August 5, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of "Long-Term Cooling Methodology," TR-0916-51299, Revision 1

REFERENCES:

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Submittal of Technical Reports supporting the NuScale Design Certification Application," dated January 9, 2017 (ML17009A490)
2. NuScale Technical Report, "Long-Term Cooling Methodology," TR-0916-51299, Revision 0, dated January, 2017 (ML17009A492)

NuScale Power, LLC (NuScale) hereby submits Revision 1 of the "Long-Term Cooling Methodology" (TR-0916-51299).

Enclosure 1 contains the proprietary version of the report entitled "Long-Term Cooling Methodology," TR-0916-51299, Revision 1. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. Enclosure 2 contains the nonproprietary version of the report entitled "Long-Term Cooling Methodology," TR-0916-51299, Revision 1.

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

If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

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Enclosure 1: "Long-Term Cooling Methodology," TR-0916-51299-P, Revision 1, proprietary version
Enclosure 2: "Long-Term Cooling Methodology," TR-0916-51299-NP, Revision 1, nonproprietary version
Enclosure 3: Affidavit of Thomas A. Bergman, AF-0719-66145

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Enclosure 1:

"Long-Term Cooling Methodology," TR-0916-51299-P, Revision 1, proprietary version

Enclosure 2:

"Long-Term Cooling Methodology," TR-0916-51299-NP, Revision 1, nonproprietary version

Licensing Technical Report

Long-Term Cooling Methodology

August 2019

Revision 1

Docket: 52-048

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Abstract

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

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

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

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

Executive Summary

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

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

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

The report describes the following NuScale-specific LTC acceptance criteria that were developed to assure that regulatory requirements of 10 CFR 50.46 are met: 1) collapsed liquid level in the reactor vessel remains above the top of the core, 2) boron concentrations in the core region remain below the boron solubility limit, and 3) fuel cladding temperatures predicted by NRELAP5 are maintained at an acceptable level.

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

The LTC EM is developed using a graded approach to the evaluation model development and assessment process (EMDAP) defined in Regulatory Guide (RG) 1.203. The approach for the long-term cooling EM utilizes the NuScale LOCA EM (Reference 8.2.1). The LTC EM focuses on the phenomena identification and ranking table (PIRT) process to identify the important parameters which are specifically addressed. An extensive PIRT was developed for LTC. Each important parameter is discussed and evaluated in this report as it relates to the LTC EM.

The LTC EM uses the proprietary NRELAP5 systems analysis computer code as the computational engine, derived from the Idaho National Laboratory RELAP5-3D© computer code. The models and correlations used by the NRELAP5 code were reviewed and determined to be

appropriate for use within the long-term cooling EM. The NRELAP5 model is validated through the assessment of NIST-1 facility tests and comparison of NRELAP5 predictions to test results. Comparison of the NRELAP5 model to the NIST-1 test results demonstrate that the NRELAP5 code adequately predicts the NPM conditions both in the RPV and the CNV.

The methodology for the NPM thermal-hydraulic response and boron precipitation evaluation are presented in this report. There are three LTC general conditions which address the thermal-hydraulic response and boron precipitation: (1) maximum cooldown to minimize the RPV core inlet region temperature for addressing boron precipitation, (2) minimum collapsed liquid level to minimize the volume of liquid in the riser region and above the active fuel to demonstrate core coverage and address boron precipitation, and (3) minimum cooldown to maximize the fuel cladding temperature.

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

- single active failure, ECCS valve failure to open is the relevant single active failure to consider in the LTC analyses
- decay heat, ranging from no decay heat to 120 percent of nominal
- DHRS operation
- reactor pool temperature, ranging from 65 degrees F to 210 degrees F
- reactor pool level, down to 55 feet (Nominal at 69 feet), 45 feet for LTC with decay heat from twelve modules
- Expansion factor used to account for compressible flow through RVVs
- non-condensable gas effect
- pressurizer level, down to 20 percent level

In all analyzed cases, the core remained covered, with greater than 2.8 feet of collapsed liquid level in the riser above the top of the core. Possible leakage from the CNV was found to have a negligible impact on the results. The cases identified as most limiting, maximum temperature with injection line break, minimum temperature with injection line break, minimum level with injection line break, and minimum level with steam generator tube failure, all showed consistently decreasing reactor coolant system (RCS) and cladding temperatures, supporting the conclusion that the ECCS is capable of providing adequate cooling for the 72 hour evaluation period.

In order to evaluate the criterion for maintaining coolable geometry, the possibility of boron precipitation is evaluated in this report. The methodology for determining boron precipitation is conservative, as it assumes the maximum boron concentration and a minimum volume that includes the core and riser region for boron mixing and that all the RPV boron remains within this region. The maximum boron concentration is shown in this report to remain below the solubility limit for the minimum RCS temperatures reached within the 72 hour evaluation period for long-term cooling.

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

1.0 Introduction

1.1 Purpose

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

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

1.2 Scope

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

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

This report summarizes the following:

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

The following LTC analysis areas are addressed in this report:

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

The following areas are outside scope of this report:

- The non-LOCA and LOCA evaluation models for the short-term time periods are covered in separate methodology reports. However the transition between short term LOCA and non-LOCA initiating events is addressed to demonstrate accurate boundary conditions are simulated upon the onset of long term cooling.
- The effects of debris on ECCS operation that are the subject of the NRC generic safety issue 191 are outside scope of this report. The NuScale design and debris loads have been assessed to ensure that the system and its components will operate as designed under long-term ECCS operating conditions. The LTC analyses are performed assuming a clean core condition without debris.
- Assessment of the NuScale design return to power due to overcooling, assuming one control rod stuck out of the core, is outside the scope of this report. For the LTC calculations in this report, the heat source is decay heat.
- Assessment of a station blackout is outside the scope of this report and covered in separate analysis.
- Analysis of long-term decay heat removal system (DHRS) performance and decay heat removal is addressed by separate analyses.
- This EM does not assess seismic issues, which are covered in separate methodologies and assessments.
- Critical heat flux (CHF) evaluation is only of interest in the short-term response of the events analyzed in this document. Short-term LOCA CHF is addressed by the NuScale LOCA EM (Reference 8.2.1). For long-term cooling, a collapsed liquid level above the top of active fuel (TAF) and acceptably low cladding temperatures calculated by NRELAP5 are considered sufficient to demonstrate that CHF does not occur.

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

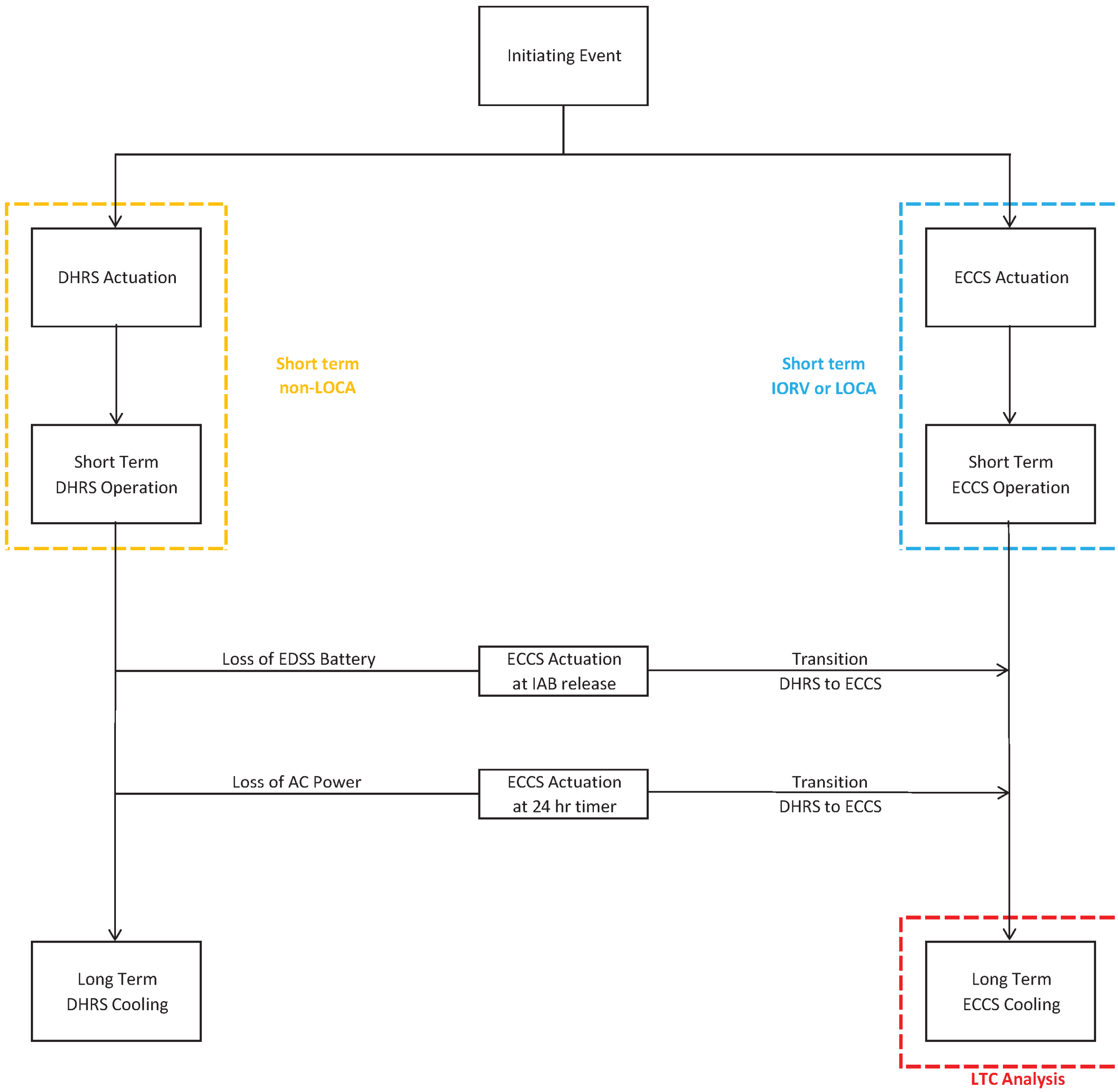

Figure 1-1 Illustration of the scope and analyses covered by long-term cooling methodology

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
AC	alternating current
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CHF	critical heat flux
CNV	containment vessel
CVCS	chemical and volume control system
DCA	Design Certification Application
DHRS	decay heat removal system
DSRS	Design Specific Review Standard
ECCS	emergency core cooling system
EM	evaluation model
EMDAP	evaluation model development and assessment process
FOM	figure of merit
GDC	Generic Design Criterion
HZP	hot zero power
IAB	inadvertent actuation block
IL	injection line
LOCA	loss-of-coolant accident
LTC	long-term cooling
NIST-1	NuScale Integral System Test
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
PDC	principal design criteria
PIRT	phenomena identification and ranking table
PZR	pressurizer
RCS	reactor coolant system
RG	Regulatory Guide
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RVV	reactor vent valve
SAF	single active failure
SG	steam generator
SGTF	steam generator tube failure
TAF	top of active fuel
UHS	ultimate heat sink

Table 1-2 Definitions

Term	Definition
C_v	Flow coefficient
“Excellent” agreement	One of the acceptance criteria defined in RG 1.203. “Excellent” agreement applies when the code exhibits no deficiencies in modeling a given behavior. Major and minor phenomena and trends are correctly predicted. The calculated results are judged to agree closely with the data. The calculation will, with few exceptions, lay within the specified or inferred uncertainty bands of the data. The code may be used with confidence in similar applications.
Figure of merit	A parameter selected to characterize the plant long-term cooling response.
Loss-of-coolant accident	Those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.
Non-LOCA transient	Reactor coolant system transients described in the NUREG-0800 Standard Review Plan Sections 15.1, 15.2, 15.4, and 15.5, and other comparable transients that may be unique to the NuScale system. Other sections in the standard review plan are specific to events with reactor coolant pumps, LOCA, radiological analysis, anticipated transient without scram, or boiling water reactors, and are outside of the scope of non-LOCA transients.
“Reasonable” agreement	One of the acceptance criteria defined in RG 1.203. “Reasonable” agreement applies when the code exhibits minor deficiencies. Overall, the code provides an acceptable prediction. All major trends and phenomena are correctly predicted. Differences between calculation and data are greater than deemed necessary for excellent agreement. The calculation will frequently lie outside but near the specified or inferred uncertainty bands of the data. However, the correct conclusions about trends and phenomena would be reached if the code was used in similar applications.

2.0 Regulatory Requirements and Roadmap

2.1 Background

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

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

- protection against design basis events is through passive means for at least 72 hours, and
- no operator actions are required for at least 72 hours for design basis events.

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

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

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

2.2 Regulatory Requirements and Guidance

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

2.2.1 Regulatory Requirements

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

of the 10 CFR 50 Appendix K requirements that apply to the long-term cooling phase are applied in the same manner as for the LOCA EM. Since the NuScale LOCA EM (Reference 8.2.1) and the LTC EM are equivalent with regard to all Appendix K requirements, no further exemptions to the Appendix K requirements are required for the LTC EM beyond those identified in Reference 8.2.1.

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

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

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

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

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

2.2.2 Regulatory Guidance

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

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

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

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

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

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

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

2.3 Acceptance Criteria and Transient Duration

2.3.1 Acceptance Criteria

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

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

1. core cooling is provided to remove decay and residual heat from the core. This acceptance criterion is demonstrated in thermal-hydraulic calculations with NRELAP5 by the following:
    a. collapsed liquid level in the reactor vessel remains above the top of the core.
    b. cladding temperatures predicted by NRELAP5 remain acceptably low.
    c. margin to the CHF predicted by NRELAP5 using a CHF correlation appropriate to the fluid conditions is maintained.
        - DSRS 15.6.5-10 states "If core uncovery is not expected during the entire period of a LOCA, the staff should ensure that a significant number of fuel rods will not be damaged because of local dryout conditions. This may be demonstrated by showing that the limiting fuel rod heat flux remains below the critical heat flux (CHF) at a given pressure after depressurization has taken place. If, however, the heat flux exceeds the CHF, further analyses should be performed to estimate the amount of fuel damage expected from "burn-out" while the bulk of the core remains covered with water during the LOCA. Fuel damage and potential for radioactivity release to the environment must be consistent with 10 CFR Part 100."
        - The NuScale LOCA EM addresses the short-term CHF response to a primary system pipe beak and ECCS actuation. No explicit CHF response is evaluated as part of the LTC calculations; maintaining a collapsed liquid level in the riser above the core, along with demonstrating that cladding temperatures remain acceptably low, are considered sufficient conditions to show MCHFR limits are not challenged. In addition, meeting the criteria that the core remain covered by collapsed liquid level in the riser and that cladding temperatures remain acceptably low assure that the PDC 35

criterion that “clad metal-water reaction is limited to negligible amounts” is met.

2. coolable geometry is maintained.

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

3. the core remains subcritical.

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

2.3.2 Transient Duration

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

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

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

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

2. DHRS cooling scenarios that transition to ECCS cooling were considered. These scenarios include:

    a. ECCS cooling that begins after a period of initial DHRS cooling, and

    b. DHRS cooling cases that transition into ECCS cooling.

    Long-term cooling analyses demonstrate that if DHRS cooling is provided until either the inadvertent actuation block (IAB) setpoint is reached or 24 hours is reached such that the ECCS timer expires, and decay heat removal transitions to ECCS cooling, then the module(s) will remain in a safe, stable condition for up to 72 hours following the event. DHRS transition cases will also include consideration for SGTF to address inventory loss prior to isolation of the SG.

2.4 Long-Term Cooling Evaluation Model Roadmap

Analyses are performed to demonstrate that a nuclear power plant can meet applicable NRC regulatory acceptance criteria for a limiting set of anticipated operational occurrences, infrequent events, and accidents. The EMDAP as defined in RG 1.203 (Reference 8.2.6) provides a structured process to establish the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems. The EM described in this report has been developed for simulating the long-term cooling capability of the NPM during long-term ECCS operation.

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

Figure 2-1 shows various elements of EMDAP as defined in RG 1.203 (Reference 8.2.6). The elements of the EMDAP and sections of this report that relate to the elements and steps of the EMDAP are summarized in Table 2-1.

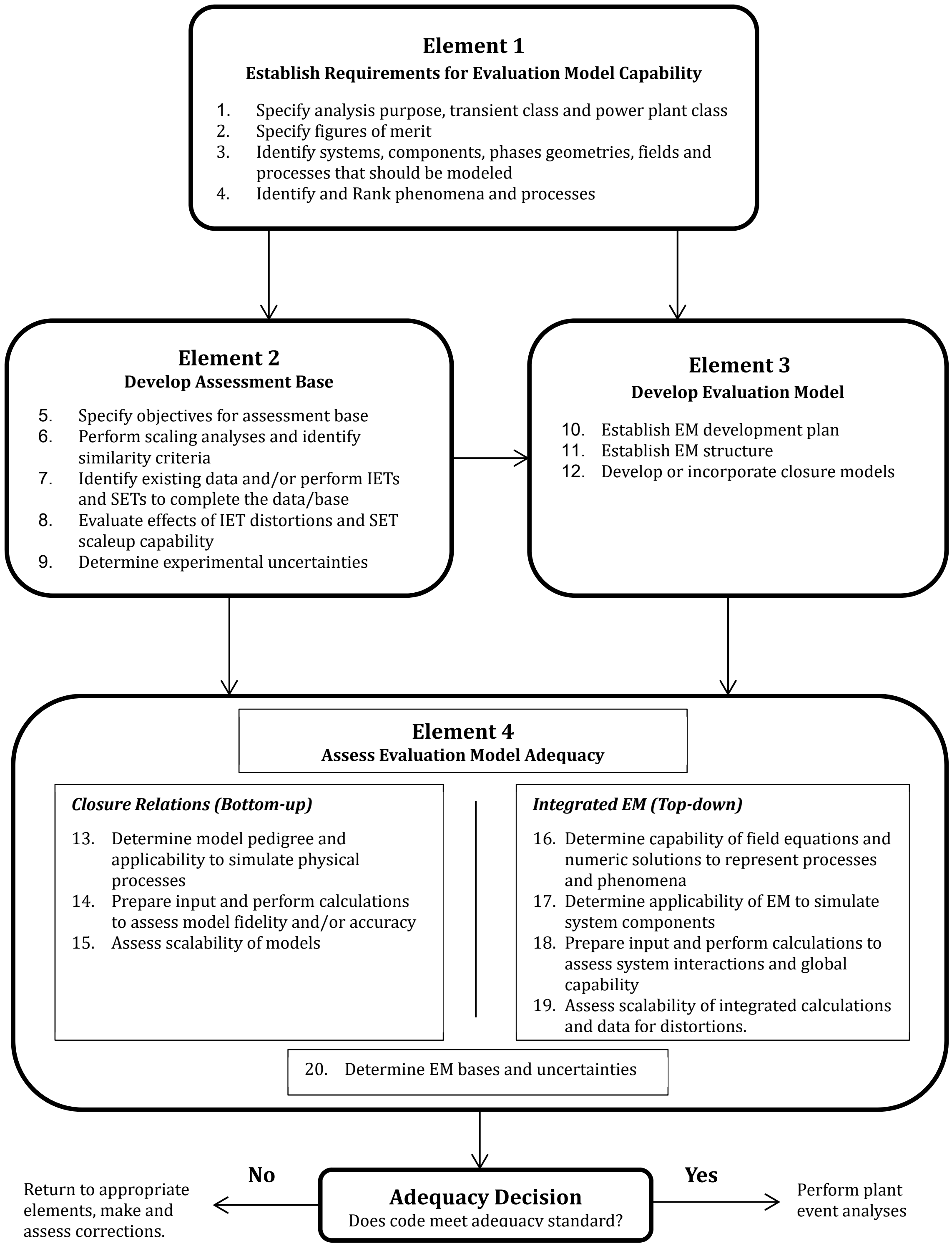

Figure 2-1 Evaluation model development and assessment process

Table 2-1 Evaluation model development and assessment process steps and associated application in the long-term cooling evaluation model

EMDAP Step	Description	EM Section
Element 1, Establish Requirements for Evaluation Model Capability		
1	Specify analysis purpose, transient class and power plant class.	The purpose of the LTC methodology is described in Section 1.1. The NuScale LOCA topical report (Reference 8.2.1) provides an overview of the NPM and a description of the plant operation. This includes the safety systems, the system logic, and operational phases which could occur in the NPM. The regulatory requirements that the methodology is designed to comply with are described in Section 2.2.
2	Specify figures of merit (FOMs).	The NuScale-specific acceptance criteria for LTC are identified in Section 2.3. Section 3.0 describes the NPM long-term cooling PIRT, including FOMs that are used to develop the PIRT.
3	Identify systems, components, phases, geometries, fields, and processes that should be modeled.	Systems, components, phases and processes are identified as a part of the LTC PIRT discussed in Section 3.0.
4	Identify and rank phenomena and processes.	Section 3.0 describes the long-term cooling PIRT.
Element 2, Develop Assessment Base		
5	Specify objectives for assessment base.	Section 3.0 describes the high ranked phenomena identified from the PIRT process and how the phenomena are addressed by NRELAP5 assessment or other approach. Many of the high ranked phenomena were assessed against experimental data as part of the NuScale LOCA EM development; additional assessments against NuScale Integral Systems Test-1 (NIST-1) test data were performed as described in Section 4.0. Other parameters are bounded or treated by a conservative methodology in the LTC analyses.
6	Perform scaling analysis and identify similarity criteria.	A scaling analysis of the LOCA and ECCS has been performed for the NPM based on the NIST-1 facility. The results of the scaling analysis are discussed in the NuScale LOCA topical report (Reference 8.2.1). Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these assessments are adequate for the LTC EM.

