SG as part of the DHRS loop for non-LOCA transients, with details provided in Section 5.3.5. The discussion in Section 5.3.5 identifies that the detailed validation is provided in Section 7.4 of the LOCA topical report (TR-0516-49422-P, Revision 3). {{

}}2(a),(c) The conclusions based on the results in EC-153070 are consistent with the conclusions discussed in the bulleted list above.

{{

}}2(a),(c)

{{

}}2(a),(c)

Therefore, no new analyses are required.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-26

Receipt Date: 04/15/2024

Question:

Please review and update Table 7-3.

Table 7-3, which lists analytical limits and actuation delays (reactor trip system and engineered safety features actuation system), does not reflect the NPM-20 design for reactor trips and ESF actuations. For example,

- Nine reactor trip functions have different setpoints (Analytical Limits)
- There is one new reactor trip function for the NPM-20, i.e., Hi RCS Avg Temp
- There are numerous changes in various ESF actuation functions and setpoints, e.g., in the ECCS, DHRS, and Demin Water Isolation System
- The Hi CNV (containment vessel) Water Level ECCS actuation function appears to be deleted for the NPM-20, apparently replaced with RPV Riser Level actuation functions.

Please provide markups to the TR accordingly.

Response:

As stated in Section 7.1.6.3 of the topical report (TR-0516-49416-P, Revision 4):

> "Examples of the analytical limits and actuation delays are summarized in Table 7-3. The values in Table 7-3 are example values used to aid the subsequent methodology discussion. Application of the methodology to a specific [NuScale Power Module (NPM)] design requires the use of analytical limits and actuation delays associated with that NPM design."

For the US460 design (i.e., NPM-20), the specific analytical limits and actuation delays are provided in Final Safety Analysis Report (FSAR) Table 15.0-7.

Because the topical report explicitly identifies Table 7-3 as an example only to aid understanding the methodology, it is not necessary for Table 7-3 to be consistent with any specific NPM design. Therefore, no changes are required.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-28

Receipt Date: 04/15/2024

Question:

Provide a methodology for establishing the effect of pressure on critical heat flux (CHF).

The TR states that the effect of pressure on critical heat flux (CHF) will be established on a case-by-case basis to determine the appropriate direction for biasing. The TR however was revised to remove discussion of various pressure effects on CHF as well as a method for biasing pressure to conservatively estimate MCHFR.

Please provide proposed markups to the TR to include the methodology for determining effect of pressure on CHF including discussion of the relative importance of various effects.

Response:

The non-loss-of-coolant accident (non-LOCA) methodology topical report, TR-0516-49416-P-A, Revision 3, was previously reviewed and approved by the NRC. Section 4.3.5 of that revision identified the conservative bias direction for determining minimum critical heat flux ratio (MCHFR) for the following four parameters: reactor power, core exit pressure, core inlet temperature, and system flow rate. The conservative bias directions for reactor power (high), core inlet temperature (high), and system flow rate (low) were all intuitive and did not require more detailed explanation. {{

}}$^{2(a),(c)}$

In Section 4.3.5 of the non-LOCA methodology topical report currently under NRC review, TR-0516-49416-P, Revision 4, the conservative bias directions for reactor power, core inlet temperature, and system flow rate, and associated rationales, have not changed from the previously approved version. However, the discussion of pressure biasing has been greatly simplified. Rather than justifying why one bias direction is always conservative, NuScale has taken a more conservative approach and identified that effect of pressure on MCHFR is to be determined on a case by case basis. In other words, there is no “a priori” assumption of the effect of pressure on MCHFR. Instead, cases with both high pressure and low pressure are used to determine which is more limiting for MCHFR. This approach ensures that the limiting MCHFR will be determined and is therefore conservative.

With this background, each aspect of the NRC audit question is addressed by providing the text of the question in indented, italicized text, followed by the NuScale response in non-indented, regular text.

> *“Provide a methodology for establishing the effect of pressure on critical heat flux (CHF).”*

Section 4.3.5 of TR-0516-49416-P, Revision 4, already provides a methodology: cases with different pressures are run and the most limiting MCHFR is selected from the different cases.

> *“The TR states that the effect of pressure on critical heat flux (CHF) will be established on a case-by-case basis to determine the appropriate direction for biasing.”*

Correct.

> *“The TR however was revised to remove discussion of various pressure effects on CHF as well as a method for biasing pressure to conservatively estimate MCHFR.”*

The removed text provided a justification for biasing pressure in one direction only. Because NuScale is no longer biasing pressure in only one direction, the text was unnecessary.

> *“Please provide proposed markups to the TR to include the methodology for determining effect of pressure on CHF including discussion of the relative importance of various effects.”*

Section 4.3.5 of TR-0516-49416-P, Revision 4, already provides a methodology: cases with different pressures are run and the most limiting MCHFR is selected from the different cases.

Additional discussion is not necessary since the limiting MCHFR is selected from the different cases regardless of the relative importance of various effects of pressure on MCHFR.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-28S

Receipt Date: 09/12/2024

Question:

As a follow-up to audit question A-NonLOCA.LTR-28, NRC sent an advanced request for additional information (RAI) on the same topic on August 29, 2024. A clarification call to discuss the RAI was held on September 12, 2024. During the clarification call, NuScale took an action to provide a revised topical report markup to address the NRC concerns from the original audit question A-NonLOCA.LTR-28 and the advanced RAI.

Response:

As described above, NuScale took an action to provide a markup to TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” to address the NRC concerns from the original audit question A-NonLOCA.LTR-28 and the advanced request for additional information (RAI). The markup is attached to this response. A summary of the changes is provided for convenience as follows:

- Section 4.3.5 is revised to replace the discussion of “case-by-case” with the explanation that the impact of pressure on critical heat flux is determined by varying the pressurizer pressure.
- Table 7-7 (decrease in feedwater temperature) is revised to indicate that both the initial pressurizer pressure and pressurizer pressure control using spray are varied {{ }}$^{2(a),(c)}$
- Table 7-14 (increase in feedwater flow) is revised to indicate that both the initial pressurizer pressure and pressurizer pressure control using spray are varied {{ }}$^{2(a),(c)}$
- Table 7-19 (increase in steam flow) is revised to indicate that both the initial pressurizer pressure and pressurizer pressure control using spray are varied {{ }}$^{2(a),(c)}$

- Table 7-24 (steam line break) is revised to indicate that both the initial pressurizer pressure and pressurizer pressure control using spray are varied {{ }}2(a),(c) In addition, pressurizer pressure control using heaters is set to nominal {{ }}2(a),(c) The initial pressurizer level is also revised to indicate that it is varied {{ }}2(a),(c) Finally, an editorial error in the initial fuel temperature bias is corrected regarding the different fuel temperatures associated with beginning-of-cycle (BOC) and end-of-cycle (EOC).
- Table 7-28 (containment flooding / loss of containment vacuum) is revised to indicate that pressurizer pressure control using spray is varied {{ }}2(a),(c) In addition, pressurizer pressure control using heaters is set to nominal {{ }}2(a),(c) Note that Table 7-28 already identified that initial pressurizer pressure is varied, so no change to that parameter is needed like it was in the other tables as described above.
- Table 7-70 (dropped control rod assemblies) is revised to indicate that the initial pressurizer pressure is varied {{ }}2(a),(c)

Markups of the affected changes, as described in the response, are provided below:

These conditions are demonstrated for a reasonable time, typically a few hundred seconds, following the last safety system actuation expected to occur in the short term transient progression, to demonstrate that core cooling is established and conditions that result in minimum margin to the acceptance criteria have occurred.

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

For NPM non-LOCA events, VIPRE-01 is used for performance of subchannel analysis calculations. Reference 6, supplemented by Reference 28, describes the subchannel analysis methodology.

In the transient subchannel analysis, the following acceptance criteria are assessed:

- MCHFR
- maximum fuel centerline temperature

For non-LOCA events that require subchannel analysis, the NRELAP5 transient analyses provide the following input from the system transient calculation to the downstream subchannel analysis:

- reactor power as a function of time
- core exit pressure as a function of time
- core inlet temperature as a function of time
- total system flow rate as a function of time

In the NRELAP5 system transient analysis, cases for downstream subchannel analysis are identified based on the conservative bias directions for the boundary condition input and considering an NPM natural circulation design. The conservative bias directions are discussed below, followed by a description of the methodology for identifying cases for downstream subchannel analysis.

As identified in Reference 6, supplemented by Reference 28, for the system transient parameters provided by NRELAP5, the conservative bias directions to minimize the CHFR are as follows:

- maximum reactor power (higher power increases the actual heat flux)
- maximum core inlet temperature (higher temperature reduces energy addition needed to raise coolant to saturated conditions)
- minimum system flow rate (minimum flow is conservative as there is less coolant flow in the reactor core available for heat transfer)

Audit Question A-NonLOCA.LTR-28

The effect of pressure on critical heat flux (CHF) is established ~~on a case-by-case basis~~by varying initial pressurizer pressure to determine the appropriate direction for biasing for events that are challenging for margin to CHF.

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Table 7-6 Acceptance criteria - decrease in feedwater temperature (Continued)

Acceptance Criteria	Discussion
CHFR	Due to the increase in reactor power and subsequent reduction of MCHFR, this acceptance criterion is challenged for the decrease in feedwater temperature event. Consequently, sensitivity cases are performed to support the follow-on MCHFR evaluation.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.1.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-7 are considered in identifying a bounding transient simulation for MCHFR.53

Audit Question A-NonLOCA.LTR-28
RAI Questions A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.~~Biased to the high condition~~	}}2(a),(c)

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Table 7-7 Initial conditions, biases, and conservatisms - decrease in feedwater temperature (Continued)

Parameter	Bias / Conservatism	Basis {{
RCS Temperature Control Automatic rod control	Varied.	
Boron concentration	Not credited.	
PZR Pressure Control PZR spray	Varied. ~~Disabled.~~	
PZR heaters	Nominal.	
PZR Level Control Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control Turbine throttle valves	Varied	
Turbine bypass valves	Disabled.	}}2(a),(c)

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Table 7-13 Acceptance criteria - increase in feedwater flow (Continued)

Acceptance Criteria	Discussion
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.
SG overfill	An increase in feedwater flow can lead to overfilling the SG. If the SG is overfilled, DHRS performance is negatively impacted. This criterion is satisfied by demonstrating the SG does not overfill and DHRS cooling is adequate, consistent with the discussion in the acceptance criterion for escalation of an AOO to an accident.

7.2.2.3 Biases, Conservatisms, and Sensitivity Studies

Audit Question A-NonLOCA.LTR-60

The biases and conservatisms indicated in Table 7-14 are considered in identifying a bounding transient simulation for criteria other than SG overfill.

Audit Questions A-NonLOCA.LTR-28, A-NonLOCA.LTR-65
RAI Questions, A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied. ~~Biased to the high condition.~~	
Initial PZR level	Biased to the high condition.	

}}2(a),(c)

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Table 7-14 Initial conditions, biases, and conservatisms - increase in feedwater flow (Continued)

Parameter	Bias / Conservatism	Basis {{
RCS Temperature Control Automatic rod control Boron concentration	 Enabled. Not credited.	
PZR Pressure Control PZR spray PZR heaters	 Varied.~~Disabled.~~ Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled to control to constant steam pressure. Disabled.	
Feedwater and Turbine Load Control		

}}2(a),(c)

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Audit Question A-NonLOCA.LTR-28
RAI Questions A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied. ~~Biased to the high condition.~~	
Initial PZR level	Biased to the high condition.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Biased to EOC conditions.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Biased to the high condition.	
SG heat transfer	Nominal	
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Steam flow increase	Varied	}}2(a),(c)

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Table 7-19 Initial conditions, biases, and conservatisms - increase in steam flow (Continued)

Parameter	Bias / Conservatism	Basis
RCS Temperature Control Automatic rod control Boron concentration	 Varied. Not credited.	{{

Table 7-23 Acceptance criteria - steam line break (Continued)

Acceptance Criteria	Discussion
Secondary pressure	Due to the depressurizing nature of this cooldown event, secondary pressure remains below the acceptance criterion for peak secondary pressure.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis. Sensitivity cases are performed to support the follow-on MCHFR evaluation.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis, however the reactivity insertion rate from the cooldown event is insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation to a more serious accident or consequential loss of functionality	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.4.3 Biases, Conservatisms, and Sensitivity Studies

RAI Questions A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

The biases and conservatisms indicated in Table 7-24, and a spectrum of break locations and sizes, are considered in identifying a bounding transient simulation for MCHFR and mass release.

In the radiological analysis, primary-to-secondary leakage is assumed, which allows primary coolant to be released with the SG break flow. The transient SG mass release can be calculated for use as an input to the downstream radiological analysis. Alternatively, bounding assumptions for primary coolant release can be used in the radiological analysis to eliminate the need for calculating SG mass release. If such bounding assumptions are used, transient analysis to maximize SG mass release is not required.

Audit Question A-NonLOCA.LTR-28

Table 7-24 Initial conditions, biases, and conservatisms - steam line break

Parameter	Bias / Conservatism	Basis(1) {{
Initial reactor power	Biased upwards to account for measurement uncertainty.	}}2(a),(c)

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Table 7-24 Initial conditions, biases, and conservatisms - steam line break (Continued)

Parameter	Bias / Conservatism	Basis(1) {{
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.~~Biased to the high condition.~~	
Initial PZR level	Varied.~~Biased to the high condition.~~	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased low for B~~E~~OC conditions and high for E~~B~~OC conditions.	
MTC	Both EOC and BOC conditions.	
Kinetics	Both EOC and BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(2)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control	Enabled. (MCHFR)	
Boron concentration	Not credited.	}}2(a),(c)

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Table 7-24 Initial conditions, biases, and conservatisms - steam line break (Continued)

Parameter	Bias / Conservatism	Basis[1] {{
PZR Pressure Control PZR spray	Varied.~~Disabled.~~	
PZR heaters	~~Enabled~~Nominal.	
PZR Level Control Charging	Not credited.	
Letdown	Disabled.	
Steam Pressure Control Turbine throttle valves	Enabled.	
Turbine bypass valves	Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Disabled.	}}2(a),(c)

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Table 7-27 Acceptance criteria - containment flooding / loss of containment vacuum

Acceptance Criteria	Discussion
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.5.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-28 are considered in order to identify the bounding transient simulation for MCHFR for the CNV flooding/loss of CNV vacuum event.

Audit Question A-NonLOCA.LTR-28
RAI Questions A-NonLOCA.LTR-31, A-NonLOCA.LTR-32, A-NonLOCA.LTR-46, A-NonLOCA.LTR-53, A-NonLOCA.LTR-56, A-NonLOCA.LTR-65

Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied	
Initial PZR level	Nominal	
Initial feedwater temperature	Nominal	
Initial fuel temperature	Nominal	
MTC	Biased to the EOC condition.	
Kinetics	Biased to the EOC condition.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Nominal	
SG heat transfer	Nominal	}}2(a),(c)

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Table 7-28 Initial conditions, biases, and conservatisms - containment flooding / loss of containment vacuum (Continued)

Parameter	Bias / Conservatism	Basis
		{{
RSV lift setpoint	Nominal	
SG tube plugging	Biased to the low condition.	
Initial containment pressure	Varied	
Initial pool temperature	Varied	
RCCW leak flow	Varied	
RCCW temperature	Varied	
RCS Temperature Control Automatic rod control	Enabled.	
Boron concentration	Not credited.	
PZR Pressure Control PZR spray	Varied.~~Disabled.~~	
PZR heaters	~~Enabled~~Nominal.	}}2(a),(c)

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Audit Question A-NonLOCA.LTR-28

Table 7-70 Initial conditions, biases, and conservatisms - control rod misoperation, dropped control rod assemblies

Parameter	Bias / Conservatism	Basis
		{{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied. ~~Biased to the high condition.~~	
Initial PZR level	Nominal.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Nominal.	
MTC	Varied.	
Kinetics	Varied.	
Decay heat	Biased to the high condition.	
Initial SG pressure	Nominal.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
Dropped CRA worth	Minimum	}}2(a),(c)

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-29

Receipt Date: 04/15/2024

Question:

Revise the TR to reference the current versions of TR-0915-17564-P-A and TR-108601-P-A. Section 4.3.1.1.2 of the TR references TR-0915-17564-P-A and TR-108601-P-A. However, there are newer versions of these twothis reference referenced that are developed for the NPM-20 design. These references need to be updated. Provide proposed markups to the TR.

Response:

Section 4.3.1.1.2 of the non-LOCA topical report (TR-0516-494160-P Revision 4) includes cross-references to References 6 and 28. As identified in Section 11.0, References 6 and 28 correspond to “Subchannel Analysis Methodology,” TR-0915-17564-P-A, Revision 2, and “Statistical Subchannel Analysis Methodology,” TR-108601-P, Revision 2, respectively. NuScale has confirmed that the information in Reference 6 for the Subchannel Analysis Methodology is accurate and current; there is no newer version and no changes are required. NuScale agrees that there is a newer revision for Reference 28 and the information is revised accordingly.

