NUSCALE POWER™

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January 9, 2017

Docket: PROJ0769

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 Technical Reports Supporting the NuScale Design Certification Application (NRC Project No. 0769).

REFERENCES: 1. NuScale Letter to the NRC, “NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application,” dated December 31, 2016.

NuScale Power, LLC (NuScale) has submitted a Design Certification Application for its Integral Small Modular Reactor design (Reference 1). The purpose of this letter is to provide the below listed technical reports. These technical reports are referenced in the NuScale Final Safety Analysis Report and provide supplementary information, data and analyses.

1) Containment Response Analysis Methodology Technical Report, TR-0516-49084

2) Long-Term Cooling Methodology, TR-0916-51299

Enclosures 1 and 2 contain the proprietary versions of the above reports. NuScale requests that the proprietary versions be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavits (Enclosures 5 and 6) support this request. Enclosure 2 has also been determined to contain Export Controlled Information. This information must be protected from public disclosure in accordance with the requirements of 10 CFR § 810. Enclosures 3 and 4 contain the non-proprietary versions of these reports.

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

Please feel free to contact Zackary Rad, Director, Regulatory Affairs at 980.349.4831 or at zrad@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
Vice President, Regulatory Affairs

Distribution: Frank Akstulewicz, NRC, TWFN-6C20
Greg Cranston, NRC, TWFN-6E55
Omid Tabatabai, NRC, TWFN- 6E55
Samuel Lee, NRC, TWFN-6C20
Rani Franovich, NRC, TWFN-6E55

Enclosure 1: Containment Response Analysis Methodology Technical Report, TR-0516-49084-P, Revision 0, proprietary version

Enclosure 2: Long-Term Cooling Methodology, TR-0916-51299-P, Revision 0, proprietary version

Enclosure 3: Containment Response Analysis Methodology Technical Report, TR-0516-49084-NP, Revision 0, nonproprietary version

Enclosure 4: Long-Term Cooling Methodology, TR-0916-51299-NP, Revision 0, nonproprietary version

Enclosure 5: Affidavit of Thomas A. Bergman, AF-0117-52660

Enclosure 6: Affidavit of Thomas A. Bergman, AF-1216-52497

Enclosure 1:

Containment Response Analysis Methodology Technical Report, TR-0516-49084-P, Revision 0, proprietary version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Enclosure 2:

Long-Term Cooling Methodology, TR-0916-51299-P, Revision 0, proprietary version

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

Enclosure 3:

Containment Response Analysis Methodology Technical Report, TR-0516-49084-NP, Revision 0, nonproprietary version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Containment Response Analysis Methodology Technical Report

January, 2017

Revision 0

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1100 NE Circle Blvd., Suite 200

Corvallis, Oregon 97330

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Abstract

This report presents the NuScale Power, LLC, methodology used to analyze the mass and energy release into the containment vessel (CNV) for the spectrum of design basis transients and accidents, and the resulting pressure and temperature response of the CNV. The NuScale Power Module (NPM) limiting peak pressure and temperature results determined using the methodology are presented.

This report demonstrates that the NuScale Power Module containment vessel design accommodates the limiting loss-of-coolant and non-loss-of-coolant events, with respect to peak accident pressure and temperature, including sufficient margin. This report also demonstrates conformance to 10 CFR 50 Appendix A, General Design Criteria (GDC) 16 and 50, and Principal Design Criterion (PDC) 38 along with compliance with relevant Acceptance Criteria given by the Design Specific Review Standard for NuScale Small Modular Reactor Design, Section 6.2.1 (Reference 7.1.4).

This report is intended to be incorporated by reference into Design Certification Application Section 6.2.

Executive Summary

This report presents the NuScale Power, LLC, (NuScale) methodology used to analyze the mass and energy release into the containment vessel (CNV) for the spectrum of design basis transients and accidents, and the resulting pressure and temperature response of the CNV. The NuScale Power Module (NPM) limiting peak pressure and temperature results determined using the methodology are presented.

The containment response analysis methodology uses the NRELAP5 thermal-hydraulic code, which is a NuScale-modified version of the RELAP5-3D© v 4.1.3 code used for loss-of-coolant accident (LOCA) and non-LOCA transient and accident analyses, including the response of the CNV.

The NRELAP5 model used to model NPM performance for primary system LOCA and emergency core cooling system valve-opening event analyses is similar with the model used in the LOCA evaluation model, described by Reference 7.2.1. The NRELAP5 model used for secondary system pipe-break analysis in the containment response analysis methodology is consistent with the non-LOCA model described by the Non-LOCA Evaluation Model Report (Ref: 7.2.2). Changes made to these models that maximize containment pressure and temperature response to primary and secondary system release events are described in this report. These changes conservatively maximize the mass and energy release and minimize the performance of the containment heat removal system and are consistent with acceptance criteria given by Design Specific Review Standard Section 6.2.1.3 (Ref: 7.1.6) and Design Specific Review Standard Section 6.2.1.4 (Ref: 7.1.7).

Other differences exist between the NRELAP5 model used to model NPM performance for primary system LOCA and emergency core cooling system valve-opening event analyses and the containment analysis model. These modeling differences, identified in Section 3.2.4.1, have a negligible impact on the CNV analysis results.

Initial and boundary conditions for the spectrum of primary system release containment response analyses and secondary system pipe break analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. These initial and boundary conditions are described in this report, along with the rationale for their selection.

The results of the NRELAP5 limiting analyses using the containment response analysis methodology are presented in this report. These analyses cover the spectrum of primary system mass and energy release scenarios for the NPM, and secondary system pipe break scenarios.

The limiting LOCA peak pressure and CNV wall temperature are a result of the reactor coolant system (RCS) injection line break. The LOCA limiting peak CNV wall temperature is approximately 523 degrees F and it results from a reactor coolant system injection line break case, with a loss of normal alternating current (AC) power. The LOCA limiting peak pressure is approximately 921 psia, which also results from a reactor coolant system injection line break case with a loss of normal AC power. The LOCA event peak CNV pressure is below the CNV design pressure of 1000 psia. The LOCA peak CNV pressure and wall temperature bound the main steamline break (MSLB) and feedwater line break (FWLB) results.

The overall limiting peak CNV accident pressure is approximately 951 psia. It results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and direct current (DC) power. The peak pressure of the limiting anticipated operational occurrence is also less than the CNV design pressure of 1000 psia. The CNV pressure for this limiting case is reduced to below 50 percent of the peak value in less than 2 hours, demonstrating adequate NPM containment heat removal.

Section 5.4 discusses margin in the NPM design that is not included in the CNV design pressure rating or modeled in the containment response analyses. Design factors conservatively not credited include CNV shell stress margins, CNV cladding material and the availability of the decay heat removal system (DHRS).

The containment response analysis methodology demonstrates that the NPM design has adequate margin to design limits and that it satisfies the requirements of General Design Criteria (GDC) 16, 50, and Principal Design Criterion (PDC) 38.

1.0 Introduction

1.1 Purpose

The purpose of this report is to present the NuScale Power, LLC, methodology used to analyze the mass and energy (M&E) release into the containment vessel (CNV) for the spectrum of design-basis transients and accidents and the resulting pressure and temperature response of the CNV, and to present the NuScale Power Module (NPM)-limiting peak pressure and temperature results that are determined using the methodology.

1.2 Scope

The scope of the Containment Response Analysis Technical Report comprises the M&E release from the spectrum of primary system and secondary system design basis transients and accidents and the resulting CNV pressure and temperature response. The duration of the analyses is sufficient to establish the CNV peak pressure and peak temperature for all events, and to demonstrate the decrease in pressure to one-half of the peak value within 24 hours. The NRELAP5 code, described in Reference 7.2.1, is used in this methodology. The simulation models used in the containment response analysis methodology are similar to the models used in the NuScale LOCA and non-LOCA methdoologies (Reference 7.2.1 and Reference 7.2.2). This report documents the differences compared to those methodologies and provides bounding analysis results for the limiting accident scenarios.

Operation at rated power is the bounding initial condition for the limiting CNV pressure and temperature event scenarios for the NPM. Operation at rated power is the bounding initial condition because it has the maximum stored energy and decay heat. For the NPM, reduced power levels and shutdown conditions are non-limiting and do not need to be analyzed specifically.

Chapter 2.0 describes the regulatory guidance that is applicable to the scope of the containment response analysis methodology and summarizes how the methodology meets the guidance. Chapter 3.0 describes the NRELAP5 computer code along with the qualification of the code for the scope of the containment response analysis methodology. Chapter 3.0 also describes the NRELAP5 model of the NPM used in the containment response analysis methodology. Chapter 4.0 describes validation and verification of the containment response analysis methodology as well as primary and secondary release event models, including the code and model qualification and conservativisms. Chapter 5.0 presents the containment response analysis methodology for primary system release events, the limiting scenarios and associated analysis results, and nominal case results demonstrating conservatism in assumed certain initial conditions. Chapter 5.0 also presents the containment response analysis methodology for secondary-system pipe breaks along with associated analysis results. Chapter 6.0 presents the report summary and conclusions.

The methodology for simulation of the longer-term M&E release and CNV and NPM response that is used for establishing the equipment qualification (EQ) pressure and

temperature envelopes, and to demonstrate the long-term cooling capabilities of the NPM, are not included within the scope of this report.

1.3 Abbreviations

Table 1-1 Abbreviations

Term	Definition
AC	alternating current
ASME	American Society of Mechanical Engineers
ANS	American Nuclear Society
BPVC	Boiler and Pressure Vessel Code
CFR	Code of Federal Regulations
CNV	containment vessel
CVCS	chemical and volume control system
DC	direct current
DHRS	decay heat removal system
DSRS	Design Specific Review Standard
ECCS	emergency core cooling system
FSAR	Final Safety Analysis Report
FWIV	feedwater isolation valve
FWLB	feedwater line break
FWRV	feedwater regulating valve
GDC	General Design Criteria
IAB	inadvertent actuation block
ID	inside diameter
LOCA	loss-of-coolant accident
M&E	mass and energy
MSIV	main steam isolation valve
MSLB	main steam line break
NIST-1	NuScale Integral System Test Facility
NPM	NuScale Power Module
NRC	U. S. Nuclear Regulatory Commission
OD	outer diameter
PDC	Principal Design Criterion
PIRT	phenomena identification and ranking table
PWR	pressurized water reactor
RCS	reactor coolant system
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RSV	reactor safety valve
RVV	reactor vent valve
SG	steam generator
SMR	small modular reactor
SRP	Standard Review Plan

2.0 Background

The CNV is a compact, steel pressure vessel that consists of an upright cylinder with top and bottom head closures. The CNV is partially immersed in a below-grade reactor pool that provides a passive heat sink and is absent of internal sumps or subcompartments that could entrap water or gases. The CNV and the reactor pool are housed within a Seismic Category 1 Reactor Building. The unique nature of the NPM design necessitates development of a specific containment response analysis methodology.

This technical report describes the thermal-hydraulic accident analysis methodology for primary and secondary system M&E releases into the CNV of the NPM, and the resulting pressure and temperature response of the CNV. This report presents the bases for the analysis methodology and results in support of Chapter 6 of the NuScale Final Safety Analysis Report (FSAR). The containment response analysis methodology and CNV peak pressure and temperature results are compared to applicable regulatory guidance, including the Design Specific Review Standard for NuScale Small Modular Reactor (SMR) Design, Section 6.2.1 (Ref: 7.1.4). A spectrum of M&E release events is analyzed that bounds all of the LOCAs and valve-opening transients in the primary system and all secondary-system pipe-break accidents. The containment response analysis methodology uses conservative initial conditions and boundary conditions to ensure overall conservative results. The limiting results are shown to be less than the design pressure (1000 psia) and the design temperature (550 degrees F) of the CNV.

The qualification of the LOCA and non-LOCA methodologies presented in References 7.2.1 and 7.2.2, in particular the comparisons to separate effects tests and integral effects tests, are applicable for the containment response analysis methodology presented in this report. The differences in the NRELAP5 simulation models used in the containment response analysis methodology as compared to the LOCA and non-LOCA models, along with the rationale for the selection of conservative initial and boundary conditions, are the subject of this report. Analysis results are presented for the limiting cases, along with nominal condition case results, demonstrating conservatism in certain initial conditions.

2.1 Regulatory Requirements

The Nuclear Regulatory Commission (NRC) regulations and regulatory guidance applicable to the containment response analysis methodology are described in this section. The elements of the containment response analysis methodology that address each of these regulations and requirements are discussed.

2.1.1 10 CFR 50 Appendix A - General Design Criteria for Nuclear Power Plants

The General Design Criteria (GDC) for Nuclear Power Plants, Appendix A to 10 CFR 50 (Ref: 7.1.2), include the NRC regulations applicable to the containment response methodology. Compliance with GDC 16 and 50 and PDC 38 is as follows:

General Design Criterion 16 - The analyses performed per the containment response analysis methodology are used to establish the limiting CNV pressure and temperature conditions resulting from the spectrum of design-basis primary system and secondary system M&E releases resulting from pipe breaks and valve actuations. The CNV is

designed to ensure that the design pressure and temperature limit are not exceeded as demonstrated by the analysis results.

Principal Design Criterion 38 - The analyses performed per the containment response analysis methodology establish the performance of NPM containment heat removal and demonstrate that the containment peak pressure and temperature are rapidly reduced. The methodology addresses LOCAs, valve-opening events and secondary pipe breaks. Following containment isolation and opening of the ECCS valves, the containment heat removal function is passive and does not require electric power. The requirement to rapidly reduce the containment pressure and temperature is demonstrated by the peak pressure decreasing to less than 50 percent of the peak value consistent with Design Specific Review Standard (DSRS) Section 6.2.1.1.A (Ref: 7.1.5). Potential single failures have been considered in the methodology, and the results of the analyses show that the safety functions can be performed including the limiting single failure.

General Design Criterion 50 - The analyses performed per the containment response analysis methodology demonstrate that sufficient margin to the CNV design pressure and temperature is maintained. The methodology explicitly models all energy sources including energy in the steam generators (SGs). However, the energy from the post-LOCA oxidation of the cladding that is typical of light water reactors is not applicable to the NuScale design and is not included. Calculated cladding temperatures for design basis LOCAs are below the level where cladding oxidation occurs on a time scale of a LOCA event for the NPM. Therefore, this requirement is satisfied by the design that precludes fuel temperature reaching critical heat flux and any significant fuel cladding heatup. For the NPM loss-of-coolant accident evaluation model core coverage and a minimum critical heat flux ratio are significantly greater than the safety limit, which precludes the occurrence of cladding oxidation (see Reference 7.2.1, Section 2.2). The NRELAP5 code and model have been assessed to experimental data to demonstrate the capability to reliably simulate the scenarios of interest. Conservative values for initial conditions and boundary conditions ensure an overall conservative analysis result.

2.1.2 Regulatory Guide 1.203

Regulatory Guide 1.203, “Transient and Accident Analysis Methods” (Ref: 7.1.3), describes a process that the NRC staff considers acceptable for industry use to develop and assess evaluation models used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design basis accident.

The containment response analysis methodology is an extension of the NuScale LOCA and non-LOCA methodologies developed following the guidance of Regulatory Guide 1.203. This report references the LOCA and non-LOCA methodologies and identifies and justifies the differences in the containment response methodology when compared to those methodologies.

2.1.3 Design Specific Review Standard for NuScale Small Modular Reactor Design

The NRC has issued “Design-Specific Review Standard for NuScale SMR Design” to guide the NRC staff review of the NuScale FSAR. This document replaces NUREG-0800, “Standard Review Plan.” The NRC staff has specified the DSRS as an acceptable method for evaluating whether an application complies with NRC regulations for NuScale small modular reactor (SMR) applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. The DSRS is used by NuScale as a guide to ensure that the containment response analysis methodology addresses all of the elements that NRC has included. Sections 2.2.3.1 through 2.2.3.4 describe how the containment response analysis methodology is consistent with the applicable DSRS guidelines, justify differences, or indicate non-applicability.

2.1.3.1 Design Specific Review Standard 6.2.1 Containment Functional Design

The DSRS Section 6.2.1, “Containment Functional Design” (Ref: 7.1.4), includes a high-level summary of an acceptable approach and content for a containment response analysis methodology, and references the lower-tier subsections with additional detail about the approach and contents. The comparison of the containment response analysis methodology to applicable content in DSRS Section 6.2.1 is provided in Table 2-1:

Table 2-1 Compliance with Design Specific Review Standard Section 6.2.1

DSRS Section 6.2.1, p. 1	**Containment Response Analysis Methodology**
The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant (LOCA), steam line, or feedwater line break accidents.	The containment response analysis methodology addresses LOCAs resulting from postulated limiting breaks, valve-opening events, main steam line break (MSLB) accidents, and feedwater line break (FWLB) accidents. A conservative approach to modeling the full spectrum of break and valve sizes and locations is included. The limiting results are less than the CNV design pressure and temperature.
The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen.	The containment response analysis methodology includes all primary system and secondary energy sources that contribute to the M&E release. The energy from the post-LOCA oxidation of the cladding that is typical of light water reactors is not applicable to the NuScale design and is not included as discussed by Section 2.1.1.
The subsequent thermodynamic effects in the containment resulting from the release of the coolant mass and energy are determined from a solution of the incremental space and time-dependent	The containment response analysis methodology uses the NRELAP5 system thermal-hydraulic analysis code. NRELAP5 solves the time-

energy, mass, and momentum conservation equations.	dependent conservation equations for mass, momentum, and energy.
DSRS Section 6.2.1, p. 2	Containment Response Analysis Methodology
GDC 50, among other things, requires that consideration be given to the potential consequences of degraded engineered safety features, such as the containment heat removal system and the ECCS, the limitations in defining accident phenomena, and the conservatism of calculation models and input parameters in assessing containment design margins.	The containment response analysis methodology models engineered safety features including NPM containment heat removal and the ECCS with conservative assumptions. Postulated single failures are considered. Initial and boundary conditions are selected to maximize containment pressure and temperature response. Margin is maintained between the analysis results and the CNV design pressure and temperature limits (See Section 5.2.2).
The regulation in 10 CFR 50 Appendix K.I.A provides the sources of energy that are required and acceptable to be included in determining the mass and energy release from loss-of-coolant accidents and secondary systems pipe ruptures.	The containment response analysis methodology includes all of the sources of energy required in Appendix K.I.A with the following exceptions to Items 4 and 5: 4) Fission Product Decay: The American Nuclear Society (ANS)-5.1-1979 decay heat standard with a two-sigma uncertainty is used rather than 120 percent of the 1971 American Nuclear Society (ANS) standard. Consistent with DSRS 6.2.1.3, Section II, Acceptance Criterion 1.C.v, the ANS-5.1-1979 standard is equal to the decay heat model given in Standard Review Plan (SRP) Section 9.2.5. 5) Metal-Water Reaction: The energy from the post-LOCA oxidation of the cladding that is typical of light water reactors is not applicable to the NuScale design and is not included as discussed in Section 2.1.1.
DSRS Section 6.2.1, p. 4	Containment Response Analysis Methodology
The temperature and pressure profiles provided in the applicant's technical submittal for the spectrum of LOCA and main steam line break accidents are acceptable for use in equipment qualification (i.e., there is reasonable assurance that the actual temperatures and pressures for the postulated accidents will not exceed these profiles anywhere within the specified environmental zones, except in the break zone).	Methodology for simulation of the M&E release and CNV response that is used for establishing the equipment qualification pressure and temperature envelopes, and to demonstrate the long-term cooling capabilities of the NPM, are outside of the scope of this report.

2.1.3.2 Design Specific Review Standard 6.2.1.1.A Containment

The DSRS Section 6.2.1.1.A, "Containment" (Ref: 7.1.5), includes content related to containment design, including some elements that are associated with the capability to withstand M&E releases. The comparison of the containment response analysis methodology to applicable content in DSRS Section 6.2.1.1.A is provided in Table 2-2:

Table 2-2 Compliance with Design Specific Review Standard Section 6.2.1.1.A

DSRS Section 6.2.1.1.A, p. 1	**Containment Response Analysis Methodology**
The temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (LOCAs) (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks	The containment response analysis methodology includes the spectrum of primary release events resulting from postulated limiting breaks (LOCAs) and valve openings, MSLB accidents, and FWLB accidents. The limiting results are less than the CNV design pressure and temperature.
The effectiveness of static (passive) and active heat removal mechanisms.	The containment response analysis methodology includes conservative modeling of passive heat removal systems (there are no active heat removal systems in the NuScale design). Specifically, conservatisms are employed in conservative assumed initial and boundary conditions, including the reactor pool to ensure a bounding peak CNV peak pressure and temperature following events involving release of mass and energy into the CNV. The performance of these systems is shown to be effective in limiting the CNV pressure and temperature response to within acceptable design limits. Conservatism in initial and boundary conditions is discussed in Section 3.5
DSRS Section 6.2.1.1.A, p. 4	Containment Response Analysis Methodology
To satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants in the design stage (i.e., at the construction permit (CP) or design certification (DC) stage) of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a LOCA, or a steam or feedwater line break. Design margins of less than 10% may be sufficient, provided appropriate justification is provided. For plants at the operating license (OL) or COL stage of review, the	For the NuScale FSAR submittal, the results of the containment response analysis methodology for the limiting event scenarios are less than the CNV design pressure and temperature. The containment design pressure provides approximately 8 percent margin to the limiting LOCA peak CNV pressure. The LOCA peak pressure bounds the steam or FWLB peak pressures. Additionally, the overall peak accident pressure resulting from an inadvertent reactor recirculation valve (RRV) opening event is approximately 5 percent below the

peak calculated containment pressure following a LOCA, or a steam or feedwater line break, should be less than the containment design pressure.	CNV design pressure. Additional margin is provided by the NPM design to satisfy the requirements of GDC 16 and 50 as discussed in Section 5.4.
To satisfy the requirements of GDC 38 to rapidly reduce the containment pressure, the containment pressure should be reduced to less than 50% of the peak calculated pressure for the design basis LOCA within 24 hours after the postulated accident. If analysis shows that the calculated containment pressure may not be reduced to 50% of the peak calculated pressure within 24 hours, the organization responsible for DSRS Section 15.0.3 should be notified.	The containment response analysis methodology is applicable to the initial CNV response and demonstrates that the peak pressure and temperature are within the CNV design limits. The methodology also demonstrates that the CNV pressure decreases to less than 50 percent of the peak pressure within 24 hours to satisfy the requirements of Principal Design Criterion 38 for rapid reduction of containment pressure. Figure 5-29 demonstrates that the CNV pressure for the limiting case is reduced to less than 50 percent of its peak value in less than two hours. This demonstrates the CNV heat removal capability.
DSRS Section 6.2.1.1.A, p. 5	Containment Response Analysis Methodology
To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the LOCA analysis should be based on the assumption of loss of offsite power and the most severe single failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure	The containment response analysis methodology models engineered safety features involving the containment heat removal function and the ECCS. Conservative assumptions regarding safety feature performance, in conjunction with conservative initial and boundary conditions, ensure that the CNV peak pressure and temperature analysis results following a primary system release are bounding (See Section 5.4). A limiting single failure is considered (See Section 5.1.1). Sensitivity cases considering the availability of power are performed to ensure that assumptions associated with availability of these systems ensure limiting peak pressure and temperature results (see Section 3.5.2). There are no emergency diesel generators associated with the NPM design. Margin is maintained between the analysis results and the CNV design pressure and temperature limits for the limiting cases.
4. To satisfy the requirements of GDC 38 and 50 with respect to the containment heat removal capability and design margin, the containment response analysis for postulated secondary system pipe ruptures should be based	The containment response analysis methodology models engineered safety features including NPM containment heat removal and the ECCS with conservative assumptions that maximize containment pressure and temperature

on the most severe single failure of the secondary system isolation provisions (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis. Acceptable methods for the calculation of the containment environmental response to main steam line break accidents are found in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."	following a secondary system pipe rupture. For postulated secondary system pipe ruptures, a limiting single failure is considered, including main steam isolation valve or feedwater isolation valve (FWIV) failure. For the NuScale design, full power and the maximum break size at each break location are the limiting conditions. Initial and boundary conditions are selected to maximize containment pressure and temperature response (See Section 3.4). Margin is maintained between the analysis results and the CNV design pressure and temperature limits. The longer-term response for equipment qualification is not in the scope of this report.

2.1.3.3 Design Specific Review Standard 6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

The DSRS Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)" (Ref: 7.1.6), includes the details of an acceptable approach and content for an M&E methodology for LOCAs. As noted, a comparison of NPM design reveals that some of the DSRS content is based on pressurized water reactor (PWR) large-break LOCA phenomena that are not applicable to the NuScale design. The comparison of the M&E methodology to applicable content in DSRS Section 6.2.1.3 is provided in Table 2-3:

Table 2-3 Compliance with Design Specific Review Standard Section 6.2.1.3

DSRS Section 6.2.1.3, p. 3	**Containment Response Analysis Methodology**
A. Sources of Energy. The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system (RCS) metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system, including the steam generator tubing and secondary water. Calculations of the energy available for release from the above sources should	The containment response analysis methodology includes reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system (RCS) metal, including the reactor vessel and reactor vessel internals; and stored energy in the secondary system, including the SG tubing and secondary water. Metal-water reaction energy is not included in the containment response analysis methodology as discussed in Section 2.1.1.

be done in general accordance with the requirements of paragraph I.A. in Appendix K to 10 CFR Part 50, “Sources of Heat during the LOCA.” However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and subsequent phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the RCS and steam generator metal and increasing the RCS and steam generator secondary mass to account for uncertainties and thermal expansion. The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, “Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters,” concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.	The containment response analysis methodology models available energy sources in accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A, with the exception of 1) metal-water reaction energy is not included, and 2) the ANS-5.1-1979 decay heat standard with a two-sigma uncertainty is used rather than a factor of 1.2 with the 1971 ANS standard. Consistent with DSRS 6.2.1.3, Section II, Acceptance Criterion 1.C.v, the ANS-5.1-1979 standard is equal to the decay heat model given in SRP Section 9.2.5. The containment response analysis methodology model of initial stored energy in the fuel is consistent with Paragraph I.A.1 of Appendix K to 10 CFR Part 50. Fuel rods are initialized at the maximum initial stored energy condition as determined by the fuel performance analysis. The fuel heat capacity values are conservatively increased to 115 percent of their nominal values to maximize fuel stored energy. The containment response analysis methodology includes conservative elements that maximize the energy release including sensible heat stored in primary and secondary metal structures, and increasing the RCS mass to account for uncertainties and thermal expansion. The secondary mass is not a significant contributor and a nominal value is used. The containment response analysis methodology does not consider the fuel cladding swelling and rupture prediction requirements of paragraph I.B in Appendix K to 10 CFR Part 50. Calculated cladding temperatures for design basis LOCAs are below the threshold for cladding swelling and rupture.
DSRS Section 6.2.1.3, p. 4	Containment Response Analysis Methodology

B. Break Size and Location i. The staff's review of the applicant's choice of break locations and types is discussed in SRP Section 3.6.2. ii. Of several breaks postulated, the break selected as the reference case should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area. iii. Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.	The containment response analysis methodology includes consideration of a spectrum of break types discussed by Section 3.2.4.1. Break locations are chosen such that M&E releases to containment are maximized. {{ }}2(a),(c)
C. Calculations In general, calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure response). The criteria given below for each phase of the accident indicate the conservatism that should exist.	The containment response analysis methodology focuses on determining the maximum post-accident containment pressure and temperature. The methodology employs conservative elements to ensure an overall conservative result.
i. Containment Analysis The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the reactor, piping and containment systems are similar to those of an approved ECCS analysis. The computer programs that are currently acceptable include CRAFT-2, and RELAP5, when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.	The M&E release determined by the containment response analysis methodology is based on the NRELAP5 computer code, and the modeling approach is similar to the NuScale LOCA evaluation model, Reference 7.2.1 that complies with the applicable portions of 10 CFR 50 Appendix K. Specific changes to the LOCA evaluation model required to convert it to a conservative methodology to model primary system mass release events are described in Section 3.2.4.1. The Moody critical flow model with a discharge coefficient of 1.0 is used for saturated two-phase critical flow.

ii. Initial Blowdown Phase Containment Design Basis	
The initial mass of water in the reactor coolant system should be based on the RCS volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.	The containment response analysis methodology assumes an initial power level of 1.02 times the rated power level. Initial RCS volume and mass are consistent with that power level.
Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.	The containment response analysis methodology uses the conservative Moody critical flow model for two-phase saturated fluid conditions consistent with Appendix K. For subcooled fluid conditions the {{ }}2(a),(c) Reference 7.2.1, Sections 8.2.2 and 8.2.3 demonstrates the adequacy of the LOCA evaluation model two-phase and single-phase choked and un-choked flow models for predictions of M&E release based on assessments of comparisons of NRELAP5 mass flow predictions to experimental data.
Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection. Calculations of heat transfer from the secondary coolant to the steam generator tubes should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.	The containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code. The LOCA evaluation model report demonstrates these correlations are applicable to the NPM design (Ref: 7.2.1). The local fluid conditions and the local heat structure surface temperatures determine the heat transfer mode. Nucleate boiling and forced convection are included in the code and are selected if the local conditions are appropriate.
Calculations of heat transfer to the containment wall from released reactor steam should be such that the heat removal from containment is conservatively underestimated so that the containment pressure is maximized. In regions where steam jetting occurs, heat transfer correlations that are based on jetting of coolant (e.g. based on forced convection) may be used as appropriate. Correlations should be appropriately conservative in regions away from jetting phenomena (e.g. based on natural convection, as appropriate). All heat transfer correlations used should be justified.	The containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code. The LOCA evaluation model report demonstrates these correlations are applicable to the NPM design (Ref: 7.2.1). The local fluid conditions and the local heat structure surface temperatures determine the heat transfer mode. Forced convection, natural convection, condensation, conduction, and nucleate boiling are included in the code and are selected if the local conditions are appropriate. Initial and boundary conditions are

Calculations of heat transferred from condensed reactor water in the containment sump into the containment wall and from the reactor vessel wall into the pooled sump water should be based on appropriate heat transfer regimes for the conditions present in containment. Heat transfer through the containment vessel wall into the Reactor Building pool should be demonstrated to conservatively underestimate heat transfer to the pool.	selected to maximize containment pressure and temperature response (See Section 3.5). Steam jetting effects are not modeled.
DSRS Section 6.2.1.3, p. 5	Containment Response Analysis Methodology
iii. Postblowdown Recirculation Phase (Cold Leg RRV Penetration Breaks Only) After initial blowdown through a failed RRV, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the RCS equalizes to the containment pressure, the water remaining in the reactor vessel should be assumed to be saturated. Justification should be provided for the duration of the recirculation period, which is the time from the end of the blowdown to the time when flow from the condensed water in the containment vessel sump comes back through the RRVs into the reactor vessel. Calculations of the refill rate should be based on the ECCS operating condition following the blowdown phase, where energy is released to the RCS primary system by the RCS metal, core decay heat, and the steam generators. The calculated ECCS conditions should conservatively maximize the containment pressure. Calculations of liquid entrainment, (i.e., the carryout rate fraction), which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the NuScale full length emergency cooling heat transfer experiments or conservatively scaled-up test results from subscale test.	The containment response analysis methodology uses the NRELAP5 code that has been determined to be capable of modeling all of the phases of the primary system release events for the NPM design as discussed by Section 3.2. NRELAP5 predicts the evolution of the primary system release event scenario, which includes the time of pressure equalization and the time at which flow of condensed water through the RRVs into the reactor vessel occurs. As discussed in Section 3.1.3, the containment response analysis methodology models applicable phenomena that contribute to maximizing the M&E release and the resulting containment pressure and temperature. The “refill rate” is only applicable to large PWRs. As discussed by the LOCA evaluation model report, the NPM design precludes core uncovery (See Reference 7.2.1). As discussed by Section 3.2.4.1, the containment response analysis methodology models applicable phenomena that contribute to maximizing the M&E release and the resulting containment pressure and temperature. The concept of carryout rate fraction that is applicable to large PWRs is not applicable to the NuScale design. As

The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the carryout rate fraction of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water. Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.	discussed by the LOCA evaluation model report, the NPM design precludes core uncovery, so there is no reflooding phase (See Reference 7.2.1). As discussed by Section 3.2.4.1, the containment response analysis methodology models applicable phenomena that contribute to maximizing the M&E release and the resulting containment pressure and temperature. The concept of steam quenching (that occurs from mixing with ECCS water) that is applicable to large PWRs is not applicable to the NuScale design because ECCS water is not injected into the core. As discussed by Section 3.2.4.1, the containment response analysis methodology models applicable phenomena that contribute to maximizing the M&E release and the resulting containment pressure and temperature. The superheating effect described is a pressurized water reactor LOCA phenomenon that has minimal applicability to the NuScale design. For the NPM design, flow of primary steam over the SG tubes results in heat transfer based on the NRELAP5 heat transfer correlation package. This allows for superheating of the steam as determined by the local conditions.
DSRS Section 6.2.1.3, p. 6	Containment Response Analysis Methodology
iv. Post-Recirculation Phase All remaining stored energy in the primary and secondary systems should be removed during the post-recirculation phase. Steam quenching on the containment vessel walls, due to pressure equalization between the reactor vessel and the containment vessel, should be justified by comparison with applicable experimental data.	The stored energy is distributed as predicted by the NRELAP5 modeling of heat transfer to and from the primary and secondary systems. The duration of the analysis is consistent with the LOCA evaluation model and the applicable figures-of-merit (See Reference 7.2.1). The containment response analysis methodology considers steam condensation on the CNV walls, as discussed by Section 3.2.4.1. The

The results of post-recirculation analytical models should be compared to applicable experimental data.	NRELAP5 code and model have been justified by comparison to applicable experimental data.
v. Decay Heat Phase The dissipation of core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in SRP Section 9.2.5. Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with the condensed water flowing from the containment sump into the reactor vessel through the RRVs.	The containment response analysis methodology models the fission product decay energy using the ANS-5.1-1979 standard plus two-sigma uncertainty. SRP Section 9.2.5 references the same ANS-5.1-1979 standard. The described steam and water mixing process does not occur in the NPM design. Water flowing through the RRVs to the core inlet is below the water mixture level in the downcomer and does not contact the steam produced by decay heat boiling.

2.1.3.4 Design Specific Review Standard 6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures

The DSRS Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures” (Ref: 7.2.2), includes the details of an acceptable approach and content for a M&E methodology for MSLBs and FWLBs. The comparison of the M&E methodology to applicable content in DSRS Section 6.2.1.4 is provided in Table 2-4:

Table 2-4 Compliance with Design Specific Review Standard Section 6.2.1.4

DSRS Section 6.2.1.4, p. 4	**Containment Response Analysis Methodology**
1. Sources of Energy. The sources of energy that should be considered in the analyses of steam and feedwater line break accidents include the stored energy in the affected helical coil SG's metal, including the vessel tubing, feedwater line, and steam line; stored energy in the water contained within the affected helical coil SG; stored energy in the feedwater transferred to the affected helical coil SG before closure of the isolation valves in the feedwater line; stored energy in the steam from the unaffected helical coil SG before the closure of the isolation valves in the helical coil SG crossover lines; and energy transferred from the primary coolant to the water in the affected helical coil SG during blowdown to include energy transferred to the draining DHRS heat exchanger water.	As discussed in Section 3.3, the containment response analysis methodology includes all of the sources of energy stored in the fluid and structures that contribute to the secondary line break scenarios. This includes energy stored in fluid contained in piping systems connected to the break flowpath into the CNV. The containment response analysis methodology considers a spectrum of pipe break sizes and various plant conditions. However, the limiting initial conditions are at 102 percent rated power as the effect of SG liquid mass inventory and feedwater flows is greatest at full power. {{ }}2(a),(c)

The steam line break accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power. The applicant need only analyze the 102-percent power condition if it can demonstrate that the feedwater flows and fluid inventory are greatest at full power.	{{ }}2(a),(c)
2. Mass and Energy Release Rate In general, calculations of the mass and energy release rates during a steam or feedwater line break accident should be performed in a conservative manner from a containment response standpoint (i.e., the postaccident containment pressure and temperature are maximized). The following criteria indicate the degree of conservatism that is desired: A. Mass release rates should be calculated using the Moody model (Ref. 6) for saturated conditions or a model that is demonstrated to be equally conservative.	The containment response analysis methodology maximizes the CNV peak pressure and temperature. The Moody critical flow model with a discharge coefficient of 1.0 is used for saturated two-phase fluid conditions. For subcooled and superheated fluid conditions the {{ }}2(a),(c) A discharge coefficient of 1.0 is used.
B. Calculations of heat transfer to the water in the affected helical coil SG should be based on nucleate boiling heat transfer.	The containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code. The non-LOCA evaluation model report demonstrates these correlations are applicable to the NPM design (Ref: 7.2.2). The local fluid conditions and the local heat structure surface temperatures determine the heat transfer mode. Nucleate boiling heat transfer is included in the code and is selected if the local conditions are appropriate. For the helical coil SG, other heat transfer modes exist as the coolant enters as subcooled liquid and exits as superheated steam. Initial and boundary conditions are selected to maximize containment pressure and temperature response (See Section 3.5).
C. Calculations of mass release should consider the water in the affected helical coil SG and feedwater line, feedwater transferred to the affected helical coil SG before the closure of the isolation valves in the feedwater lines and upon flooding with the DHRS heat exchanger inventory in the affected loop, and steam in the helical coil SG.	The containment response analysis methodology includes the water inventory stored in piping systems connected to the break flowpath into the CNV. The closure of isolation valves, with consideration of a single failure, determines which sources of water contribute to the M&E release to ensure limiting CNV peak pressure and temperature results.
D. If liquid entrainment is assumed in the steam line breaks, experimental data should support the predictions of the liquid entrainment model. A spectrum of steam line breaks should be analyzed, beginning with the double-ended break (DEB) and	The containment response analysis methodology uses the two-phase flow and heat transfer models in the NRELAP5 code. The depressurization of the SG secondary will cause flashing in addition to the increase in primary-to-secondary heat transfer. The initial liquid inventory in the SG

decreasing in area until no entrainment is calculated to occur. This will allow selection of the maximum release case. If no liquid entrainment is assumed, a spectrum of the steam line breaks should be analyzed beginning with the DEB and decreasing in area until it has been demonstrated that the maximum release rate has been considered	secondary will boil and flash, and additional inventory will result from continued feedwater flow and from liquid in connecting pipes. The net effect may include some liquid entrainment in the break flow that is time dependent. An interfacial drag multiplier is available as a junction component option in NRELAP5 to minimize liquid entrainment.
E. Feedwater flow to the affected helical coil SG should be calculated considering the diversion of flow from the other helical coil SG between the two feedwater pipes to the common header with inlets to the helical coil SG on opposite sides of the reactor vessel, feedwater flashing, and increased feedwater pump flow caused by the reduction in helical coil SG pressure. An acceptable method for computing feedwater flow is to assume all feedwater travels to the helical coil SG at the pump run-out rate before isolation. After isolation, the unisolated feedwater mass should be added to the available inventory in the helical coil SG.	The containment response analysis methodology includes the water inventory stored in in piping systems connected to the break flowpath into the CNV. The increase in feedwater flow due to the depressurization of the helical coil SG is considered. The closure of isolation valves with consideration of a single failure determines which sources of water contribute to the M&E release. The net feedwater addition is calculated using conservative modeling assumptions.
DSRS Section 6.2.1.4, p. 5	Containment Response Analysis Methodology
iii. Single-Failure Analyses Steam and feedwater line break analyses should assume a single active failure in the steam or feedwater line isolation provisions to maximize the containment peak pressure and temperature. For the assumed failure of a safety-related steam or feedwater line isolation valve, operation of nonsafety-related equipment may be relied upon as a backup to the safety-related equipment.	The containment response analysis methodology considers single failures that affect the isolation of the main steam lines and feedwater lines. Non-safety valves are credited for isolation as a backup.

3.0 Analysis

3.1 Modeling Software

The containment response analysis methodology uses the NRELAP5 system thermal-hydraulic code, which is a NuScale-modified version of the RELAP5-3D© v 4.1.3 code. NRELAP5 is used for all LOCA and non-LOCA transient and accident analyses, including the response of the CNV. The NRELAP5 simulation model used for the containment response analysis methodology is also similar to the NRELAP5 simulation models used for the LOCA and non-LOCA methodologies, which are presented in References 7.2.1 and 7.2.2. The phenomena identification and ranking tables (PIRT) developed for the LOCA and non-LOCA methodologies are applicable to the containment response analysis methodology. The qualification of the LOCA and non-LOCA methodologies, in particular the comparisons to separate effects tests and integral effects tests, applicable to the containment response analysis methodology are presented in Section 4.1. The differences in the NRELAP5 simulation models used in the containment response analysis methodology as compared to the LOCA and non-LOCA models, along with the rationale for selection of conservative initial and boundary conditions, are the subject of this report.

3.2 NRELAP5 Base Simulation Model Development

3.2.1 RELAP5-3D©

RELAP5-3D©, version 4.1.3 was used as the baseline development platform for the NRELAP5 code. RELAP5-3D© was procured by NuScale and subsequently features were added to address unique aspects of the NuScale design and licensing methodology. The following is a brief description of the RELAP5-3D© code.

The RELAP5-3D© code has been developed for best-estimate transient simulation of light water RCSs during postulated accidents. The code models the coupled behavior of the RCS and the core for LOCAs and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems.

The RELAP5-3D© code is based on a non-homogeneous and non-equilibrium model for the two-phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. The code includes many generic component models from which general systems can be simulated. The component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor kinetics, electric heaters, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and noncondensable gas transport.

3.2.2 RELAP5-3D© Quality Assurance

NuScale Power procured RELAP5-3D© v.4.1.3 from the Idaho National Laboratory through a commercial-grade dedication process that complies with NQA-1-2008 and NQA-1a-

2009 requirements. The commercial-grade dedication evaluation determined that verification of certain of the critical characteristics required testing. Eleven test cases were identified for verification, figures of merit, and acceptance criteria. Included were models of the NPM along with NuScale proprietary test programs, legacy tests, and special feature tests. These cases constitute the matrix for commercial-grade dedication acceptance testing as discussed by the LOCA evaluation model report (Reference 7.2.1), Section 6.1.2.

RELAP5-3D© v.4.1.3 was then placed under the NuScale quality assurance program as NRELAP5 Version 0.0. Subsequent NRELAP5 versions were developed and placed under the NuScale Quality Assurance Program including the technical code revisions listed in Table 3-3 along with code corrections and administrative code revisions.

3.2.2.1 NRELAP5

NRELAP5 is NuScale's proprietary system thermal-hydraulic computer code for use in engineering design and analysis. NRELAP5 was developed at NuScale, using RELAP5-3D© v.4.1.3 as the initial baseline. Chapter 6 of the LOCA Evaluation Model (Ref: 7.2.1) is a summary of the RELAP5-3D© code and the revisions incorporated by NuScale to produce the NRELAP5 code used in both the LOCA Evaluation Model and the Non-LOCA Evaluation Model (Ref: 7.2.2). The new models in NRELAP5 are listed in Table 3-3 along with the application in the containment response analysis methodology.

Table 3-1 New NRELAP5 models

New Model	**Application in Containment Response Analysis Methodology**
Condensation heat transfer • {{ }}2(a),(c)	Used for condensation heat transfer on the CNV inside diameter and inside the decay heat removal system (DHRS) heat exchanger tubes
Critical flow • Moody critical flow model for two-phase flow conditions	Used for two-phase saturated critical flow
Helical coil SG component • Heat transfer correlation • Friction correlation	Used for modeling the helical coil SGs
Pool heat transfer • Churchill-Chu natural convection correlation correction to use bulk fluid properties • Cooper pool boiling correlation • Rohsenow pool boiling correlation	Churchill-Chu is used for modeling the CNV outside diameter (OD), the reactor pressure vessel (RPV) outside diameter, and outside the DHRS heat exchanger tubes (vertical surfaces only).

New Model	Application in Containment Response Analysis Methodology
Interfacial drag multiplier • Input multiplier added to allow minimizing liquid entrainment in break and valve flow	Used in containment response analysis methodology to evaluate effect of liquid entrainment on break and valve flow
Void drift velocity • Kataoka-Ishii alternative formulation set to default	Used for two-phase flow
Critical Heat Flux • Reyes correlation • Electric Power Research Institute correlation with counter current flow limitation or Groenveld as interpolation point for zero flow • Chang correlation • 2006 Groenveld tables • Extended Hench-Levy correlation	The 2006 Groenveld tables are used in the containment response analysis methodology. CHF does not occur for all LOCA and non-LOCA scenarios in the containment response analysis methodology.
Dynamic gap conductance • Dynamic gap conductance model with optional pellet axis offset capability	Not used in the containment response analysis methodology
Boric acid solubiity • Compare boric acid concentration to solubility limit	Not used in the containment response analysis methodology
Decay heat • 1971 ANS Standard including actinides	Not used in the containment response analysis methodology

3.2.3 NRELAP5 Simulation Models

This section presents the NRELAP5 simulation models of the NPM that are used for the containment response analysis methodology. The NRELAP5 models developed for the LOCA and non-LOCA evaluation models are used to develop the primary system (LOCA and valve opening events) and secondary system (MSLB and FWLB events) M&E release and containment response models, respectively. Substantive changes to the NRELAP5 LOCA model are limited to those necessary for containment response analysis applications. Other changes are made to the LOCA model that do not significantly impact primary system release event containment analysis results (see Section 3.2.4.1). Changes to the NRELAP5 non-LOCA model are limited to those necessary for containment response analysis applications.

3.2.3.1 NRELAP5 LOCA Evaluation Model

The NRELAP5 loss-of-coolant accident model input file is developed from engineering drawings, calculations, and reference documents. These sources of information provide the numerical information necessary to develop a complete thermal-hydraulic simulation model of the NuScale SMR per the input file specification. The types of required information fall into the following NRELAP5 input categories:

- Thermal-hydraulic fluid volumes and connecting heat structures
  - reactor vessel primary loop
    - lower plenum
    - core
    - riser
    - pressurizer
    - SG primary side
    - downcomer
  - reactor kinetics
  - reactor vessel secondary system
    - SG secondary
    - main steam piping
    - feedwater piping
  - CNV
  - reactor pool
  - DHRS
  - ECCS
  - Chemical and volume comtrol system (CVCS) piping for RCS injection, discharge and pressurizer spray supply
- Material properties
- Control systems
  - normal control systems
    - pressurizer pressure
    - pressurizer level
    - T_{avg}
    - steam pressure
    - turbine load
  - reactor protection system
  - engineered safety feature controls

The NRELAP5 NuScale Power Module model from which LOCA runs are initiated is described in the LOCA Evaluation Model in detail (Reference 7.2.1, Section 5.3) and is summarized in this report. The objectives of the NRELAP5 loss-of-coolant accident model are to analyze the LOCA break spectrum for the NPM and to demonstrate compliance with 10 CFR 50 Appendix K.

Figure 3-1 is a simplified diagram of the nodalization selected to enable modeling of the phenomena that were determined to be important for the spectrum of LOCA scenarios. The LOCA primary system release scenarios start with the blowdown of the primary inventory through the pipe break into the CNV. The reactor trips on high CNV pressure, which causes a turbine trip along with main steam isolation and feedwater isolation. The primary system depressurizes as the CNV pressurizes, and the coolant inventory accumulates in the CNV. Steam released into the CNV condenses on the CNV inner surface that is cooled by conduction and convection to the reactor pool. When the primary system inventory reaches the low level setpoint, or the CNV level reaches the high level setpoint, and the pressure drop across the ECCS valves is less than the inadvertent actuation block (IAB) release pressure, the ECCS valves open. Opening of the reactor vent valves (RVVs) increases the primary depressurization rate and completes equalization of primary and secondary pressures. Opening of the RRVs establishes a flowpath for the inventory in the CNV to flow by gravity into the RPV for core cooling. The flowpaths through the break plus the RVV, and the flowpath through the RRV provide abundant core cooling that is sufficient to keep the core covered by a two-phase mixture that prevents any heatup of the fuel rod cladding.

The NRELAP5 loss-of-coolant accident model includes the following additions to obtain a conservative LOCA analysis that meets the Appendix K requirements:

- conservative initial conditions at 102 percent of rated power level
- with or without loss of normal alternating current (AC) power
- high core power peaking factors
- break junction modeling for the various break locations
- Moody critical flow option
- ANS 1973 decay heat standard with 1.2 factor and actinides
- limiting single failure assumption
- ECCS actuation with conservative performance
- conservative CNV modeling
- conservative reactor pool modeling
- conservative setpoints and actuation delays

The LOCA evaluation model nodalization and each of these conservative LOCA modeling elements are evaluated in Section 3.2.4.1 for use in the primary system release event containment response analysis methodology. The adequacy of the NRELAP5 code and the LOCA model for modeling the primary system M&E scenarios is addressed in Sections 4.1 and 4.2.

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}}2(a),(c)

Figure 3-1 NRELAP5 NuScale Power Module noding diagram

3.2.3.2 NRELAP5 Non-Loss-of-Coolant Accident Evaluation Models

The NRELAP5 non-LOCA models are summarized in this section. The objectives of the NRELAP5 non-LOCA models are to analyze the spectrum of non-LOCA transients and accidents for the NuScale SMR, and to demonstrate compliance with the regulatory acceptance criteria.

3.2.3.2.1 **Inadvertent Operation of Emergency Core Cooling System**

The inadvertent operation of ECCS events include the inadvertent opening of an RVV or an RRV. Both events involve an initial primary system M&E release through the inadvertently opened valve into the CNV, and a subsequent actuation of the remaining ECCS valves that results in a second M&E release into the CNV. The FSAR Section 15.6.6 describes the methodology for analyzing these events and is the starting point for developing the valve opening event models in the primary system containment response analysis methodology.

3.2.3.2.2 Secondary System Pipe Breaks

The NRELAP5 non-LOCA model is the starting point for developing the MSLB and FWLB models in the containment response analysis methodology. Figure 3-2 shows the non-LOCA NRELAP5 nodalization diagram.

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}}2(a),(c)

Figure 3-2 NRELAP5 nodalization for non-loss-of-coolant accident evaluation model

The FSAR Chapter 15 MSLB and FWLB scenarios start with the blowdown of the secondary inventory through the pipe break and into the CNV. The reactor trips on high CNV pressure, and that causes a turbine trip along with main steam isolation and feedwater isolation. One SG depressurizes as the CNV pressurizes, and an equilibrium is approached. The DHRS actuates and transfers decay heat to the reactor pool. Steam released into the CNV condenses on the CNV inner surface that is cooled by conduction and convection to the reactor pool.

The safety concern for the FSAR Chapter 15 main steam line break scenario is the module response to the resulting overcooling, and the key boundary condition for the main steam line large-break scenario is the feedwater supplied to the affected SG. A single failure of the FWIV on the affected SG results in a continuation of feedwater flow until a delayed isolation occurs on feedwater regulating valve (FWRV) closure. The MSLB inside containment analysis includes the following modeling considerations:

- break modeling with {{ }}2(a),(c)
- reactor trip on high CNV pressure
- main steam isolation valves (MSIVs) actuation
- feedwater isolation and regulating valves actuation
- feedwater pump dynamic response
- feedwater pipe inventory flashing
- DHRS actuation
- with or without loss of normal AC and direct current (DC) electrical power
- limiting single failure

The differences in the NRELAP5 MSLB modeling for the containment reponse analysis methodology that focus on a conservative analysis of the CNV peak pressure and temperature response are detailed in Section 3.2.4.2.

The safety concern for the FSAR Chapter 15 FWLB scenario is the module response to the overheating caused by a loss of the SG heat sink and the resulting primary system and secondary system pressurization. The key boundary conditions are the DHRS performance, which limits the peak secondary pressure, and the reactor safety valve (RSV) capacity, which limits the peak primary pressure.

A single failure of the MSIV on the intact SG results in a small decrease in secondary inventory during the transition to DHRS operation, and a conservative minimum secondary heat sink. The FSAR Chapter 15 FWLB inside containment analysis includes the following modeling considerations:

- break modeling with {{ }}2(a),(c)

- reactor trip on high CNV pressure
- MSIVs actuation
- FWIVs actuation
- feedwater pump dynamic response
- feedwater pipe inventory flashing
- DHRS actuation
- with or without loss of normal AC and direct current (DC) electrical power
- limiting single failure

The differences in the NRELAP5 feedwater line break modeling for the containment reponse analysis methodology that focus on a conservative analysis of the CNV peak pressure and temperature response are detailed in Section 3.4.2. The adequacy of the NRELAP5 code and the non-LOCA models for evaluation of the secondary system release scenarios is addressed in Sections 4.1 and 4.2.

3.2.4 Containment Reponse Analysis Base Model Development

3.2.4.1 NRELAP5 Primary Release Event Analysis Model

Overview

The NRELAP5 model used to model NPM performance for primary system (LOCA and ECCS valve opening) release event analyses is similar to the model used in the LOCA evaluation model described in Section 3.2.3.1. The following substantive differences are related to the objective of determining the maximum containment peak pressure and peak temperature scenarios. This is accomplished by conservatively maximizing the M&E release and minimizing containment heat removal. Figure 3-3 is an illustration of the NPM during power operation that shows the main design features. Figure 3-4 illustrates the ECCS mode of operation and shows the RVVs and RRVs along with the CNV and reactor pool that provide containment heat removal and ultimate heat sink. The nodalization diagram in Figure 3-1 plus the changes described in this section constitute the NRELAP5 model used to simulate primary release scenarios resulting from bounding breaks and valve opening events. The following modification is included in the primary release event containment response analysis model:

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}}2(a),(c)

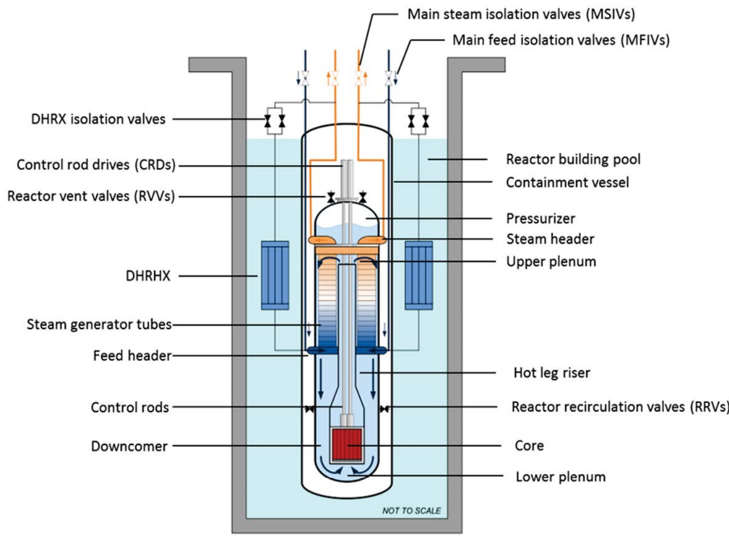

Figure 3-3 NuScale module during power operation

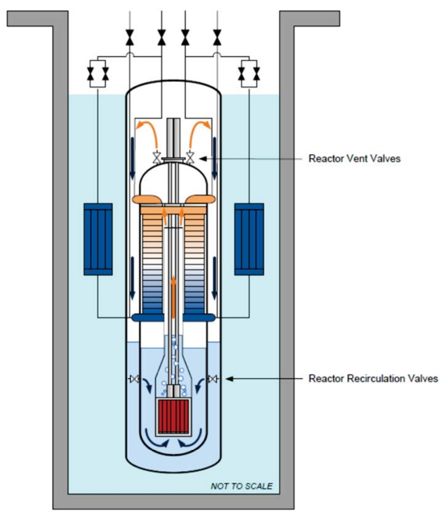

Figure 3-4 NuScale module during emergency core cooling system operation

LOCA Pipe Break and Valve Opening Modeling

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}}2(a),(c)

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}}2(a),(c)

Figure 3-5 NRELAP5 nodalization for reactor coolant system discharge line break loss-of-coolant accident

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}}2(a),(c)

Figure 3-6 NRELAP5 nodalization for reactor coolant system injection line break loss-of-coolant accident

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}}2(a),(c)

Figure 3-7 NRELAP5 nodalization for pressurizer spray supply line break and RPV high point vent degasification line loss-of-coolant accident

Conservative modeling of the LOCA pipe break spectrum and the valve opening events to ensure a bounding M&E release includes the following elements:

- all break locations are considered

- maximum credible break size at each location
- critical flow with discharge coefficient of 1.0
- saturated liquid - Moody critical flow
- subcooled liquid – {{ }}2(a),(c)
- modified pressure volume work term
- maximum RRV and RVV flow areas
- liquid entrainment evaluated by use of interfacial drag multiplier in upper riser, riser upper plenum, pressurizer baffle, pressurizer, and downcomer

Containment Vessel and Reactor Pool Models

The CNV nodalization in the NRELAP5 loss-of-coolant accident and valve opening event containment response analysis model (Figure 3-6, Component 500) is modified compared to the LOCA evaluation model. The following substantive changes were made to the LOCA evaluation model to maximize M&E release and consequential containment pressure and temperature.

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}}2(a),(c)

In addition to the five substantive modifications described above, the following non-subtantive modifications are to the LOCA evaluation model:

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}}2(a),(c)

A sensitivity analysis demonstrates that the above modifications to the LOCA evaluation model have a negligible effect on CNV analysis results.

The CNV is maintained at a partial vacuum with an assumed high initial pressure (e.g. 2.0 psia), so the effects of non-condensable gases are small but are modeled. Heat transfer to the CNV vessel inside diameter (ID) is initially by radiative heating from the RPV, and that modeling sets the initial CNV wall temperature distribution. The LOCA or valve opening event M&E release into the CNV results in a rapid heating and pressurization of the CNV. The steam is condensed on the CNV inside diameter and the condensate film flows downward and forms a pool in the bottom of the CNV. As the CNV pool level rises boiling occurs on the RPV surface.

Heat transfer from the CNV outside diameter to the reactor pool initially maintains the vessel at a low temperature except for the upper section of the vessel that is above the pool surface elevation. Following the LOCA or valve opening event, the condensing of steam and convection from the CNV pool increases the vessel temperature, and heat transfer from the CNV outside diameter to the reactor pool increases. Heat transfer on the CNV outside diameter is by pool convection and pool nucleate boiling, except for the upper section that is not submerged in the reactor pool. In the upper section heat transfer is by convection to air (See Figure 3-8).

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}}2(a),(c)

Figure 3-8 NRELAP5 reactor pool model

Conservative modeling of the heat transfer to and from the CNV inside diameter, and from the CNV outside diameter to the reactor pool, to ensure a bounding peak CNV pressure and temperature response following a LOCA or valve opening event, includes the following elements:

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}}2(a),(c)

Table 3-2 shows the heat transfer correlations and models for all of the processes that could impact the CNV peak pressure and temperature response. These correlations and models, along with their applications, are described in greater detail in the LOCA Evaluation Model Report (Reference 7.2.1).

Table 3-2 Containment vessel and reactor pool heat transfer modeling

Heat Transfer Process	**Correlation/Model**
Radiant heating from RPV outside diameter to CNV inside diameter	Radiation enclosure model
Convection from RPV outside diameter to CNV pool	<u>Vertical Sufaces</u> {{ }}2(a),(c) <u>Non-Vertical Surfaces</u> {{ }}2(a),(c)
Condensation on CNV inside diameter	{{ }}2(a),(c)
Interphase heat transfer	Default model based on flow regimes
Convection from CNV outside diameter to reactor pool	<u>Vertical Sufaces</u> {{ }}2(a),(c) <u>Non-Vertical Surfaces</u> {{ }}2(a),(c)
Reactor pool mixing	No mixing is modeled
Reactor pool cooling to ambient	Assumed adiabatic
Reactor pool mixing with other modules	No mixing with other modules is modeled

3.2.4.2 NRELAP 5 Secondary System Break Analysis Model

Overview

The NRELAP5 model used for secondary system pipe break analysis in the containment response analysis methodology is similar to the NRELAP5 model used in the non-LOCA accident FSAR Chapter 15 methodology (Section 3.2.3.2). The differences are related to the objective of determining the maximum containment peak pressure and peak temperature scenarios. This is accomplished by conservatively maximizing the M&E release, and minimizing containment heat removal.

The following changes were made to the non-LOCA evaluation model to maximize M&E release and consequential containment pressure and temperature.

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}}2(a),(c)

Figure 3-3 is an illustration of the NuScale Power Module during power operation that shows the main design features including the DHRS that actuates for secondary line breaks. For some secondary line break scenarios actuation of the DHRS results in a slow cooldown of the primary system and an eventual opening of the ECCS valves and a second M&E release, when a loss of power to the ECCS valve actuator solenoid occurs. Additions and modifications to this model for the secondary system M&E release analysis are the feedwater system model, the pipe break model, the CNV and the reactor pool model. These modifications to the model are described below.

Feedwater System Model

The feedwater system is an important boundary condition for the secondary system M&E release analyses. The initial secondary inventory in the helical coil SG is small and does not by itself cause a significant CNV pressurization following a secondary line break. The main source of mass is the feedwater system due to an assumed single failure of the FWIV on the affected helical coil SG. Also, the feedwater pump is assumed to respond to the decrease in helical coil SG pressure by a corresponding increase in feedwater flow. Feedwater flow continues to supply the affected helical coil SG until the FWRV automatically closes to back up the FWIV.

Secondary Pipe Break Model

The secondary pipe break spectrum modeling in the containment response analysis methodology is the same as in the Non-LOCA Methodology, with the limiting break size being the double-ended break. Figure 3-9 shows the NRELAP5 model of the MSLB. The break is modeled by closing the normal flow path (Valve 910) and by opening two break junctions (Valves 911 and 912) that start the break flow to the CNV at the appropriate elevations. Figure 3-10 depicts the NRELAP5 model of the FWLB. The break is modeled by closing the normal flow path (Valve 913) and by opening two break junctions (Valves 914 and 915) that start the break flow to the CNV at the appropriate elevations. Main steam isolation valve closure isolates the unaffected SG from the affected SG. A single failure of one MSIV to close is addressed by automatic closure of the secondary MSIV on each steam line.

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}}2(a),(c)

Figure 3-9 Main steam line break model

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}}2(a),(c)

Figure 3-10 Feedwater line break model

Conservative modeling of the secondary pipe breaks to ensure a bounding M&E release includes the following elements:

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}}2(a),(c)

- {{

}}2(a),(c)

Containment Vessel and Reactor Pool Models

The CNV and reactor pool models for the MSLB and FWLB containment response analysis methodology are the same as the modeling for LOCA. Refer to Section 3.2.4.1.

3.3 Containment Response Analysis Methodology for Primary System Release Events

Section 3.3 presents the details of the containment response analysis methodology for primary system releases resulting from primary system breaks and valve opening events. The NRELAP5 computer code described in Section 3.2.2.1 and the LOCA containment response analysis model described in Section 3.2.4.1 are applied using the methodology in this section to meet the NRC regulations and regulatory guidance in Section 2.0.

3.3.1 Primary System Mass and Energy Release Methodology

3.3.1.1 Loss-of-Coolant Accident Scenario Phenomena Identification and Ranking Table Results

NuScale has performed and documented a PIRT for the LOCA scenarios resulting from primary system breaks and ECCS valve opening events. Loss-of-Coolant Accident Evaluation Model Report (Reference 7.2.1), Chapter 4.0, summarizes the LOCA phenomena identification and ranking table. The results of the LOCA phenomena identification and ranking table were used in the development of the NRELAP5 code, the NRELAP5 LOCA model, and the LOCA evaluation model.

The results of the LOCA scenario PIRT are directly applicable to the primary system M&E release and resultant CNV pressure and temperature response that are the focus of the containment response methodology. The basis for this statement is that “CNV pressure and temperature” is a figure-of-merit in the LOCA phenomena identification and ranking table. Therefore, the LOCA scenario PIRT is also considered to be the LOCA containment response analysis methodology PIRT.

3.3.1.2 Module Response

The typical response of the NPM to a primary system M&E release is characterized by a simultaneous depressurization of the primary system and pressurization of the CNV. The module response depends on the size of the break or valve opening, the location of the release as that determines if the release is steam or liquid or two-phase, and the timing of the M&E releases. The resulting high containment pressure signal causes an immediate actuation of the following safety features:

- containment isolation, including
  - closure of MSIVs
  - closure of FWIVs
  - closure of backup MSIVs (non-safety)
  - closure of FWRVs (non-safety)
- reactor trip
- turbine trip
- DHRS actuation

Any steam that is released through the break or valve condenses on the cold inner surface of the CNV. Condensate and any unflashed break liquid accumulates into a pool on the bottom of the CNV. The primary system level decreases due to the break or valve flow. The ECCS actuates on the following conditions:

- low RPV level
- high CNV level
- loss of normal AC power and the highly reliable DC power system

The following design criteria govern RVVs and RRVs opening:

- If the pressure differential across the valves is greater than the IAB threshold when the ECCS signal actuates, then the valves stay closed until the pressure differential decreases to below the IAB release pressure
- If the pressure differential across the valves has decreased to below the IAB threshold pressure when the ECCS signal actuates, then the valves open and the IAB release pressure is not used

Opening of the RVVs increases the depressurization rate, and the primary system and CNV pressures approach equalization. As the pressures equalize, the break/valve flow decreases. With pressure equalization and the increase in the CNV pool level, flow through the RRVs into the reactor vessel starts to provide long-term core cooling via recirculation. This terminates the reactor vessel level decrease prior to core uncovery. Heat transfer to the CNV wall and to the reactor pool eventually exceeds the energy addition from the break flow and the RVV flow. When this occurs the period of peak

containment pressure and temperature have been completed, and a gradual depressurization and cooling phase begins.

3.3.1.3 Event Scenarios and Break Spectrum

The postulated primary system M&E release events include the following pipe break accidents and valve actuations. For the valve opening events, the specific FSAR events that result in actuation of that valve are listed.

- Pipe breaks (LOCAs)
  - FSAR 15.6.5 - RCS discharge line break LOCA {{ }}2(a),(c)
  - FSAR 15.6.5 - RCS injection line break LOCA {{ }}2(a),(c)
  - FSAR 15.6.5 - Pressurizer spray supply line break LOCA {{ }}2(a),(c)
  - FSAR 15.6.5 – RPV high point degasification line LOCA {{ }}2(a),(c)
- RSV actuation {{ }}2(a),(c)
  - FSAR 15.6.1 – Inadvertent RSV opening
- RVV actuation ({{ }}2(a),(c))
  - FSAR 15.6.6 – Inadvertent RVV opening
- RRV actuation {{ }}2(a),(c)
  - FSAR 15.6.6 – Inadvertent RRV opening

The RPV high point degasification line, the pressurizer spray supply line, and the RSVs are all located near the top of the RPV. A LOCA in the RPV high point degasification line is the largest break size in this location and is analyzed in the containment response analysis methodology. The other two are non-limiting and are not analyzed.

One RVV or one RRV can open as an initiating event due to an assumed mechanical failure. The RVVs and RRVs all open following ECCS signal actuation and when the IAB design criteria discussed in Section 3.3.1.2 are met .

The RPV high point degasification line break LOCA differs in that the break flow will be steam. The RCS break locations differ in that the discharge line connects to the downcomer, and the injection line connects to the riser. These three break locations plus the valve opening event locations fulfill the adequacy of the break spectrum with regard to location.

The adequacy of the break spectrum with regard to break size is important in the timing of the ECCS valve opening, as the second M&E release resulting from the opening of the three RVVs is the dominant event for CNV pressure and temperature response. First, the maximum break size at each location is analyzed to ensure the maximum initial M&E release rate into the CNV during the first phase of CNV pressurization. Then, the sensitivity of the opening time of the three RVVs is addressed by analysis of a range of IAB release pressures for each break location. In this manner a lower IAB release pressure results in

a delay in the RVV opening time. This is similar to a break size sensitivity because a range of break sizes would result in a range of depressurization rates and RVV opening times. However, by using the maximum break size for all cases the maximum initial M&E release rate is used for all cases. This approach fulfills the adequacy of the break spectrum with regard to break size.

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}}2(a),(c)

In summary, the limiting postulated primary system M&E release scenarios consist of an initiating anticipated operational occurrence or accident, which may include a pipe break or RVV or RRV valve opening, with a resultant an ECCS actuation signal causing all RVVs and RRVs to open after the IAB design criteria discussed in Section 3.3.1.2 are met. Table 3-3 shows the primary system M&E release scenarios that are used to determine the limiting cases.

Table 3-3 Primary system mass and energy release scenarios

Iniating Event	**Subsequent RVV and RRV Actuations on ECCS**	**Analysis Case**
LOCA in RCS discharge line from downcomer	Three RVVs and two RRVs actuate	1
LOCA in RCS injection line from riser	Three RVVs and two RRVs actuate	2
LOCA in RPV High Point Degasification Line near top of vessel	Three RVVs and two RRVs actuate	3
RVV opens due to a mechanical failure	Two RVVs and two RRVs actuate	4

Iniating Event	Subsequent RVV and RRV Actuations on ECCS	Analysis Case
RRV opens due to a mechanical failure	Three RVVs and one RRV actuate	5

3.3.1.4 Identification of Bounding Events

The bounding events for peak CNV pressure and for peak CNV temperature are identified by analyzing the spectrum of scenarios in Table 3-3 with conservative initial conditions and boundary conditions. Sensitivity studies are used to determine the bounding conditions and assumptions for the limiting cases. This is further discussed in Section 5.1.1.

3.4 Secondary System Containment Response Analysis Methodology

Section 3.4 presents the details of the containment response analysis methodology for the secondary system pipe break accidents. The NRELAP5 computer code described in Section 3.2.2.1 and the secondary system containment response analysis model described in Section 3.2.4.2 are applied using the methodology in this section to meet the NRC regulations and regulatory guidance discussed in Section 2.0. The methodology for the main MSLB and the FWLB accident analyses is presented.

3.4.1 Steamline Break Mass and Energy Release Methodology

3.4.1.1 Non-Loss-of-Coolant Accident Event Phenomena Identification and Ranking Table Results

NuScale has performed and documented a PIRT for the non-LOCA events. The results of the non-LOCA phenomena identification and ranking table are summarized in the non-LOCA evaluation model (Ref: 7.2.2). The results of the non-LOCA phenomena identification and ranking table are directly applicable to the secondary system M&E release and CNV pressure and temperature response that are the focus of the containment response analysis methodology. The basis for this statement is that {{

}}2(a),(c) Therefore the non-LOCA phenomena identification and ranking table is also considered to be the secondary system containment response analysis PIRT.

