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NUREG-1953

Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom

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Errata

NUREG-1953, “Confirmatory Thermal-Hydraulic to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models – Surry and Peach Bottom,” published September 2011

Page 10 incorrectly lists the modeled power level for Peach Bottom as 3,458 MWt. The modeled power was in fact the correct power level (as of 2011) of 3,514 MWt.

The following equations were inadvertently omitted from page 13:

$$Cumulative\ Probability\ of\ Failure = 1 - (1 - P_D)^n, \text{ or}$$

$$Cumulative\ Probability\ of\ Failure = 1 - (1 - P_D^{initial})(1 - P_D^{subsequent})^{n-1}$$

Footnote 20 (pg. 39) should be read with the knowledge that the MELCOR model would not account for the effect of ambient pump room temperature on the pump bearing temperature.

The Peach Bottom analyses that credit reactor core isolation cooling (RCIC) in automatic mode use a setpoint for high-vessel-level shutoff which is 10 inches (0.25m) below the actual plant value. This is not expected to have a significant effect on the presented results, in that it will affect the specifics of the frequency and duration of RCIC duties, but not whether or not RCIC is able to maintain core cooling.

Some plots in the report have accident parameter signatures (i.e., time histories) that are partially or completely obscured by overlying signatures (e.g., the figure at the top of page A-7). In viewing these figures, the reader must use cues to recognize this overlap. For example, in the figure at the bottom of page A-7, it can be discerned that the 3 steam generators' pressures overlie one another because: (i) there is no departure at time zero, where the three pressures would be roughly the same and (ii) the curves do diverge at the end of the simulation.

Similarly, some plots contain signatures where the values are zero for the entire simulation. These are included because the zero value in and of itself provides information (e.g., confirms that a system was appropriately disabled for a simulation where it is assumed to be unavailable, shows that a low-pressure pump was dead-headed for the entire simulation). Such situations can generally be discerned by discolored x-axes.

Finally, the following abbreviation was inadvertently omitted:

- SPR – Containment spray

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Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom

Prepared by:

H. Esmaili,[1] D. Helton,[1] D. Marksberry,[1] R. Sherry (NRC retired),[1]
P. Appignani,[1] D. Dube,[2] M. Tobin[1]

R. Buell,[3] T. Koonce,[3] J. Schroeder[3]

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[1] Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

[2] Office of New Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

[3] Idaho National Laboratory
P.O. Box 1625
Idaho Falls, ID 83415

ABSTRACT

In a limited number of cases, thermal-hydraulic success criteria from the suite of standardized plant analysis risk (SPAR) models have apparent inconsistencies when compared to counterpart licensee probabilistic risk assessments (PRAs), other relevant SPAR models (i.e., models for similar plants), or relevant engineering studies. These inconsistencies are a natural outcome of the SPAR development process, and often reflect the apparent inconsistencies seen across licensee PRAs for similar plants. Even so, the U.S. Nuclear Regulatory Commission (NRC) staff wants to strengthen the technical basis for the SPAR models by performing targeted additional engineering analysis. The identified success criteria are for both pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). This report describes MELCOR analyses performed to augment the technical basis for supporting or modifying these success criteria. The success criteria contained herein are intended to be confirmatory in nature, and while suitable for their intended use in supporting the SPAR models they are not intended to be used by licensees for risk-informed licensing submittals.

This report first provides a basis for using a core damage surrogate of 2,200 degrees Fahrenheit (1,204 degrees Celsius) peak cladding temperature. Following this discussion are descriptions of the major plant characteristics for the two plants used for this analysis (Surry Power Station and Peach Bottom Atomic Power Station) and the MELCOR models used to represent these plants. Finally, the report presents the results of many MELCOR calculations and compares these results to the corresponding sequences and success criteria in the SPAR models for Surry and Peach Bottom.

The results provide additional timing information for many sequences, confirm many of the existing SPAR model modeling assumptions, and support a few specific changes. Specific changes that have been made to the SPAR models as a result of these analyses are:

- For six SPAR models corresponding to three-loop “high-head” Westinghouse PWRs:
  - Reduction of the adequate venting capability for feed and bleed from two power-operated relief valves (PORVs) to one PORV.
  - Adjustment of the sufficient injection flow during the early stages of a large-break loss-of-coolant accident from two accumulators to one accumulator or one high-head safety injection pump.
- For SPAR models corresponding to BWR Mark Is and Mark IIs:
  - Credit for two control rod drive (CRD) pumps providing adequate core cooling flow following the initial successful operation of the high-pressure coolant injection system or reactor core isolation cooling system.
  - Credit for one CRD pump providing adequate core cooling for injection late in the accident sequence (if not already included).

Some additional changes supported by the MELCOR analysis are not implemented because they are limited by other SPAR modeling assumptions (e.g., timing of core damage relative to battery depletion for station blackout sequences).

FOREWORD

The U.S. Nuclear Regulatory Commission's standardized plant analysis risk (SPAR) models are used to support a number of risk-informed initiatives. The fidelity and realism of these models is ensured through a number of processes, including cross-comparison with industry models, review and use by a wide range of technical experts, and confirmatory analysis. The following report, prepared by staff in the Office of Nuclear Regulatory Research in consultation with staff from the Office of Nuclear Reactor Regulation, experts from Idaho National Laboratory, and the agency's senior reactor analysts, represents a major confirmatory analysis activity.

One of the key strengths and challenges of probabilistic risk assessment (PRA) models is the integration of modeling capability from different disciplines, including human performance, thermal-hydraulics, severe accident progression, nuclear analysis, fuels behavior, structural analysis, and materials analysis. This report investigates the thermal-hydraulic aspects of the SPAR models, with the goal of further strengthening the technical basis for decisionmaking that relies on the SPAR models. This analysis employs the MELCOR computer code, using plant models developed as part of the State-of-the-Art Reactor Consequence Analyses project. This report uses these models for a number of scenarios with different assumptions. In many cases, the operator response is not modeled in order to establish minimal equipment needs or bounding operator action timings. The report clearly articulates all assumptions and limitations.

The analyses summarized in this report provide the basis for confirming or changing success criteria in the SPAR models for the Surry Power Station and Peach Bottom Atomic Power Station. Further evaluation of these results was also performed to extend the results to similar plants. In addition, future work is planned to perform similar analysis for other design classes. In addition, work is planned to scope other aspects of this topical area, including the degree of variation typical in common PRA sequences and the quantification of conservatisms associated with core damage surrogates. The confirmation of success criteria and other aspects of PRA modeling using the agency's state-of-the-art tools (e.g., the MELCOR computer code) is expected to receive continued focus as the agency moves forward in this area.

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ABBREVIATIONS AND ACRONYMS

ac	alternating current
ACC	accumulator
ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AFW	auxiliary feedwater
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BAF	bottom of active fuel
BRK	break
BTU/hr	British thermal units per hour
BWR	boiling-water reactor
C	Celsius
CDF	core damage frequency
CET	core exit thermocouple
CFR	*Code of Federal Regulations*
CL-A	cold leg of loop A
cm	centimeter
COR	MELCOR core package
CRD	control rod drive injection
CST	condensate storage tank
CVH	control volume hydrodynamics (MELCOR package)
dc	direct current
DC	downcomer
Dt	delta temperature
DW	drywell
ECA	emergency contingency action
ECCS	emergency core cooling system
eCST	emergency condensate storage tank
EOP	emergency operating procedure
EPRI	Electric Power Research Institue
F	Fahrenheit
F/hr	Fahrenheit per hour
ft	feet
ft^3	cubic feet
FW	feedwater
gal	gallon(s)
gpm	gallon(s) per minute
HCTL	heat capacity temperature limit
HHSI	high-head safety injection
HL-x	hot leg of loop x
HPCI	high-pressure coolant injection
hr or h	hour(s)
in.	inch(es)
inj or Inj	injection
IORV	inadvertently open relief valve
JP	jet pump
K	Kelvin
kg/s	kilogram(s) per second
lb/s	pound(s) per second
LBLOCA or LLOCA	large-break loss-of-coolant accident
LHSI	low-head safety injection
LOCA	loss-of-coolant accident
LOFT	loss-of-fluid test
LOMFW	loss of main feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
m	meter(s)
m^3	cubic meter(s)
m^3/min	cubic meter(s) per minute
m^3/s	cubic meter(s) per second
MAAP	Modular Accident Analysis Program
MBLOCA or MLOCA	medium-break loss-of-coolant accident
MCP	main coolant pump
MD-AFW	motor-driven auxiliary feedwater
MELCOR	Not an acronym
MFW	main feedwater
min	minute(s)
mm	millimeter(s)
MPa	megapascal(s)
MSIV	main steam isolation valve
MW	megawatt(s)
MWt	megawatt(s) thermal
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
PCT	peak cladding temperature
PORV	power- (or pilot-) operated relief valve
PRA	probabilistic risk assessment
PRT	pressurizer relief tank
PRZ	perssurizer
psi	pound(s) per square inch
psia	pound(s) per square inch absolute
psid	pound(s) per square inch differential

psig	pound(s) per square inch gage
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
rec or Recir	recirculation
RHR	residual heat removal
RPV	reactor pressure vessel
RWST	refueling water storage tank
SBLOCA or SLOCA	small-break loss-of-coolant accident
SC	success criteria
scfm	standard cubic foot/feet per minute
sec or s	second(s)
SG	steam generator
SG-x	steam generator in loop x
SGTR	steam generator tube rupture
SI	safety injection
SOARCA	State-of-the-Art Reactor Consequence Analyses
SP	suppression pool
SPC	suppression pool cooling
SPAR	standardized plant analysis risk
SRV	safety relief valve
SV	safety valve
TAF	top of active fuel
Tavg	loop average temperature
TC	thermocouple
TCL	cladding termperature
TCV	turbine control valve
TD-AFW	turbine-driven auxiliary feedwater
TFU	fuel temperature
TLIQ	liquid temperature
TRACE	TRAC/RELAP5 Advanced Computational Engine
TSAT	saturation temperature
TVAP	vapor temperature
WOG	Westinghouse Owners Group
WW	wetwell

1. INTRODUCTION AND BACKGROUND

The success criteria in the U.S. Nuclear Regulatory Commission's (NRC's) standardized plant analysis risk (SPAR) models are largely based on the success criteria used in the associated licensee probabilistic risk assessment (PRA) model.[1] Licensees have used a variety of methods to determine success criteria, including conservative design-basis analyses and more realistic best-estimate methods. Consequently, in some situations plants that should behave similarly from an accident sequence standpoint have different success criteria for specific scenarios. This issue has been recognized for some time, but until recently the infrastructure was not in place at the NRC to support refinement of these success criteria.

To facilitate improvements in this area, the NRC staff ran MELCOR calculations for specific sequences to provide the basis for confirming or changing the corresponding SPAR models. This analysis used the Surry Power Station (Surry) and the Peach Bottom Atomic Power Station (Peach Bottom). The staff chose these plants because of the availability of mature and well-exercised MELCOR input models arising from the State-of-the-Art Reactor Consequence Analyses (SOARCA) project. The sequences analyzed are not necessarily the most probable sequences because of the assumed unavailability of systems or the assumed lack of operator action. This situation is an appropriate effect of the nature of this work (i.e., the informing of particular pieces of the PRA model). In all cases, this report gives these assumptions in the results description.

This report summarizes the analyses that have been performed, including the following topics:

- the basis for the core damage definition employed
- major plant characteristics for Surry and Peach Bottom
- a description of the two MELCOR models used
- results of various MELCOR calculations
- application of the MELCOR results to the Surry and Peach Bottom SPAR models, as well as to the SPAR models for other similar plants

The success criteria contained herein are intended to be confirmatory in nature, and while suitable for their intended use in supporting the SPAR models they are not intended to be used by licensees for risk-informed licensing submittals.

1 In some cases, success criteria are based on other sources, such as NRC studies (e.g., NUREG/CR-5072, "Decay Heat Removal Using Feed and Bleed for U.S. Pressurized Water Reactors," issued June 1988 (NRC, 1988)).

2. DEFINITION OF CORE DAMAGE

To perform supporting analysis of success criteria, it is necessary to define what is meant by core damage (i.e., sequence success versus failure) because no universal quantitative definition of core damage exists. The American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," issued March 2009 (ASME/ANS, 2009) defines core damage as "uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects." The standard later requires the analysis to specify the plant parameters used to determine core damage in Section 2-2.3, "Supporting Requirement SC-A2" (ASME/ANS, 2009). The core damage surrogate provides the linkage between the qualitative definition above and the quantitative, measurable computer code outputs. The surrogate is necessary since a full Level 3 PRA is not being performed.

For this analysis, the staff ran a number of MELCOR calculations to identify a realistically conservative core damage surrogate. This report does not thoroughly describe the MELCOR models used for this part of the project for the following reasons:

- All results are relative, meaning that a change in the model would generally not be expected to affect the delta-time between the surrogate core damage definition and the onset of rapid cladding oxidation (which is in fact another surrogate, as described further below).

- The model is based on the general-purpose models used in the SOARCA project, which will be documented thoroughly as part of that project.

The analysis used MELCOR version 1.8.6 (NRC, 2005) to assess several possible surrogate definitions for a variety of pressurized-water reactor (PWR) and boiling-water reactor (BWR) accident sequences. For the PWR (Surry Power Station), the following sequences were analyzed:

- station blackout with a 182 gallons per minute (gpm) (0.689 cubic meters per minute (m^3/min)) per reactor coolant pump (RCP) seal leak rate[2]

- station blackout with a 500 gpm (1.89 m^3/min) per RCP seal leak rate

- hot-leg loss-of-coolant accident (LOCA) for 2-inch (in.) (5.1-centimeter (cm)), 4-in.(10.2-cm), and 10-in. (25.4-cm) equivalent diameter break sizes

For the BWR (Peach Bottom Atomic Power Station), the following sequences were analyzed:

- station blackout

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[2] Note that the seal leakage assumptions used in this analysis differ from those used in the SOARCA project (see additional discussion in Section 6.4). Also note that the leakage rate provided here is the leakage rate at full system pressure. As the system depressurizes, the leak rate will decrease.

- recirculation line LOCA for 2-in. (5.1-cm), 6-in. (15.2-cm), and 10-in. (25.4-cm) equivalent diameter break sizes

Because no universal definition of core damage exists, the definition used here for comparison with the surrogates will be the temperature at which the transition occurs in the Urbanic-Heidrick zirconium/water reaction correlation (i.e., a peak cladding temperature (PCT) of approximately 1,580 degrees Celsius (C) (2,876 degrees Fahrenheit (F)) to 1,600 degrees C (2,912 degrees F)). This is the point at which the reaction becomes more energetic, and significant oxidation of the cladding is more likely.

A number of potential surrogates that have traditionally been used in PRAs, several of which are called out in the PRA standard (Section 2-2.3) (ASME/ANS, 2009) were considered. These included various parameters associated with collapsed reactor vessel water level, peak core-exit thermocouple temperature, and PCT. Figure 1 shows the results of the MELCOR calculations to investigate these surrogates. The ordinate axis is the time that the proposed surrogate (e.g., 1,204 degrees C (2,200 degrees F)) is reached, relative to the time that the zirconium/water transition temperature range (1,580 degrees C to 1,600 degrees C) is reached. In all cases but one (the surrogate representing a core exit thermocouple temperature greater than 1,200 degrees F plus a 30-minute offset), the proposed surrogate is reached before the oxidation transition temperature (see "Time Rapid Core Damage" in Figure 1). A PCT of 1,204 degrees C (2,200 degrees F) achieves all of the following characteristics:

- It always precedes oxidation transition.
- It is not overly conservative.
- It is equally applicable for both PWRs and BWRs.
- The timing between 1,204 degrees C (2,200 degrees F) and oxidation transition is relatively similar among the different sequences analyzed.
- It is consistent with the criteria contained in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" (10 CFR, 2007).

With regard to the latter bullet, the conservatism (i.e., safety margin) in 10 CFR 50.46 is due to uncertainty in large-break loss-of-coolant accident (LBLOCA) thermal-hydraulic analysis. For PRA usage, the margin has, in part, a different reason: the desire to have a specific criterion that can be used for all sequences combined with overall analysis uncertainty. For the reasons stated above, a PCT of 1,204 degrees C (2,200 degrees F) is the surrogate used to define core damage for the MELCOR analyses in this report.

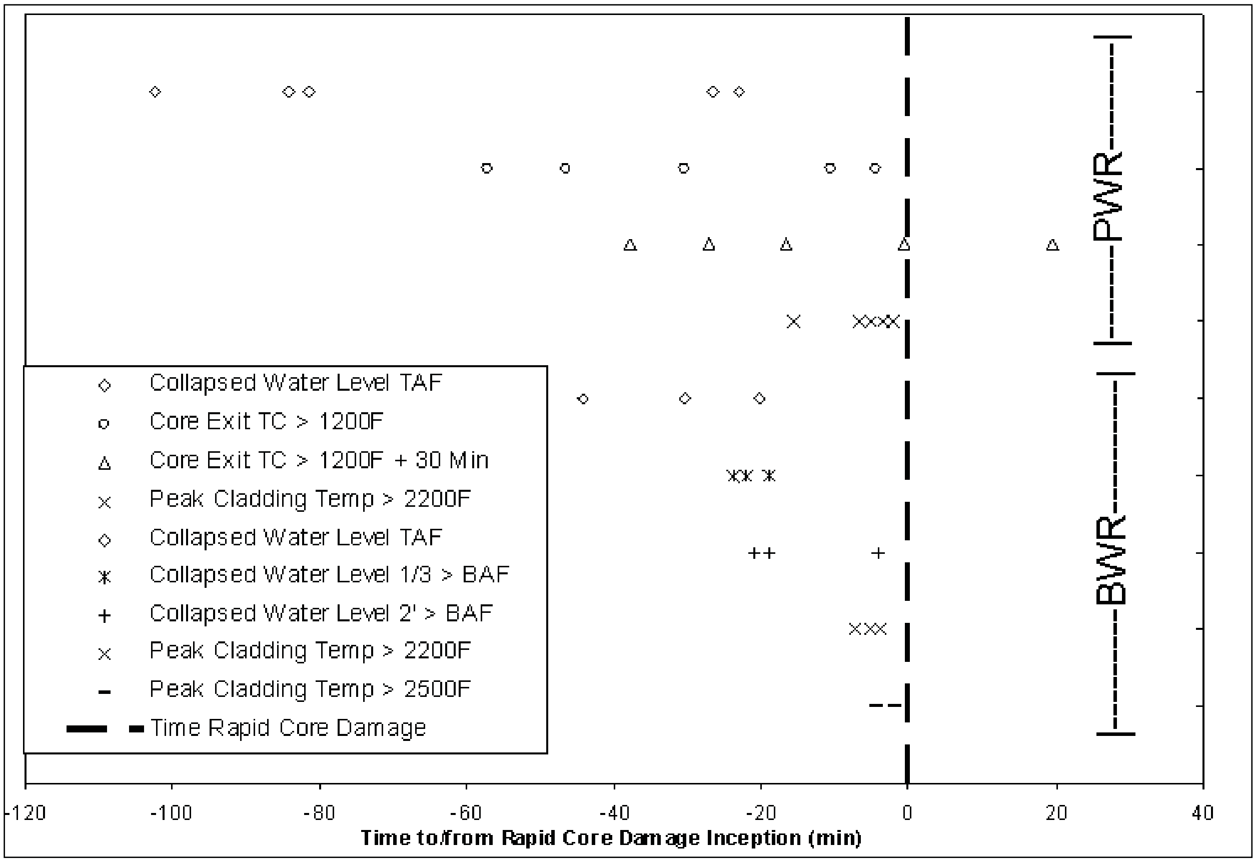

Figure 1 Summary of Core Damage Surrogate Calculations (1,200 °F = 649 °C; 2,200 °F = 1,204 °C; 2,500 °F = 1,371 °C)

3. RELATIONSHIP TO THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS/AMERICAN NUCLEAR SOCIETY PROBABILISTIC RISK ASSESSMENT STANDARD

Core damage specification is one of several aspects of success criteria analysis covered by the ASME/ANS PRA standard (ASME/ANS, 2009). Although the present project is confirmatory in nature, it is still prudent to cross-check the effort against the PRA standard requirements (see Table 1). Capability Category II is used for comparison, since this is the category identified in Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," issued March 2009, as current industry good practice (NRC, 2009). Because the current report focuses primarily on the actual thermal-hydraulic and accident progression analysis and defers the actual PRA model changes for a subsequent report, there are some cases where the comparison to the standard has limited applicability. Table 1 notes these instances as appropriate.

Table 1 Comparison of this Project to the ASME/ANS PRA Standard

PRA Standard Supporting Requirement for Capability Category II	This Project
SC-A1: Use provided core damage definition or justify the definition used.	The standard provides a qualitative core damage definition. The definition used here is believed to be consistent with the definition, but is necessarily quantitative. The basis for the definition (in terms of quantitative accident analysis and comparison of alternatives) is provided. Sensitivity calculations of direct current (dc) power recovery during station blackout have demonstrated that there is not excessive margin in the definition used.
SC-A2: Specify the quantitative surrogate used for core damage and provide basis.	
SC-A3: Specify success criteria for each safety function for each accident sequence.	The existing SPAR model essentially satisfies the requirement. Any changes proposed to the success criteria should not inappropriately remove criteria for important safety functions; this is believed to be the case.
SC-A4: Identify systems shared by units and how they perform during initiating events affecting both units.	In the context of this project, this requirement only applies to changes in which the success criteria is modified to include systems that are shared by multiple units that were not previously in the success criteria. This is not believed to be the case for any of the changes proposed.
SC-A5: Specify the mission times being used (and use appropriate mission times).	These calculations use an overall mission time of 24 hours, when appropriate. For most calculations, either a stable condition has been reached before 24 hours or core damage has been predicted before 24 hours.
SC-A6: Confirm that the bases for the success criteria are consistent with the operating philosophy of the plant.	Many of the specific sequences that are being quantified assume few operator actions. By design, these sequences presume a lack of operator action and do not agree with the operating philosophy of the plant (e.g., emergency operating procedures (EOPs)). In cases in which operator action is being modeled, and in all cases involving system operation, significant effort has been made to ensure that the analyses appropriately mimic the operation of the plant. Cases with ambiguity or limitations are noted. Additional effort has been taken to look at the EOPs, have senior staff review the analyses, have lead SOARCA analysts review the analyses, and so forth.
SC-B1: Use realistic generic analyses evaluations.	For this project, the use of realistic plant-specific analyses means that Capability Category III is being met, though the last clause in Category III about using no assumptions that could yield conservative criteria is debatable.

PRA Standard Supporting Requirement for Capability Category II	This Project
SC-B2: Do not use expert judgment except when sufficient information / analytical methods are unavailable.	Other than cases in which MELCOR models are based on expert judgment, or judgment is used for selecting operator timings, these analyses do not use expert judgment. Some judgment will be inevitable when the analyses are translated to specific changes in the success criteria for other, similar plants.
SC-B3: Use analysis that is appropriate to the scenario and contains the necessary level of detail.	This requirement is clearly met by the use of MELCOR on a sequence-by-sequence basis for the sequences being studied.
SC-B4: Use appropriate models and codes, and use them within their limits of applicability.	MELCOR is not formally assessed in the same manner as a design-basis analysis code, but it does undergo some of the same steps (e.g., comparison of results against relevant experimental results). The documentation for this project provides some high-level information about this assessment but does not attempt to make a comprehensive argument for MELCOR's applicability. In general, MELCOR is considered an appropriate tool for this application. In the case in which its applicability is most ambiguous (i.e., LBLOCA), the extent of calculation margin is addressed.
SC-B5: Confirm that the analyses results are reasonable and acceptable.	All analyses have been reviewed by multiple experienced engineers to confirm that the results are reasonable and acceptable. In addition, the results for many analyses have been compared to similar analyses performed by the SOARCA project. The SOARCA lead PWR analyst reviewed all results in the interim report. Results for station blackout were compared to similar Westinghouse calculations. Results for Surry feed and bleed were compared to similar TRACE calculations.
SC-C1: Document the analyses to support PRA applications, upgrades, and peer review.	The analyses are being comprehensively documented. The judgment used in applying the analyses as the basis for making specific SPAR model changes will be documented separately.
SC-C2: Document the overall analysis comprehensively, including consideration of a provided list of documentation areas.	In general, the level of documentation being provided with these analyses is consistent with this Supporting Requirement. The one area that is currently weak is the discussion of limitations of MELCOR. Specific MELCOR applicability assessments for each initiator are beyond the scope of this confirmatory analysis.
SC-C3: Document the sources of model uncertainty and related assumptions.	This has not been formally done, except that a general sense of modeling uncertainty prompted some of the additional analyses (e.g., RCP seal LOCA model). Another aspect that has received consideration is the relationship between uncertainty and the margin in a given calculation. For example, MELCOR may have higher uncertainty in the modeling of LBLOCAs. Of the 15 Surry LOCA cases with a break size ≥15 cm (6 in.), the highest PCT for a case that was deemed to be successful is 812 degrees C (1,494 degrees F), about 400 degrees C below the core damage definition. This suggests that, for these cases, a higher degree of uncertainty is acceptable because there is significant margin.

4. MAJOR PLANT CHARACTERISTICS

The following subsections describe the aspects of the analyzed plants that are germane to the analysis performed in this report.

4.1 Surry Power Station

To the level of detail needed for this analysis, Surry Units 1 and 2 were considered to be identical. Each unit is a three-loop Westinghouse with a subatmospheric containment. Each has three high-head safety injection (HHSI) pumps and two low-head safety injection (LHSI) pumps. The latter are also required for high-pressure recirculation (in order to provide sufficient net positive suction head (NPSH) to the high-head pumps when using the containment sump as a water source). The minimum technical specification refueling water storage tank (RWST) volume is 387,100 gallons (gal) (1,470 cubic meters (m^3)). The water source for the emergency core cooling system (ECCS) automatically transfers from the RWST to the containment sump when the RWST water level drops below 13.5 percent.[3] This transfer operation takes 2.5 minutes because of the time it takes for the sump isolation valves to fully open.[4]

The containment spray system in injection mode relies on two pumps rated at 3,200 gpm (12.1 m^3/min) per pump (which includes approximately 300 gpm (1.14 m^3/min) per pump of bleed-off flow[5]) and draws from the RWST. Containment spray automatically actuates at 25 pounds per square inch absolute (psia) (0.17 megapascal (MPa)) containment pressure, and the operators are directed by the EOPs to secure (and reset) containment sprays once containment pressure drops back below 12 psia (0.083 MPa). The containment spray system in recirculation mode uses four pumps (two in containment and two outside of containment) that are each rated at 3,500 gpm (13.2 m^3/min) and take suction from the containment sump.[6] Table 2 summarizes major plant characteristics.

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3 Note that the relationship between RWST volume and percent inventory is not intuitive because zero percent corresponds to about 14,000 gal (53 m^3), 13.5 percent corresponds to 66,000 gallons (250 m^3), about 97 percent corresponds to the technical specification limit, and 100 percent corresponds to 399,000 gal (1,510 m^3). Also note that switchover requires two-out-of-four RWST low level signals coincident with Recirculation Mode Transfer switches selected in the proper position.

4 The MELCOR input model does not model the effects of this delay in terms of RWST inventory reduction.

5 This bleed-off flow goes to the suction of the outside containment recirculation spray pumps to ensure that adequate NPSH is available.

6 Note that successful sump recirculation function requires containment heat removal through the recirculation spray system.

Table 2 Major Plant Characteristics for Surry

Characteristic	Value
Design Type	Three-loop Westinghouse
Containment Type	Subatmospheric
Power Level	2,546 MWt[1]
Number of HHSI Pumps	Three
Number of HHSI Trains	Two
Shutoff Head for HHSI	5,905 ft / 2,560 psi (17.65 MPa)
Lowest PORV Opening/Closing Setpoint	2,350 psi (16.2 MPa) / 2,260 psi (15.6 MPa)
Number of Cold-Leg Accumulators	One per loop (three total)
Nominal Operating Pressure	2,250 psia (15.5 MPa)
RWST Volume Technical Specification	387,000 gal (1,470 m^3)

[1] The power level used in this report is the power level before the October 2010 measurement uncertainty recapture power uprate of 1.6%. In general, the results in this report are not expected to be sensitive to a power change of this amount, or the potential adjustments to protection system, control system, and operating procedure set-points associated with the change. However, this is a qualitative assertion, as only the calculations in Section 6.2 consider a higher power level (and in that case it is a much higher power level).

4.2 Peach Bottom Atomic Power Station

As with Surry, to the level of detail needed for this analysis, Peach Bottom Units 2 and 3 were considered to be identical. Both are General Electric BWR/4s with Mark-I containment. Peach Bottom's reactor core isolation cooling (RCIC) system has a capacity of 600 gpm (2.3 m^3/min) at 150 to 1,150 pounds per square inch gage (psig) (1.0 to 7.9 MPa). The high-pressure coolant injection (HPCI) system capacity is 5,000 gpm (18.9 m^3/min). The condensate storage tank (CST) is the preferred source until a low level in the CST (less than 5 feet (1.5 meters)) causes an automatic switchover to the suppression pool. The RCIC and HPCI turbines will automatically trip with a high turbine exhaust pressure of 50 psig and 150 psig (0.34 and 1.03 MPa), respectively. RCIC and HPCI systems will automatically isolate with a low steamline pressure of 75 psig (0.51 MPa). RCIC and HPCI pump bearings are rated for 210 degrees F (99 degrees C). The high-capacity low-pressure coolant injection (LPCI) system has a shutoff head of 295 psig (2.0 MPa). The volume of the CST is 200,000 gal (756 m^3). The suppression pool has a technical specification maximum temperature limit of 95 degrees F (35 degrees C) and a volume of 127,300 cubic feet (3,605 m^3). Major plant characteristics are summarized in Table 3.

Table 3 Major Plant Characteristics for Peach Bottom

Characteristic	Value
Design Type	General Electric Type 4
Containment Type	Mark 1
Power Level	3,458 MWt
RCIC Capacity	600 gpm (2.27 m^3/min)
HPCI Capacity	5,000 gpm (18.9 m^3/min)
Lowest SRV Opening/Closing Setpoint in Relief Mode	1,133.5 psid[1] (7.81 MPa) / 1,099.5 psid (7.58 MPa)
Nominal Operating Pressure	1,050 psia (7.24 MPa)
Suppression Pool Inventory	952,000 gal (3,605 m^3)

[1] Pounds per square inch differential (psid) is the differential pressure in psi between the main steamline and the wetwell.

5. MELCOR MODEL

5.1 Plant Representation

The Surry and Peach Bottom models used for this analysis are based on the models utilized in the SOARCA study. Efforts to ensure that the models appropriately reflect the as-built, as-operated plant included discussions with plant operation and engineering staff, site visits, and review of plant documentation and operating procedures. Detailed documentation of the models will be provided in the near future as part of that project and is therefore not duplicated in this report. In some cases, additional information (e.g., additional containment spray trip logic) was added to the SOARCA model to address systems and sequence characteristics needed for this study that were not needed for the SOARCA study. For RCP seal leakage, the models used in this analysis differ from those used in the SOARCA analysis. The modeling of RCP seal leakage is described in the section on the Surry station blackout analysis later in this report (Section 6.4). Below is a brief overview of the Surry and Peach Bottom models, followed by some discussion of MELCOR's validation base.

Appendix A of this report outlines the basic features of the Surry model. Included are the reactor trip signals modeled, the ECCS injection setpoints, the HHSI and LHSI pump curves, details of the switchover of ECCS suction from the RWST to the containment sump, accumulator characteristics, containment spray system characteristics, containment fan cooler characteristics, and relief valve setpoints.

Figure 2 shows a plan view of the MELCOR model for the Surry reactor coolant system (RCS). All three RCS loops are modeled individually. The detailed nodalization of the RCS loop piping as well as the reactor core and vessel upper plenum allows modeling of the in-vessel and hot-leg counter-current natural circulation during core heatup. This feature has been shown to be relevant even within the temperature ranges of interest in the analysis (i.e., those preceding core damage). The RCPs are tripped on power failure or voiding (related to pump vibration) in the loop.[7] The core region is nodalized into 10 axial thermal response nodes (the MELCOR core package (COR)) mapped to 5 axial hydrodynamic volumes (the MELCOR control volume hydrodynamics package (CVH)), and is comprised of 5 radial rings. Safety systems are modeled using injection points, and the relevant portions of the reactor protection system and control systems are modeled using MELCOR control functions. For the secondary side, both turbine-driven auxiliary feedwater (TD-AFW) and motor-driven auxiliary feedwater (MD-AFW) are modeled (including provisions for water level control). The core decay power is based on a number of ORIGEN calculations for each radial ring. The containment is divided into nine control volumes representing the major compartments. Containment sprays and fan coolers are also modeled.

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[7] Since the present analyses do not credit operator actions to trip the RCPs early in the transient (for cases in which procedures would direct this action), a global void fraction in the vicinity of the pumps of 10 percent is selected to represent a condition in which pump cavitation would prompt shutdown of these pumps. A system-level code such as MELCOR does not have the capability to directly model actual pump performance under degraded conditions.

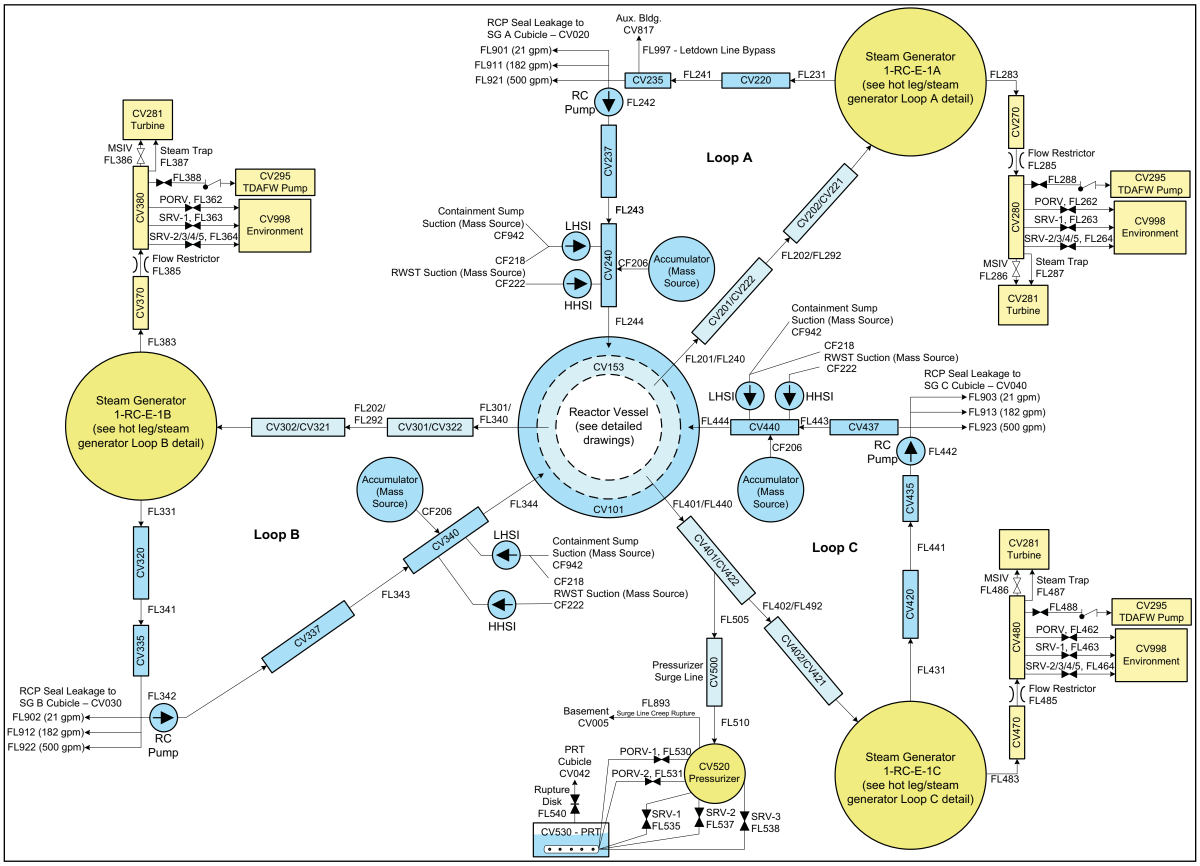

Figure 2 Plan View of the Surry MELCOR RCS Model

Figure 3 shows a schematic of the Peach Bottom MELCOR model, including the reactor pressure vessel (RPV), wetwell, and safety systems. The drywell (not shown) has four control volumes representing the pedestal, lower drywell, upper drywell, and upper head regions. The vessel (excluding the core region) is represented by seven control volumes with connections to various safety systems, including control rod drive injection (CRD), RCIC, HPCI, low-pressure core spray (LPCS), and residual heat removal (RHR) (vessel injection and containment cooling modes). The models for HPCI and RCIC include separate control volumes for the turbine exhausting into the suppression pool. All safety relief valves (SRVs), including dedicated automatic depressurization system (ADS) valves, are modeled with flowpaths on two steamlines (a single steamline A and a combined steamline for B, C, and D). The core nodalization is similar to the Surry model, with 10 axial levels (with a 2:1 COR:CVH ratio) and five radial rings. Like the Surry model, the core decay power is based on a number of ORIGEN calculations for each radial ring. Because very few changes were made to the SOARCA model, Appendix B of this report does not include the same introductory plant model information for Peach Bottom as Appendix A does for Surry.

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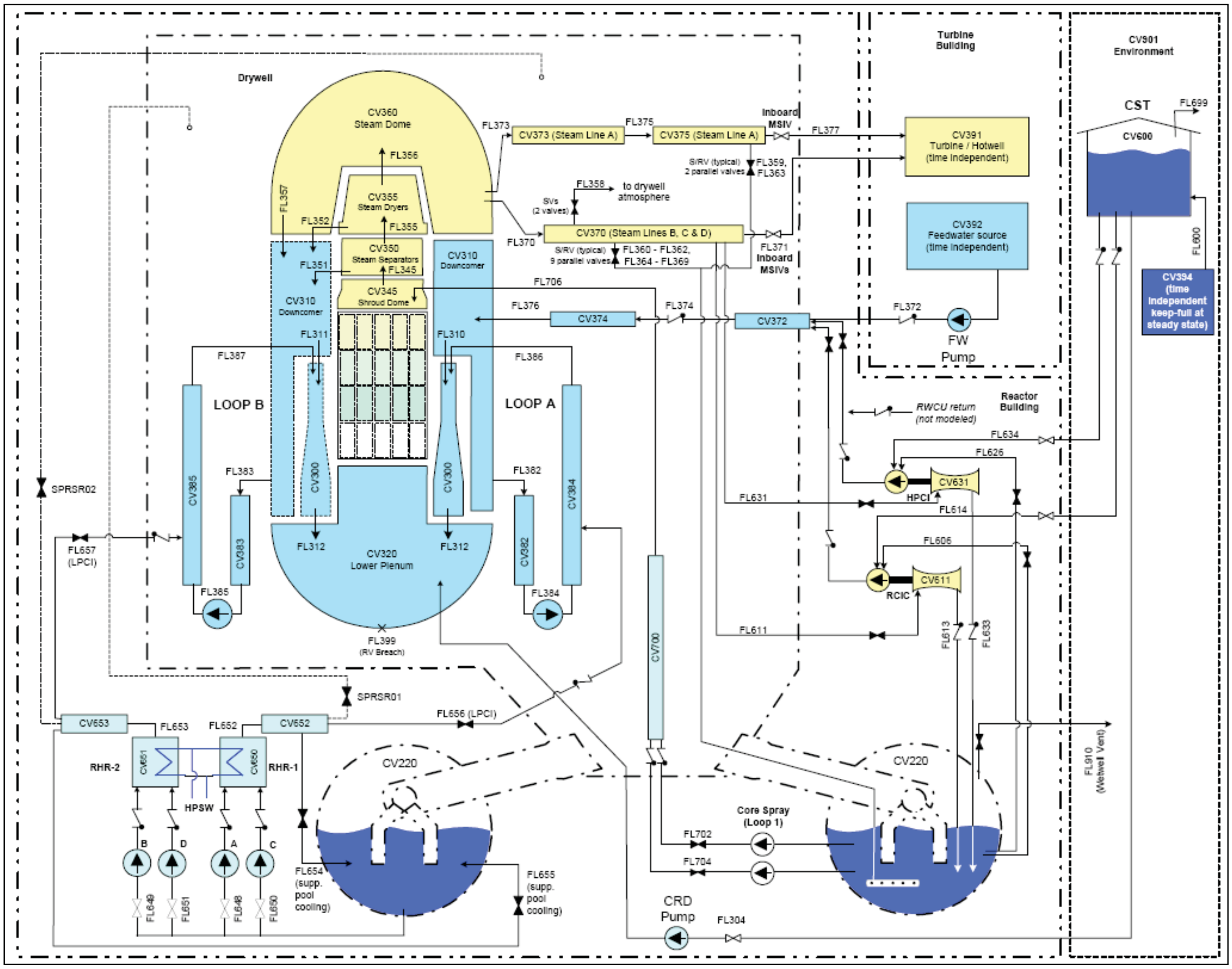

Figure 3 Schematic of the Peach Bottom RCS Nodalization

To model failure of pressurizer power-operated relief valves (PORVs) or SRVs, one of three approaches is used, as designated in the boundary condition descriptions for each case: (1) the relief valve cannot stick open, (2) the relief valve sticks open on the first lift, or (3) the relief valve sticks open after *n* lifts, where *n* is a user-prescribed number. The purpose of the third approach is to provide intermediate results (relative to the two extremes), for assessing the variation in plant response. Generally speaking, the SPAR models treat the situation in a binary fashion—the valve is either stuck open or it is not.

For the purposes of this analysis, a simplified treatment of valve cycling and failure is adopted for this intermediate situation. Table 4 and Table 5 provide a synopsis of the basis for the values used for Surry, including the specific value used for each type of valve. These tables also provide comparative values from the Surry Individual Plant Examination and a more recent relief valve reliability study (NRC, 2011). The values in Table 5 are tabulated using the following formulas:

, or

where P_D equals the probability of failure per demand and n equals the number of lifts. This report used a median value (cumulative probability equal to 0.5). Two key limitations associated with the way this report treats failures due to valve cycling are (1) the use of a constant failure probability per demand and (2) the assumption that the failure probability is the same regardless of whether the valve is passing steam, water, or a two-phase mixture.

Table 4 Comparison of Failure per Demand Probabilities for Surry Stuck-Open Valves

	Probability of Sticking Open per Demand			
Valve	**Surry Individual Plant Examination**	**NUREG/CR-7037 Automatic Demand (Initial/Subsequent)**	**NUREG/CR-7037 Liquid Demand (Initial/Subsequent)[1]**	**Circa 2006 Surry PRA (used in the present analyses)**
Pressurizer PORV	0.0123	0.00495 / 0.00275	0.0625 / 0.000715	0.0028
Main Steamline PORV	0.0123	0.00295 / 0.0109	N/A	0.0058
Pressurizer SRV	0.0123	0.5 / N/A[2]	N/A	0.0027
Main Steamline SRV	0.0123	0.0270 / 0.00254	N/A	0.0027

1 No liquid demands were witnessed for the main steamline valves or for the SRVs, so no estimates could be given.

2 Note that only four demands were observed. No subsequent demands on the pressurizer SRV were witnessed, so no estimates could be given.

Table 5 Comparison of Number-of-Lifts Values for Surry Stuck-Open Valves

	# of Lifts for Cumulative Probability of Sticking Open = 0.5			
Valve	**Surry Individual Plant Examination**	**NUREG/CR-7037 Automatic Demand**	**NUREG/CR-7037 Liquid Demand**	**Circa 2006 Surry PRA (used in the present analyses)**
Pressurizer PORV	56	251	880	247
Main Steamline PORV	56	64	N/A	119
Pressurizer SRV	56	1[2]	N/A	256
Main Steamline SRV	56	263	N/A	256

1 No liquid demands were witnessed for the main steamline valves or for the SRVs, so no estimates could be given.

2 Note that only four initial demands and no subsequent demands on the pressurizer SRV were witnessed, so this value is highly sensitive to the availability of sparse data.

The value used for Peach Bottom was 187 lifts, which corresponds to a cumulative failure probability of 0.5 for a probability of failure per demand of 0.0037, in comparison to the values from NUREG/CR-7037, which gave an observed behavior after scram value of 956 lifts. (These values correspond to a failure per demand of 7.08×10^{-4} and 7.25×10^{-4}, for initial and subsequent demands.) For the liquid demand, the observed behavior in NUREG/CR-7037 after scram had a value of 79 lifts, which corresponds to a failure per demand of 8.77×10^{-3}. For both Surry and Peach Bottom, the values used for failure of the relief valves due to cycling may differ from the values used in the SOARCA study. It is also useful to point out that the valve temperatures associated with the high-temperature seizure failure mechanism being considered under the SOARCA study correspond to fuel temperatures reached after significant heatup (generally at or beyond the time of initial core damage). Since the present study only considers the phase of the accident up and until the start of core damage, this valve failure mechanism is not believed to be relevant for this analysis.

5.2 MELCOR Validation

The MELCOR code is designed to run best-estimate accident simulations (NRC, 2005). The code has been assessed against a number of experiments and plant calculations. The current test suite for MELCOR contains over 170 separate input decks. MELCOR has been used for final safety analysis report audit calculations (related to engineered safety feature design and performance, containment design and performance, design-basis accident analysis, and severe accident analysis); the post-September 11, 2001, security assessments; and the SOARCA project. It has also been used to assess significance determination process issues. For these reasons, it is an ideal tool to use in this project.

Specific experiments and plant calculations relevant to this project for which MELCOR has been assessed include the following:

- Quench experiment 11, simulating a small-break loss-of-coolant accident (SBLOCA) with late vessel depressurization to investigate response of overheated rods under flooding conditions (Hering, 2007)

- the Three Mile Island Unit 2 accident (NRC, 1980)

- loss-of-fluid test (LOFT) LP-FP-2, simulating an LBLOCA (Adams, 1985)

- Russian Academy of Sciences MEI experiments involving a spectrum of LOCA sizes to study critical flow and vessel response (e.g., Dementiev, 1977)

- NEPTUN experiments to test pool boiling models and void fraction treatment (NRC, 1992)

- General Electric level swell and vessel blowdown experiments characterizing single- and two-phase blowdown, liquid carryover, and water level swell (e.g., Appendix A to NRC, 1981)

- General Electric Mark III tests with steam blowdown into the suppression pool investigating vent clearing and heat transfer models

- containment thermal-hydraulic phenomena studied in various experimental facilities, including Nuclear Power Engineering Corporation for mixing and stratification (e.g., NUPEC, 1993), Heissdampfreaktor for blowdown into containment, and Carolinas-Virginia Tube Reactor for steam condensation in the presence of noncondensables (SNL, 2008)

- small-scale experiments to test condensation models, including Wisconsin flat plate experiments (e.g., Huhtinemi, 1993) and Dehbi tests

6. MELCOR RESULTS

The detailed results for Surry and Peach Bottom are provided in Appendices A and B, respectively.[8] The following subsections summarize these results in a standard format: (1) a brief description of the scenario, (2) a list of key assumptions and operator actions, (3) a table of results, and (4) a table of the timing to key events.

The analysis evaluated a number of different scenarios. The following scenarios were analyzed for Surry:

- SBLOCAs to investigate the time available until RWST depletion and core damage
- feed and bleed (during loss of all feedwater) to investigate the number of pressurizer PORVs and HHSI pumps needed
- steam generator tube rupture (SGTR) events to provide updated accident sequence timings
- station blackout events to provide updated accident sequence timings
- medium- and large-break LOCAs to look at the systems needed for successful inventory control during the injection phase

The following scenarios were analyzed for Peach Bottom:

- inadvertent open relief valve cases to investigate the effects of various sources of high-pressure injection
- station blackout events to investigate the time for alternating current (ac) power recovery, the time for suppression pool heatup, and the times associated with the loss of turbine-driven high-pressure systems

In many cases, the analyzed sequence progressions make assumptions about the unavailability of systems and about operator actions that are not taken. These assumptions often stem from the particular sequence in the event tree that is being studied, which may not be the most probable sequence. In other cases, these characteristics are not included because of resource constraints. In all cases, the relevant subsections below note these assumptions. Section 6 of this report places these analyses in the context of the associated SPAR models.

6.1 Small-Break Loss-of-Coolant Accident Dependency on Sump Recirculation (Surry)

This series of cases investigates the timing to RWST depletion (and thus switchover to recirculation) for SBLOCAs in which operators take very few actions. In reality, the operators would enter procedure E-0, "Reactor Trip or Safety Injection" (e.g., verify reactor and turbine trip, verify mitigative system availabilities and alignments), transition to E-1, "Loss of Reactor or Secondary Cooling" (e.g., reduce RCS injection flow, initiate evaluation of plant status), and

[8] Plots of reactor vessel water level in Appendices A and B show the actual water level (i.e., they include two-phase effects where appropriate).

later transition to ES-1.2, “Post LOCA Cooldown and Depressurization” (e.g., dump steam to condenser, depressurize RCS to refill pressurizer).

The varied parameters are break size (0.5 in. (1.3 cm), 1 in. (2.5 cm), and 2 in. (5.1 cm)), the assumption on relief valve sticking, and containment spray function (available or not available). In all 12 cases investigated, the break location is the horizontal section of the cold leg. In addition, sensitivity cases were conducted to look at the effects of securing HHSI pumps (Cases 2a and 6a) and performing secondary-side cooldown (Cases 2b and 6b). These sensitivity cases demonstrate the impact of HHSI and secondary-side cooldown on RCS pressure and RHR entry timing. Because of project resource considerations, the modeling uses a simplified scoping approach and does not necessarily represent the actual plant operating procedures.[9] For this reason, the results should be used with caution. Results are provided in Table 6, Table 7, and Table 8. In addition to the key timing tables below, plots for various results of interest are provided in Appendix A, Section A.2.

For the 2-in. (5.1-cm) breaks investigated, the RCS depressurizes as a result of the break. The loss of high-head injection following RWST depletion (high-head recirculation was not modeled) further reduces the primary side pressure to less than the maximum pressure for LHSI recirculation; thus, HHSI recirculation is not necessary. The same is true for 0.5-in. (1.3-cm) breaks when the PORV is assumed to stick open after 247 lifts (see Table 5) because this causes the 0.5-in. (1.3-cm) break to become a 1.9-in. (4.8-cm) break.[10] Note that operator action to reduce injection (in response to PORV cycling) and thus limit pressurizer PORV cycling was not modeled. Also note that some cases do include throttling HHSI for the purpose of scoping operator actions to depressurize and cool down. For the 0.5-in. (1.3-cm) cases in which the PORV does not stick open, the system does not depressurize. Finally, for the 1-in. (2.54-cm) cases, the break is not large enough to cause depressurization (because of HHSI injection) and the PORV does not open. As a result, the system pressure is still high at the time of RWST depletion. Loss of HHSI at RWST depletion causes depressurization, but not enough to allow for LHSI recirculation.

Key assumptions and operator actions in these calculations include the following:

- For the 0.5-in. (1.3-cm) breaks, the PORV sticks open after 247 cycles unless (1) it does not lift that many times (Case 6b) or (2) noted otherwise (Cases 7 and 8).

- Operators do not throttle injection for the purpose of preventing valve chattering, which is relevant for 0.5-in. (1.3-cm) breaks.

- Operators do not take action to refill the RWST.

- Prior to RWST depletion, operators secure containment sprays (and reset to allow subsequent actuation) in accordance with the EOPs after containment pressure drops below 12 psia (0.083 MPa).

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9 Specifically, for these cases the model assumes that at 30 minutes the third HHSI pump is secured and the steam generator (SG) PORV is opened on all three SGs to an opening fraction that will result in a cooldown of approximately 100 degrees F per hour (55.6 degrees C per hour) on the secondary side (corresponding to a similar cooldown on the primary side). This differs from the operating procedures that utilize more complex approaches depending on the exact situation (e.g., isolating HHSI, establishing and controlling normal charging flow, using pressurizer sprays).

10 The equivalent diameter of the PORV is 1.39 in. (3.53 cm).

- RCPs trip at 10-percent voiding (see Section 5.1 for more information on this modeling assumption and Appendix D for more information on its effect).

- HHSI recirculation is not modeled. Operator actions for manual cooldown and depressurization are not modeled, except in a simplified manner for sensitivity Cases 2b and 6b.

- MD-AFW and TD-AFW is available.

- Accumulators are available. Note that this assumption is not typical for SBLOCA success criteria analysis. It is not expected to affect the end-state results, but could affect some intermediate timings. See Appendix D for additional information.

Table 6 Surry SBLOCA Sump Recirculation Results

<table>
<tr><th>Case</th><th>Size (inch)[5]</th><th>HHSI Pumps</th><th>PORV Treatment</th><th>Sprays</th><th>Secondary-Side Cooldown</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td rowspan="4">1</td><td rowspan="2">3</td><td rowspan="6">N/A</td><td>0</td><td rowspan="3">No</td><td>9.2[1]</td><td>11.9[1]</td></tr>
<tr><td>2</td><td rowspan="3">2</td><td>7.3[1]</td><td>9.9[1]</td></tr>
<tr><td>2a[2]</td><td>3/1</td><td>7.9[1]</td><td>10.0[1]</td></tr>
<tr><td>2b[3]</td><td>3/1/0</td><td>Yes</td><td>No[4]</td><td>No[4]</td></tr>
<tr><td>3</td><td rowspan="2">2</td><td rowspan="4">3</td><td>0</td><td rowspan="5">No</td><td>No</td><td>No</td></tr>
<tr><td>4</td><td>2</td><td>No</td><td>No</td></tr>
<tr><td>5</td><td rowspan="6">0.5</td><td rowspan="3">Sticks open after 247 lifts</td><td>0</td><td>No</td><td>No</td></tr>
<tr><td>6</td><td rowspan="3">2</td><td>No</td><td>No</td></tr>
<tr><td>6a[2]</td><td>3/1</td><td>8.8[1]</td><td>9.6[1]</td></tr>
<tr><td>6b[3]</td><td>3/1/0</td><td>N/A</td><td>Yes</td><td>No[4]</td><td>No[4]</td></tr>
<tr><td>7</td><td rowspan="2">3</td><td rowspan="2">Does not stick open</td><td>0</td><td rowspan="2">No</td><td>17.8[1]</td><td>25.1[1]</td></tr>
<tr><td>8</td><td>2</td><td>14.4[1]</td><td>21.4[1]</td></tr>
</table>

[1] Core damage is an artifact of the assumed unavailability of HHSI recirculation.
[2] It is assumed that two HHSI pumps are secured at 15 minutes.
[3] It is assumed that two HHSI pumps are secured at 15 minutes, and the third pump is secured at 30 minutes, followed by secondary-side cooldown at 100 degrees F per hour (55.6 degrees C per hour).
[4] These cases reach RHR entry conditions (both temperature and pressure) before heatup.
[5] 1 in. = 2.54 cm; 2 in. = 5.1 cm; 0.5 in. = 1.3 cm.

Table 7 Surry SBLOCA Sump Recirculation Key Timings (Cases 1–4)

Event	Case 1 (hr)	Case 2 (hr)	Case 2a (hr)	Case 2b (hr)	Case 3 (hr)	Case 4 (hr)
Reactor trip	0.03	0.03	0.03	0.03	0.01	0.01
HHSI injection	0.03	0.03	0.03	0.03	0.01	0.01
LHSI injection	-	-	-	2.02	-	-
First actuation of contain. sprays	-	2.65	3.29	-	-	1.76
RWST depletion (<13.5%)	5.83	4.30	5.80	-	3.12	2.63
Spray recirculation	-	4.30	5.80	-	-	2.63
LHSI recirculation	-	-	-	-	3.38	2.86
Accumulator starts to inject	6.00	4.52	5.83	0.82	0.23	0.23
RCP trip (10% void)	7.38	5.76	6.73	1.41	-	-
Core uncovery	9.23	7.32	7.9	-	-	-
Core damage (max. temp. >2,200 °F)[1]	11.9	9.93	10.0	-	-	-

[1] 2,200 °F = 1,204 °C

Table 8 Surry SBLOCA Sump Recirculation Key Timings (Cases 5–8)

Event	Case 5 (hr)	Case 6 (hr)	Case 6a (hr)	Case 6b (hr)	Case 7 (hr)	Case 8 (hr)
Reactor trip	0.01	0.01	0.01	0.01	0.01	0.01
HHSI injection	0.01	0.01	0.01	0.01	0.01	0.01
LHSI injection	-	-	-	3.49	-	-
PORV stuck open	0.83	0.83	4.65	-	-	-
First actuation of contain. sprays	-	2.20	5.30	-	-	3.23
RWST depletion (<13.5%)	4.14	3.43	7.45	-	8.17	5.52
Spray recirculation	-	3.43	7.45	-	-	5.53
LHSI recirculation	4.72	3.97	-	-	26.6	-
Accumulator starts to inject	4.14	3.43	7.14	1.11	8.28	5.65
RCP trip (10% void)	-	4.68	5.00	13.8	11.7	10.3
Core uncovery	-	-	8.77	-	17.8	14.4
Core damage (max. temp. >2,200 °F)[1]	-	-	9.61	-	25.1	21.4

[1] 2,200 °F = 1,204 °C

6.2 Feed-and-Bleed Power-Operated Relief Valve Success Criteria (Surry)

The initiating event of interest for these calculations is loss of main feedwater (LOMFW). Additionally, auxiliary feedwater is assumed unavailable. The parameter of interest is how many pressurizer PORVs need to be available for the feed-and-bleed procedure to be effective at removing decay heat. The injection source is HHSI (initially from RWST) and the bleed path is the PORVs. Repeated actuation of the PORV leads to an increase in the pressure in the pressurizer relief tank (PRT). Following failure of the PRT rupture disk, primary side coolant exiting the PORV passes into containment, resulting in an increase in containment pressure. Containment sprays actuate once containment pressure reaches the containment spray setpoint.

For these analyses, no operator actions are modeled except for securing containment sprays. Regarding the actual expected operator response for a loss of all feedwater event, the operators would enter E-0, “Reactor Trip or Safety Injection” (e.g., verify reactor and turbine trip, verify mitigative system availabilities and alignments), transition to ES-0.1, “Reactor Trip Response” (e.g., attempt to establish feed flow, control pressurizer pressure), and later enter FR-H.1, “Response to Loss of Secondary Heat Sink” (e.g., establish feed from condensate system, manually initiate bleed and feed) based on the associated critical safety function status tree. For the purpose of determining the effectiveness of a single PORV for removing decay heat, the lack of operator action is conservative (i.e., delayed initiation of HHSI). However, these results should be used with caution for determining the time to RWST depletion (and thereby switchover to recirculation) because for that aspect this assumption may be nonconservative (i.e., earlier initiation of HHSI may lead to earlier RWST depletion depending on the interplay with containment spray actuation).

The cause of the reactor trip is varied for three cases to scope the effect of the different trip criteria that exist for the set of high-head three-loop Westinghouse plants in operation. In all cases, safety injection (SI) does not start until an auto-SI signal occurs due to high containment pressure. The power level is also varied to scope the effect of higher decay power, because Surry has the lowest power level of the high-head three-loop Westinghouse plants in operation. The cases that used a power level of 13.9 percent higher than Surry’s power level correspond to a power level of 2,900 megawatts thermal (MWt), which corresponds to the upper range of the three-loop plants.

The analysis performed here demonstrates that one PORV provides a sufficient bleed path to maintain quasi-steady conditions on the primary side.[11] Further, it is not necessary for the operators to manually open the PORV, as the HHSI at Surry will cause the valve to automatically open due to high pressure. Even in the absence of operator action, the capacity of one HHSI pump is sufficient to remove decay heat for either the Surry or elevated (e.g., Virgil C. Summer Nuclear Station) power levels. Nevertheless, it is important to note that other differences between Surry and the higher power-level three-loop plants (most notably the type of steam generator (SG)) have not been addressed.

In the absence of further operator action, these cases do eventually proceed to core damage in these analyses because HHSI recirculation (which would actuate upon RWST depletion) is not modeled. However, at least 8 hours is available prior to RWST depletion, and an additional 3.5 to 4 hours is available until core damage occurs. This timing information can be used to inform related sequences that include human failure events associated with refilling the RWST or aligning the HHSI water source to the containment sump. In addition to the results and key timings in Table 9 and Table 10 below, plots for various results of interest are provided in Appendix A, Section A.3.

Key assumptions and operator actions in these calculations include the following:

- Prior to RWST depletion, operators secure containment sprays (and reset to allow subsequent actuation) in accordance with the EOPs after containment pressure drops below 12 psia (0.083 MPa).

- HHSI recirculation is not modeled; thus, the time to core damage is driven by RWST depletion (the timing of which is affected by the assumption that operators do not take early action to start HHSI).