In addition, to the change associated with this audit question, NuScale is updating other references in Section 11.0 to the latest version, as applicable.

Markups of the affected changes, as described in the response, are provided below:

11.0 References

1. U.S. Nuclear Regulatory Commission, “Transient and Accident Analysis Methods,” Regulatory Guide 1.203, December 2005.

2. NuScale Power, LLC, “Loss-of-Coolant Accident Evaluation Model,” TR-0516-49422-P, Revision 3.

Audit Question A-NonLOCA.LTR-29

3. NuScale Power, LLC, “NuScale Power, LLC Quality Assurance Program Description,” MN-122626-A, Revision 1~~0~~.

4. U.S. Code of Federal Regulations, “General Design Criteria for Nuclear Power Plants,” Appendix A, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50 Appendix A).

5. U.S. Code of Federal Regulations, “Contents of applications; technical information,” Section 52.47, Part 52 Chapter I, Title 10, “Energy,” (10 CFR 52.47).

6. NuScale Power, LLC, “Subchannel Analysis Methodology,” TR-0915-17564-P-A, Revision 2.

7. Not Used

8. NuScale Power, LLC, “Accident Source Term Methodology,” TR-0915-17565-P-A, Revision 4.

9. U.S. Nuclear Regulatory Commission, “Standard Review Plan, Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment,” NUREG-0800, Chapter 15, Section 15.6.2, Rev. 2, July 1981.

10. NuScale Power, LLC, “Evaluation Methodology for Stability Analysis of the NuScale Power Module,” TR-0516-49417-P-A, Revision 1.

11. S.M. Modro et al., “Multi-Application Small Light Water Reactor Final Report,” Idaho National Engineering and Environmental Laboratory, INEEL/EXT-04-01626, December 2003.

12. Electric Power Research Institute, “RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,” Volumes 1-5, EPRI NP-7450-A, Revision 10, September 2014.

13. Moghanaki, S. K. and Rahgoshay, M., “Pressurizer Modeling: Using Different Thermodynamic Models and Comparing Results with RELAP Code Results,” Applied Mechanics and Materials (Volumes 423-426), pp. 1444-1448, 2013.

14. U.S. Nuclear Regulatory Commission, “Combined License Applications for Nuclear Power Plants,” Regulatory Guide 1.206, June 2007.

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15. U.S. Nuclear Regulatory Commission, “Standard Review Plan, Transient and Accident Analyses,” NUREG-0800, Chapter 15, Revision 3, March 2007 and various Chapter 15 subsections.

16. U.S. Code of Federal Regulations, “Definitions,” Section 50.2, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50.2).

17. American National Standards Institute / American Nuclear Society, “Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems,” ANSI/ANS-58.9-2002 R2015.

18. U.S. Nuclear Regulatory Commission, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” SECY-94-084, March 1994.

19. U.S. Code of Federal Regulations, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Appendix B, Part 50 Chapter I, Title 10, “Energy,” (10 CFR 50 Appendix B).

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

Audit Question A-NonLOCA.LTR-29

21. NuScale Power, LLC, “Rod Ejection Accident Methodology,” TR-0716-50350-P, Revision 3~~2~~.

22. NuScale Power, LLC, “Applicability of AREVA Fuel Methodology for the NuScale Design,” TR-0116-20825-P-A, Revision 1.

23. NuScale Power, LLC, “Nuclear Analysis Codes and Methods Qualification,” TR-0616-48793-P-A, Revision 1.

24. Kim, S.J., “Turbulent film condensation of high pressure steam in a vertical tube of passive secondary consensation system,” PhD thesis, Korea Advanced Institute of Science and Technology, 2000.

25. SwUM-0304-17023, Revision 8, NRELAP5 Version 1.4 Theory Manual.

26. NuScale Power, LLC, “Extended Passive Cooling and Reactivity Control Methodology,” TR-124587-P, Revision 0.

27. U.S. Code of Federal Regulations, “Contents of applications; technical information,” Section 52.137, Part 52 Chapter I, Title 10, “Energy,” (10 CFR 52.137).

Audit Question A-NonLOCA.LTR-29

28. NuScale Power, LLC, “Statistical Subchannel Analysis Methodology,” TR-108601-P-A (Supplement 1 to TR-0915-17564-P-A, Revision 2), Revision 4~~2~~.

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29. SwUM-0304-17023, Revision 10, NRELAP5 Version 1.6 Theory Manual.

Audit Question A-NonLOCA.LTR-29

30. NuScale Power, LLC, “Methodology for the Determination of the Onset of Density Wave Oscillations (DWO),” TR-131981-P, Revision 1~~0~~.

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-30

Receipt Date: 04/15/2024

Question:

{{

}}$^{2(a),(c)}$ Therefore, the applicant is expected to provide the bases for applicability of the EM for RCS pressurization and heat-up events, such as positive reactivity insertion and decrease in heat removal events. Comparisons of predicted and measured (or benchmarked) RCS pressure response for rapid pressurization events are needed.

Provide the validation of the EM for RCS pressurization and heat-up events, such as positive reactivity insertion and decrease in heat removal events. For example, provide NRELAP5 code validation for a limiting RCS pressurization transient in well scaled test facilities. The code validations must show the similarities between the test facilities/apparatus and the reactor design. The results of the code validation must include the biases and uncertainties of the code in predicting the system phenomena on both the primary and secondary sides. The dependence of code biases on major parameters must be evaluated to ensure the code not to bias on the non-conservative direction. Sensitivity studies may be performed in lieu of tests.

Provide proposed markups to the TR.

Response:

As discussed in Section 5.3.7.2 of the non-loss-of-coolant accident (non-LOCA) topical report, TR-0516-49416-P Rev. 4, the purpose of the NIST-2 testing was to enhance the validation basis for the decay heat removal system (DHRS). The NIST-2 DHRS testing did include a loss of feedwater (LOFW) as part of the testing sequence and therefore has sometimes been referred to as an LOFW test. However, it is emphasized that the focus of the testing was on DHRS performance after operation of DHRS was underway. The specific initiating events and associated response prior to DHRS operation was not the focus of the testing. Therefore, comparison of the NIST-2 DHRS test response to the Final Safety Analysis Report (FSAR) Chapter 15 LOFW results is not appropriate. The NRELAP5 validation previously performed and included in the the non-LOCA topical report that was reviewed and approved by the NRC (TR-0516-49416-P-A Rev. 3) demonstrated applicability of the evaluation model to predict the reactor coolant system pressurization response during non-LOCA events. Justification is provided below.

{{

}}2(a),(b),(c),ECI

Contrast this to the sequence of events for the Chapter 15 LOFW in FSAR Table 15.2-17. In the Chapter 15 LOFW, the LOFW occurs at time zero but reactor trip does not occur until 13 seconds later. As a result, there is 13 seconds where the reactor is still generating full power

even though there is no feedwater being supplied to the steam generators. The mismatch between power being produced in the primary and power being removed by the secondary results in a significant heatup and pressurization of the primary during this period as shown in FSAR Figures 15.2-31 and 15.2-28, respectively. The pressurization results in the high pressurizer pressure analytical limit being reached and reactor trip occurs. Following reactor trip, power decreases rapidly to decay heat levels. The power reduction eliminates the mismatch between primary side power generation and secondary side heat removal, and pressure responds by decreasing rapidly as shown in FSAR Figure 15.2-28. After this rapid decrease, decay heat generation results in a smaller increase in pressure. The DHRS actuation valves fully open at 43 seconds per FSAR Table 15.2-17. The primary pressure is stable but somewhat elevated until effective DHRS cooling is established around 100 seconds as shown in FSAR Figure 15.2-29 and Figure 15.2-28. Once DHRS is in operation, pressure beings to slowly decrease as DHRS begins to effectively remove decay heat.

{{

}}$^{2(a),(c)}$ This similarity during DHRS operation is expected as the intent of the testing was to focus on the period of DHRS operation as stated above.

Regarding the validation of NRELAP5 for pressurization and heatup events, the validation was previously performed and included in the the non-LOCA topical report that was reviewed and approved by the NRC (TR-0516-49416-P-A Rev. 3). {{

}}$^{2(a),(c)}$ As has been described in several previous audit responses, NuScale performed an applicability assessment and identified no new phenomena resulting from the design changes from the US600 design to the US460 design. Therefore, the existing validation basis {{

}}$^{2(a),(c)}$ remains applicable. This audit question also requested information regarding code uncertainties associated with the above validation basis. As in the US600 design that was previously approved by the NRC, the maximum reactor coolant system pressure in the US460 design is limited by the reactor safety valves (RSVs) and there is little transient-dependent over-shoot as demonstrated by the limiting results presented in FSAR

Section 15.2. Therefore, code uncertainty has little impact on figures of merit associated with pressurization and heatup, provided the modeling of the RSV capacity and lift pressure are conservative.

Because the validation results for pressurization and heatup events were previously reviewed by the NRC during approval of TR-0516-49416-P-A Rev. 3, are maintained in TR-0516-49416-P-A Rev. 4, and remain applicable, no new validation or markups are required.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-33

Receipt Date: 04/29/2024

Question:

Provide methodology for performing sensitivity studies on the impacts of the axial power shape primary and secondary system pressure, flow and fluid temperature responses.

Section 4.3.1.1.2 states: “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, use of a nominal center-peaked average axial power shape input is appropriate for the system transient analyses.” These statements indicate that sensitivity studies are used to determine the impacts of the axial power shape on primary and secondary system pressure, flow and fluid temperature responses. However, the TR does not provide information on what sensitivities need to be performed and the methods for performing these sensitivity studies and the acceptance criteria.

Please provide proposed markups to the TR to include this information.

Response:

The discussion of axial power shape in Section 4.3.1.1.2 in the current revision of the non-loss-of-coolant accident (non-LOCA) topical report (TR-0516-49416-P, Revision 4) is identical to the discussion in Revision 3 of the non-LOCA topical report that was previously reviewed and approved by the NRC (other than an editorial change to add a reference to the subchannel topical report supplement).

As stated in Section 3.4.3.1 of the safety evaluation report (SER) for Revision 3 of the non-LOCA topical report:

> “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.”

Based on the applicability review demonstrating the similarity between the NuScale Power Module (NPM) for the US460 design (i.e., NPM-20) and the NPM for the US600 design (i.e., NPM-160), no changes were made to the methodology for the axial power shape and sensitivity studies on the axial power shape were not repeated.

Because (1) the treatment of axial power shape has not changed since the prior NRC approval, (2) the NRC previously audited the axial power shape sensitivity studies, and (3) no new sensitivity studies have been performed, there is no reason to provide further information about the topic. No markup is provided because the level of information in Revision 4 of the topical report is consistent with what was sufficient for prior NRC approval of Revision 3.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-34

Receipt Date: 04/29/2024

Question:

Provide the methodology for performing the sensitivity study on the energy deposition factor and how is the bounding energy deposition factor is determined.

The TR state that a bounding high energy deposition factor that results in all energy being deposited in the fuel is used in the non-LOCA analyses and sensitivity studies on this parameter confirm that its effect on the system response is not significant with respect to demonstrating margin to acceptance criteria for NPM events addressed by the non-LOCA evaluation model due to the MPS design and selection of the analytical limits to actuate reactor trip. However, the TR does not include information on how to perform sensitivity studies on this parameter, nor on how the bounding factor is determined.

Please provide proposed markups to the TR.

Response:

As described in Section 4.3.1.1.3 of the non-loss-of-coolant accident (non-LOCA) topical report (TR-0516-49416-P, Revision 4), the bounding high energy deposition factor "results in **all** energy being deposited in the fuel," (emphasis added). In other words, the energy deposition factor is 1.0. {{

}}$^{2(a),(c)}$

The discussion of energy deposition factor in Section 4.3.1.1.3 in the current revision of the non-LOCA topical report is identical to the discussion in Revision 3 of the non-LOCA topical report that was previously reviewed and approved by the NRC (other than an editorial change to replace "reactor trip system" with "MPS"). {{

}}$^{2(a),(c)}$

{{

}}2(a),(c)

As stated in Section 3.4.3.1 of the safety evaluation report (SER) for Revision 3 of the non-LOCA topical report:

> “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.”

Based on the applicability review demonstrating the similarity between the NuScale Power Module (NPM) for the US460 design (i.e., NPM-20) and the NPM for the US600 design (i.e., NPM-160), no changes were made to the energy deposition factor and sensitivity studies on the deposition factor were not repeated.

Because (1) the treatment of energy deposition factor has not changed since the prior NRC approval, (2) the NRC previously audited the energy deposition factor sensitivity studies, and (3) no new sensitivity studies have been performed, there is no reason to provide further information about the topic. No markup is provided because the level of information in Revision 4 of the topical report is consistent with what was sufficient for prior NRC approval of Revision 3.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-35

Receipt Date: 04/29/2024

Question:

Clarify how the addition of boron baskets, venturis to the RVVs and RRVs, and/or removal of the IABs from RVVs impact the NRELAP5 models for the Non-LOCA events as identified in Section 6.

Currently, the models are similar to models for Rev 3. The staff notes that significant design changes exist between the NPM-20 and NPM-160; Rev. 3 of this TR is developed for NPM-160. {{

}}$^{2(a),(c)}$

Please provide proposed markups to the TR.

Response:

The boron baskets, reactor vent valves (RVVs) and associated components, and reactor recirculation valves (RRVs) and associated components, are features that only perform a function during emergency core cooling system (ECCS) operation. Analysis of Chapter 15 events that include ECCS operation is outside the scope of the non-loss-of-coolant accident (non-LOCA) topical report and is performed using other methodologies as described in Section 1.2 of the non-LOCA topical report. Therefore, the performance of these features is not relevant to the non-LOCA topical report.

The boron baskets' role in reactivity control is addressed in the extended passive cooling and reactivity control methodology (XPC) topical report. The non-LOCA topical report provides an appropriate cross-reference (i.e., Reference 26) to the XPC topical report for this reason. The venturis in the RVVs and RRVs are valve internal components, and therefore only have an impact when the valves are open. Events where RVVs and RRVs open, either as the initiating event or as part of ECCS actuation in response to the initiating event, are addressed in the

LOCA topical report, with long-term ECCS cooling addressed separately in the XPC topical report. The non-LOCA topical report provides appropriate cross-references to the LOCA topical report and XPC topical report for this reason (i.e., References 2 and 26, respectively). {{

}}2(a),(c)

The modeling of the containment vessel is described in Section 6.1.7 and shown in Figure 6-16 of the non-LOCA topical report. {{

}}2(a),(c)

The containment volume is an input parameter {{

}}2(a),(c) The containment volume for the NPM-160 is used for analysis of the NPM-160. Likewise, the containment volume for the NPM-20 is used for analysis of the NPM-20. The impact of the NPM-20 design features (different from NPM-160) identified in the audit question on the containment volume is small relative to the total containment volume. In addition, the containment volume has minimal impact on non-LOCA events in the Final Safety Analysis Report (FSAR) Chapter 15. The non-LOCA events where containment volume may be relevant are those that may credit the high containment pressure signal: steam line breaks inside containment (FSAR Section 15.1.5), loss of containment vacuum and containment flooding (FSAR Section 15.1.6), and feedwater line breaks inside containment (FSAR Section 15.2.8). For the double-ended guillotine steam line and feedwater line breaks inside containment, the pressurization of the containment is rapid and results in immediate reactor trip (i.e., within the first few seconds), demonstrating that the results are insensitive to small changes in containment volume. For the loss of containment vacuum and containment flooding, most cases, including the limiting case presented in FSAR Section 15.1.6, do not result in the high containment pressure signal being reached, demonstrating that the results are also insensitive to small changes in containment volume. Therefore, containment volume is not a key parameter for non-LOCA analysis and small changes in volume due to the design features described in this audit question, from whether explicitly accounting for them or not, have no impact.

Note that containment volume is a more important consideration when determining peak containment pressure and temperature {{

}}2(a),(c) This consideration is outside the scope of the non-LOCA topical report.

Because the NPM-20-specific design features identified in the audit question do not function during non-LOCA analysis and their impact on containment volume as an input parameter is minimal, the features do no impact the evaluation model described in the non-LOCA topical report. No markups are required.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-36

Receipt Date: 05/06/2024

Question:

The TR states: “For transients where operation of the normal rod control function provides more adverse consequences of the transient, the control rod movement is modeled to increase the power response during the transient.” However, the TR does not include methodology for achieving this goal. Please explain how to model the control rod movement, i.e., a methodology, to ensure that the modeling of rod movement results in more adverse consequences of the transient. For the same event, the staff notes that revision 3 of this TR states: “For transients where operation of the normal rod control function provides more adverse consequences of the transient, the highest integral control rod bank worth (over time in cycle) is modeled, in conjunction with the maximum control bank withdrawal speed and bounding input for MTC, to increase the power response to the modeled control rod movement.” The applicant needs to revise the TR to include the same level of specifications for the evaluation methods or new approach that includes sufficient conservatisms together with the technical bases.