3.4.1.2 Module Response

The NPM initially responds to a MSLB inside the CNV with a simultaneous depressurization of the secondary system and a pressurization of the CNV. Feedwater flow out the break increases due to the decrease in backpressure and due to flashing of the feedwater pipe inventory. The resulting high containment pressure signal causes an immediate actuation of the following safety features:

- containment isolation including
  - closure of primary main steam isolation valves

- closure of FWIVs
- closure of backup main steam isolation valves (non-safety)
- closure of FWRVs (non-safety)

- reactor trip
- DHRS actuation
- turbine trip

As the secondary system depressurizes, the feedwater pump flowrate increases in response to the decrease in SG pressure. Closure of the MSIVs separates the affected SG from the unaffected SG, thereby reducing the mass and energy release. Actuation of the DHRS on the unaffected SG establishes long-term decay heat removal. Closure of the FWIVs terminates the supply of secondary inventory and the affected SG boils dry. The initial primary system transient is a moderate overcooling event that does not result in ECCS actuation. Steam that is released through the break condenses on the cold inner surface of the CNV. Condensate accumulates into a pool on the bottom of the CNV. As the break flow decreases, the CNV pressure and temperature decrease and the time period of the peak values is completed. The peak pressure and temperature are significantly less than for a LOCA due to the smaller secondary inventory that is released prior to feedwater isolation.

The typical MSLB scenario is more severe when a single failure is considered. The limiting single failure is a failure of the FWIV to close on the affected SG. Closure of the FWRV is credited in this scenario, but the much longer stroke time results in a higher CNV peak pressure and temperature. Isolation of the feedwater ends the mass and energy release, and the CNV pressure and temperature then decrease due to heat transfer to the reactor pool through the CNV. When this occurs the period of peak containment pressure and temperature have been completed, and a gradual depressurization and cooling phase begins via the DHRS.

The above MSLB scenario is made more adverse by assuming a loss of normal AC and DC power (concurrent with the break), which results in an ECCS signal. Subsequent primary system depressurization resulting from heat transfer via the DHRS along with a loss of power to the pressurizer heaters leads to ECCS actuation when the pressure differential decreases to below the IAB release pressure. Opening of the RVVs results in a second M&E release from the primary system, and the peak CNV pressure and temperature from this second release may be higher than the initial peak from the secondary system M&E release.

3.4.1.3 Limiting Event Description

The limiting MSLB event is a double-ended rupture of the largest main steam line (12 in. Schedule 120 / 10.75 in. ID), which is a break area of 0.6303 ft^2. Both SGs blow down into the CNV until the MSIVs close. After the initiation of the break there are two potential limiting events depending on the evolution of the scenario with continued normal AC power, or following a loss of normal AC and DC power.

For the scenario with continued normal AC power, the affected SG continues to blow down until feedwater is isolated including a single failure of the FWIV on the affected SG. This results in an extended period of feedwater delivery until the FWRV closes. The availability of power to the pressurizer heaters maintains primary system pressure and there is no ECCS actuation. The peak CNV pressure and temperature occurs as a result of the blowdown of the affected SG, and then the event is terminated.

For the scenario with a loss of normal AC and DC power concurrent with the break, the feedwater pump stops and the delivery of feedwater to the affected SG is less than the case with continued normal AC power. The loss of normal AC and DC power causes an ECCS actuation signal and a loss of power to the pressurizer heaters. With DHRS actuation the primary system begins a gradual cooldown and depressurization. The IAB prevents the ECCS valves from opening until the pressure differential decreases to below the IAB release pressure. Opening of the RVVs initiates a primary system M&E release with the CNV pre-heated and pressurized from the initial MSLB M&E release. This second M&E release has the potential to produce the peak CNV pressure and wall temperature results. Continued heat transfer through the CNV wall to the reactor pool results in a gradual cooldown and depressurization.

Analysis of the two above scenarios has determined that the case with continued normal AC power results in the peak CNV pressure and peak CNV temperature results.

3.4.2 Feedwater Line Break Mass and Energy Methodology

3.4.2.1 Module Response

The NPM initially responds to an FWLB inside the CNV with a reduction in the secondary heat sink due to the loss of feedwater flow, a depressurization of the affected SG as it blows down, and a pressurization of the CNV. Feedwater flow out the break increases due to the decrease in backpressure and due to flashing of the feedwater pipe inventory. The resulting high containment pressure signal causes an immediate actuation of the following safety features:

- containment isolation including
  - closure of primary main steam isolation valves
  - closure of FWIVs
  - closure of backup main steam isolation valves (non-safety)
  - closure of FWRVs (non-safety)
- reactor trip
- DHRS actuation
- turbine trip

Closure of the MSIVs separates the affected SG from the unaffected SG, thereby reducing the mass and energy release. Actuation of the DHRS on the unaffected SG establishes long-term decay heat removal. Closure of the FWIVs terminates the supply of feedwater

to the break, and the affected SG dries out and ends the secondary mass and energy release. The primary system transient is initially a moderate overheating event that is stabilized by DHRS heat transfer, and does not result in ECCS actuation. Any steam that is released through the break condenses on the cold inner surface of the CNV. Condensate accumulates along with unflashed break liquid into a pool on the bottom of the CNV. As the break flow decreases, the CNV pressure and temperature decrease and the time period of the peak values is completed.

The typical FWLB scenario is potentially more severe when a single failure is considered. The postulated single failures are a failure of the FWIV to close, or a failure of the MSIV to close, on the affected SG. Closure of the nonsafety-related FWRV, or closure of the non-safety secondary MSIV to close, is credited in this scenario, but the longer stroke times result in a higher CNV peak pressure and temperature. Isolation of the feedwater ends the secondary system mass and energy release, and the CNV pressure and temperature then decrease due to heat transfer to the reactor pool through the CNV and via the DHRS. When this occurs the period of peak containment pressure and temperature have been completed, and a gradual depressurization and cooling phase begins.

The above FWLB scenario is made more adverse by assuming a loss of normal AC and DC power concurrent with turbine trip that results in an ECCS actuation signal. The loss of pressurizer heaters causes a gradual primary system depressurization during the DHRS cooldown, and subsequent opening of the RVVs when the pressure differential decreases to the IAB release pressure. Opening of the RVVs initiates a second M&E release.

3.4.2.2 Limiting Event Description

The limiting FWLB event is a double-ended rupture of the largest feedwater pipe (5 in. Schedule 120 / 4.563 in. ID), which is a break area of 0.1136 ft^2. The affected SG and its feedwater pipe blow down into the CNV. The unaffected SG responds to the depressurization of the affected SG until the MSIV closes. The feedwater piping on the affected SG then continues to blow down until feedwater is isolated by FWIV closure.

A single failure of the FWIV to close on the affected SG is mitigated by closure of the FWRV. The limiting case also assumes a loss of normal AC and DC power at time of turbine trip, and that results in ECCS signal actuation and a loss of power to the pressurizer heaters. With DHRS actuation the primary system begins a gradual cooldown and depressurization. The IAB prevents the ECCS valves from opening until the pressure differential eventually decreases to below the IAB release pressure. Opening of the RVVs combines a subsequent primary system M&E release with the initial feedwater line break M&E release and results in a significantly more severe CNV pressure and temperature response.

Analysis of the above scenarios has determined that the case with loss of normal AC and DC power and ECCS actuation results in the peak CNV pressure and peak CNV temperature. A single failure of the FWIV on the affected steam generatior to close, and minimum initial primary system pressure, are included in the limiting case based on sensitivity analysis results.

3.5 Initial and Boundary Conditions

3.5.1 Primary System Release Event Initial Conditions

Initial conditions for the spectrum of primary system release containment response analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the initial conditions is consistent with the guidance in DSRS Section 6.2.1.3. The selection process ensures that energy sources are maximized and energy sinks are minimized. Table 3-4 presents the primary system initial conditions for the primary system release containment response analyses.

Table 3-4 Primary system initial conditions

Parameter	Conservative containment response analysis methodology Initial Condition	Rationale
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		}}2(a),(c)

The initial conditions in the secondary system, in particular {{

}}2(a),(c) The SG initial conditions result from the NRELAP5 initialization process and are consistent with the conservative primary system initial conditions.

The initial conditions for the CNV and the reactor pool are shown in Table 3-5. These initial conditions ensure that the CNV heat sink is minimized so that the peak containment pressure and temperature are modeled conservatively.

Table 3-5 Containment vessel and reactor pool initial conditions

Parameter	Initial Condition Assumption	Rationale
{{		
		}}2(a),(c)

3.5.2 Primary System Release Event Boundary Conditions

Boundary conditions for the spectrum of primary system M&E release analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the boundary conditions is consistent with the guidance in DSRS Section 6.2.1.3. The selection process ensures that energy sources are maximized and energy sinks are minimized. Due to the simplicity of the NPM design there are few postulated single failures for the primary system M&E release scenarios. Failure of ECCS valves to open would obviously reduce the M&E release and are not limiting. Failures of MSIVs or FWIVs to close are analyzed as sensitivity studies, but they have minimal effect on the CNV pressure and temperature response as the secondary system response is not

important. Table 3-6 presents the boundary conditions for the LOCA containment response analyses.

Table 3-6 Primary system boundary conditions

Parameter	Boundary Condition Assumption	Rationale
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		}}2(a),(c)

Parameter	Boundary Condition Assumption	Rationale
{{		
		}}2(a),(c)

Parameter	Boundary Condition Assumption	Rationale
{{		}}2(a),(c)

3.5.3 Main Steam Line Break Initial Conditions

Initial conditions for the MSLB containment response analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the initial conditions is consistent with the guidance in DSRS Section 6.2.1.4. The selection process ensures that energy sources are maximized and energy sinks are minimized. Table 3-4 presents the primary system initial conditions used for primary system release containment response analyses. {{

}}2(a),(c) Table 3-5 presents the CNV and reactor pool initial conditions used by the LOCA containment response analyses that are also used by the MSLB containment response analyses. Table 3-7 presents the secondary system initial conditions used by the MSLB containment response analyses.

Table 3-7 Secondary system initial conditions

Parameter	Initial Condition Assumption	Rationale
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		}}2(a),(c)

3.5.4 Main Steam Line Break Boundary Conditions

Boundary conditions for the MSLB mass and energy release analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the boundary conditions is consistent with the guidance in DSRS Section 6.2, and specifically DSRS Section 6.2.1.4. The selection process ensures that energy sources are maximized and energy sinks are minimized.

The largest break size is assumed to maximize the secondary system M&E release rate into the CNV and thereby maximize the resulting CNV pressurization and temperature increase. However, a subsequent primary system M&E release following ECCS actuation and delayed opening of the three RVVs may result in the peak CNV pressure and temperature response for some scenarios. Also, opening of the RVVs depends on the IAB

design criteria in Section 3.3.1.2 being satisfied, and that may not occur until the DHRS has been operating for some period of time. As the DHRS cools the primary system, a delayed M&E release through the RVVs will be smaller, and the second CNV pressurization will be lower. Furthermore, the steam line break CNV pressure and temperature response remains bounded by the LOCA. Therefore, the maximum MSLB size is bounding and a break spectrum analysis is not necessary.

Due to the simplicity of the NPM design, there are few postulated single failures for the secondary system M&E release scenarios. Failure of ECCS valves to open would obviously reduce the M&E release and are not analyzed. Failures of MSIVs or FWIVs to close are analyzed as sensitivity studies. Table 3-6 presented the boundary conditions for the primary system containment response analysis methodology, and they are the same for the MSLB containment response analysis methodology except for those presented in Table 3-8.

Table 3-8 Boundary conditions for the main steam line break containment response analysis methodology

Parameter	**Boundary Condition Assumption**	**Rationale**
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		}}2(a),(c)

Parameter	Boundary Condition Assumption	Rationale
{{		
		}}2(a),(c)

3.5.5 Feedwater Line Break Initial Conditions

Initial conditions for the FWLB mass and energy release analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the initial conditions is consistent with the guidance in DSRS Section 6.2, and DSRS Section 6.2.1.4 specifically. The selection process ensures that energy sources are maximized and energy sinks are minimized. Table 3-4 presents the primary system initial conditions used by the LOCA containment response analyses. {{

}}2(a),(c) Table 3-5 presents the CNV and reactor pool initial conditions used by the LOCA containment response analyses, and these initial conditions are also used by the FWLB containment response analyses. Table 3-7 presents the secondary system initial conditions used by the MSLB containment response analyses, and these initial conditions are also used by the FWLB containment response analyses.

3.5.6 Feedwater Line Break Boundary Conditions

Boundary conditions for the FWLB mass and energy release analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the boundary conditions is consistent with the guidance in DSRS Section 6.2, and specifically DSRS Section 6.2.1.4. The selection process ensures that energy sources are maximized and energy sinks are minimized. Section 3.4.4 and Table 3-8 presented the boundary conditions used by the MSLB containment response analyses, these boundary conditions are also used by the FWLB containment response analyses, with the exception of the single failure evaluation that is discussed below.

The largest break size is assumed to maximize the initial M&E release into the CNV. However, it is the subsequent second M&E release following ECCS actuation and opening of the three RVVs that results in the peak CNV pressure and temperature response. Also, opening of the RVVs depends on the pressure differential decreasing to below the IAB release pressure, and that may not occur until DHRS has been operating for some period of time. Therefore, the initial break size is unimportant as the secondary M&E release is similar, and the sequence of events leading to the opening of the RVVs is similar. Furthermore, the feedwater line break CNV pressure and temperature response is bounded by the LOCA. Therefore, a break spectrum analysis is not necessary.

Due to the simplicity of the NPM design, there are few postulated single failures for the secondary system M&E release scenarios. Failure of ECCS valves to open would obviously reduce the M&E release and are not analyzed. Failures of a MSIV or a FWIV to close are analyzed as sensitivity studies, with the FWIV failure identified as limiting.

4.0 Qualification and Assessment

4.1 Assessment of Methodology and Data

4.1.1 Primary System Release Effects Code and Model Qualification

The NRELAP5 code has been qualified or assessed to the separate effects and integral effects tests as described by LOCA Evaluation Model Report (Reference 7.2.1), Chapter 7.0 to demonstrate the capability to simulate LOCAs in the NPM. The results of the NRELAP5 comparisons to data establish the capability of the code to model the NPM design for the LOCA analysis. The most important assessment activities were those comparing to integral LOCA tests conducted in the NIST-1 facility.

The following two key known scaling distortions are relevant to the scope of the containment response analysis methodology:

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  }}2(a),(c)

Neither of the above phenomena have an impact on the peak CNV pressure. The first distortion is addressed by the containment response analysis methodology by closure of the MSIVs. The second distortion is addressed by the overall conservative modeling of CNV heat transfer in the containment response analysis methodology, which includes use of conservative initial conditions and boundary conditions that are discussed in Section 3.4.

The LOCA Evaluation Model Report (Reference 7.2.1, Section 8.2) also presents the evaluation of the adequacy of the NRELAP5 code and LOCA Evaluation Model for modeling LOCAs in the NPM. The following action was identified as needed to address adequacy issues relative to the containment response analysis methodology:

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  }}2(a),(c)

No additional qualification activities were performed for the LOCA containment response analysis methodology as the LOCA evaluation model qualification activities addressed in LOCA Evaluation Model Report (Ref: 7.2.1) are adequate.

4.1.2 Secondary System Pipe Break Effects Code and Model Qualification

The LOCA-event code and model qualification as described in Section 4.1.1, which credits the LOCA Evaluation Model qualification activities, is generally applicable to the secondary system M&E release events.

Additional NRELAP5 code and model qualification activities were included in the Non-LOCA Evaluation Model (Reference 7.2.2) with the focus being DHRS and SG heat transfer as they are of greater importance during non-LOCA events. The following NIST-1 facility and testing distortions are applicable to secondary M&E release containment analysis methodology:

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}}2(a),(c)

The secondary system M&E releases consist of the MSLB and the FWLB events, and both of these events involve asymmetric responses in the two SGs. Following break initiation the affected SG blows down into the CNV until the feedwater supply has been isolated. The unaffected SG is isolated from the affected SG following closure of the MSIVs, and then provides decay heat removal via the DHRS. A second M&E release for these events occurs for cases that include ECCS actuation on loss of normal AC and DC power coincident with the pipe break. The primary pressure gradually decreases during the DHRS cooldown phase, and the ECCS valves open when the differential pressure decreases to below the IAB release pressure.

The NRELAP5 code and the containment response analysis model for the NPM are fully capable of modeling the secondary system M&E releases without directly applicable NIST-1 test data. The large body of NIST-1 separate effects and LOCA integral tests have demonstrated the capability of NRELAP5 to adequately model the NPM design. There are no additional phenomena associated with secondary M&E releases, and no additional qualification activities were performed for the secondary containment response analysis methodology.

Additional justification for the above position is that the secondary system M&E release analyses for the NPM demonstrate that they are non-limiting compared to the primary system containment response analyses. This justification is further supported by the overall conservatism in the containment response analysis methodology.

4.2 Testing Results

4.2.1 NuScale Integral System Test Facility Testing

A scaled facility of the NPM was constructed at Oregon State University, referred to as the NuScale Integral System Test Facility-1, or NIST-1, facility, to assist in validation of the NRELAP5 system thermal-hydraulic code. The facility is designed to perform various tests, including LOCA tests. A detailed description of NIST-1, the NRELAP5 model of the facility, and the NRELAP5 validation testing is provided in Reference 7.2.1, Section 7.5.

The NRELAP5 predictions of CNV pressure, level and temperature documented in Reference 7.2.1 show good fidelity to NIST-1 experimental measurements as follows.

The CNV level and pressure response is predicted with reasonable to excellent agreement to RCS discharge line break experimental measurements as discussed by Reference 7.2.1, Section 7.5.6.

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}}2(a),(c)

The CNV pressure response is predicted with reasonable to excellent agreement to spurious RVV opening experimental data as discussed by Reference 7.2.1, Section 7.5.8.

A separate high pressure condensation test described by Reference 7.2.1, Section 7.5.4 demonstrates that NRELAP5 has the capability to predict condensation rates for various pressures with reasonable to excellent agreement to experimental data.

5.0 Results

5.1 Primary System Release Scenario Containment Response Analysis

This section presents the results of the NRELAP5 limiting analyses of the spectrum of primary system M&E release scenarios for the NPM, listed in Table 3-3, and secondary system break scenarios that are determined using the containment response analysis methodology presented earlier in this report. The case labels from Table 3-3 are used in the following discussion.

5.1.1 Analysis Approach

The approach to determine the limiting peak CNV pressure event from the the spectrum of primary mass and energy release scenarios for the NPM, listed in Table 3-3, and the limiting peak CNV temperature for each primary release event was as follows:

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}}2(a),(c)

5.1.2 Reference Analysis and Sensitivity Results

The following insights were obtained from the results of the NRELAP5 analyses of the five primary system M&E release cases and associated sensitivity studies.

- The peak CNV pressure scenario is the RRV release (Case 5). The RRV mass and energy release causes an initial heatup and pressurization of the CNV, and then ECCS actuation results in a second M&E release with all three RVVs and second RRV opening that pressurizes the CNV to the highest peak pressure.
- The peak CNV wall temperature scenario is the CVCS injection line LOCA (Case 2). The break in this location combines a high temperature liquid initial M&E release followed by a high temperature M&E release through all three RVVs following an ECCS actuation signal.
- The sensitivity parameters have only a small effect on the peak CNV pressure (~+18 psi / ~+1.9 percent) and temperature (~+5 degrees F / ~+1 percent) results of the limiting cases. No single failures had a significant impact on the results for the limiting cases. The loss of power sensitivity that results in early ECCS actuation, and the IAB release pressure sensitivity that affects the timing of the opening of the ECCS valves, were the more important sensitivity parameters.
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}}2(a),(c)

5.1.3 Primary Release Scenario Pressure and Temperature Results

The initial conditions used by NRELAP5 analyses for each of the five cases in Table 3-4 are shown in Table 5-1. The initial condition values in the second column of Table 5-1 are the nominal values plus the uncertainty or conservative allowance in parentheses. The assumed parameter values are consistent with the methodology as discussed by Section 3.5.1 and maximize heat sources while minimizing heat sinks. The decay heat conservatively used by these analyses is 120 percent of the 1979 ANS standard rather than the methodology assumption (1979 ANS standard plus 2-sigma uncertainty). The 120 percent assumption bounds the required 2-sigma uncertainty required by the containment response analysis methodology (See Table 3-6).

Table 5-1 Initial conditions for primary system release event analyses

Parameter	Conservative Containment Response Analysis Methodology Initial Condition
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	}}2(a),(c)

The results of each of the five primary containment response analysis release analysis cases are summarized in the following sections, with more detailed results and discussion provided for the limiting CNV peak pressure scenario (Case 5 - RRV loss-of-coolant accident), and for the limiting CNV peak temperature scenario (Case 2 – RCS injection line break LOCA).

5.1.3.1 Case 1: Reactor Coolant System Discharge Line Break Loss-of-Coolant Accident

The LOCA in the RCS discharge line initiates an M&E release from the downcomer into the CNV. The sequence of events is shown in Table 5-2. The CNV pressure response and temperature response are shown in Figures 5-1 and 5-2. The peak CNV pressure is 884 psia for the reference case, and 906 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, 1200 psid IAB release pressure). The peak CNV temperature is 499 degrees F for the reference case, and 502 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC power, 900 psid IAB release pressure, high-biased CNV level signal, high primary system flow, RRV single failure). Case 1 is non-limiting.

Table 5-2 Case 1 sequence of events - reactor coolant system discharge line break loss-of-coolant accident

Peak CNV Pressure Case Time (sec)	**Event**	**Peak CNV Temperature Case Time (sec)**
50	LOCA in RCS discharge line Loss of normal AC and DC power • ECCS actuation signal • Reactor trip • Containment isolation • MSIV closure • FWIV closure For peak pressure case Loss of normal AC power only for peak temperature case	Same
N/A	High CNV pressure resulting in • Containment isolation • MSIV closure • FWIV closure • Reactor trip For peak temperature case	51 - 53
151	ECCS valve opening on pressure difference below IAB release pressure for peak pressure case	N/A
N/A	ECCS actuation on high CNV level for peak temperature case	495
N/A	ECCS valve opening for peak temperature case	498
169	Peak CNV pressure (906 psia)	N/A
N/A	Peak CNV temperature (502 degrees F)	610
~2300	CNV pressure decreases to <50% of peak pressure	N/A

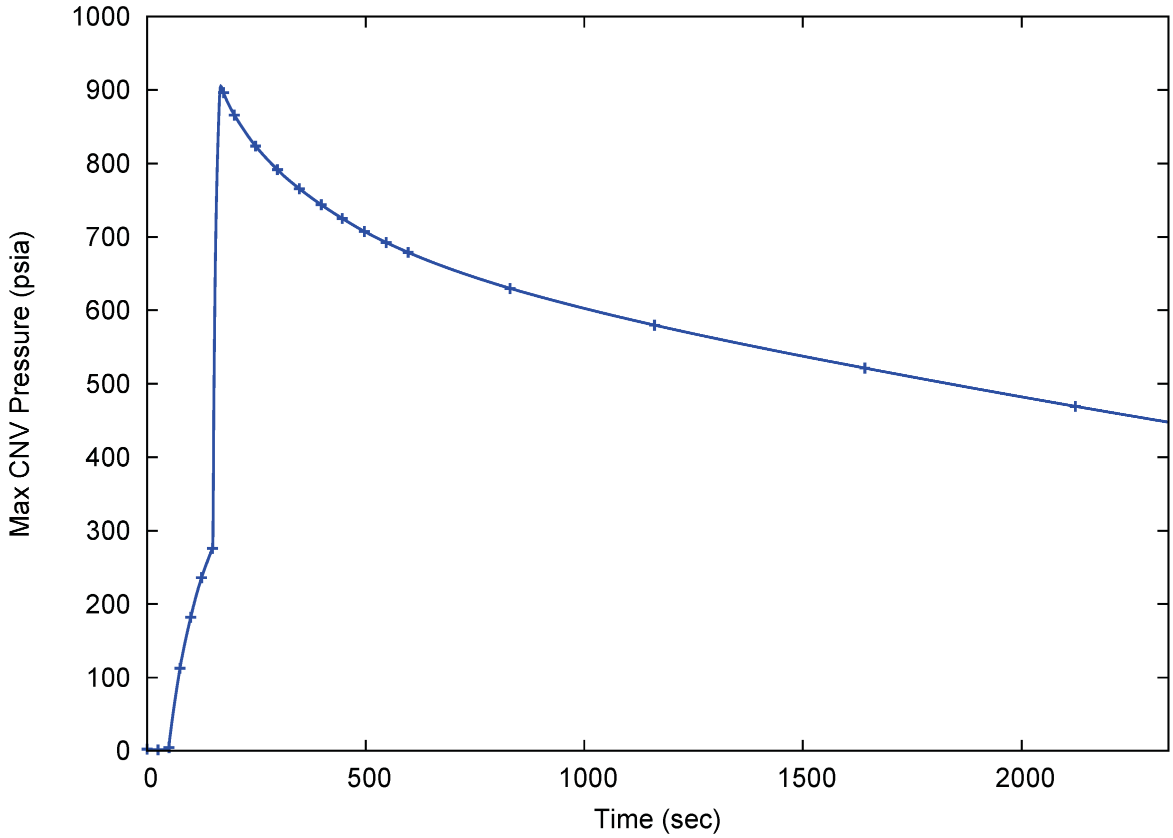

Figure 5-1 Case 1 containment vessel pressure – reactor coolant system discharge line break loss-of-coolant accident

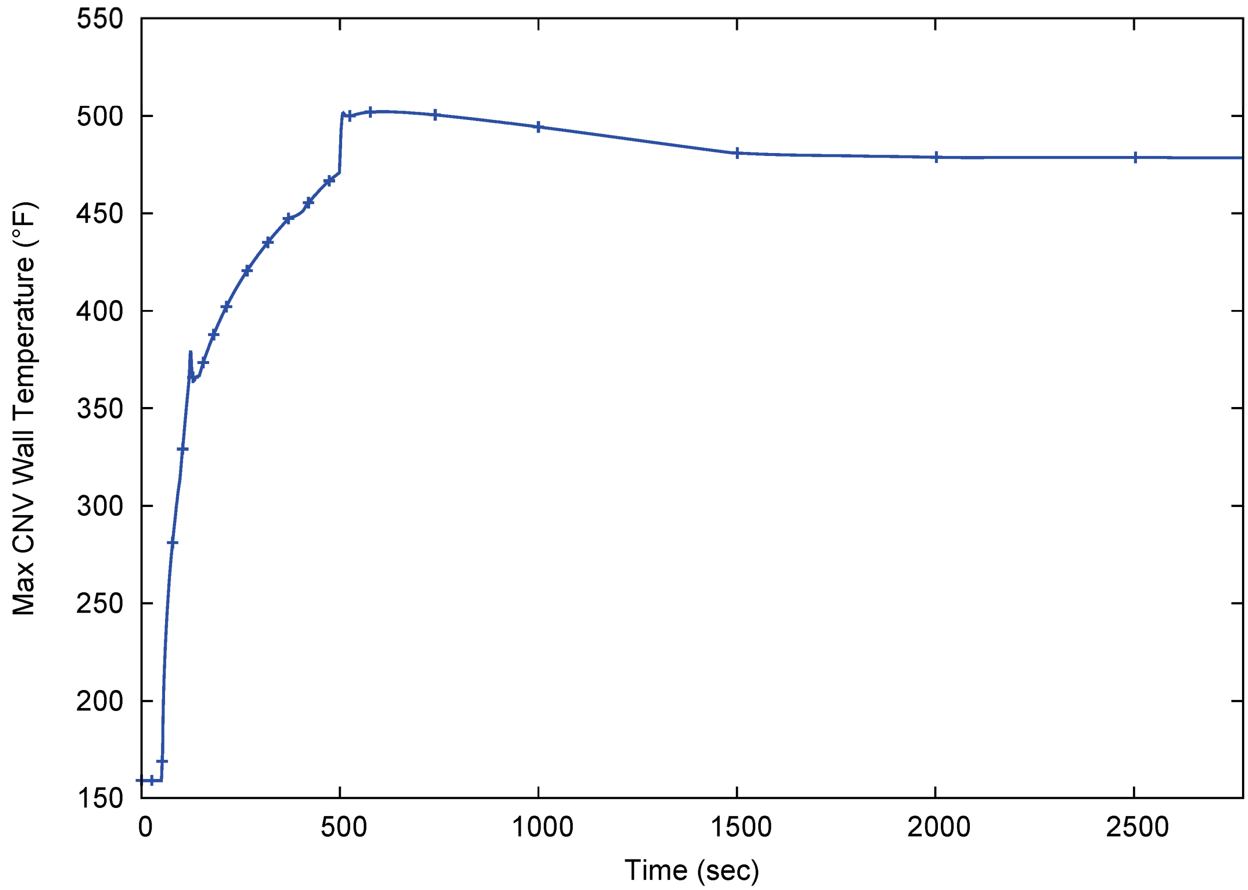

Figure 5-2 Case 1 containment vessel wall temperature - reactor coolant system discharge line break loss-of-coolant accident

5.1.3.2 Case 2: Limiting Loss-of-Coolant Event - Reactor Coolant System Injection Line Break Loss-of-Coolant Accident

The LOCA in the RCS injection line initiates an M&E release from the riser into the CNV. The results of the primary release event M&E release break spectrum analysis and sensitivity analyses have determined that Case 2 is the limiting LOCA peak pressure and overall limiting CNV wall temperature event. In addition, the analyses have shown that the Case 2 peak pressure results and CNV wall temperature results are ~1.6 and ~4.2 percent higher, respectively, than the next highest result (Case 1); therefore, there is confidence that the overall limiting break location and scenario has been identified. The sequence of events is shown in Table 5-3 and detailed results for key parameters are shown in Figures 5-3 through 5-16. The peak CNV wall temperature is 518 degrees F for the reference case, and 523 degrees F with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +5 degrees F (~1 percent) increase are: 1) the timing of ECCS valve opening as determined by the IAB setpoints and 2) the assumption of a loss of normal AC power. There was a small, adverse impact from the assumed single

failure of one RRV failing to open. The peak CNV pressure is 904 psia for the reference case, and 921 psia with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +17 psi (~2 percent) increase are: 1) the timing of ECCS valve opening as determined by the ECCS actuation setpoint, 2) the assumption of a loss of normal AC power, and 3) high primary system flow. There was a small, adverse impact from the assumed single failure of one RRV failing to open. The detailed discussion of the Case 2 results that follow are for the limiting peak CNV pressure and temperature cases.

The sequence of events (Table 5-3) show that in the first seconds following the occurrence of a LOCA in the RCS injection line many automatic responses occur to transition the module from full power operation to an alignment that mitigates the initial LOCA blowdown phase. The break flow into the CNV causes a rapid pressurization that reaches the 9.5 psia high pressure setpoint. The following automatic actions occur on high CNV pressure:

- containment isolation resulting in MSIV and FWIV closure
- reactor trip
- turbine trip
- DHRS actuation (Note: DHRS actuation is conservatively not credited in the primary system containment response analysis methodology)

As a conservative assumption a loss of normal AC power is also assumed to occur at the time of the break and the ECCS signal is actuated on high CNV level or low RPV level. In the containment response analysis methodology the ECCS setpoints are important analysis input as they determine the time of the second primary system M&E release into the CNV via the RVVs. The peak CNV pressure and peak CNV wall temperature occur following this RVV actuation, after the CNV has been preheated by the initial LOCA M&E release.

Following the alignment of the module for the LOCA blowdown phase, the primary system pressure and inventory decrease due to the loss of inventory through the LOCA. The CNV pressurizes and the steam condenses on the cold ID of the CNV. The condensate flows down the CNV walls and accumulates in a pool in the CNV lower head. The cold CNV wall absorbs the energy of the condensed steam and starts to heat up by conduction. Eventually the energy is transferred through the CNV wall to the reactor pool, and the pool temperature slowly increases. For the peak CNV wall temperature case, the ECCS signal actuates on high CNV level at 872 seconds, and the opening of the ECCS valves occurs at 1158 seconds (after the IAB release pressure is reached). The ECCS actuation and opening of the three RVVs and one RRV causes the peak CNV wall temperature to occur at 1180 seconds. For the peak pressure case, the ECCS signal actuates on high CNV level at 1078 seconds, and the opening of the ECCS valves occurs at 1081 seconds. The ECCS actuation and opening of the three RVVs and one RRV causes the peak CNV pressure to occur at 1100 seconds. Then, as flow through the RVVs dimishes, the primary and CNV pressures converge, and continued heat transfer to the CNV leads to a gradual cooldown and depressurization phase. Pressure equalization enables recirculation flow from the CNV pool through the RRVs to establish the long-term cooling recirculation alignment.

The primary system response for the RCS injection line LOCA CNV peak pressure case is shown in Figures 5-3 through 5-9. Figure 5-3 shows the primary pressure response. The initial depressurization phase due to the LOCA is followed by the rapid depressurization when the RVVs open. Figures 5-4 and 5-5 show the inventory in the pressurizer and in the riser. These figures show the expected trend of a decreasing level in the primary followed by a stabilization in inventory, with some liquid holdup in the pressurizer. A sensitivity study that decreased the interphase drag in the upper riser, riser upper plenum, pressurizer baffle, pressurizer, and the downcomer, with the intent of reducing liquid entrainment, showed that there was no adverse impact on the peak CNV pressure for this case. Figure 5-6 shows the primary coolant temperatures at six locations. Following ECCS actuation the temperatures converge and the cooldown proceeds. Figure 5-7 shows the LOCA and ECCS mass flowrates although the spike in mass release when ECCS valves open is not shown because of reduced plot frequency. Figures 5-8 and 5-9 show the integrated LOCA and ECCS mass flowrate and energy flowrate. Based on the integrated mass and energy flow rate plots, it is evident that the ECCS flow through the three RVVs into the CNV is significant. It is this M&E flow spike that causes the peak CNV pressure and wall temperatures to occur shortly thereafter as shown in Table 5-3.

The CNV and reactor pool responses for the RCS injection line LOCA peak pressure case are shown in Figures 5-10 to 5-15. Figure 5-10 shows the CNV pressure response and the limiting LOCA value of 921 psia. This NRELAP5 analysis result is approximately 8% below the CNV design pressure of 1000 psia. This is a key result of this limiting LOCA containment pressure response analysis case. Pressure increases rapidly to the peak value immediately following opening of the RVVs. Figure 5-11 shows the CNV liquid level increase as the unflashed break flow and condensed steam accumulates. Figure 5-12 shows the CNV vapor temperature. {{

}}2(a),(c) Figure 5-13 shows the temperature profile across the CNV wall at the 45 foot elevation. There is a large temperature gradient across the CNV wall. Figure 5-14 shows the reactor pool temperatures for a range of elevations. The reactor pool temperature does not increase significantly for the short duration of these analyses. From Figures 5-13 and 5-14 it is evident that the NPM design provides an effective heat sink for these short-term M&E analyses. Even with the conservative initial reactor pool level of 55 ft above the pool floor and a temperature of 140 degrees F assumed in this analysis, the peak CNV wall temperature remains within the design limit.

Figure 5-15 shows the energy balance during the CVCS injection line LOCA and the trends of the heat sources and sinks. At approximately 1400 seconds, the energy release from the LOCA and the RVV valves decreases to below the energy transferred through the CNV wall. The CNV wall then continues to provide a strong heat sink for the sustained cooldown and depressurization of the module.

As demonstrated by Table 5-3, the event progression for the RCS injection line LOCA peak pressure case and the peak CNV wall temperature case are similar. Accordingly, only the

CNV pressure and wall temperature figures will be presented for the peak CNV wall temperature case. Figure 5-16 shows the CNV pressure response for the RCS injection line LOCA peak wall temperature case. Figure 5-17 shows the CNV wall temperature response for the RCS injection line LOCA peak CNV wall temperature case and the overall limiting value of 523 degrees F. This limiting NRELAP5 is less than the CNV design temperature of 550 degrees F. This is a key result of this limiting containment wall temperature response analysis case.

Table 5-3 Case 2 sequence of events for limiting containment vessel temperature event - reactor coolant system injection line break loss-of-coolant accident

Peak CNV Pressure Case Time (sec)	**Event**	**Peak CNV Temperature Case Time (sec)**
50	LOCA in RCS injection line Loss of normal AC power	Same
52 - 55	High CNV pressure resulting in • Containment isolation • MSIV closure • FWIV closure • Reactor trip • Turbine trip	Same
1078	ECCS actuation on high CNV level	872
1081	ECCS valve opening	1158
1100	Peak CNV pressure (921 psia)	N/A
N/A	Peak CNV temperature (523 degrees F)	1180
~3000	CNV pressure decreases to <50% of peak pressure	N/A

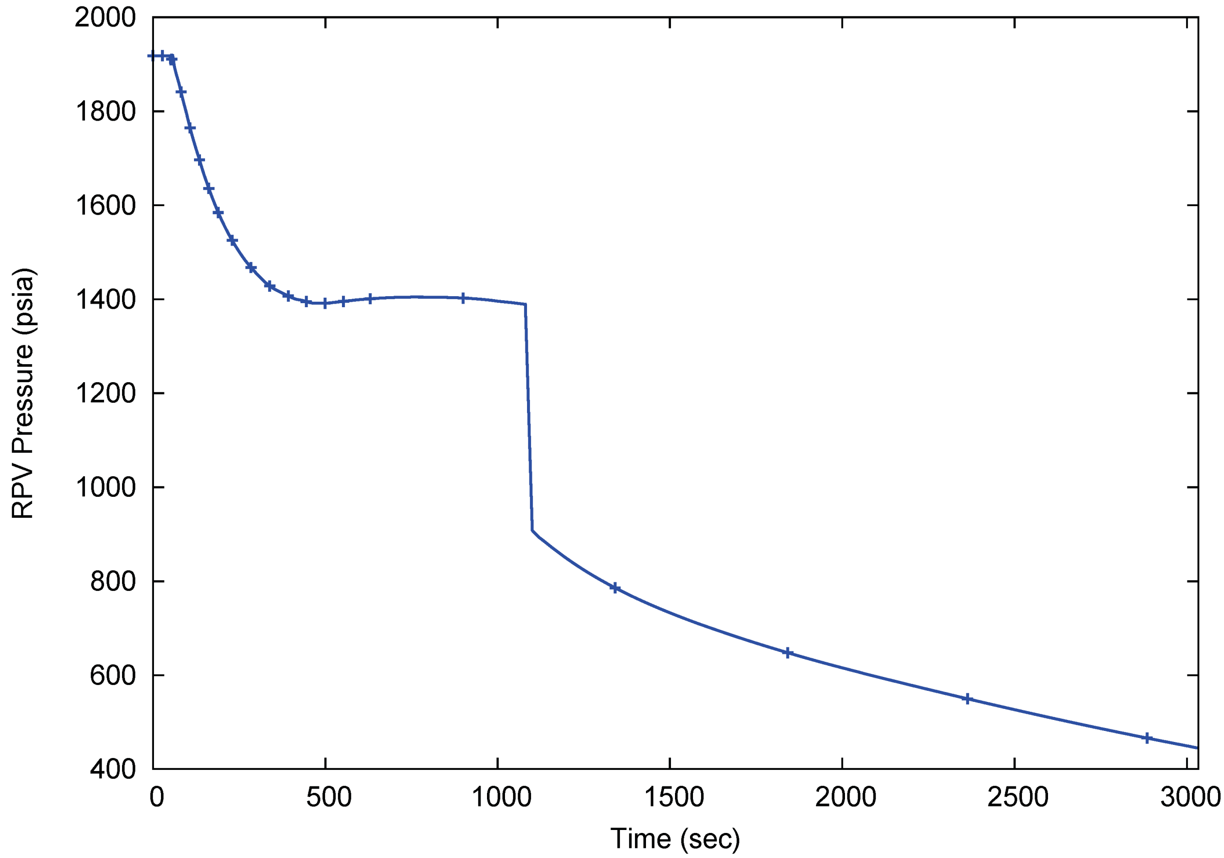

Figure 5-3 Case 2 primary system pressure – reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

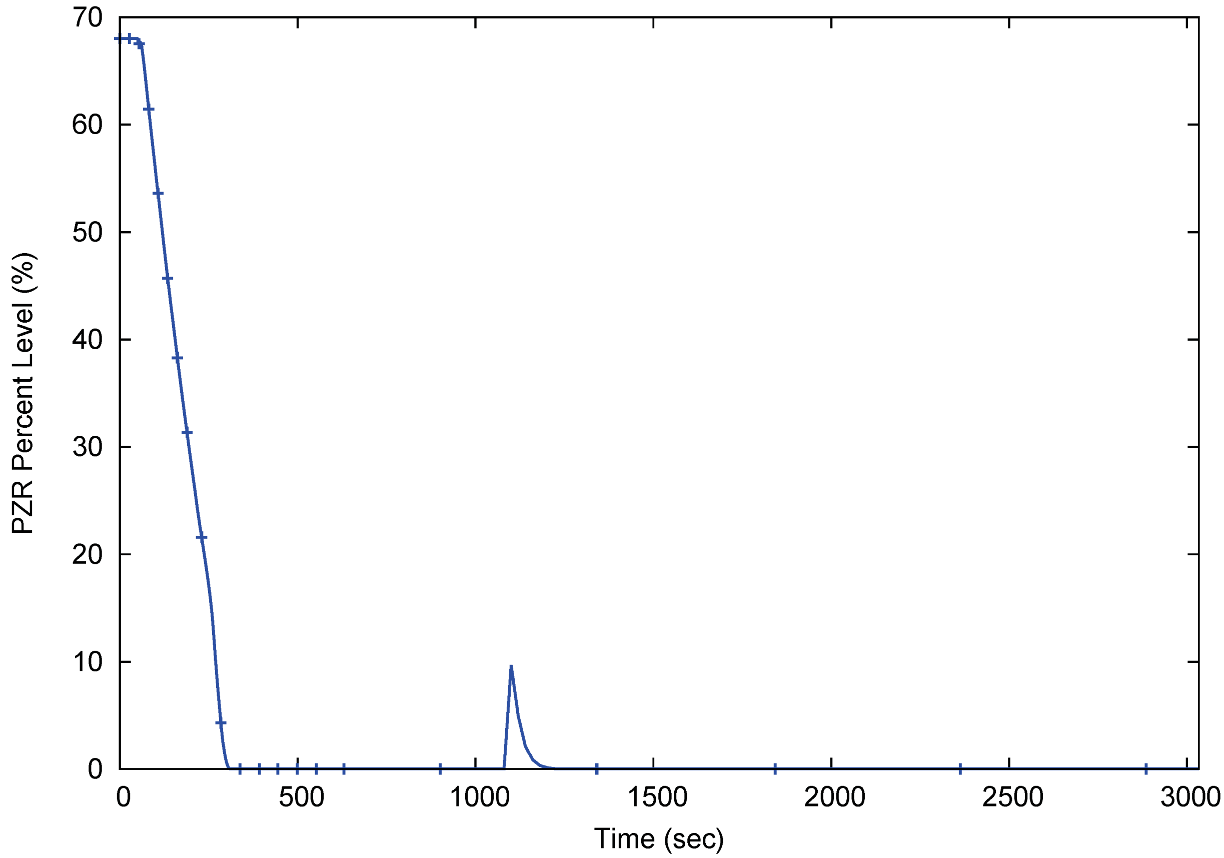

Figure 5-4 Case 2 pressurizer level - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

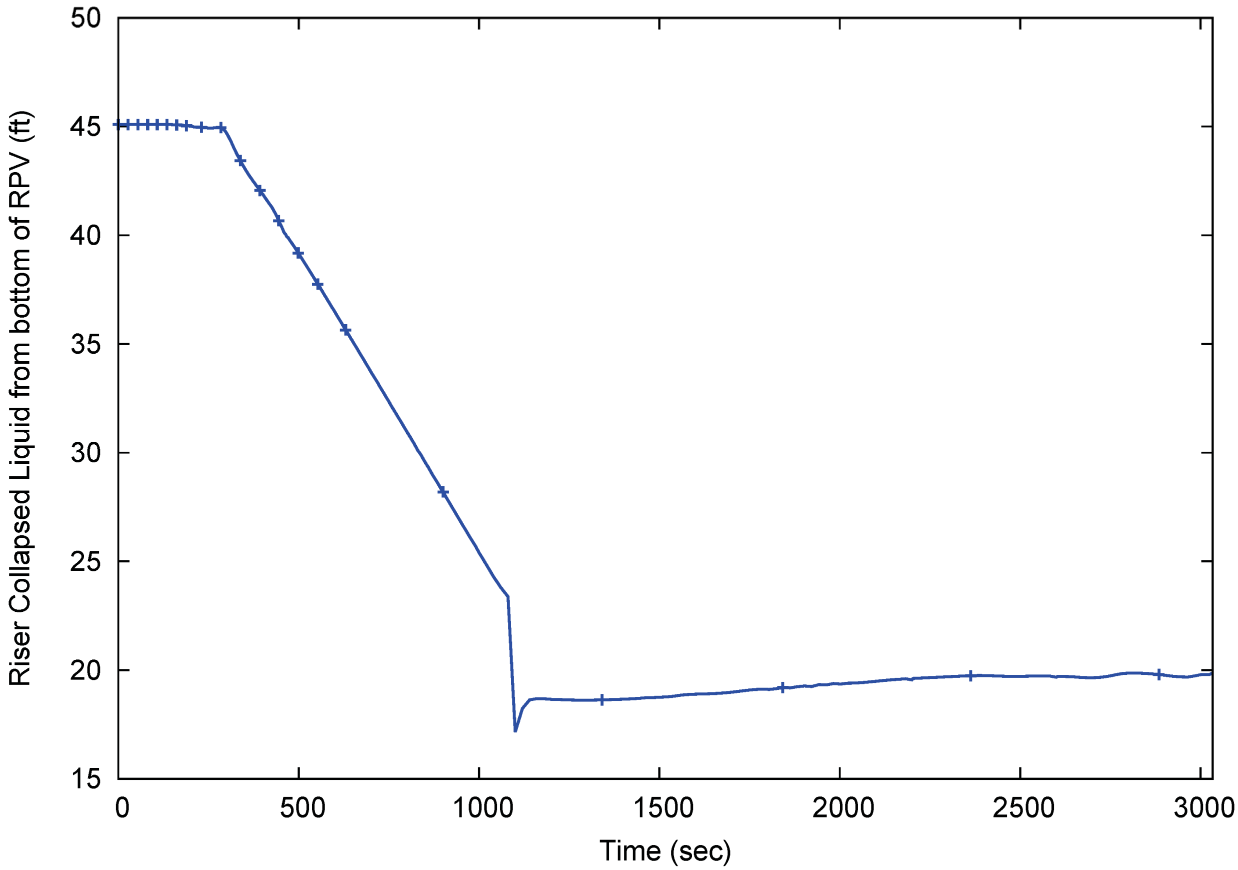

Figure 5-5 Case 2 riser level - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

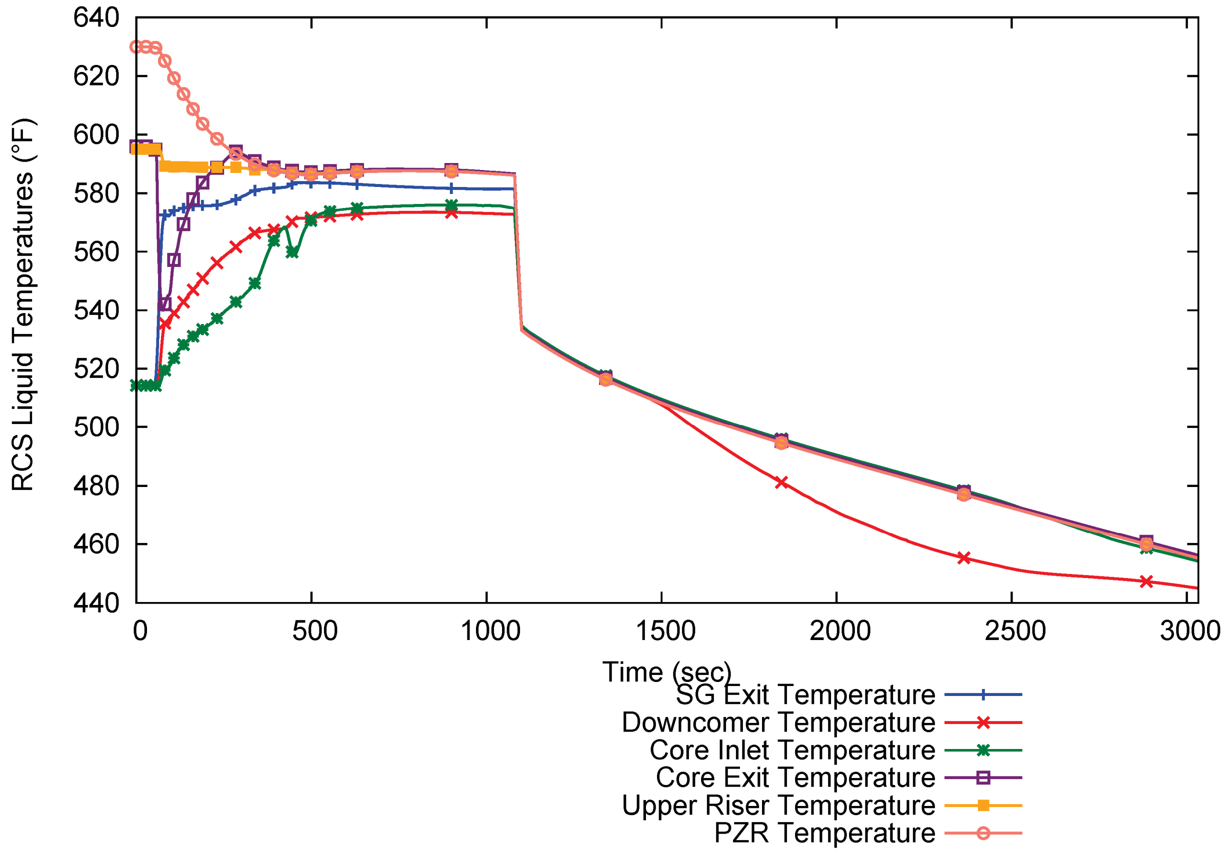

Figure 5-6 Case 2 primary temperatures - reactor coolant system injection line break loss-of-coolant accident

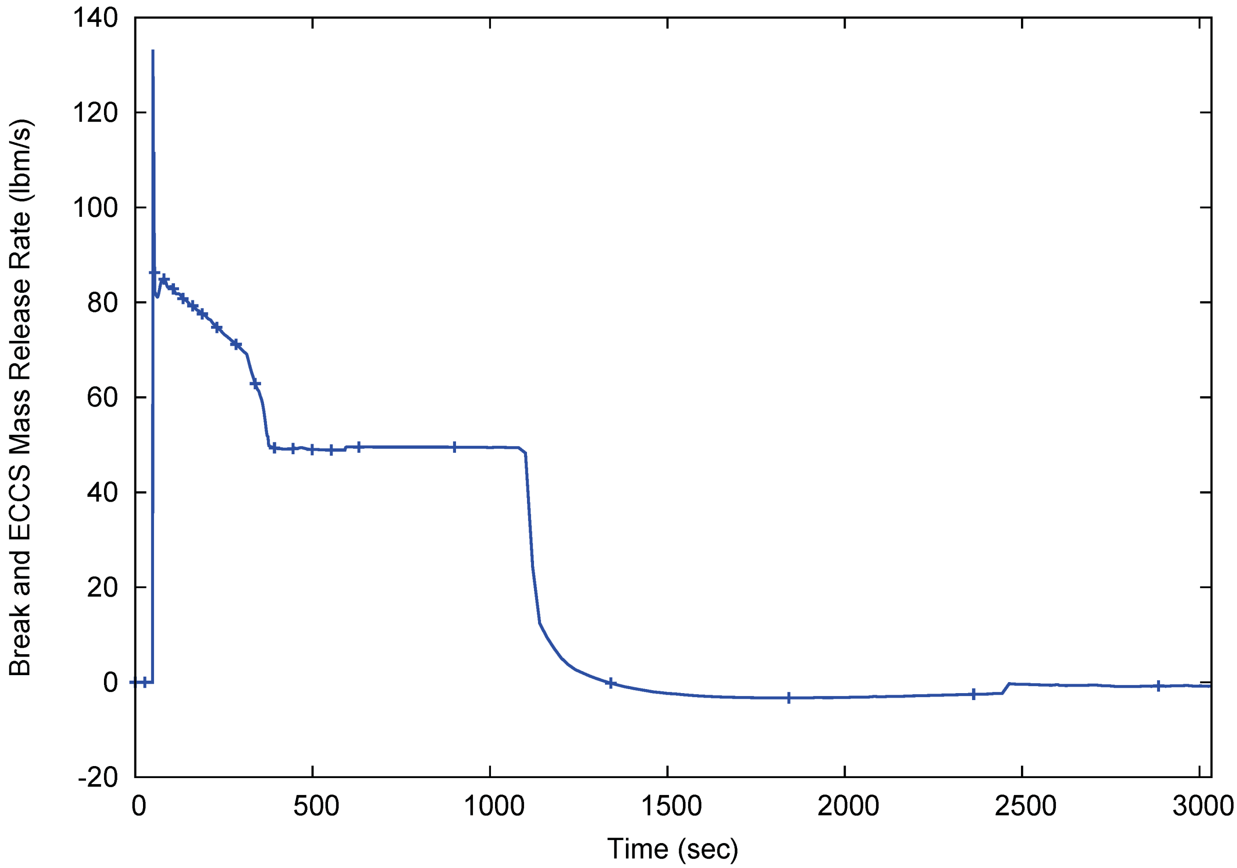

Figure 5-7 Case 2 break and emergency core cooling system mass flowrate - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

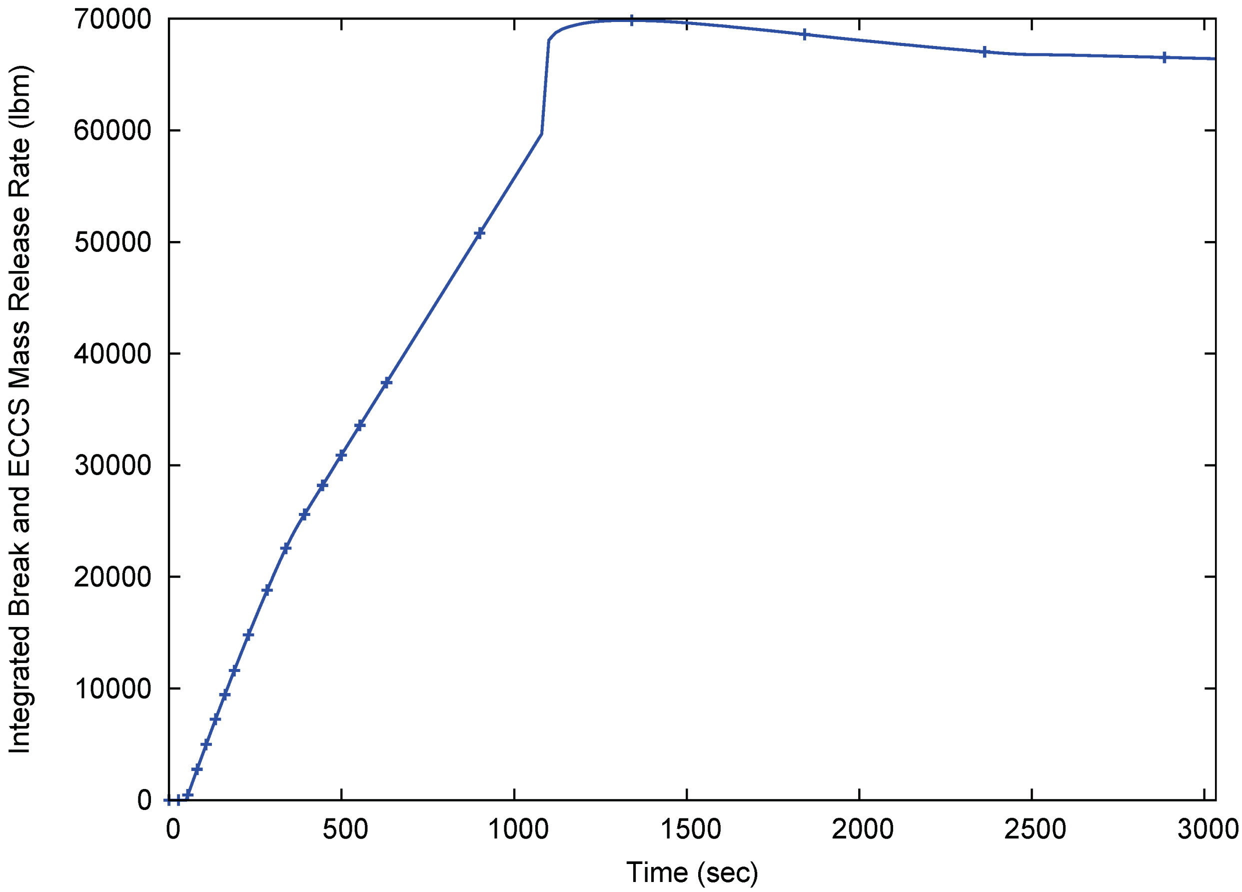

Figure 5-8 Case 2 integrated loss-of-coolant accident and emergency core cooling system mass release - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

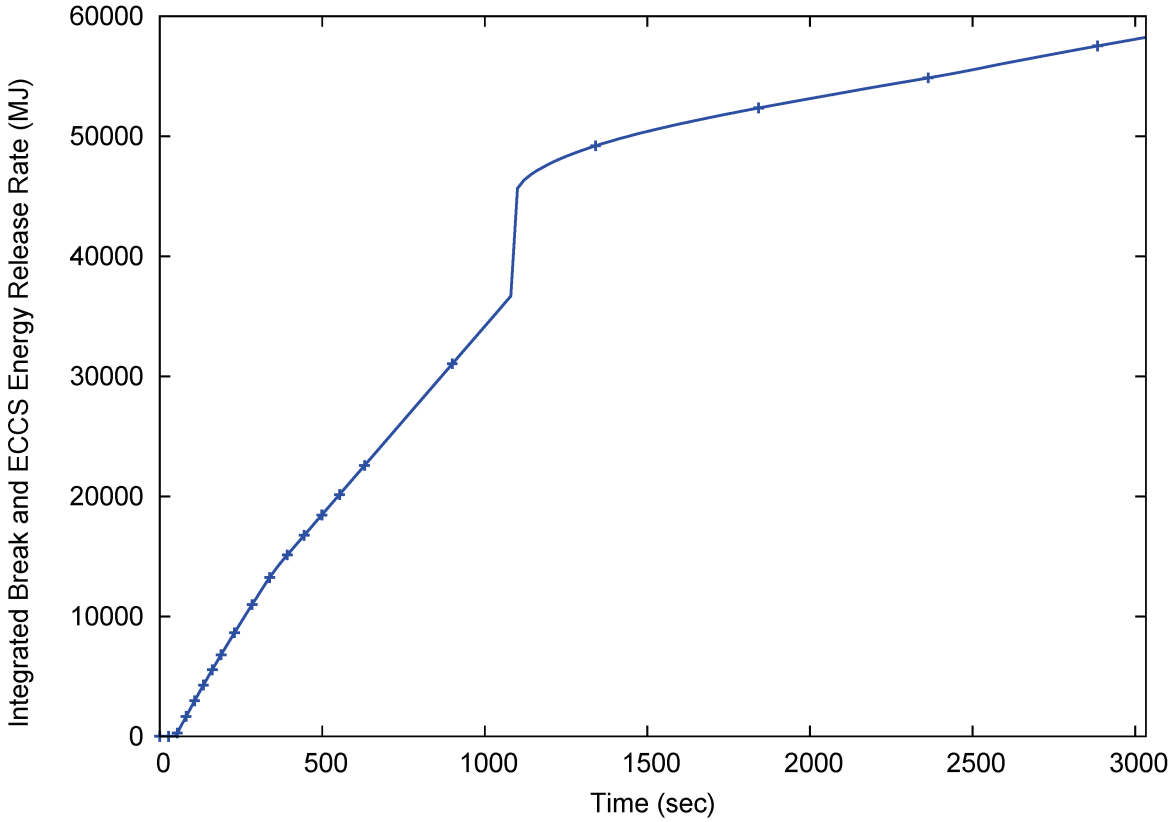

Figure 5-9 Case 2 integrated loss-of-coolant accident and emergency core cooling system energy release - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

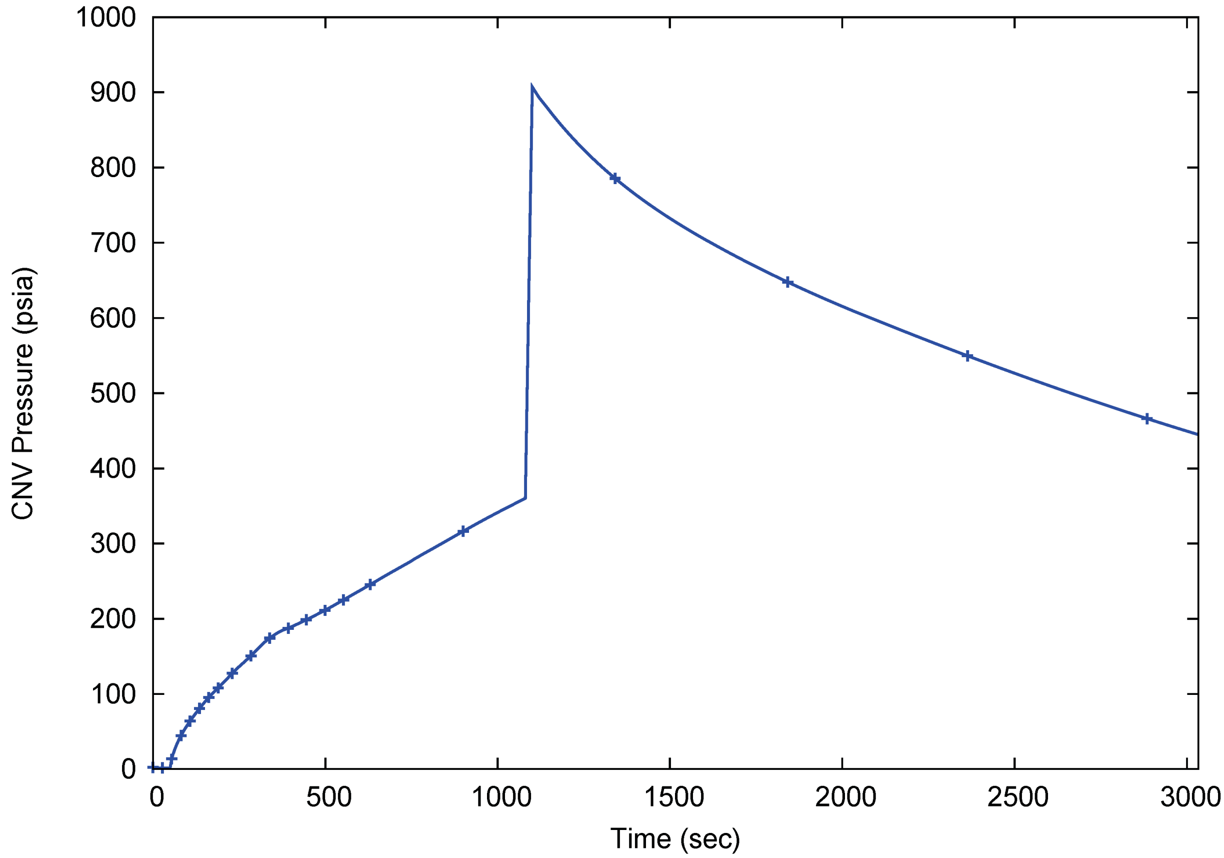

Figure 5-10 Case 2 containment vessel pressure - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

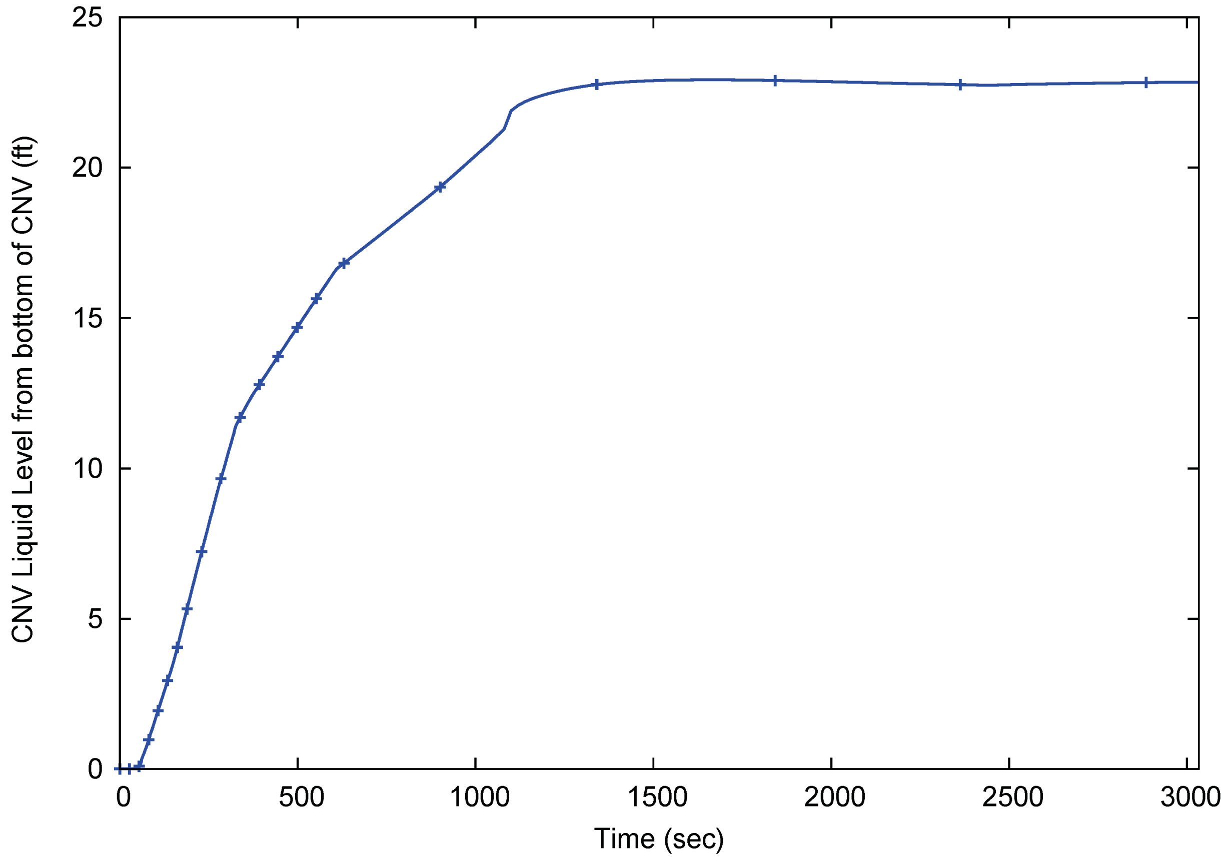

Figure 5-11 Case 2 containment vessel level - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

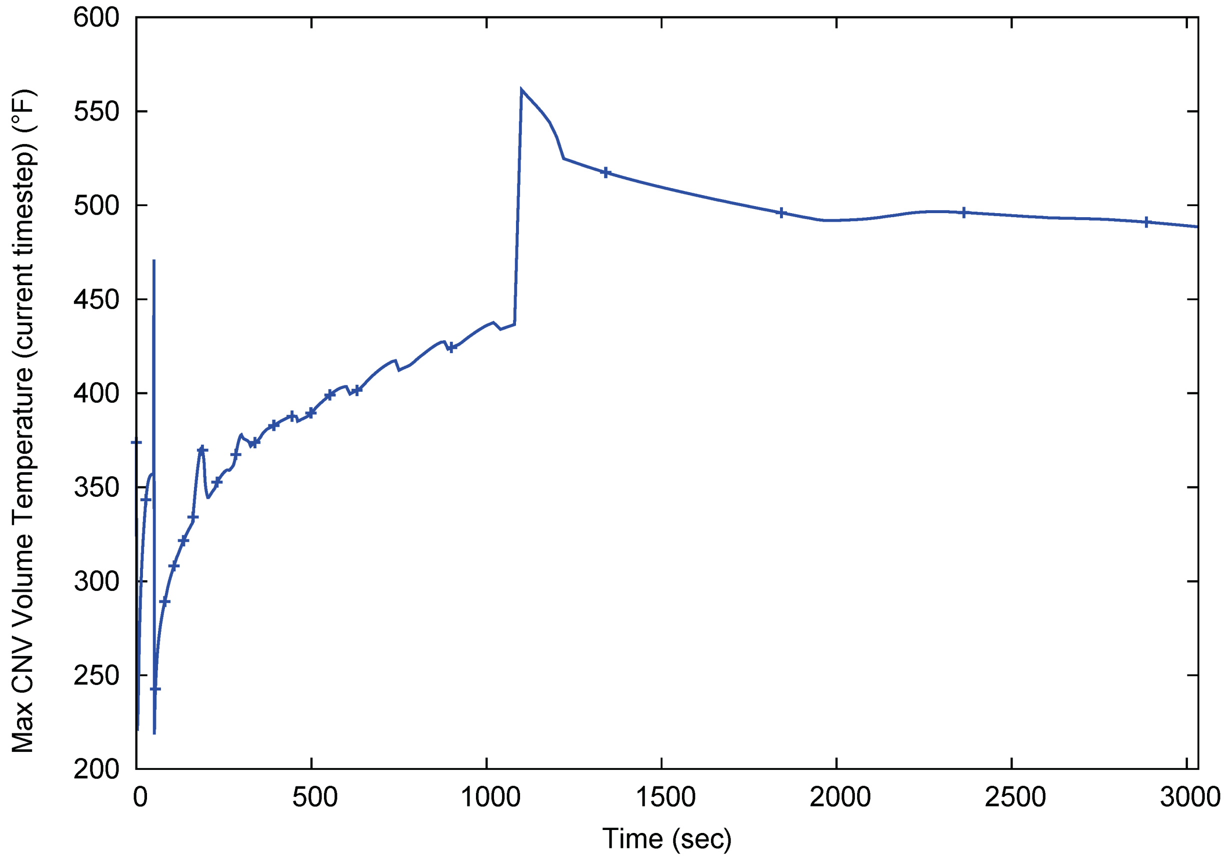

Figure 5-12 Case 2 containment vessel vapor temperature - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