EMDAP Step	Description	EM Section
7	Identify existing data and perform integral effects tests (IETs) and separate effects tests (SETs) to complete database.	The NuScale LOCA topical report (Reference 8.2.1) and Section 4.0 of this report provide the results of the NRELAP5 validation against the SETs and IETs.
8	Evaluate effects of IET distortions and SET scaleup capability.	In the NuScale LOCA topical report (Reference 8.2.1), a bottom-up assessment of NRELAP5 closure models and correlations essential to simulate high-ranked PIRT phenomena for LOCA events is presented; this assessment addresses the fidelity of the models and correlations to the appropriate fundamental or SET data. In Reference 8.2.1, a top-down assessment of the NRELAP5 governing equations and numerics is presented. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these assessments are adequate for the LTC evaluation model.
9	Determine experimental uncertainties.	The NuScale LOCA topical report (Reference 8.2.1) and Section 4.0 of this report address experimental uncertainties for NRELAP5 assessments against the SETs and IETs.
Element 3, Develop Evaluation Model		
10	Establish EM development plan.	The NRELAP5 development plan includes programming standards and procedures, quality assurance procedures, and configuration control, which are summarized in Reference 8.2.1.
11	Establish EM structure.	The NuScale LOCA topical report (Reference 8.2.1) provides a summary of NRELAP5 models and correlations. For LTC analysis, the plant model is described in Section 4.0. The LTC methodology for thermal-hydraulic calculations is described in Section 5.0 and the methodology for boron precipitation analysis is described in Section 6.0.
12	Develop or incorporate closure models.	The NuScale LOCA topical report (Reference 8.2.1) provides a summary of NRELAP5 models and correlations. A full description of the closure models and the associated equations used in the LTC evaluation model is provided in the NRELAP5 theory and users manuals (Reference 8.2.8).
Element 4, Assess Evaluation Model Adequacy Closure Relations (Bottom-up)		

EMDAP Step	Description	EM Section
13	Determine model pedigree and applicability to simulate physical processes.	The NuScale LOCA topical report (Reference 8.2.1) includes a bottom-up assessment of important NRELAP5 models/correlations essential to simulate high-ranked PIRT phenomena for LOCA events, including discussion of model pedigree and applicability. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these assessments are adequate for the LTC evaluation model.
14	Prepare input and perform calculations to assess model fidelity and/or accuracy.	Reference 8.2.1 and Section 4.0 summarize the results of comparison of NRELAP5 against the selected SETs and IETs including evaluation of code fidelity and accuracy.
15	Assess scalability of models.	The NuScale LOCA topical report (Reference 8.2.1) includes discussion on scalability of NRELAP5 models and correlations that are essential to simulate high-ranked PIRT phenomena for LOCA events. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these assessments are adequate for the LTC EM.
Element 4, Assess Evaluation Model Adequacy Integrated EM (Top-down)		
16	Determine capability of field equations and numeric solutions to represent processes and phenomena.	NRELAP5 field equations and the numeric solution scheme are discussed in Reference 8.2.1 and evaluated for their applicability to NPM LOCA phenomena. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these assessments are adequate for the LTC EM.
17	Determine applicability of EM to simulate system components.	The applicability of the NuScale LOCA EM to simulate the NPM system and components is demonstrated by assessment of NRELAP5 against NuScale design-specific SETs and IETs as summarized in Reference 8.2.1 and Section 4.0.
18	Prepare input and perform calculations to assess system interactions and global capability.	The NuScale LOCA topical report (Reference 8.2.1) and Section 4.0 summarize the results of assessment of NRELAP5 against NIST-1 IET data.

EMDAP Step	Description	EM Section
19	Assess scalability of integrated calculations and data for distortions.	The NuScale LOCA topical report (Reference 8.2.1) provides an evaluation of scaling distortions between the NIST-1 LOCA IET data and the NPM design. The scalability of the EM to represent NPM LOCA phenomena and processes is presented therein. Considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions in the LTC plant transient calculations, these NuScale LOCA EM assessments are adequate for the LTC EM.
20	Determine EM biases and uncertainties.	For the LTC system transient analyses, suitably conservative input is specified in the plant calculations as described in Section 5.0 and Section 6.0, considering the effects on the appropriate acceptance criteria.

3.0 Phenomena Identification and Ranking Table

3.1 Phenomena Identification and Ranking Table Process

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

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

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

3.2 Figures of Merit

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

- CHFR: The ratio of the heat flux needed to cause CHF phenomena to the actual local heat flux of a fuel rod. Since the core remains covered with water throughout the event, clad does not significantly heat up. Therefore collapsed liquid level with a long-term decreasing trend in fuel cladding temperature is identified as the surrogate FOM for demonstrating acceptable cladding integrity.
- Coolant collapsed level: The coolant level that results if all voids in the vapor-phase coolant are collapsed. If the core remains covered, significant clad heatup is avoided and it is evident that 10 CFR 50.46 criteria of adequate LTC is established.
- Subcriticality: The condition of a nuclear reactor system, in which nuclear fuel no longer sustains a fission chain reaction (that is, the reaction fails to initiate its own repetition, as it would in a reactor's normal operating condition). A reactor becomes subcritical when its fission events fail to release a sufficient number of neutrons to sustain an ongoing series of reactions, possibly as a result of increased neutron leakage or poisons. The scope of this report is limited to the evaluation of fission product decay heat loads (i.e. subcritical core configuration). Evaluation of extended cooldown loss of shutdown condition is not within the scope of this report.

3.3 Highly Ranked Phenomena

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

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

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

As discussed in the LOCA evaluation model, NRELAP5 is NuScale's system thermal-hydraulics code used to calculate the NPM system response during the LOCA long-term cooling event progression. The NRELAP5 code has been assessed against several separate effects and integral effects tests as part of the code development and development of the NuScale LOCA evaluation model to demonstrate the capability to simulate the NPM response to LOCA events (Reference 8.2.1).

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

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4.0 NRELAP5 Applicability to Long-Term Cooling Analysis

The LTC EM is developed from the NuScale LOCA EM described in Reference 8.2.1. Specifically, the LTC model is derived from the coarser nodalized LOCA model described in Section 9.6.1 of Reference 8.2.1. The coarser LOCA model is selected to improve calculation performance over long term, quasi-steady state conditions where the fidelity of finer model nodalization is not required. This section describes the LTC model, how the LTC EM was developed and the differences between the LTC EM and the NuScale LOCA EM described in Reference 8.2.1. This section also validates the LTC EM for use in LTC assessments by benchmarking to the NIST-1 facility test results.

4.1 Summary of the Long-Term Cooling Model

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

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

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4.2 NRELAP5 Validation and Assessments for Long-Term Cooling

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

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

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4.2.2 NIST-1 Facility NRELAP5 Model

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4.2.3 Integral Assessment of NIST-1 HP-19a

4.2.3.1 Purpose of Assessment

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

4.2.3.2 HP-19a Test Progression

The test consists of the following:

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4.2.3.3 NRELAP5 Prediction of HP-19a

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

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

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

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

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

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

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

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

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

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

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

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

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

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

4.2.4 Integral Assessment of NIST HP-19b

4.2.4.1 Purpose of Assessment

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

4.2.4.2 HP-19b Test Progression

The test consists of the following:

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4.2.4.3 NRELAP5 Prediction of HP-19b

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

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

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

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

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

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

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

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

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

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

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

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

4.2.5 Conclusions from Integral Test Assessments

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

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

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Figure 4-17 Loss-of-coolant accident evaluation model nodalization consistency comparison: pressurizer pressure through 1 hour

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Figure 4-18 Loss-of-coolant accident evaluation model nodalization consistency comparison: pressurizer pressure

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Figure 4-19 Loss-of-coolant accident evaluation model nodalization consistency comparison: containment pressure through 1 hour

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Figure 4-20 Loss-of-coolant accident evaluation model nodalization consistency comparison: containment pressure

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Figure 4-21 Loss-of-coolant accident evaluation model nodalization consistency comparison: riser collapsed liquid level relative to the top of active fuel through 1 hour

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

Figure 4-22 Loss-of-coolant accident evaluation model nodalization consistency comparison: riser collapsed liquid level relative to the top of active fuel

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Figure 4-23 Loss-of-coolant accident evaluation model nodalization consistency comparison: pressurizer level

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Figure 4-24 Loss-of-coolant accident evaluation model nodalization consistency comparison: containment collapsed liquid level relative to the top of active fuel

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Figure 4-25 Loss-of-coolant accident evaluation model nodalization consistency comparison: core inlet temperature

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Figure 4-26 Loss-of-coolant accident evaluation model nodalization consistency comparison: core outlet temperature

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

Figure 4-27 Loss-of-coolant accident evaluation model nodalization consistency comparison - injection line break: riser collapsed liquid level relative to the top of active fuel

5.0 Long-Term Cooling Methodology and Evaluation

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

The methodology to address maintaining a coolable geometry by precluding boron precipitation is described in Section 6.0. The results presented in Section 5.6 for the minimum RCS temperatures will be considered in the Section 6.0 analyses for limiting boron solubility conditions.

5.1 Long-Term Cooling Heat Removal Methodology

For the LTC phase of heat removal the acceptance criteria addressed are: (1) collapsed liquid level is maintained above the active fuel, and (2) fuel cladding temperature is maintained at an acceptable level.

These criteria are demonstrated with the basic established conditions below:

- Maximum temperature: Achieved by minimum cooldown and demonstrates that the fuel cladding temperature is maintained at an acceptable level.
- Minimum temperature: Achieved by maximum cooldown and demonstrates that the collapsed liquid level is maintained above the active fuel and that the minimum temperature supports the criteria that no boron precipitation occurs during the LTC evaluation period.
- Minimum level: Achieved by maximum cooldown with minimum initial RCS inventory and maximum inventory loss to the CNV, and demonstrates that the collapsed liquid level is maintained above the active fuel. Due to the conservative assumption that all boron is concentrated in the core and riser regions, these conditions also support the criterion that no boron precipitation occurs during the LTC evaluation period.

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

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

transition cases also include consideration for SGTF to address inventory loss prior to isolation of the SG.

Analysis of LTC only credits long term decay heat removal through the ECCS for the purpose of demonstrating that the top of active fuel remains covered. LTC conditions are also evaluated with DHRS enabled to demonstrate minimum temperature requirements are met for boron precipitation concerns. The sequence of events leading to long-term ECCS cooling are described below.

1. ECCS valves open. This may occur after short term DHRS cooling in the event of a non-LOCA transient in conjunction with a loss of normal AC power.

2. RCS level begins to drop while CNV level rises.

3. Minimum level in the RCS occurs. The minimum level reached occurs during the short term LOCA phase.

4. Condensation in the CNV increases CNV level.

5. Recirculation flow from the CNV to the RPV through the RRV is established.

6. Long term levels stabilize, assuming the pool boundary condition is constant. The stabilization of the long term levels begins the long-term cooling phase of the event.

5.2 Events Evaluated for Long Term Cooling

The ECCS is designed to operate following a LOCA event, after the inadvertent opening of an RPV valve (IORV), or if power to the ECCS valve actuators is lost and the system has depressurized to the IAB release pressure. Therefore, a series of both LOCA and non-LOCA events are identified to evaluate long term ECCS cooling acceptability. For non-LOCA events, emphasis is placed on events which reduce reactor inventory, such as a small line break outside containment or a steam generator tube failure. A DHRS cooldown event and loss of feedwater event were also evaluated to confirm that events which reduce primary inventory are limiting. The following events were evaluated as part of the LTC evaluation:

- LOCA Spectrum – The full LOCA break spectrum was evaluated. This includes break locations at the discharge line, injection line, high point vent line, and pressurizer spray line, {{ }}2(a),(c)

  The most limiting collapsed riser levels occur for small LOCA breaks immediately after ECCS actuation. Since this time range is covered by LOCA methodology, minimum level during this time is not considered limiting for LTC analysis. Instead, limiting minimum level is determined in the hours following ECCS actuation during the characteristic level depression seen in this time range and consistent with the definition of the NPM conditions for LTC.

- IORV – The inadvertent opening of an RVV or RRV was considered in LTC analysis. The RSV was not evaluated as the valve size is bounded between the RVV and LOCA steam space breaks.

- Steam Generator Tube Failure – The SGTF transient was included in LTC analysis to evaluate the impact of RCS inventory lost to the secondary system. The break was modeled at the top of the steam generator in order to minimize return flow into the RPV.

- DHRS Cooldown – These events generically evaluated the transition from DHRS to ECCS cooling either if DC power is lost and IAB release pressure is reached, or 24 hours after losing AC power. These transients were initiated by a loss of AC and DC power at time zero. For the 24 hour transition cases, the ECCS logic was modified to actuate after 24 hours rather than on IAB release pressure.

  A subset of these cases is also performed at an initial PZR level of 20% to provide a bounding evaluation of small line breaks outside containment. This level corresponds to containment system isolation on the low-low pressurizer level signal, which would isolate the break and prevent further inventory loss.

- Loss of Feedwater – The LOFW non-LOCA event was selected to demonstrate that the module temperature and pressure response prior to reactor trip has little influence on long term conditions. This event was modeled by setting feedwater flow to zero at event initiation. A loss of AC power is assumed at reactor trip which allows ECCS actuation after 24 hours.

5.3 Long Term Cooling Analysis Assumptions

5.3.1 Electric Power Availability

For LTC analysis, availability of electric power is considered for its impact on ECCS actuation timing. For non-LOCA events, ECCS actuation can only occur at the IAB release pressure if DC power to the valve actuators is lost, or 24 hours after losing AC power. The following scenarios are considered:

- Loss of AC and DC power at time zero was evaluated for all cases unless specified otherwise. Losing DC power at time zero causes ECCS actuation once the IAB release pressure is reached. This timing is earlier relative to actuation on the RPV or CNV level signals for LOCA or the 24 hour timer for non-LOCA, and earlier actuation is limiting for minimum collapsed level.

- Loss of AC power was evaluated at time zero and at reactor trip for some DHRS cooldown and LOFW cases to confirm they are not limiting. These cases actuate ECCS 24 hours after losing AC power.

5.3.2 Single Failure Evaluation

The failure of one ECCS division (i.e., one RVV and one RRV) was considered when evaluating sensitivity to minimum ECCS capacity. When maximum ECCS capacity was evaluated, all ECCS valves were assumed to open. Single failures in the secondary system were not considered as these have little influence on the long term results.

5.3.3 Multi-module Consideration

In the NuScale plant design, up to twelve modules may be operating. The safety systems credited for mitigation of the design basis events are module-specific except for the shared reactor pool portion of the UHS. Long term cooling analysis evaluated a single module response to demonstrate that the acceptance criteria are met. The LTC analyses considered a range of reactor pool boundary conditions to sufficiently address the effects of one or more modules, up to all twelve modules, transferring decay heat into the reactor pool.

5.3.4 Long Term Cooling Evaluation Period

LTC analysis is limited to three days. This timeframe is considered acceptable because: (1) the most severe conditions will have been captured within the 72 hour window analyzed, and any conditions that could reasonably be expected to occur beyond this time period are thus bounded by the 72 hour calculation, and (2) after 72 hours, operator actions can be credited.

5.4 Initial Conditions and Biases

As stated in Section 5.1, three scenarios are defined to evaluate LTC acceptance criteria: minimum level, minimum temperature, and maximum temperature. Specific key conditions for these scenarios are defined in Table 5-1. Some cases feature minor variations from these conditions (defined on a case specific basis in the following sections) to evaluate parameter sensitivity.

Table 5-1 Default scenario initial conditions and biases

Scenario	Reactor Power (%)(1)	Decay Heat (multiplier)	RCS Avg. T. (°F)	RCS P. (psia)	PZR Level (%)	Pool T. (°F)	Pool Level (ft)	Non-condensable Gas (lbm)	ECCS Capacity (Area and Cv)	Expansion Factor (Y)	Single Failures	DHRS Enabled for LOCA(2)
Minimum Level	102	1.2 (LOCA) 1.0 (nonLOCA)	555	1780	52	65	69	0	minimum	0.7	RVV/ RRV	false
Minimum Temperature	102	0.8	535	1780	68	65	69	0	maximum	1.0	none	true
Maximum Temperature	102	1.2 (LOCA) 1.0 (nonLOCA)	555	1920	52	210	55	~131	minimum	0.7	RVV/ RRV	false

(1) Lower power, down to 13% initial power, is considered as a separate sensitivity for the minimum temperature cases.
(2) DHRS is always enabled for all non-LOCA events.

5.5 Sensitivity Considerations

The parameters considered as part of the sensitivity analysis are based on the findings in the PIRT from Section 3.0 and are conservatively applied in Section 5.6 depending on the requirements of the specific scenario. These parameters are as follows:

- decay heat, ranging from no decay heat to 120 percent of nominal
- reactor pool temperature, ranging from 65 degrees F to 210 degrees F
- reactor pool level, ranging from 55 feet to 69 feet
- sensitivity to 45 feet is evaluated to address the possible boil-off of pool liquid due to long-term cooling from all twelve modules providing decay heat to the pool
- non-condensable gas effect
- pressurizer level, down to 20 percent of nominal
- Expansion factor used to account for compressible flow through RVVs
- DHRS operation
- reactor pool temperature is modeled as constant through assuming a very large pool volume

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

5.6 Demonstration of Limiting Results

Three scenarios were established to determine the limiting conditions that could develop during the LTC phase. Results are demonstrated for the LOCA injection line break and the SGTF event as these were found to be limiting for collapsed liquid level. The IORV, DHRS cooldown, and LOFW events were non-limiting for collapsed level and core temperatures, and detailed results are not discussed. The following cases are presented:

- maximum temperature with injection line break
- minimum temperature with injection line break
- minimum level with injection line break
- minimum level with SGTF

A more detailed description of each case is provided in Section 5.6.1 through Section 5.6.4. The transient response is simulated for 12.5 hours following event initiation. This time range is sufficient to evaluate the influence of each initiating event on the mid-term LTC response. During this time, minimum collapsed level occurs coincident to a local peak in differential pressure between the RPV and CNV. After passing this peak, the differential pressure between the vessels follows a continually decreasing trend as energy is removed from the module. As the differential pressure decreases, the required static head in containment to drive recirculation flow through the RRVs is also reduced, allowing inventory to accumulate back inside the RPV. Riser level continues to recover toward long-term equilibrium which is a function of decay heat, heat transfer from containment to the UHS, total ECCS capacity, and RRV elevation.

Decay heat continually decreases overtime, reducing the pressure differential between the CNV and RPV over the long-term. Accumulation of non-condensible gases overtime is not modeled. Instead, depending on the conservative direction for the specific evaluation, the LTC analyses assume either zero non-condensible gas is present or assume all non-condensible gas dissolved in the RCS, in the pressurizer vapor space, in containment, in the control rod drive mechanisms, and in the RCS degasification line are instantly transported to the containment vessel at transient initiation. Additionally, the reactor pool temperature is fixed during the transient calculation. Therefore, heat transfer from containment to the UHS is only a function of temperature inside containment. Finally, total ECCS capacity and RRV elevation remain fixed during the transient calculation.

It is concluded that once minimum level is reached and level recovery begins, there is no evolving mechanism which would cause the increasing trend in collapsed level to reverse over the long term. System pressures and temperatures also follow a continually decreasing trend with decay heat over time. Since transient minimum collapsed level has been captured and there are no mechanisms to change the cooldown trajectory, explicit transient calculations past 12.5 hours are not required.

Instead, a following state-point analysis is performed to save calculation time. This is done by taking module conditions at the end of 12.5 hours, setting core power to a constant value corresponding to decay heat levels at 72 hours, and then allowing system conditions to converge to equilibrium. The state-point analysis results provide final module conditions without needing to explicitly model the quasi-equilibrium, long term response as decay heat slowly decreases to the 72 hour value. The primary purpose of these calculations is to find the limiting minimum core inlet temperature which occurs at 72 hours for boron precipitation analysis. Long term maximum cladding temperature and collapsed level are also evaluated to confirm that acceptance criteria remain satisfied.

Table 5-2, below verifies that the state-point analysis estimating the quasi-steady cooldown is appropriate by showing that the predicted conditions at 72 hours for the state-point approximation is within 1 psi for system pressures, 1 degree F for system temperatures, and within 0.1 feet for the collapsed liquid level above the core for the 25% and 100% injection line break cases. The 25% power scenario results confirm that the state-point results converge to the transient 72 hour solution regardless of differing initial conditions. Final module conditions at 72 hours are discussed in Section 5.6.5. These calculations demonstrate the trend in both the RCS temperatures (including fuel cladding temperatures) and collapsed liquid level in the riser.

Table 5-2 Maximum temperature 100% injection line break transient and state-point results at 72 hours

Case	Core Inlet T (°F)	Lower RPV P (psia)	Collapsed Riser Level (ft)	CNV T at RRV (°F)	Lower CNV P (psia)
100% power transient	270	50	9.0	264	51
100% power state point	271	51	9.0	264	52
25% power transient	245	37	9.9	244	37
25% power state point (initial conditions from 25% power transient)	245	37	9.9	245	37
25% power state point (initial conditions from 100% power transient)	245	37	9.8	245	37

5.6.1 Maximum Temperature

The maximum temperature scenario evaluates the combined impact of all conditions that result in a slower cooldown rate in order to maximize cladding temperature. In general, conditions are biased to maximize RCS energy and minimize heat transfer to the UHS. Specific conditions include:

- minimum ECCS capacity, including {{ }}2(a),(c), and the single failure of one RVV and RRV to open
- DHRS operation disabled
- decay heat with 1.2 multiplier
- RCS conditions biased to maximize initial energy and minimize initial inventory
- non-condensable gas modeled inside containment
- a maximum reactor pool temperature of 210 °F and minimum reactor pool level of 55 feet

Results are presented for the LOCA 100% and 5% injection line breaks. Figure 5-1 shows continually decreasing core inlet temperature post-ECCS actuation. Figure 5-2 shows that the maximum temperature scenario does not challenge cladding temperature. The long-term maximum cladding temperature is seen to decrease to a level well below those seen in the short term. The cladding temperature follows a decreasing trend with saturation temperature. Figure 5-3 and Figure 5-4 demonstrate that the core remains covered at all times, and that long term collapsed level is established well above the top of active fuel. The 100% IL break case shows a long term level which is lower than the 5% IL break. Due to the conservative modeling of the IL break at the same axial elevation of the RRV, the 100% break is sufficiently large to establish a small liquid recirculation path from the riser to containment in the long term. This results in a conservative prediction for long term collapsed riser level and core inlet temperature for the 100% IL break relative to the 5% IL break. Figure 5-5 through Figure 5-8 show that long term pressure follows a decreasing trend, with system pressures converging to the same value for the 100% IL break and 5% IL break cases. Figure 5-9 demonstrates that stable ECCS cooling is established through the LTC phase.

The results demonstrate that cladding temperature follows a long term decreasing trend and that the core remains covered at all times, and all acceptance criteria are satisfied for the maximum temperature scenario. State point conditions at 72 hours are presented in Section 5.6.5 for the 100% IL break case.