- The PORV is aligned for automatic operation and opens when the RCS pressure increases above the high pressure setpoint (i.e., no manual operator action). For all calculations performed, the PORV had cycled roughly 200 times as of the time of core damage (i.e., fewer times than required for the valve to stick open for the cycling failure model used in this report). Were high-head recirculation to have been modeled, the valve would have eventually reached the required number of cycles for failure. Such a failure (if treated) would probably not impact the ability of the high-head safety injection pumps to maintain primary-side inventory, based on a qualitative assessment of the results in this section, as well as Cases 27 and 29 of Section 6.5.

- Manual RCS depressurization and cooldown is not modeled.

- RCPs trip at 10-percent voiding; in actuality, Function Restoration Procedure FR-H.1 would have the operators stop all RCPs. See additional information in Appendix D regarding the sensitivity to this assumption.

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11 Note that for Cases 2 and 3, SRV1 briefly lifts because of the actuation of HHSI (PORV 2 was disabled for the calculation). This brief actuation is judged to be inconsequential to the overall progression of the event.

Table 9 Surry Feed-and-Bleed PORV Success Criteria Results

Case	Power Level[1]	Cause of Reactor Trip[2]	Cause of SI	# HHSI Pumps	# of Pressurizer PORVs	Core Uncovery (hr)	Core Damage
1	100%	MFW trip	High Cont. Press.	1	1	No[3]	No[3]
2		Low SG level + feed/steam mismatch				1.65	No[3]
3	113.9%	Low-low SG level				1.60	No[3]

1 100 percent equals 2,546 MWt (Surry), and 113.9 percent equals 2,900 MWt (Beaver Valley, Harris, and Summer); 2,900 MWt is the highest present power level of the three-loop Westinghouse plants.

2 Low SG level is <19 percent of narrow-range span, while low-low SG level is <16 percent of narrow-range span, based on Technical Specification 2.3-3 (January 2008).

3 Core uncovery and damage late in the simulation are artifacts of the assumed unavailability of HHSI recirculation.

Table 10 Surry Feed-and-Bleed PORV Success Criteria Key Timings

Event[1]	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)
MFW, MD-AFW, TD-AFW unavailable	0	0	0
Reactor trip	0	0.008 (29 s)	0.008 (27 s)
SG dryout	1.11	0.63	0.58
PRT rupture disk open	1.56	0.97	0.93
SI signal (containment pressure >1.22 bars)	1.96	1.36	1.29
RCP trip (10% void)	2.05	1.43	1.35
First actuation of containment sprays (containment pressure >1.72 bars)	3.84	3.24	3.17
RWST depletion (<13.5%)	9.43	8.35	8.24
Core uncovery	10.90[2]	1.65 / 9.54[2]	1.60 / 9.42[2]
Core damage (max. temp. >2,200 °F)	13.53	11.80	11.68

1 1.22 bars = 0.122 MPa; 1.72 bars = 0.172 MPa; 2,200 °F = 1,204 °C.

2 For Case 1, the core comes close to uncovering around the time of SI actuation, then later does uncover after the loss of HHSI. For Cases 2 and 3, the core uncovers early in the accident, recovers prior to significant heatup, and later uncovers again (due to the loss of HHSI).

6.3 Steam Generator Tube Rupture Event Tree Timing (Surry)

These calculations assess the time available to take corrective actions for events involving spontaneous (as opposed to accident-induced or consequential) tube rupture events. In addition to the results and key timings in Table 11 and Table 12 below, plots for various results of interest are provided in Appendix A, Section A.4. For reference, the effective leak size of a one-tube rupture is about a 1-in. (2.5 cm) effective diameter. Past operating experience for SGTR events suggests that, in some cases, the time between the initiating event and initiation of RHR can be significant (e.g., this timing ranges from 3.25 hours to 21.5 hours for the events covered in a study conducted in the mid-1990s)[12]. Here, very few operator actions are assumed. In reality, the operators would be expected to enter E-0, “Reactor Trip or Safety Injection” (e.g., verify reactor and turbine trip, verify mitigative system availabilities and alignments), transition to E-3, “Steam Generator Tube Rupture” (e.g., initiate RCS cooldown, depressurize RCS and terminate SI to minimize primary-to-secondary leakage), and later transition to one of three post-SGTR procedures (based on plant conditions).

12 “Steam Generator Tube Failures,” NUREG/CR-6365, April 1996.

Even with few operator actions assumed, the results provided below show that the availability of secondary-side heat removal allows a substantial amount of time for corrective actions. At 24 hours, the fuel temperatures for all five cases are stable at less than 550 degrees F (288 degrees C), although additional actions would be eventually required (e.g., refilling the CST). For the first three cases, the faulted SG relief valves are not allowed to stick open, despite cycling a large number of times (e.g., >15,000). For Cases 4 and 5, the faulted SG relief valve sticks open after 119 cycles (see Table 5), which occurs within the first hour for both cases. Even in these cases, the availability of SI early in the accident and MD-AFW later in the accident results in times to core damage greater than 24 hours.

Key assumptions and operator actions in these calculations include the following:

- Main steamline isolation valves close on reactor trip.
- Operators secure either one or two HHSI pumps at 15 minutes (depending on the case) and manually control auxiliary feedwater to maintain SG level (standard practice).
- For Cases 1 through 3, the faulted SG PORV does not stick open regardless of the number of lifts and regardless of whether it passes water. In all other situations, the SG PORVs stick open after 119 cycles (see Table 5).
- HHSI recirculation is not modeled.
- RCPs trip at 10-percent voiding.
- Manual isolation of the faulted SG is not assumed (i.e., operators fail to perform this action).
- Manual actions to model long-term heat removal (EOP Emergency Contingency Action (ECA) 3.1/3.2) are not modeled.

Table 11 Surry SGTR Results

<table>
<tr><th>Case</th><th>No. Tubes</th><th>HHSI Pumps</th><th>SG PORV Treatment</th><th>TD-AFW</th><th>MD-AFW</th><th>Nominal Break Flow Prior to Loss of HHSI (kg/sec)</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td>1</td><td rowspan="2">3/2</td><td rowspan="3">Does not stick open[1]</td><td colspan="2" rowspan="5">Yes[2]</td><td>30</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>2</td><td>5</td><td>50 – 60</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>3</td><td rowspan="2">1</td><td>3/1</td><td>23</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>4</td><td rowspan="2">3/2</td><td rowspan="2">Sticks open after 119 lifts</td><td>30 – 40</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>5</td><td>5</td><td>60 - 70</td><td>No[3]</td><td>No[3]</td></tr>
</table>

1 Logic was added to address numerical instability (by limiting the flow area to smooth the liquid flow through the faulted SG PORV).
2 TD-AFW is lost within the first hour for all cases due to flooding of the steamline.
3 The response is based on a 24-hour mission time.

Table 12 Surry SGTR Key Timings

Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)	Case 4 (hr)	Case 5 (hr)
Reactor Trip	0.048	0.012	0.048	0.048	0.012
HHSI initiates (3 pumps)	0.051	0.013	0.051	0.051	0.013
1 of 3 HHSI pumps secured	0.25	0.25	N/A	0.25	0.25
2 of 3 HHSI pumps secured	N/A	N/A	0.25	N/A	N/A
TD-AFW shut down[1]	0.70	0.32	0.75	0.70	0.32
Faulted SG PORV stuck open	N/A	N/A	N/A	0.76	0.35
RWST depletion (<13.5%)[2]	10.68	5.58	14.06	8.41	4.69
Accumulator injection	N/A	N/A	N/A	8.62	0.94
RCP trip (10% void)	17.81	11.71	20.20	12.44	5.02
Emergency CST empty[3]	> 24 hours	> 24 hours	> 24 hours	> 24 hours	22.20
Core damage	> 24 hours				

1 TD-AFW shuts down due to filling of the steamline and flooding of the pump.

2 Recall that since the RCS leak location is the ruptured SG tube(s), a substantial amount of water is expelled from the system via the SG relief valves (rather than into containment) and is thus unavailable for containment sump recirculation.

3 Depletion of the emergency CST (96,000 gal (363 m^3)), which is the normal injection source for AFW, stops MD-AFW.

6.4 Pressurized-Water Reactor Station Blackout (Surry)

A number of simulations were run for station blackout sequences to investigate the effects of RCP seal failures, SRV operation, and TD-AFW availability and operation on the time available to recover ac power and re-establish core cooling. Along with the above variations in system conditions and responses, some other factors that affect the time to core damage are the time to battery depletion (loss of direct current (dc) power), the time to depletion of the emergency CST tank (for cases with TD-AFW available), the system pressure, and the occurrence of natural circulation (Case 4). Cases 4 and 6 assume dc power is always available, which mimics successful “blind feeding” of the SGs using TD-AFW following the loss of dc (see (West., 2008) for more information on this topic). Meanwhile, Cases 9 and 10 assume the loss of TD-AFW at 4 hours, which equals the station blackout coping time for Surry from NUREG-1776, “Regulatory Effectiveness of the Station Blackout Rule,” issued August 2003 (NRC, 2003a).

In the EOPs, the operators would first enter E-0, “Reactor Trip or Safety Injection” (e.g., verify reactor and turbine trip, verify ac emergency buses energized), which would direct them to ECA-0.0, “Loss of All AC Power” (verify AFW flow, try to restore power to any ac emergency bus). If ac power is recovered, the operators will transition to ECA-0.1, “Loss of All AC Power Recovery without SI Required” and/or ECA-0.2, “Loss of All AC Power Recovery with SI Required” (e.g., restore necessary injection sources, restore component cooling). If ac power is not recovered and the core-exit thermocouples rise past 1,200 degrees F (649 degrees C), the operators will transition to SACRG-1, “Severe Accident Control Room Guideline Initial Response” (e.g., check if RCS should be depressurized, determine containment spray requirements).

The Surry SPAR model does not credit operation of auxiliary feedwater following battery depletion. Further, the SPAR model assumes core damage at the time of battery depletion (i.e., no further opportunity for recovering ac power and averting core damage). This assumption exists because dc power is an integral part of ac power recovery, in that it provides the control power to operate electrical distribution system breakers in order to bring electrical power into the power block following a station blackout. Alternate sources of dc control power are required once batteries are depleted in a station blackout sequence, but this issue is not further explored in this report.

The RCP seal leakage rates and timing are taken from the Westinghouse Owners Group (WOG) 2000 seal leakage model for "new" high-temperature seals used in the current Surry SPAR model, which is described in WCAP 15603, "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," issued May 2003 (West., 2003), as modified by the NRC staff's associated April 2003 safety evaluation report (NRC, 2003b).[13] The safety evaluation report for WCAP-15603 makes a few modifications to the WCAP-15603 model, including the disallowance of credit for the third RCP seal. The resulting model has outcomes associated with four possible leakage rates for use in PRAs, with the onset of increased leakage occurring at 13 minutes in all cases. Table 13 reproduces the leakage rates and their conditional probabilities, along with some associated timings from the Westinghouse Emergency Response Guidelines as reproduced in the Surry SPAR v3.52 model documentation of July 2008. The current analysis ran cases for three of these leakage sizes (21 gpm per pump (0.079 m^3/min), 182 gpm per pump (0.689 m^3/min), and 500 gpm per pump (1.89 m^3/min)).[14]

Table 13 Reactor Coolant Pump Seal Leakage Details

Seq. #	Leak Rate at >13 Minutes (gpm)[2]	Conditional Probability	Time to Core Uncovery Based on Westinghouse Emergency Response Guidelines[1]	
			Without Depressurization	With Depressurization
1	21	0.79	~13 hours	~22 hours
3	76	0.01	~7 hours	~9 hours
2	182	0.1975	~3 hours	~5 hours
4	480	0.0025	~2 hours	~2.5 hours

1 Assumes availability of TD-AFW

2 21 gpm = 0.079 m^3/min; 76 gpm = 0.29 m^3/min; 182 gpm = 0.689 m^3/min; 480 gpm = 1.82 m^3/min

The results of the present analysis are in good agreement with those from the Westinghouse Emergency Response Guidelines (Table 13). For analogous cases (i.e., those with TD-AFW available and no secondary-side depressurization) the following conditions apply:

- Time to core uncovery is about 1.5 hours for the largest leakage rate of 500 gpm/RCP (1.89 m^3/min/RCP), as compared to 2 hours in the Westinghouse calculations.
- Time to core uncovery is about 4 hours for the intermediate leakage rate of 182 gpm/RCP (0.68 m^3/min/RCP), as compared to 3 hours in the Westinghouse calculations.
- Time to core uncovery is about 13 hours for the normal leakage rate of 21 gpm/RCP (0.079 m^3/min/RCP), which is identical to the Westinghouse calculations.

13 This is the same model that is invoked in a later PRA guidance topical report, WCAP-16141, "WOG 2000 RCP Seal Leakage PRA Model Implementation Guidelines for Westinghouse PWRs," issued August 2003.

14 In accordance with convention, these leak rates correspond to full system pressure. Actual leak rates will be substantially lower once system pressure decreases. Note that the figures for RCP seal leakage in Appendix A are designed to demonstrate this fact. An unfortunate side effect of plotting these leakage rates as a volumetric flow rate (as opposed to a mass flow rate) is that the plots go off scale once the flow becomes two phase.

The current MELCOR calculations demonstrate an additional 0.5 to 3 hours between the time of core uncovery and the time of core damage.

Topical report WCAP-16396-NP, "WOG 2000 Reactor Coolant Pump Seal Performance for Appendix R Solutions," issued January 2005 (West., 2005), discusses why the NRC's safety evaluation of the WOG 2000 model—and the WOG 2000 model itself—result in conservative estimates of RCP seal leak rates. These conservatisms are associated with both the leak rates assumed and the timing of seal failure (which is reported to vary from 8 minutes to 40 minutes, as compared with the 13 minutes used in the WOG 2000 model). This topical report quantitatively assesses the effects of these conservatisms on accident progression timings (specifically, the time for loss of pressurizer level and core uncovery). The topical report concludes that the conservatisms can substantially affect the assessment of coping strategies, but that the conservatisms are "unlikely to affect any conclusions drawn from PRA models for internal events from at-power conditions" (West., 2005) These conclusions led to the decision not to request NRC review of a less conservative model. If applied in this case, these conclusions suggest that the timings to core damage calculated are conservative, but that these conservatisms will not affect the overall conclusions drawn from the models. Even so, the potential conservatisms could affect intermediate PRA results, such as the human error probability associated with a particular action.

For the timing of ac power recovery needed to avert core damage, two sensitivity cases were run for Case 1:

- recovery of HHSI at 2.14 hours (i.e., at the onset of core damage based on a PCT of 2,200 degrees F (1,204 degrees C))

- recovery of HHSI at 1.64 hours (i.e., half an hour before core damage)

As shown in Figure 5, the sensitivity case in which HHSI was recovered at 2.14 hours occurred too late to avert fuel melting. For the case in which HHSI was recovered at 1.64 hours, recovery of injection was sufficient to avert fuel melting.[15] A best-estimate time could be developed by running calculations using an intermediate time (e.g., 15 minutes) for this case, as well as running similar sensitivities for other cases. In addition to the results and key timings in Table 14 through Table 17, and Figure 4 below, plots for various results of interest are provided in Appendix A, Section A.5.

Key assumptions and operator actions in these calculations include the following:

- Operators manually control auxiliary feedwater to maintain SG level (standard practice).

- DC power is always available for control of TD-AFW for Cases 4 and 6 (i.e., mimics successful blind feeding).

- Operator actions to refill the emergency CST are not modeled.

- SRV sticks open on the first lift for some cases (as specified below).

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15 Note that the sensitivity studies correspond to a case in which the seals fail at 13 minutes. As such, failure of the seals from thermal shock upon recovery of seal cooling is not pertinent to this particular case. In addition, Surry has proceduralized isolation of the RCP seals as part of ECA-0.0, "Loss of All AC Power."

- For cases with RCP seal failure, failure is assumed to occur at 13 minutes.[16]
- Manual operator actions for rapid secondary-side depressurization are not modeled.

Table 14 Surry Station Blackout Results

<table>
<tr><th>Case</th><th>Seal Leakage Rate[1] after Failure (gpm[3] per pump)</th><th>Seal Failure Time (min)</th><th>SRV Stuck Open</th><th>TD-AFW</th><th>ac/dc</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td rowspan="4">500</td><td rowspan="4">13</td><td rowspan="6">N/A[2]</td><td rowspan="3">Fails to start</td><td>-</td><td>1.4</td><td>2.1</td></tr>
<tr><td>1a</td><td>ac recovery at 2.1 hours</td><td>1.4</td><td>2.1</td></tr>
<tr><td>1b</td><td>ac recovery at 1.6 hours</td><td>1.4</td><td>-</td></tr>
<tr><td>2</td><td>Available</td><td rowspan="7">-</td><td>1.6</td><td>2.3</td></tr>
<tr><td>3</td><td rowspan="4">21</td><td rowspan="4">-</td><td>Fails to start</td><td>2.3</td><td>3.4</td></tr>
<tr><td>4</td><td>Available; successful blind feeding</td><td>13.3</td><td>16.3</td></tr>
<tr><td>5</td><td rowspan="2">1st lift</td><td>Fails to start</td><td>2.1</td><td>2.6</td></tr>
<tr><td>6</td><td>Available; successful blind feeding</td><td>13.0</td><td>13.8</td></tr>
<tr><td>7</td><td rowspan="2">182</td><td rowspan="2">13</td><td rowspan="3">N/A[2]</td><td>Fails to start</td><td>2.0</td><td>3.1</td></tr>
<tr><td>8</td><td>Available</td><td>3.9</td><td>4.8</td></tr>
<tr><td>9</td><td rowspan="2">21</td><td rowspan="2">-</td><td rowspan="2">Available; lost at 4 hours</td><td rowspan="2">dc lost at 4 hours</td><td>8.4</td><td>10.9</td></tr>
<tr><td>10</td><td>1st lift</td><td>8.1</td><td>8.8</td></tr>
</table>

1 The leakage rate provided here is the leakage rate at full system pressure. As the system depressurizes, the leak rate decreases.

2 The model is set to stick the valve open after 256 lifts, but the valve does not lift that many times for these calculations.

3 500 gpm = 1.89 m^3/min;182 gpm = 0.689 m^3/min; 21 gpm = 0.076 m^3/min.

16 Note that this differs from the seal failure model used in the SOARCA project, which employed a more mechanistic approach (saturated conditions at the pump) to model the timing of seal failure.

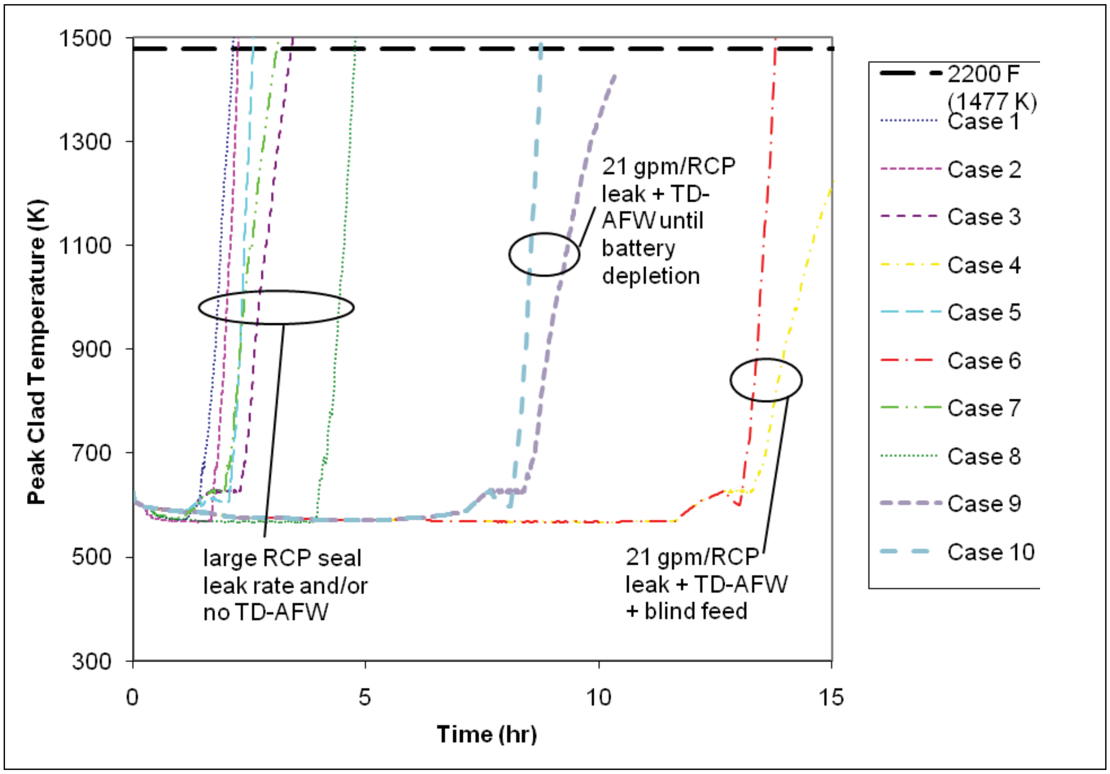

Figure 4 PCT Signatures for all Surry Station Blackout Cases

Table 15 Surry Station Blackout Key Timings (Cases 1–2)

Event[1]	Case 1 (hr)	Case 1a (hr)	Case 1b (hr)	Case 2 (hr)
Reactor trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0	0	0
Seal leakage (21 gpm/pump)	0	0	0	0
Seal failure (500 gpm/pump)	0.22	0.22	0.22	0.22
Primary side SG tubes water level starts to decrease	0.52	0.52	0.52	0.52
Primary side SG tubes dry	0.96	0.96	0.96	0.98
SG dryout	1.16	1.16	1.16	-
Core uncovery	1.40	1.40	1.40	1.63
Gap release	1.92	1.92	-	2.15
Core damage (max. temp. >2,200 °F)	2.14	2.14	-	2.25

[1] 500 gpm = 1.89 m^3/min; 21 gpm = 0.076 m^3/min; 2,200 °F = 1,204 °C.

Table 16 Surry Station Blackout Key Timings (Cases 3–6)

Event[1]	Case 3 (hr)	Case 4 (hr)	Case 5 (hr)	Case 6 (hr)
Reactor trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0	0	0
Seal leakage (21 gpm/pump)	0	0	0	0
Primary side SG tubes water level starts to decrease	1.92	5.38	1.52	5.42
Emergency CST depleted	-	7.97	-	7.97
Primary side SG tubes dry	2.03	11.30	1.66	11.30
SG dryout	1.19	11.77	1.19	11.80
SRV sticks open	N/A	N/A	1.45	12.71
Core uncovery	2.28	13.31	2.06	13.03
Gap release	2.96	14.83	2.42	13.60
Core damage (max. temp. >2,200 °F)	3.40	16.33	2.57	13.80

[1] 21 gpm = 0.076 m^3/min; 2,200 °F = 1,204 °C.

Table 17 Surry Station Blackout Key Timings (Cases 7–10)

Event[1]	Case 7 (hr)	Case 8 (hr)	Case 9 (hr)	Case 10 (hr)
Reactor trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0	0	0
Seal leakage (21 gpm/pump)	0	0	0	0
Seal failure (182 gpm/pump)	0.22	0.22	-	-
TD-AFW assumed lost at battery depletion	-	-	4	4
Primary side SG tubes water level starts to decrease	1.04	1.01	5.62	5.63
Primary side SG tubes dry	1.52	2.22	6.58	6.58
SG dryout	1.22	-	7.13	7.12
SRV sticks open	N/A	N/A	N/A	7.67
Core uncovery	1.98	3.88	8.37	8.10
Gap release	2.63	4.00	9.48	8.59
Core damage (max. temp. >2,200 °F)	3.09	4.77	10.85	8.77

[1] 182 gpm = 0.689 m^3/min; 21 gpm = 0.076 m^3/min; 2,200 °F = 1,204 °C.

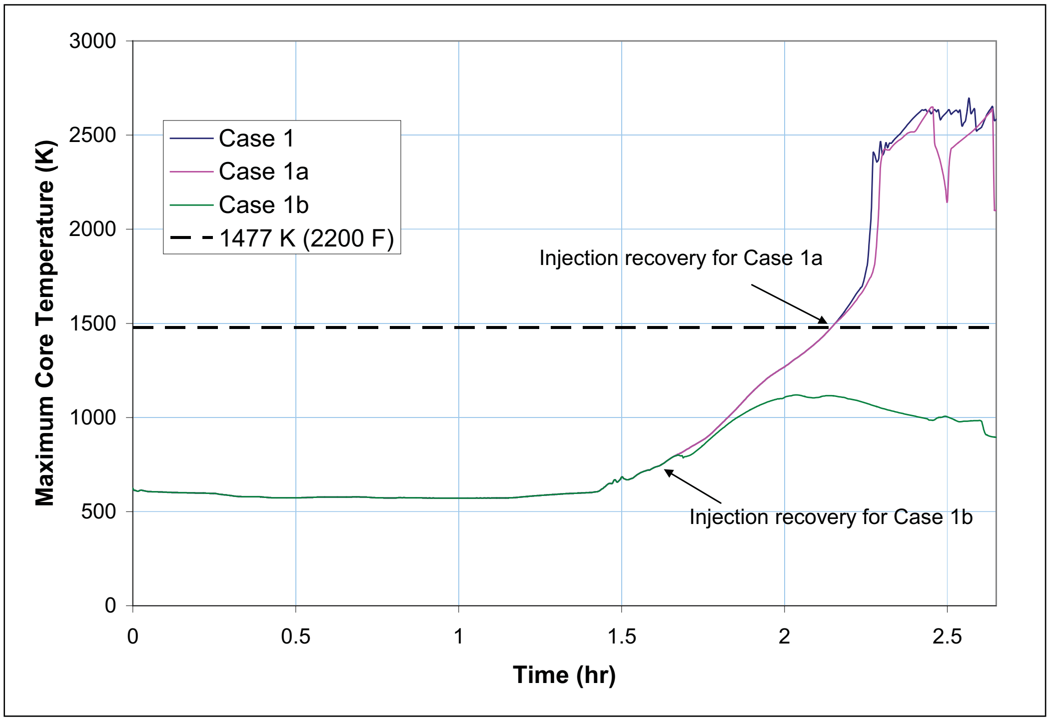

Figure 5 Surry Injection Recovery Sensitivity Cases

6.5 Pressurized-Water Reactor Medium- and Large-Break Loss-of-Coolant Accident Initial Response (Surry)

The final set of Surry sequences investigated combinations of accumulators, HHSI, and LHSI for a spectrum of LOCA break sizes for the early phase of the accident (e.g., the first few hours). Break sizes from 2 in. (5.1 cm) to a double-ended break were analyzed, as shown in Table 19. Although some calculations are simulated into the long-term cooling phase, the calculations are only intended to inform success criteria for the early injection phase of the accident.

By convention, the breakdown in the LOCA spectrum for most Westinghouse PWRs is 0.5 in. (1.3 cm) to 2 in. (5.1 cm) (SBLOCA), 2 in. (5.1 cm) to 6 in. (15.2 cm) (medium-break LOCA (MBLOCA)), and 6 in. (15.2 cm) and greater (LBLOCA). The break location for the current analyses is always the horizontal section of the cold leg in the pressurizer loop. Very few operator actions are modeled. In reality, the operators would enter E-0, “Reactor Trip or Safety Injection” (e.g., verify reactor and turbine trip, verify mitigative system availabilities and alignments) and transition to E-1, “Loss of Reactor or Secondary Coolant” (e.g., check if containment sprays should be secured and reset, check if accumulators should be isolated). Depending on the course of the accident, the operators would then transition to one of several ES-1.x series supplemental emergency procedures.

As can be seen below, some of these accidents progress very quickly, with core uncovery taking place within the first minute (for LBLOCAs). Since quickly evolving accidents can be more challenging to simulate from a thermal-hydraulic standpoint, it is of interest to look at the degree of margin between the PCT (for cases that are deemed successful) and the core damage definition being used. Table 18 presents these figures, demonstrating that the highest MBLOCA PCT (for a success case) is 483 degrees F (268 degrees C) from the core damage definition used here, and the highest LBLOCA PCT (for a success case) is 706 degrees F (392 degrees C) from the core damage definition. This demonstrates that there is significant margin in these cases, which helps to counteract the additional model uncertainty that might be expected for these quickly evolving accidents. In addition to the key timings in Table 20 through Table 26 below, plots for various results of interest are provided in Appendix A, Section A.6.

Table 18 PCT Ranges for Accumulator Success Cases

Range of Break Size	Range of PCT for Success Cases	Range of Margin: 2,200 °F PCT (1,204 °C PCT)
MBLOCA (2 in. to 6 in.)	575–1,717 °F (302–936 °C)	483–1,625 °F (268–902 °C)
LBLOCA (6 in. to double-ended)	575–1,494 °F (302–812 °C)	706–1,625 °F (392–902 °C)

The results in Table 19 are distilled here to identify the minimal equipment needed to avoid core damage *during the injection phase*. For MBLOCAs, the minimal equipment is the following:

- For 6-in. (15.2-cm) breaks, the analyses demonstrate that any two of the following three would be adequate: one HHSI, one accumulator in an intact loop, and one LHSI, with or without AFW.

- For 4-in. (10.2-cm) breaks, Case 13 demonstrates that one accumulator in an intact loop and one LHSI are not adequate, leaving two remaining success paths that are successful for this break size: one HHSI and one accumulator in an intact loop, or one HHSI and one LHSI, with or without AFW.

- For 2-in. (5.1-cm) breaks, both of the above criteria are sufficient, with or without AFW.

The resulting minimal equipment success criteria *for the injection phase* for MBLOCAs is one HHSI and either one accumulator in an intact loop or one LHSI. Note that the former criterion would not be sufficient for the recirculation phase because LHSI is necessary to accomplish HHSI recirculation. AFW is not needed for success for an MBLOCA *for the injection phase*; the break size is large enough to remove decay heat.

For LBLOCAs, the minimal equipment is the following:

- For 6-in. (15.2-cm) breaks, the analyses demonstrate that any two of the following three would be adequate: one HHSI, one accumulator in an intact loop, and one LHSI, with or without AFW.

- For 8-in. (20.3-cm) breaks, Cases 3, 18, and 23 confirm the above.

- For 10-in. (25.4-cm) breaks, Cases 4, 19, and 24 confirm the above.

- For a double-ended break, Case 10 demonstrates that only LHSI is necessary. A case was not run to determine if one HHSI and one accumulator in an intact loop would have been sufficient. As noted above, such a combination would not permit recirculation.

The resulting minimal equipment success criteria *for the injection phase* for LBLOCAs are one LHSI and either one accumulator in an intact loop or one HHSI. AFW is not needed for success for an LBLOCA; the break size is large enough to remove decay heat and the system fully depressurizes.

Key assumptions and operator actions in these calculations include the following:

- The break is in the horizontal section of the cold leg in the pressurizer loop.
- The RCPs trip at 10-percent voiding.
- HHSI recirculation is not modeled. Operator actions to depressurize and perform secondary side cooldown are not modeled.
- Containment sprays are available for all cases (same actuation pressure and operator actions to secure as in Section 6.1 and 6.2).

Table 19 Surry MBLOCA and LBLOCA Results

Case	Break Size (in.)[4]	# HHSI Pumps	# Accum.	# LHSI Pumps	AFW?[1]	Time of Initial Core Uncovery (hr)	Core Damage During Injection Phase? (hr)
9	2	1	0	0	Yes	0.42	No[2]
15		0	2	1		0.41	0.73
20		1	1	0		0.42	No[2]
21		1	0	1		0.42	No[3]
27		1	1	0	No	0.38	No[2]
29		1	0	1		0.38	No[3]
1	4	1	0	1	Yes	0.09	No
11		1	0	0		0.09	No[2]
12		0	0	1		0.10	0.27
13		0	1	1		0.10	0.27
14		0	2	1		0.10	No
22		1	1	0		0.09	No[2]
25		1	0	1	No	0.09	No
28		1	1	0		0.09	No[2]
2	6	1	0	1	Yes	0.04	No
5		0	0	1		0.04	0.16
6		0	1	1		0.04	No
7		1	0	0		0.07	0.28
8		1	1	0		0.08	No[2]
16		1	0	1	No	0.04	No
17		1	1	0		0.06	No[2]
26		0	1	1		0.04	No
3	8	1	0	1	Yes	0.02	No
18		1	1	0		0.01	No[2]
23		0	1	1		0.03	No
4	10	1	0	1		0.01	No
19		1	1	0		0.01	No[2]
24		0	1	1		0.02	No
10	Double-ended	0	0	1		0.02	No

1 Conventionally, AFW is not needed for success for an LBLOCA; the break size is large enough to remove decay heat and the system fully depressurizes.

2 Note that core damage eventually occurs (or would occur, in cases in which the calculation was terminated early) because of the inability to go to HHSI recirculation (due to the unavailability of LHSI)

or, more directly, from the lack of a low-pressure injection source. Recall that the present calculations are focused only on the injection phase success criteria.

3 For these cases, core damage eventually occurs because HHSI recirculation is not modeled, and the pressure is not sufficiently low prior to core damage to allow for LHSI recirculation.

4 2 in. = 5.1 cm; 4 in. = 10.2 cm; 6 in. = 15.2 cm; 8 in. = 20.3 cm; 10 in. = 25.4 cm.

Table 20 Surry MBLOCA and LBLOCA Key Timings (2-in. Breaks)

Event	Case 9 (hr)	Case 15 (hr)	Case 20 (hr)	Case 21 (hr)	Case 27 (hr)	Case 29 (hr)
Reactor trip	0.01	0.003	0.01	0.01	0.01	0.01
HHSI injection	0.01	-	0.01	0.01	0.01	0.01
RCP trip (10% void)	0.28	0.07	0.28	0.28	0.18	0.17
First actuation of containment sprays	1.14	-	1.21	1.14	0.94	0.94
Core uncovery (water < TAF)	0.42	0.41	0.42	0.42	0.38	0.38
LHSI injection	-	-	-	6.39	-	6.17
Maximum cladding temperature timing (max. temperature)	0.44 (592 K)	0.73 (1,477 K[1])	0.44 (592 K)	0.44 (592 K)	0.40 (592 K)	0.40 (592 K)
Core covered	0.87	N/A	0.8	0.87	0.75	0.75

1 Actual peak temperature would be higher; this value corresponds to the surrogate used in this project for core damage—2,200 °F (1,204 °C).

Table 21 Surry MBLOCA and LBLOCA Key Timings (4-in. Breaks Group 1)

Event	Case 1 (hr)	Case 11 (hr)	Case 12 (hr)	Case 13 (hr)
Reactor trip	0.003	0.003	0.003	0.003
HHSI injection	0.003	0.004	-	-
RCP trip (10% void)	0.04	0.04	0.04	0.04
First actuation of containment sprays	0.08	0.08	0.07	0.07
Core uncovery (water < TAF)	0.09	0.09	0.10	0.10
LHSI injection	0.29	-	0.33	0.45
Maximum cladding temperature timing (max. temperature)	0.34 (982 K)	0.53 (1,209 K)	0.27 (1,477 K[1])	0.27 (1,477 K[1])
Core covered	0.38	>0.83	N/A	N/A

1 Actual peak temperature would be higher; this value corresponds to the surrogate used in this project for core damage—2,200 °F (1,204 °C).

Table 22 Surry MBLOCA and LBLOCA Key Timings (4-in. Breaks Group 2)

Event	Case 14 (hr)	Case 22 (hr)	Case 25 (hr)	Case 28 (hr)
Reactor trip	0.003	0.003	0.003	0.003
HHSI injection	-	0.004	0.004	0.004
RCP trip (10% void)	0.04	0.04	0.04	0.03
First actuation of containment sprays	0.07	0.08	0.08	0.07
Core uncovery (water < TAF)	0.10	0.09	0.09	0.09
LHSI injection	0.73	-	0.30	-
Maximum cladding temperature timing (max. temperature)	0.73 (1,183 K)	0.21 (807 K)	0.32 (1,054 K)	0.26 (721 K)
Core covered	0.79	0.39	0.39	0.41

Table 23 Surry MBLOCA and LBLOCA Key Timings (6-in. Breaks Group 1)

Event	Case 2 (hr)	Case 5 (hr)	Case 6 (hr)	Case 7 (hr)
Reactor trip	0.002	0.002	0.002	0.002
HHSI injection	0.002	-	-	0.002
RCP trip (10% void)	0.02	0.02	0.02	0.02
First actuation of containment sprays	0.03	0.03	0.03	0.03
Core uncovery (water < TAF)	0.04	0.04	0.04	0.07
LHSI injection	0.13	0.14	0.18	-
Maximum cladding temperature timing (maximum temperature)	0.15 (774 K)	0.16 (1,477 K[1])	0.16 (990 K)	0.28 (1,477 K[1])
Core covered	0.19	N/A	0.20	N/A

[1] Actual peak temperature would be higher; this value corresponds to the surrogate used in this project for core damage—2,200 °F (1,204 °C).

Table 24 Surry MBLOCA and LBLOCA Key Timings (6-in. Breaks Group 2)

Event	Case 8 (hr)	Case 16 (hr)	Case 17 (hr)	Case 26 (hr)
Reactor trip	0.002	0.002	0.002	0.002
HHSI injection	0.002	0.002	0.002	-
RCP trip (10% void)	0.02	0.02	0.02	0.02
First actuation of containment sprays	0.03	0.03	0.03	0.03
Core uncovery (water < TAF)	0.08	0.04	0.06	0.04
LHSI injection	-	0.13	-	0.18
Maximum cladding temperature timing (maximum temperature)	0.04 (592 K)	0.152 (775 K)	0.04 (575 K)	0.13 (931 K)
Core covered	0.10	0.19	0.12	0.22

Table 25 Surry MBLOCA and LBLOCA Key Timings (8-in. Breaks)

Event	Case 3 (hr)	Case 18 (hr)	Case 23 (hr)
Reactor trip	0.002	0.002	0.002
HHSI injection	0.002	0.002	-
RCP trip (10% void)	0.009	0.009	0.01
First actuation of containment sprays	0.01	0.01	0.01
Core uncovery (water < TAF)	0.02	0.01	0.03
LHSI injection	0.07	-	0.08
Maximum cladding temperature timing (maximum temperature)	0.10 (851 K)	0.40 (1,085 K)	0.07 (792 K)
Core covered	0.14	0.91	0.11

Table 26 Surry MBLOCA and LBLOCA Key Timings (≥10-in. Breaks)

Event	Case 4 (hr)	Case 19 (hr)	Case 24 (hr)	Case 10 (hr)
Reactor trip	0.001	0.001	0.001	0.001
HHSI injection	0.001	0.001	-	-
RCP trip (10% void)	0.008	0.008	0.006	0.001
First actuation of containment sprays	0.008	0.008	0.008	0.005
Core uncovery (water < TAF)	0.01	0.008	0.02	0.022
LHSI injection	0.04	-	0.05	0.005
Maximum cladding temperature timing (maximum temperature)	0.08 (850 K)	0.30 (835 K)	0.04 (640 K)	0.036 (1,043 K)
Core covered	0.12	0.87	0.06	0.053

6.6 Inadvertent Open Relief Valve Success Criteria (Peach Bottom)

The first scenario of interest for Peach Bottom deals with an inadvertent/stuck-open relief valve. For most of these simulations, at time zero the reactor trips, feedwater trips, and a safety relief valve (SRV1) opens. In actuality, the plant would not be expected to trip, but would instead be manually tripped by the operators due to high suppression pool temperature if they were unable to reclose the stuck-open valve. Two sensitivity cases scope the effects of the assumption that the reactor and feedwater trip at time zero. LPCI is available for all cases and the main steam isolation valves (MSIVs) close shortly after reactor trip (see Table 28)[17]. The availability of RCIC, HPCI, and CRD injection is varied to assess their effects.

This analysis models very few operator actions. In reality, the operators would execute their procedures. A number of different procedure paths are possible, depending on available equipment. In general, the following procedures would apply:

- Conditions will prompt the operators to attempt to reclose the open SRV.
- High suppression pool temperature will prompt the operators to start the residual heat removal system in suppression pool cooling mode in accordance with T-102, "Primary Containment Control."
- Low vessel level will prompt the alignment or recovery of frontline injection sources (e.g., RCIC), and, if insufficient, alternative injection sources (e.g., high-pressure service water) in accordance with T-101, "RPV Control," and T-111, "Level Restoration," along with supporting procedures.
- If conditions continue to degrade, the operators will perform an emergency depressurization to allow low-pressure injection.

The calculations summarized in Table 27 and Table 28 demonstrate that any of the injection options considered will prevent heatup before depressurization to LPCI entry. In the case of HPCI, the injection capacity is such that depressurization to LPCI entry does not occur for 9 hours. For cases with only CRD injection, CRD prevents significant heatup even when the second CRD pump is not started until 20 minutes after the initiating event. For cases with no high-pressure injection, the system still depressurizes to LPCI entry conditions before core damage would occur, with a maximum cladding temperature of 939 degrees C (1,722 degrees F).

The above results are subject to the assumption that suppression pool temperature is not significantly elevated by the time of natural depressurization to LPCI conditions, such that low-pressure injection drawing from the suppression pool would be unavailable due to NPSH concerns (i.e., the somewhat stylized nature of assuming the reactor trips at time zero). To investigate this assumption, two sensitivity cases were run in which the reactor continues to operate at power until an automatic trip signal is reached. These sensitivity cases were run for the more limiting of the CRD cases (Case 4). In Case 4a, feedwater is tripped at time zero, and in Case 4b, feedwater continues to run. For Case 4a, the reactor automatically tripped at eight seconds, leading to a PCT that is 110 degrees C higher than in Case 4, but still more than 500 degrees C below the onset of core damage. For Case 4b, because of the continued

[17] Due to the way the MSIVs are modeled in the MELCOR model, MSIVs effectively close when the pressure in the main steamline drops below 994 psia (6.85 MPa), or due to low RPV water level (for Case 4a).

availability of feedwater, there is no core cooling concern and the reactor does not automatically trip until 46 minutes. By the time the reactor trips (on high drywell pressure), the suppression pool temperature has already exceeded multiple technical specification limits (as prescribed in Section 3.6.2.1) that would have prompted operator action. Specifically, at 95 degrees F (35 degrees C), the technical specifications initiate increased monitoring and action to reduce temperature. At 110 degrees F (43 degrees C), the technical specifications require that the reactor be tripped "immediately." Finally, at 120 degrees F (49 degrees C), the technical specifications require the reactor to be depressurized with a completion time for this action of 12 hours (NRC, 2003c). The significance of this case is that despite the assumption that the operators do not take the above actions, the suppression pool temperature does not reach the NPSH limit until greater than 5 hours (the CRD continues to provide sufficient injection after this point).

In addition to the key timing tables below, plots for various results of interest are provided in Appendix B, Section B.1.

Key assumptions and operator actions in these calculations include the following:

- Operator actions to reclose the SRV, start RHR in suppression pool cooling mode, and perform an emergency depressurization are either not initiated or are unsuccessful.
- Reactor trip (mimicking an early manual scram), feedwater trip, and one SRV stuck open occurs at time zero (except for Cases 4a and 4b).
- RCIC is run in inventory control mode.
- Post-scram CRD flow ranges from 110 gpm (0.416 m^3/min) at high pressure (1,020 psia (7.0 MPa)) to 180 gpm (0.681 m^3/min) at low pressure (14.7 psia (0.1 MPa)) for one pump, or 210 gpm (0.795 m^3/min) to 300 gpm (1.14 m^3/min) for two pumps[18].
- RCIC and HPCI isolate on low steamline pressure of 75 psig (0.52 MPa).

[18] No operator action is required to achieve the flow rate cited for one pump. The pre-SCRAM flow rate is 60 gpm (0.227 m^3/min) for one pump. SCRAM results in the automatic opening of inlet valves on the individual CRD hydraulic control units, which increases the flow to 110 gpm (0.416 m^3/min) for one pump. The operators can also open throttle valves to increase the flow further, but this action is not considered in this analysis.

Table 27 Peach Bottom Inadvertent Open SRV Results

Case	RCIC	HPCI	CRD	LPCI	LPCS	ac/dc	FW, SPC, ADS	Core Uncovery (hr)	Core Damage (hr)
1	Yes	No	No	Yes	No	ac/dc	No	No	No
2	No	Yes						No	No
3		No	1 at t = 0 and 2 at t = 10 min					0.41	No
4			1 at t = 0 and 2 at 20 min after scram					0.37	No
4a[1]								0.29	No
4b[1]							FW	No	No
5			No				No	0.32	No

[1] For this case, the reactor was allowed to scram based on a reactor protection system trip signal, rather than at time t = 0.

Table 28 Peach Bottom Inadvertent Open SRV Key Timings (Cases 1–5)

Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)	Case 4 (hr)	Case 4a (hr)	Case 4b (hr)	Case 5 (hr)
SRV1 open	0	0	0	0	0	0	0
Reactor trip	0	0	0	0	< 0.01[1]	0.76	0
MSIVs close	< 0.01	< 0.01	< 0.01	< 0.01	< 0.01	0.79	0
Downcomer level first reaches L2	0.07	0.07	0.07	0.07	0.03	N/A	0.07
RCIC/HPCI first started (CST injection mode)	0.08	0.08	-	-	-	-	-
2nd CRD pump started	-	-	0.17	0.33	0.33	1.09	-
Downcomer level reaches L1	0.37	8.93	0.32	0.32	0.24	N/A	0.26
Downcomer level below TAF	0.37	8.93	0.35	0.33	0.25	N/A	0.28
Suppression pool temp. >110 degrees F[3]	0.40	0.61	0.42	0.42	0.41	0.30	0.40
LPCI first started	0.51	8.93	0.59	0.58	0.53	N/A	0.57
RCIC/HPCI pump isolation: low steamline pressure < 0.52 MPa (75 psig)	0.82	5.59	-	-	-	-	-
HCTL limit reached[3] (no action taken)	4.5	4.0	> 1[2]	> 1[2]	5.0	0.57	> 1[2]
RHR pump isolation - NPSH	9.6	11.1	> 1[2]	> 1[2]	> 10[2]	5.4	> 1[2]
Maximum cladding temperature timing (max. temperature)	No heatup	No heatup	0.78 (786 K)	0.76 (830 K)	0.67 (941 K)	No heatup	0.75 (1,212 K)

[1] Reactor trips at 8 seconds on low RPV level.
[2] The simulation was stopped prior to reaching this condition.
[3] The HCTL limit is based on suppression pool temperature, suppression pool level, and RPV pressure.

6.7 Boiling-Water Reactor Station Blackout (Peach Bottom)

These calculations investigate variations in the availability of injection sources, the behavior of the SRVs (failure to close), manual operator actions to implement heat capacity temperature limit (HCTL)-based depressurization[19], and the time to battery depletion. For reference, the Peach Bottom coping time listed in NUREG-1776 is 8 hours (NRC, 2003a). Here, very few operator actions are modeled. In reality, the operators would enter special event procedure SE-11, “Station Blackout,” based on plant conditions. This procedure would have the operators attempt to recover ac power from the grid and diesel generators and request configuration of the Conowingo station blackout line. The procedure would also direct the operators to shed loads to extend battery availability, take steps to extend HPCI or RCIC operation, and depressurize once plant conditions permitted. Concurrently, the EOPs would direct the operators to maintain level, stabilize pressure, and cooldown, as achievable.

A sensitivity case was performed to look at the effect of recovery, similar to the Surry station blackout sensitivities described in Section 6.4. Except as noted, most cases assume that dc power is always available, which is an intentional modeling artifact to investigate timing. No EOP manual actions are modeled except for HCTL-based depressurization.

For cases in which both HPCI and RCIC are unavailable, core damage occurs at 0.8 or 1.2 hours, depending on the assumption about a stuck-open SRV. Recovery of injection at the time of core damage was demonstrated to quickly arrest heatup. For cases in which dc is lost after 2 hours, core damage occurs at 4 to 5 hours. For cases in which the SRV sticks open after 187 lifts (see Table 5) or HCTL depressurization is performed, core damage ranges from 7 to 11 hours. (Note that the operators would initiate HCTL depressurization to protect containment even without a low-pressure injection source.) For cases in which the SRV does not stick open and HCTL depressurization is not performed, RCIC or HPCI fails after approximately 14 or 16 hours (depending on which is assumed available) because of CST depletion, and core damage occurs after 19 hours. In these cases, switchover to the suppression pool is not permitted because the NPSH limit has already been exceeded[20]. Considering all cases, the time lag from uncovery of the top of active fuel (TAF) to the time of core damage ranges from 0.5 to 1.8 hours. In addition to the results and key timings in Table 29 to Table 32 below, plots for various results of interest are provided in Appendix B, Section B.2.

Key assumptions and operator actions in these calculations include the following:

- RCIC and HPCI (when available) are run in inventory control mode.
- DC power is always available for control of HPCI and RCIC, except as noted.
- Post-accident alignment of CRD is not credited.

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19 For an SRV sticking open due to cycling, the lowest setpoint SRV (SRV/1) is the relevant SRV. For HCTL depressurization, the highest setpoint SRV (SRV/11) is the relevant SRV. Note that in the MELCOR model these valves have the same effective leak area.

20 For cases such as these where RCIC or HPCI is operated for an extended period of time without room cooling, failure due to pump bearing temperature can become a concern. However, for the analysis performed here, temperatures did not approach the pump bearing rating assumed in the MELCOR model (210 degrees F (99 degrees C)).

Table 29 Peach Bottom Station Blackout Results

Case	RCIC	HPCI	ac/dc	SRV Sticks Open?	HCTL Depress ?	Core Uncovery (hr)	Core Damage (hr)
1	No	No	-	No[1]	No	0.5	1.2
1a			ac recovery at 1.2 hr	No		0.5	1.2[2]
2			-	At t = 0		0.3	0.8
3	Yes		dc is always available	No		17.7	19.4
4					Yes	6.0	7.2
5			2 hr of dc		No	3.3	4.3
6			dc is always available	At 187 lifts		6.0	7.2
7	No	Yes		No		17.5	19.3
8					Yes	9.3	10.8
9			2 hr of dc		No	3.8	4.9
10			dc is always available	At 187 lifts		9.2	10.7

1 For this case, the SRV does not stick open until after core damage, so this assumption does not affect the outcome.

2 Recovery of injection upon reaching 2,200 degrees F (1,204 degrees C) quickly arrests further heatup.

Table 30 Peach Bottom Station Blackout Key Timings (Cases 1, 1a, and 2)

Event	Case 1 (hr)	Case 1a (hr)	Case 2 (hr)
Reactor trip, MSIV closure	0	0	0
Downcomer level reaches L2	0.16	0.16	0.16
Downcomer level reaches L1	0.50	0.50	0.27
Downcomer level below TAF	0.50	0.50	0.27
Gap release: 900 °C (1,652 °F)	1.02	1.02	0.69
Core damage: max. temp. >1,204 °C (2,200 °F)	1.17	1.17	0.79
HPCI, RCIC, CRD injection start	-	1.17	-
ADS actuated	-	1.24	-
Downcomer level recovers above TAF	-	1.27	-
SRV sticks open due to high # of cycles	1.75	-	-

Table 31 Peach Bottom Station Blackout Key Timings (Cases 3–6)

Event	Case 3 (hr)	Case 4 (hr)	Case 5 (hr)	Case 6 (hr)
Reactor trip, MSIV closure	0	0	0	0
Downcomer level first reaches L2	0.16	0.16	0.16	0.16
RCIC started (CST injection mode)	0.17	0.17	0.17	0.17
RCIC fails due to loss of dc	-	-	2.00	-
HCTL limit reached	2.46 (no action taken)	2.46	2.46 (no action taken)	2.46 (no action taken)
SRV sticks open due to high # of cycles	-	-	-	2.47
RCIC NPSH limit exceeded[1]	12.67	-	-	-
RCIC pump isolation: low steam line pressure < 0.52 MPa (75 psig)	-	3.90	-	3.92
RCIC injection ends due to CST level < 5 ft (1.5 m)	14.43	-	-	-
Downcomer level reaches L1	17.68	5.61	3.25	5.62
Downcomer level below TAF	17.68	5.61	3.25	5.62
Gap release: 900 °C (1,652 °F)	19.06	6.99	4.04	7.00
Core damage max. temp. >1,204 °C (2,200 °F)	19.42	7.17	4.25	7.18
Exhaust pressure exceeded: 0.35 MPa (50 psig)	20.14	-	-	-

[1] Switchover to the suppression pool is not permitted after this point.

Table 32 Peach Bottom Station Blackout Key Timings (Cases 7–10)

Event	Case 7 (hr)	Case 8 (hr)	Case 9 (hr)	Case 10 (hr)
Reactor trip, MSIV closure	0	0	0	0
Downcomer level first reaches L2	0.16	0.16	0.16	0.16
HPCI started (CST injection mode)	0.17	0.17	0.17	0.17
HPCI fails due to loss of dc	-	-	2.00	-
SRV sticks open due to high # of cycles	-	-	-	2.53
HCTL limit reached	2.67 (no action taken)	2.67	2.67 (no action taken)	2.67 (no action taken)
HPCI NPSH limit exceeded[1]	12.07	-	-	-
HPCI pump isolation: low steam line pressure < 0.52 MPa (75 psig)	-	5.72	-	5.61
HPCI injection ends due to CST level < 5 ft (1.5 m)	16.05	-	-	-
Downcomer level reaches L1	17.53	8.97	3.82	8.94
Downcomer level below TAF	17.53	9.06	3.82	8.94
Gap release: 900 °C (1,652 °F)	18.96	10.59	4.63	10.46
Core damage max. temp. >1,204 °C (2,200 °F)	19.31	10.8	4.85	10.68
Exhaust pressure exceeded: 1.04 MPa (150 psig)	-	-	-	-

[1] Switchover to the suppression pool is not permitted after this point.

7. APPLICATION OF MELCOR RESULTS TO SURRY AND PEACH BOTTOM SPAR MODELS

Table 33 and Table 34 below map the MELCOR calculations presented in Section 6 with the most closely corresponding SPAR model[21] sequences and provide the relative risk contribution of these sequences. Note that at the initiator heading level (e.g., LOMFW), the right-most column gives the relative contribution of all SPAR sequences from that initiator class (e.g., 9.97 percent), while the subsequent rows give the relative contributions from the subset of sequences studied in this report (e.g., LOMFW-16 = 9.32 percent). Regarding loss of offsite power / station blackout, the initiator class relative contribution is for all loss of offsite power events (e.g., switchyard centered), whereas the analyses in this report focus on station blackout events. Finally, for the station blackout sequences, the nomenclature of having multiple sequence numbers reflects transfers amongst two or more event trees. For instance, "LOOP-17-45" indicates the sequence with end-state #17 from the LOOP event tree, which transfers to the SBO event tree and results in end-state #45 from that event tree. All relevant event trees are provided in Appendix C.

It is also of interest to look at the quantitative timings to core uncovery and ac power recovery used in the Surry SPAR model relative to those from the present analysis (as provided in Section 6.4). Table 35 provides this comparison. A key difference between the SPAR model and the present analyses arises for sequences with AFW available and a stuck-open relief valve. SPAR assumes that the relief valve sticks open early in the event, whereas in the present analyses, the relief valves are not challenged (when AFW is available) until much later (e.g., 8 hours). This difference results in a very large delta in the time to core damage. A second key difference is the SPAR assumption that offsite power must be recovered before battery depletion (i.e., no opportunity for preventing core damage following battery depletion), as compared to the present analysis in which the calculation is continued beyond battery depletion until the core damage surrogate is reached.

21 When the majority of the analysis documented in this report was performed, the active versions of the SPAR models were v3.52 (Surry) and v3.50 (Peach Bottom). These are the models that are discussed in this section. In the intervening evaluation and documentation phase of the project, these models were updated to the 8.x models presently used with SAPHIRE8, which included some data and unrelated modeling changes. As such, the relative risk contribution and sequence numbering for particular sequences would be different for these newer models, but the overall concepts and proposed modifications discussed in this section are unchanged. The actual model changes made as a result of this project were implemented in December 2010 through February 2011.

Table 33 Mapping of MELCOR Analyses to the Surry 1 & 2 SPAR (v3.52) Model

SPAR Sequence (see App. C)	**MELCOR Calculations**	**Percentage as Part of Initiator Class CDF (Internal Events)**	**Percentage as Part of Total Internal Event CDF**
SBLOCA—Section 6.1 of this report			**2.05%**
SLOCA-1	Cases 2b, 6b	N/A—Success Path	N/A—Success Path
SLOCA-9	Cases 1, 2, 2a, 3, 4, 5, 6, 6a, 7, 8	1.05%	0.02%
LOMFW Feed and Bleed—Section 6.2 of this report			**9.97%**
LOMFW-16[1]	All Cases	93.39%	9.32%
SGTR—Section 6.3 of this report			**13.83%**
SGTR-12	All Cases	37.26%	5.15%
LOOP / Station Blackout—Section 6.4 of this report			**43.69%**
LOOP-17-42	Cases 6, 10	0.11%	0.05%
LOOP-17-15-7	Case 4	<0.01%	<0.01%
LOOP-17-15-10	Case 9	0.06%	0.03%
LOOP-17-21	Case 8	0.05%	0.02%
LOOP-17-39	Case 2	<0.01%	<0.01%
LOOP-17-45	Cases 1, 3, 5, 7	6.51%	2.85%
MBLOCA—Section 6.5 of this report			**1.70%**
MLOCA-6	Cases 1, 2, 7, 8, 9, 11, 20, 21, 22	69.21%	1.18%
MLOCA-9	Cases 16, 17, 25, 27, 28, 29	<0.01%	<0.01%
MLOCA-14	Cases 14, 15	<0.01%	<0.01%
MLOCA-16	Cases 5, 6, 12, 13, 26	17.41%	0.30%
LBLOCA—Section 6.5 of this report			**0.06%**
LLOCA-8	Cases 2, 3, 4, 5, 6, 7, 8, 10, 16, 17, 18, 19, 23, 24, 26	3.50%	<0.01%

[1] The feed-and-bleed fault tree is used for many event trees. The relative contribution of the LOMFW sequence studied to the overall core damage frequency (CDF) is on the same order of magnitude or higher than the frequency associated with other sequences that include a failure of feed and bleed. The only other sequence with a higher CDF is a loss of ac bus 1J (22 percent higher). In addition, there is a non-station-blackout LOOP sequence that includes failure of feed and bleed, and the summation of the four types of LOOP (e.g., switchyard centered) results in a CDF equivalent to the LOMFW sequence. Note that the latter sequence uses a modified fault tree (FAB-L) specific to the LOOP event tree. All other sequences that include failure of feed and bleed are a factor of four or more lower.

Table 34 Mapping of MELCOR Analyses to the Peach Bottom 2 SPAR (v3.50) Model

SPAR Sequence (See App. C)	**MELCOR Calculations**	**Percentage as Part of Initiator Class CDF (Internal Events)**	**Percentage as Part of Total Internal Event CDF**
Inadvertently Open Relief Valve—Section 6.6 of this report			**2.86%**
IORV-14	Cases 1, 2	N/A—Success Path	N/A—Success Path
IORV-44	Cases 3, 4, 4a, 4b, 5	4.47%	0.13%
LOOP / Station Blackout—Section 6.7 of this report			**5.75%**
LOOP-31-9	Cases 3, 4	<0.01%	<0.01%
LOOP-31-30	Case 5	16.86%	0.97%
LOOP-31-45	Case 8	<0.01%	<0.01%
LOOP-31-51	Cases 7, 9	0.51%	0.03%
LOOP-31-57	Cases 1, 1a	2.14%	0.12%
LOOP-31-59-6	Cases 6, 10	0.01%	<0.01%
LOOP-31-59-7	Case 2	0.04%	<0.01%

Table 35 Comparison of Surry Station Blackout Results to the SPAR Model

	SPAR (v3.52) Model			**This Report**	
Conditions	**Sequence #**	**SPAR Basis for Time to Core Uncovery (hr)**	**Required Time for Power Recovery (hr)**	**Time to Core Uncovery (hr)**	**Time to Core Damage (hr)**
AFW available w/ stuck-open SRV w/ 21 gpm/RCP leak	LOOP-17-42	0.5	1	8–13	9–14
AFW available w/o stuck-open SRV w/ 21 gpm/RCP leak	LOOP-17-15-7/10	15	4[1]	8–13	11–16
AFW available w/o stuck-open SRV w/ 182 gpm/RCP leak	LOOP-17-21	3	3	4	5
AFW available w/o stuck-open SRV w/ 500 gpm/RCP leak	LOOP-17-39	2	2	1.6	2.3
AFW unavailable	LOOP-17-45	0.5	1	1.4–2.3	2.1–3.4

1 SPAR assumes a maximum time to recover power from station blackout of 4 hours, which is related to assumed battery depletion (and an assumed inability to control AFW or restore offsite power following loss of dc).

Table 36 and Table 37 below (1) summarize the scenarios that have been investigated, (2) recap the boundary and initial condition variations studied using MELCOR, (3) highlight the relevant parts of the existing Surry and Peach Bottom SPAR success criteria, and (4) discuss potential changes to these models based on the MELCOR analysis (including identifying whether these changes have or have not been made). In addition, the table identifies cases in which these results were applied to SPAR models for other, similar plants.

