November 26, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of "Containment Response Analysis Methodology, Technical Report," TR-0516-49084, Revision 2

REFERENCES:
1. Letter from NuScale Power, LLC to the Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Technical Reports Supporting the NuScale Design Certification Application," dated January 9, 2017 (ML17009A490)
2. Letter from NuScale Power, LLC to the Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of 'Containment Response Analysis Methodology,' TR-0516-49084, Revision 1," dated June 13, 2019 (ML19164A145)

NuScale Power, LLC (NuScale) hereby submits Revision 2 of the "Containment Response Analysis Methodology," TR-0516-49084.

Enclosure 1 contains the proprietary version of the report titled "Containment Response Analysis Methodology." NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version of the report entitled "Containment Response Analysis Methodology."

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

If you have any questions, please contact Rebecca Norris at 541-602-1260 or at rnorris@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: "Containment Response Analysis Methodology Technical Report," TR-0516-49084-P, Revision 2, proprietary version
Enclosure 2: "Containment Response Analysis Methodology Technical Report," TR-0516-49084-NP, Revision 2, nonproprietary version
Enclosure 3: Affidavit of Zackary W. Rad, AF-1119-68069

Enclosure 1:

"Containment Response Analysis Methodology Technical Report," TR-0516-49084-P, Revision 2, proprietary version

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Enclosure 2:

"Containment Response Analysis Methodology Technical Report," TR-0516-49084-NP, Revision 2, nonproprietary version

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Containment Response Analysis Methodology Technical Report

November 2019

Revision 2

Docket: 52-048

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Department of Energy Acknowledgement and Disclaimer

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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 to 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 similar to 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).

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 526 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 internal pressure is approximately 959 psia, which results from a reactor coolant system injection line break case with a loss of normal AC and direct current (DC) power. The LOCA event peak CNV pressure is below the CNV design pressure of 1050 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 994 psia, which is approximately 5 percent below the containment design pressure of 1050 psia. It results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power, considering an inadvertent actuation block (IAB) release pressure

range of 950 psi +/- 50 psi. 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 atmospheric pressure acting against the CNV exterior surface 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. 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 (1050 psia) and the design temperature (550 degrees F) of the CNV.

The qualification of the LOCA, valve opening event and non-LOCA methodologies presented in Reference 7.2.1 and Reference 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, valve opening event 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.

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, valve opening event and non-LOCA methodologies developed following the guidance of Regulatory Guide 1.203. This report references the LOCA, valve opening event 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 energy, mass, and momentum conservation equations.	The containment response analysis methodology uses the NRELAP5 system thermal-hydraulic analysis code. NRELAP5 solves the time-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	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.

will not exceed these profiles anywhere within the specified environmental zones, except in the break zone).	

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	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 overall limiting peak CNV accident pressure is approximately 994 psia, which is approximately 5 percent below the containment design pressure of 1050

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 peak calculated containment pressure following a LOCA, or a steam or feedwater line break, should be less than the containment design pressure.	psia. It results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power considering an IAB release pressure range of 950 +/- 50 psia. 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 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.”	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 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 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 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. 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 fuel thermal conductivity values are conservatively decreased to 85 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	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.

accident indicate the conservatism that should exist.	
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. Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison 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	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. The initial RCS volume conservatively includes an allowance for RCS thermal expansion. 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. The containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code. The LOCA evaluation model report demonstrates these

natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam. 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. 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.	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. 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 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.	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.
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.	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.
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 concept of carryout rate fraction that is applicable to large PWRs is not applicable to the NuScale design. As 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 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.	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.
Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.	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 results of post-recirculation analytical models should be compared to 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 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. 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.	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)
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 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	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 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 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, valve opening event and non-LOCA methodologies, which are presented in Reference 7.2.1 and Reference 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 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 the LOCA Evaluation Model, the Evaluation Model of the valve opening events (Reference 7.2.1, Appendix B), 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

New Model	Application in Containment Response Analysis Methodology
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).
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. 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 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. Reference 7.2.1, Appendix B 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 or low steam line 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, subsequent to feedwater isolation, 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 MSLB 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 low steam line 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 DC electrical power
- limiting single failure

The differences in the NRELAP5 MSLB modeling for the containment response analysis methodology that focus on a conservative analysis of the CNV peak pressure and temperature response are detailed in Section 3.4.1.

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 DC electrical power
- limiting single failure

The differences in the NRELAP5 feedwater line break modeling for the containment response 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 NPM geometry inputs and conservative fuel inputs in the containment response analysis model are consistent with those used by the LOCA Evaluation Model. 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:

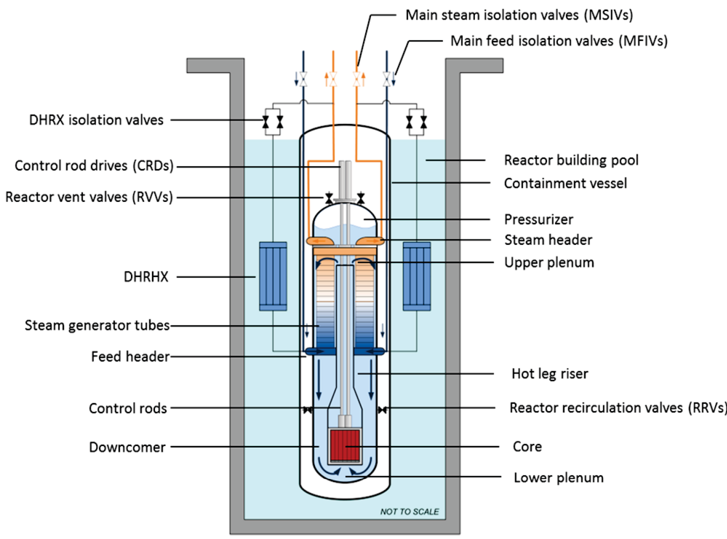

Figure 3-3   NuScale module during power operation

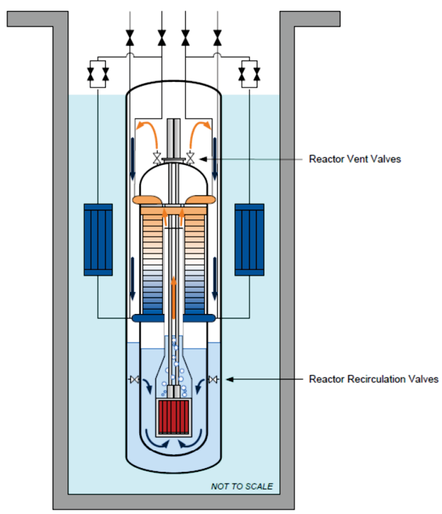

Figure 3-4   NuScale module during emergency core cooling system operation

LOCA Pipe Break and Valve Opening Modeling

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}}2(a),(c)

{{

}}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 consistent with the LOCA evaluation model.

The CNV is maintained at a partial vacuum with an assumed high initial pressure (e.g. 3.0 psia), and with the maximum total mass of noncondensables that could exist within the CNV during operation (approximately 66 lbm), in order to capture the effects of CNV non-condensable gases. Also, during a LOCA or valve opening event, the maximum total mass of noncondensables that could exist within the RPV during operation are released to the CNV model (approximately 65 lbm), in order to capture the effects of RPV non-condensable gases. 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. The initial CNV wall temperatures above the pool level are maintained at 240 degrees F, which bounds the maximum CNV wall temperature that can exist during normal operation. In the upper section heat transfer is neglected due to the presence of insulation on the CNV upper head (See Figure 3-8).

The initial CNV wall temperature is conservatively addressed based on the assumed reactor pool level. At normal operation conditions, there is very little heat loss from the RPV to CNV, such that the CNV metal below the pool surface is very close to the pool temperature. This region of the CNV heat structure is initialized within NRELAP5 based

on explicit treatment of the pool heat transfer on the CNV outer surface. The region of the CNV heat structure above the pool level is conservatively given an adiabatic outer surface boundary condition.

The CNV wall temperatures below the pool level are initialized based on the 110°F temperature modeled in the reactor pool. During the NRELAP5 steady state initialization, {{

}}2(a),(c)

The CNV wall temperatures above the pool level are initialized at a 240°F temperature, with an adiabatic boundary condition applied on the CNV wall heat structure outer surface, to conservatively not credit any heat transfer between the CNV and pool at any elevation above the pool level.

The 240°F temperature value applied at the CNV wall heat structures above the pool level is based on {{

}}2(a),(c)

assuming an initial CNV wall temperature of 240°F above the pool level for the CNV analysis is conservative.

{{

}}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:

- {{

}}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 considered in analysis. This was not included in the model since inclusion of a radiation enclosure model has a negligible impact on CNV peak pressure and temperature results.
Convection from RPV outside diameter to CNV pool	Vertical Sufaces {{ }}2(a),(c) Non-Vertical Surfaces {{ }}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	Vertical Sufaces {{ }}2(a),(c) Non-Vertical Surfaces {{ }}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.

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, subsequent to feedwater isolation, 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.

<u>Feedwater System Model</u>

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.

<u>Secondary Pipe Break Model</u>

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.

{{

}}2(a),(c)

Figure 3-9 Main steam lIne break model

{{

}}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:

- {{

}}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. As discussed in Reference 7.2.1, Appendix B.7, there are no significant differences in physics phenomena between the LOCA and valve opening events for the NuScale NPM. Therefore, the high-ranked phenomena from the LOCA PIRT also apply to the valve opening events.

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

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:

- 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. As discussed in FSAR Section 6.3.2.2, the threshold pressure for IAB operation to prevent spurious opening of the main ECCS valve is 1300 psid. Therefore, the IAB prevents the main valve from opening for all reactor pressures 1300 psid and greater, with respect to containment. Given an initial IAB block, the IAB

releases at 950 psid +/- 50 psi once reactor pressure is reduced. The IAB does not prevent the main valve from opening for initial pressures of 900 psid and below.

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 fully 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
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)

{{ }}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
- 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 changed by assuming a loss of normal AC and DC power (concurrent with the break), which results in an ECCS signal and DHRS actuation following secondary system isolation. 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 close to 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 and DC 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.

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
- 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

For each feedwater train, the FW line geometry inside CNV changes from one 5” Schedule 120 line (between the FWIV and the FW tee) to two 4” Schedule 120 lines (between the FW tee and the FW plenum). The 0.1433 ft^2 FW line break area used in the CNV analysis represents the total area of two 4” Schedule 120 lines between the FW tee and the FW plenum. The maximum break area of a single FW line inside CNV is actually 0.1136 ft^2, corresponding to one 5” Schedule 120 line between the FWIV and the FW tee. However, since these two geometries are located at the same region (i.e. the FW tee), assuming a larger FW break size (0.1433 ft^2) is acceptable since it conservatively maximizes mass release to the CNV.

The limiting FWLB event is a double-ended rupture of the largest feedwater pipe with a break area of 0.1433 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 MSIV to close on the affected SG is mitigated by closure of the backup MSIV. The limiting case also assumes a loss of normal AC and DC power at event initiation, and that results in ECCS signal actuation and a loss of power to the pressurizer heaters. With DHRS actuation, subsequent to feedwater isolation, 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.

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

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

}}2(a),(c)

{{

}}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)

Parameter	Initial Condition Assumption	Rationale
{{		}}2(a),(c)
{{		
		}}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 are analyzed as sensitivity studies, and failure of MSIVs or FWIVs to close are considered, but they have minimal effect on the CNV pressure and temperature response as the secondary system is immediately isolated for the primary side events. 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**
{{		
		}}2(a),(c)
{{		
		}}2(a),(c)

Parameter	Boundary Condition Assumption	Rationale
{{		
		}}2(a),(c)
{{		
		}}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
{{		
		}}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 is considered for the scenarios in which ECCS actuation occurs. 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
{{		
		}}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 is considered for the scenarios in which ECCS actuation occurs. Failures of a MSIV or a FWIV to close are analyzed as sensitivity studies.

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. Reference 7.2.1 Appendix B describes extension of the LOCA EM for application to valve opening events. 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.

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}}2(a),(c)

No additional qualification activities were performed for the LOCA or valve opening 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, for the LOCA EM, 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.

Reference 7.2.1, Appendix B.7 provides additional NRELAP5 assessment results for an updated spurious RVV opening test. The updated RVV spurious opening test provides better understanding of the impact of different ECCS orifice sizes on RVV opening events. The CNV pressure response is predicted with reasonable to excellent agreement.

Reference 7.2.1, Appendix C provides additional NRELAP5 assessment results for a spurious RRV opening event test. The CNV pressure response is predicted with reasonable agreement.

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)

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}}2(a),(c)

The threshold pressure for IAB operation to prevent spurious opening of the main ECCS valve is 1300 psid. Therefore, the IAB prevents the main valve from opening for all reactor pressures 1300 psid and greater, with respect to containment. Given an initial IAB block, the IAB releases at 950 psid +/- 50 psi once reactor pressure is reduced. The IAB does not prevent the main valve from opening for initial pressures of 900 psid and below.

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}}2(a),(c)

5.1.2 Base Case 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 and temperature 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)

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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 CNV peak pressure is 705 psia for the base case, and 943 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse IAB release pressure, low-biased

High CNV Level setpoint, fine CNV volume nodalization). The peak CNV wall temperature is 492 degrees F for the base case, and 510 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse IAB release pressure, low-biased High CNV Level setpoint, fine CNV volume nodalization). The results of this case are adequately representative of the RCS discharge line break, although they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The effect of the IAB release pressure range of 950 psi +/- 50 psi, including effect of valves opening at different pressures within that range, has been evaluated. Case 1 is non-limiting and was confirmed to be non-limiting in comparison to the RRV opening event.

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)**
0	LOCA in RCS discharge line For peak pressure case • Loss of normal AC and DC power • FW/MS isolation • Reactor trip For peak temperature case • Same	Same
1	High CNV pressure resulting in For peak pressure case • Containment isolation For peak temperature case • Same	Same
92	ECCS actuation on IAB release pressure	Same
95	ECCS valve opening on IAB release pressure	Same
109	Peak CNV temperature reached: For peak pressure case: 510 °F For peak temperature case: Same	Same
112	Peak CNV pressure is reached: For peak pressure case: 943 psia For peak temperature case: Same	Same
~1900	CNV pressure decreases to <50% of peak pressure	Same

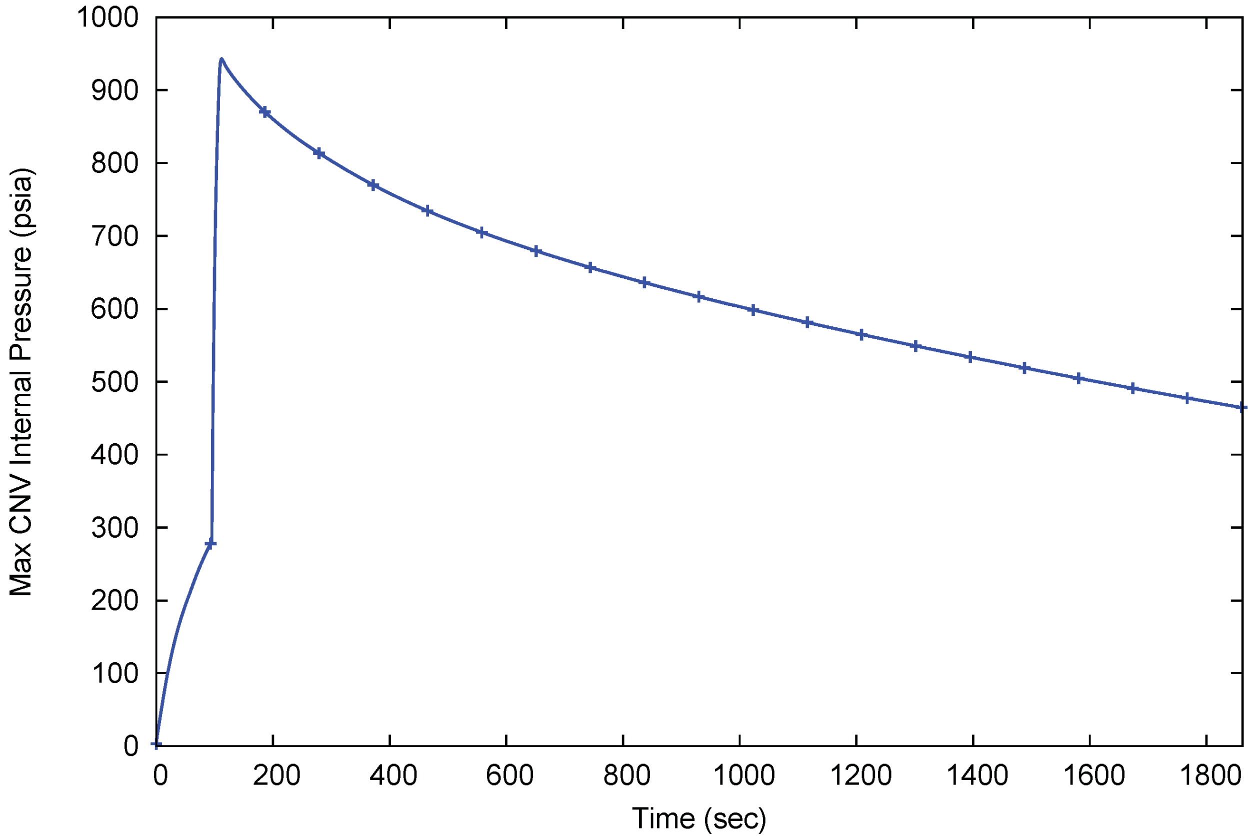

Figure 5-1 Case 1 containment vessel pressure – reactor coolant system discharge line break loss-of-coolant accident

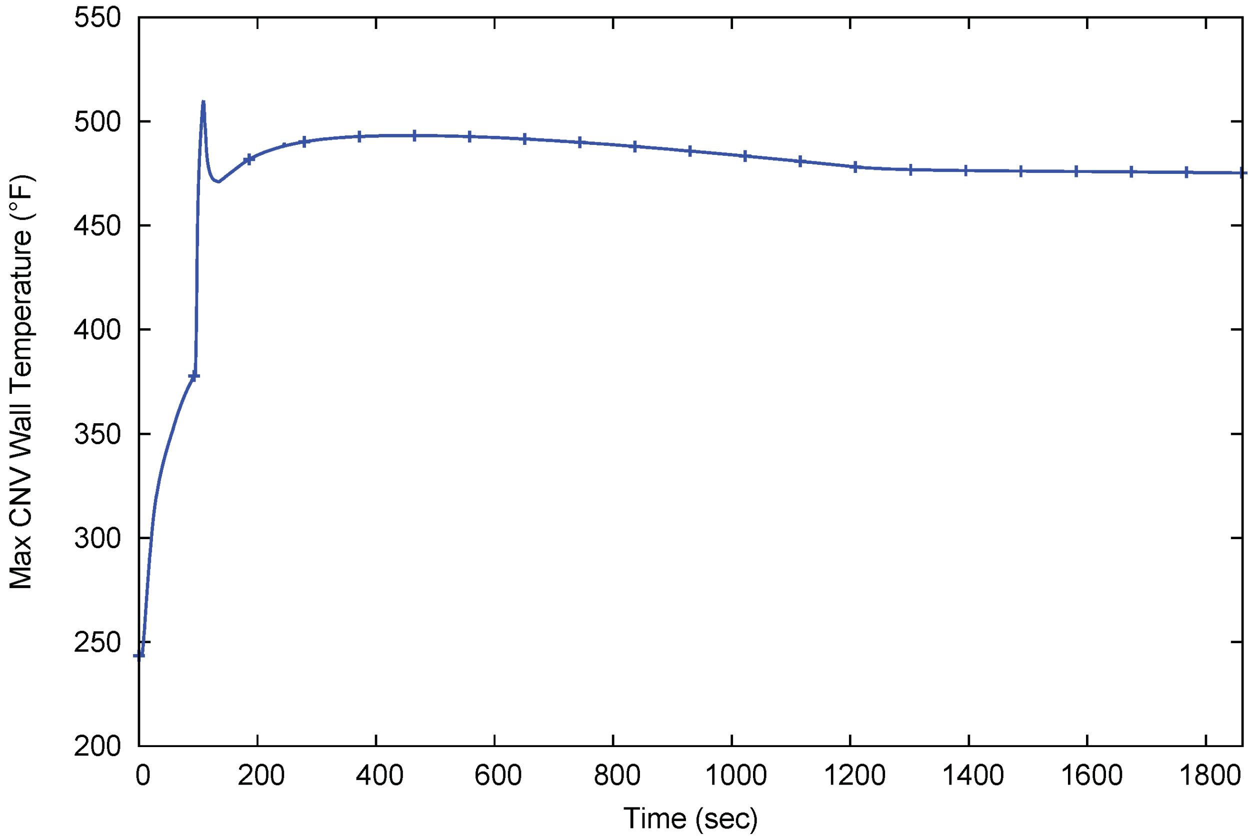

Figure 5-2 Case 1 containment vessel wall temperature - reactor coolant system discharge line break loss-of-coolant accident

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 ~1.7 and ~3.1 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 514 degrees F for the base case, and 526 degrees F with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +12 degrees F (~2.4 percent) increase are: (1) the timing of ECCS valve opening as determined by the IAB release and high CNV level setpoints (2) the assumption of a loss of normal AC power (3) fine CNV volume & heat structure nodalization and (4) single failure of one RRV to open. The peak CNV

pressure is 894 psia for the base case, and 959 psia with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +65 psi (~7.3 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 and DC power (3) single failure of one RRV to open (4) fine CNV volume & heat structure nodalization and (5) the RPV noncondensable release to CNV. The effect of the IAB release range of 950 psi +/- 50 psi was evaluated for Case 2 and determined to be equivalent to or non-limiting compared to the previously analyzed range of 1000-1200 psi. 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
- reactor trip

As a conservative assumption, either a loss of normal AC power or a loss of normal AC and DC power is also assumed to occur at the time of the break and the ECCS signal is actuated on high CNV level or IAB release pressure. 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 ECCS valves. The peak CNV pressure and peak CNV wall temperature occur following the ECCS valve 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 952 seconds, and the opening of the ECCS valves occurs at 955 seconds (after a 3-second signal delay). The ECCS actuation and opening of the three RVVs and one RRV causes the peak CNV wall temperature to occur at 978 seconds. For the peak pressure case, the ECCS signal actuates on IAB release pressure at 364 seconds, and the opening of the ECCS valves occurs at 367 seconds. The ECCS actuation and opening of the three RVVs and one RRV causes the peak CNV pressure to occur at 385 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 including the spike in mass release when ECCS valves open. 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 959 psia. This NRELAP5 analysis result is approximately 9% below the CNV design pressure of 1050 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 65 ft above the pool floor and a temperature of 110 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 1000 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 526 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)**
0	LOCA in RCS injection line For peak pressure case: • Loss of normal AC power • FW/MS isolation • Reactor trip For peak temperature case: • Loss of normal AC power • FW/MS isolation	0
3	High CNV pressure resulting in For peak pressure case: • Containment isolation For peak temperature case: • Containment isolation • Reactor trip	3
364	ECCS actuation on For peak pressure case: • IAB release pressure For peak pressure case: • high CNV level	952
367	ECCS valve opening	955
385	Peak CNV pressure reached: For peak pressure case: 959 psia For peak temperature case: 939 psia	967
384	Peak CNV temperature reached: For peak pressure case: 509 °F For peak temperature case: 526 °F	978
2200	CNV pressure decreases to <50% of peak pressure	~2500

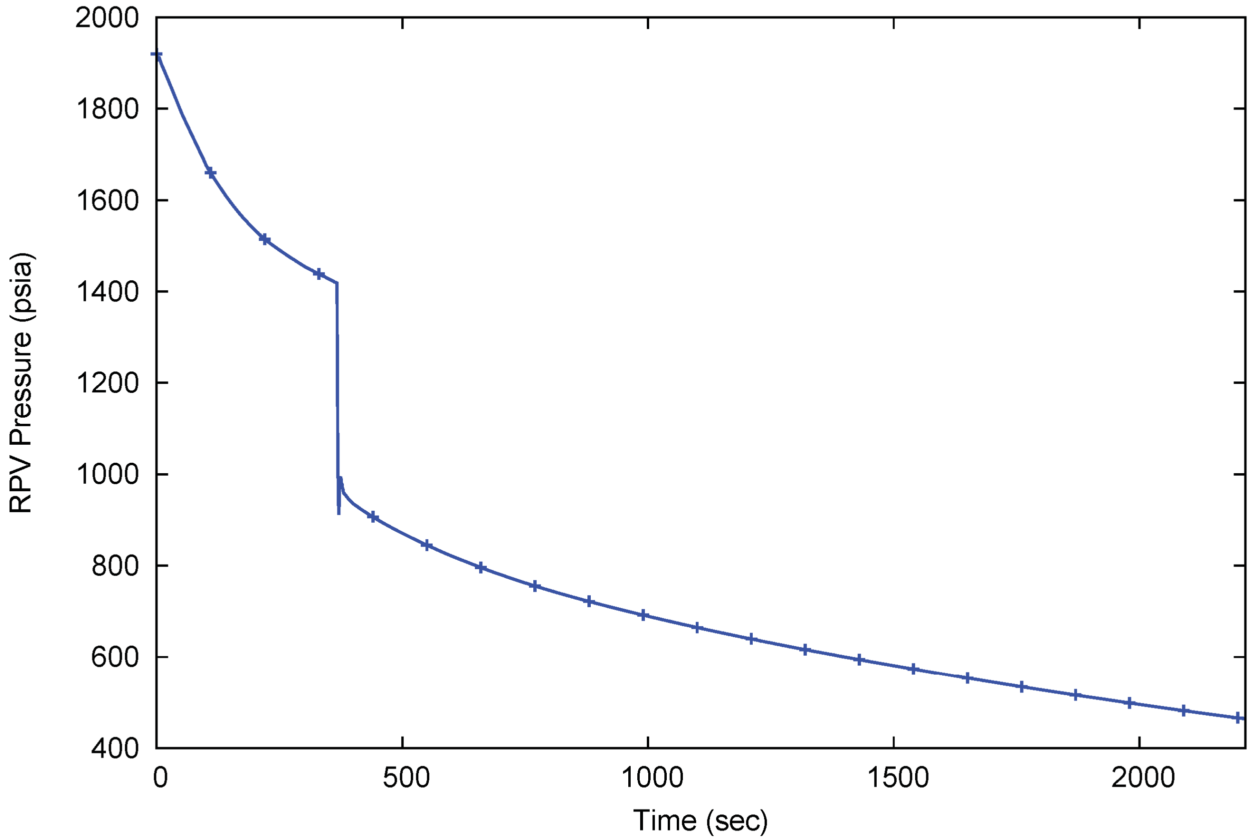

Figure 5-3 Case 2 primary system pressure – reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

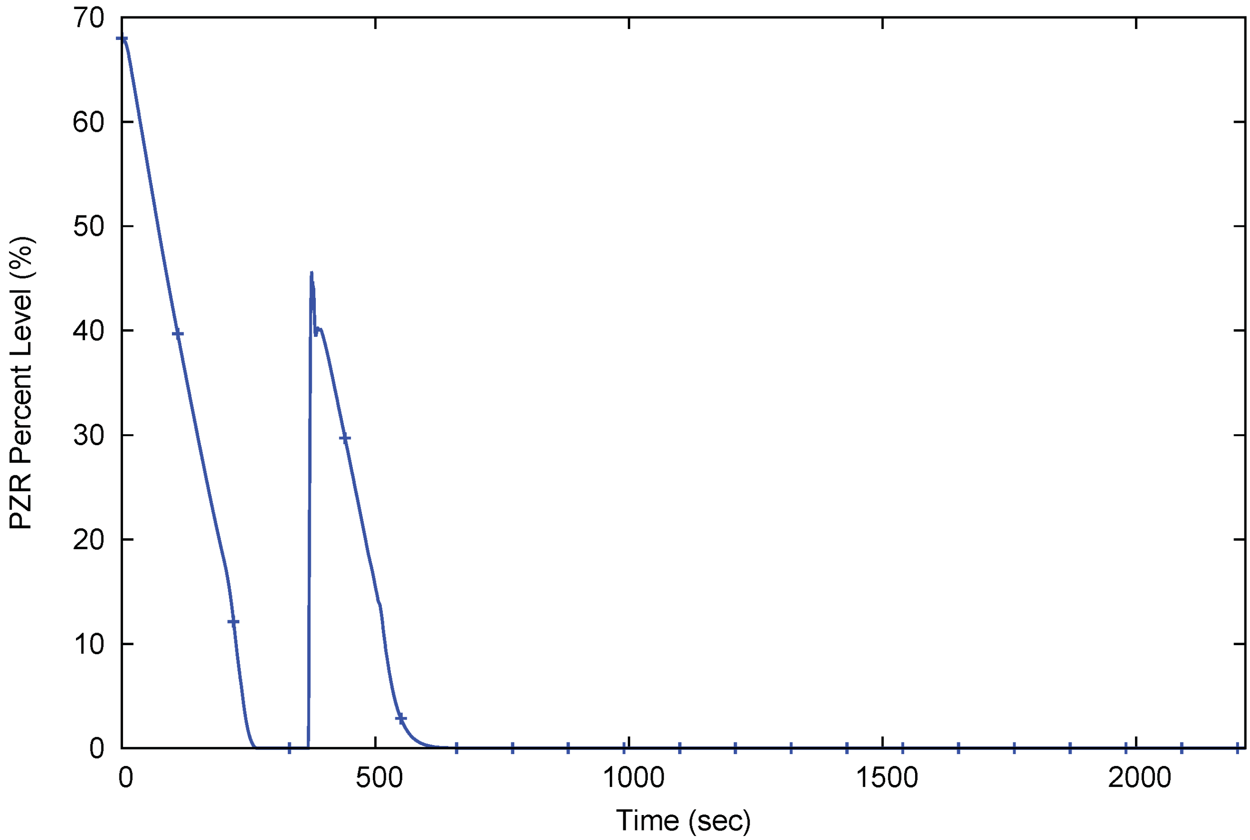

Figure 5-4 Case 2 pressurizer level - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

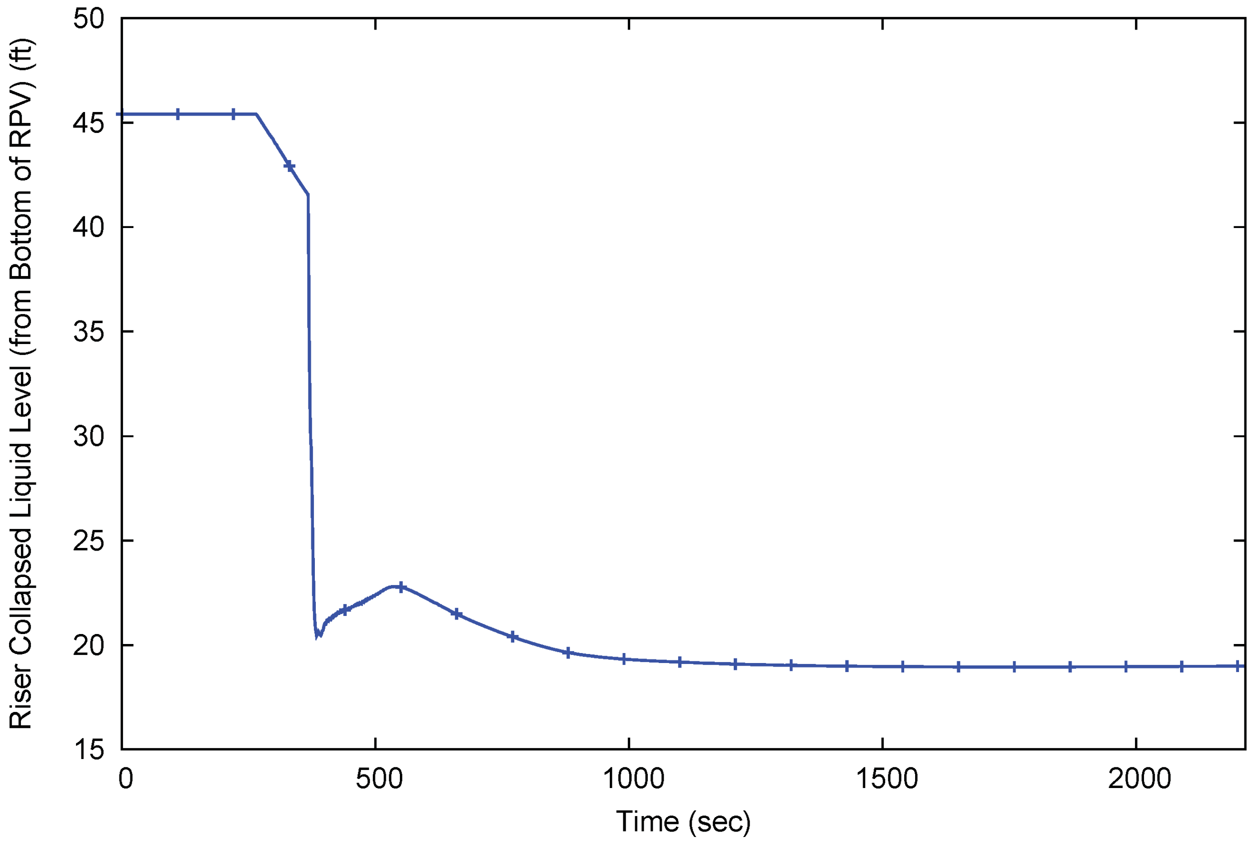

Figure 5-5 Case 2 riser level - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