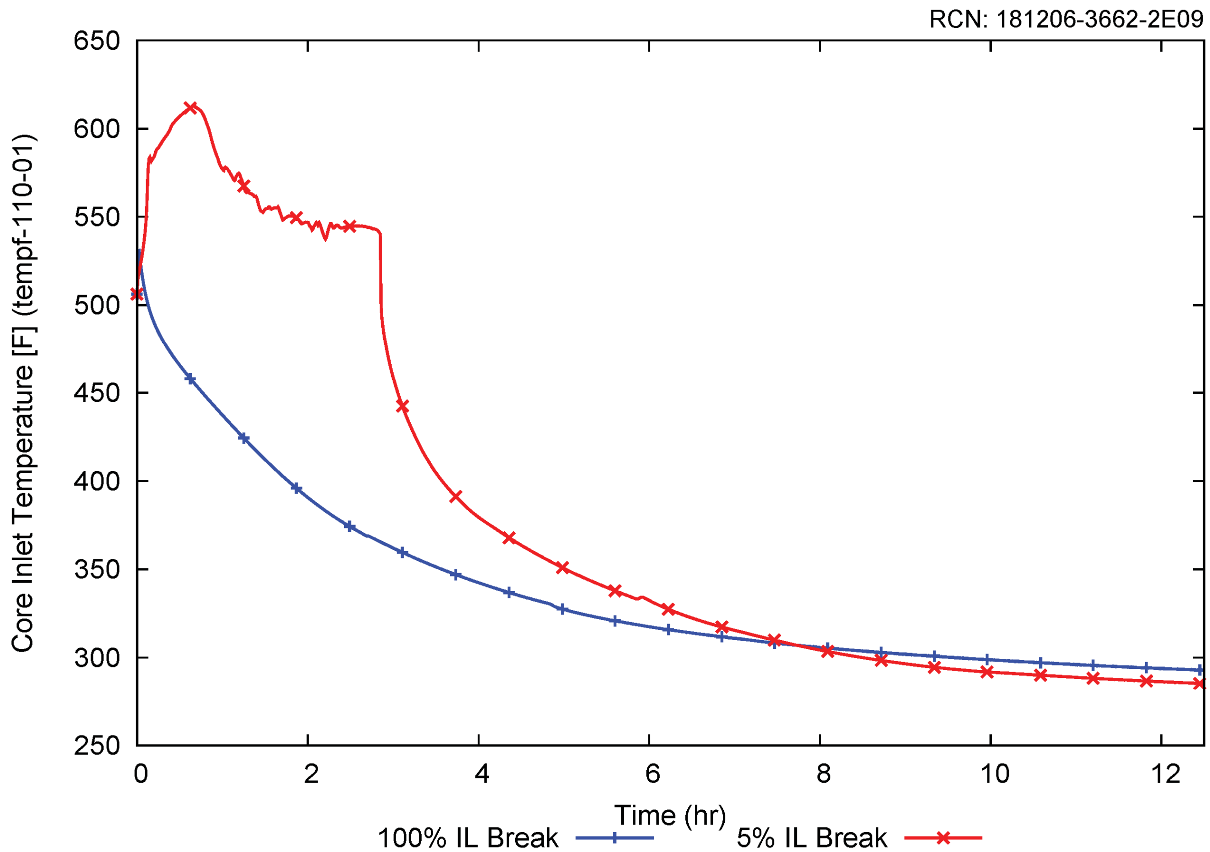

Figure 5-1 Maximum temperature injection line break: core inlet temperature

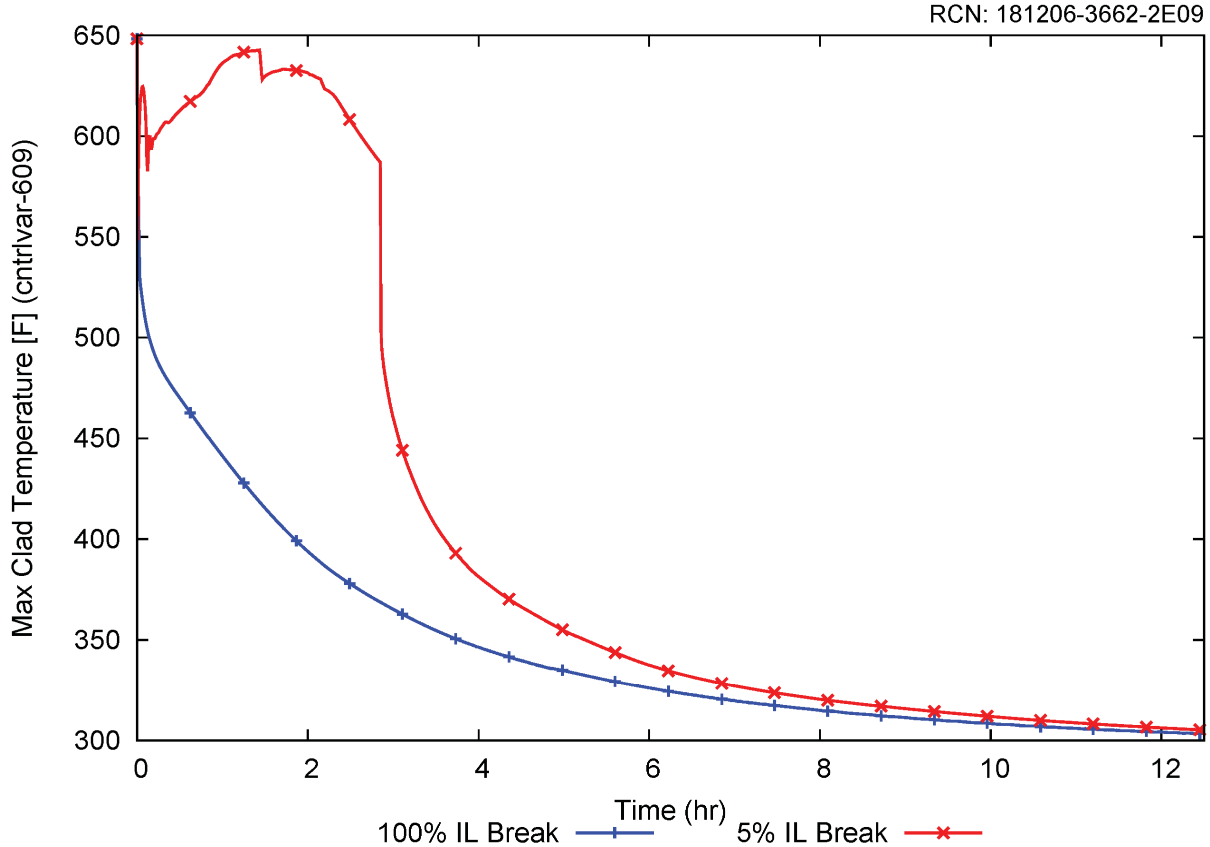

Figure 5-2 Maximum temperature injection line break: maximum cladding temperature

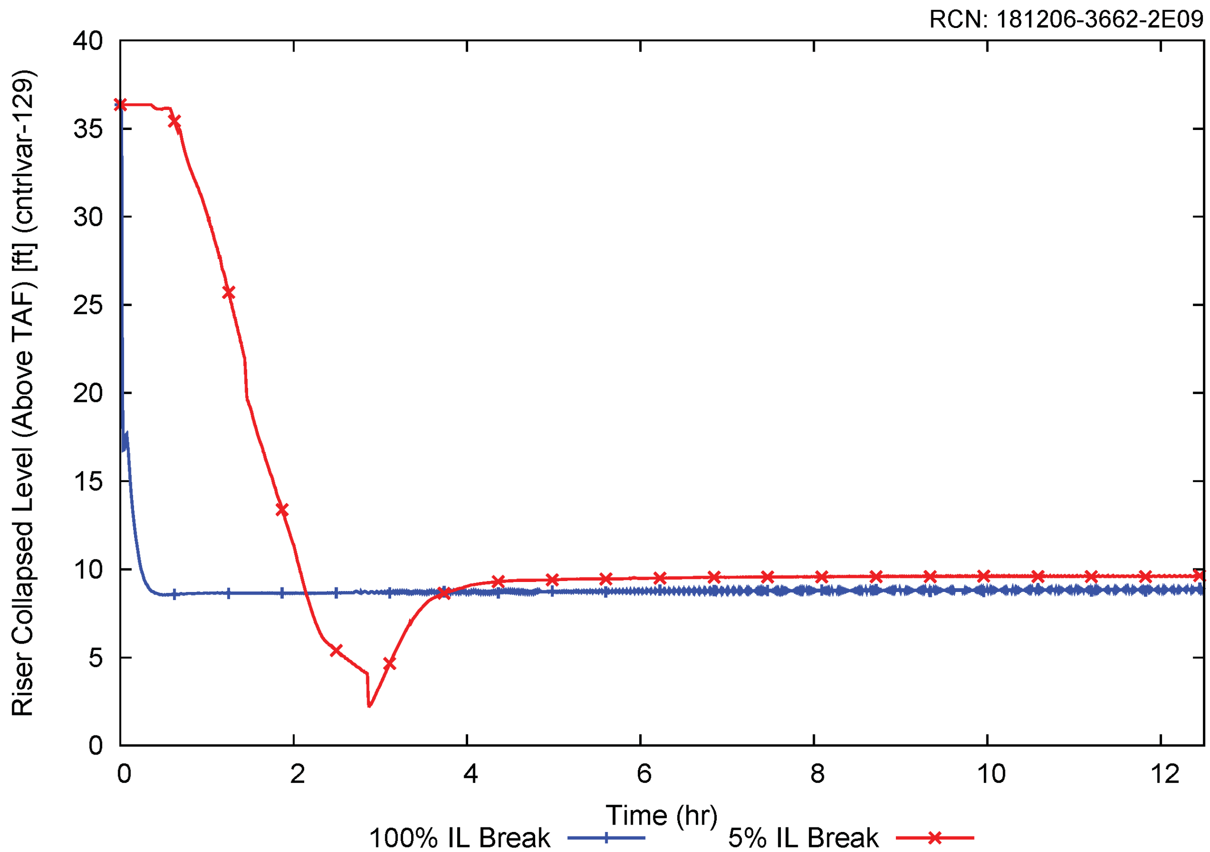

Figure 5-3 Maximum temperature injection line break: riser collapsed liquid level above top of active fuel

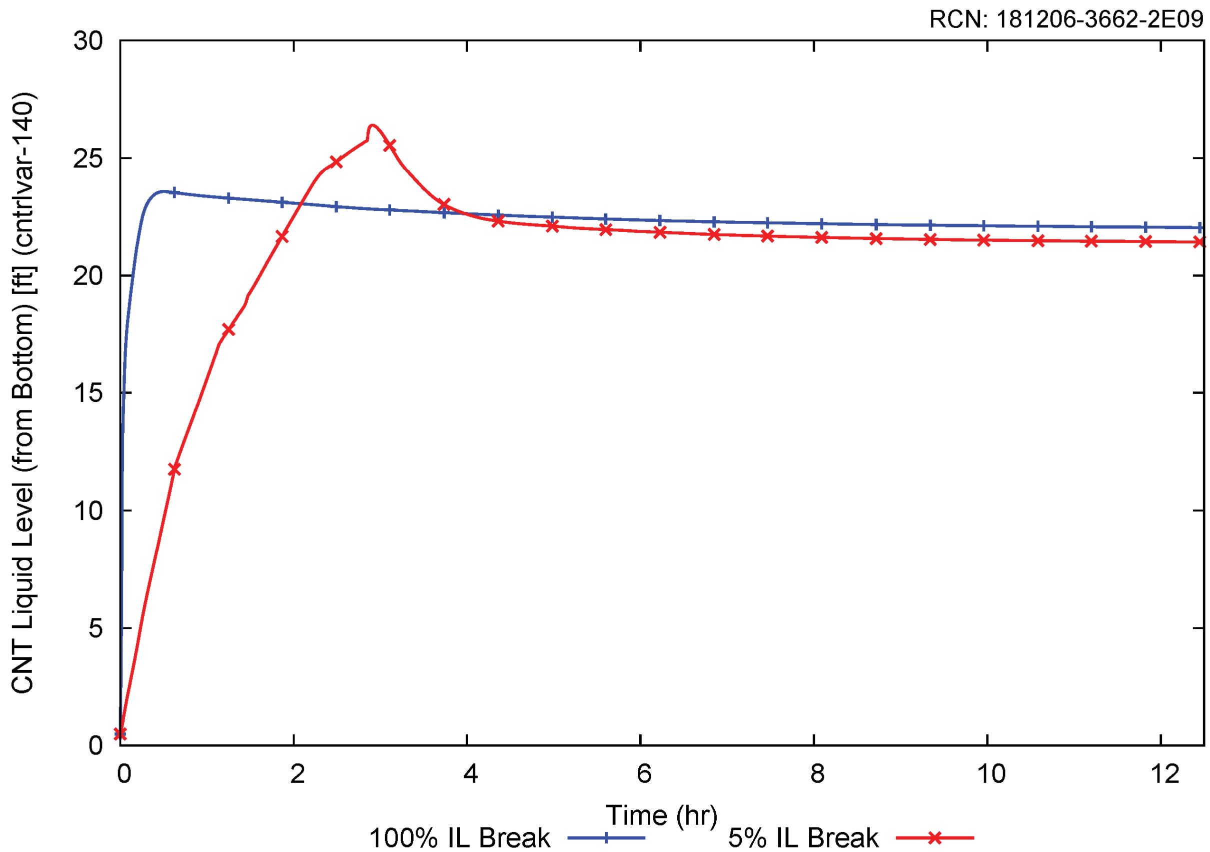

Figure 5-4 Maximum temperature injection line break: containment liquid level above bottom

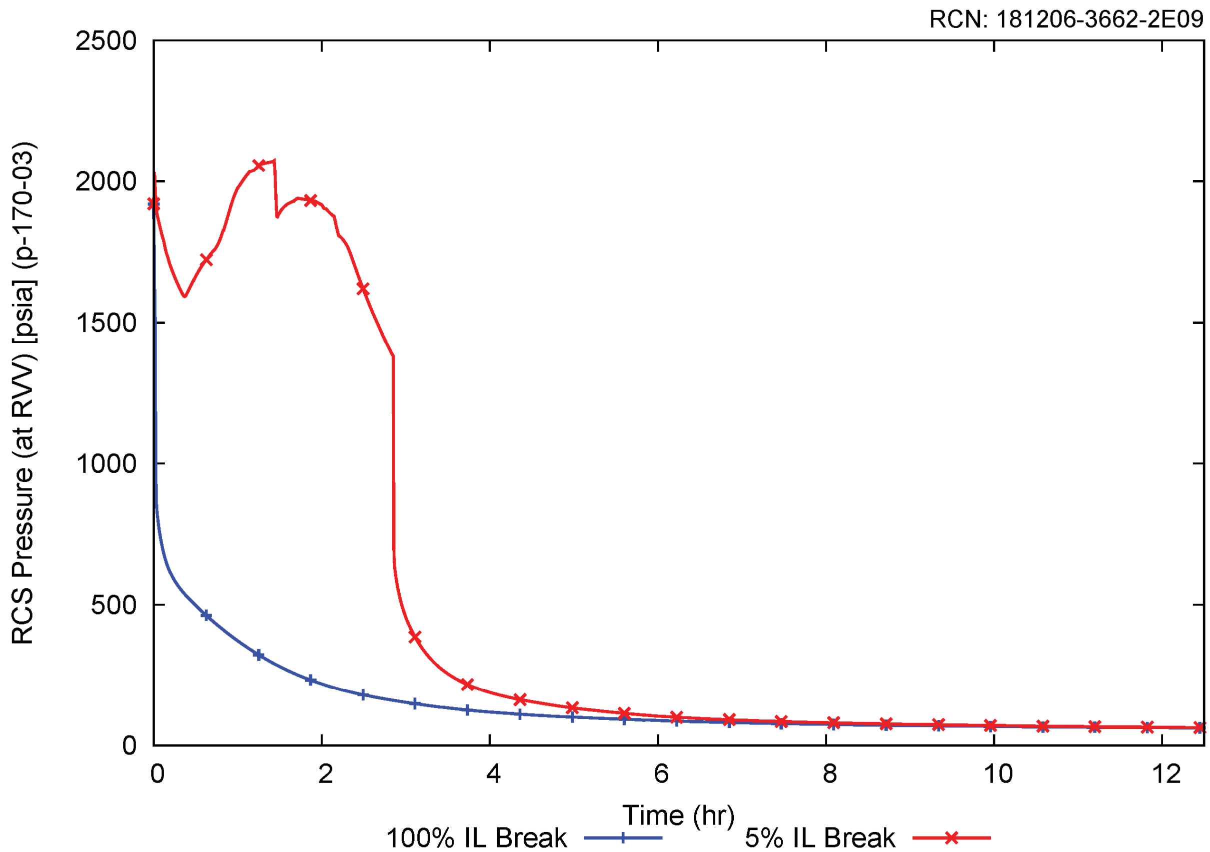

Figure 5-5 Maximum temperature injection line break: RCS pressure at RVV

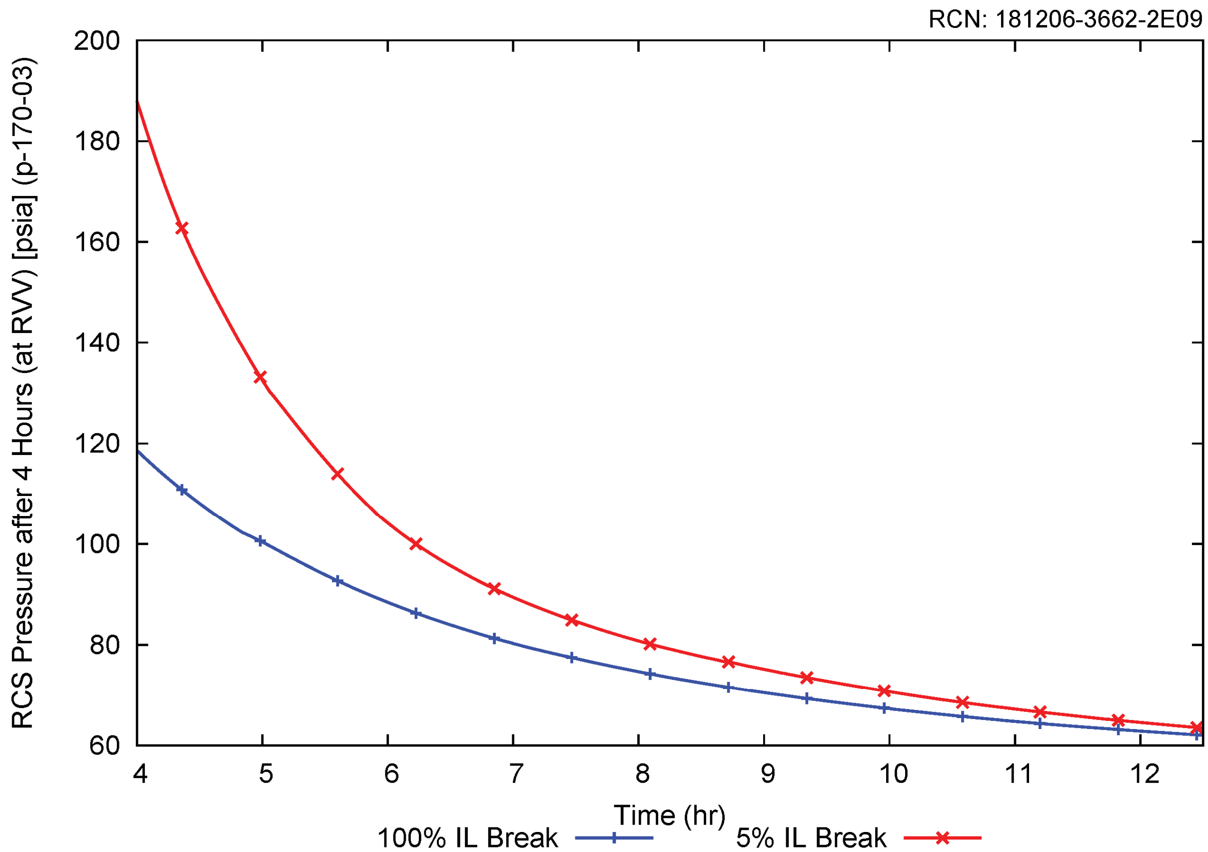

Figure 5-6 Maximum temperature injection line break: RCS pressure at RVV after 4 hours

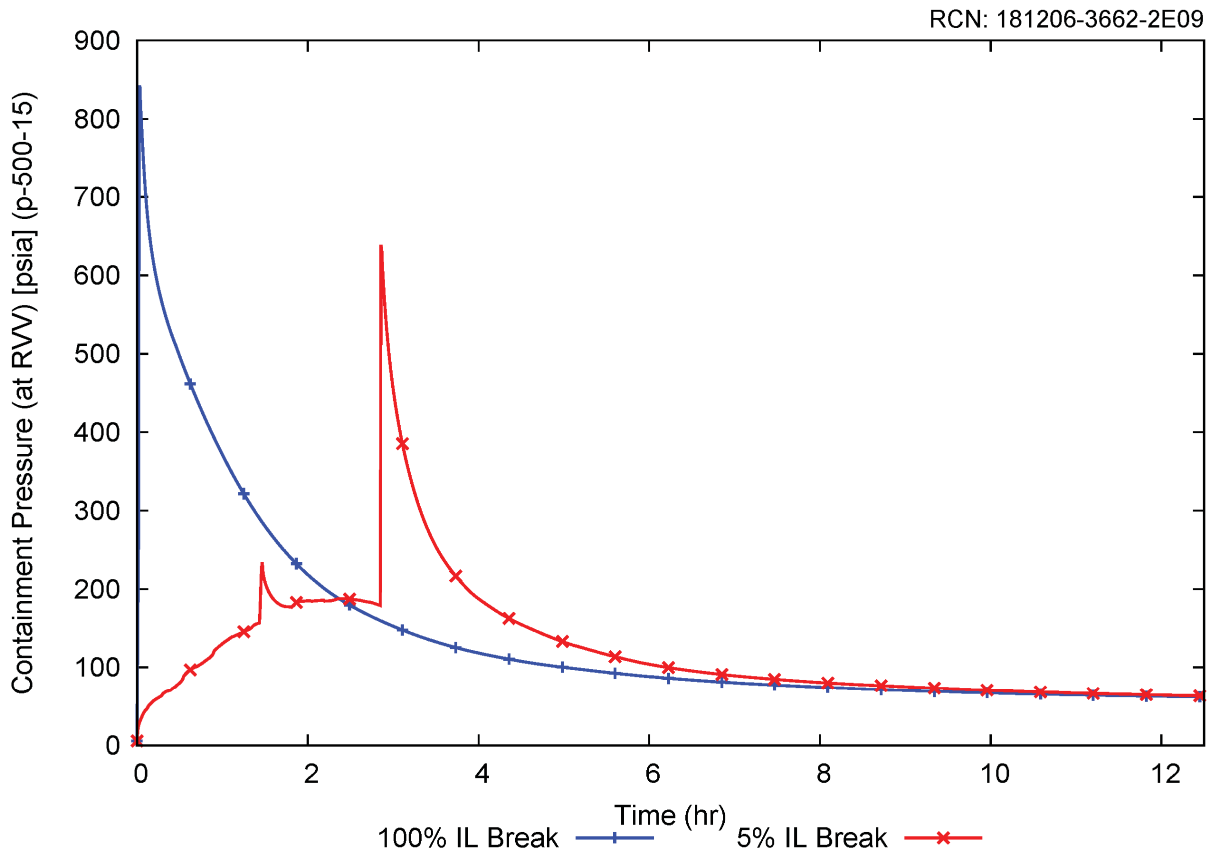

Figure 5-7   Maximum temperature injection line break: containment pressure at RVV