Table 36 Potential Success Criteria Updates Based on Surry Results

Initiator/Aspect of Interest	MELCOR Variations	Affected Portion of Existing SPAR Model	Proposed/Actual Changes
SBLOCA (Section 6.1)	• Break size: 0.5, 1, 2 in. (1.3, 2.5, 5.1 cm) • # of containment spray pumps operating: 0, 2 • PORV treatment: sticks open at 247 lifts, does not stick open	SBLOCA sequence timing and mitigation success criteria	For sequences without modeling of controlled cooldown via operator action, it has not been demonstrated that all break sizes will depressurize to RHR conditions before RWST depletion, or even core damage. Thus, HHSI recirculation is still required. Sensitivity studies have been performed for investigating the effects of controlled cooldown, but these calculations are not sufficient to justify changes to the SPAR models. These calculations demonstrate that the time between RWST depletion and core damage can be substantial, but are subject to the assumption that accumulators are available (see App. D).
Feed and Bleed (Section 6.2)	• Power level • Reactor trip signal	Success criteria for feed and bleed: 2 PORVs and 1 HHSI train	The analysis supports reduction of the number of required PORVs for Surry and similar plants[1] from 2 to 1. This change (which has been made) aligns the SPAR success criteria with the significance determination process notebooks and the licensee PRAs for all of these plants.
SGTR (Section 6.3)	• # of tubes ruptured: 1, 5 • # of HHSI pumps secured: 1, 2 • Faulted SG PORV treatment: sticks open at 119 lifts, does not stick open	SGTR event tree timing	The analysis performed demonstrates that (1) a single HHSI pump is sufficient for adequate injection and (2) significant time (>24 hours) exists before core damage will occur (for the conditions studied), even with very little operator action and even though the RWST is depleted much earlier. The former item confirms the current treatment of HHSI in the success criteria. The latter item suggests that some specific sequences for which the failure to refill the RWST is an important factor may warrant revisiting in the context of a specific application, particularly in light of the fact that some of these sequences include human error probabilities that are driven by time-sensitive performance shaping factors.

Initiator/Aspect of Interest	MELCOR Variations	Affected Portion of Existing SPAR Model	Proposed/Actual Changes
Station Blackout (Section 6.4)	• RCP seal leakage rate: 21, 182, 500 gpm/pump (0.076, 0.689, 1.89 m^3/min) • SRV stuck-open: 1st lift, never • TD-AFW: available, unavailable, blind-feeding success • dc power: unavailable, depletes at 4 hr, always available	Time to recover ac power (and re-establish AFW cooling and RCS makeup capability)	Table 35 provides a comparison of the timings between SPAR and the MELCOR analyses. In many cases, the MELCOR results confirm the current modeling assumption. In other cases, the timings suggest a potential to reduce conservatism in the context of a specific application. Sensitivity cases for this scenario suggest that recovery of ac power at 30 minutes or more prior to core damage provides adequate time (from a thermal-hydraulic standpoint, as opposed to a system alignment standpoint) to establish high-pressure injection and stop fuel heatup. The timing window needed for low-pressure sequences would be expected to be shorter, owing to the higher capacity injection that would be available.
MBLOCA (Section 6.5)[2]	• Break size: 2, 4, 6, 8, 10 in., double-ended (5.1, 10.2, 15.2, 20.3, 25.4 cm) • # of HHSI pumps: 0, 1 • # of LHSI pumps: 0, 1 • # of accumulators: 0, 1, 2 • AFW availability	Success criteria for the injection phase for the MBLOCA event tree: 1 HHSI train and (1 accumulator in each intact loop *or* 1 AFW train)	The MELCOR analysis suggests possible refinement of the success criteria. Nonetheless, the decision was made to retain the existing SPAR success criteria.
LBLOCA (Section 6.5)[3]		Success criteria for inventory control during injection phase for the LBLOCA event tree: 1 accumulator in each intact loop and 1 low-pressure injection train	Based on the MELCOR analyses, the success criteria for the early stages of an LBLOCA has been changed for Surry and similar plants[1] from two accumulators to one accumulator (in an intact loop) or one HHSI train.

1 In this case, similar plants are those with high-volume/high-head SI (chemical and volume control system) pumps (150 gpm (0.568 m^3/min) at 2,500 psi (17.2 MPa)), large-volume SGs (series 51 and F) and core thermal power ≤ 2,900 MWt; plants in this category are Beaver Valley 1 & 2, J.M. Farley 1 & 2, North Anna 1 & 2, Harris, Summer, and Surry 1 & 2.

2 Historically. MBLOCAs have been 2-in. (5.1-cm) to 6-in. (15.2-cm) equivalent diameter (NRC, 1990) and (NRC, 1999) Appendix J.

3 Historically, LBLOCAs have been greater than 6-in. (15.2-cm) equivalent diameter (NRC, 1990) and (NRC, 1999) Appendix J.

Table 37 Potential Success Criteria Updates Based on Peach Bottom Results

Class	MELCOR Variations	Affected Portion of Existing SPAR Model	Proposed Changes
IORV (Section 6.6)	• Injection source: RCIC, HPCI, CRD, none • Timing of 2nd CRD pump initiation: 10 min, 20 min • Reactor/ feedwater trip: time zero vs. automatic	Effectiveness of injection source for core cooling until low-pressure pumps can provide makeup	For RCIC and HPCI, the calculation confirms the treatment in the SPAR models. For CRD, the SPAR models for BWR Mark I and Mark II plants have been changed to credit: (1) two CRD pump flow for adequate core cooling following initial HPCI/RCIC success and (2) one CRD pump flow for adequate core cooling late in the event (if not already credited).
Station Blackout (Section 6.7)	• Injection: HPCI, RCIC, none • Operator actions: HCTL depress., none • SRV behavior: stuck open at t = 0, stuck open at 187 lifts, never sticks • Recovery time: 1.2 hours, never • dc power: unavailable, depletes at 2 hours, always available	Time to recover ac power (and reestablish core cooling)	In general, the SPAR modeling conventions related to battery depletion limit the incorporation of insights from the MELCOR analyses. Some specific observations are provided below for consideration in the context of a specific application: • For complete loss of ac and dc, calculations suggest that credit for recovery of offsite power can be extended to 1 hour (currently, the SPAR models give credit for 30 minutes). • For complete loss of ac and dc commensurate with a stuck-open SRV, calculations suggest that credit for recovery of offsite power can be extended to 30 minutes (currently, none of the SPAR models give credit). • For cases in which dc power is always available and RCIC/HPCI is lost because of NPSH, current SPAR models agree with these results. • For cases with 2 hours of dc, calculations suggest that 2 hours can be credited for boiloff (currently, none of the SPAR models gives any credit for boiloff). • For the maximum time for injection without suppression pool cooling (HCTL depressurization cases), the SPAR models agree with these calculations, with the exception of Grand Gulf and Nine Mile Point 2 for RCIC.

8. CONCLUSION

This project defined a realistically conservative core damage definition surrogate based on accident simulations. The project performed MELCOR analyses for two plants (Surry and Peach Bottom), looking at a range of initiating events and sequences. These results have been mapped to specific, realized changes for relevant SPAR models. The NRC is continuing to work in this area and continues to seek opportunities to engage internal and external stakeholders.

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APPENDIX A

SURRY MELCOR ANALYSES

A.1 <u>Summary of Surry Model Changes</u>

A.1.1 Surry Model

The Surry input deck is generally consistent with the model used in the State-of-the-Art Reactor Consequence Analyses (SOARCA) project. For the present application, however, a number of modifications are made to control the reactor trip and the engineered safety features, including the emergency core cooling system (ECCS) and containment sprays. In addition, the treatment of reactor coolant pump (RCP) seal leakage is also different. The general setpoints and operation of the systems are described below.

Reactor Trip

Table 1 indicates the conditions for reactor trip (i.e., if any condition becomes true, then the reactor is tripped).

Table 1 Conditions for Reactor Trip

	Condition	**Comments**
1	Loss of power	
2	ECCS actuation	See ECCS signals
3	MFW trip[1]	Scram or loss of power or manual
4	TCV closure	
5	RCP trip	Loss of power or loop void >10%
6	HHSI activation	ECCS signal + power available
7	LHSI activation	ECCS signal + power available
8	High RCS pressure	>2,400 psia (16.55 MPa)
9	Low RCS pressure	<1,815 psia (12.51 MPa)
10	High PRZ level	>44.97 ft (26.26 m in MELCOR model)
11	Low PRZ level	<12.51 ft (16.36 m in MELCOR model)
12	High loop Dt	>75 °F (41.67 °C)
13	Manual	Time based

[1] Several different configurations were found just among the three-loop high-head Westinghouse plants in terms of whether a main feedwater (MFW) trip would result in a turbine trip and subsequent reactor trip. The trip in the Surry MELCOR model is potentially dated. For these reasons, additional calculations were run in the loss of all feedwater section of this report to address the (perhaps more common) situation in which a reactor trip would not occur until the reactor protection system trip signal(s) related to steam generator water level.

Emergency Core Cooling System

The high-head safety injection (HHSI), as well as the low-head safety injection (LHSI) and containment sprays in the injection mode, draw water from the refueling water storage tank (RWST). Once the RWST is depleted, the LHSI suction is switched over to the containment sump.

The ECCS actuation signals for HHSI and LHSI are as follows (i.e., if any of the conditions are satisfied, then both systems are activated):

- pressurizer (PRZ) pressure (less than 1,775 pounds per square inch gage (psig) (12.2 megapascals (MPa)))

- high steamline differential pressure (greater than 120 pounds per square inch differential (psid) (0.83 MPa))

- high containment pressure (greater than 17.7 pounds per square inch absolute (psia) (0.122 MPa))

- manual operator action

In addition, power must be available.

The condition for high steam flow and either low steamline pressure (less than 525 psig (3.62 MPa)) or low average temperature (Tavg) (543 degrees Fahrenheit (F) (284 degrees Celsius (C))) is not modeled.

High-Head Safety Injection

HHSI flows are delivered to the cold legs of the Surry model (control volumes 240/340/440). All three HHSI pumps at Surry are assumed to start on HHSI activation.[1] Total HHSI flow is portioned equally between the three cold legs. HHSI pump performance is given in Table 2 below as HHSI flow per pump (gallons per minute (gpm)) (Byron Jackson Test T-30705-3, 5-13-69).

Table 2 High-Head Safety Injection Flow per Pump

Feet (Meters)	gpm (m^3/min)	Comment
0 (0)	615 (2.33)	Runout
1,600 (488)	550 (2.08)	
2,500 (762)	500 (1.89)	
3,275 (998)	450 (1.70)	
3,950 (1,204)	400 (1.51)	
4,500 (1,372)	350 (1.32)	
4,950 (1,509)	300 (1.14)	
5,300 (1,615)	250 (0.946)	
5,600 (1,707)	200 (0.757)	
5,800 (1,768)	150 (0.568)	Rated
5,900 (1,798)	100 (0.379)	
5,905 (1,800)	0 (0)	Shutoff

Low-Head Safety Injection

LHSI flows are delivered to the cold legs of the Surry model (control volumes 240/340/440). The two LHSI pumps at Surry are assumed to start on LHSI activation. Total LHSI flow is portioned equally between the three cold legs. LHSI pump performance is given in Table 3 below as LHSI flow per pump (gpm) (Byron Jackson Test T-31192-1, 11-10-69).

[1] The present MELCOR model assumes that all three HHSI pumps inject upon receiving a safety injection (SI) signal (one pump on the H bus and two pumps on the J bus). This (perhaps atypical) capability is based on interactions with the licensee and is corroborated by particular references (e.g., the emergency operating procedures). The model does not account for the potential reduction in overall flow injection created by three pumps injecting through two trains. However, this modeling assumption is actually conservative in the present analysis because the three-pump alignment is only used for the small- break loss-of-coolant accident (LOCA) and steam generator tube rupture scenarios, in which the effect of RWST depletion and lack of system depressurization are more relevant than the core cooling (because adequate core cooling would be provided by fewer pumps).

Table 3 Low-Head Safety Injection Flow per Pump

Feet (Meters)	gpm (m^3/min)	Comment
0 (0)	4,000 (15.1)	Runout
188 (57.3)	4,000 (15.1)	
213 (64.9)	3,500 (13.2)	
240 (73.2)	3,000 (11.4)	
269 (82.0)	2,500 (9.46)	
296 (90.2)	2,000 (7.57)	
321 (97.8)	1,500 (5.68)	
342 (104)	1,000 (3.79)	
356 (109)	500 (1.89)	
365 (111)	0 (0)	Shutoff

The RWST level must also be above 13.5 percent (RWST-to-sump switchover starts at 13.5 percent and takes 2.5 minutes). After LHSI from the RWST is terminated, a model is activated for LHSI from the reactor sump using the same pump curve. Sump water availability and water temperature are checked.

Accumulators

Accumulators are also modeled as mass and enthalpy injected into cold leg component control volumes 240, 340, and 440. The initial water volume per accumulator is 975 cubic feet (ft^3) (27.6 cubic meters (m^3)) with an initial nitrogen cover gas volume of 475 ft^3 (13.5 m^3). The minimum operating pressure is given as 600 psig (4.137 MPa). All three accumulators are assumed to behave identically in that they are all modeled by a single set of control functions.

Containment Sprays

The injection sprays use two pumps that can operate at 2,900 gpm each (a rated flow of 3,200 gpm per pump minus bleed-off flow of 300 gpm per pump) (11 m^3/min). The droplet size released by the spray headers is 1 millimeter (mm). The pumps deliver water from the RWST at 45 degrees F (280.4 Kelvin (K)), the maximum temperature allowed by the technical specifications, until the RWST water reaches 13.5 percent. The following three headers are in the dome:

(1) the first at 95.50 feet (ft) (29.1 meter (m)) elevation with 88 nozzles
(2) the second at 142.40 ft (43.4 m) elevation with 73 nozzles
(3) the third at 143.75 ft (43.82) elevation with 73 nozzles

The delay from spray signal to full operation is less than 15 seconds. The recirculation sprays are modeled by two pumps identical to the injection mode. The cooler duty is 55,534,520 British thermal units per hour (BTU/hr) each (two per pump, four total), which translates to 16.276 megawatts (MW) per cooler (65.1 MW total). Headers are common with those of the injection system. The containment sprays are initiated at a pressure of 25 psia (0.17 MPa) and are secured (while in injection mode) when the pressure is less than 12 psia (0.0827 MPa).

Containment Fan Coolers

The recirculation system has three 75,000-standard cubic feet per minute (scfm) recirculation fans (35.4 cubic meters per second (m^3/s). The total volumetric flow rate is 225,000 scfm (106.2 cubic meters per second (m^3/s)). The system is supplied with 2,000 gpm (135.91 kilograms per second (kg/s)) at 70 degrees F (294 K) component cooling water until containment temperature and pressure are high enough or until pumps become submerged, at which point the systems goes to chilled cooling water. MELCOR monitors the liquid level, vapor temperature, and pressure in the lower dome (control volume 50) and initiates the fans when the correct parameters are met. The fan inlet and discharge is within the “basement” (control volume 5). The MELCOR input model contains a “low capacity” fan and a “high capacity” fan. Both recognize the same parameters, with the difference being that the high-capacity fan is turned on at a higher temperature. The low-capacity fans have a secondary coolant mass flow rate of 300 pounds per second (lb/s) (135.9 kg/s), and the high-capacity fan has a secondary coolant mass flow rate of 831 lb/s (377.1 kg/s).

Reactor Coolant Pumps

The pumps operate at a rated head of 280 ft (85.3 m) of water at 650 degrees F (343 degrees C), 2,235 psig, which is 6.80×10^5 pascals. The pumps are tripped on either loss of power or high void (assumed to be 10 percent in the MELCOR model). Appendix D provides more information with respect to the sensitivity of the calculations to this latter assumption.

Power-Operated Relief Valve and Safety Relief Valve Setpoints

The opening and closing pressures for the group of pressurizer power-operated relief valves (PORVs) and safety relief valves (SRVs) modeled in the MELCOR input are given in Table 4 below.

Table 4 Opening and Closing Pressures for PORV and SRV

	Opening Pressure in MPa (psi)	**Closing Pressure in MPa (psi)**
PORV-1	16.2 (2,350)	15.55 (2,255)
PORV-2	16.3 (2,364)	15.65 (2,270)
SRV-1	17.23 (2,499)	16.54 (2,399)
SRV-2	17.33 (2,514)	16.64 (2,413)
SRV-3	17.43 (2,528)	16.74 (2,428)

A.2 Small-Break Loss-of-Coolant Accident Dependency on Sump Recirculation

Analysis Summary

Table 5 through Table 7 provide results for this portion of the analysis.

Table 5 Surry SBLOCA Sump Recirculation Results

<table>
<tr><th>Case</th><th>Size (inch)[5]</th><th>HHSI Pumps</th><th>PORV Treatment</th><th>Sprays</th><th>Secondary-Side Cooldown</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td rowspan="4">1</td><td rowspan="2">3</td><td rowspan="6">N/A</td><td>0</td><td rowspan="3">No</td><td>9.2[1]</td><td>11.9[1]</td></tr>
<tr><td>2</td><td rowspan="3">2</td><td>7.3[1]</td><td>9.9[1]</td></tr>
<tr><td>2a[2]</td><td>3/1</td><td>7.9[1]</td><td>10.0[1]</td></tr>
<tr><td>2b[3]</td><td>3/1/0</td><td>Yes</td><td>No[4]</td><td>No[4]</td></tr>
<tr><td>3</td><td rowspan="2">2</td><td rowspan="4">3</td><td>0</td><td rowspan="5">No</td><td>No</td><td>No</td></tr>
<tr><td>4</td><td>2</td><td>No</td><td>No</td></tr>
<tr><td>5</td><td rowspan="6">0.5</td><td rowspan="3">Sticks open after 247 lifts</td><td>0</td><td>No</td><td>No</td></tr>
<tr><td>6</td><td rowspan="3">2</td><td>No</td><td>No</td></tr>
<tr><td>6a[2]</td><td>3/1</td><td>8.8[1]</td><td>9.6[1]</td></tr>
<tr><td>6b[3]</td><td>3/1/0</td><td>N/A</td><td>Yes</td><td>No[4]</td><td>No[4]</td></tr>
<tr><td>7</td><td rowspan="2">3</td><td rowspan="2">Does not stick open</td><td>0</td><td rowspan="2">No</td><td>17.8[1]</td><td>25.1[1]</td></tr>
<tr><td>8</td><td>2</td><td>14.4[1]</td><td>21.4[1]</td></tr>
</table>

1 Core damage is an artifact of the assumed unavailability of HHSI recirculation.
2 It is assumed that two HHSI pumps are secured at 15 minutes.
3 It is assumed that two HHSI pumps are secured at 15 minutes, and the third pump is secured at 30 minutes, followed by secondary-side cooldown at 100 degrees F (55.6 degrees C) per hour).
4 These cases reach RHR entry conditions (both temperature and pressure) before heatup.
5 1 in. = 2.54 cm; 2 in. = 5.1 cm; 0.5 in. = 1.3 cm.

Table 6 Surry SBLOCA Sump Recirculation Key Timings (Cases 1–4)

Event	Case 1 (hr)	Case 2 (hr)	Case 2a (hr)	Case 2b (hr)	Case 3 (hr)	Case 4 (hr)
Reactor trip	0.03	0.03	0.03	0.03	0.01	0.01
HHSI injection	0.03	0.03	0.03	0.03	0.01	0.01
LHSI injection	-	-	-	2.02	-	-
First actuation of contain. sprays	-	2.65	3.29	-	-	1.76
RWST depletion (<13.5%)	5.83	4.30	5.80	-	3.12	2.63
Spray recirculation	-	4.30	5.80	-	-	2.63
LHSI recirculation	-	-	-	-	3.38	2.86
Accumulator starts to inject	6.00	4.52	5.83	0.82	0.23	0.23
RCP trip (10% void)	7.38	5.76	6.73	1.41	-	-
Core uncovery	9.23	7.32	7.9	-	-	-
Core damage (max. temp. >2,200 °F)[1]	11.9	9.93	10.0	-	-	-

1 2,200 °F = 1,204 °C.

Table 7 Surry SBLOCA Sump Recirculation Key Timings (Cases 5–8)

Event	Case 5 (hr)	Case 6 (hr)	Case 6a (hr)	Case 6b (hr)	Case 7 (hr)	Case 8 (hr)
Reactor trip	0.01	0.01	0.01	0.01	0.01	0.01
HHSI injection	0.01	0.01	0.01	0.01	0.01	0.01
LHSI injection	-	-	-	3.49	-	-
PORV stuck open	0.83	0.83	4.65	-	-	-
First actuation of contain. sprays	-	2.20	5.30	-	-	3.23
RWST depletion (<13.5%)	4.14	3.43	7.45	-	8.17	5.52
Spray recirculation	-	3.43	7.45	-	-	5.53
LHSI recirculation	4.72	3.97	-	-	26.6	-
Accumulator starts to inject	4.14	3.43	7.14	1.11	8.28	5.65
RCP trip (10% void)	-	4.68	5.00	13.8	11.7	10.3
Core uncovery	-	-	8.77	-	17.8	14.4
Core damage (max. temp. >2,200 °F)[1]	-	-	9.61	-	25.1	21.4

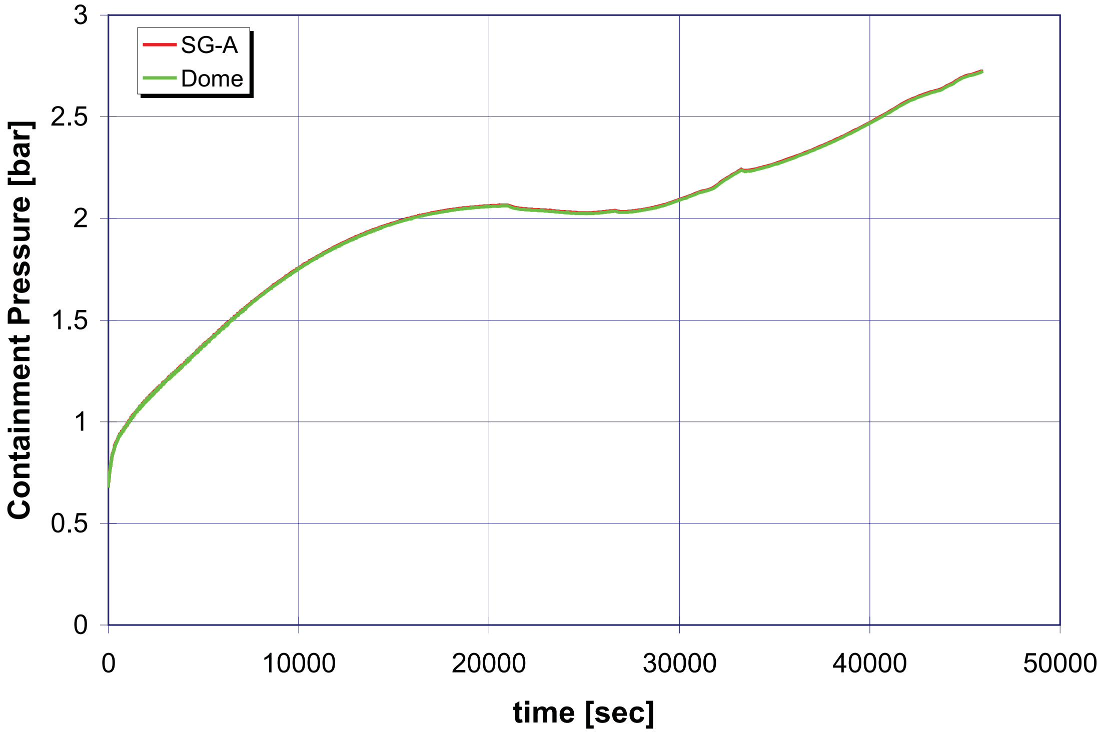
[1] 2,200 °F = 1,204 °C.

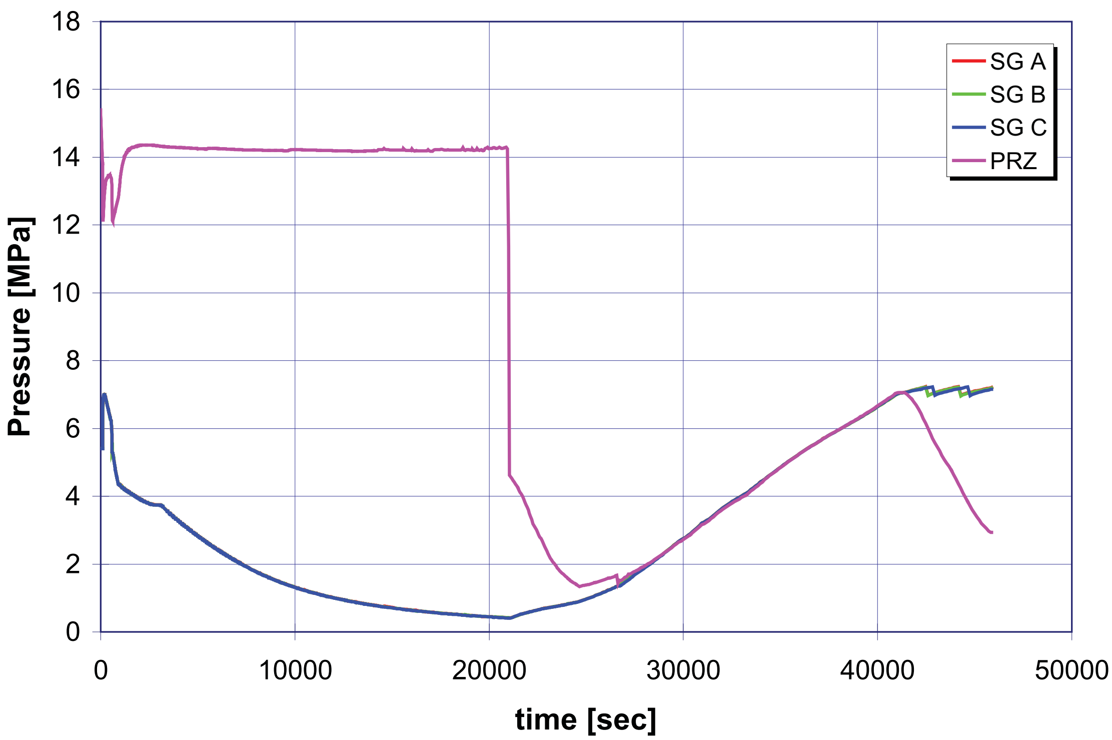
Notes

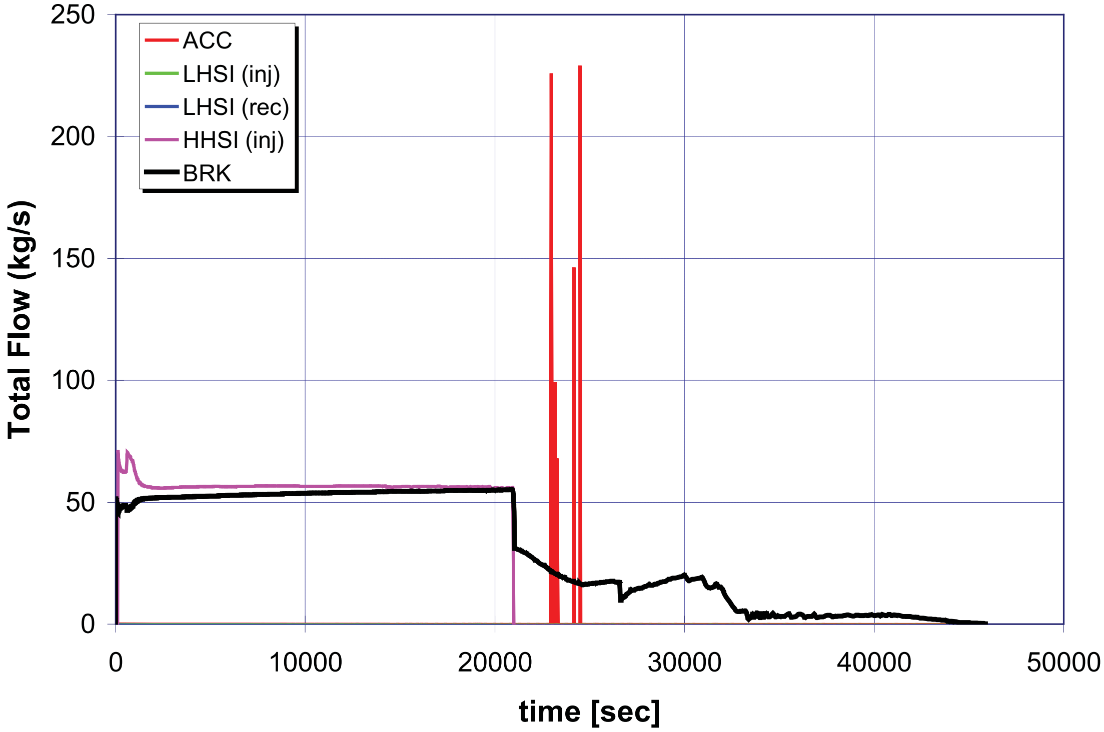
For Cases 5 and 6, PORV1 cycles initially and then gets stuck open because of the number of cycles (247 cycles). The equivalent diameter for the PORV is 1.387 inches, so it depressurizes and goes to LHSI recirculation mode.

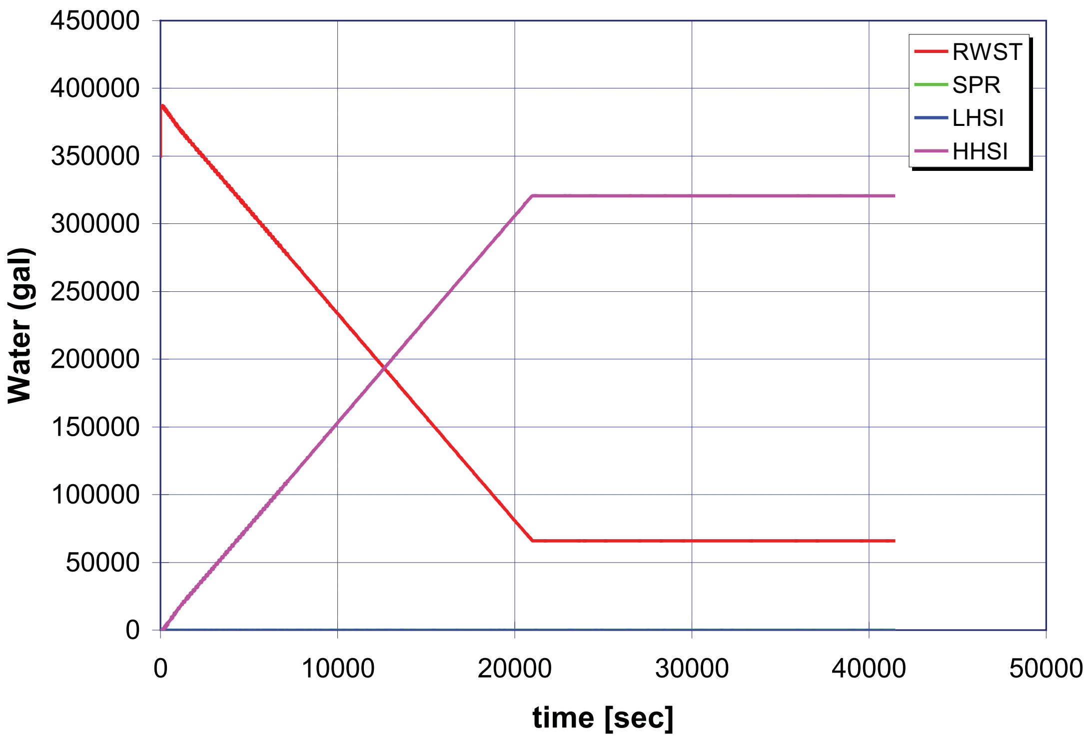
Cases 2a, 2b, 6a, and 6b are sensitivity calculations to demonstrate the impact of HHSI injection and secondary cooldown on reactor coolant system (RCS) pressure and to determine the residual heat removal (RHR) entry conditions. They may not represent actual plant operating procedures.

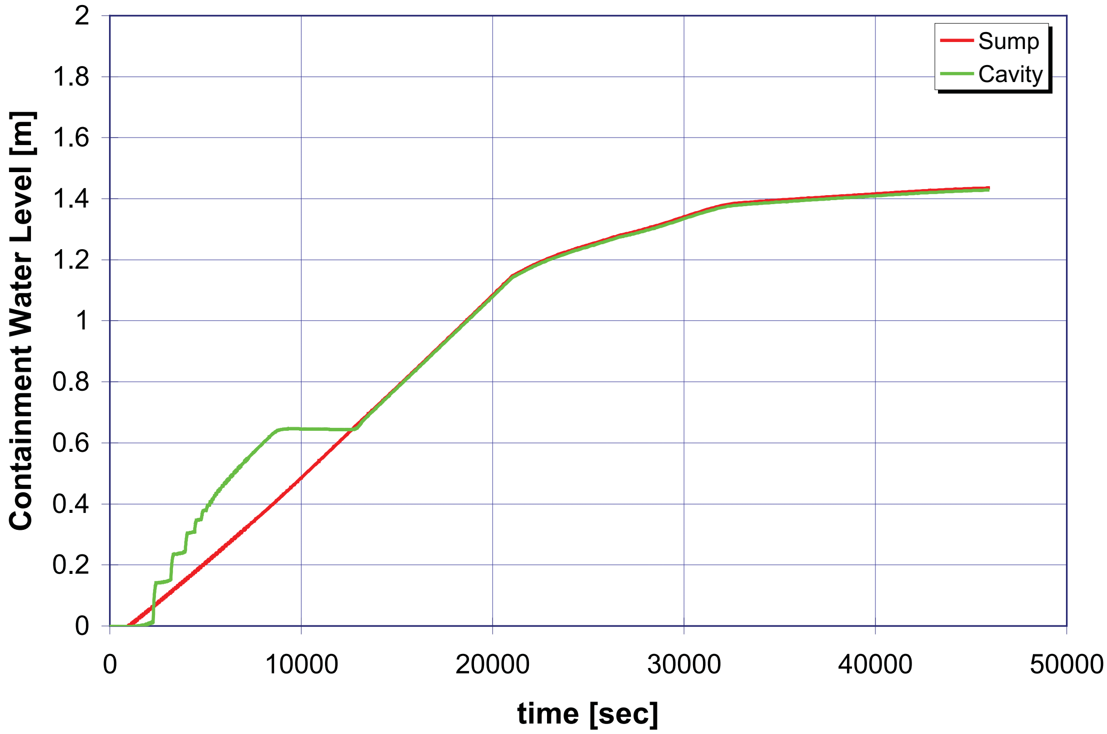
A.2.1 Case 1: 1-Inch Break LOCA without Sprays

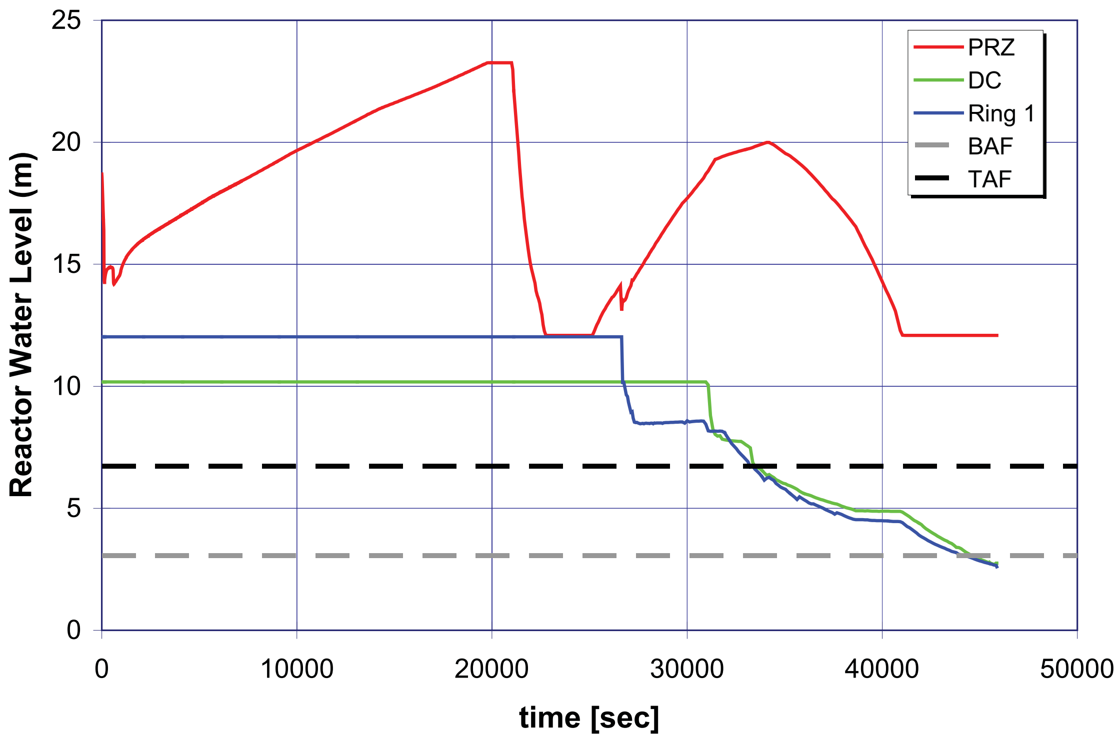


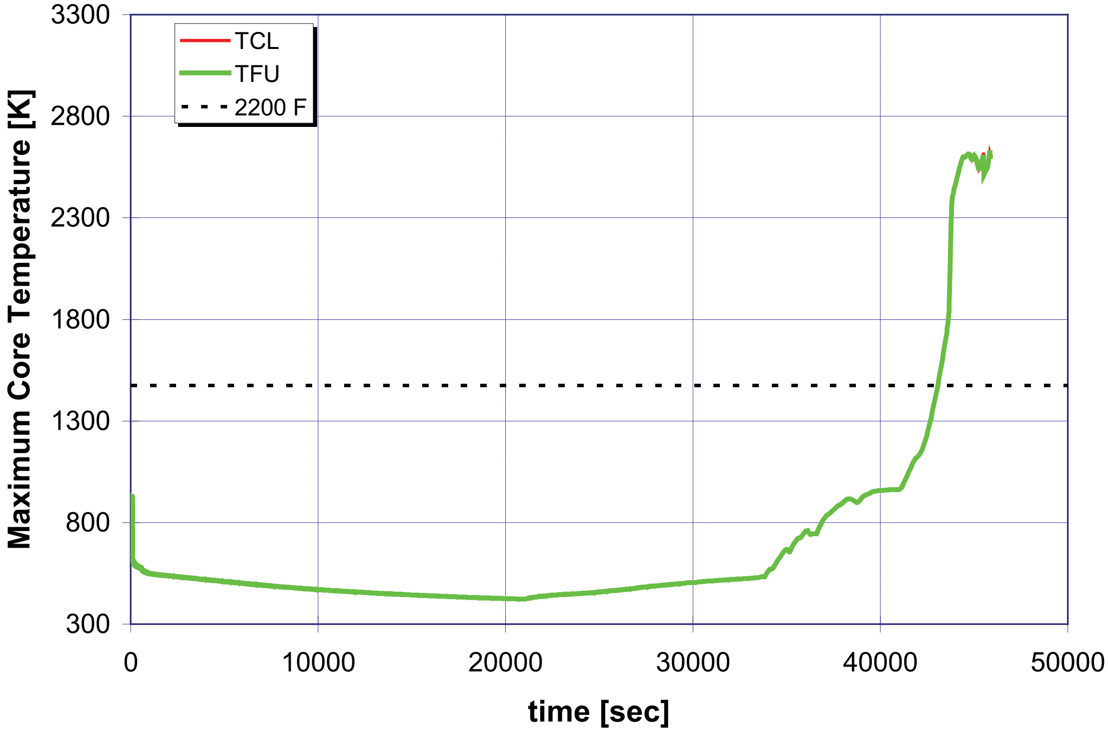



Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
0
50
100
150
200
250
0
10000
20000
30000
40000
50000

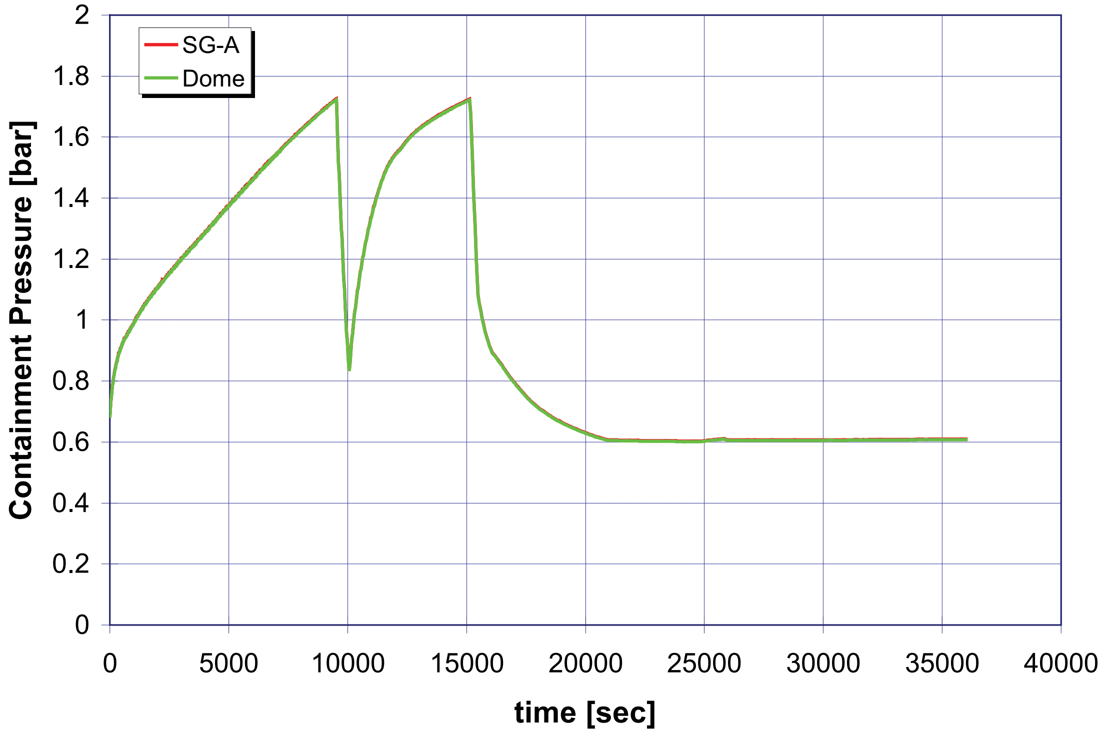

Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
10000
20000
30000
40000
50000

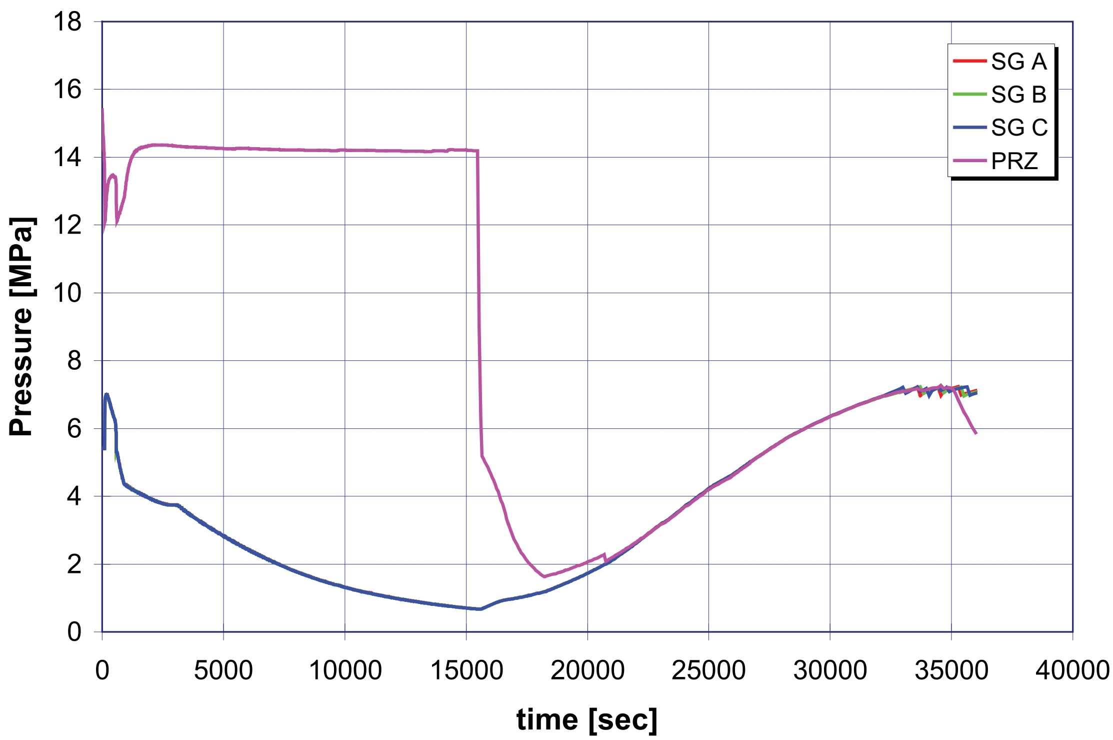

Containment Water Level [m]
time [sec]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
10000
20000
30000
40000
50000

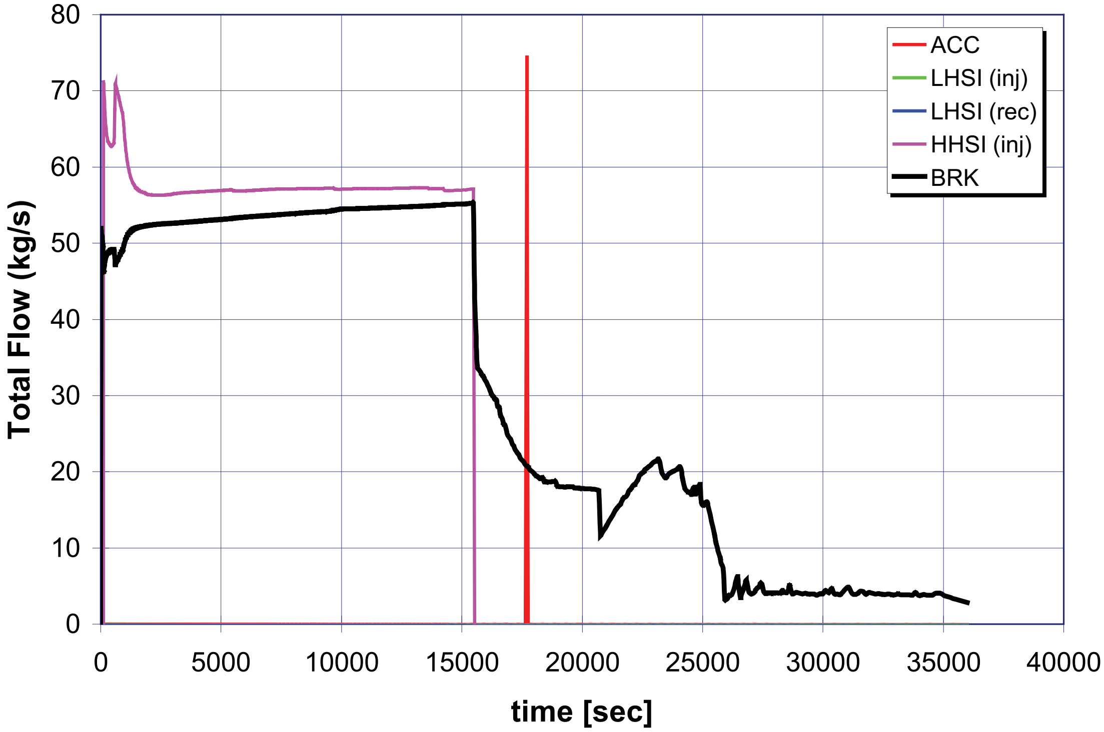

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
5
10
15
20
25
0
10000
20000
30000
40000
50000


TCL
TFU
2200 F
Maximum Core Temperature [K]
3300
2800
2300
1800
1300
800
300
0
10000
20000
30000
40000
50000
time [sec]

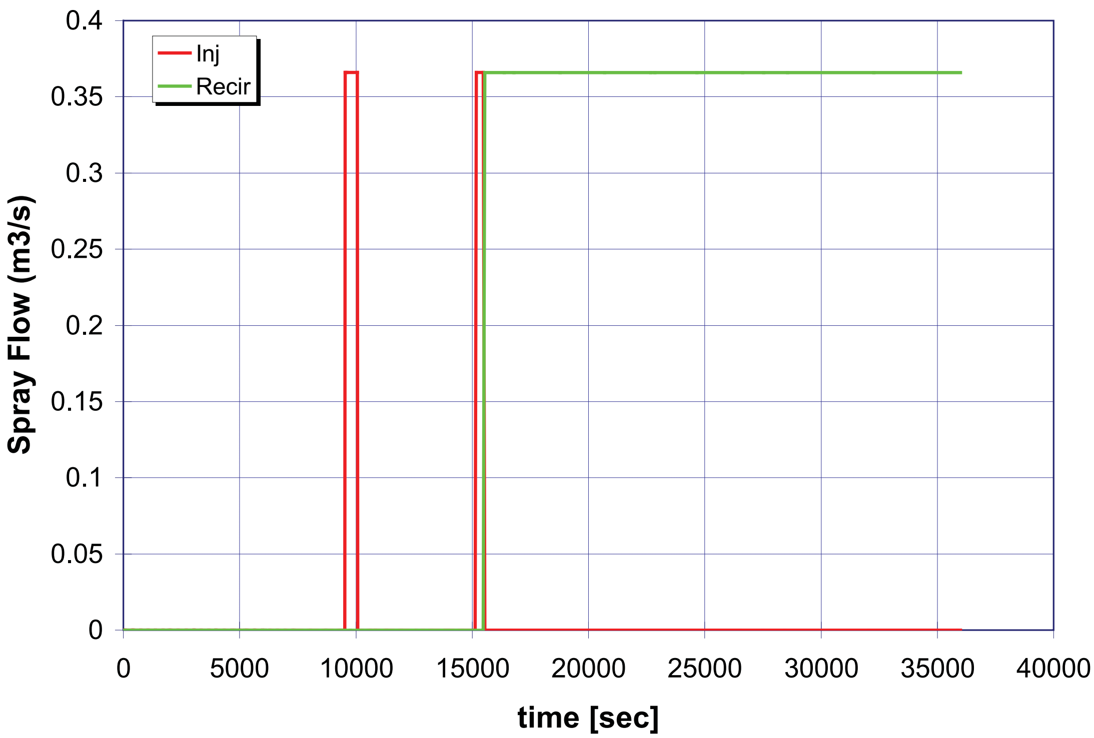
A.2.2 Case 2: 1-Inch Break LOCA with Sprays



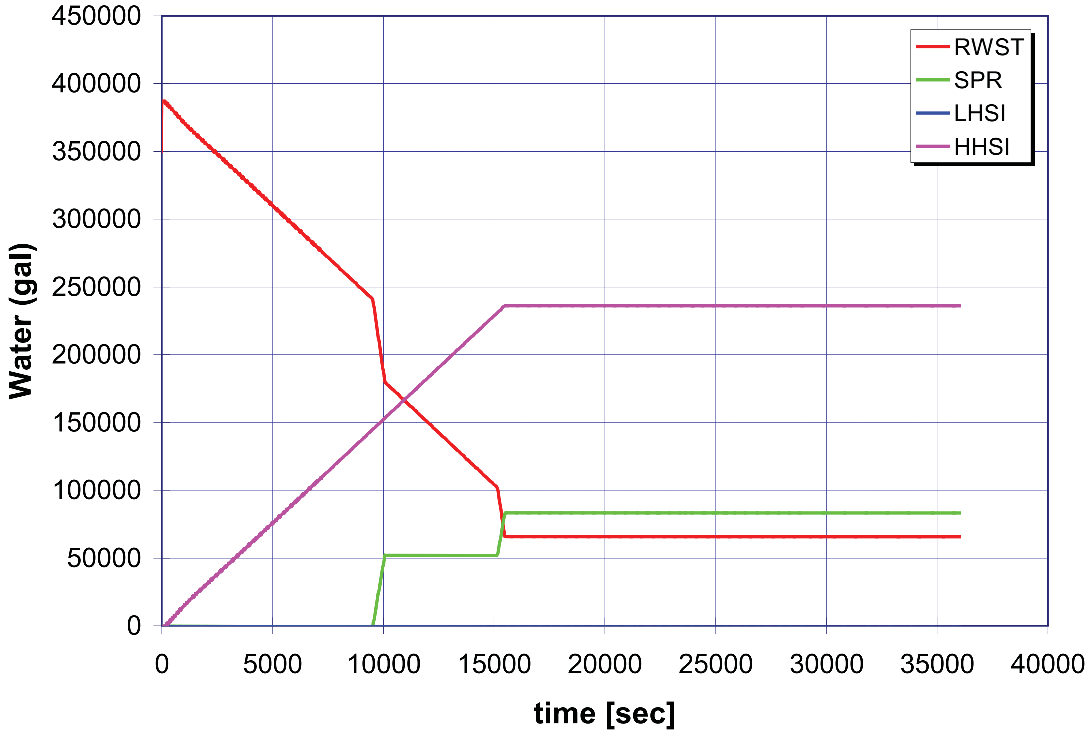




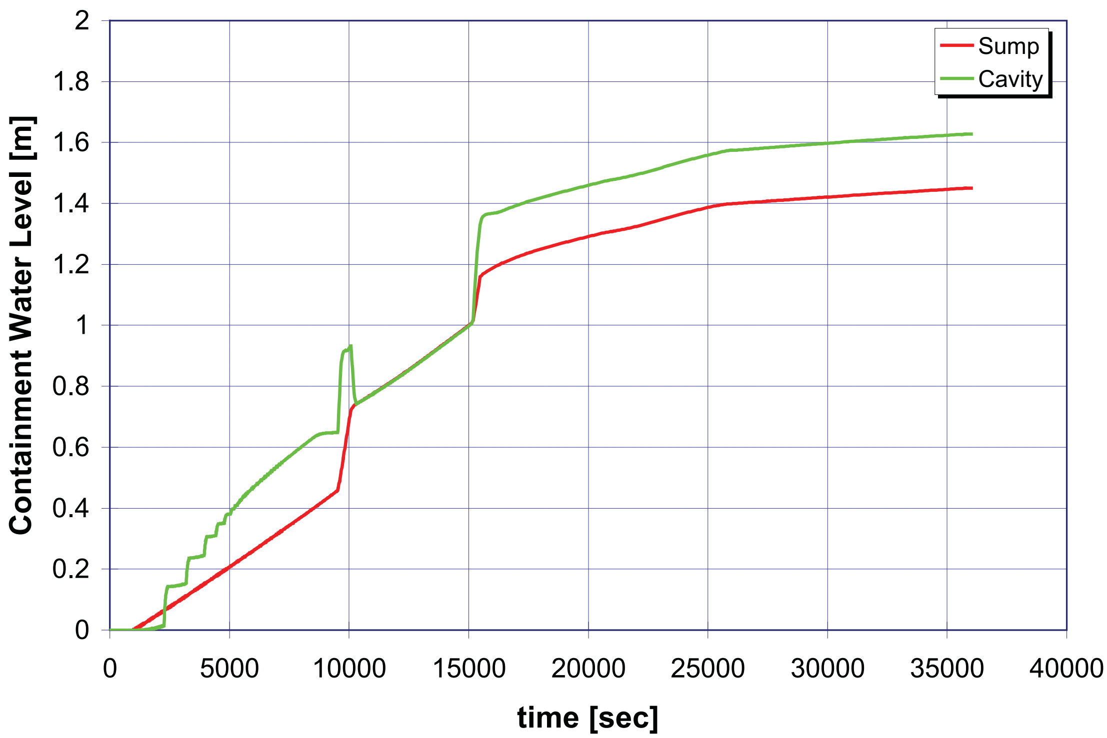
80
70
60
50
40
30
20
10
0
Total Flow (kg/s)
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



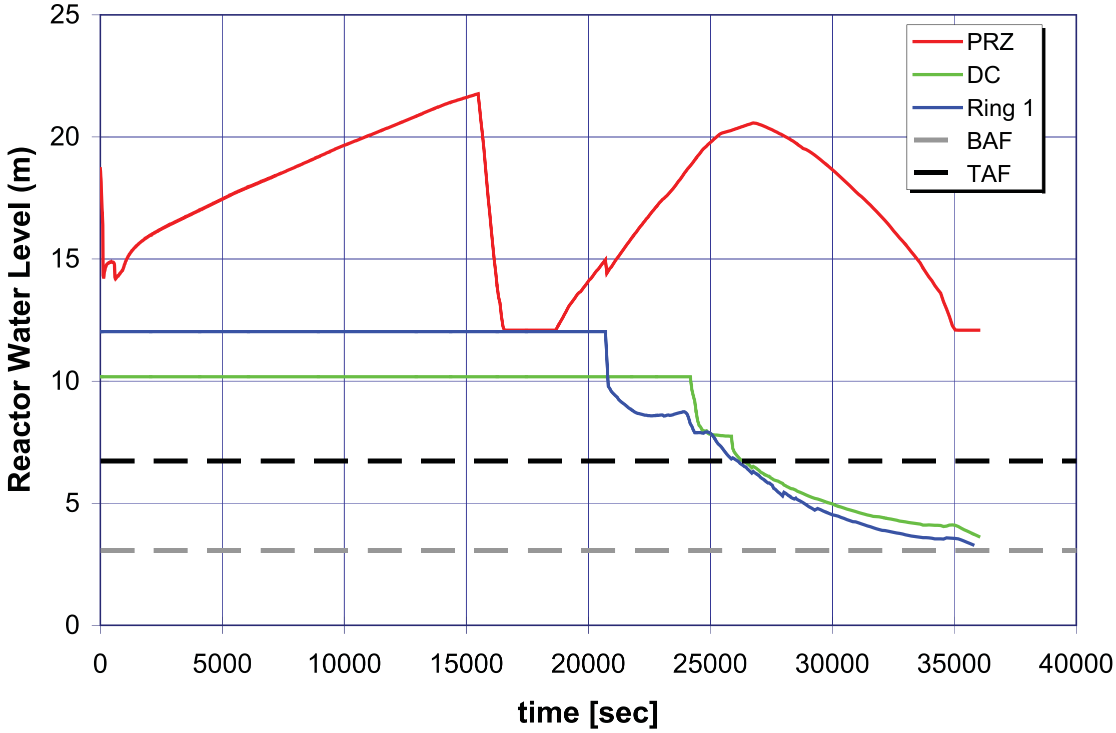
0.4
0.35
0.3
0.25
0.2
0.15
0.1
0.05
0
Spray Flow (m3/s)
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
Inj
Recir