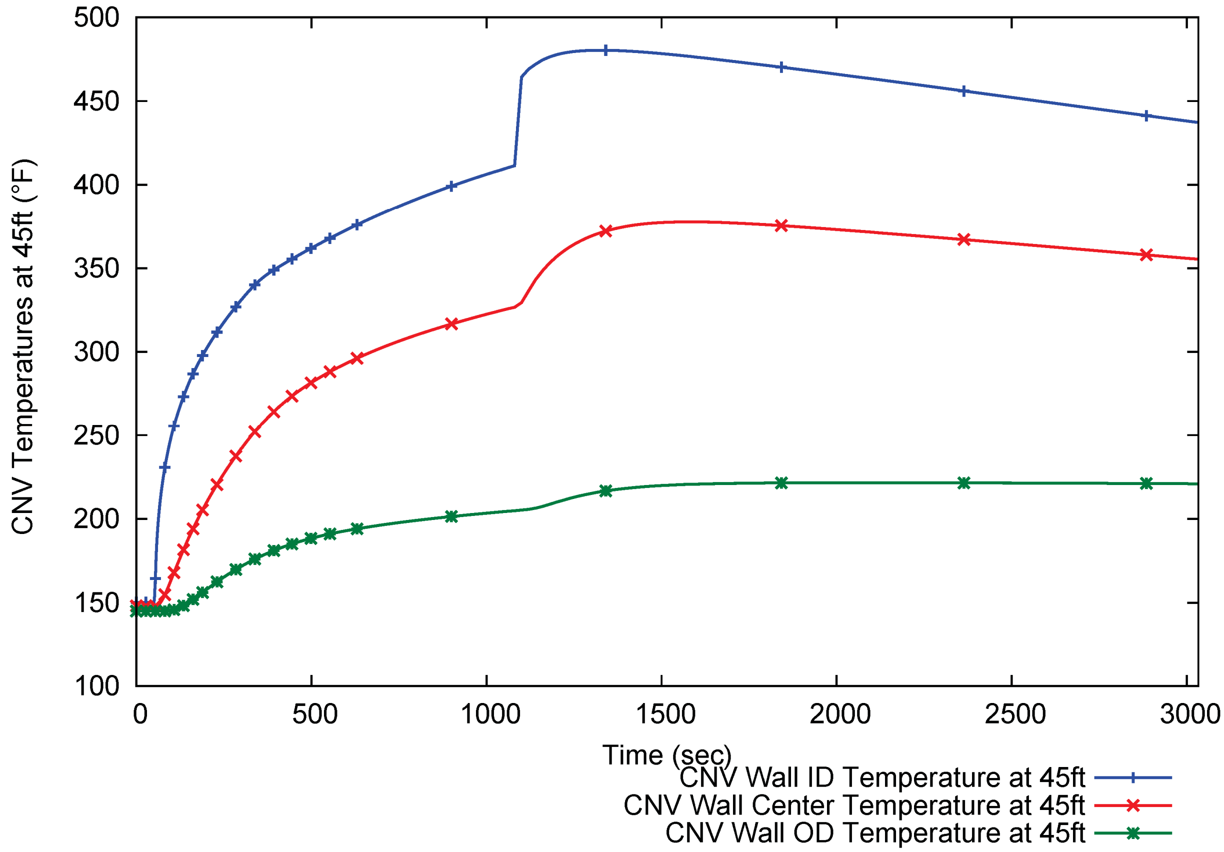

Figure 5-13 Case 2 containment vessel wall temperature profile -reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

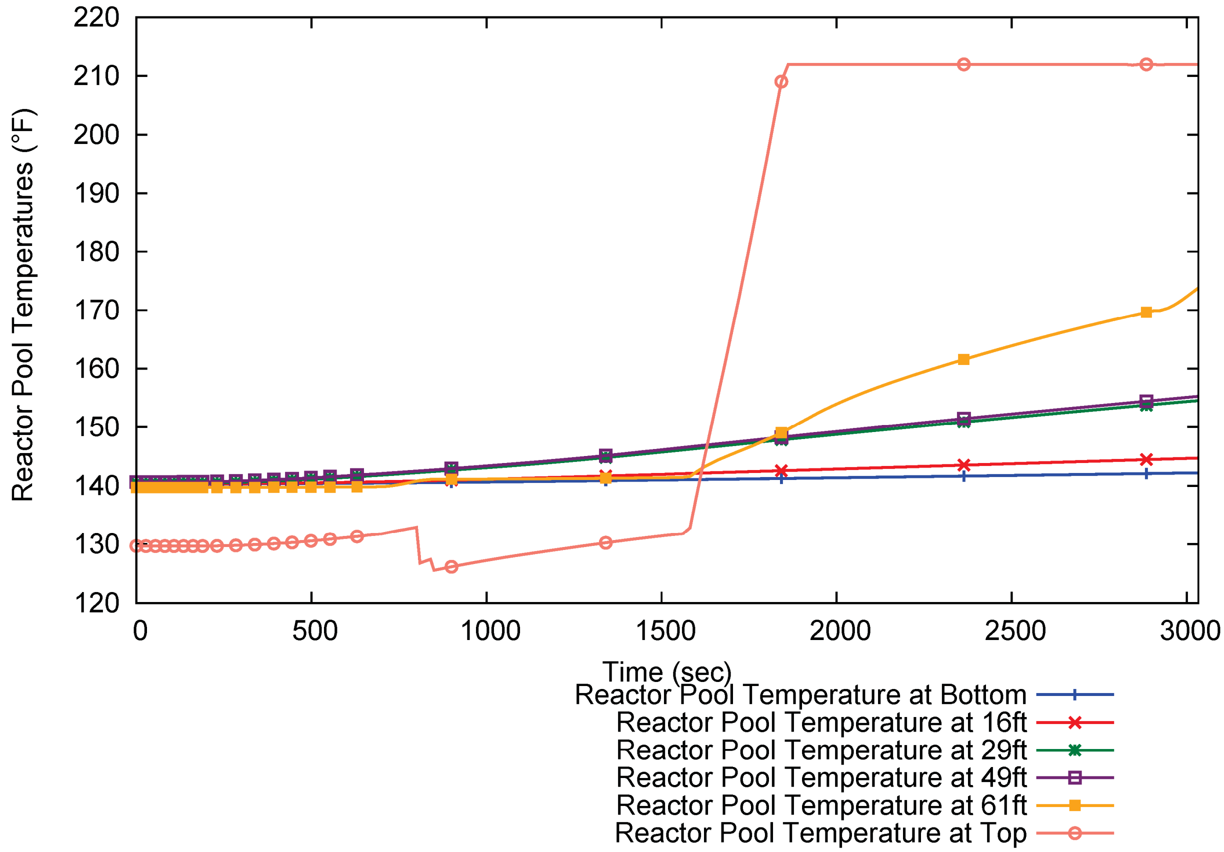

Figure 5-14 Case 2 reactor pool temperatures - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

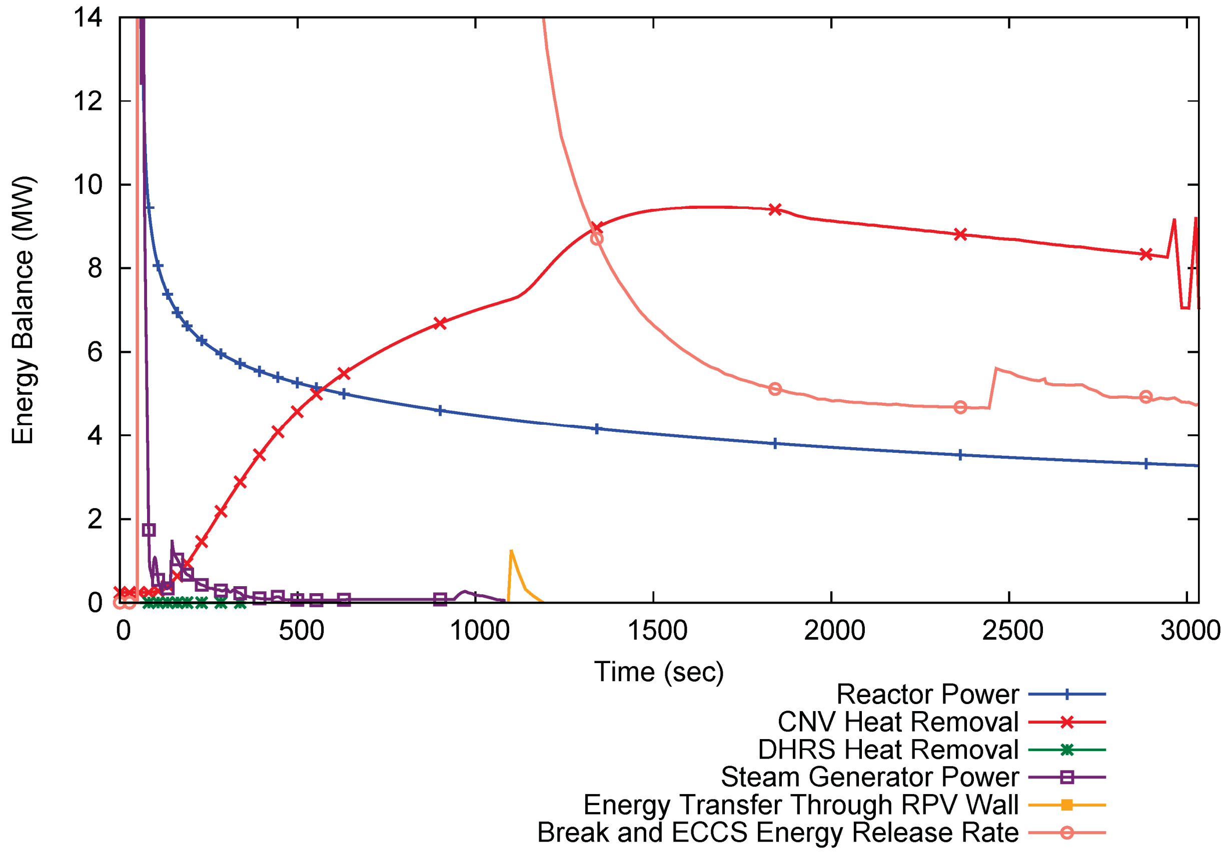

Figure 5-15 Case 2 energy balance - reactor coolant system injection lIne break loss-of-coolant accident (peak pressure case)

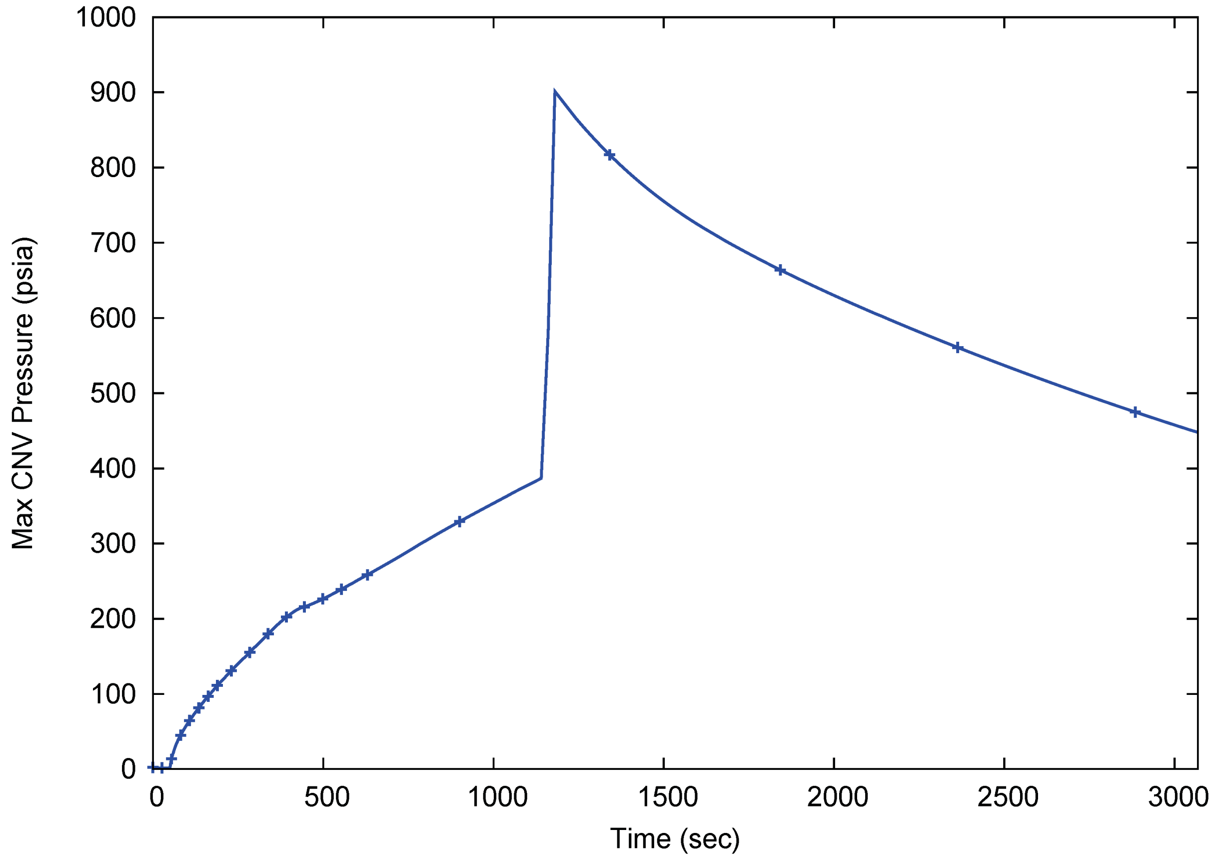

Figure 5-16 Case 2 containment vessel pressure - reactor coolant system injection line break loss-of-coolant accident (peak CNV wall temperature case)

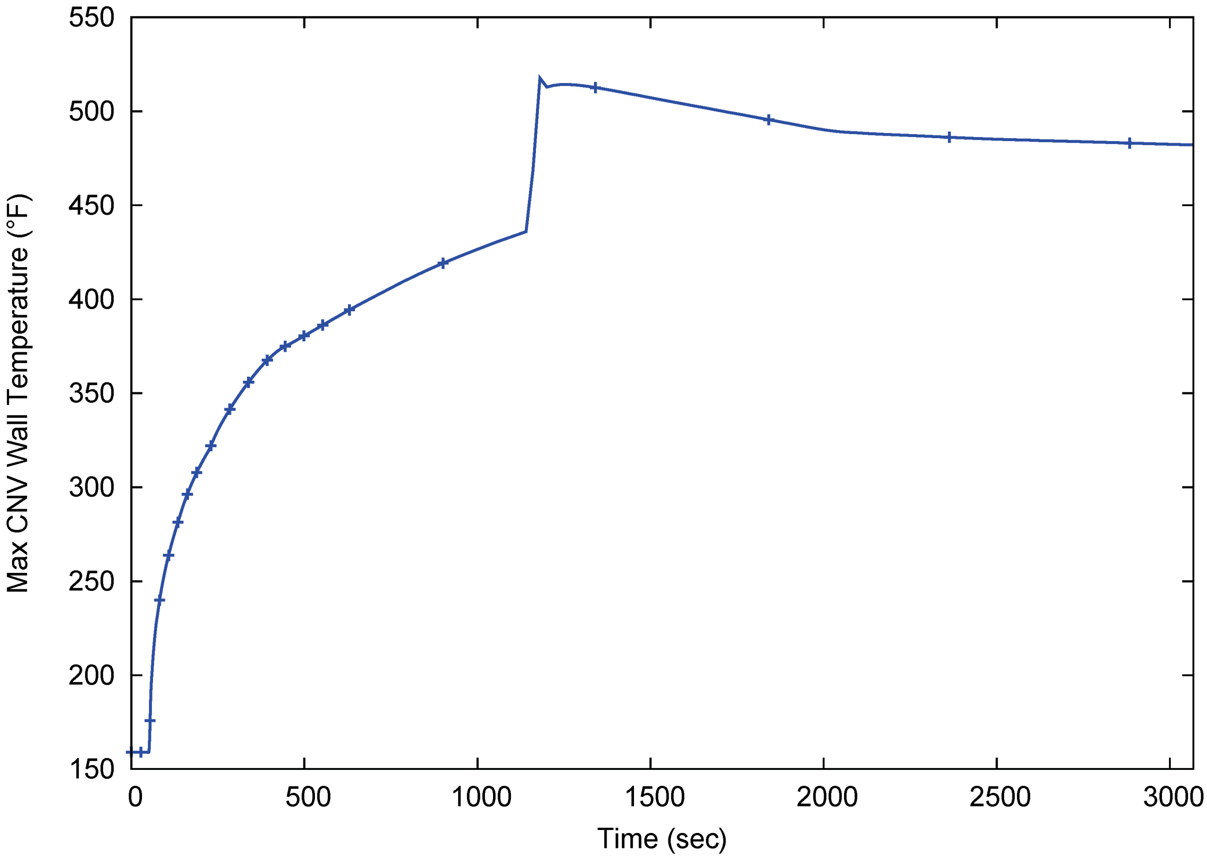

Figure 5-17 Case 2 containment vessel peak wall temperature - reactor coolant system injection line break loss-of-coolant accident (peak CNV wall temperature case)

5.1.3.3 Case 3: Reactor Pressure Vessel High Point Degasification Vent Line Loss-of-Coolant Accident

The LOCA in the RPV high point degasification line initiates an M&E release from the top of the pressurizer into the CNV. The sequence of events is shown in Table 5-4. The CNV pressure response and temperature response are shown in Figures 5-18 and 5-19. The peak CNV pressure is 586 psia for the reference case, and 864 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, 1200 psid IAB release pressure). The peak CNV wall temperature is 476 degrees F for the reference case, and 484 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, 900 psid IAB release pressure, RRV single failure). Case 3 is non-limiting.

Table 5-4 Case 3 sequence of events - RPV high point degasification line break loss-of-coolant accident

Peak CNV Pressure Case Time (sec)	Event	Peak CNV Temperature Case Time (sec)
50	LOCA in RPV high point degasification line Loss of normal AC and DC power • ECCS actuation signal • Reactor trip • Containment isolation • MSIV closure • FWIV closure	Same
98	ECCS valves open on below IAB release pressure	142
119	Peak CNV pressure (864 psia)	N/A
N/A	Peak CNV temperature (484 degrees F)	437
~2300	CNV pressure decreases to <50% of peak pressure	N/A

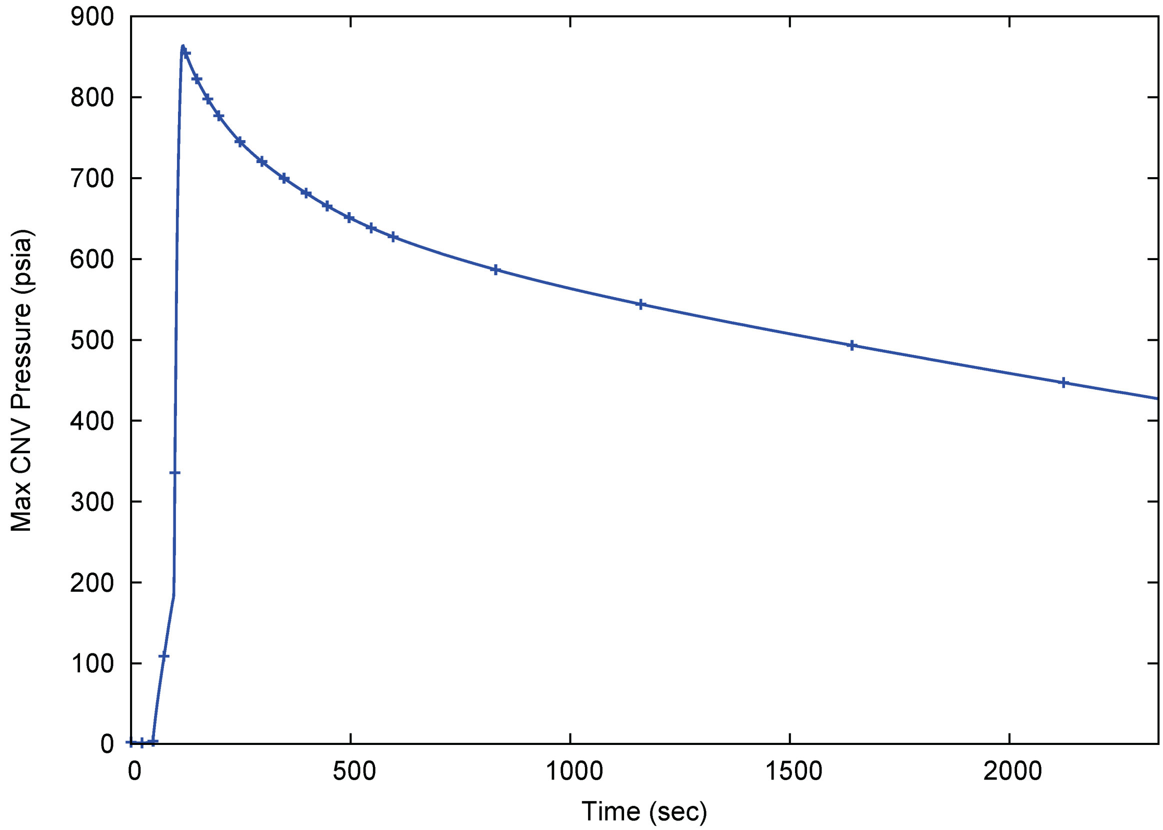

Figure 5-18 Case 3 containment vessel pressure - pressurizer spray supply line break loss-of-coolant accident

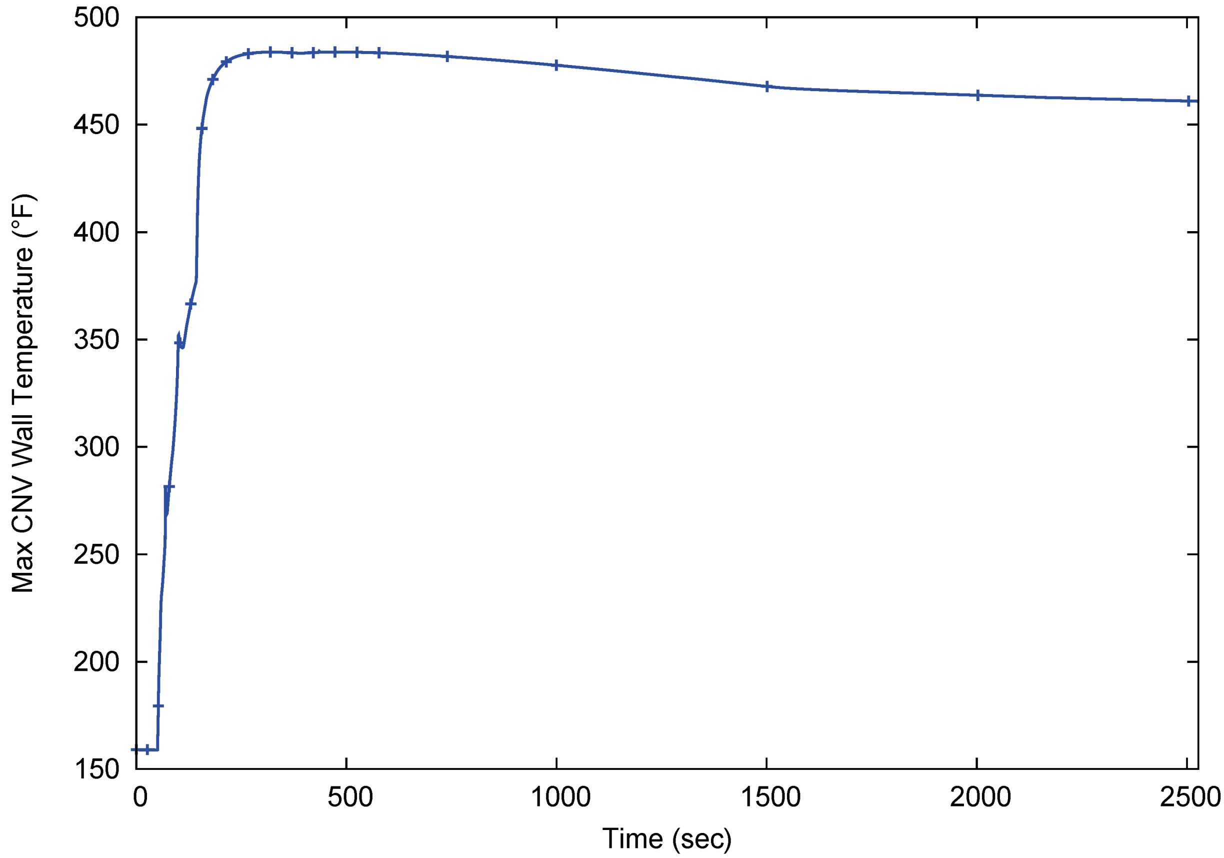

Figure 5-19 Case 3 containment vessel wall temperature – pressurizer spray supply line break loss-of-coolant accident

5.1.3.4 Case 4: Inadvertent Reactor Vent Valve Opening Anticipated Operational Occurrence

The inadvertent RVV actuation anticipated operational occurrence (AOO) initiates an M&E release from the top of the pressurizer into the CNV. The sequence of events is shown in Table 5-5. The CNV pressure response and temperature response are shown in Figures 5-20 and 5-21. The peak CNV pressure is 802 psia for the reference case, and 880 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, 1200 psid IAB release pressure, minimum primary system flow). The peak CNV temperature is 481 degrees F for the reference case, and 482 degrees F for the case with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, 900 psid IAB release pressure). There was no sensitivity to any of the single failures. Case 4 is non-limiting.

Table 5-5 Case 4 sequence of events – inadvertent reactor vent valve opening event

Peak CNV Pressure Case Time (sec)	Event	Peak CNV Temperature Case Time (sec)
50	Inadvertent RVV actuation Loss of normal AC and DC power • ECCS actuation signal • Reactor trip • Containment isolation • MSIV closure • FWIV closure	Same
61	ECCS valves opening on below IAB release pressure	68
79	Peak CNV pressure (880 psia)	N/A
N/A	Peak CNV temperature (482 degrees F)	256
~2200	CNV pressure decreases to <50% of peak pressure	N/A

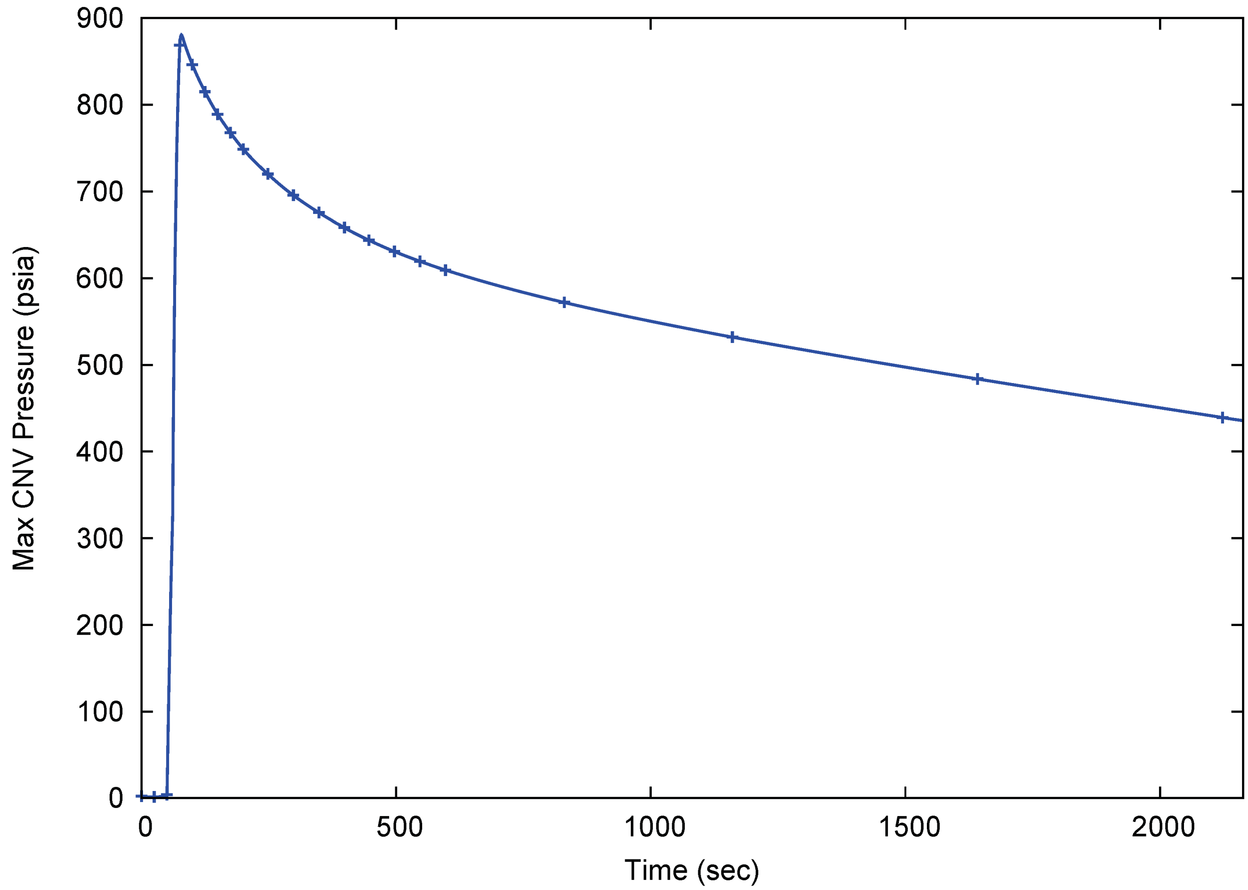

Figure 5-20 Case 4 containment vessel pressure – inadvertent reactor vent valve opening event

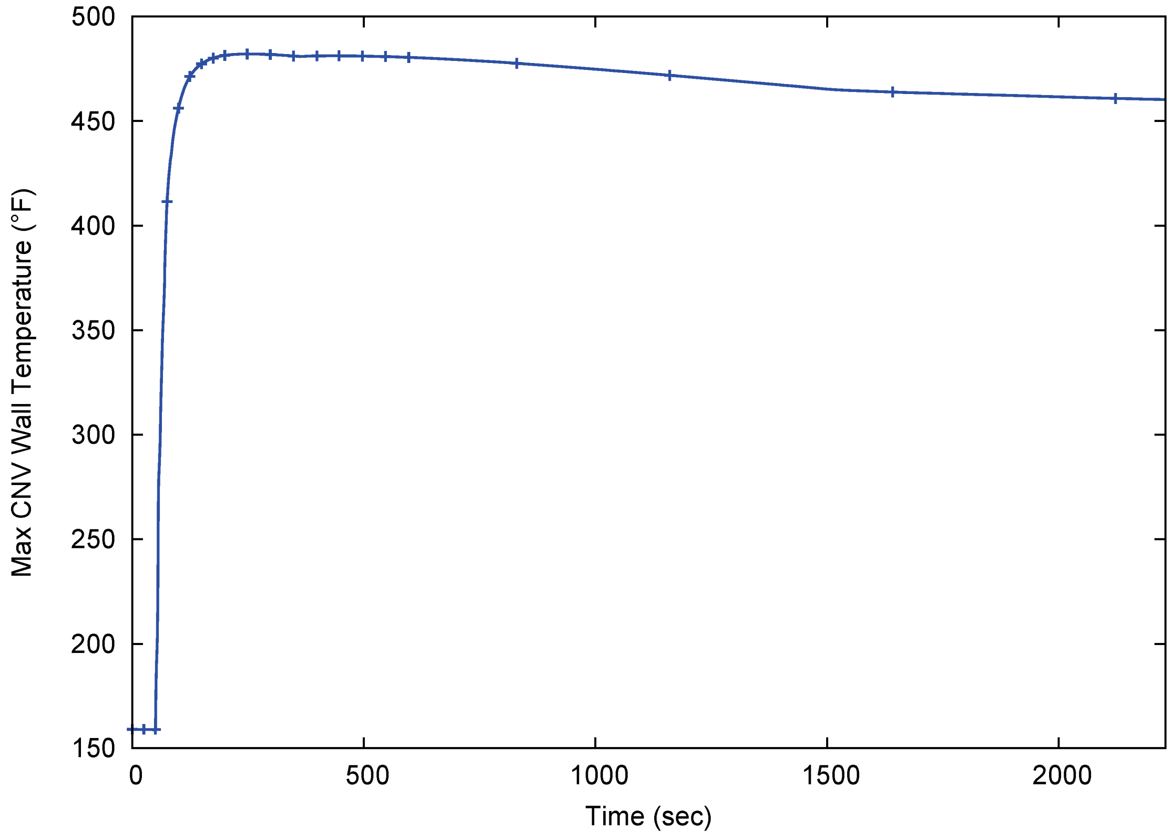

Figure 5-21 Case 4 containment vessel wall temperature – inadvertent reactor vent valve opening event

5.1.3.5 Case 5: Limiting Overall Containment Vessel Pressure Event - Inadvertent Reactor Recirculation Valve Opening Anticipated Operational Occurrence

The inadvertent RRV actuation initiates an M&E release from the downcomer into the CNV. The results of the primary release event M&E release break spectrum analysis and sensitivity analyses have determined that this AOO (Case 5) results in the limiting peak CNV pressure for all postulated events. The sequence of events is shown in Table 5-6, and detailed results for key parameters are shown in Figures 5-22 through 5-35. The peak CNV pressure is 933 psia for the reference case, and 951 psia with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +18 psi (~1.9 percent) increase are 1) the timing of the ECCS valve opening as determined by the IAB release pressure setpoint; 2) the assumption of a loss of normal AC and DC power; and 3) minimum primary system flow. There was no adverse impact from the single failure sensitivity studies. The peak CNV temperature is 495 degrees F for the reference case, and 506 degrees F with the combined effect of the adverse sensitivity parameters. The Case 5 peak CNV pressure case is the same case as the Case 5 peak CNV wall temperature case.

The sequence of events (Table 5-6) shows that in the first seconds following the occurrence of an inadvertent RRV event many automatic responses occur to transition the

module from full-power operation to an alignment that mitigates the initial blowdown phase. The break flow into the CNV causes a rapid pressurization that reaches the 9.5 psia high pressure setpoint. The following automatic actions occur on high CNV pressure:

- containment isolation resulting in MSIV and FWIV closure
- reactor trip
- turbine trip
- DHRS actuation (Note: DHRS actuation is not credited in the primary system containment response analysis methodology)

As a conservative assumption, a loss of normal AC and DC power is also assumed to occur at the time of the break. This results in an ECCS signal. However, RRVs and RVVs opening does not occur until the differential pressure across the valve decreases to below the IAB release pressure. In the containment response analysis methodology the IAB release pressure is an important analysis input as it determines the second primary system M&E release into the CNV through the RVVs and the second RRV. The peak CNV pressure and peak CNV wall temperature occur following the RVVs opening after the CNV has been preheated by the initial M&E release.

Following the alignment of the module for blowdown, the primary system pressure and inventory decrease due to the loss of inventory. The CNV pressurizes and the steam condenses on the cold interior wall of the CNV. The condensate flows down the CNV walls and accumulates along with unflashed break liquid in a pool in the CNV lower head. The cold CNV wall absorbs the energy of the condensed steam and starts to heat up by conduction. Eventually the energy is transferred through the CNV wall to the reactor pool, and the pool temperature slowly increases. Opening of the RVVs occurs at 127 seconds when the RCS pressure decreases to below the 1000 psid IAB release pressure, determined by the results of sensitivity analyses. Opening of the three RVVs and the second RRV results in the peak CNV pressure and wall temperature at 143 and 189 seconds, respectively. As flow through the RVVs dimishes, the primary and CNV pressures converge, and continued heat transfer to the CNV leads to a gradual cooldown and depressurization phase. Pressure equalization enables recirculation flow from the CNV pool through the RRVs to establish the long-term cooling recirculation alignment.

The primary system response for the Case 5 inadvertent RRV opening event is shown in Figures 5-22 through 5-28. Figure 5-22 shows the primary pressure response. The initial depressurization phase due to the RRV opening is continued by the rapid depressurization when the RVVs open. Figures 5-23 and 5-24 show the inventory in the pressurizer and in the riser. These figures show the expected trend of a decreasing level in the primary followed by a stabilization in inventory, with some liquid holdup in the pressurizer. A sensitivity study that decreased the interphase drag in the upper riser, riser upper plenum, pressurizer baffle, pressurizer, and the downcomer with the intent of reducing liquid entrainment, showed that there was no adverse impact on the peak CNV pressure for this case. Figure 5-25 shows the primary coolant temperatures at six locations. Following ECCS actuation the temperatures converge and the cooldown proceeds. Figure 5-26 shows the RRV opening and ECCS mass flowrates. It is evident that the ECCS flow immediately following ECCS actuation, mainly the flow through the three RVVs into the

CNV, is significant. It is this flow spike that causes the peak CNV pressure and wall temperatures to occur shortly thereafter as shown in Table 5-6. Figures 5-27 and 5-28 show the integrated LOCA and ECCS mass flowrate and energy flowrate.

The CNV and reactor pool response for the Case 5 inadvertent RRV actuation LOCA is shown in Figures 5-29 to 5-34. Figure 5-29 shows the CNV pressure response and how pressure rapidly increases to the limiting peak value of 951 psia. This limiting NRELAP5 result can be compared to the CNV design pressure of 1000 psia. This is a key result of this limiting containment response analysis case. Figure 5-29 also demonstrates the long term cooling capability of the UHS. CNV pressure is reduced to below 50 percent of the peak value within two hours of accident initiation.

Figure 5-30 shows the CNV liquid level increase as the unflashed break flow and condensed steam accumulates. Figure 5-31 shows the CNV vapor temperature. Initially, flashing of the break flow at low CNV pressure results in a temperature decrease. {{

}}2(a),(c) Figure 5-32 shows the peak CNV wall temperature and the limiting value of 506 degrees F. Figure 5-33 shows the temperature profile across the CNV wall at the 45 foot elevation. There is a large temperature gradient across the CNV wall. Figure 5-34 shows the reactor pool temperatures for a range of elevations. The reactor pool temperature does not increase significantly for the short duration of these M&E release analyses. From Figures 5-31 through 5-34 it is evident that the CNV wall is the significant heat sink in the short-term. Even with the conservative initial reactor pool level of 55 ft above the pool floor and a temperature of 140 degrees F assumed in these analyses, the CNV wall is capable of maintaining the peak CNV pressure within the design limit.

Figure 5-35 shows the energy balance during the RRV loss-of-coolant accident and the trends of the heat sources and sinks. At approximately 700 seconds the energy release from the LOCA and the RVV valves decreases to below the energy transferred through the CNV wall. The CNV wall then continues to provide a strong heat sink for the sustained cooldown and depressurization of the module.

Table 5-6 Case 5 sequence of events – inadvertent reactor recirculation valve opening event

Time (sec)	Event
50	Inadvertent RRV Actuation Loss of normal AC and DC power • ECCS actuation signal • Reactor trip • Containment isolation • MSIV closure • FWIV closure
127	ECCS valve opening on differential pressure below IAB release pressure (<1000psid)
143	Peak CNV pressure
189	Peak CNV temperature
~2100	CNV pressure decreases to <50% of peak pressure

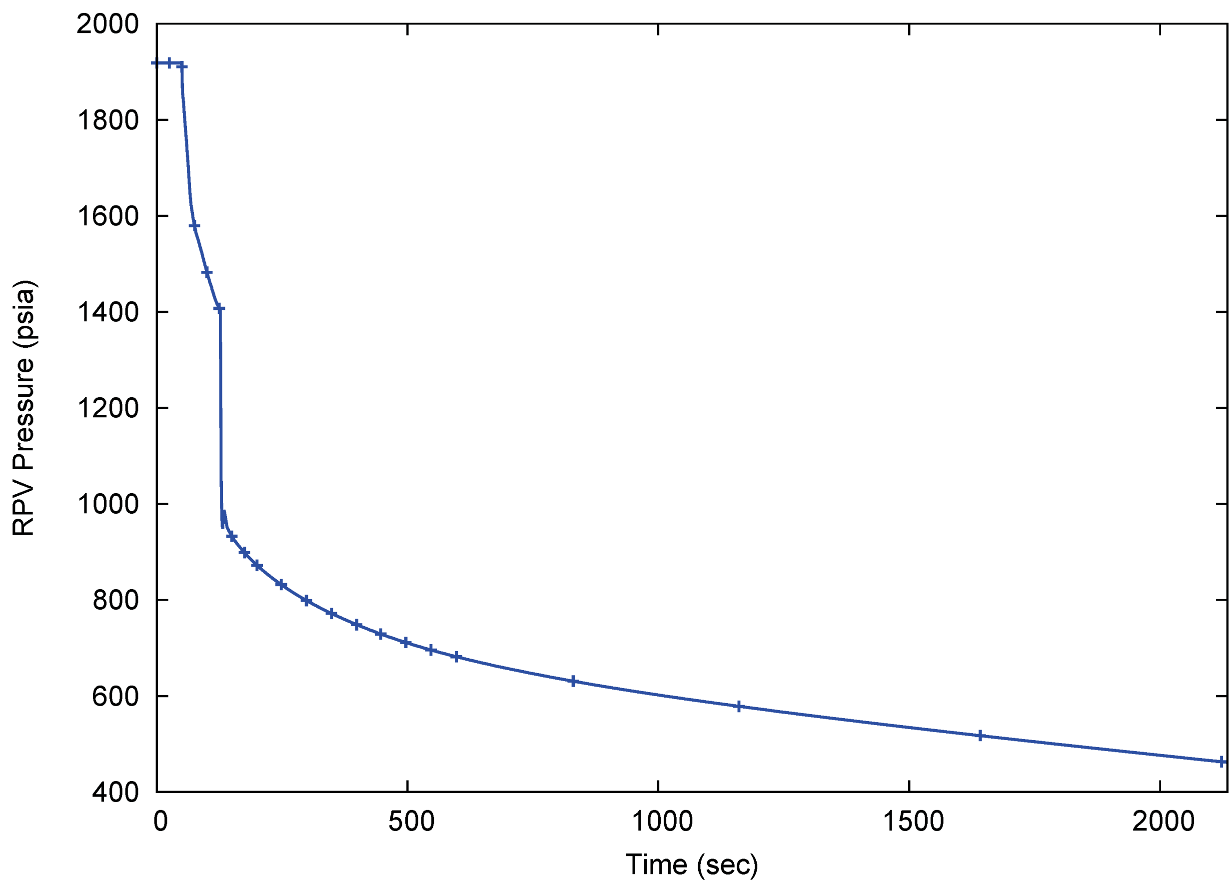

Figure 5-22 Case 5 primary pressure - inadvertent reactor recirculation valve opening event

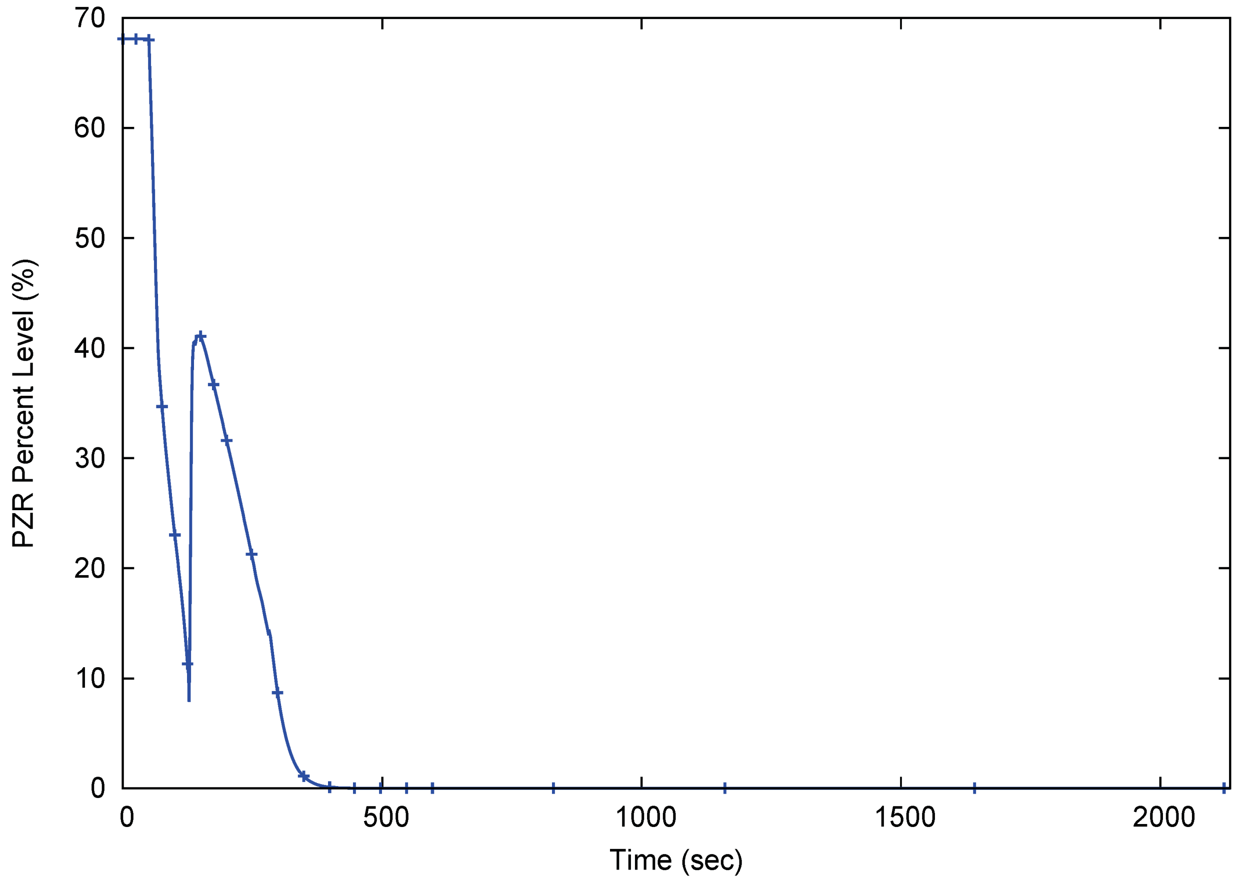

Figure 5-23 Case 5 pressurizer level - inadvertent reactor recirculation valve opening event

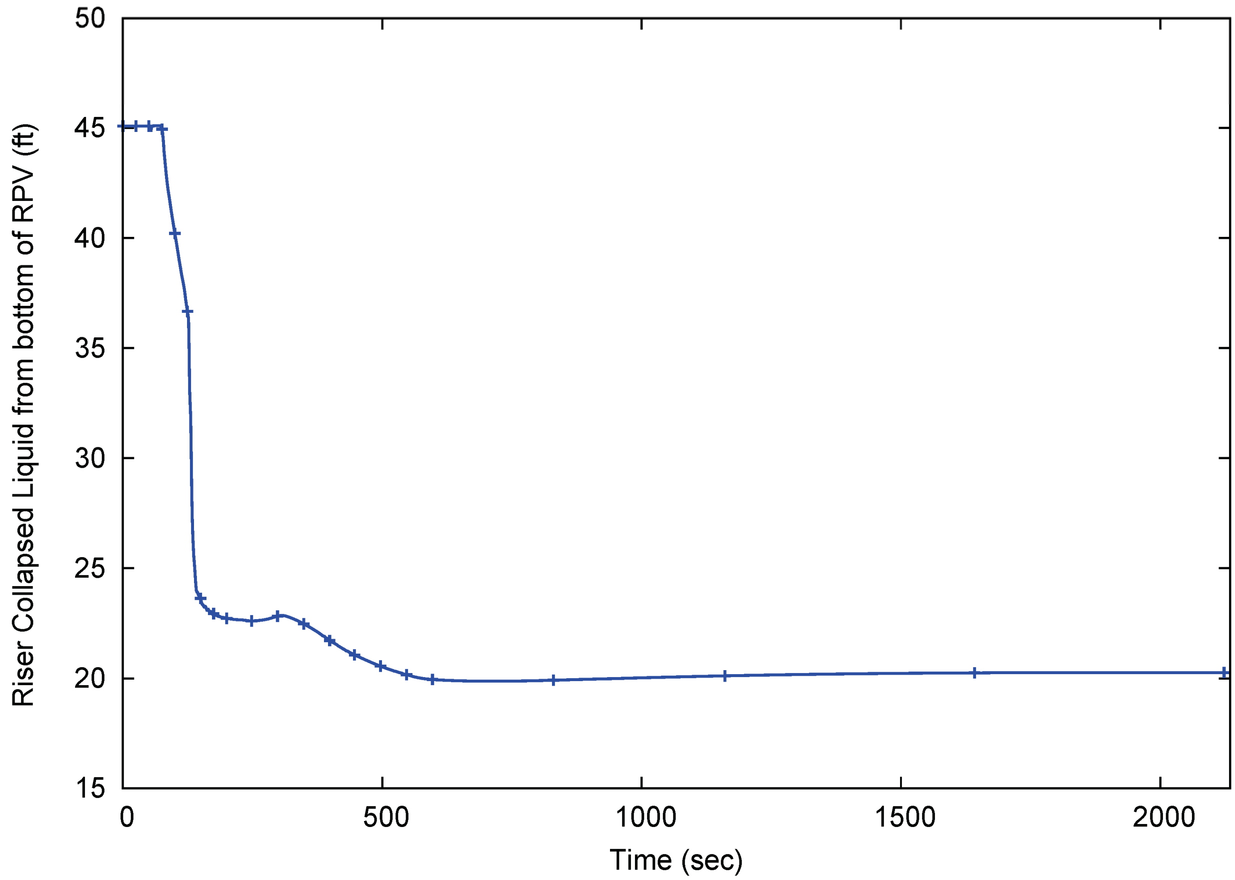

Figure 5-24 Case 5 riser level - inadvertent reactor recirculation valve opening event

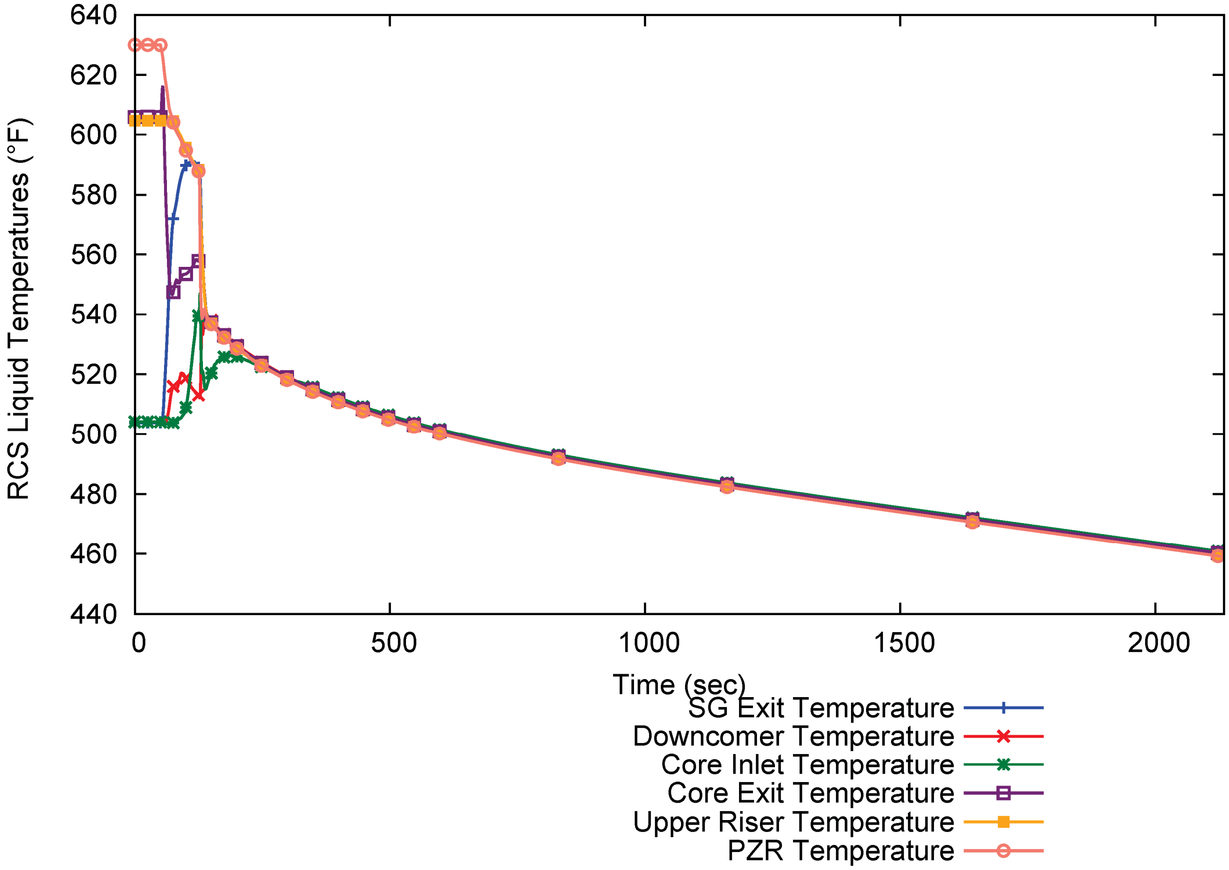

Figure 5-25 Case 5 primary temperature - inadvertent reactor recirculation valve opening event

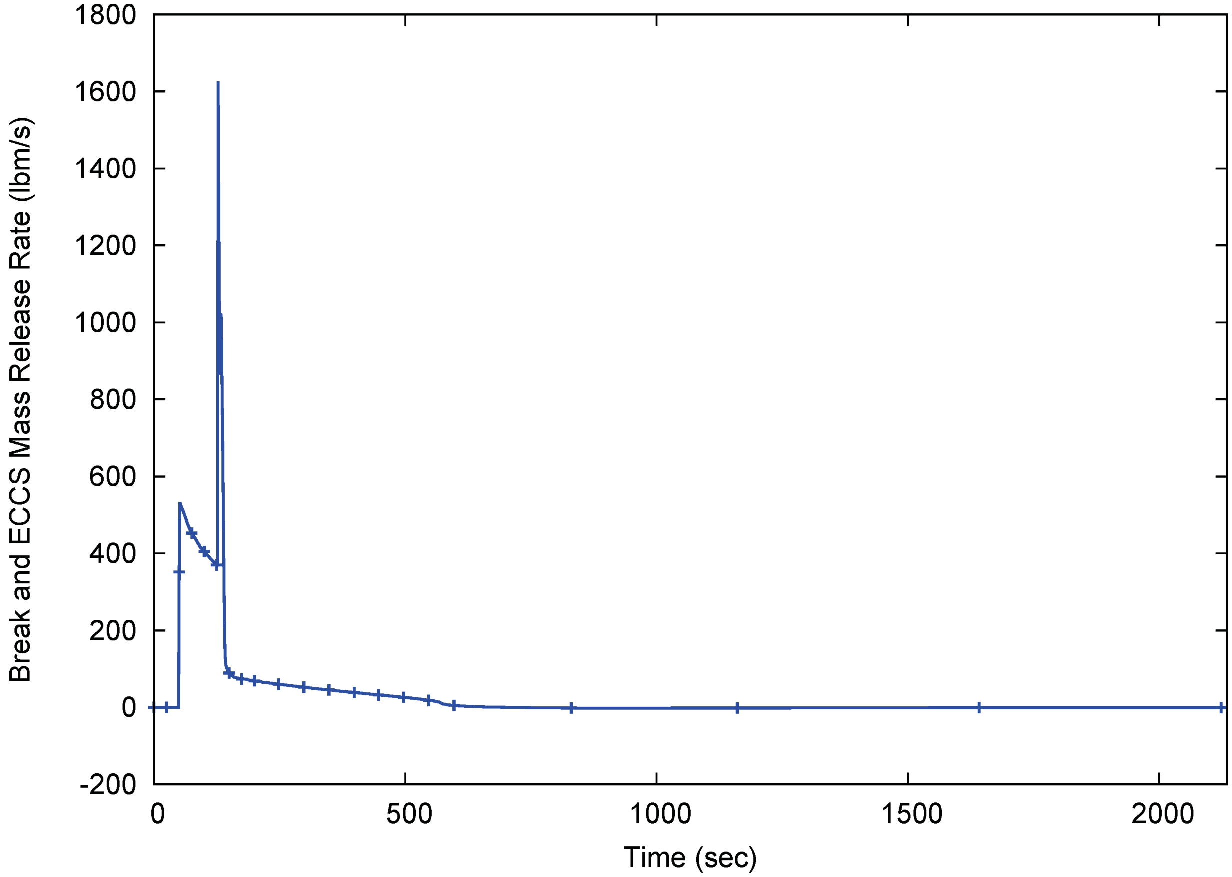

Figure 5-26 Case 5 loss-of-coolant accident and emergency core cooling system flowrate - inadvertent reactor recirculation valve opening event

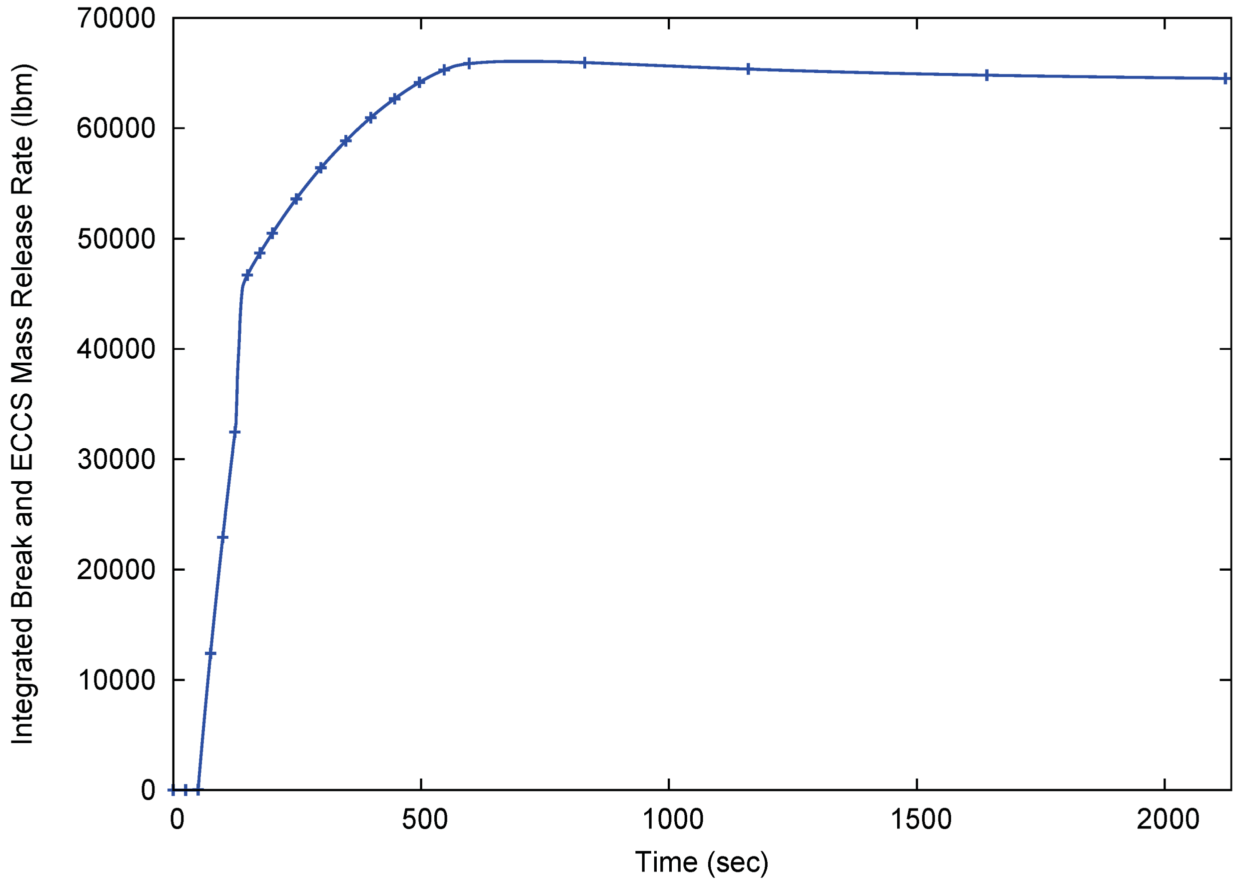

Figure 5-27 Case 5 integrated loss-of-coolant accident and emergency core cooling system mass flow rate - inadvertent reactor recirculation valve opening event

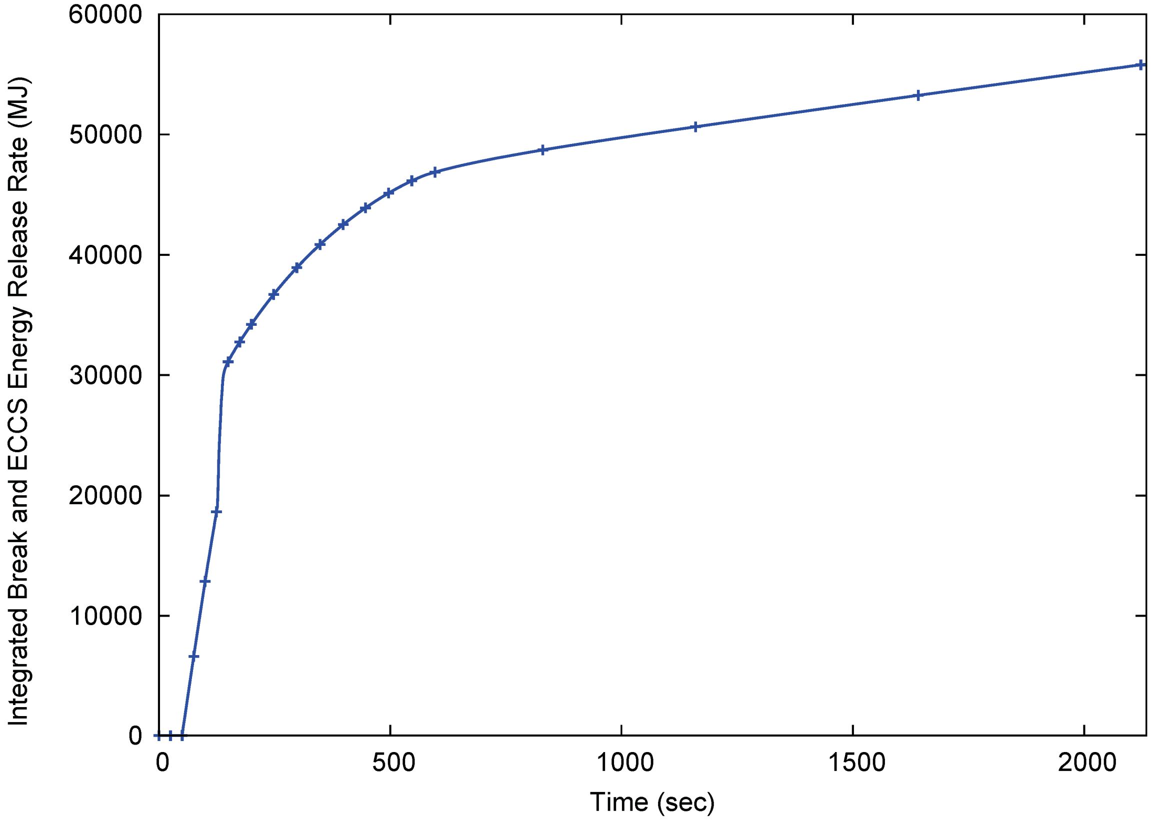

Figure 5-28 Case 5 integrated loss-of-coolant accident and emergency core cooling system energy release - inadvertent reactor recirculation valve opening event

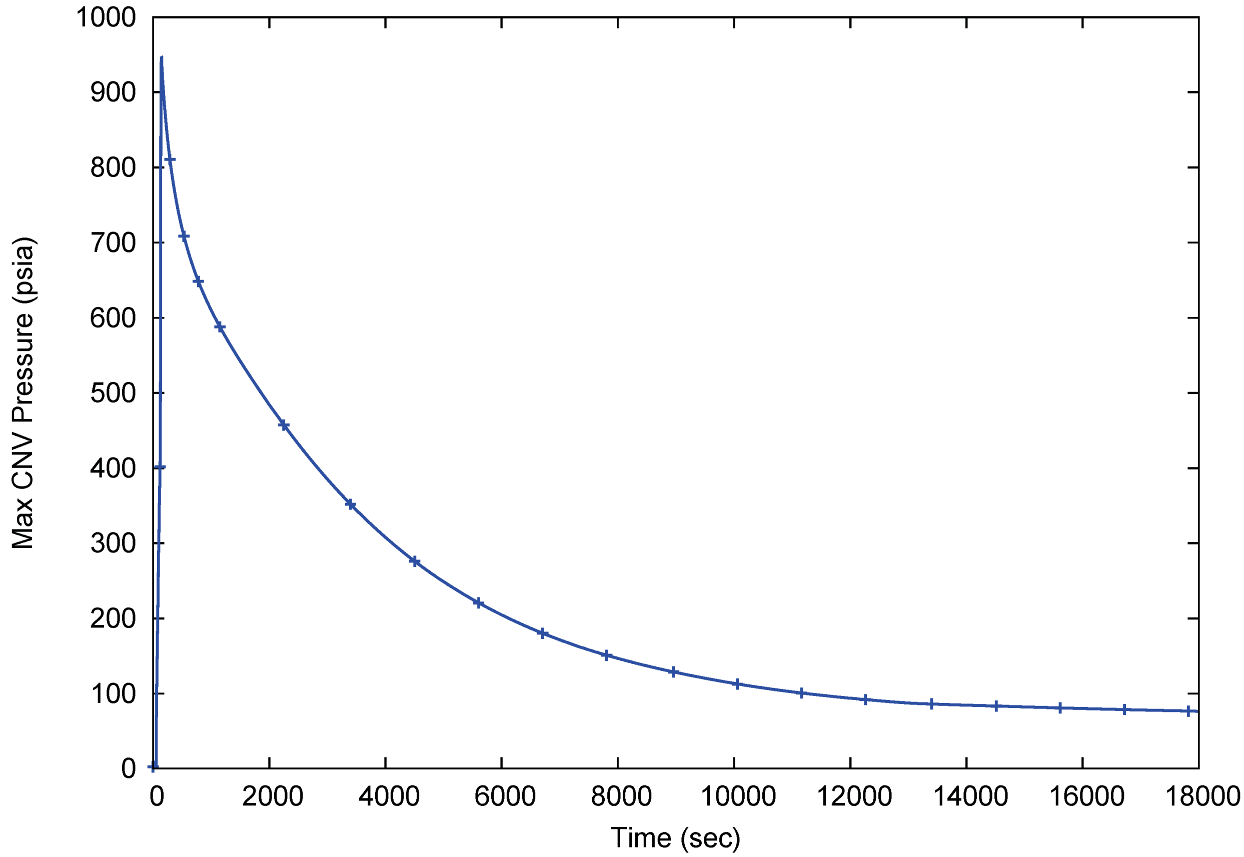

Figure 5-29 Case 5 containment vessel pressure - inadvertent reactor recirculation valve opening event (overall limiting pressure case)

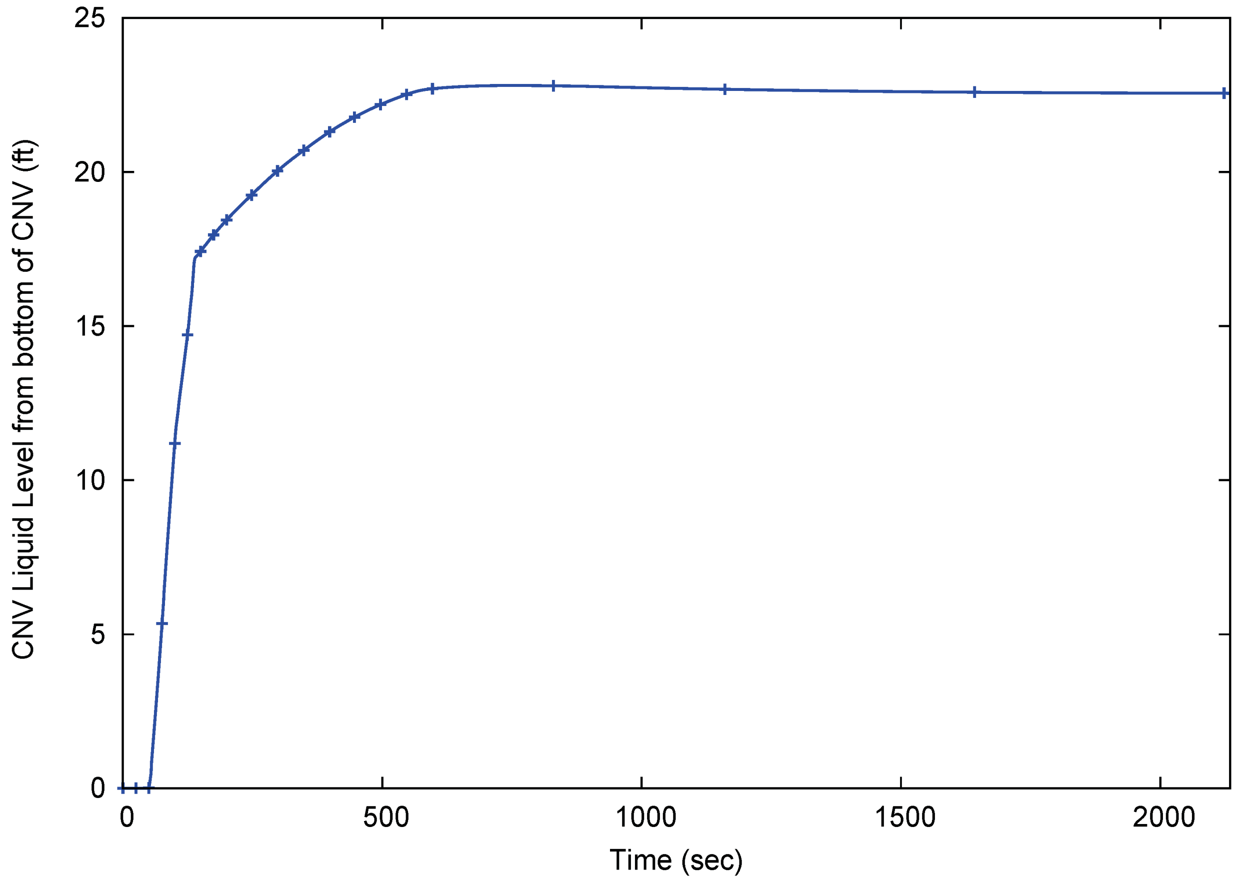

Figure 5-30 Case 5 containment vessel level - inadvertent reactor recirculation valve opening event