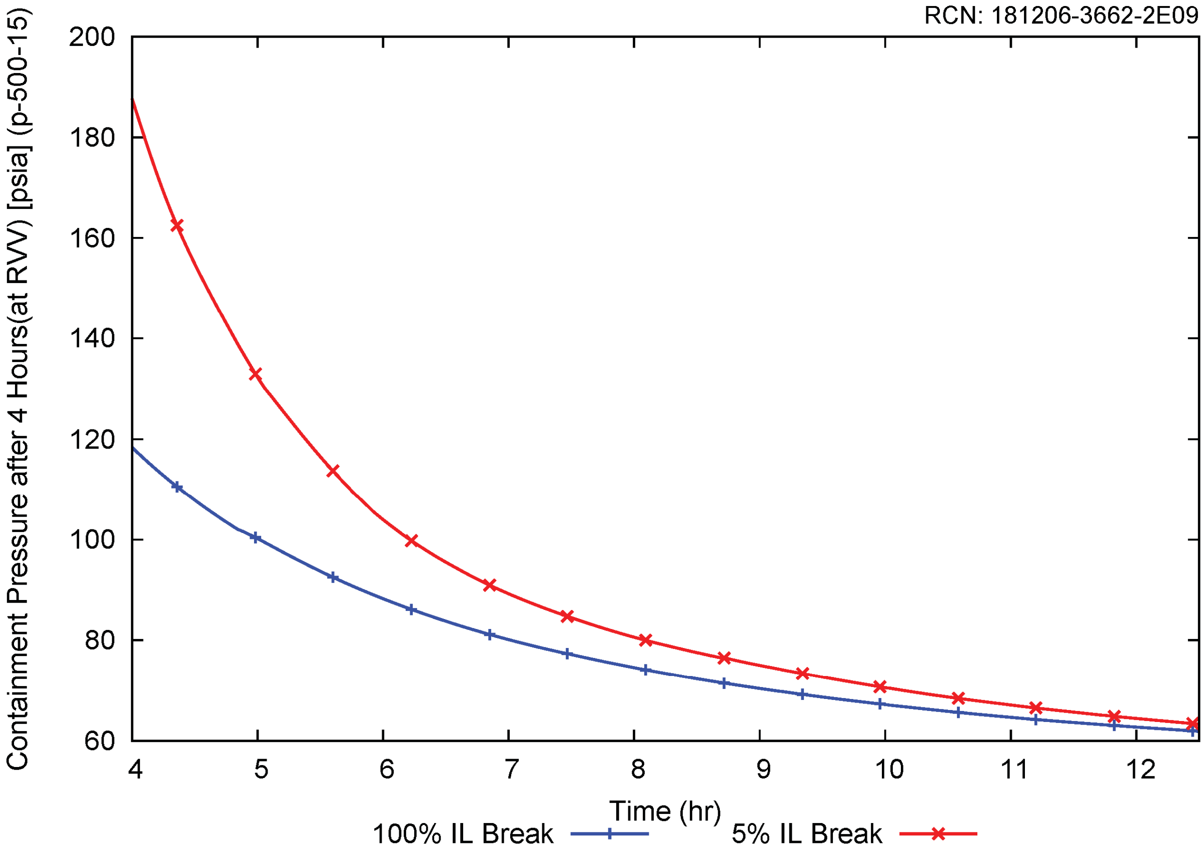

Figure 5-8 Maximum temperature injection line break: containment pressure at RVV after 4 hours

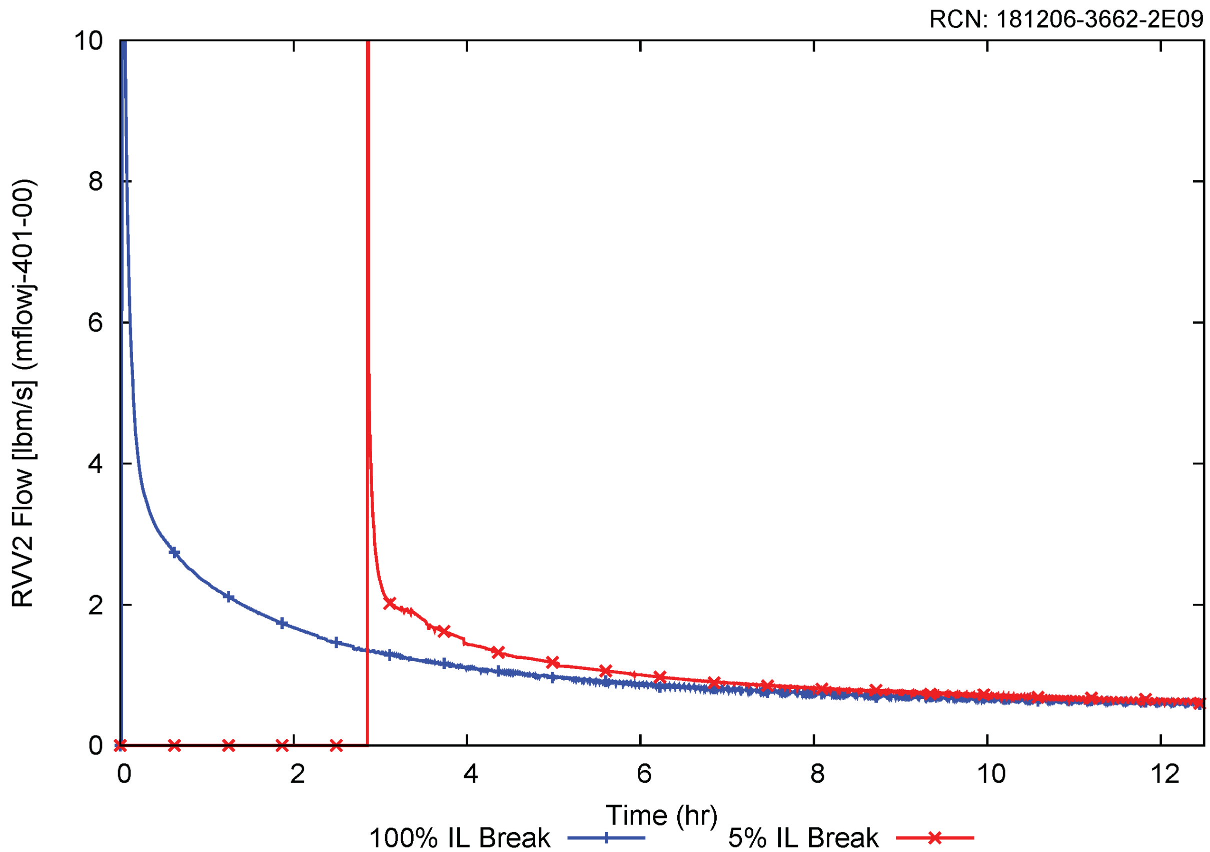

Figure 5-9 Maximum temperature injection line break: RVV2 flow

5.6.2 Minimum Temperature

The minimum temperature scenario evaluates the combined impact of all conditions that result in a faster cooldown rate in order to minimize core liquid temperature and demonstrate that boron precipitation is precluded. In general, conditions are biased to minimize RCS energy and maximize heat transfer to the UHS. Specific conditions include:

- maximum ECCS capacity, including maximum valve area and flow coefficients, no expansion factor penalty applied to RVVs, and no single failure
- DHRS operation is enabled for entire transient duration
- decay heat with 0.8 multiplier RCS conditions are biased to minimize initial energy
- RCS inventory is maximized which increases total boron mass in the system
- zero non-condensable gas is modeled
- a minimum reactor pool temperature of 65 °F and maximum reactor pool level of 69 feet

Results are presented for the LOCA 100% and 5% injection line breaks. Figure 5-10 and Figure 5-11 show that temperatures rapidly drop over the first few hours, then continue on a gradually decreasing trend. Figure 5-12 and Figure 5-13 show that after an initial decrease, long term level is quickly established after approximately four hours. Figure 5-14 through Figure 5-17 show that the RCS and CNV pressures become sub-atmospheric over the long term. Figure 5-18 demonstrates that stable ECCS cooling is established through the LTC phase. System temperatures, pressures, level, and ECCS flow are shown to converge toward the same value regardless of initiating break size. Collapsed level and core inlet temperature remain sufficiently high to preclude boron precipitation, and all acceptance criteria are satisfied for the minimum temperature scenario. State point conditions at 72 hours are presented in Section 5.6.5 for the 100% IL break case.

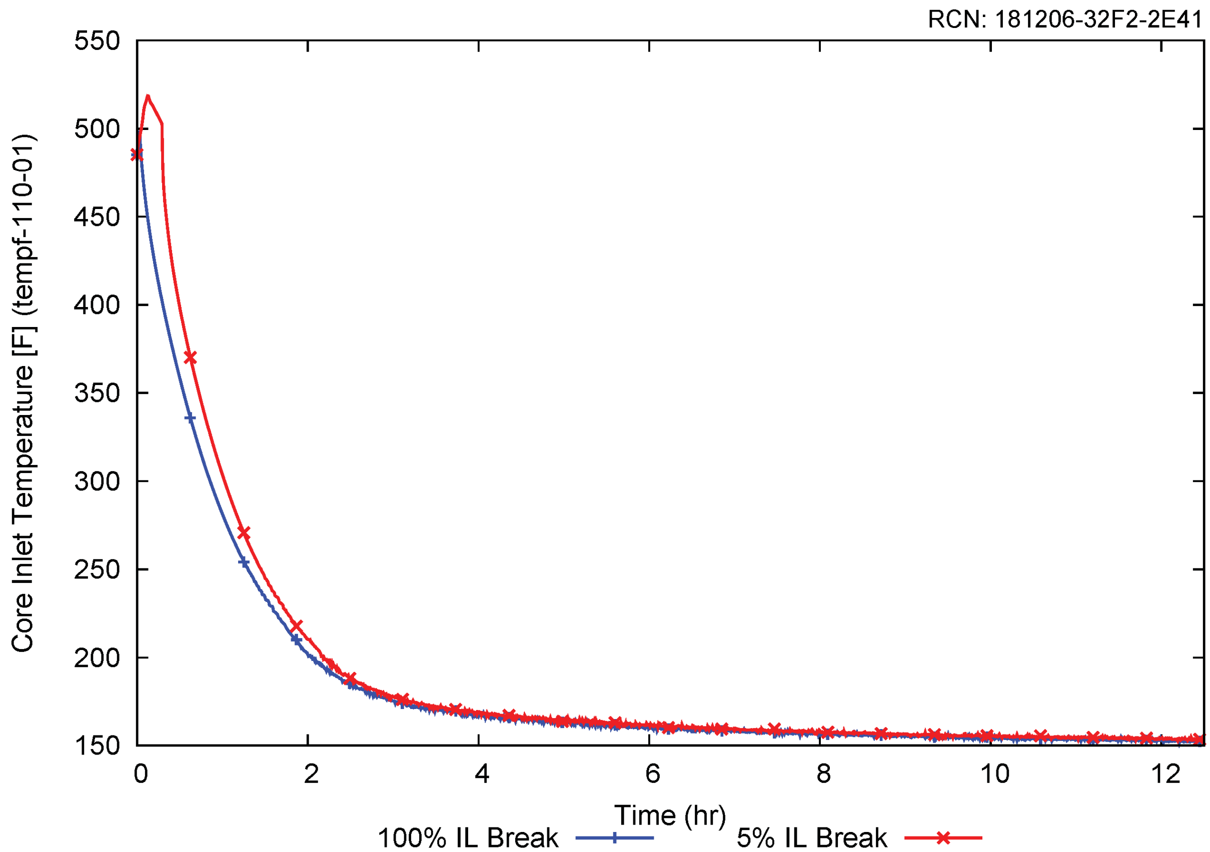

Figure 5-10 Minimum temperature injection line break: core inlet temperature

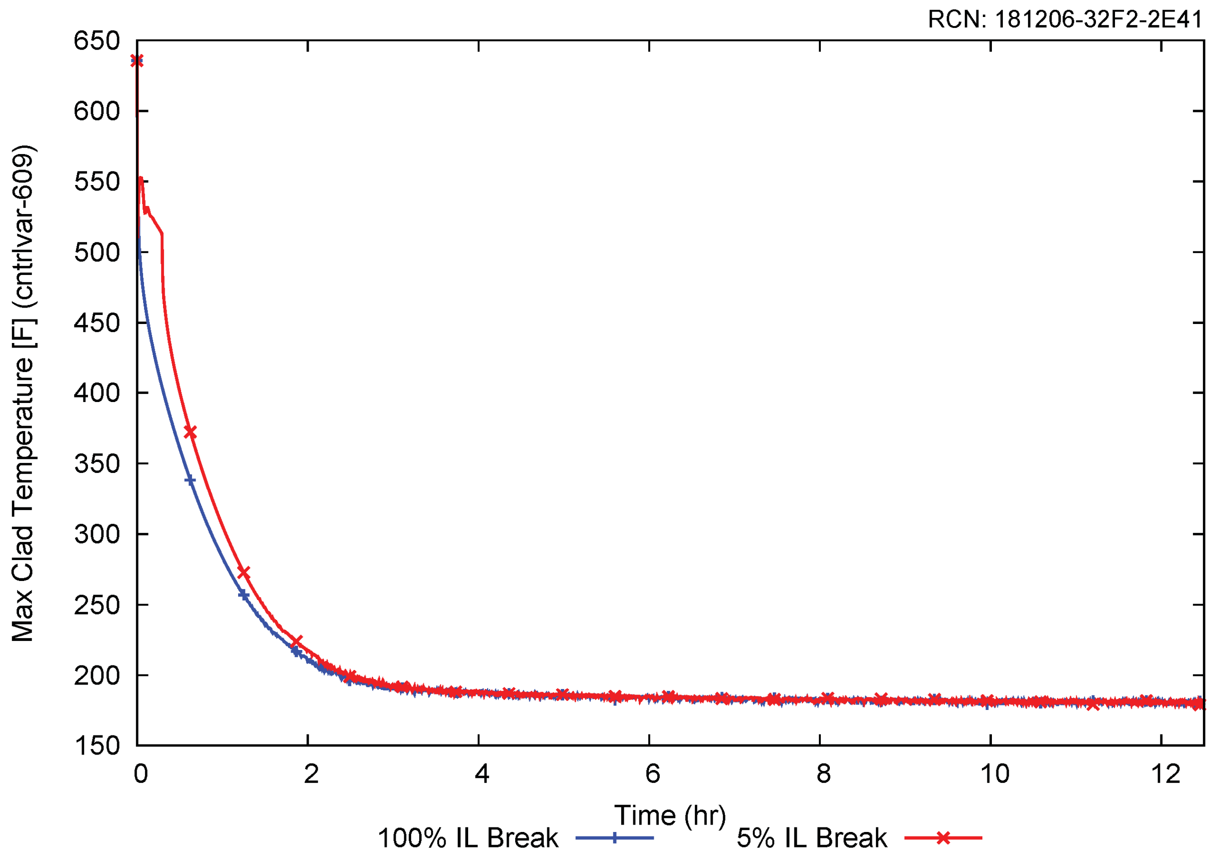

Figure 5-11 Minimum temperature injection line break: maximum cladding temperature

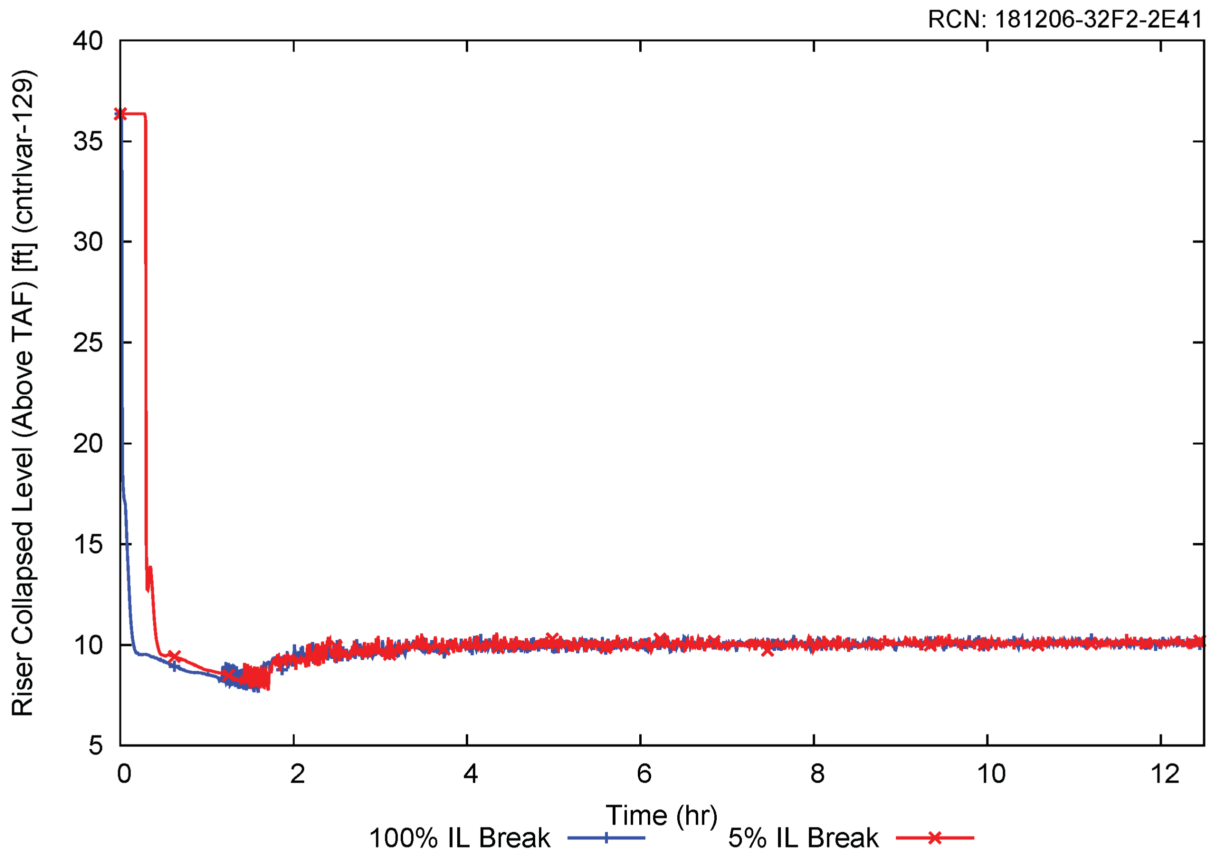

Figure 5-12 Minimum temperature injection line break: riser collapsed liquid level above top of active fuel

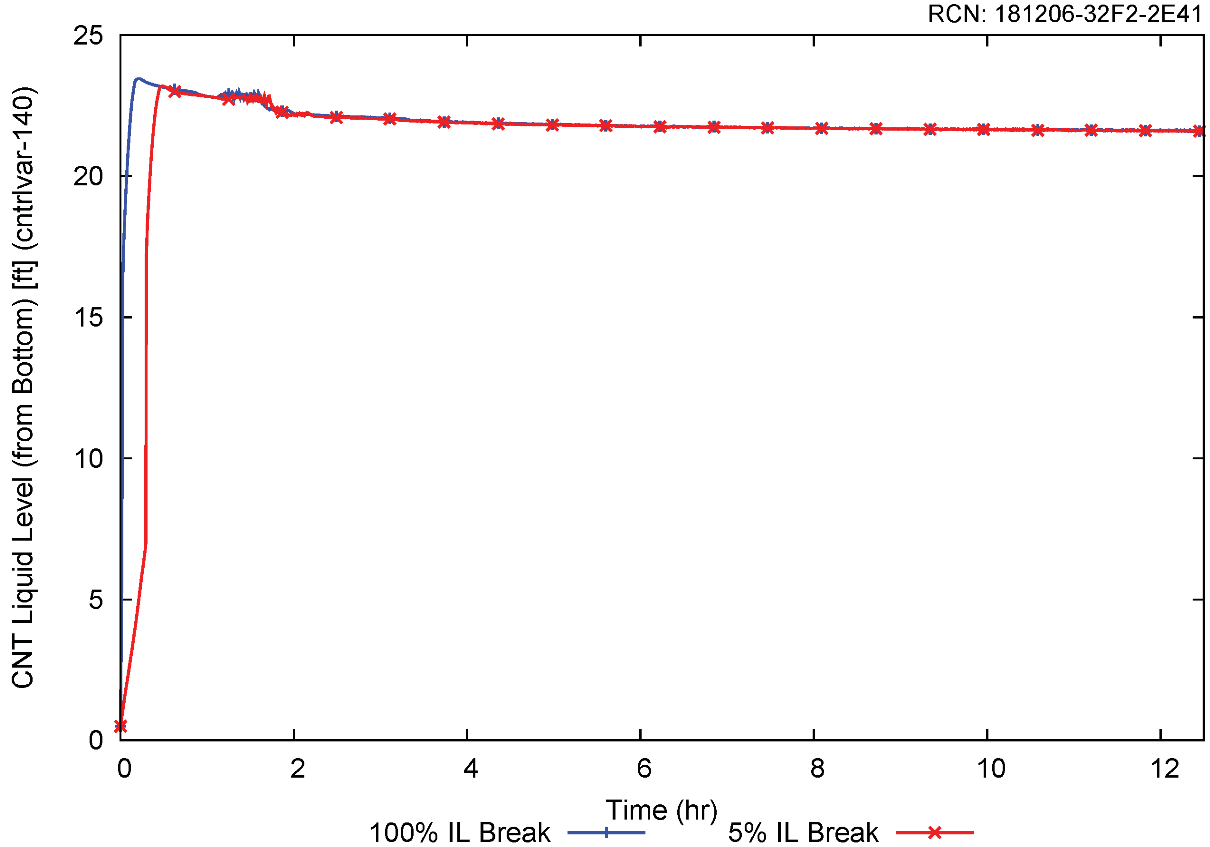

Figure 5-13 Minimum temperature injection line break: containment liquid level above bottom