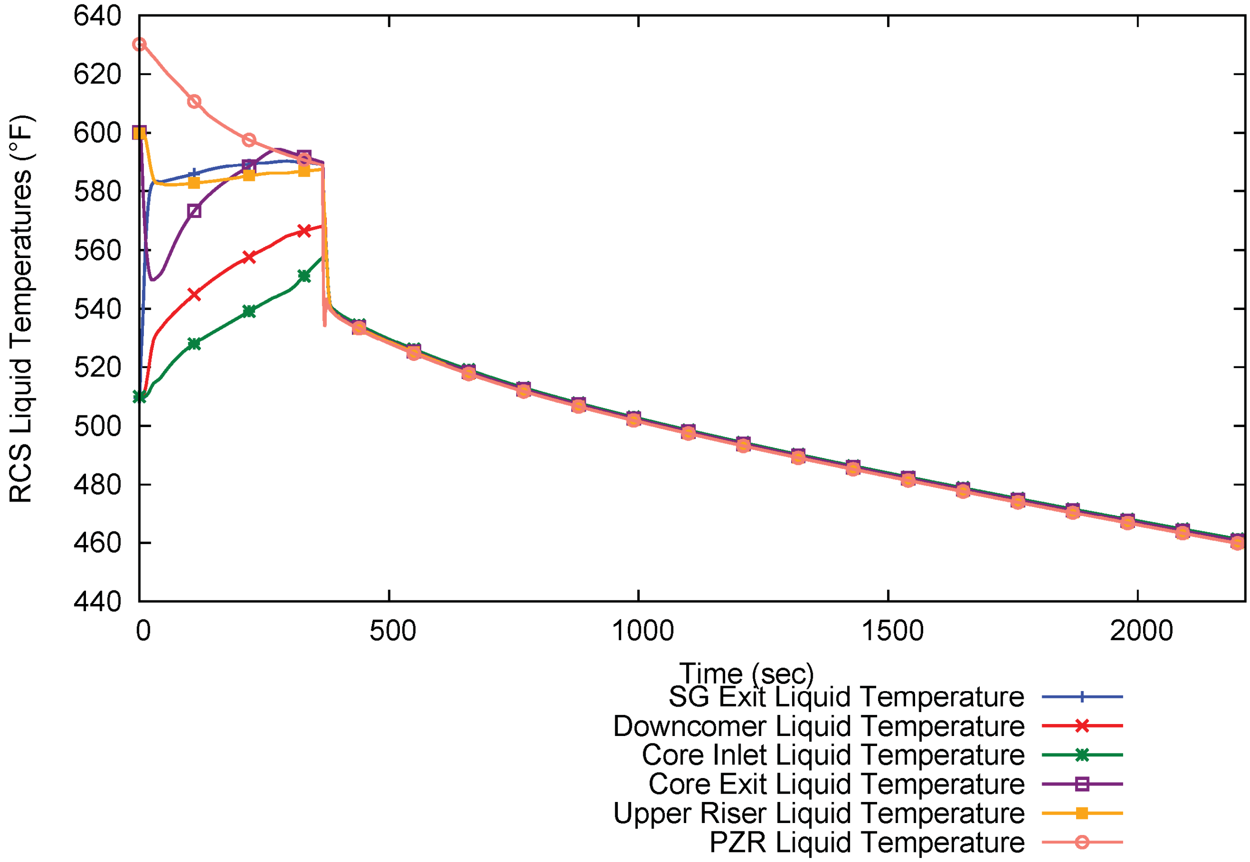

Figure 5-6 Case 2 primary temperatures - reactor coolant system injection line break loss-of-coolant accident

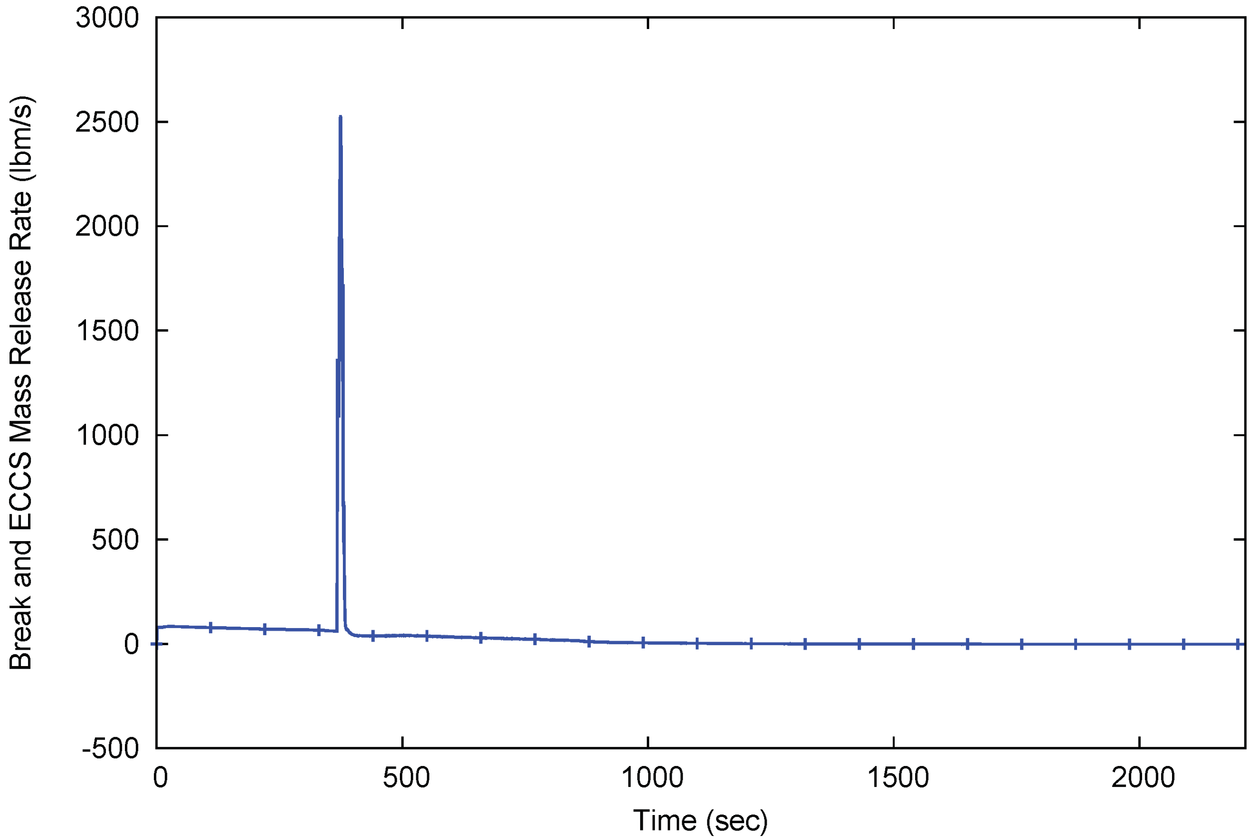

Figure 5-7 Case 2 break and emergency core cooling system mass flowrate - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

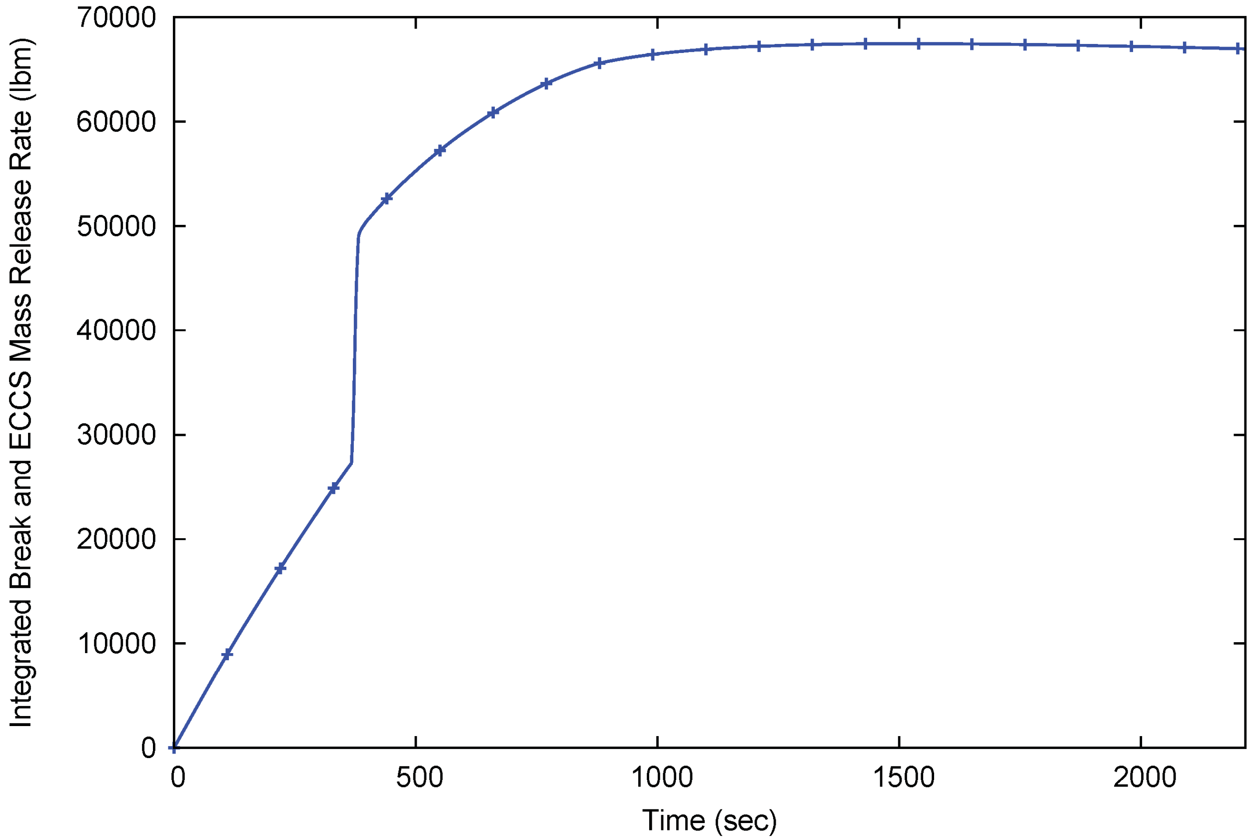

Figure 5-8 Case 2 integrated loss-of-coolant accident and emergency core cooling system mass release - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

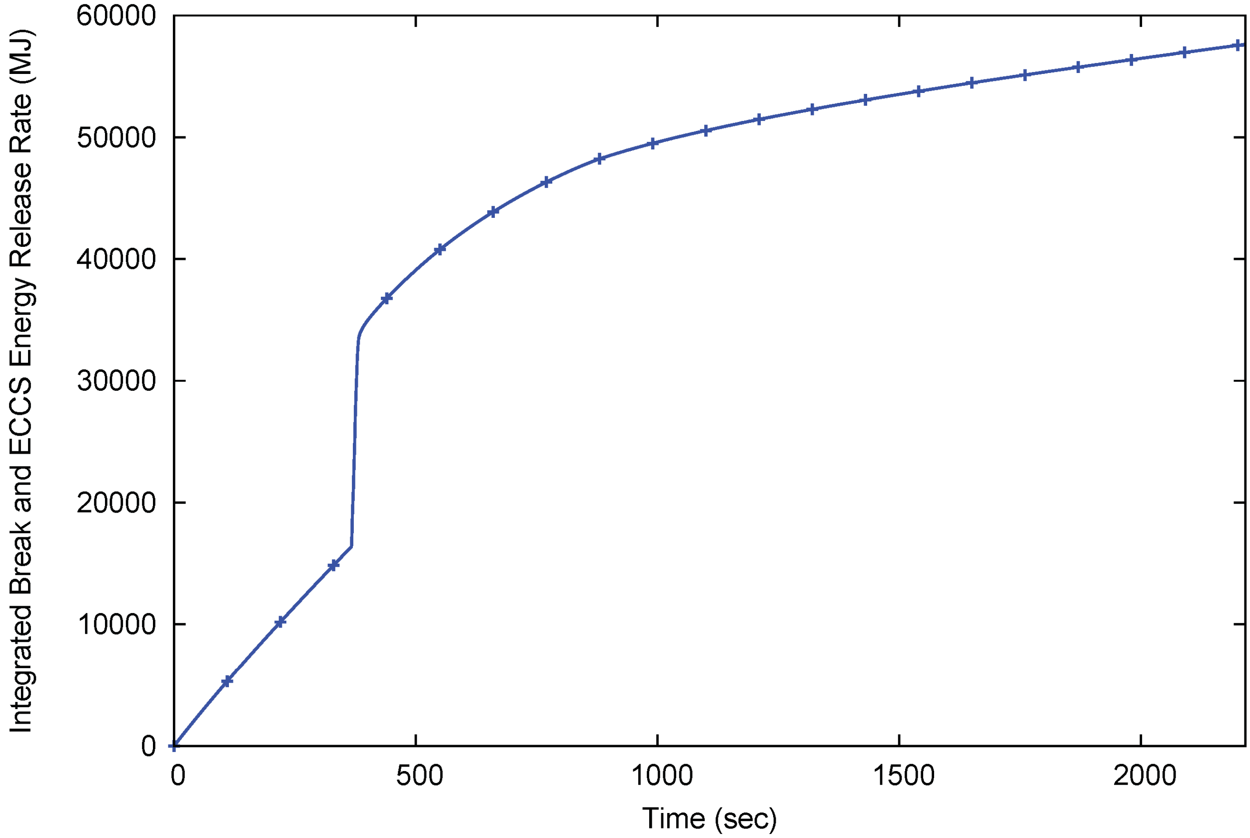

Figure 5-9 Case 2 integrated loss-of-coolant accident and emergency core cooling system energy release - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

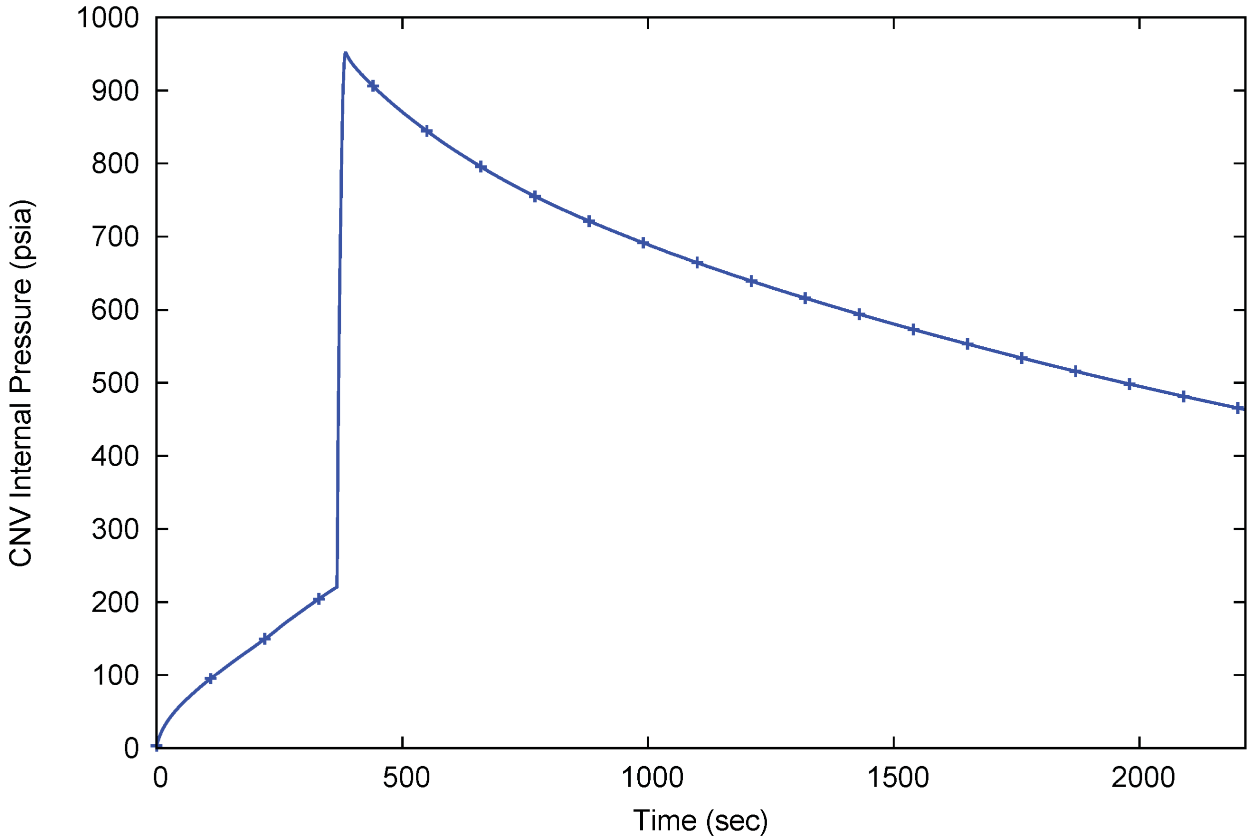

Figure 5-10 Case 2 containment vessel pressure - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

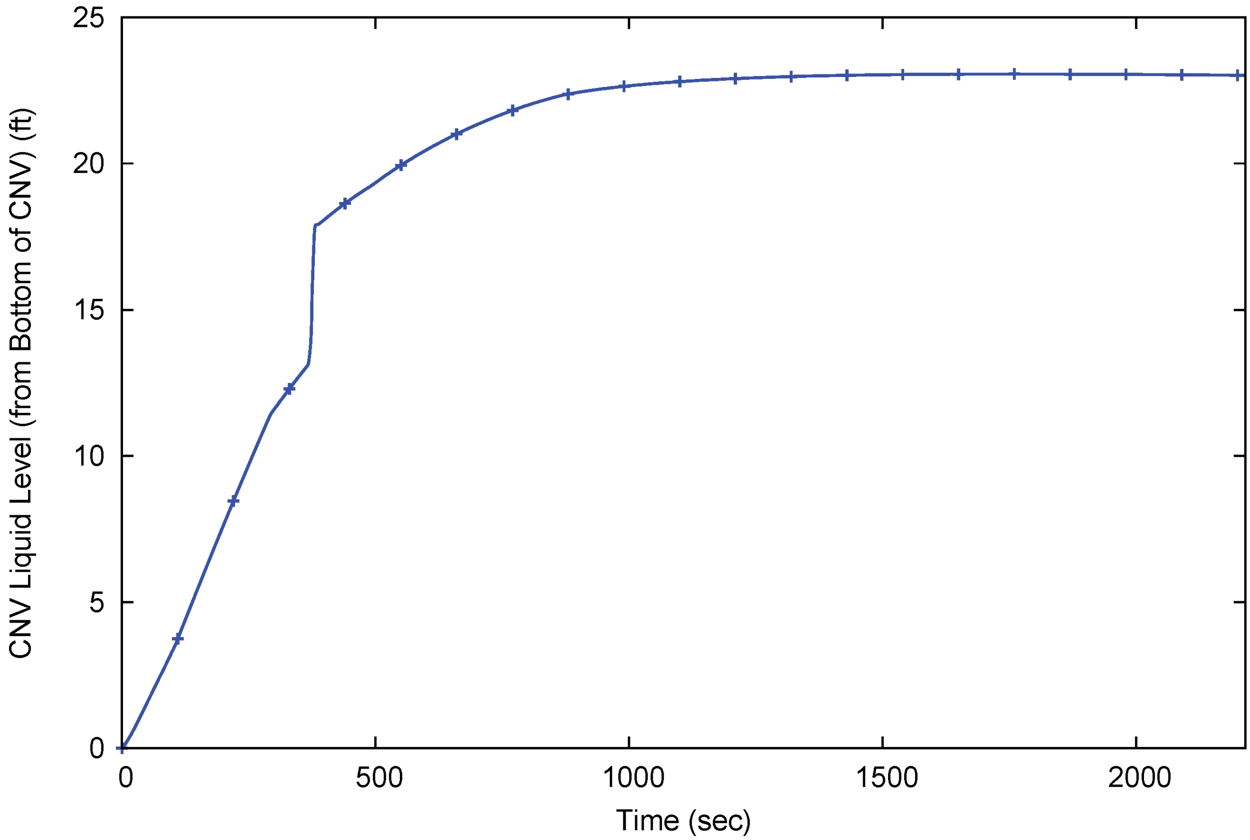

Figure 5-11 Case 2 containment vessel level - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

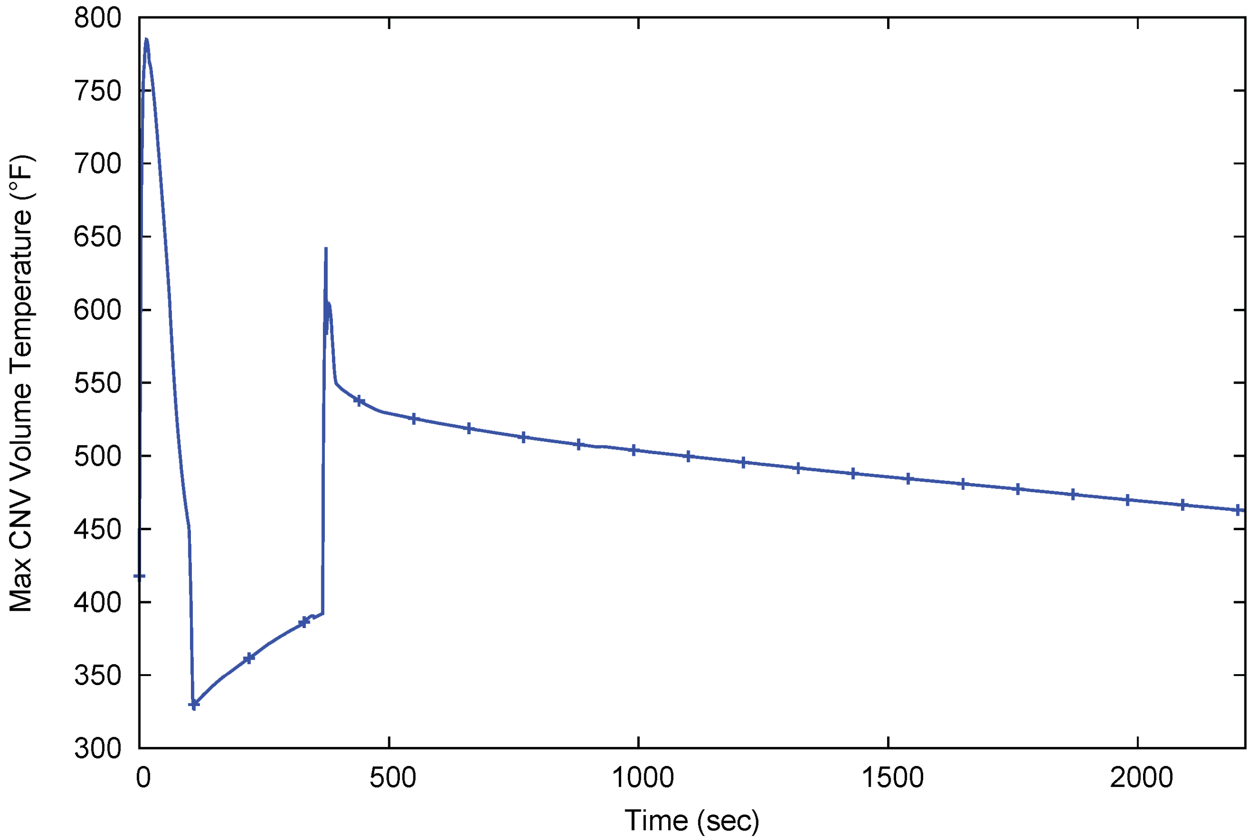

Figure 5-12 Case 2 containment vessel vapor temperature - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

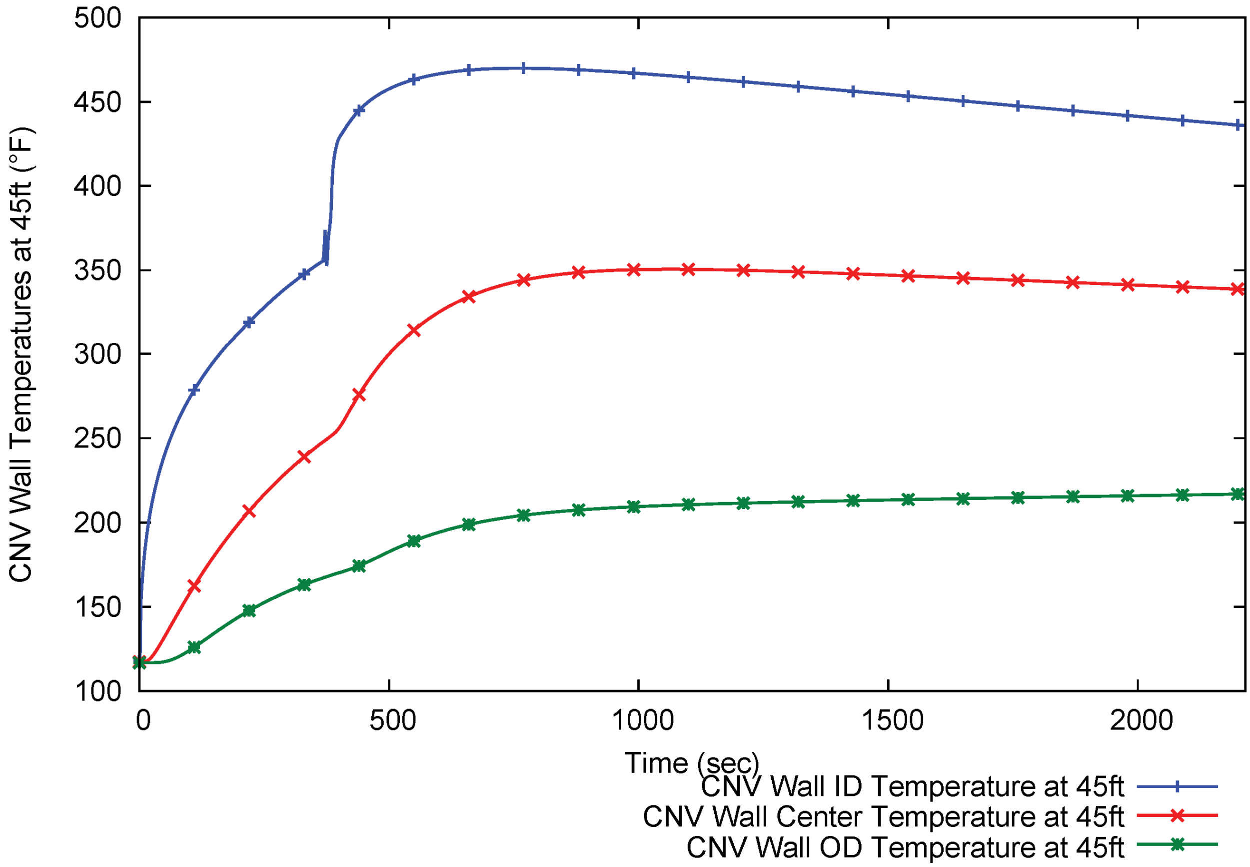

Figure 5-13 Case 2 containment vessel wall temperature profile -reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

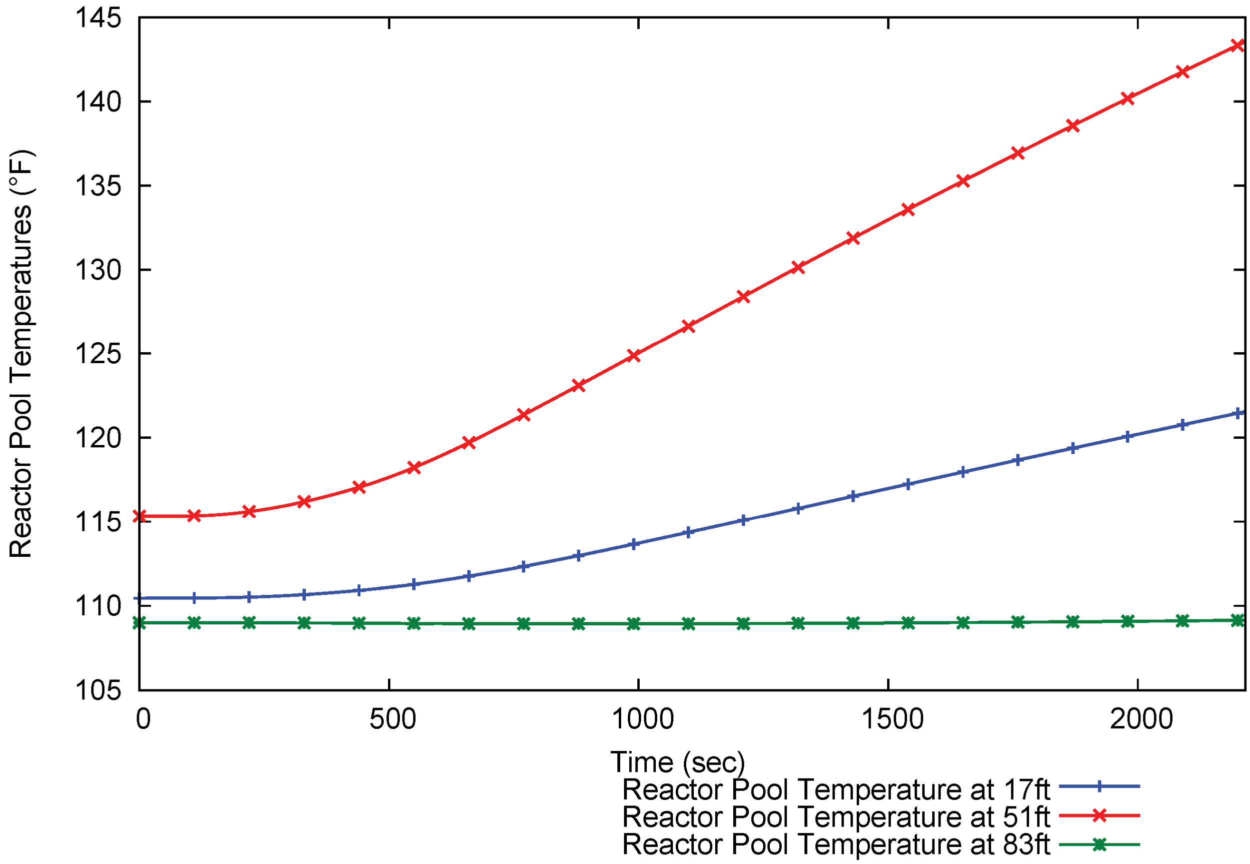

Figure 5-14 Case 2 reactor pool temperatures - reactor coolant system injection line break loss-of-coolant accident (peak pressure case)