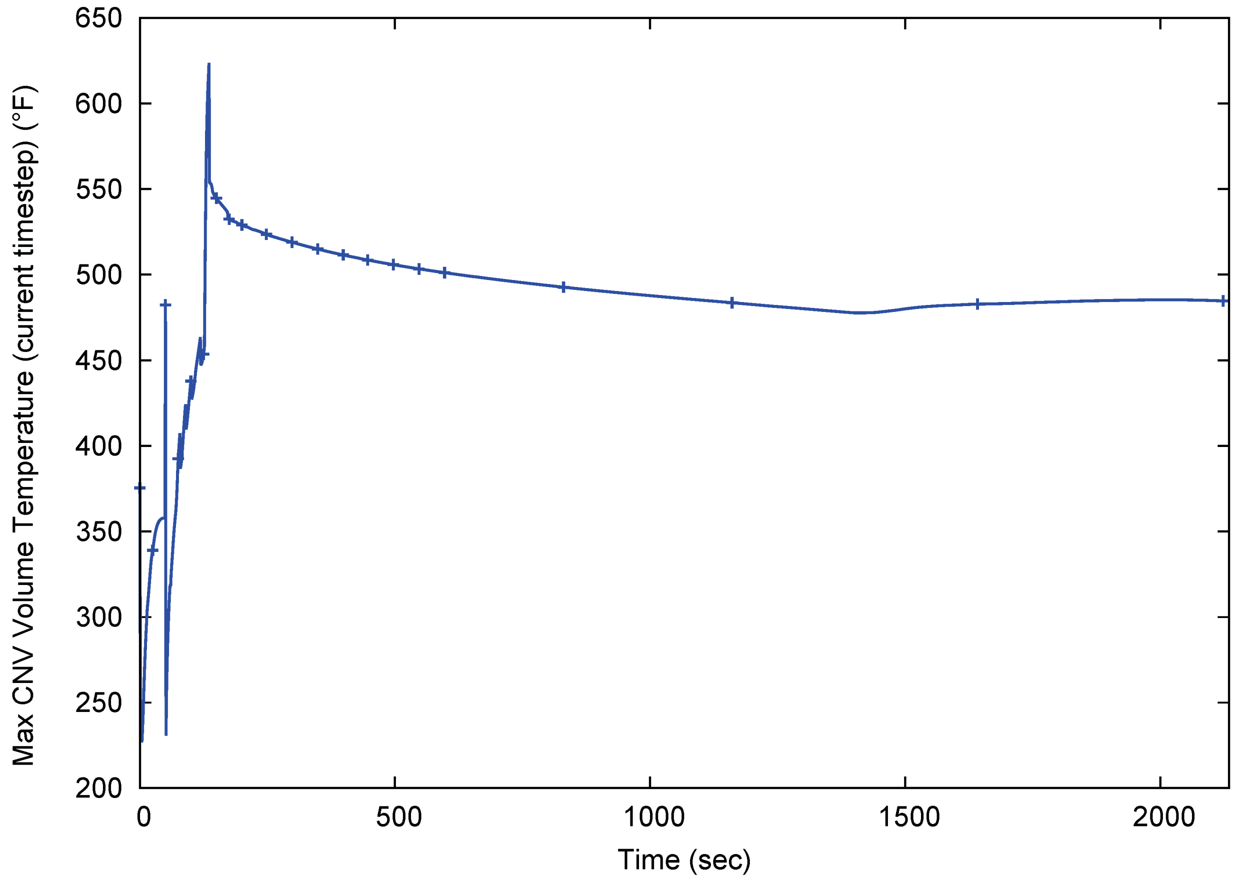

Figure 5-31 Case 5 containment vessel vapor temperature - inadvertent reactor recirculation valve opening event

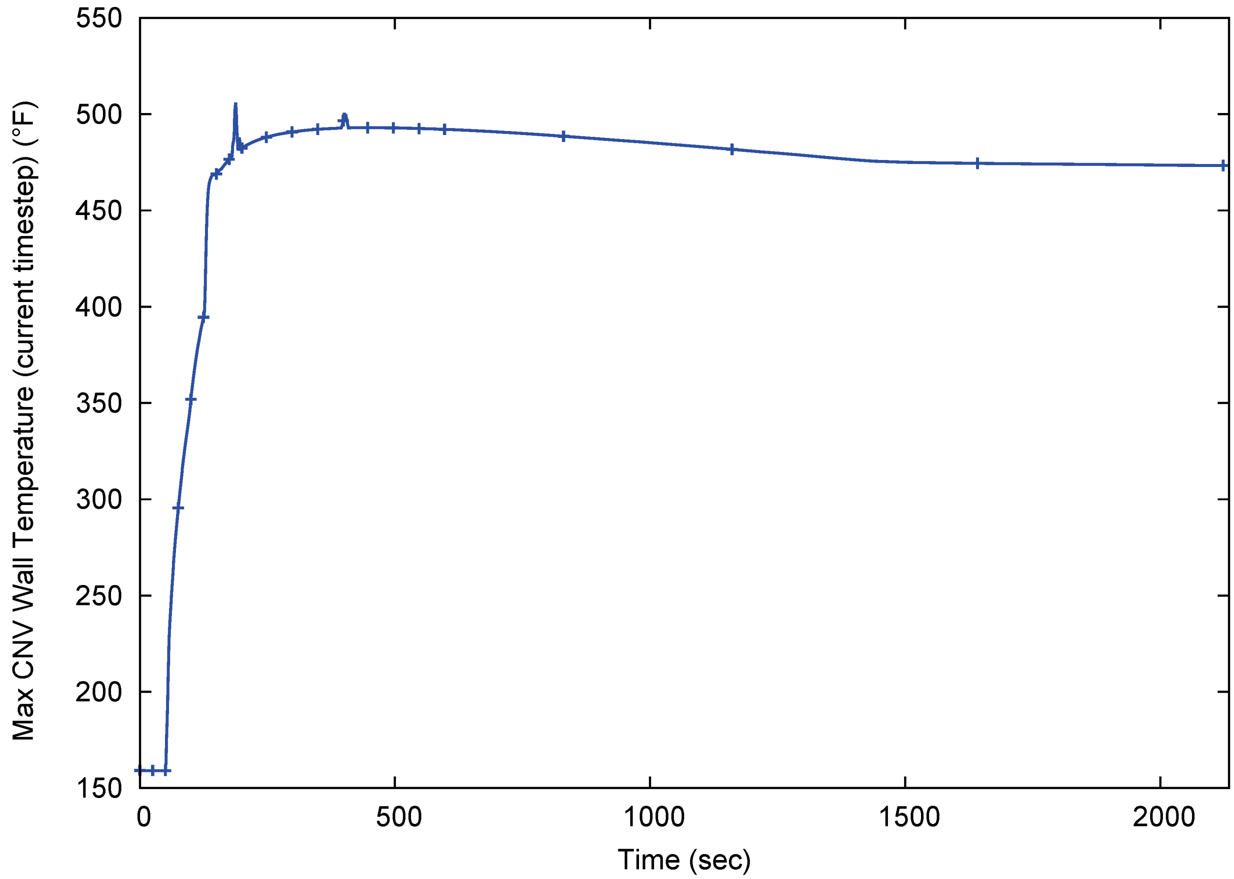

Figure 5-32 Case 5 containment vessel wall temperature - inadvertent reactor recirculation valve opening event

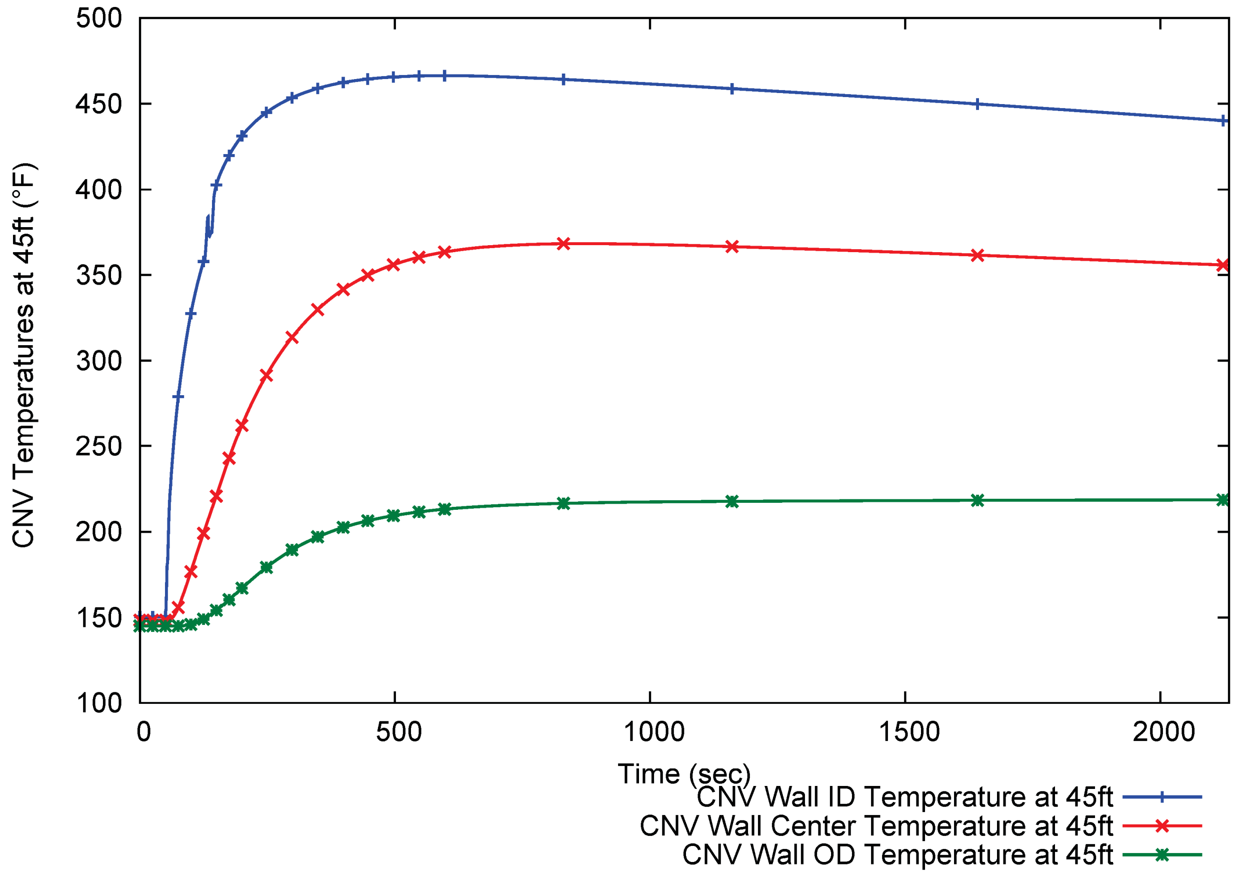

Figure 5-33 Case 5 containment vessel wall temperature profile - inadvertent reactor recirculation valve opening event

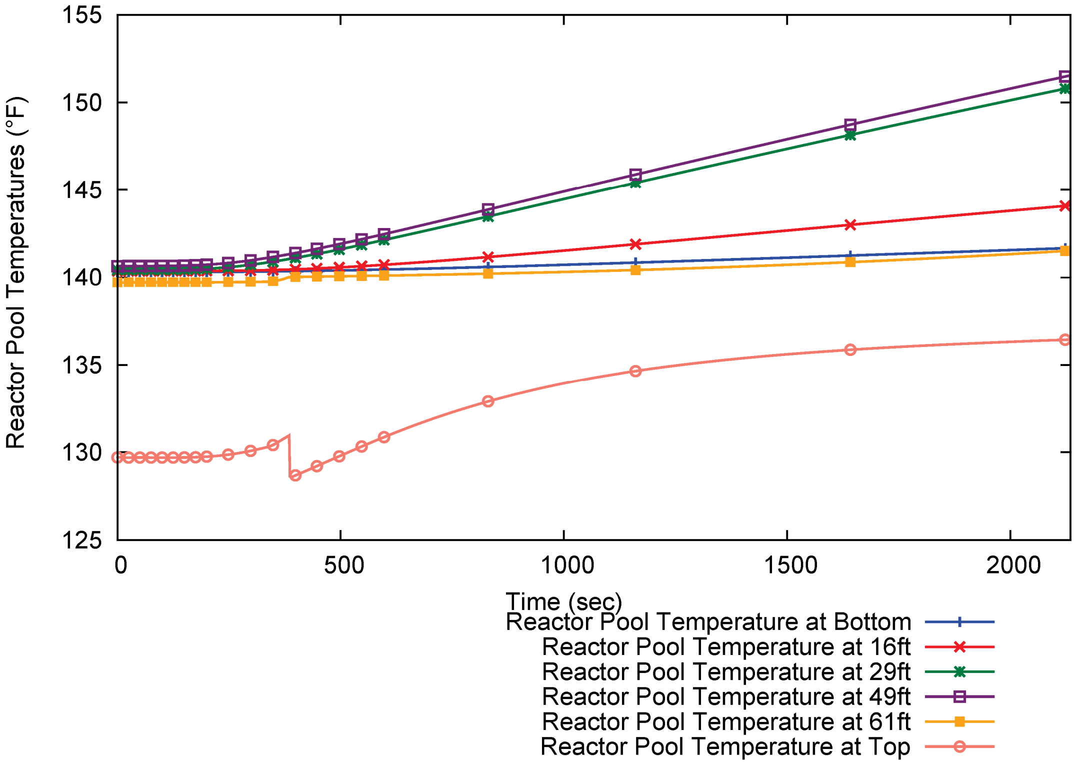

Figure 5-34 Case 5 reactor pool temperature - inadvertent reactor recirculation valve opening event

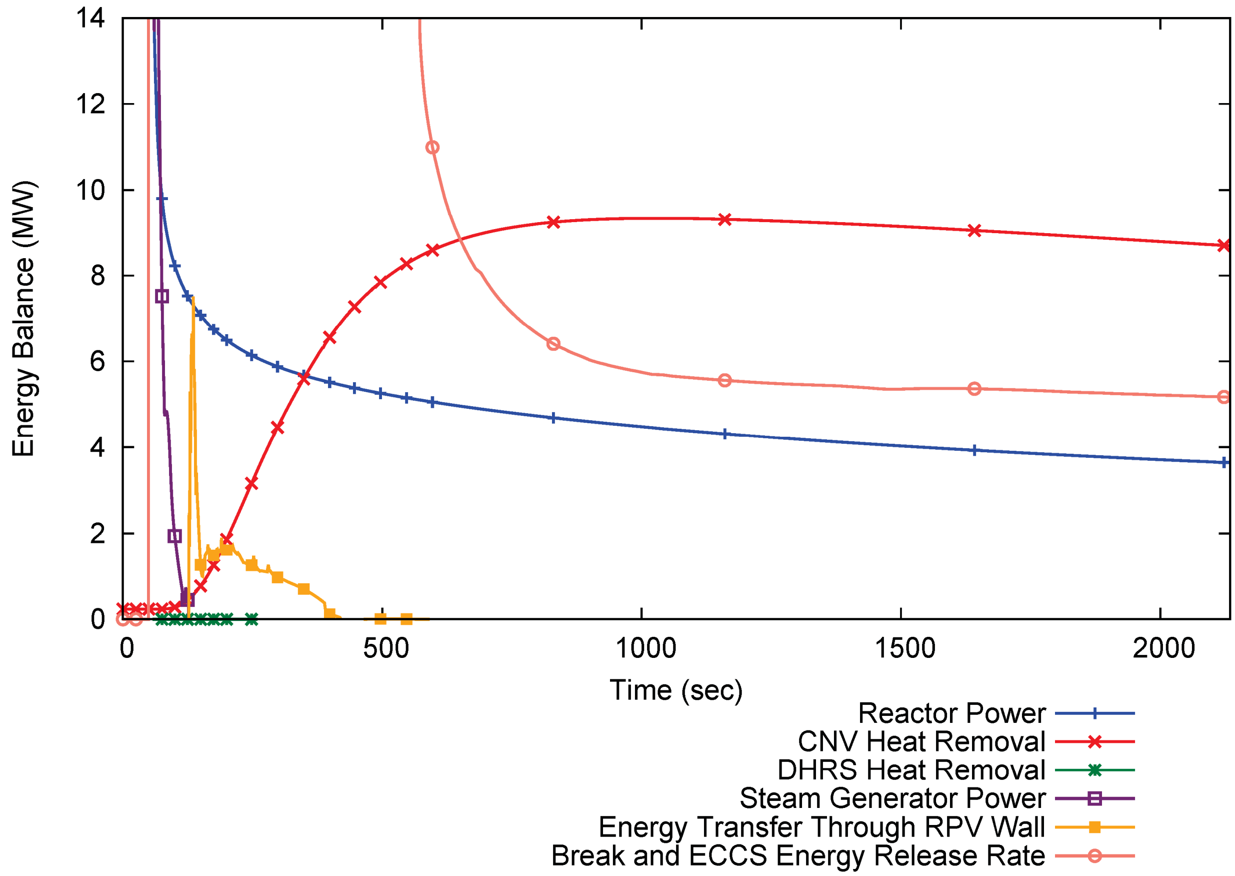

Figure 5-35 Case 5 energy balance - inadvertent reactor recirculation valve opening event

5.2 Main Steamline Break Pressure and Temperature Results

The sequence of events (Table 5-7) show that in the first seconds following the occurrence of a MSLB many automatic responses occur to transition the module from full power operation to an alignment that mitigates the secondary system blowdown. The break flow into the CNV causes a rapid pressurization that reaches the 9.5 psia high pressure setpoint. The following automatic actions occur on high CNV pressure:

- containment isolation including MSIV and FWIV closure
- reactor trip
- turbine trip
- DHRS actuation

Following the alignment of the module to mitigate the secondary blowdown, the secondary system pressure and inventory decrease due to the loss of inventory through the break. With continued normal AC power the feedwater pump initially continues to operate and supply the SGs. Feedwater isolation then terminates the supply of feedwater to the affected SG and effectively mitigates the event. The CNV pressurizes and the steam

condenses on the cold ID of the CNV. The condensate flows down the CNV walls and accumulates in a pool in the CNV lower head. The cold CNV wall absorbs the energy of the condensed steam and starts to heat up by conduction. Eventually the energy is transferred through the CNV wall to the reactor pool, and the pool temperature slowly increases.

The module response for the MSLB is shown in Figures 5-36 through 5-51. Figure 5-36 shows the SG pressure response with the affected SG (SG2) depressurizing via blowdown out the break into the CNV. The unaffected SG (SG1) initially depressurizes until the MSIV closes, and then gradually pressurizes following DHRS actuation. Figure 5-37 shows the primary system temperature response due to the initial secondary system blowdown and then following DHRS actuation. Figure 5-38 shows the primary system pressure response with the initial depressurization following secondary system blowdown, and then the pressure increasing from operation of the pressurizer heaters during DHRS operation. Figure 5-39 shows that the pressurizer level rapidly decreases during the initial overcooling, and then gradually decreases in response to the decrease in primary temperatures during DHRS operation. Figures 5-40 through 5-42 show the secondary system mass release, the integrated mass release, and the integrated energy release into the CNV, respectively. The liquid entrainment in the break flow was negligible, and therefore the sensitivity study on interphase drag upstream of the break flow was not necessary.

The CNV and reactor pool responses for the MSLB are shown in Figures 5-43 to 5-48. Figure 5-43 shows the CNV pressure response. The pressure rapidly increases to the limiting peak value of 419 psia at 41 seconds. This limiting NRELAP5 result can be compared to the CNV design pressure of 1000 psia, and to the limiting primary release event result. The MSLB result is bounded by the limiting LOCA (Case 2) and overall limiting primary release event result (Case 5). This is a key result in this MSLB containment response analysis.

Figure 5-44 shows the CNV vapor temperature. {{

}}2(a),(c) Figure 5-45 shows the peak CNV wall temperature and the limiting value of 427 degrees F at 46 seconds. This limiting NRELAP5 result can be compared to the CNV design temperature of 550 degrees F, and to the limiting LOCA result. The MSLB result is bounded by the limiting primary release event result (Case 2). This is a key result in this MSLB containment response analysis.

Figure 5-46 shows the CNV level response. Figure 5-47 shows the temperature profile across the CNV wall. There is a large temperature gradient. Figure 5-48 shows the reactor pool temperatures for a range of elevations. The reactor pool temperature does not increase significantly for the short duration of these analyses. From these results it is evident that the CNV wall is the significant heat sink for these containment response analyses. Even with the conservative initial reactor pool level of 55 ft above the pool floor

and a temperature of 140 degrees F assumed in these analyses, the CNV wall is capable of maintaining the peak CNV pressure and temperature within the design limit.

Figure 5-49 shows the energy balance during the MSLB and the trends of the heat sources and sinks. At approximately 500 seconds the energy release from the MSLB has dimished, and the energy transfer through the CNV wall and from the DHRS to the pool dominate. This energy balance is consistent with the cooldown of the primary system shown in Figure 5-37.

Table 5-7 Main steam line break sequence of events

Time (sec)	**Event**
0	MSLB
0 – 2	Low steam line pressure resulting in • Reactor trip • Turbine trip • MSIV closure • FWIV closure • DHRS actuation High CNV pressure resulting in • Containment isolation
32	Closure of FWRV complete
41	Peak CNV pressure
46	Peak CNV temperature
~200	CNV pressure decreases to <50% of peak pressure

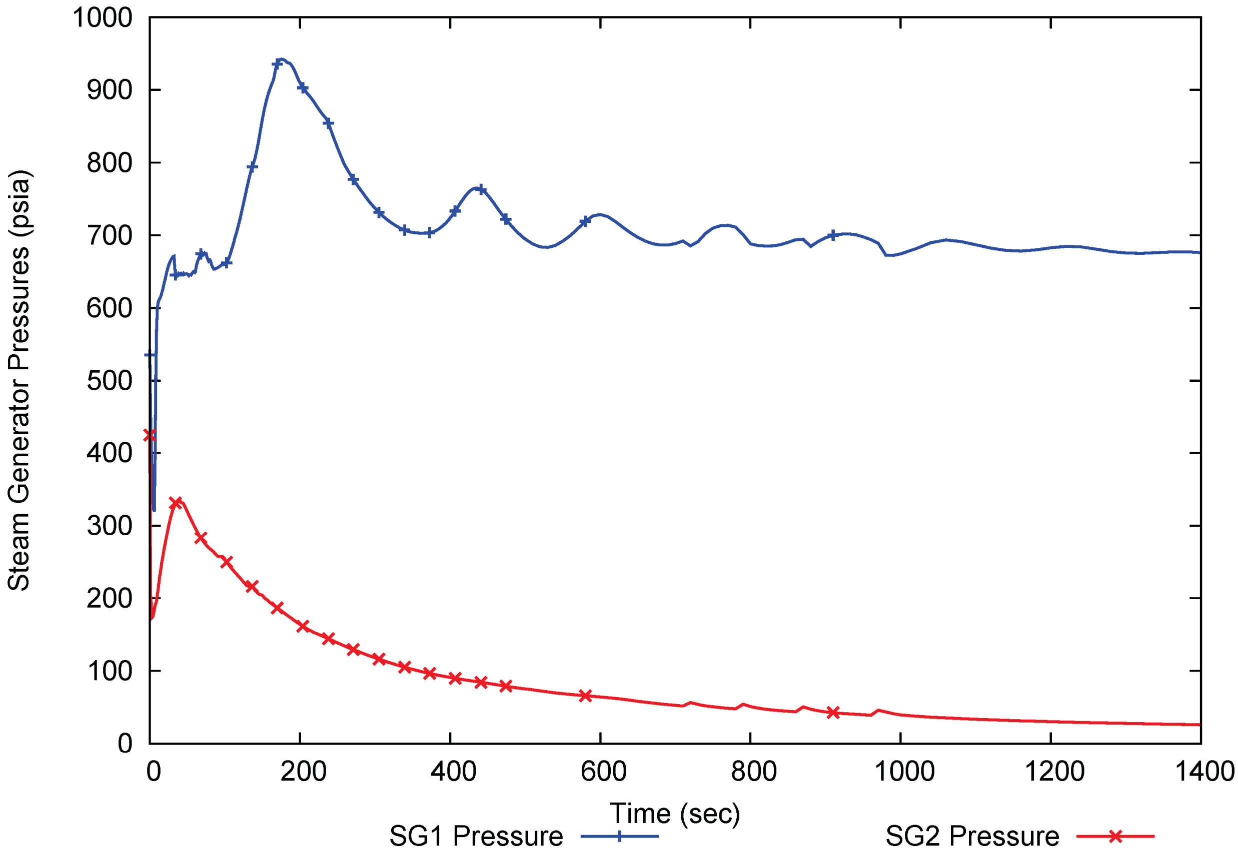

Figure 5-36 Main steam line break steam generator pressure

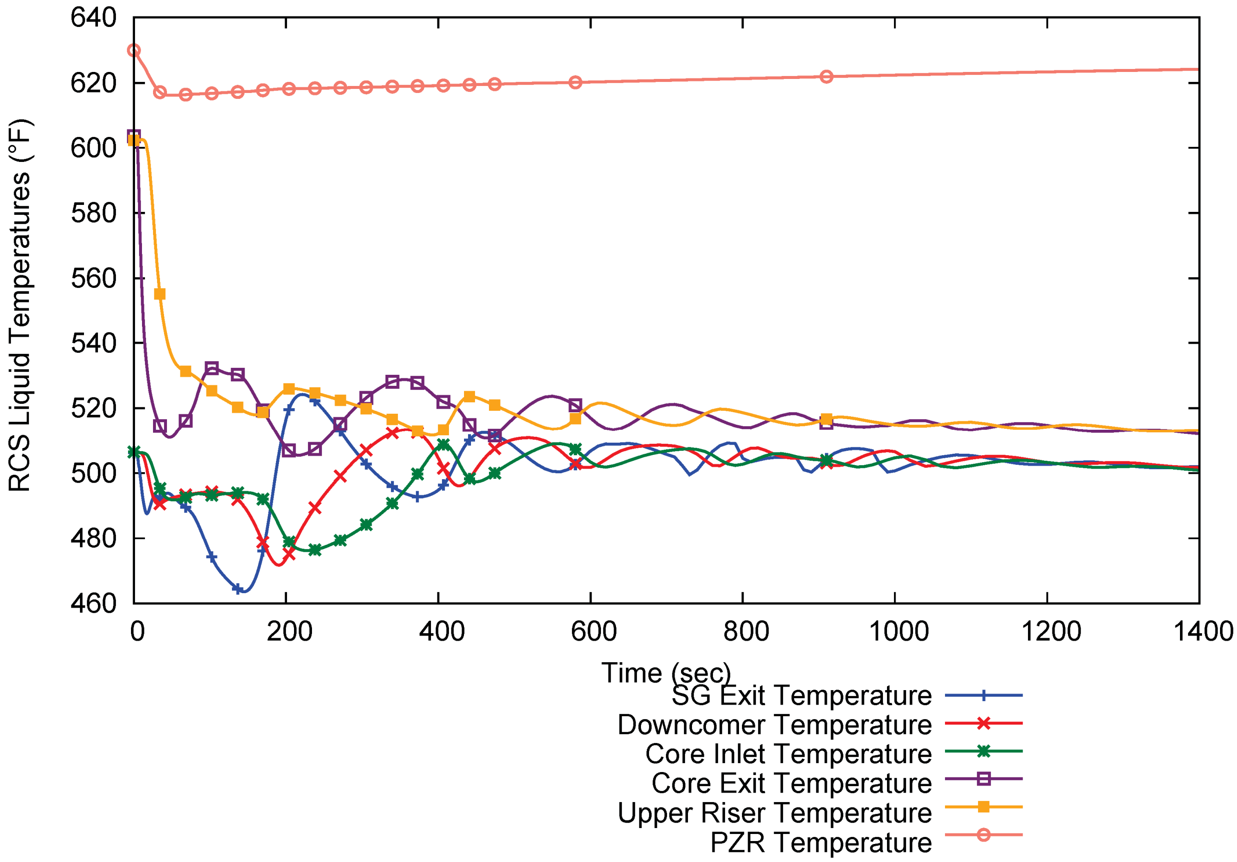

Figure 5-37 Main steam line break primary temperature

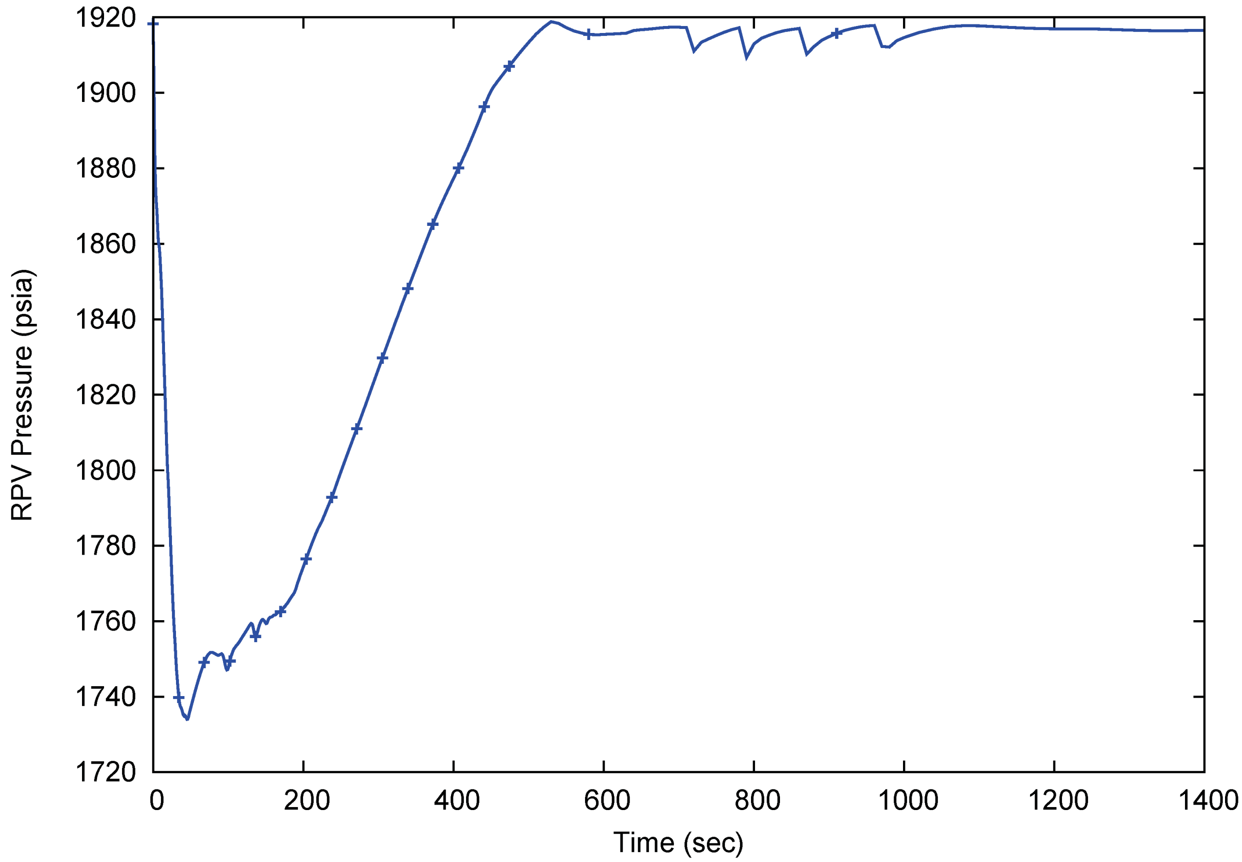

Figure 5-38 Main steam line break primary system pressure

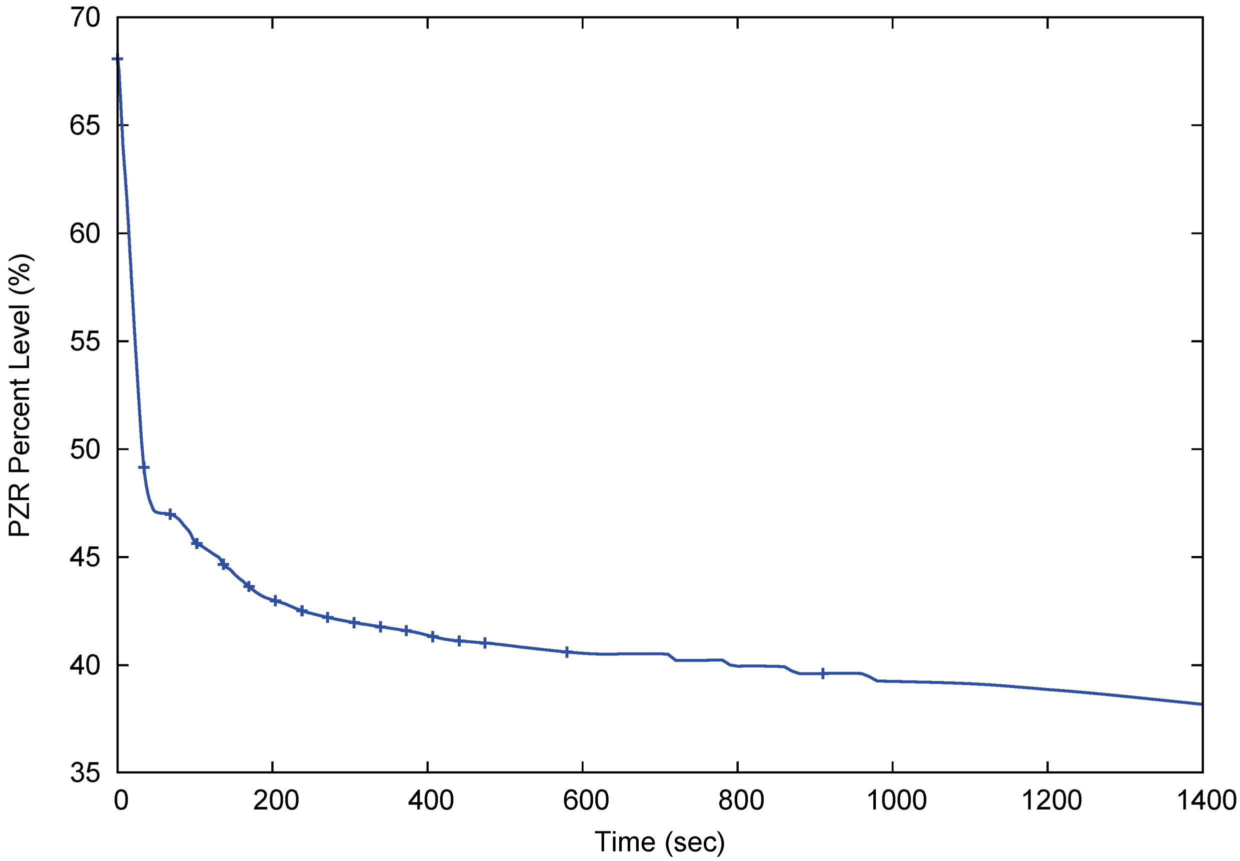

Figure 5-39 Main steam line break pressurizer level

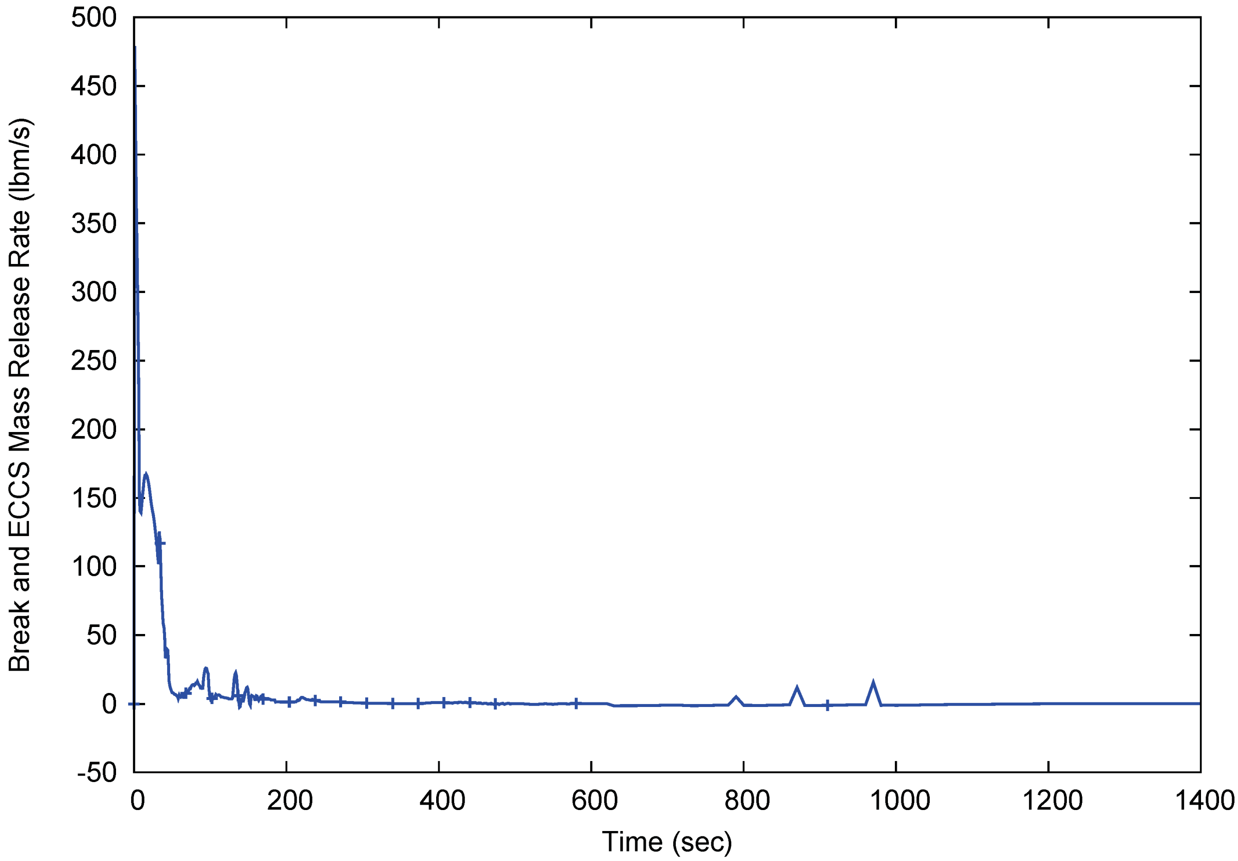

Figure 5-40 Main steam line break and emergency core cooling system flowrate

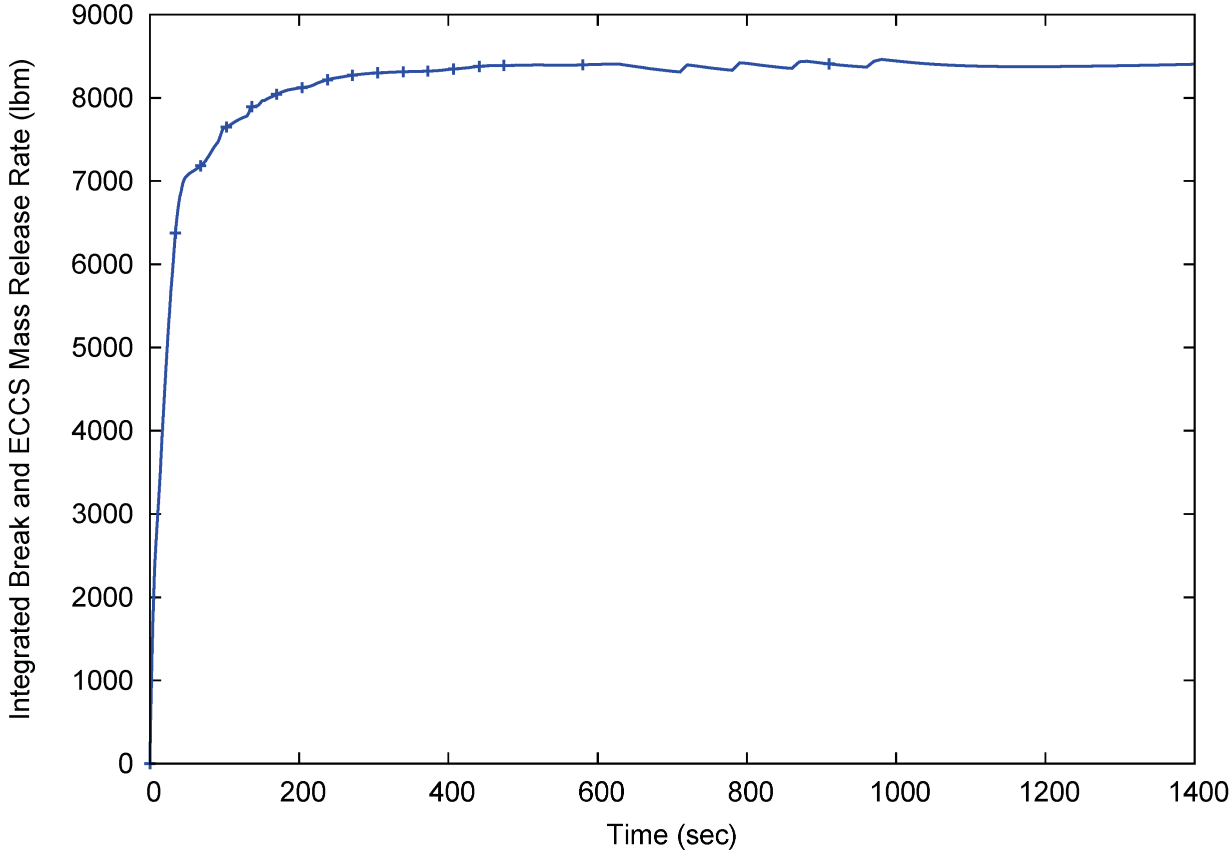

Figure 5-41 Main steam line break and emergency core cooling system integrated mass release

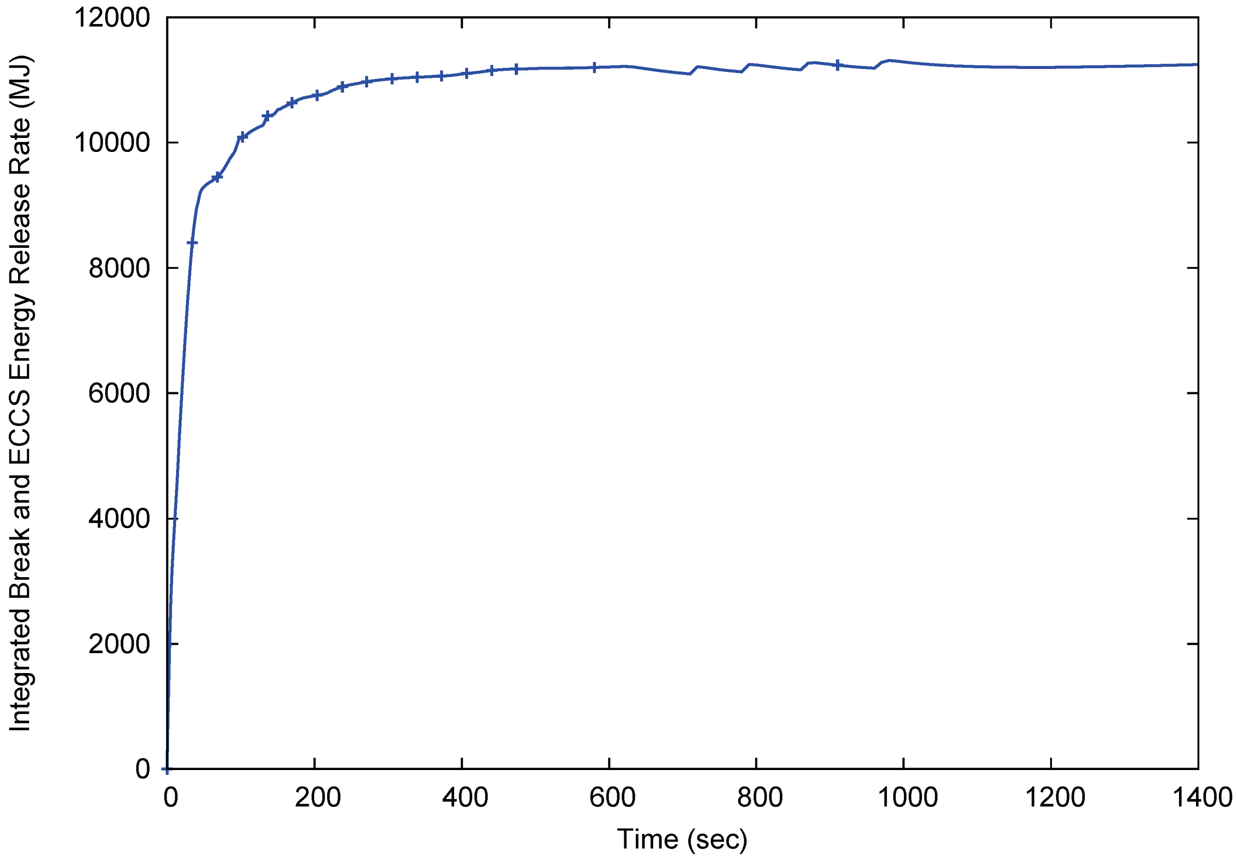

Figure 5-42 Main steam line break integrated energy release

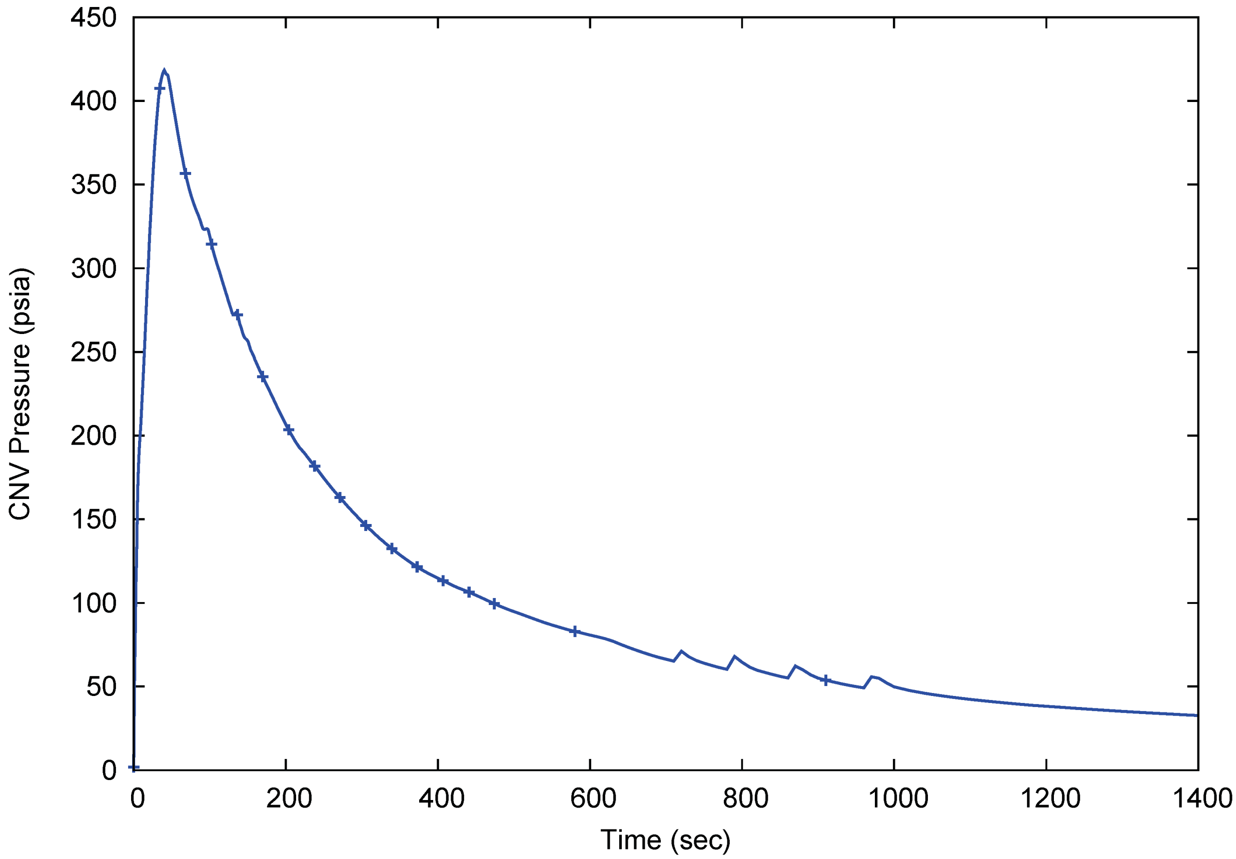

Figure 5-43 Main steam line break containment vessel pressure

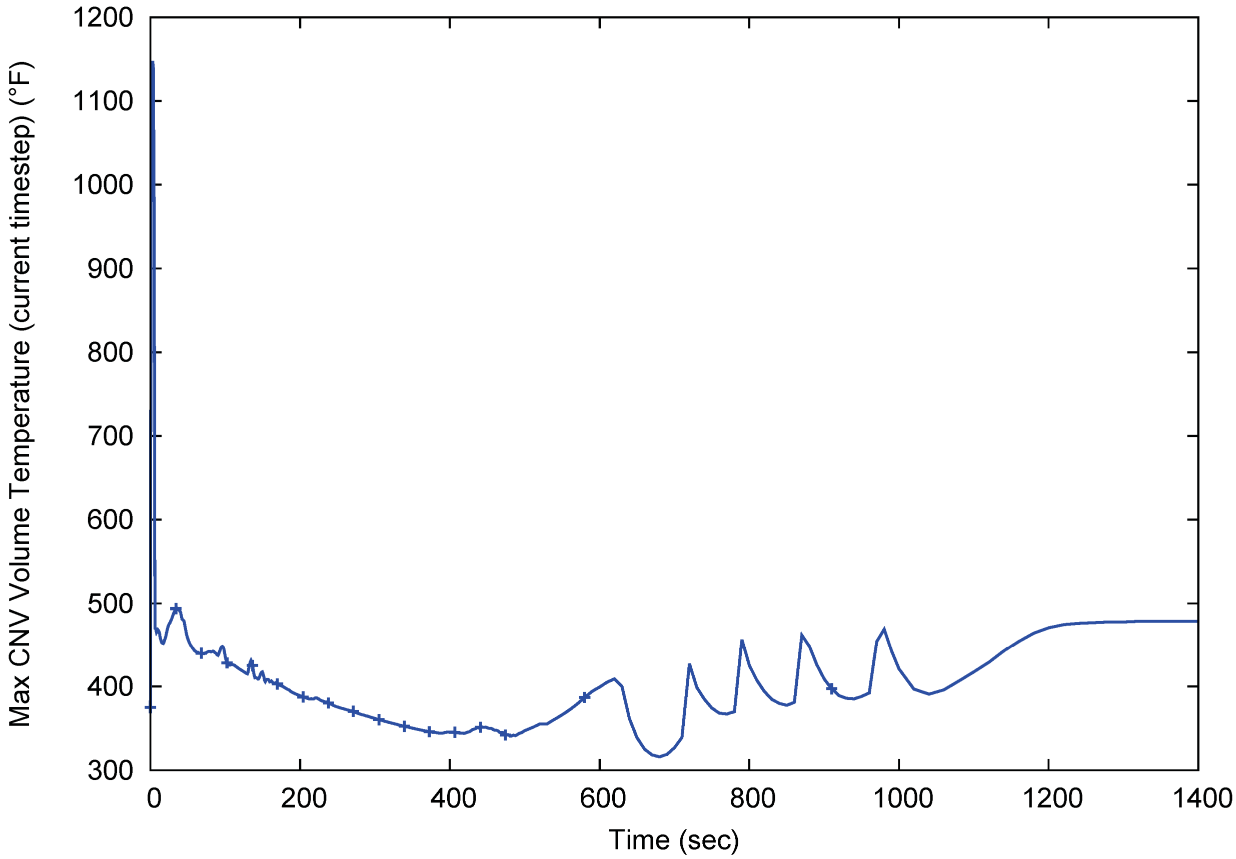

Figure 5-44 Main steam line break containment vessel vapor temperature

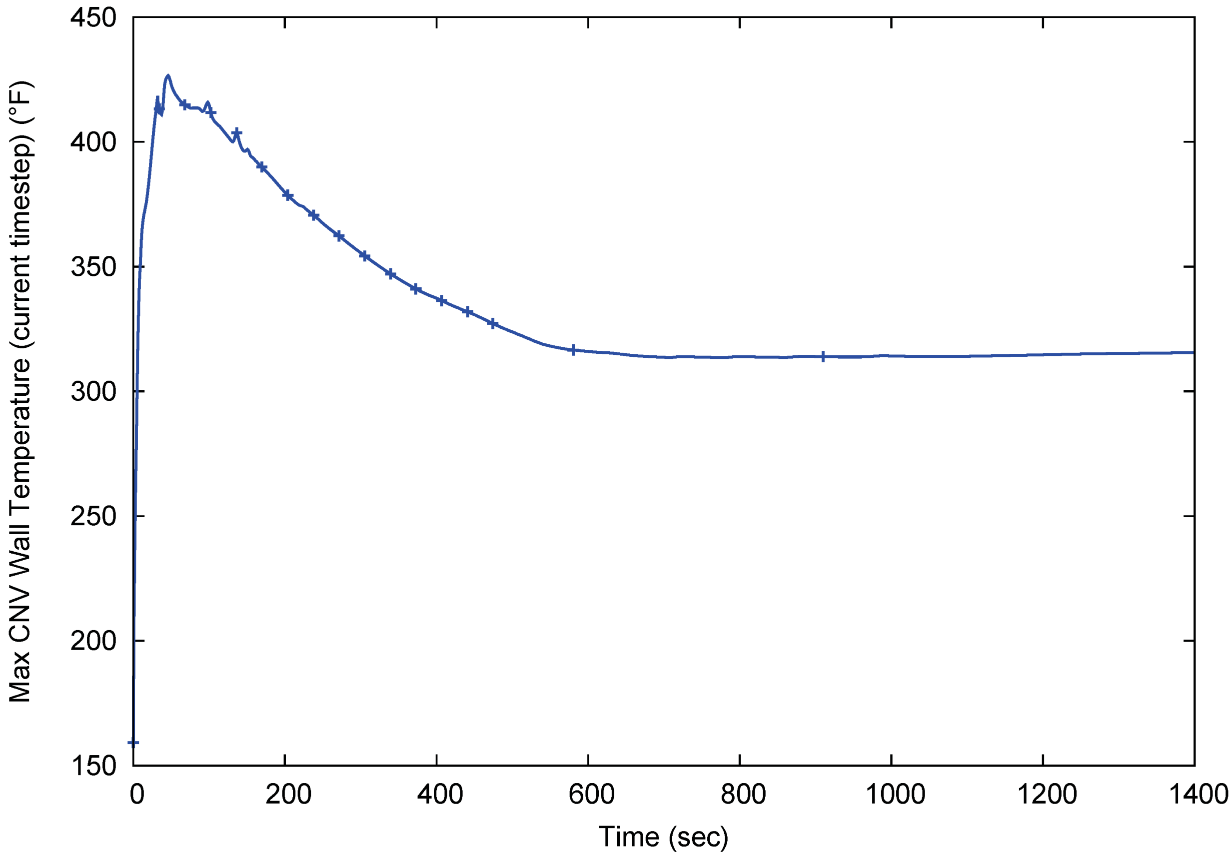

Figure 5-45 Main steam line break containment vessel wall temperature

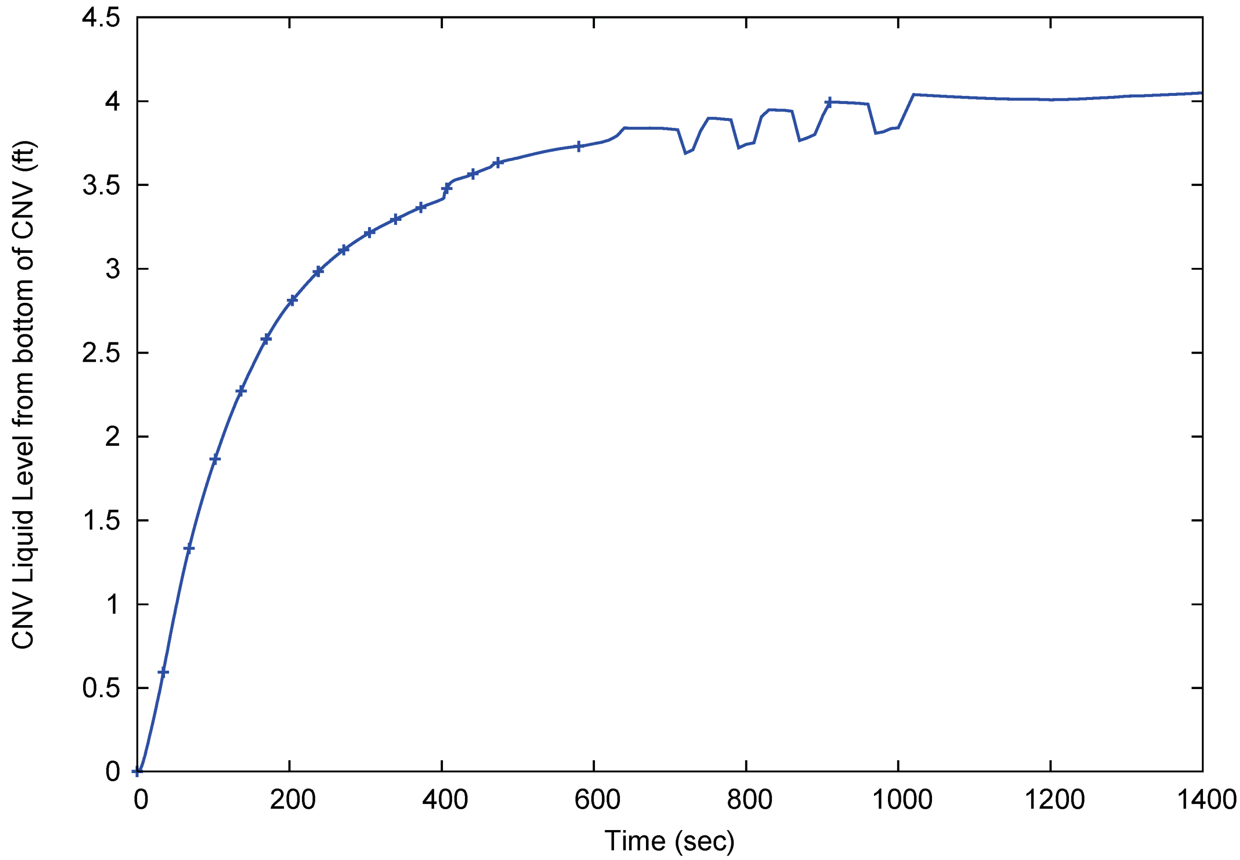

Figure 5-46 Main steam line break containment vessel level

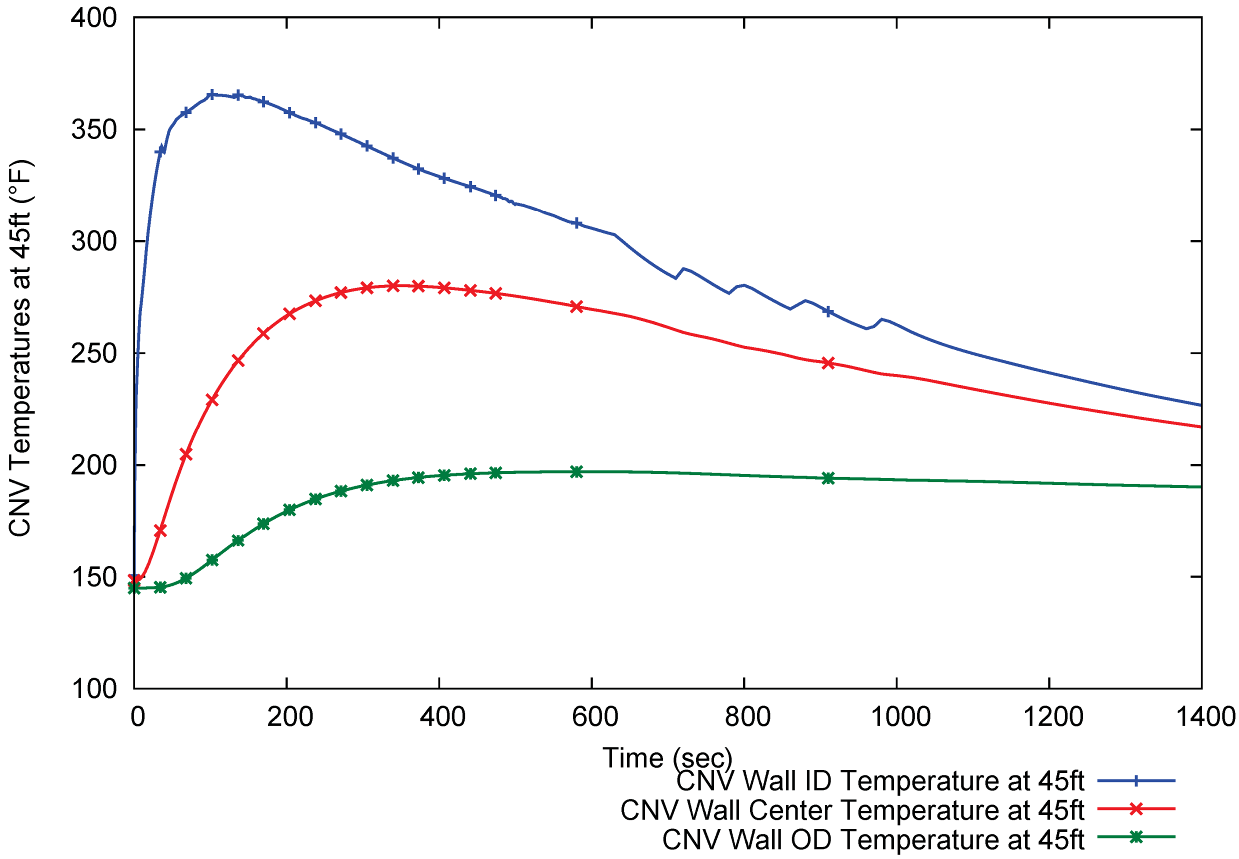

Figure 5-47 Main steam line break containment vessel wall temperature profile

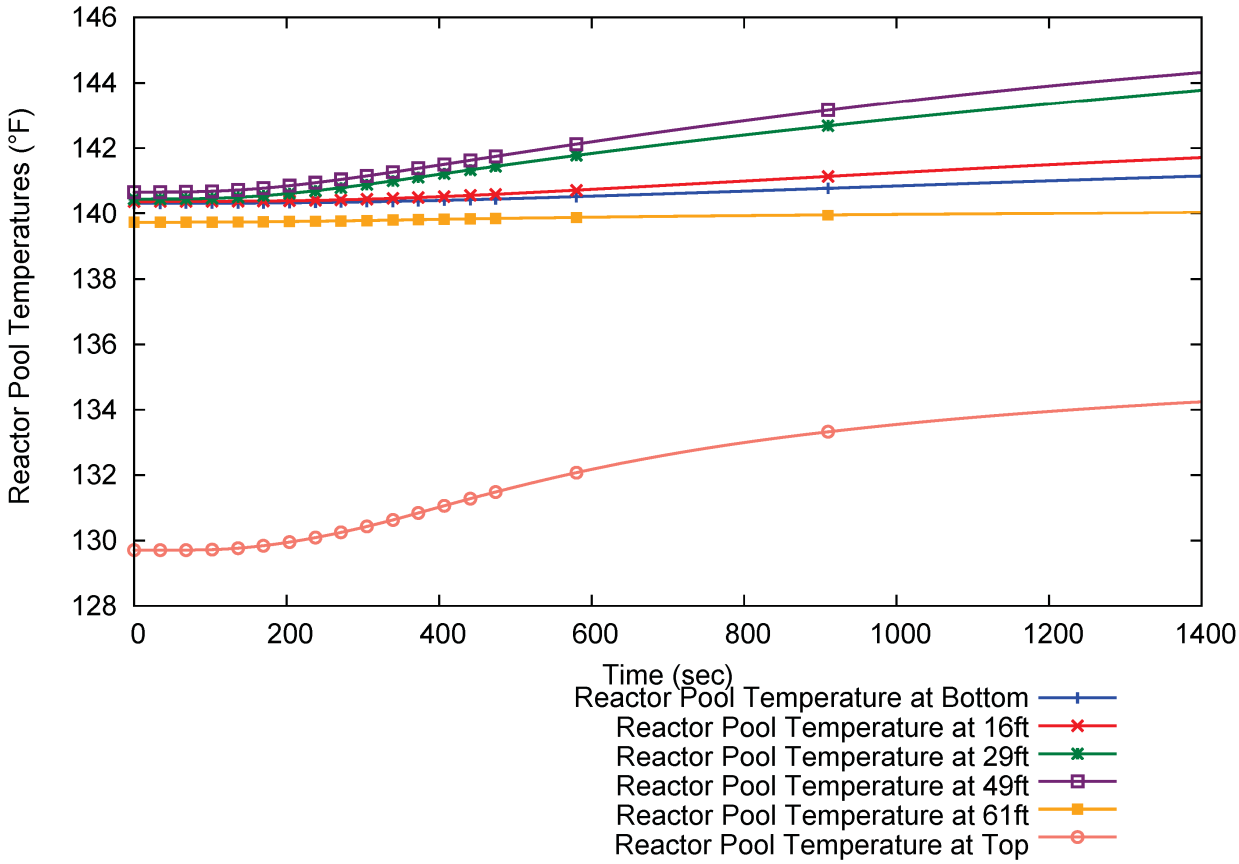

Figure 5-48 Main steam line break reactor pool temperature

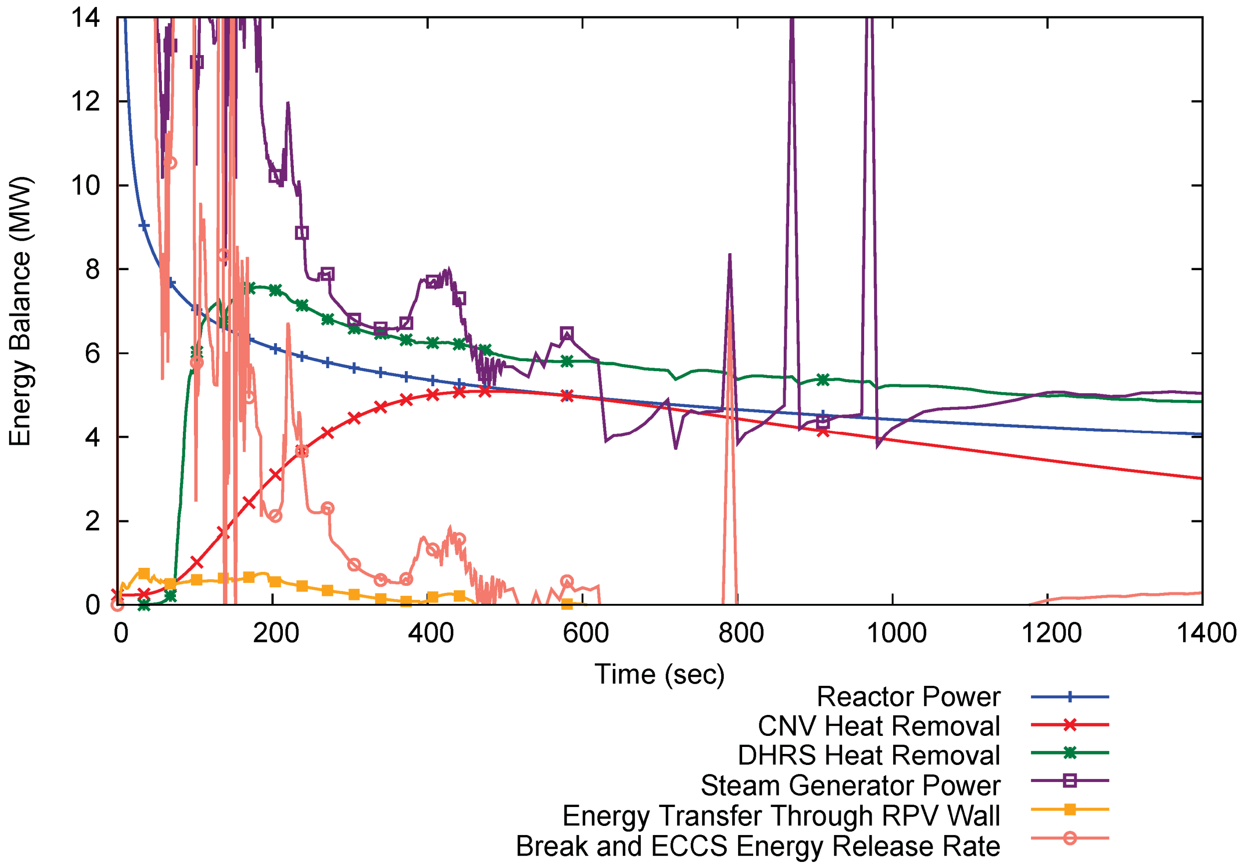

Figure 5-49 Main steam line break energy balance

5.3 Feedwater Line Break Pressure and Temperature Results

The sequence of events (Table 5-8) show that in the first seconds following the occurrence of an FWLB many automatic responses occur to transition the module from full power operation to an alignment that mitigates the initial secondary system blowdown phase. The break flow into the CNV causes a rapid pressurization that reaches the 9.5 psia high pressure setpoint. The following automatic actions occur on high CNV pressure:

- containment isolation including MSIV closure and FWIV closure
- DHRS actuation
- reactor trip
- turbine trip

As a conservative assumption a loss of normal AC and DC power is also assumed to occur at the time of turbine trip. This results in an ECCS signal. However, opening of the emergency core cooling system RRVs and RVVs does not occur until the pressure

differential decreases to below the IAB release pressure. In the containment response analysis methodology the IAB release pressure is an important analysis input as it determines the second M&E release into the CNV via the RVVs. A higher IAB release pressure results in an earlier opening of the ECCS valves when the RCS is hotter. The peak CNV pressure and peak CNV wall temperature occur following the RVV actuation, after the CNV has been preheated by the initial M&E release. Sensitivity studies of single failures have determined that a failure of a FWIV to close had an adverse impact on the CNV peak pressure and temperature results.

Following the alignment of the module to mitigate the initial secondary blowdown phase, the secondary system pressure and inventory decrease due to the loss of inventory. The CNV pressurizes and the steam condenses on the cold ID of the CNV. The condensate flows down the CNV walls and accumulates with unflashed secondary break liquid in a pool in the CNV lower head. The cold CNV wall absorbs the energy of the condensed steam and starts to heat up by conduction. Eventually the energy is transferred through the CNV wall to the reactor pool, and the pool temperature slowly increases. After the end of the secondary blowdown phase decay heat removal is via the DHRS. Opening of the ECCS valves occurs at 12,012 seconds when the pressure differential decreases to below the 1200 psid IAB release pressure. This causes the peak CNV pressure (442 psia) and the peak CNV wall temperature (414 degrees F) at ~12,058 and ~12,200 seconds, respectively. As flow through the RVVs dimishes, the primary system and CNV pressures converge, and continued heat transfer to the CNV leads to a gradual cooldown and depressurization phase. Pressure equalization enables recirculation flow from the CNV pool through the RRVs to establish the long-term cooling recirculation alignment.

The module response for the FWLB analysis is shown in Figure 5-50 through Figure 5-64. Figure 5-50 shows the SG pressure response with the affected SG (SG2) depressurizing via blowdown out the break into the CNV and stabilizing at a pressure of approximately 30 psia. The unaffected SG (SG1) pressure fluctuates in response to DHRS heat transfer. The affected SG repressurizes by reverse break flow on ECCS valve opening. Then, both SGs depressurize as ECCS heat transfer dominates. Figure 5-51 shows the gradual primary system cooldown due to DHRS, and the increase in the cooldown rate with the opening of the ECCS valves. Figure 5-52 shows the relatively steady pressurizer level decrease during DHRS cooling and then a rapid level decrease when ECCS valves open. Figure 5-53 shows the riser level remaining full until the ECCS valves open, and then level rapidly decreases before stabilizing. Primary system pressure (Figure 5-54) gradually decreases during the DHRS cooldown period due to loss of pressurizer heaters and then rapidly depressurizes on ECCS valves opening. Figure 5-55 through Figure 5-57 show the break and ECCS mass release, the integrated mass release, and the integrated energy release into the CNV, respectively. The FWLB flow rate and integrated mass release is not significant due to the small SG inventory. Due to the insignificance of the secondary break flow, the effect of liquid entrainment is also insignificant. The primary system M&E release through the three RVVs is the significant M&E release event for the FWLB accident.