Please provide proposed markups.

Response:

The audit question identifies a change made in TR-0516-49416-P, “Non-Loss-of-Coolant Accident Analysis Methodology,” between Revision 3 (previously approved by the NRC) and Revision 4 (currently under review). Relevant excerpts from TR-0516-49416-P are identified below.

Section 4.3.1.1.1, “Reactor Kinetics Model” in NRC-approved Revision 3:

> “For transients where operation of the normal rod control function provides more adverse consequences of the transient, the highest integral control rod bank worth (over time in cycle) is modeled, in conjunction with the maximum control bank withdrawal speed and

bounding input for MTC, to increase the power response to the modeled control rod movement."

Section 4.3.1.1.1, "Reactor Kinetics Model" in Revision 4:

"For transients where operation of the normal rod control function provides more adverse consequences of the transient, the control rod movement is modeled to increase the power response during the transient."

In the previous approach, maximum rod worth and maximum rod speed were both used to maximize the reactivity insertion associated with the control system response to temperature error (i.e., a temperature difference from the control setpoint). In the current approach, the maximum rod worth continues to be used. However, the rod speed is treated nominally. The use of a nominal rod speed, while a change to TR-0516-49416-P Section 4.3.1.1.1, is consistent with the discussion in TR-0516-49416-P Section 6.3.1.3 that states "[t]he control rod controller uses design data to model a calculated rate of reactivity insertion due to maximum or nominal rod movement rates." The TR-0516-49416-P Section 6.3.1.3 discussion of using maximum or nominal rod movement rates in Revision 4 is unchanged from Revision 3 that was reviewed and approved by the NRC.

Final Safety Analysis Report (FSAR) Section 15.4.1.2 identifies that the maximum rod speed is 15 in/min and the nominal rod speed is 6 in/min. In the rod control system model, the rod speed increases as the absolute value of the temperature error increases, up to the nominal rod speed of 6 in/min for absolute value temperature errors of 5 degrees F and larger. {{

}}$^{2(a),(c)}$

As a result of this modeling approach, the reactivity insertion caused by the rod control system (when enabled) varies with the size of the temperature decrease (i.e., cooldown). {{

}}$^{2(a),(c)}$

Although the rod speed is limited to the nominal 6 in/min rather than the maximum 15 in/min, the overall approach is still conservative. The rod control design prohibits rod withdrawal at full

power as described in TR-0516-49416-P Sections 6.3.1.3 and 7.1.2. However, this design feature is ignored in the analyses where automatic rod control is assumed to exacerbate the transient. {{

}}[2(a),(c)] The result is that power can increase well above 100 percent due to the combination of the initiating event cooldown and the rod control response. For example, FSAR Figures 15.1-3, 15.1-12, 15.1-23, and 15.1-31 show a reactor power response above 100 percent power that is primarily driven by the rod control modeling with withdrawal allowed above 100 percent power.

Sections 4.3.1.1.1, 6.3.1.3, and 7.1.2 of TR-0516-49416-P are revised aMarkups of the affected changes, as described in the response, are provided below. Note that while the associated document itself is identified as having export controlled information, none of the attached markup pages from the document contain export controlled information. shown in the attached markups to provide details to ensure the conservatism of the modeling approach.

Markups of the affected changes, as described in the response, are provided below:

- Effective delayed neutron decay constant of group i (λ_i)
- Prompt neutron generation time (Λ)
- Reactivity feedback
- Reactivity changes from control rod movement (normal controls), or scram
- Decay heat model input

In the non-LOCA evaluation model, the input to the point kinetics model is specified to give a conservatively high power response prior to actuation of reactor scram. The power response is biased by the input accounting for the reactivity feedback effects of moderator temperature, fuel temperature, and the normal control rod movement. After a reactor scram signal, the negative reactivity associated with insertion of the control and safety banks is conservatively modeled as described in Section 7.1.5.

The moderator temperature coefficient (MTC) is a measure of the relative change in reactivity associated with a change in moderator (coolant) temperature. The Doppler temperature coefficient (DTC) is a measure of the relative change in the reactivity as the fuel temperature changes. In the non-LOCA evaluation model, reactivity feedback effects from moderator temperature changes and fuel temperature changes are conservatively bounded as described in Section 7.1.5. Negative reactivity insertion due to void generation impacts on moderator density is conservatively neglected.

Audit Question A-NonLOCA.LTR-36

As described in Section 6.0 and Section 7.1.2, the rod control system and associated control logic are incorporated into the NRELAP5 model to allow simulation of reactivity changes (negative or positive) associated with normal control rod movement in response to postulated transients. In cooldown events, the normal rod control function attempts to increase average temperature by withdrawing control rods and therefore adding positive reactivity. For transients where operation of the normal rod control function provides more adverse consequences of the transient, the control rod movement is modeled to increase the power response during the transient. A conservative power response is ensured by using maximum rod worth and maximum rod speed for the control rod movement. A conservative power response is also ensured when using nominal rod speed if the 'insert only' mode is neglected as described in Section 6.3.1.3.

The decay heat power contribution is conservatively bounded high or low, as appropriate for the specific transient, as described in Section 7.1.5.

Appropriate input based on the core design is used for other parameters needed as input to the point reactor kinetics model.

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with a control rod controller and a boron controller. The control rod controller uses design data to model a calculated rate of reactivity insertion due to maximum or nominal rod movement rates. {{

}}2(a),(c) Neither controller accounts for all the actual core physics including the effect of xenon or other decay products or poisons that could be expected with control rod repositioning.

Audit Question A-NonLOCA.LTR-36

The average coolant temperature is controlled by adjusting core power, which is accomplished by moving the control rods or changing the boron concentration of the reactor coolant. The choice of which method is based on the desired rate of change for core power. The control rods are moved to achieve faster power changes to meet the target average coolant temperature; slower power changes are accomplished by changing the boron concentration of the reactor coolant. At full power, the rod control system is set to 'insert only' mode to prevent automatic withdrawal of the control rods during a transient. If this design feature is neglected in the model, the rod control system will allow control rods to be withdrawn even if reactor power is at or above full power, resulting in a conservative power response to a transient.

6.3.1.4 Steam Pressure Control (Nonsafety-related)

In an NPM design, the turbine throttle and bypass valves are used to control steam pressure at the programmed values, {{

}}2(a),(c)

6.3.1.5 Feedwater and Turbine Load Control (Nonsafety-related)

An NPM prototypic control scheme design for the feedwater system is based on turbine load demand. The feedwater pumps are variable speed and can provide variable flow for module operations over a wide range of power without adjustments to the feedwater regulating valve. {{

}}2(a),(c)

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Core Average Coolant Temperature Control

Audit Question A-NonLOCA.LTR-36

The average coolant temperature is controlled by moving the control rods or changing the boron concentration of the reactor coolant. This selection is based on the desired rate of change for core power. Hence, the control rods are moved to achieve faster power changes to meet the target average coolant temperature; slower power changes are accommodated by changing the boron concentration of the reactor coolant. At full power, the rod control system is set to 'insert only' mode to prevent automatic withdrawal of the control rods during a transient. Neglecting this feature, if assumed, is conservative.

Steam Pressure Control

Steam pressure is controlled to the desired value using the turbine throttle valves or the turbine bypass valves. The effect of these valves to a change in steam pressure is considered for the non-LOCA transient analyses.

Turbine Load Control

The mass flow rate and pressure provided by the feedwater pump is used to meet the desired turbine load, which reflects the power generation rate. The impact of the feedwater pumps continuing to operate until the feedwater line is isolated is considered for the non-LOCA transient analyses when DHRS is actuated.

Containment pressure control

The containment pressure is established at sub-atmospheric conditions via operation of the containment evacuation system. The impact of this system continuing to operate is considered for the non-LOCA transient analyses.

7.1.3 Loss of Power Conditions

This section defines the term "loss of normal power" as applied to an NPM; describes the various power supplies (AC and DC); and, explains how the loss of these power supplies is treated by the non-LOCA transient analyses.

7.1.3.1 Background

Chapter 15 of the SRP (Reference 15) does not ordinarily consider a loss of offsite power for events that require a malfunction of an active system for which power must be available; however, exceptions are made for some reactivity initiated events. The role of offsite power is less defined for an NPM plant than for traditional plants for several reasons, but the use of natural circulation for normal operation and safety systems is the fundamental reason. Consequently, these design features limit the impact of a power loss to an NPM plant compared to a traditional plant design that relies on forced circulation.

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-37

Receipt Date: 05/06/2024

Question:

Provide a methodology that must be used to determine the time-in-cycle core average burnup is used to calculate the fuel thermal conductivity as a function of temperature or incorporate the approach in the revision 3 of this TR with updated burnup values with which the fuel thermal conductivities are calculated.

TR Section 4.3.1.2.2, “Fuel Rod Material Properties,” states: “the fuel pellet properties, UO2 properties are used to reflect the core average response. Since the fuel pellet thermal conductivity is a function of burnup and fuel temperature, and the fuel pellet thermal conductivity degrades with burnup, a representative time-in-cycle core average burnup is used to calculate the fuel thermal conductivity as a function of temperature.” However, the TR does not provide a methodology for determining this value. The staff asked the same question during its review of Revision 3 of this TR (ML19039A090). {{

}}$^{2(a),(c),ECI}$

Please provide proposed markups to the TR.

Response:

Section 2.2.3 of EC-0000-7782, Revision 3, “NPM-20 NRELAP5 Basemodel” (previously provided in the electronic reading room (eRR)) describes the treatment of fuel exposure in the NRELAP5 basemodel for the NuScale Power Module (NPM) for the US460 Standard Design Approval Application (i.e., the NPM-20). {{ }}$^{2(a),(c),ECI}$

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}}2)a,(c),ECI the burnup limit of approximately 60 GWd/MT described in Final Safety Analysis Report Chapter 4.

The treatment of fuel exposure in the NPM-20 basemodel in EC-0000-7782 is then expanded in the non-loss-of-coolant accident (non-LOCA) basemodel in EC-0000-8507, Revision 2, "NPM-20 NRELAP5 Non-LOCA Model" (provided in the eRR with this response).
{{

}}2)a,(c),ECI The tables of biases in Section 7.2 of TR-0516-49416-P, Revision 4, "Non-Loss-of-Coolant Accident Methodology," identify fuel temperature is biased low, nominal, or high, depending on the event. {{

}}2)a,(c),ECI

The methodology in Section 4.3.1.2.2 of TR-0516-49416-P, Revision 4, specifies that a representative time-in-cycle core average burnup be used. {{

}}2)a,(c),ECI

As identified by the reference in this audit question, the previous version of TR-0516-49416-P that was reviewed and approved by the NRC (i.e., Revision 3) did not provide a more detailed discussion of the burnup or a specific burnup value in the topical report itself. Instead, the information was provided by NuScale in support of an NRC audit. Consistent with the approach taken during the review and approval of TR-0516-49416-P-A, Revision 3, NuScale has provided

the more detailed information in this audit response and is not including the information in the topical report itself. Therefore, no markups are required.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-38

Receipt Date: 05/06/2024

Question:

Please submit or provide a summary of DHRS scaling analyses from the following three supporting documents for the Non-LOCA TR:

1. ER-126485 Rev. 0, “NRELAP5 Applicability to NPM-20 Non-LOCA Analysis,” and
2. EE-107403 Rev. 2, “NIST-2 SG-DHRS IET Pretest Predictions and Scaling Analyses.”
3. EC-127301, “NIST-2 Non-LOCA IET Data Evaluation and NRELAP5 Assessment R2.”

Please provide proposed markups to the TR to include a summary of this information.

Response:

Each of the three documents identified in the audit question has previously been provided in the electronic reading room (eRR). A summary of the DHRS scaling analyses, including markups to the topical report, was included in the NuScale response to the NRC request for supplemental information 17 (RSI-17), submitted under the following:

> NuScale letter LO-145424, “NuScale Power, LLC Submittal of Responses to NRC RSIs on Topical Report ‘Methodology for the Determination of the Onset of Density Wave Oscillations (DWO),’ TR-131981, Revision 0,” dated July 14, 2023, available in ADAMS at ML23195A091.

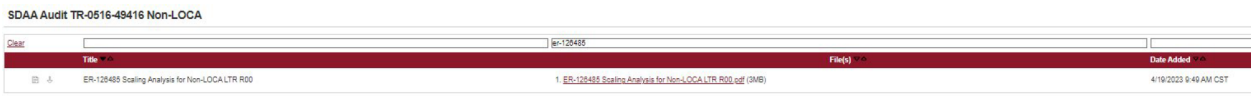
Feedback received on 6/18/2024 states the following:

“NuScale needs to provide a summary of the DHRS scaling analyses from the following three supporting documents for the Non-LOCA TR:

1. ER-126485 Rev. 0, “NRELAP5 Applicability to NPM-20 Non-LOCA Analysis”
2. EE-107403 Rev. 2, “NIST-2 SG-DHRS IET Pretest Predictions and Scaling Analyses”
3. EC-127301, “NIST-2 Non-LOCA IET Data Evaluation and NRELAP5 Assessment R2”

Please provide proposed markups to the TR that include a summary of the information.”

As identified in the initial audit response posted to the eRR on 5/9/2024, the initial audit question and the feedback received 6/18/2024 are the same. Please see the certrec entries which identify the documents available within the Non-LOCA eRR:


SDAA Audit TR-0516-49416 Non-LOCA

Clear | er-126485

Title	File(s)	Date Added
ER-126485 Scaling Analysis for Non-LOCA LTR R00	1. ER-126485 Scaling Analysis for Non-LOCA LTR R00.pdf (3MB)	4/19/2023 9:49 AM CST

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-39

Receipt Date: 05/13/2024

Question:

TR Section 7.2.10 (Loss of Normal Feedwater Flow) does not include descriptions of the events that may cause loss of normal feedwater flow AOO. The scenarios should be presented as they establish the limiting case for peak RCS pressure, peak SG pressure, and minimum MCHFR.

Response:

The non-loss-of-coolant accident (non-LOCA) topical report (TR-0516-49416-P, Revision 4) describes that “[a] postulated fault results in a partial or complete loss of feedwater flow, and the water in the steam generators boils off” in Section 7.2.10.1. Although the possible causes of the loss of feedwater flow are not described, the statement is identical to the statement in Section 7.2.10.1 of the previous revision of the non-LOCA topical report that was reviewed and approved by the NRC (TR-0516-49416-P-A, Revision 3). Because the information requested to be provided in the topical report in this audit question was not required to be included in the topical report to support the previous NRC review and approval, no changes to the topical report are required.

Note that potential causes of a loss of feedwater event are described in Final Safety Analysis Report (FSAR) Section 15.2.7.1. There is nothing specific about these potential causes that impact the methodology. Therefore, inclusion of the specific potential causes in the topical report is not necessary to describe the methodology for analyzing the event.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-40

Receipt Date: 05/28/2024

Question:

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}}2(a),(c)

Please provide proposed markups to the TR to include the requested information.

Response:

This is a revised response to audit question A-NonLOCA.LTR-40 to address NRC feedback received on July 12, 2024; the original response posted to the electronic reading room (eRR) on June 10, 2024 is unchanged in the eRR. The revised portion of the response begins with the section identified as “Response to Additional NRC Feedback.”

The TR-0516-49416-P, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 4, Section 4.3.5 identifies that VIPRE-01 is used for the subchannel analysis of non-loss-of-coolant accident (non-LOCA) events. The acceptance criteria of minimum critical heat flux ratio (MCHFR) and maximum fuel centerline temperature are therefore evaluated with VIPRE-01.

Section 4.3.5 of TR-0516-49416-P describes that NRELAP5 is used to provide input to VIPRE-01. {{

}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

Response to Additional NRC Feedback

On July 12, 2024, the NRC provided the following feedback:

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}}2(a),(c)

In response to the NRC feedback, Section 4.3.5 of TR-0516-49416-P, Revision 4, is revised as shown in the attached markups {{

}}2(a),(c)

Markups of the affected changes, as described in the response, are provided below:

limiting initialization for a given transient progression to ensure the maximum power and core inlet fluid temperature are reached prior to reactor trip system actuation. For example, in the case of a heatup event, the RCS increases in temperature, causing a pressurizer insurge and subsequent increase in pressure. The limiting CHF scenario is the transient progression that results in the highest core outlet temperature at the time of reactor trip on high pressure, which is generally the faster heatups where the pressurizer initialization is biased to delay the high pressure trip.

Audit Question A-NonLOCA.LTR-40

For some transients, a spectrum of cases is analyzed from the limiting initialization.
{{

Audit Question A-NonLOCA.LTR-23

}}2(a),(c)

After the system transient analysis calculations are performed and assessed, for events that require subchannel analysis, a number of cases are identified as limiting for MCHFR. For the limiting cases selected, the required system transient parameters are tabulated as a function of time for input to the downstream subchannel analysis calculations, to calculate margin to CHF.