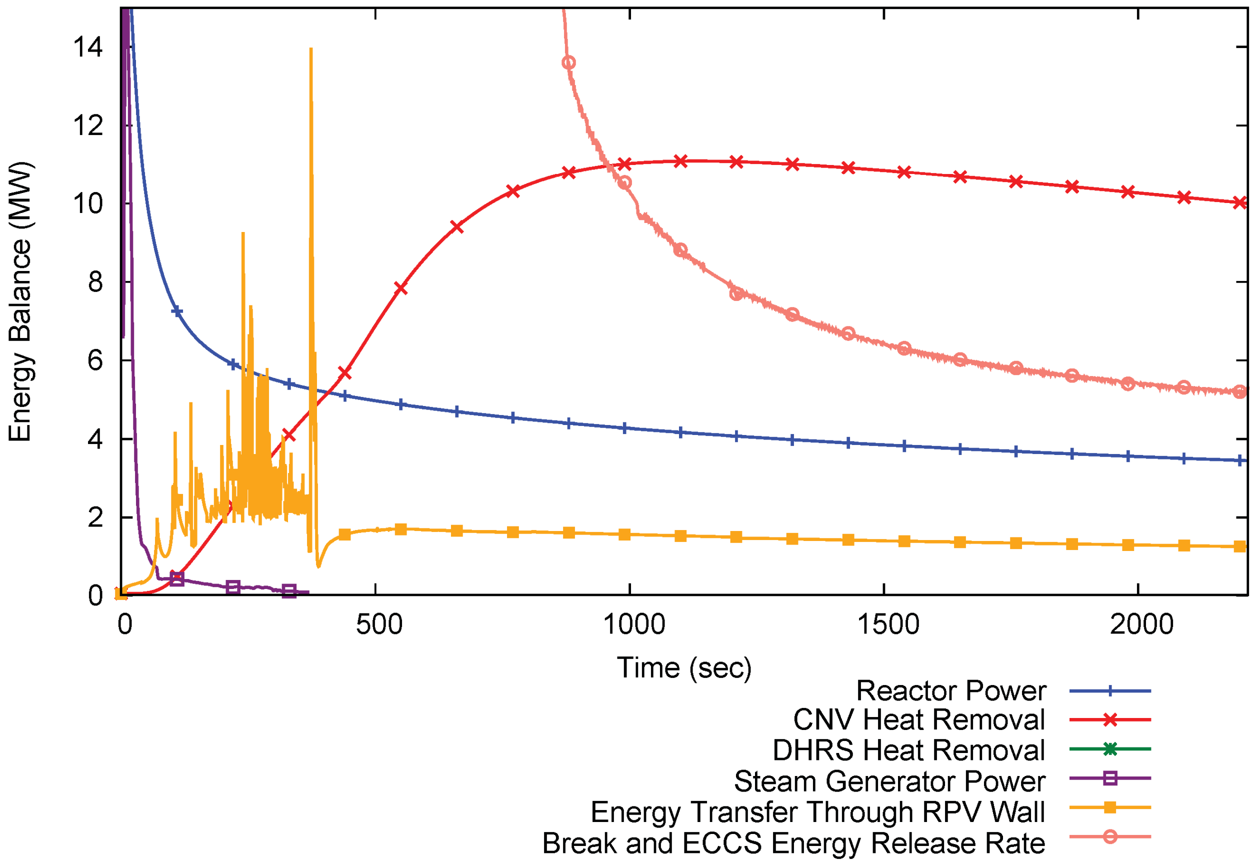

Figure 5-15 Case 2 energy balance - reactor coolant system injection lIne break loss-of-coolant accident (peak pressure case)

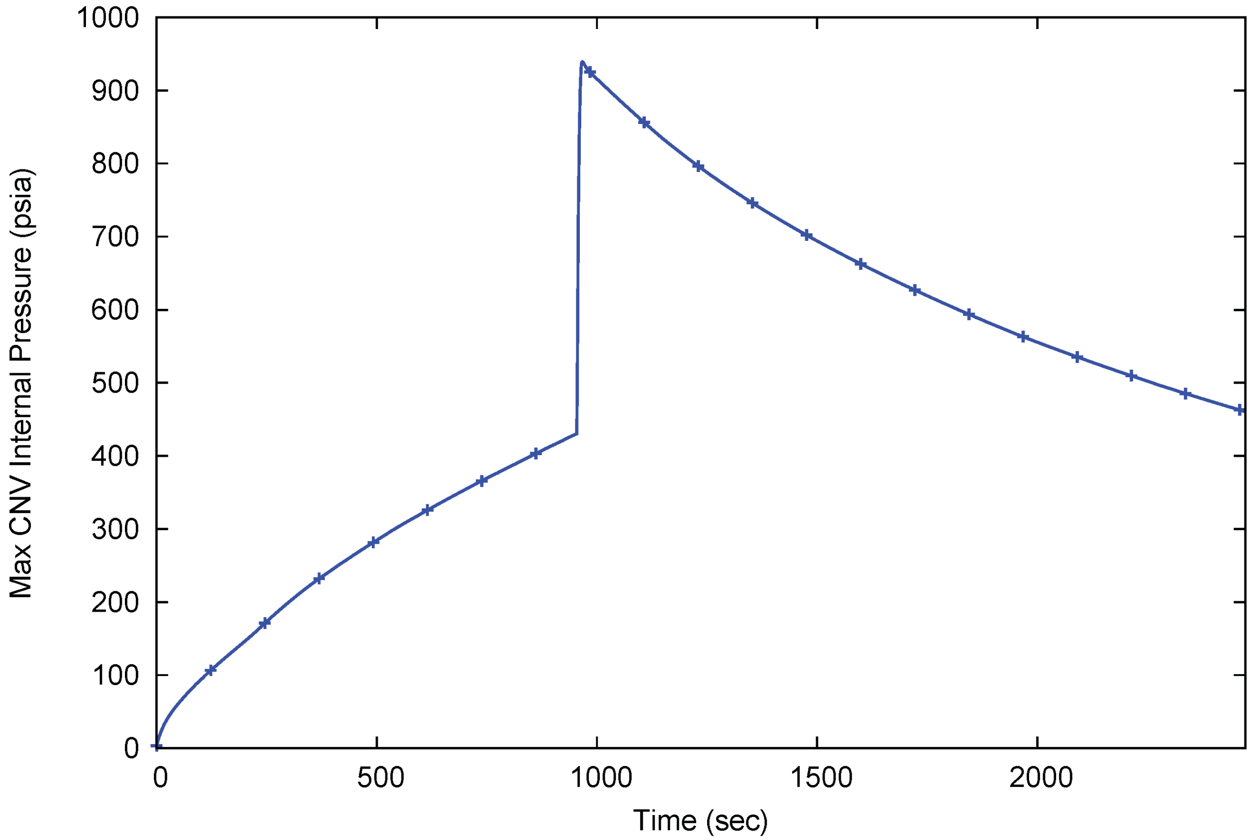

Figure 5-16 Case 2 containment vessel pressure - reactor coolant system injection line break loss-of-coolant accident (peak CNV wall temperature case)

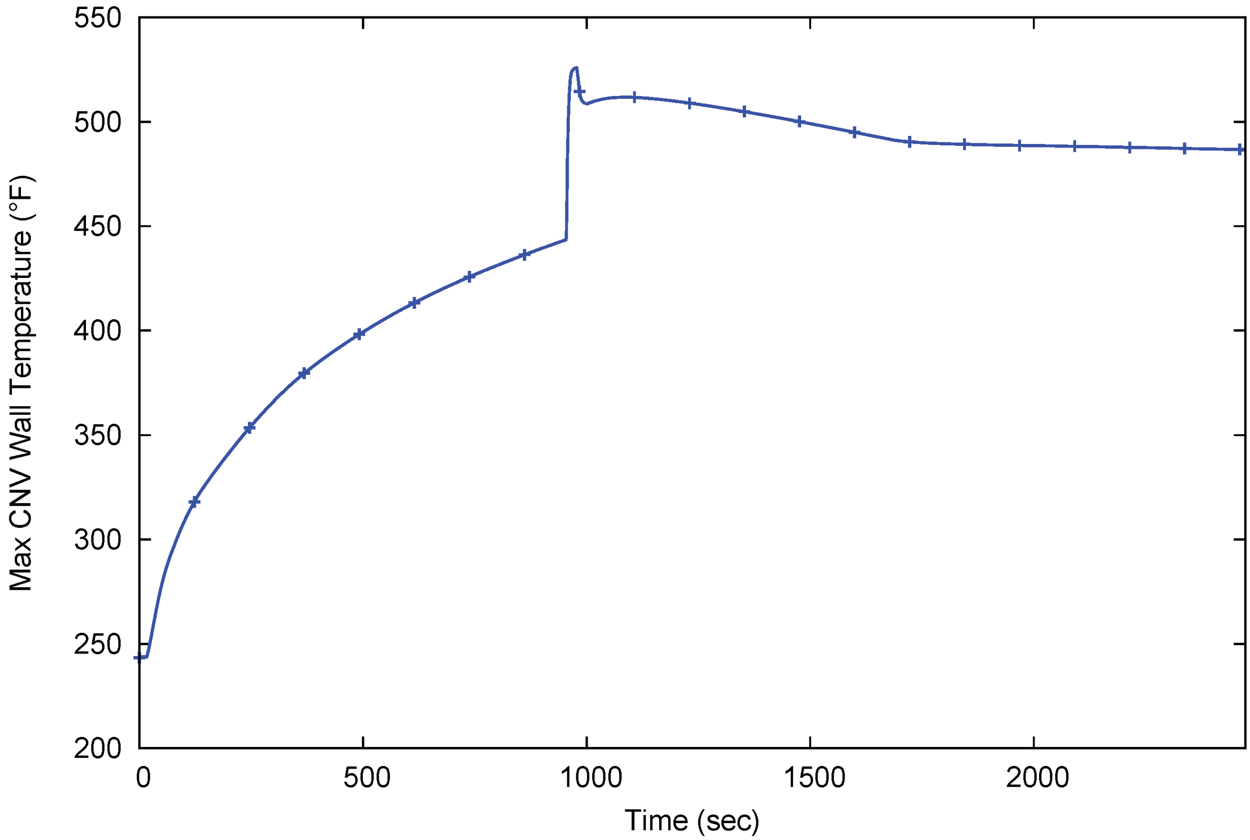

Figure 5-17 Case 2 containment vessel peak wall temperature - reactor coolant system injection line break loss-of-coolant accident (peak CNV wall temperature case)

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 CNV peak pressure is 554 psia for the base case, and 901 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse IAB release pressure, high RCS flow, fine CNV volume nodalization). The peak CNV wall temperature is 471 degrees F for the base case, and 489 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse IAB release pressure, high RCS flow, fine CNV volume nodalization). The results of this case are adequately representative of the inadvertent RVV opening event, although they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The effect of the IAB release pressure range of 950 psi +/- 50 psi, including effect of valves opening at different pressures within that range, has been evaluated. Case 3 is non-limiting and was confirmed to be non-limiting in comparison to the RRV opening event.

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)
0	LOCA in RPV high point degasification line For peak pressure case only: • Loss of normal AC and DC power • FW/MS isolation • Reactor trip	0
1	High CNV pressure resulting in For peak pressure case: • Containment isolation For peak temperature case: • Reactor trip • FW/MS isolation • Loss of normal AC and DC power assumed at turbine trip	1
58	ECCS actuation on: IAB release pressure	106
61	ECCS valve opening	109
82	Peak CNV pressure reached: For peak pressure case: 901 psia For peak temperature case: 894 psia	128
454	Peak CNV temperature reached: For peak pressure case: 487 °F For peak temperature case: 489 °F	478
~1900	CNV pressure decreases to <50% of peak pressure	~2000

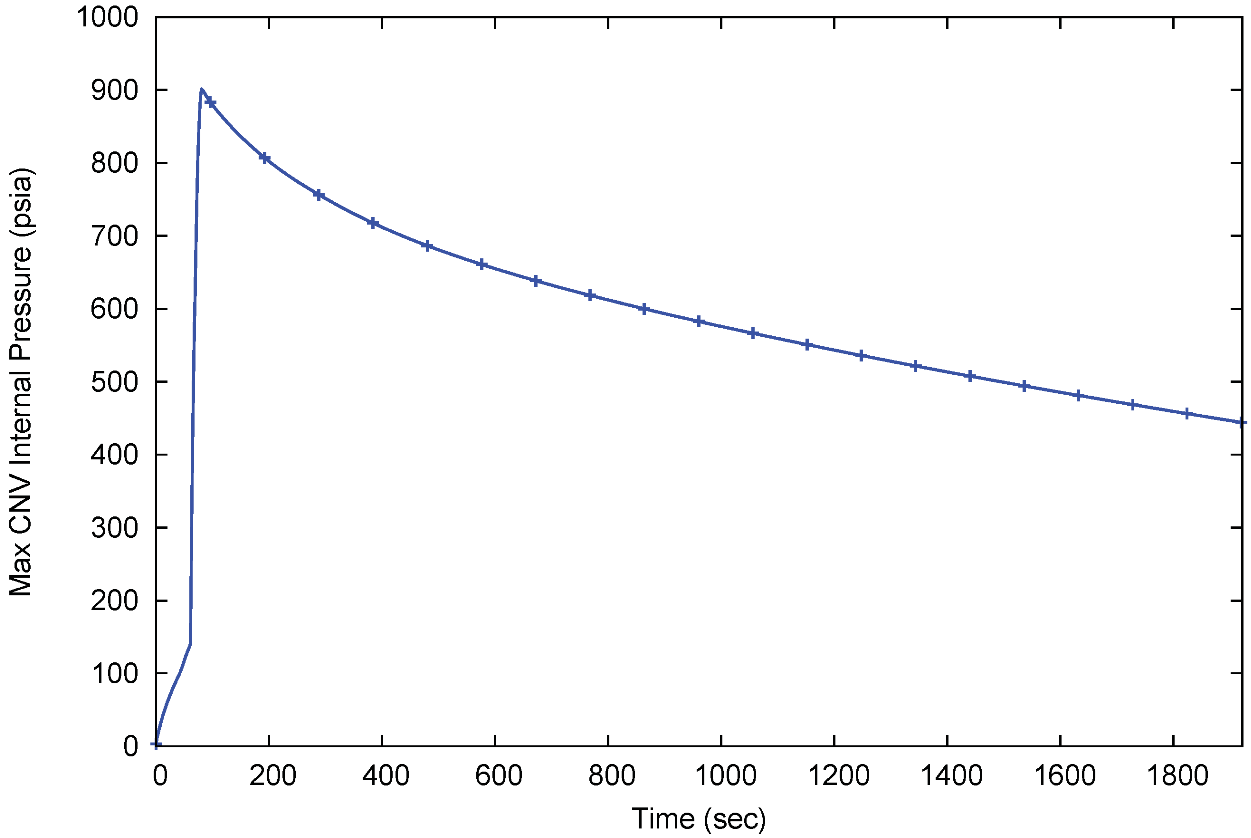

Figure 5-18 Case 3 containment vessel pressure - high point vent line break loss-of-coolant accident

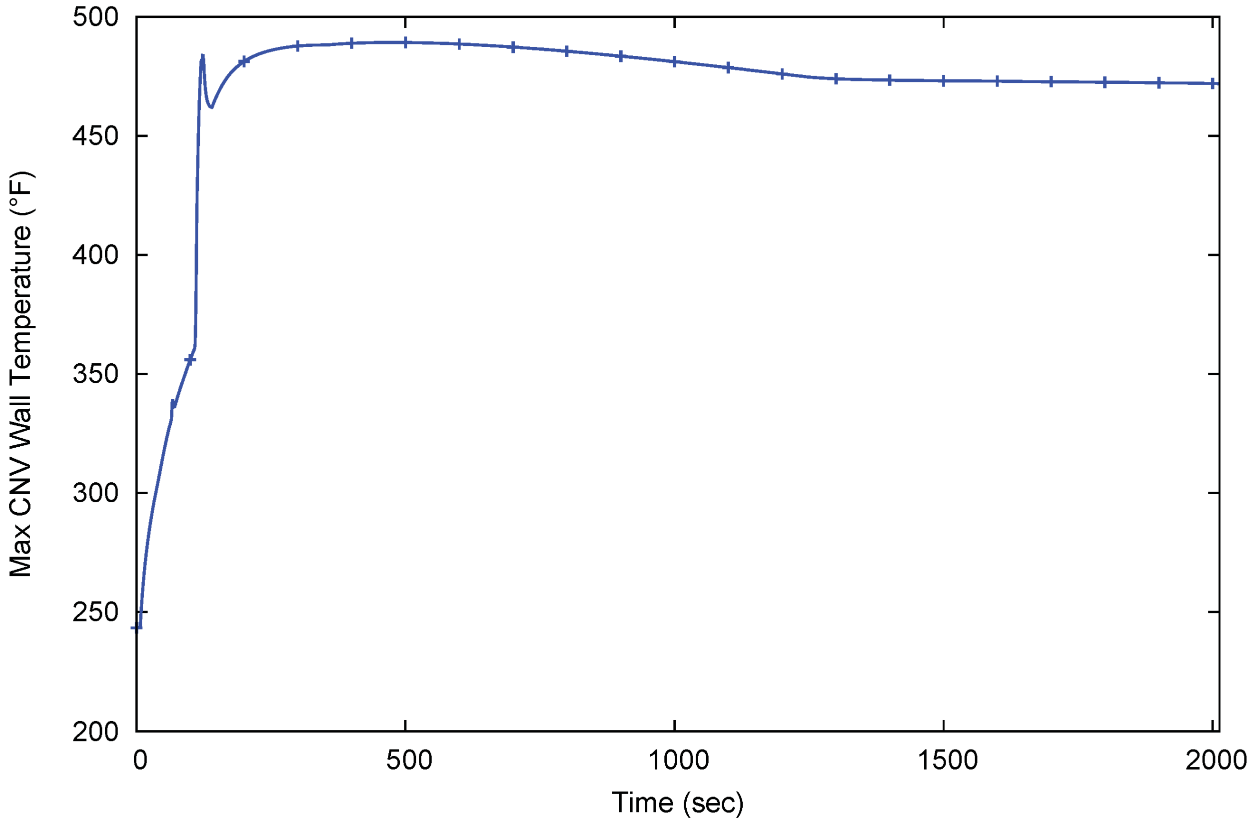

Figure 5-19 Case 3 containment vessel wall temperature – high point vent line break loss-of-coolant accident

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 CNV peak pressure is 856 psia for the base case, and 911 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse IAB release pressure, low RCS flow, fine CNV heat structure & reactor pool nodalization). The peak CNV temperature is 483 degrees F for the base case, and 486 degrees F for the case with the combined effect of the adverse sensitivity parameters (normal AC and DC power available, fine CNV volume nodalization). The results of this case are adequately representative of the high point line break, even though they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The effect of the IAB release pressure range of 950 psi +/- 50 psi, including effect of valves opening at different pressures within that range, has been evaluated. Case 4 is non-limiting and was confirmed to be non-limiting in comparison to the RRV opening event.

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)**
0	Inadvertent RVV actuation For peak pressure case only: • Loss of normal AC and DC power • FW/MS isolation • Reactor trip	0
0.2	High CNV pressure resulting in For peak pressure case: • Containment isolation For peak temperature case: • Containment isolation • Reactor trip • FW/MS isolation	0.2
7	ECCS actuation on: IAB release pressure	n/a
10	ECCS valves opening	n/a
27	Peak CNV pressure reached For peak pressure case: 911 psia For peak temperature case: 855 psia	57
361	Peak CNV temperature reached For peak pressure case: 482 °F For peak temperature case: 486 °F	437
~1700	CNV pressure decreases to <50% of peak pressure	~2000

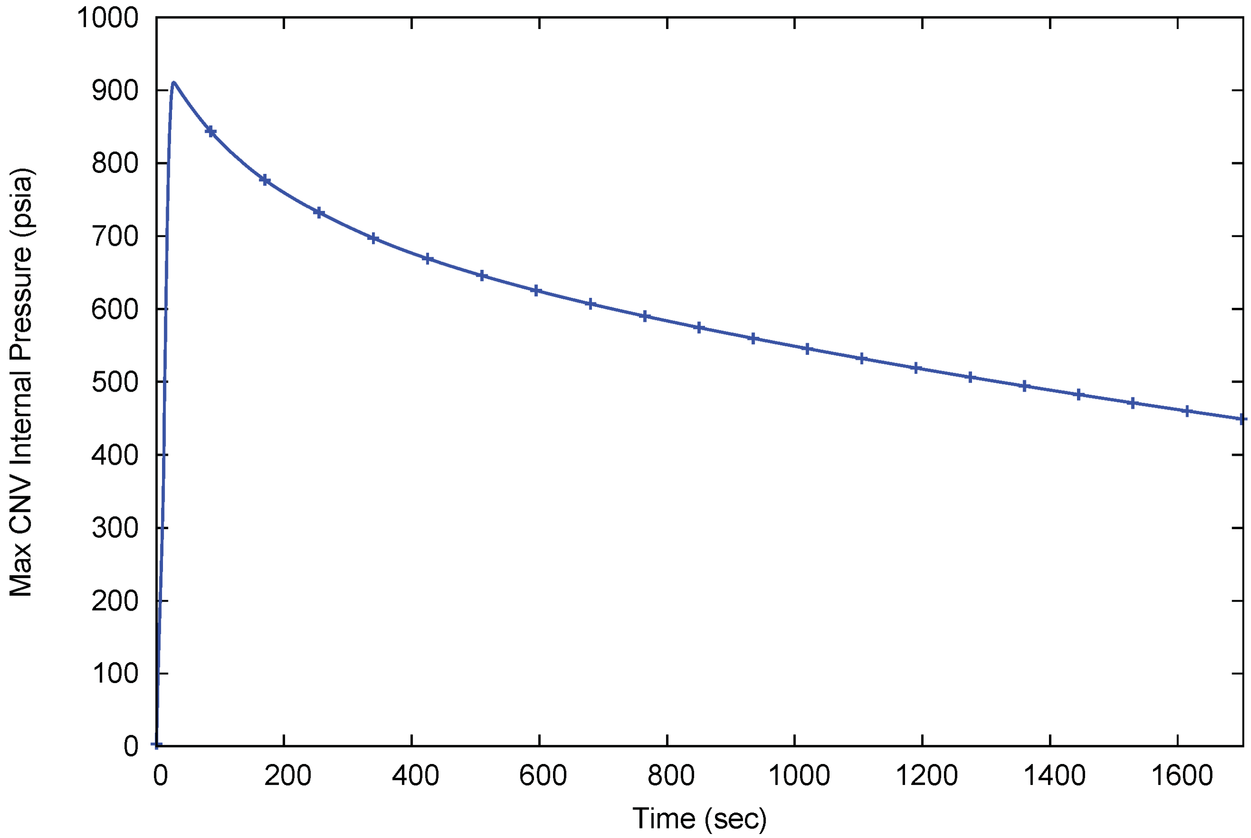

Figure 5-20 Case 4 containment vessel pressure – inadvertent reactor vent valve opening event

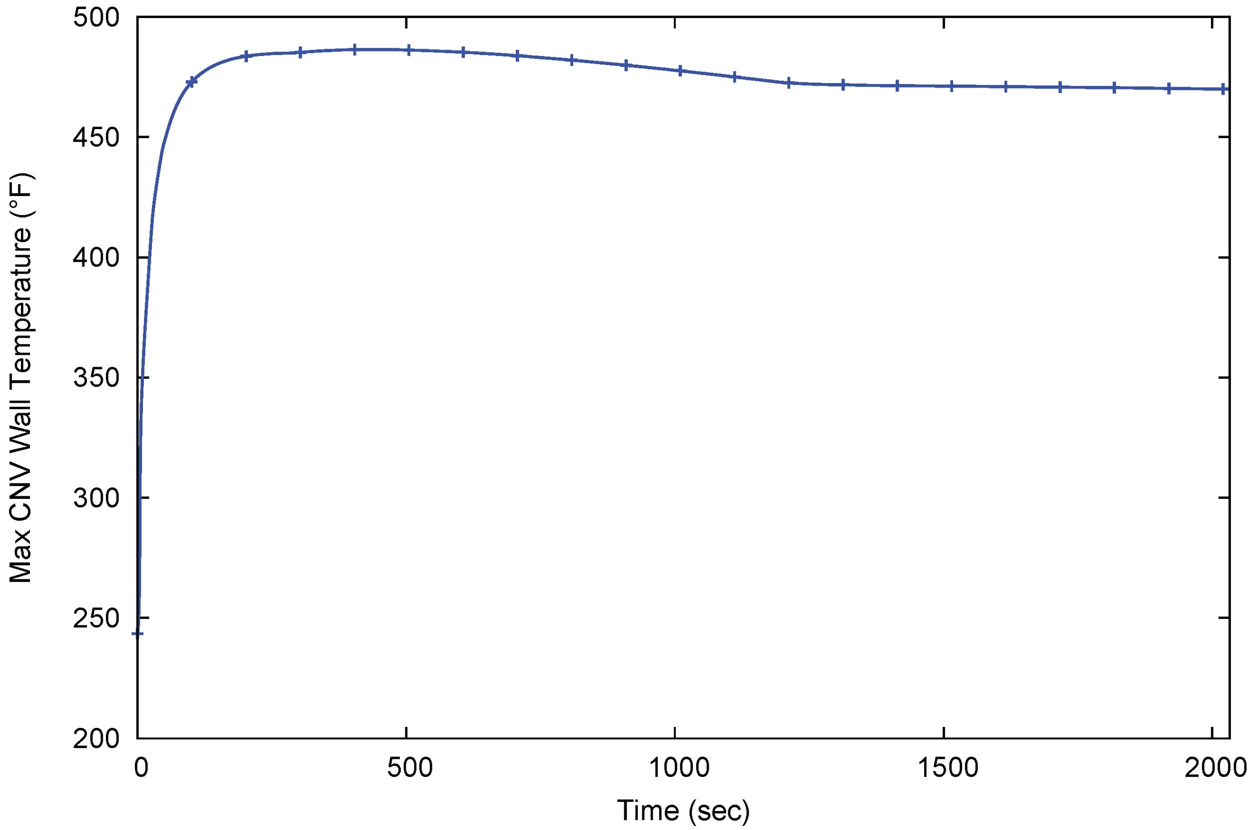

Figure 5-21 Case 4 containment vessel wall temperature – inadvertent reactor vent valve opening event

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 limiting case, which accounts for the IAB release pressure of 950 psid +/- 50 psi and the potential for ECCS valves to open at different differential pressures over this range, is summarized in FSAR Section 6.2.

The following discussion reflects the RRV opening event analysis when the ECCS valves are assumed to open at 1000 psid, resulting in a peak pressure of 986 psia. This case is representative of the RRV opening event. The sequence of events for this representative case is shown in Table 5-6, and detailed results for key parameters are shown in Figures 5-22 through 5-35. The CNV peak pressure is 941 psia for the base case, and 986 psia with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +45 psi (~4.8 percent) increase are: (1) the timing of the ECCS valve opening as determined by the IAB release and high CNV level setpoint; (2) the assumption of a loss of normal AC and DC power; (3) single failure of an RRV; (4) fine

CNV volume & heat structure and reactor pool nodalization; (5) fast RPV non-condensable release to CNV; and (6) low RCS flow. The peak CNV temperature is 492 degrees F for the base case, and 512 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC power, low-biased high CNV level setpoint, single failure of an RRV, fine CNV volume & reactor pool nodalization).

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

For the peak temperature case, a loss of normal AC power is assumed to occur at the time of the break. RRVs and RVVs opening does not occur until the high CNV level setpoint is reached. In the containment response analysis methodology the high CNV level setpoint 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 wall temperature occurs following the RVVs opening after the CNV has been preheated by the initial M&E release.

For the peak pressure case, 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 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 ECCS valves occurs at 171 seconds for the peak temperature case (when the high CNV level setpoint is reached) and at 77 seconds for the peak pressure case (when the RCS pressure decreases to an adverse IAB release pressure), as determined by the results of sensitivity analyses. For the peak temperature case, opening of the three RVVs and the second RRV results in the peak CNV pressure and wall temperature at 182 and 180 seconds, respectively. For the peak pressure case, opening of the three RVVs and the second RRV results in the peak CNV pressure and wall temperature at 91 and 596 seconds, respectively. As flow through the RVVs diminishes, 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 representative Case 5 inadvertent RRV opening event (peak pressure case) 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 representative Case 5 inadvertent RRV opening event 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 986 psia. This limiting NRELAP5 result can be compared to the CNV design pressure of 1050 psia. 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 492 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 65 ft above the pool floor and a temperature of 110 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 opening event and the trends of the heat sources and sinks. At approximately 750 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

Peak CNV Pressure Case Time (sec)	Event	Peak CNV Temperature Case Time (sec)
0	Inadvertent RRV actuation: For peak temperature case • Loss of normal AC power • FW/MS isolation For peak pressure case • Loss of normal AC and DC power • FW/MS isolation • Reactor trip	0
0.4	High CNV pressure resulting in: For peak temperature case • Containment isolation • Reactor trip For peak pressure case • Containment isolation	0.4
74	ECCS actuation on : For peak temperature case • high CNV level For peak pressure case • IAB release pressure	168
77	ECCS valve opening	171
91	Peak CNV pressure reached: For peak pressure case: 986 psia For peak temperature case: 967 psia	182
596	Peak CNV temperature reached: For peak pressure case: 492 °F For peak temperature case: 512 °F	180
~1800	CNV pressure decreases to <50% of peak pressure	~1800

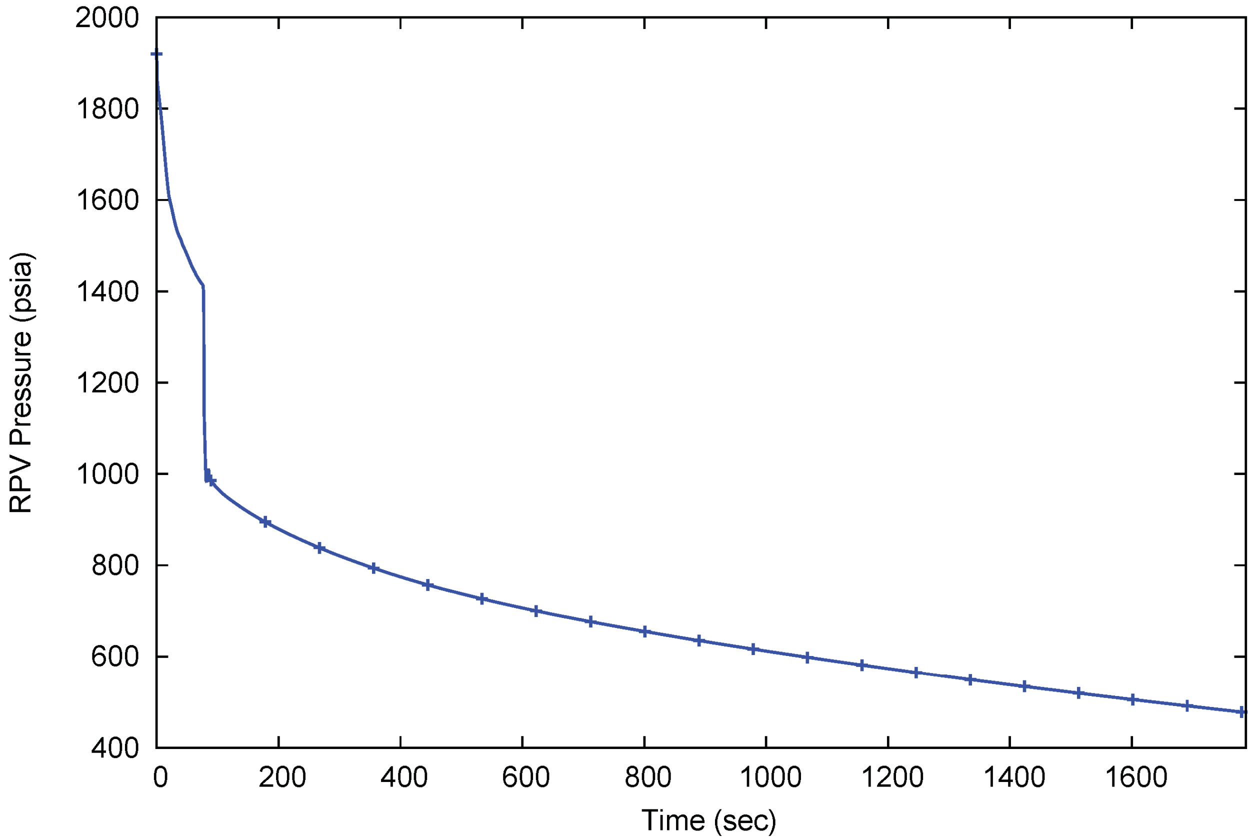

Figure 5-22 Case 5 primary pressure - inadvertent reactor recirculation valve opening event