The CNV and reactor pool responses for the FWLB are shown in Figure 5-58 through Figure 5-63. Figure 5-58 shows the CNV pressure response. The initial M&E release results in the CNV pressurizing to ~60 psia before heat transfer to the CNV wall results in pressure stabilizing at ~15 psia. Then pressure rapidly increases to the limiting peak value

of 442 psia following opening of the RVVs. This limiting NRELAP5 result can be compared to the CNV design pressure of 1000 psia and to the limiting MSLB and primary release event results. The FWLB peak CNV pressure result is higher than the MSLB result, but is bounded by the limiting LOCA results. This is a key result in this FWLB containment response analysis.

Figure 5-59 shows the CNV vapor temperature. {{

}}2(a),(c) Figure 5-60 shows the peak CNV wall temperature and the limiting value of 414 degrees F. This limiting NRELAP5 result can be compared to the CNV design temperature of 550 degrees F, and to the limiting MSLB and LOCA results. The FWLB is bounded by both the MSLB result and the limiting primary release event results. This is a key result in this FWLB containment response analysis.

Figure 5-61 shows the CNV level response with an initial level increase following the initial M&E release, and the second level increase following the delayed opening of the ECCS valves. Figure 5-62 shows the temperature profile across the CNV wall at the 45 foot elevation. A significant temperature gradient exists. Figure 5-63 shows the reactor pool temperature for a range of elevations. Clearly the reactor pool temperature does not increase significantly through the time of peak CNV pressure and temperature. Even with the conservative initial reactor pool level of 55 ft above the pool floor and a temperature of 140 degrees F assumed in these analyses, the CNV wall is capable of maintaining the peak CNV pressure and temperature within the design limit.

Figure 5-64 shows the energy balance during the FWLB and the trends of the heat sources and sinks. The DHRS and CNV wall heat sinks combine to exceed the ECCS energy release and results in a sustained cooldown of the primary system as shown in Figure 5-51.

Table 5-8 Feedwater line break sequence of events

Time (sec)	Event
0	FWLB

0 – 2.6	High CNV pressure resulting in
	• Containment isolation
	  ○ MSIV closure
	  ○ FWIV closure
	  ○ DHRS actuation
	• Reactor trip
	• Turbine trip
	• Loss of normal AC and DC power on turbine trip resulting in ECCS actuation signal

10	Peak CNV pressure from secondary M&E release
12,012	ECCS valve opening on differential pressure below IAB release pressure (<1200psid)
~12,058	Peak CNV pressure
~12,200	Peak CNV temperature
~13,000	CNV pressure decreases to <50% of peak pressure

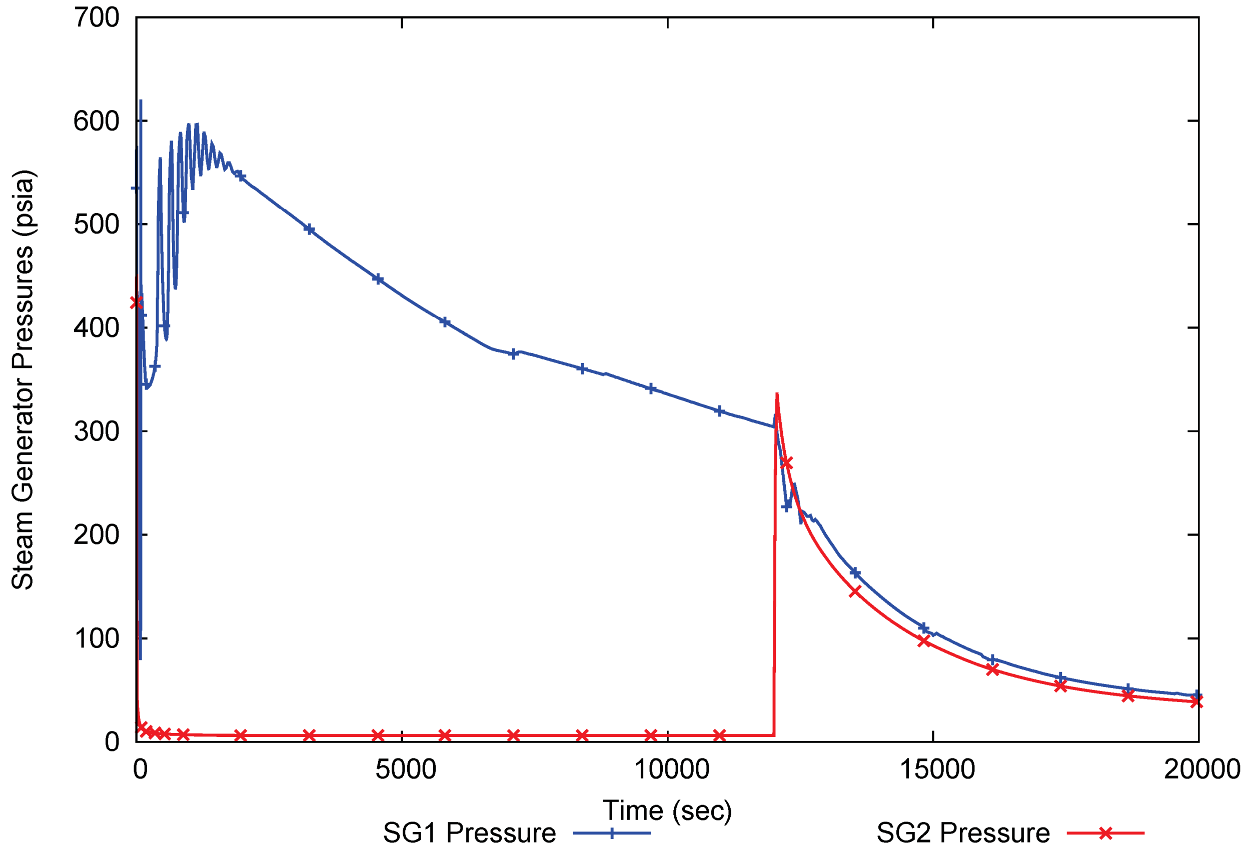

Figure 5-50 Feedwater line break steam generator pressure

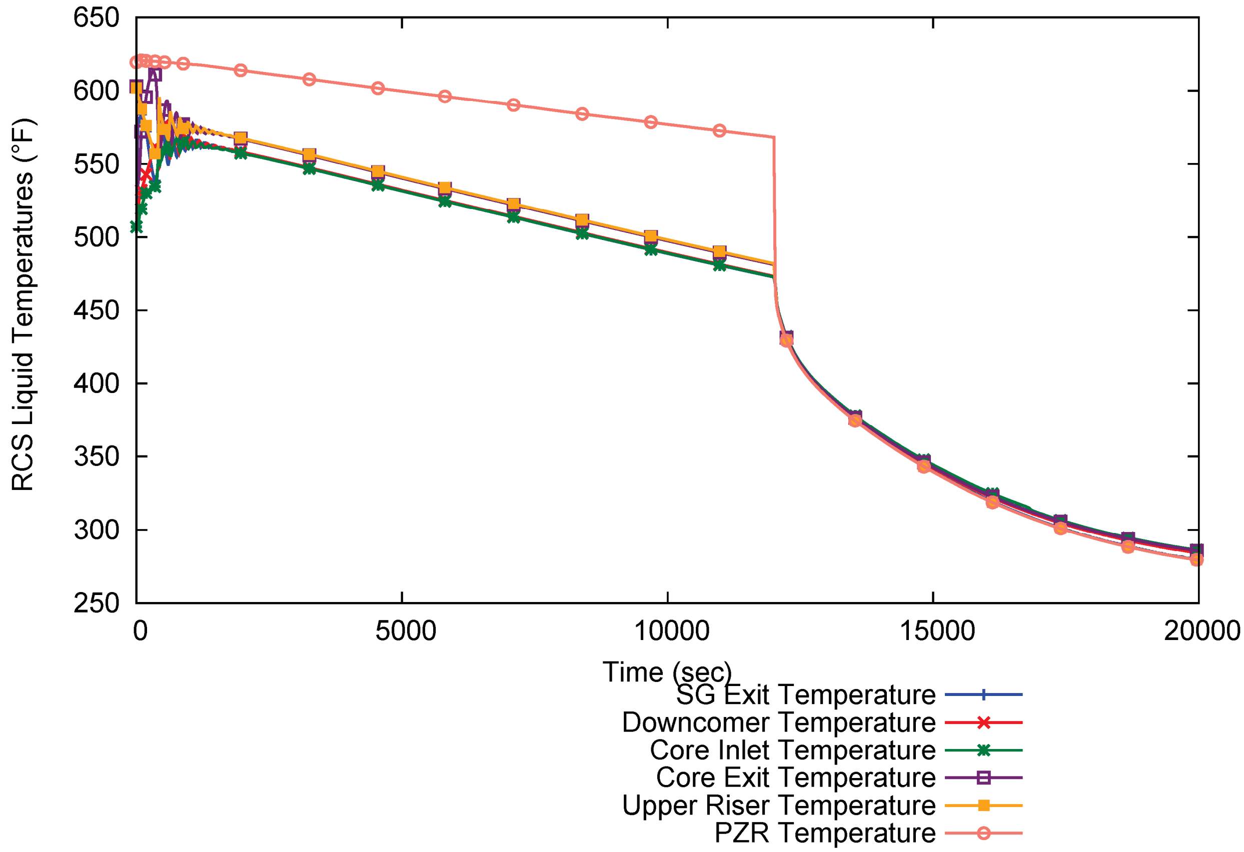

Figure 5-51 Feedwater line break primary temperature

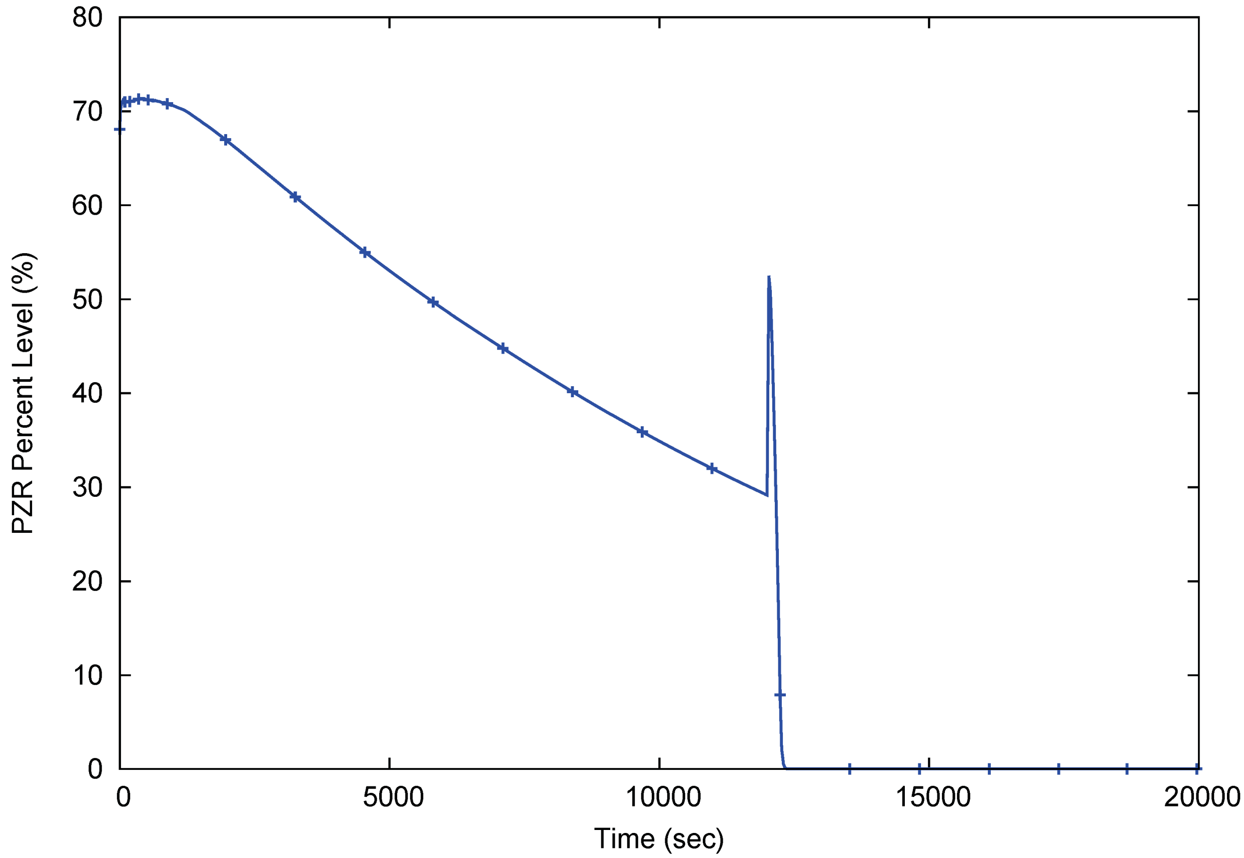

Figure 5-52 Feedwater line break pressurizer level

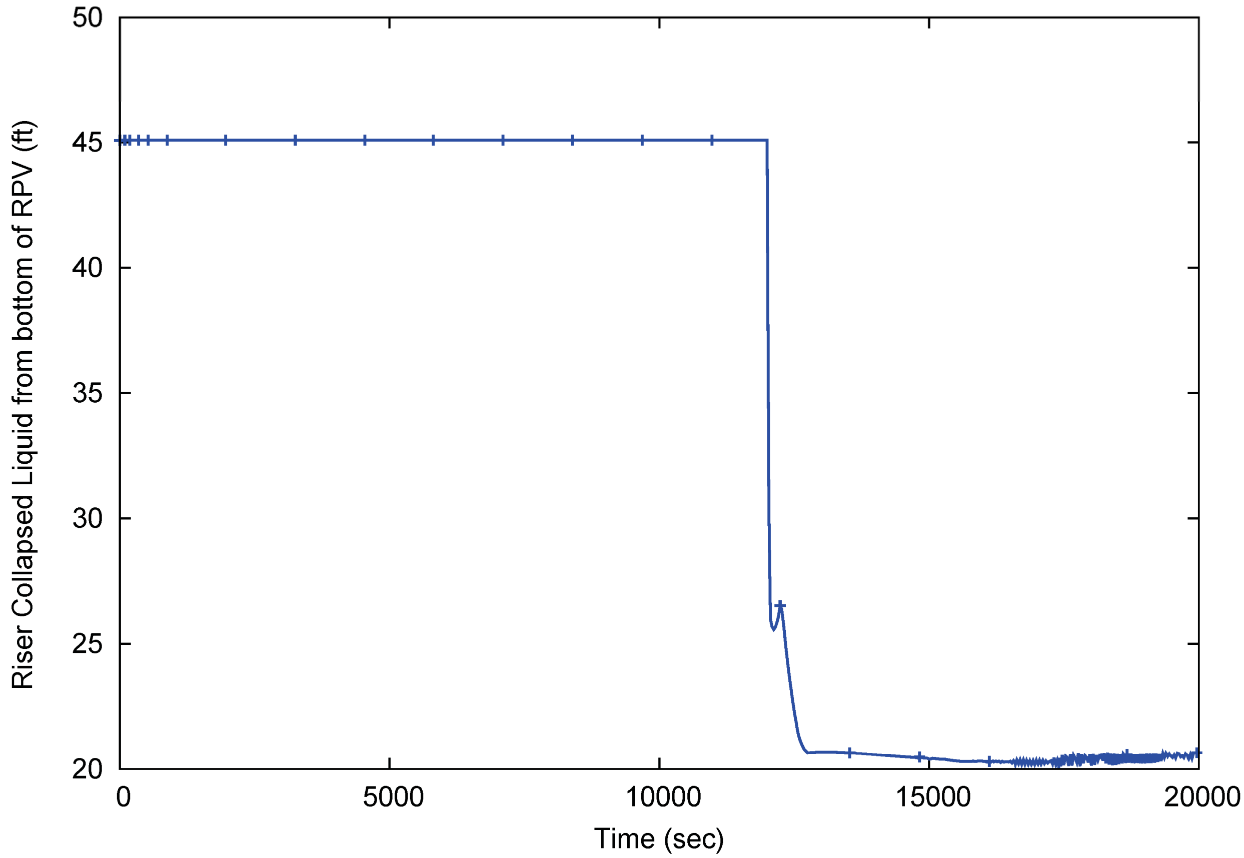

Figure 5-53 Feedwater line break riser level

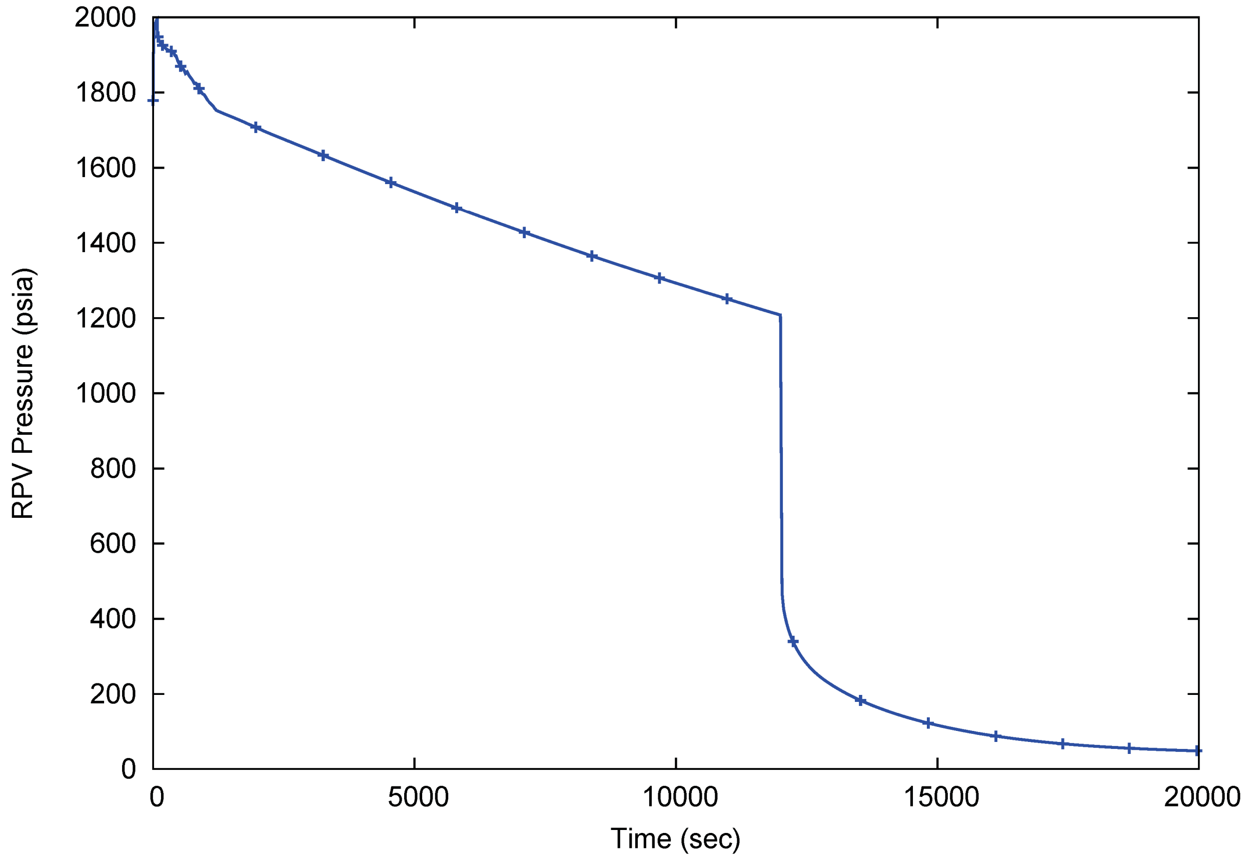

Figure 5-54 Feedwater line break primary system pressure

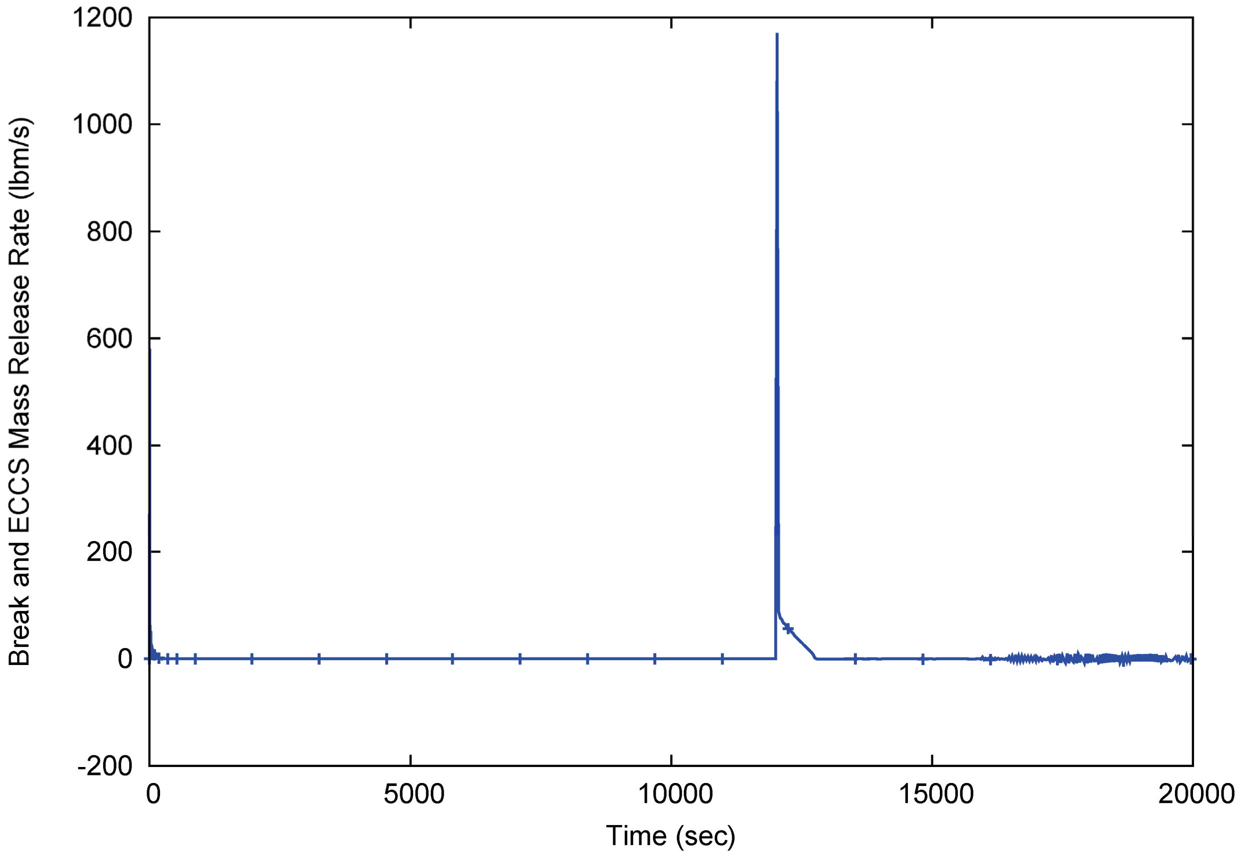

Figure 5-55 Feedwater line break and emergency core cooling system flowrate

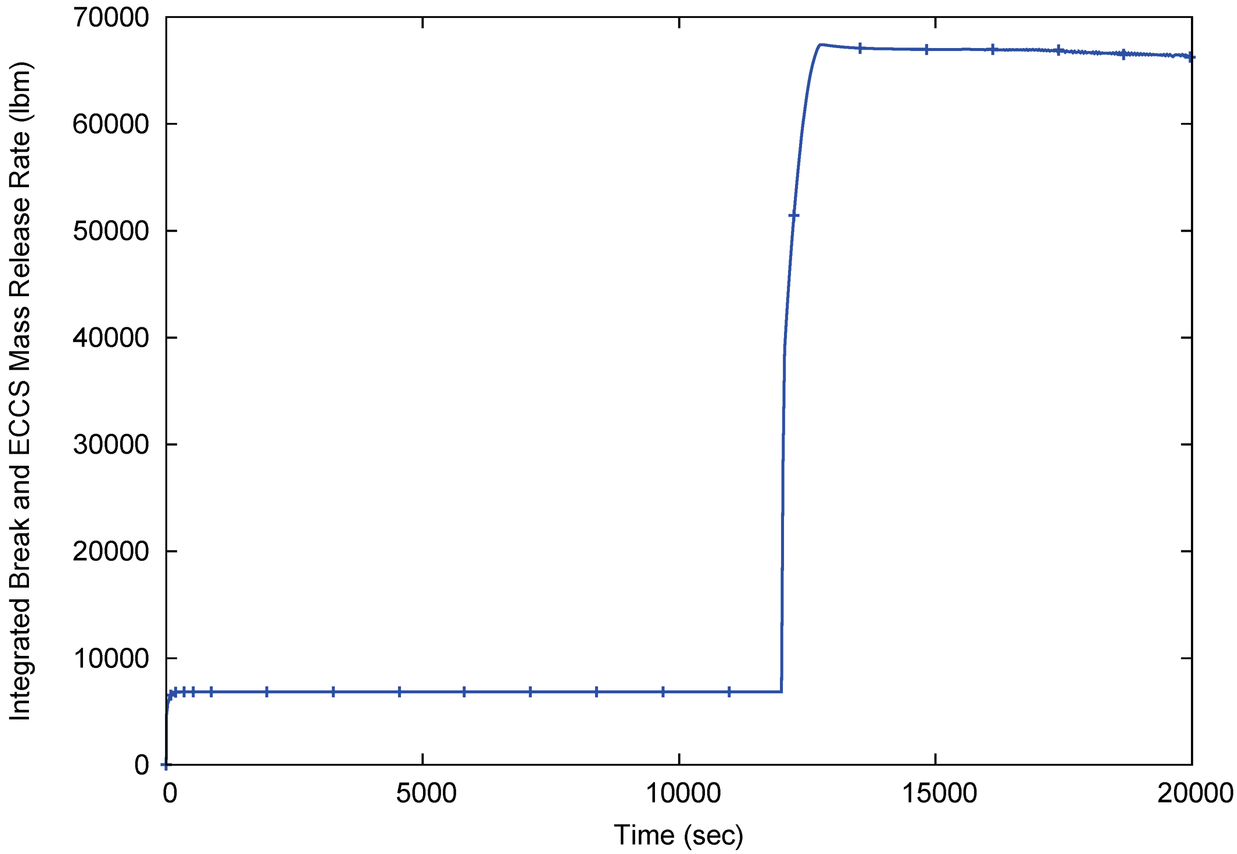

Figure 5-56 Feedwater line break and ECCS integrated mass release

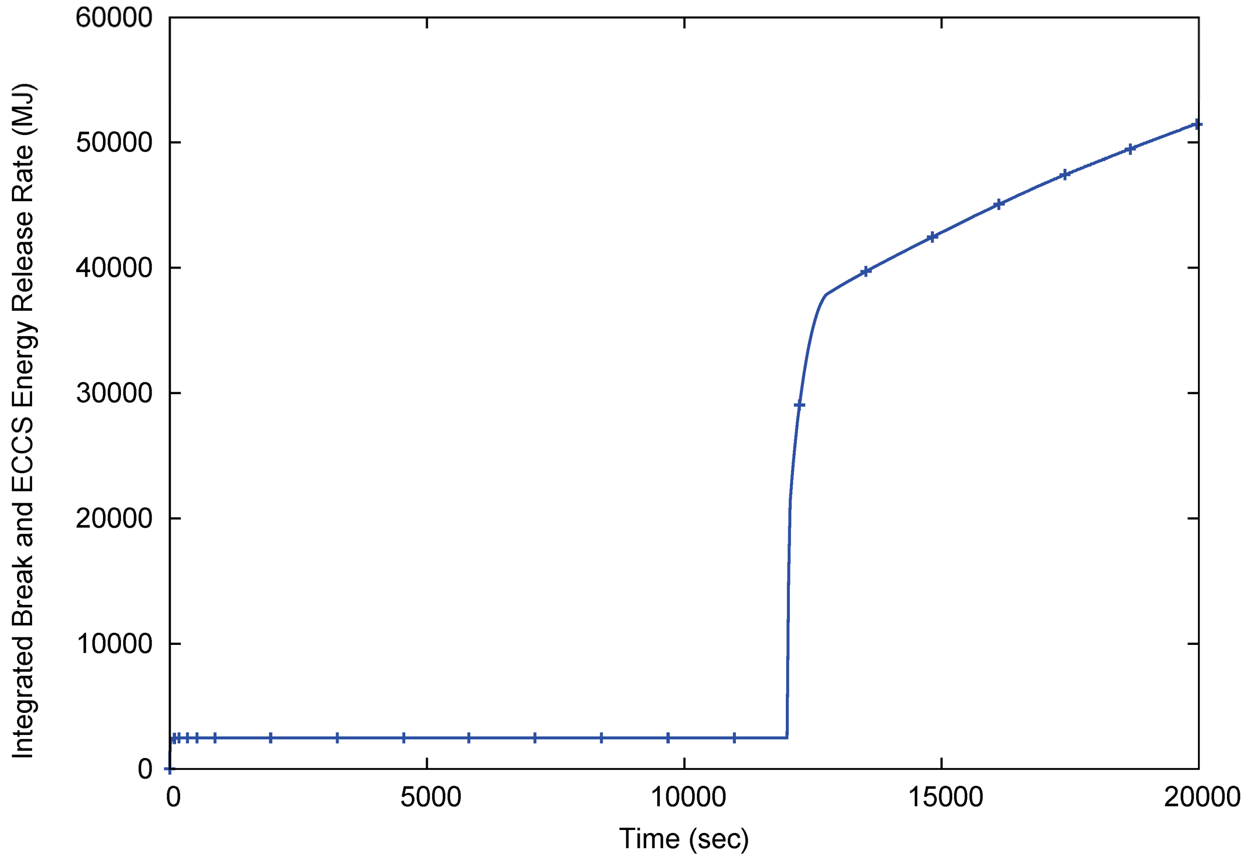

Figure 5-57 Feedwater line break and ECCS integrated energy release

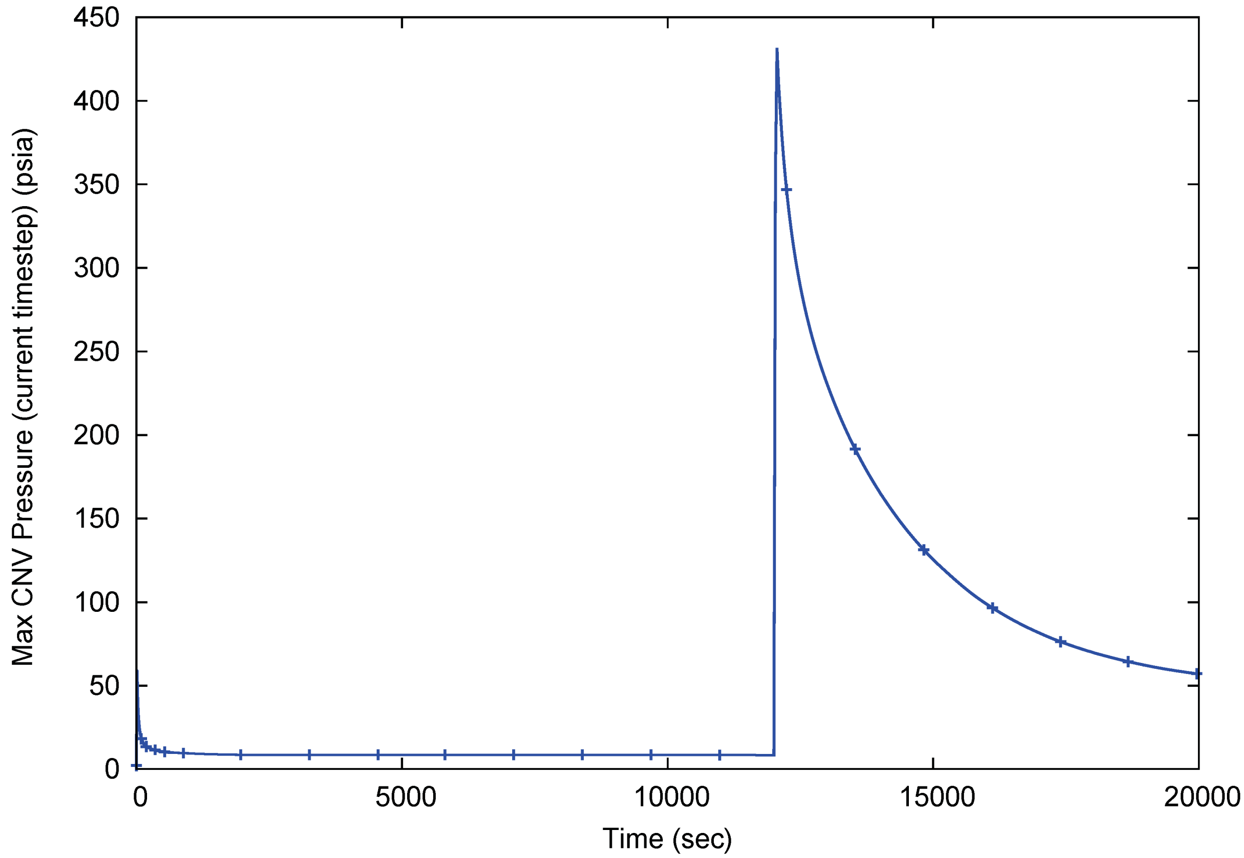

Figure 5-58 Feedwater line break containment vessel pressure

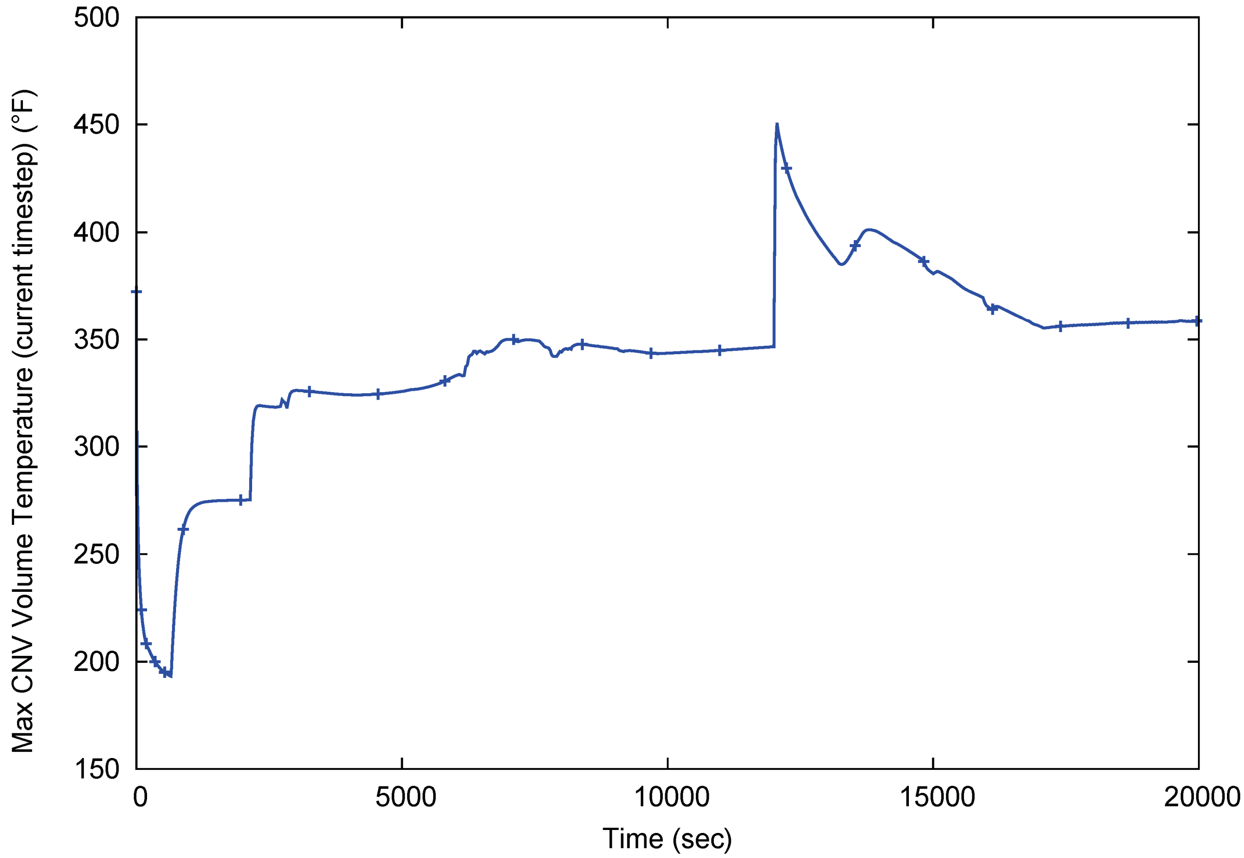

Figure 5-59 Feedwater line break containment vessel vapor temperature

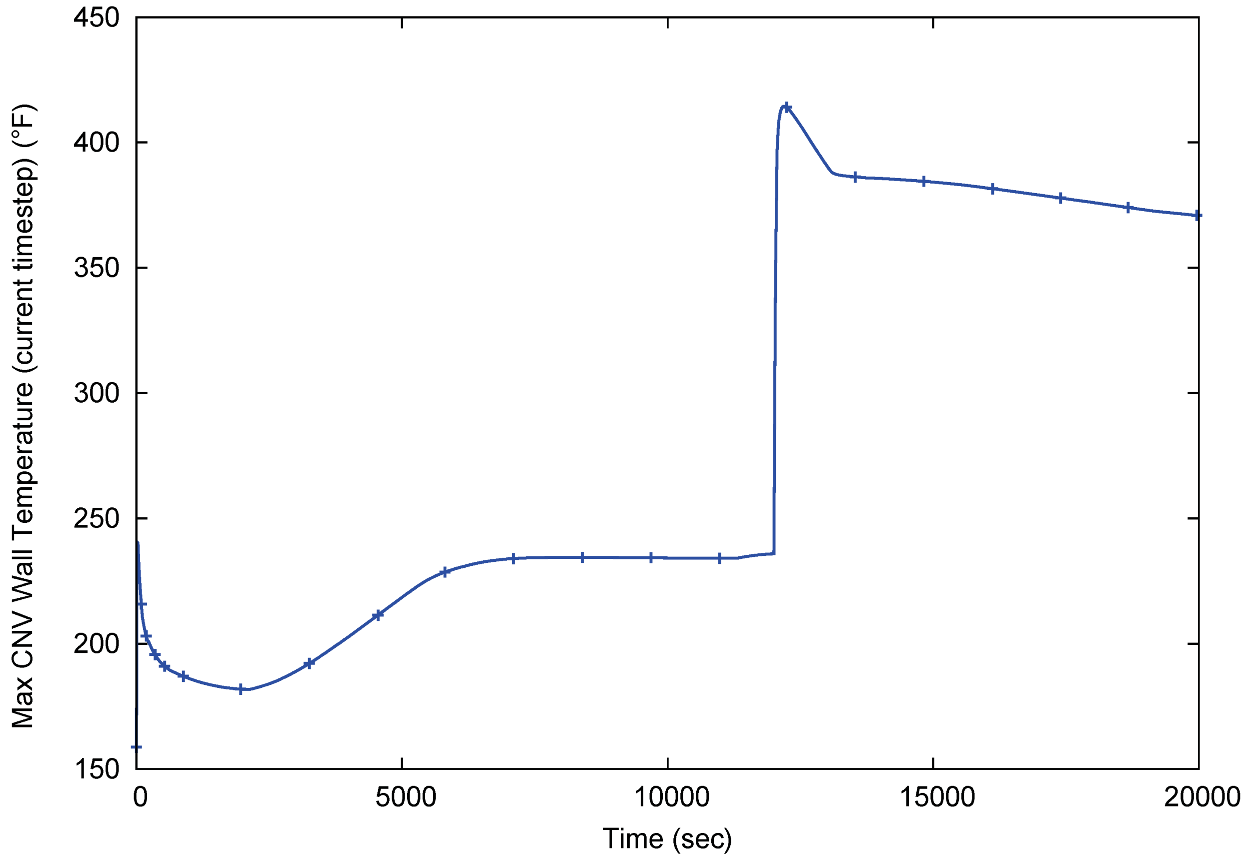

Figure 5-60 Feedwater line break containment vessel wall temperature

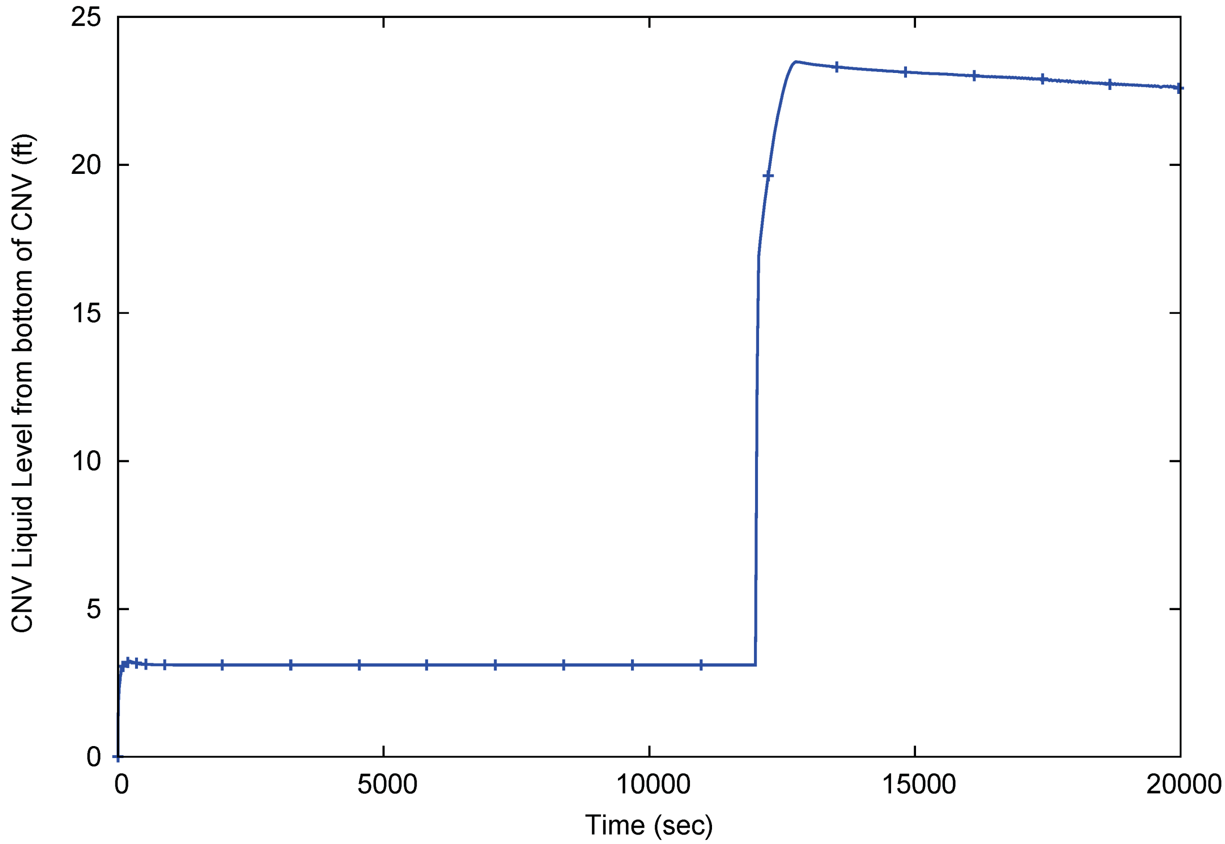

Figure 5-61 Feedwater line break containment vessel level

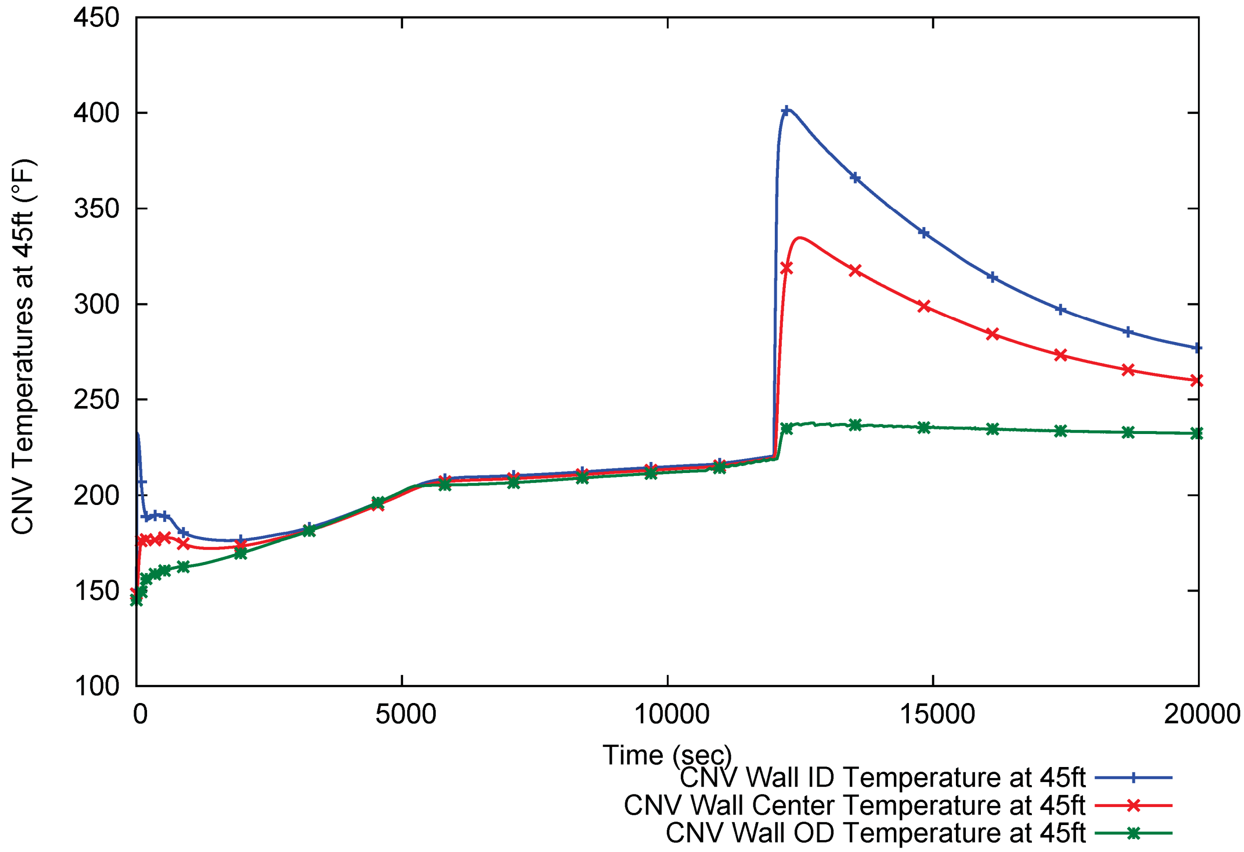

Figure 5-62 Feedwater line break containment vessel wall temperature profile

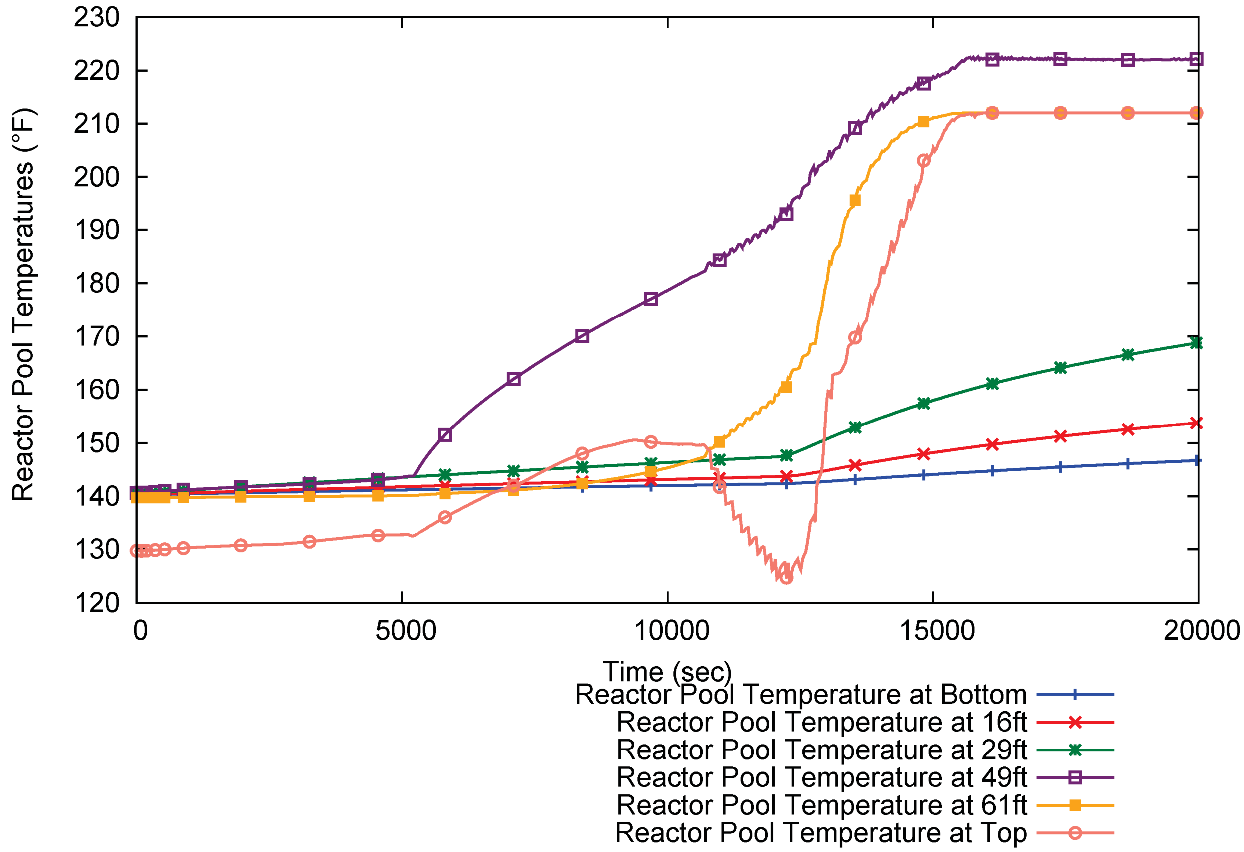

Figure 5-63 Feedwater line break reactor pool temperature

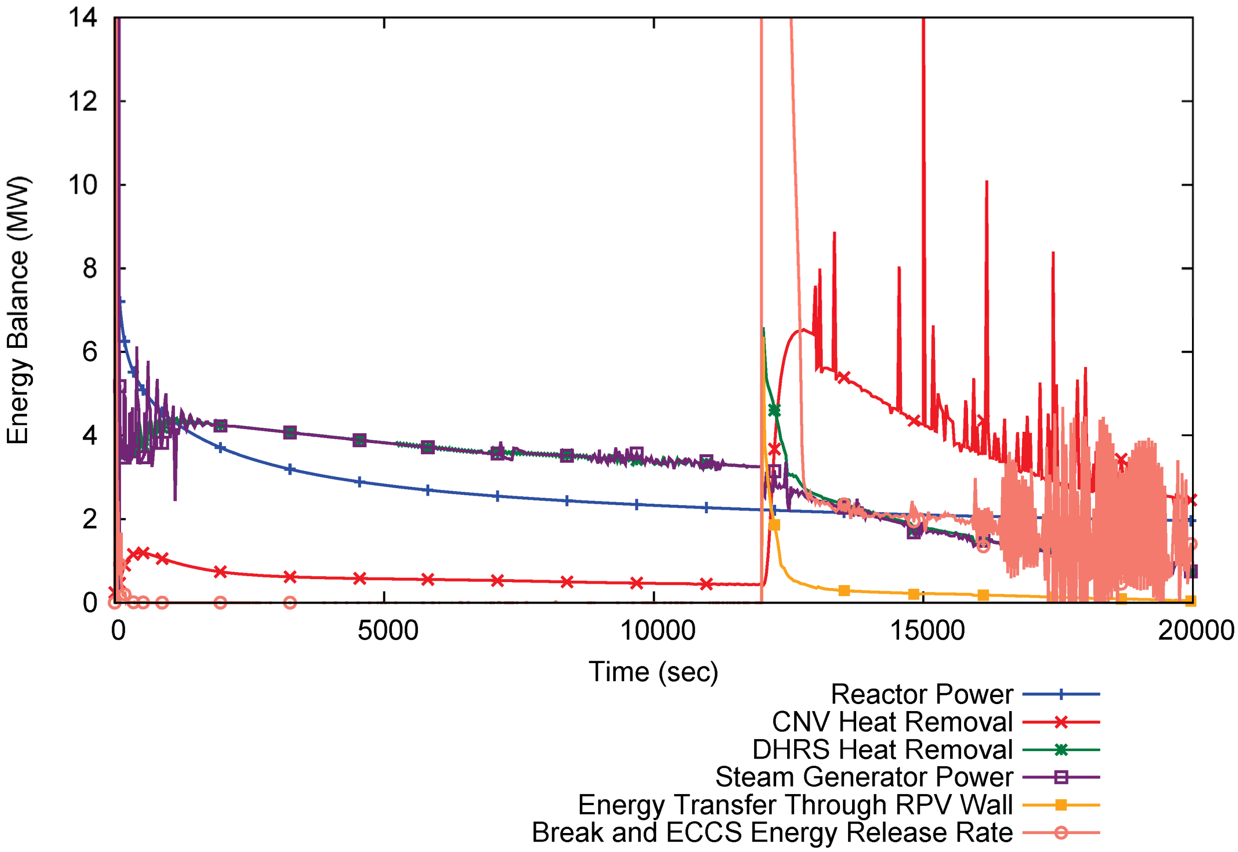

Figure 5-64 Feedwater line break energy balance

5.4 Margin Assessment

The following subsections discuss the analytical and design margin incorporated into the NPM design. Section 5.4.1 describes margin inherent in the enhanced requirements imposed on the CNV as an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Class 1 vessel. Section 5.4.2 describes conservative modeling assumptions in the containment peak pressure and temperature analysis, along with the results of a nominal case quantifying the conservatism resulting from some of these assumptions.

5.4.1 Containment Vessel Design Pressure Margin

The CNV consists of an upper section and a lower section joined at the closure flange about one third of the way up the vessel. The CNV upper section is made of SA-508 Grade 3, Class 2 low alloy steel with 0.125-in. stainless steel cladding on the inside surfaces and 0.250-in. stainless steel cladding on the outside surface. The top head of the upper section is a torispherical head and the vessel shell has an OD of 176.75 in. and a base metal wall thickness of 3.00 in. The CNV lower section is made of SA-965 FXM-19 austenitic stainless steel and does not require any cladded surfaces. The bottom head on the lower section is a torispherical head and the vessel shell consists of three regions. The bottom

region, in the area of the fuel, has an OD of 134.50 in. and a wall thickness of 3.00 in., the middle region is a transition between the diameters of the bottom and top region. The top region has an OD of 176.50 in. with a wall thickness of 3.250 in. Containment vessel shell-to-shell, shell-to-head, and penetration-to-shell welds are high-quality, full-penetration welds performed in the vessel fabrication shop. Material properties of the structural portions of the CNV are summarized in Table 5-9.

Table 5-9 Containment vessel material properties

Material	**Material Property (ksi)**					
	S_m		**S_y**		**S_u**	
	140 F	550 F	140 F	550 F	140 F	550 F
SA-508 Grade 3, Class 2	30.0	30.0	62.5	55.4	90.0	90.0
SA-965 FXM-19	33.2	29.45	49.8	38.1	99.7	88.4

The CNV has a design pressure of 1,000 psia and design temperature of 550 degrees F. The limiting LOCA peak containment pressure is 921 psia and the overall peak containment pressure is 951 psia. The design pressure meets the requirements of ASME Boiler and Pressure Vessel Code (BPVC), Section III, Paragraph NCA-2142.1(a) and NB-3112.1(a) by bounding the most severe Level A service level pressure and ASME BPVC, Section III, Paragraph NE-7120(b) by not exceeding service limits specified in the design specification.

The CNV is designed, fabricated, inspected and tested as an ASME Code Class 1 component in accordance to ASME BPVC, Section III, Subsection NB. Pressure boundary forgings and weld filler materials are tested for mechanical and fracture toughness to the requirements of ASME BPVC, Section III, Article NB-2000. The CNV is a high-quality, shop-fabricated vessel, made to the requirements of ASME BPVC, Section III, Article NB-4000, with low-alloy welds post weld heat treated. Many requirements between an NB and MC vessel are similar. However, one substantive difference between an NB and MC class vessel is in weld inspection. The main welds forming the pressure boundary shell are Category A, B and C full penetration, butt welds. In an NB class vessel these welds are required to have a volumetric and either liquid penetrant or magnetic particle inspection performed per ASME BPVC, Section III, Subarticle NB-5200. The corresponding welds in an MC class vessel only require a fully radiographed inspection per ASME BPVC, Section III, Subarticle NE-5200. By only performing a radiograph examination of the weld, all potential flaws in the weld may not be detected, which reduces the quality and potentially the strength of the weld.

After fabrication of the CNV is completed, a shop hydrostatic test of the vessel is performed per ASME BPVC, Section III, Article NB-6000. Before hydrostatic testing, 100 percent of the pressure boundary welds are inspected. Inspection is performed to ASME BPVC, Section III, Subsubarticle NB-5280 and Subarticle IWB-2200 using examination methods of ASME BPVC Section V except as modified by ASME BPVC, Section III, Paragraph NB-5111. The hydrostatic test is done to a minimum pressure of 1,265 psia (125 percent) and a maximum pressure of 1,340 psia at a minimum temperature of 70

degrees F and a maximum temperature of 140 degrees F. The hydrostatic pressure and temperature is held for a minimum of 10 minutes. The pressure is then decreased to design pressure and held for a minimum of four hours and then inspected for leaks. After the test is completed, the pressure boundary welds are inspected again to the same requirements used before the test. The NB-6000 hydrostatic test is performed to a greater pressure then required by NE-6000. NE-6321 specifies a minimum test pressure of only 110 percent and NE-6322 a maximum of 116 percent. The NuScale CNV is tested to a pressure margin 15 percent greater than conventional steel containment structures.

The ASME BPVC provides allowable stress limits to prevent gross failure of vessels. Analysis of primary stresses for the CNV shows that the most limiting cross sections occur in the general section of the shells. Table 5-10 summarizes the design condition (1,000 psia and 550 degrees F) stress results for the upper and lower CNV. The table shows that the upper and lower CNV stresses in the shells have 3.8 percent and 9.5 percent margin to the ASME allowable stresses, respectively. The upper and lower CNV rated design pressure could be increased by the margins to 1,038 psia and 1,095 psia, respectively. A comparison of the limiting LOCA peak pressure (921 psia) and the maximum overall peak containment pressure (951 psia) to the increased pressure ratings of the upper and lower shells shows is provided by Table 5-10, when the margin between the maximum membrane stresses to the ASME allowable stresses is considered.

Table 5-10 Design condition stress summary

CNV Section	Maximum Membrane Stress (ksi)	Allowable Membrane Stress, S_m (ksi)	Stress to allowable margin (%)	Increased CNV rated pressure (psia)	Increased LOCA peak pressure margin (%)	Increased overall peak pressure Margin
Upper CNV shell	28.86	30.00	3.8	1,038	13.7	9.1
Lower CNV shell	26.64	29.45	9.5	1,095	18.9	15.1

Additionally, the upper CNV shell is low alloy steel with stainless steel cladding on both sides. The base metal of the upper section is 3.00 in. Cladding adds an additional 0.375 in. to the thickness. This is an increase of 12.5 percent in thickness and therefore an increase in pressure capacity. The increase in thickness proportionally increases pressure capacity of the CNV. While in the ASME Code calculations cladding is neglected, the additional material does provide additional margin. When this margin is considered the increased pressure rating of the upper CNV becomes 1.125*1,038 psia = 1,168 psi. The margin of maximum containment peak pressure (951 psia) to the increased rated pressure of the upper CNV is 22.8. percent.

This demonstrates additional margin in the CNV design that is not considered by the containment design pressure rating.

5.4.2 Decay Heat Removal System Availability

The LOCA and AOO Case 2 and Case 5 are performed with and without DHRS available to estimate the impact of DHRS availability on the CNV peak pressure response. The DHRS is conservatively not credited in the design basis containment response analysis cases. The NRELAP5 code has not been validated to cover DHRS performance during LOCAs or valve opening events. However, the DHRS is a single-failure proof safety-related system that can be credited in the future, with additional NRELAP5 validation, if the CNV pressure margin is reduced for any reason (design changes). The results of the DHRS available cases indicate that more than 25 psia additional margin could be gained by credit for DHRS availability.

5.4.3 Conclusion

The NPM design provides sufficient margin to satisfy the requirements of GDC 16 and 50. The CNV response to the limiting LOCA event and AOO transient are conservatively calculated and demonstrate that the peak calculated pressures are below the CNV design pressure and decrease in pressure to one-half of the peak value within 24 hours.

Further assurance of sufficient margin is provided through consideration of the robust design and construction of the NPM and conservative assumptions related to the crediting of the DHRS system in the containment response analysis. The determination of NPM design pressure, in accordance with ASME Class 1 criteria, is conservative relative to Class MC and CC containments. This design pressure does not consider the additional margin provided by the internal and external cladding of the upper CNV shell. The effect of DHRS actuation in reducing peak containment pressure was not credited in the containment response analysis.

The containment response analysis methodology, analysis results and further conservatisms related to design and system operation provide assurance that the NPM design demonstrates sufficient margin to satisfy the requirements of GDC 16 and 50.

6.0 Summary and Conclusions

This report presents the NuScale containment response analysis methodology for determining primary system and secondary system mass and energy releases and the resultant CNV pressure and temperature response for the NPM. A spectrum of LOCAs and ECCS valve opening events were analyzed along with the MSLB and FWLB accidents. The scope of the methodology is the short-term CNV response for comparison to the CNV pressure and temperature design limits. Equipment qualification and the long-term NPM response are not in the scope of this report.

The containment response analysis methodology uses the NRELAP5 code, which originates from the RELAP5-3D© code. The NRELAP5 code includes new capabilities added by NuScale to enable modeling of the design features and transient response of the NPM. The NRELAP5 model of the NPM used in the containment response analysis methodology is based on the NuScale LOCA and non-LOCA evaluation models with limited revisions and additions necessary for application in the containment response analysis methodology. NuScale has completed LOCA and non-LOCA phenomena identification and ranking tables. The results of the PIRTs have been used in the development of the NRELAP5 code and model. The NRELAP5 LOCA and non-LOCA models have been assessed by comparison to generic separate effects tests and intergral effects test, as well as to the NuScale design-specific NIST-1 facility separate effects and integral LOCA tests.

The containment response analysis methodology is shown to meet the intent of Section 6.2 of the NuScale DSRS. Based on the systematic application of conservative initial conditions and boundary conditions in the containment response analysis methodology, the margin in the containment response analysis methodology is judged to be sufficient.

Conservative NRELAP5 demonstration analyses of the containment response analysis methodology have been performed for a spectrum of primary system LOCAs and ECCS valve opening events, and for the MSLB and FWLB accident secondary system events. Sensitivity studies have been used to identify the bounding scenarios and trends. The following insights were obtained:

- The bounding scenarios for both peak CNV pressure and temperature were determined to be primary system release events. The secondary system break events may include ECCS actuation, which essentially combines an initial secondary system M&E release with a subsequent primary system M&E release, but they are non-limiting scenarios.
- The limiting M&E release scenario is characterized by an initial heatup and pressurization of the CNV due to the LOCA or ECCS valve opening, and then the subsequent opening of the RVVs on following the pressure differential decreasing to below the IAB release pressure. It is the second M&E release that drives the CNV to the peak CNV pressure and peak CNV wall temperature results.
- The heat capacity of the CNV wall, rather than heat transfer to the reactor pool, provides the short-term heat sink to limit the peak CNV pressure and temperature.

- For the limiting cases the results of the sensitivity studies, including postulated single failures and the loss of normal AC and DC power, showed only a limited impact (~1 percent) on the key figures-of-merit. The loss of normal AC and DC power and the timing of ECCS valve opening were the most important sensitivity parameters.

The limiting LOCA peak pressure and CNV wall temperature are a result of the RCS injection line break. The LOCA limiting peak CNV wall temperature is approximately 523 degrees F and it results from a reactor coolant system injection line break case, with a loss of normal AC power. The LOCA limiting peak pressure is 921 psia which also results from a reactor coolant system injection line break case, with a loss of normal AC power. The LOCA event peak CNV pressure is below the CNV design pressure of 1000 psia. The LOCA peak CNV pressure and wall temperature bound the main steamline break and FWLB results.

The limiting peak CNV accident pressure is approximately 951 psia and it results from an inadvertent reactor recirculation valve opening AOO, with a loss of normal AC and DC power. This peak pressure is also less than the CNV design pressure of 1000 psia. The CNV pressure for this limiting case is reduced to below 50 percent of the peak value in less than 2 hours, demonstrating adequate NPM containment heat removal.

Section 5.4 discussed margin in the NPM design that is not included in the CNV design pressure rating or modeled in the containment response analyses. Design factors conservatively not credited include CNV shell stress margins, CNV cladding material and the availability of the DHRS.

The containment response analysis demonstrates that the NPM design has adequate margin to design limits and that it satisfies the requriements of GDC 16 and 50 and PDC 38.

7.0 References

7.1 Source Documents

7.1.1 U.S. Code of Federal Regulations, Title 10, Part 50.

7.1.2 U.S Code of Federal Regulations, “Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants, “ (10CFR50, Appendix A).

7.1.3 U.S. Nuclear Regulatory Commission, “Transient and Accident Analysis Methods,” Regulatory Guide 1.203, December 2005.

7.1.4 U.S. Nuclear Regulatory Commision, “Design Specific Review Standard for NuScale SMR Design,” Section 6.2.1, June 2016.

7.1.5 U.S. Nuclear Regulatory Commision, “Design Specific Review Standard for NuScale SMR Design,” Section 6.2.1.1.A, June 2016.

7.1.6 U.S. Nuclear Regulatory Commision, “Design Specific Review Standard for NuScale SMR Design,” Section 6.2.1.3, June 2016.

7.1.7 U.S. Nuclear Regulatory Commision, “Design Specific Review Standard for NuScale SMR Design,” Section 6.2.1.4, June 2016.

7.2 Reference Documents

7.2.1 NuScale Power, LLC, “LOCA Evaluation Model,” TR-0516-49422, Revision 0.

7.2.2 NuScale Power, LLC, “Non-LOCA Transient Analysis Methodology Report,” TR-0516-49426, Revision 0.

8.0 Appendicies

8.1 Mass and Energy Input

The purpose of this Appendix is to present the mass and energy release to the CNV during the limiting LOCA event (Case 2), the limiting overall peak CNV pressure event (Case 5) and the limiting secondary system release event (MSLB).

8.2 Heat Sink Tables

The purpose of this Appendix is to present the passive heat sink characteristics credited in the containment response analysis methodology.

Table 8-1 Limiting Peak Pressure Case – Mass and Energy Release

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
0	0	0
49.245	0	0
50.25	351.711	173140
51.255	529.189	260397
62.31	495.158	247671
95.475	411.279	236802
105.525	397.222	227323
116.58	380.903	219530
127.635	1626.91	1491710
128.64	1413.56	1311580
129.645	1182.23	1109160
130.65	967.717	909117
131.655	863.712	803803
132.66	1023.01	771943
133.665	1019.46	743749
134.67	967.35	687226
135.675	831.284	601624
136.68	716.093	524738
137.685	560.892	432894
138.69	419.776	344836
139.695	279.342	247789
140.7	188.447	170871
141.705	137.069	122905
186.93	72.4145	55850.6
348.37	45.3034	33856.7
412.53	36.9198	28001.5
540.02	0.111389	270.432

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
650.05	1.6522	8387.4
650.075	-1.33437	-1593.41
760.13	-1.02407	-1227.47
870.185	11.8632	15318.7
970.235	15.5462	20038
1160.4	-0.0610871	-56.3615
1380.95	0.184891	248.364
1601.5	-0.804867	5088.77
1794.2	-0.524	-650.898
1802	-0.637231	5023.34
2014.75	0.0259352	42.1415
2170	0.390625	507.728
2180	107.388	130405
2200.05	-0.454126	-558.524
2280.25	0.151932	207.155
2400.55	-0.301595	-369.278
2420.6	-0.231089	-275.562
2440.65	46.8948	57547.2
2460.7	-0.187912	-228.765
2580	0.263031	343.004
2841.65	0.166304	218.097
3042.15	0.143518	188.538
3242.65	0.129225	169.854
3500	0.118755	3483.25

1. RVV opens at 50 seconds.

Table 8-2 Limiting Secondary Break Peak Pressure Mass and Energy Release

Time (s)(2)	Mass Release (lbm/s)	Energy Release (Btu/s)
0	0	0
1.005	479.103	592441
2.01	416.448	513402
3.015	367.703	456779
4.02	324.901	408889
5.025	287.698	364725
6.03	246.597	315160
7.035	148.581	190756
12.06	156.577	198539
17.085	165.819	205905

Time (s)(2)	Mass Release (lbm/s)	Energy Release (Btu/s)
23.115	144.182	180805
29.145	119.093	150507
35.175	98.1639	105865
38.19	58.7877	74755.5
41.205	33.1988	42901.4
47.01	13.0776	16917.1
53.04	6.97991	9046.27
64.095	6.28984	8112.73
75.15	12.7395	16425.3
86.205	13.2017	17118.6
97.26	21.8136	28220.1
108.315	6.32603	8364.87
119.37	4.47799	6003.89
130.425	11.6693	15248.7
141.48	1.94533	2718.69
151.53	0.421951	754.803
162.585	3.36566	4553.49
173.64	3.96629	5316.12
184.695	2.78703	3807.61
195.75	1.35812	1975.04
206.805	1.43866	2075.6
217.86	3.43159	4624.47
228.915	2.97928	4039.31
239.97	2.26944	3130.48
250.02	1.47545	2115.27
261.075	1.45841	2092.53
272.13	1.53005	2181.9
283.185	0.866597	1330.24
294.24	0.691397	1103.58
305.295	0.536189	902.676
316.35	0.419932	751.601
327.405	0.344147	652.188
338.46	0.311589	608.824
348.51	0.255509	534.707
359.565	0.227776	496.142
370.62	0.287188	569.137
381.675	0.55293	909.054
392.73	0.991042	1465.49
403.785	0.853209	1280.26

Time (s)[2]	Mass Release (lbm/s)	Energy Release (Btu/s)
414.84	0.800428	1206.54
425.895	1.06917	1547
436.95	0.887626	1300.05
447	0.81661	1205.05
458.055	0.20107	417.002
469.11	0.394275	665.475
480.165	-0.118122	9.48294
491.22	0.308655	551.042

2. Break initiated at 0 seconds.

8.2.1 Listing of Passive Heat Sinks

The containment vessel shell is the only passive heat sink credited in the containment response analysis methodology.