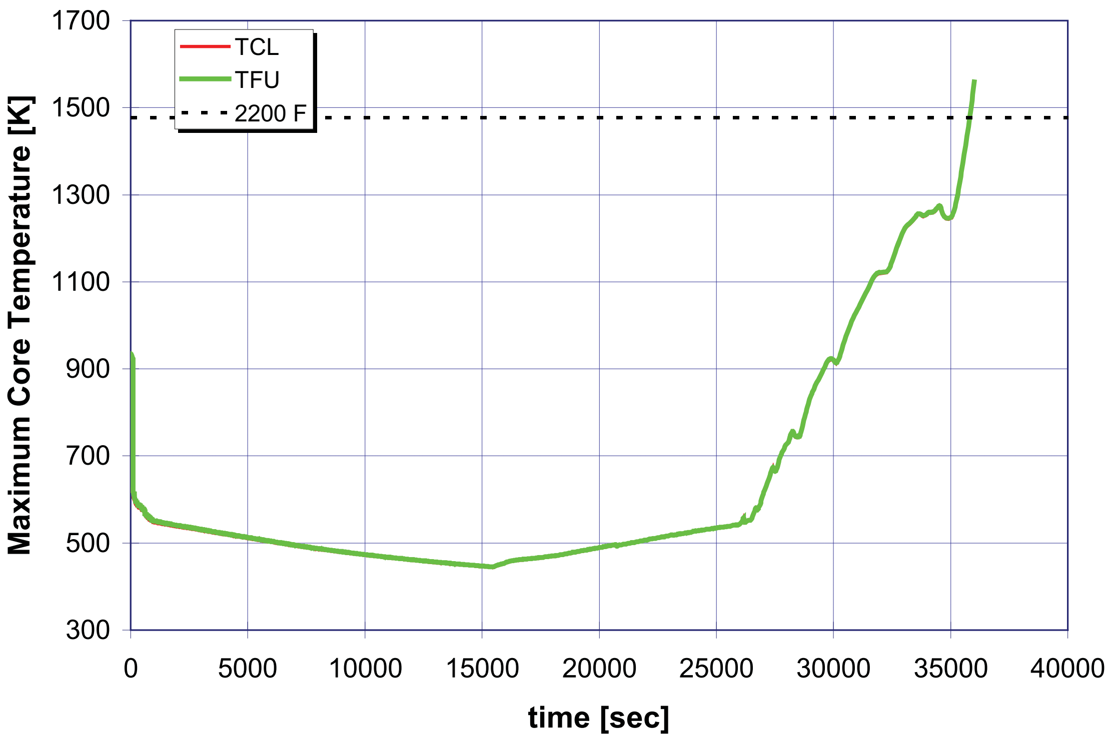


450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
RWST
SPR
LHSI
HHSI

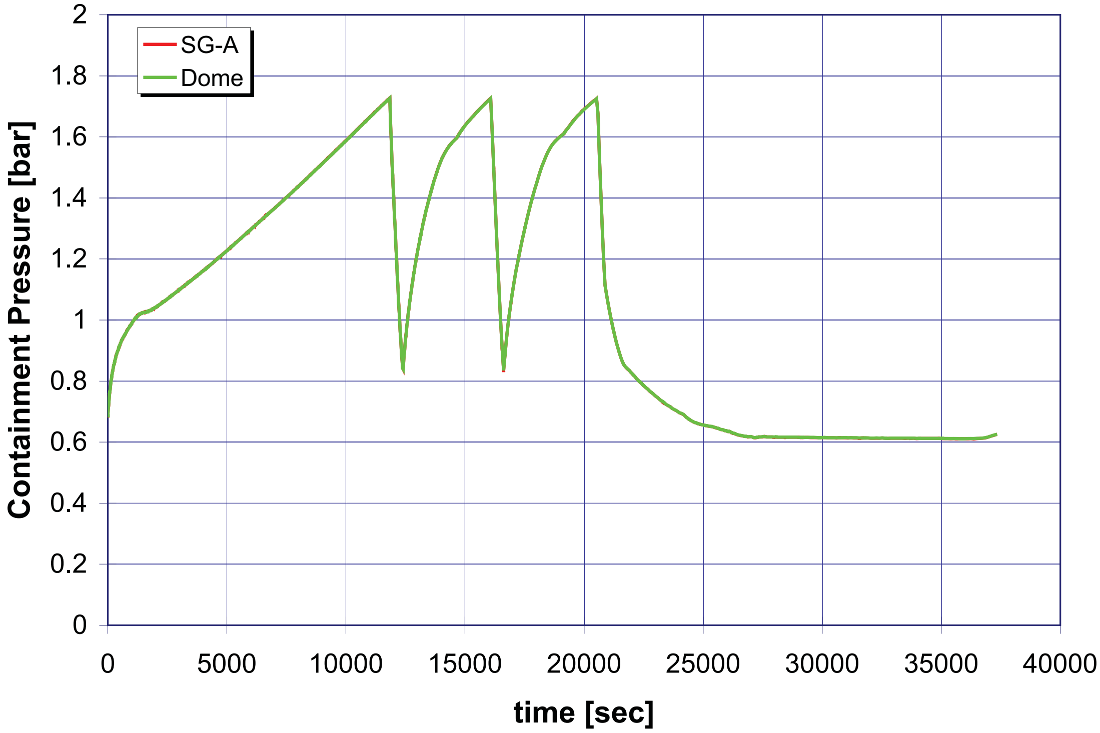


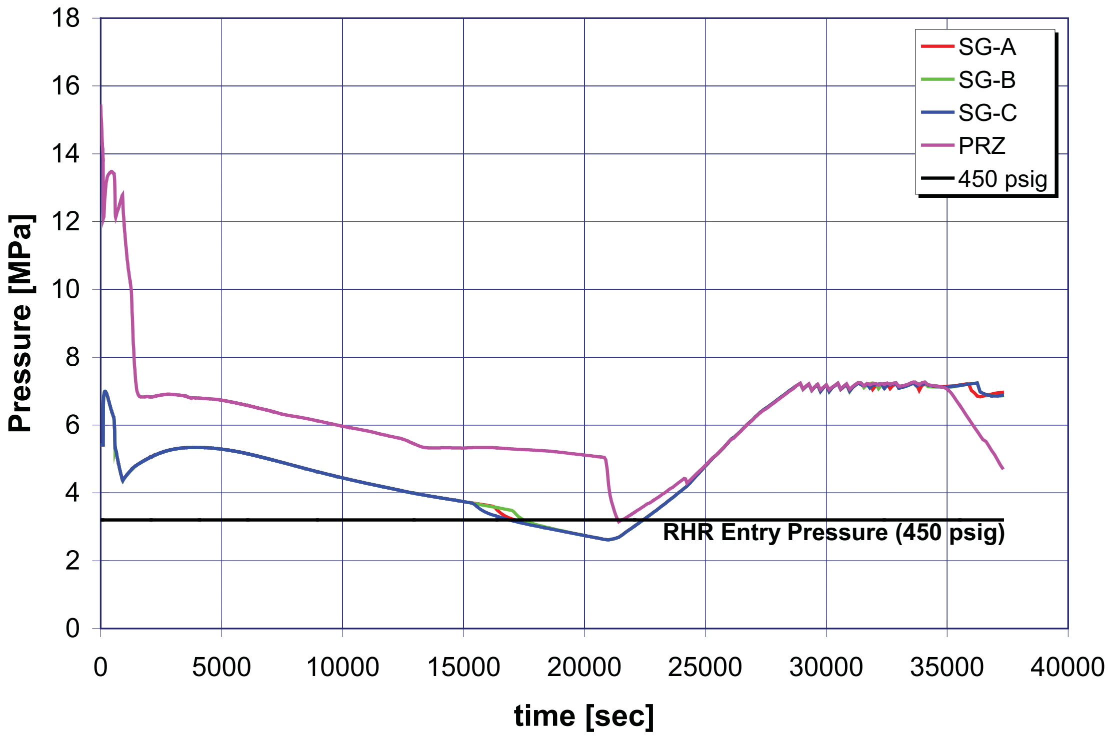
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
Containment Water Level [m]
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
Sump
Cavity

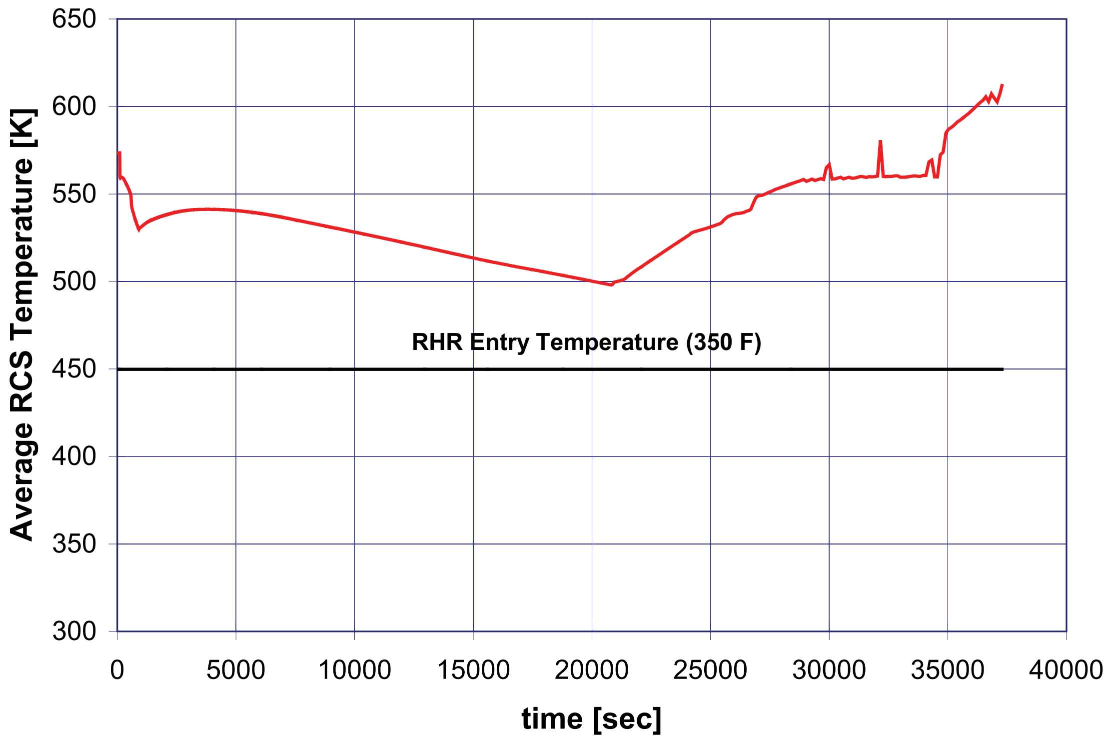

Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
PRZ
DC
Ring 1
BAF
TAF

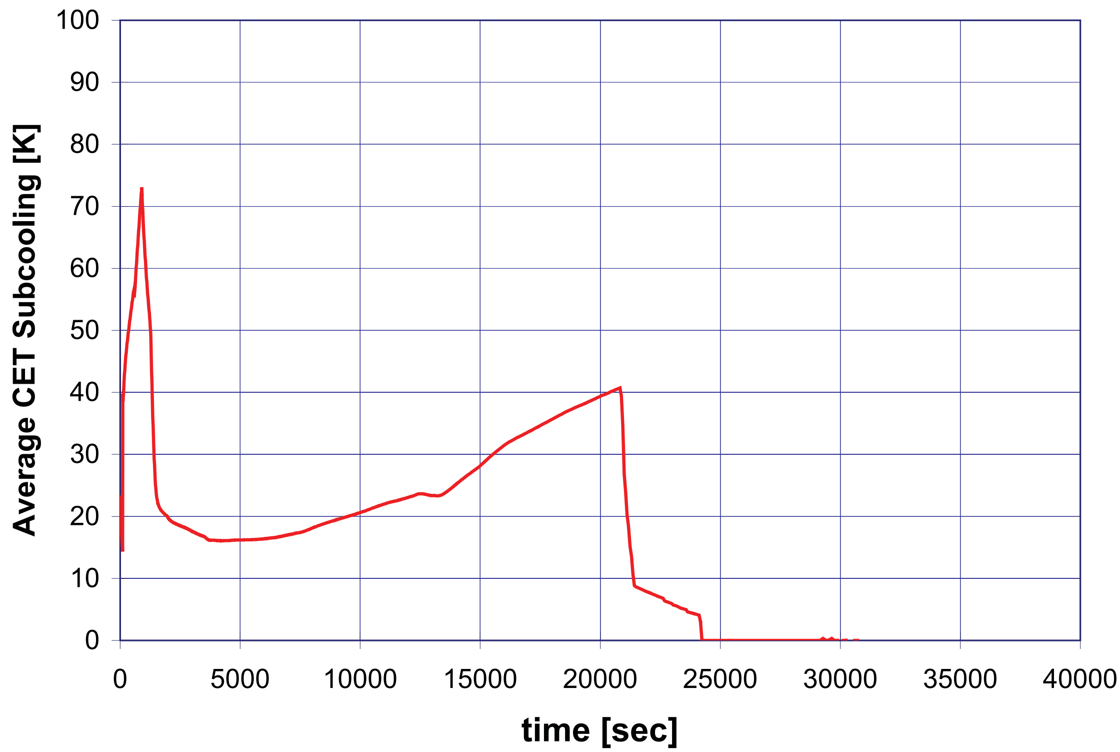

Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
TCL
TFU
2200 F

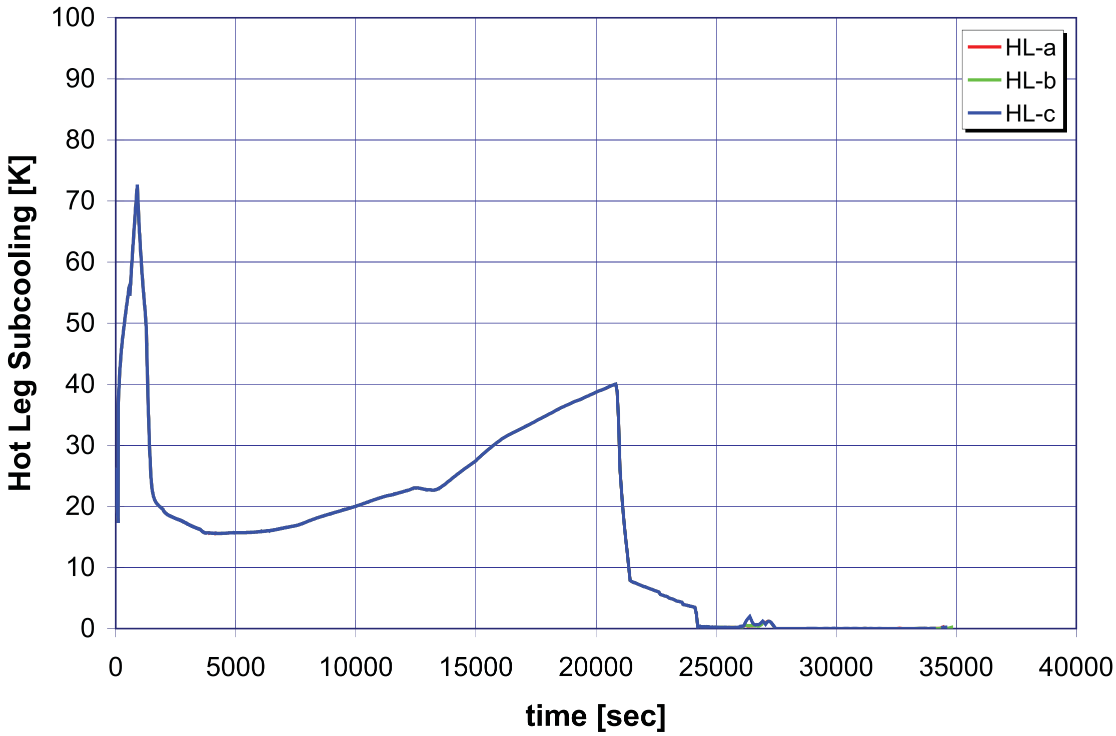
A.2.2.1 Case 2a: 1-Inch Break LOCA with Sprays and Secure Two HHSI Pumps



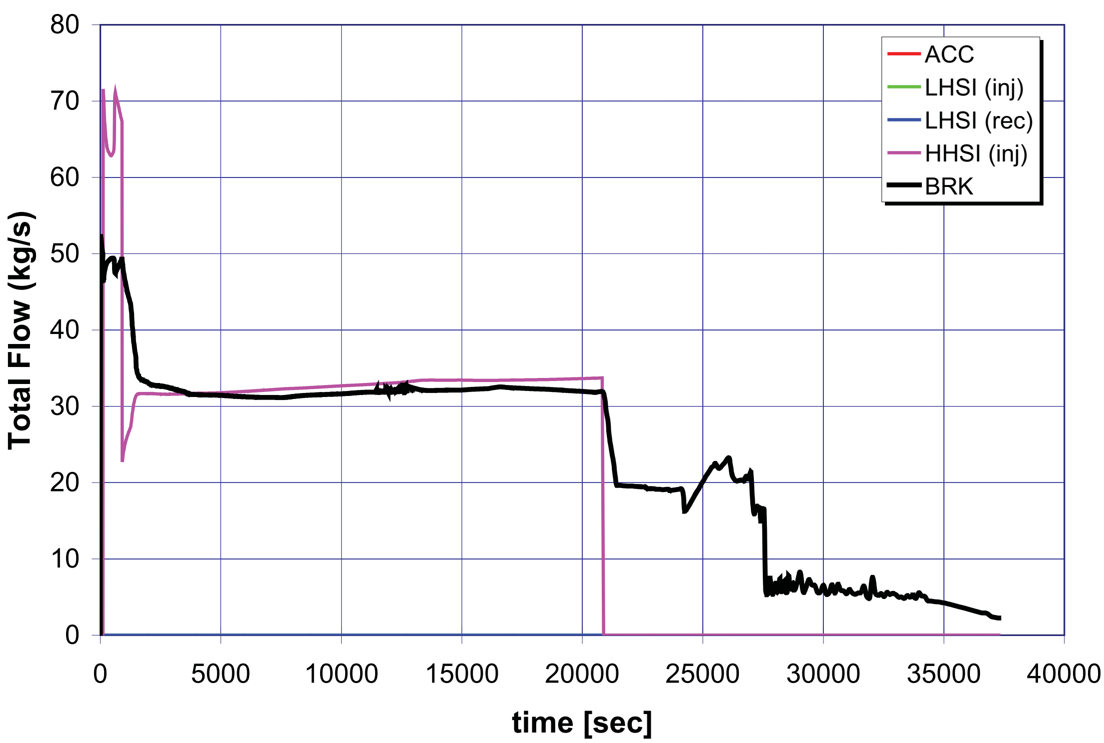



650
600
550
500
450
400
350
300
Average RCS Temperature [K]
RHR Entry Temperature (350 F)
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]

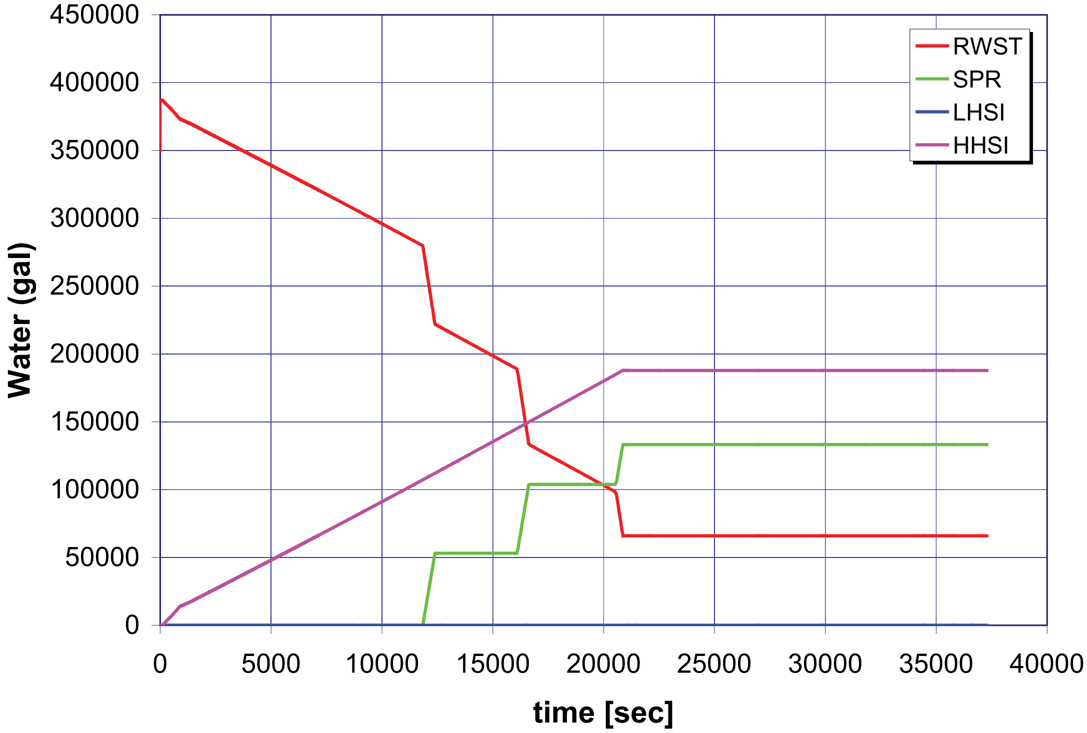

100
90
80
70
60
50
40
30
20
10
0
Average CET Subcooling [K]
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]

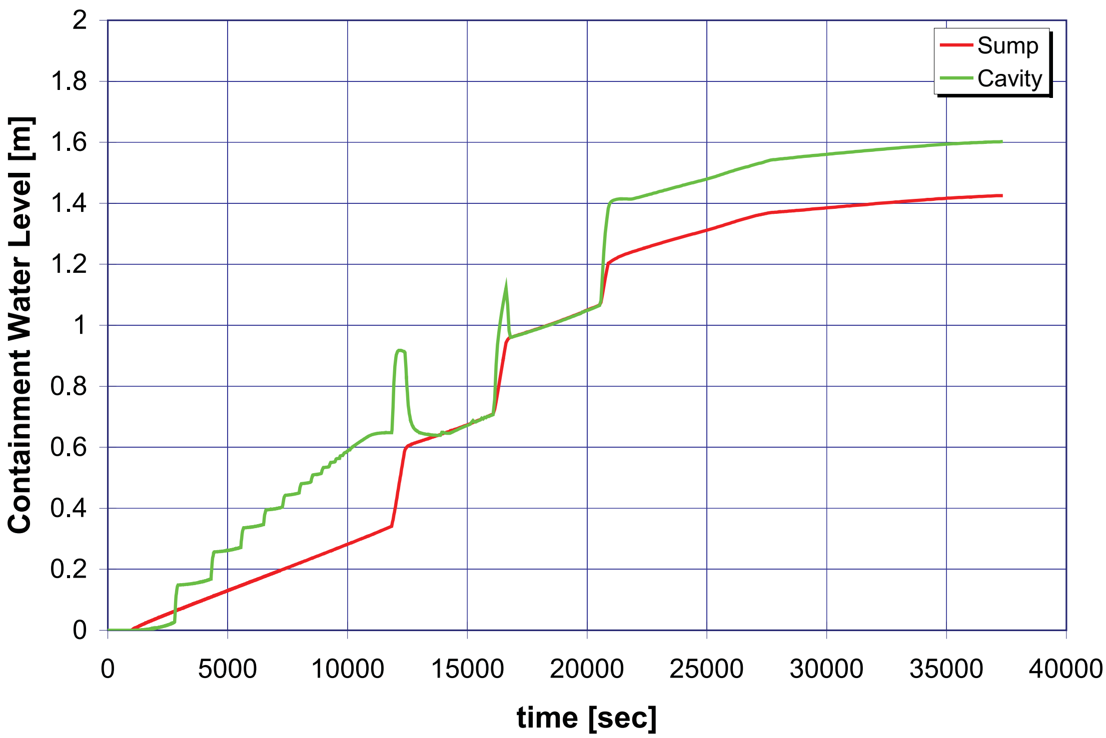


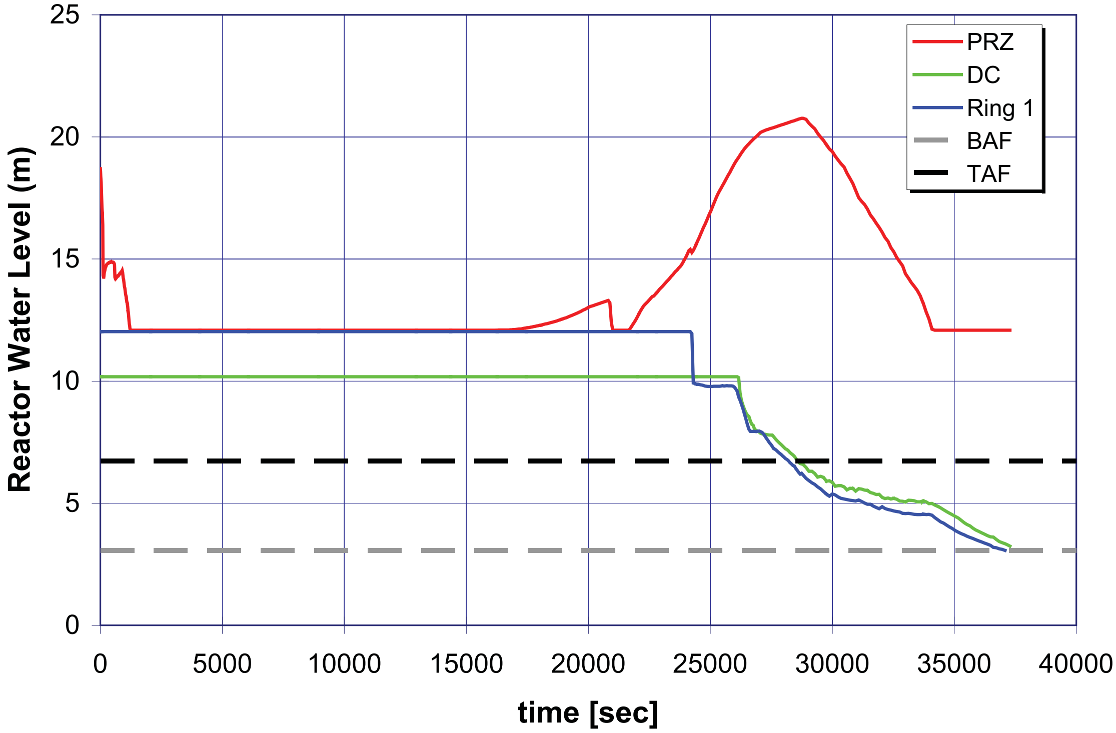
100
90
80
70
60
50
40
30
20
10
0
Hot Leg Subcooling [K]
HL-a
HL-b
HL-c
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]



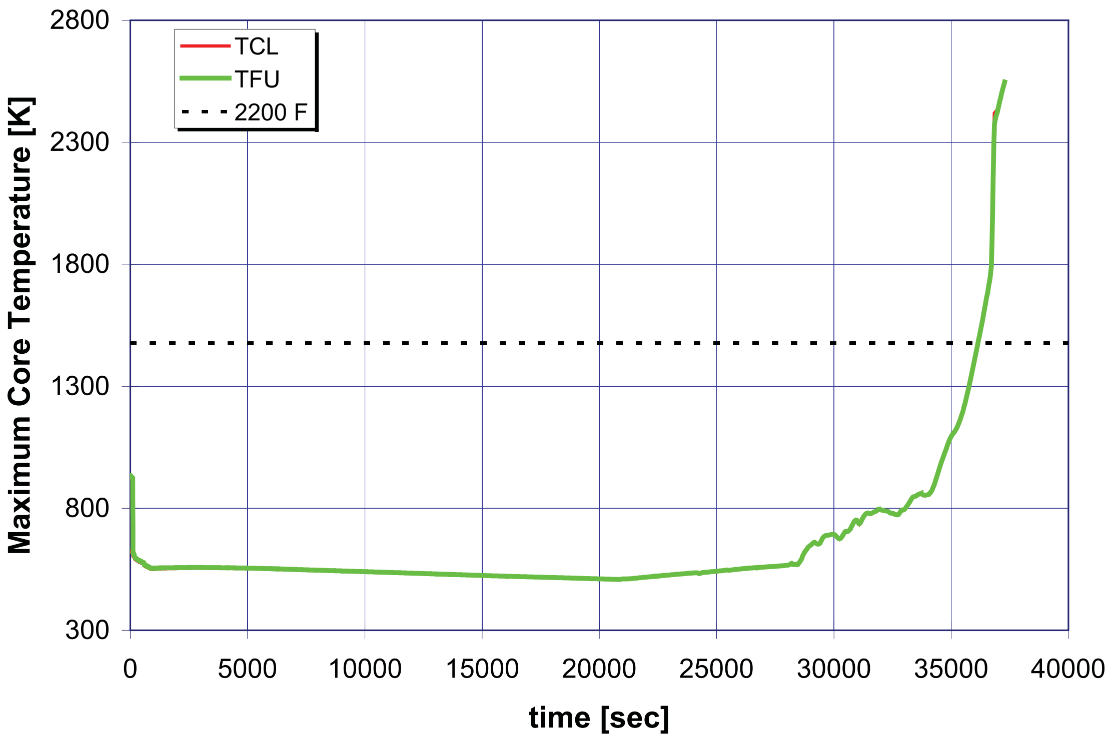
80
70
60
50
40
30
20
10
0
Total Flow (kg/s)
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]


RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
5000
10000
15000
20000
25000
30000
35000
40000


Sump
Cavity
Containment Water Level [m]
time [sec]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
5000
10000
15000
20000
25000
30000
35000
40000

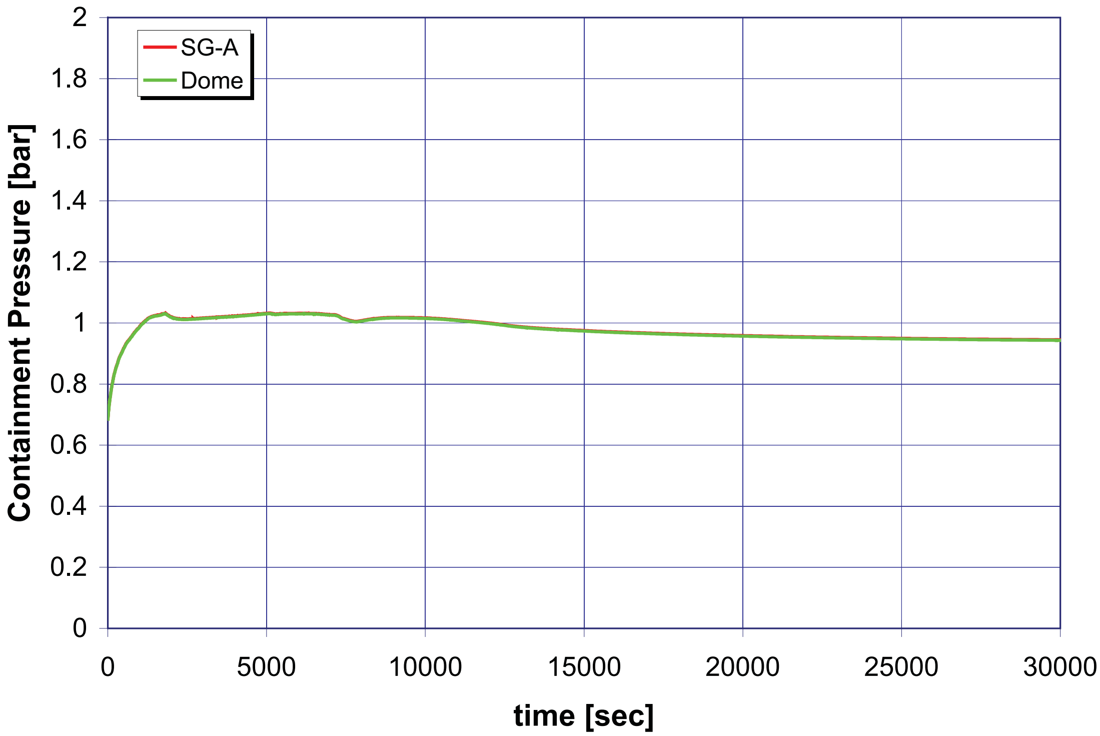


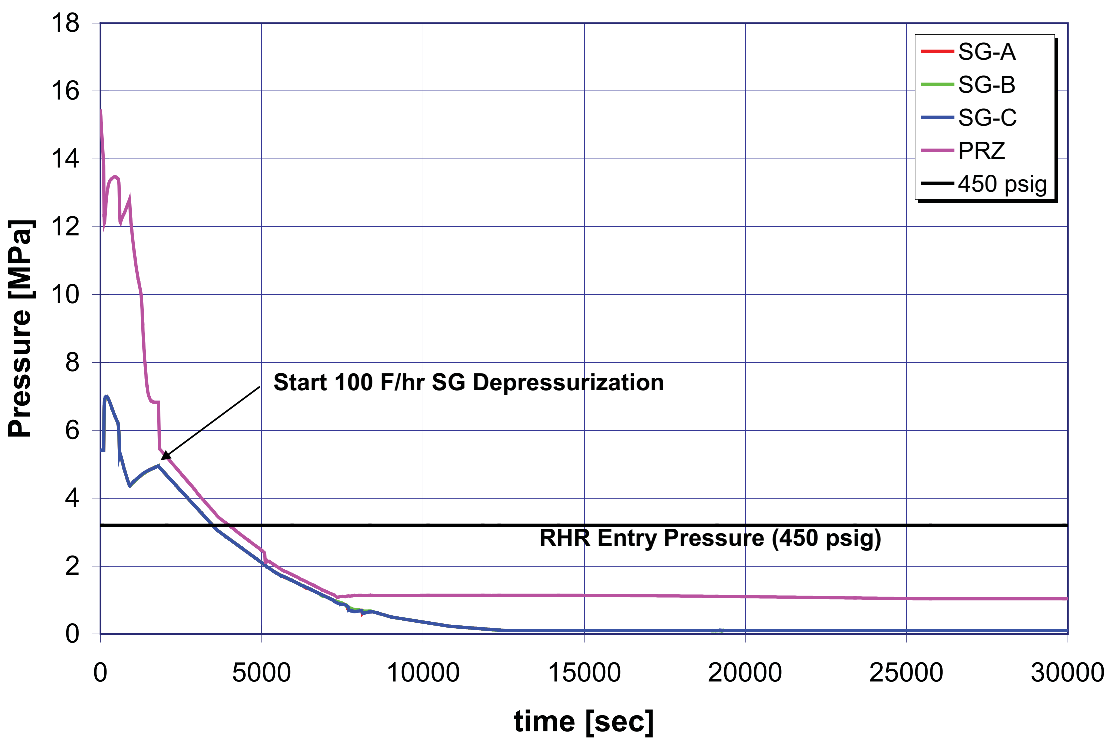
Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
PRZ
DC
Ring 1
BAF
TAF

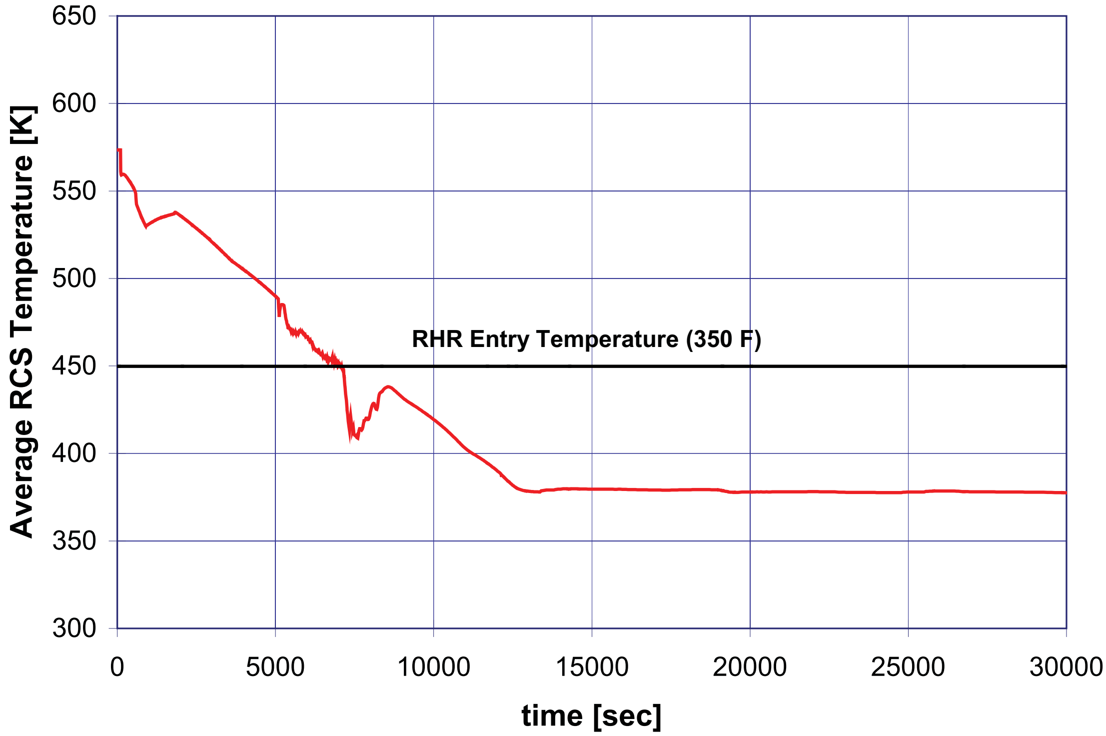


Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
TCL
TFU
2200 F

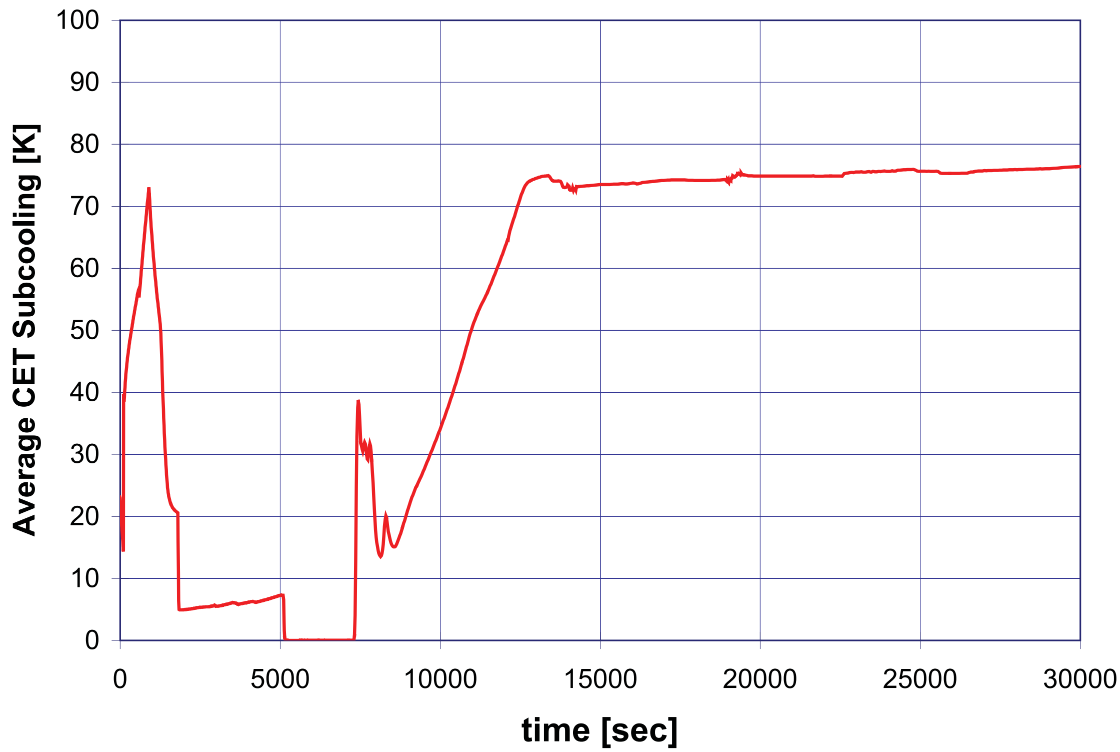
A.2.2.2 Case 2b: 1-Inch Break LOCA with Sprays, Secure HHSI Pumps, and Secondary Cooldown



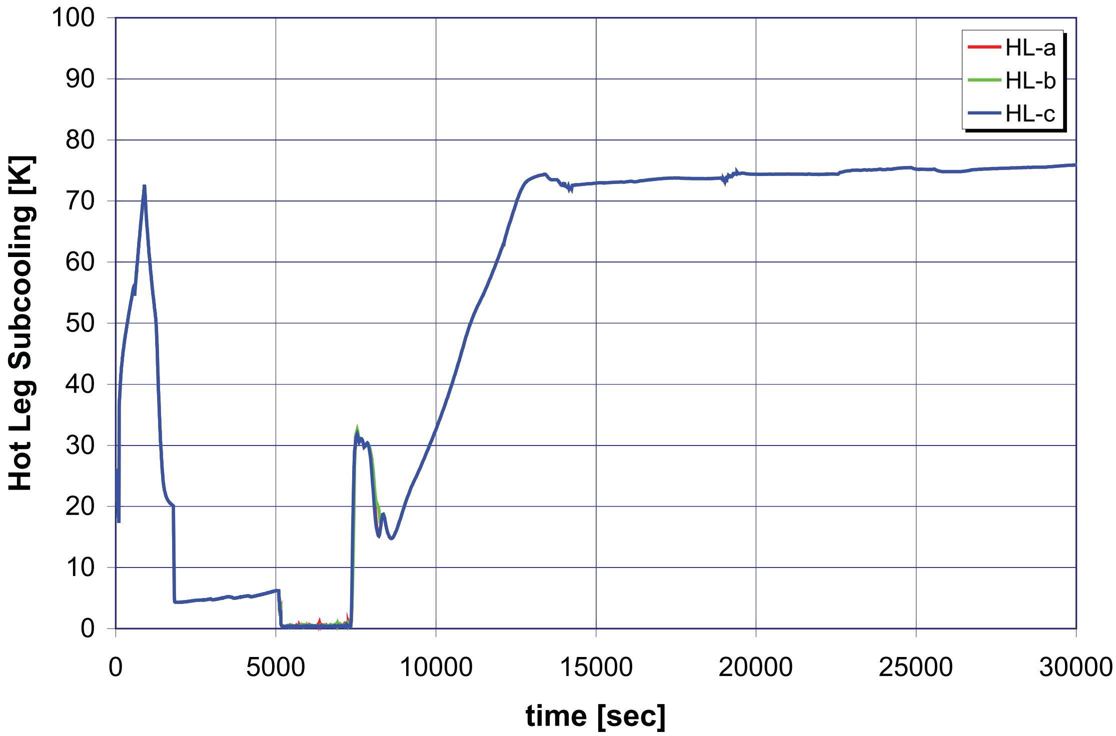




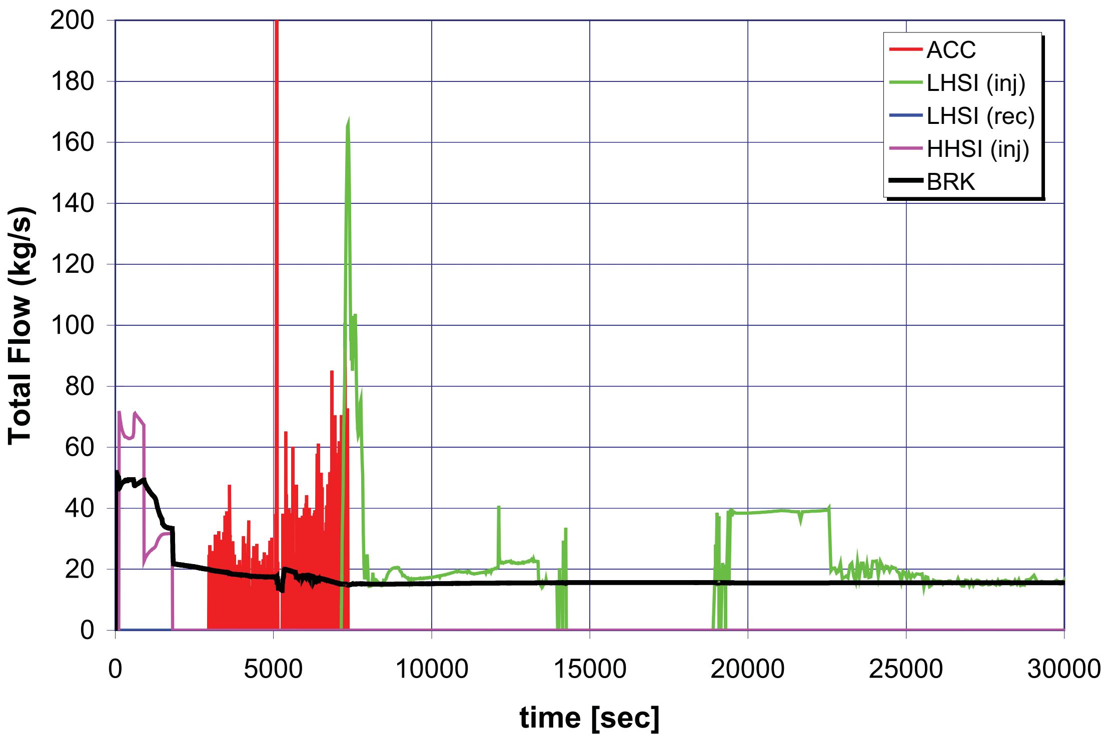
Average RCS Temperature [K]
650
600
550
500
450
400
350
300
RHR Entry Temperature (350 F)
0
5000
10000
15000
20000
25000
30000
time [sec]



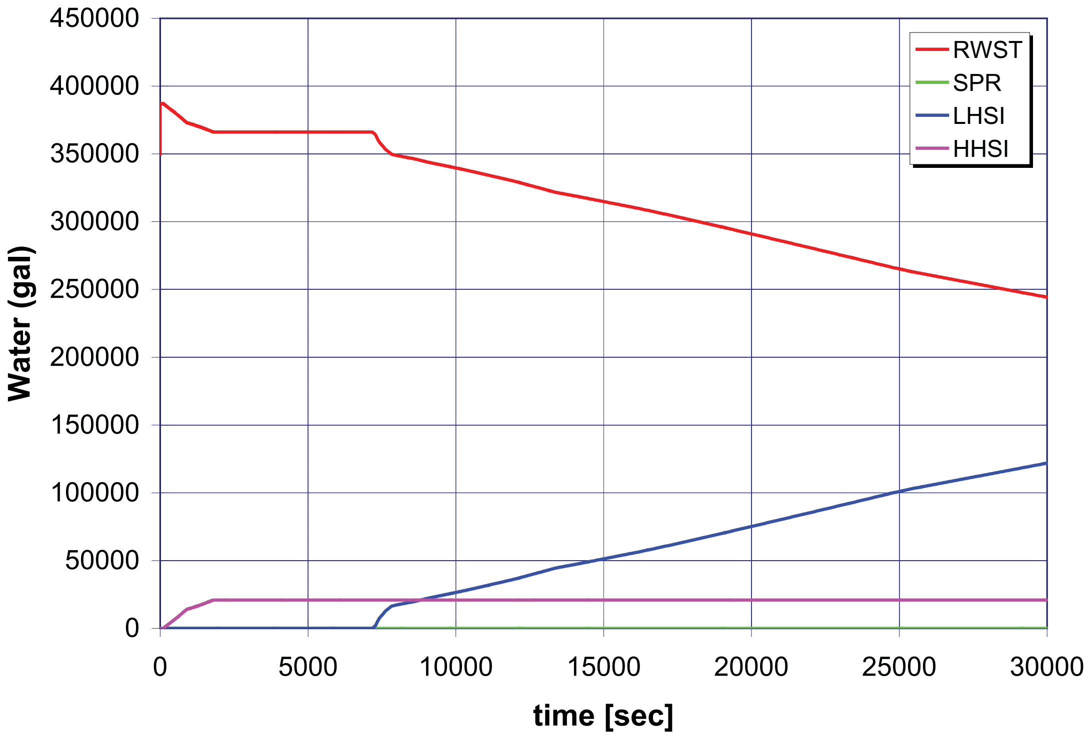
Average CET Subcooling [K]
100
90
80
70
60
50
40
30
20
10
0
0
5000
10000
15000
20000
25000
30000
time [sec]

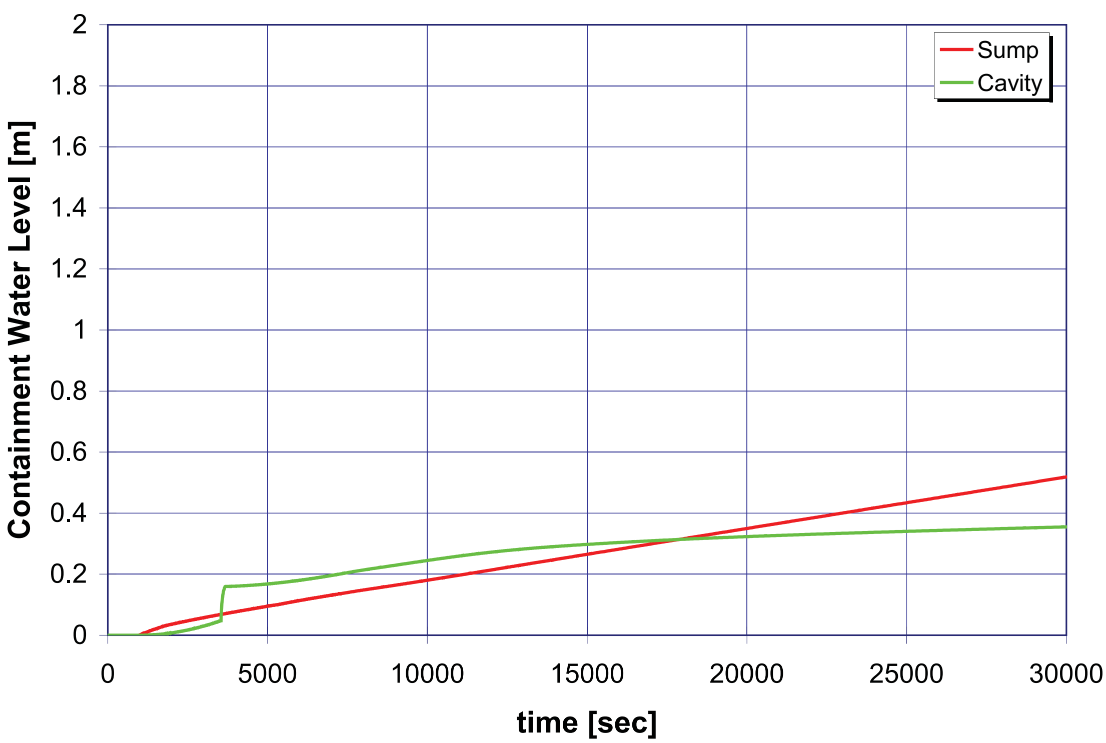


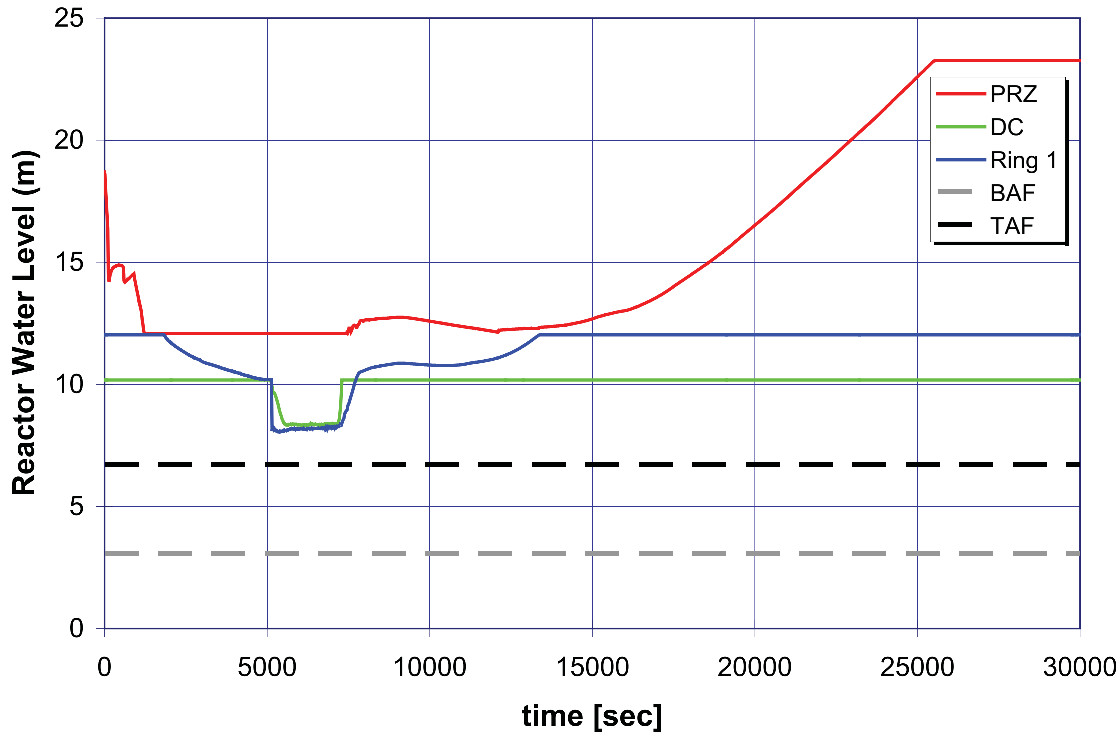
HL-a
HL-b
HL-c
Hot Leg Subcooling [K]
time [sec]
0
10
20
30
40
50
60
70
80
90
100
0
5000
10000
15000
20000
25000
30000



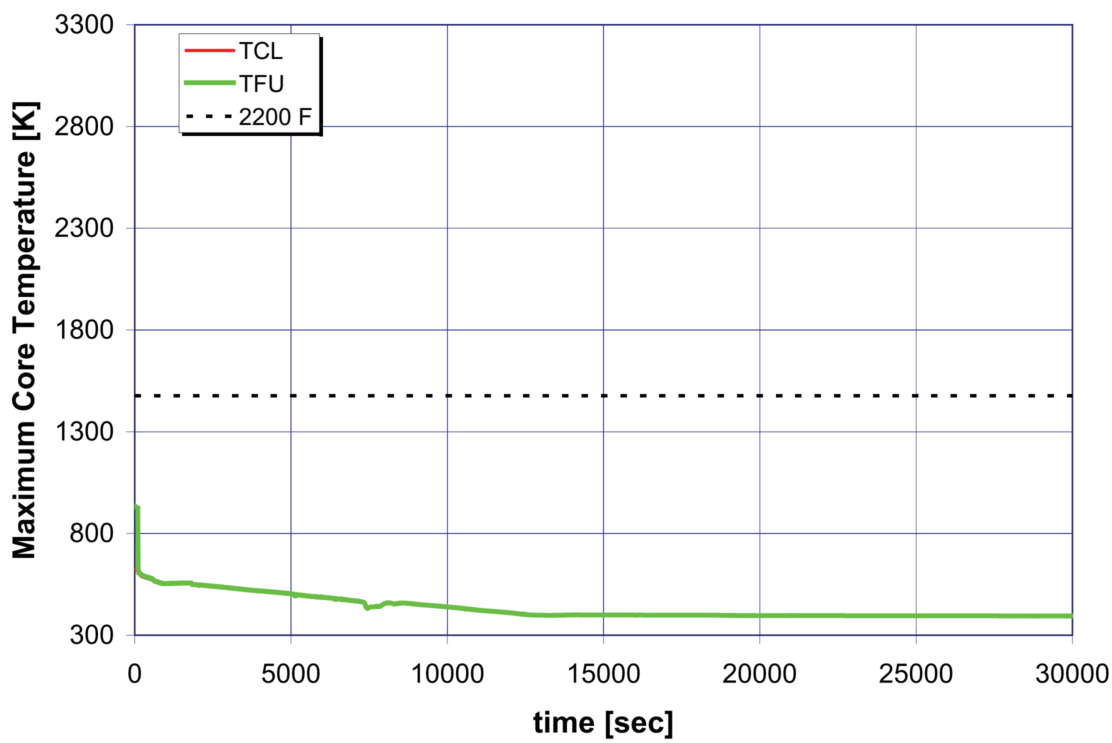
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
0
20
40
60
80
100
120
140
160
180
200
0
5000
10000
15000
20000
25000
30000


Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
5000
10000
15000
20000
25000
30000


Containment Water Level [m]
time [sec]
Sump
Cavity
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
5000
10000
15000
20000
25000
30000

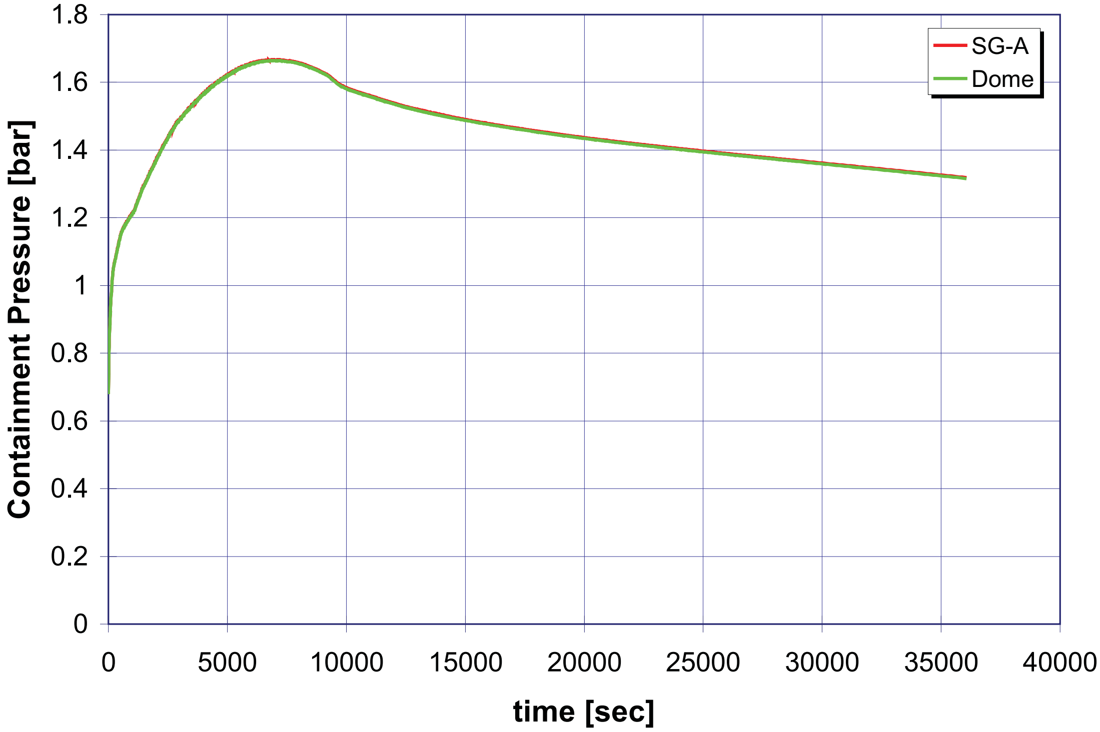


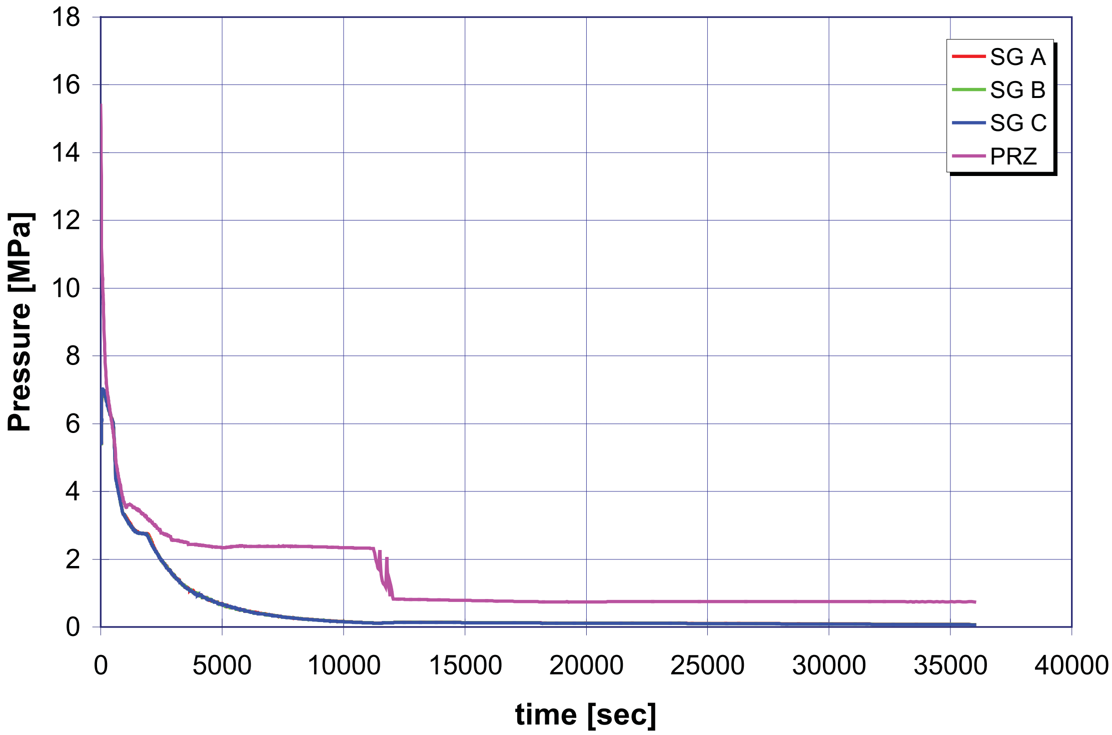
Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
time [sec]
PRZ
DC
Ring 1
BAF
TAF