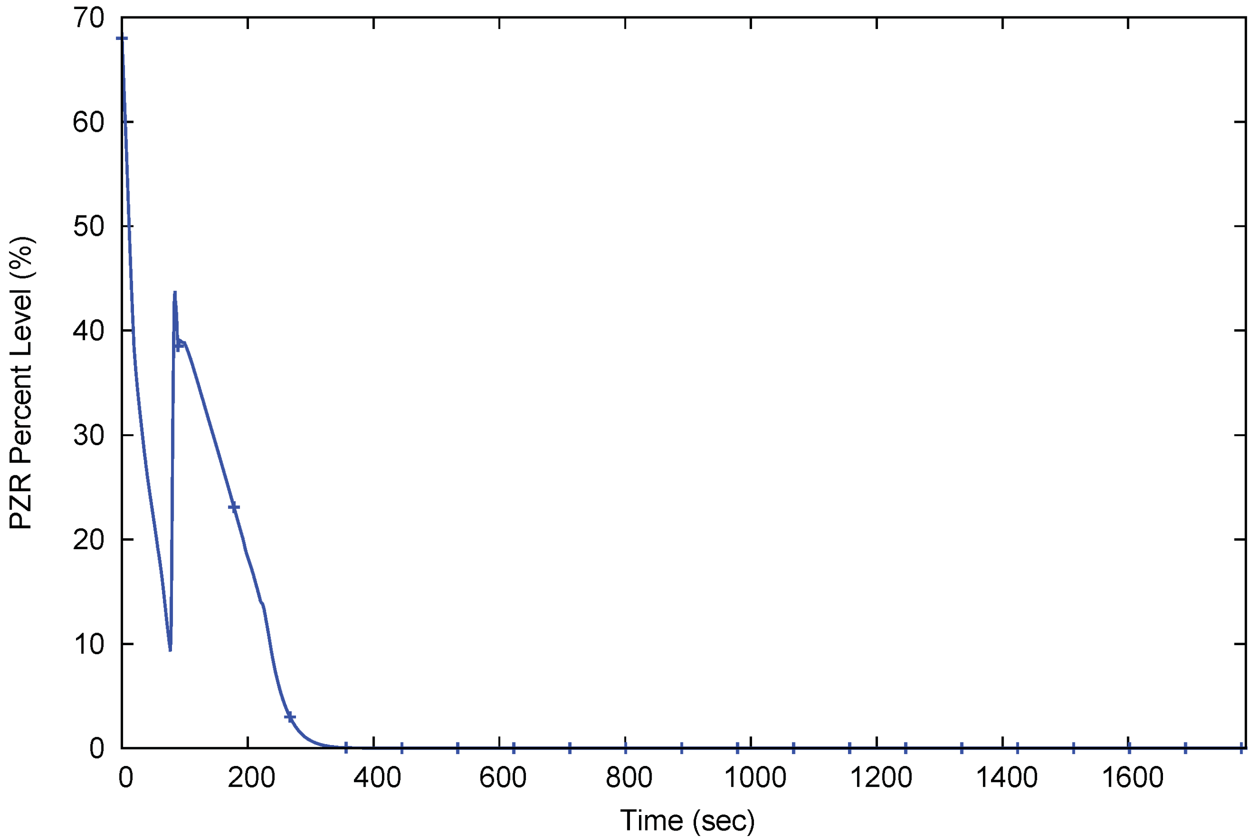

Figure 5-23 Case 5 pressurizer level - inadvertent reactor recirculation valve opening event

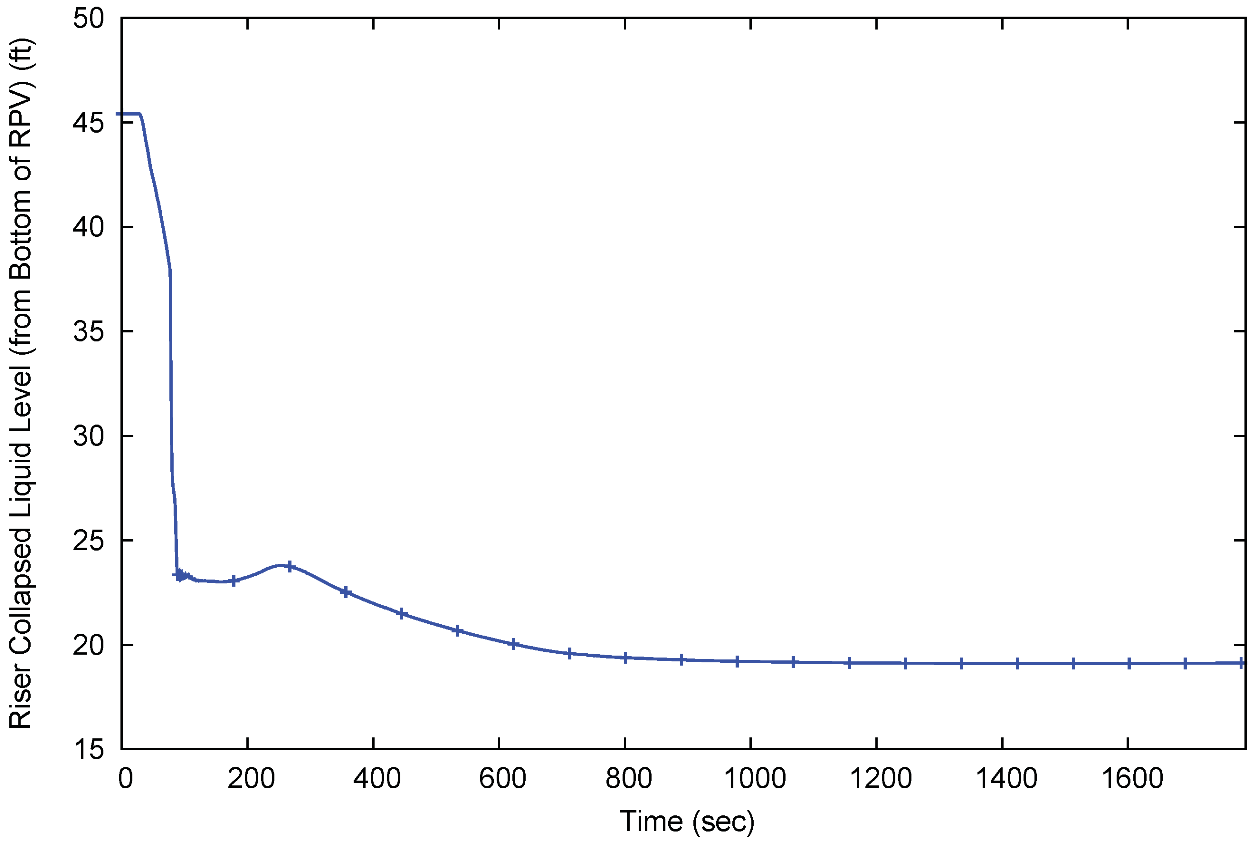

Figure 5-24 Case 5 riser level - inadvertent reactor recirculation valve opening event

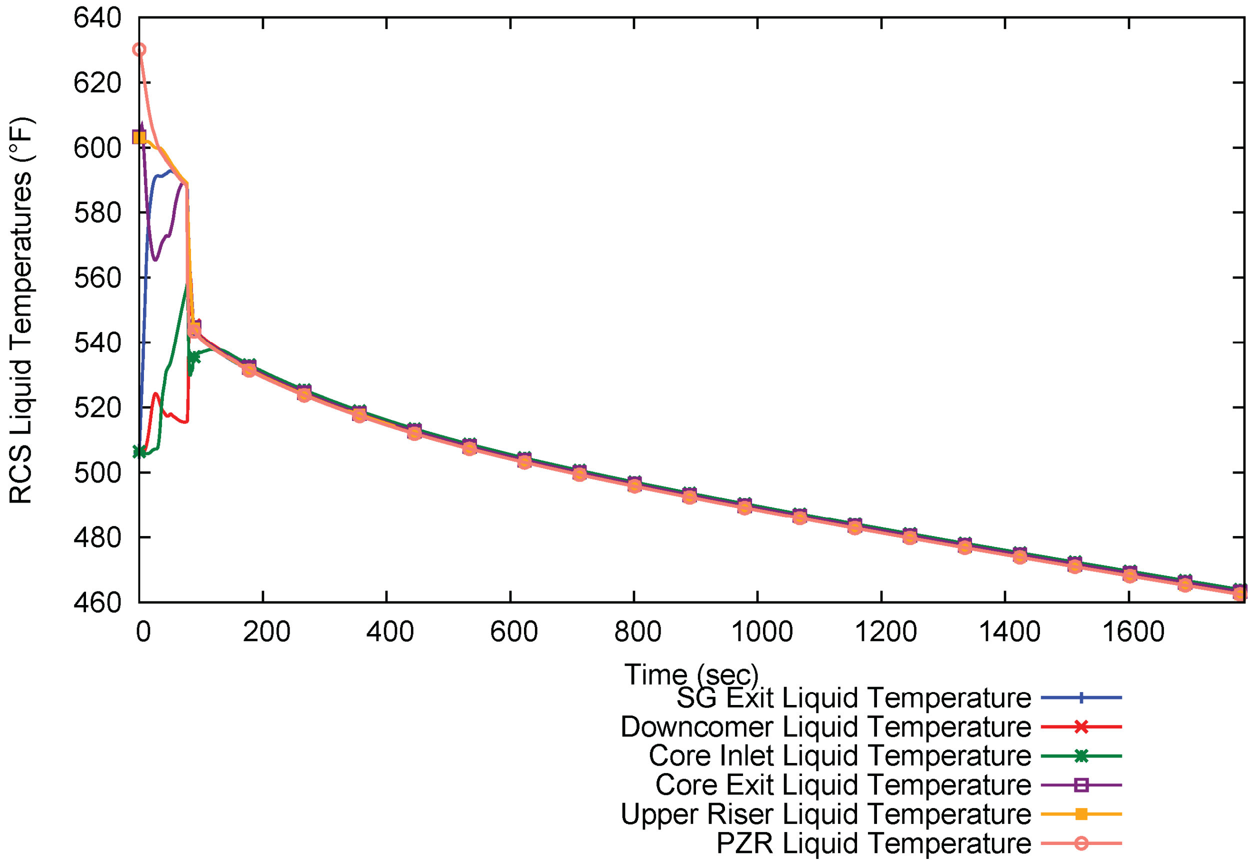

Figure 5-25 Case 5 primary temperature - inadvertent reactor recirculation valve opening event

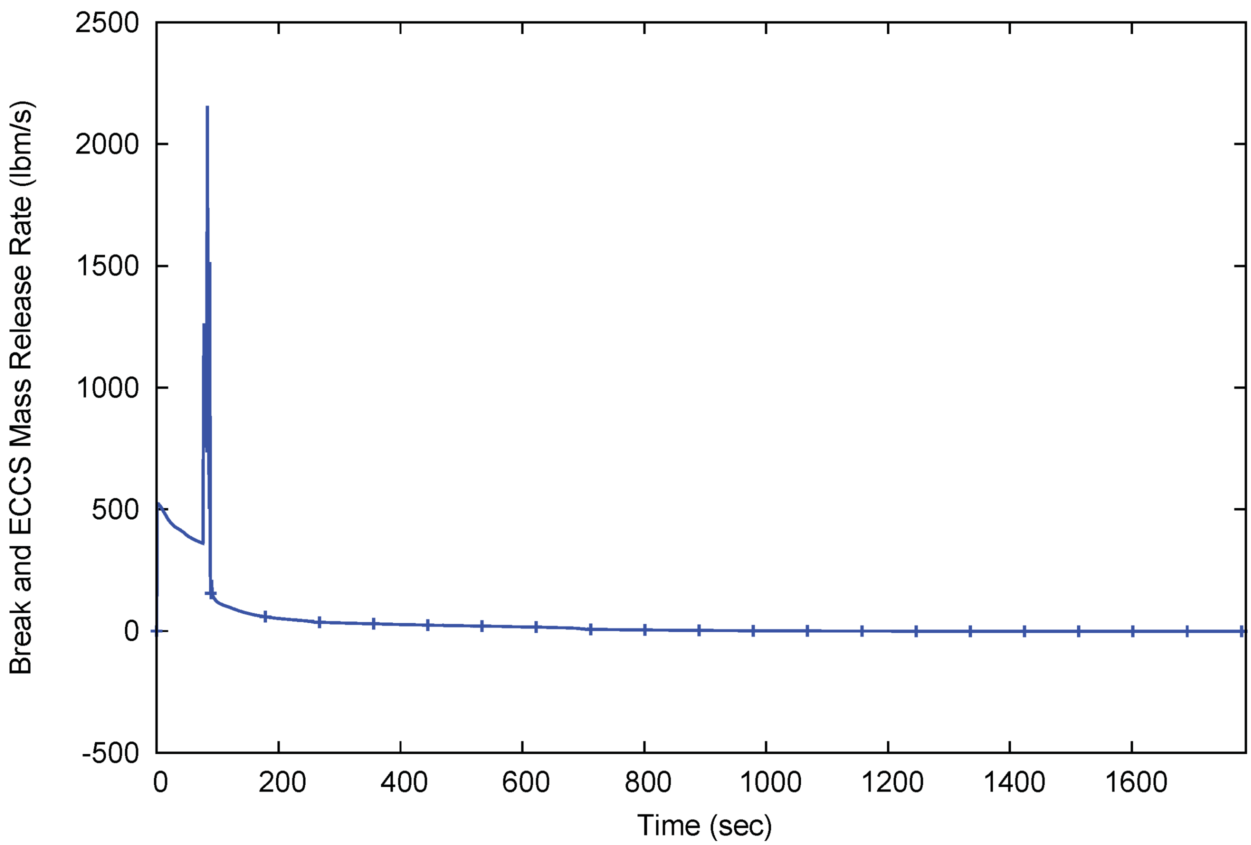

Figure 5-26 Case 5 loss-of-coolant accident and emergency core cooling system flowrate - inadvertent reactor recirculation valve opening event

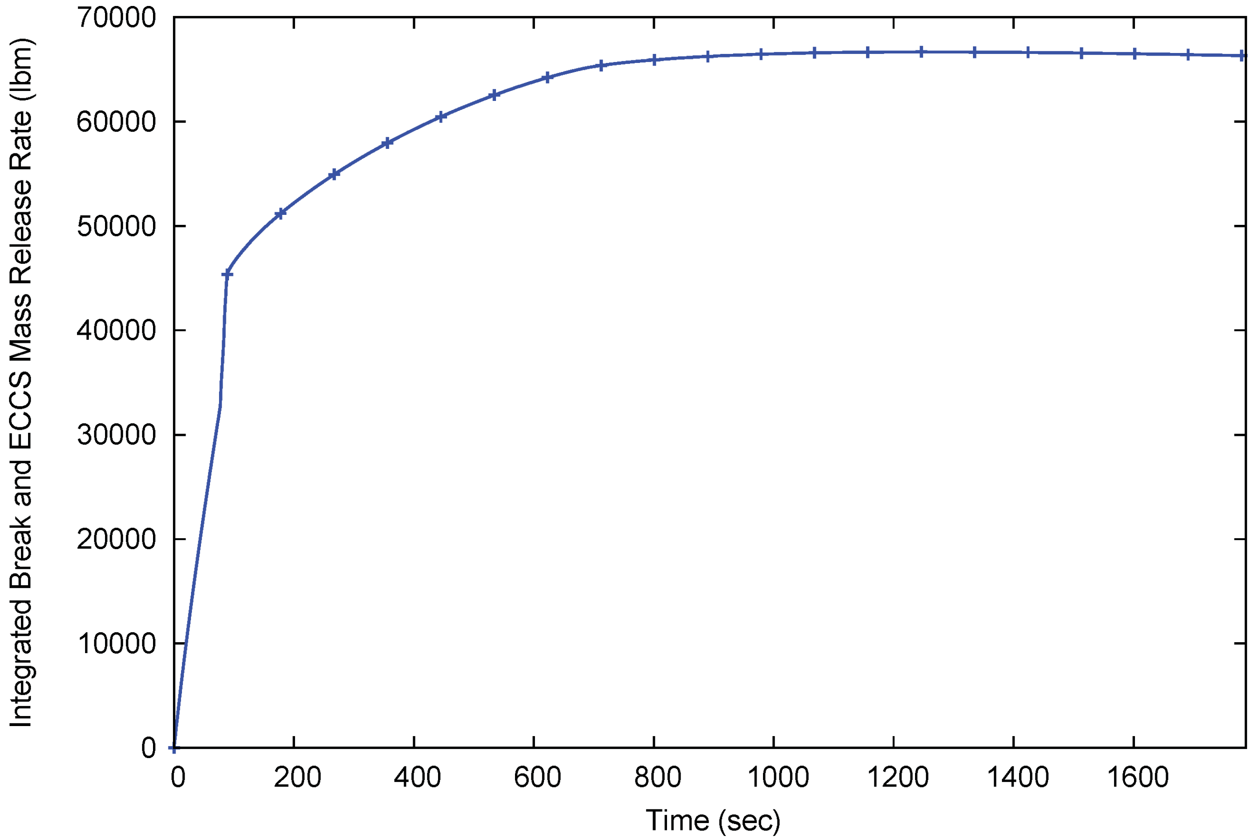

Figure 5-27 Case 5 integrated loss-of-coolant accident and emergency core cooling system mass flow rate - inadvertent reactor recirculation valve opening event

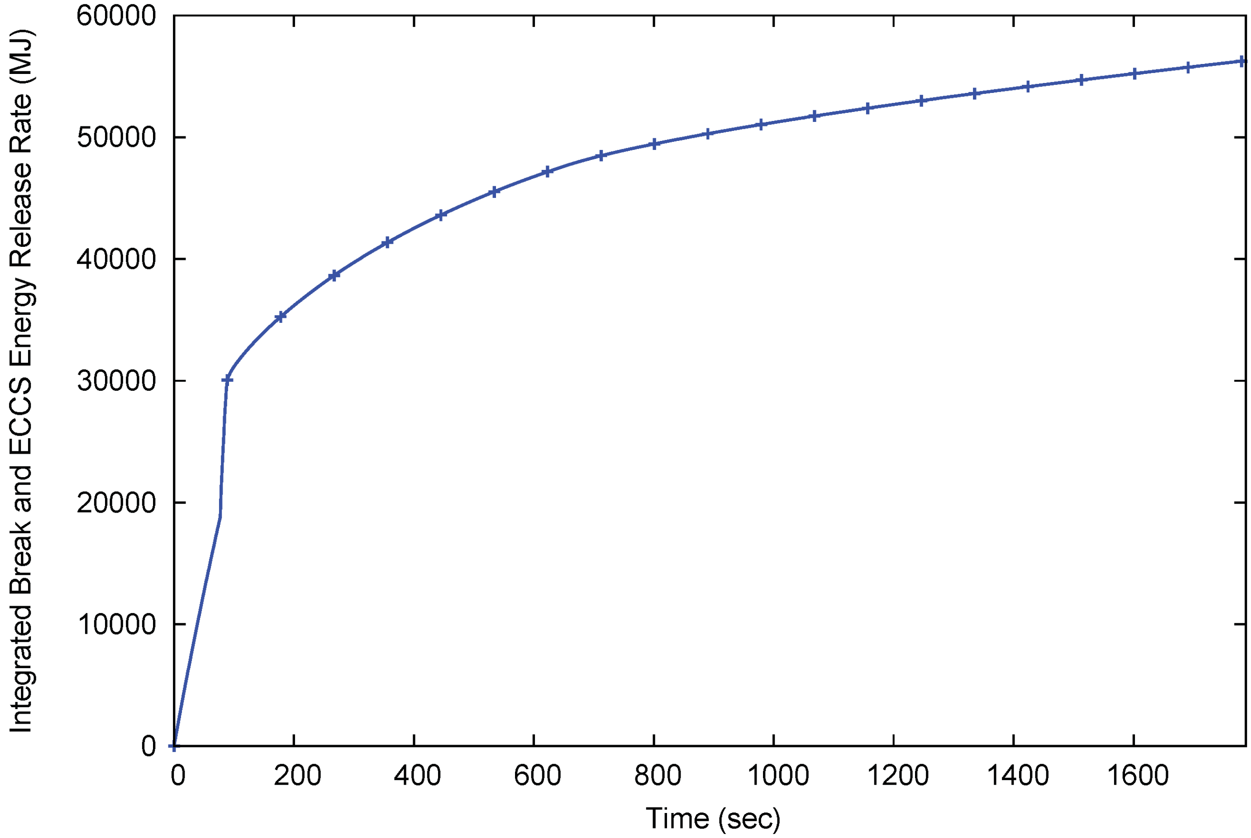

Figure 5-28 Case 5 integrated loss-of-coolant accident and emergency core cooling system energy release - inadvertent reactor recirculation valve opening event

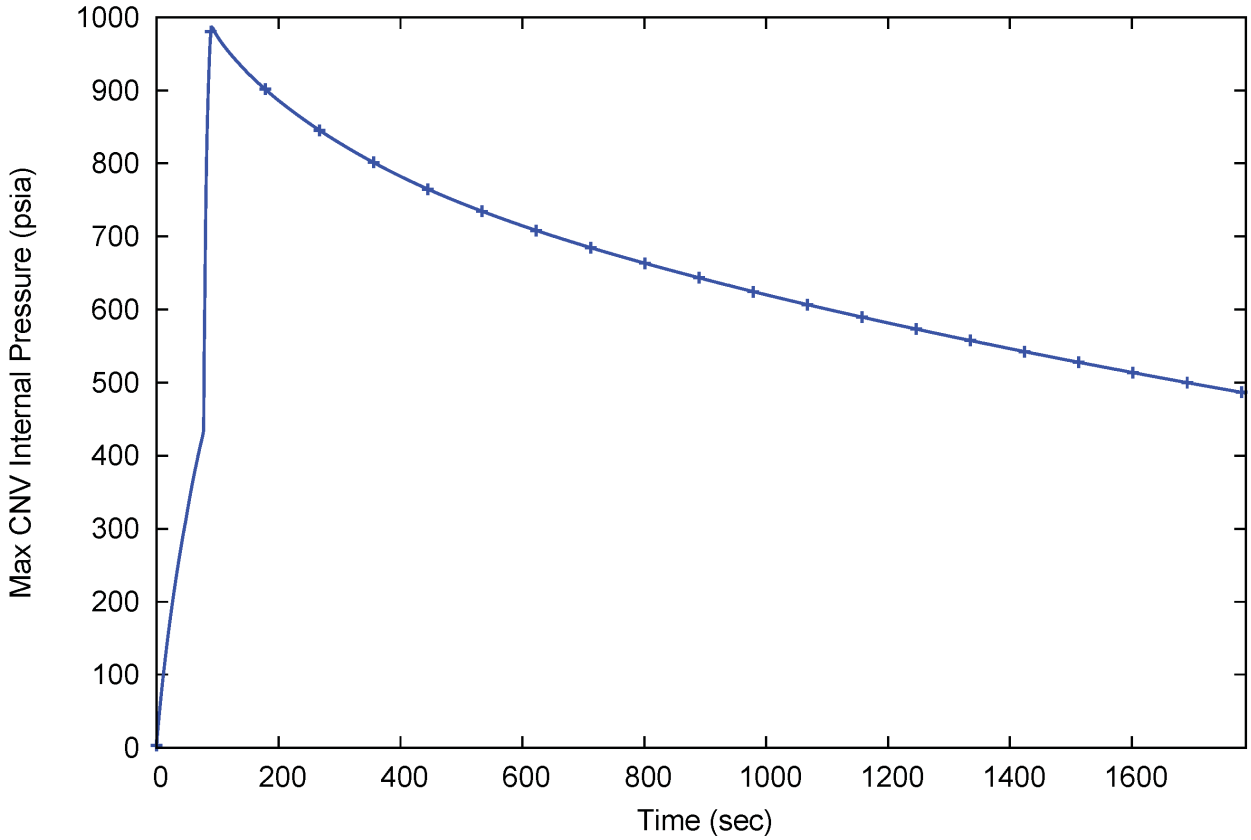

Figure 5-29 Case 5 containment vessel pressure - inadvertent reactor recirculation valve opening event (representative peak pressure)

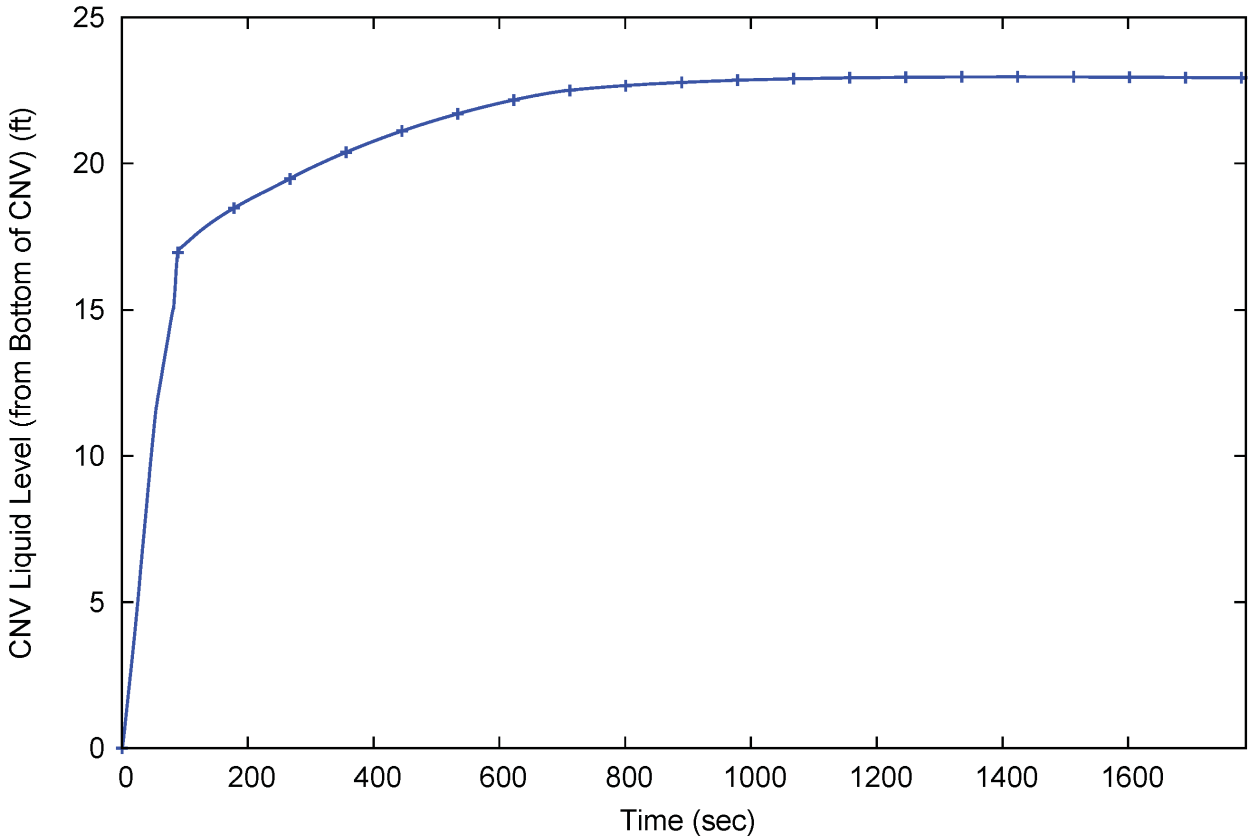

Figure 5-30 Case 5 containment vessel level - inadvertent reactor recirculation valve opening event

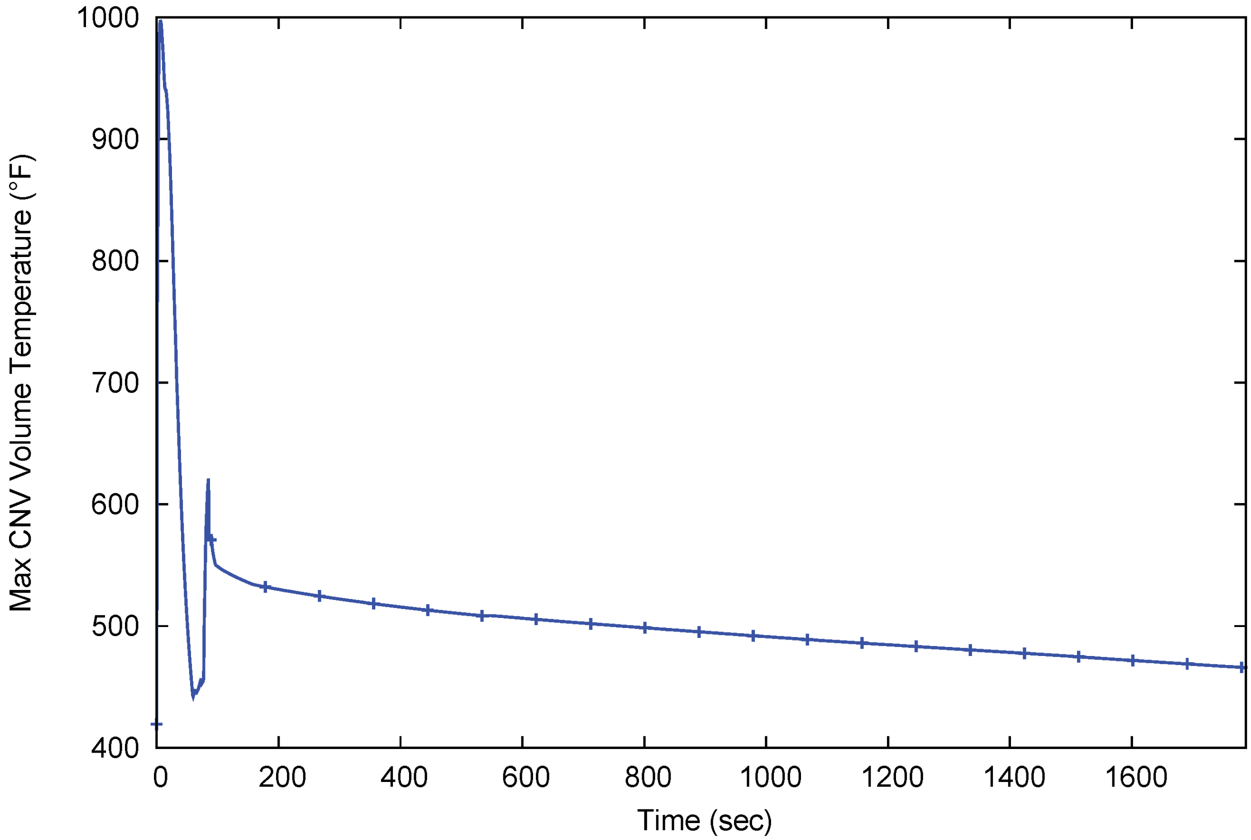

Figure 5-31 Case 5 containment vessel vapor temperature - inadvertent reactor recirculation valve opening event