8.2.2 Modeling of Passive Heat Sinks

Table 8-3 Passive heat sinks

Passive Heat Sink (Vessel steel plate)	Material	Thickness, in	Group	Exposed Surface Area by Thickness Group, ft2	Shell Volume, ft3	Total Mass, lbm	Total Surface Area, ft2
{{							
				.			
							}}2(a),(c)

8.2.3 Thickness Groups

Table 8-4 Thickness groups

Material	Group Designation	Thickness Range, in
SA-240 304L (Stainless Steel)	1	{{ }}2(a),(c)
SA-240 304L (Stainless Steel)	2	{{ }}2(a),(c)
SA-240 304L (Stainless Steel), SA-508 Grade 3 (Carbon Steel)	3	{{ }}2(a),(c)

8.2.4 Properties of Passive Heat Sink Materials

Table 8-5 Physical properties of passive heat sink materials

Material	Density, lbm/ft3	Specific Heat, Btu/lbm-°F	Thermal Conductivity, Btu/hr-ft-°F
SA-240 304L (Stainless Steel)	501.12	0.1137	8.6
SA-508 Grade 3 (Carbon Steel)	483.84	0.1067	23.7

Enclosure 4:

Long-Term Cooling Methodology, TR-0916-51299-NP, Revision 0, nonproprietary version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Long-Term Cooling Methodology

January 2017

Revision 0

Docket: PROJ0769

NuScale Power, LLC

1100 NE Circle Blvd., Suite 200

Corvallis, Oregon 97330

www.nuscalepower.com

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Abstract

This report presents (1) the NuScale Power, LLC, methodology used to evaluate the emergency core cooling system (ECCS) long-term cooling capability of the NuScale Power Module (NPM) after a successful initial short-term response to a design basis event, and (2) evaluation results demonstrating satisfactory ECCS performance during long-term cooling. The report includes discussion on the transition to long-term cooling for events that assume the use of the decay heat removal system (DHRS) as well as those that actuate the ECCS early in a design basis event. This report is applicable to long-term cooling capability following both loss-of-coolant (LOCA) and non-LOCA design basis events.

The long-term cooling methodology is an extension of the NuScale LOCA evaluation model (EM) (Reference 8.2.1), and thus uses a graded approach to the EM development and assessment process (EMDAP) defined in Regulatory Guide 1.203. The phenomena of high importance developed in the long-term cooling phenomena identification and ranking table (PIRT) analysis performed for long-term cooling EM are discussed in this report.

The long-term cooling evaluation results demonstrate ECCS conformance with the acceptance criteria in 10 CFR 50.46(b)(4) and 10 CFR 50.46(b)(5) for coolable geometry and long-term cooling for the long-term cooling phase when stable natural circulation has developed through the ECCS configuration. This report also demonstrates conformance to NuScale Principal Design Criterion 35 along with compliance with relevant Acceptance Criteria given by the Design Specific Review Standard for NuScale Small Modular Reactor Design, Sections 6.3 and 15.6.5 (Reference 8.2.3 and Reference 8.2.4, respectively).

This report provides information supplementing NuScale Final Safety Analysis Report Section 6.2, Section 6.3, Section 15.0, and Section 15.6.5.

Executive Summary

The NuScale Power Module (NPM) is designed to successfully cool down after experiencing an initiated event and transition to a long-term cooling condition. The purpose of this report is to define the evaluation model (EM) for evaluating long-term cooling and demonstrate that ECCS performance meets the regulatory criteria during long-term cooling in a conservative fashion. The long-term cooling (LTC) analyses are performed for 72 hours to demonstrate that the module(s) will remain in a safe, stable condition with the ECCS operating without credit for normal AC power, the nonsafety-related DC power system, or any operator action. The LTC EM is developed to conservatively model the long term global heat removal capabilities of the emergency core cooling system (ECCS) and the reactor pool. This methodology ensures that the criteria of 10 CFR 50.46(b)(4) and 10 CFR 50.46(b)(5) are met.

In addition, this evaluation demonstrates conformance with the ECCS *Principal Design Criterion (PDC)* 35, as described in the NuScale Final Safety Analysis Report (FSAR) Section 3.1.

Additional regulatory guidance for the design of the ECCS is found in the Design Specific Review Standard (DSRS) for NuScale Small Modular Reactor Design, Section 6.3 relating to gravitational head providing sufficient core cooling for 72 hours, without operator actions and without nonsafety-related onsite or offsite power. The NuScale DSRS Section 15.6.5 refers to the evaluation of post-LOCA long-term cooling for decay heat removal by assuring boric acid precipitation is prevented for all break locations and sizes and asks the reviewer to verify that procedures are in place to assure boron precipitation is mitigated. DSRS Section 15.6.5 also specifies that steam generator tube failure (SGTF) be reviewed for the potential coolant inventory loss from the reactor vessel to the secondary side.

The report describes the following NuScale-specific acceptance criteria that were developed to assure that regulatory requirements of 10 CFR 50.46 are met: 1) collapsed liquid level in the reactor vessel remains above the top of the core, 2) cladding temperatures predicted by NRELAP5 remain acceptably low, 3) margin to the critical heat flux (CHF) predicted by NRELAP5 using a CHF correlation appropriate to the fluid conditions is maintained, 4) coolable geometry is maintained, and 5) the core remains subcritical. The third criterion is demonstrated by the first two criteria and showing that the minimum critical heat flux ratio (MCHFR) prediction for the short term LOCA is acceptable. The fifth criteria is not applicable to the LTC cooling condition since no mechanism to push a large volume of diluted water into the core inlet exists, and therefore no credible mechanism for recriticality due to boron dilution exists.

The long-term phase of core cooling starts once the ECCS is actuated and the NPM is configured such that steam from the pressurizer region is released to the containment vessel (CNV) through the reactor vent valves (RVVs) and condenses on the CNV wall collecting in the bottom of the CNV, then flowing through the reactor recirculation valves (RRVs) to the core inlet. This recirculation flow loop continues as the NPM is cooled. The long-term cooling configuration is reached through both LOCA and non-LOCA initiating events.

The LTC EM is developed using a graded approach to the evaluation model development and assessment process (EMDAP) defined in Regulatory Guide (RG) 1.203. The approach for the long-term cooling EM utilizes the NuScale LOCA EM (Reference 8.2.1). The LTC EM focuses on the phenomena identification and ranking table (PIRT) process to identify the important

parameters which are specifically addressed. An extensive PIRT was developed for LTC. Each important parameter is discussed and evaluated in this report as it relates to the LTC EM.

The LTC EM uses the proprietary NRELAP5 systems analysis computer code as the computational engine, derived from the Idaho National Laboratory RELAP5-3D© computer code. The models and correlations used by NRELAP5 were reviewed and, where appropriate, modified for use within the long-term cooling EM. The NRELAP5 model is validated through the assessment of NIST-1 facility tests and comparison of NRELAP5 predictions to test results. Comparison of the NRELAP5 model to the NIST-1 test results demonstrate that the NRELAP5 code adequately predicts the NPM conditions both in the RPV and the CNV.

The methodology for the NPM thermal-hydraulic response and boron precipitation evaluation are presented in this report. There are two LTC general conditions which address the thermal-hydraulic response and boron precipitation: maximum cooldown to minimize the RPV core liquid volume in the riser region for addressing boron precipitation and the minimum collapsed liquid level above the active fuel, and minimum cooldown to maximize the fuel cladding temperature.

The methodology is demonstrated in the report by presenting the limiting results of a base LOCA case for the letdown line break (LDBRK) utilizing conservative worst case conditions determined by sensitivity calculations. In addition the SGTF results are presented. Sensitivity cases performed considered the following assumptions:

- single active failure, ECCS valve failure to open is the relevant single active failure to consider in the LTC analyses
- decay heat, ranging from no decay heat to 120 percent of nominal
- heat transfer from the RPV to CNV, ranging from adiabatic to 1000 percent of nominal
- heat transfer from the CNV to pool, ranging from 20 percent to 1000 percent
- reactor pool temperature, ranging from 40 degrees F to 210 degrees F
- reactor pool level, down to 45 feet (Nominal at 69 feet)
- reactor pool volume effect on calculated pool temperature heatup from initial conditions
- non-condensable gas effect
- pressurizer level, down to 20 percent level

In all analyzed cases, the core remained covered, with greater than 2.25 feet of collapsed liquid level in the riser above the top of the core. Possible leakage from the CNV was found to have a negligible impact on the results. The cases identified as most limiting, the minimum cooldown, maximum cooldown, and SGTF with decay heat removal system (DHRS), all showed consistently decreasing reactor coolant system (RCS) and cladding temperatures, supporting the conclusion that the ECCS is capable of providing adequate cooling for the 72 hour evaluation period.

In order to evaluate the criterion for maintaining coolable geometry, the possibility of boron precipitation is evaluated in this report. The methodology for determining boron precipitation is

conservative, as it assumes the maximum boron concentration and a minimum volume that includes the core and riser region for boron mixing and that all the RPV boron remains within this region. The maximum boron concentration is shown in this report to remain below the solubility limit for the minimum RCS temperatures reached within the 72 hour evaluation period for long-term cooling.

The long-term cooling methodology, boron precipitation methodology, and analysis results presented in this report provide supplemental information designed to inform the NRC's evaluation of NuScale Final Safety Analysis Report Sections 6.2, 6.3, 15.0 and 15.6.5.

1.0 Introduction

1.1 Purpose

The purpose of this report is to present the NuScale evaluation model (EM) used to evaluate the long term module response during emergency core cooling system (ECCS) operation and to present evaluation results demonstrating satisfactory ECCS performance during long-term cooling. This report describes the ECCS long-term cooling (LTC) analysis scope, acceptance criteria, and methodology for demonstrating that the acceptance criteria are met for the NuScale Power Module (NPM).

The LTC analysis scope is defined based on the applicable regulatory requirements, NuScale-specific requirements for the design, and considering relevant aspects of the NuScale design that affect the long-term transient progression.

1.2 Scope

In the NPM, the ECCS is designed to operate following a loss-of-coolant accident (LOCA) or after the inadvertent opening of a valve that allows release of primary reactor coolant into containment, or if power to the ECCS valve actuators is lost and the reactor coolant system (RCS) is at sufficiently low pressure. Due to the unique NuScale ECCS design, these different scenarios are considered in the analysis of the ECCS long-term cooling.

The long-term cooling phase of decay heat removal is defined as beginning when ECCS actuates to open the RVVs and RRVs, the recirculation flow is established, and the pressures and levels in containment and the RPV approach a stable condition (Reference 8.2.1, Section 4.2).

This report summarizes the following:

- long term NuScale design basis event progression following ECCS actuation
- regulatory requirements and NuScale-specific design requirements applicable to LTC
- LTC acceptance criteria
- NuScale LTC phenomena identification and ranking table (PIRT)
- analysis tools, qualification of the tools, and methodology for demonstrating that the LTC acceptance criteria are met
- results of the LTC analyses.

The following LTC analysis areas are addressed in this report:

- demonstration of long-term core cooling following ECCS actuation
- evaluation for boron precipitation

The following areas are outside scope of this report:

- The non-LOCA and LOCA evaluation models for the short-term time periods are covered in separate methodology reports. However, the transition between a non-LOCA initiating event, including steam generator tube failure (SGTF) events, and ECCS long-term cooling is considered in this report.

- The effects of debris on ECCS operation that are the subject of the NRC generic safety issue 191 are outside scope of this report. The NuScale design and debris loads have been assessed to ensure that the system and its components will operate as designed under long-term ECCS operating conditions. The LTC analyses are performed assuming a clean core condition without debris.

- Assessment of the NuScale design return to power due to overcooling, assuming one control rod stuck out of the core, is outside the scope of this report. For the LTC calculations in this report, the heat source is decay heat.

- Assessment of a station blackout is outside the scope of this report and covered in separate analysis.

- Analysis of long-term decay heat removal system (DHRS) performance and decay heat removal is addressed by separate analyses.

- This EM does not assess seismic issues, which are covered in separate methodologies and assessments.

- Critical heat flux (CHF) evaluation is only of interest in the short-term response of the events analyzed in this document. Short-term LOCA CHF is addressed by the NuScale LOCA EM (Reference 8.2.1). For long-term cooling, a collapsed liquid level above the top of active fuel (TAF) and acceptably low cladding temperatures calculated by NRELAP5 are considered sufficient to demonstrate that CHF does not occur.

The ECCS long-term cooling analyses provided in this report address all design basis events that evolve to the configuration where operation of ECCS is needed for long-term cooling, as illustrated in Figure 1-1. These analyses are relevant for both LOCA and non-LOCA initiated events.

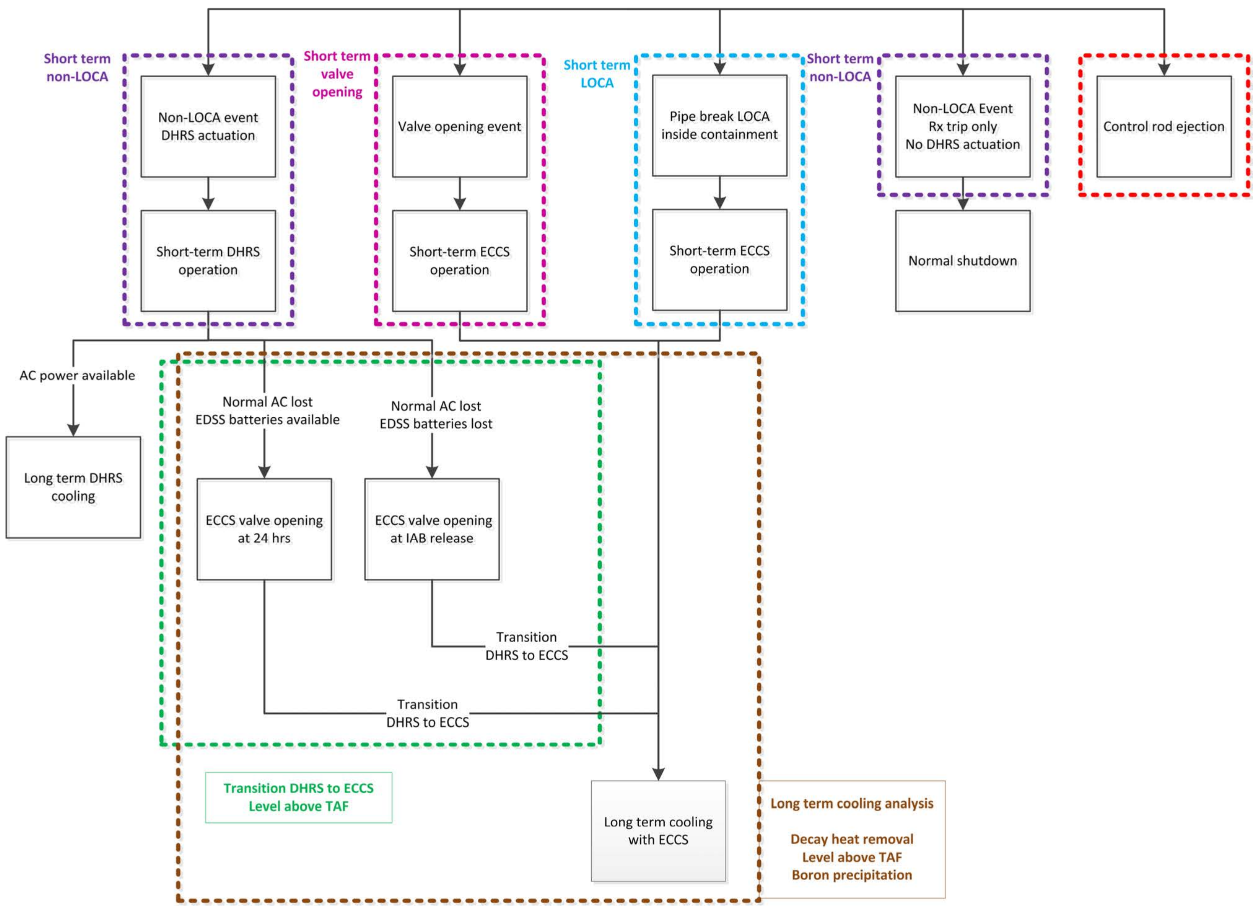

Figure 1-1 Illustration of the scope and analyses covered by long-term cooling methodology

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
AC	alternating current
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CHF	critical heat flux
CNV	containment vessel
CVCS	chemical and volume control system
DCA	Design Certification Application
DHRS	decay heat removal system
DSRS	Design Specific Review Standard
ECCS	emergency core cooling system
EM	evaluation model
EMDAP	evaluation model development and assessment process
FOM	figure of merit

Term	Definition
GDC	Generic Design Criterion
HZP	hot zero power
IAB	inadvertent actuation block
LDBRK	letdown line break
LOCA	loss-of-coolant accident
LTC	long-term cooling
NIST-1	NuScale Integral System Test
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
PDC	principal design criteria
PIRT	phenomena identification and ranking table
PZR	pressurizer
RCS	reactor coolant system
RG	Regulatory Guide
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RVV	reactor vent valve
SAF	single active failure
SG	steam generator
SGTF	steam generator tube failure
TAF	top of active fuel
UHS	ultimate heat sink

Table 1-2 Definitions

Term	Definition
C_v	Flow coefficient
“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 will, with few exceptions, lay within the specified or inferred uncertainty bands of the data. The code may be used with confidence in similar applications.
Figure of merit	A parameter selected to characterize the plant long-term cooling response.
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.
Non-LOCA transient	Reactor coolant system transients described in the NUREG-0800 Standard Review Plan Sections 15.1, 15.2, 15.4, and 15.5, and other comparable transients that may be unique to the NuScale system. 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 the scope of non-LOCA transients.

Term	Definition
“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 will frequently lie 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 was used in similar applications.

2.0 Regulatory Requirements and Roadmap

2.1 Background

In the NPM design, there are two systems that may perform the safety-related functions of decay heat and residual heat removal following an anticipated operational occurrence or accident. The DHRS provides decay and residual heat removal while RCS inventory is retained inside the reactor pressure vessel. If RCS inventory is redistributed between the reactor pressure vessel and the containment vessel (CNV), due to a pipe break LOCA or RCS valve opening event, or opening of the ECCS valves, the ECCS provides decay and residual heat removal. The scope of this EM addresses ECCS long-term cooling.

As an advanced passive plant design, the NuScale plant is designed such that:

- protection against design basis events is through passive means for at least 72 hours, and

- no operator actions are required for at least 72 hours for design basis events.

Therefore, in the NuScale design, after initial operation of the ECCS, the safety-related systems continue to provide decay and residual heat removal, without operator actions, for at least 72 hours for design basis events.

In the NPM, the ECCS is designed to operate following a LOCA or after the inadvertent opening of a valve that allows release of primary reactor coolant into containment, or if power to the ECCS valve actuators is lost and the RCS is at sufficiently low pressure. Due to the unique NuScale ECCS design, these different scenarios are considered in the analysis of the ECCS long-term cooling. Ultimately these scenarios will converge towards a similar long-term cooling transient.

For the design basis safety analyses, reactor trip and actuation of the passive safety systems to mitigate the event will generally occur early in the transient progression. Analysis of the short-term design basis event progression is performed following the appropriate methodology. This report addresses the interface with the short-term analyses, the acceptance criteria applicable to the longer term transient progression to LTC with ECCS, and how these acceptance criteria are met for the NuScale design.

2.2 Regulatory Requirements and Guidance

The NRC regulations and regulatory guidance applicable to the LTC methodology are described in this section. The elements of the LTC methodology that address each of these regulations and guidance documents are discussed.

2.2.1 Regulatory Requirements

10 CFR 50.46 (a) provides two options for an acceptable NuScale LOCA EM. Paragraph 50.46(a)(i) allows for a best-estimate approach to be followed and Paragraph 50.46.(a)(ii) allows for the conservative deterministic approach detailed in 10 CFR 50 Appendix K. As the LTC EM is an extension of the NuScale LOCA EM (Reference 8.2.1), the disposition of the 10 CFR 50 Appendix K requirements that apply to the long-

term cooling phase are applied in the same manner as for the LOCA EM. Since the NuScale LOCA EM (Reference 8.2.1) and the LTC EM are equivalent with regard to all Appendix K requirements, no further exemptions to the Appendix K requirements are required for the LTC EM beyond those identified in Reference 8.2.1.

The NuScale Principal Design Criterion (PDC) 35, based on General Design Criterion 35, establishes the required safety function of the ECCS, as described in FSAR Section 3.1 of the NuScale DCA. The portion of the PDC of interest to the LTC methodology is identical to 10 CFR 50, Appendix A, General Design Criterion 35, and states:

> *A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*
>
> *Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.*

10 CFR 50.46(b) implements GDC 35, and thus NuScale PDC 35, by establishing specific acceptance criteria for ECCS cooling performance. The applicable regulatory criteria from 10 CFR 50.46(b) regarding long-term ECCS performance (Reference 8.2.2) include the following:

> (4) Coolable geometry.
>
> *Calculated changes in core geometry shall be such that the core remains amenable to cooling.*
>
> (5) Long-term cooling.
>
> *After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.*

10 CFR 50.46 applies to ECCS performance following a LOCA. For the NPM, the long-term core cooling ECCS requirements following a LOCA are fulfilled through the actuation of the passive ECCS. While 10 CFR 50.46 does not address ECCS performance associated with non-LOCA events for long-term core cooling, the ECCS removes residual and core decay heat whenever the NPM transitions to the ECCS configuration.

2.2.2 Regulatory Guidance

NRC review guidance regarding the ECCS requirements in DSRS Section 6.3 (Reference 8.2.3) includes the following from page 6.3-2.

> *For advanced passive reactors that rely on gravitational head to provide ECCS injection to the reactor coolant system (RCS), the RCS should be designed such that the available gravitational head is sufficient to provide adequate core cooling when depressurized.*
>
> *For advanced reactors which rely on passive safety-related systems and equipment to automatically establish and maintain safe-shutdown conditions for the plant, these passive safety systems must be designed with sufficient capability to maintain safe shutdown conditions for 72 hours, without operator actions and without nonsafety-related onsite or offsite power.*

The following review guidance from DSRS Section 15.6.5 (Reference 8.2.4) refers to the evaluation of post-LOCA long-term cooling for decay heat removal, and for assessment of boric acid precipitation.

> *An evaluation of post-LOCA long-term cooling should also be performed to identify the operator actions to successfully control and prevent boric acid precipitation. Analyses of small break LOCAs should be performed to identify the timing for boric acid precipitation. A spectrum of small breaks should also be analyzed to identify other means to control boric acid precipitation when RCS pressure remains too high to enable flushing of the core. All equipment and operator action times should also be clearly identified in the analyses.*

From the DSRS page 15.6.5-4, the reactor systems review of this section includes the following.

> *F. The results of the post-LOCA long-term cooling analyses to assure that an acceptable model has been employed to identify the timing of boric acid precipitation for all break locations and sizes. The review will also verify that an adequate procedure has been devised to control boric acid precipitation for all breaks to assure long-term cooling.*
>
> *and,*
>
> *Steam generator tube rupture events shall also be reviewed as part of the LOCA break spectrum analysis. The reviewer shall review the potential coolant inventory loss from reactor vessel to the secondary side.*

The transition of an event such as an SGTF or small pipe break outside of containment to cooling by the ECCS with reduced reactor coolant inventory is dispositioned in this report. In the NuScale design, with normal AC power available an SGTF event will result in the actuation of the DHRS; the inventory reduction from the primary to the secondary is detected and isolated before the ECCS is actuated. The short-term event progression of an SGTF is analyzed using the non-LOCA analysis methodology described in Reference 8.2.5. Similarly, in the NuScale design, with normal AC power available a break in a small pipe outside of containment will result in the actuation of the DHRS; the inventory reduction from the primary to the secondary is detected and isolated by closing the containment isolation valves before the ECCS is actuated. The short-term event progression of a small pipe break outside of containment is analyzed using the non-LOCA analysis methodology

described in Reference 8.2.5. If normal power is assumed to be lost, an SGTF or a small pipe break outside of containment will transition to cooling by the ECCS.

2.3 Acceptance Criteria and Transient Duration

2.3.1 Acceptance Criteria

The NuScale-specific acceptance criteria for the LTC analysis and the transient duration for which the acceptance criteria are demonstrated are defined in this section.

The NuScale-specific acceptance criteria for the ECCS long-term cooling analyses are:

1. core cooling is provided to remove decay and residual heat from the core.
   This acceptance criterion is demonstrated in thermal-hydraulic calculations with NRELAP5 by the following:

   a. collapsed liquid level in the reactor vessel remains above the top of the core.

   b. cladding temperatures predicted by NRELAP5 remain acceptably low.

   c. margin to the CHF predicted by NRELAP5 using a CHF correlation appropriate to the fluid conditions is maintained.

      - DSRS 15.6.5-10 states “If core uncovery is not expected during the entire period of a LOCA, the staff should ensure that a significant number of fuel rods will not be damaged because of local dryout conditions. This may be demonstrated by showing that the limiting fuel rod heat flux remains below the critical heat flux (CHF) at a given pressure after depressurization has taken place. If, however, the heat flux exceeds the CHF, further analyses should be performed to estimate the amount of fuel damage expected from “burn-out“ while the bulk of the core remains covered with water during the LOCA. Fuel damage and potential for radioactivity release to the environment must be consistent with 10 CFR Part 100.”

      - The NuScale LOCA EM addresses the short-term CHF response to a primary system pipe beak and ECCS actuation. No explicit CHF response is evaluated as part of the LTC calculations; maintaining a collapsed liquid level in the riser above the core, along with demonstrating that cladding temperatures remain acceptably low, are considered sufficient conditions to show MCHFR limits are not challenged. In addition, meeting the criteria that the core remain covered by collapsed liquid level in the riser and that cladding temperatures remain acceptably low assure that the PDC 35 criterion that “clad metal-water reaction is limited to negligible amounts” is met.

2. coolable geometry is maintained.

   This acceptance criterion is demonstrated by the boron precipitation analysis that demonstrates that the boron concentration in the core region remains below the solubility limit.

3. the core remains subcritical.

   In the long-term cooling analyses, it is demonstrated that decay heat is removed and the core remains cooled. Temperature changes or changes in core boron concentration can affect reactivity. Reactivity effects due to cooldown, assuming the worst control rod stuck out of the core, are outside scope of this report. During the long-term cooling phase, boiling in the core region is expected to concentrate boron in the liquid in the core and riser region. After ECCS valves open and recirculation is established, liquid from containment enters the reactor pressure vessel through the reactor recirculation valves, circulates into the core region, and vapor is vented into containment through the reactor vent valves where it condenses on the containment wall. The boron concentration of liquid in containment may be lower than the boron concentration of liquid in the core/riser region. However, since flow rates from containment into the reactor pressure vessel through the recirculation valves are low and the boron concentration in the core region will tend to increase due to boiling in the core region, no credible means of introducing a large slug of deborated water unmixed into the core region has been identified for the NuScale design. Therefore, for the long-term cooling analyses, the core heat source is decay heat and not any additional heat due to a possible recriticality once the NPM has begun heat removal in the long-term cooling phase.

2.3.2 Transient Duration

The ECCS cooling evaluation can be broken into three stages: (1) blowdown, (2) ECCS depressurization, and (3) LTC. Consistent with the NuScale LOCA EM topical report (Reference 8.2.1), the transition from LOCA to LTC occurs once natural circulation between the RPV and containment has been established and the pressure and liquid levels in the CNV and the RPV approach a stable equilibrium condition. This natural circulation pattern consists of coolant upflow through the core producing steam, steam leaving the RPV through the reactor vent valves (RVVs) and condensing on the cool containment shell, and the condensate being returned from the containment pool to the RPV through the reactor recirculation valves (RRVs). This is a natural transition point into LTC as all LOCA events will evolve to this condition. The assessment of LTC then covers the progression of the event from this point forward.

The LTC analyses are performed for 72 hours to demonstrate that the module(s) will remain in a safe, stable condition with the ECCS operating without credit for normal AC power, the nonsafety-related DC power system, or any operator action.

The ECCS long-term cooling analyses address the following scenarios:

1. ECCS cooling that begins during the short-term event progression. LTC begins where the NuScale LOCA EM analysis ends when ECCS recirculation flow (RCS steam is released to the CNV through the RVVs, condensed on the CNV walls, and condensed liquid re-enters the RPV through the RRVs) and pressures and levels in the RPV and CNV approach a stable equilibrium condition.

   DHRS cooling transition into ECCS cooling is considered. Long- term cooling analyses demonstrate that if DHRS cooling is provided until either the inadvertent actuation block (IAB) setpoint is reached or 24 hours is reached such that the

ECCS timer expires, and decay heat removal transitions to ECCS cooling, then the module(s) will remain in a safe, stable condition for up to 72 hours following the event. DHRS transition cases will also include consideration for SGTF to address inventory loss prior to isolation of the SG.

2.4 Long-Term Cooling 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 anticipated operational occurrences, infrequent events, and accidents. The EMDAP as defined in RG 1.203 (Reference 8.2.6) 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 in this report has been developed for simulating the long-term cooling capability of the NPM during long-term ECCS operation.

NRELAP5 is the thermal-hydraulics code used to assess the ECCS performance of the NPM during LTC. The NuScale LOCA evaluation model (Reference 8.2.1) was developed following the EMDAP guidelines of RG 1.203 (Reference 8.2.6). Phenomena identified as high-ranked for ECCS long-term cooling were evaluated with respect to the high-ranked phenomena identified as part of the NuScale LOCA EM development. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC calculations (see Section 3.0 and Section 5.0), a graded approach to the EMDAP is applied for development of the LTC evaluation model.

Figure 2-1 shows various elements of EMDAP as defined in RG 1.203 (Reference 8.2.6). 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.

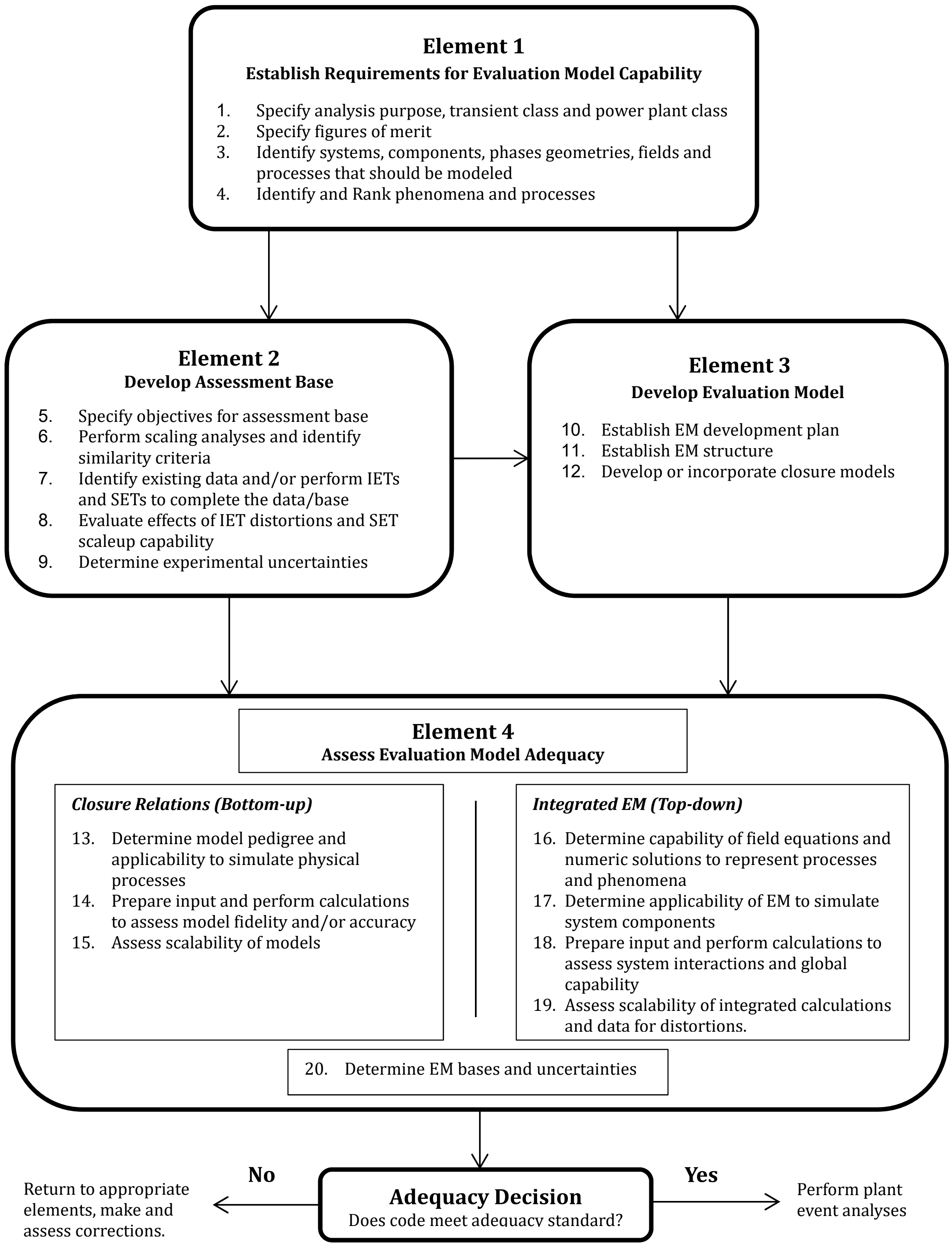

Figure 2-1 Evaluation model development and assessment process

Table 2-1 Evaluation model development and assessment process steps and associated application in the long-term cooling 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 LTC methodology is described in Section 1.1. The NuScale LOCA topical report (Reference 8.2.1) provides an overview of the NPM and a description of the plant operation. This includes the safety systems, the system logic, and operational phases which could occur in the NPM. The regulatory requirements that the methodology is designed to comply with are described in Section 2.2.
2	Specify figures of merit (FOMs).	The NuScale-specific acceptance criteria for LTC are identified in Section 2.3. Section 3.0 describes the NPM long-term cooling PIRT, including FOMs that are used to develop the 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 LTC PIRT discussed in Section 3.0.
4	Identify and rank phenomena and processes.	Section 3.0 describes the long-term cooling PIRT.
Element 2, Develop Assessment Base		
5	Specify objectives for assessment base.	Section 3.0 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 NuScale LOCA EM development; additional assessments against NuScale Integral Systems Test-1 (NIST-1) test data were performed as described in Section 4.0. Other parameters are bounded or treated by a conservative methodology in the LTC analyses.
6	Perform scaling analysis and identify similarity criteria.	A scaling analysis of the LOCA and ECCS has been performed for the NPM based on the NIST-1 facility. The results of the scaling analysis are discussed in the NuScale LOCA topical report (Reference 8.2.1). Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these assessments are adequate for the LTC EM.

EMDAP Step	Description	EM Section
7	Identify existing data and perform integral effects tests (IETs) and separate effects tests (SETs) to complete database.	The NuScale LOCA topical report (Reference 8.2.1) and Section 4.0 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 the NuScale LOCA topical report (Reference 8.2.1), a bottom-up assessment of NRELAP5 closure models and correlations essential to simulate high-ranked PIRT phenomena for LOCA events is presented; this assessment addresses the fidelity of the models and correlations to the appropriate fundamental or SET data. In Reference 8.2.1, 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 LTC plant transient calculations, these assessments are adequate for the LTC evaluation model.
9	Determine experimental uncertainties.	The NuScale LOCA topical report (Reference 8.2.1) and Section 4.0 of this report address 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 8.2.1.
11	Establish EM structure.	The NuScale LOCA topical report (Reference 8.2.1) provides a summary of NRELAP5 models and correlations. For LTC analysis, the plant model is described in Section 4.0. The LTC methodology for thermal-hydraulic calculations is described in Section 5.0 and the methodology for boron precipitation analysis is described in Section 6.0.
12	Develop or incorporate closure models.	The NuScale LOCA topical report (Reference 8.2.1) provides a summary of NRELAP5 models and correlations. A full description of the closure models and the associated equations used in the LTC evaluation model is provided in the NRELAP5 theory and users manuals (Reference 8.2.8).
Element 4, Assess Evaluation Model Adequacy Closure Relations (Bottom-up)		
13	Determine model pedigree and applicability to simulate physical processes.	The NuScale LOCA topical report (Reference 8.2.1) includes a bottom-up assessment of important NRELAP5 models/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 LTC plant transient calculations, these assessments are adequate for the LTC evaluation model.

EMDAP Step	Description	EM Section
14	Prepare input and perform calculations to assess model fidelity and/or accuracy.	Reference 8.2.1 and Section 4.0 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.	The NuScale LOCA topical report (Reference 8.2.1) includes discussion on scalability of 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 LTC plant transient calculations, these assessments are adequate for the LTC 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 8.2.1 and evaluated for their applicability to NPM LOCA phenomena. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these assessments are adequate for the LTC EM.
17	Determine applicability of EM to simulate system components.	The applicability of the NuScale LOCA EM to simulate the NPM system and components is demonstrated by assessment of NRELAP5 against NuScale design-specific SETs and IETs as summarized in Reference 8.2.1 and Section 4.0.
18	Prepare input and perform calculations to assess system interactions and global capability.	The NuScale LOCA topical report (Reference 8.2.1) and Section 4.0 summarize the results of assessment of NRELAP5 against NIST-1 IET data.
19	Assess scalability of integrated calculations and data for distortions.	The NuScale LOCA topical report (Reference 8.2.1) 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 LTC plant transient calculations, these NuScale LOCA EM assessments are adequate for the LTC EM.
20	Determine EM biases and uncertainties.	For the LTC system transient analyses, suitably conservative input is specified in the plant calculations as described in Section 5.0 and Section 6.0, considering the effects on the appropriate acceptance criteria.

3.0 Phenomena Identification and Ranking Table

3.1 Phenomena Identification and Ranking Table Process

The purpose of the NuScale LTC PIRT is to provide an assessment of the relative importance of phenomena and processes that may occur in the NuScale module during LTC in relation to specified FOMs. This assessment is part of the process prescribed by Regulatory Guide 1.203 (Reference 8.2.6).

The current NuScale LTC PIRT has been developed by a panel of experts for the NPM and is built upon the state-of-knowledge at the time of its development. A comprehensive, integrated PIRT was performed for LTC based on the full event progression. The PIRT panel considered the NPM design to identify systems, components, and subcomponents of the design for which phenomena were assessed. The panel then followed the PIRT process. Phenomena were identified and ranked considering their level of importance relative to identified figures-of-merit (FOM) for LTC.

The panel established a knowledge ranking for each of the phenomena. The knowledge level is on a 1 to 4 scale; 4 represents well-known and easily modeled phenomena, while 1 represents a parameter that is not understood and can be difficult to sufficiently model.

3.2 Figures of Merit

During post-LOCA long-term cooling, there are three identified FOMs to which the identified phenomena are compared.

- CHFR: The ratio of the heat flux needed to cause CHF phenomena to the actual local heat flux of a fuel rod. Since the core remains covered with water throughout the event, clad does not significantly heat up. As discussed in Section 2.3, during long-term cooling, maintaining a collapsed liquid level in the riser above the core and demonstrating cladding temperatures remain acceptably low indicate that minimum CHFR is not challenged.

- Coolant collapsed level: The coolant level that results if all voids in the vapor-phase coolant are collapsed. If the core remains covered, significant clad heatup is avoided and it is evident that 10 CFR 50.46 criteria of adequate LTC is established.

- Subcriticality: The condition of a nuclear reactor system, in which nuclear fuel no longer sustains a fission chain reaction (that is, the reaction fails to initiate its own repetition, as it would in a reactor's normal operating condition). A reactor becomes subcritical when its fission events fail to release a sufficient number of neutrons to sustain an ongoing series of reactions, possibly as a result of increased neutron leakage or poisons. Note that for long-term cooling, the core heat source is decay heat.

3.3 Highly Ranked Phenomena

The following sub-sections summarize the phenomena that were ranked high importance by the PIRT panel for the NuScale LTC assessment. The knowledge level assigned by the

PIRT panel and the systems/components where the phenomena were ranked as high importance is also included.

The LTC PIRT was a comprehensive, integrated PIRT for LOCA long-term cooling phase of the event progression. The NPM systems and components, and the relevant phenomena were considered in detail.

As discussed in the LOCA evaluation model, NRELAP5 is NuScale's system thermal-hydraulics code used to calculate the NPM system response during the LOCA long-term cooling event progression. 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 the NPM response to LOCA events (Reference 8.2.1).

How the highly ranked phenomena are addressed in the LTC evaluation model is discussed.

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}}2(a),(c)

3.3.4 {{ }}2(a),(c)

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3.3.5 {{ }}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

3.3.42 {{ }} 2(a),(c)

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}}2(a),(c)

4.0 NRELAP5 Applicability to Long-Term Cooling Analysis

The LTC EM uses an earlier version of the NuScale LOCA EM than the EM described in Reference 8.2.1. This section describes the LTC model, how the LTC EM was developed and the differences between the LTC EM and the NuScale LOCA EM described in Reference 8.2.1. This section also validates the LTC EM for use in LTC assessments by benchmarking to the NIST-1 facility test results.

4.1 Summary of the Long-Term Cooling Model

The NRELAP5 LTC model input file is developed from engineering drawings, calculations, and reference documents. These sources of information provide the numerical information necessary to develop a complete thermal-hydraulic simulation model of the NPM. The types of required information fall into the following NRELAP5 input categories:

- thermal-hydraulic fluid volumes and connecting heat structures
  - reactor vessel primary loop
    - lower plenum
    - core
    - riser
    - pressurizer
    - SG primary side
    - downcomer
  - reactor kinetics
  - reactor vessel secondary system
    - SG secondary
    - steam lines
    - feedwater lines
  - CNV
  - reactor pool
  - DHRS
  - ECCS
  - chemical and volume control system (CVCS) piping for RCS injection, discharge, and pressurizer spray lines
- material properties
- control systems
  - simplified control systems for initialization
    - pressurizer pressure
    - pressurizer level
    - vessel average temperature
    - steam pressure
    - turbine load
  - reactor protection system
  - engineered safety feature controls

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}}2(a),(b),(c)

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}}2(a),(b),(c)

4.2 NRELAP5 Validation and Assessments for Long-Term Cooling

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}}2(a),(b),(c)

4.2.1 Long-Term Cooling Tests at the NIST-1 Facility

The description of the NIST-1 facility is provided within the NuScale LOCA topical report (Reference 8.2.1).

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}}2(a),(b),(c)

4.2.2 NIST-1 Facility NRELAP5 Model

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}}2(a),(b),(c)

4.2.3 Integral Assessment of NIST-1 HP-19a

4.2.3.1 Purpose of Assessment

The HP-19a test results provide a better understanding of phenomena related to an ECCS reactor vent valve spurious opening (without DHRS). The focus in this report is on LTC period.

4.2.3.2 HP-19a Test Progression

The test consists of the following:

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}}2(a),(b),(c)

- {{

}}2(a),(b),(c)

4.2.3.3 NRELAP5 Prediction of HP-19a

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}}2(a),(b),(c)

{{

}}2(a),(b),(c),ECI

Figure 4-1 HP19a transient long-term cooling containment vessel level comparison

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}}2(a),(b),(c),ECI

Figure 4-2 HP19a transient long-term cooling reactor pressure vessel level comparison

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}}2(a),(b),(c),ECI

Figure 4-3 HP19a transient long-term cooling containment vessel pressure comparison

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}}2(a),(b),(c),ECI

Figure 4-4 HP19a transient long-term cooling reactor pressure vessel pressure comparison

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}}2(a),(b),(c),ECI

Figure 4-5 HP19a transient long-term cooling pool level comparison

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}}2(a),(b),(c),ECI

Figure 4-6 HP19a transient long-term cooling lower pool temperature

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}}2(a),(b),(c),ECI

Figure 4-7 HP19a transient long-term cooling pool middle temperature comparison

{{

}}2(a),(b),(c),ECI

Figure 4-8 HP19a transient long-term cooling pool upper temperature comparison

4.2.4 Integral Assessment of NIST HP-19b

4.2.4.1 Purpose of Assessment

The HP-19b test results provide a better understanding of phenomena related to an ECCS reactor vent valve spurious opening (without DHRS), with the presence of non-condensible gas. The focus of this report is on the LTC period.

4.2.4.2 HP-19b Test Progression

The test consists of the following:

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}}2(a),(b),(c)

4.2.4.3 NRELAP5 Prediction of HP-19b

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}}2(a),(b),(c)

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}}2(a),(b),(c)

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}}2(a),(b),(c),ECI

Figure 4-9 HP19b Transient long-term cooling containment vessel level comparison

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}}2(a),(b),(c),ECI

Figure 4-10 HP19b transient long-term cooling reactor pressure vessel level comparison

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}}2(a),(b),(c),ECI

Figure 4-11 HP19b transient long-term cooling containment vessel pressure comparison

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}}2(a),(b),(c),ECI

Figure 4-12 HP19b transient long-term cooling reactor pressure vessel pressure comparison

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}}2(a),(b),(c),ECI

Figure 4-13 HP19b transient long-term cooling pool level comparison

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}}2(a),(b),(c),ECI

Figure 4-14 HP19b transient long-term cooling pool lower temperature comparison

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}}2(a),(b),(c),ECI

Figure 4-15 HP19b transient long-term cooling pool middle temperature comparison

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}}2(a),(b),(c),ECI

Figure 4-16 HP19b transient long-term cooling pool upper temperature comparison

4.2.5 Conclusions from Integral Test Assessments

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}}2(a),(b),(c)

Considering the validation presented in Reference 8.2.1, and this assessment, NRELAP5 is capable of adequately predicting the key parameters of RPV and CNV pressure and level during the LTC timeframe.

4.3 Loss-of-Coolant Accident / Long-Term Cooling Consistency Evaluation

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}}2(a),(b),(c)

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}}2(a),(c)

Figure 4-17 Long-term cooling, loss-of-coolant accident evaluation model nodalization consistency comparison: pressurizer pressure

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}}2(a),(c)

Figure 4-18 Long-term cooling, loss-of-coolant accident evaluation model nodalization consistency comparison: containment pressure

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}}2(a),(c)

Figure 4-19 Long-term cooling, loss-of-coolant accident evaluation model nodalization consistency comparison: riser collapsed liquid level relative to the top of active fuel

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}}2(a),(c)

Figure 4-20 Long-term cooling, loss-of-coolant accident evaluation model nodalization consistency comparison: riser collapsed liquid level relative to the top of active fuel

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}}2(a),(c)

Figure 4-21 Long-term cooling, loss-of-coolant accident evaluation model nodalization consistency comparison: pressurizer level

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}}2(a),(c)

Figure 4-22 Long-term cooling, loss-of-coolant accident evaluation model nodalization consistency comparison: containment collapsed liquid level relative to the top of active fuel

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}}2(a),(c)

Figure 4-23 Long-term cooling, loss-of-coolant accident evaluation model nodalization consistency comparison: core inlet temperature

5.0 Long-Term Cooling Methodology and Evaluation

Section 3.0 describes the important phenomena and parameters to evaluate the FOM, which are tied to the acceptance criteria delineated in Section 2.0. This section establishes the LTC decay heat removal methodology.

The methodology to address maintaining a coolable geometry by precluding boron precipitation is described in Section 6.0. The results presented in Section 5.3 for the minimum RCS temperatures will be used in the Section 6.0 analyses for limiting boron solubility conditions.

5.1 Long-Term Cooling Heat Removal Methodology

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}}2(a),(c)

Section 2.3 establishes that the ECCS long-term cooling analyses address the following scenarios:

- ECCS cooling begins during the short-term event progression. Long-term cooling begins where the NuScale LOCA EM analysis ends when ECCS recirculation flow (RCS steam is released to the CNV through the RVVs, condensed on the CNV walls, and condensed liquid re-enters the RPV through the RRVs), pressures and levels in the RPV and CNV approach a stable equilibrium condition.
- Transition from DHRS cooling to ECCS cooling is considered. LTC analyses demonstrate that if DHRS provides passive decay heat removal until either the IAB setpoint is reached or 24 hours is reached such that the ECCS timer expires, and then decay heat removal transitions to ECCS, the module(s) will remain in a safe, stable condition for up to 72 hours following the event. Decay Heat Removal System transition cases also include consideration for SGTF to address inventory loss prior to isolation of the SG.

In order to demonstrate sufficient capability to maintain safe shutdown conditions for 72 hours without operator actions and without nonsafety-related power, two events are considered to ensure that the ECCS can adequately provide long-term core cooling. The two events are:

- LOCA in conjunction with a loss of normal AC power scenario: The CVCS letdown line break at the RPV is analyzed. It is modeled as a double-ended guillotine break of the CVCS discharge pipe at the RPV. At the time of the letdown break initiation, the normal letdown flow path is closed and the break paths are instantly opened.

  Based on the results of the LOCA analysis of a spectrum of various break sizes and conditions, the long term trend of all break sizes and locations converge. Therefore, use of only the CVCS letdown line break in the sensitivity calculations is appropriate and satisfactory.

- SGTF, leading to short-term DHRS cooling which then transitions to cooling by ECCS, either on IAB release differential pressure or 24 hours when the ECCS timer expires.

Analysis of LTC only considers long term decay heat removal via the ECCS. The sequence of events leading to long-term ECCS cooling are described below.

1. ECCS valves open. This may occur after short term DHRS cooling in the event of a non-LOCA transient in conjunction with a loss of normal AC power.
2. RCS level begins to drop while CNV level rises.
3. Minimum level in the RCS occurs. The minimum level reached occurs during the short term LOCA phase.
4. Condensation in the CNV increases CNV level.
5. Recirculation flow from the CNV to the RPV through the RRV is established.
6. Long term levels stabilize, assuming the pool boundary condition is constant. The stabilization of the long term levels begins the long-term cooling phase of the event.

Analysis of the sensitivities is limited to three days. This timeframe is considered acceptable because: (1) the most severe conditions will have been captured within the 72 hour window analyzed, and any conditions that could reasonably be expected to occur beyond this time period are thus bounded by the 72 hour calculation, and (2) after 72 hours, operator actions can be credited. When realistic initial operating levels, temperatures, and decay heat are considered, the scope of the LTC analyses bound the expected level decrease in the reactor pool ultimate heat sink over 30 days, for up to twelve modules transferring decay and residual heat to the reactor pool.

In the NuScale plant design, up to twelve modules may be operating. The safety systems credited for mitigation of the design basis events are module-specific except for the shared reactor pool portion of the UHS. The LTC analyses consider a range of reactor pool boundary conditions to sufficiently address the effects of one or more modules, up to all twelve modules, transferring decay heat into the reactor pool.

These calculations demonstrate the trend in both the RCS temperatures (including fuel cladding temperatures) and collapsed liquid level in the riser. For reference in the figures, {{ }}2(a),(c),ECI The single sensitivity cases show collapsed

liquid level in the riser relative to the inside bottom of the RPV; the remaining cases show the level relative to TAF. These parameters are used to confirm both the capability of the ECCS to provide long-term cooling as well as show the conditions do not reach those that would induce precipitation of boric acid. For the limiting cases, additional plots are shown for RCS core inlet temperature, RCS pressure, CNV pressure, CNV level, cladding temperature, and flow through one of the RVVs.

5.2 Sensitivity Considerations

The base case to demonstrate LTC acceptability is a letdown line break LOCA case assuming nominal conditions, {{ }}[2(a),(c)] Non-LOCA DHRS to ECCS transition cases were also evaluated. The limiting non-LOCA event is the SGTF event.

The parameters that are evaluated as part of the sensitivity analysis are based on the findings in the PIRT from Section 3.0. These parameters are as follows:

- single active failure, ECCS valve failure to open is the relevant single active failure to consider in the LTC analyses
- decay heat, ranging from no decay heat to 1 0 percent of nominal
- heat transfer from the RPV to CNV, ranging from adiabatic to 1000 percent of nominal
- heat transfer from the CNV to reactor pool, ranging from 20 percent to 1000 percent
- reactor pool temperature, ranging from 40 degrees F to 210 degrees F
- reactor pool level, down to 45 feet
- reactor pool volume effect on calculated pool temperature heatup from initial conditions
- non-condensable gas effect
- pressurizer level, down to 20 percent of nominal

In addition, inventory loss through possible containment leakage is also considered and inventory loss due to containment leakage is found to be insignificant. With conservative assumptions of saturated vapor and an inlet pressure of 1000 psia, the calculated leakage resulted in a decrease in riser level of 0.41 inches per 24 hours. Over 72 hours, the resultant loss of 1.23 inches of collapsed liquid level in the riser region has no impact on the conclusions drawn from the analysis.

5.3 Demonstration of Limiting Results

Combined effect cases were evaluated to determine the limiting conditions that could develop during the LTC phase. These cases are as follows:

- minimum cooldown with letdown line break (LDBRK)
  - {{

    }}2(a),(c)
- maximum cooldown with LDBRK
  - {{

    }}2(a),(c)
- maximum cooldown with SGTF
  - includes 20 percent pressurizer level
  - 100 percent of the ANS decay heat standard, including the actinide contribution, is the conservatively high decay heat assumed for this scenario.

Table 5-1 describes the cases that were evaluated utilizing the criteria described in this list. A more detailed description of each case is provided in Sections 5.3.1 through 5.3.4.

Table 5-1 Descriptions of long-term cooling single parameter cases

Sensitivity Group	Case Name	Description
LDBRK Min Cooldown Sens.	LDBRK.mincool.RRV.RVV.45ft.hiNCG3	{{ }}2(a),(c)
	LDBRK.mincool.RRV.RVV.55ft.hiNCG3	{{ }}2(a),(c)
LDBRK Max Cooldown Sensitivities: Nom PZR Level, 0.8x Decay Heat	LDBRK.maxcool.DK0.8.RRV.RVV.new.rvv.fix.att2.full	{{ }}2(a),(c)
LDBRK Max Cooldown Sensitivities: Nom PZR Level, 1.2x Decay Heat	LDBRK.maxcool.DK1.2.RRV.RVV.new.rvv.fix.att2.full	{{ }}2(a),(c)
LDBRK Max Cooldown Sensitivities: 20% PZR Level, 1.2x Decay Heat	LDBRK.maxcool.DK1.2.RRV.RVV.new.rvv.fix.lvl20	{{ }}2(a),(c)

Sensitivity Group	Case Name	Description
SGTF Max Cooldown Sensitivities: Min PZR Level	SGTF.LtMxT.ecc24.LTC.poolTmin.compr.lvl20	{{ }}2(a),(c)

5.3.1 Minimum Cooldown Rate

To evaluate the combined impact of all conditions that result in a slower cooldown rate, the base case ('LDBRK') was re-performed with the following changes that were based on the results of the single effect sensitivities described in the previous sections:

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}}2(a),(c)

Figure 5-1 through Figure 5-14 show that the minimum cooldown rate does not result in a larger RCS temperature and pressure, or a larger CNV pressure compared to the base case. In addition, the long-term collapsed liquid level in the riser is not significantly different with the slower cooldown rate, indicating a long-term level difference of less than a foot. The minimum collapsed liquid level for this condition was found to be nonlimiting, however the fuel cladding temperature was found to be highest at 72 hours compared to the other limiting cases.

In Figure 5-1 through Figure 5-14, the long-term maximum cladding temperature is seen to decrease to a level well below those seen in the short term. The cladding temperature follows the saturation temperature. This indicates CHF does not occur during LTC with minimum cooldown rate. Although Figure 5-1, Figure 5-2, Figure 5-3, and Figure 5-6 indicate that the system pressures and therefore saturation temperatures are calculated to increase at the end of the 72 hour period, the results of the limiting case assuming an initial pool level of 45 feet demonstrate that even assuming a lower reactor pool level, sufficient decay heat is removed during the 72 hour time period (Figure 5-8 through Figure 5-10, and Figure 5-13).

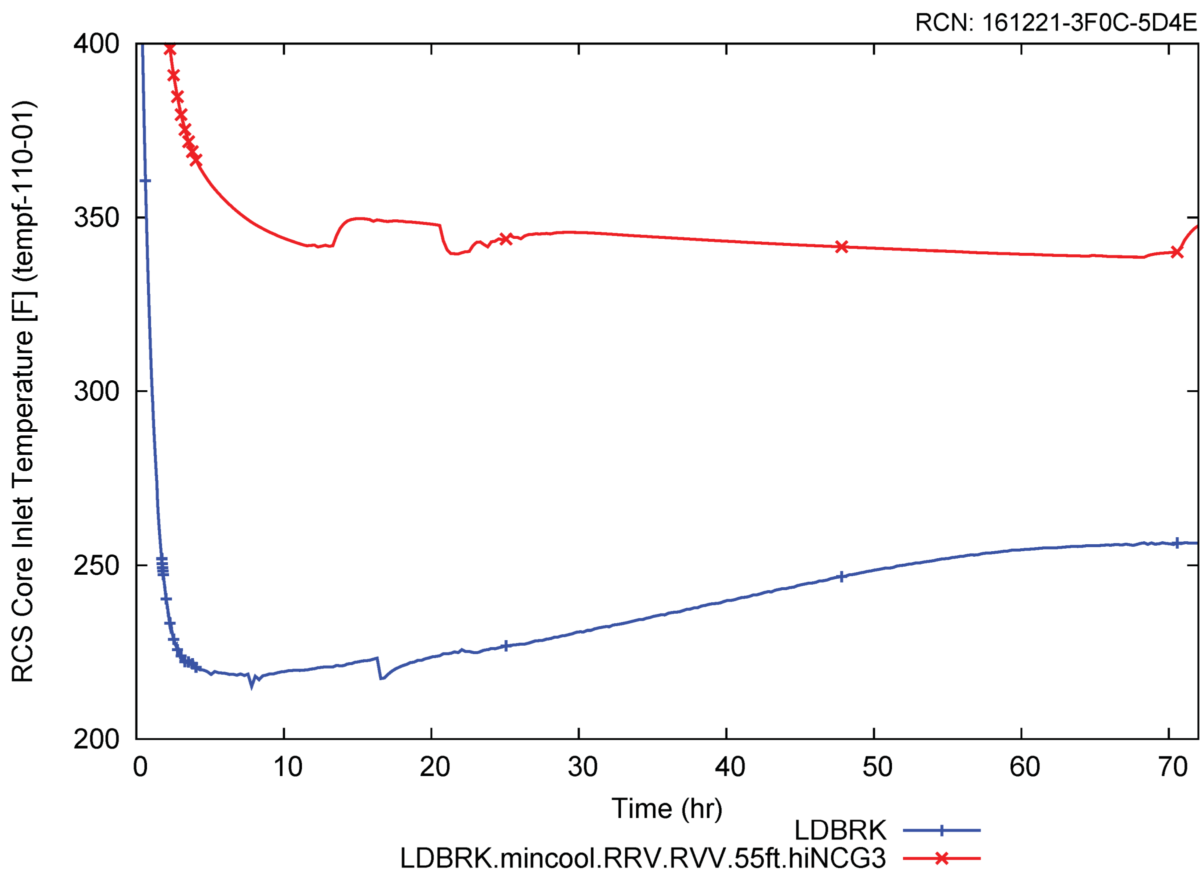

Figure 5-1 Reactor coolant system core inlet temperature for minimum cooldown rate, 55 feet level

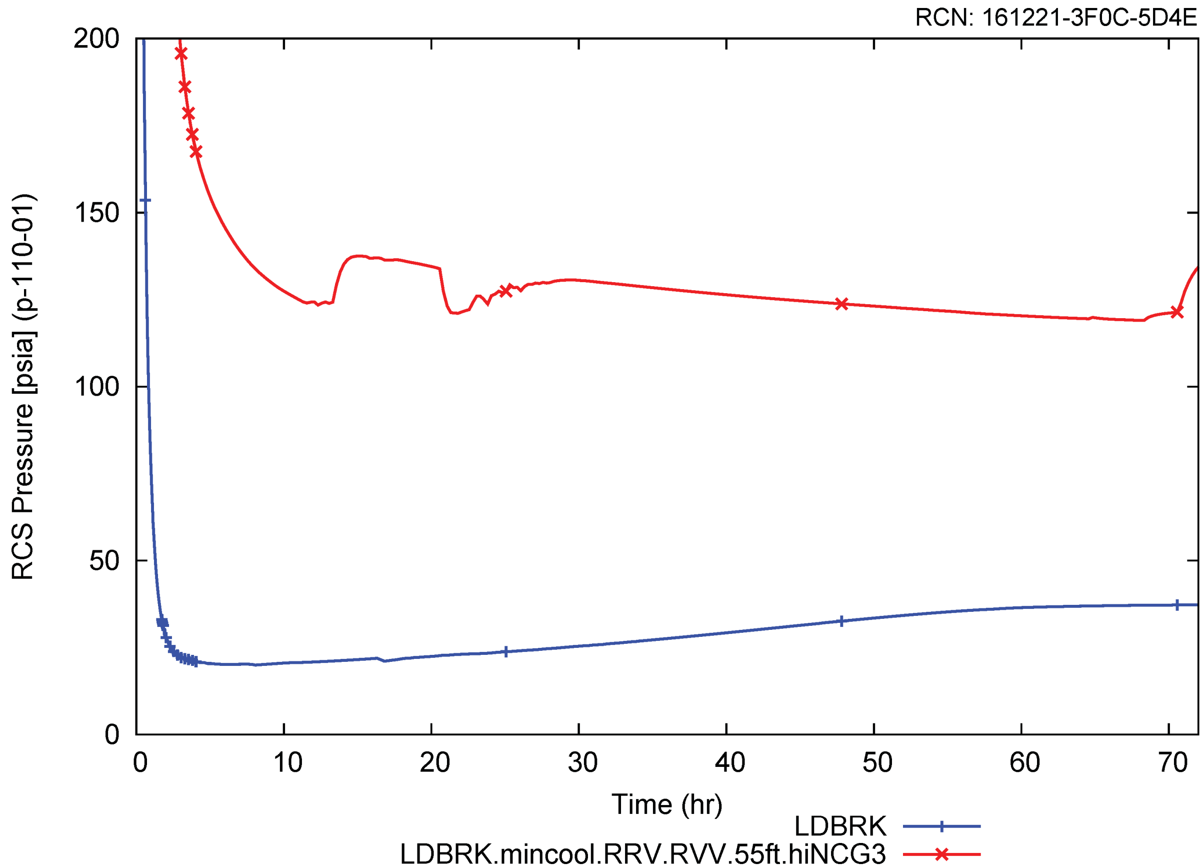

Figure 5-2 Reactor coolant system pressure for minimum cooldown rate, 55 feet level

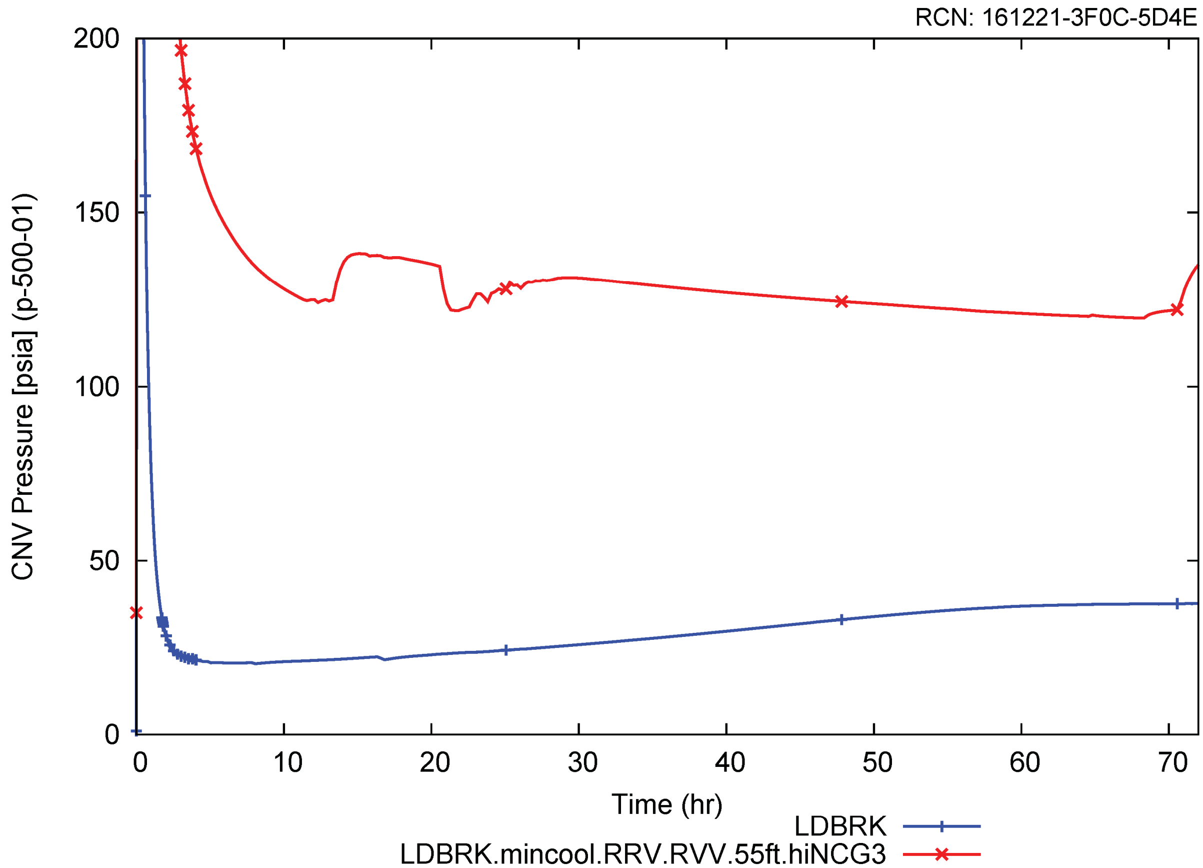

Figure 5-3 Containment vessel pressure for minimum cooldown rate, 55 feet level

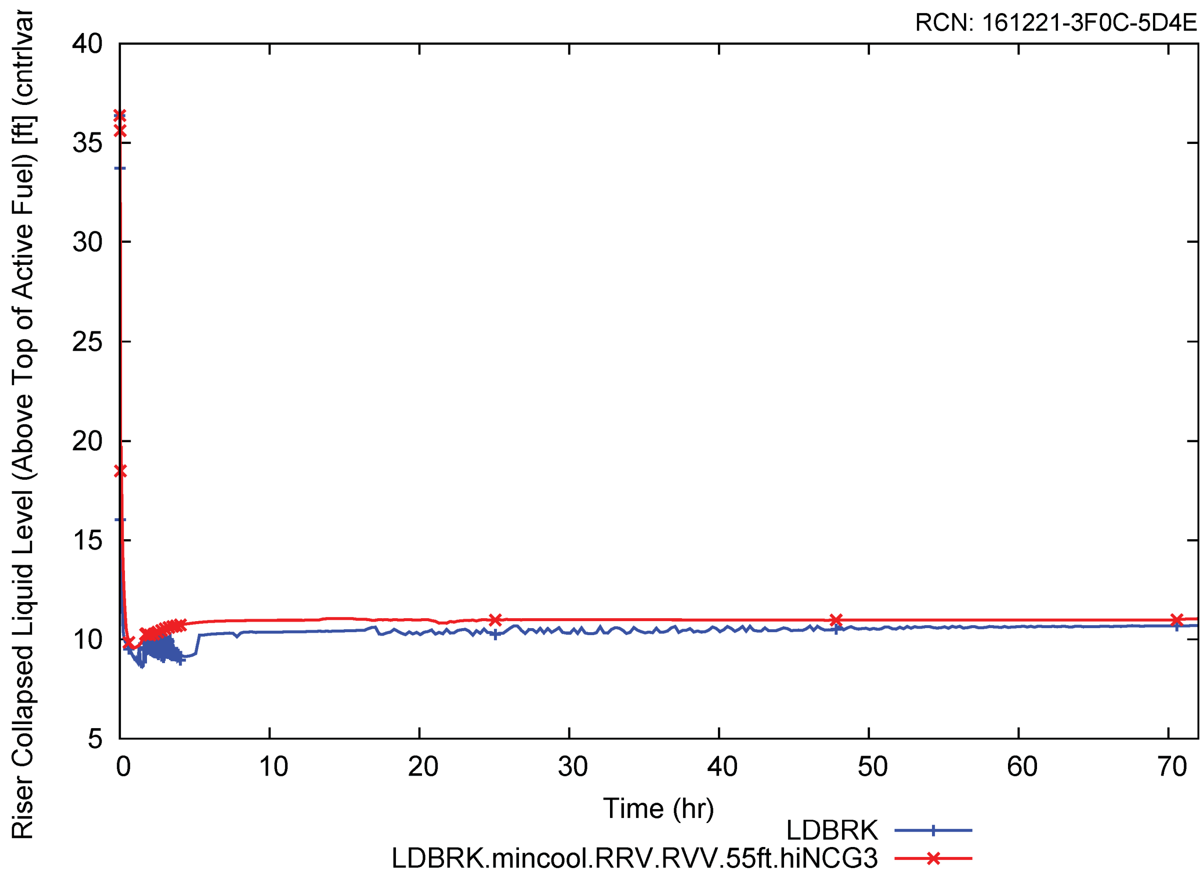

Figure 5-4 Reactor coolant system collapsed liquid level above top of active fuel for minimum cooldown rate, 55 feet level