The system transient parameters are provided for subchannel analysis for sufficient time for the subchannel analyses to demonstrate that the MCHFR has occurred, typically 10-15 seconds following reactor trip.

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-41

Receipt Date: 06/10/2024

Question:

Tables 7-58, 7-62, and 7-66 discuss loss of AC power scenarios for the uncontrolled control rod assembly (CRA) bank withdrawal from subcritical or low power startup conditions, uncontrolled CRA bank withdrawal at power, and control rod misoperation events. However, no discussion of loss of normal DC power is provided. Provide topical report markups discussing the loss of normal DC power scenarios considered for these transients.

Response:

TR-0516-49416-P, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 4, Tables 7-58, 7-62, and 7-66 discuss loss of alternating current (AC) power but do not discuss direct current (DC) power. The discussion of AC power and not DC power in these tables is the same as in Tables 7-58, 7-62, and 7-66 of TR-0516-49416-P-A, Revision 3, that was previously reviewed and approved by the NRC.

In the NRC’s Safety Evaluation Report (SER) for TR-0516-49416-P-A, Revision 3, a limitation and condition was included that stated that “[a]n 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,” with emphasis added.

The process being used for TR-0516-49416-P, Revision 4 is the same as previously used for TR-0516-49416-P-A, Revision 3. The event-specific information is provided in the Final Safety Analysis Report (FSAR). The NRC has already requested additional information on loss of power scenarios for the events associated with Tables 7-58, 7-62, and 7-66 in audit question

A-15.4.1-2. The NuScale response and supplemental response to audit question A-15.4.1-2 provides the necessary information supporting the FSAR.

Therefore, there is no need to revise Tables 7-58, 7-62, and 7-66 of TR-0516-49416-P, Revision 4, because the information is provided in the FSAR rather than the topical report.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-43

Receipt Date: 06/17/2024

Question:

Several sections of TR-0516-49416, Revision 4, include language regarding operator actions, the implementation of which has not clearly been consistent with Limitation & Condition 5 of TR-0516-49416-P-A, Revision 3, which states, “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.” The staff intends to place a similar L/C on the currently submitted Non-LOCA LTR revision. As such, NuScale is requested to propose markups to remove or sufficiently reword any and all statements that explicitly or implicitly discuss operator actions. For example, the statements could be reworded to clearly identify this assumption within the context of event escalation. For example, if the initiating event is a small leak/break it may be assumed the leak/break does not get worse during the transient timeframe.
Operator actions are considered to mitigate design basis events if the action is relied on in the analysis within the 72 hours following the initiating event to improve the results relative to the applicable figures of merit for a particular set of initial conditions.
There are sections throughout the LTR that contain such statements, and the staff has specifically noted the ones below that are necessary to be removed or significantly reworded (if reworded it is necessary to mention that any credit must be sought through specific NRC approval via the design review for such credit). If there are other such statements in other sections, they should also be addressed:

- TR Section 4.3.6 states that input to radiological analysis may assume valve isolation times based on MPS response or operator actions, as operator actions may be taken to prevent abnormal operating events from resulting in more severe events.
- TR Section 7.1.7 states that operator actions credited for the non-LOCA transient analyses

are typically justified and consistent with plant operating procedures. For an NPM, there are no occasions where operator action is credited for event mitigation by the non-LOCA transient analyses. Operator actions taken to prevent abnormal operating events from resulting in more severe events are excluded from consideration. For example, very small leaks of reactor coolant from the CVCS that do not result in automatic reactor trip for more than 30 minutes are considered an abnormal operating event where operators are expected to identify and isolate the leak before it results in a more severe event..

- TR Section 7.2.16 states that surveillance of the pool boron concentration at appropriate intervals prevents prevent boron dilution in Mode 5. This implies that the operator needs to identify and isolate the source of the diluted water.

Response:

TR-0516-49416, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” is revised as indicated in the attached markups.

accident radiological analysis. The conservative bias directions for the transient analysis input to the accident radiological analysis are:

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

  Basis: For a constant radionuclide concentration in the RCS, the greater the mass released outside of containment, the more severe the radiological consequences.

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

  Basis: Reference 8 describes how iodine spiking is accounted for in the accident radiological analyses. During a transient progression, changes in reactor power, RCS average temperature, or RCS pressure could result in iodine spiking, changing the radionuclide concentration in the RCS. Consistent with Standard Review Plan Section 15.6.2 (Reference 9), for accident radiological calculations assuming a coincident iodine spike (iodine spike occurring during the event), the iodine spiking is assumed to begin at the time of reactor trip as the result of the reactor shutdown or depressurization of the primary system. In some cases, particularly for smaller breaks in RCS piping, the time between reactor trip and isolation of the break flow could be extended compared to larger break sizes. The increased time between reactor trip and isolation increases the time of mass release when the RCS radionuclide inventory reflects iodine spiking.

Audit Question A-NonLOCA.LTR-43

For each transient analysis case identified, the input provided for accident radiological analysis includes:

- Time of reactor trip if it is calculated to occur
- Isolation time, at which point release of RCS fluid outside of containment is stopped~~. The isolation may be due to MPS response or due to operator action (as identified in Section 7.1.7, operator actions may be taken to prevent abnormal operating events from resulting in more severe events).~~
- RCS fluid mass release outside of containment as a function of time
- System transient response parameters such as
  - Reactor power as a function of time
  - RCS average temperature as a function of time
  - RCS pressure as a function of time
  - Secondary side feedwater and steam flow rates as a function of time

As an alternative to transient analysis, the accident radiological analysis can use bounding values for both mass release and isolation times. An example bounding approach is provided as follows. If the MPS of an NPM design includes reactor trip and isolation setpoints based on pressurizer level, these setpoints effectively limit the RCS inventory that can be released to the difference between a high pressurizer level and the pressurizer level associated with the isolation, and limit the time between

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reverse flow is experienced during a break in the feedwater system piping. (Section 7.1.4)

Audit Question A-NonLOCA.LTR-43

~~Operator actions credited for the non-LOCA transient analyses are typically justified and consistent with plant operating procedures.~~ For an NPM, there are no occasions where operator action is credited for event mitigation by the non-LOCA transient analyses. ~~Operator actions taken to prevent abnormal operating events from resulting in more severe events are excluded from consideration. For example, very small leaks of reactor coolant from the CVCS that do not result in automatic reactor trip for more than 30 minutes are considered an abnormal operating event where operators are expected to identify and isolate the leak before it results in a more severe event.~~

7.2 Event Specific Methodology

The non-LOCA event simulations are performed using conservative methodologies. Pertinent event-specific methodologies, as well as inputs and results for a representative NPM for non-LOCA event simulations are presented herein, and compared with the regulatory acceptance criteria listed in Table 7-4. Section 4.1 contains additional discussion of Chapter 15 design basis events and acceptance criteria.

All acceptance criteria are considered for each event, and the criteria with the potential for being challenged are identified and evaluated in further detail (i.e., overcooling events do not challenge the acceptance criterion for primary side pressure, but may challenge the CHFR acceptance criteria). An event-specific parameter that is relevant to the acceptance criterion may be described as “challenging” in the event-specific summary, however, it is recognized that the parameter may not present the worst challenge for any event.

Table 7-4 Regulatory acceptance criteria

Description	AOO Criteria	IE Criteria	Accident Criteria
RCS pressure	≤ 110% of design	≤ 120% of design	≤ 120% of design
SG pressure	≤ 110% of design	≤ 120% of design	≤ 120% of design
CHFR(1)	> Limit	Note (3)	Note (3)
Maximum fuel centerline temperature(1)	≤ Limit	Note (3)	Note (3)
Containment integrity(2)	< Limits	< Limits	< Limits
Escalation of an AOO to an accident (AOO) or consequential loss of system functionality (IE or accident)?	No	No	No
Dose(1)	Normal operations	< Limit	< Limit

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determine the shutdown margin available after isolation of the DWS, and the time at which the shutdown margin would be lost if the dilution source is not terminated.

Mode 4 (Transition)

In this mode of plant operation, all CVCS connections to an NPM are disconnected, isolated, or locked out. Thus, the possibility of a design-basis inadvertent decrease in boron concentration is precluded.

Mode 5 (Refueling)

Audit Question A-NonLOCA.LTR-43

In this mode of plant operation, the Technical Specifications require the pool boron concentration to be sufficient to have appropriate shutdown margin. For some NPM designs, the Technical Specifications also require the pool level to be maintained within a narrow range. Surveillance of the boron concentration, and level if applicable, of the refueling pool is performed at appropriate intervals is expected to prompt operator actions in accordance with Technical Specifications during an inadvertent dilution of the pool; such operator actions are not credited. ~~to prevent significant inadvertent dilution from flow paths to the reactor pool, or proximate water sources such as fire mains or feedwater piping.~~

The relevant acceptance criteria, SAF, and LOP scenarios are listed in Table 7-72.

Table 7-72 Acceptance criteria, single active failure, loss of power scenarios - inadvertent decrease in boron concentration

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	MCHFR is challenged during this reactivity anomaly event. (Reactivity insertion rates are insufficient to challenge fuel centerline temperature.)
No single failure	The challenging cases typically occur when all equipment is operational.
No loss of power	The challenging cases typically occur when AC power is available to the CVCS equipment for the event duration.

7.2.16.2 Acceptance Criteria

Evaluation of the most challenging case(s) relative to the acceptance criteria is presented in Table 7-73.

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-44

Receipt Date: 06/17/2024

Question:

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}}2(a),(c),ECI

Please provide proposed markup to the TR.

Response:

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}}2(a),(c),ECI

{{

}}2(a),(c),ECI

Markups of the affected changes, as described in the response, are provided below:

5.3.6 Heat Transfer Correlation Comparison

5.3.6.1 Background

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}}2(a),(c)

5.3.6.2 Computational Fluid Dynamics Calculations

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A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

}}2(a),(c),ECI

A-NonLOCA.LTR-2

5.3.6.2.1 Model 1 Analysis and Results

A-NonLOCA.LTR-2, A-NonLOCA.LTR-44

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}}2(a),(c),ECI

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{{

A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

A-NonLOCA.LTR-2

}}2(a),(c),ECI

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}}2(a),(c),ECI

A-NonLOCA.LTR-2

Table 5-18 ~~Comparison of correlation to CFD results – CFD model 2~~

~~{{~~

~~}}2(a),(c)~~

A-NonLOCA.LTR-2

~~{{~~

~~}}2(a),(c)~~

A-NonLOCA.LTR-2

5.3.6.2.4 Computational Fluid Dynamics Conclusions

A-NonLOCA.LTR-2, A-NonLOCA.LTR-44

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}}2(a),(c),ECI

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-44S

Receipt Date: 07/29/2024

Question:

Audit question A-NonLOCA.LTR-44 was received by NuScale on June 17, 2024. NuScale provided a response on July 16, 2024. On July 29, 2024, NuScale received the following written feedback from the NRC:

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}}2(a),(c),ECI

Response:

This response is a supplement to the July 16, 2024 response to address the feedback from July 29, 2024; the July 16, 2024 response remains unchanged in the electronic reading room (eRR).

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}}2(a),(c),ECI

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}}[2(a),(c),ECI]

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}}2(a),(c),ECI

Other than the two changes above specific to this audit question, the rest of the changes shown in the markup pages come from the response to audit question A-NonLOCA.LTR-2 that was previously accepted and closed by the NRC. The changes should be retained because they were previously deemed necessary by the NRC to close audit question A-NonLOCA.LTR-2.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-45

Receipt Date: 06/17/2024

Question:

The current revision of the non-LOCA LTR contains provisions to provide bounding mass release values to the radiological analysis rather than mass release values based on transient results. However, the topical report only provides an example for an unspecified transient but does not provide a method to select bounding values for particular transients. Therefore, NRC staff approval will be of use of bounding values in lieu of values derived from transient analysis, but the method by which applicants and licensees select bounding values and the justification that said values are bounding is not reviewed and approved with this topical report and will be subject to application-specific NRC staff review. If more extensive approval is desired, provide LTR markups that specify the method by which bounding values are selected for specific transients with either justification that this method results in bounding values, or description of checks that will be performed (e.g., comparison with transient results as suggested in the LTR example) to provide assurance that values are bounding.

Response:

In Revision 4 of TR-0516-49416-P, “Non-Loss-of-Coolant Accident Analysis Methodology,” NuScale revised Section 4.3.6 to allow for use of bounding values for radiological analysis as an alternative to providing event-specific transient results into the downstream radiological analysis. Section 4.3.6 identified an example method to determine bounding values. There are other methods that could be used to determine bounding values, depending on the level of conservatism desired. The specific bounding values, and the justification of those values, is provided on an application-specific basis. In the case of the Standard Design Approval Application (SDAA) for the US460 design, the bounding values and associated justification are provided in Chapter 15 of the Final Safety Analysis Report (FSAR).

Given that Section 4.3.6 of TR-0516-49416-P only describes an example, NuScale has no objection to the approach identified by the NRC in the audit question. Therefore, NuScale proposes no changes to TR-0516-49416-P in response to this audit question.

Note that the change to TR-0516-49416-P was to add the option to use bounding values as an alternative to the previously approved event-specific transient methodology (i.e., not to replace it). Therefore, the NRC approval of the use of bounding values should not negate the validity of the previously approved event-specific transient methodology.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-49

Receipt Date: 06/24/2024

Question:

The NRC staff notes that several parameters identified in TR 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. The TR also states: “The initial RCS flow rate is biased to either end of the range expected for normal operation. As an example, consider a normal operational range of 535 kg/s to 690 kg/s for RCS flow rate at 100 percent RTP. For this situation, the initial RCS flow rate is set to either 535 kg/s or 690 kg/s.”

Please provide a method that can be used to determine how to count for the interdependence of the initial flow rate and when to set the flow to 535 kg/s or 690 kg/s.

Please provide proposed markup to the TR.

Response:

The NRC is correct that the reactor coolant system (RCS) flow for a natural circulation NuScale Power Module (NPM) is related to the power input by the reactor, the heat removal by the steam generators (SGs), and the hydraulic characteristics of the circuit. Given that reactor power and SG conditions are also set as input for the initial conditions, the RCS flow initial condition is set by adjusting the form loss coefficients in the RCS loop. If low RCS flow is desired as the initial condition, high loss values are selected. If high RCS flow is desired as the initial condition, low loss values are selected. {{

}}$^{2(a),(c)}$

Note that the values of 535 kg/s and 690 kg/s discussed in Section 7.1.6.1 of TR-0516-49416, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology” are identified as examples only. The specific range of RCS flow applicable to an NPM is found in the Final Safety Analysis Report (FSAR) associated with the NPM design. For the Standard Design Approval Application (SDAA) for the US460 design, the RCS flow rate range is identified in FSAR Table 15.0-6. The basis provided with the values provides information consistent with this response regarding the relationship between RCS flow, power, and hydraulic resistance, as well as information regarding biasing the resistance to achieve the desired flow.

Section 7.1.6.1 of TR-0516-49416, Revision 4, does not specify a generic direction of flow biasing to apply. The direction of flow biasing to apply (e.g., low, nominal, high) is addressed in the bias tables associated with each specific event in Section 7.2 of TR-0516-49416, Revision 4. Markups to TR-0516-49416, Revision 4, Section 7.1.6.1 are not required because the process described here is consistent with the process used in the previously approved version of the topical report, TR-0516-49416-A, Revision 3. In the NRC’s prior approval of TR-0516-49416-A, Revision 3, no additional explanation was required to be included in Section 7.1.6.1.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-54

Receipt Date: 06/24/2024

Question:

Please make available for audit the sensitivity studies performed for Main Steam Isolation Valve(s) Closure event. TR Section 72.8 states: "Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently,sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event."

Response:

The quoted discussion from Section 7.2.8 of TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” provides the methodology for the analysis. The methodology identifies that sensitivity studies are performed *as needed*. The focus of this overheating event is on system pressure response.

The application of this methodology for the US460 Standard Design Approval Application (SDAA) is documented in the Final Safety Analysis Report (FSAR). The event corresponding to Section 7.2.8 of TR-0516-49416-P is identified in FSAR Section 15.2.4. The analysis supporting FSAR Section 15.2.4 is documented in EC-117482, Revision 0, “NPM-20 Main Steam Isolation Valve Closure Transient Analysis.” EC-117482 was previously provided in the electronic reading room (eRR) for FSAR Chapter 15 for audit. Section 3.3.4 of EC-117482 identifies the sensitivity studies performed and considers parameters consistently with those identified in Table 7-40 of TR-0516-49416-P as being “varied.” The sensitivity results are summarized in Tables 5-1 and 5-2. The limiting peak system pressures of the cases in EC-117482 are reported in FSAR Table 15.2-12.