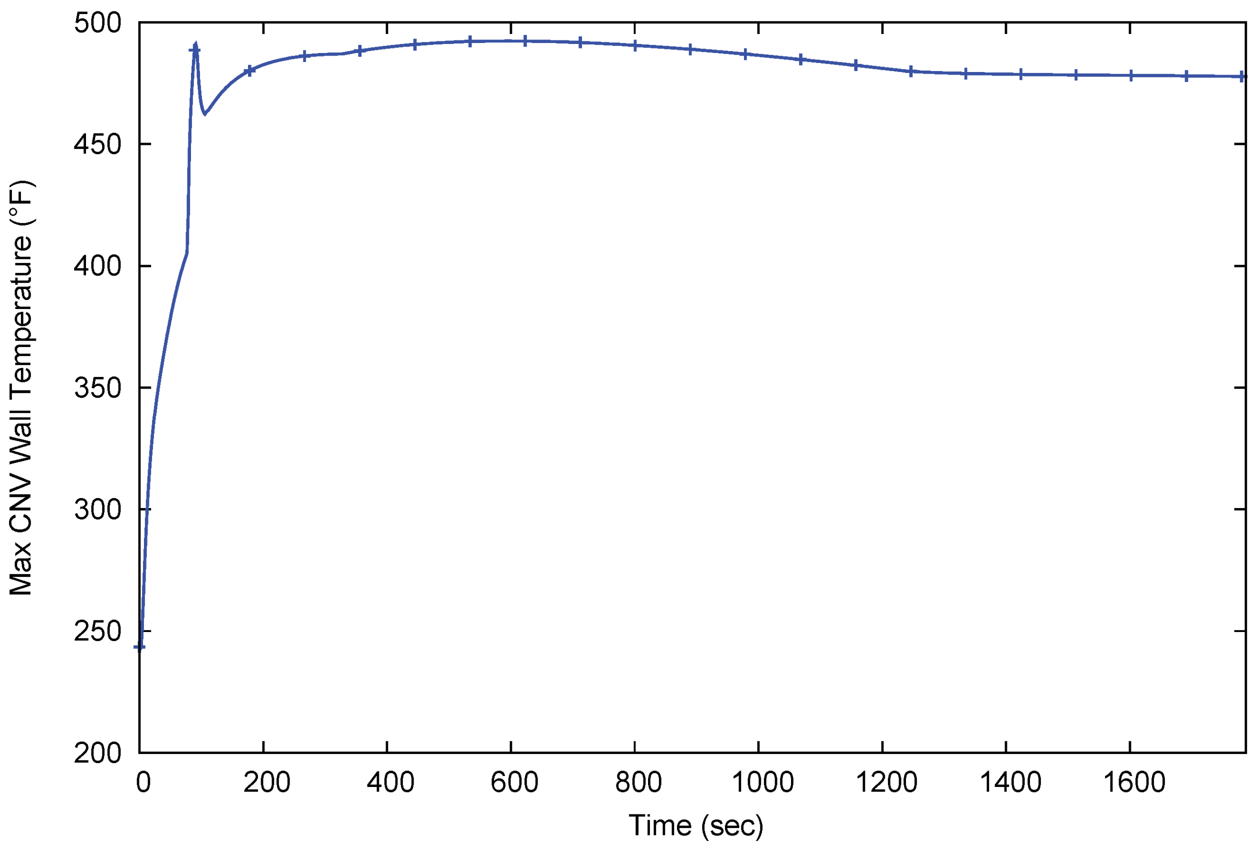

Figure 5-32 Case 5 containment vessel wall temperature - inadvertent reactor recirculation valve opening event

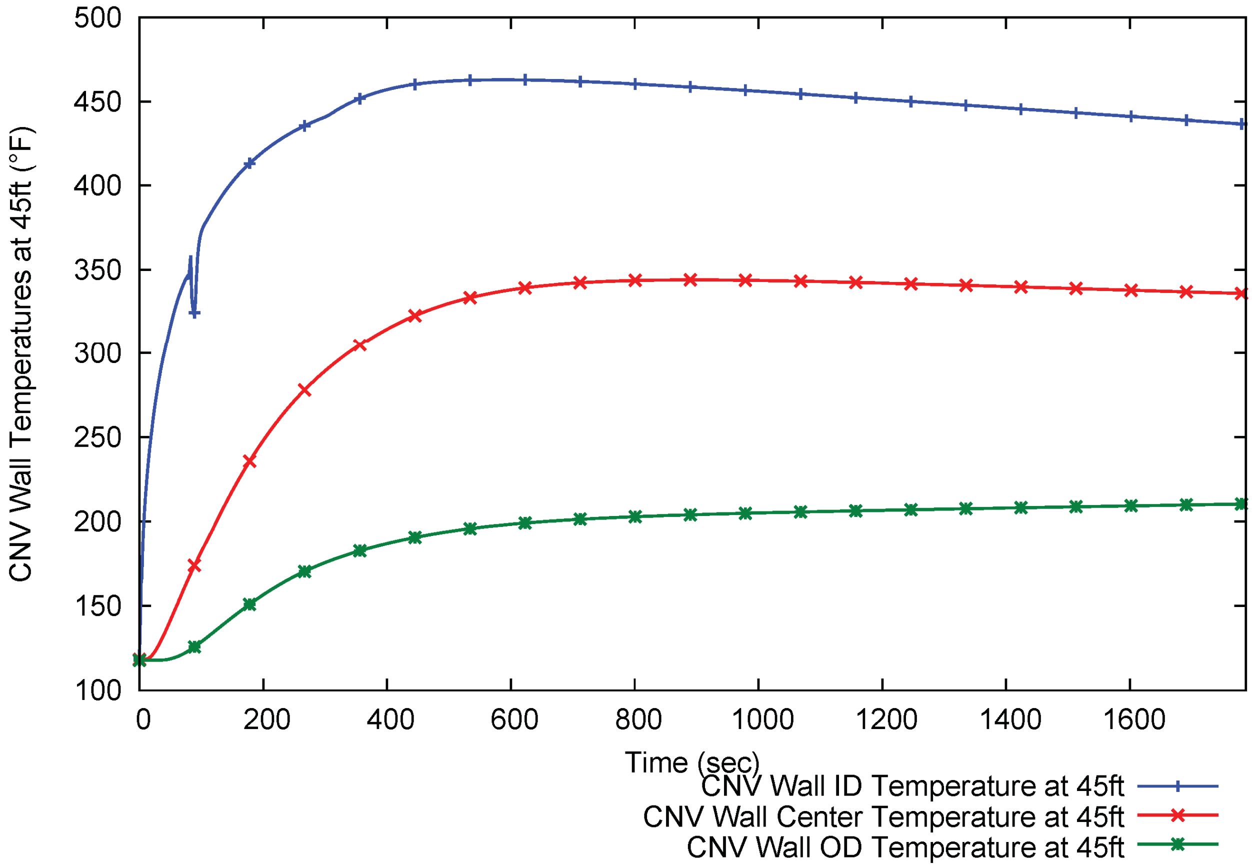

Figure 5-33 Case 5 containment vessel wall temperature profile - inadvertent reactor recirculation valve opening event

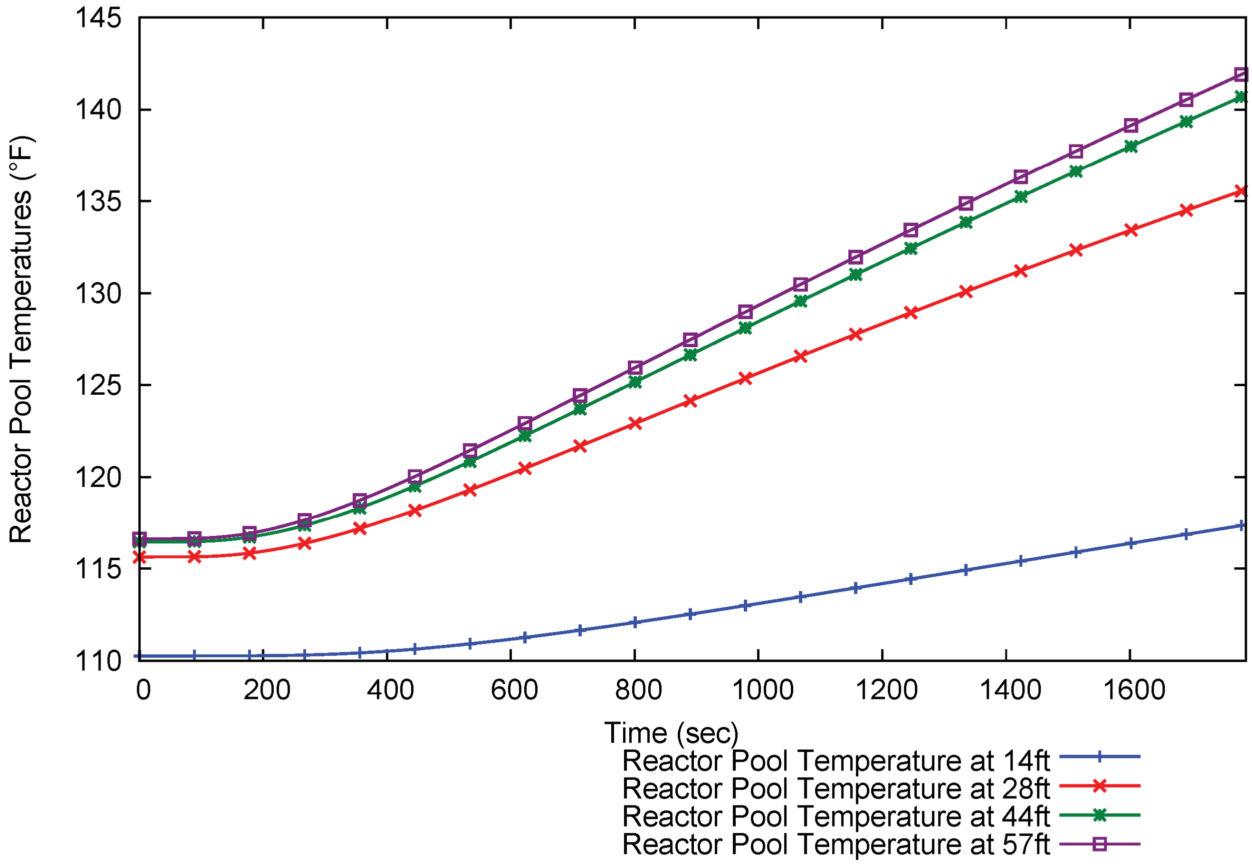

Figure 5-34 Case 5 reactor pool temperature - inadvertent reactor recirculation valve opening event

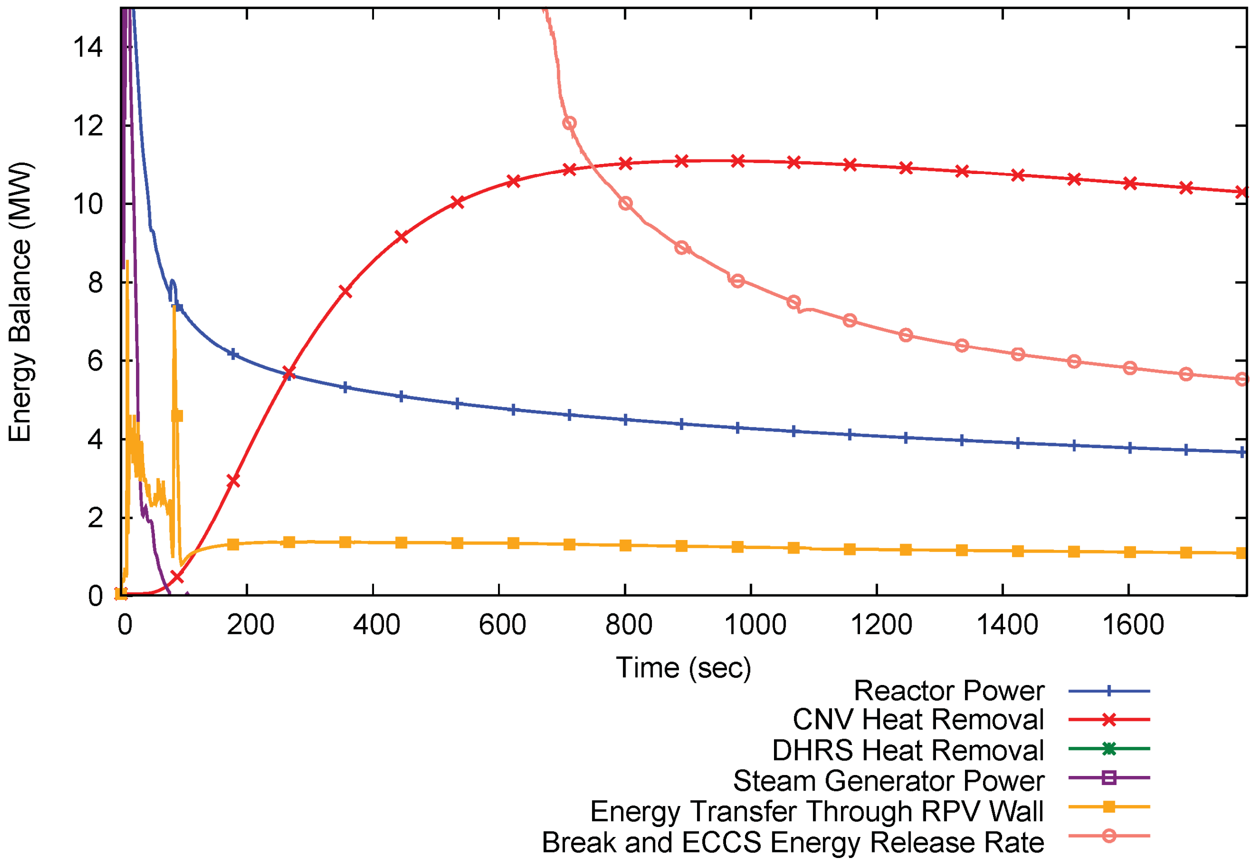

Figure 5-35 Case 5 energy balance - inadvertent reactor recirculation valve opening event

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 causes a rapid SG depressurization that reaches the low steam line pressure setpoint. The following automatic actions occur on low steam line pressure:

- MSIV and FWIV closure
- reactor trip
- turbine trip

Immediately following the low steam line pressure signal, the high CNV pressure signal is reached, resulting in containment isolation. 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 isolation. Figure 5-37 shows the primary system temperature response due to the initial secondary system blowdown and then following secondary side isolation. Figure 5-38 shows the primary system pressure response with the initial depressurization following secondary system blowdown, and then the pressure increasing following secondary side isolation. Figure 5-39 shows that the pressurizer level rapidly decreases during the initial overcooling, and then gradually increases in response to the increase in primary temperatures following secondary side isolation. 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 449 psia at 42 seconds. This limiting NRELAP5 result can be compared to the CNV design pressure of 1050 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 428 degrees F at 41 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 65 ft above the pool floor and a temperature of 110 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 300 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 – 4	Low steam line pressure resulting in • Reactor trip • Turbine trip • MSIV closure • FWIV closure High CNV pressure resulting in • Containment isolation
34	Closure of FWRV complete
42	Peak CNV pressure
41	Peak CNV temperature
~200	CNV pressure decreases to <50% of peak pressure

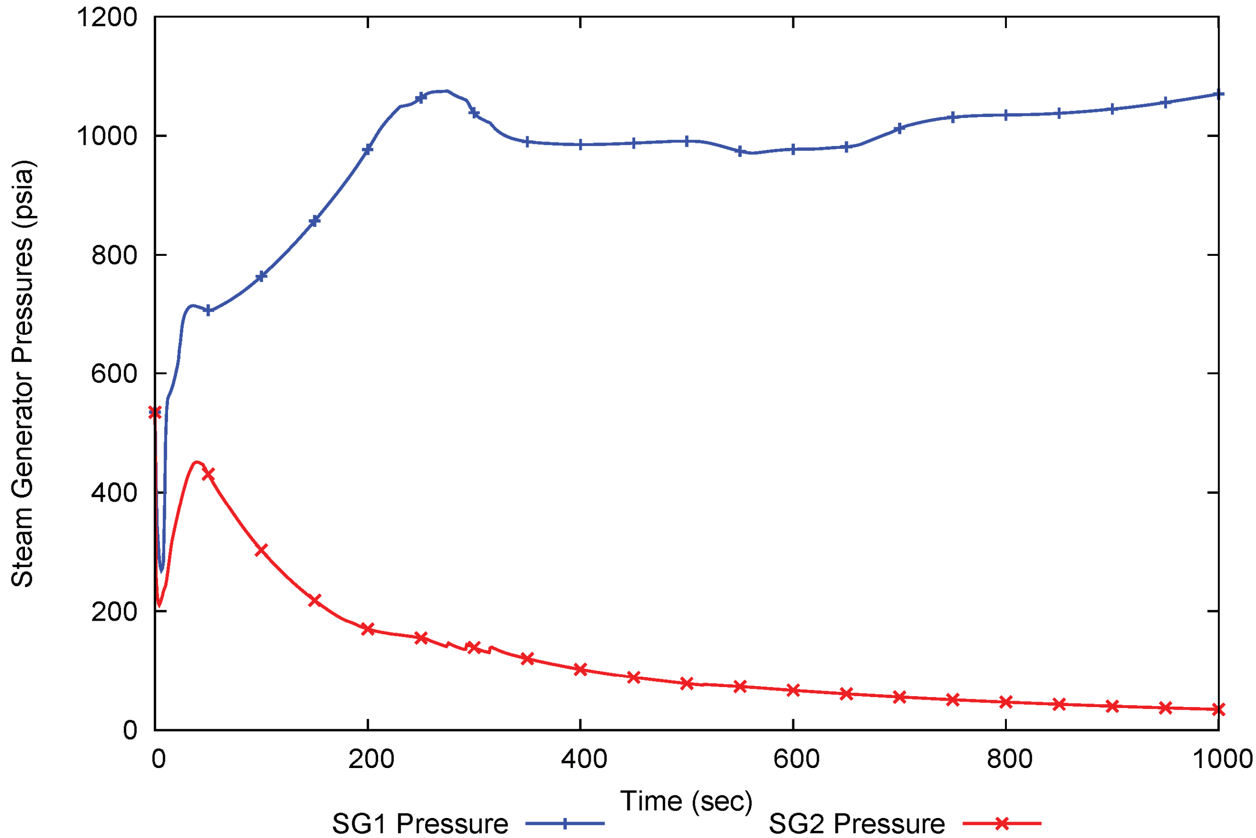

Figure 5-36 Main steam line break steam generator pressure

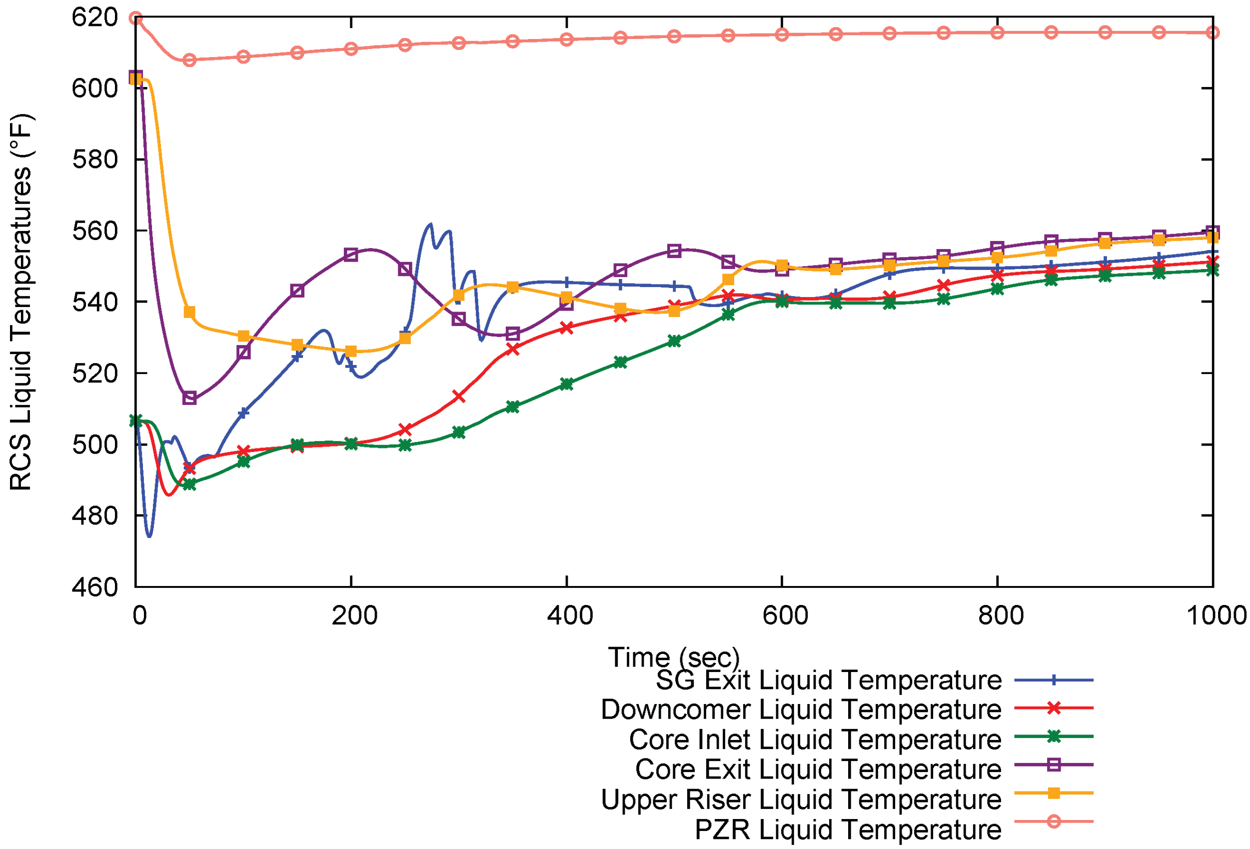

Figure 5-37 Main steam line break primary temperature

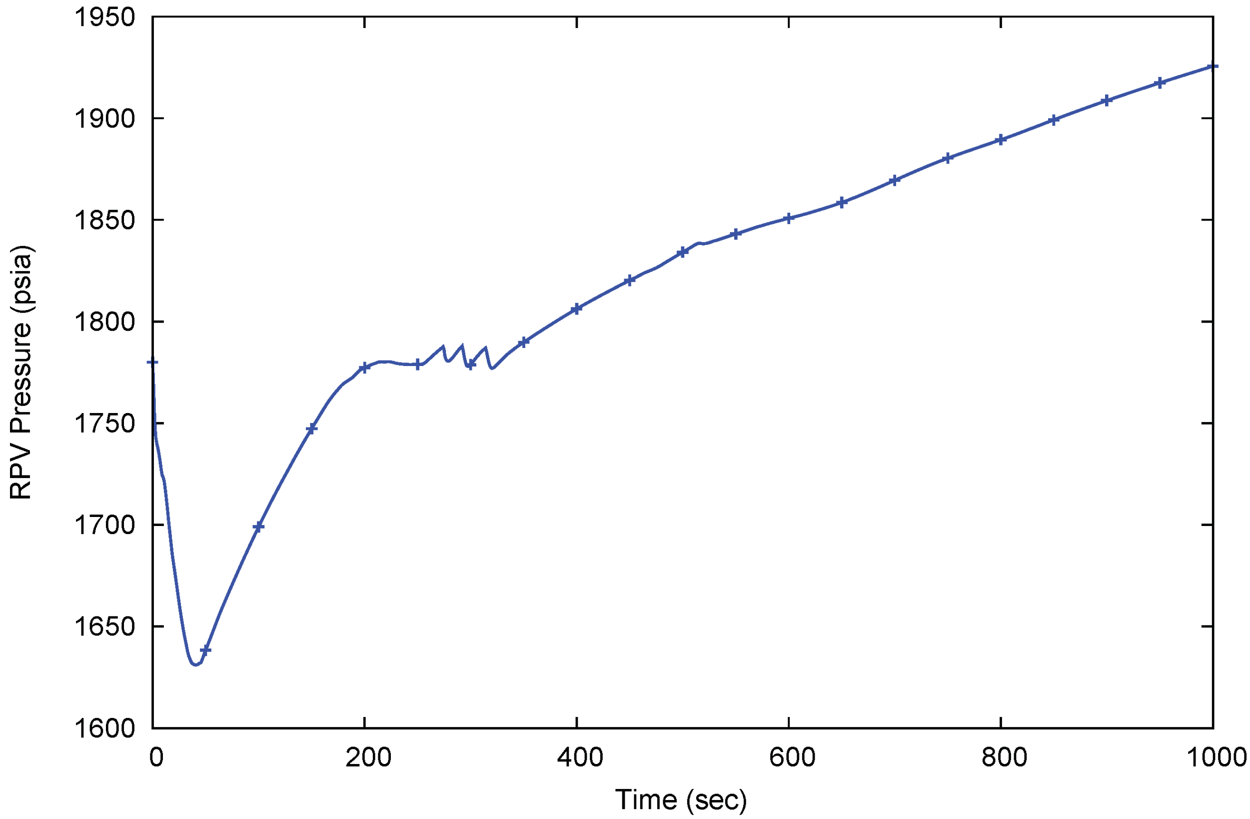

Figure 5-38 Main steam line break primary system pressure

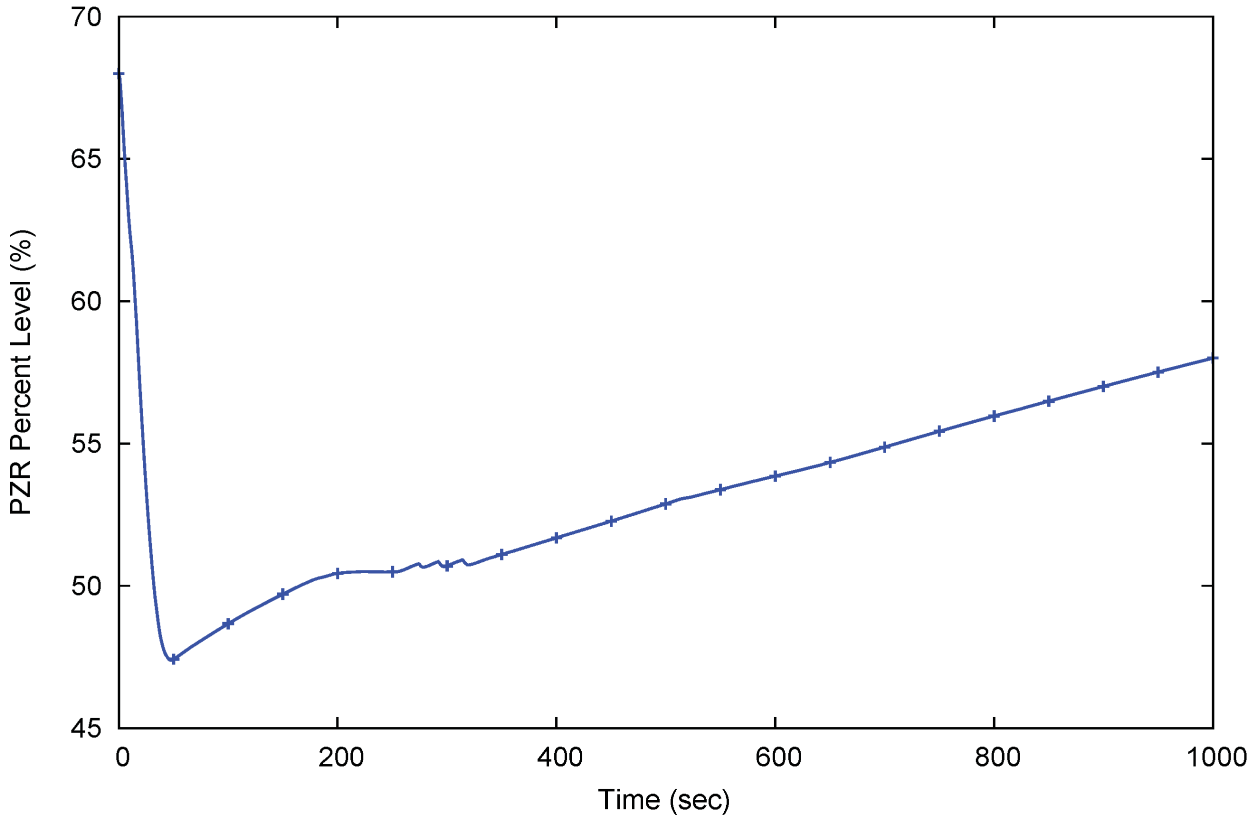

Figure 5-39 Main steam line break pressurizer level

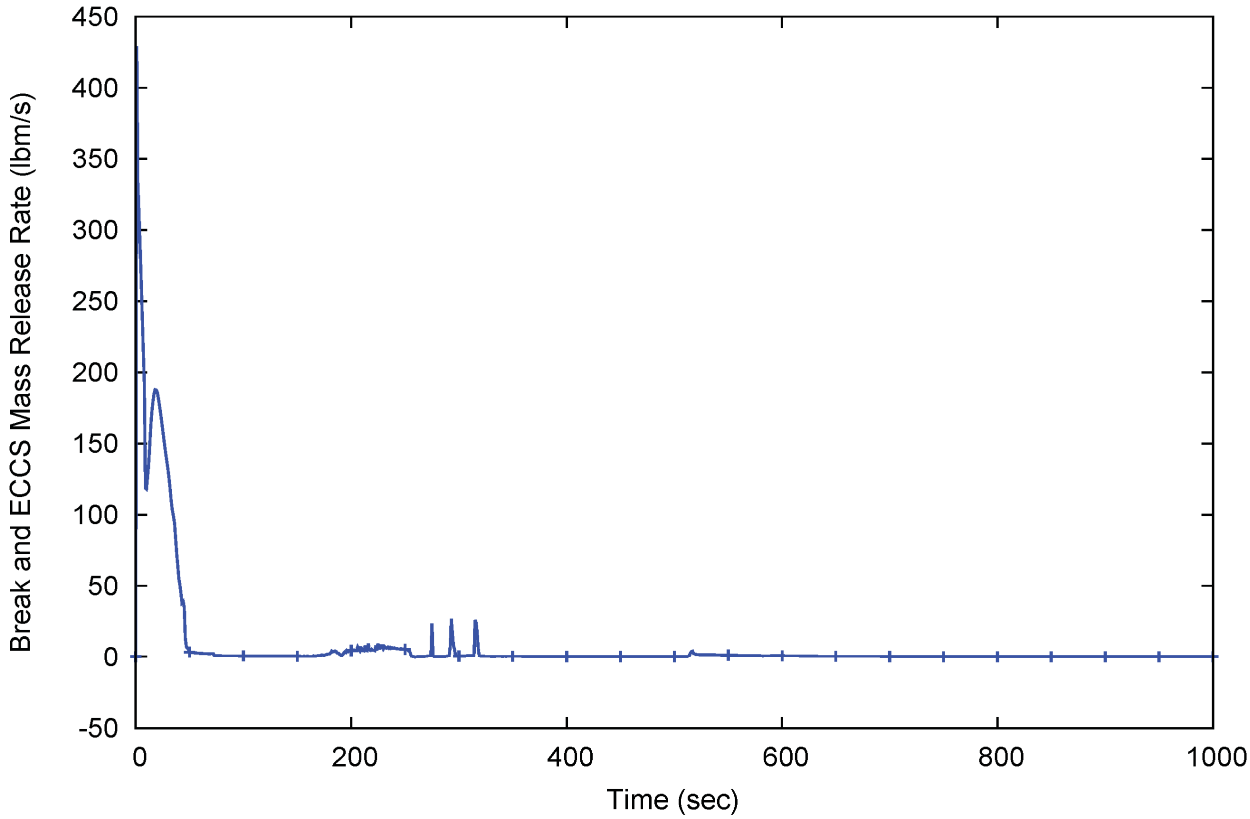

Figure 5-40 Main steam line break and emergency core cooling system flowrate

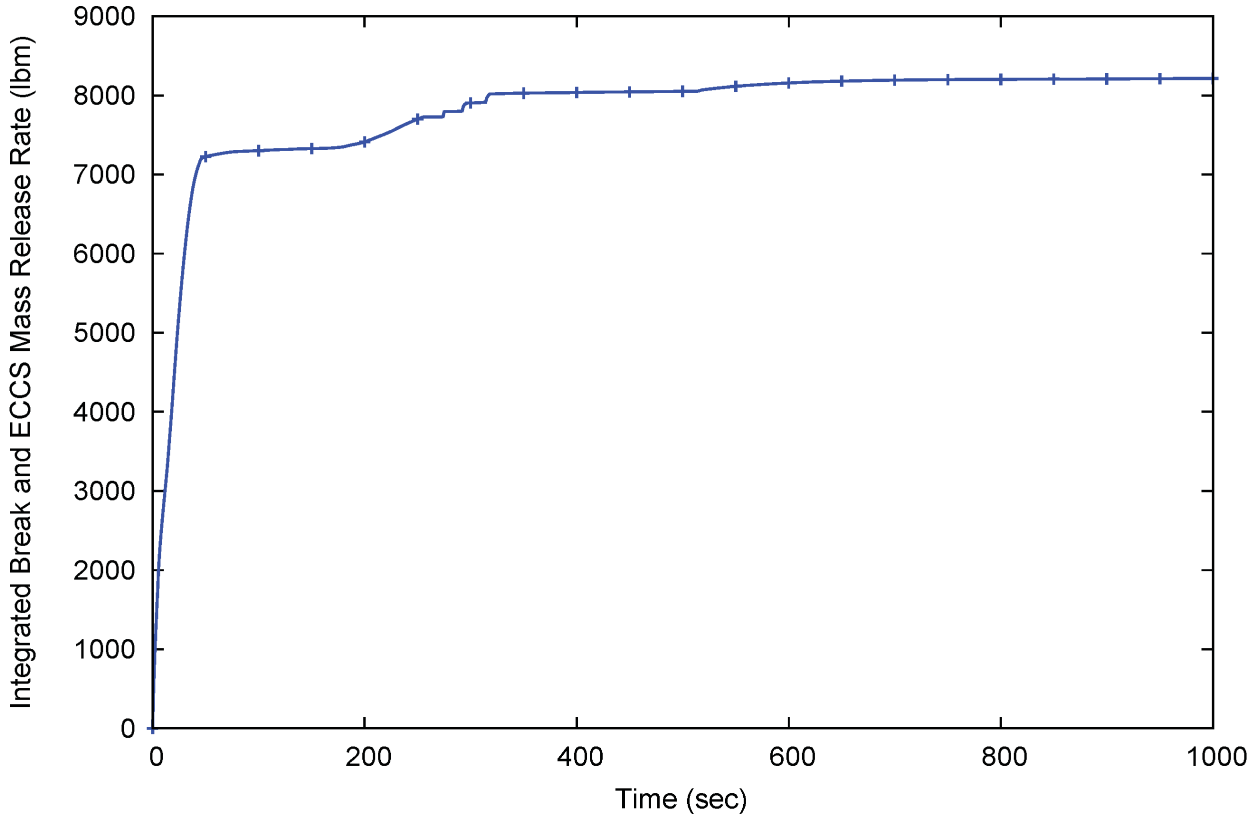

Figure 5-41 Main steam line break and emergency core cooling system integrated mass release