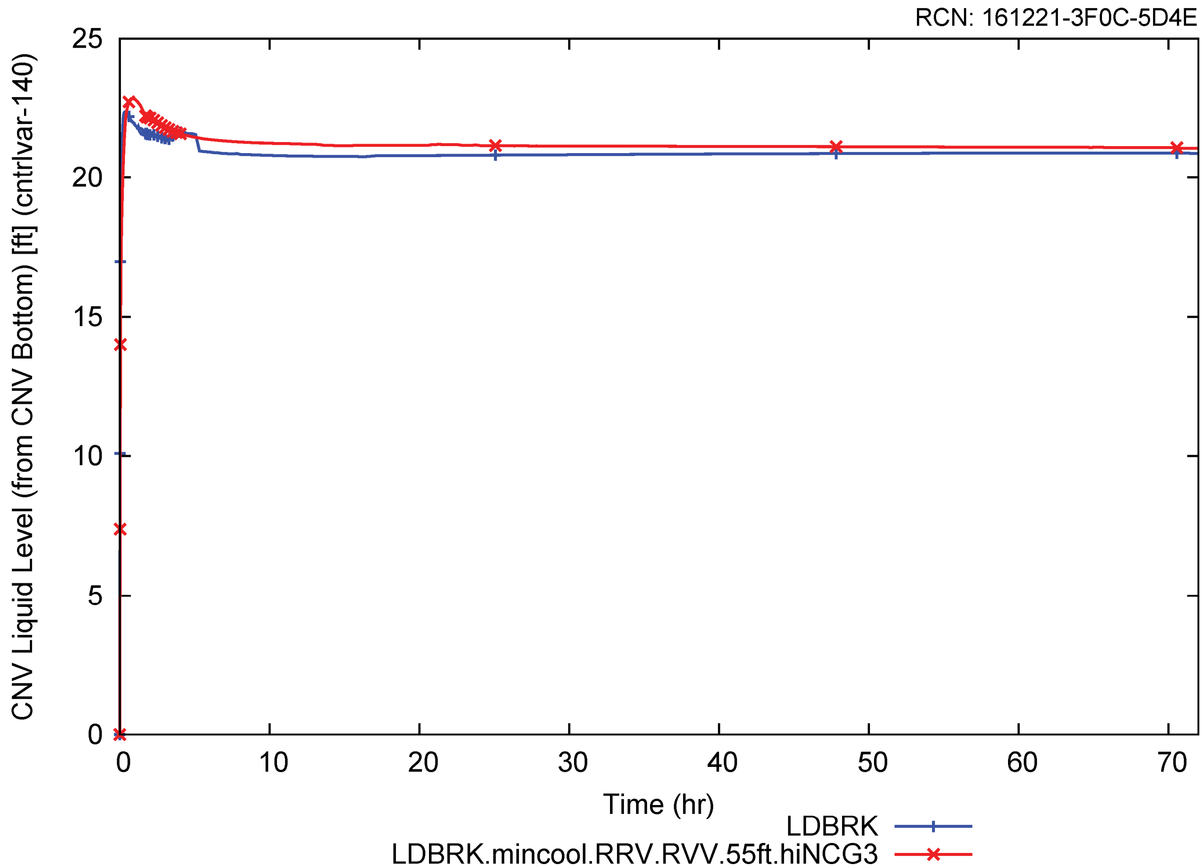

Figure 5-5 Containment vessel level for minimum cooldown rate, 55 feet level

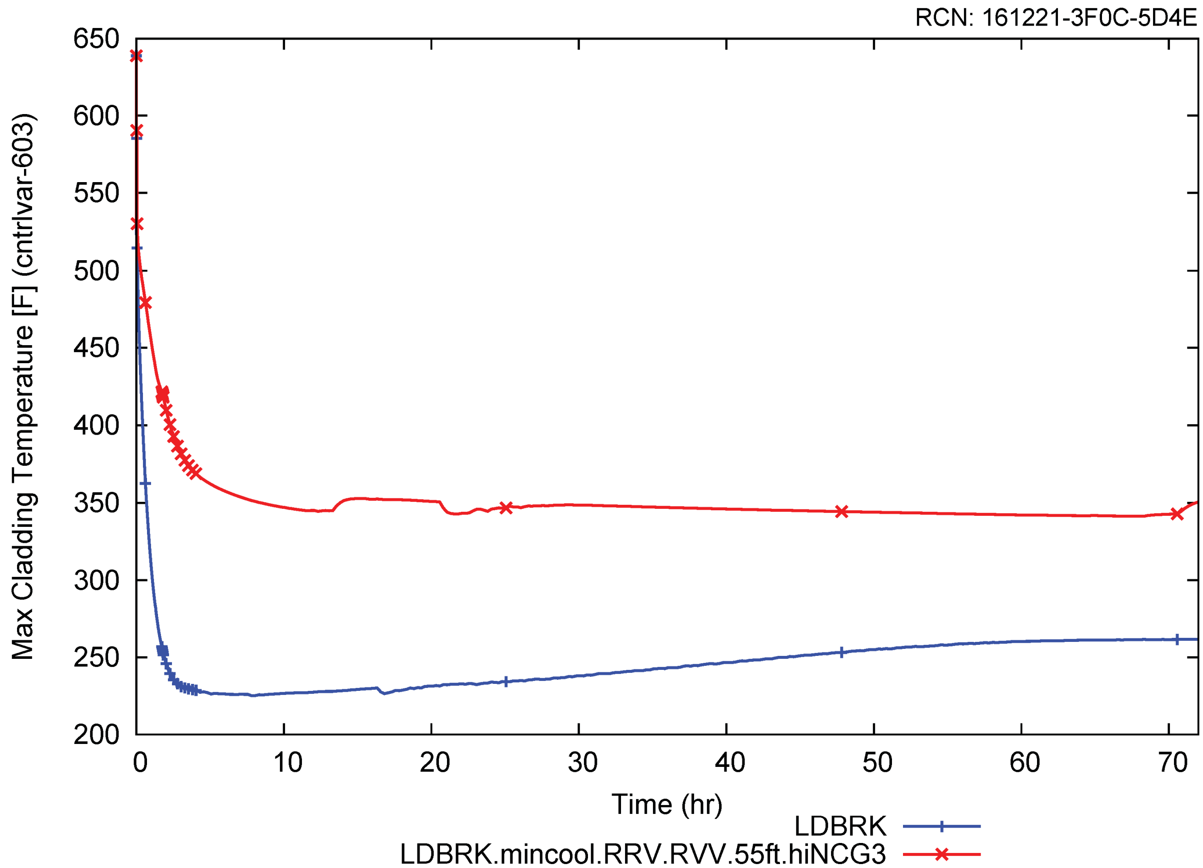

Figure 5-6   Peak cladding temperature for minimum cooldown rate, 55 feet level

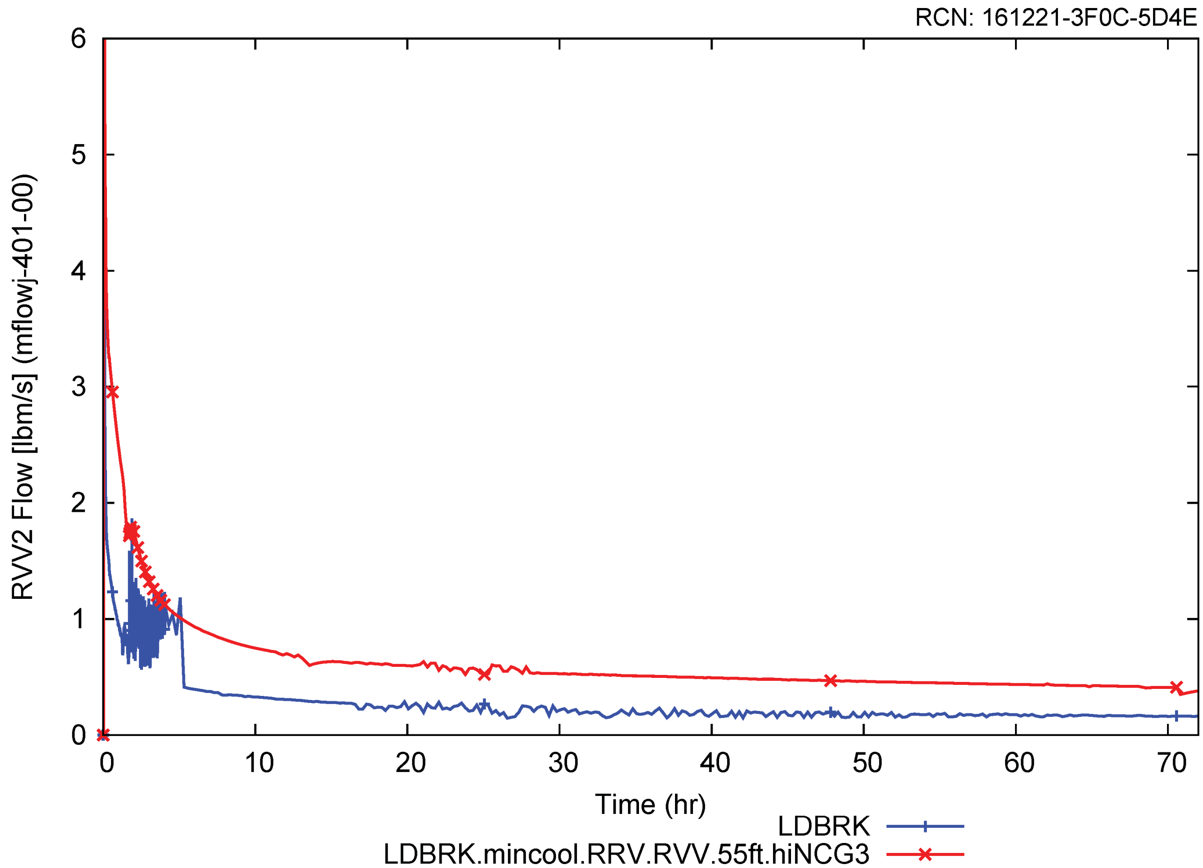

Figure 5-7 Flow through single reactor vent valve for minimum cooldown rate, 55 feet level

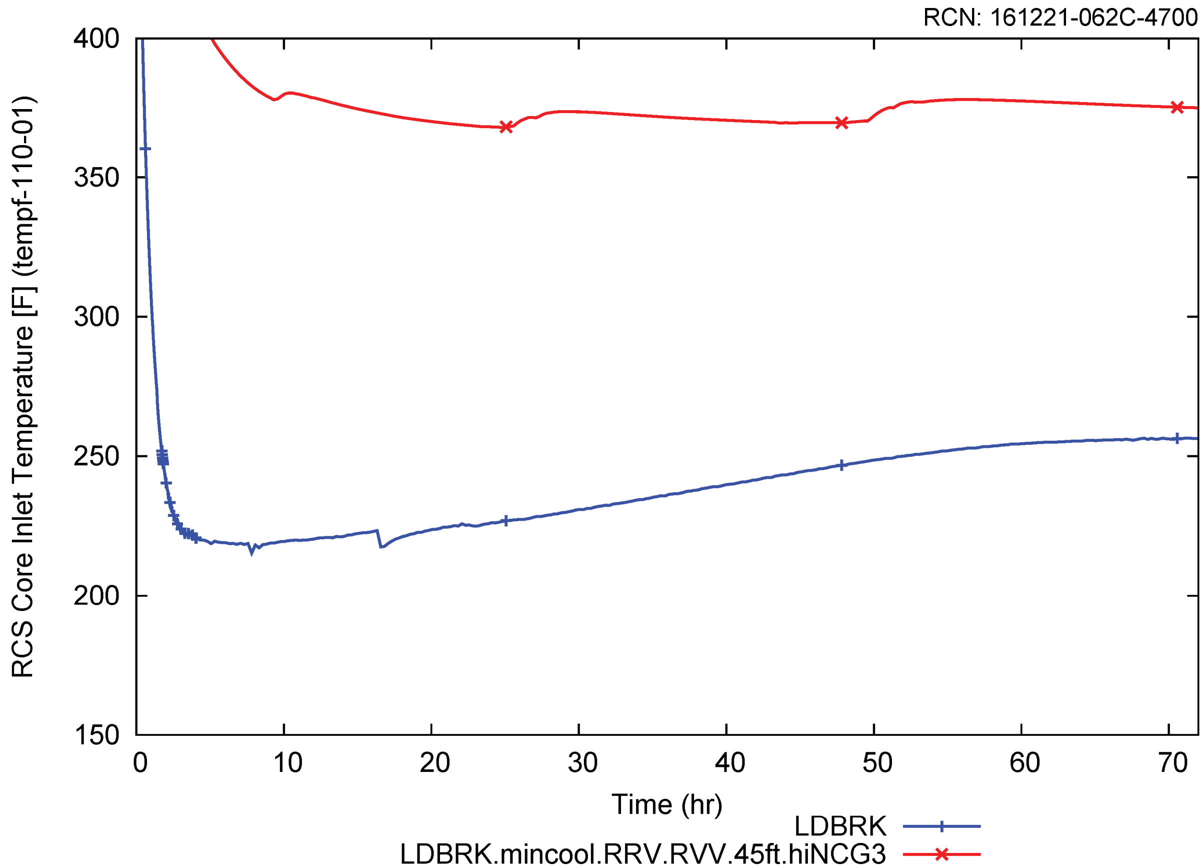

Figure 5-8 Reactor coolant system core inlet temperature for minimum cooldown rate, 45 feet level

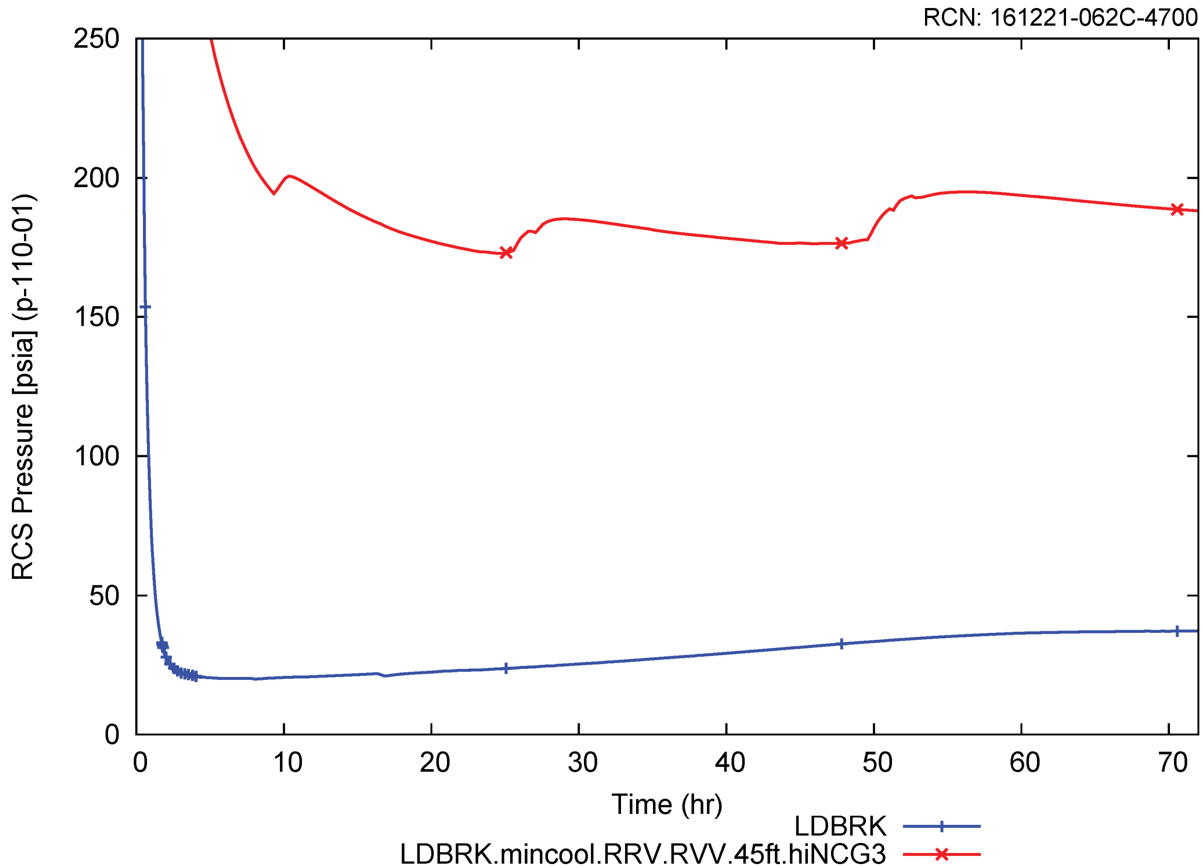

Figure 5-9   Reactor coolant system pressure for minimum cooldown rate, 45 feet level

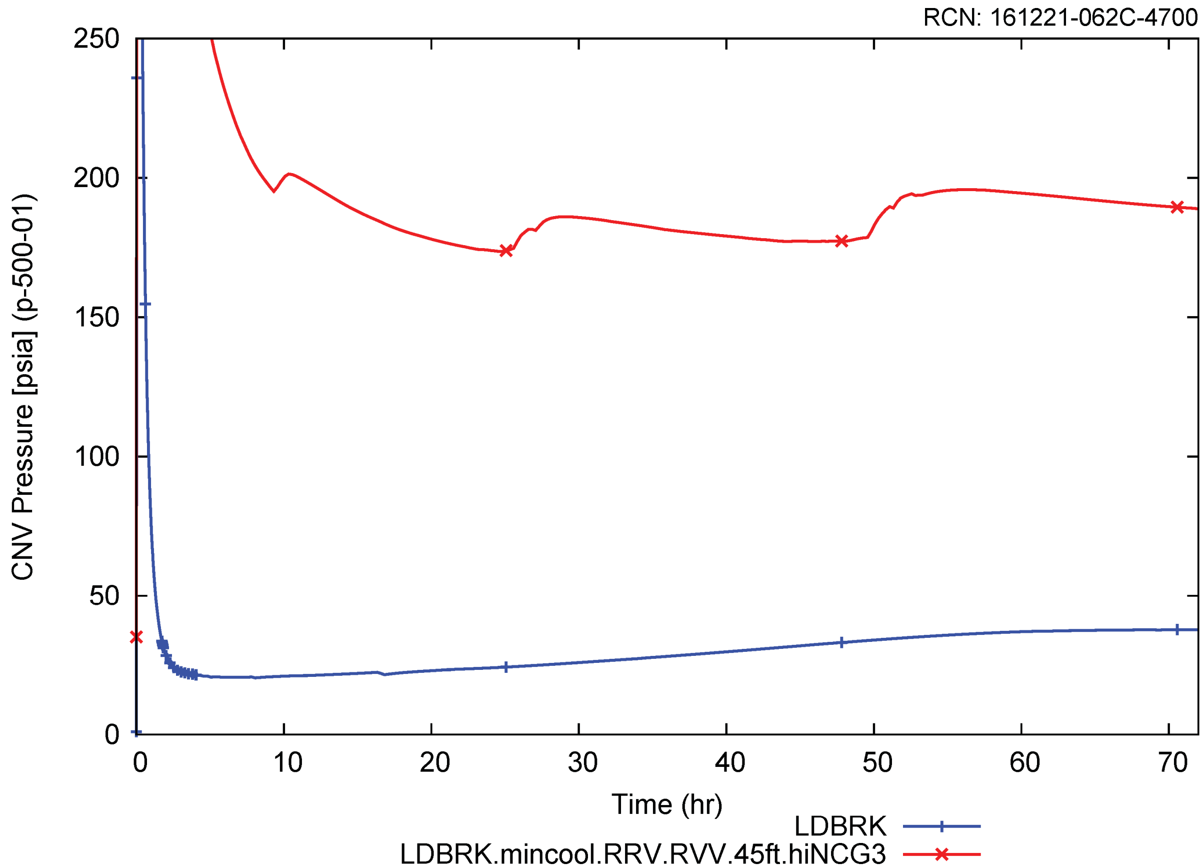

Figure 5-10 Containment vessel pressure for minimum cooldown rate, 45 feet level

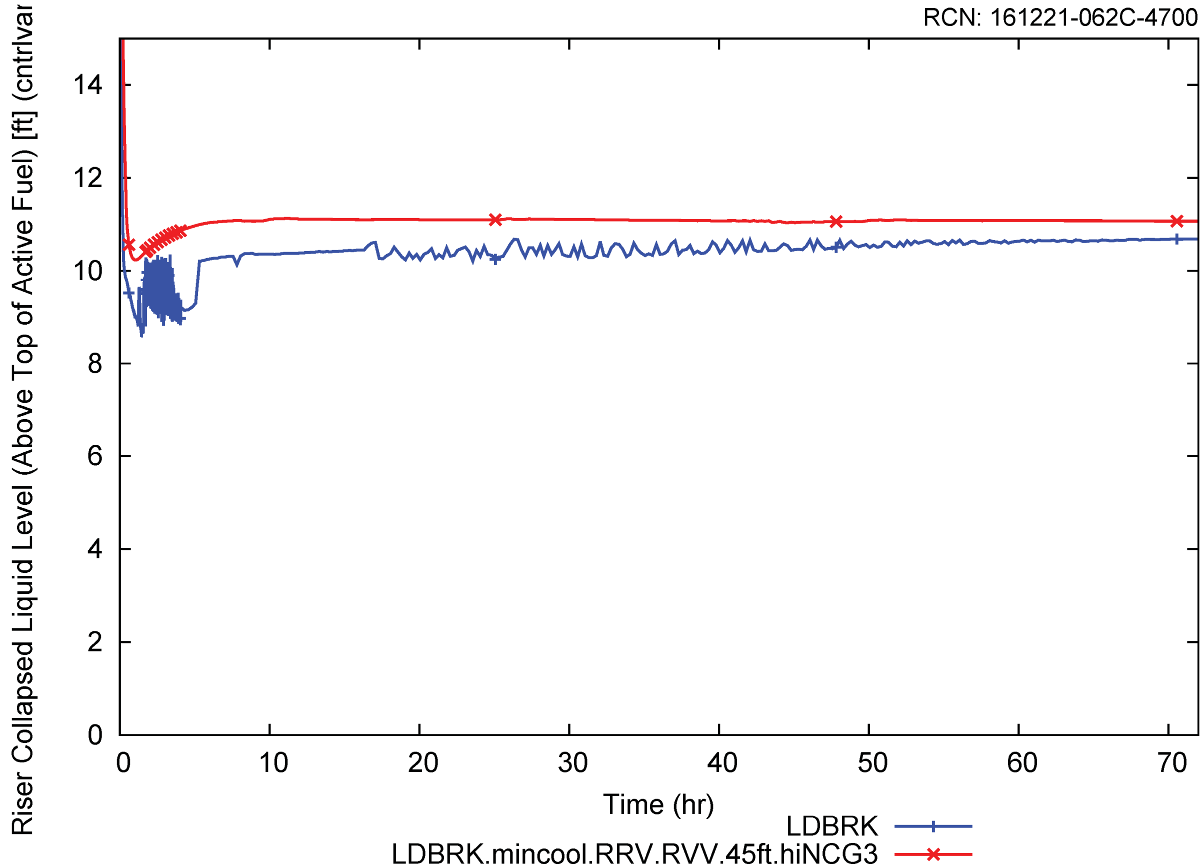

Figure 5-11 Reactor coolant system collapsed liquid level above top of active fuel for minimum cooldown rate, 45 feet level

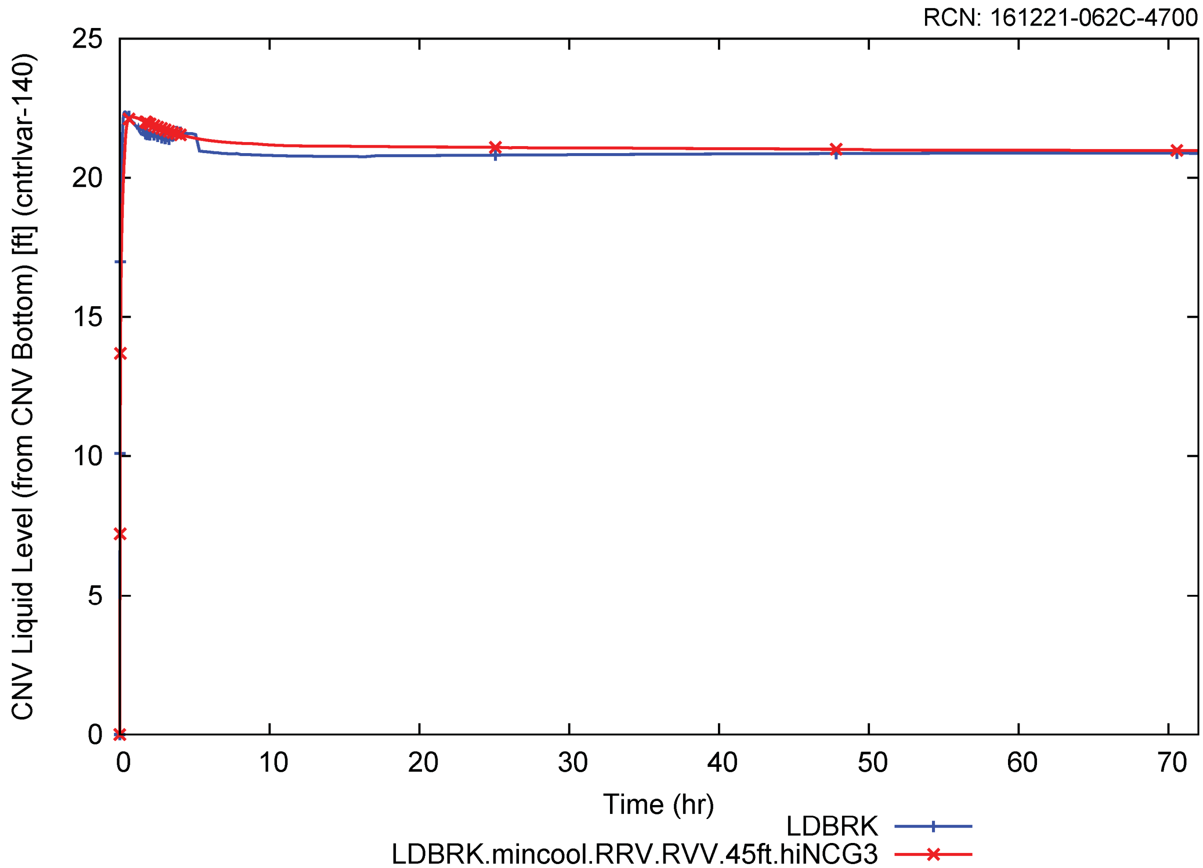

Figure 5-12 Containment vessel level for minimum cooldown rate, 45 feet level

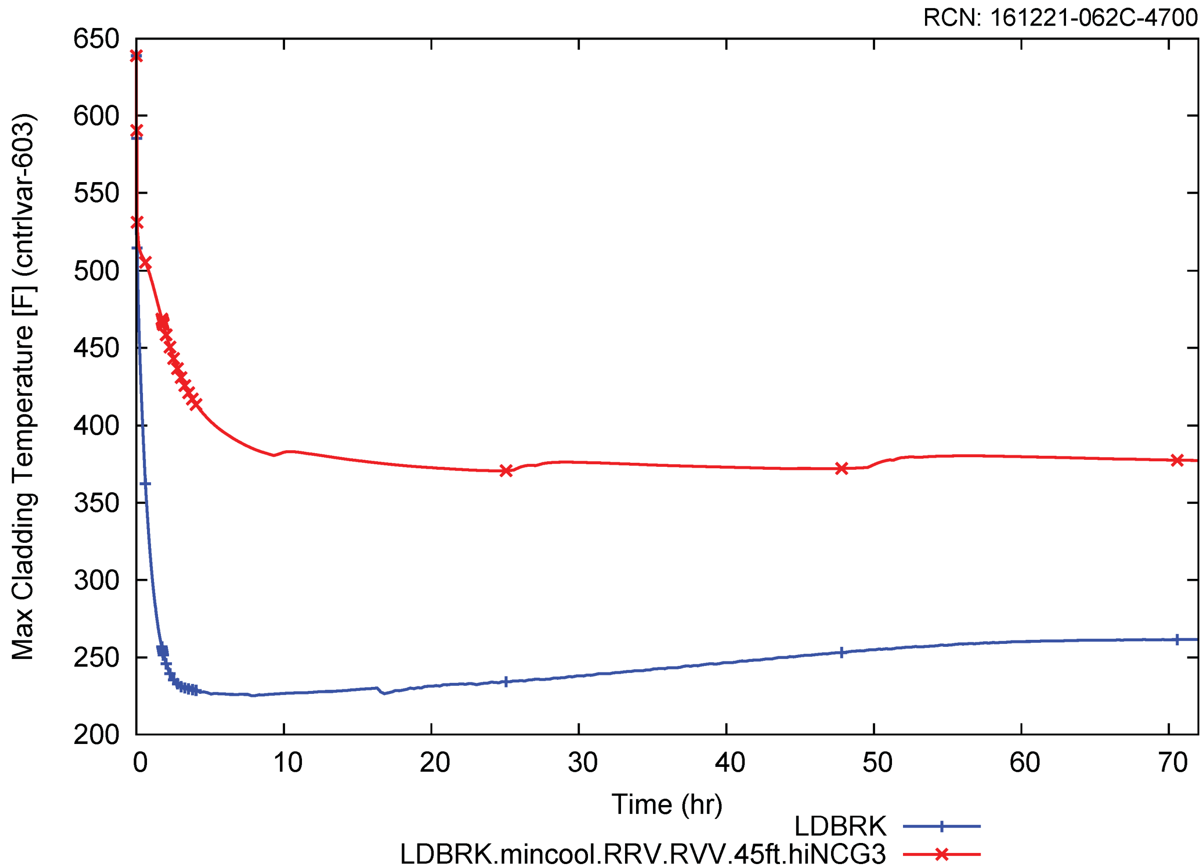

Figure 5-13 Peak cladding temperature for minimum cooldown rate, 45 feet level

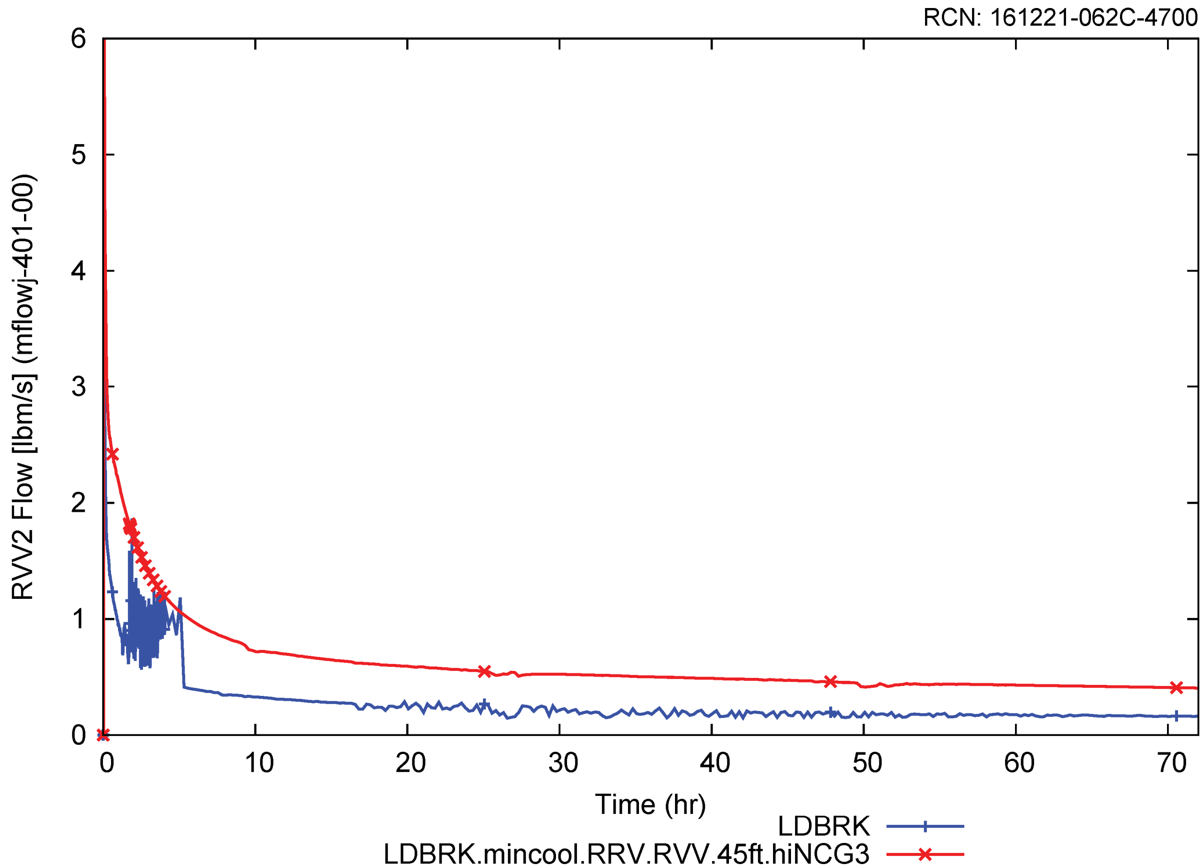

Figure 5-14 Flow through single reactor vent valve for minimum cooldown rate, 45 feet level

5.3.2 Maximum Cooldown Rate

To evaluate the combined impact of all conditions that result in a faster cooldown rate, the base case letdown break was re-performed with the following changes that were based on the results of the single effect sensitivities described in the previous sections:

- {{

}}2(a),(c)

- {{

}}2(a),(c)

In addition to the changes listed above, for some of the runs it was necessary to isolate heat transfer to the secondary side (SG tubes and DHRS) in order to allow for code convergence. Although this is not consistent with biasing for maximum cooldown, the effect was found to be negligible based on sensitivity calculations.

Based on the results presented in this section, the maximum cooldown presents the limiting conditions in terms of collapsed level and minimum RCS temperature. The RCS and CNV pressures become sub-atmospheric, further reducing the temperature by significantly decreasing the saturation temperature. However, for all of the cases presented here, adequate core cooling is maintained as the collapsed level does not fall below the TAF, and the RCS temperature remains stable and acceptably low. The minimum collapsed liquid level was found to be 2.271 feet above the TAF for the 1.2 multiplier decay heat case.

Figure 5-15 through Figure 5-35 demonstrate that the long-term maximum cladding temperature decreases to a level well below those seen in the short-term LOCA results. This indicates CHF does not occur during LTC with maximum cooldown rate conditions.

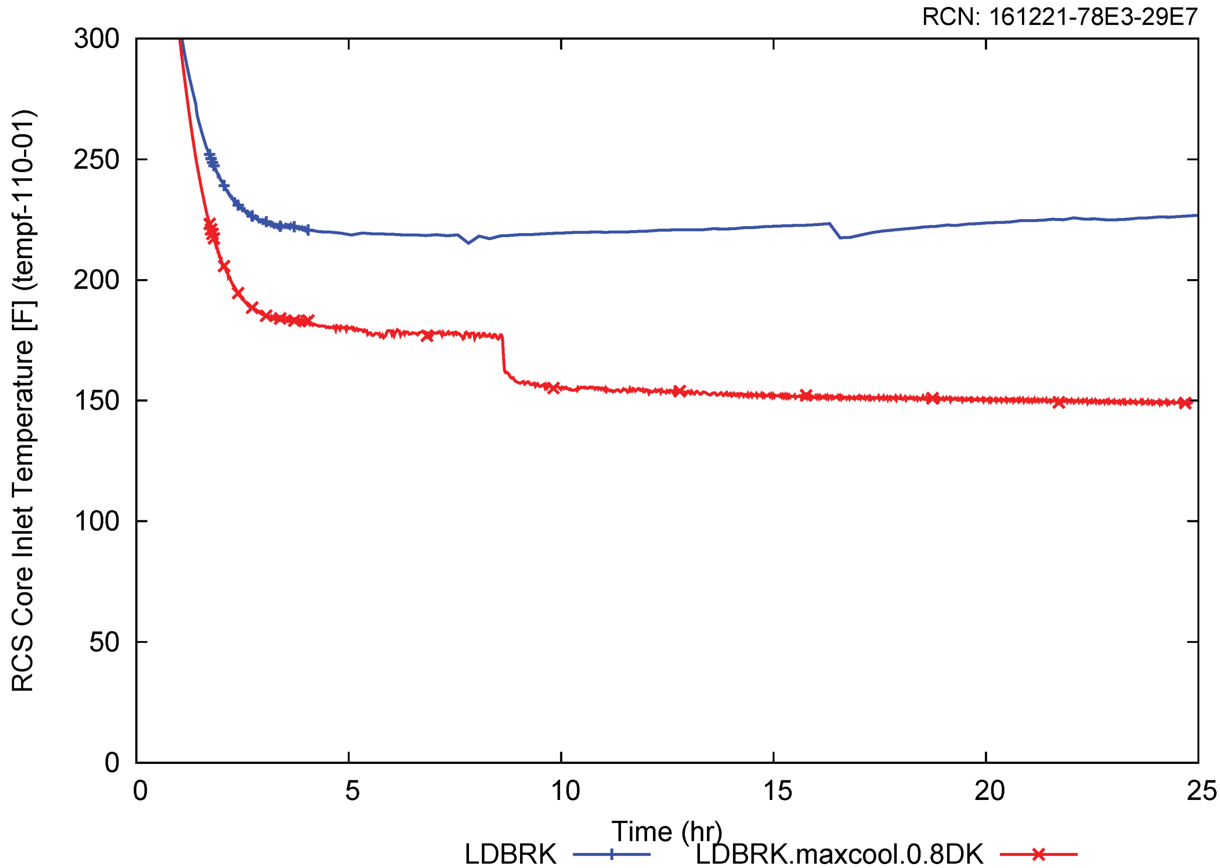

Figure 5-15 Reactor coolant system core inlet temperature for maximum cooldown, 0.8 decay heat

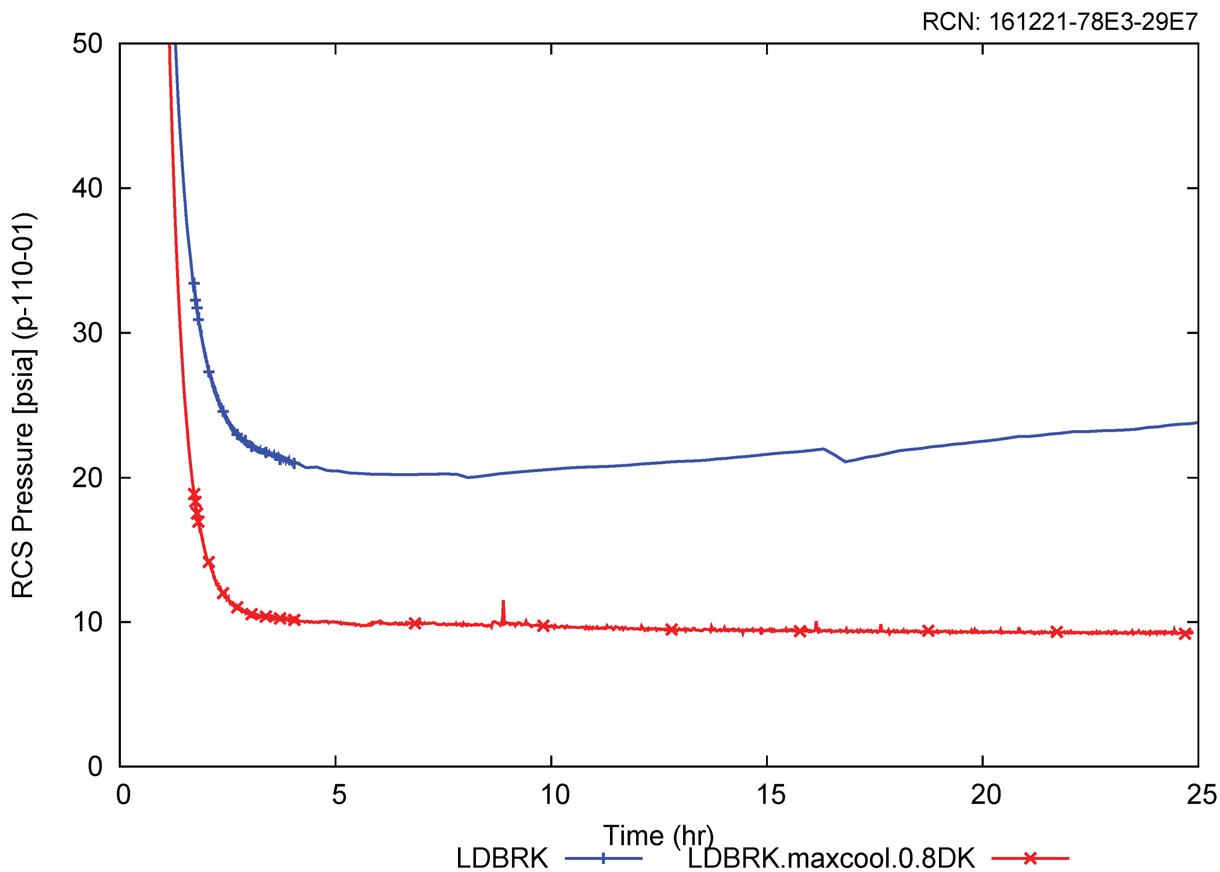

Figure 5-16 Reactor coolant system pressure for maximum cooldown, 0.8 decay heat

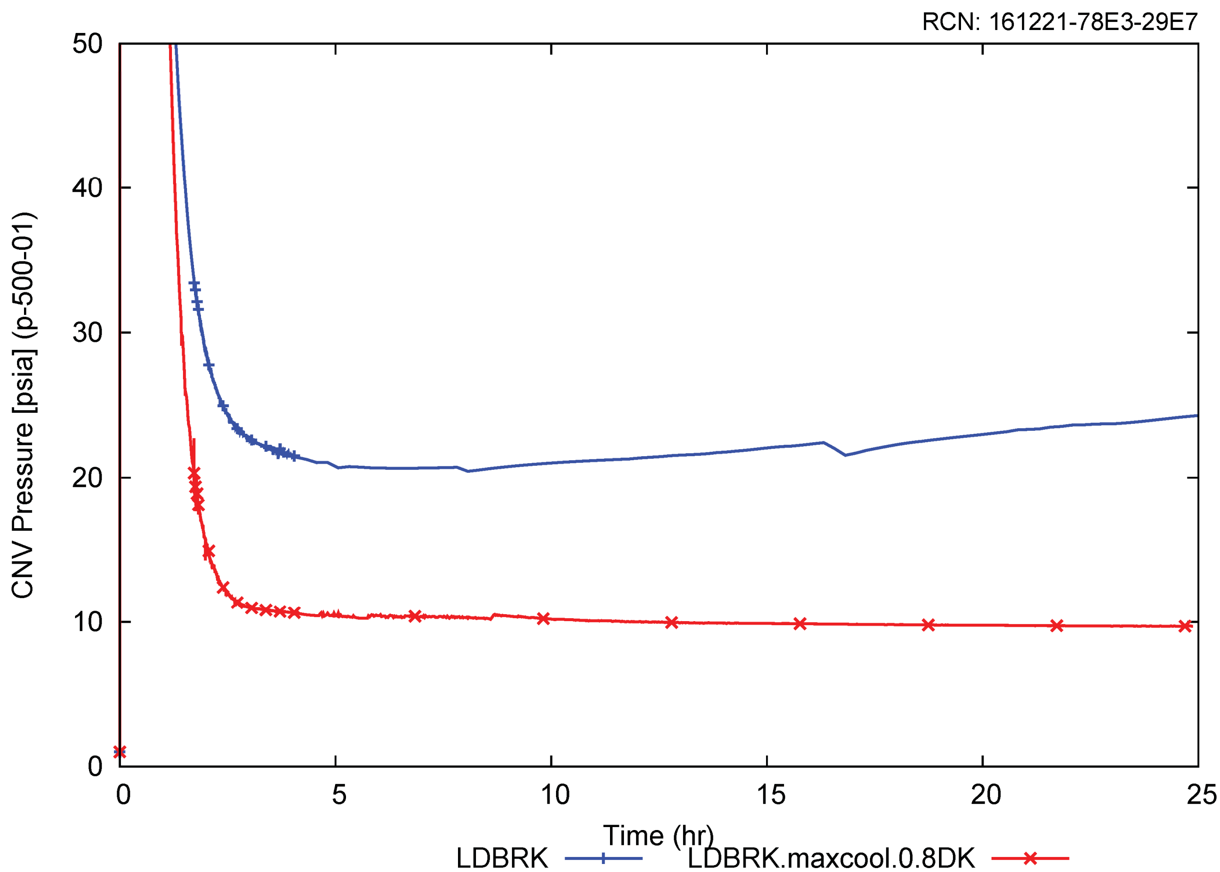

Figure 5-17 Containment vessel pressure for maximum cooldown, 0.8 decay heat

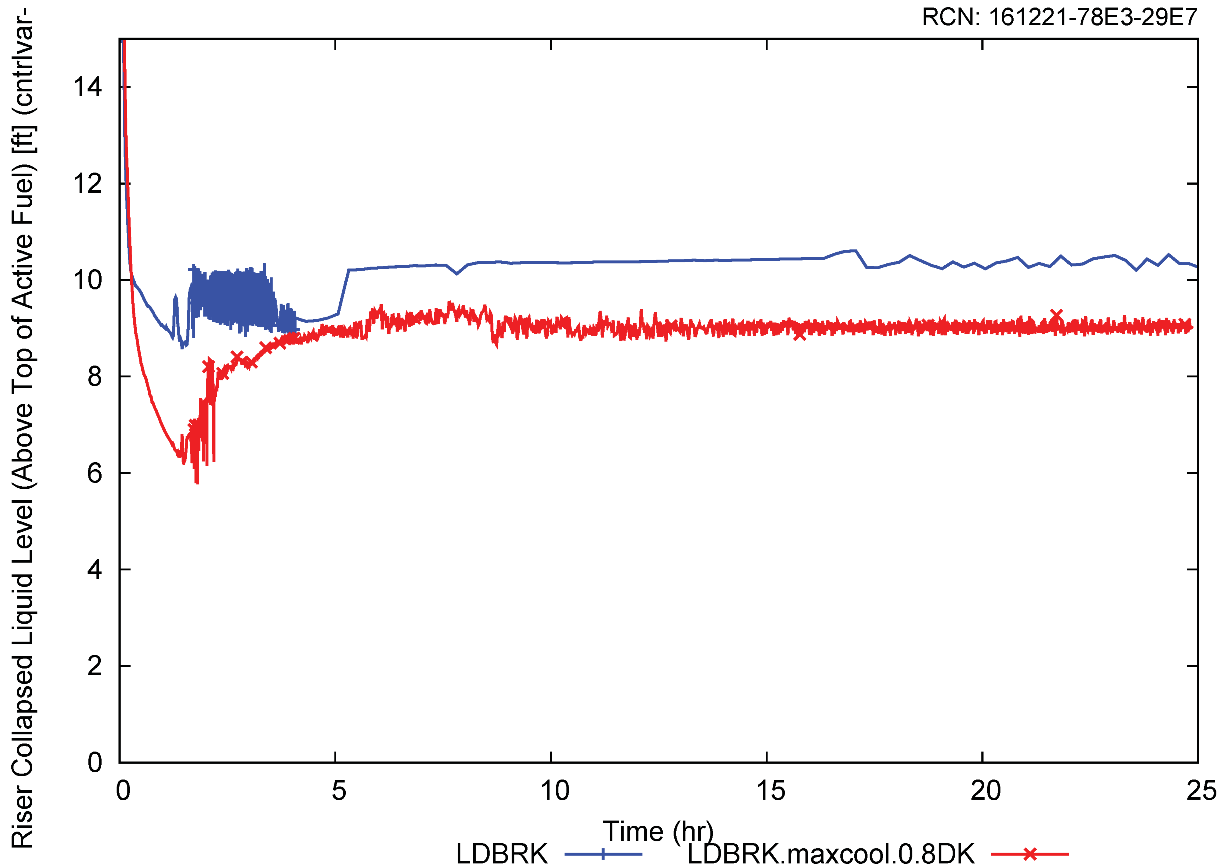

Figure 5-18 Reactor coolant system collapsed liquid level above top of active fuel for maximum cooldown, 0.8 decay heat

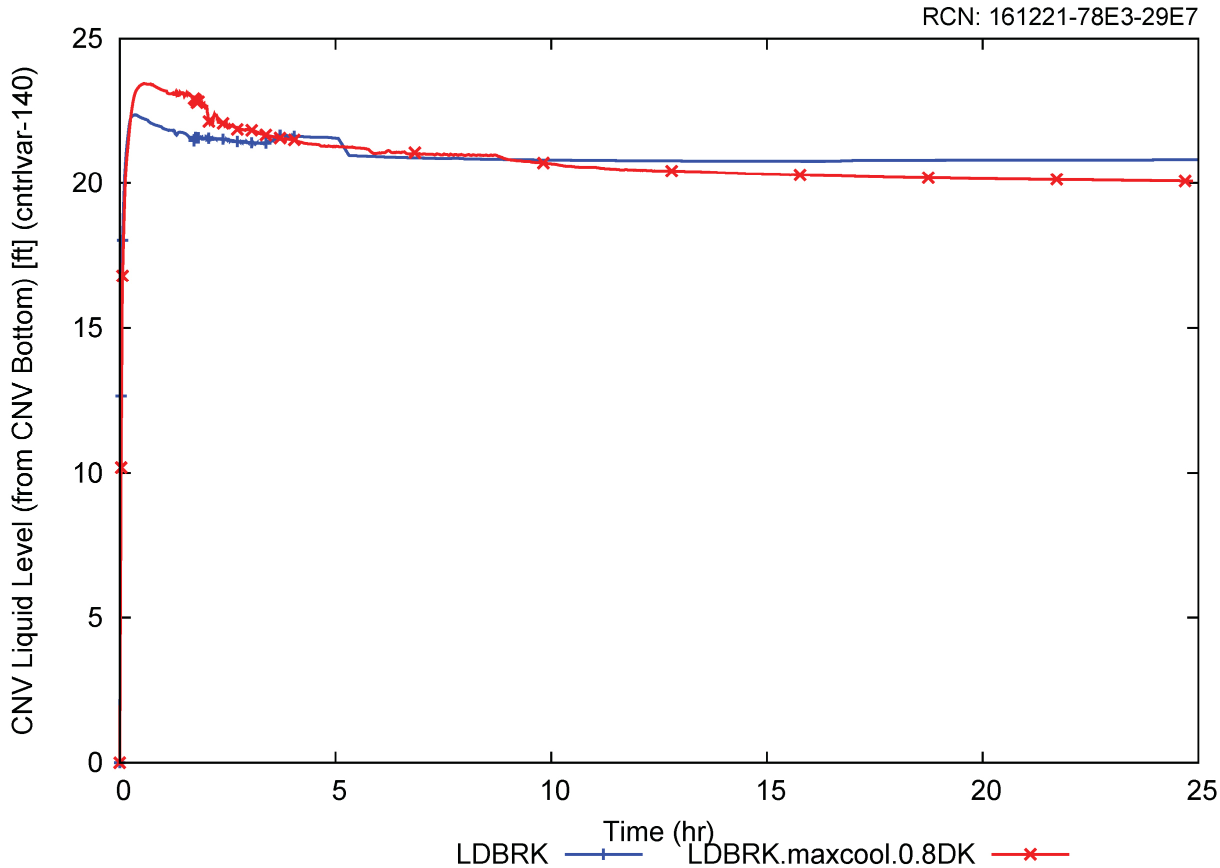

Figure 5-19 Containment vessel level for maximum cooldown, 0.8 decay heat

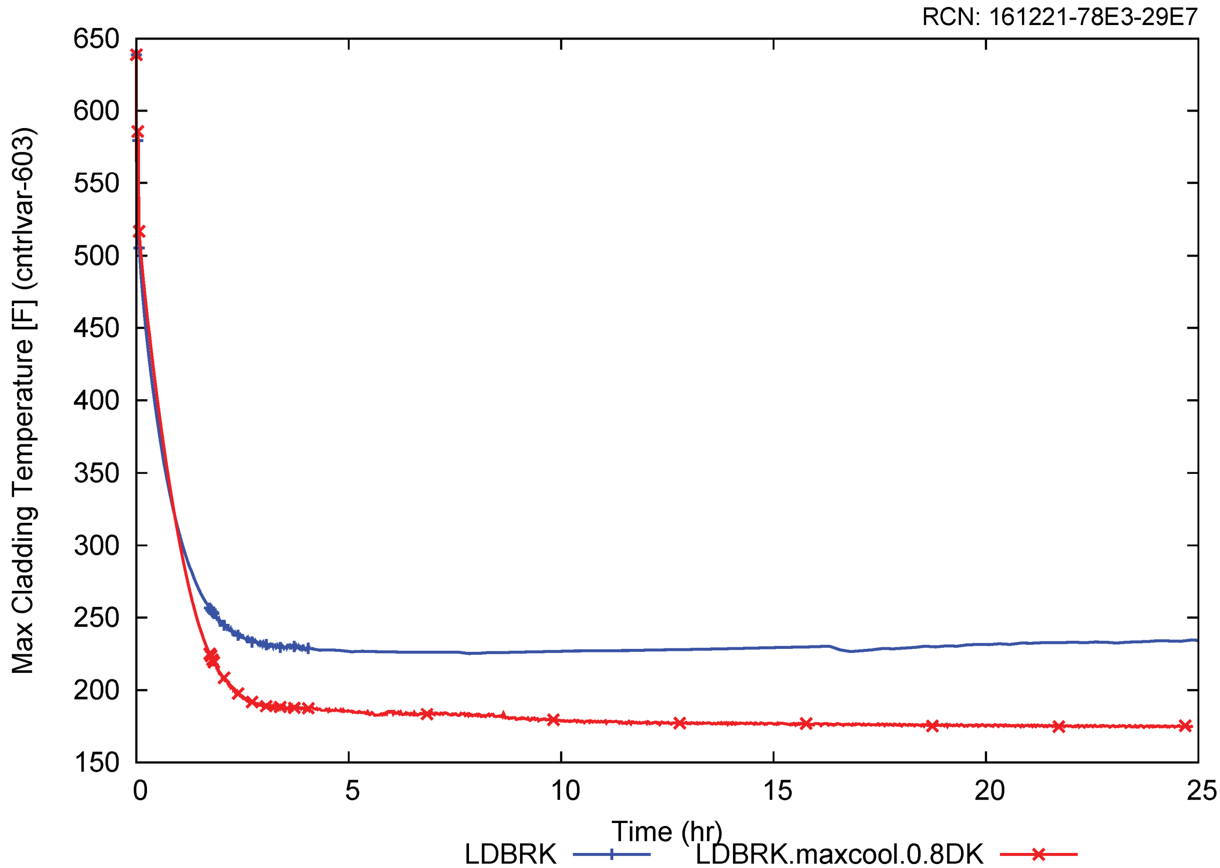

Figure 5-20 Peak cladding temperature for maximum cooldown, 0.8 decay heat

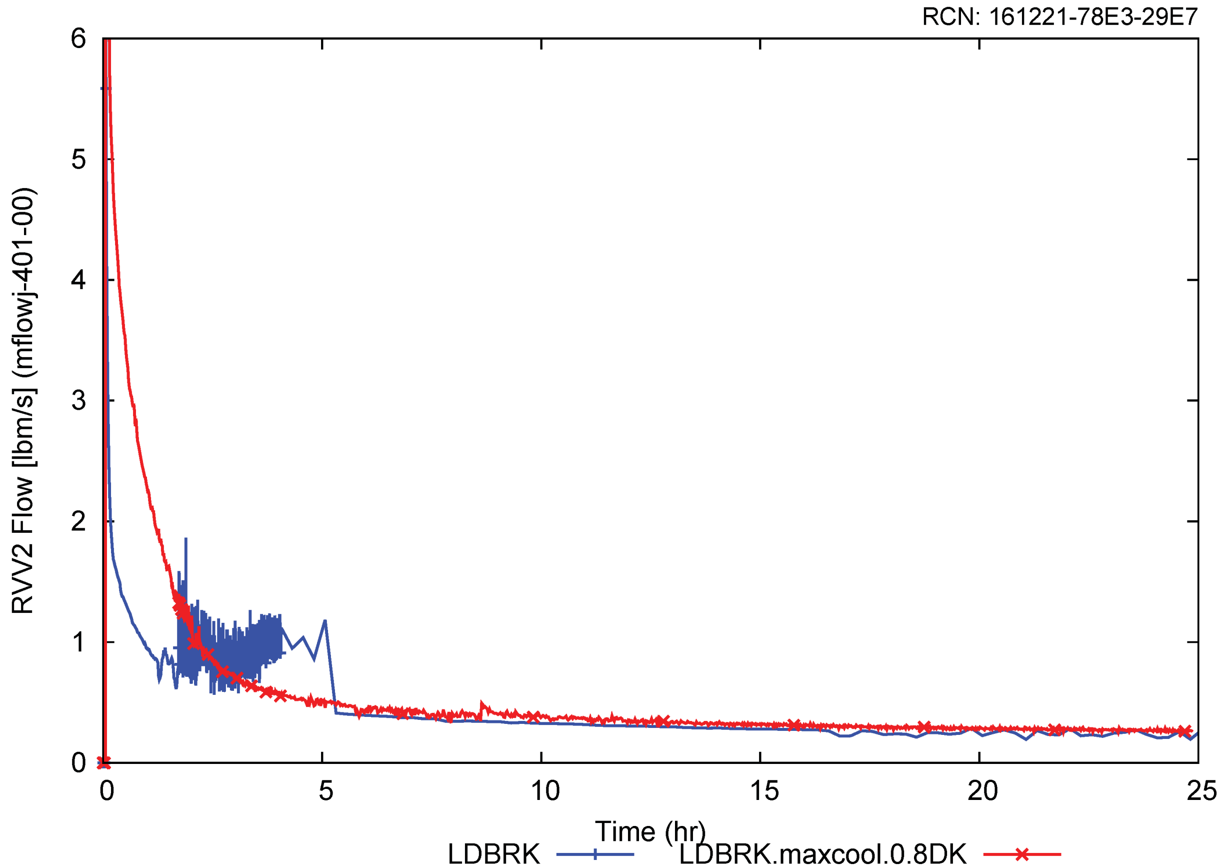

Figure 5-21 Flow through single reactor vent valve for maximum cooldown, 0.8 decay heat

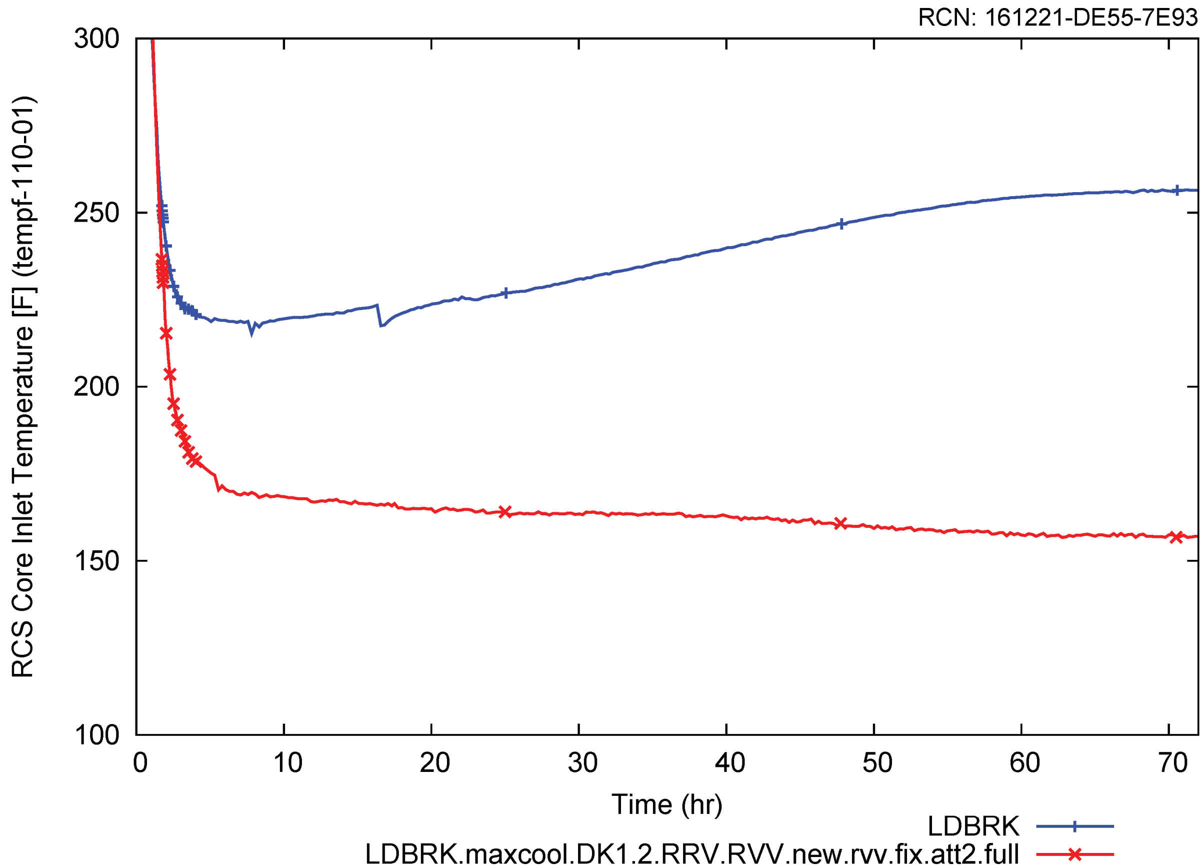

Figure 5-22 Reactor coolant system core inlet temperature for maximum cooldown, 1.2 decay heat

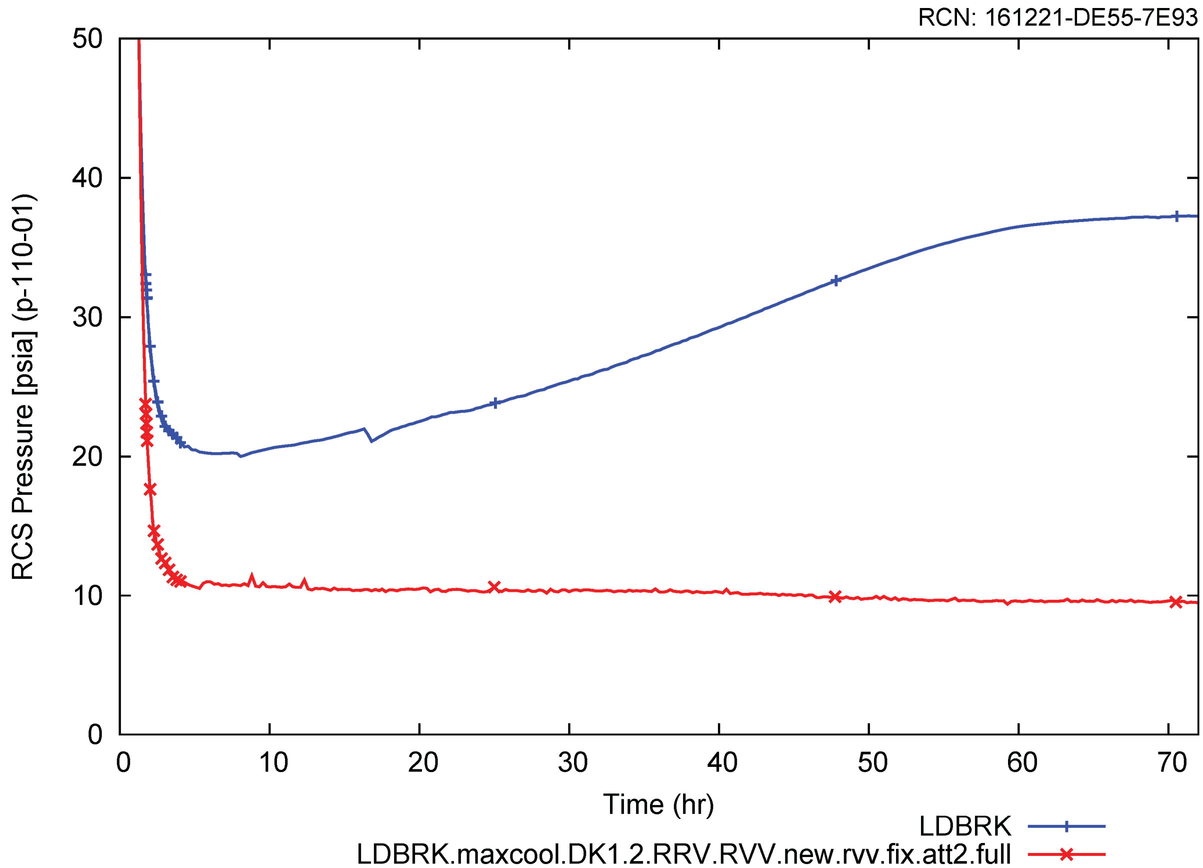

Figure 5-23 Reactor coolant system pressure for maximum cooldown, 1.2 decay heat

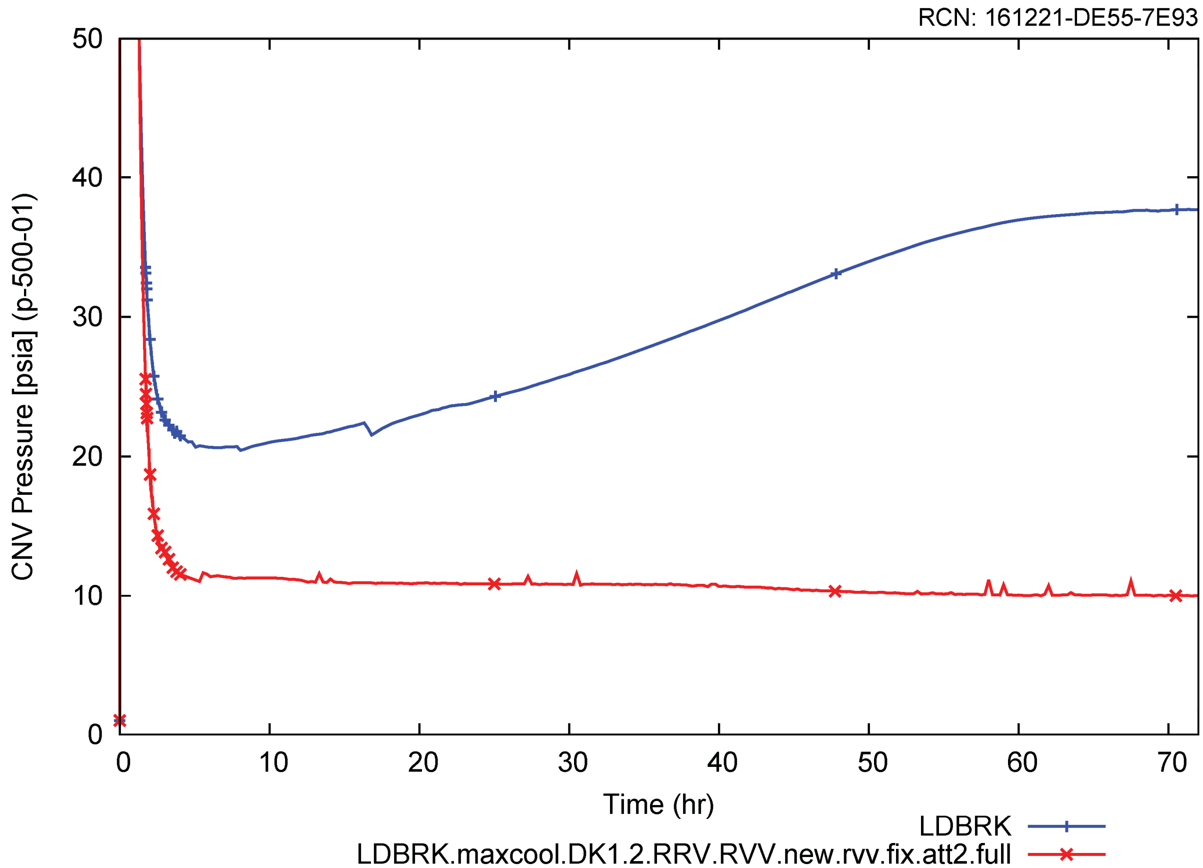

Figure 5-24 Containment vessel pressure for maximum cooldown, 1.2 decay heat

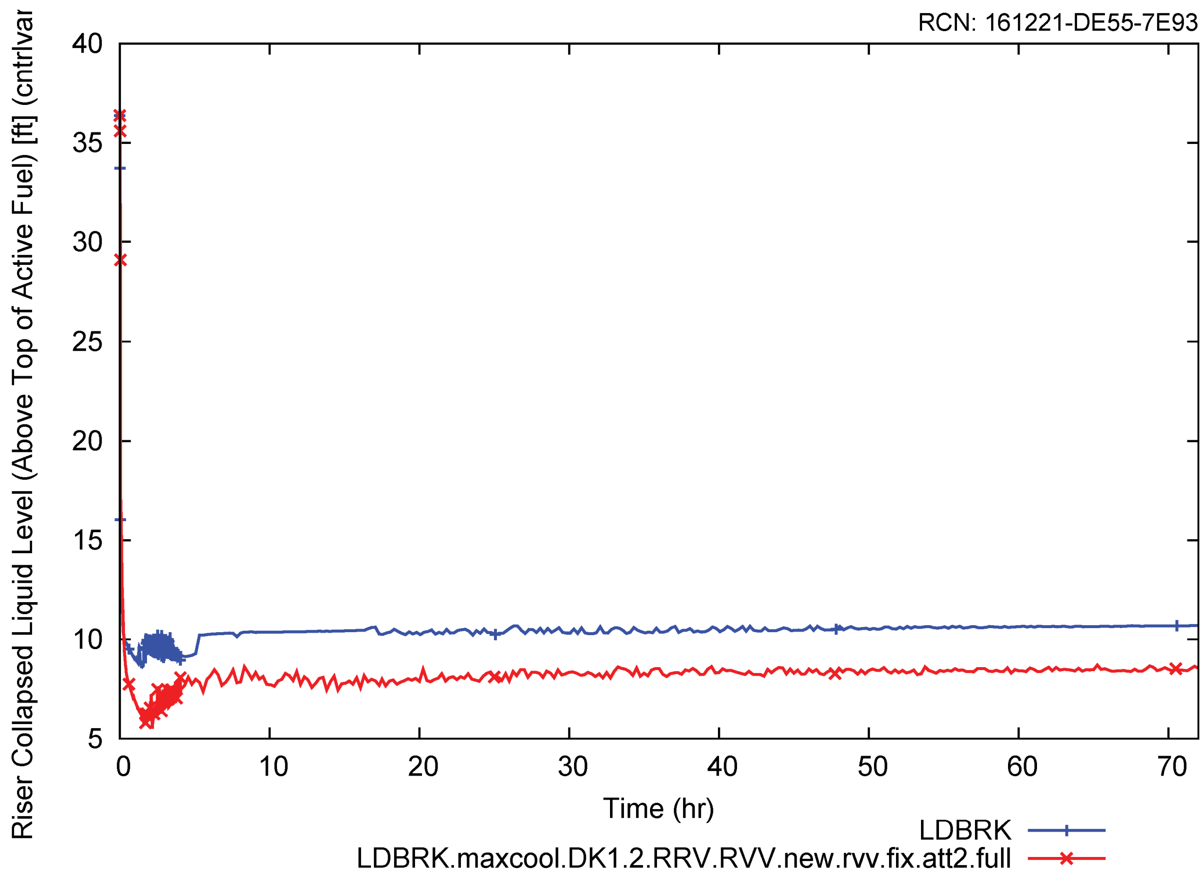

Figure 5-25 Reactor coolant system collapsed liquid level above top of active fuel for maximum cooldown, 1.2 decay heat