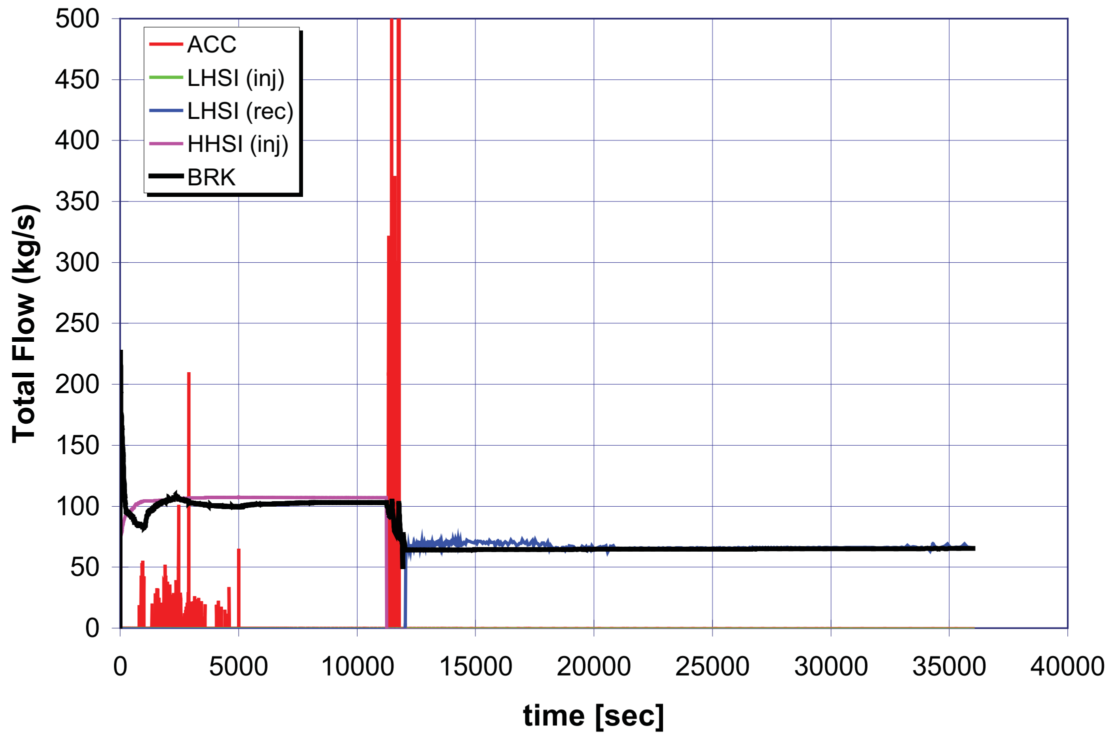


Maximum Core Temperature [K]
3300
2800
2300
1800
1300
800
300
0
5000
10000
15000
20000
25000
30000
time [sec]
TCL
TFU
2200 F

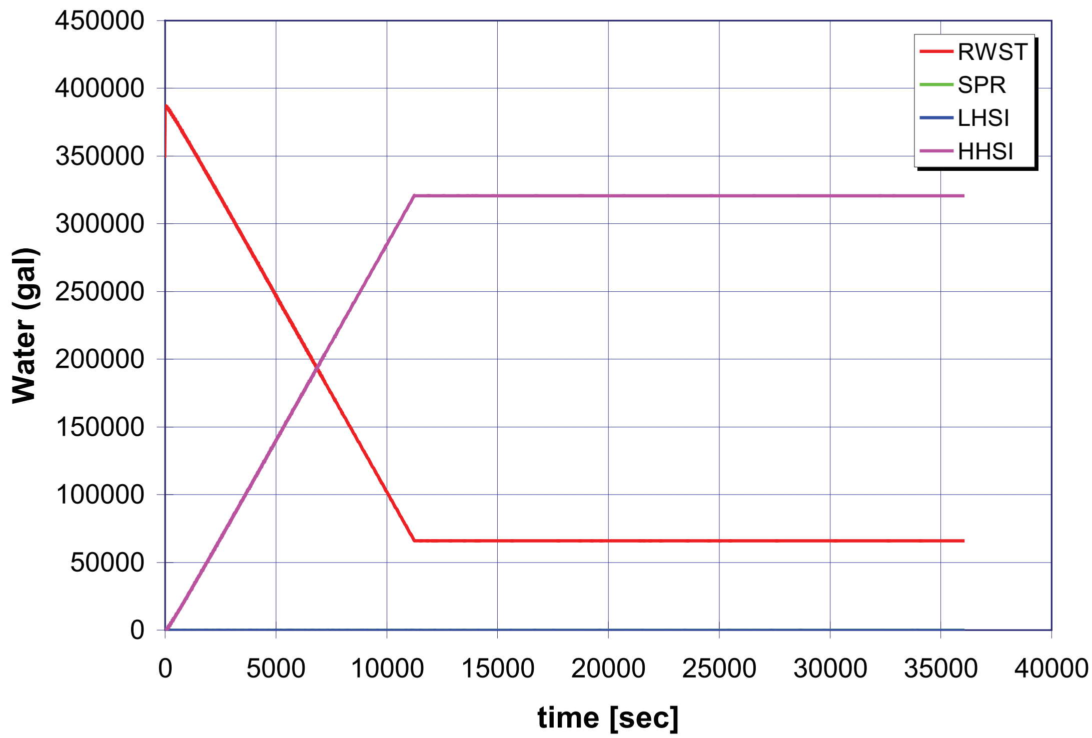
A.2.3 Case 3: 2-Inch Break LOCA without Sprays



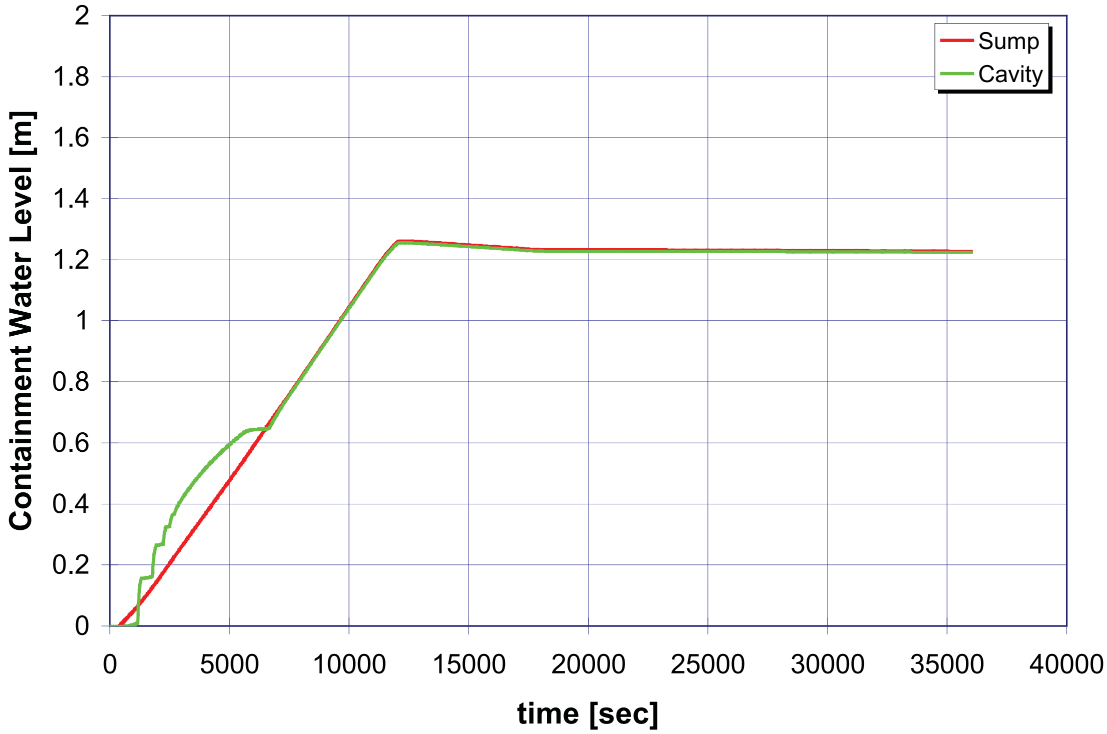


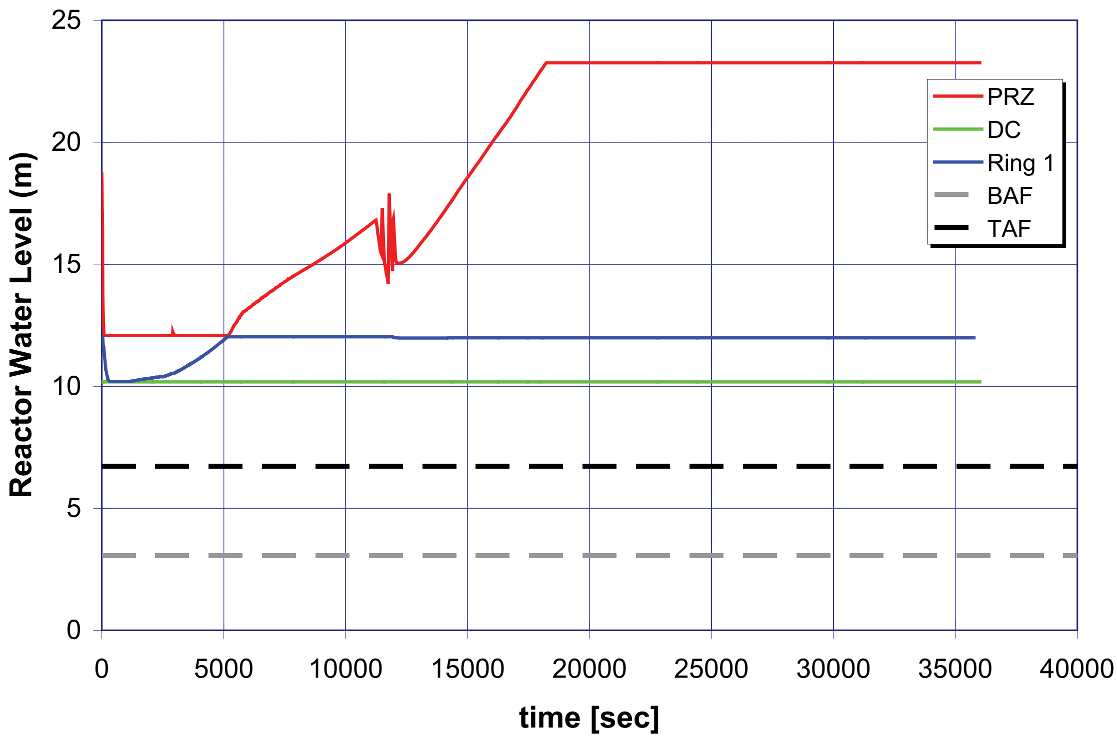


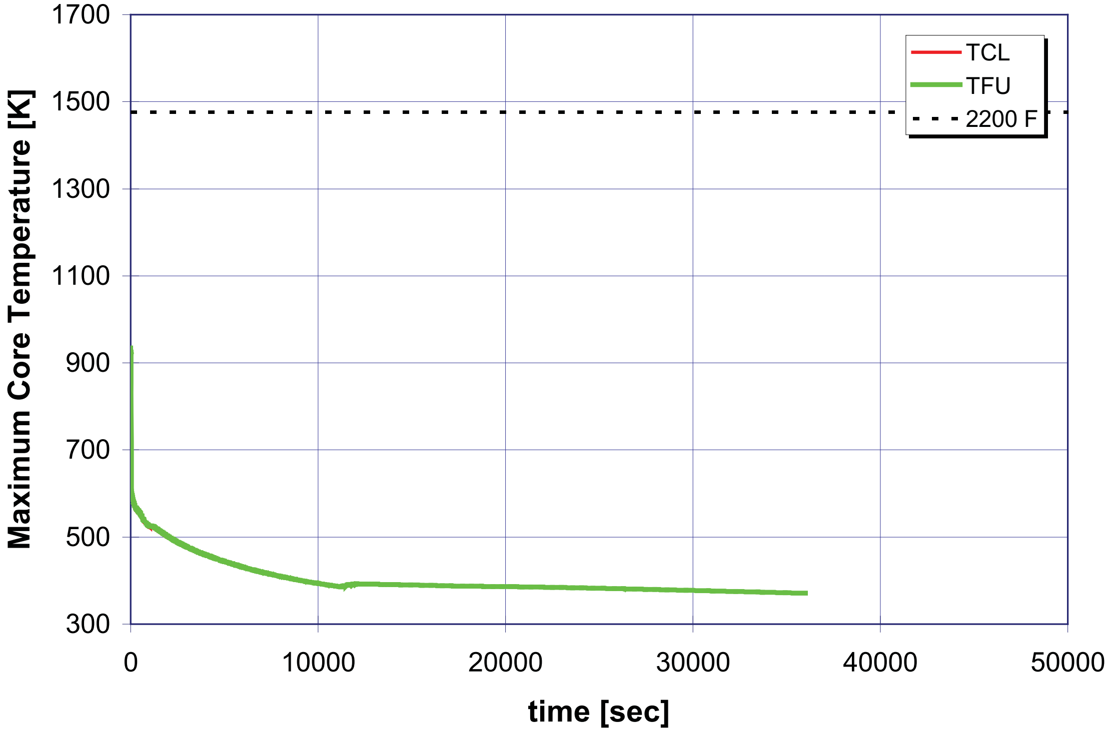
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]



RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]

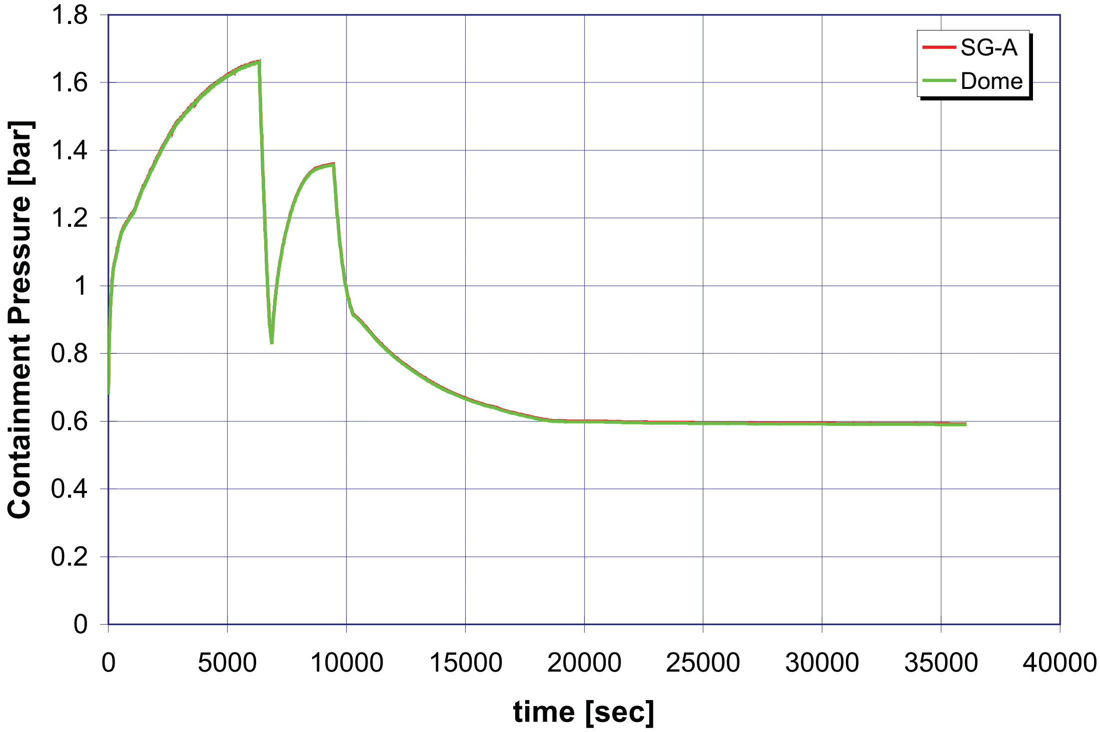

Containment Water Level [m]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
Sump
Cavity

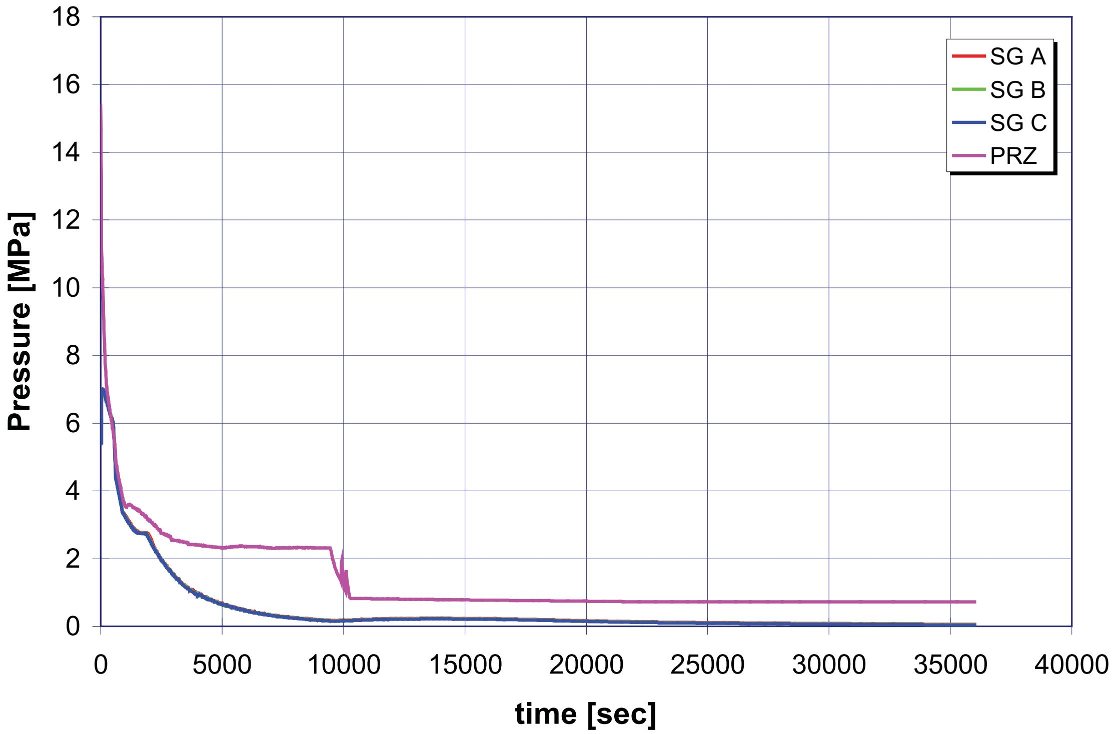

Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
PRZ
DC
Ring 1
BAF
TAF

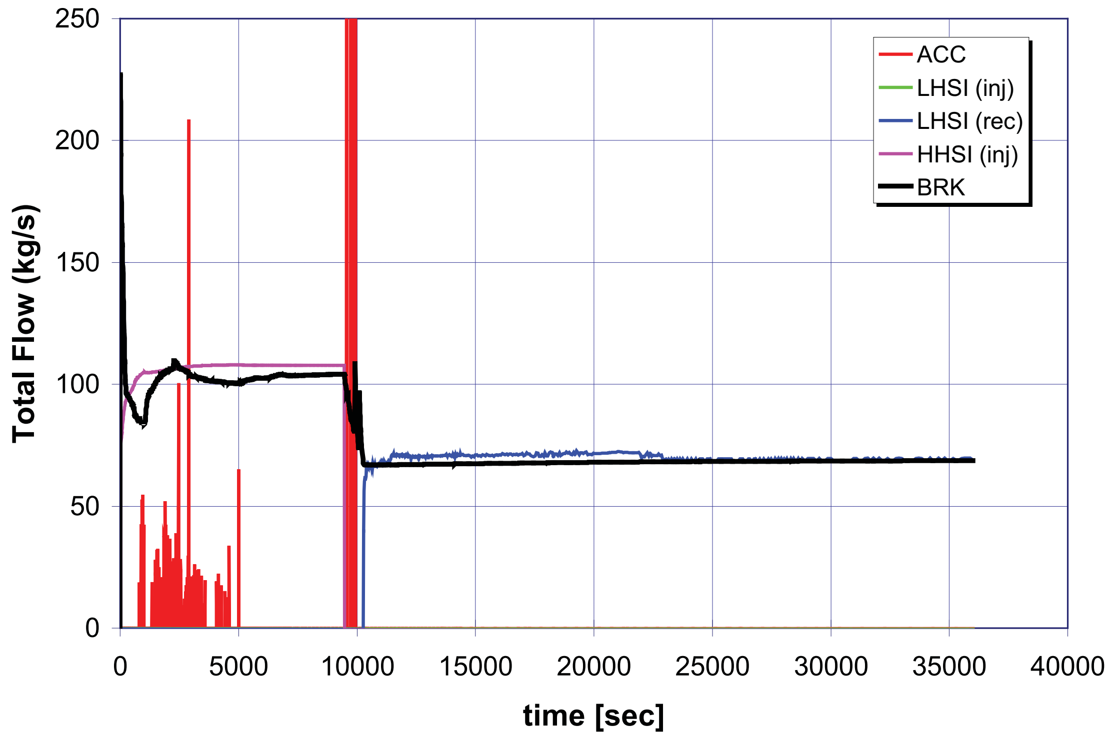


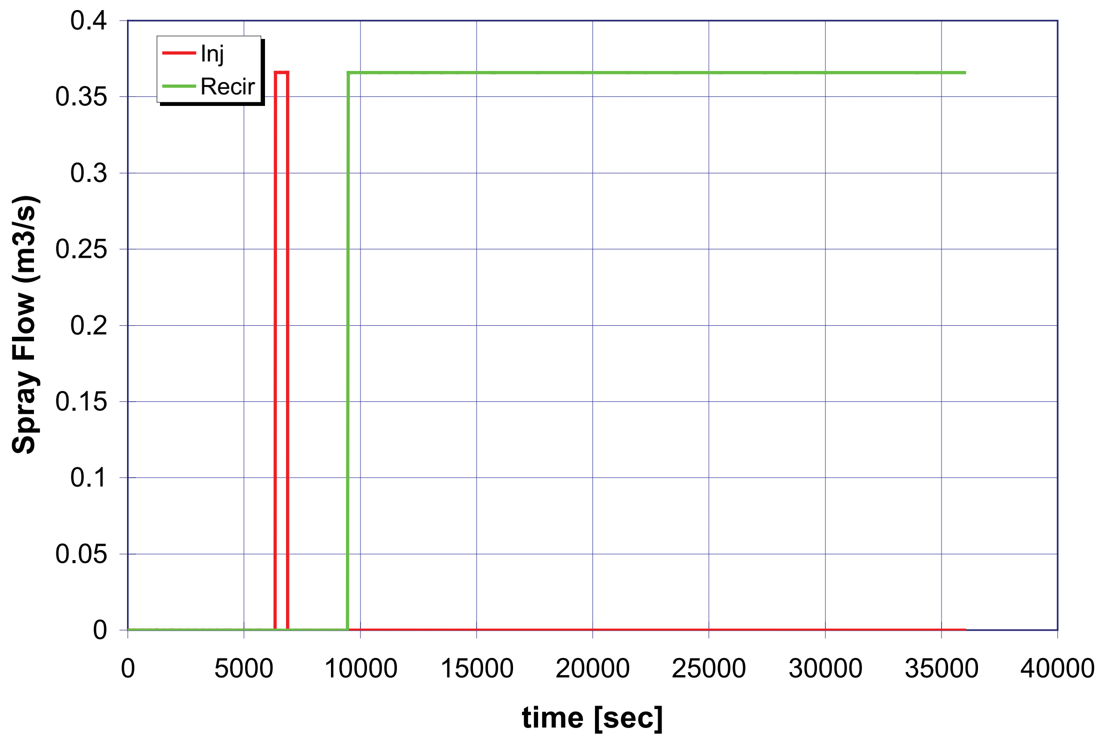
1700
1500
1300
1100
900
700
500
300
Maximum Core Temperature [K]
0
10000
20000
30000
40000
50000
time [sec]
TCL
TFU
2200 F

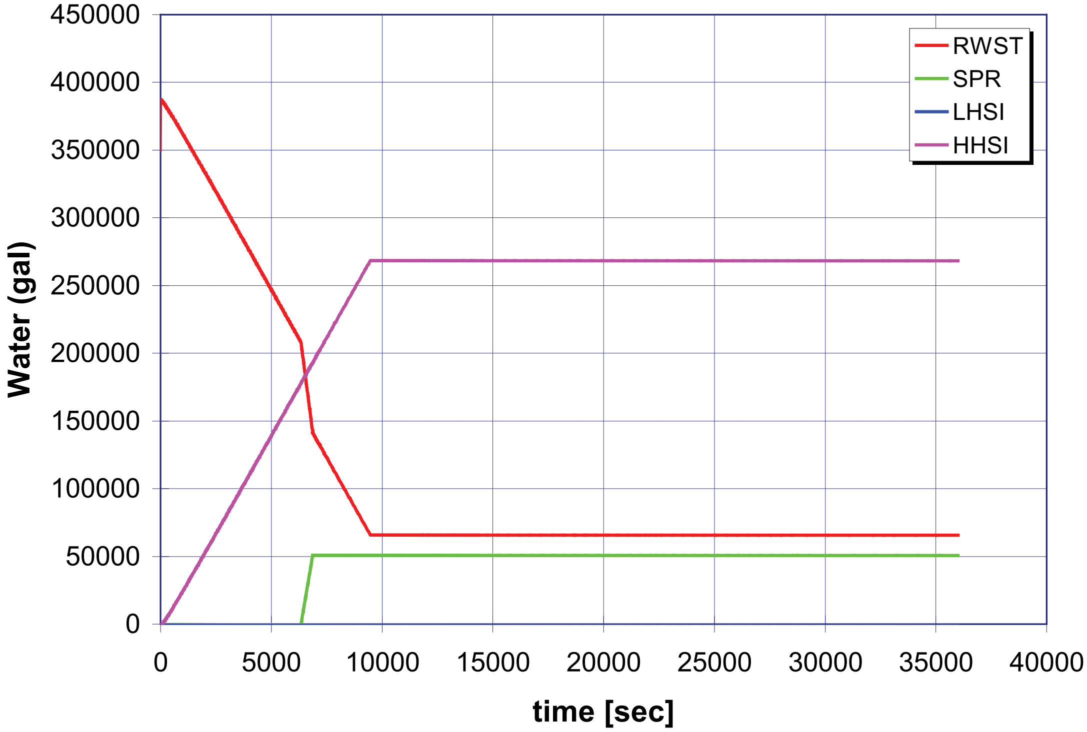
A.2.4 Case 4: 2-Inch Break LOCA with Sprays

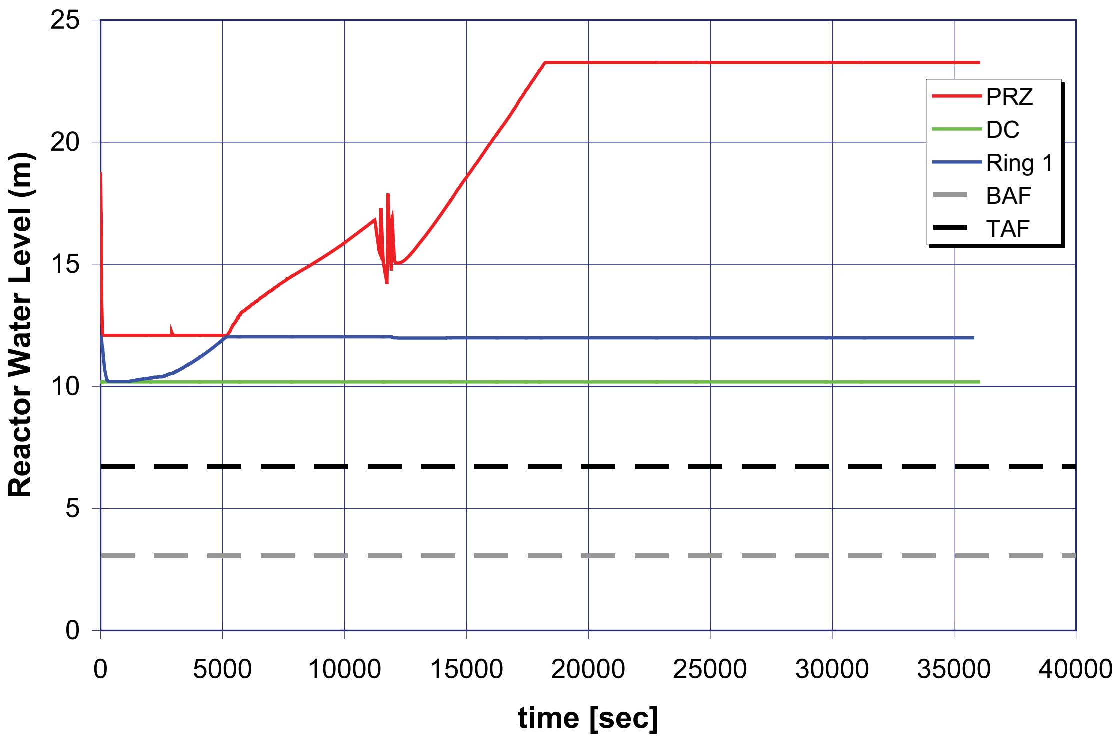


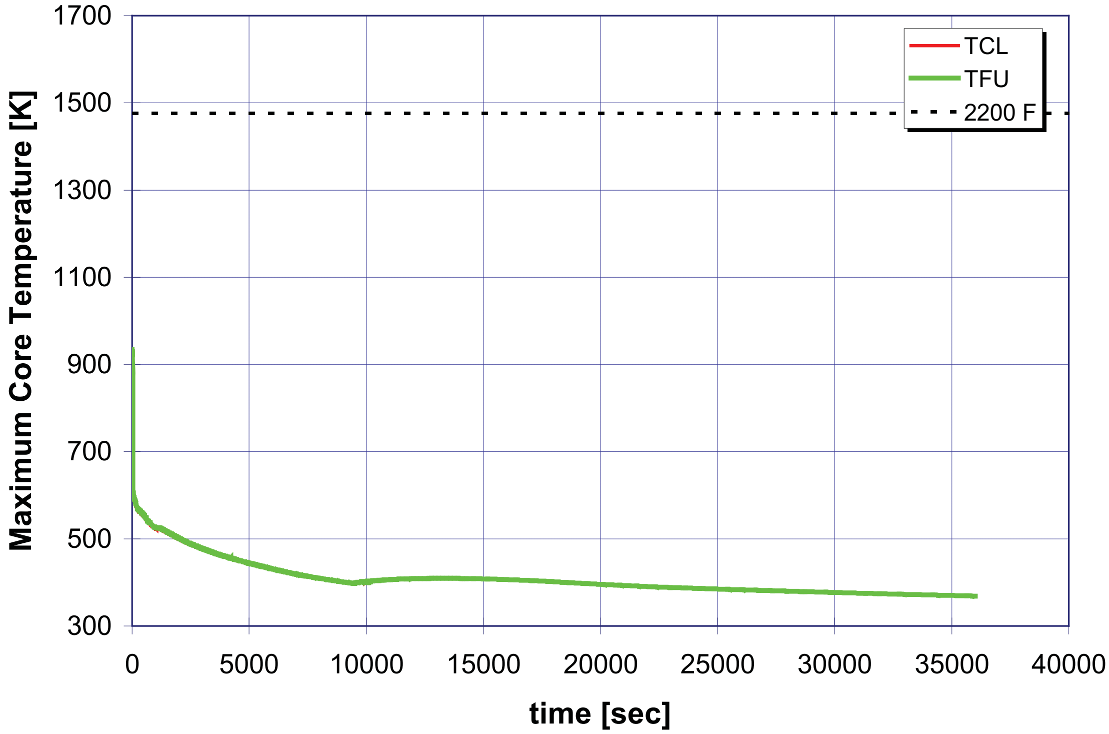



ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
250
200
150
100
50
0
0
5000
10000
15000
20000
25000
30000
35000
40000


Inj
Recir
Spray Flow (m3/s)
time [sec]
0.4
0.35
0.3
0.25
0.2
0.15
0.1
0.05
0
0
5000
10000
15000
20000
25000
30000
35000
40000

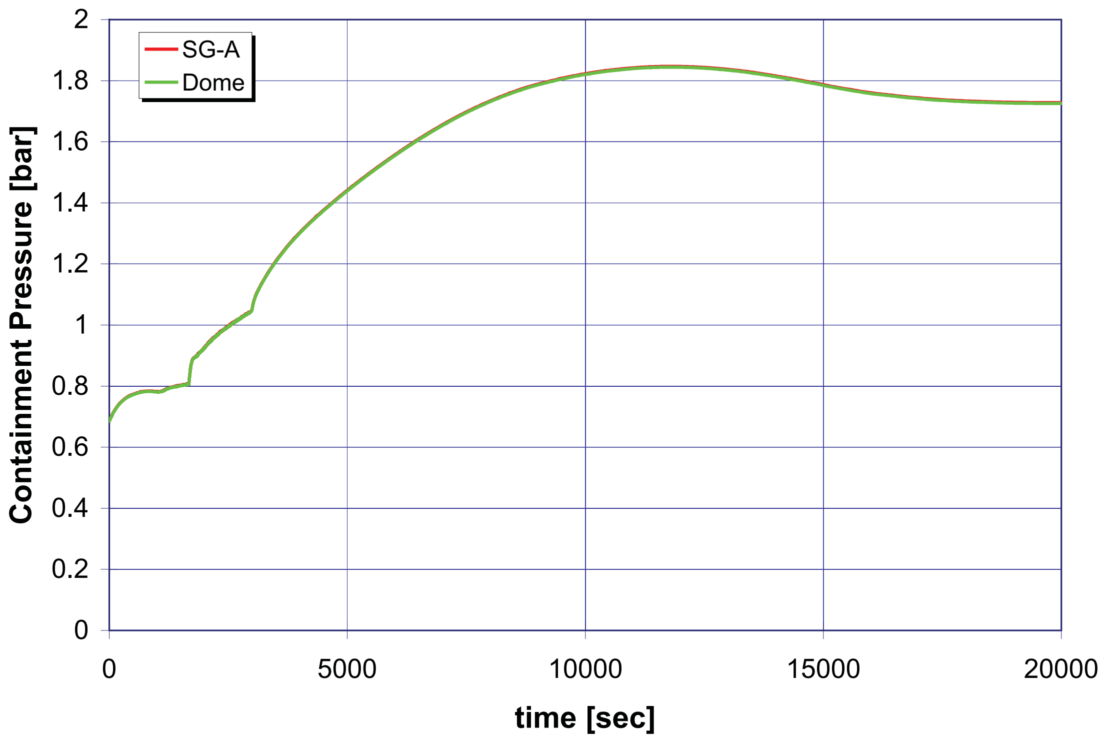

Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
5000
10000
15000
20000
25000
30000
35000
40000

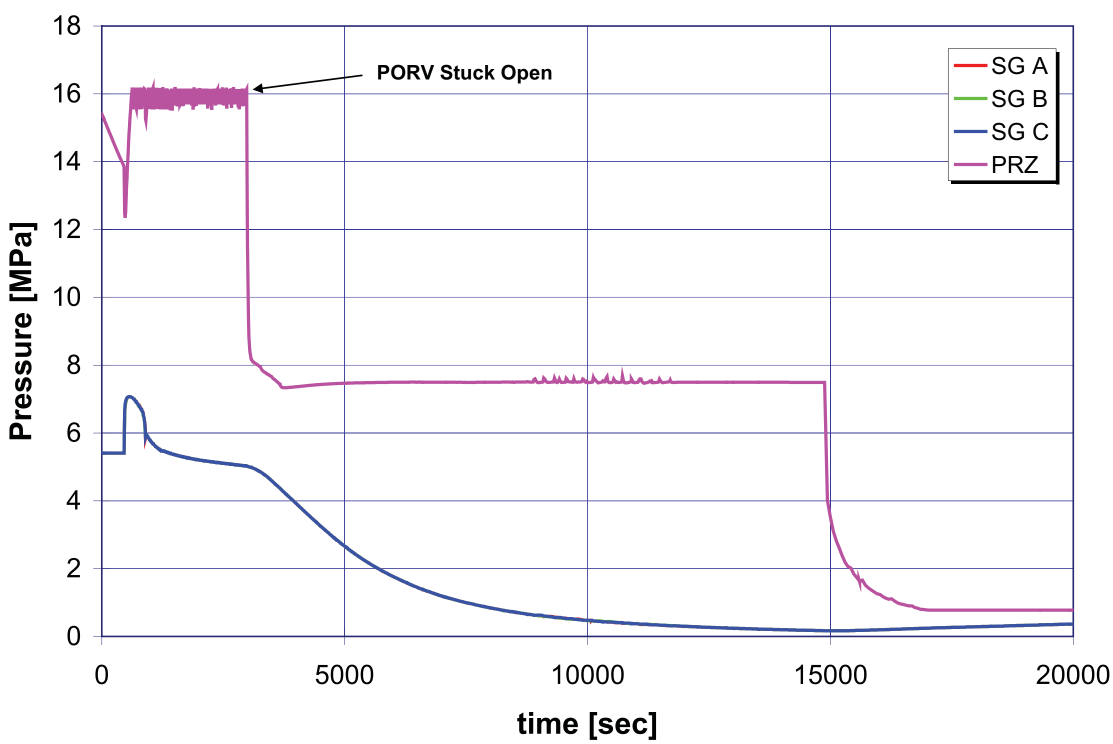

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
25
20
15
10
5
0
5000
10000
15000
20000
25000
30000
35000
40000

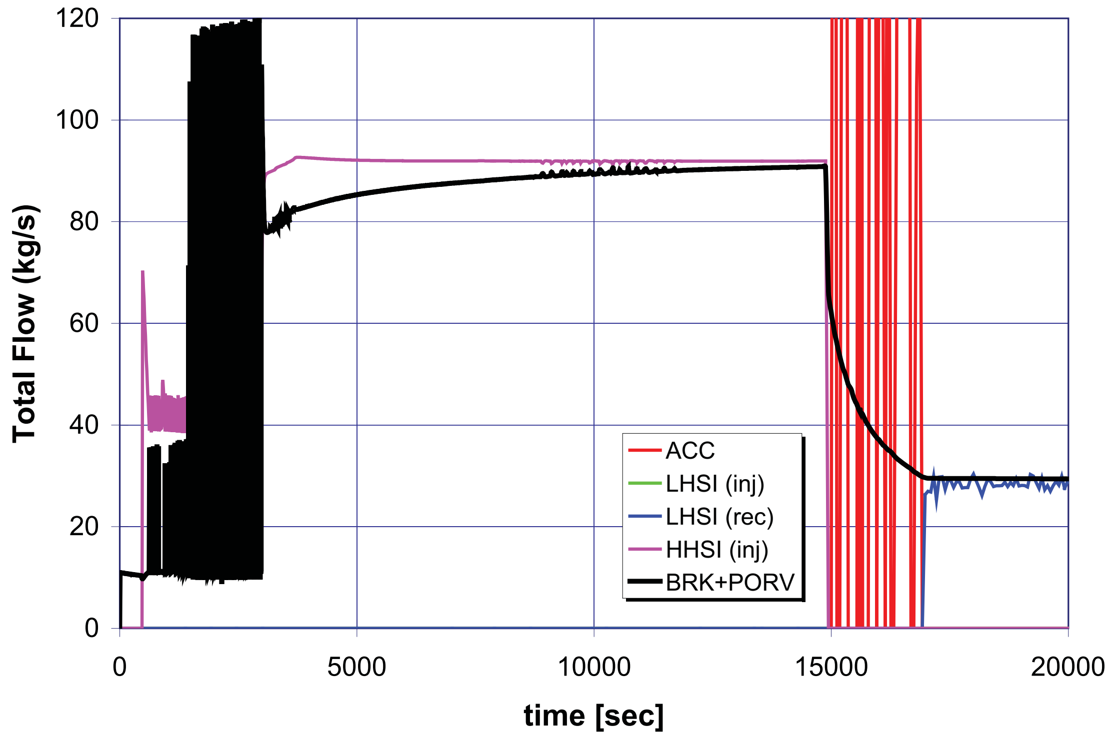


1700
1500
1300
1100
900
700
500
300
0
5000
10000
15000
20000
25000
30000
35000
40000
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F

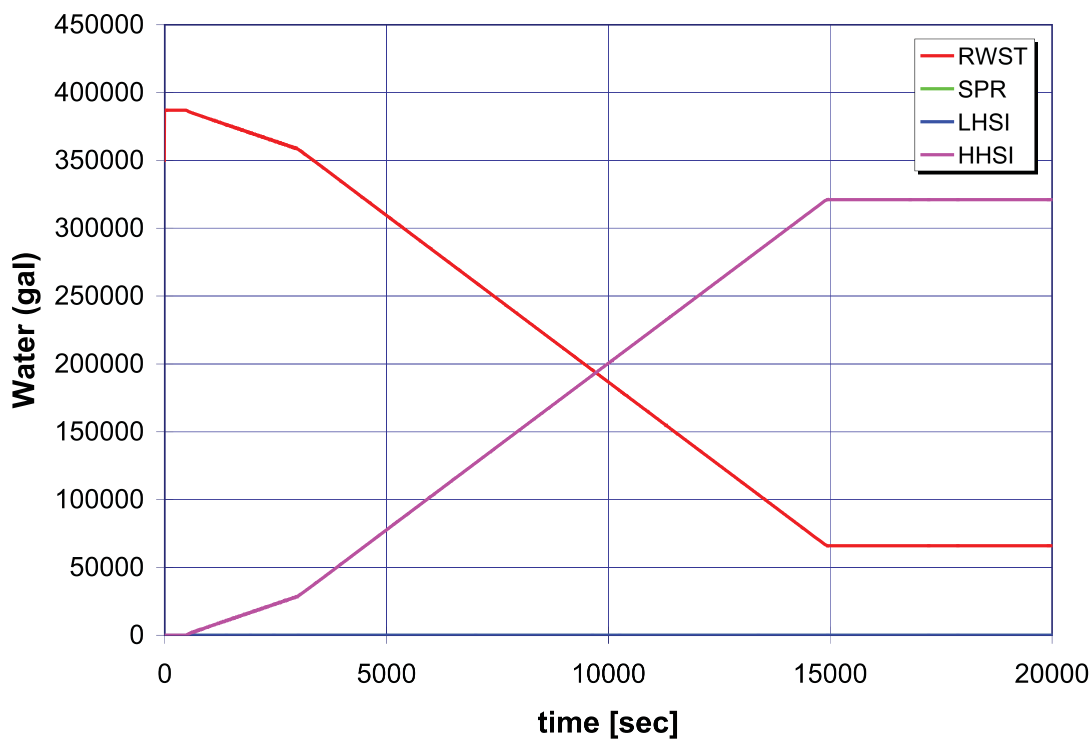
A.2.5 Case 5: 0.5-Inch Break LOCA without Sprays



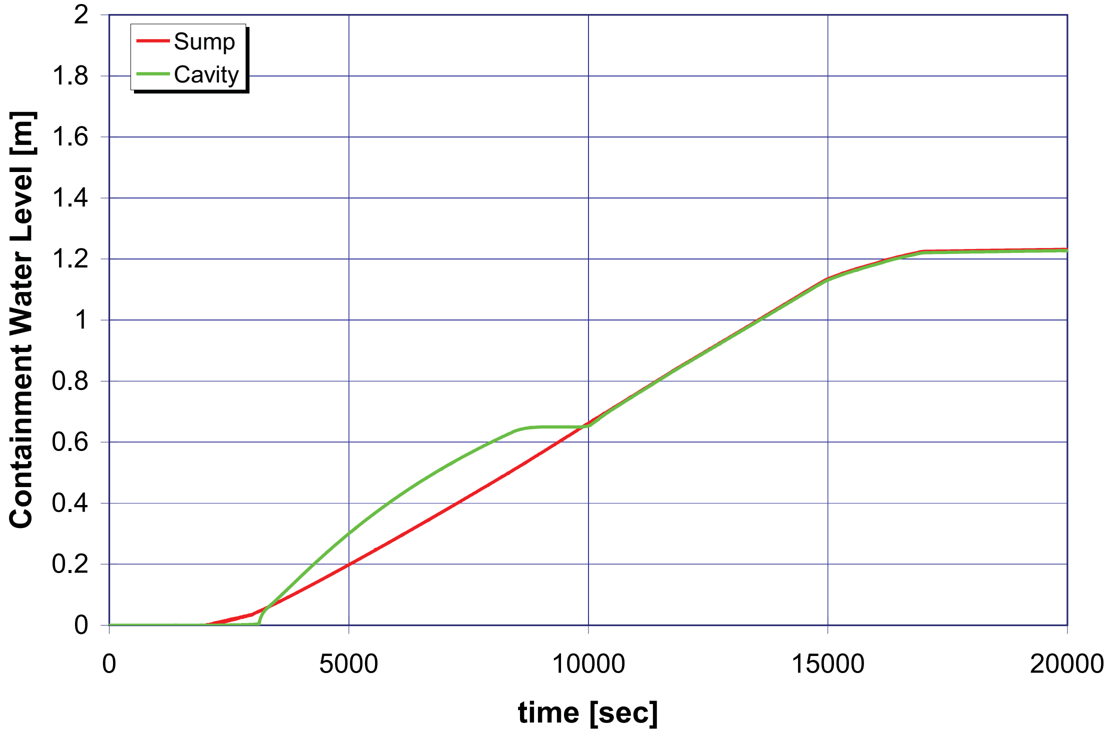


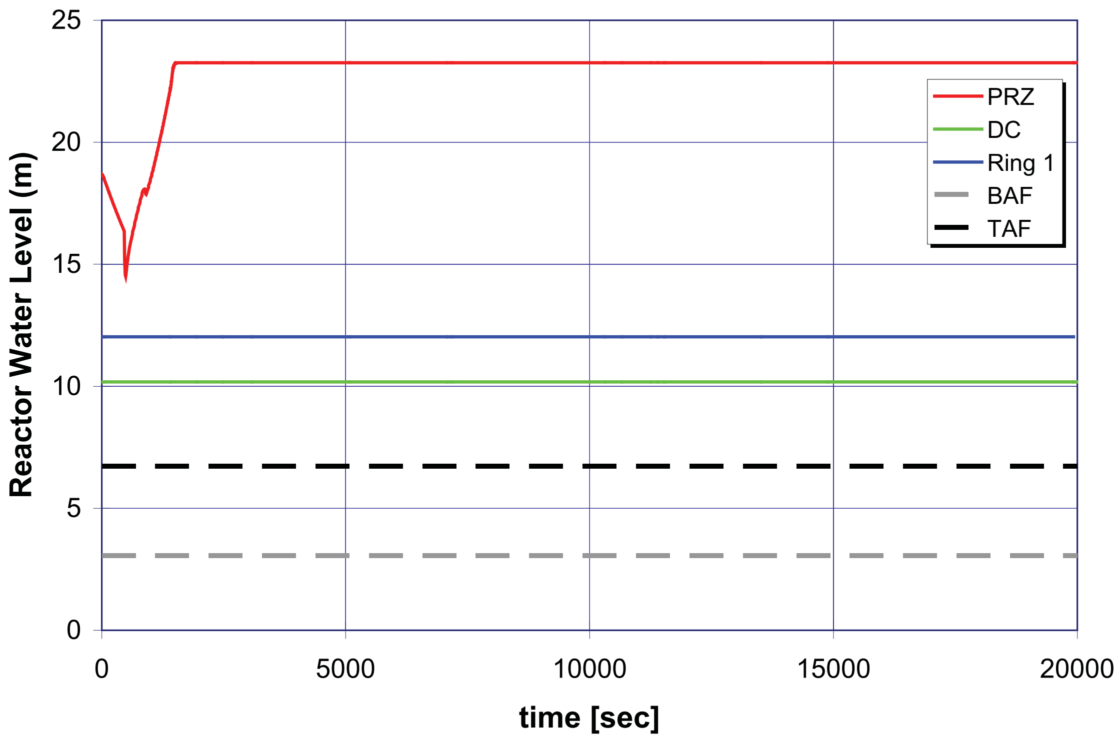


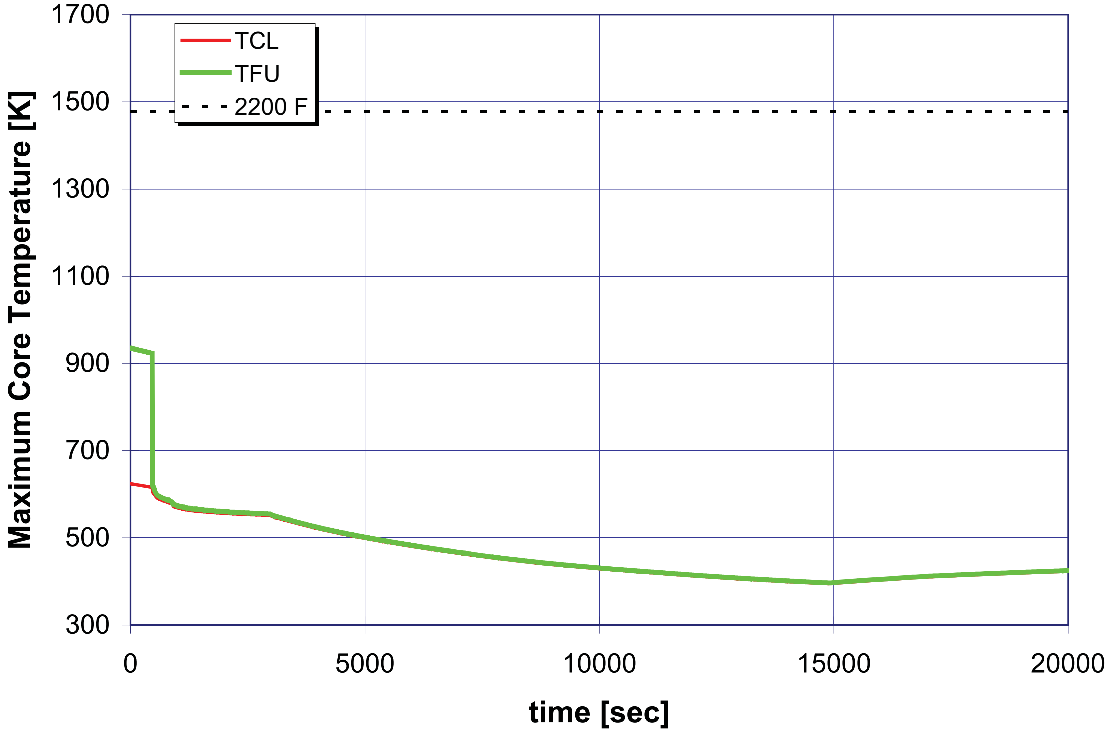
Total Flow (kg/s)
120
100
80
60
40
20
0
0
5000
10000
15000
20000
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK+PORV



Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
5000
10000
15000
20000
time [sec]
RWST
SPR
LHSI
HHSI

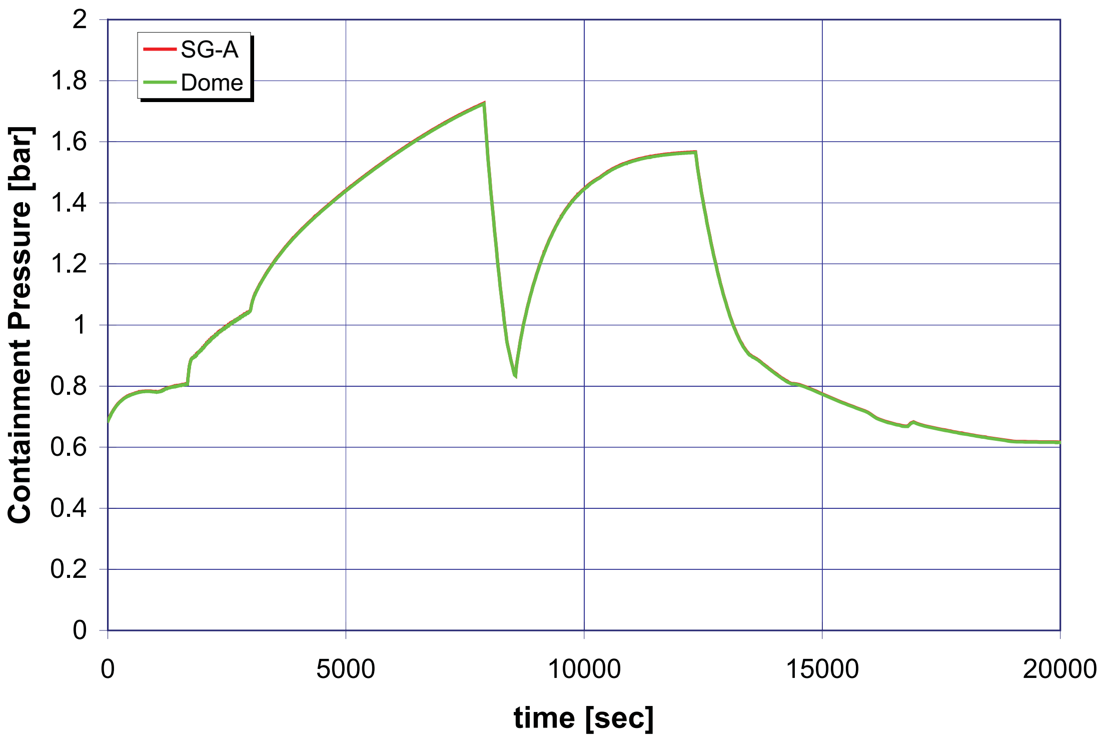

Sump
Cavity
Containment Water Level [m]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
5000
10000
15000
20000
time [sec]

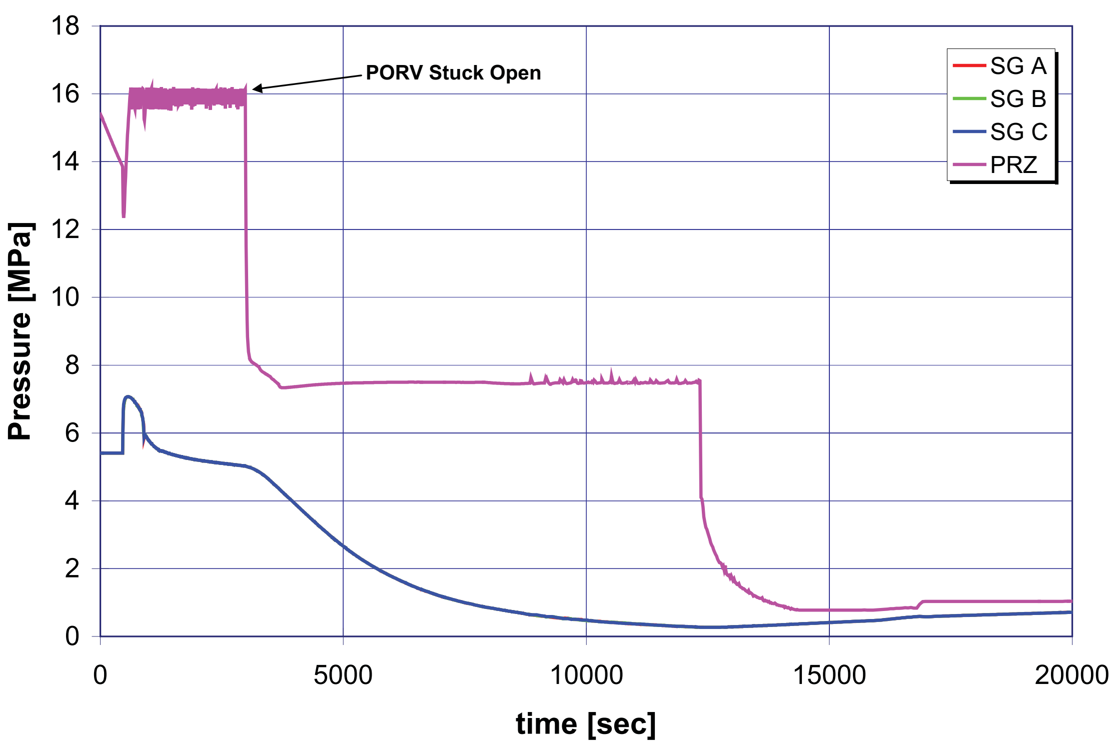

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
time [sec]

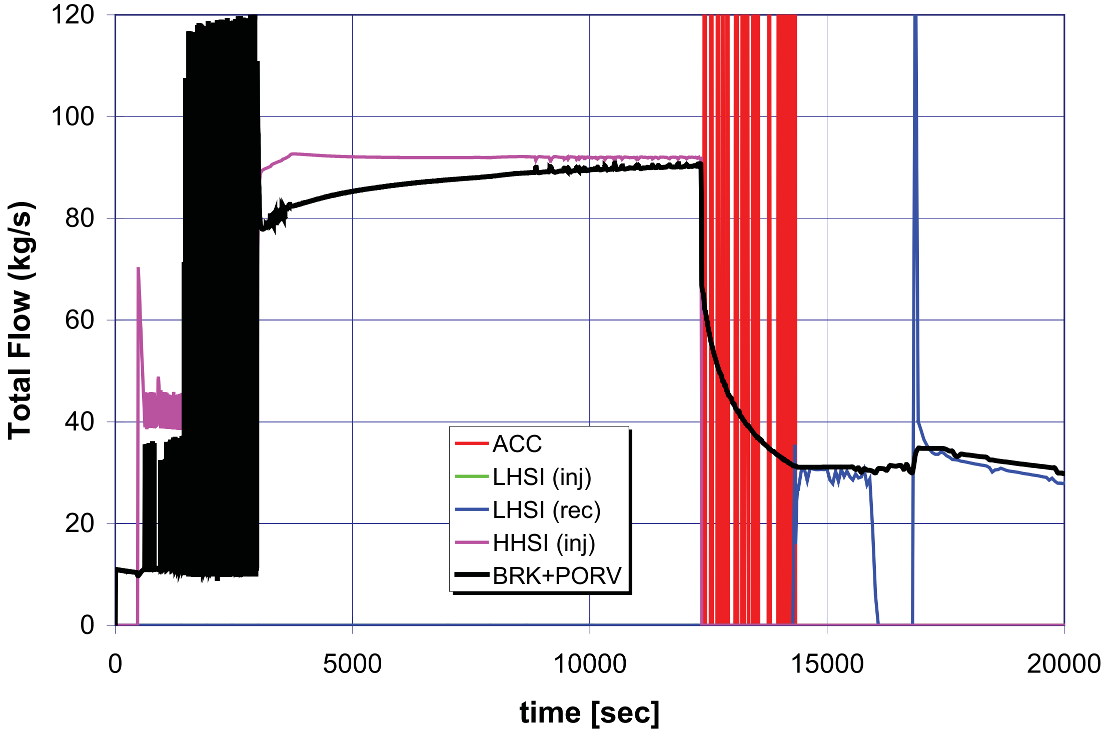


1700
1500
1300
1100
900
700
500
300
Maximum Core Temperature [K]
0
5000
10000
15000
20000
time [sec]
TCL
TFU
2200 F

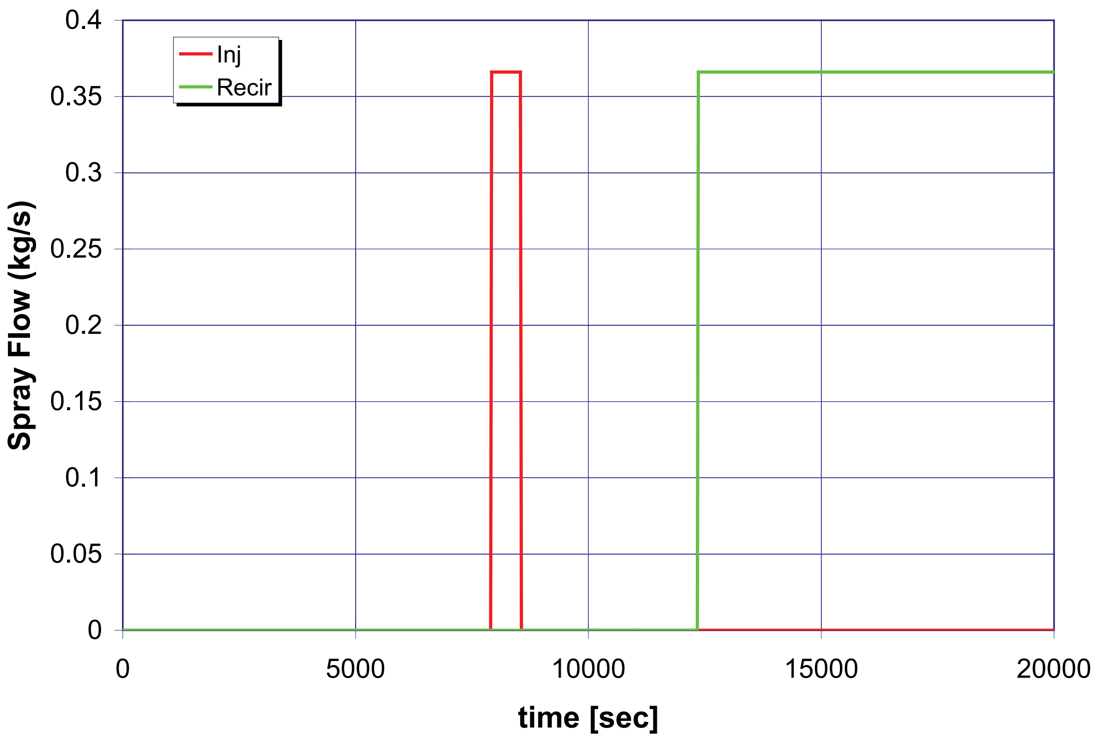
A.2.6 Case 6: 0.5-Inch Break LOCA with Sprays