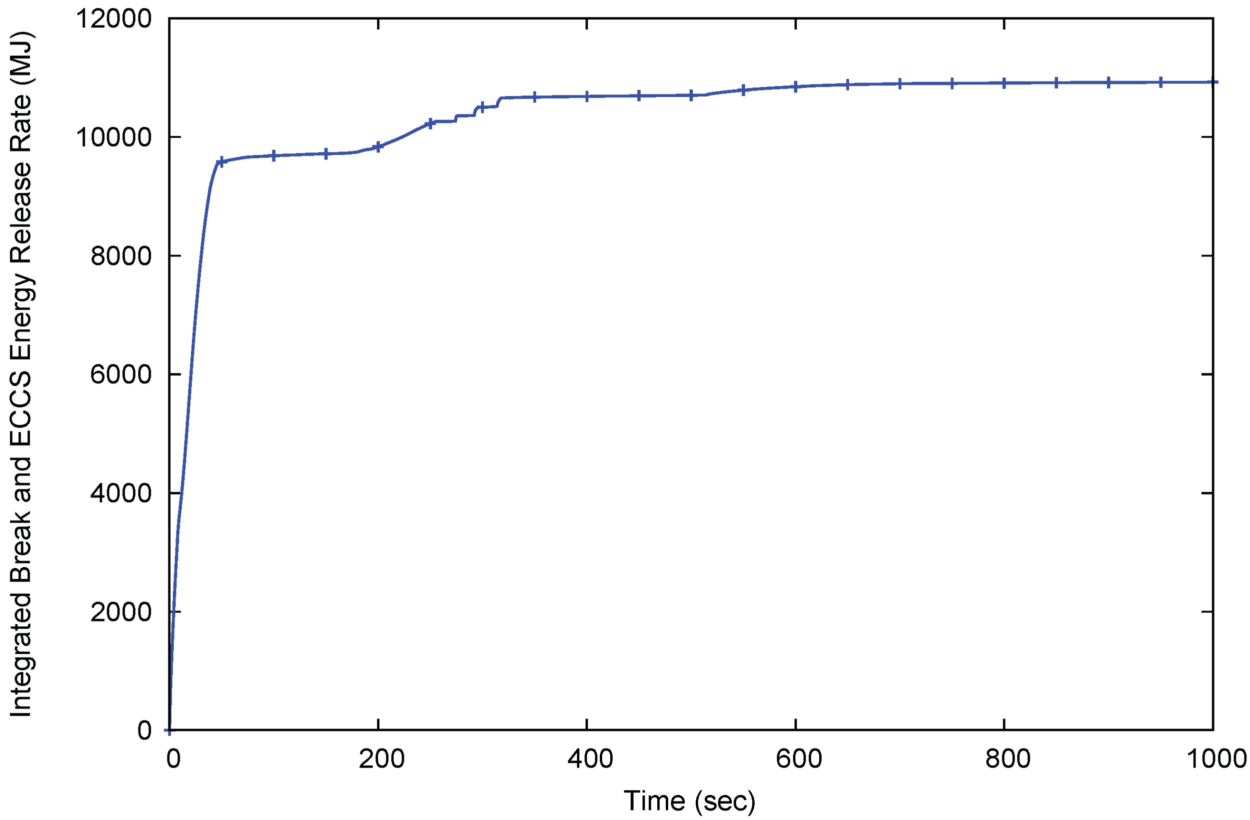

Figure 5-42 Main steam line break integrated energy release

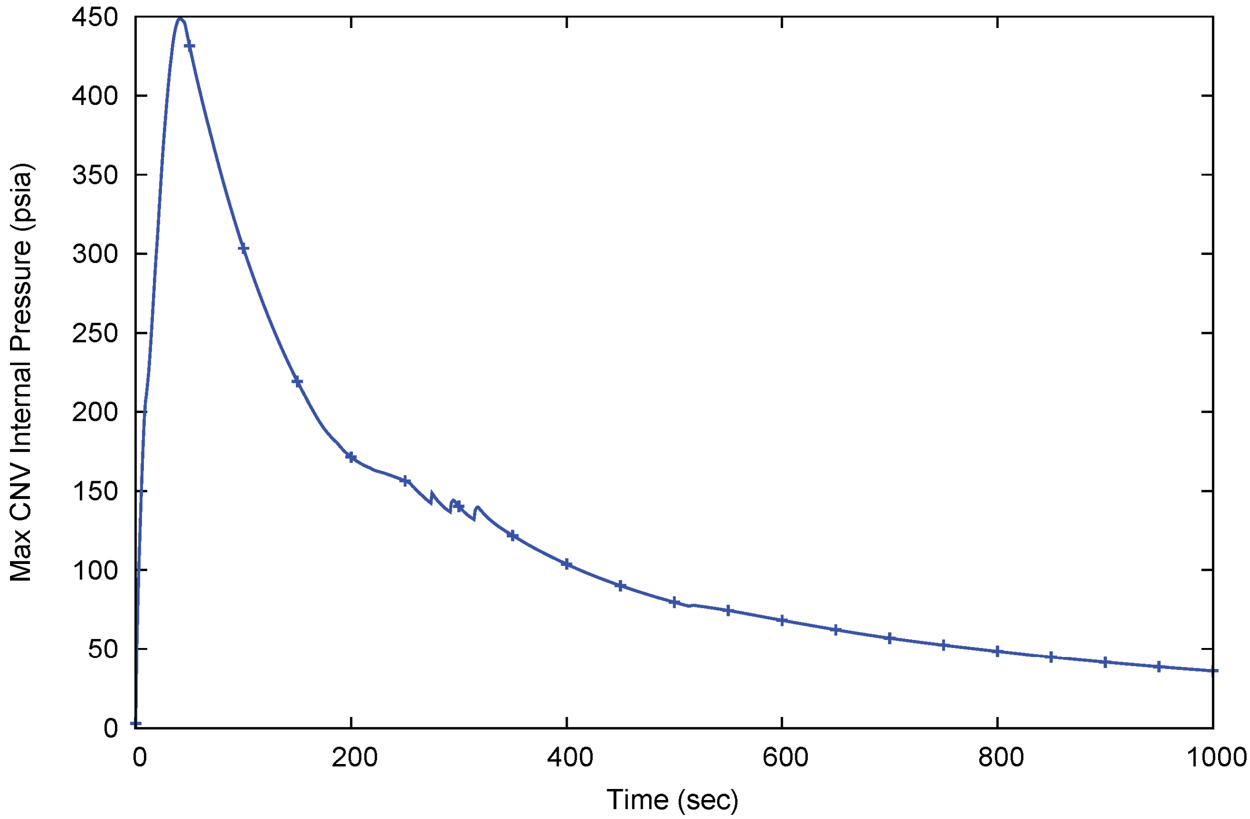

Figure 5-43 Main steam line break containment vessel pressure

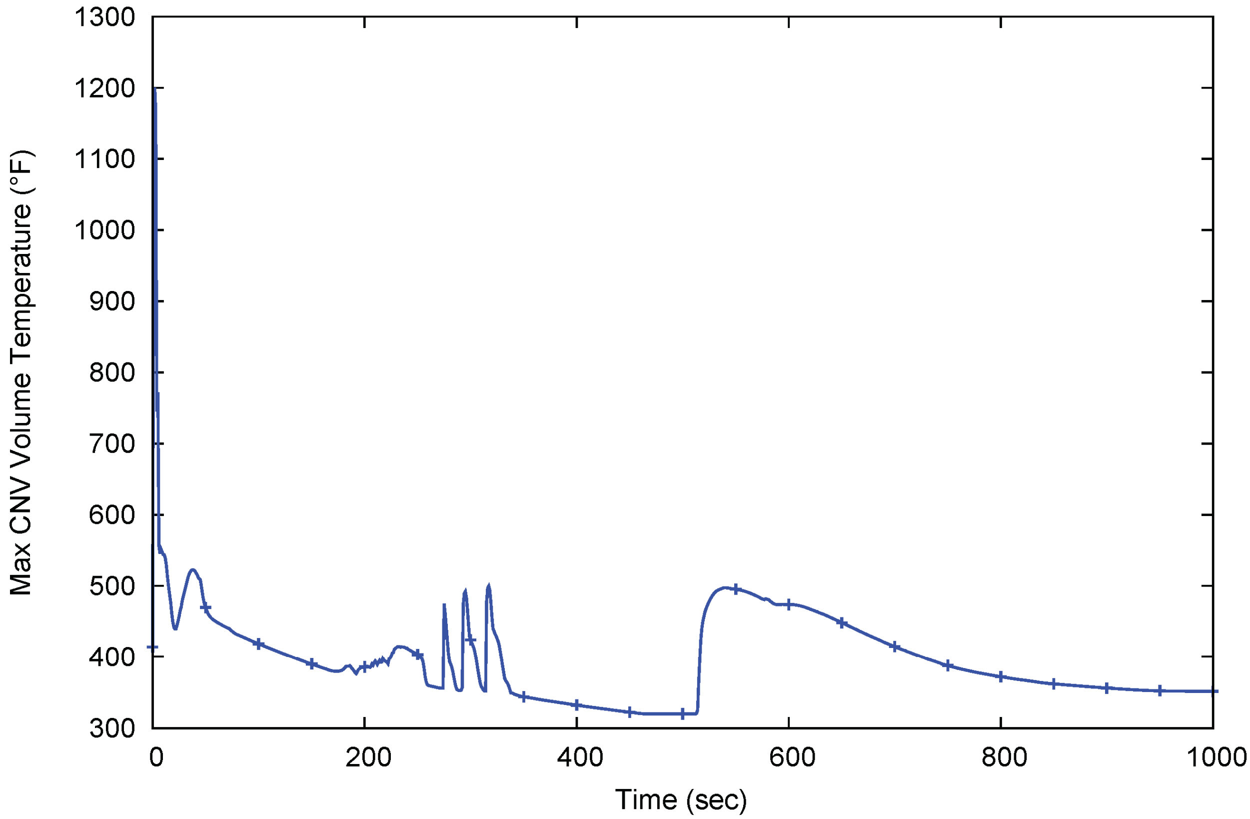

Figure 5-44 Main steam line break containment vessel vapor temperature

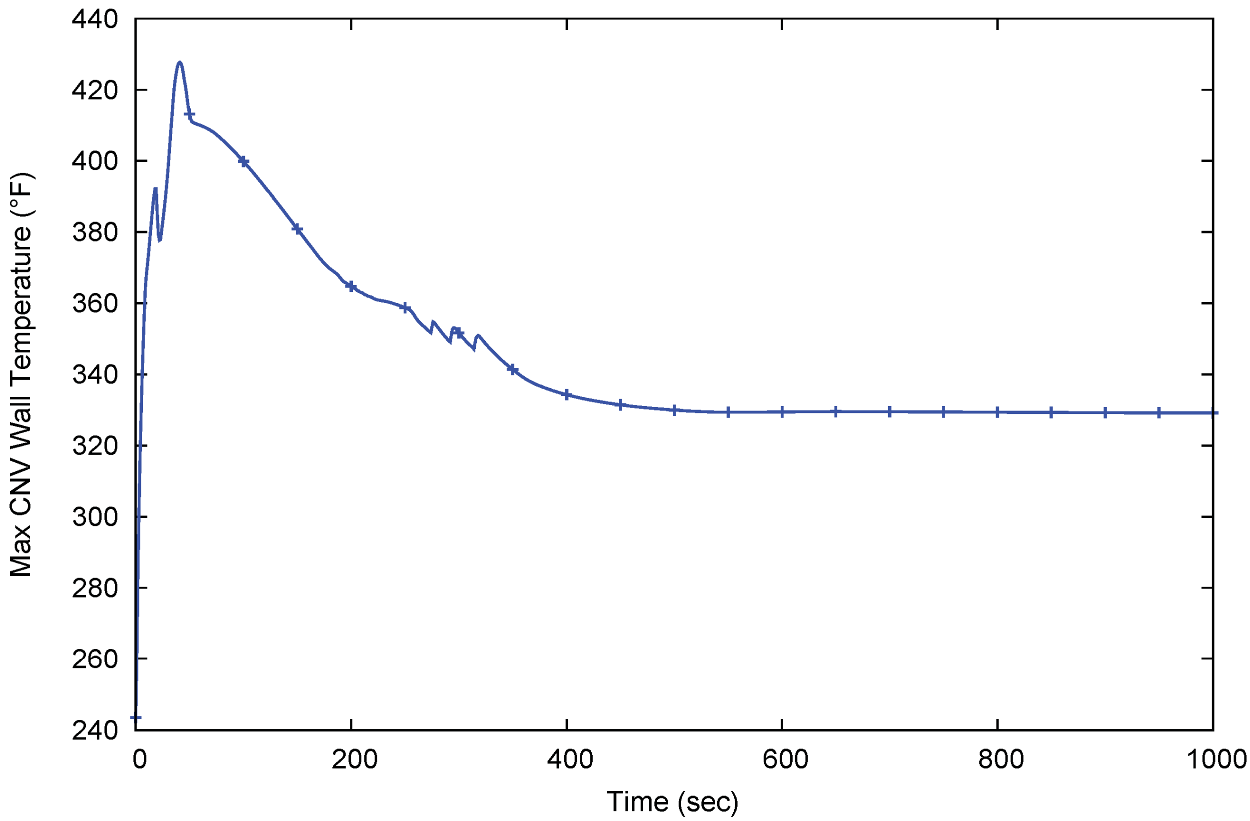

Figure 5-45 Main steam line break containment vessel wall temperature

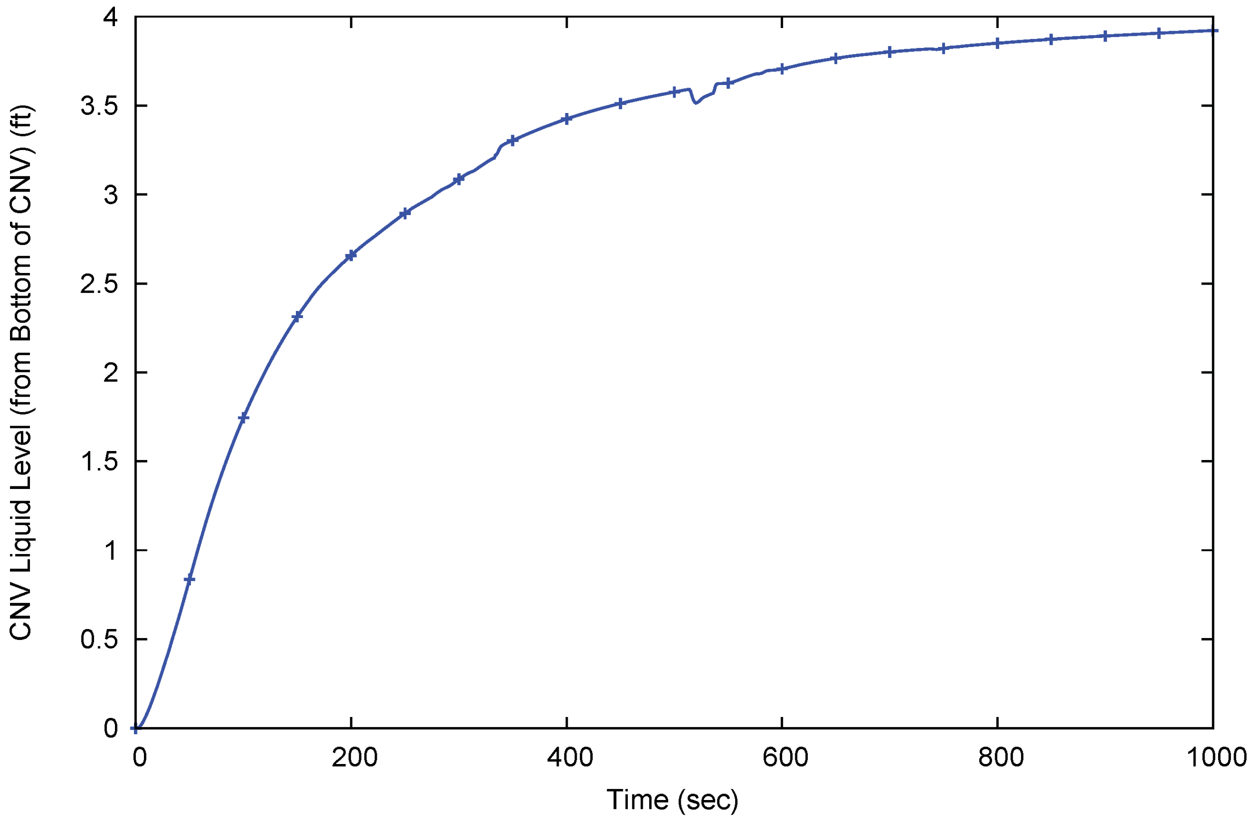

Figure 5-46 Main steam line break containment vessel level

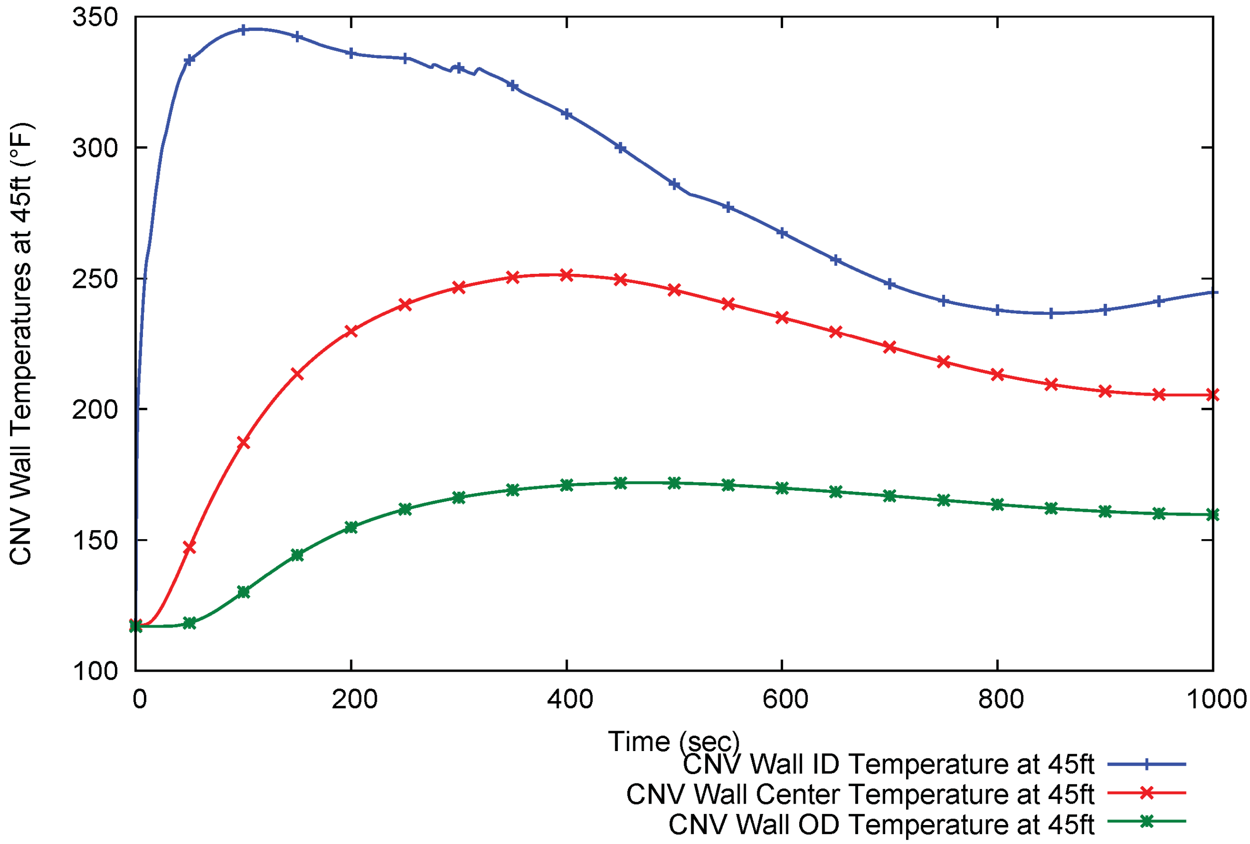

Figure 5-47 Main steam line break containment vessel wall temperature profile

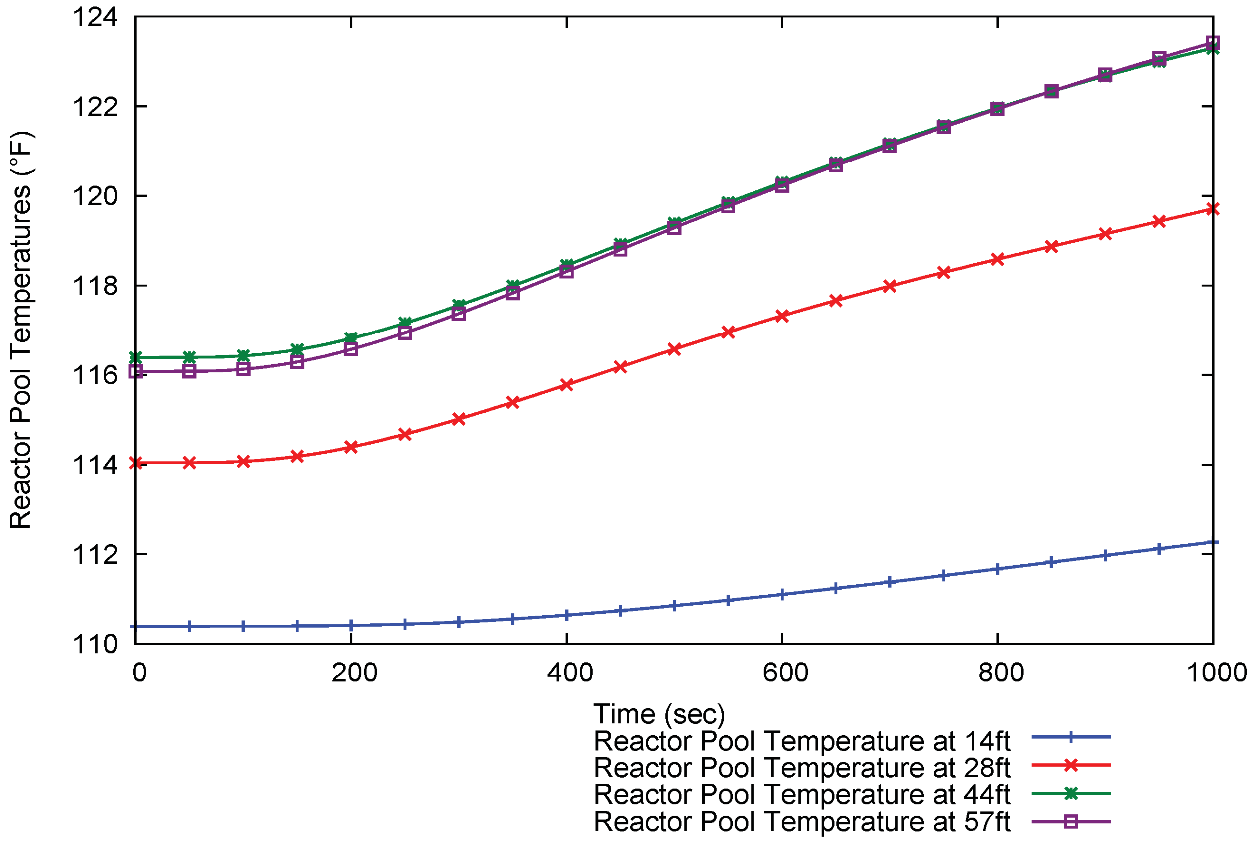

Figure 5-48 Main steam line break reactor pool temperature

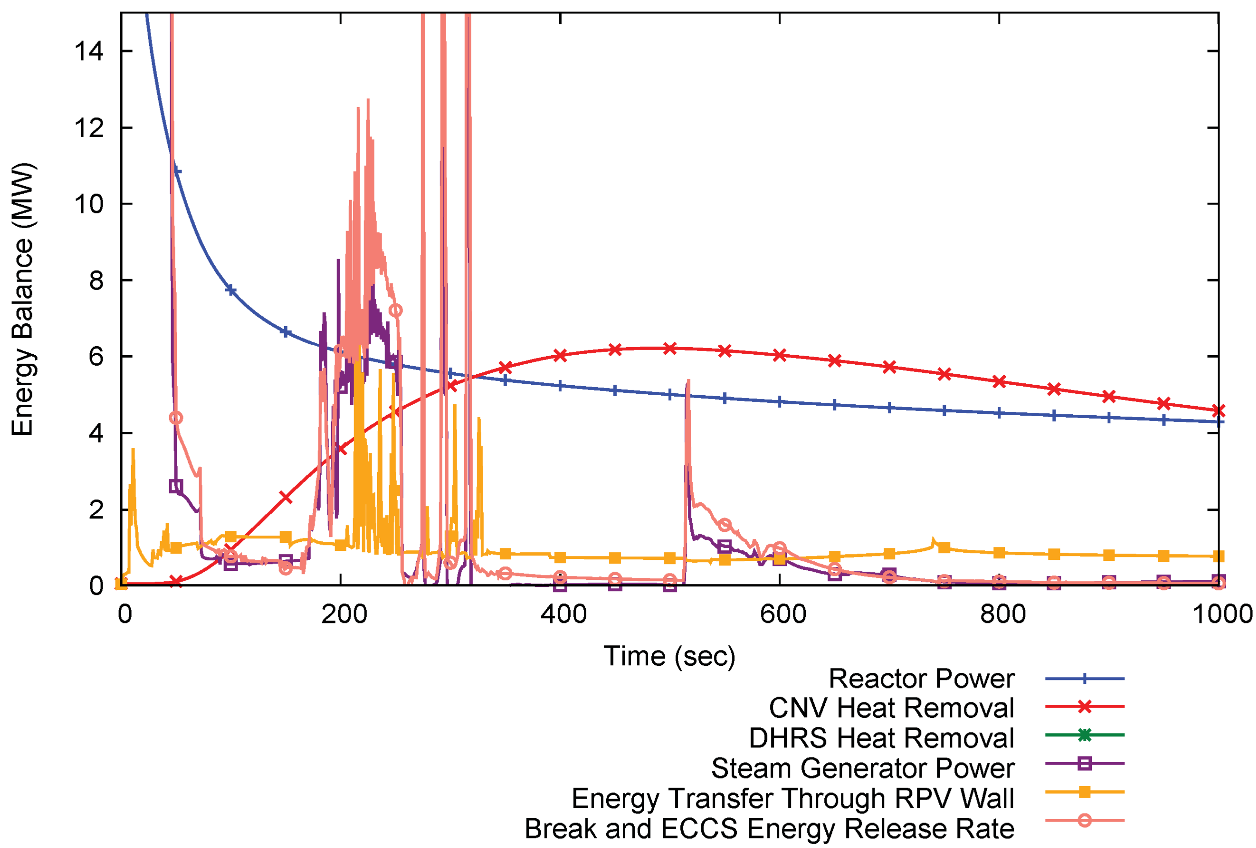

Figure 5-49 Main steam line break energy balance

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. The following automatic actions occur following an assumed loss of normal AC and DC power at the time of event initiation in the limiting case:

- 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 assumed to occur at the time of event initiation. 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 results of this case are adequately representative of the feedwater line break case although they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The IAB release pressure range of 950 psi +/- 50 psi was evaluated and determined to be non-limiting. 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 MSIV 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 11,566 seconds when the pressure differential decreases to below the 1200 psid IAB release pressure. This causes the CNV peak pressure (416 psia) and the peak CNV wall temperature (407 degrees F) at ~11,600 and ~11,870 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 low pressure . 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 416 psia following opening of the RVVs. This limiting NRELAP5 result can be compared to the CNV design pressure of 1050 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 407 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 65 ft above the pool floor and a temperature of 110 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 • Loss of normal AC and DC power at event initiation resulting in ECCS actuation signal • Reactor trip • Turbine trip
0 – 2	High CNV pressure followed by: • Containment isolation • MSIV closure • FWIV closure • DHRS actuation
~12	Peak CNV pressure from secondary M&E release
11,566	ECCS valve opening on differential pressure below adverse IAB release pressure
~11,600	Peak CNV pressure
~11,870	Peak CNV temperature
~13,000	CNV pressure decreases to <50% of peak pressure