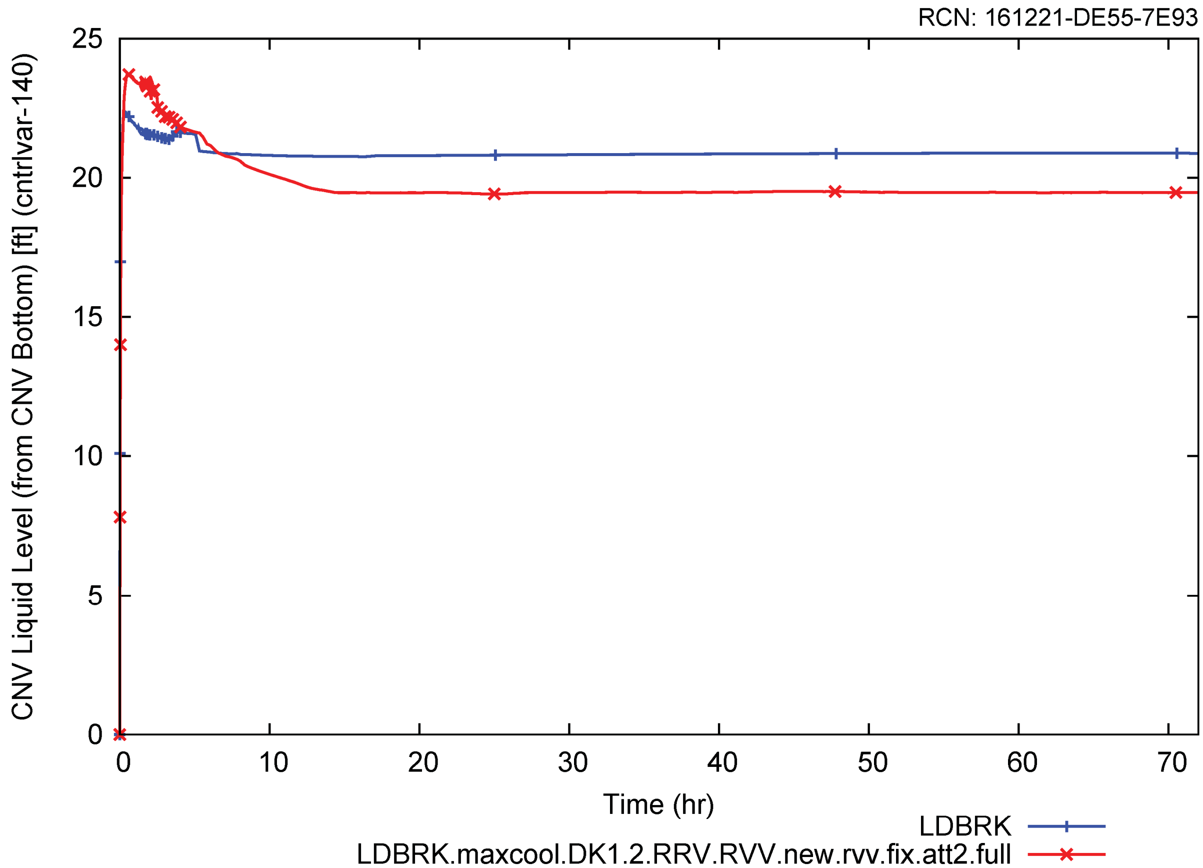

Figure 5-26 Containment vessel level for maximum cooldown, 1.2 decay heat

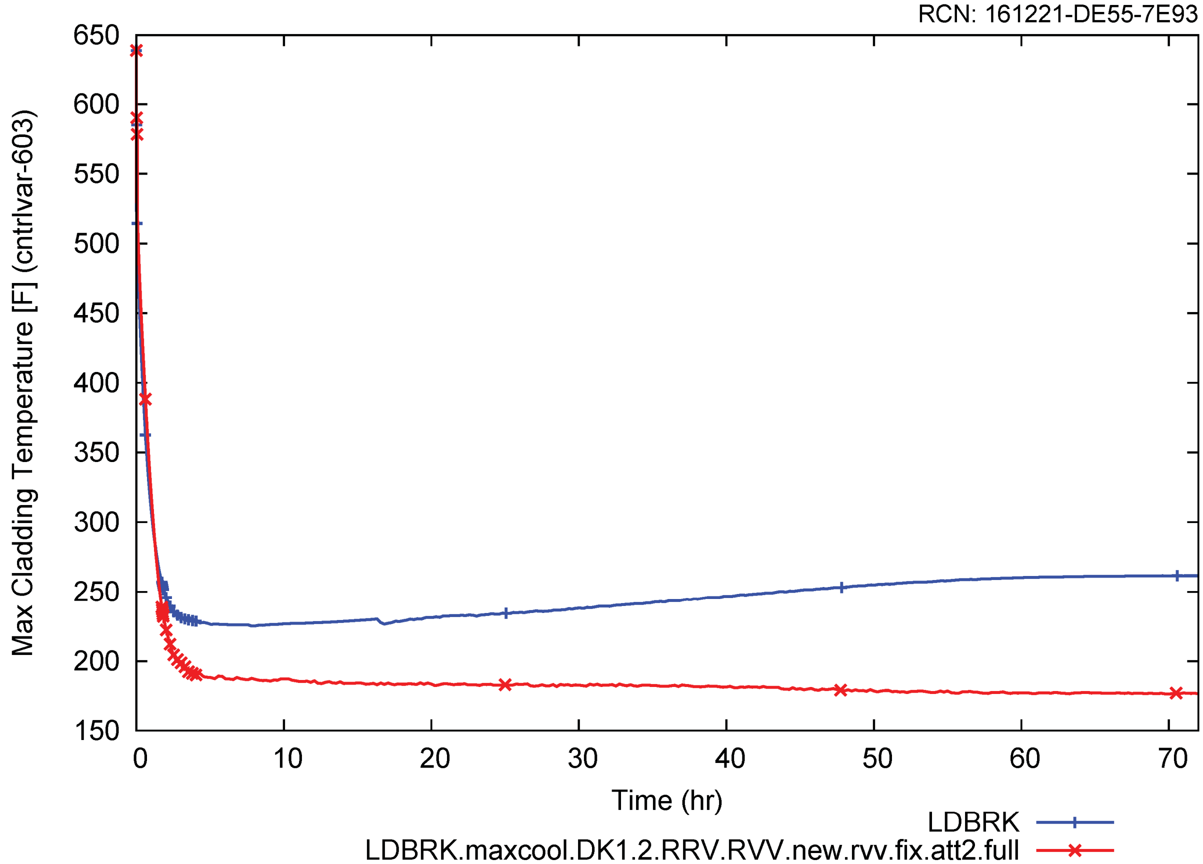

Figure 5-27 Peak cladding temperature for maximum cooldown, 1.2 decay heat

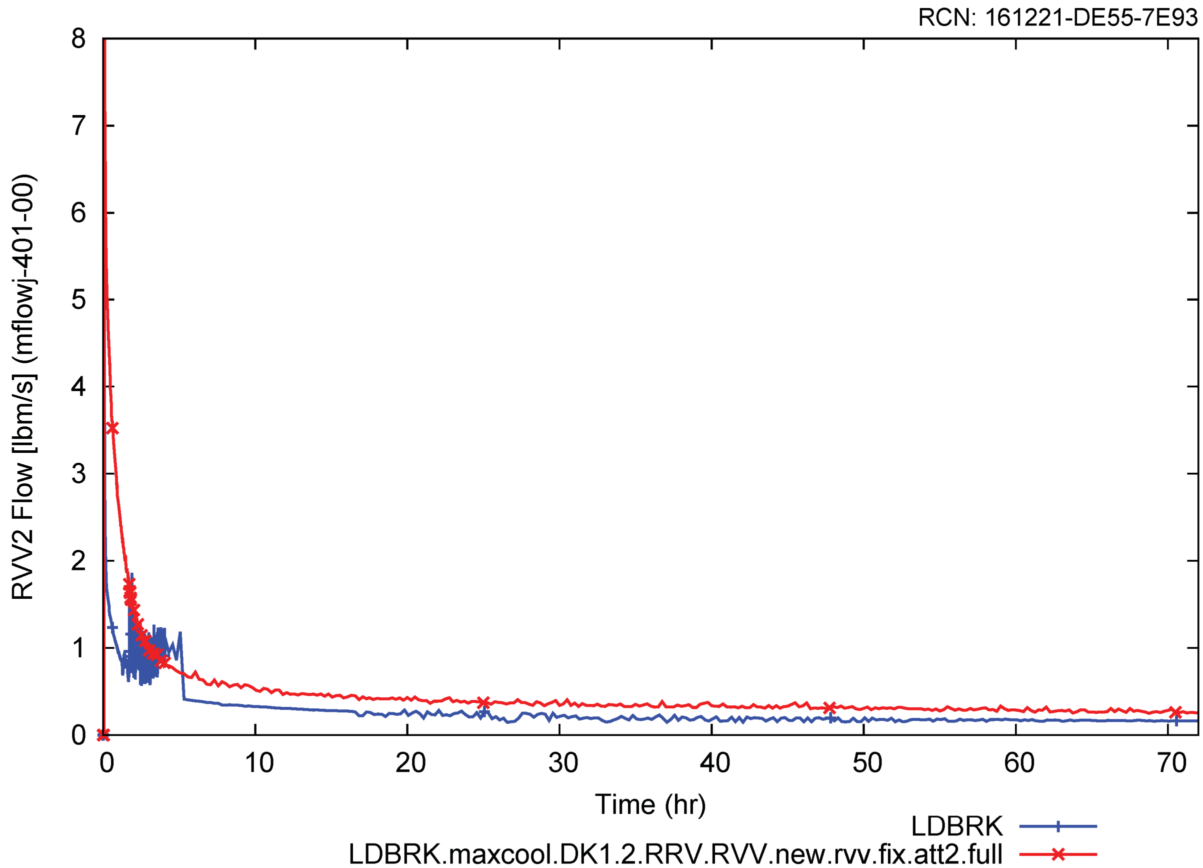

Figure 5-28 Flow through single reactor vent valve for maximum cooldown, 1.2 decay heat

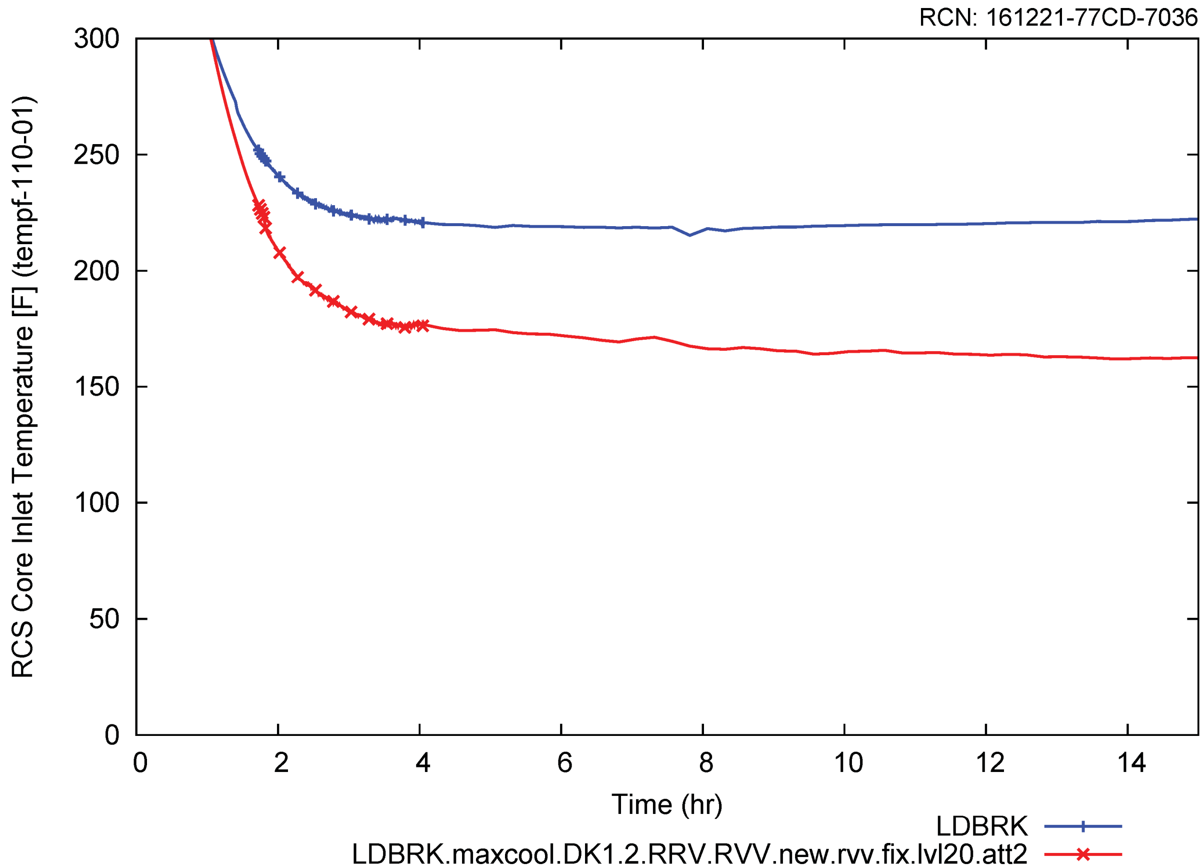

Figure 5-29 Reactor coolant system core inlet temperature for maximum cooldown, 1.2 decay heat, 20% pressurizer level

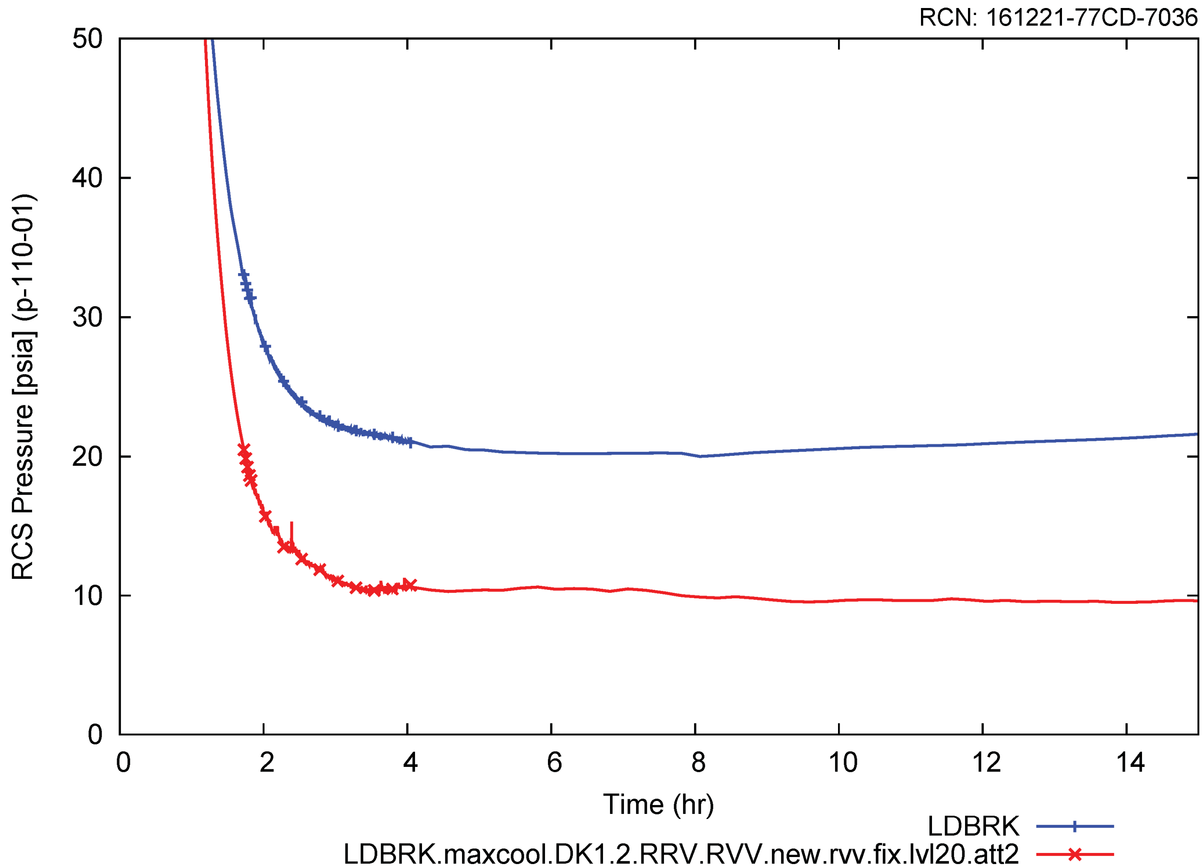

Figure 5-30 Reactor coolant system pressure for maximum cooldown, 1.2 decay heat, 20% pressurizer level

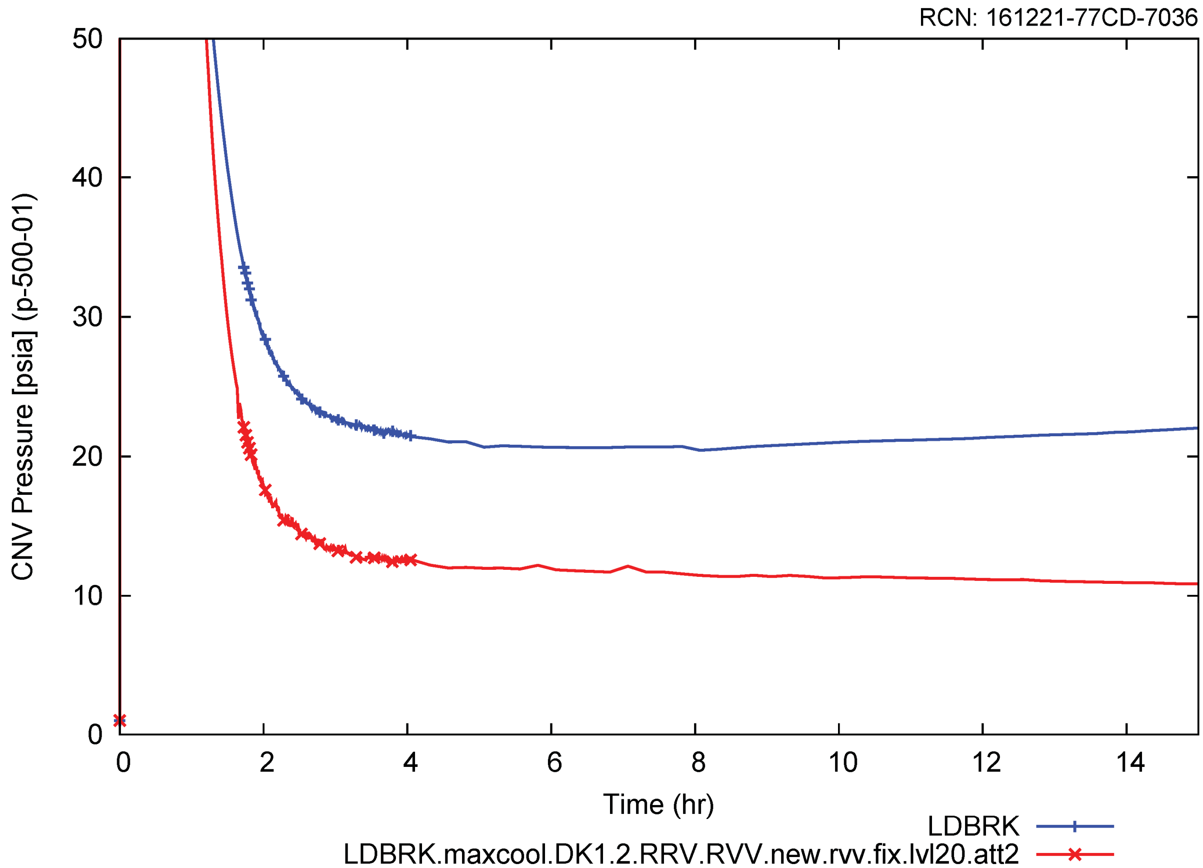

Figure 5-31 Containment vessel pressure for maximum cooldown, 1.2 decay heat, 20% pressurizer level

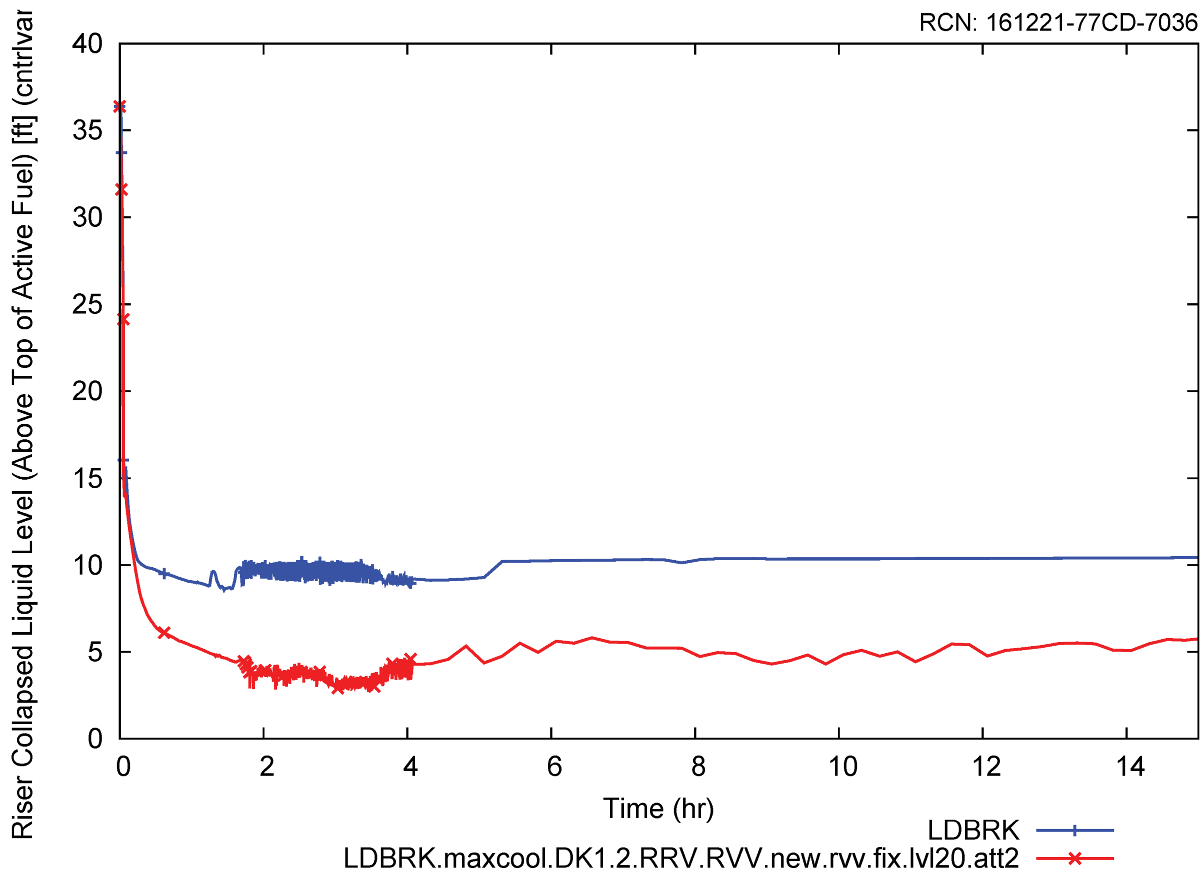

Figure 5-32 Reactor coolant system collapsed liquid level above top of active fuel for maximum cooldown, 1.2 decay heat, 20% pressurizer level

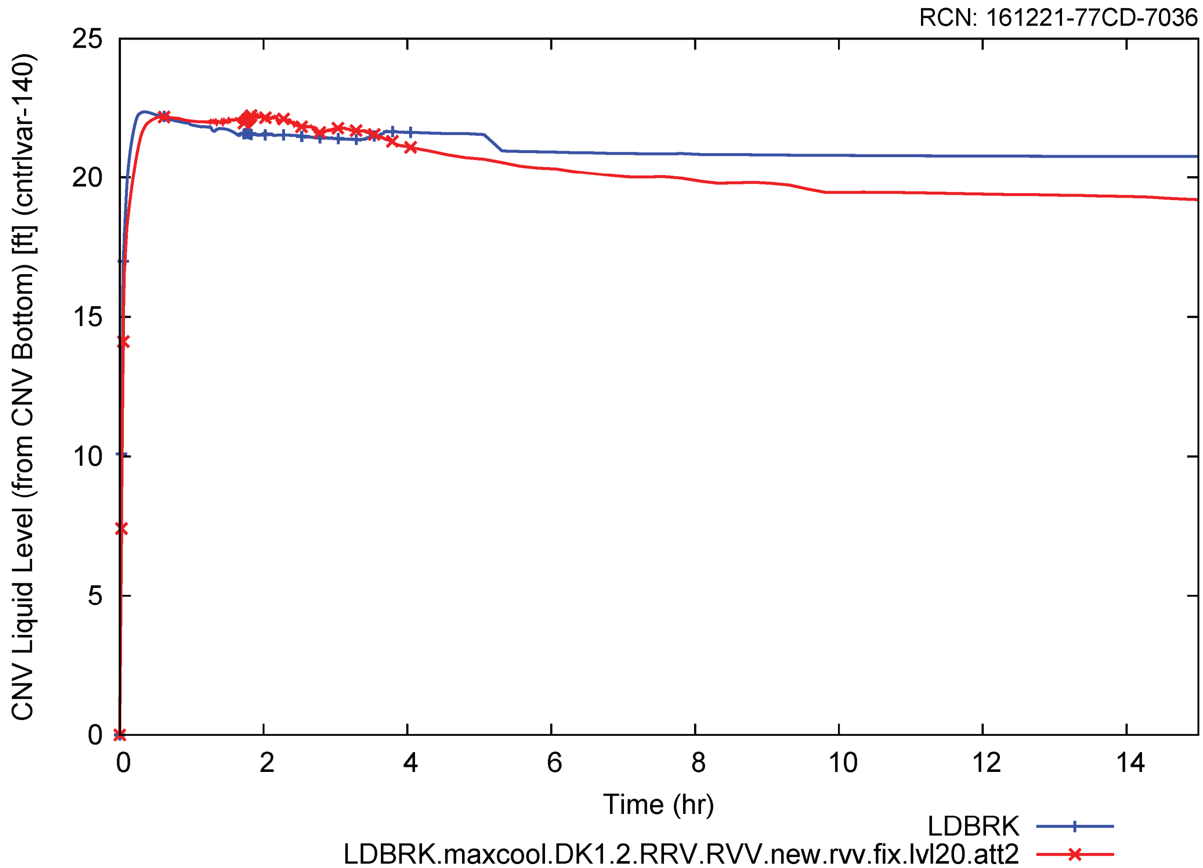

Figure 5-33 Containment vessel level for maximum cooldown, 1.2 decay heat, 20% pressurizer level

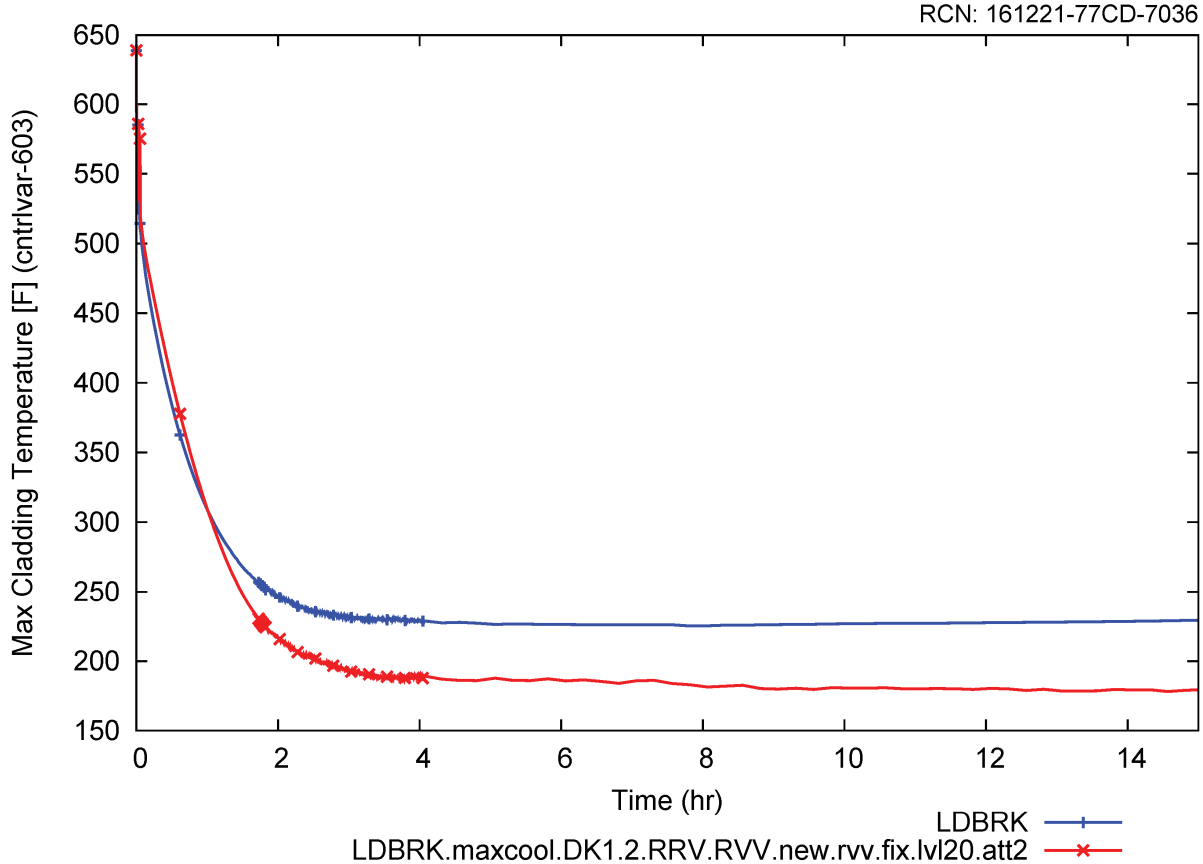

Figure 5-34 Peak cladding temperature for maximum cooldown, 1.2 decay heat, 20% pressurizer level

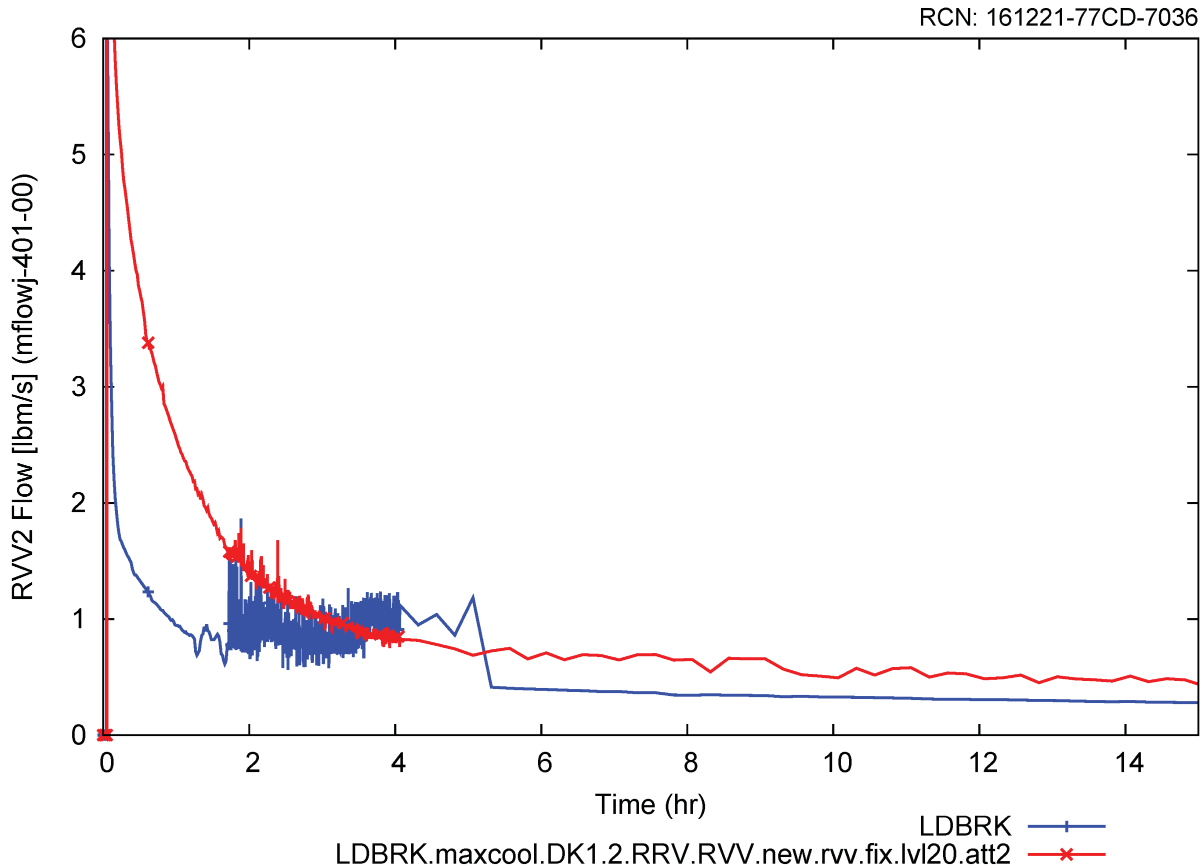

Figure 5-35 Flow through single reactor vent valve for maximum cooldown, 1.2 decay heat, 20% pressurizer

5.3.3 Decay Heat Removal System to Emergency Core Cooling System Transition with Maximum Cooldown Conditions for Steam Generator Tube Failure

In addition to the letdown break scenario, non-LOCA initiating events with DHRS active to remove decay heat until ECCS actuation (i.e. at the IAB release pressure or at 24 hours) were considered. Figure 5-36 through Figure 5-42 present results from the SGTF initiating event with loss of normal AC power and DHRS active to remove decay heat until ECCS valve opening at 24 hours. This scenario was run with max cooldown rate conditions and 20 percent initial pressurizer level, since the previous sensitivities demonstrate this condition to be most limiting with respect to adequacy of ECCS in maintaining core cooling.

As illustrated by the results presented in this section, and sensitivities where ECCS valves opened at the IAB release pressure, the effects of SGTF and DHRS with the maximum cooldown case does not significantly affect the previous maximum cooldown conclusions. Adequate core cooling is maintained even with the additional inventory loss of the SGTF. The

minimum collapsed liquid level for the SGTF maximum cooldown cases were non-limiting compared to the results presented in Section 5.3.2.

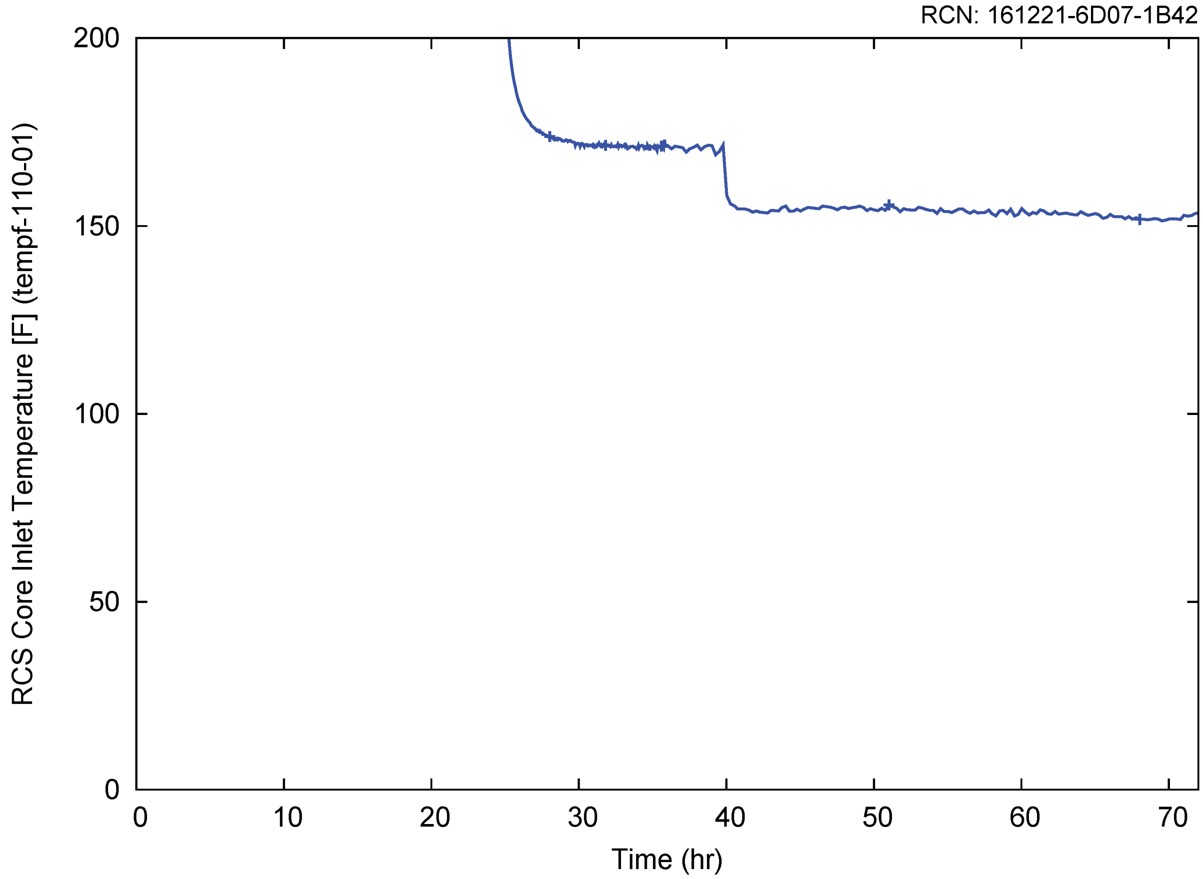

Figure 5-36 Reactor coolant system core inlet temperature for maximum cooldown with steam generator tube failure

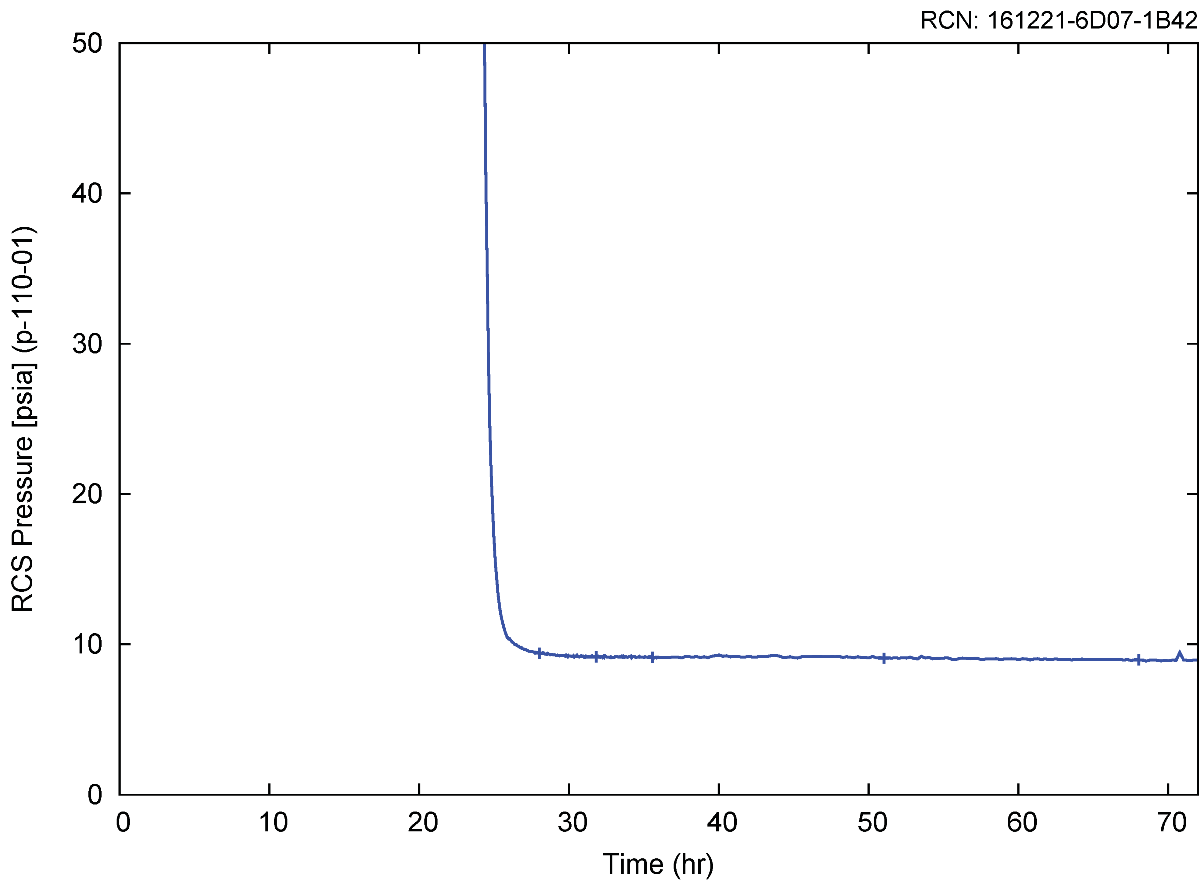

Figure 5-37 Reactor coolant system pressure for maximum cooldown with steam generator tube failure

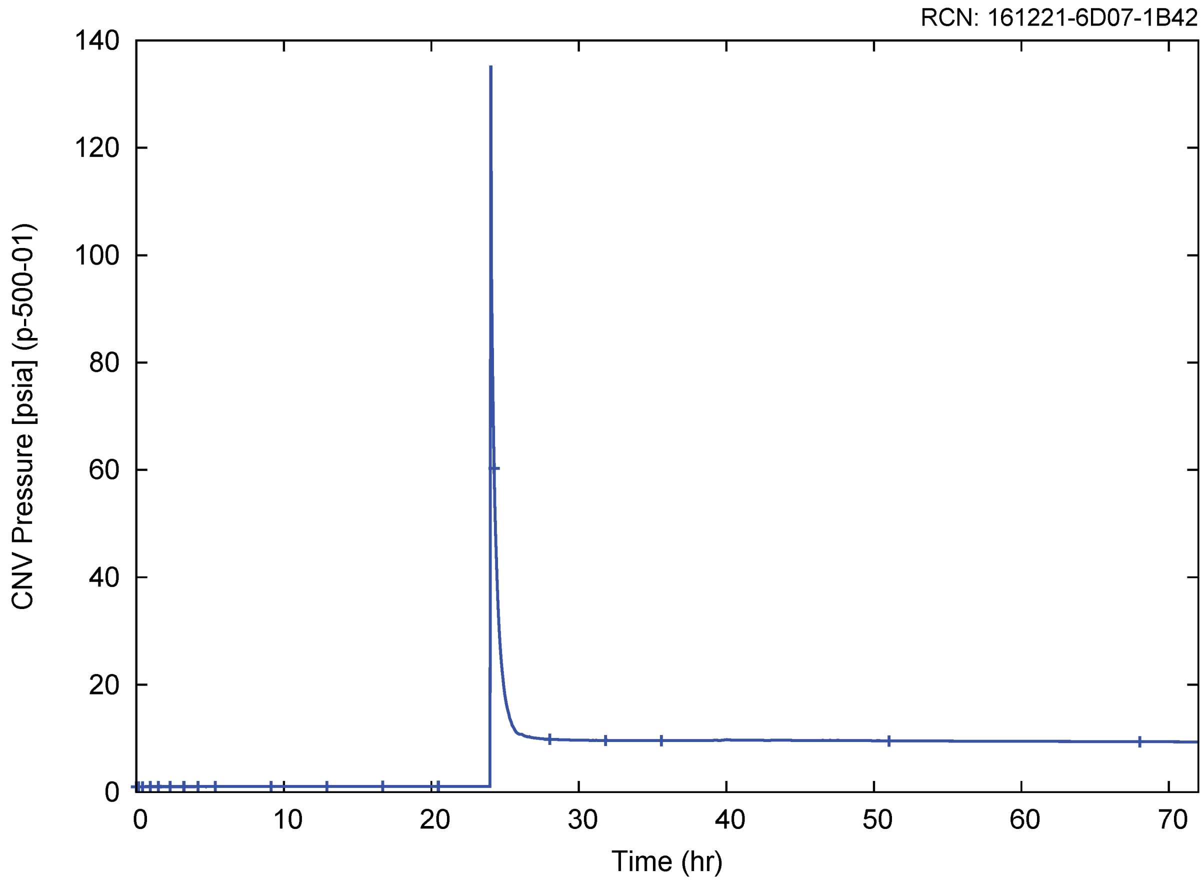

Figure 5-38 Containment vessel pressure for maximum cooldown with steam generator tube failure

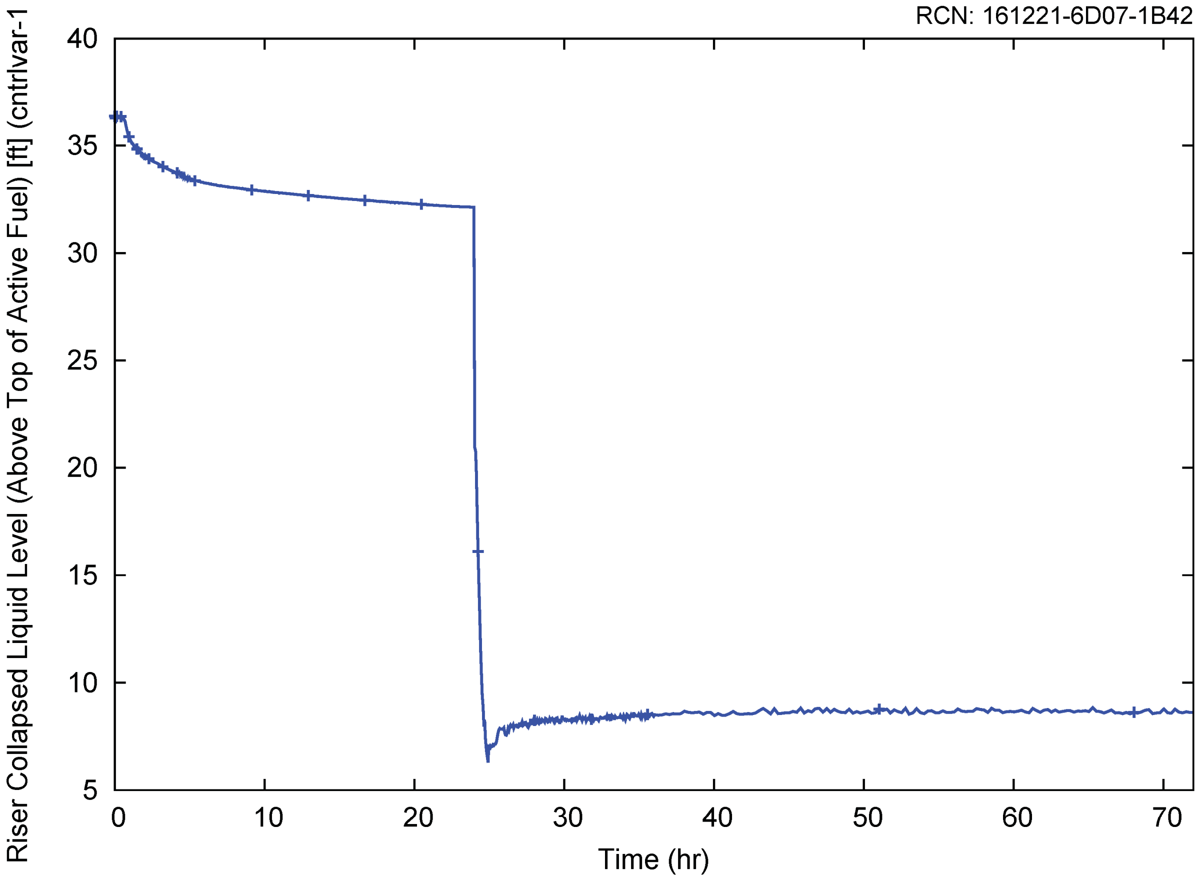

Figure 5-39 Reactor coolant system collapsed liquid level above top of active fuel for maximum cooldown with steam generator tube failure

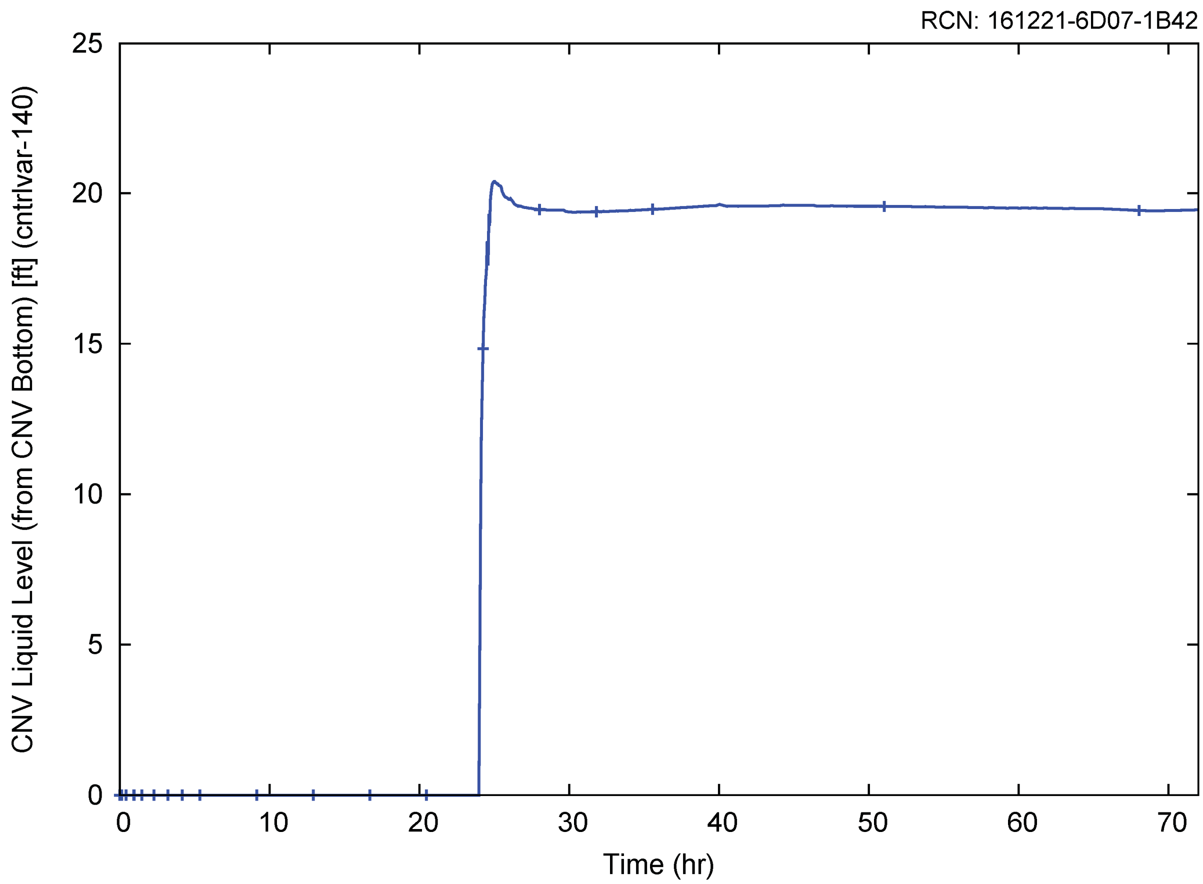

Figure 5-40 Containment vessel level for maximum cooldown with steam generator tube failure

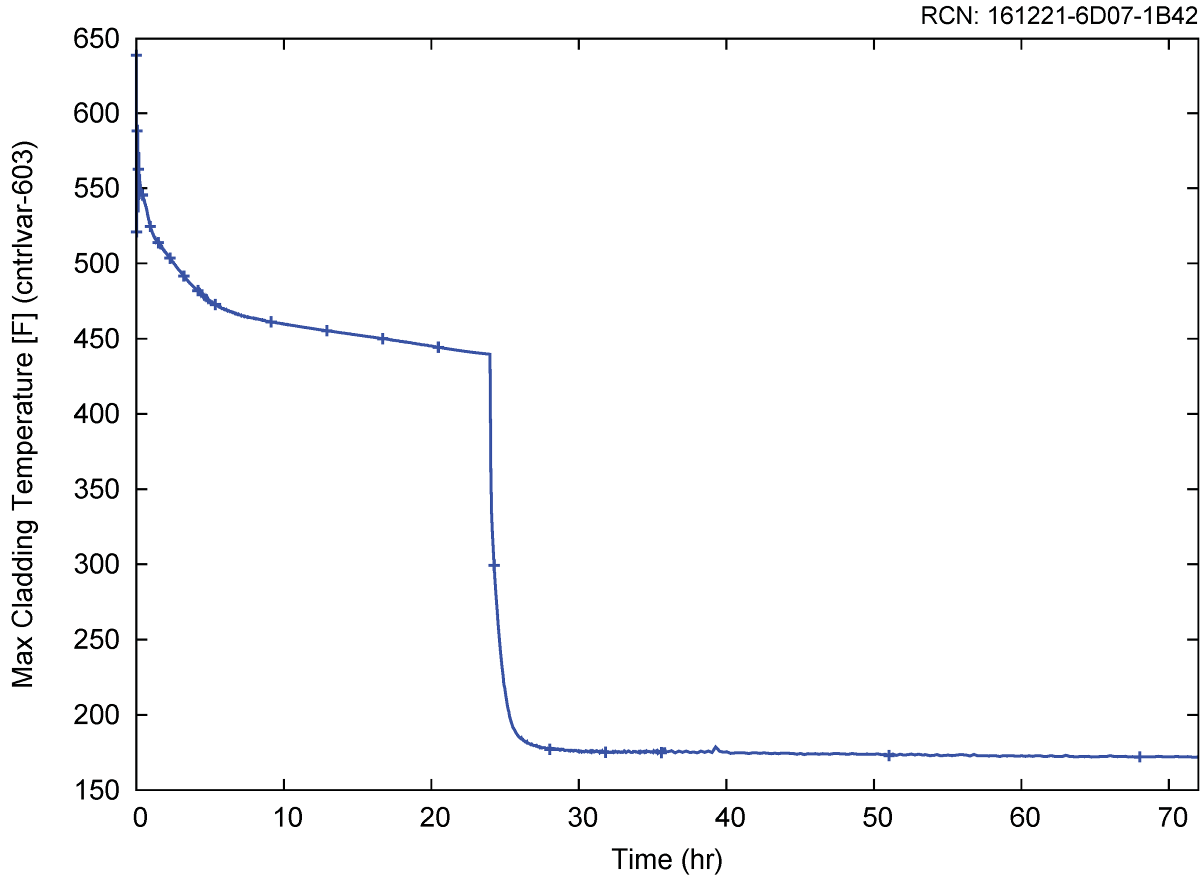

Figure 5-41 Peak cladding temperature for maximum cooldown with steam generator tube failure

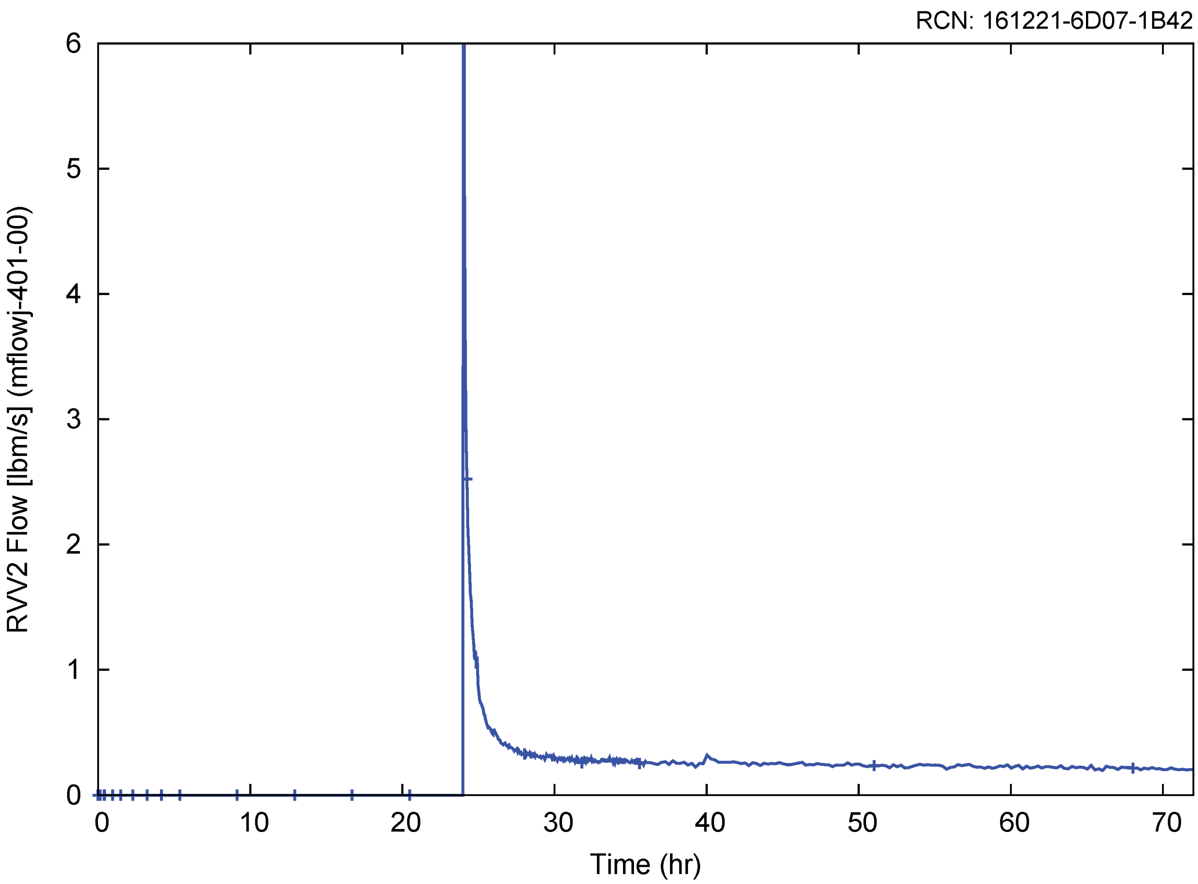

Figure 5-42 Flow through single reactor vent valve for maximum cooldown with steam generator tube failure

5.3.4 Summary and Conclusions

Decay heat removal via ECCS, regardless of the short term initiating conditions and events, is acceptable compared to the regulatory requirements outlined in Section 2.2 and the acceptance criteria outlined in Section 2.3. The reduction of the long-term peak cladding temperature for all limiting conditions (minimum cooldown, maximum cooldown, pressurizer level, DHRS availability, and SGTF) compared to the initial temperature provides justification that CHF does not occur. Stable long-term collapsed liquid levels in the riser in conjunction with maintaining positive flow through the RVV provide assurance that core flow is continuous and positive during LTC.

The limiting case for the minimum collapsed liquid level above the active fuel is produced by the maximum cooldown case with 1.2 times decay heat with the initial pressurizer level at a minimum of 20 percent. The minimum collapsed liquid level was found to be 2.271 feet above the TAF. The limiting case for maximum fuel cladding temperature is produced by the minimum cooldown case with the initial reactor pool level at 45 feet.

The boron precipitation methodology presented in Section 6.0 uses the RCS inlet core temperatures and the minimum RCS liquid level above the TAF to evaluate acceptance criteria for determining that a coolable geometry is maintained.

6.0 Boron Precipitation Methodology and Analysis Results

The NPM uses boron for core reactivity control during normal operation. During the long-term cooling phase of ECCS operation, boiling in the core region is expected to concentrate boron in the liquid in the core and riser region. After ECCS valves open and recirculation is established, liquid from containment enters the RPV through the RRVs, circulates into the core region, and vapor is vented into containment through the RVVs where it condenses on the containment wall. Over time, the vapor venting from the RPV into containment will result in increased boron concentration in the RPV and decreased boron concentration in the fluid in containment. In the NPM design, the collapsed liquid level remains above the top of the core during the long-term cooling phase. Therefore, the concentration of the boron in the reactor vessel core and riser region is analyzed to demonstrate that boron precipitation does not occur and coolable geometry is maintained.

6.1 General Approach and Acceptance Criteria

A simplified, conservative mixing volume approach is used to demonstrate that following an event that transitions to long-term ECCS cooling, the boron concentration of the liquid in the core and riser region remains below the solubility limit and therefore boron precipitation does not occur and coolable geometry is maintained. The mixing volume credited in the boron precipitation analysis is the liquid volume in the core and riser region, based on the collapsed liquid level above the bottom of the core calculated by NRELAP5 in the long-term cooling calculations. The core inlet temperature predicted by NRELAP5 is compared to the solubility temperature for the calculated core and riser region collapsed liquid level. The maximum allowable boron concentration during operation is conservatively assumed. A simplified, conservative analysis is performed where it is assumed that the mass of boron initially in the RCS is located in the liquid in the core and riser region; liquid in containment, the downcomer, and the lower plenum are assumed to be entirely diluted. In reality, after the initial blowdown of liquid and vapor into containment, it would take time for the boron concentration in the core and riser region to increase due to vapor venting. This time-dependent transport of boron from liquid in the downcomer and containment is conservatively neglected.

The boron solubility curve that specifies the acceptance criterion of allowable concentration of boric acid as a function of temperature is shown in Figure 6-1.

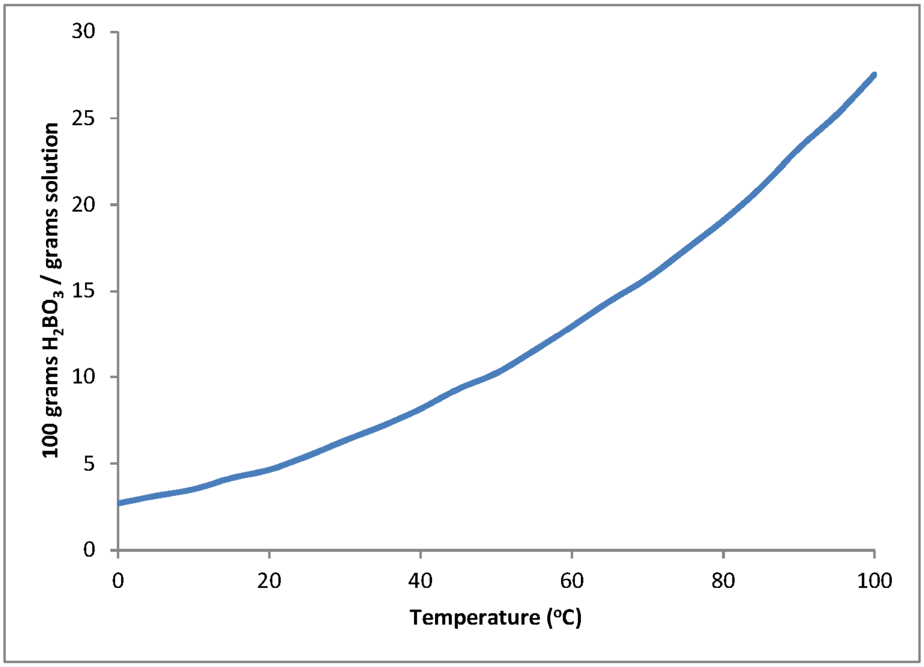

Figure 6-1 Percent boric acid at solubility limit as a function of temperature

6.2 Methodology

The determination of the precipitation temperature for a given mixing volume starts with the calculation of the entire mass of boron in the reactor coolant system (RCS). Then, a corresponding concentration is calculated for the mixing volume. Next, the precipitation temperature is obtained for the mixing volume concentration using the boron precipitation curve.

First, these calculations are performed for various mixing volumes corresponding to various elevations of liquid levels above the core. Finally, the level above the TAF, and the core inlet temperature, from the NRELAP5 long-term cooling calculation are used with the calculated solubility temperature to demonstrate that boron precipitation does not occur.

6.2.1 Calculate Total Boron Mass

The maximum allowable boron concentration in the RCS during operation is conservatively assumed. The initial mass of water in the RCS is determined assuming the maximum pressurizer level. The initial RCS density is taken as the density at the vessel average temperature. This implicitly assumes that the RCS fluid volume at T_{cold} conditions is equal to the fluid volume at T_{hot} conditions. This approximation is acceptable because, in the NuScale design, the limiting maximum allowable boron concentration is at hot zero power (HZP) conditions. At HZP conditions, there is minimal temperature rise across the core, and therefore a uniform density is appropriate.

Given an initial boron concentration, C_{rv}, the mass of boron in the reactor vessel is

$$M_{Brv} = \frac{C_{rv} * M_{Wrv}}{10^6 - C_{rv}}$$

where,

$M_{Brv} = mass\ of\ boron\ in\ RCS, l$

$C_{rv} = initial\ concentration\ of\ boron\ in\ RCS, ppm$

$M_{Wrv} = initial\ mass\ of\ water\ in\ RCS, lb$

6.2.2 Calculate Mass of Fluid in Mixing Volume

With the collapsed liquid level in the core and riser region from the inside bottom of vessel, the mixing volume is calculated based on the RCS geometry.

The density of the collapsed liquid is 53.351 lbm/ft^3 at HZP conditions. With the density ($\rho_{l,mv}$) and volumes (V_{mv}) known, the mass of the mixing volume can be solved using the following equation.

$$M_{mv} = V_{mv} * \rho_{l,mv}$$

The mass of the mixing volume changes with the height of the collapsed liquid level.

6.2.3 Calculate Boron Concentration in Mixing Volume

The mixing volume boron and boric acid concentrations are expressed in ppm as

$$C_B = \frac{M_{Brv} * 10^6}{M_{Brv} + M_{mv}}$$

$$C_{H_3BO_3} = C_B \frac{MW_{H_3BO_3}}{MW_B}$$

where $MW_{H_3BO_3}$ and MW_B are the molecular weights of boric acid and boron, respectively. C_B and $C_{H_3BO_3}$are the concentration of boron and boric acid, respectively. The molecular weights are 61.8 g/mol for boric acid and 10.8 g/mol for boron.

The boric acid concentration in weight percent (wt%) is expressed as

$$C_{H_3BO_3}(wt\%) = \frac{C_{H_3BO_3}}{10^6} * 10^2 = \frac{C_{H_3BO_3}}{10^4}$$

6.2.4 Assess Margin to Boron Precipitation

Finally, the level above the TAF, and the core inlet temperature, from the NRELAP5 long-term cooling calculation are used with the calculated solubility temperature to demonstrate that boron precipitation does not occur.

The NRELAP5 calculations demonstrate that the minimum collapsed level in the core and riser region occurs relatively early in the transient following ECCS valve opening (within approximately 1-2 hours in the letdown break calculations). Longer term, the RCS and containment levels equilibrate and the core and riser level increases from the minimum. At the time of minimum core and riser level, the core inlet temperature remains fairly high, and decreases to a quasi-steady condition at the end of the calculation. Therefore, two points are considered in the boron precipitation analysis: the point of minimum level, and the calculation end point where minimum temperature occurs.

6.3 Results

Table 6-1 shows the temperature at which boron will precipitate for mixing volumes encompassing the core and the riser fluid corresponding to the given fluid levels above TAF. These assume the HZP conditions of 1800 ppm boron concentration, 420 degrees F RCS average temperature, and 1850 psia RCS pressure. Based on the results shown in Section 5.0, collapsed liquid levels below 1foot above TAF are not reached, and have not been presented below.

Table 6-1 Boron precipitation temperatures for various collapsed liquid levels above top of active fuel

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}}2(a),(c)

{{

}}2(a),(c)

From the cases considered in the boron precipitation analysis, the lowest level reached was 2.271 feet above TAF in a LDBRK case biased for maximum cooldown, 1.2 multiplier on the decay heat, failure of one RRV and one RVV, and minimum initial pressurizer level of 20 percent.

The core conditions for this case showed a core inlet and core outlet temperature of approximately 185 degrees F at about 2.3 hours after break initiation. Using the Table 6-1 entry at 2.25 feet shows the critical boron solubility temperature is 178 degrees F. From this, boron precipitation will not occur during the level-limiting LTC event.

Margin to the solubility temperature at the end of the cooling calculation was demonstrated for all cases considered in the boron precipitation analysis. For example, the core inlet temperature at 72 hours is 157 degrees F for a LDBRK case biased for maximum cooldown, 1.2 multiplier on decay heat, failure of one RRV and one RVV. At this temperature, the given case showed a collapsed liquid level of 8.50 feet above the TAF. Using Table 6-1, a temperature of 107 degrees F would initiate boron precipitation if the

collapsed liquid level were 8.5 feet above TAF. Therefore, adequate margin to the solubility temperature is demonstrated.

Cases with lower decay heat were also considered in the boron precipitation analysis and were not limiting due to minimal impact on core inlet temperatures and higher long-term level.

6.4 Conclusions

Based on the results shown above, boron precipitation will not occur during any postulated condition in a long-term cooling scenario. Sufficient margin exists both to cover the low levels reached immediately following ECCS activation as well as the low temperatures reached during extended ECCS operation assuming conservative plant conditions.

7.0 Summary and Conclusions

This report documents the analytical methodology for long-term ECCS operation, either as an extension to a LOCA, or as a result of ECCS activation following a non-LOCA event when loss of normal AC power is assumed.

The applicable regulatory requirements from 10CFR50.46, NuScale PDC 35, and the regulatory guidance from the NuScale DSRS have been addressed by the long-term cooling methodology. This methodology utilizes the NuScale LOCA EM described in Reference 8.2.1, and was developed in accordance with to RG 1.203.

The LTC methodology is informed by comprehensive work with NRELAP5 parametric calculations, exploring an extensive set of sensitivities for effect on the FOMs as defined in the PIRT. These sensitivities used appropriate ranges of controlling parameters. The LTC methodology includes the evaluation of margin to boron precipitation.

Bounding evaluations were performed with a limiting set of assumptions and initial conditions based on sensitivity results. The cases identified as most limiting, the minimum cooldown, maximum cooldown, and SGTF with DHRS, demonstrated that the collapsed liquid level remains above the TAF with acceptably low RCS and cladding temperatures, showing that the ECCS capability to provide core cooling for an extended period is adequate. In addition, boron precipitation was evaluated and it was demonstrated not to occur for the range of conditions evaluated for long-term cooling, thereby demonstrating that the core remains in a coolable geometry.

8.0 References

8.1 Source Documents

8.1.1 American Society of Mechanical Engineers, *Quality Assurance Program Requirements for Nuclear Facility Applications*, ASME NQA-1-2008, ASME NQA-1a-2009 Addenda, as endorsed by Regulatory Guide 1.28, Revision 4.

8.1.2 *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (10 CFR 50 Appendix B).

8.1.3 NuScale Topical Report, "NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant," NP-TR-1010-859-NP-A, Revision 3.

8.2 Referenced Documents

8.2.1 NuScale Topical Report, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 0.

8.2.2 *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities." Part 50, Title 10, Section 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light-Water Nuclear Power Reactors," (10 CFR 50.46).

8.2.3 U.S. Nuclear Regulatory Commission, "Emergency Core Cooling System," Design-Specific Review Standard for NuScale SMR Design, Section 6.3, Revision 0, June 2016.

8.2.4 U.S. Nuclear Regulatory Commission, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," Design-Specific Review Standard for NuScale SMR Design, Section 15.6.5, Revision 0, June 2016.

8.2.5 NuScale Topical Report, "Non-LOCA Methodologies," TR-0516-49416, Revision 0.

8.2.6 U.S. Nuclear Regulatory Commission, "Transient and Accident Analysis Methods," Regulatory Guide 1.203, Revision 0, December 2005.

8.2.7 ISA, ANSI/ISA 75.01.01-2007, "Flow Equations for Sizing Control Valves."

8.2.8 SwUM-0304-17023, Revision 4, NRELAP5 Version 1.3 Theory Manual.

Enclosure 5:

Affidavit of Thomas A. Bergman, AF-0117-52660

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 Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

(1) I am the Vice President 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 technical report reveals distinguishing aspects about the method by which NuScale develops its containment response analysis.

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 technical report entitled "Containment Response Analysis Methodology Technical Report." 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 9, 2017.

Thomas A. Bergman

Enclosure 6:

Affidavit of Thomas A. Bergman, AF-1216-52497

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 Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

(1) I am the Vice President 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 report reveals distinguishing aspects about the method by which NuScale evaluates its long term cooling capability.

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 "Long-Term Cooling Methodology". 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).

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(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 9, 2017.

Thomas A. Bergman