In summary, the sensitivity studies described by Section 7.2.8 of TR-0516-49416-P have been previously made available for audit as part of the implementation of the methodology.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-57

Receipt Date: 06/28/2024

Question:

FSAR Section 15.2.7 describes three different event sequences, (i.e., pump failures, valve malfunctions, and loss of AC), for the loss of FW flow event. However, there is no discussion of the three potential initiating events and their sequences in the TR. As discussed in FSAR Section 15.2.7.2, these events establish the limiting case for peak RCS pressure, peak SG pressure, and minimum MCHFR respectively. Please revise TR Section 7.2.10 to include discussions of the three event sequences to make the TR consistent with FSAR Section 15.2.7 analyses.

Please provide proposed markup to the TR.

Response:

This audit question requesting update to Section 7.2.10 is the same as audit question A-NonLOCA.LTR-39. The audit question A-NonLOCA.LTR-39 was sent to NuScale on May 10, 2024, responded to by NuScale on May 15, 2024, and closed by the NRC on June 6, 2024.

Because this issue has already been raised, responded to, and resolved, no further response is necessary. See the response to audit question A-NonLOCA.LTR-39 for more information.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-58

Receipt Date: 06/28/2024

Question:

The staff notes that the basis for PZR Pressure Control PZR spray in Table 7-7 and Table 7-19 of Version 3 of the TR states: {{

}}$^{2(a),(c)}$ These bases are removed from Table 7-7 and Table 7-19 of Revision 4 of the TR. Please clarify if the statement in Version 3 of the TR remains true for the NRELAP5 code. If it is true, please revise the basis for PZR Pressure Control PZR spray in Table 7-7 and Table 7-19 of the TR to the same language as approved in Revision 3 {{

}}$^{2(a),(c)}$

Provide proposed markup to the TR.

Response:

TR-0516-49416-P-A, Revision 3, “Non-Loss-of-Coolant Accident Analysis Methodology,” Section 4.3.5 includes an extensive discussion of the effect of pressure on critical heat flux (CHF). Generally, there is a negative trend in CHF versus pressure (CHF is more limiting as pressure decreases) that changes to a positive trend at low pressures. These trends are consistent with CHF test data. {{

}}$^{2(a),(c)}$

For this reason, the discussion in Tables 7-7, 7-14, and 7-19 of TR-0516-49416-P-A, Revision 3, identifies that {{

}}2(a),(c)

The Standard Design Approval Application (SDAA) for the US460 design uses NPM-20. The NPM-20 has a higher power, higher normal operating pressure, and higher flow than the NPM-160 of the DCA. {{

}}2(a),(c) The sensitivity study results supporting this trend are shown in EC-117453, Revision 2, "Steady-State Subchannel Analysis," by comparison of Tables 4-1 and 4-2. {{

}}2(a),(c) EC-117453 was previously provided in the electronic reading room (eRR) with the response to audit question A-5.4.5-1. Additionally, sensitivities in EC-103521, Revision 3, "VIPRE-01 Sensitivity Analysis for the Core-250B design," Section 4.6 and Figure 4-19 demonstrate that {{

}}2(a),(c) EC-103521 was previously provided in the eRR for the "Statistical Subchannel Audit."

{{

}}2(a),(c) Therefore, TR-0516-49416-P, Revision 4, is accurate for NPM-20.

With the above background, NuScale provides the following response to the NRC request. The NRC request is identified by the indented, italicized text, with the NuScale response identified by the non-indented, non-italicized text.

> *"Please clarify if the statement in Version 3 of the TR remains true for the NRELAP5 code."*

The statement in Sections 4.3.5 and Tables 7-7, 7-14, and 7-19 of TR-0516-49416-P-A, Revision 3, are accurate for the NPM-160 design {{

}}2(a),(c)

"If it is true, please revise the basis for PZR Pressure Control PZR spray in Table 7-7 and Table 7-19of the TR to the same language as approved in Revision 3 ..."

The statements in Sections 4.3.5 and Tables 7-7, 7-14, and 7-19 of TR-0516-49416-P-A, Revision 3, are {{ }}2(a),(c) revised in Revision 4. No further modification to Revision 4 is required.

{{

}}2(a),(c)

{{

}}2(a),(c)

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-59

Receipt Date: 06/28/2024

Question:

LTR Table 7-64 states that letdown is modeled as “disabled” for the uncontrolled CRA withdrawal at power event {{

}}$^{2(a),(c)}$ Provide support for this basis for NRC staff review or revise this disposition in the LTR.

Response:

TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” Table 7-64 identifies initial conditions, biases, and conservatisms to be applied for analyses of the uncontrolled control rod bank withdrawal at power. Table 7-64 identifies that letdown is disabled {{

}}$^{2(a),(c)}$

{{

}}$^{2(a),(c)}$ Pressurizer spray exists specifically to control pressurizer pressure and maintain it within a range below the high pressurizer pressure reactor trip signal. By varying pressurizer spray modeling for this event (on or off), different sequences of events can be

obtained. {{

}}2(a),(c)

Demonstration of the effectiveness of the application of this methodology can be seen in the analysis in Section 15.4.2 of the Final Safety Analysis Report (FSAR) for the US460 design. Table 15.4-4 provides the sequence of events for the limiting MCHFR case (a case with spray modeled as on) and shows that reactor trip occurs on high power, not high pressurizer pressure. Figure 15.4-8 shows that, for this case, the reactor coolant system (RCS) hot temperature is also near the analytical limit of 620 degrees F at the time of reactor trip on high power {{

}}2(a),(c) Figure 15.4-7 shows the pressurizer pressure during the transient is maintained below the high pressurizer pressure reactor trip signal. The “sawtooth” behavior in Figure 15.4-7 indicates the operation of pressurizer spray: pressure increases until spray turns on, decreases during spray operation until spray turns off, then increases again. {{

}}2(a),(c) power and fluid temperature to continue increasing as shown in Figure 15.4-6 until the high power trip occurs. Although a sequence of events table and transient figures are not provided for the limiting linear heat generation rate (LHGR) case (LHGR is used as the figure of merit instead of fuel centerline temperature as discussed in the FSAR), the case associated with the limiting LHGR result in Table 15.4-7 {{

}}2(a),(c)

The analysis supporting FSAR Section 15.4.2 is EC-0000-8672, Revision 0, “Uncontrolled Control Rod Assembly Bank Withdrawal at Power Transient Analysis,” previously provided in the electronic reading room (eRR). {{

}}2(a),(c)

Based on this information, NuScale has addressed the underlying concern in this audit question that control systems that affect pressurizer pressure should be considered for their potential

{{

}}2(a),(c)

Note that audit question A-5.4.5-1 requested further information about pressurizer spray setpoints based on the observed “sawtooth” behavior in FSAR Figure 15.4-7. {{

}}2(a),(c)

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-59S

Receipt Date: 08/02/2024

Question:

Audit question A-NonLOCA.LTR-59 was received by NuScale on June 28, 2024. NuScale provided a response on July 22, 2024. On August 2, 2024, NuScale received the following written feedback from the NRC:

{{

}}2(a),(c)

Response:

This response is a supplement to the July 22, 2024 response to address the feedback from August 2, 2024; the July 22, 2024 response remains unchanged in the electronic reading room (eRR).

{{

}}2(a),(c)

NuScale agrees that high power level and high RCS hot temperature trips may not mitigate cases initiated from lower power levels. However, the design also includes reactor trips based on high average RCS temperature and high power rate. These trips are available to mitigate cases initiated from lower power levels. {{

}}2(a),(c) NuScale analyzes a wide range of reactivity insertion ranges in order to identify the transition points between the various trip signals. Limiting conditions often occur at such transition points. Inclusion of low power cases {{ }}2(a),(c) is therefore evidence that the analysis includes appropriate consideration of the range of reactivity insertions. {{

}}2(a),(c)

{{

}}2(a),(c)

In conclusion, uncontrolled control rod assembly bank withdrawals initiated from power levels below rated power includes a variation of assumptions and initial conditions sufficient to identify potential impacts on figures of merit {{

}}2(a),(c)

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-59S2

Receipt Date: 09/12/2024

Question:

As a follow-up to audit question A-NonLOCA.LTR-59, NRC sent an advanced request for additional information (RAI) on the same topic on August 29, 2024. A clarification call to discuss the advanced RAI was held on September 12, 2024. During the clarification call, NuScale took an action to provide a revised topical report markup to address the NRC concerns from the original audit question A-NonLOCA.LTR-59 and the advanced RAI.

Response:

As described above, NuScale took an action to provide a markup to TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” to address the NRC concerns from the original audit question A-NonLOCA.LTR-59 and the advanced request for additional information (RAI). The markup is attached to this response. A summary of the changes is provided for convenience as follows: {{

}}2(a),(c)

Markups of the affected changes, as described in the response, are provided below:

Table 7-63 Acceptance criteria - uncontrolled control rod bank withdrawal at power

Acceptance Criteria	Discussion
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.14.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-64 are considered in order to identify the bounding transient simulation for MCHFR and maximum fuel centerline temperature.

Audit Question A-NonLOCA.LTR-59

Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	Varied. RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied. Biased to the high condition.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	}}2(a),(c)

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Table 7-64 Initial conditions, biases, and conservatisms - uncontrolled control rod bank withdrawal at power (Continued)

Parameter	Bias / Conservatism	Basis
PZR Pressure Control PZR spray PZR heaters	 Varied. Nominal.	{{ }}2(a),(c)
PZR Level Control Charging Letdown	 Not credited. Varied.~~Disabled.~~	

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-61

Receipt Date: 06/28/2024

Question:

The Non-LOCA TR references “Rod Ejection Accident Methodology,” TR-0716-50350-P, Revision 2. The correct reference should be Revision 3.

Response:

The requested markup was previously provided in the response to audit question A-NonLOCA.LTR-29. Audit question A-NonLOCA.LTR-29 was previously closed by the NRC.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-62

Receipt Date: 06/28/2024

Question:

TR Section 6.1.3 states: {{

}}$^{2(a),(c)}$

Please provide proposed markups to the TR.

Response:

The purpose of the hot channel heat structure described in TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” Section 6.1.3 is to allow critical heat flux (CHF) to be calculated in NRELAP5. {{

}}$^{2(a),(c)}$

Table 1: {{ }}[2(a),(c)]

{{

}}[2(a),(c)]

{{

}}[2(a),(c)]

TR-0516-49416-P, Revision 4, Section 6.1.3 is revised as indicated in the attached markups to {{ }}2(a),(c) consistent with the response above.

{{

}}2(a),(c)

Markups of the affected changes, as described in the response, are provided below:

6.1.2 Core Kinetics

The separable point kinetics model is used to calculate reactivity feedback to the core power from the moderator, the fuel, and decay heat. The various point kinetic parameters are input based on the fuel burnup (for example, new core, BOC, or EOC) and control rod insertion amounts assumed for the analysis. NRELAP5 assumes an infinite core operating time at the initialized power when determining the decay heat power. The fission product decay type is specified as 'gamma-ac' with the 'ans73' model, which calculates decay heat in accordance with the 1973 ANS standard while adding the contribution from actinides. A fission product yield factor of 1.0 is specified in the base models, which can be changed to suit the scenario being analyzed.

Control variable inputs to the core kinetics model are used to simulate control rod movement and reactivity feedback from a variety of parameters including, but not limited to fuel temperature, moderator temperature, and moderator density. Trips are used to enable or disable each reactivity feedback mechanism as needed. The fuel and moderator feedback controllers utilize volume weighting of the fuel and moderator temperatures to specify bounding reactivity feedback input; volume weighting is used for consistency with how the reactivity feedback design limits are confirmed in the core design.

A scram table is used to simulate control rod insertion following reactor trip. Appropriately conservative scram curves are developed based on the core time-in-life, the initial power level, the location of the control bank, and other relevant factors to preserve the minimum shutdown margin.

6.1.3 Fuel Rod Design Input

{{

the fuel/pellet region, one interval for the gas gap and }}$^{2(a),(c)}$ Fuel performance data (Section 4.3.1.2) is incorporated into the NRELAP5 non-LOCA model via material thermal property tables for the UO_2 fuel region, the gas gap, and the cladding. Because the density of UO_2 changes with burnup, the thermal property tables can be revised as needed to match time in cycle.

Audit Question A-NonLOCA.LTR-62

As discussed in Section 4.3.1.1, the core power distribution is based on a nominal average axial power shape with power distributed solely in the fuel pellet. {{

}}$^{2(a),(c)}$

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{{

}}2(a),(c) The gap thermal conductivity is calculated at bounding values burnup to ensure that the fuel volume average temperature is appropriately bounded compared to fuel performance design data. This bounding method also accounts for gap closure over the fuel cycle.

6.1.4 Secondary System

6.1.4.1 Feedwater System

In an NPM design two feedwater lines penetrate the CNV immediately downstream of the FWIVs. Each feedwater line splits into two lines before connecting to the SGs. {{

Audit Question A-NonLOCA.LTR-63

Audit Question A-NonLOCA.LTR-63

}}2(a),(c)

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-68

Receipt Date: 06/28/2024

Question:

Discussion of the Mode 5 boron dilution transient in LTR Section 7.2.16.1 does not provide a method for evaluating this event that does not depend on operator response, as it states that surveillance of the boron concentration is performed at appropriate intervals to prevent significant inadvertent dilution to the reactor pool. NuScale has provided some markup to address this issue in the context of the Chapter 15 review. Provide LTR markup.

Response:

The requested markup was already provided in the response to audit question A-NonLOCA.LTR-43.

No changes to the SDAA are necessary.

January 14, 2025 Docket No. 052-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of the NRC's Request for Docketing of Resolved Audit Responses

The purpose of this letter is to provide NuScale's response to the NRC's request to docket certain audit responses. The responses to the individual audit items are provided in the attachments.

This letter contains NuScale's responses to the following audit questions from the NRC:

Non-LOCA

Information related to NuScale Topical Report: "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416, Revision 0.
Five Items:

- A-NonLOCA.LTR-46 (proprietary and nonproprietary)
- A-NonLOCA.LTR-46S (proprietary and nonproprietary)
- A-NonLOCA.LTR-56 (proprietary and nonproprietary)
- A-NonLOCA.LTR-63 (proprietary and nonproprietary)
- A-NonLOCA.LTR-66 (nonproprietary)

The audit items specified above have been attached to the NRC EIE submittal. The responses that are labeled as proprietary include both proprietary and nonproprietary versions. The enclosed affidavit supports this request.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Thomas Griffith at 541-452-7813 or tgriffith@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 14, 2025.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: Affidavit of Mark W. Shaver, AF-178079

Enclosure 1:

Affidavit of Mark W. Shaver, AF-178079

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NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying responses reveals distinguishing aspects about the process by which NuScale developed the Standard Design Approval Application for the US460 design.

NuScale has performed significant research and evaluation to develop a basis for this process and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the attached responses entitled NuScale Power, LLC Submittal of the NRC's Request for Docketing of Resolved Audit Responses. The attachments contain the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

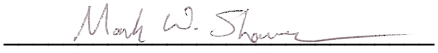
(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 14, 2025.



Mark W. Shaver

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-46

Receipt Date: 06/24/2024

Question:

In a number of event specific tables in the Non-LOCA TR (e.g., Turbine Trip, MSIV Closure, Loss of Normal Feedwater Transient Analysis, and Inadvertent Operation of the Decay Heat Removal System) that indicate “Initial conditions, biases, and conservatisms”, the SG heat transfer and/or DHRS heat transfer is set to “nominal”, {{

}}$^{2(a),(c)}$ This treatment is unchanged from the previous revision of the non-LOCA LTR. {{

}}$^{2(a),(c)}$

Response:

The events provided as examples in the audit question (i.e., turbine trip, main steam isolation valve (MSIV) closure, loss of normal feedwater (FW), and inadvertent operation of the decay heat removal system (DHRS)) are examples of heatup events where primary and secondary pressures are the key figures of merit. Section 4.2 of TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” describes how the need to perform sensitivities is related to the available margin. An example is given in Section 4.2 for maximum secondary system pressure, with the conclusion that “extensive sensitivity calculations to maximize secondary side pressure are not necessary for the non-LOCA transient analysis calculations.”

In the NuScale Power Module (NPM) design, primary side (reactor coolant system (RCS)) pressure during design-basis events (a DBE) is limited by the lifting of the reactor safety valves (RSVs). The lift setpoint of the RSVs is selected to ensure RCS pressure has adequate margin to acceptance criteria. Therefore, the biases applied to the lift setpoint of the RSVs is ultimately what determines the maximum RCS pressure during a DBE. The biases of other parameters may have minor effects that result in small differences in RCS pressure, but not enough to cause acceptance criteria to be exceeded. A similar discussion is provided in Section 7.2.6.3 of TR-0516-49416-P:

> "The maximum RCS pressure is limited by the RSV lifting and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases."