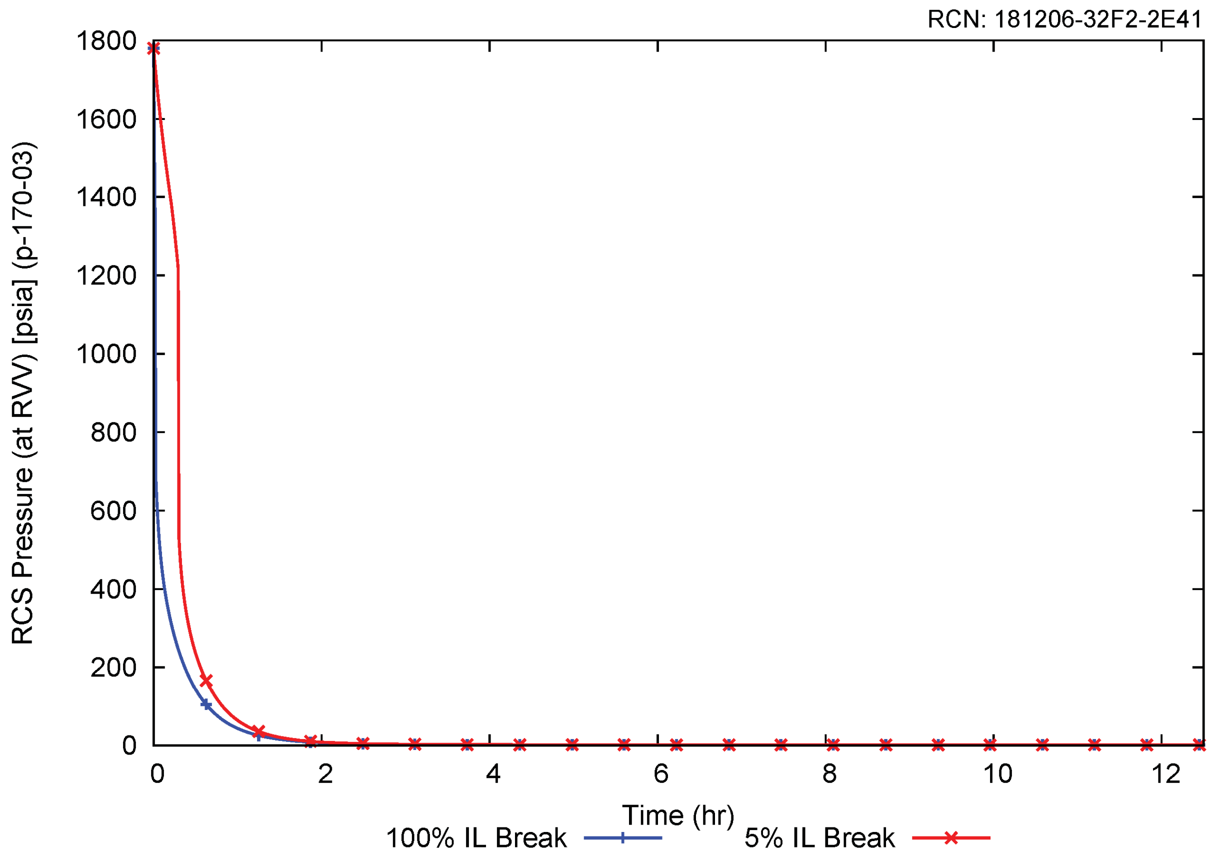

Figure 5-14 Minimum temperature injection line break: RCS pressure at RVV

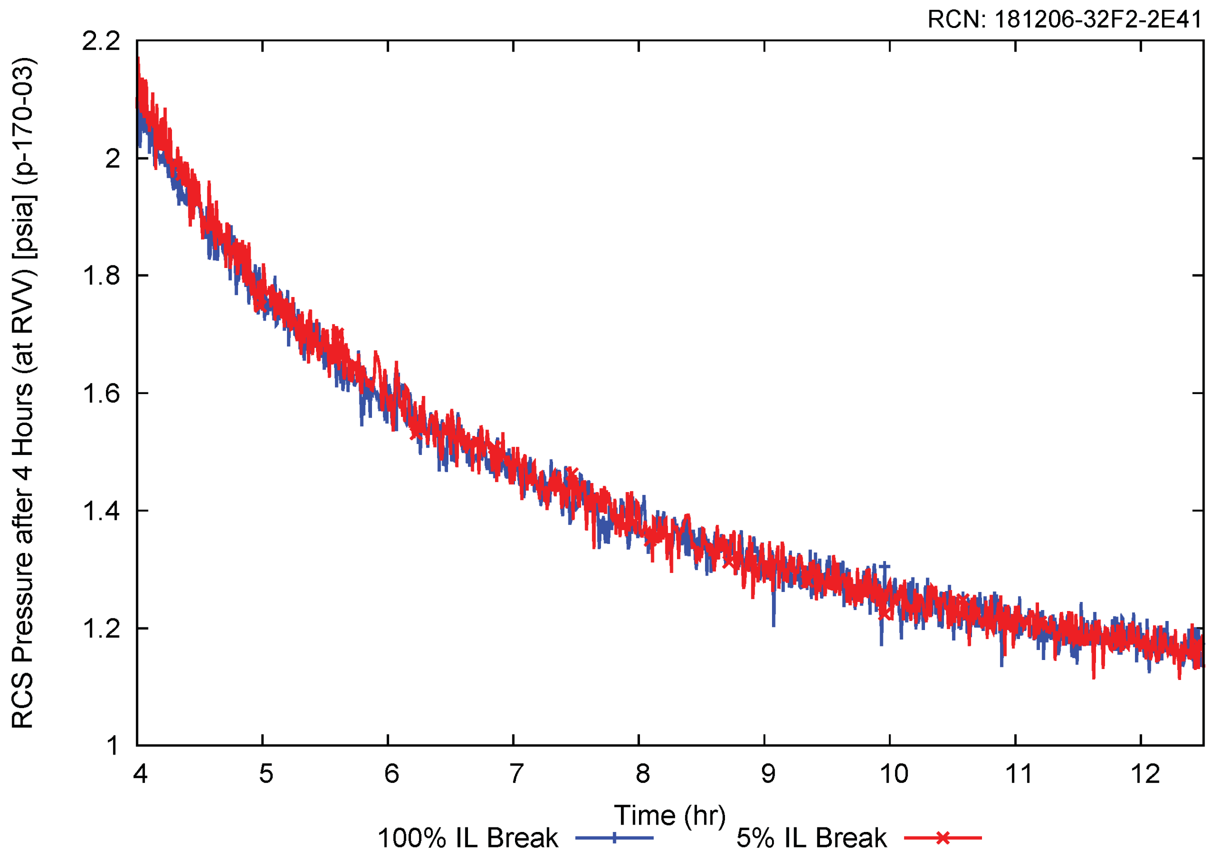

Figure 5-15 Minimum temperature injection line break: RCS pressure at RVV after 4 hours

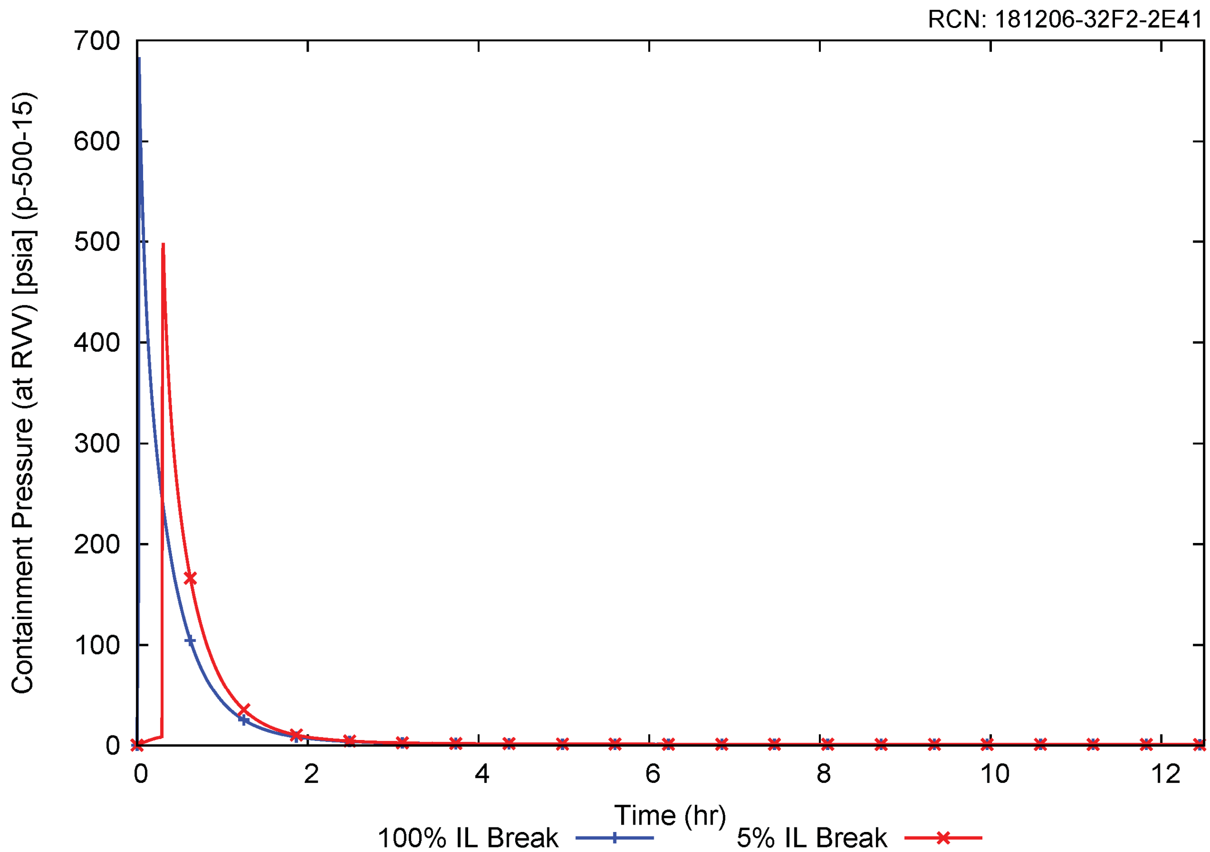

Figure 5-16 Minimum temperature injection line break: containment pressure at RVV

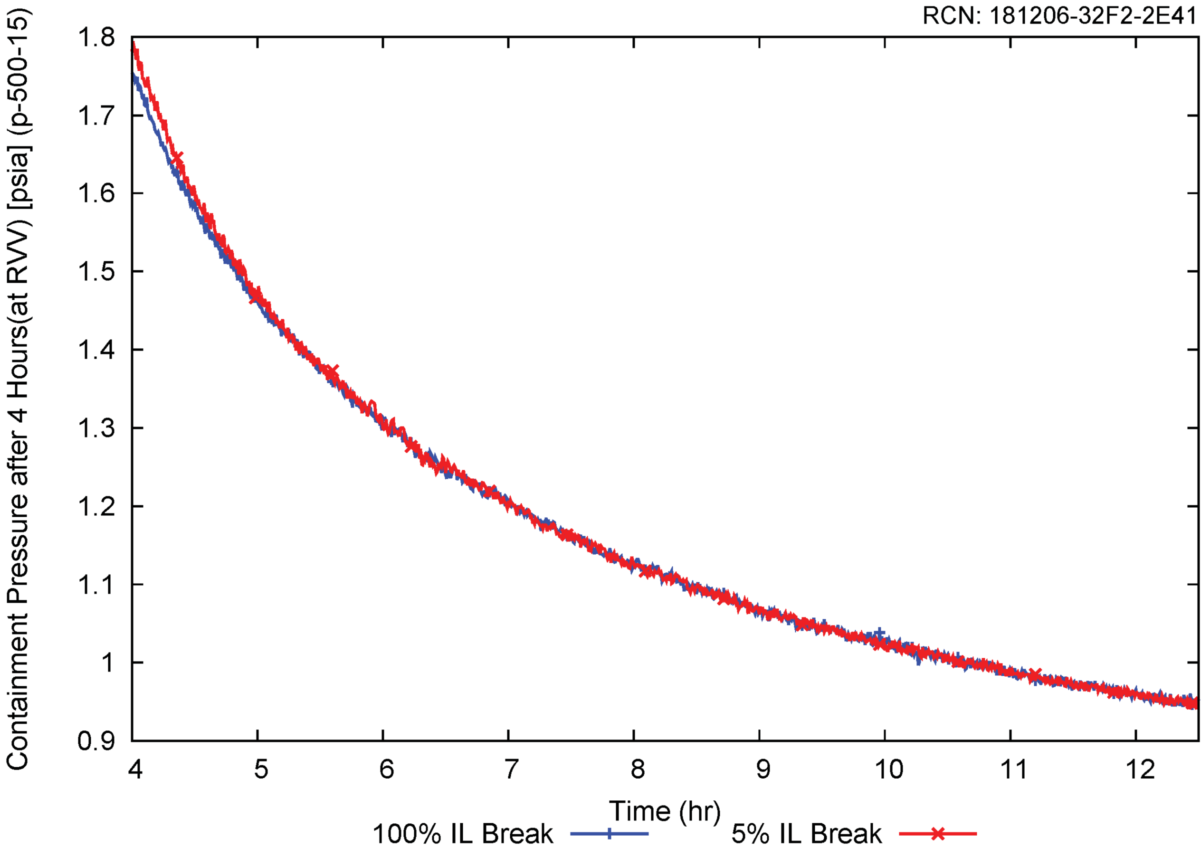

Figure 5-17 Minimum temperature injection line break: containment pressure at RVV after 4 hours

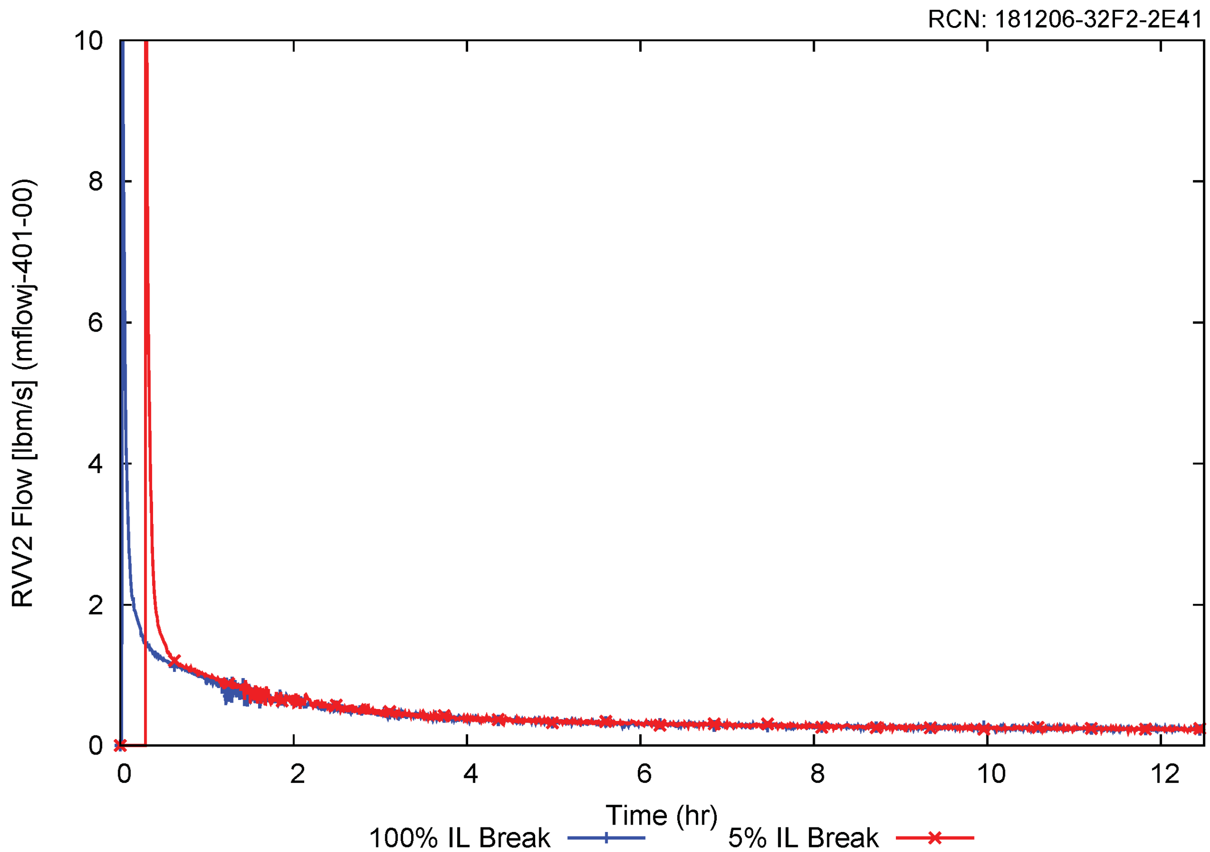

Figure 5-18 Minimum temperature injection line break: RVV2 flow

5.6.3 Minimum Level

The minimum level scenario evaluates the combined impact of all conditions that reduce collapsed liquid level above the top of active fuel. Level is minimized when the long term differential pressure between the RPV and CNV is maximized. Under this condition, sufficient inventory must accumulate inside containment to create enough static head to allow coolant recirculation back into the RPV through the RRVs. As containment inventory increases, RPV inventory coincidently decreases. In general, this scenario is achieved by maximizing heat transfer from containment to the UHS, minimizing ECCS capacity, and maximizing RCS energy. Specific conditions include:

- minimum ECCS capacity, including minimum valve area and flow coefficients, expansion factor of Y=0.7 applied to RVVs, and the single failure of one RVV and RRV to open
- DHRS operation is disabled (DHRS would provide an additional means of RPV pressure relief which is non-conservative for minimizing level)
- decay heat with 1.2 multiplier
- initial RCS temperature is maximized
- initial RCS inventory is minimized
- zero non-condensable gas is modeled
- a minimum reactor pool temperature of 65 °F and maximum reactor pool level of 69 feet

Results are presented for the LOCA 100% IL break, which is the limiting collapsed level case for LTC analysis, and the 5% IL break. Figure 5-21 shows the minimum collapsed level for the 100% IL break is 2.8 feet above TAF and occurs approximately 3.6 hours after ECCS actuation. It is noted that Figure 5-22 shows the 5% IL break has an overall lower minimum level than the 100% IL break, however the rapid drop in level occurs coincident with ECCS actuation and is characteristic of small line LOCA breaks. As this sudden reduction in level occurs before recirculation flow is established through the RRVs, the minimum level for the 5% IL break is covered by the LOCA EM and is not considered limiting for LTC analysis. Both cases show similar trends for long term level recovery. System temperatures and pressures and fuel temperatures are shown to converge toward the same value regardless of initiating break size (Figure 5-19 and Figure 5-20 and Figure 5-23 through Figure 5-26). Figure 5-27 demonstrates that stable ECCS cooling is established through the LTC phase.

The results demonstrate that the core remains covered at all times, and collapsed level and core inlet temperature remain sufficiently high to preclude boron precipitation at all times during the LTC phase. All acceptance criteria are satisfied for the minimum level scenario. State point conditions at 72 hours are presented in Section 5.6.5 for the 100% IL break case.