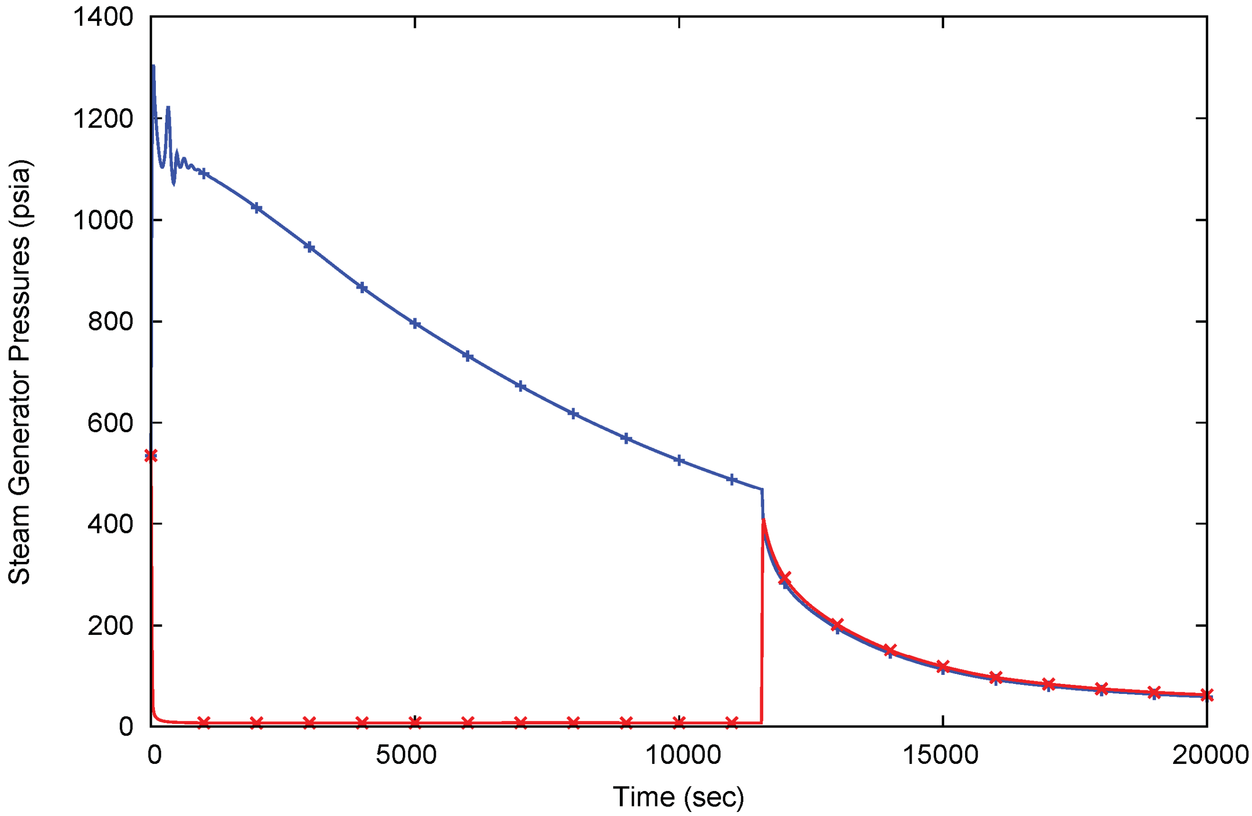

Figure 5-50 Feedwater line break steam generator pressure

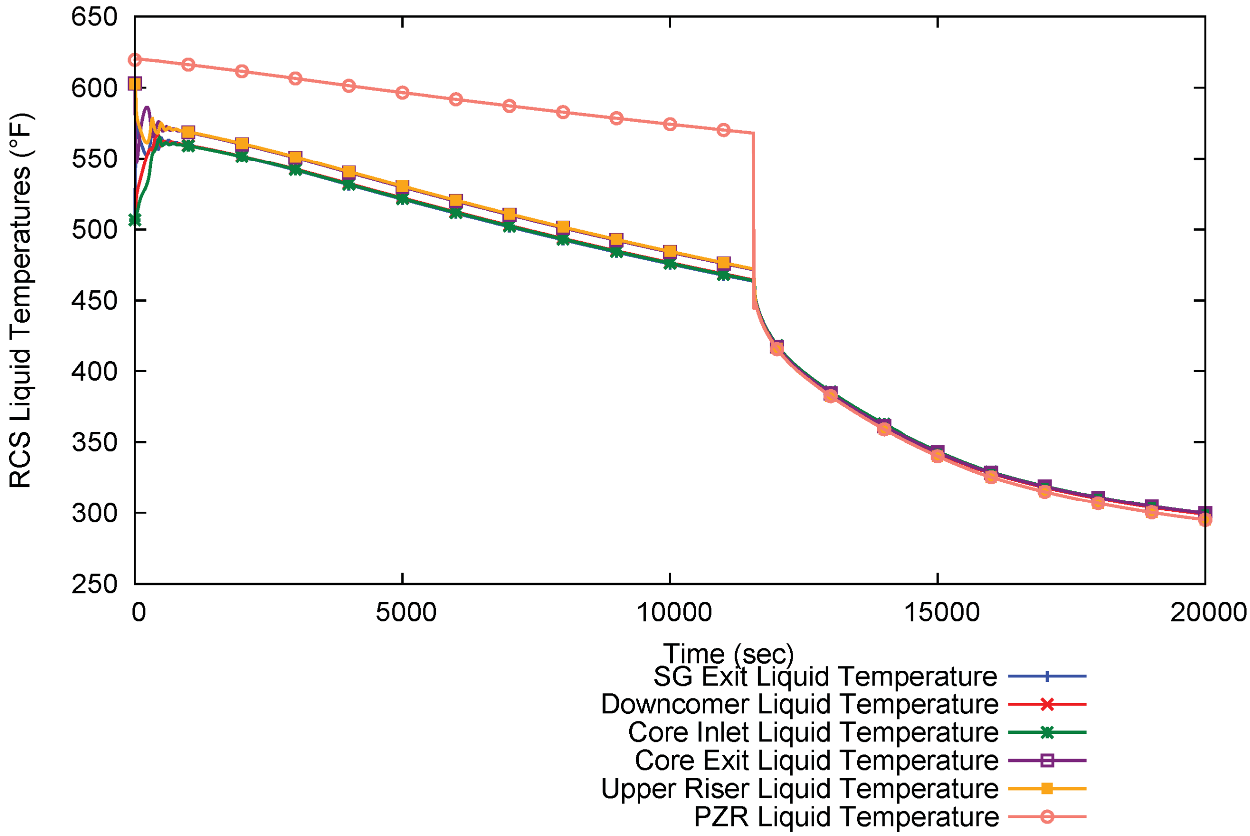

Figure 5-51 Feedwater line break primary temperature

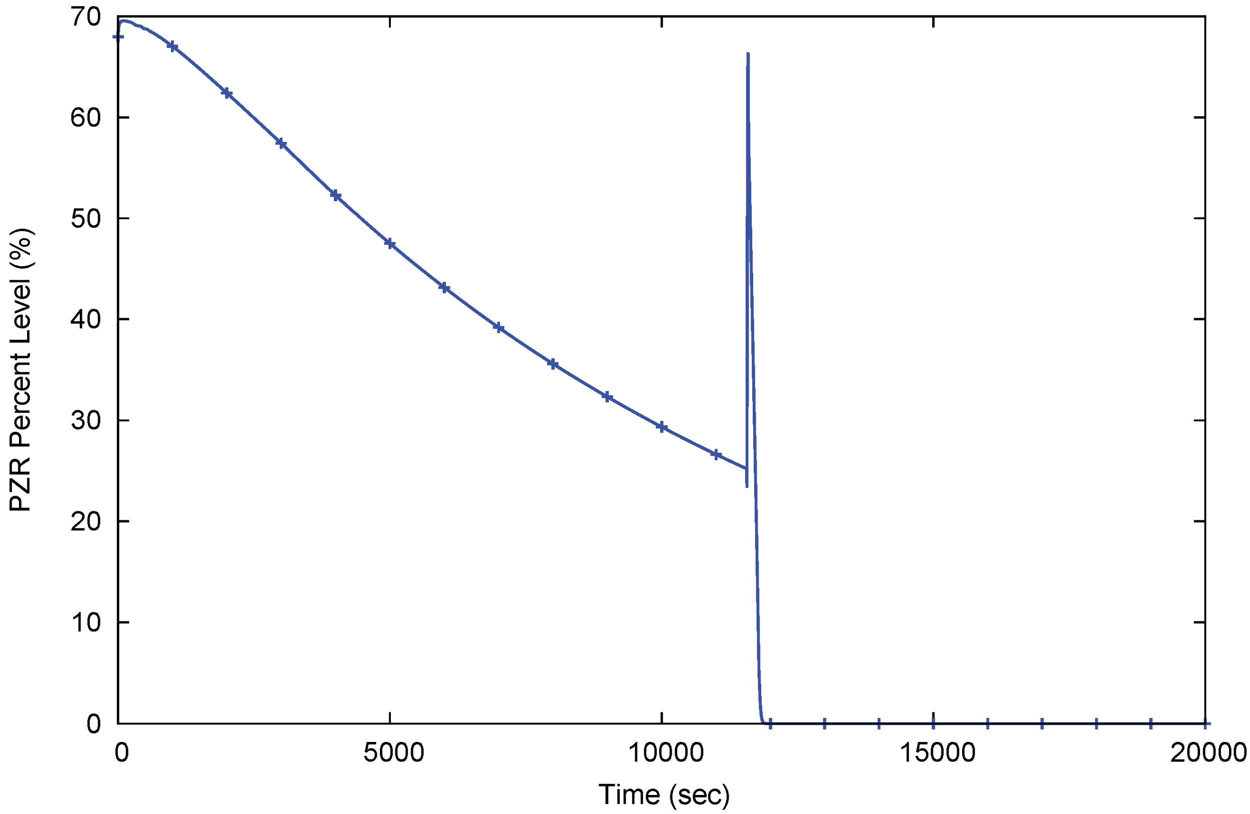

Figure 5-52 Feedwater line break pressurizer level

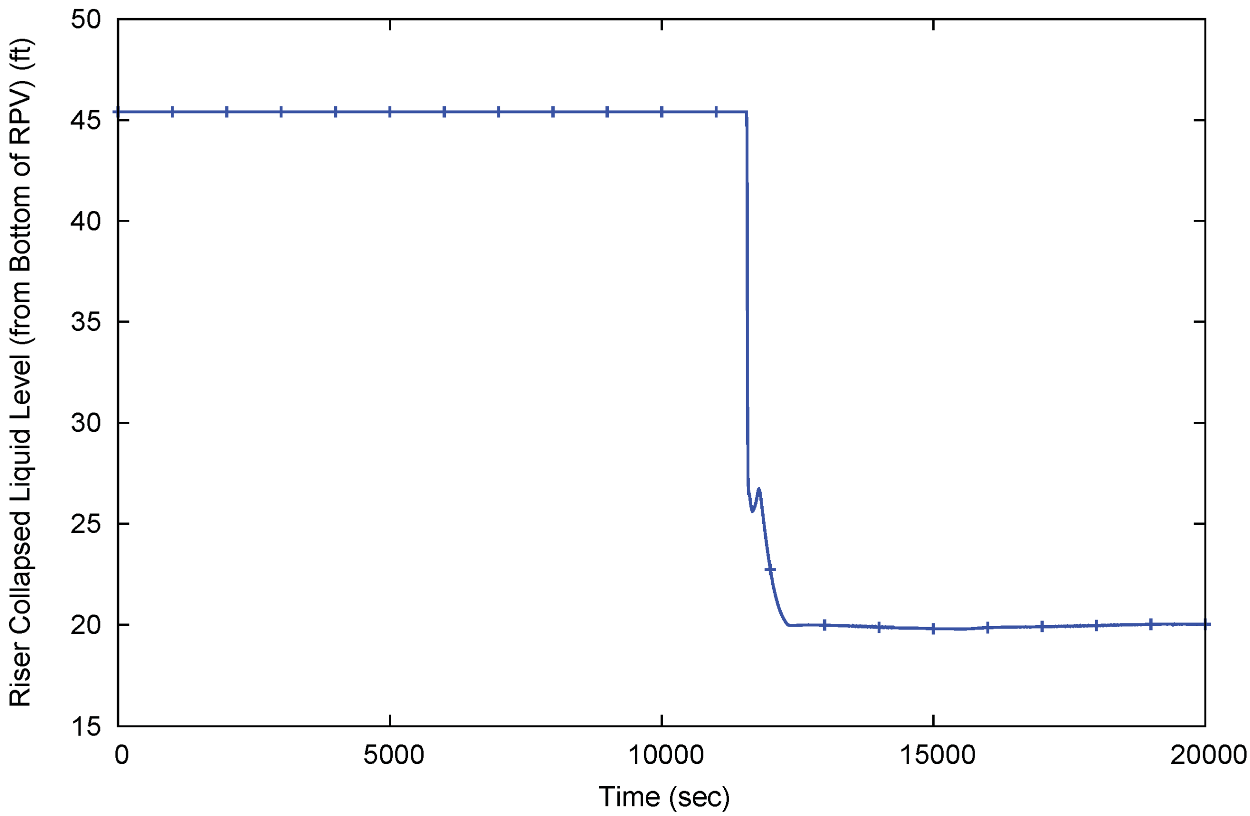

Figure 5-53 Feedwater line break riser level

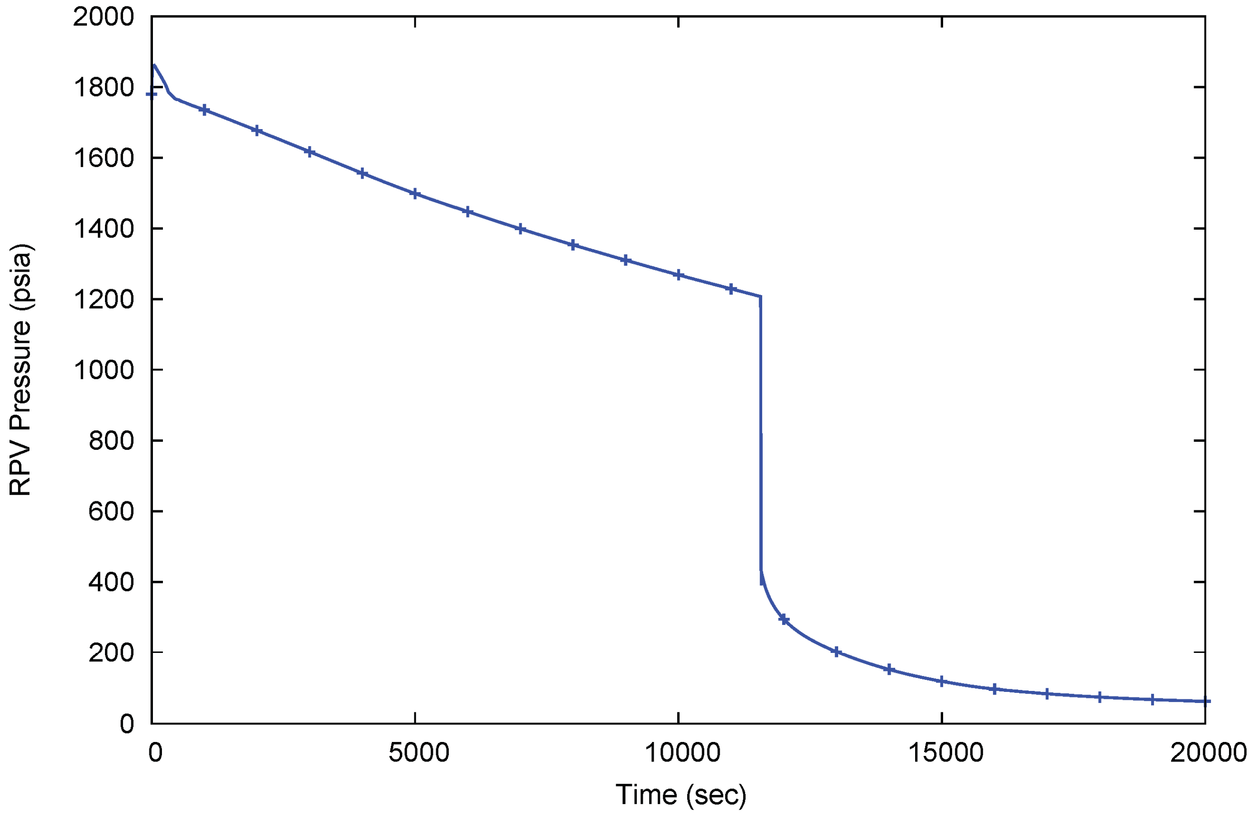

Figure 5-54 Feedwater line break primary system pressure

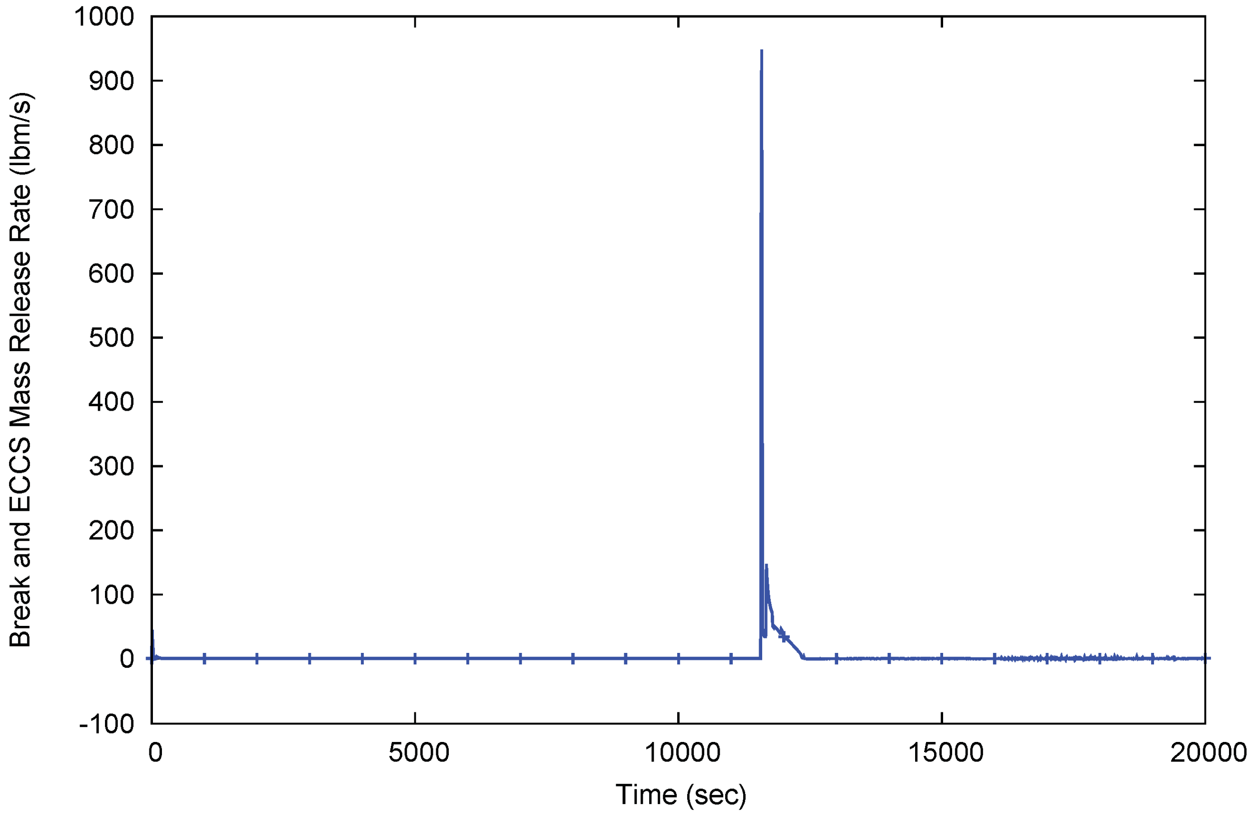

Figure 5-55 Feedwater line break and emergency core cooling system flowrate

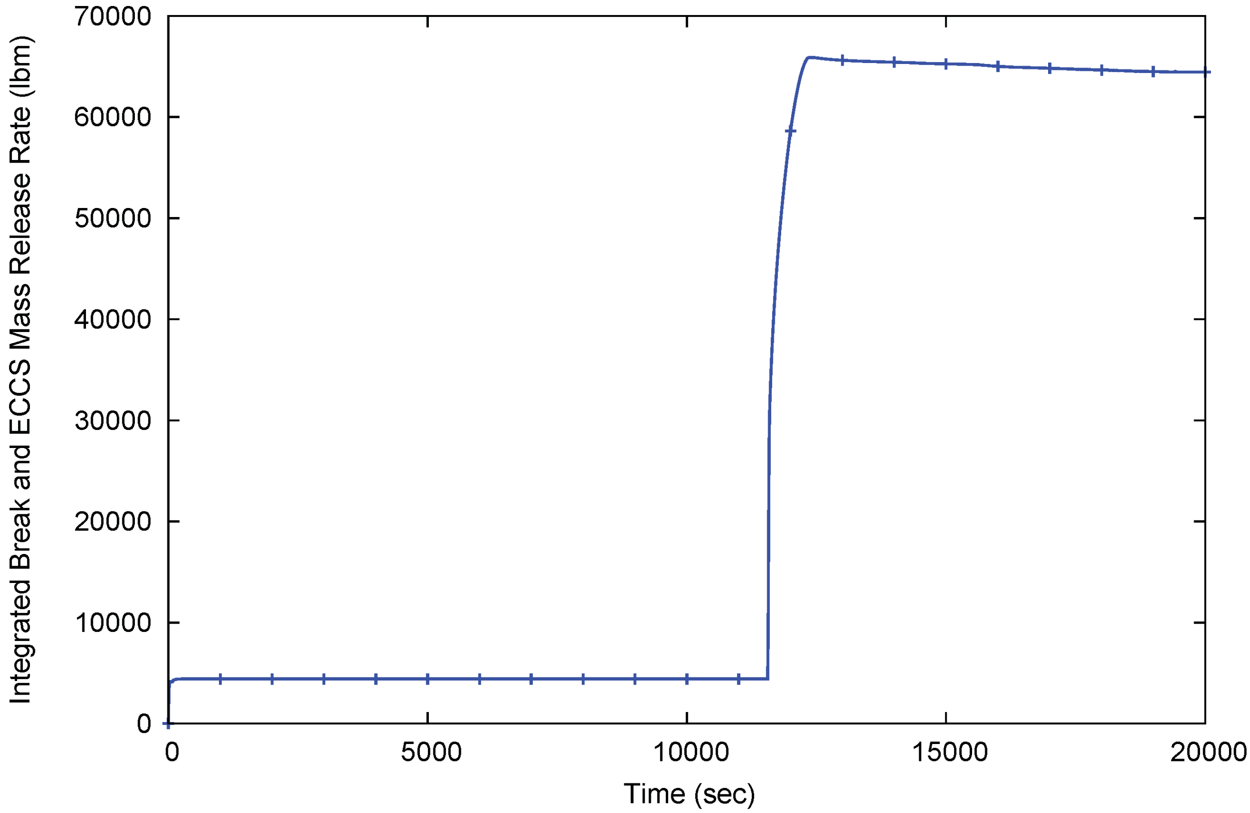

Figure 5-56 Feedwater line break and ECCS integrated mass release

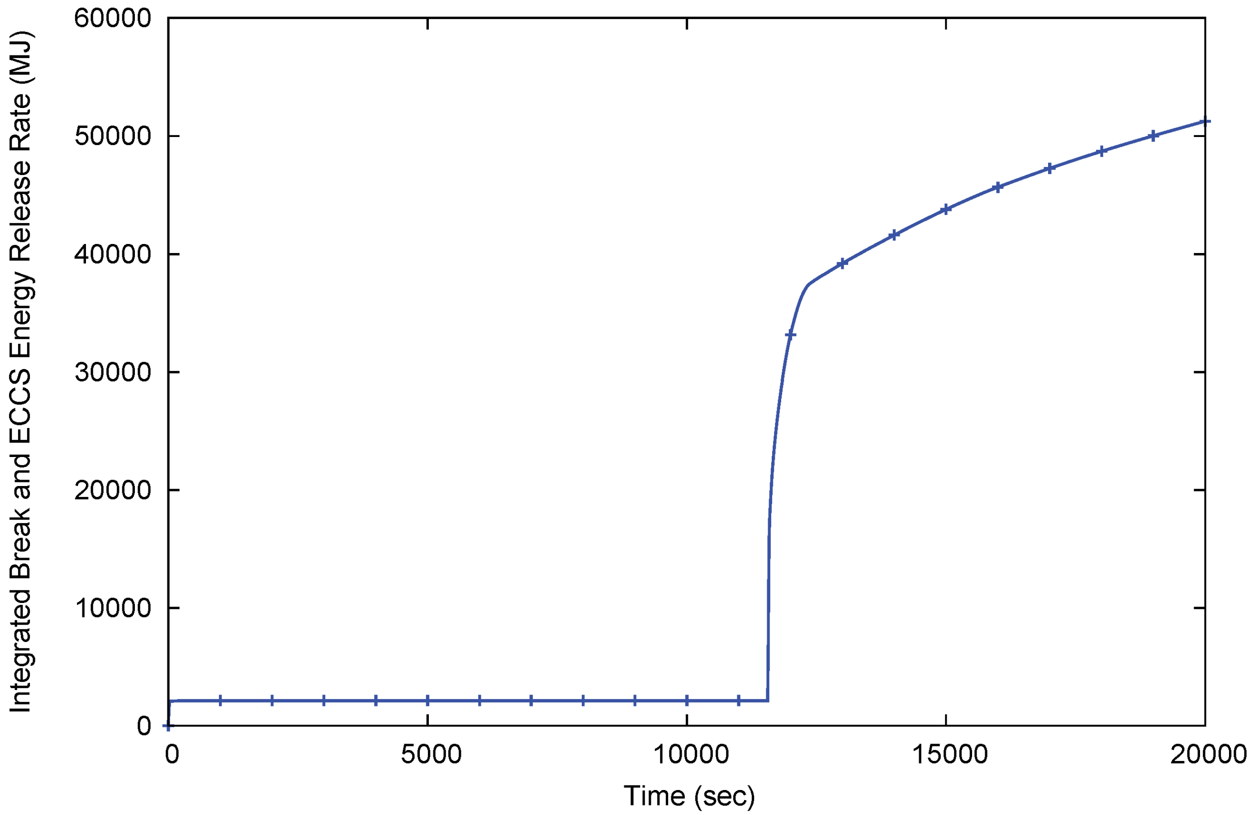

Figure 5-57 Feedwater line break and ECCS integrated energy release

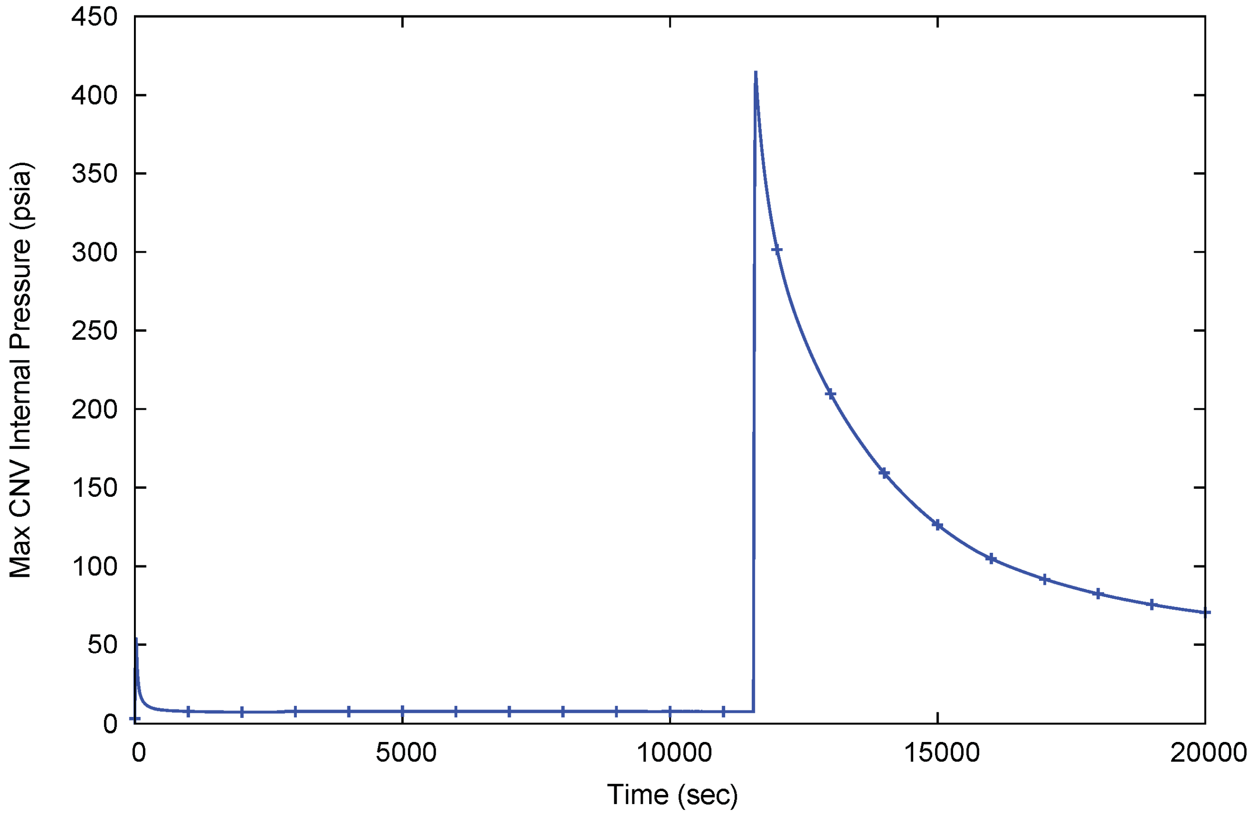

Figure 5-58 Feedwater line break containment vessel pressure

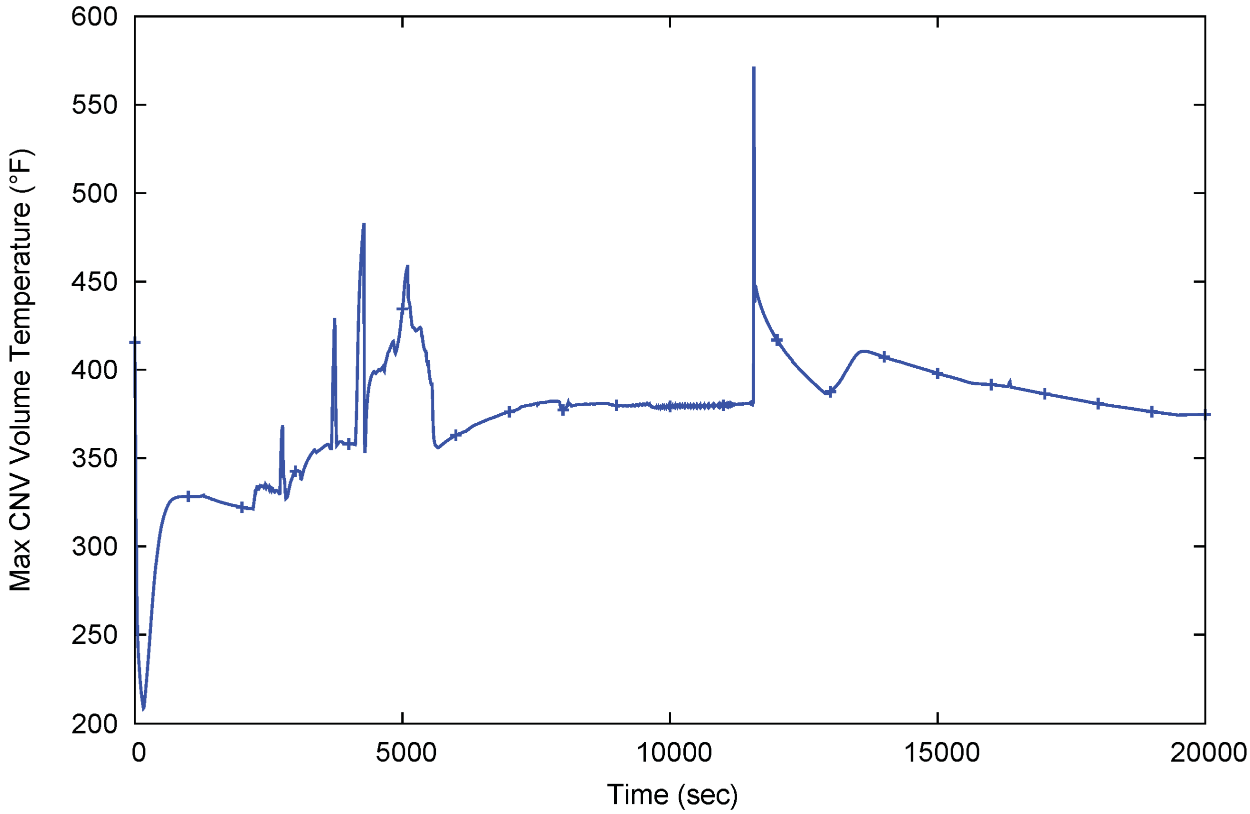

Figure 5-59 Feedwater line break containment vessel vapor temperature

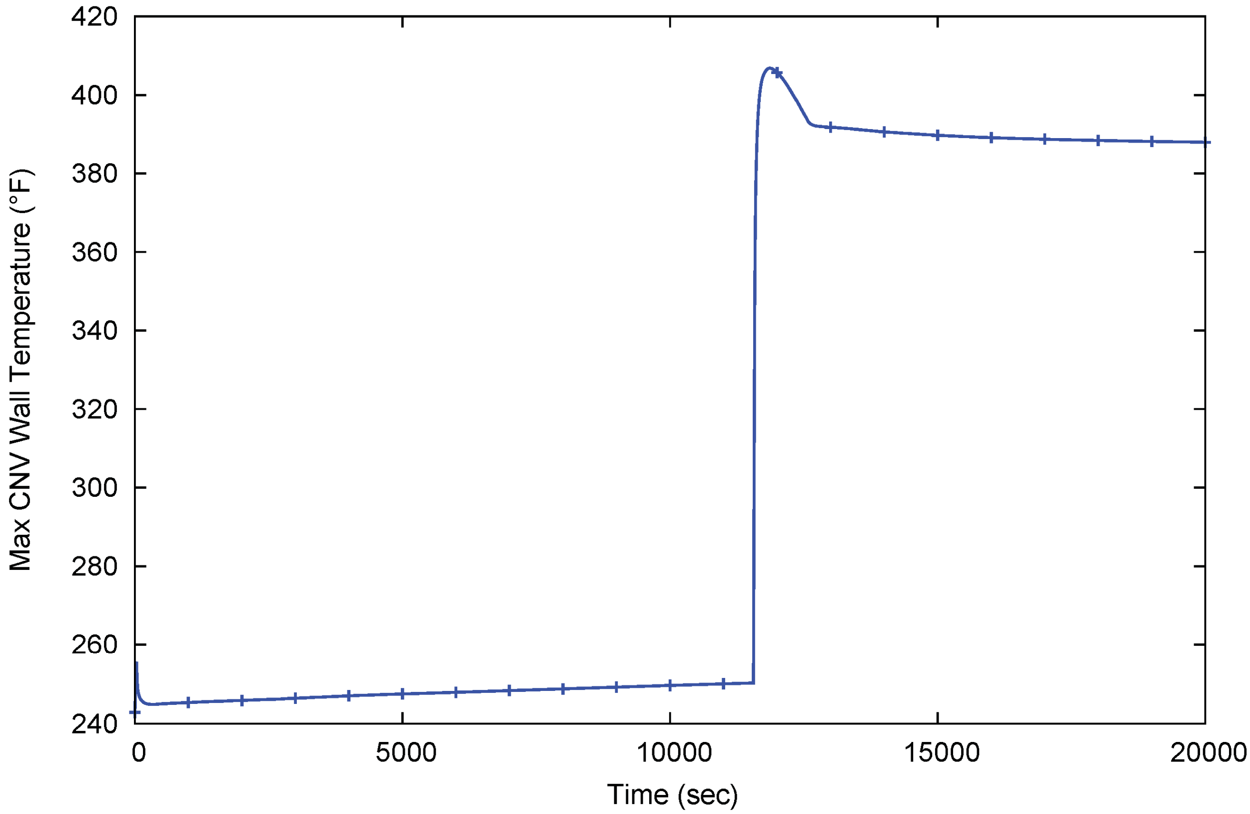

Figure 5-60 Feedwater line break containment vessel wall temperature

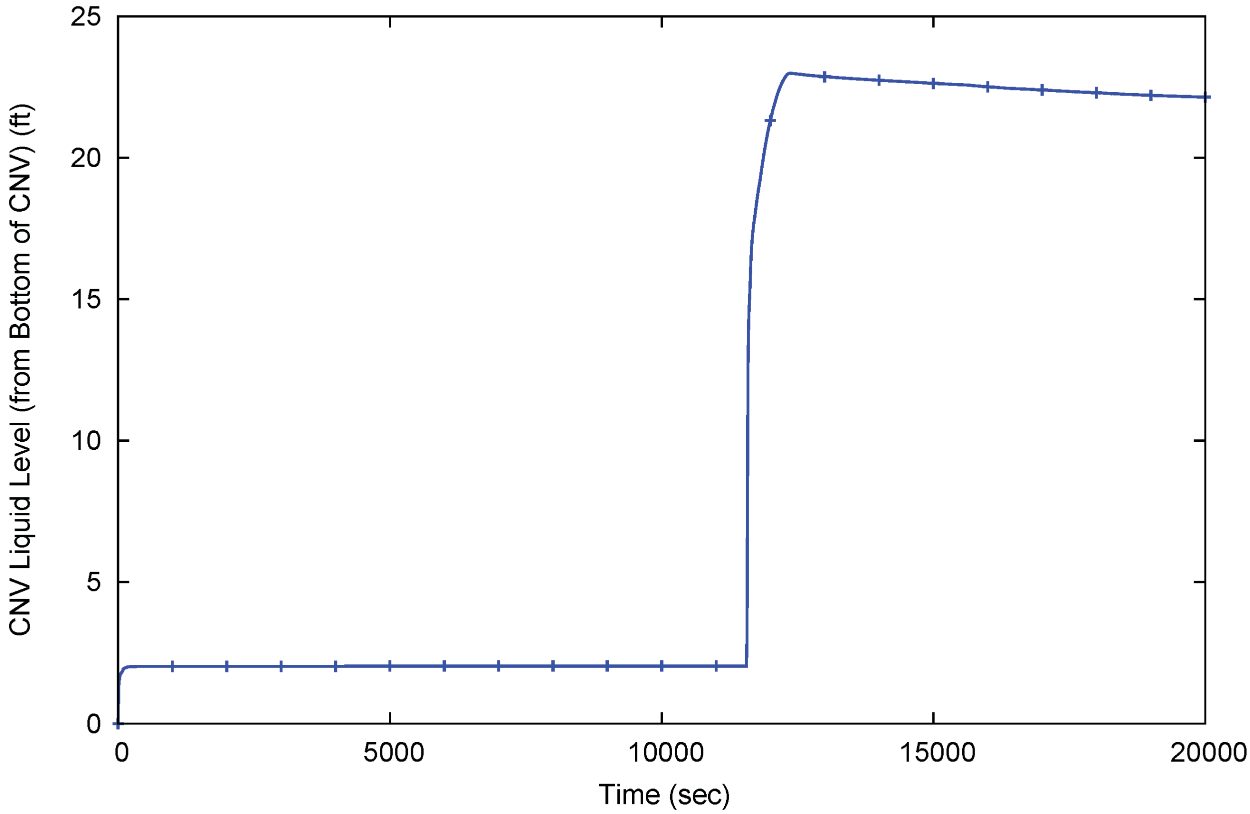

Figure 5-61 Feedwater line break containment vessel level

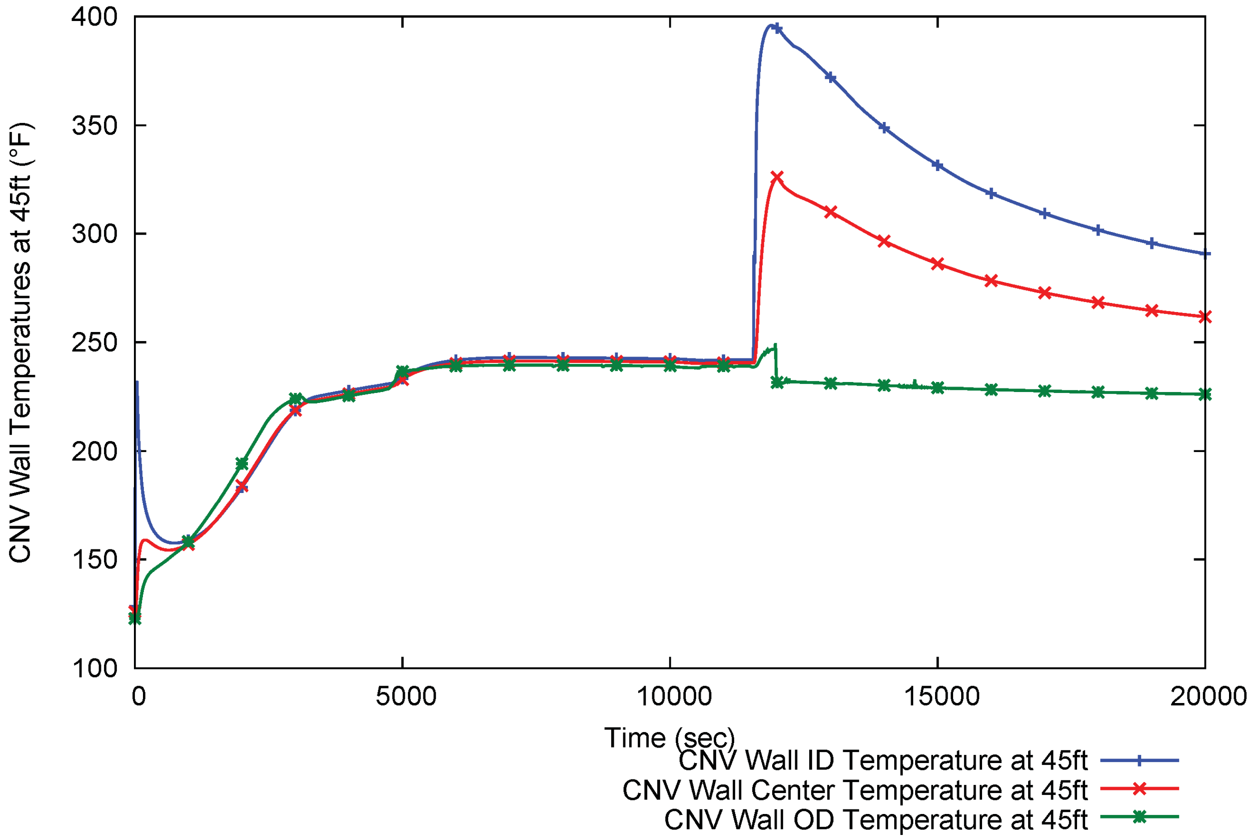

Figure 5-62 Feedwater line break containment vessel wall temperature profile

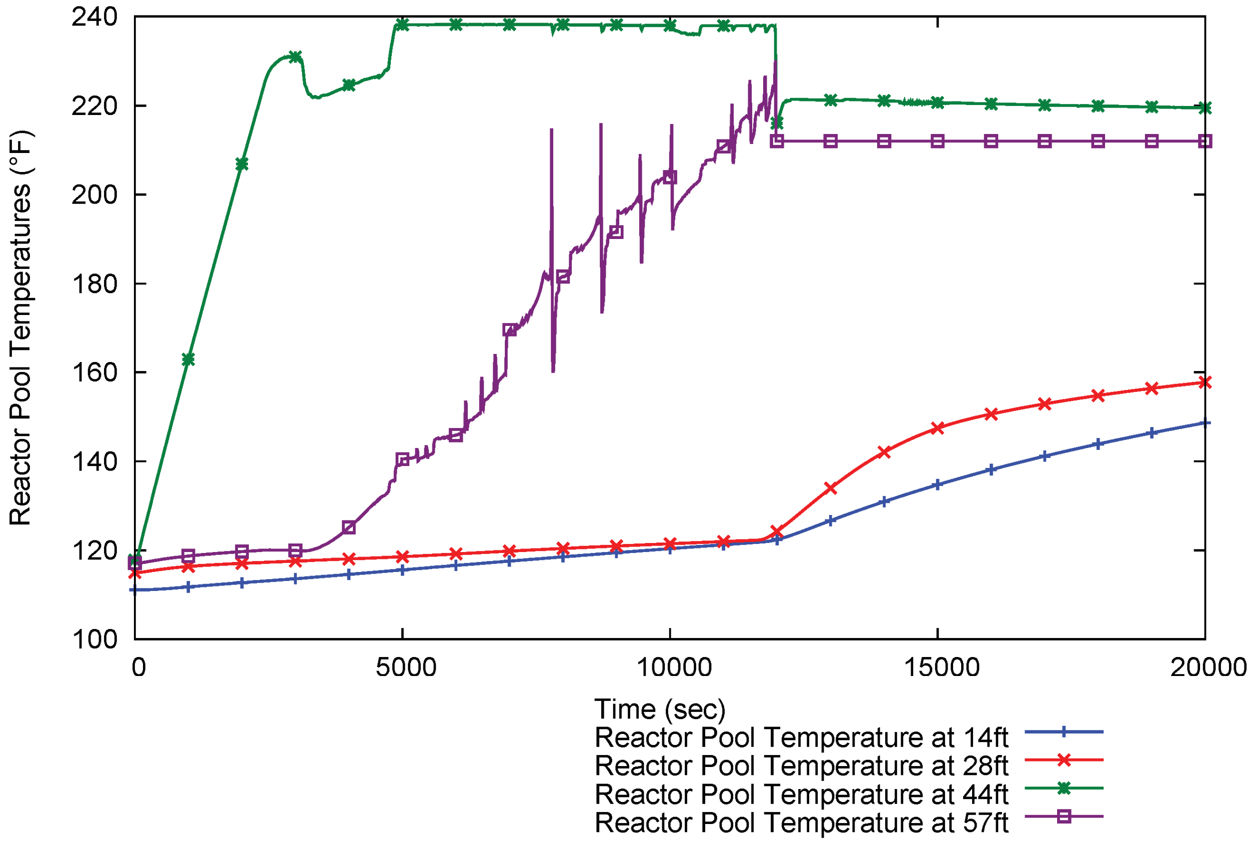

Figure 5-63 Feedwater line break reactor pool temperature

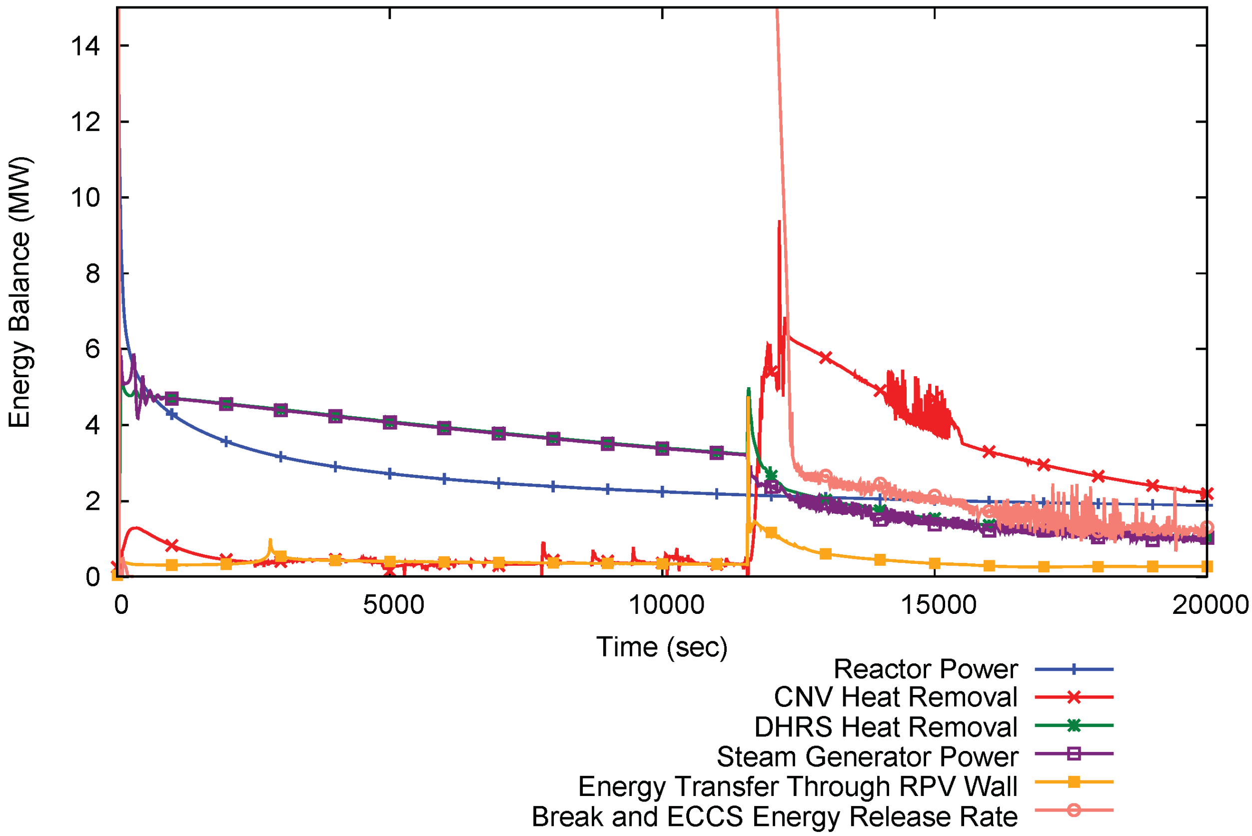

Figure 5-64 Feedwater line break energy balance

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.

5.4.1 Hydrostatic Pressure

The overall limiting peak CNV peak pressure results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power. The overall limiting CNV peak pressure is 994 psia, which is approximately 5 percent below the design pressure of 1050 psia, which occurs at a CNV elevation at the bottom of the CNV. The peak pressure occuring in the vapor space of the CNV is 987 psia; the difference is due to the hydrostatic pressure of liquid accumulation within the CNV at the time of peak pressure. The reactor pool hydrostatic head, which acts against the CNV exterior surface, provides additional margin that is not credited by the CNV response analysis methodology.

5.4.2 Decay Heat Removal System Availability

The LOCA (Case 2) and AOO (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 about 37 psi 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 LOCA peak pressure and the AOO peak pressure analyses demonstrate that sufficient margin to the CNV design pressure of 1050 psia is available to address the acceptance criteria given by DSRS Section 6.2.1.1.A (See Table 2-2). 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 hydrostatic head and availability of the DHRS in the containment response analysis.

Sensitivity studies determined an approximate 8 psi increase in the CNV peak pressure, documented in FSAR Section 6.2, for the limiting inadvertent opening of a reactor recirculation valve event if a lower IAB release pressure (listed in Section 5.1.1) is considered. The sensitivity calculations considered effects of different ECCS valves opening at different differential pressures over this range. The limiting pressure result, accounting for different ECCS valves opening at different pressures, of 994 psia is presented in FSAR Section 6.2. Considering the maximum pressure, and the margin not credited by the analysis discussed above, sufficient margin is provided to satisfy the requirements of GDC 16 and 50.

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, 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 reactor coolant system (RCS) injection line break. The LOCA limiting peak CNV wall temperature is approximately 526 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 959 psia, which results from a reactor coolant system injection line break case, with a loss of normal AC and DC power. The LOCA event peak CNV pressure is below the CNV design pressure of 1050 psia. The LOCA peak CNV pressure and wall temperature bound the main steamline break (MSLB) and feedwater lind break (FWLB) results.

The overall limiting event for peak CNV pressure is approximately 994 psia, which is approximately 5 percent below the containment design pressure of 1050 psia. It results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power considering an IAB release pressure range of 950 +/- 50 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 static water pressure 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 maximum temperature case), the overall peak CNV pressure event (Case 5 representative pressure case) and the limiting secondary system release event (MSLB), up to the time that the peak pressure is reduced to one half its value. The mass and energy releases provided in Table 8-1 are representative of the RRV opening event and reflect the case where all ECCS valves open at 1000 psid, resulting in peak containment pressure of 986 psia. The limiting peak pressure and temperature results are presented in FSAR Section 6.2.

Table 8-1 Case 5 Representative Peak Pressure Case – Mass and Energy Release

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
0.00	0.00	0.00
1.00	520.85	257627.45
2.00	523.17	258750.86
3.00	521.93	258164.72
4.00	519.26	256962.72
5.00	515.99	255607.71
6.00	512.44	254280.63
7.00	508.73	253062.33
8.00	504.94	251984.33
9.00	500.89	250943.23
10.00	497.07	250168.15
11.00	492.84	249319.52
12.00	488.50	248522.89
13.00	483.98	247730.73
14.00	479.49	247027.59
15.00	475.20	246477.05
16.00	470.79	245889.66
17.00	466.32	245272.77
18.00	461.90	244650.14
19.00	457.90	244207.37
20.00	454.78	244180.29
40.00	412.85	228110.54
60.00	378.12	216874.65
61.00	377.02	215906.03
62.00	375.81	215107.74
63.00	374.48	214449.07
64.00	373.13	213733.48
65.00	371.90	212887.33

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
66.00	370.89	212000.58
67.00	369.82	211170.35
68.00	368.63	210506.14
69.00	367.35	209910.68
70.00	366.22	209185.02
71.00	365.31	208336.35
72.00	364.44	207525.80
73.00	363.47	206812.03
74.00	362.44	206204.10
75.00	361.43	205589.78
76.00	360.48	204939.76
77.00	1105.21	1079703.71
78.00	1263.49	1270075.68
79.00	1169.96	1206666.55
80.00	976.53	1004884.36
81.00	838.30	909863.68
82.00	735.11	697397.33
83.00	2156.32	1334963.85
84.00	1611.53	1038460.23
85.00	1230.48	794378.32
86.00	502.46	523334.05
87.00	1515.48	927023.91
88.00	231.91	254414.90
89.00	155.94	177983.24
90.00	209.84	172368.89
91.00	161.79	130981.20
92.00	144.75	115382.50
93.00	139.61	110392.72
94.00	136.23	106487.96
95.00	132.60	102460.47
96.00	128.69	98461.93
97.00	126.62	96438.44
98.00	123.56	93419.72
99.00	120.86	90594.20
100.00	118.67	88154.41
101.00	116.65	86096.24
102.00	115.36	84829.11
103.00	114.12	83538.32
104.00	112.74	82081.88
105.00	111.91	81198.07
106.00	111.10	80460.96

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
107.00	110.12	79604.71
108.00	108.94	78503.95
109.00	107.37	77011.06
110.00	106.08	75843.21
111.00	105.14	75092.50
112.00	104.38	74545.22
113.00	104.15	74478.90
114.00	103.05	73560.53
115.00	102.05	72792.56
116.00	101.19	72219.92
117.00	100.41	71743.60
118.00	99.60	71236.55
119.00	98.66	70603.35
120.00	97.70	70001.60
140.00	79.37	58965.27
160.00	67.06	51663.90
180.00	58.34	46108.22
200.00	51.94	41845.43
220.00	47.36	38792.00
240.00	43.26	36013.35
260.00	37.48	32408.39
280.00	35.65	30650.56
300.00	33.86	29094.61
320.00	32.43	27834.92
340.00	30.90	26514.72
360.00	29.81	25566.75
380.00	28.61	24579.84
400.00	27.29	23548.94
420.00	26.15	22680.53
440.00	25.18	21945.98
460.00	24.11	21085.17