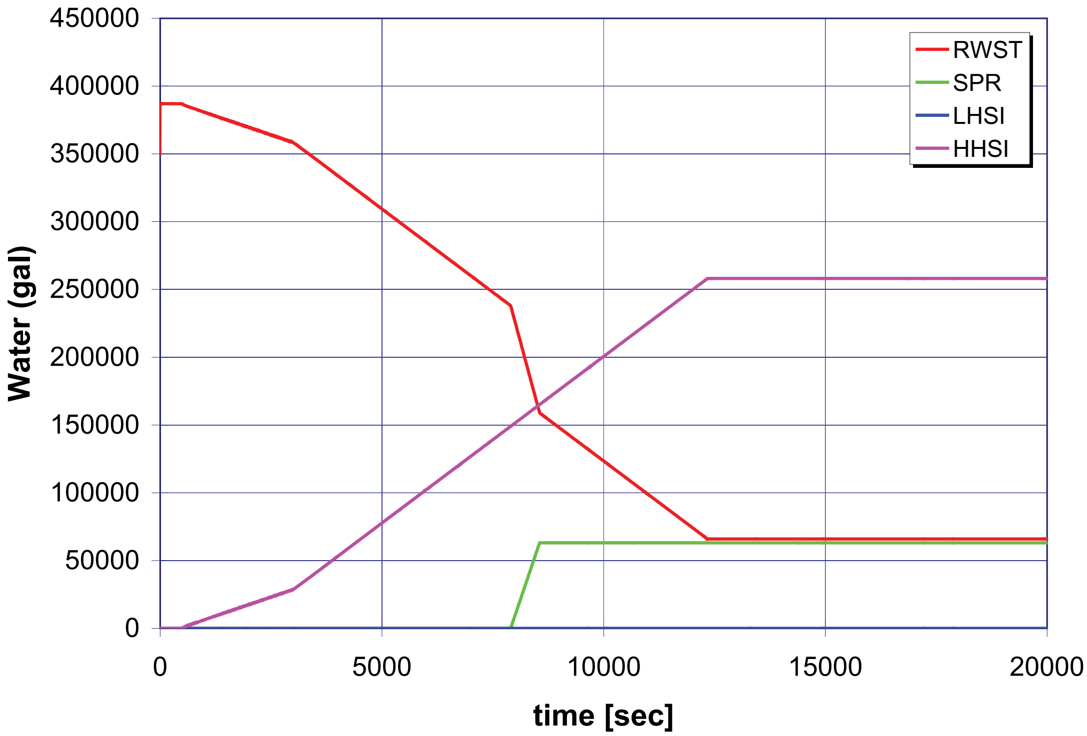




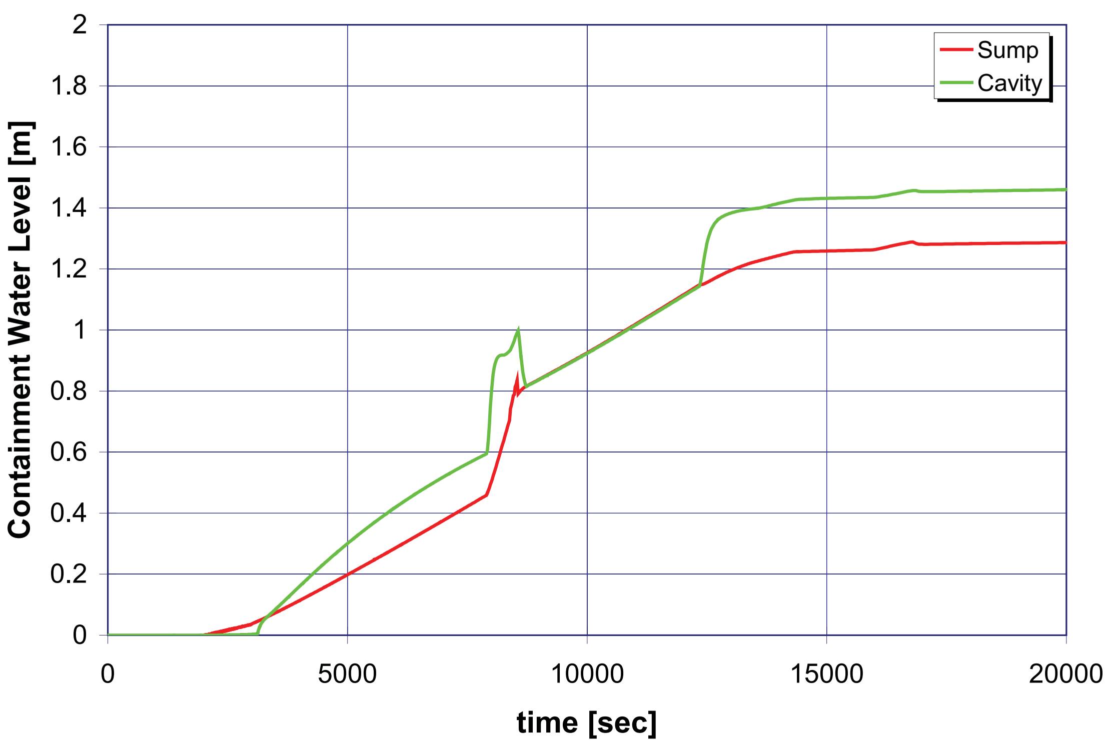
120
100
80
60
40
20
0
Total Flow (kg/s)
0
5000
10000
15000
20000
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK+PORV



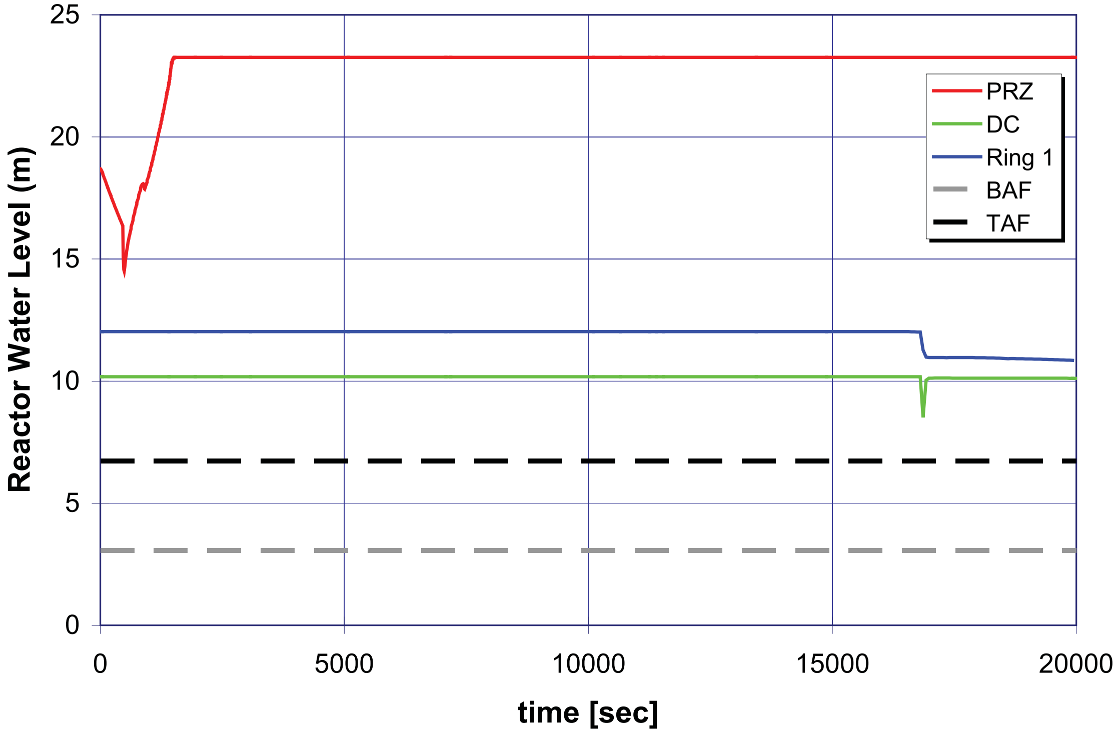
0.4
0.35
0.3
0.25
0.2
0.15
0.1
0.05
0
Spray Flow (kg/s)
0
5000
10000
15000
20000
time [sec]
Inj
Recir



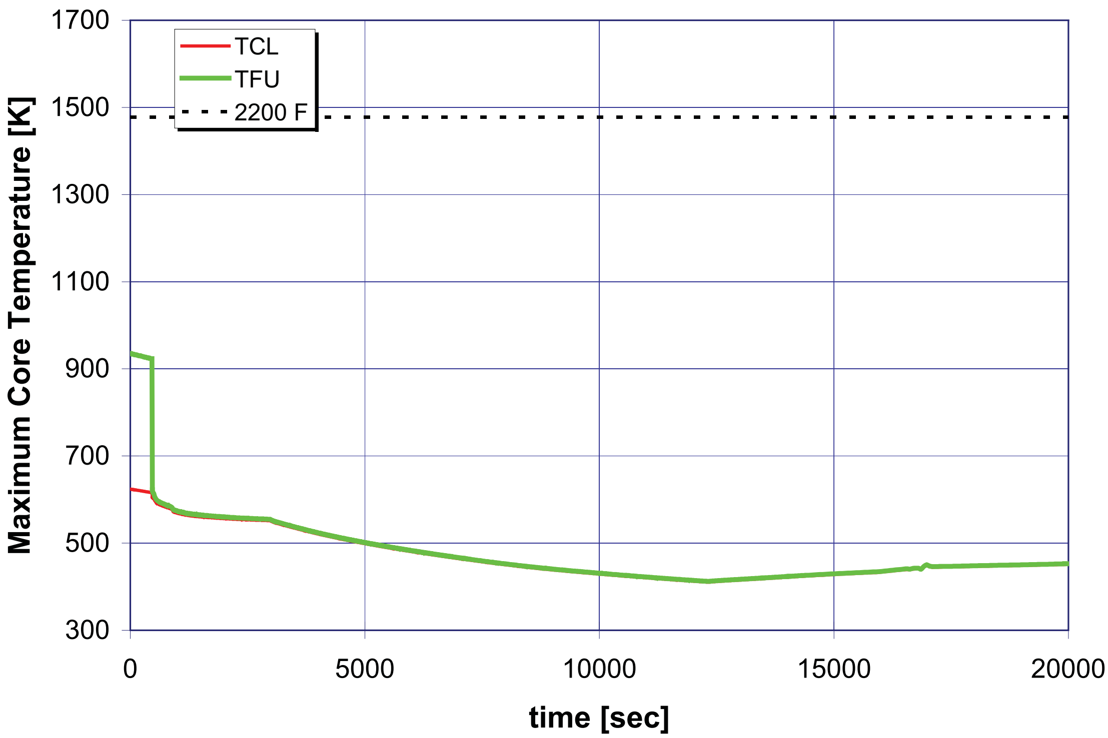
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
5000
10000
15000
20000



Containment Water Level [m]
time [sec]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000

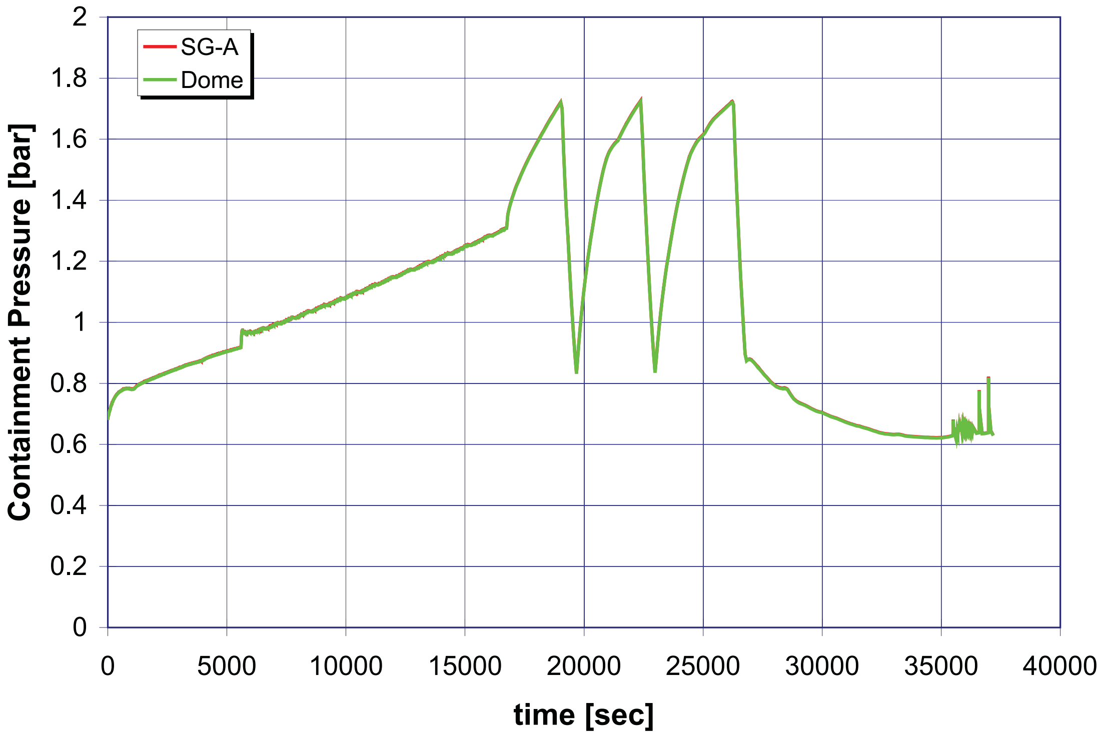


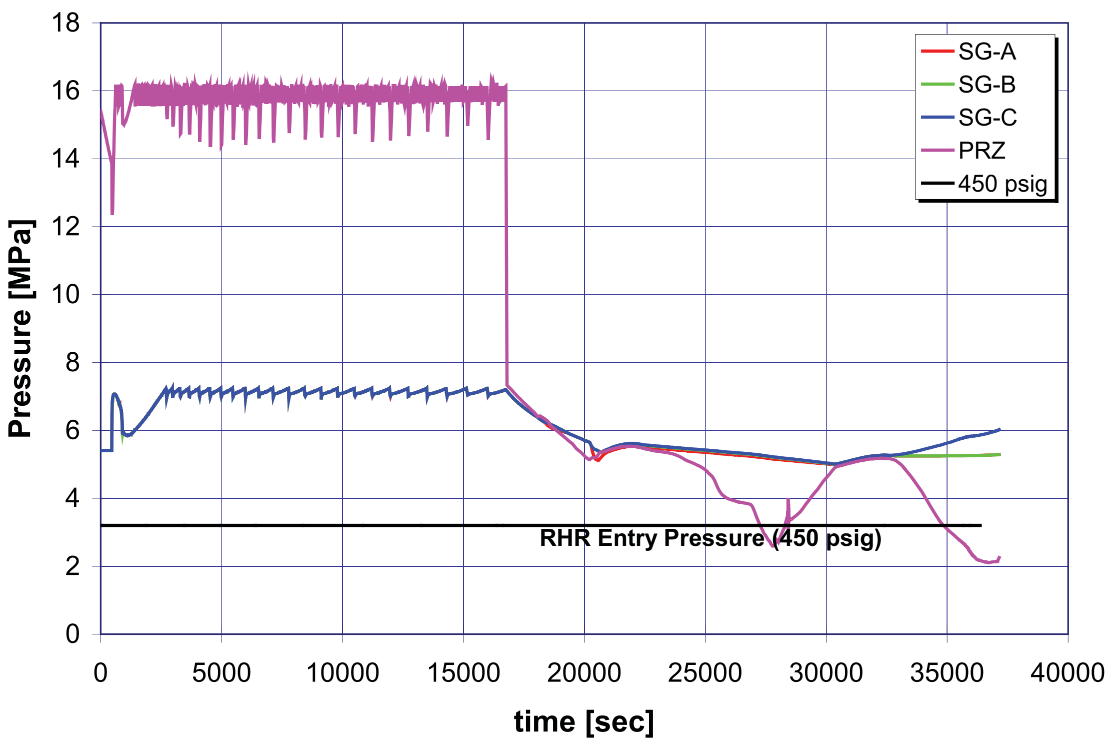
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
5
10
15
20
25
0
5000
10000
15000
20000

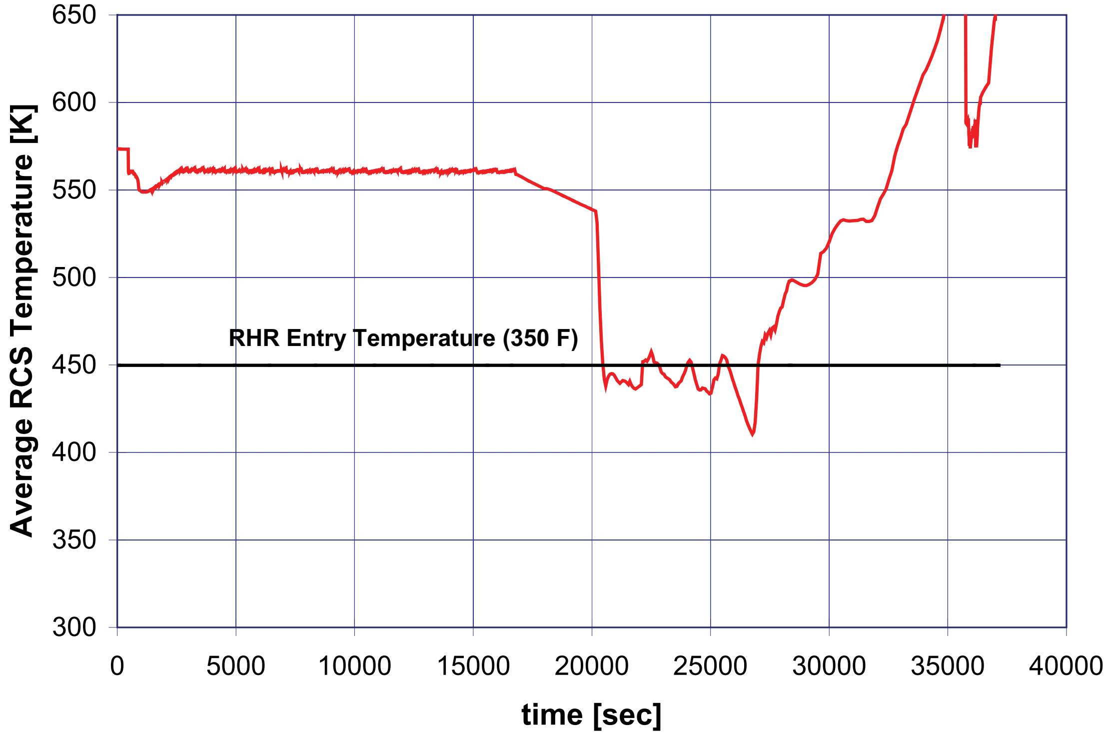


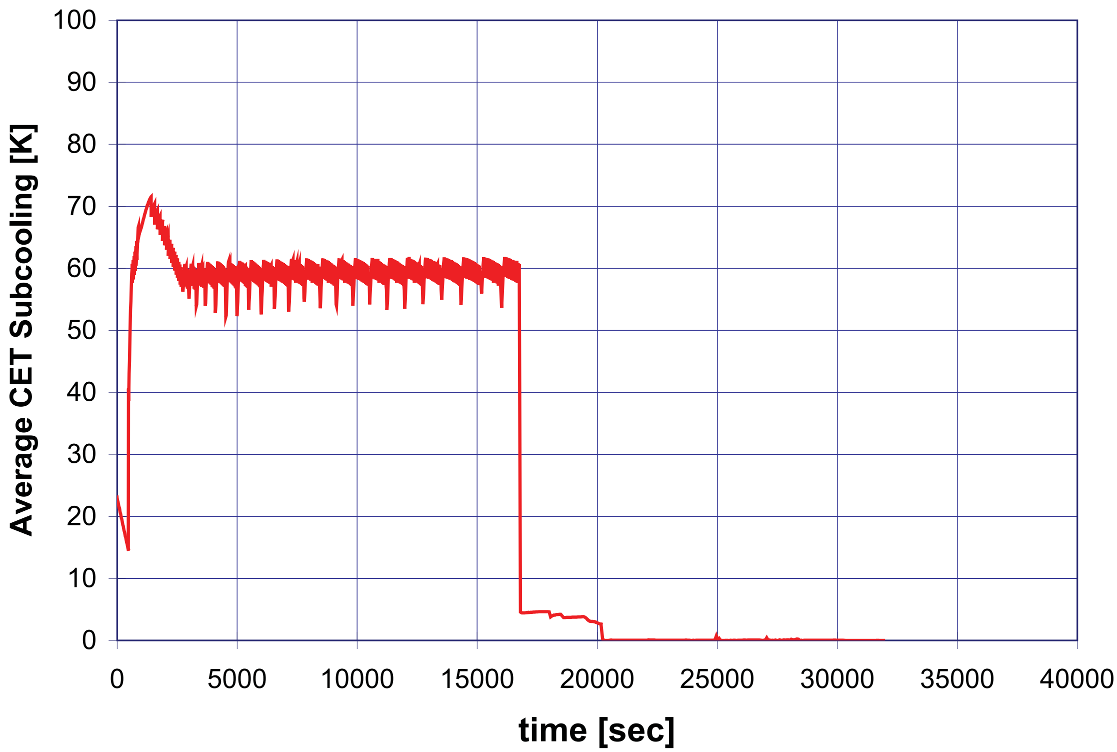
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
500
700
900
1100
1300
1500
1700
0
5000
10000
15000
20000

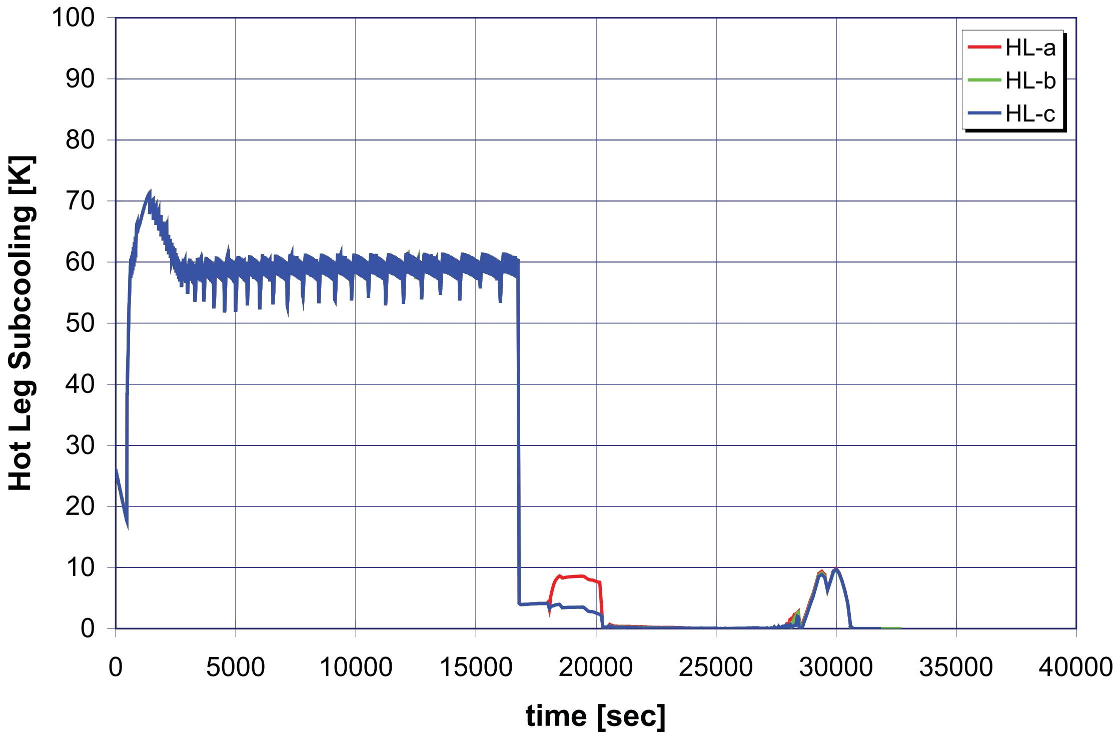
A.2.6.1 Case 6a: 0.5-Inch Break LOCA with Sprays and Secure HHSI Pumps

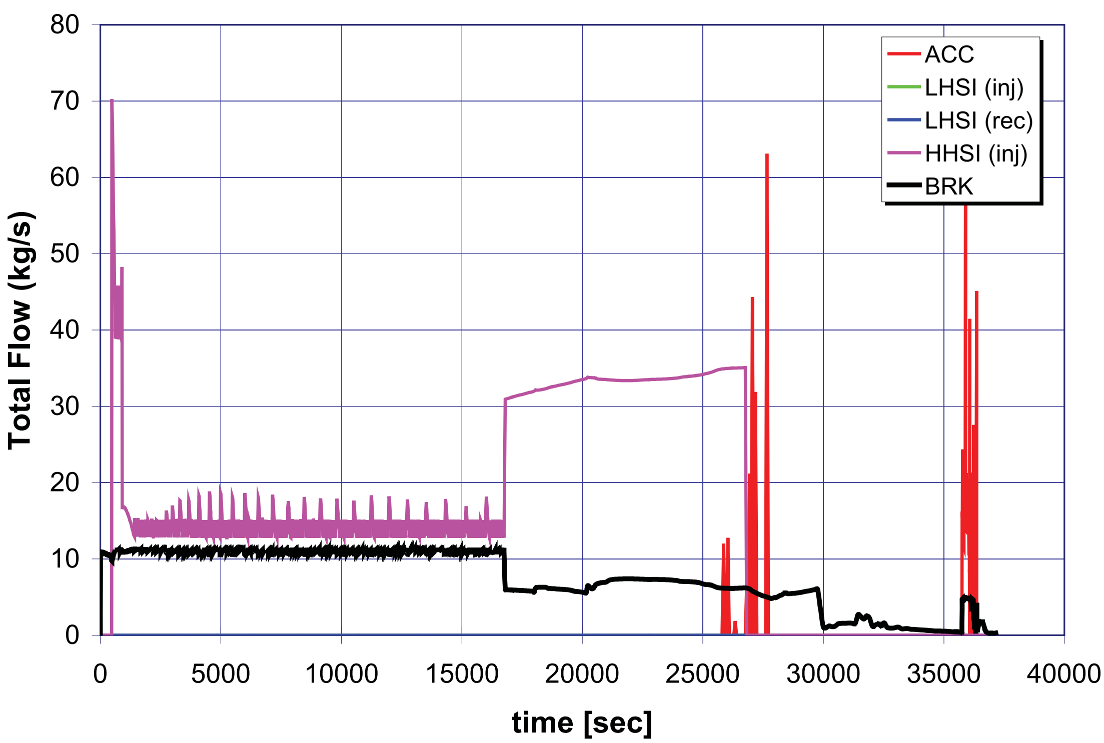


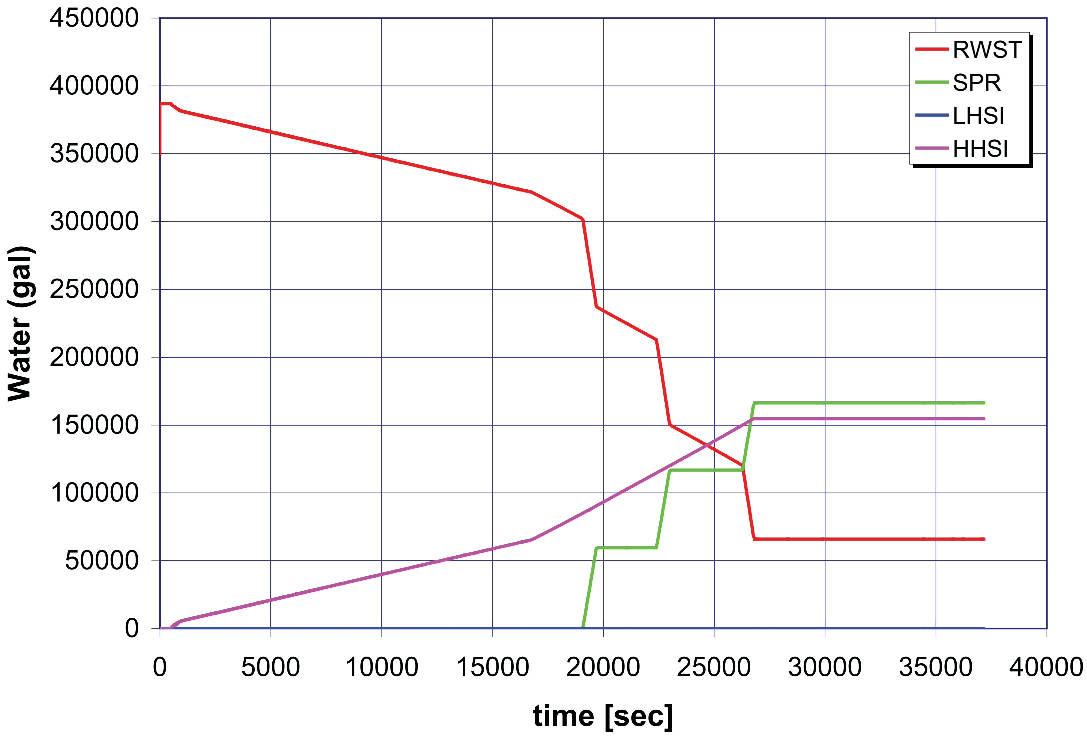



Average RCS Temperature [K]
650
600
550
500
450
400
350
300
RHR Entry Temperature (350 F)
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]

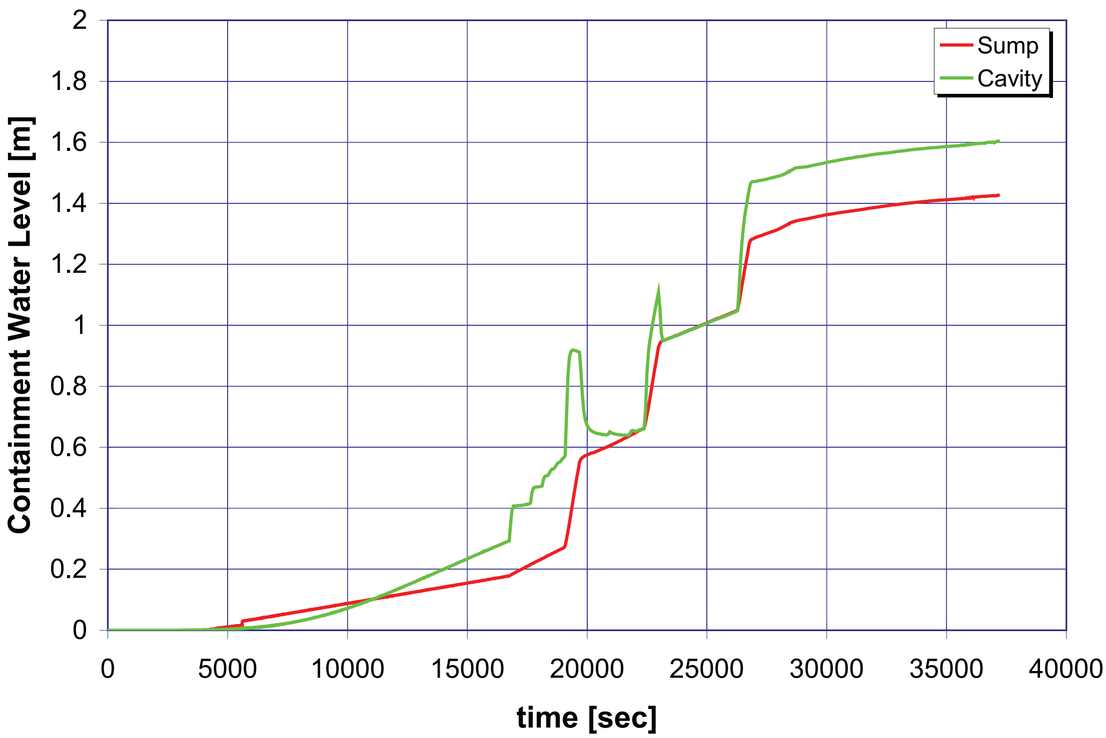

Average CET Subcooling [K]
100
90
80
70
60
50
40
30
20
10
0
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]


Hot Leg Subcooling [K]
time [sec]
HL-a
HL-b
HL-c
0
10
20
30
40
50
60
70
80
90
100
0
5000
10000
15000
20000
25000
30000
35000
40000

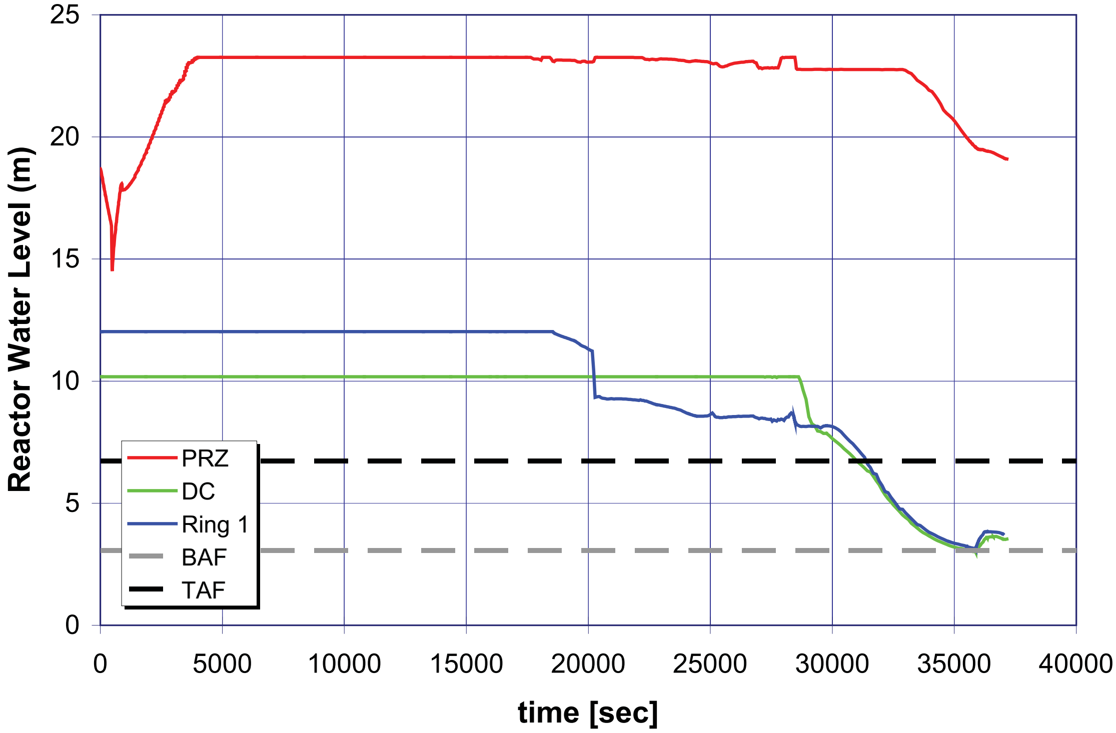

Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
0
10
20
30
40
50
60
70
80
0
5000
10000
15000
20000
25000
30000
35000
40000



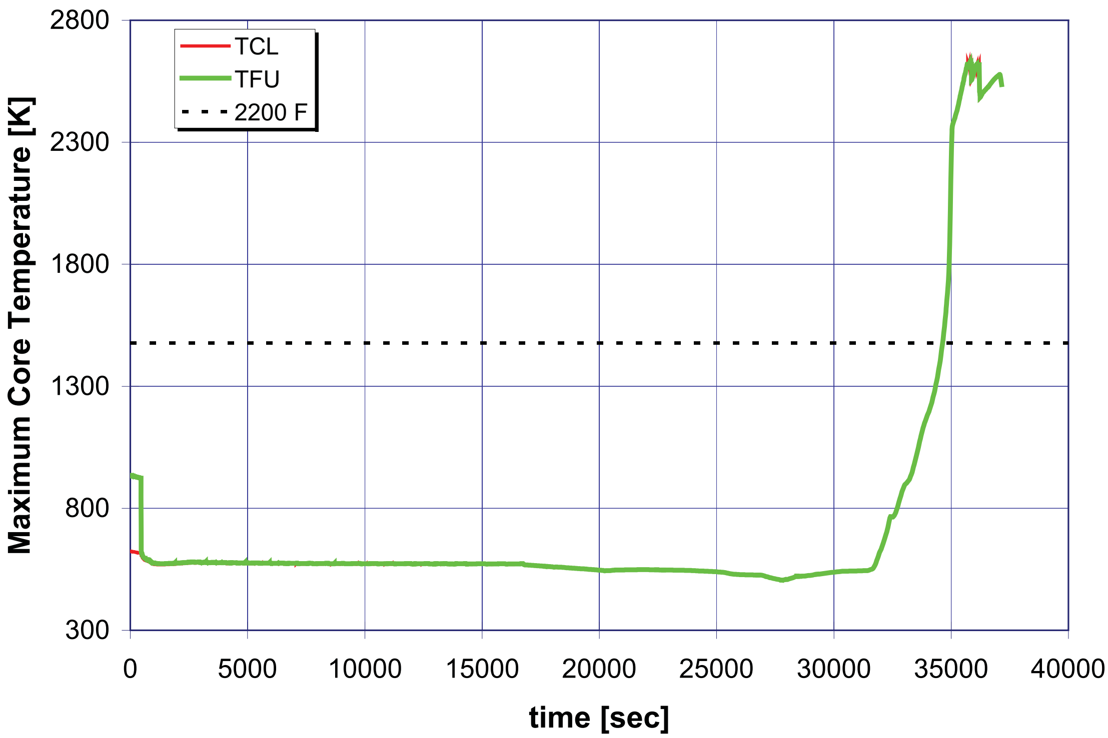
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI



Containment Water Level [m]
time [sec]
Sump
Cavity

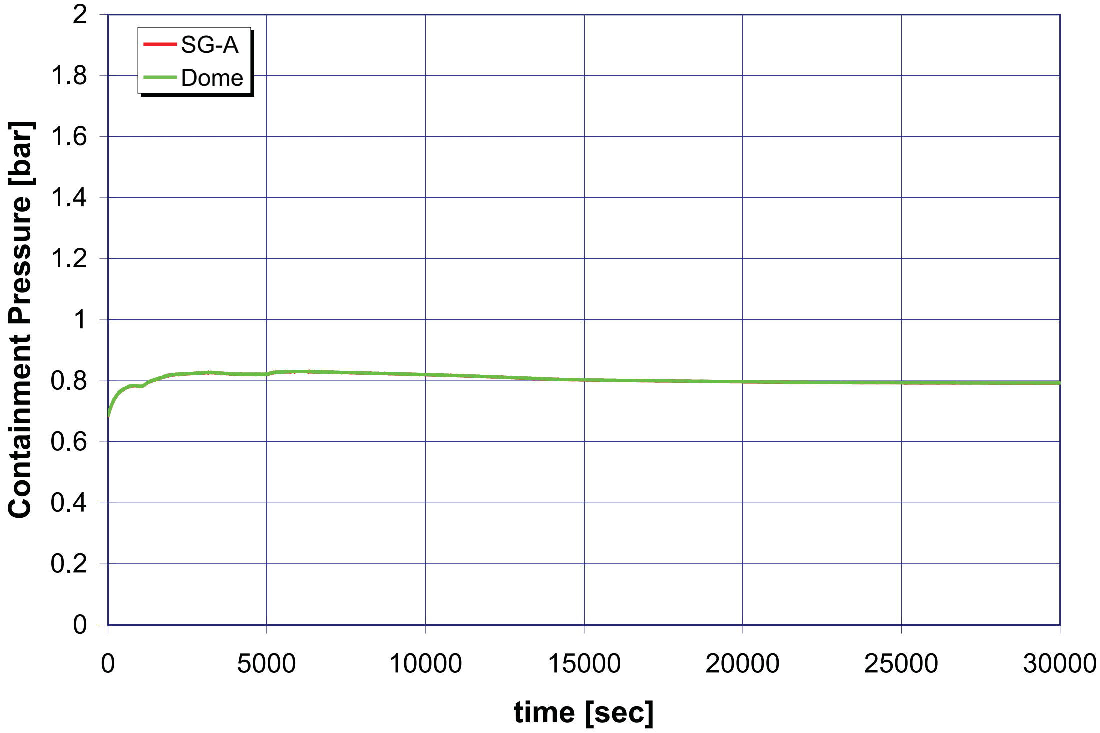


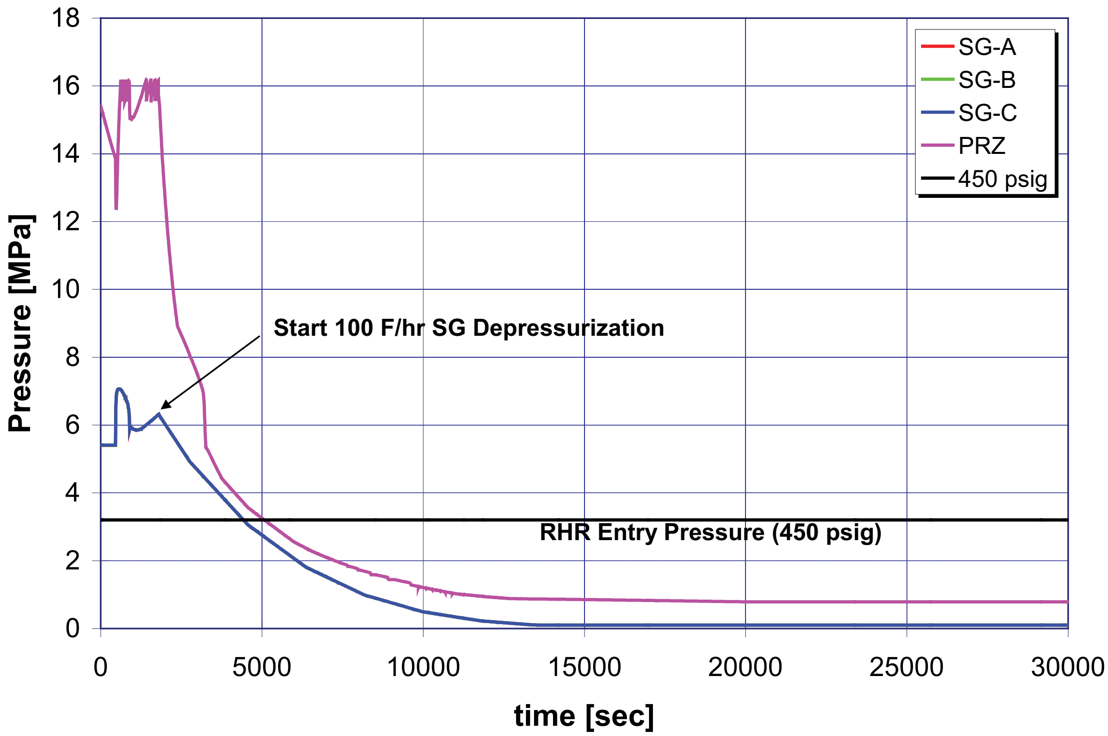
Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
PRZ
DC
Ring 1
BAF
TAF

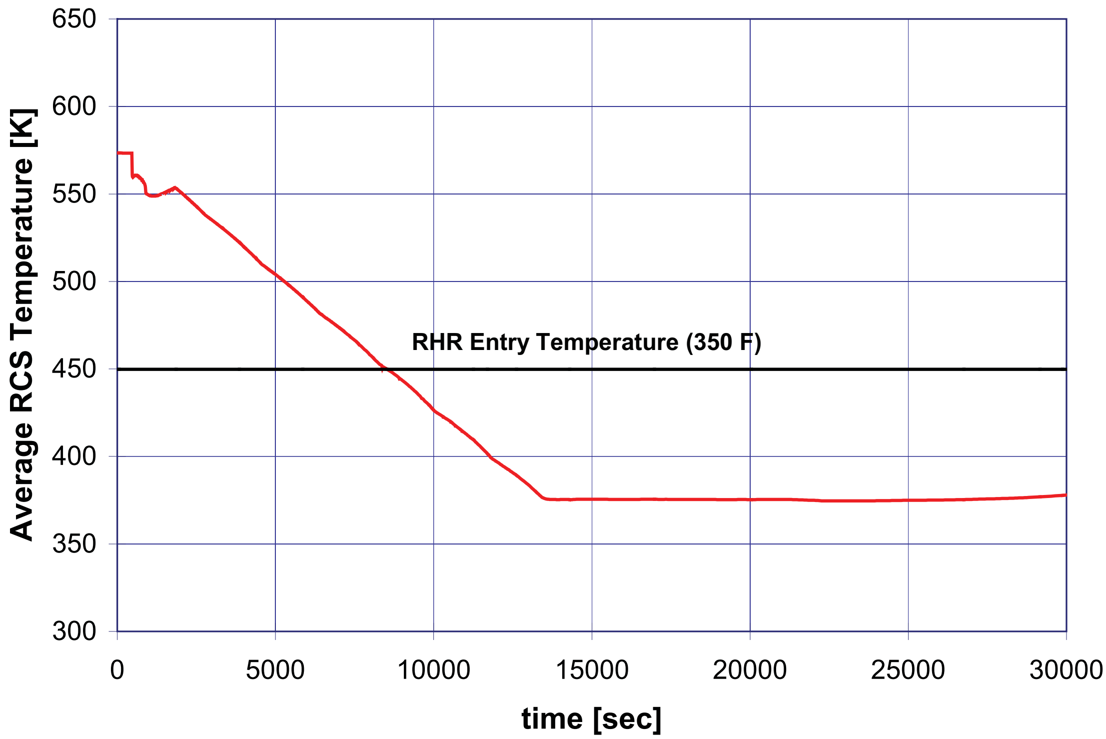


Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
TCL
TFU
2200 F

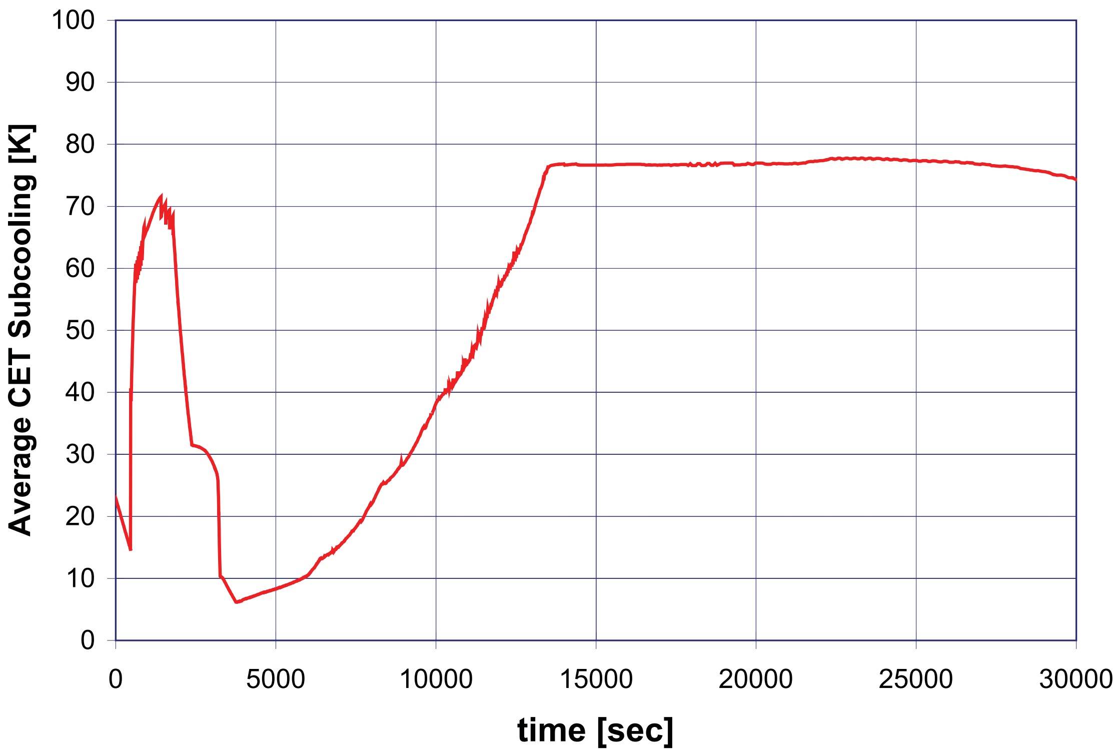
A.2.6.2 Case 6b: 0.5-Inch Break LOCA with Sprays, Secure HHSI Pumps, and Secondary Cooldown



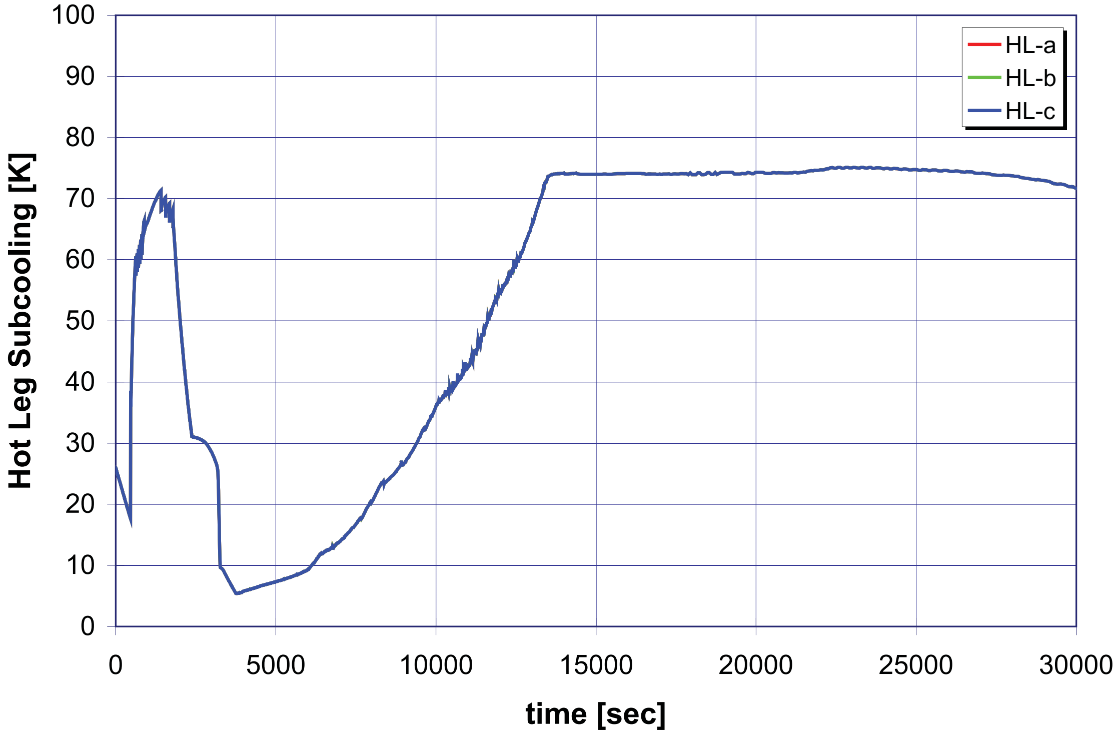


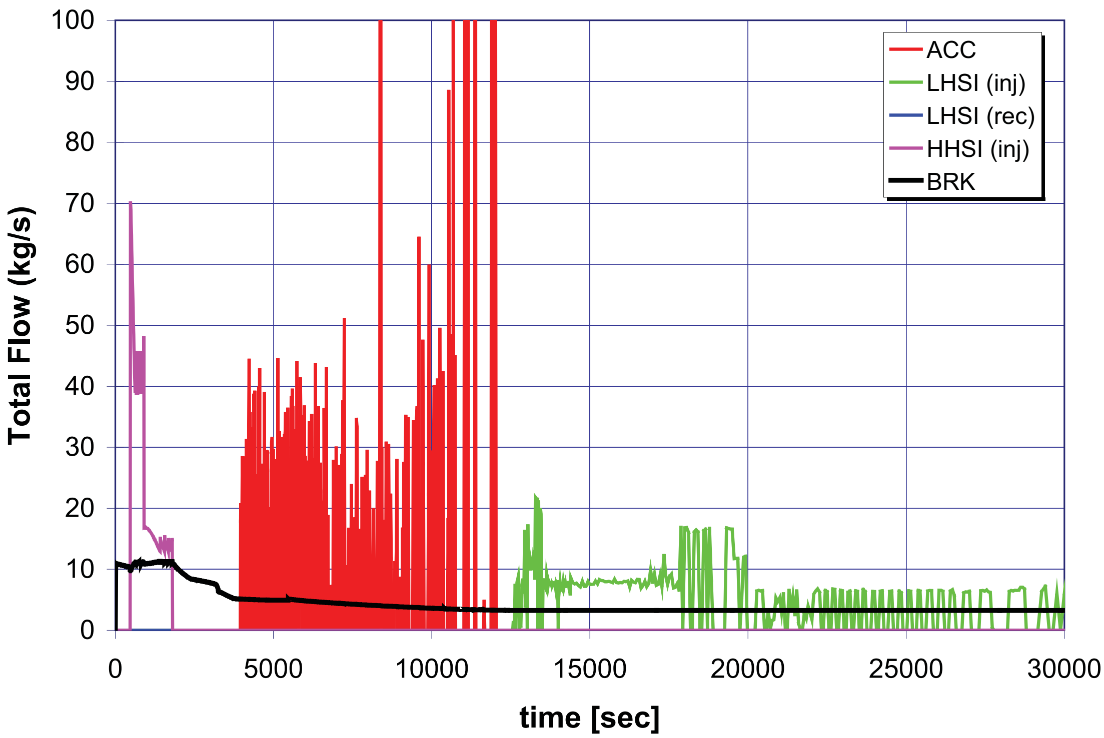


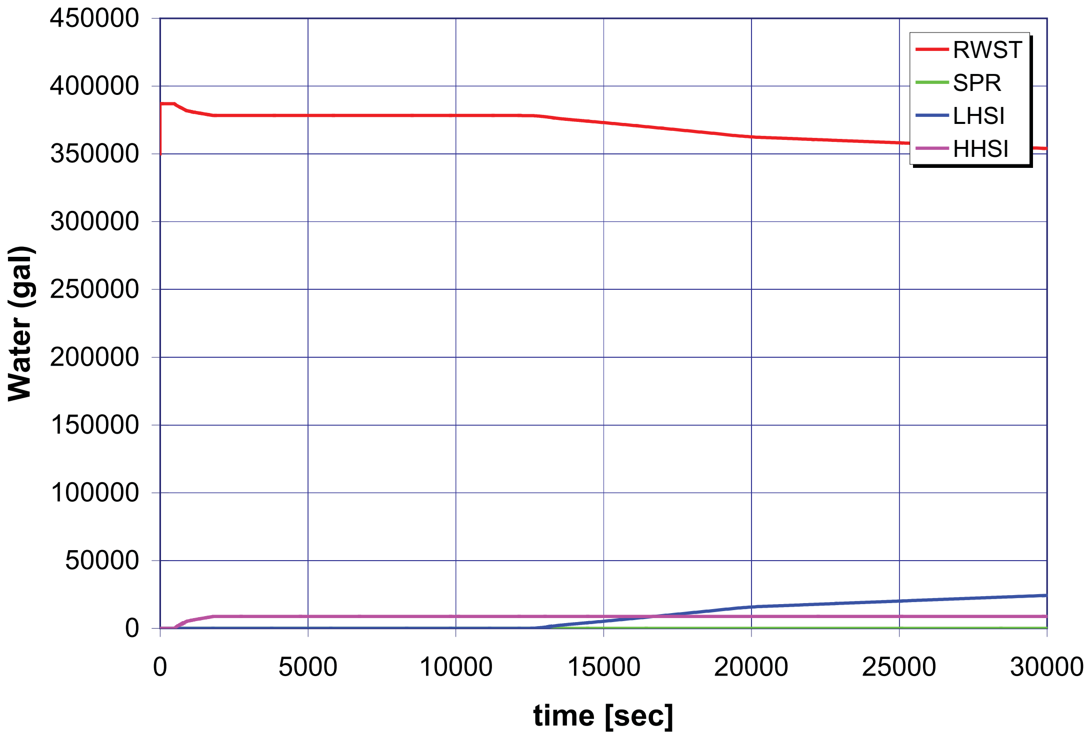
650
600
550
500
450
400
350
300
Average RCS Temperature [K]
RHR Entry Temperature (350 F)
0
5000
10000
15000
20000
25000
30000
time [sec]

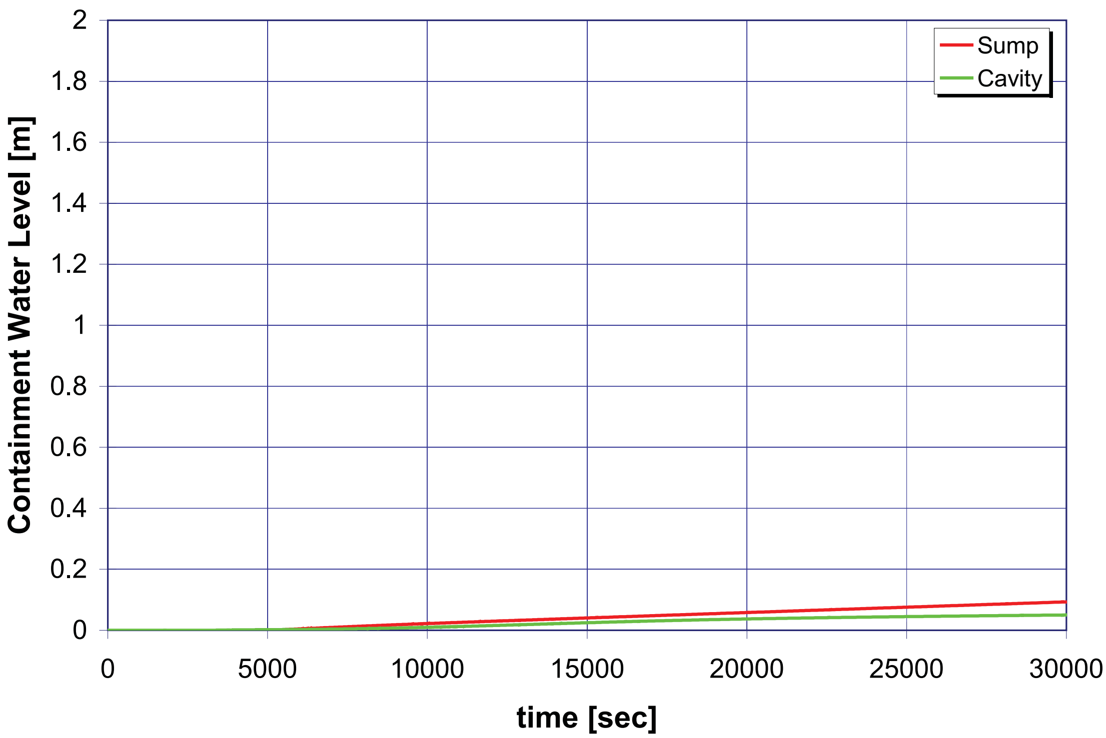


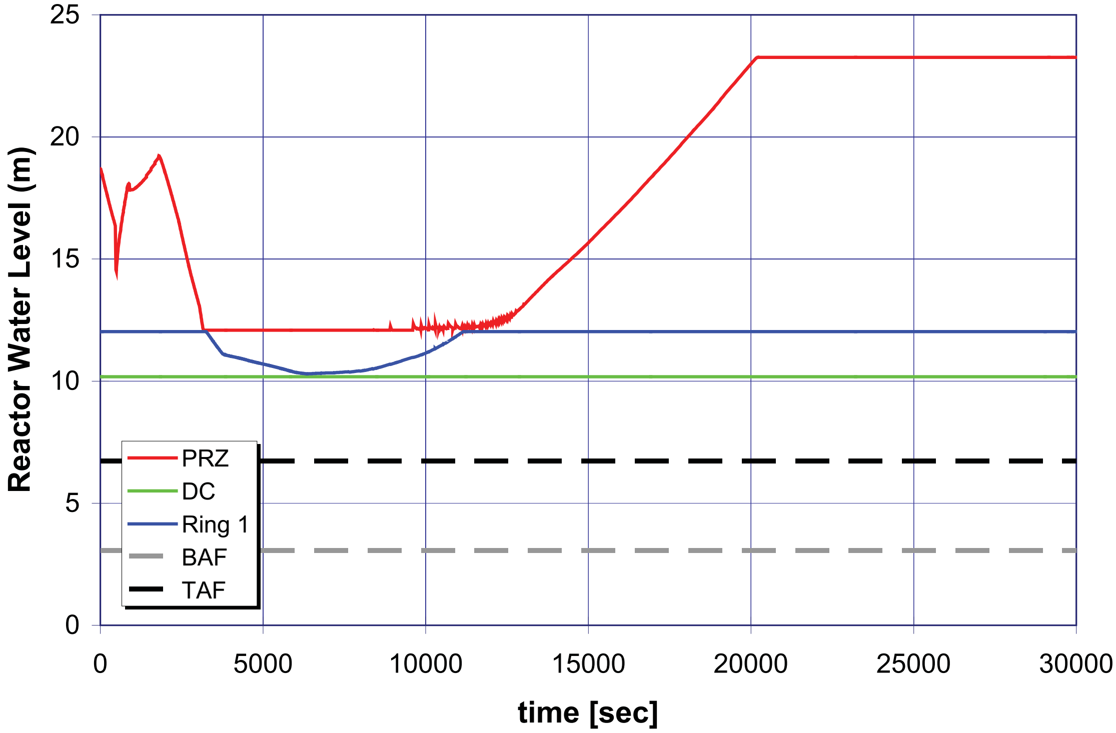
100
90
80
70
60
50
40
30
20
10
0
Average CET Subcooling [K]
0
5000
10000
15000
20000
25000
30000
time [sec]