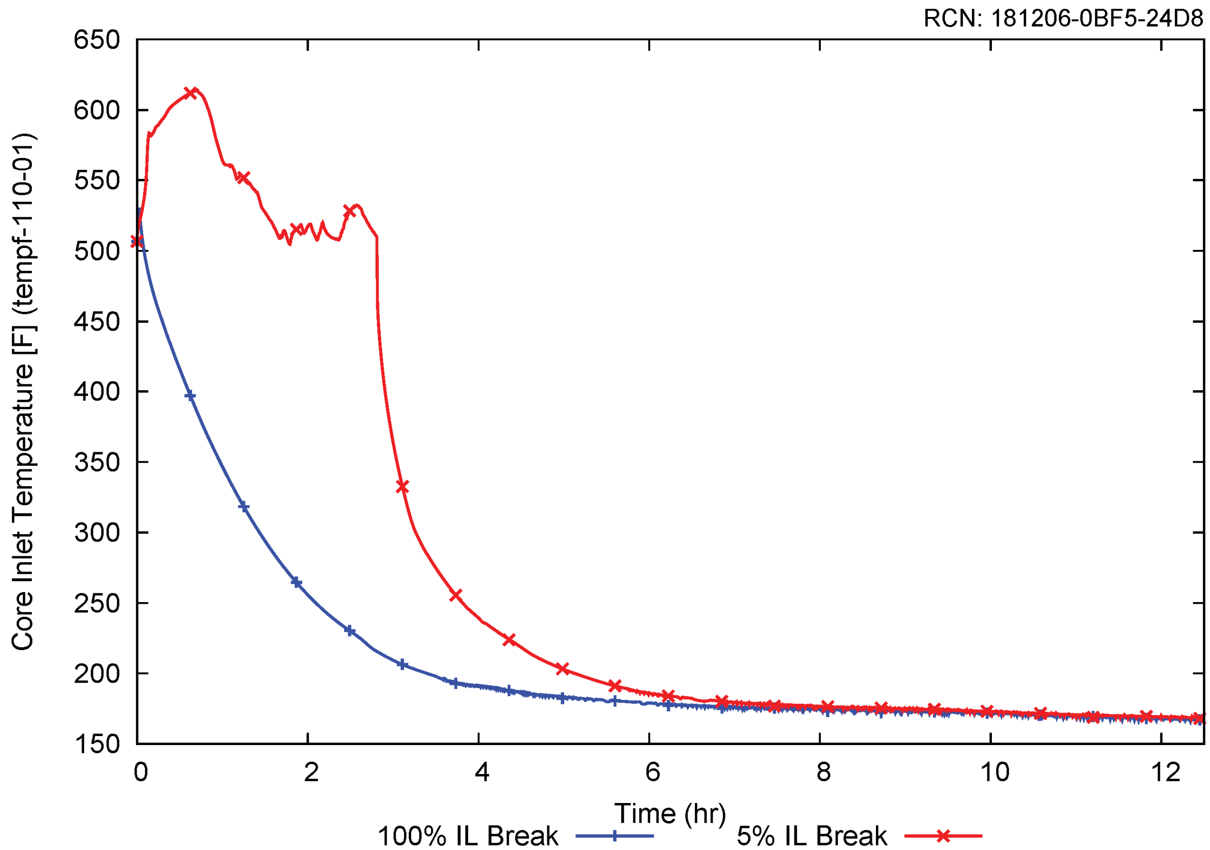

Figure 5-19 Minimum level injection line break: core inlet temperature

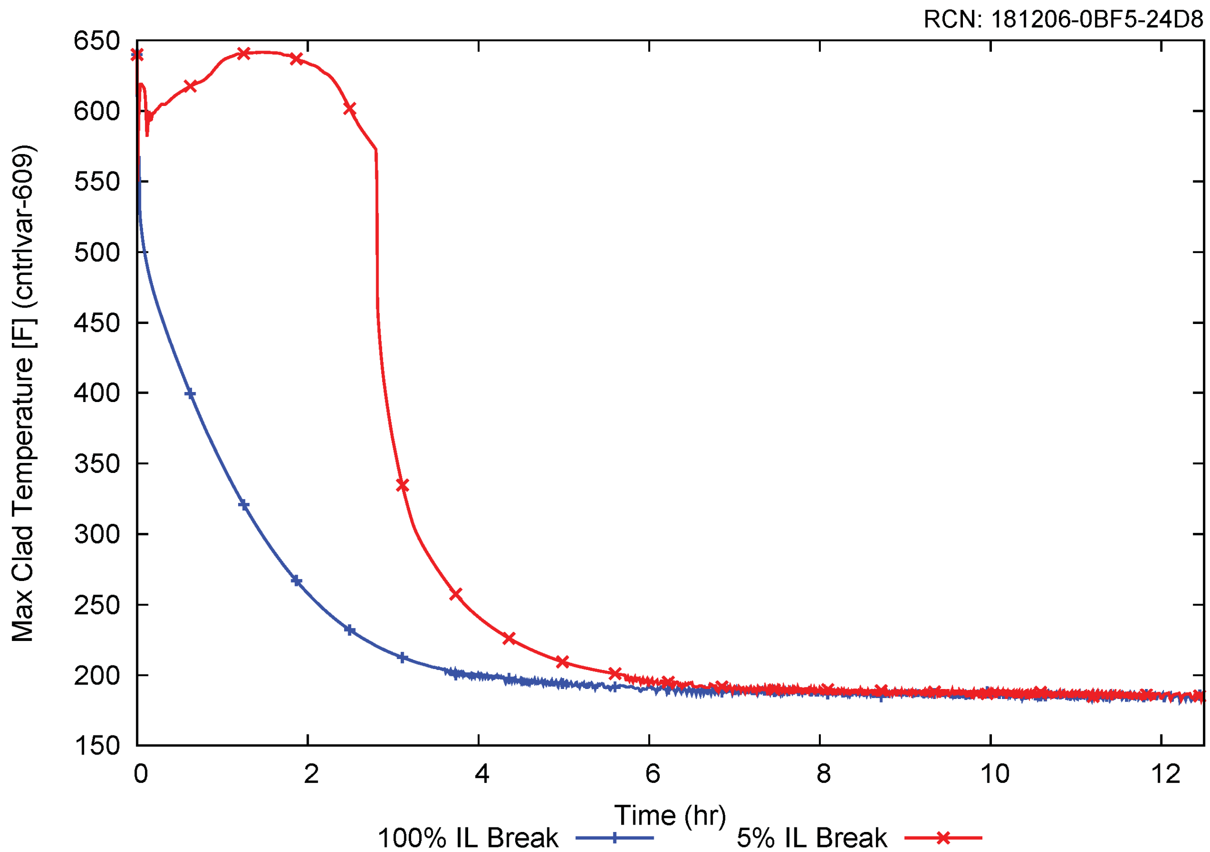

Figure 5-20 Minimum level injection line break: maximum cladding temperature

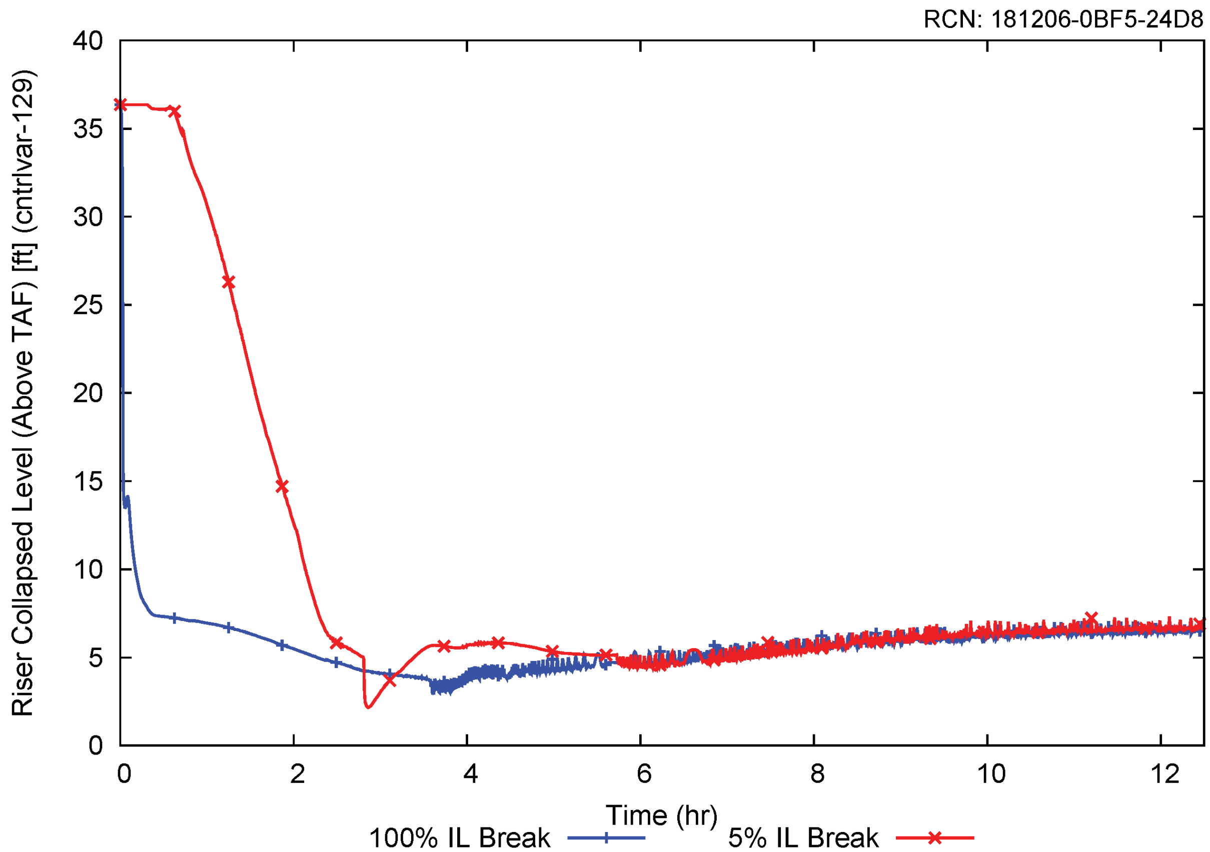

Figure 5-21 Minimum level injection line break: riser collapsed liquid level above top of active fuel

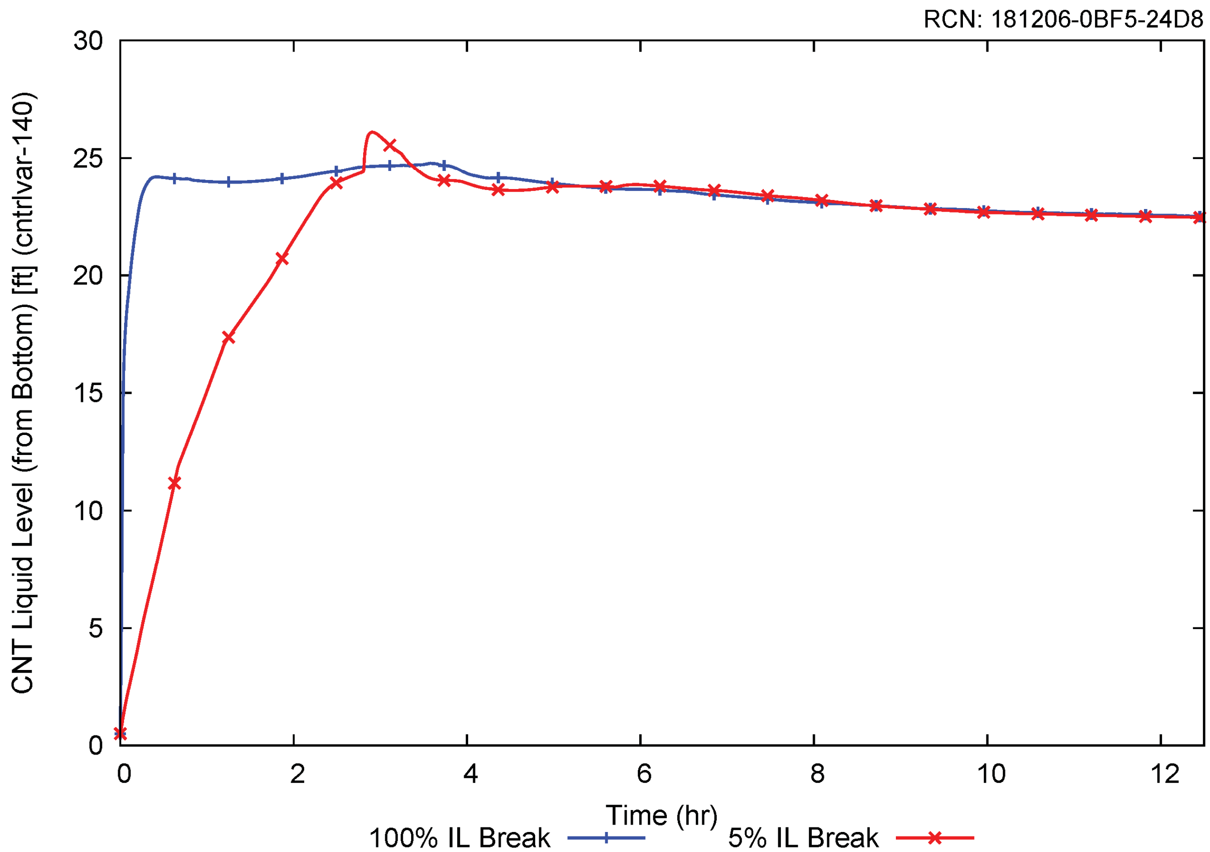

Figure 5-22 Minimum level injection line break: containment liquid level above bottom

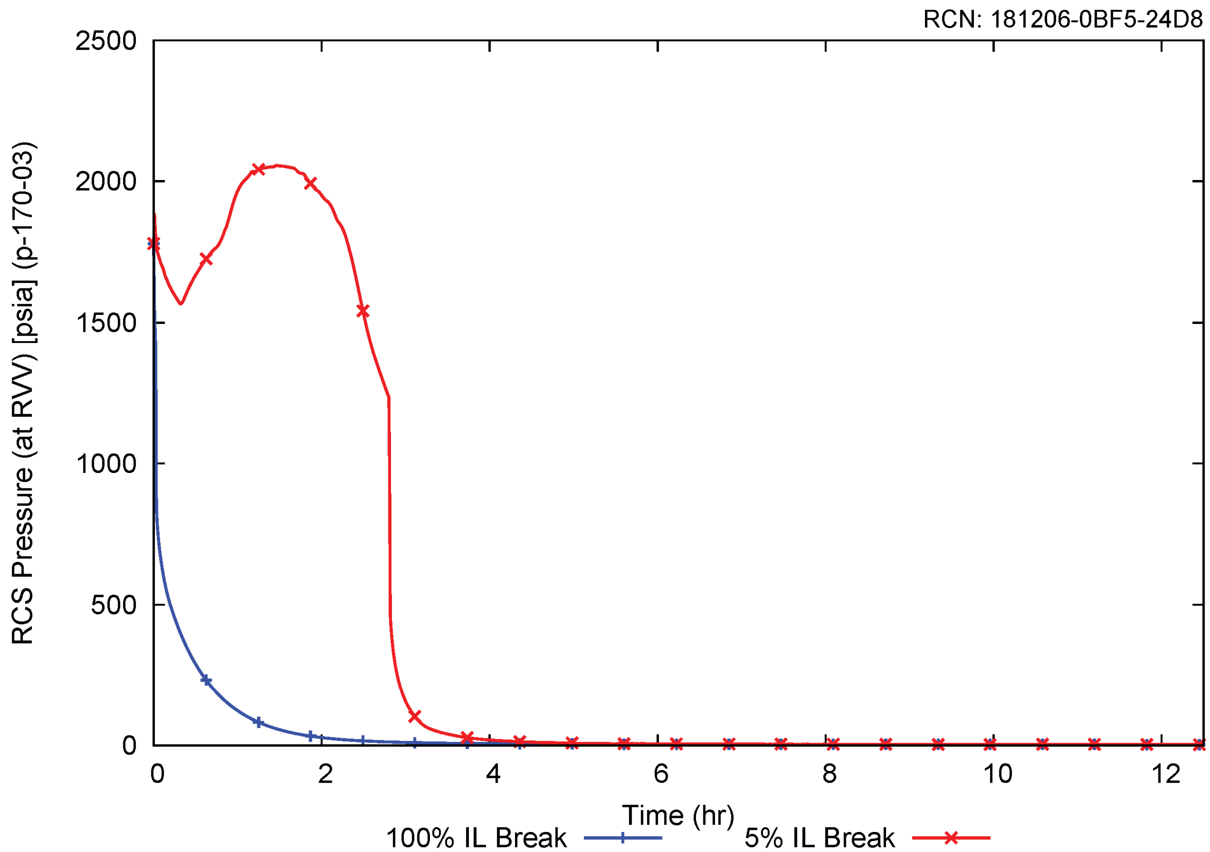

Figure 5-23 Minimum level injection line break: RCS pressure at RVV

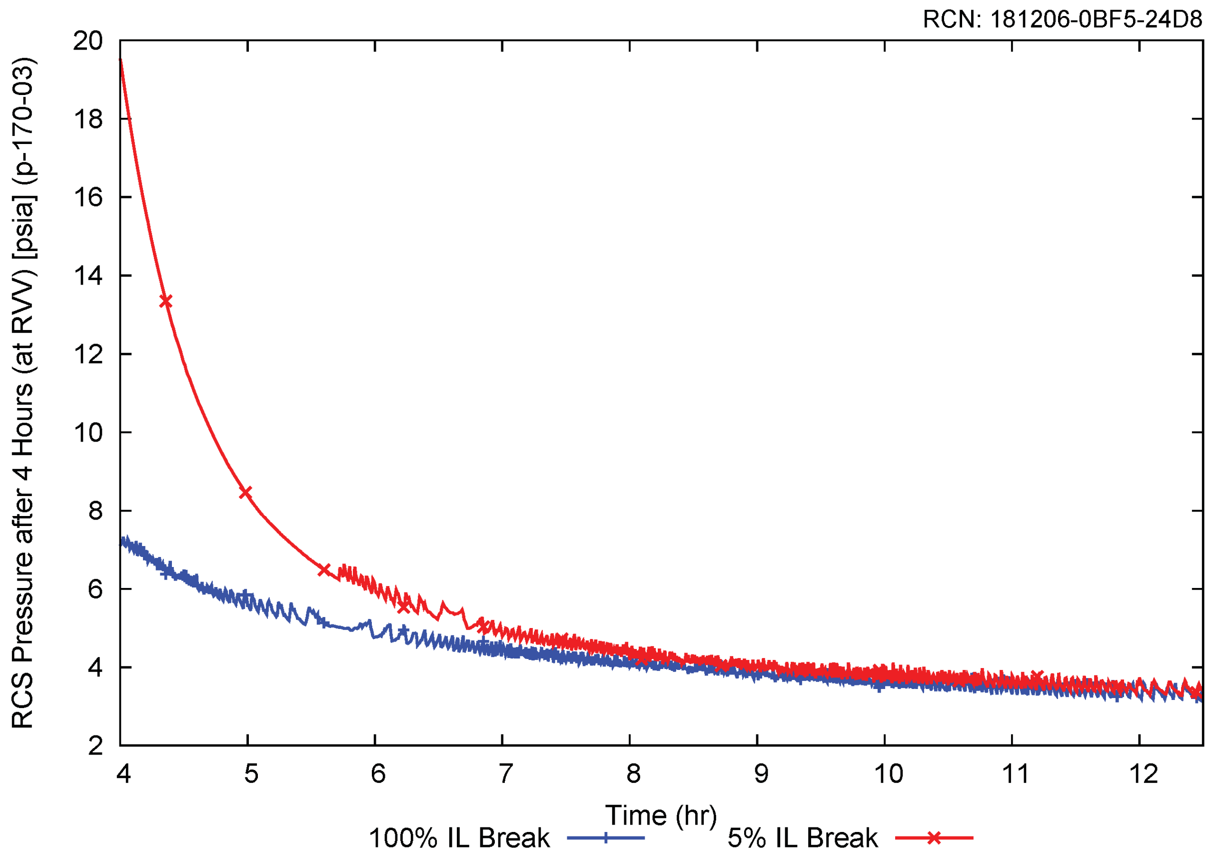

Figure 5-24 Minimum level injection line break: RCS pressure at RVV after 4 hours

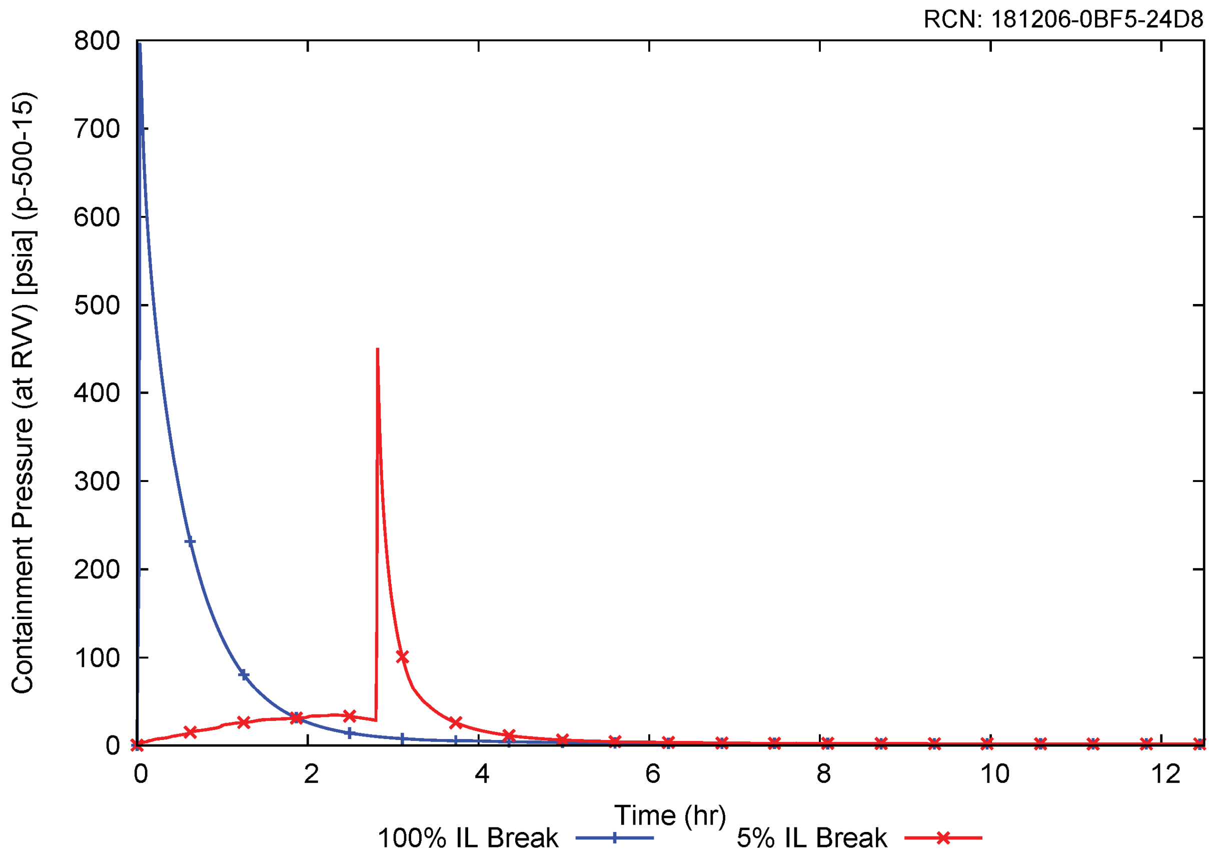

Figure 5-25 Minimum level injection line break: containment pressure at RVV

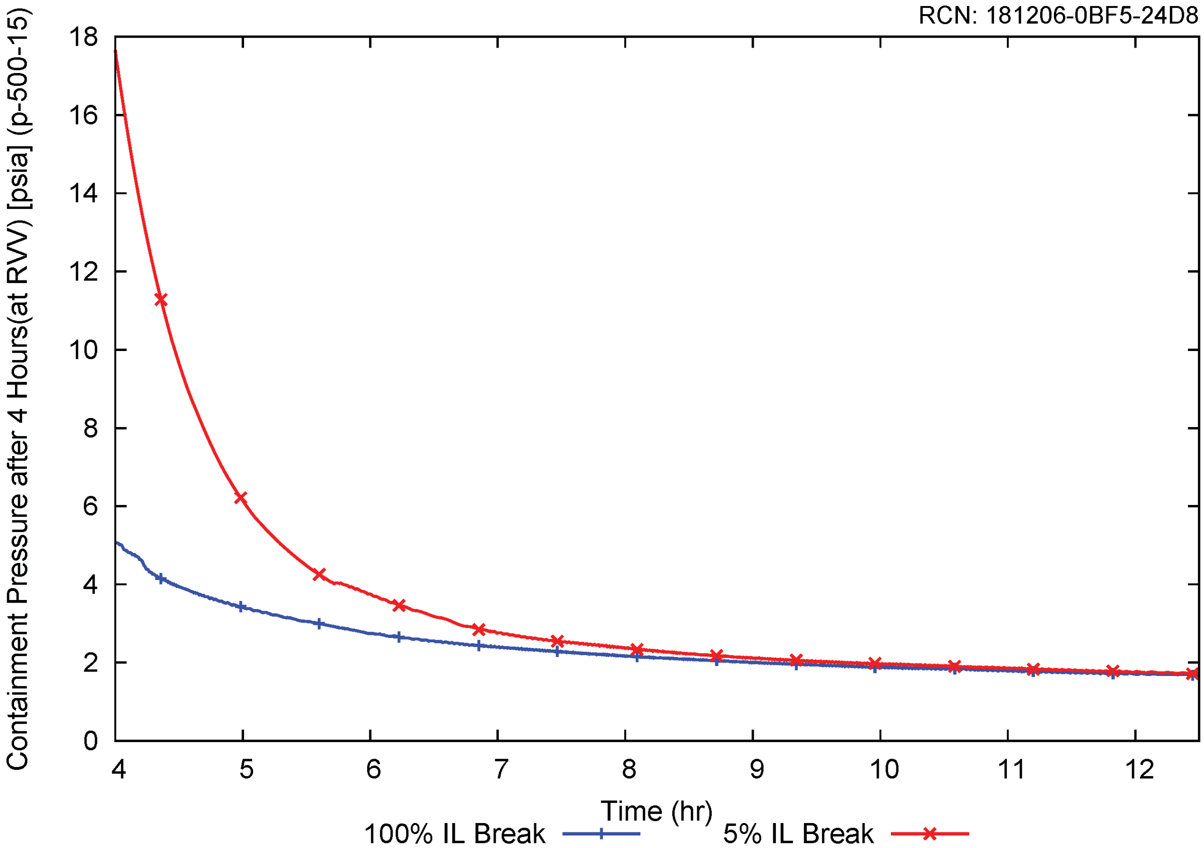

Figure 5-26 Minimum level injection line break: containment pressure at RVV after 4 hours