Consistent with the above discussion, the heatup events appropriately apply a high bias for the RSV lift setpoint. Using a high bias for the RSV lift setpoint delays the opening of the RSV until RCS pressure is higher, resulting in a higher peak RCS pressure. The application of a the high RSV lift setpoint is identified in the tables in TR-0516-49416-P as indicated below:

- Turbine trip, loss of external load, and loss of condenser vacuum – Table 7-32
- MSIV closure – Table 7-40
- Loss of nonemergency alternating current power – Table 7-44
- Loss of normal FW – Table 7-48
- Inadvertent operation of DHRS – Table 7-52
- Feedwater system pipe break – Table 7-56

These events also apply a nominal bias for steam generator (SG) heat transfer as shown in the same identified tables. {{

}}$^{2(a),(c)}$

The justification can be further illustrated by a review of the limiting RCS pressures for the heatup events in the Final Safety Analysis Report (FSAR) for the US460 as shown in Table 1.

Table 1: Limiting Reactor Coolant System Pressure for Heatup Events in FSAR

FSAR Section	Event	Peak RCS Pressure (psia)	Margin to Acceptance Criteria (psia)
15.2.1 / 15.2.2 / 15.2.3	Turbine trip, loss of external load, and loss of condenser vacuum	2310	110
15.2.4	MSIV closure	2313	107
15.2.6	Loss of nonemergency AC power	2314	106
15.2.7	Loss of normal FW	2314	106
15.2.8	Feedwater system pipe break	2320	100
15.2.9	Inadvertent operation of DHRS	2317	103

As shown in Table 1, the variation in the peak RCS pressure for the events is less than 10 percent of the margin to the acceptance criteria. Despite the large difference in initiating event and resultant event sequences, there is minimal difference in the calculated peak RCS pressure. This demonstrates that the RSV successfully mitigates the increasing pressure with a similar amount of overshoot, regardless of the specific sequence resulting in the RSV lifting. FSAR Table 5.2-2 identifies the lift pressure and tolerance associated with the RSVs. Based on these values in FSAR Table 5.2-2, and the application of the high bias as previously discussed, the first RSV opens during the events in Table 1 at 2266 psia. Given the margin of at least 100 psia for the events, and considering that the application of the high bias on the RSV is worth 66 psia, two conclusions can be drawn: the margin to acceptance criteria is significant and RSV biasing is the dominant factor to determine peak RCS pressure.

{{

}}2(a),(c)

{{

}}2(a),(c)

Table 2: {{ }}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

Regardless of the event or specific biases applied, heatup events produce the same peak RCS pressure that is primarily determined by the capacity and lift pressure of the RSVs. Applying a high bias to RSV lift setpoint is the dominant contributor to margin. For events where the RSV lift setpoint is not reached, the specific biases may have a more significant impact on peak RCS pressure; however, since the RCS pressure remains below the RSV lift setpoint, these events do not challenge acceptance criteria by definition.

In summary, regarding RCS pressure, the use of nominal SG heat transfer bias for the heatup events in TR-0516-49416-P Tables 7-32, 7-40, 7-44, 7-48, 7-52, and 7-56 is reasonable.
{{

}}2(a),(c) TR-0516-49416-P is revised as shown in the attached markups to add further technical basis consistent with this response.

{{

}}2(a),(c) The analyses associated with the methodology in TR-0516-49416-P are run long enough to demonstrate a stable condition, as shown by trends in RCS pressure and temperature. Other aspects of long-term DHRS cooling are addressed in TR-124587-P, Revision 0, “Extended Passive Cooling and Reactivity Control Methodology.” {{

}}2(a),(c) Note that the impact of high SG levels on DHRS performance is explicitly assessed as part of the SG overfill analysis associated with the increase in feedwater flow event. For example, FSAR Figure 15.1-8 shows high SG levels associated with the increase in feedwater flow event {{ }}2(a),(c) Audit question

A-NonLOCA.LTR-60 identifies that TR-0516-49416-P Section 7.2.2 for the increase in feedwater flow event does not identify limiting bias directions for the SG overfill analysis. NuScale will address the impact of SG heat transfer bias on potential SG overfill conditions with the response to audit question A-NonLOCA.LTR-60.

Markups of the affected changes, as described in the response, are provided below.

DHRS condensers and the energy is transferred to the reactor pool UHS. The maximum pressure in the SG secondary side is limited to the saturation pressure at the temperature of the RCS fluid on the SG primary side. Therefore, the maximum secondary pressure is affected by the secondary side inventory and the primary side conditions at the time of DHRS actuation, and is less sensitive to a specific initiating event. The SG design pressure is significantly higher than pressures expected during DHRS operation. The margin to the SG design pressure is physically limited, based on the primary side conditions. The representative transient results in Section 8.0 demonstrate that significant margin to the maximum SG pressure acceptance criterion is maintained for all types of events. Therefore, extensive sensitivity calculations to maximize secondary side pressure are not necessary for the non-LOCA transient analysis calculations.

Audit Question A-NonLOCA.LTR-46

Another example is RCS pressure. In the NPM, operation of an RSV at its lift setpoint is sufficient to mitigate an increasing pressure with minimal overshoot. This design feature ensures that the maximum RCS pressure does not significantly exceed the RSV lift setpoint. With appropriate selection of the RSV lift setpoint, including consideration of uncertainty, the maximum RCS pressure will be well below the acceptance criterion. Modification of other biases can generate many cases with similar maximum RCS pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

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Table 7-31 Acceptance criteria - turbine trip / loss of external load (Continued)

Acceptance Criteria	Discussion
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation and continued FW operation, in addition to the actual turbine trip or loss of external load. The DHRS heat removal is limited by the DHRS condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.6.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-32 are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Audit Question A-NonLOCA.LTR-46

Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased to the high condition to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	}}2(a),(c)

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Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Biased to the high condition.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the high condition	
MTC	Consistent with BOC conditions.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
Steam generator heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	}}2(a),(c)

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Table 7-32 Initial conditions, biases, and conservatisms - turbine trip / loss of external load (Continued)

Parameter	Bias / Conservatism	Basis {{
PZR Pressure Control PZR spray PZR heaters	 Disabled. Nominal.	
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Disabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Disabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

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7.2.8.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-40 are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Audit Question A-NonLOCA.LTR-46

Table 7-40 Initial conditions, biases, and conservatisms - main steam isolation valve closure

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the high condition	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
Steam generator heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	 }}2(a),(c)

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Table 7-40 Initial conditions, biases, and conservatisms - main steam isolation valve closure (Continued)

Parameter	Bias / Conservatism	Basis {{
PZR Pressure Control PZR spray PZR heaters	Disabled. Nominal.	
PZR Level Control Charging Letdown	Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may

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have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

Table 7-41 Not Used

7.2.9 Loss of Nonemergency AC Power

The methodology used to simulate a postulated loss of nonemergency (normal) AC power for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.9.1 General Event Description

The low voltage AC electrical distribution system (ELVS) supplies AC power to plant motors, heaters, packaged equipment, and battery chargers. Loss of normal AC power to the station auxiliaries can result from electrical grid-related failures, failures in plant or switchyard equipment, or external weather events. The nonsafety-related EDNS and EDAS/EDSS may remain available via battery operation; the primary loads for these systems include the module control system (MCS) and the MPS. Loss of the EDNS and/or EDAS/EDSS batteries with the loss of normal AC power is considered as described in Section 7.1.3. A loss of AC power results in a pressurization of the secondary system and overheating of the RCS. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition. Table 7-42 lists the relevant acceptance criteria, SAF, and LOP scenarios.

The electrical system design may vary by NPM design. Review of the electrical system is performed to determine the impact on plant equipment from the loss of power. In general, the following typically occur from the loss of power (AC or DC or both):

- turbine trip occurs
- the feedwater pumps and CVCS pumps stop
- the PZR heaters turn off
- control rods begin to drop due to loss of power to control rod drive mechanisms (CRDMs)
- the MPS actuates reactor trip, DHRS, and various system isolations within 60 seconds after event initiation (if not already actuated during that time), due to loss of AC power to the EDAS/EDSS battery chargers
- the MPS actuates ECCS within 24 hours after event initiation, due to loss of AC power to the EDAS/EDSS battery chargers

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Table 7-43 Acceptance criteria - loss of normal AC power (Continued)

Acceptance Criteria	Discussion
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.9.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-44 are considered in identifying the bounding transient simulation for primary and secondary pressure.

Audit Question A-NonLOCA.LTR-46

Table 7-44 Initial conditions, biases, and conservatisms - loss of normal AC power

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Biased to the high condition.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the high condition	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal.	}}2(a),(c)

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Table 7-44 Initial conditions, biases, and conservatisms - loss of normal AC power (Continued)

Parameter	Bias / Conservatism	Basis
		{{
Feedwater and Turbine Load Control feedwater pump speed	Enabled.	
CNV Pressure Control CNV evacuation system	Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)
2. Loss of normal AC power initiating event results in a loss of system function by loss of power to the system thereby making the system control not relevant to the event.

Audit Question A-NonLOCA.LTR-46

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

Table 7-45 Not Used

7.2.10 Loss of Normal Feedwater Flow

The methodology used to simulate a postulated loss of normal feedwater flow for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below.

7.2.10.1 General Event Description

A postulated fault results in a partial or complete loss of feedwater flow, and the water in the steam generators boils off. The loss of steam generators as a heat sink leads to a rise in the RCS temperature and pressure until the reactor typically

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Table 7-47 Acceptance criteria - loss of normal feedwater flow (Continued)

Acceptance Criteria	Discussion
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.10.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms indicated in Table 7-48 are considered in identifying a bounding transient simulation for primary and steam generator pressure.

Audit Question A-NonLOCA.LTR-46

Table 7-48 Initial conditions, biases, and conservatisms - loss of normal feedwater flow

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	
Initial RCS flow rate	Varied.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased to the high condition.	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	}}2(a),(c)

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1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest primary and secondary pressures, varying the magnitude of the feedwater flow rate decrease (other parameters are biased as indicated in Table 7-48). The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

Table 7-49 Not Used

7.2.11 Inadvertent Decay Heat Removal System Actuation

The methodology used to simulate a postulated inadvertent DHRS actuation for an NPM, and an evaluation of the acceptance criteria for an AOO listed in Table 7-4, are presented below. This event is unique to the NPM designs.

7.2.11.1 General Event Description

Inadvertent actuation of the DHRS may result from either an unexpected DHRS valve actuation or a spurious DHRS actuation signal. Due to the DHRS design and configuration, several scenarios exist for consideration. The specific scenarios considered are discussed below. The sequence of events and the MPS signals that are reached first depend on the scenario. Reactor trip and transition to stable DHRS flow terminates the transients with the NPM in a safe, stable condition. The relevant acceptance criteria, SAF, and LOP scenarios are listed in Table 7-50.

Scenario 1: The unexpected opening of a single DHRS valve can occur at full power or reduced power conditions. At low power conditions, a portion of the DHRS liquid inventory drains into the feedwater line, which momentarily increases feedwater flow and causes overcooling. This overcooling event is not considered further because it is bounded by other, more limiting overcooling events (i.e., increase in feedwater flow). The most challenging conditions for this heatup scenario occur at full power with initial conditions biased in the conservative directions. Since feedwater flow tends to increase in response to the reduced steam enthalpy and turbine load, limiting the feedwater response maximizes the heatup.

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Table 7-51 Acceptance criteria - inadvertent decay heat removal system actuation

Acceptance Criteria	Discussion
Primary pressure	Primary pressure quickly rises to the peak value, then drops as the lowest setpoint RSV lifts to reduce pressure.
Secondary pressure	Peak secondary pressurization is largely a function of DHRS actuation. The DHRS heat removal is limited by the DHR condenser so some pressurization is expected for every actuation of this system.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis.
Maximum fuel centerline temperature	This criterion is evaluated by downstream subchannel analysis.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.11.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-52 are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Audit Question A-NonLOCA.LTR-46

Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation

Parameter	Bias / Conservatism	Basis
		{{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Varied.	}}2(a),(c)

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Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation (Continued)

Parameter	Bias / Conservatism	Basis {{
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Nominal.	
Initial fuel temperature	Biased to the high condition	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
Steam generator heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	
SG tube plugging	Biased to the low condition.	
RCS Temperature Control Automatic rod control Boron concentration	 Disabled. Not credited.	
PZR Pressure Control PZR spray	 Varied.	}}2(a),(c)

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Table 7-52 Initial conditions, biases, and conservatisms - inadvertent decay heat removal system actuation (Continued)

Parameter	Bias / Conservatism	Basis
PZR heaters	Nominal.	{{
PZR Level Control Charging Letdown	 Not credited. Disabled.	
Steam Pressure Control Turbine throttle valves Turbine bypass valves	 Enabled. Disabled.	
Feedwater and Turbine Load Control feedwater pump speed	 Disabled.	
CNV Pressure Control CNV evacuation system	 Enabled.	}}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

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7.2.12.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-56 are considered in identifying the bounding transient simulation for primary and steam generator pressure.

Audit Question A-NonLOCA.LTR-46

Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break

Parameter	Bias / Conservatism	Basis {{
Initial reactor power	RTP biased upwards to account for measurement uncertainty.	
Initial RCS average temperature	Biased to the high condition.	
Initial RCS flow rate	Biased to the low condition.	
Initial PZR pressure	Varied.	
Initial PZR level	Varied.	
Initial feedwater temperature	Varied.	
Initial fuel temperature	Biased to the high condition.	
MTC	Consistent with BOC kinetics.	
Kinetics	Biased to BOC conditions.	
Decay heat	Biased to the high condition.	
Initial SG pressure(1)	Varied.	
SG heat transfer	Nominal.	
RSV lift setpoint	Biased to the high condition.	}}2(a),(c)

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Table 7-56 Initial conditions, biases, and conservatisms - feedwater line break (Continued)

Parameter	Bias / Conservatism	Basis
CNV Pressure Control CNV evacuation system	Enabled.	{{ }}2(a),(c)

1. {{ }}2(a),(c)

Audit Question A-NonLOCA.LTR-46

Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event. The maximum RCS pressure is limited by the RSV lifting, with its applied lift pressure bias, and therefore many cases may have similar peak pressures. Extensive sensitivity studies are not required to investigate the small differences between those cases.

Table 7-57 Not Used

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

The methodology used to simulate a postulated uncontrolled CRA bank withdrawal from subcritical or low power startup conditions for an NPM, and an evaluation of the acceptance criteria listed for an AOO in Table 7-4, are presented below.

The range of initial power levels associated with low power startup conditions for an NPM is based on the low setting for the high power signal (below 15 percent RTP for the example in Table 7-3). When core power reaches the low setting level, a hold point is established to alter the high power setting. Thus, low power startup conditions exist until reactor power reaches the low setting level.

7.2.13.1 General Event Description and Methodology

The limiting event consequences to an uncontrolled CRA bank withdrawal from subcritical or low power startup conditions typically (for most PWR designs) occur for cases with very low initial power levels (~1 Watt). The primary reason for this

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-46S

Receipt Date: 08/09/2024

Question:

Audit question A-NonLOCA.LTR-46 was received by NuScale on June 24, 2024. NuScale provided a response on July 22, 2024. On August 9, 2024, NuScale received the following written feedback from the NRC:

{{

}}$^{2(a),(c)}$ However, the LTR does not define clear criteria to show how these determinations are made in accordance with the evaluation model. For figures of merit associated with the non-LOCA topical report, identify the thresholds that determine whether margins to acceptance criteria are sufficient and the thresholds that determine whether figures of merit are insensitive to biases, initial conditions and analysis assumptions.

{{

}}$^{2(a),(c)}$ It is unclear to NRC staff whether these additional sensitivity studies (i.e., sensitivity studies not indicated by event-specific tables) are requirements of the evaluation model and, if so, what indicates that additional sensitivity studies are required. Clarify whether and in what circumstances sensitivity studies beyond those indicated in event-specific "initial conditions, biases, and conservatisms" tables are required by the non-LOCA LTR evaluation model.

Provide LTR markup as needed.

Response:

This response is a supplement to the July 22, 2024 response to address the feedback from August 9, 2024; the July 22, 2024 response remains unchanged in the electronic reading room (eRR). The context of the original audit question is provided:

{{

}}2(a),(c) The original audit question requested that NuScale "demonstrate that this basis is acceptable." NuScale has demonstrated the basis for the statement in TR-0516-49416-P, Revision 4. {{

}}2(a),(c) Specifically, the operation of the reactor safety valve (RSV) at its biased high setpoint effectively determines the peak RCS pressure. This technical basis was added to TR-0516-49416-P, Revision 4 as indicated in the markup in the original response.

To address the additional NRC feedback, each aspect of the feedback is addressed by providing the text of the feedback in indented, italicized text, followed by the NuScale response in un-indented, regular text.

{{

}}2(a),(c) However, the key technical basis for the conclusion, as specified in the markups provided with the original audit response, is that the RSV lift limits the pressurization of the RCS. The RSV design ensures that the

maximum RCS pressure does not significantly exceed the RSV lift setpoint. Modification of initial and boundary conditions in the calculations can generate many cases with similar maximum RCS pressure due to this design feature.