480.00	23.28	20464.64
500.00	22.39	19824.76
520.00	21.45	19148.74
540.00	20.52	18497.43
560.00	19.61	17896.37
580.00	18.64	17263.95
600.00	17.69	16652.23
620.00	16.71	16050.69
640.00	15.59	15391.62
660.00	14.30	14661.37

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
680.00	12.98	13926.77
700.00	8.86	11910.40
720.00	7.42	11136.13
740.00	6.43	10578.58
760.00	5.72	10159.74
780.00	5.16	9842.23
800.00	4.63	9508.67
820.00	4.17	9219.58
840.00	3.77	8959.33
860.00	3.42	8737.91
880.00	3.09	8525.88
900.00	2.79	8327.20
920.00	2.56	8181.55
940.00	2.29	8002.94
960.00	2.05	7836.06
980.00	1.77	7617.34
1000.00	1.63	7541.69
1020.00	1.43	7411.43
1040.00	1.25	7284.12
1060.00	1.08	7165.44
1080.00	0.71	6883.31
1100.00	0.69	6923.14
1120.00	0.57	6829.92
1140.00	0.45	6740.14
1160.00	0.34	6655.13
1180.00	0.23	6569.41
1200.00	0.12	6484.89
1220.00	0.02	6404.77
1240.00	-0.07	6334.75
1260.00	-0.16	6270.01
1280.00	-0.23	6208.31
1300.00	-0.31	6149.76
1320.00	-0.37	6094.65
1340.00	-0.44	6042.12
1360.00	-0.50	5993.93
1380.00	-0.55	5944.68
1400.00	-0.61	5895.22
1420.00	-0.66	5849.06
1440.00	-0.71	5805.89
1460.00	-0.75	5766.42
1480.00	-0.79	5728.05

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
1500.00	-0.83	5689.85
1520.00	-0.87	5652.53
1540.00	-0.91	5616.71
1560.00	-0.94	5581.06
1580.00	-0.97	5547.13
1600.00	-1.01	5513.44
1620.00	-1.04	5476.91
1640.00	-1.07	5442.10
1660.00	-1.10	5406.73
1680.00	-1.13	5375.81
1700.00	-1.15	5347.79
1720.00	-1.17	5319.46
1740.00	-1.19	5292.06
1760.00	-1.21	5265.43
1780.00	-1.23	5239.46
1787.10	-0.90	5371.96

1. RRV opens at 0 seconds.

Table 8-2 Limiting Peak Wall Temperature Case - Mass and Energy Release

Time (s)(1)	Mass Release (lbm/s)	Energy Release (Btu/s)
0.00	0.00	0.00
1.00	79.18	48243.13
2.00	79.30	48314.98
3.00	79.57	48482.36
4.00	79.92	48690.46
5.00	80.29	48914.73
6.00	80.64	49119.33
7.00	80.83	49230.89
8.00	80.91	49270.28
9.00	80.89	49252.24
10.00	80.81	49187.48
11.00	80.70	49080.36
12.00	80.59	48944.34
13.00	80.50	48782.97
14.00	80.43	48601.19
15.00	80.38	48398.72
16.00	80.37	48197.13
17.00	80.45	48031.03
18.00	80.63	47922.32
19.00	80.88	47846.57
20.00	81.14	47787.04
40.00	82.67	46764.30
60.00	81.49	45804.67
80.00	80.30	44989.66
100.00	78.89	44200.28
120.00	77.54	43524.76
140.00	76.25	42962.65
160.00	74.95	42446.83
180.00	73.72	41987.01
200.00	72.58	41572.17
220.00	72.65	41690.47
240.00	71.49	41296.20
260.00	70.43	40984.25
280.00	69.50	40786.43
300.00	68.37	40548.94
320.00	66.60	39942.51
340.00	64.62	38980.23
360.00	64.08	38649.04
380.00	63.87	38472.63

Time (s)(1)	Mass Release (lbm/s)	Energy Release (Btu/s)
400.00	63.87	38426.77
420.00	63.75	38342.67
440.00	63.54	38216.47
460.00	63.37	38113.27
480.00	63.22	38024.68
500.00	63.11	37955.82
520.00	63.02	37892.62
540.00	62.94	37835.79
560.00	62.87	37788.37
580.00	62.82	37745.00
600.00	62.77	37705.34
620.00	62.70	37657.63
640.00	62.66	37616.46
660.00	62.61	37577.23
680.00	62.57	37538.52
700.00	62.51	37491.14
720.00	62.47	37449.59
740.00	62.43	37409.24
760.00	62.38	37367.66
780.00	62.32	37313.38
800.00	62.23	37246.63
820.00	62.17	37187.16
840.00	62.12	37138.68
860.00	62.10	37095.76
880.00	62.10	37068.80
900.00	62.08	37029.61
920.00	62.03	36977.23
940.00	61.95	36914.79
950.00	61.85	36852.69
951.00	61.84	36846.42
952.00	61.83	36840.27
953.00	61.82	36834.32
954.00	61.81	36828.52
955.00	1352.36	1370453.47
956.00	1193.26	1214639.97
957.00	996.84	1039740.72
958.00	841.22	878708.54
959.00	677.92	705524.91
960.00	553.42	574452.22
961.00	488.95	500751.61
962.00	378.97	386018.43

Time (s)(1)	Mass Release (lbm/s)	Energy Release (Btu/s)
963.00	286.76	292965.95
964.00	206.23	210777.85
965.00	146.88	149539.51
966.00	104.54	106021.53
967.00	77.48	80524.71
968.00	64.94	67469.99
969.00	56.93	58992.81
970.00	51.97	53697.29
971.00	48.08	49749.34
972.00	45.23	46977.89
973.00	41.99	43648.53
974.00	39.38	41252.21
975.00	38.24	40286.01
976.00	36.20	38304.67
977.00	34.52	36582.13
978.00	32.77	35013.81
979.00	32.66	35063.81
980.00	30.51	33275.33
1000.00	7.61	16951.11
1020.00	4.87	14668.09
1040.00	3.50	13415.22
1060.00	2.36	12351.95
1080.00	1.38	11456.21
1100.00	0.53	10619.46
1120.00	-0.19	9953.30
1140.00	-0.71	9465.05
1160.00	-1.09	9093.51
1180.00	-1.49	8686.42
1200.00	-1.89	8306.51
1220.00	-2.31	7904.29
1240.00	-2.65	7592.29
1260.00	-2.97	7271.04
1280.00	-3.19	7042.75
1300.00	-3.38	6834.40
1320.00	-3.48	6721.77
1340.00	-3.67	6547.47
1360.00	-3.78	6403.45
1380.00	-3.90	6250.04
1400.00	-3.96	6129.98
1420.00	-4.00	6046.52
1440.00	-4.11	5896.39

Time (s)(1)	Mass Release (lbm/s)	Energy Release (Btu/s)
1460.00	-4.17	5796.09
1480.00	-4.17	5752.75
1500.00	-4.25	5623.53
1520.00	-4.26	5552.52
1540.00	-4.27	5462.45
1560.00	-4.30	5384.27
1580.00	-4.31	5317.01
1600.00	-4.31	5267.64
1620.00	-4.34	5193.43
1640.00	-4.24	5213.03
1660.00	-4.28	5114.53
1680.00	-4.30	5051.19
1700.00	-4.27	5022.07
1720.00	-4.27	4961.03
1740.00	-4.20	4963.80
1760.00	-4.10	4979.97
1780.00	-3.99	4982.59
1800.00	-3.79	5020.53
1820.00	-3.51	5109.37
1840.00	-3.34	5139.58
1860.00	-3.25	5130.42
1880.00	-3.15	5149.35
1900.00	-3.12	5103.51
1920.00	-3.11	5060.80
1940.00	-3.10	5028.42
1960.00	-3.08	4996.18
1980.00	-3.04	4966.92
2000.00	-3.09	4867.16
2020.00	-3.38	4746.53
2040.00	-3.43	4697.04
2060.00	-3.29	4710.26
2080.00	-3.16	4724.90
2100.00	-3.04	4760.70
2120.00	-2.93	4781.73
2140.00	-2.86	4788.56
2160.00	-2.82	4767.95
2180.00	-2.78	4736.15
2200.00	-2.73	4727.76
2220.00	-2.61	4763.77
2240.00	-2.55	4748.07
2260.00	-2.51	4731.84

Time (s)(1)	Mass Release (lbm/s)	Energy Release (Btu/s)
2280.00	-2.46	4724.87
2300.00	-2.40	4733.99
2320.00	-2.34	4744.78
2340.00	-2.24	4783.38
2360.00	-2.24	4743.72
2380.00	-2.23	4713.57
2400.00	-2.18	4719.62
2420.00	-2.34	4626.02
2440.00	-2.26	4644.15
2460.00	-2.05	4698.36
2472.90	-2.26	4601.93

1. Break initiated at 0 seconds.

Table 8-3 Limiting Secondary Break Peak Pressure Mass and Energy Release

Time (s) [(1)]	Mass Release (lbm/s)	Energy Release (Btu/s)
0.00	0.00	0.00
1.00	429.13	529471.38
2.00	333.15	413594.14
3.00	311.07	393366.92
4.00	290.94	371550.11
5.00	264.05	339779.42
6.00	238.67	309056.90
7.00	216.91	282123.38
8.00	192.24	250588.45
9.00	118.64	154655.48
10.00	118.04	154001.57
15.00	173.00	218097.88
16.00	180.15	224434.34
17.00	184.92	227523.77
18.00	187.88	229107.05
19.00	187.54	226782.47
20.00	185.84	223894.90
25.00	159.89	196109.64
30.00	131.22	164114.01
35.00	99.41	126511.96
40.00	55.26	71015.79
45.00	34.88	44867.91
46.00	13.36	17175.87
47.00	7.15	9186.32
48.00	6.09	7825.33
49.00	5.59	7178.04
50.00	3.25	4170.70
55.00	2.81	3593.22
60.00	2.54	3252.16
80.00	0.73	943.37
100.00	0.57	725.24
120.00	0.48	609.39
140.00	0.49	623.51
160.00	0.35	444.30
180.00	2.14	2752.23
196.55	4.19	5420.36

1. Break initiated at 0 seconds.

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.

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 8-4 Passive heat sinks

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

8.2.3 Thickness Groups

Table 8-5 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-965 FXM-19 (Stainless Steel), SA-508 Grade 3 (Carbon Steel)	3	{{ }}2(a),(c)

8.2.4 Properties of Passive Heat Sink Materials

Table 8-6 Physical properties of passive heat sink materials

Material	**Density, lbm/ft^3**	**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
SA-965 FXM-19 (Stainless Steel)	487.296	0.1142	6.4

Enclosure 3:

Affidavit of Zackary W. Rad, AF-1119-68069

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying 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 titled "Containment Response Analysis Methodology," TR-0516-49084, Revision 2. 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 November 26, 2019.

Zackary W. Rad