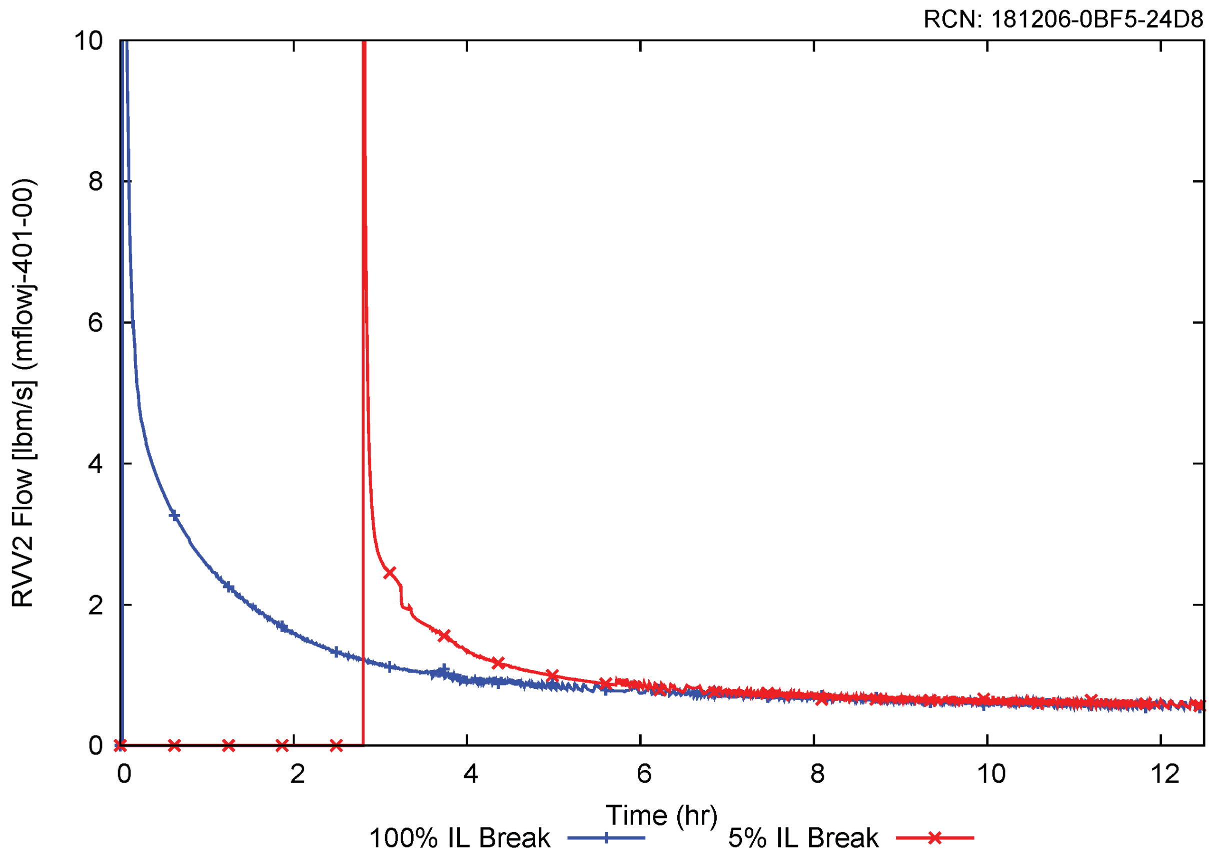

Figure 5-27 Minimum level injection line break: RVV2 flow

5.6.4 Steam Generator Tube Failure with Minimum Level Conditions

The SGTF event was identified as the limiting non-LOCA case in terms of minimum collapsed riser level. This scenario was run with the minimum level conditions identified in Section 5.6.3, except that a decay multiplier of 1.0 was applied, and the DHRS is not disabled for non-LOCA events. A loss of normal AC and DC power is assumed at event initiation, causing ECCS actuation once the system has depressurized to the IAB release setpoint.

Except for ECCS actuation occurring later in time, Figure 5-28 through Figure 5-36 demonstrate similar trends in long term conditions as seen for the LOCA IL break presented in Section 5.6.3. Adequate core cooling is maintained even with the additional inventory loss of the SGTF. The minimum collapsed liquid level for the SGTF event was non-limiting compared to the results presented in Section 5.6.3.

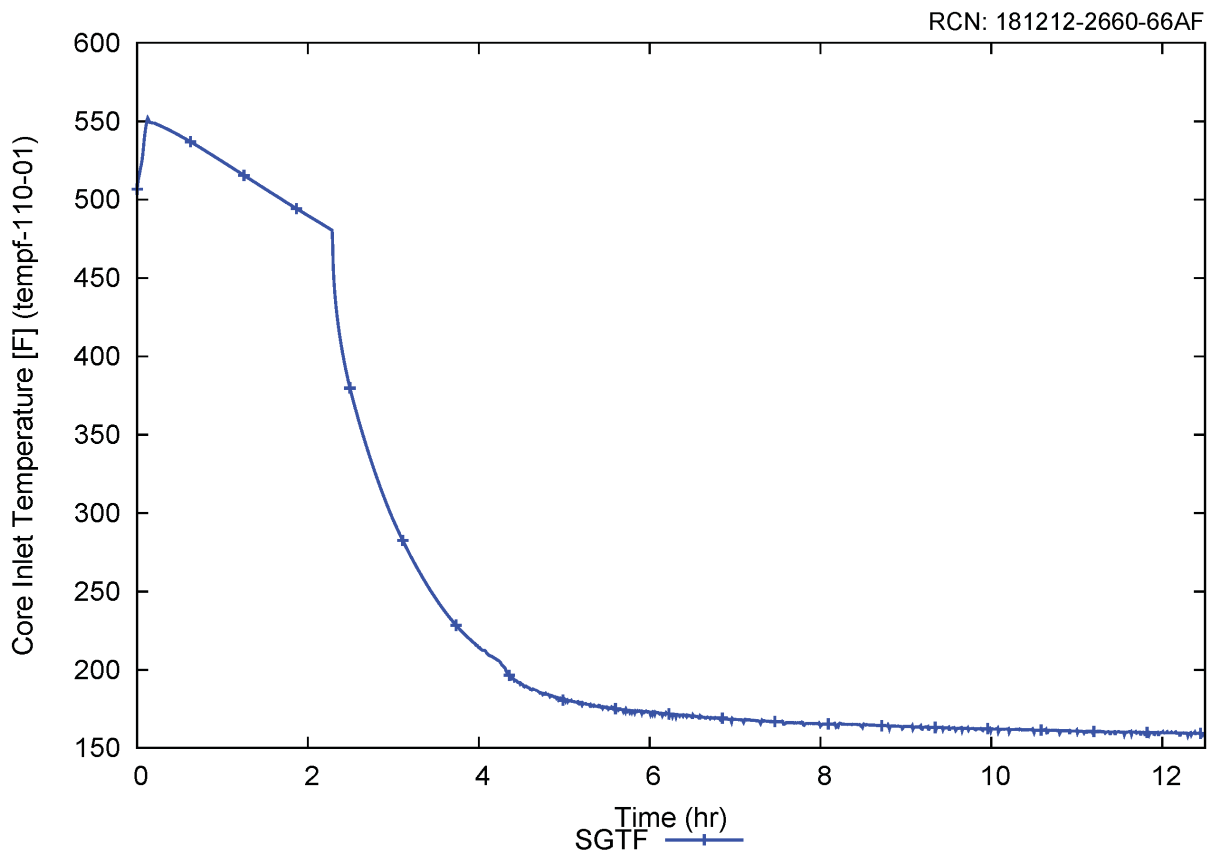

Figure 5-28 Minimum level steam generator tube failure: core inlet temperature

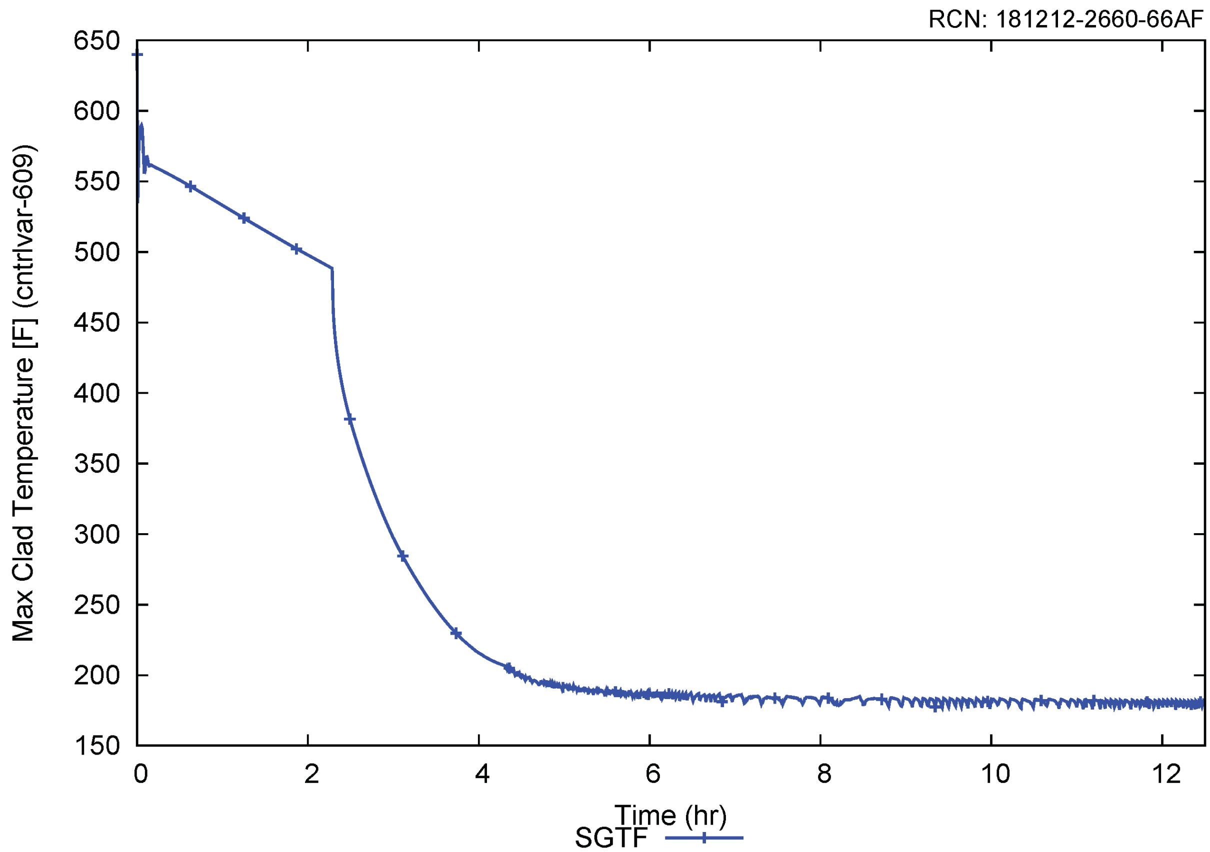

Figure 5-29 Minimum level steam generator tube failure: maximum cladding temperature

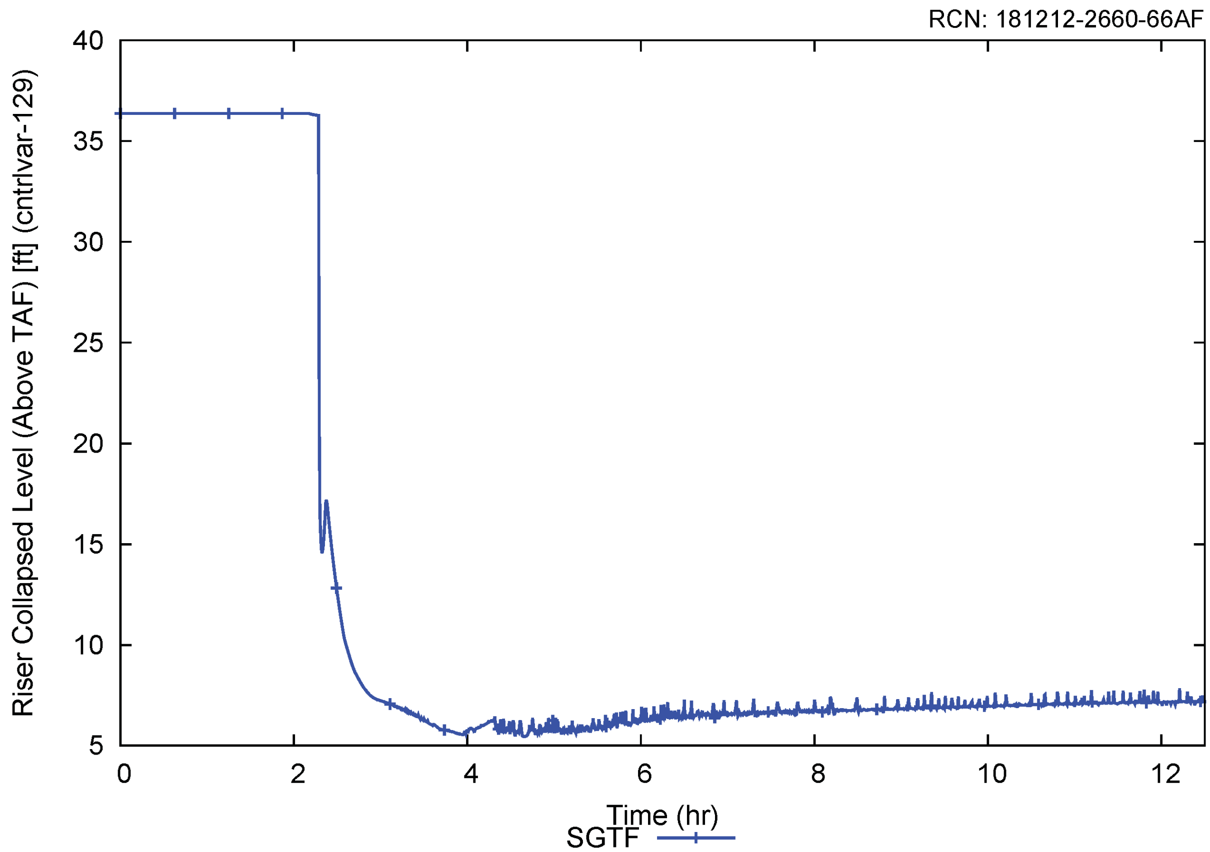

Figure 5-30 Minimum level steam generator tube failure: riser collapsed liquid level above top of active fuel

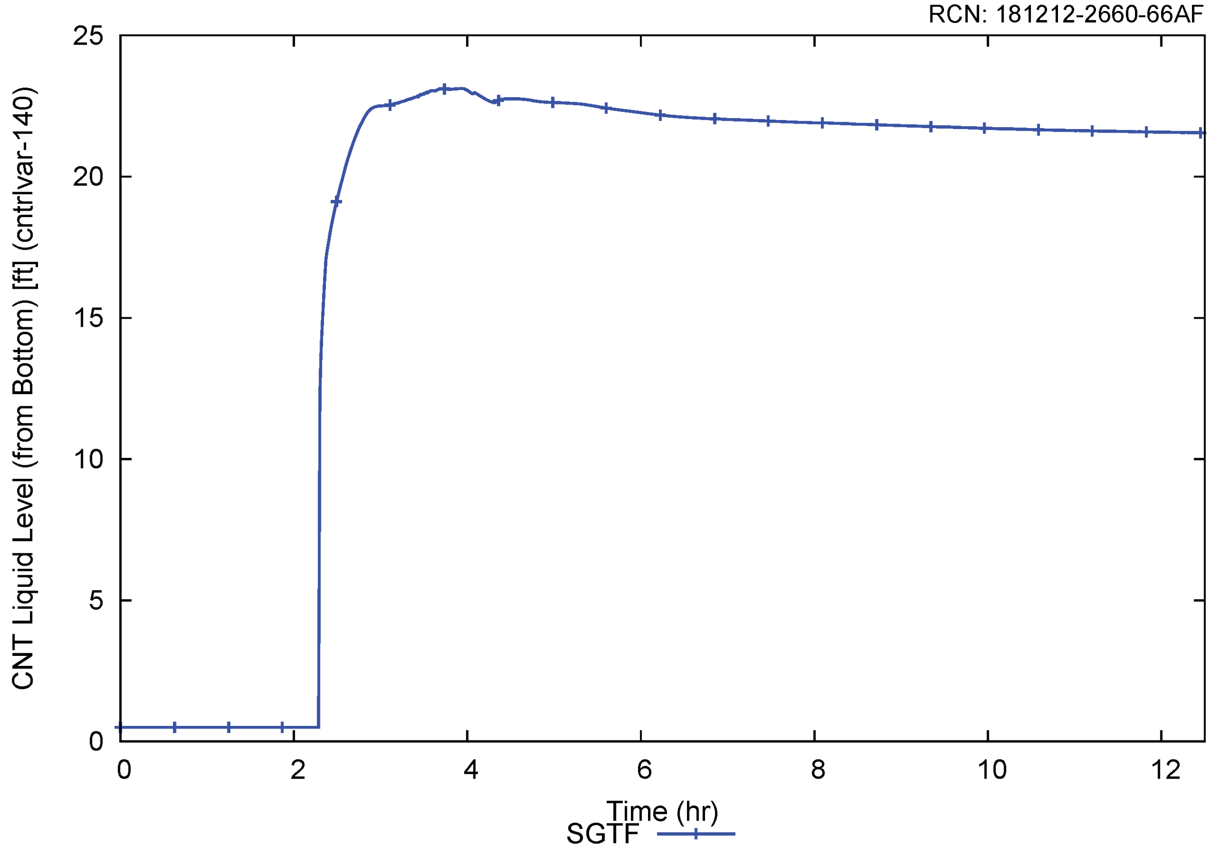

Figure 5-31 Minimum level steam generator tube failure: containment liquid level above bottom

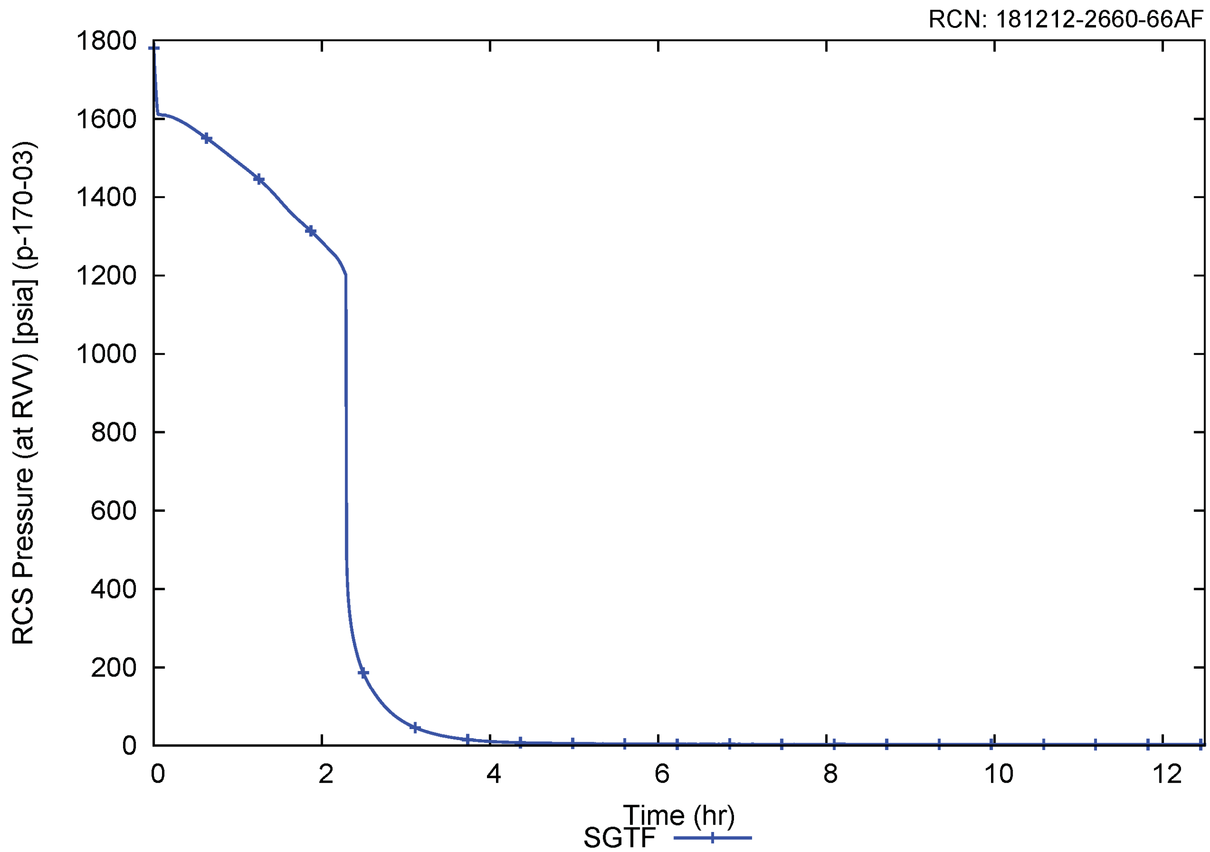

Figure 5-32 Minimum level steam generator tube failure: RCS pressure at RVV

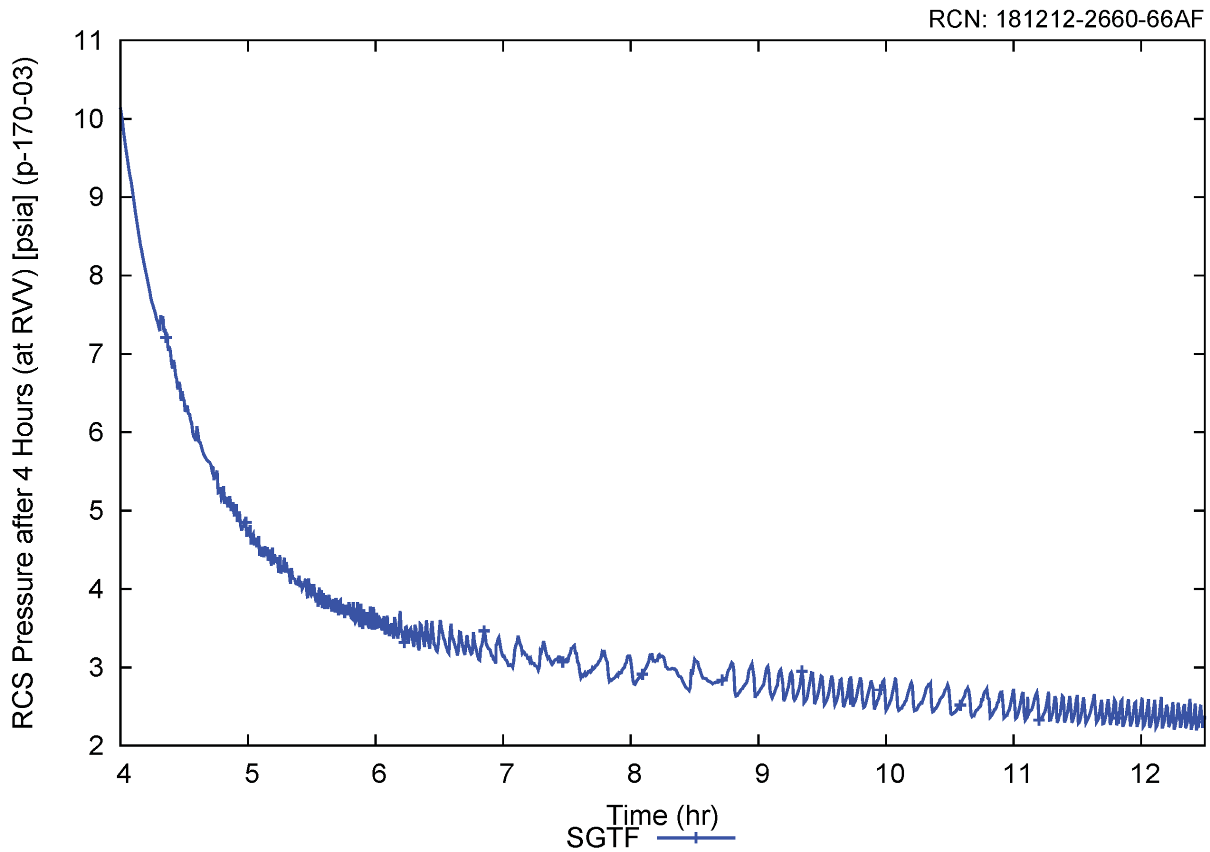

Figure 5-33 Minimum level steam generator tube failure: RCS pressure at RVV after 4 hours