> *"However, the LTR does not define clear criteria to show how these determinations are made in accordance with the evaluation model. For figures of merit associated with the non-LOCA topical report, identify the thresholds that determine whether margins to acceptance criteria are sufficient and the thresholds that determine whether figures of merit are insensitive to biases, initial conditions and analysis assumptions."*

Based on this NRC feedback, NuScale concludes that the NRC feedback is concerned about how "insensitivity" will be judged in the future. The NRC feedback requests additional specific detailed information be added to TR-0516-49416-P to determine the size of the change relative to the margin that can be considered insensitive. NuScale believes this is unnecessary. This level of detail was not necessary in the previous revision of the NRC-approved topical report. {{

}}$^{2(a),(c)}$ The NRC feedback now requests specific detailed information be added to TR-0516-49416-P to determine the size of the change relative to the margin that can be considered insensitive. NuScale believes this is unnecessary because the insensitivity of the results can be shown regardless of the margin. {{

}}$^{2(a),(c)}$

{{

}}2(a),(c)

As described in TR-0516-49416-P, and the associated markups in the original response, it is a characteristic of the design of the NuScale Power Module (NPM) that peak RCS pressure does not challenge acceptance criteria. The peak pressures for the heatup events were summarized in Table 1 of the original response. It is worth noting that the peak RCS pressures (ranging from 2310 to 2320 psia) are slightly above the nominal setpoint for opening of the second reactor safety valve (RSV) (2290 psid per Final Safety Analysis Report Table 5.2-2). {{

}}2(a),(c) *It is unclear to NRC staff whether these additional sensitivity studies (i.e., sensitivity studies not indicated by event-specific tables) are requirements of the evaluation model and, if so, what indicates that additional sensitivity studies are required. Clarify whether and in what circumstances sensitivity studies beyond those indicated in event-specific "initial conditions, biases, and conservatisms" tables are required by the non-LOCA LTR evaluation model."*

The sensitivity studies performed on {{ }}2(a),(c) are in addition to those required by TR-0516-49416-P, Revision 4, "Non-Loss-of-Coolant Accident Analysis Methodology." These sensitivity studies are not part of the implementation of the methodology in TR-0516-49416-P. The additional sensitivity studies are studies included at the discretion of the analyst for information. It is acceptable for an implementing analysis to perform additional sensitivity studies or include additional information beyond what is required by the methodology. Because inclusion of any such additional studies or information is, by definition, beyond the scope of the methodology, additional information on this decision-making process is not included in TR-0516-49416-P.

"Provide LTR markup as needed."

There are no additional markups to TR-0516-49416-P as a result of this supplemental response.

No changes to the SDAA are necessary.

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-56

Receipt Date: 06/28/2024

Question:

TR Sections 7.2.2.3 and 7.2.3.3 state “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response for this overcooling event.”

TR Section 7.2.4.3 states “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the lowest CHFR response and challenging mass releases for this overcooling event.”

TR Sections 7.2.6.3, 7.2.8.3, 7.2.9.3, and 7.2.12.3, state “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events). Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this overheating event.”

TR Section 7.2.10.3 states “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest primary and secondary pressures, varying the magnitude of the feedwater flow rate decrease (other parameters are biased as indicated in Table 7-48).”

TR Section 7.2.11.3 states “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, sensitivity studies are performed to identify cases with the highest pressure responses for this event.”

TR Section 7.2.13.3 states “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, a sensitivity study is generally performed to identify cases for lowest MCHFR and highest fuel centerline temperature for this reactivity event.”

TR Sections 7.2.14.3 and 7.2.15.3, state: “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event. Consequently, a sensitivity study is generally performed to identify cases for lowest MCHFR for this reactivity event.”

{{

}}2(a),(c)

TR Section 7.2.17.3 states “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event, including cases with the highest pressure responses for this inventory increase event.”

TR Sections 7.2.18.3 and 7.2.19.3 state “Sensitivity studies are performed as needed to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event.”

However, the TR does not provide instructions for how and when to perform the sensitivity studies.

Please provide proposed markups to the TR to include this information.

Response:

The audit question identifies sections in TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” that contain the phrase “Sensitivity studies are performed as needed...” Note that Sections 7.2.1.3 and 7.2.5.3 of TR-0516-49416-P, Revision 4, also contain the phrase, but were not identified in the audit question. The following response is applicable to these two additional sections although they were not included in the audit question.

Each use of the phrase identified in the audit question is directly preceded by a table that identifies the biases or conservatism to be applied to specific parameter. These tables include one or more biases or conservatisms that state “varied.” The use of “varied” in the tables for a specific parameter indicates that different values of that parameter should be used in the implementing analysis. {{ }}2(a),(c)

{{

}}2(a),(c) In other words, use of “varied” in the tables means that a sensitivity study is performed on that parameter. Therefore, the sensitivity studies to be performed as needed are those that are indicated as needed by the tables. Based on this discussion, TR-0516-49416-P, Revision 4, already identifies “when” to perform the sensitivity studies.

The “how” of performing the sensitivity studies is consistent with generic safety analysis practice: if a specific parameter has a range of initial values, the high and low ends of the range are considered. Multiple examples of how to perform such sensitivity studies given a range of initial values are provided in TR-0516-49416-P, Revision 4, Section 7.1.6.1. {{

}}2(a),(c) Therefore, there is also sufficient information existing in TR-0516-49416-P, Revision 4, to identify “how” to perform the sensitivity studies.

In addition, the downstream application of the methodology always has the option to perform additional sensitivity studies beyond the minimum required.

It is important to note that each of use of the phrase “Sensitivity studies are performed as needed...” in TR-0516-49416-P, Revision 4, also existed in the TR-0516-49416-P-A, Revision 3 that was previously reviewed and approved by the NRC. Therefore, the uses of this phrase are not changes or departures from what was previously reviewed and approved by the NRC. For this reason, further information is not required to be added to TR-0516-49416-P, Revision 4, to describe how and when to perform sensitivity studies.

{{

}}2(a),(c) Note that TR-0516-49416-P-A, Revision 3, Section 7.2.16.3, identified that the sensitivity studies of interest were for initial power and dilution flow rate. TR-0516-94916-P, Revision 4, Section 7.2.16.1, does identify that different initial power levels and dilution flow rates are considered, although this is not identified in Table 7-74 or the Section 7.2.16.3 discussion. The topical report is revised as indicated in the attached markup to resolve this minor inconsistency.

Markups of the affected changes, as described in the response, are provided below.

Table 7-73 Acceptance criteria - inadvertent decrease in boron concentration

Acceptance Criteria	Discussion
Primary pressure	This criterion is not evaluated because an NPM does not introduce more challenging conditions for primary pressure compared to other AOOs.
Secondary pressure	This criterion is not evaluated because an NPM does not introduce more challenging conditions for secondary pressure compared to other AOOs.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis; or by demonstrating that a loss of shutdown margin does not occur.
Maximum fuel centerline temperature	This criterion is not evaluated because the reactivity insertion rates are insufficient to challenge the temperature limit.
Containment integrity	Containment integrity is evaluated by a separate analysis methodology.
Escalation of an AOO to an accident	This criterion is satisfied by demonstrating stable RCS flow rates and constant or downward trending RCS and DHRS (if actuated) pressures and temperatures exist at the end of the transient, all acceptance criteria evaluated in the transient analysis are met, and shutdown margin is maintained at the end of the transient. RCS conditions during extended DHRS cooling are addressed in a separate analysis.

7.2.16.3 Biases, Conservatisms, and Sensitivity Studies

The biases and conservatisms presented in Table 7-74 are considered in order to identify the bounding conditions for loss of shutdown margin.

Audit Question A-NonLOCA.LTR-56

Table 7-74 Initial conditions, biases, and conservatisms - inadvertent decrease in boron concentration

Parameter	Bias / Conservatism	Basis
Initial reactor power	~~Excluded~~Varied.	{{ }}2(a),(c)
Initial RCS average temperature	Biased to the high condition.	{{ }}2(a),(c)
Initial RCS flow rate	Biased to the low condition.	{{ }}2(a),(c)
Initial PZR pressure	Nominal.	{{ }}2(a),(c)

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Table 7-74 Initial conditions, biases, and conservatisms - inadvertent decrease in boron concentration (Continued)

Parameter	Bias / Conservatism	Basis
Initial PZR level	Excluded.	Not part of mixing model.
Initial feedwater temperature	Excluded.	Not part of mixing model.
Initial fuel temperature	Excluded.	Not part of mixing model.
MTC	Excluded.	Not part of mixing model.
Kinetics	Biased to BOC conditions.	{{ }}2(a),(c)
Decay heat	Excluded.	Not part of mixing model.
Initial SG pressure	Excluded.	Not part of mixing model.
SG heat transfer	Excluded.	Not part of mixing model.
RSV lift setpoint	Excluded.	Not part of mixing model.
SG tube plugging	Excluded.	Does not alter active RCS volume.
Shutdown margin	Biased to the low condition.	{{ }}2(a),(c)
Initial boron concentration	Biased to the high condition	{{ }}2(a),(c)
Boron worth	Biased to the high condition.	{{ }}2(a),(c)
Active RCS volume	Biased to the low condition.	{{ }}2(a),(c)
Makeup flow rate	~~Biased to the high condition~~Varied.	{{ }}2(a),(c)
Makeup temperature	Biased to the low condition.	{{ }}2(a),(c)
RCS Temperature Control		
Automatic rod control	Excluded.	Not part of mixing model.
Boron concentration	Not credited.	{{ }}2(a),(c)
PZR Pressure Control		
PZR spray (normal)	Excluded.	Not part of mixing model.
(bypass)	Excluded.	Not part of mixing model.
PZR heaters (non-prop.)	Excluded.	Not part of mixing model.
(prop.)	Excluded.	Not part of mixing model.

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-63

Receipt Date: 06/28/2024

Question:

Non-LOCA TR r4 Section 6 contains numerous instances of Figures and descriptions applicable to the NPM-160 design but inapplicable to the NPM-20 design. Update Section 6 such that it accurately describes the NRELAP5 non-LOCA base model "nl20r1-model.i" (r1) that is associated with EC-0000-8507 "NPM-20 NRELAP5 Non-LOCA Model" (r1).

Response:

The figures and descriptions provided in TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” are intended to be typical descriptions of the NuScale NRELAP5 plant model that is not specific to the details of the NuScale Power Module (NPM) that may vary between the NPM-160 and NPM-20. To this end, TR-0516-49416-P Section 6.0 states:

> *“The base model for a specific NPM design reflects the parameters and features of that design.”*

Specific design features that differ between NPM designs do not change the methodology used to model important parts of the design for non-loss-of-coolant accident (non-LOCA) analysis. TR-0516-49416-P, Revision 4, Section 6.1 includes the following that is unchanged from Revision 3 (i.e., the version previously reviewed and approved by the NRC):

> *“Where precise noding is described in Section 6.1, the specified level of detail is considered the minimum level of detail required for the component of interest. Should additional detail be needed in the future, the relevant benchmarks, sensitivity studies, and transient analyses will be reviewed for continued applicability. If necessary, the relevant benchmarks, sensitivity studies, and transient analyses will*

be revised to demonstrate the higher level of detail for the component of interest is applicable to an NPM."

Precise noding details that are important, considering the non-LOCA evaluation model validation basis and relationship to the subchannel analysis, are unchanged between Revision 3 and Revision 4 of TR-0516-49416-P, and have been used in the application of the methodology to the NPM-20 design. These precise noding details include: {{

}}[2(a),(c)]

NRELAP5 provides the user considerable flexibility in using pipe, branch and various junction or valve components to model a physical system. Equivalent modeling can be accomplished with different combinations of components. For this reason, it is not necessary for the figures in Section 6 to match one specific design. Instead, the figures in Section 6 are identified as typical. The figures in Section 6 have corresponding text discussion in Section 6. The text in Section 6 of the topical report was reviewed to determine if additional clarification was necessary due to design changes between NPM-160 and NPM-20 or due to basemodel changes. As a result of the review, TR-0516-49416-P Section 6 is revised as shown in the attached markup to provide additional clarification related to model development when different design details are evaluated. In some cases, the revised text allows for different modeling options to be used. The reason for allowing multiple options is to retain the modeling option previously reviewed and approved in TR-0516-49416 Revision 3, while allowing for the alternative approach used in the basemodel for NPM-20. With these changes, Section 6 accurately reflects the NRELAP5 basemodel for NPM-20, consistent with the request in this audit question.

Markups of the affected changes, as described in the response, are provided below.

Steam Generator Primary

The SG is a helical coiled, once-through HX, with the primary system on the shell side and the secondary system on the tube side. The primary system water is cooled as it flows over the outer surfaces of the SG tubes before passing over the feedwater plena that are located above the elevation of the conical transition riser fairing in the RPV. On the secondary side, feedwater enters the bottom of the SG tubes via the feedwater plena and is heated as it flows upward, with superheated steam exiting the tops of the tubes. Two independent sets of interwoven SG tube banks occupy the SG region, each having independent feedwater and steam plena. If the tube banks experience different secondary side conditions, the primary coolant does not experience any corresponding asymmetries because of the interwoven design of the helical coiled tubes.

The heat transfer to the SG tubes occurs in the upper downcomer. The heat transfer and pressure drop resulting from the presence of the SG tubes was assessed with data from SIET TF-2 (Section 5.3.5). This assessment used special heat transfer model options to determine the methodology for accurately predicting the pressure drop in the region. {{

}}2(a),(c)

Core and Lower Plenum

Audit Question A-NonLOCA.LTR-63

{{

}}2(a),(c)

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{{

}}2(a),(c)

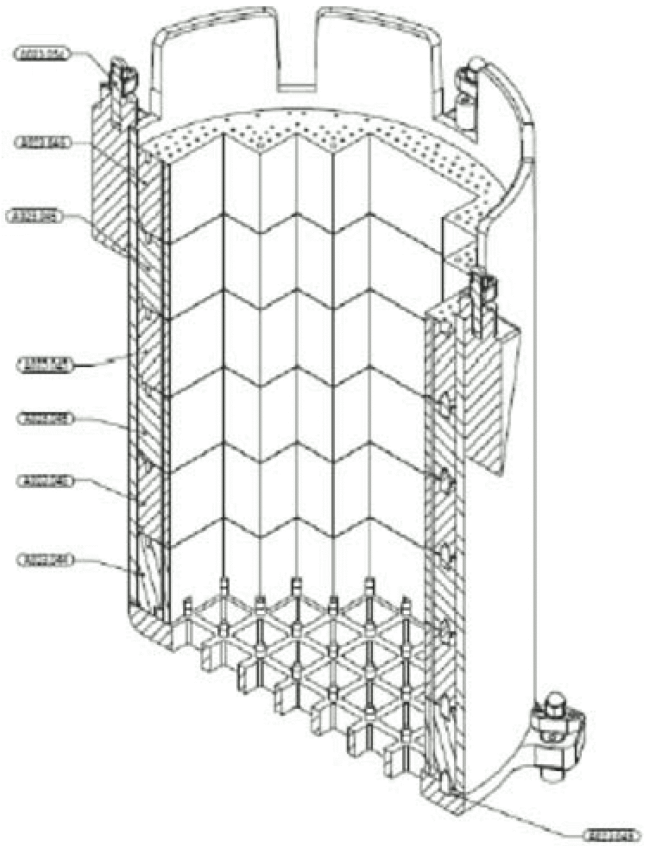
Figure 6-5 Typical core and lower plenum model

{{

}}2(a),(c)

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Figure 6-6 Reflector / core bypass without fuel assemblies (for illustration only)



Lower Riser

Audit Question A-NonLOCA.LTR-63

{{

}}2(a),(c)

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Upper Riser

Audit Question A-NonLOCA.LTR-63

{{

}}2(a),(c) Normal flow in the riser is single-phase subcooled water. Transients that involve RPV depressurization or inventory loss can result in flashing and two-phase flow in the riser region.

{{

}}2(a),(c)

Figure 6-9 Typical reactor pressure vessel upper riser model

{{

}}2(a),(c)

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Pressurizer

{{

Audit Question A-NonLOCA.LTR-63

}}2(a),(c)

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{{

}}2(a),(c) The gap thermal conductivity is calculated at bounding values burnup to ensure that the fuel volume average temperature is appropriately bounded compared to fuel performance design data. This bounding method also accounts for gap closure over the fuel cycle.