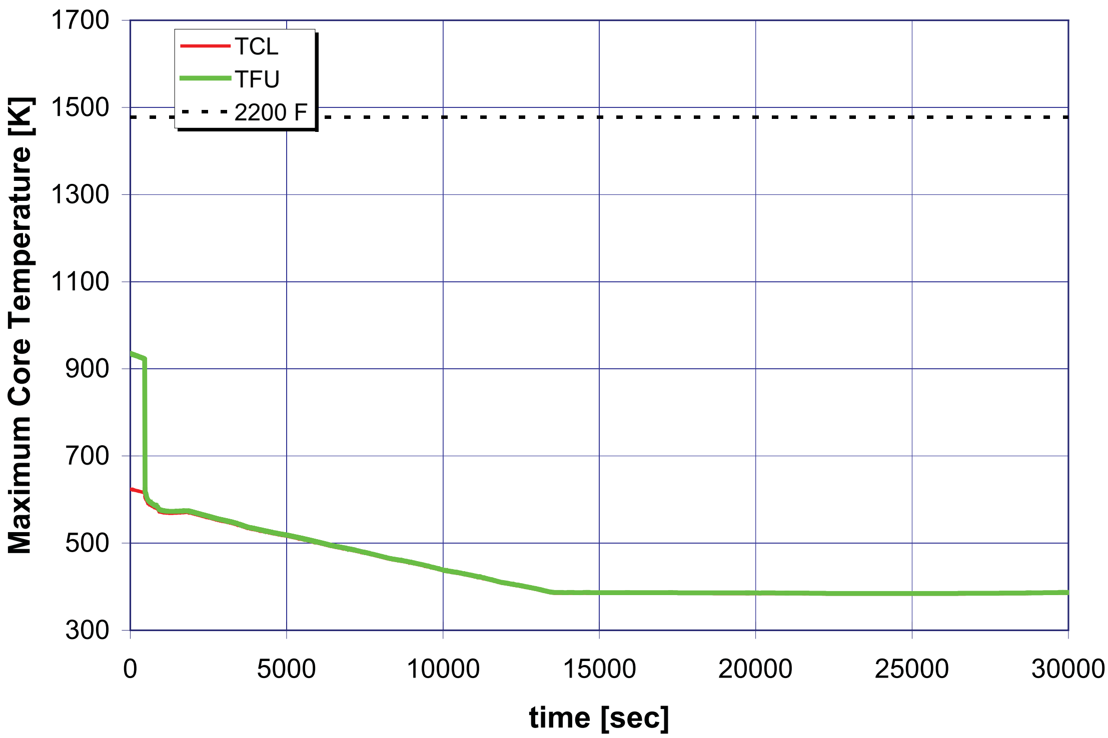

Hot Leg Subcooling [K]
time [sec]
HL-a
HL-b
HL-c
0
10
20
30
40
50
60
70
80
90
100
0
5000
10000
15000
20000
25000
30000


Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
0
10
20
30
40
50
60
70
80
90
100
0
5000
10000
15000
20000
25000
30000

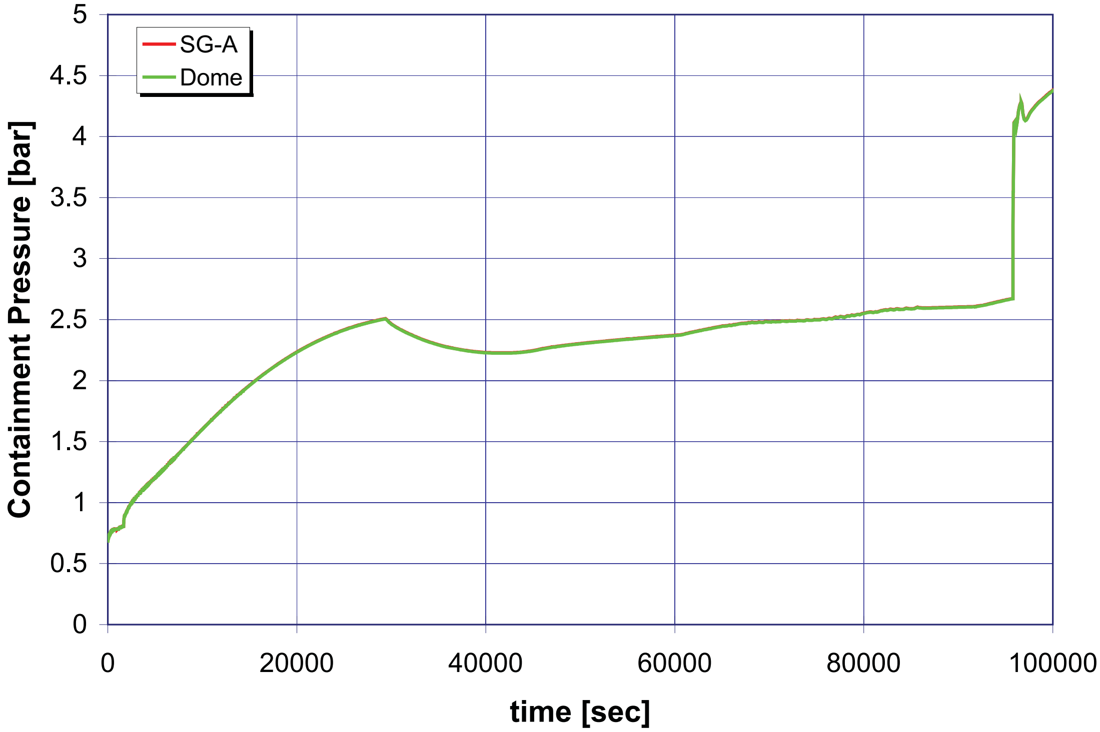

Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI

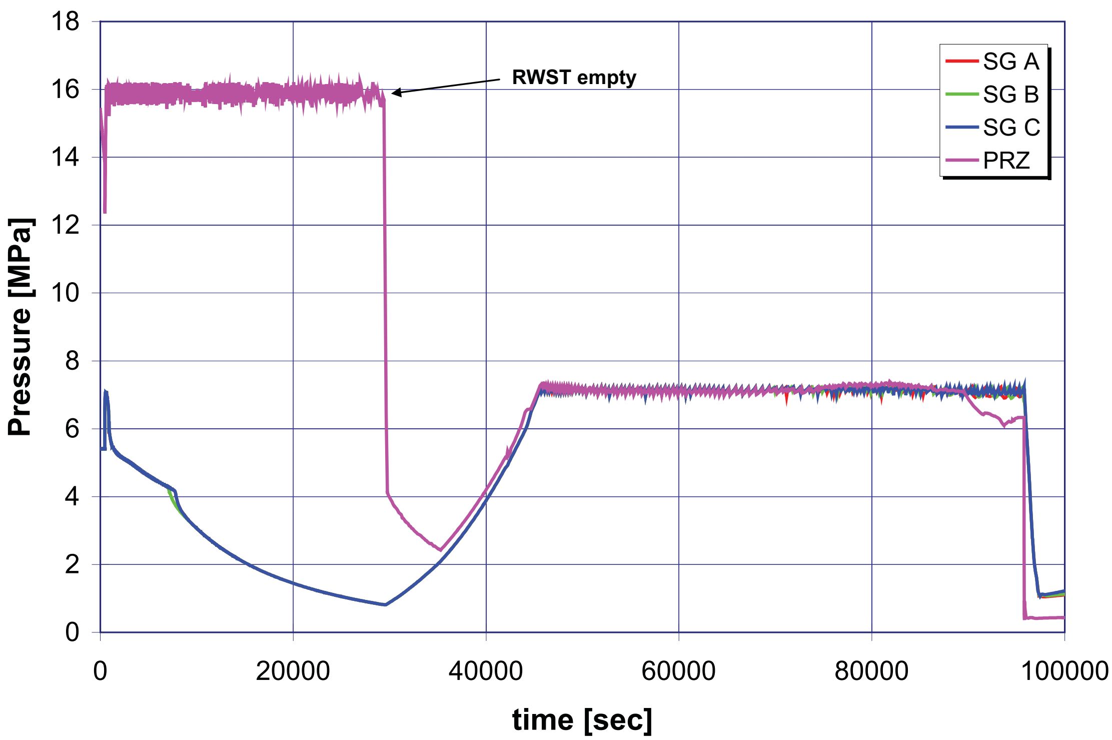

Containment Water Level [m]
time [sec]
Sump
Cavity

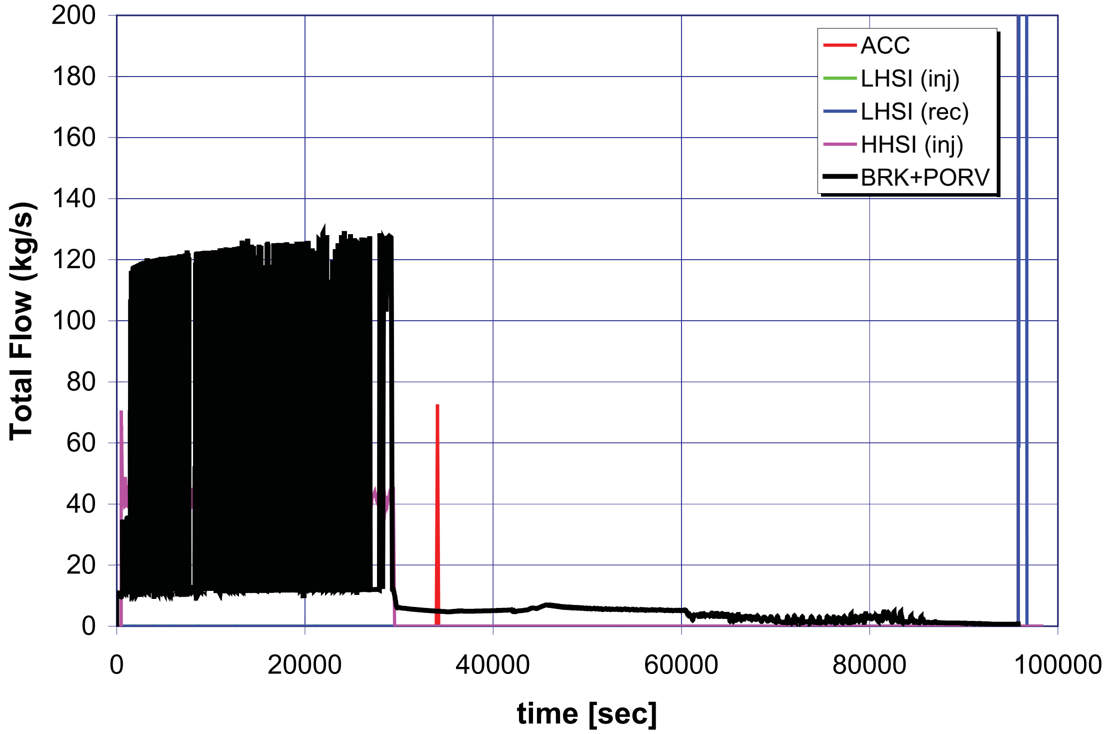

Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
time [sec]
PRZ
DC
Ring 1
BAF
TAF


Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
5000
10000
15000
20000
25000
30000
time [sec]
TCL
TFU
2200 F

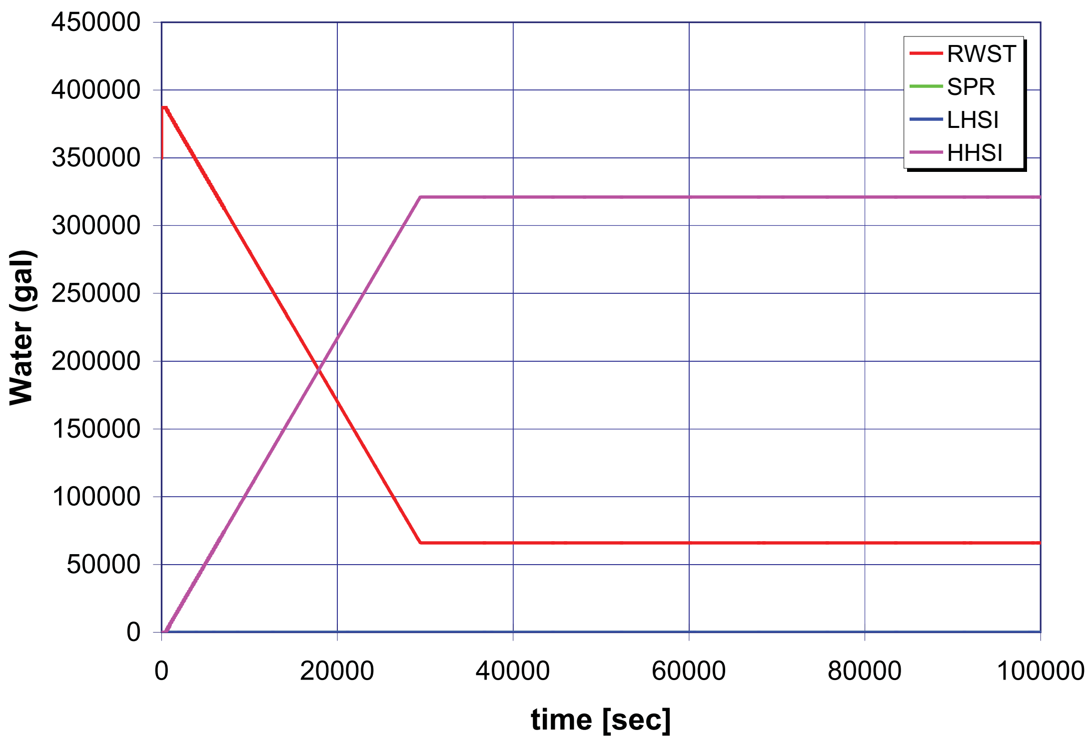
A.2.7 Case 7: 0.5-Inch Break LOCA without Sprays and PRZ PORV Not Stuck Open



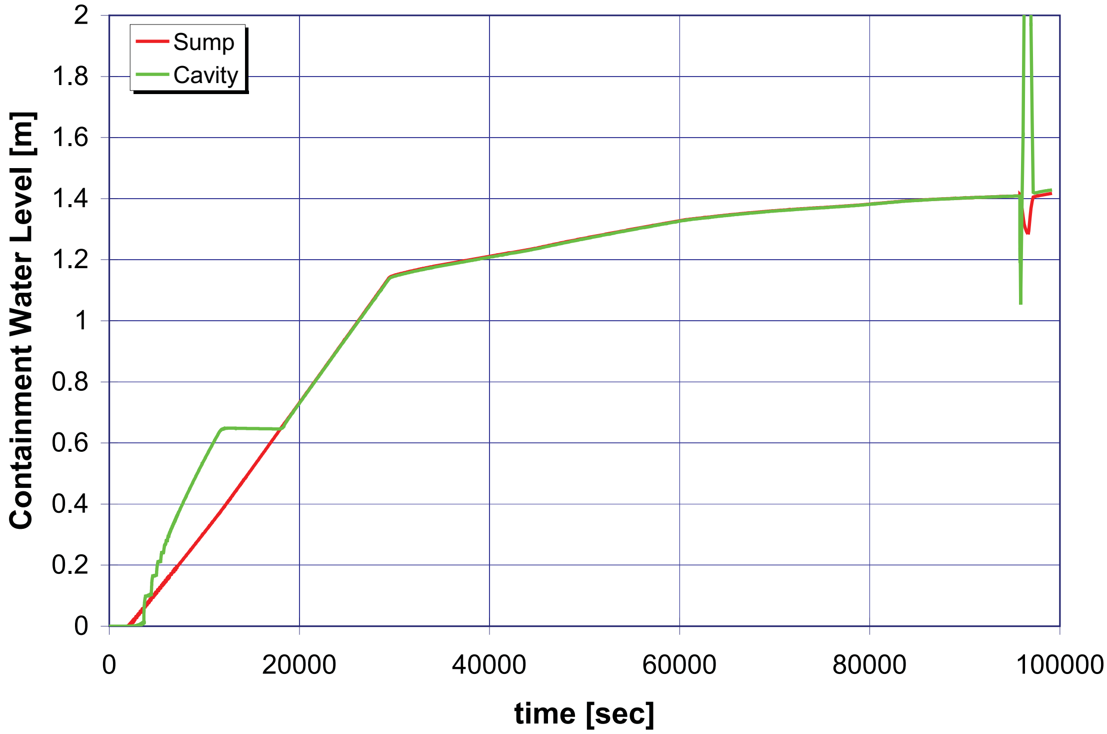


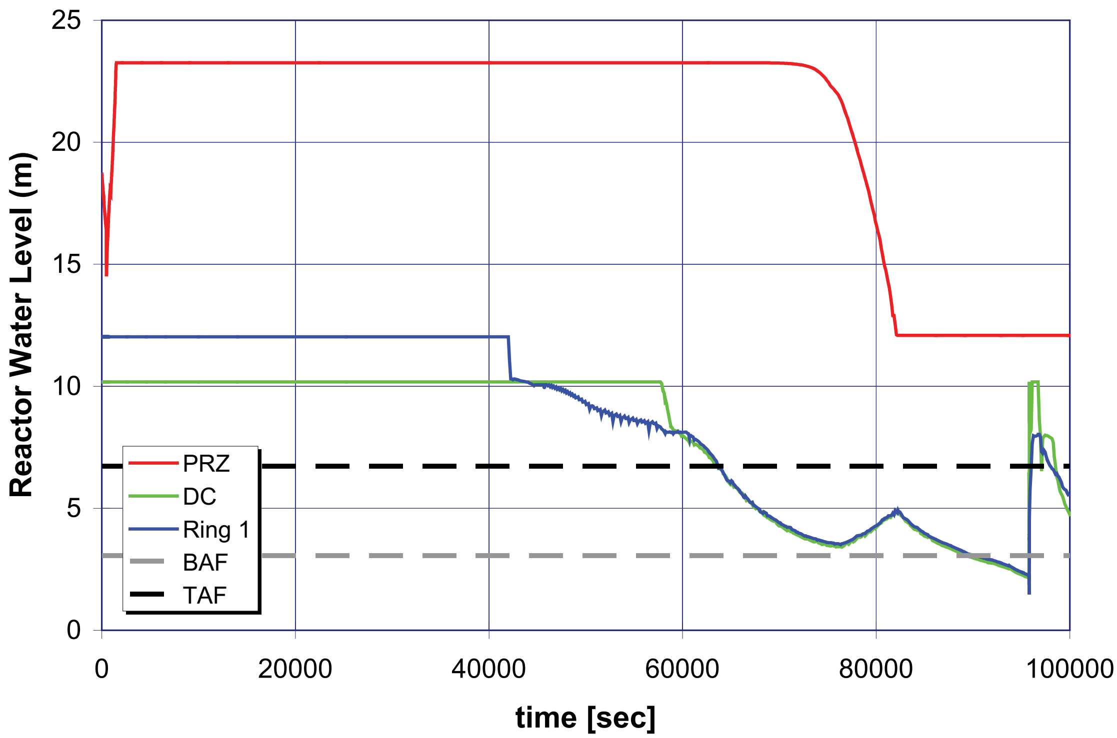


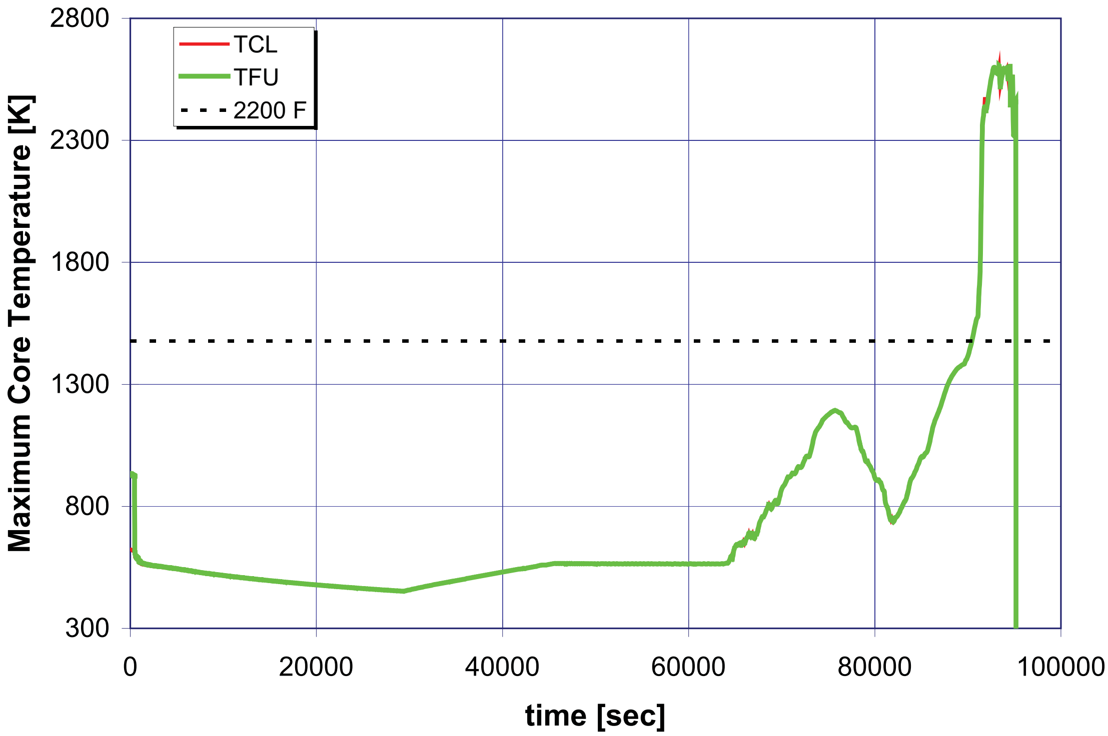
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK+PORV
200
180
160
140
120
100
80
60
40
20
0
0
20000
40000
60000
80000
100000



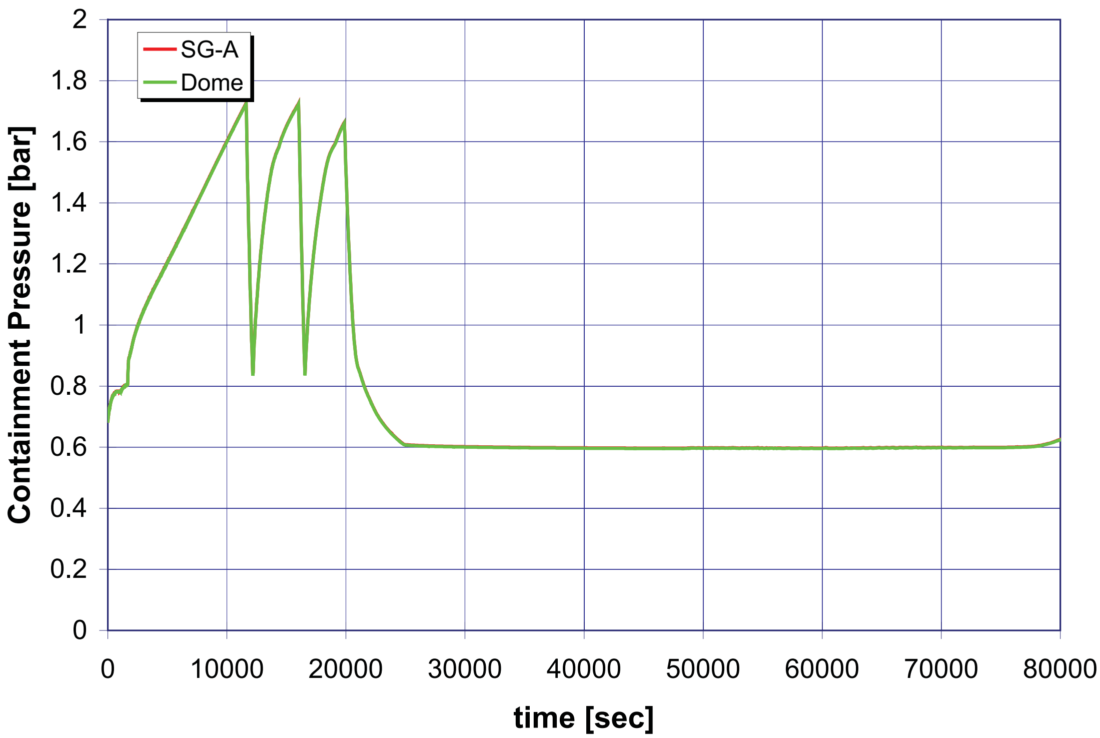
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
20000
40000
60000
80000
100000

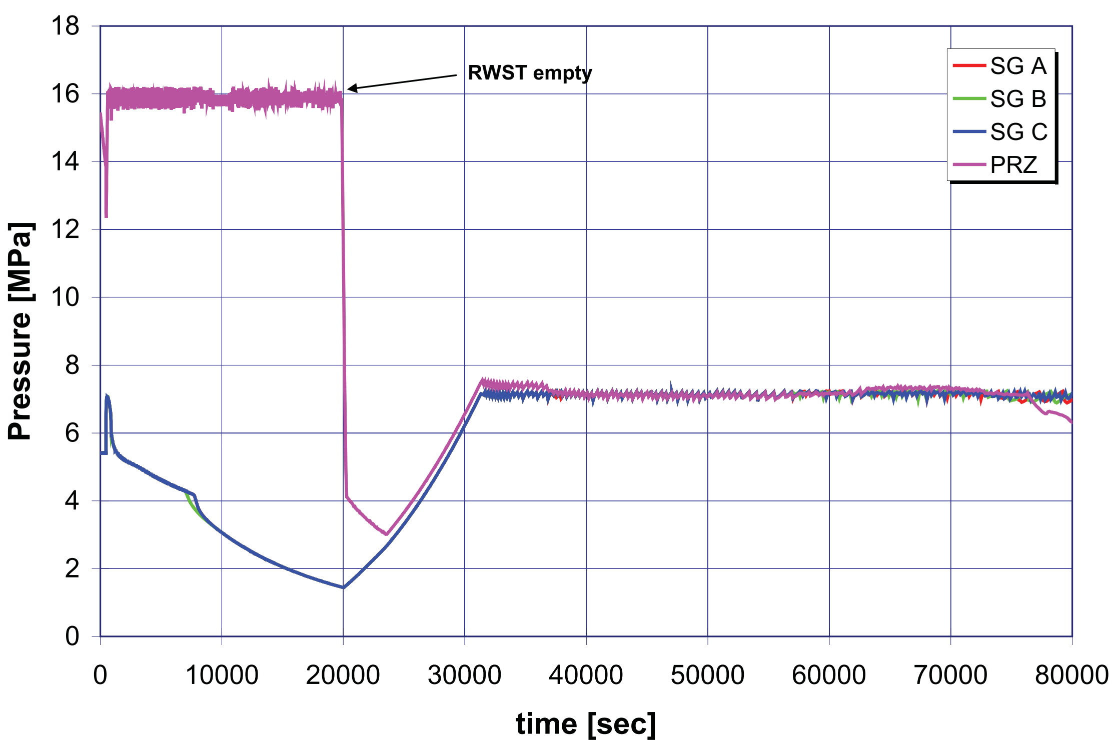

Sump
Cavity
Containment Water Level [m]
time [sec]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
20000
40000
60000
80000
100000

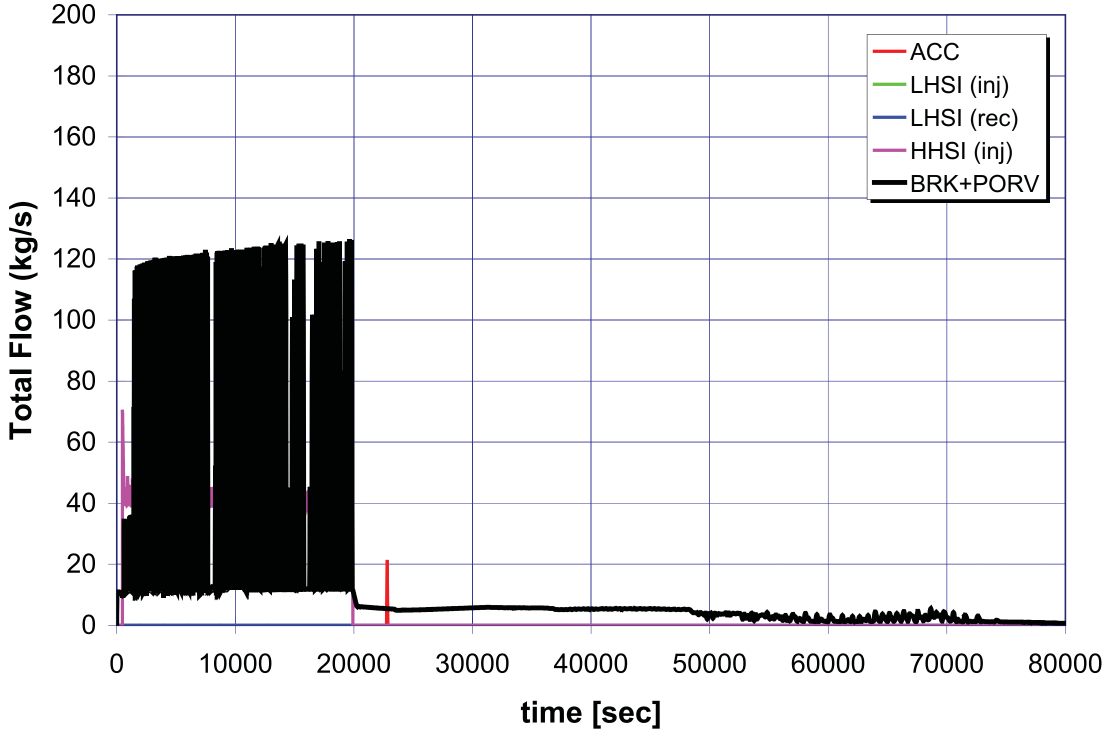

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
25
20
15
10
5
0
20000
40000
60000
80000
100000


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
20000
40000
60000
80000
100000
time [sec]

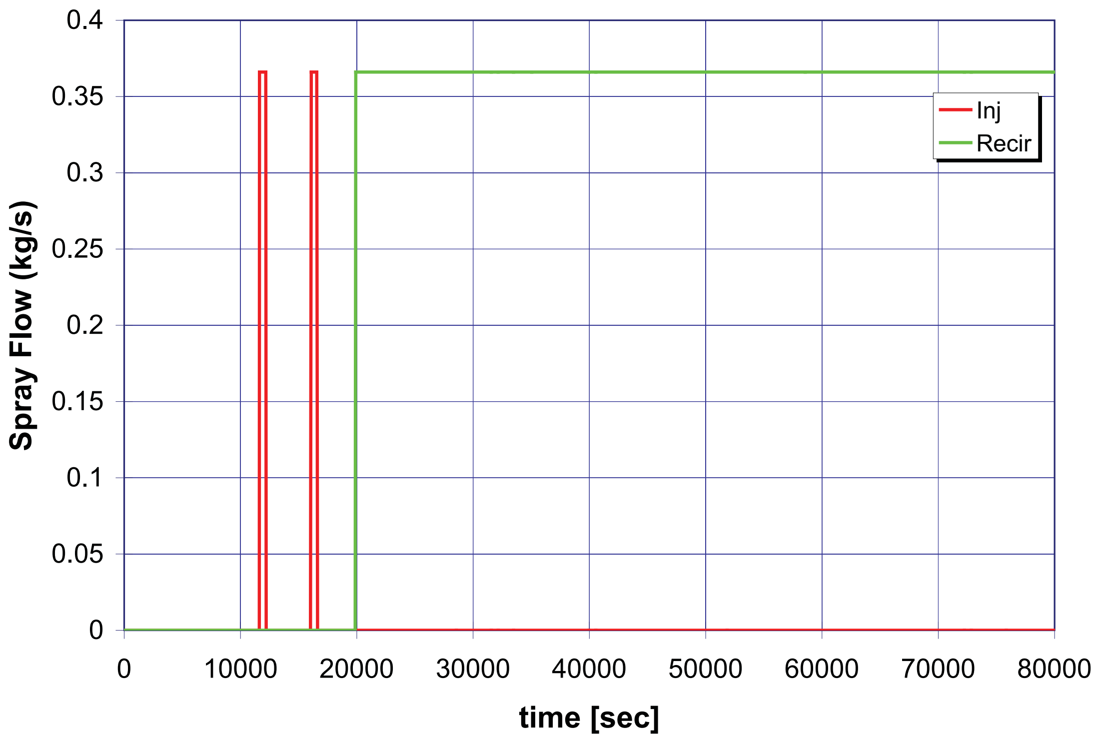
A.2.8 Case 8: 0.5-Inch Break LOCA with Sprays and PRZ PORV Not Stuck Open



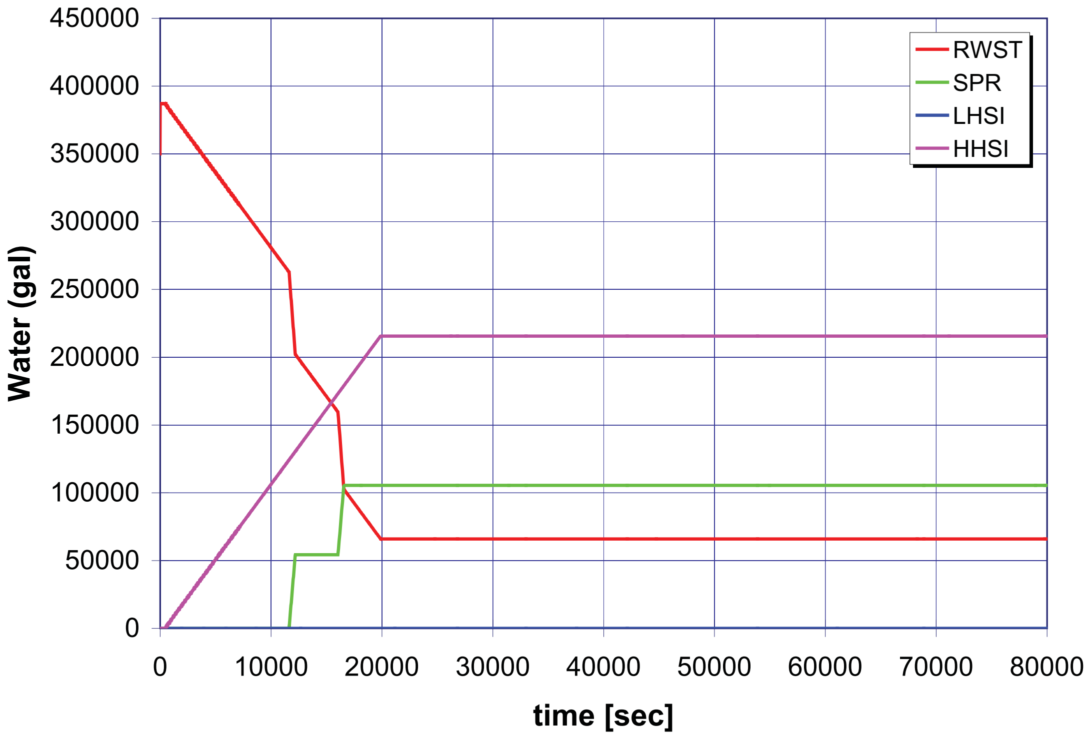


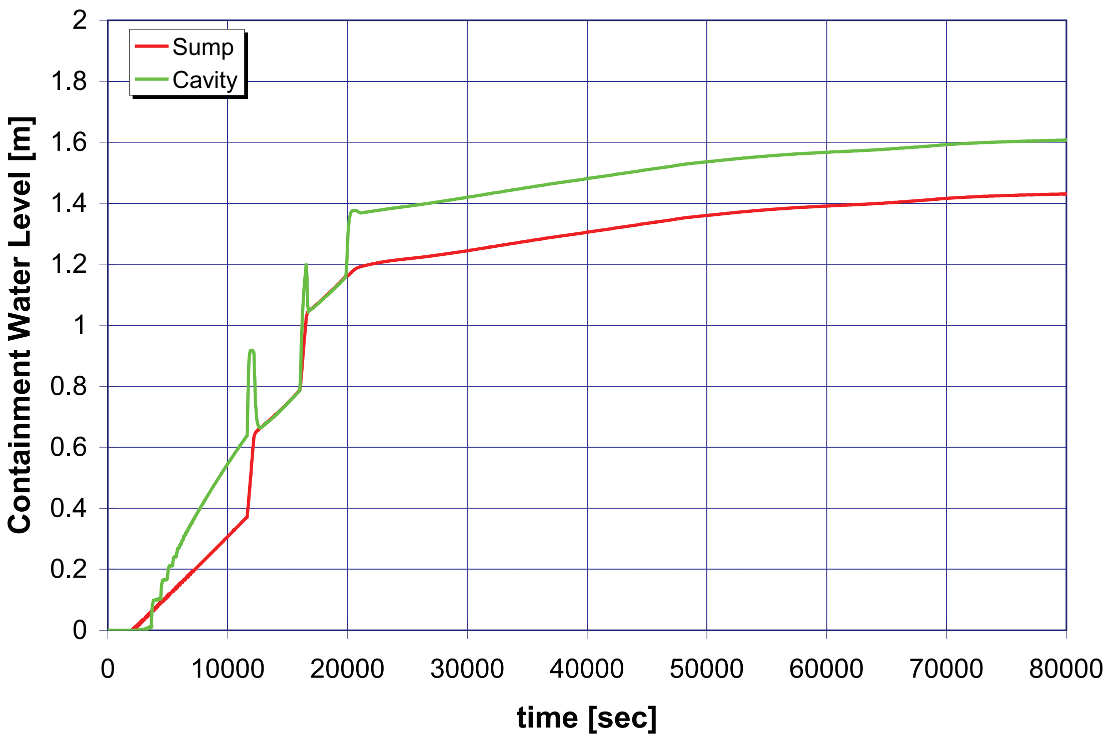


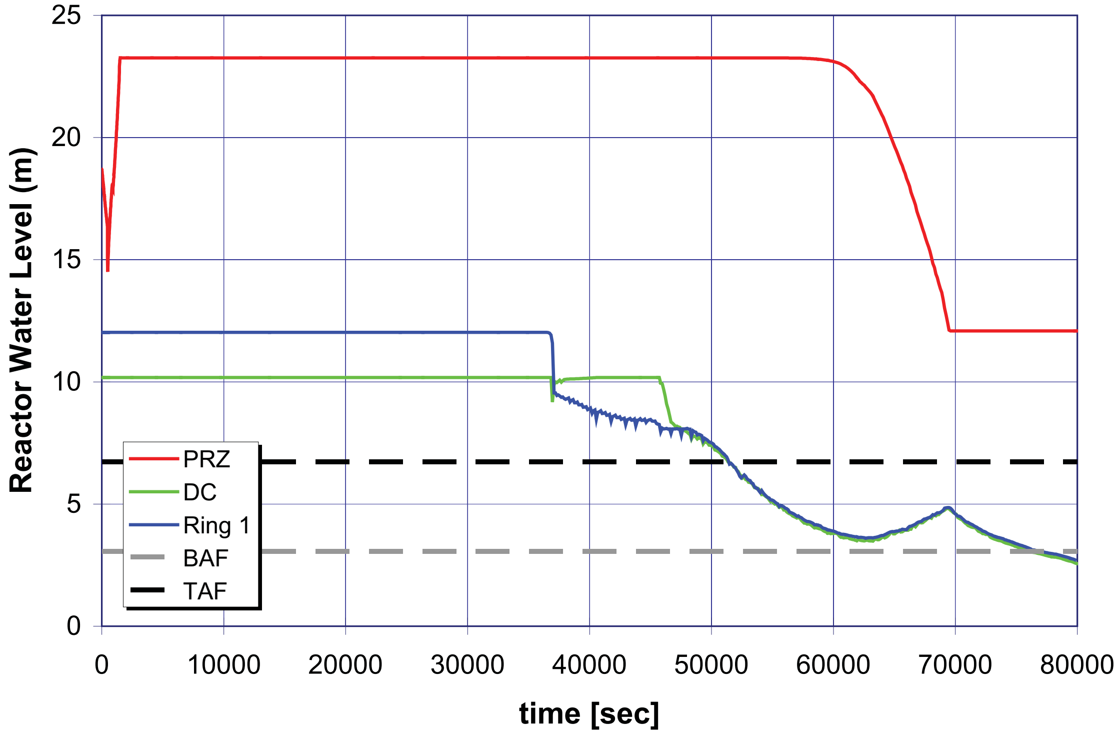
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK+PORV
Total Flow (kg/s)
time [sec]

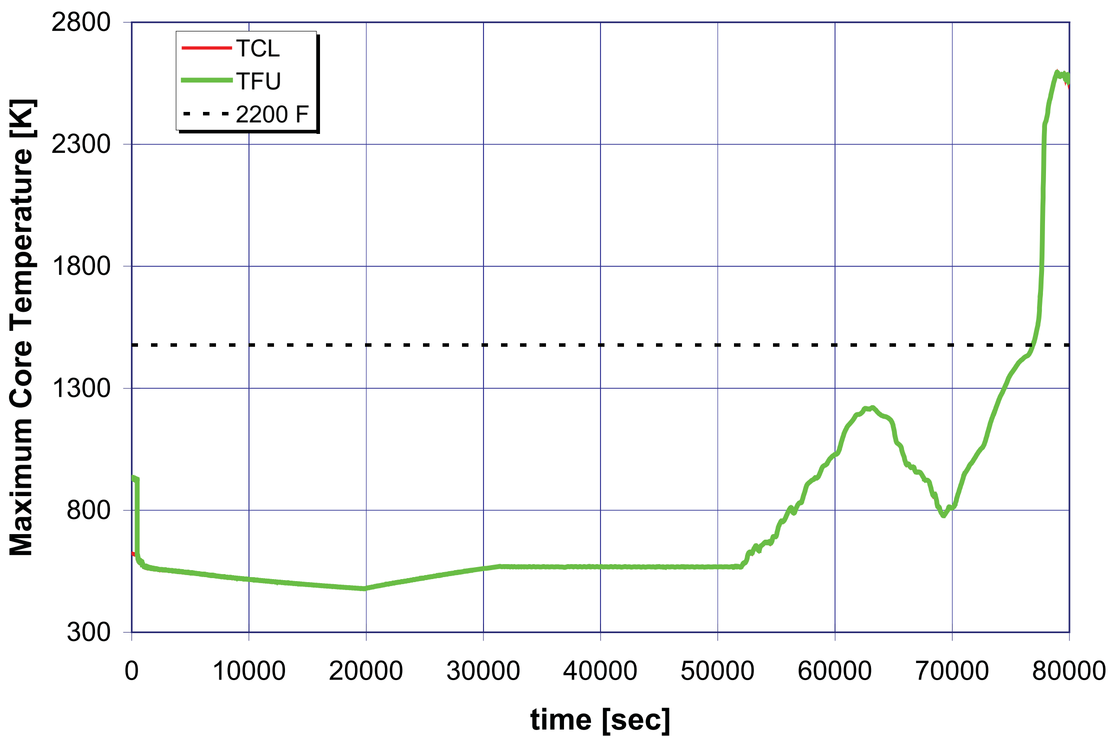


Inj
Recir
Spray Flow (kg/s)
time [sec]


RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]


Sump
Cavity
Containment Water Level [m]
time [sec]


Reactor Water Level (m)
25
20
15
10
5
0
PRZ
DC
Ring 1
BAF
TAF
0
10000
20000
30000
40000
50000
60000
70000
80000
time [sec]


Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
TCL
TFU
2200 F
0
10000
20000
30000
40000
50000
60000
70000
80000
time [sec]

A.3 Feed-and-Bleed PORV Success Criteria

Analysis Summary

Table 8 and Table 9 provide results for this portion of the analysis.

Table 8 Surry Feed-and-Bleed PORV Success Criteria Results

<table>
<tr><th>Case</th><th>Power Level[1]</th><th>Cause of Reactor Trip[2]</th><th>Cause of SI</th><th># HHSI Pumps</th><th># of Pressurizer PORVs</th><th>Core Uncovery (hr)</th><th>Core Damage</th></tr>
<tr><td>1</td><td rowspan="2">100%</td><td>MFW trip</td><td rowspan="3">High Cont. Press.</td><td rowspan="3">1</td><td rowspan="3">1</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>2</td><td>Low SG level + feed/steam mismatch</td><td>1.65</td><td>No[3]</td></tr>
<tr><td>3</td><td>113.9%</td><td>Low-low SG level</td><td>1.60</td><td>No[3]</td></tr>
</table>

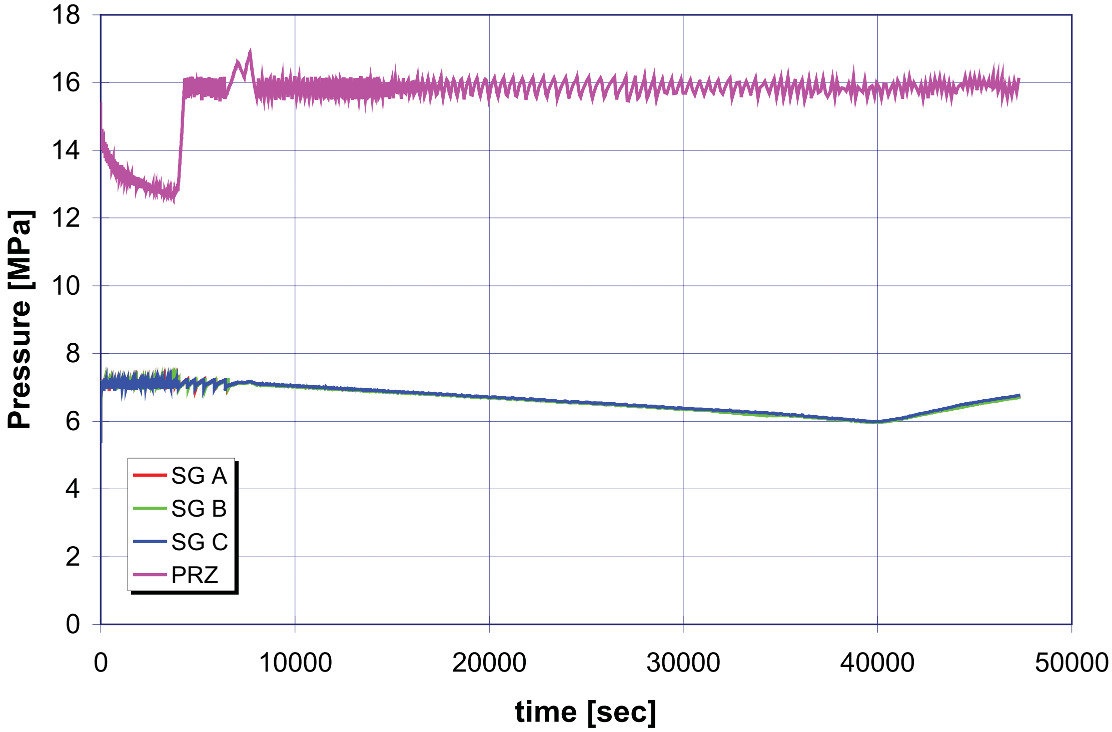
1 100% equals 2,546 MWt (Surry) and 113.9% equals 2,900 MWt (Beaver Valley, Harris, and Summer); 2,900 MWt is the highest present power level of the three-loop Westinghouse plants.
2 Low SG level is <19% of narrow-range span, while low-low SG level is <16% of narrow-range span, based on Technical Specification 2.3-3 (NRC, 2003).
3 Core uncovery and damage late in the simulation are artifacts of the assumed unavailability of HHSI recirculation.

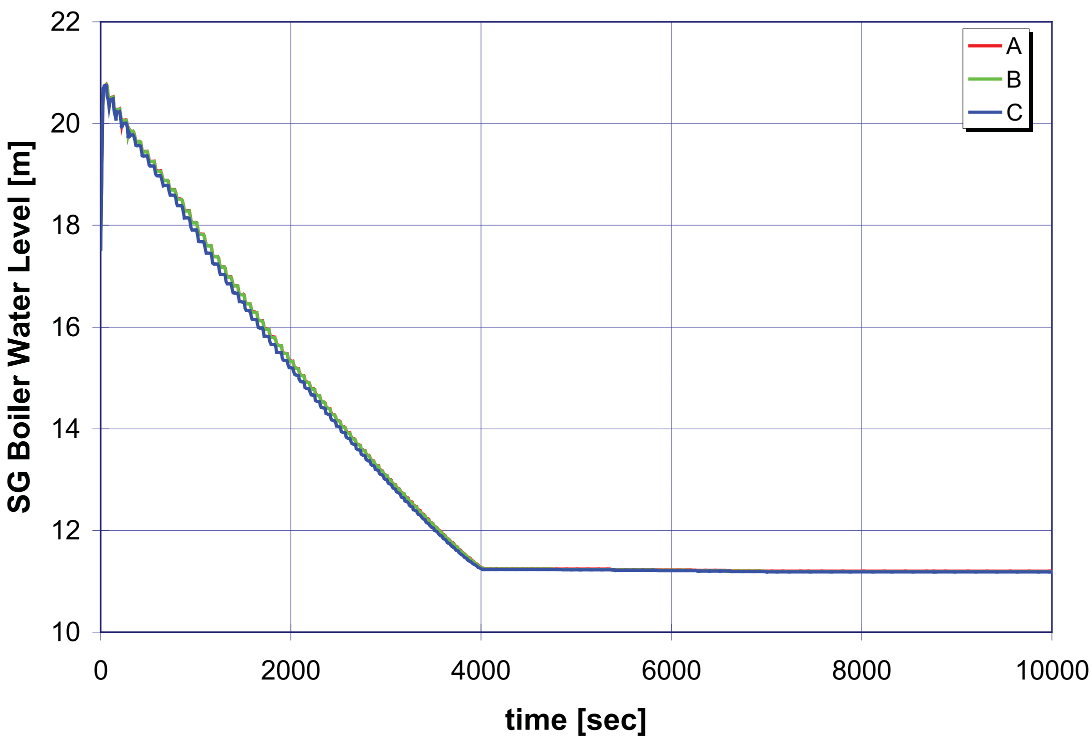
Table 9 Surry Feed-and-Bleed PORV Success Criteria Key Timings

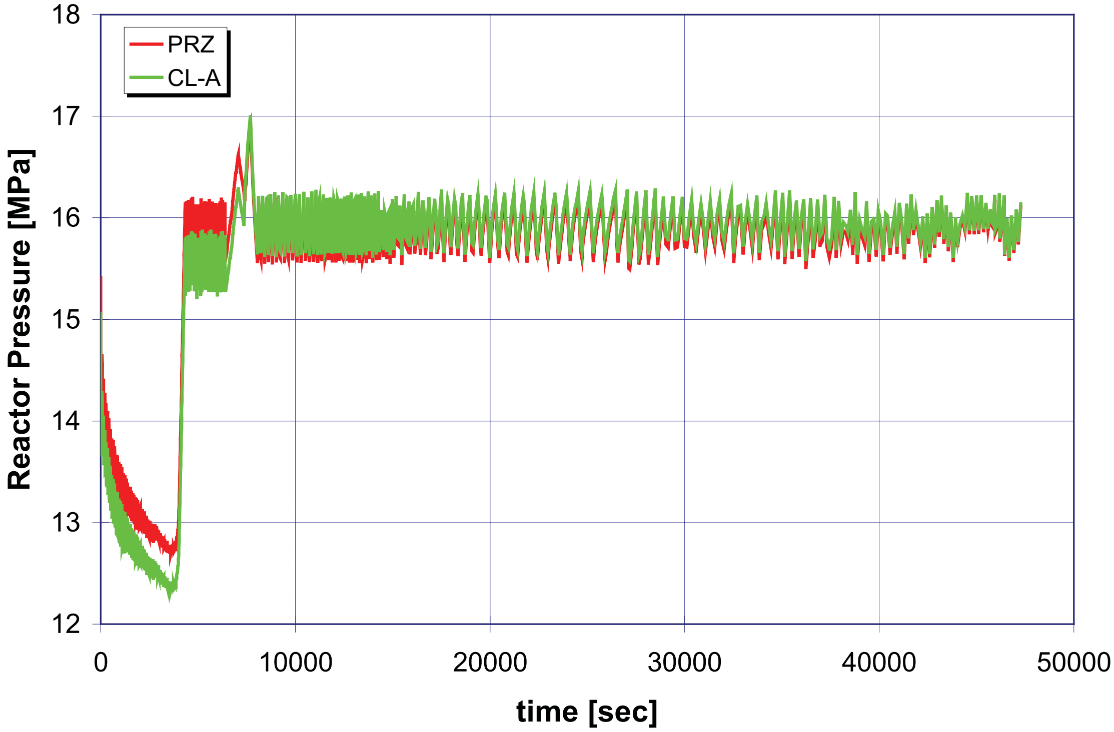
Event[1]	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)
MFW, MD-AFW, TD-AFW unavailable	0	0	0
Reactor trip	0	0.008 (29 s)	0.008 (27 s)
SG dryout	1.11	0.63	0.58
PRT rupture disk open	1.56	0.97	0.93
SI signal (containment pressure >1.22 bars)	1.96	1.36	1.29
RCP trip (10% void)	2.05	1.43	1.35
First actuation of containment sprays (containment pressure >1.72 bars)	3.84	3.24	3.17
RWST depletion (<13.5%)	9.43	8.35	8.24
Core uncovery	10.90[2]	1.65 / 9.54[2]	1.60 / 9.42[2]
Core damage (max. temp. >2,200 °F)	13.53	11.80	11.68

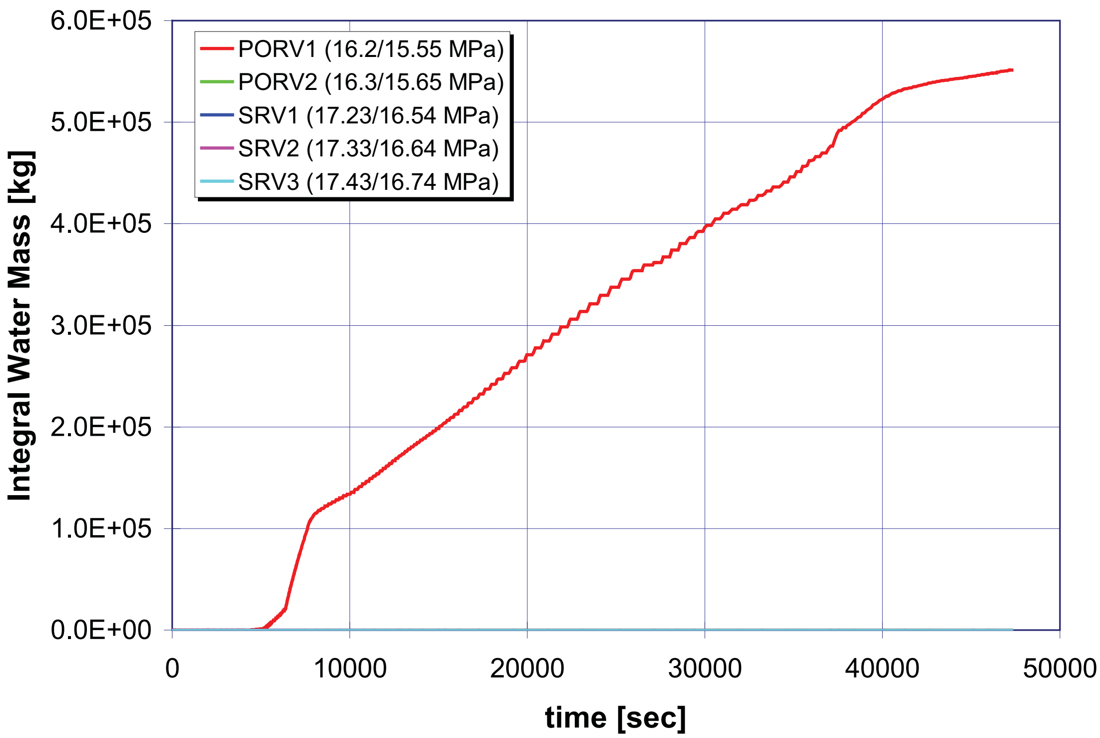
1 1.22 bars = 0.122 MPa; 1.72 bars = 0.172 MPa; 2,200 °F = 1,204 °C.
2 For Case 1, the core comes close to uncovering around the time of SI actuation and then later does uncover after the loss of HHSI. For Cases 2 and 3, the core uncovers early in the accident, recovers prior to significant heatup, and later uncovers again (due to the loss of HHSI).