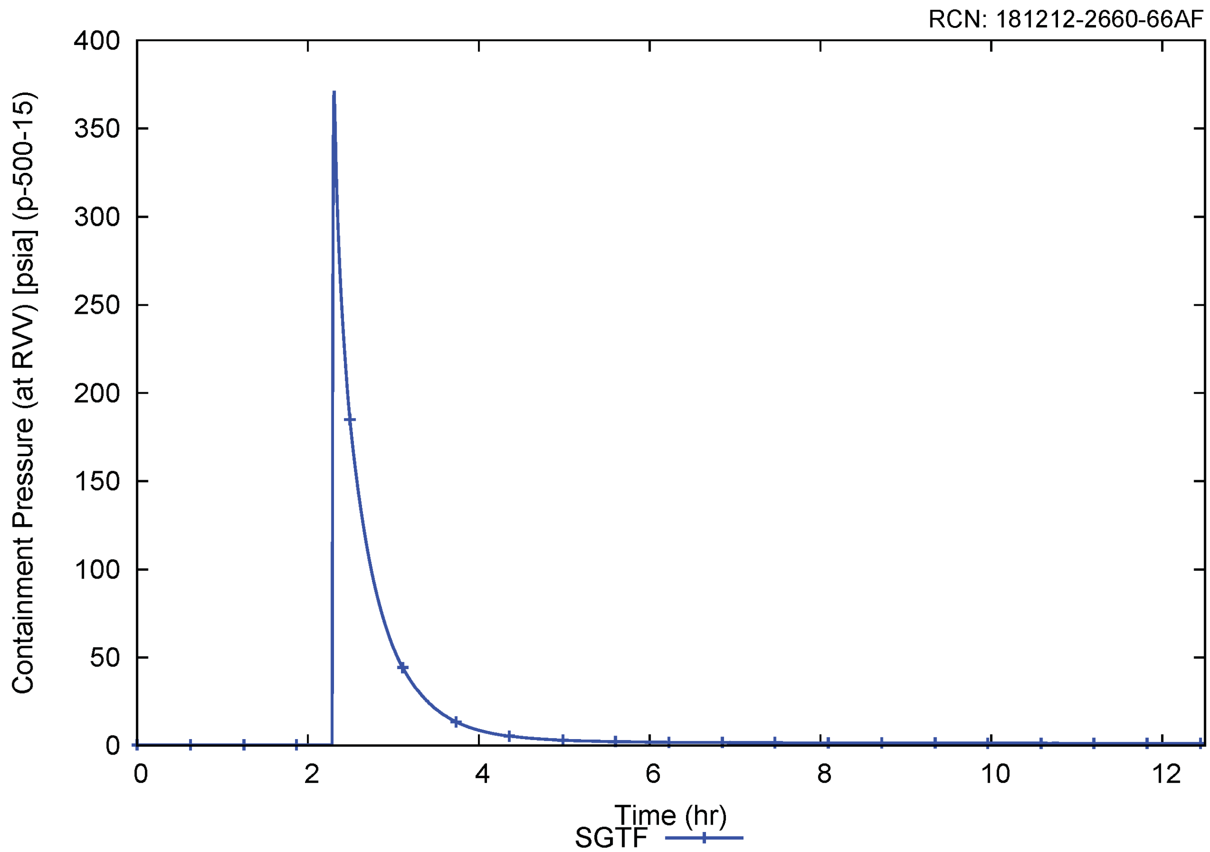

Figure 5-34 Minimum level steam generator tube failure: containment pressure at RVV

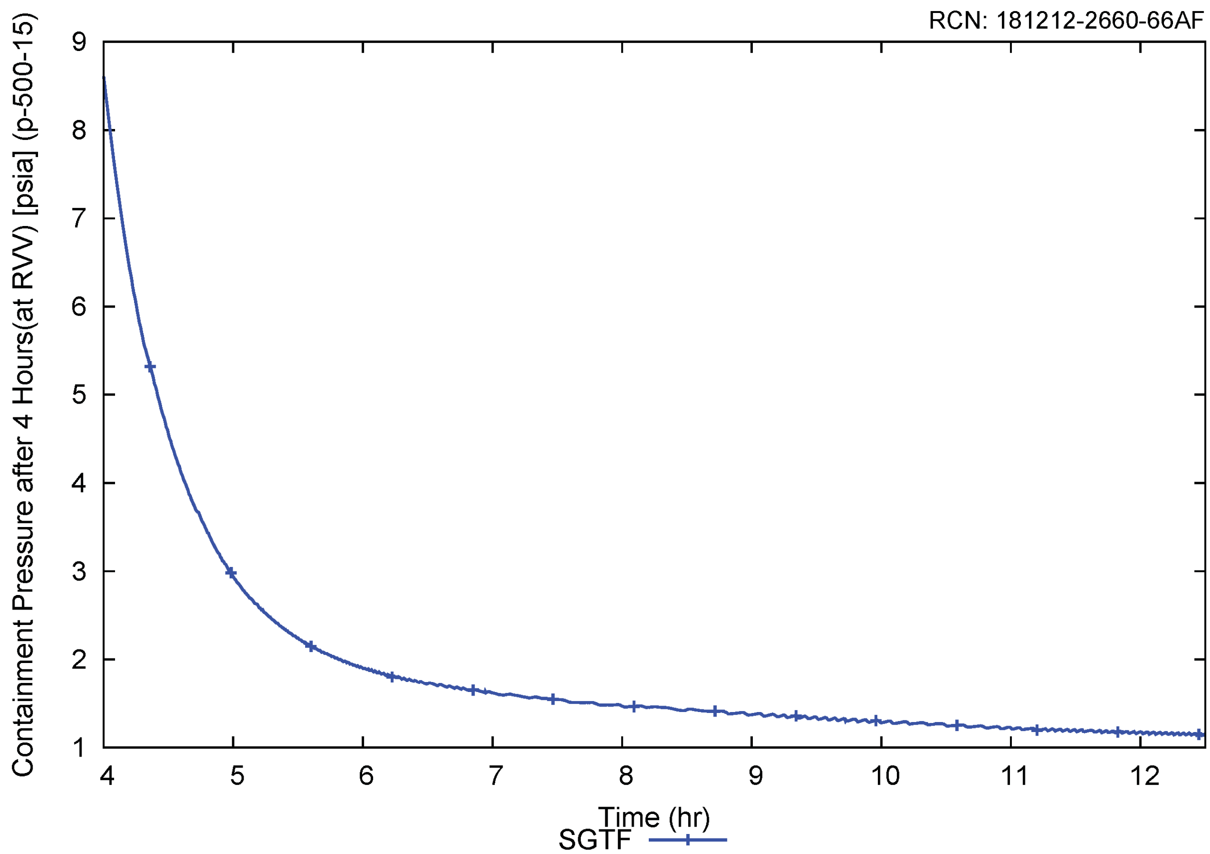

Figure 5-35 Minimum level steam generator tube failure: containment pressure at RVV after 4 hours

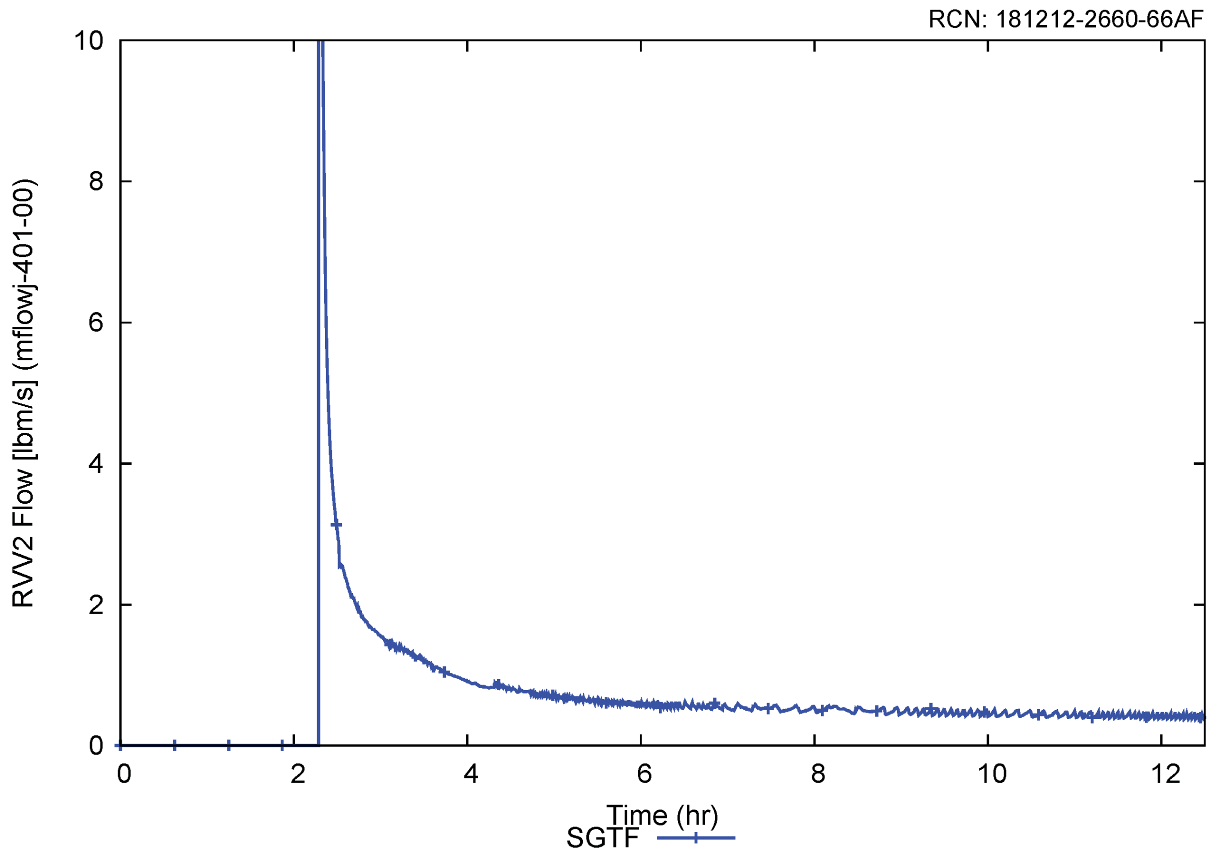

Figure 5-36 Minimum level steam generator tube failure: RVV2 flow

5.6.5 State-point Evaluation at 72 Hours

State-point module conditions after 72 hours are presented in Table 5-3. Results are included for the following events: the IL break cases presented in Sections 5.6.1, 5.6.2, and 5.6.3, the maximum temperature IL break case with reactor pool level set to 45 feet, and three minimum temperature cases which assume an initial reactor power of 13%. The following conclusions are drawn:

- While a reactor pool level of 45 feet did increase core inlet temperature by 10°F relative to the 55 feet case, this increase is not significant enough to impact the conclusion that cladding temperature is not challenged during LTC.
- Collapsed level above TAF was maintained for all cases after 72 hours. Limiting minimum level occurred earlier in the LTC phase with level recovering to equilibrium by 72 hours.
- Margin to boron precipitation was demonstrated for all cases. The low power initial condition results in lower core inlet temperature after 72 hours.

Table 5-3 Results of state-point analysis at 72 hours

Case	Core Inlet Temperature (°F)		Collapsed Riser Level above TAF (ft)		Boron Precipitation Margin (°F)	
	Transient at 12.5 Hours	State-point at 72 Hours	Transient at 12.5 Hours	State-point at 72 Hours	Transient at 12.5 Hours	State-point at 72 Hours
Maximum Temperature IL break	292.8	270.4	8.9	9.1	208.9	187.8
Minimum Temperature IL break	152.8	140.4	10.0	10.4	73.1	62.3
Minimum Level IL break	165.3	154.5	7.3 (1)	8.0	76.2	69.0
Maximum Temperature IL break, 45 feet reactor pool level (2)	-	280.3	-	9.2	-	197.6
Minimum Temperature IL break, 13% initial power (2)	-	94.3	-	10.4	-	16.6
Minimum Temperature SGTF, 13% initial power (2)	-	112.1	-	10.1	-	33.3
Minimum Temperature DHRS cooldown, 13% initial power (2)	-	116.8	-	10.4	-	39.2

(1) Minimum collapsed riser level was 2.8 feet and occurred approximately 3.6 hours after ECCS actuation.
(2) A 12 hour transient simulation for these cases was not performed. Limiting conditions are only important at the end of the LTC phase at 72 hours.

5.6.6 Summary and Conclusions

Evaluation of LTC conditions following a variety of LOCA and non-LOCA initiating events was performed for three scenarios, maximum temperature, minimum temperature, and minimum level, which are challenging to different acceptance criteria. Detailed results from a subset of these events are provided in Section 5.6.1 through Section 5.6.4. The LTC acceptance criteria for cladding temperature, level above TAF, and margin to boron precipitation are satisfied for all cases. The following conclusions are drawn from the LTC analysis results.

- All maximum temperature cases showed decreasing cladding temperatures over the long term, with final cladding temperature remaining well below operating temperature at full power.
  - Generally, the LOCA spectrum cases result in higher cladding temperatures than the non-LOCA cases by 12.5 hours.
  - The maximum temperature cases are not challenging for collapsed level and boron precipitation.
  - Sensitivity results indicate that including non_condensable gases inside containment is limiting for clad temperature.
  - Sensitivity to reactor pool level of 45 feet was evaluated. While final clad temperature was 24 °F higher than at 12.5 hours, all acceptance criteria remained satisfied and overall module conditions were not significantly affected. The 55 feet pool initial level was generally applied in the LTC analysis since this is the credible condition at transient initiation.
  - Section 5.6.5 demonstrates acceptable cladding temperatures after 72 hours for the maximum temperature scenario, including both the 45 feet and 55 feet reactor pool level cases.
- All minimum temperature cases showed margin to boron precipitation over the long term as core inlet temperature decreased.
  - Generally, all LOCA and non-LOCA events converged towards the same temperatures and pressures after 12.5 hours.
  - Collapsed level by 12.5 hours was most challenged by non-LOCA events in which RCS inventory is lost to the secondary system. Such cases were also limiting for boron precipitation after 12.5 hours due to higher boric acid concentration. All cases demonstrated margin to the collapsed level and boron precipitation criteria.
  - Margin to boron precipitation was demonstrated for the limiting minimum temperature case after 72 hours, which included an initial reactor power of 13% (Section 5.6.5).

- All minimum level cases showed the core remained covered at all times during the transient and that boron precipitation driven by increased boric acid concentration with decreasing level is precluded.

  - The LOCA injection line breaks were limiting for minimum level. The most limiting case is the 100% IL break with a minimum level of 2.8 feet above the TAF occurring approximately 3.6 hours after ECCS actuation.

  - All cases show long term level recovery, where final level after 12.5 hours is higher than the transient minimum level.

  - The timing of minimum boron precipitation margin generally occurred shortly after the timing of minimum level. Margin is reduced as boric acid concentration in the mixing volume increases.

  - The non-LOCA SGTF cases were limiting for boron precipitation due the worst case combination of collapsed level and core inlet temperature.

  - Section 5.6.5 demonstrates margin to collapsed level and boron precipitation after 72 hours for the minimum level scenario.

6.0 Boron Precipitation Methodology and Analysis Results

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

6.1 General Approach and Acceptance Criteria

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

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

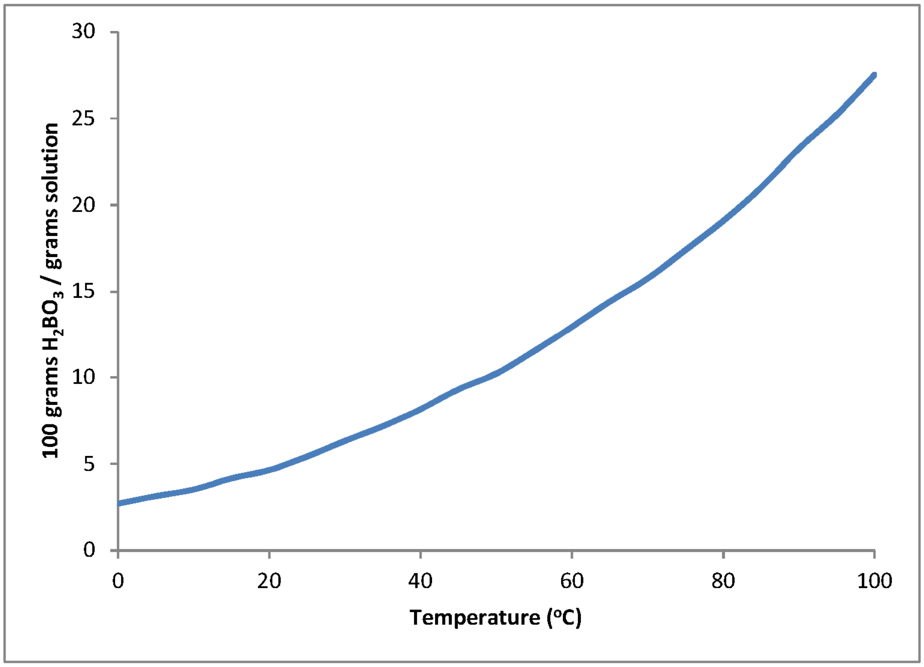

Figure 6-1 Percent boric acid at solubility limit as a function of temperature

6.2 Methodology

The determination of the precipitation temperature for the liquid mass in the core and riser regions starts with the calculation of the entire mass of boron in the reactor coolant system (RCS). Then, a corresponding boric acid concentration is calculated for the mixing volume. Next, the precipitation temperature is obtained for the mixing volume concentration using the boron precipitation curve. Finally, core inlet temperature is compared to the precipitation temperature to determine margin to boron precipitation.

These calculations are performed by control variables built into the NRELAP5 LTC model. This allows total boron mass and boric acid concentration in the mixing volume to be calculated based on case-specific conditions.

6.2.1 Calculate Total Boron Mass

The maximum allowable boron concentration in the RCS during operation is conservatively assumed. The initial mass of water in the RCS is calculated on a case-specific basis.

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

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

where,

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

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

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

6.2.2 Calculate Mass of Fluid in Mixing Volume

The total liquid mass inside the core and riser regions is captured dynamically by the NRELAP5 LTC model.

6.2.3 Calculate Boron Concentration in Mixing Volume

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

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

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

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

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

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

6.2.4 Assess Margin to Boron Precipitation

The precipitation temperature as a function of boric acid concentration is compared to the core inlet temperature. Margin is demonstrated by ensuring core inlet temperature remains greater than the precipitation temperature.

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

6.3 Results

The results for the LTC analysis cases considered in the boron precipitation analysis are given in Table 5-3 in Section 5.6.5. Using the input from the boron solubility curve in Figure 6-1, NRELAP5 predicted that the case which produced the minimum margin of 16.6 degrees F to boron precipitation was a 13% power IL break case biased for minimum temperature at 72 hours. Full power cases with lower decay heat were considered in the boron precipitation analysis and were found to be non-limiting due to minimal impact on core inlet temperatures offset by higher long-term level.

6.4 Conclusions

Based on the results shown above, boron precipitation will not occur during any postulated condition in a long-term cooling scenario.

7.0 Summary and Conclusions

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

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

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

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

8.0 References

8.1 Source Documents

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

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

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

8.2 Referenced Documents

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

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

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

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

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

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

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

8.2.8 SwUM-0304-17023, Revision 6, NRELAP5 Version 1.4 Theory Manual.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0719-66145

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

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

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

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

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

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

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

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

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the method by which NuScale develops its long term cooling capability.

NuScale has performed significant research and evaluation to develop a basis for this method and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed report entitled "Long-Term Cooling Methodology," TR-0916-51299-P, Revision 1. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon

the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR § 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 5, 2019.

Thomas A. Bergman