6.1.4 Secondary System

6.1.4.1 Feedwater System

In an NPM design two feedwater lines penetrate the CNV immediately downstream of the FWIVs. Each feedwater line splits into two lines before connecting to the SGs. {{

Audit Question A-NonLOCA.LTR-63

Audit Question A-NonLOCA.LTR-63

}}2(a),(c)

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{{

Audit Question A-NonLOCA.LTR-63

}}2(a),(c)

Figure 6-12 Typical main steam system model

{{

}}2(a),(c)

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6.1.5 Decay Heat Removal System

Audit Question A-NonLOCA.LTR-63

An NPM incorporates two separate DHRSs that are treated individually in the NRELAP5 model. {{

}}2(a),(c) Figure 6-13 shows the typical nodalization for DHRS loop 1 {{

}}2(a),(c). Loop 2 is modeled similarly. While each DHRS line in an NPM features two parallel actuation valves, {{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-63

The number of hydrodynamic volumes in the DHRS condenser piping and HX regions are based on results from NRELAP5 assessments using data from the NIST-1 facility (Section 5.3.2). {{

}}2(a),(c)

Audit Question A-NonLOCA.LTR-63

In an actual NPM, the DHRS ~~heat exchanger is~~condensers are located in the reactor cooling pool. {{

}}2(a),(c) The long-term use of DHRS is addressed separately in Reference 26.

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{{

Audit Question A-NonLOCA.LTR-63

}}2(a),(c)

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6.1.8 Reactor Cooling Pool

Audit Question A-NonLOCA.LTR-63

{{

}}2(a),(c)

6.2 Material Properties

Thermal properties (thermal conductivity and volumetric heat capacity) are specified by user input in the NRELAP5 non-LOCA base models for the following materials typically used in the heat structures. These material properties may be amended or revised as the NPM design evolves:

1. fuel cladding (AREVA's M5® cladding)
2. inconel 690 (SG tubes)
3. uranium dioxide (UO_2)
4. stainless steel (SA-240 304L)
5. fuel-to-cladding gas gap (initially pressurized helium at BOC; mixture of fission product gases and helium after irradiation)
6. carbon steel (SA-508)
7. martensitic stainless steel (SA-336, F6NM)
8. austenitic stainless steel (SA-965, FXM-19)
9. {{

}}2(a),(c)

6.3 Control Systems

With its combination of trips, control functions, and user-defined tables, NRELAP5 provides flexibility to accurately simulate plant control and protection system responses during both steady-state and transient operation. The NRELAP5 non-LOCA base models contains logic for "normal controls" that simulate normal operational plant response, as well as user-convenience controls that make it easier to initialize the model for particular

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6.3.1.3 Reactor Coolant System Temperature Control (Nonsafety-related)

The MCS model controls RCS average temperature by changing reactivity in the core to increase or decrease core power. This reactivity change is accomplished with a control rod controller and a boron controller. The control rod controller uses design data to model a calculated rate of reactivity insertion due to maximum or nominal rod movement rates. {{

}}2(a),(c) Neither controller accounts for all the actual core physics including the effect of xenon or other decay products or poisons that could be expected with control rod repositioning.

Audit Question A-NonLOCA.LTR-36

The average coolant temperature is controlled by adjusting core power, which is accomplished by moving the control rods or changing the boron concentration of the reactor coolant. The choice of which method is based on the desired rate of change for core power. The control rods are moved to achieve faster power changes to meet the target average coolant temperature; slower power changes are accomplished by changing the boron concentration of the reactor coolant. At full power, the rod control system is set to 'insert only' mode to prevent automatic withdrawal of the control rods during a transient. If this design feature is neglected in the model, the rod control system will allow control rods to be withdrawn even if reactor power is at or above full power, resulting in a conservative power response to a transient.

6.3.1.4 Steam Pressure Control (Nonsafety-related)

Audit Question A-NonLOCA.LTR-63

In an NPM design, the turbine throttle and bypass valves are used to control steam pressure at the programmed values, {{

}}2(a),(c)

6.3.1.5 Feedwater and Turbine Load Control (Nonsafety-related)

Audit Question A-NonLOCA.LTR-63

An NPM prototypic control scheme design for the feedwater system is based on turbine load demand. The feedwater pumps are variable speed and can provide variable flow for module operations over a wide range of power without

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adjustments to the feedwater regulating valve. {{

Audit Question A-NonLOCA.LTR-63

}}2(a),(c)

6.3.1.6 Containment Pressure Control (Nonsafety-related)

The containment pressure is established at sub-atmospheric conditions via operation of the containment evacuation system. The impact of this system continuing to operate is considered for the non-LOCA transient analyses.

6.3.2 Module Protection System (Safety-related)

6.3.2.1 Analytical Limits and Delays

The MPS implemented in the NRELAP5 base models is intended for the purposes of performing safety analysis transient simulations. As such, the logic and actuation points are based on the NPM safety analysis analytical limits. Fixed delay times are specified considering different sensor response times. {{

}}2(a),(c) In addition to the sensor delays, a given safety signal is subject to instrumentation string delays, an MPS processing delay, and an actuation delay. The NRELAP5 non-LOCA model incorporates the methodology assumption that a bounding total for these additional delays is applied as a signal delay in addition to the individual sensor delay.

The MPS actuation signals in an NPM design are typically based on the following types of parameters:

- power range power
- power range power rate
- intermediate range log power rate

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Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-66

Receipt Date: 06/28/2024

Question:

TR Table 7-22 indicates MSIV single failure and FWIV single failure are potentially bounding for mass releases for break locations outside the CNV if mass release is calculated for use in the downstream radiological analysis. This implies that MSIV single failure and FWIV single failure are not bounding for some scenarios. Please clarify when MSIV single failure and FWIV single failure are not bounding.
Please provide proposed markups to the TR.

Response:

TR-0516-49416-P, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 4, Table 7-22 provides the acceptance criteria, single active failure, and loss of power scenarios for the steam line break event. Table 7-22 identifies that the single active failures of the NuScale Power Module (NPM) designs that could affect the steam line break event progression do not affect the limiting minimum critical heat flux (MCHFR) scenarios. The limiting MCHFR condition occurs around the time of reactor trip, at or before secondary side isolation or decay heat removal system (DHRS) actuation.

In the NRC’s Safety Evaluation Report (SER) for TR-0516-49416-P-A, Revision 3, a limitation and condition was included that stated that “[a]n 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,” with emphasis added. Under the assumption that the same limitation and condition would be imposed for TR-0516-49416-P, Revision 4, the single failures in the tables in Section 7.2 of TR-0516-49416-P,

Revision 4, are those that are typically limiting. The specific single failures that are limiting for each event are identified in the design-specific Final Safety Analysis Report (FSAR).

For the steam line break, Table 7-22 provides a discussion of why the single failures typically do not impact MCHFR as discussed above. Single failures do impact the mass release. However, the TR-0516-49416-P, Revision 4, methodology identifies that the radiological consequence analyses can be performed using bounding input that does not require determination of a specific release quantity. In the case where the bounding input is used, the single failure assumptions associated with mass release are irrelevant. In the case where the bounding input is not used, the single failures that are typically limiting for mass release are identified in Table 7-22: main steam isolation valve (MSIV) for breaks outside containment and feedwater isolation valve (FWIV) for breaks inside containment.

Consistent with the previous SER limitation and condition, the FSAR for the US460 design identifies the limiting single failure for each event. The steam line break is described in FSAR Section 15.1.5. The discussion in FSAR Section 15.1.5.2 addresses possible single failures and identifies the single failures that are typically limiting, consistent with Table 7-22 of TR-0516-49416-P, Revision 4. The subsequent discussion in FSAR Section 15.1.5.3.2 identifies the specific single failure limiting for each case. Finally, the discussion in FSAR Section 15.1.5.3.3 identifies the impact of the assumed single failure on the results for each case. The single failures for each case are also identified in some of the sequence of events tables, such as FSAR Tables 15.1-11 and 15.1-12. The FSAR Section 15.1.5 discussion does not include mass release cases because the bounding approach is used for the US460 radiological consequence analysis of the steam line break (FSAR Section 15.0.3.7.3). Therefore, single failures related to mass release are not applicable.

The FSAR section 15.1.5.3.2 discussion of the limiting MCHFR case identifies that this case is a break outside containment and single failure of an MSIV to close is modeled. Table 15.1-13 shows the sequence of events for this case. The high power signal actuates reactor trip and MCHFR is reached at 34 seconds. The low pressurizer pressure signal actuates secondary side isolation at 85 seconds. The modeled single active failure of an MSIV to close occurs well after the time of MCHFR. Failure of an MSIV to close in this case does affect the available DHRS capacity because the break is located between the MSIV and secondary MSIV as described in FSAR 15.1.5.3.2. Therefore, the modeled failure of the MSIV on the broken loop to close disables one train of DHRS in this case, similar to the limiting reactor coolant system pressure case, but has no impact on the figure of merit for this case (i.e., MCHFR).

In summary, the discussion of the typical limiting single failures in Table 7-22 is consistent with the FSAR Section 15.1.5 discussion of the typical limiting single failures. The identification of the specific limiting single failures is outside the approved scope of TR-0516-49416-P, Revision 4, and is addressed in the FSAR instead. However, TR-0516-49416-P, Revision 4, is revised as shown in the attached markup to clarify the wording for consistency with this response and other similar tables in the report.

Markups of the affected changes, as described in the response, are provided below:

Audit Question A-NonLOCA.LTR-66

Table 7-22 Acceptance criteria, single active failure, loss of power scenarios - steam line break

Acceptance Criteria / Single Active Failure / Loss of Power Scenarios of Interest	Discussion
MCHFR	Critical heat flux is potentially challenged for this overcooling event. (Reactivity insertion rates from the overcooling event are insufficient to challenge fuel centerline temperature.)
Radiological consequences	A postulated break in the main steam line is evaluated for radiological consequences.
Failure of one MSIV to close on the train with break	MSIV single failure typically has no effect on MCHFR since DHRS actuation is not before the time when MCHFR occurs. MSIV single failure is typically limiting~~potentially bounding~~ for mass releases for break locations outside the CNV if mass release is calculated for use in the downstream radiological analysis.
Failure of one FWIV to close on the train with the break	FWIV single failure typically has no effect on MCHFR since DHRS actuation is not before the time when MCHFR occurs. FWIV single failure is typically limiting~~potentially bounding~~ for mass releases for break locations inside the CNV if mass release is calculated for use in the downstream radiological analysis.
No loss of power	Loss of power scenarios typically terminate feedwater and/or trip the reactor, thus mitigating the overcooling event.

7.2.4.2 Acceptance Criteria

Evaluation of the most challenging case relative to the acceptance criteria is presented in Table 7-23.

Table 7-23 Acceptance criteria - steam line break

Acceptance Criteria	Discussion
Primary pressure	Due to the depressurizing nature of this cooldown event, primary pressure remains below the acceptance criterion for peak primary pressure.
Secondary pressure	Due to the depressurizing nature of this cooldown event, secondary pressure remains below the acceptance criterion for peak secondary pressure.
Critical heat flux ratio	This criterion is evaluated by downstream subchannel analysis. Sensitivity cases are performed to support the follow-on MCHFR evaluation.

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February 04, 2025 Docket No. 052-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of the NRC's Request for Docketing of Audit Response

The purpose of this letter is to provide NuScale's response to the NRC's request to docket an audit response. The response to the individual audit item is provided in the attachment.

This letter contains NuScale's response to the following audit question from the NRC:

Non-LOCA

Information related to NuScale Topical Report: "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416, Revision 4.
One Item:
A-NonLOCA.LTR-20 (proprietary and nonproprietary)

The audit item specified above has been attached to the NRC EIE submittal. The response includes both proprietary and nonproprietary versions. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 1) supports this request.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Thomas Griffith at 541-452-7813 or tgriffith@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 04, 2025.

Sincerely,

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: Affidavit of Mark W. Shaver, AF-179125

Enclosure 1:

Affidavit of Mark W. Shaver, AF-179125

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NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying response reveals distinguishing aspects about the process by which NuScale developed the Standard Design Approval Application for the US460 design.

NuScale has performed significant research and evaluation to develop a basis for this process and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the attached response entitled "A-NonLOCA.LTR-20". The attachment contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 04, 2025.

Mark W. Shaver

Response to NuScale Topical Report Audit Question

Question Number: A-NonLOCA.LTR-20

Receipt Date: 04/01/2024

Question:

{{

}}$^{2(a),(c)}$ The SG-DHRS loop inventory is determined when the SG-DHRS loop is isolated for DHRS operation. In various Chapter 15 events, the loop isolation timing varies depending on the event progression and plant response. Provide a bounding SG-DHRS loop inventory analysis for applicable Chapter 15 events to ensure sufficient DHRS heat removal capacity.

Response:

{{

}}$^{2(a),(c)}$ The results show that there is significant DHRS heat transfer capability across a wide range of loop inventory. The results also show that the DHRS behavior for the SDAA is consistent with the DCA. {{

}}$^{2(a),(c)}$

{{

}}2(a),(c) The loop inventory during a transient event is a function of the initial inventory in the SG tubes, relevant single failures, the isolation timing, and, importantly, the transient boundary conditions modeled for the condensate feedwater (FW) pump response. Figure 1 of the file shows the condensate FW pump design limits and NRELAP5 model implementation for Chapter 15 events where a detailed FW pump response is modeled. The Chapter 15 events that result in a single train of DHRS conservatively model the FW pump response and also model conservative isolation times as specifically described in the Final Safety Analysis Report (FSAR) for those events. Sufficient DHRS heat removal capacity is demonstrated in Chapter 15 events by showing that specified acceptable fuel design limits (SAFDLs) are met and a safe, stabilized condition is achieved.

Section 15.1.2 of the FSAR addresses the anticipated operational occurrence (AOO) consisting of an increase in FW flow. The analysis of this event explicitly considers the potential for loop inventory to affect DHRS performance. Tables 15.1-4 and 15.1-5 show the sequence of events for the limiting minimum critical heat flux ratio (MCHFR) and SG overfill cases, respectively.

For MCHFR cases, Section 15.1.2.2 states that, “In order to bound the possible FW flow increase scenarios, a spectrum of FW flow increases are analyzed to demonstrate that limiting conditions for MCHFR are reached.” Figure 15.1-8 through Figure 15.1-16 of SDAA show results for the limiting MCHFR case. The limiting FW flow increase is a near instantaneous 15-percent increase of normal FW flow. Section 15.1.2.3.3 concludes that, “Passive DHRS cooling is established, and the transient is terminated with the NPM in a safe, stable condition.”

For SG overfill cases, Section 15.1.2.2 identifies that the increase in FW flow could overfill the SG that increased SG levels can degrade DHRS performance (as shown in Slide 11). Therefore, Section 15.1.2.3.3 describes the conservative assumptions to increase SG level and possibly reduce DHRS heat removal. For the limiting SG overfill case, slow closure (i.e., maximum closing time) of secondary isolation valves is modeled and single failure of a feedwater isolation valve is modeled, allowing FW flow into the affected loop to continue until the feedwater regulating valve closes. The results show that “the SG does not overfill as shown in Figure 15.1-18” and that the “[e]valuation of cases biased for SG overfill demonstrate

adequate DHRS heat removal capability is available under these conditions as shown in Figure 15.1-17."

Section 15.1.5 of the FSAR addresses the postulated accident (PA) consisting of a main steam line break (MSLB). The analysis of the MSLB event explicitly considers the potential for DHRS performance to be affected. Figures 15.1-29 through 15.1-43 show the MSLB results for various cases. Some of the cases result in disabling an entire train of DHRS, including the limiting reactor coolant system (RCS) pressure case as described in Section 15.1.5.3.3. The RCS average temperature response for the limiting RCS pressure case in Figure 15.1-32 demonstrate there is adequate DHRS heat removal capability even with only a single train of DHRS. Note that the supporting calculations for FSAR Section 15.1.5 have been provided in the eRR for Chapter 15 with the response to audit question A-15.1.1-5.

The effect of loop inventory on DHRS performance has been considered in Chapter 15 based on the conservative assumptions used for single failure, isolation times, and FW pump response in relevant events. The results demonstrate that DHRS performance is adequate for the predicted range of DHRS loop inventories during those events.

In summary, each aspect of the audit question is addressed by providing the text of the question in indented, italicized text, followed by the NuScale response in regular text.

> *"The SG-DHRS loop inventory is determined when the SG-DHRS loop is isolated for DHRS operation."*

NuScale agrees.

> *"In various Chapter 15 events, the loop isolation timing varies depending on the event progression and plant response."*

NuScale agrees.

> *"Provide a bounding SG-DHRS loop inventory analysis for applicable Chapter 15 events to ensure sufficient DHRS heat removal capacity."*

Relevant Chapter 15 events where DHRS inventory could be negatively impacted due to loop inventory (e.g., increase in FW flow in Section 15.1.2) are already analyzed to explicitly account for the effect of loop inventory during these events by the assumed single failure, modeled isolation times, and FW pump response. Adequate DHRS heat removal capacity is demonstrated in Chapter 15 for the relevant events.

No changes to the SDAA are necessary.

Enclosure 3:

Affidavit of Mark W. Shaver, AF-182333

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairsof NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the process by which NuScale develops its Non-Loss-of-Coolant Accident Analysis Methodology.

NuScale has performed significant research and evaluation to develop a basis for this method and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed report entitled, "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-A, Revision 5. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 16, 2025.

Mark W. Shaver

Mark W. Shaver