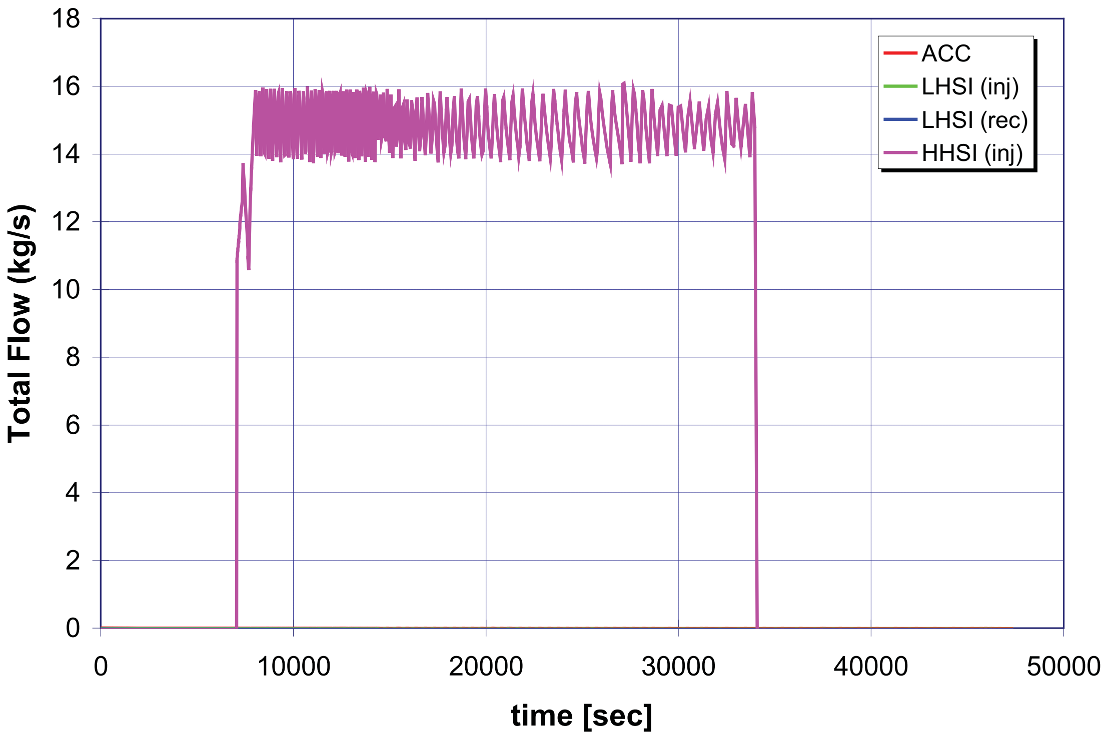
A.3.1 Case 1: 100-Percent Power, Reactor Trip at Time Equals Zero



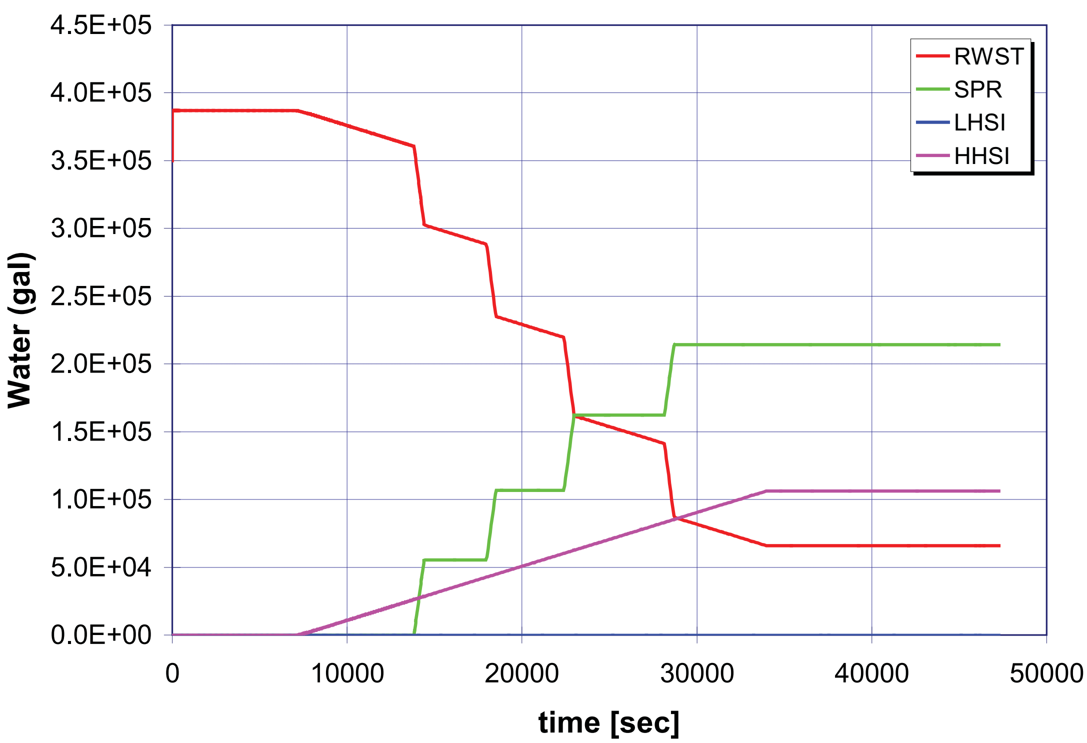



PRZ
CL-A
Reactor Pressure [MPa]
time [sec]
18
17
16
15
14
13
12
0
10000
20000
30000
40000
50000

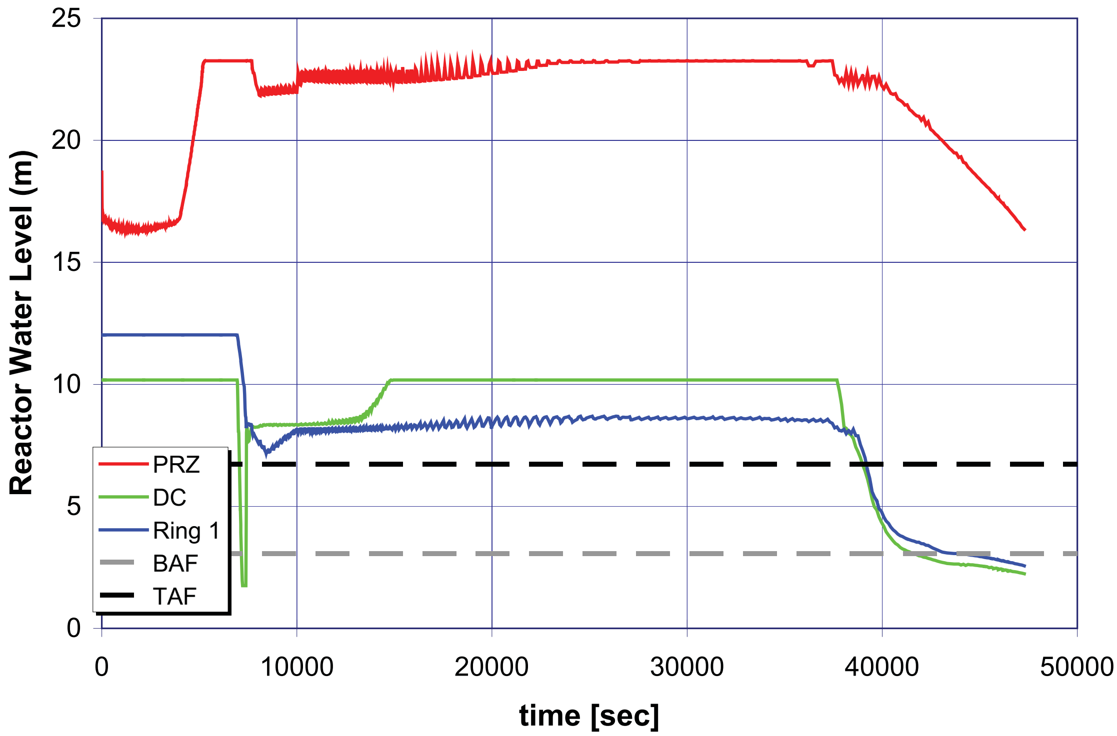

PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
Integral Water Mass [kg]
time [sec]
6.0E+05
5.0E+05
4.0E+05
3.0E+05
2.0E+05
1.0E+05
0.0E+00
0
10000
20000
30000
40000
50000



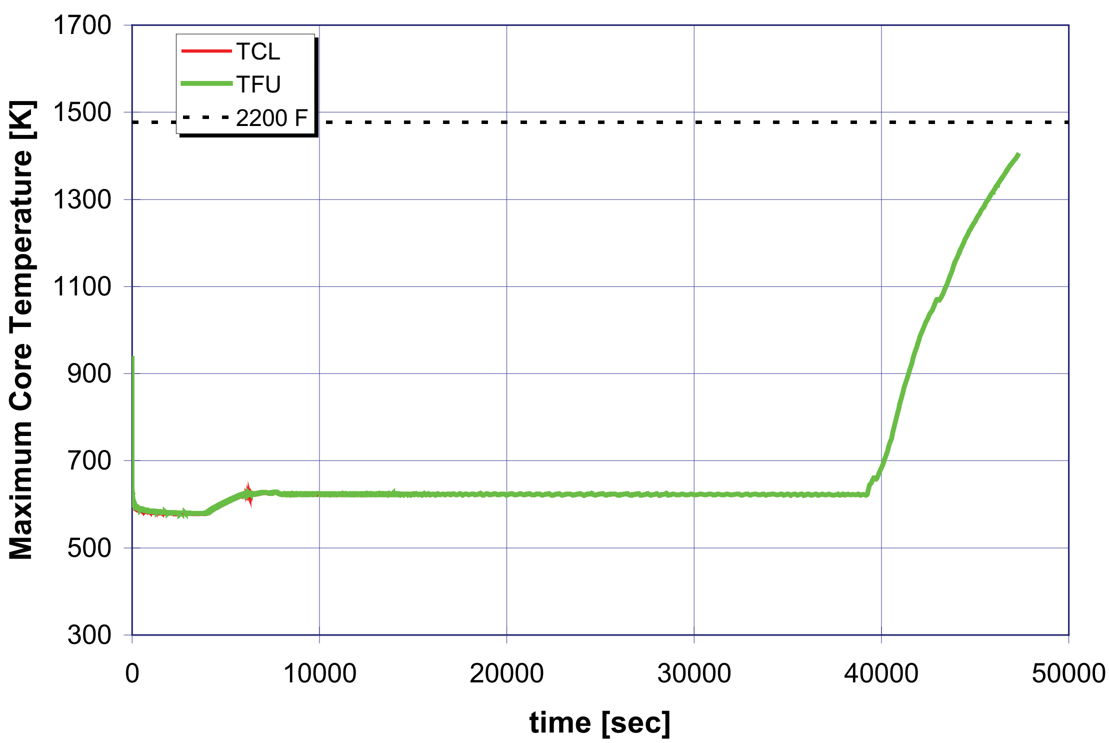
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
0
2
4
6
8
10
12
14
16
18
0
10000
20000
30000
40000
50000



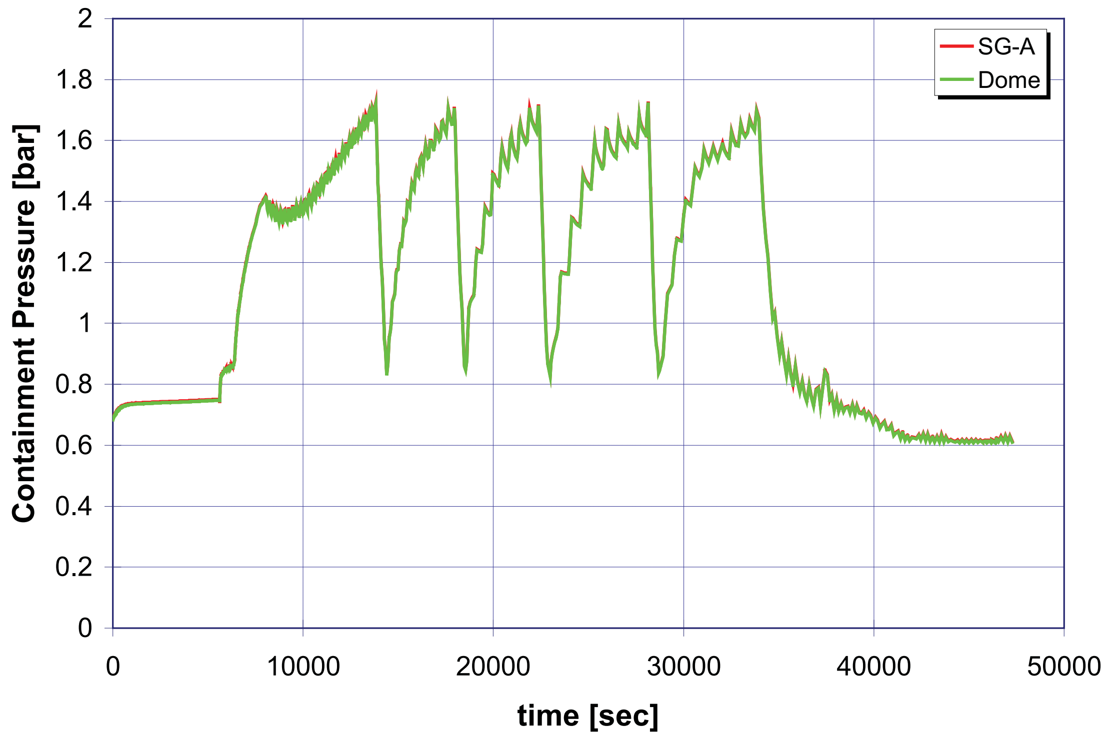
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
0.0E+00
5.0E+04
1.0E+05
1.5E+05
2.0E+05
2.5E+05
3.0E+05
3.5E+05
4.0E+05
4.5E+05
0
10000
20000
30000
40000
50000



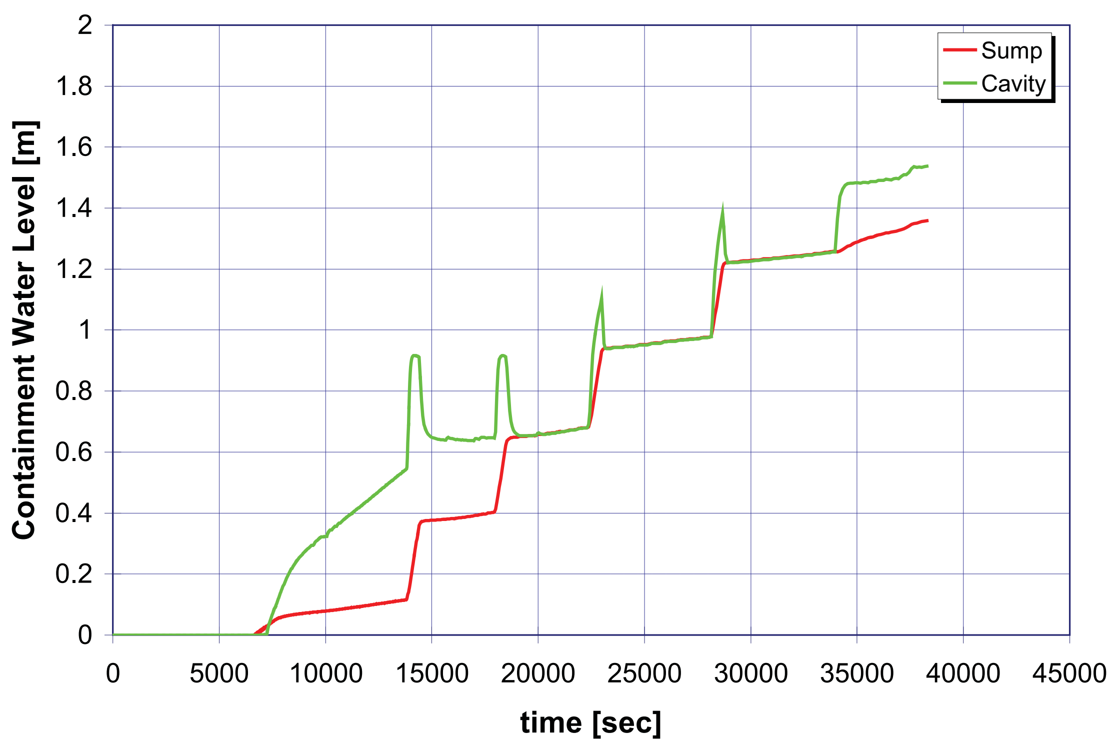
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
5
10
15
20
25
0
10000
20000
30000
40000
50000



Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
500
700
900
1100
1300
1500
1700
0
10000
20000
30000
40000
50000

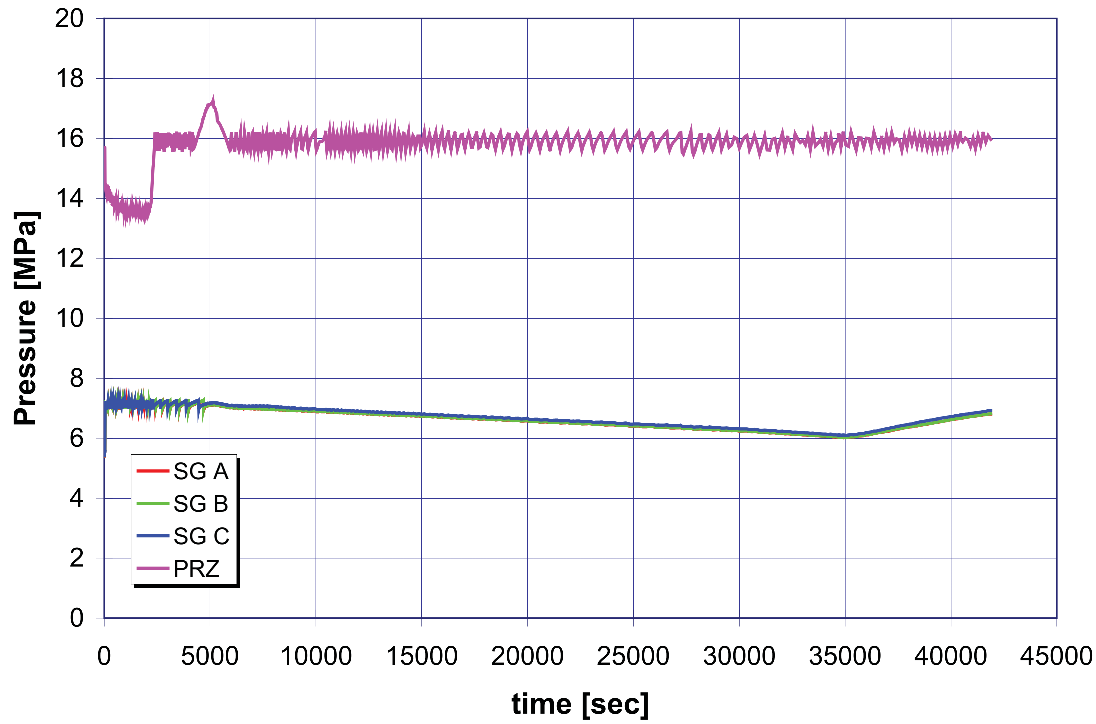


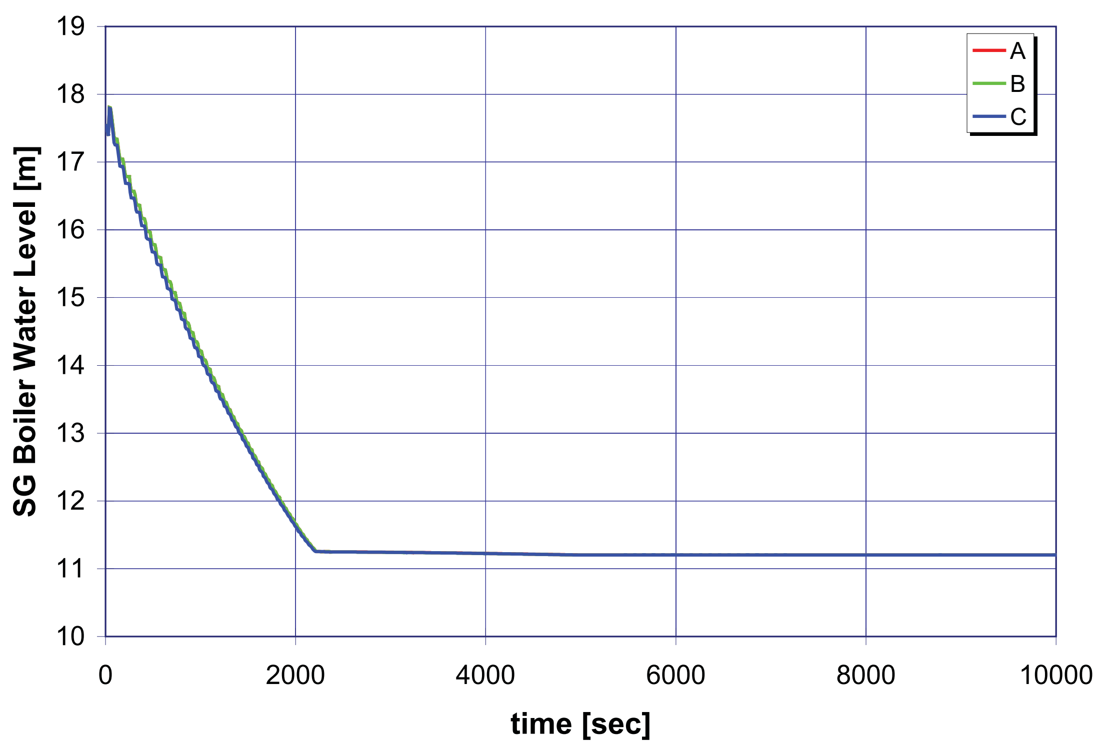
SG-A
Dome
Containment Pressure [bar]
time [sec]

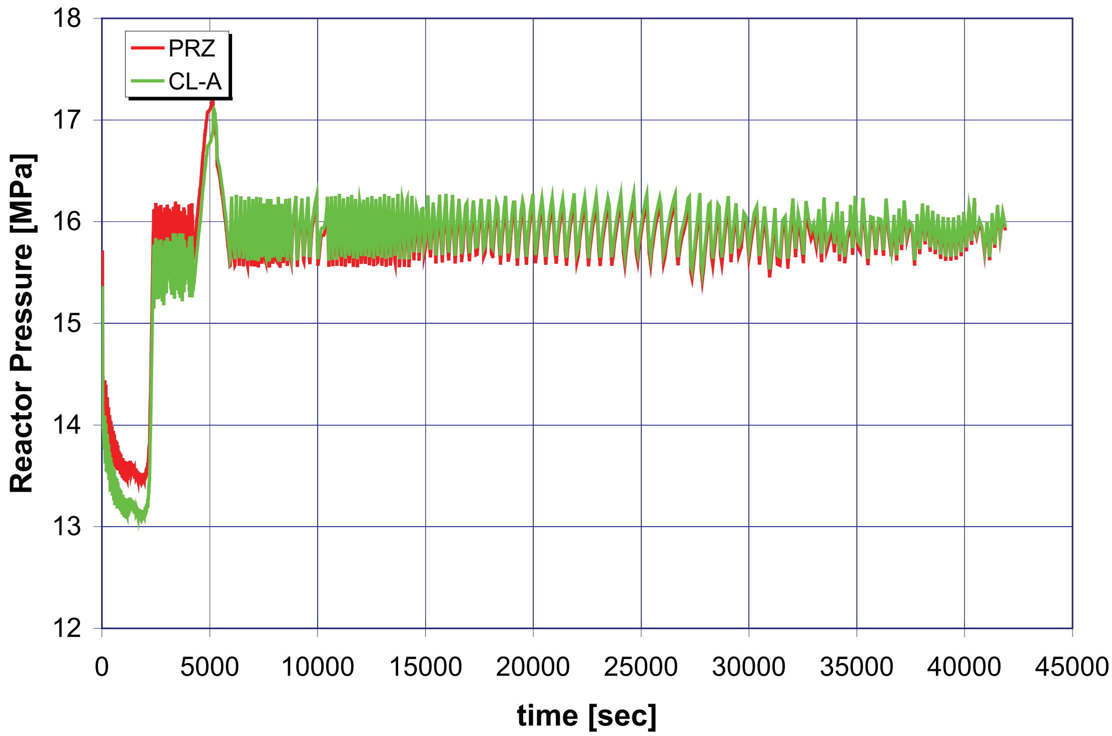


Sump
Cavity
Containment Water Level [m]
time [sec]

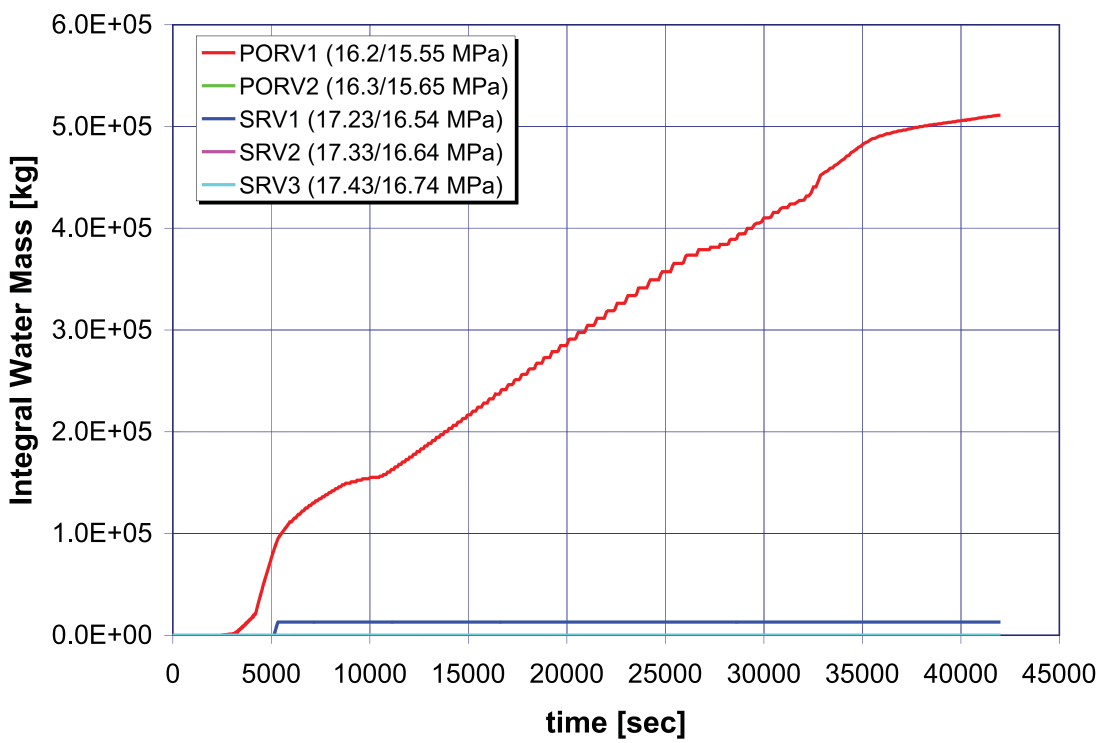
A.3.2 Case 2: 100-Percent Power, Anticipatory Reactor Trip



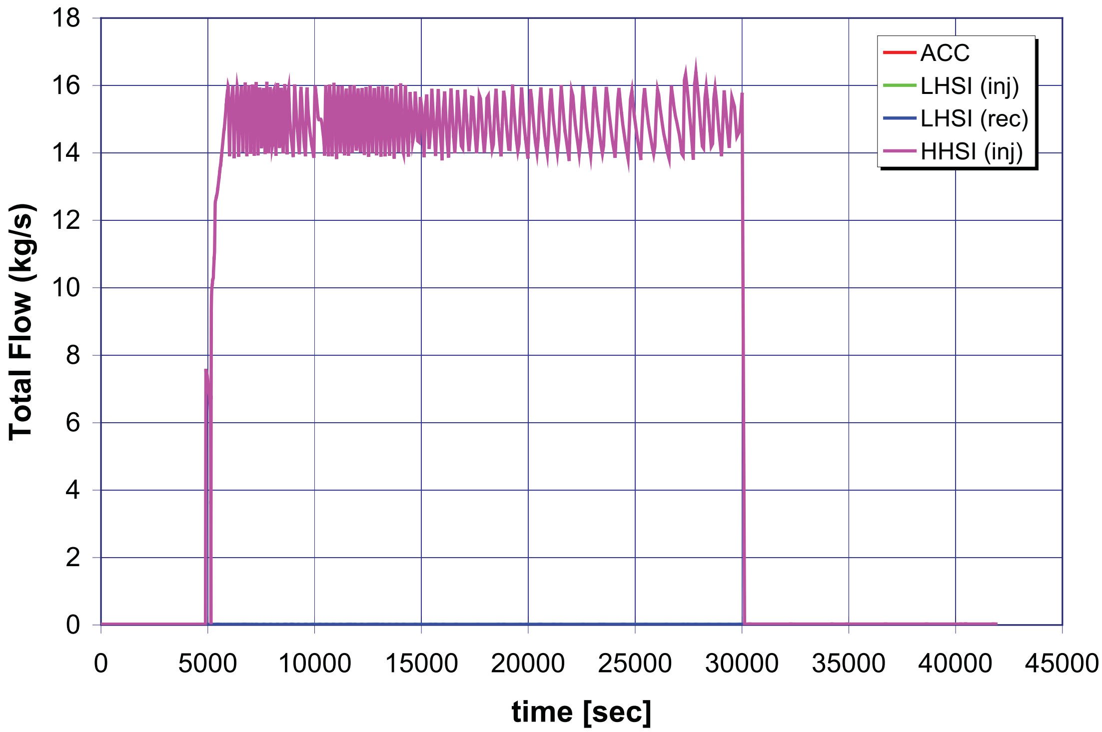




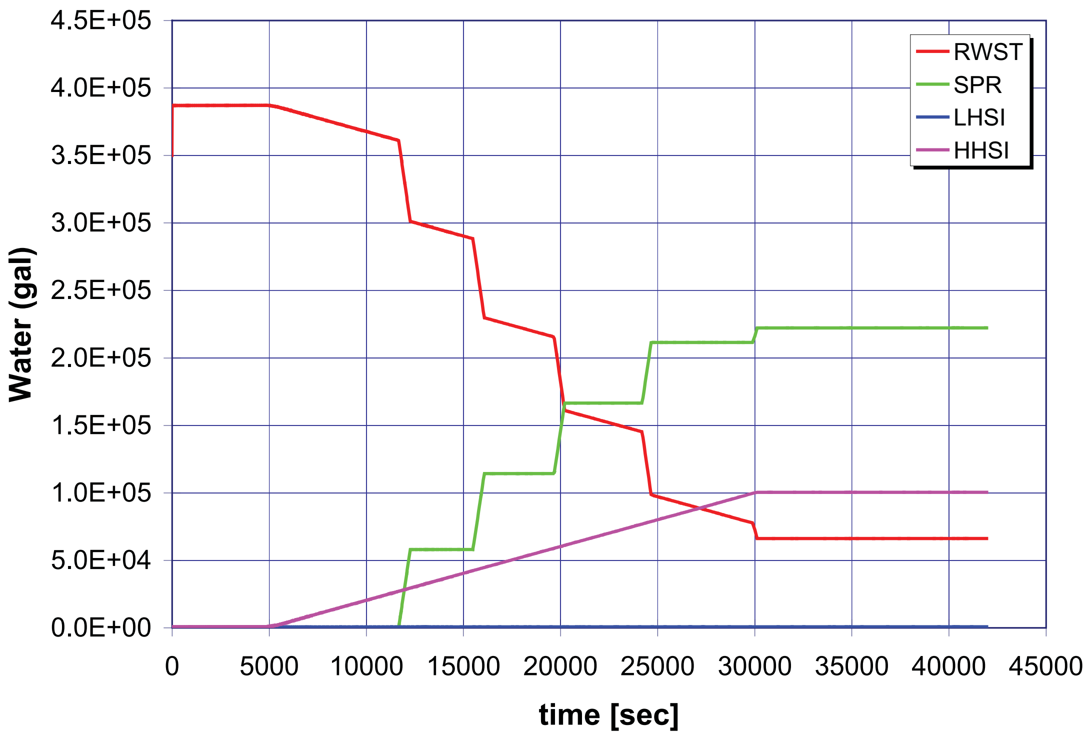
18
17
16
15
14
13
12
Reactor Pressure [MPa]
PRZ
CL-A
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
time [sec]



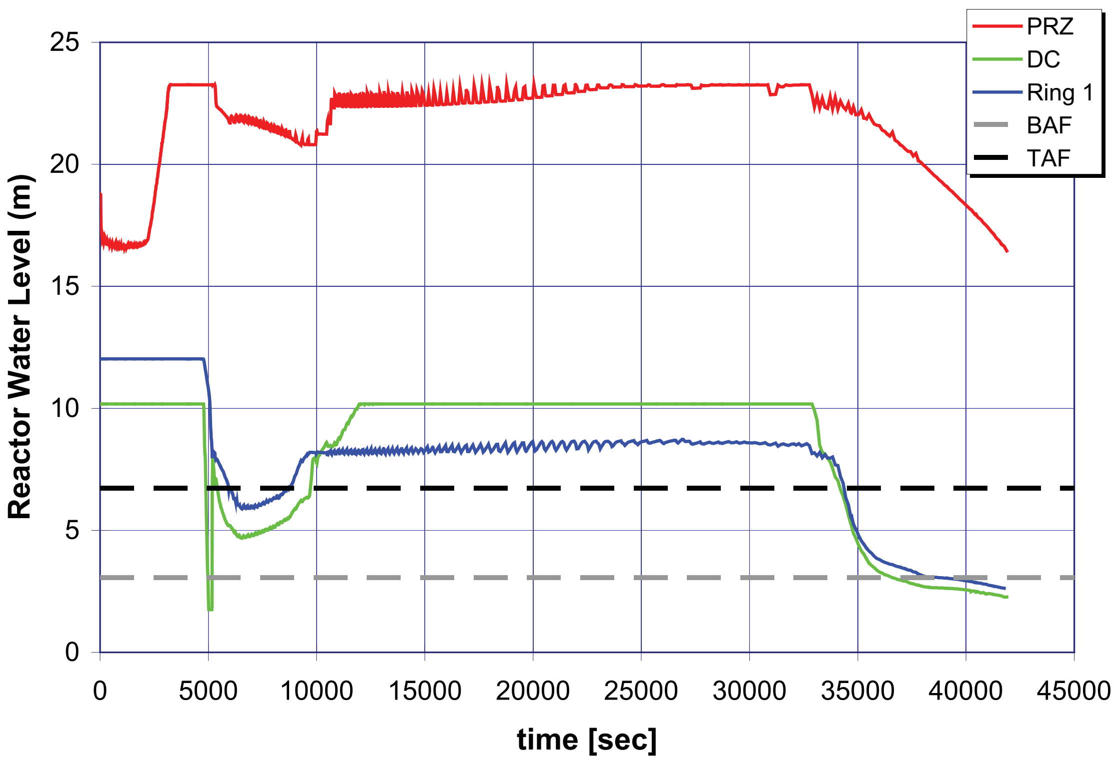
6.0E+05
5.0E+05
4.0E+05
3.0E+05
2.0E+05
1.0E+05
0.0E+00
Integral Water Mass [kg]
PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
time [sec]



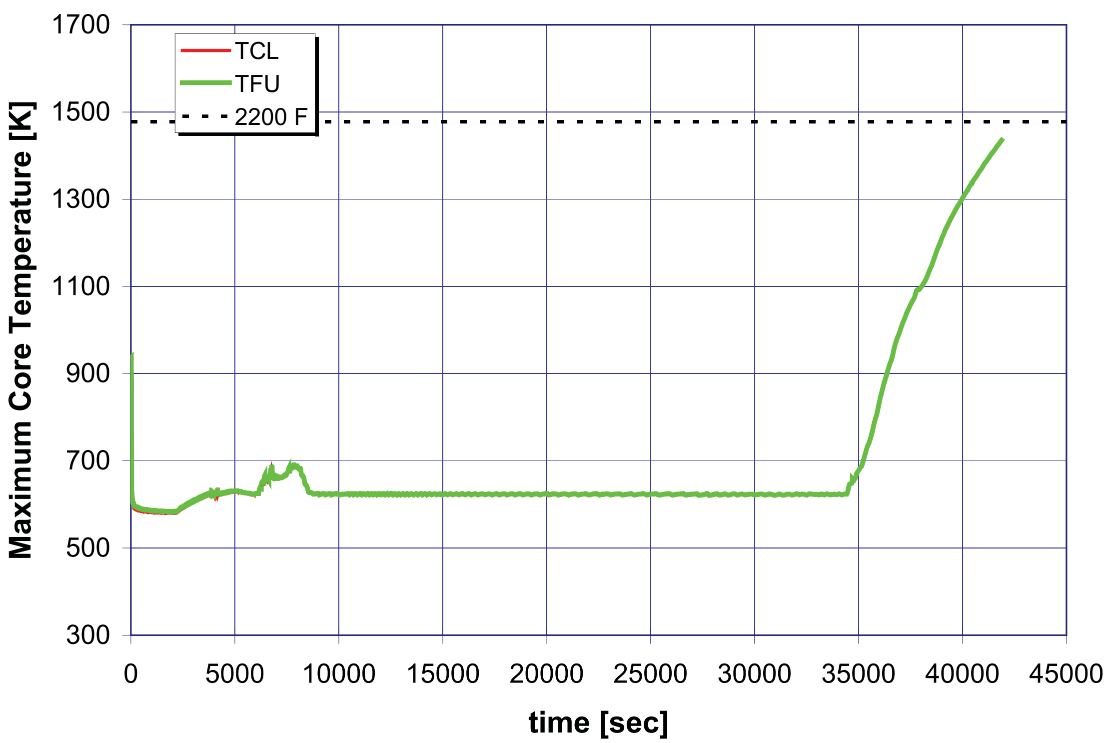
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)



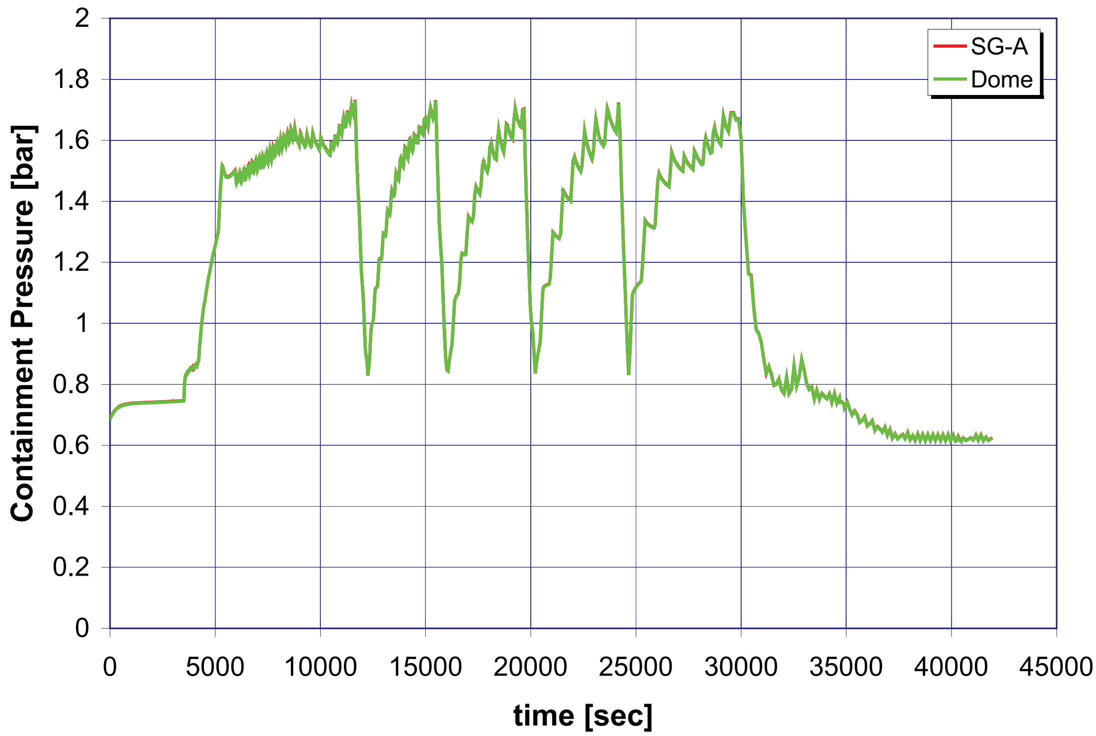
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI



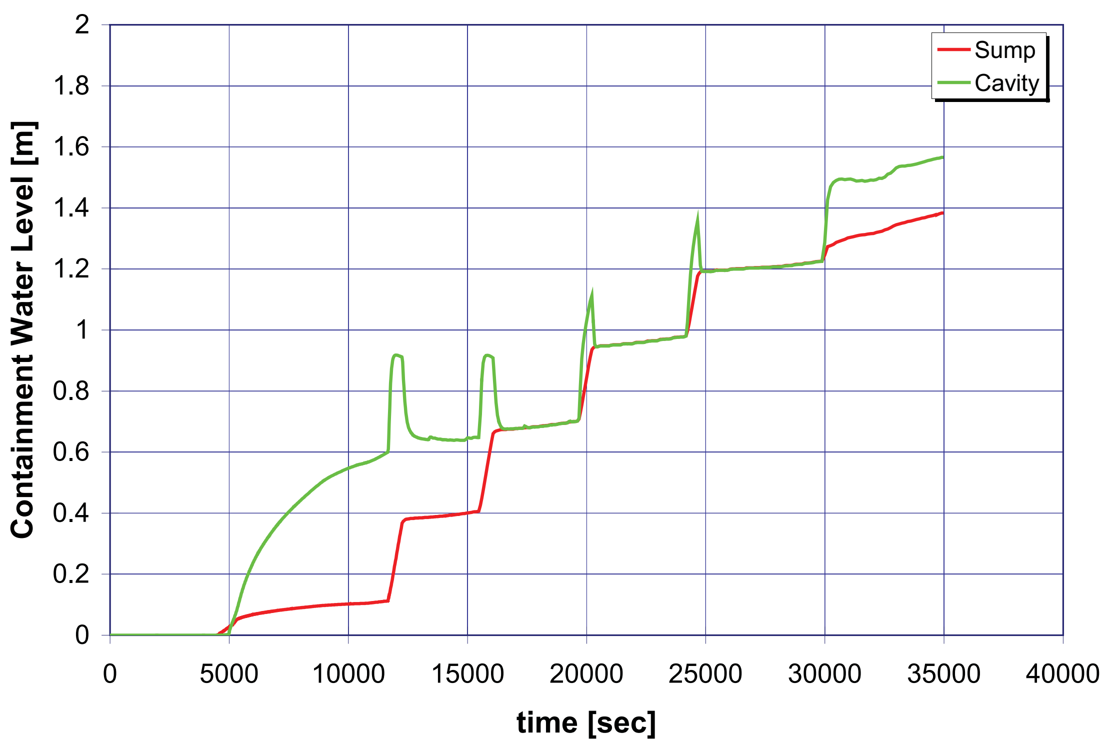
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
5
10
15
20
25
0
5000
10000
15000
20000
25000
30000
35000
40000
45000



Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
500
700
900
1100
1300
1500
1700
0
5000
10000
15000
20000
25000
30000
35000
40000
45000

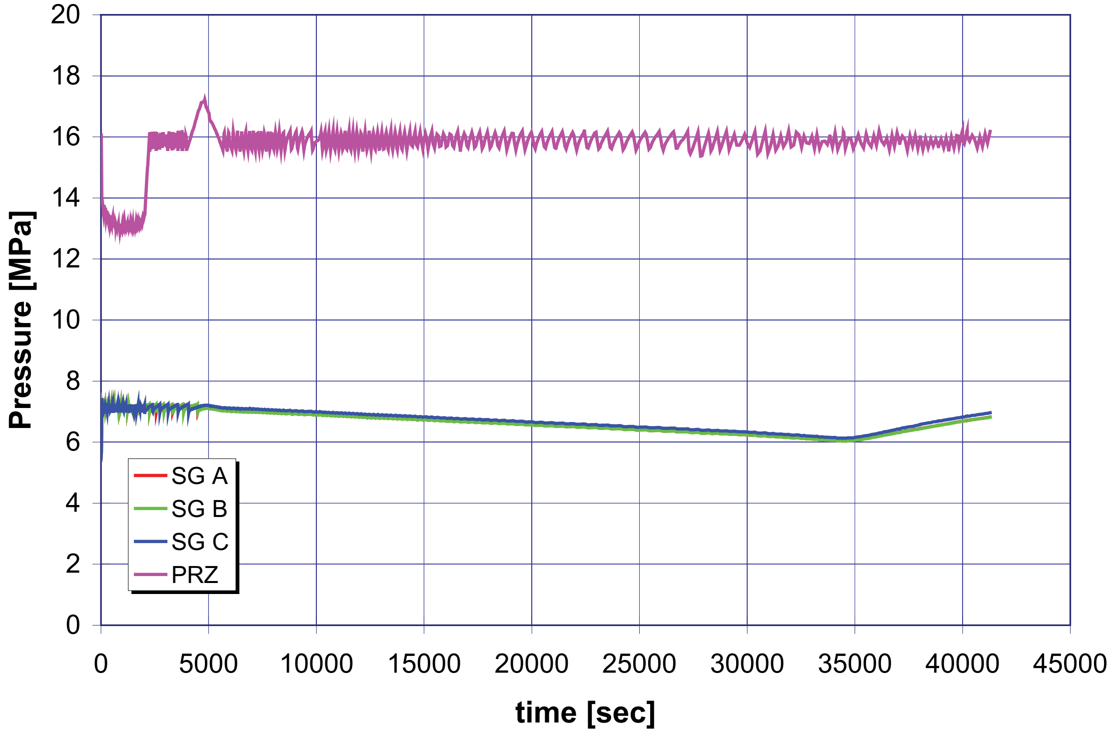


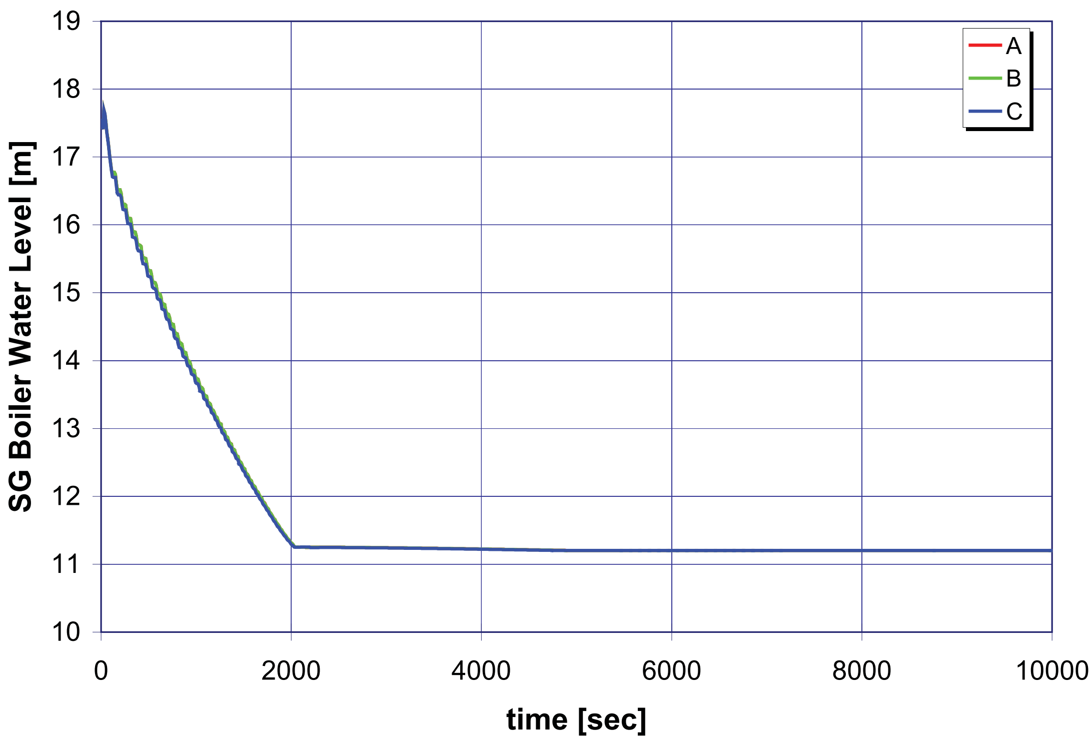
Containment Pressure [bar]
time [sec]
SG-A
Dome
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000
35000
40000
45000

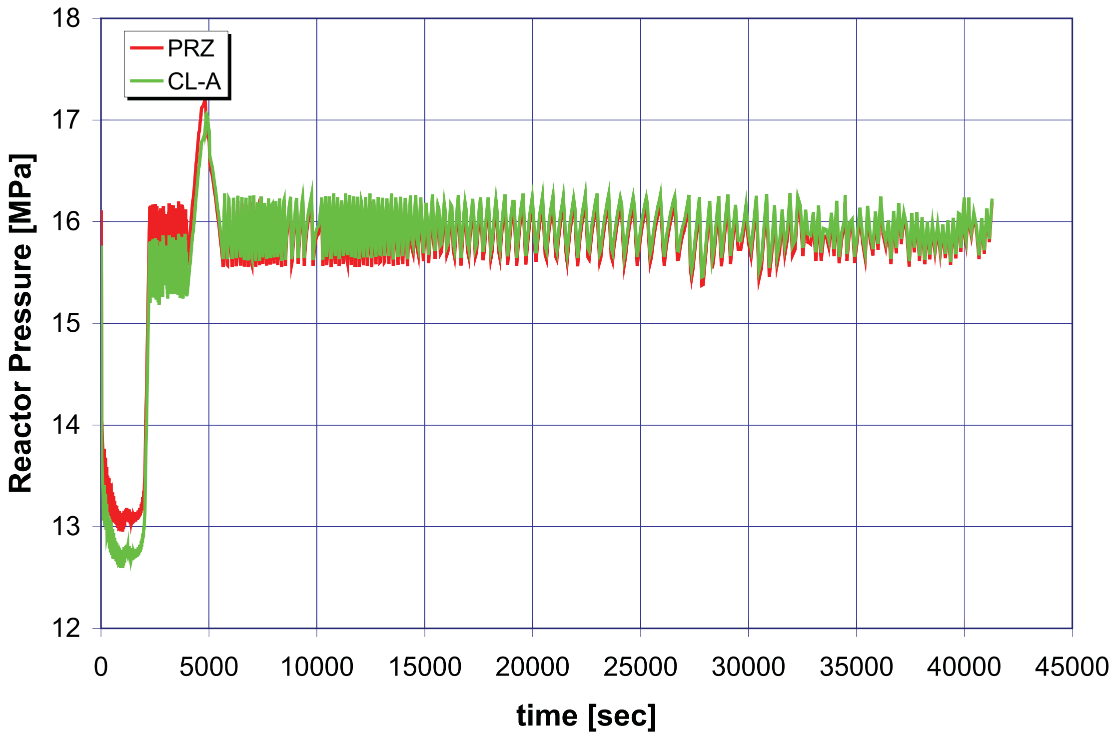


Containment Water Level [m]
time [sec]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000
35000
40000

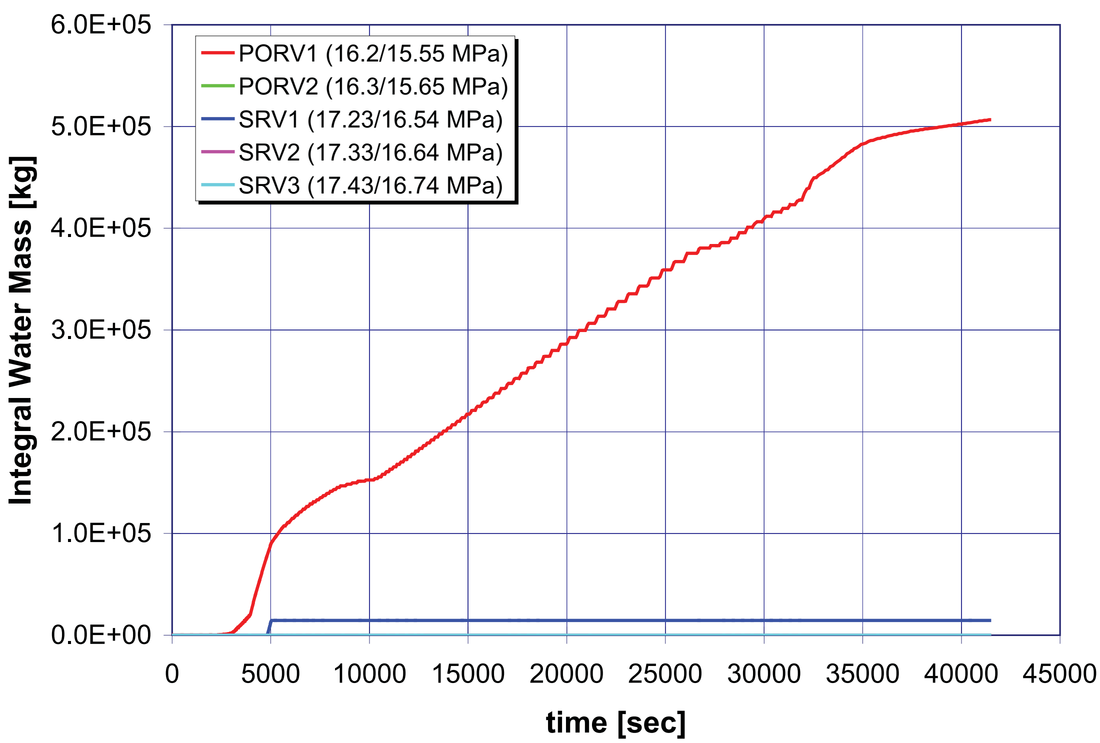
A.3.3 Case 3: 113.9-Percent Power Level, Reactor Trip on Low-Low SG Level



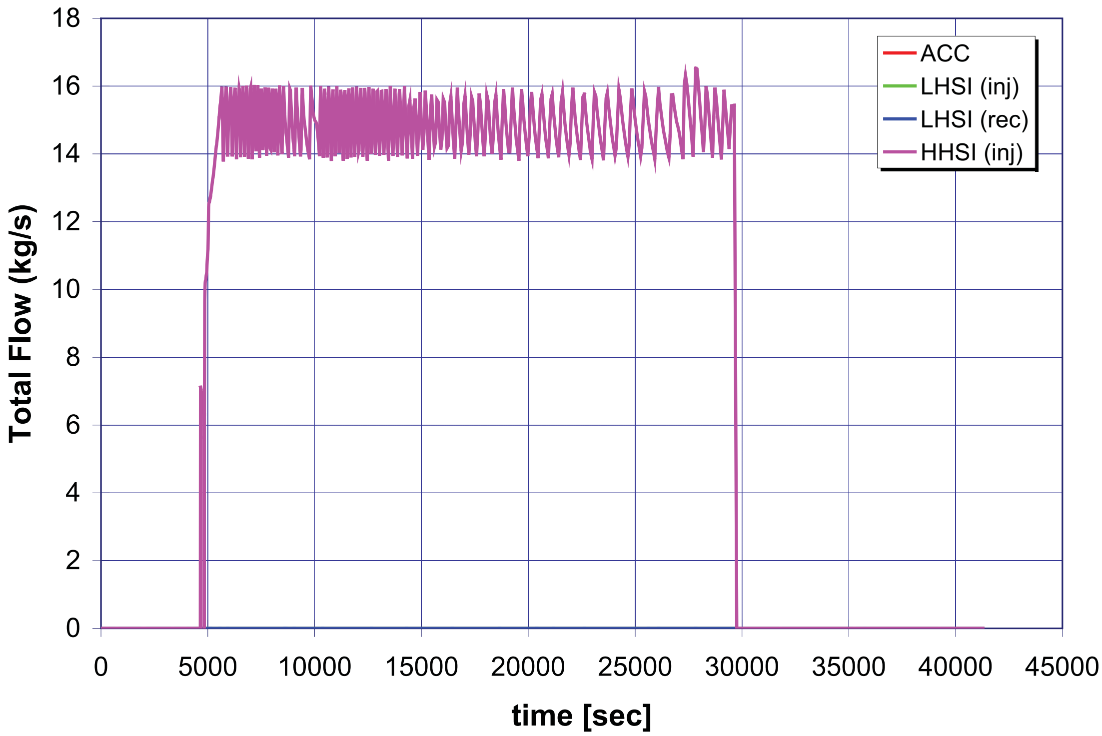




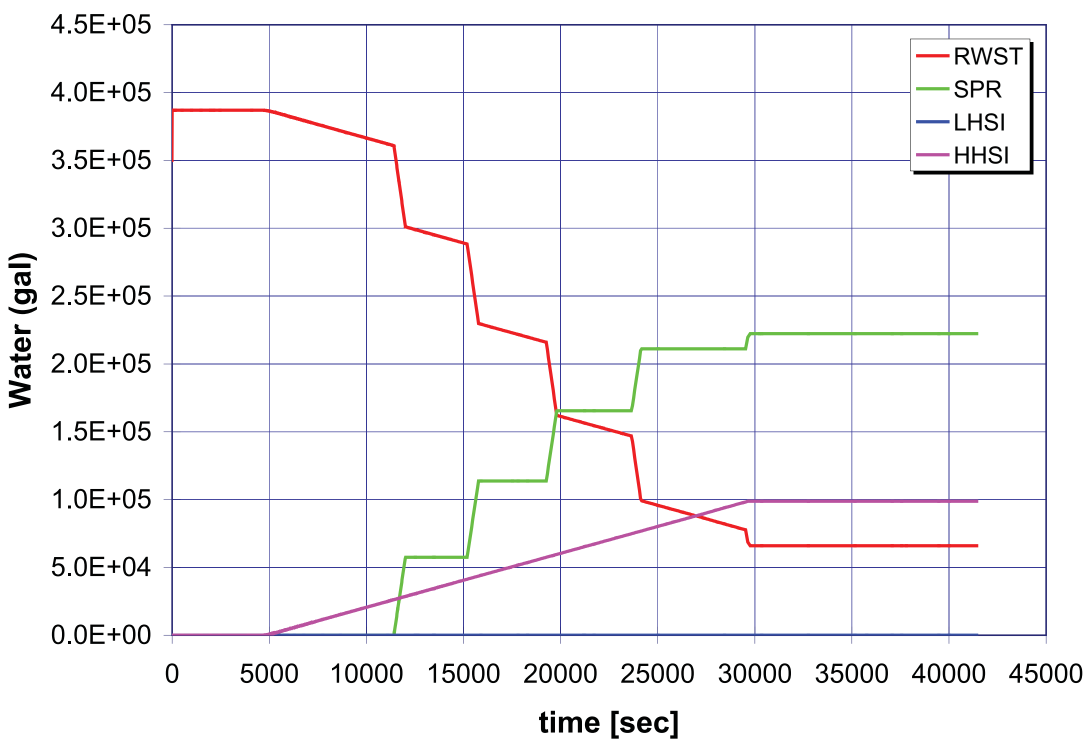
PRZ
CL-A
Reactor Pressure [MPa]
time [sec]
18
17
16
15
14
13
12
0
5000
10000
15000
20000
25000
30000
35000
40000
45000



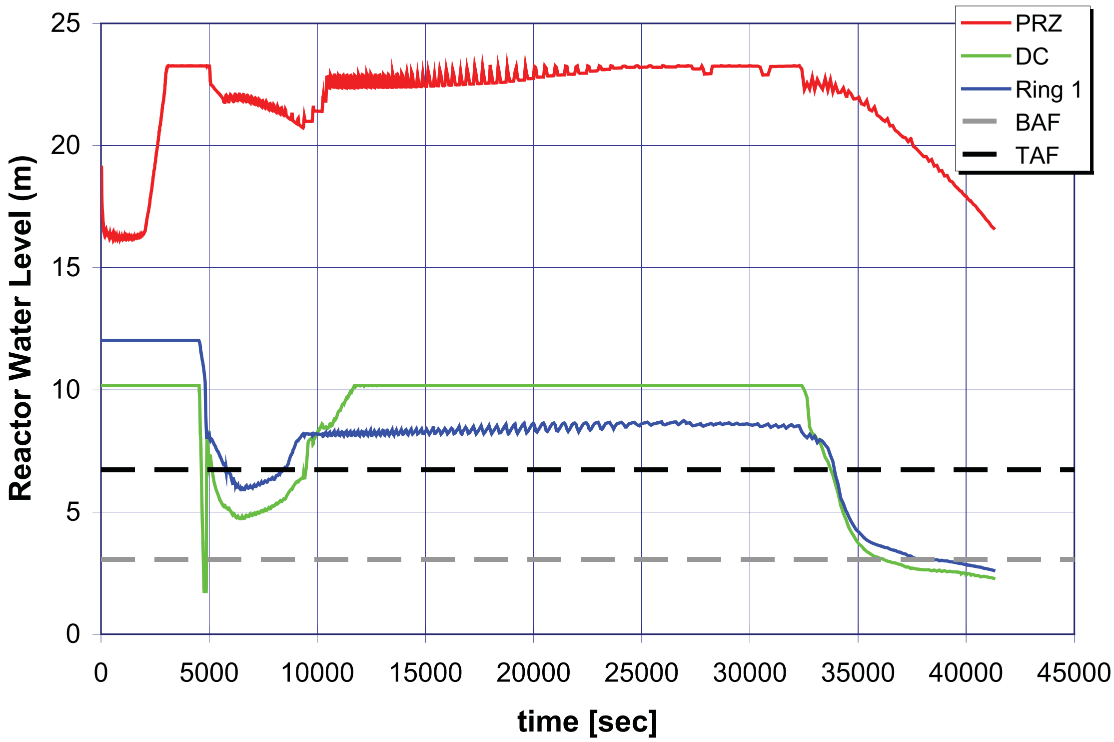
PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
Integral Water Mass [kg]
time [sec]
6.0E+05
5.0E+05
4.0E+05
3.0E+05
2.0E+05
1.0E+05
0.0E+00
0
5000
10000
15000
20000
25000
30000
35000
40000
45000

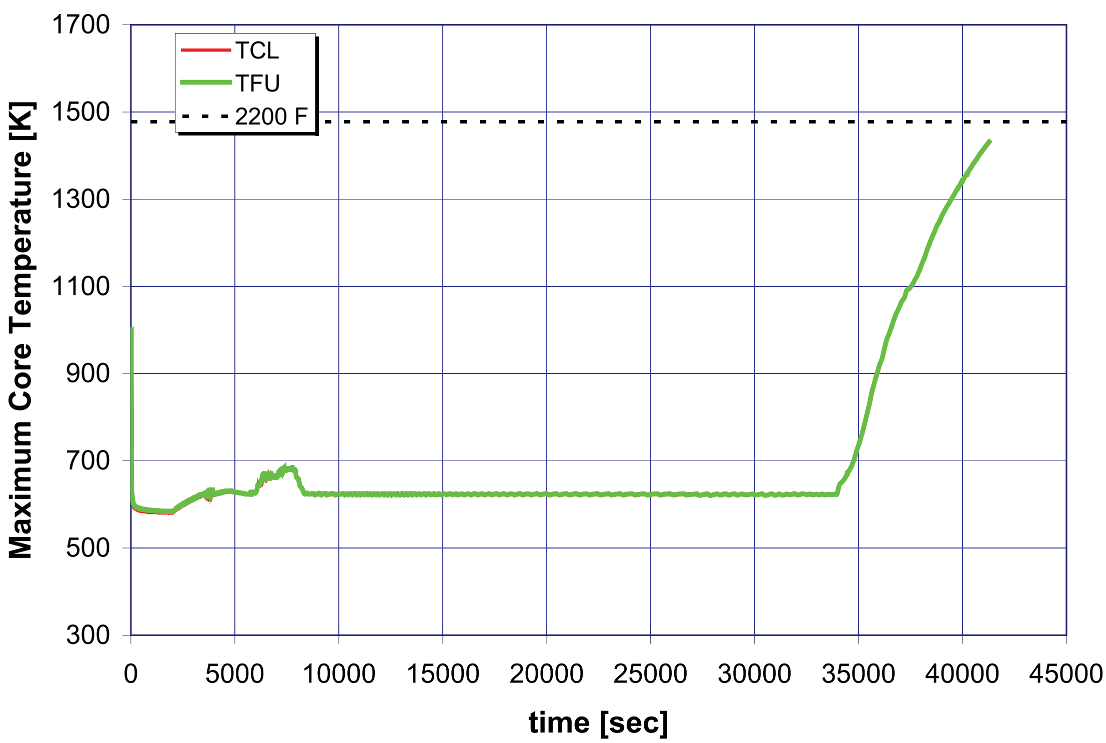


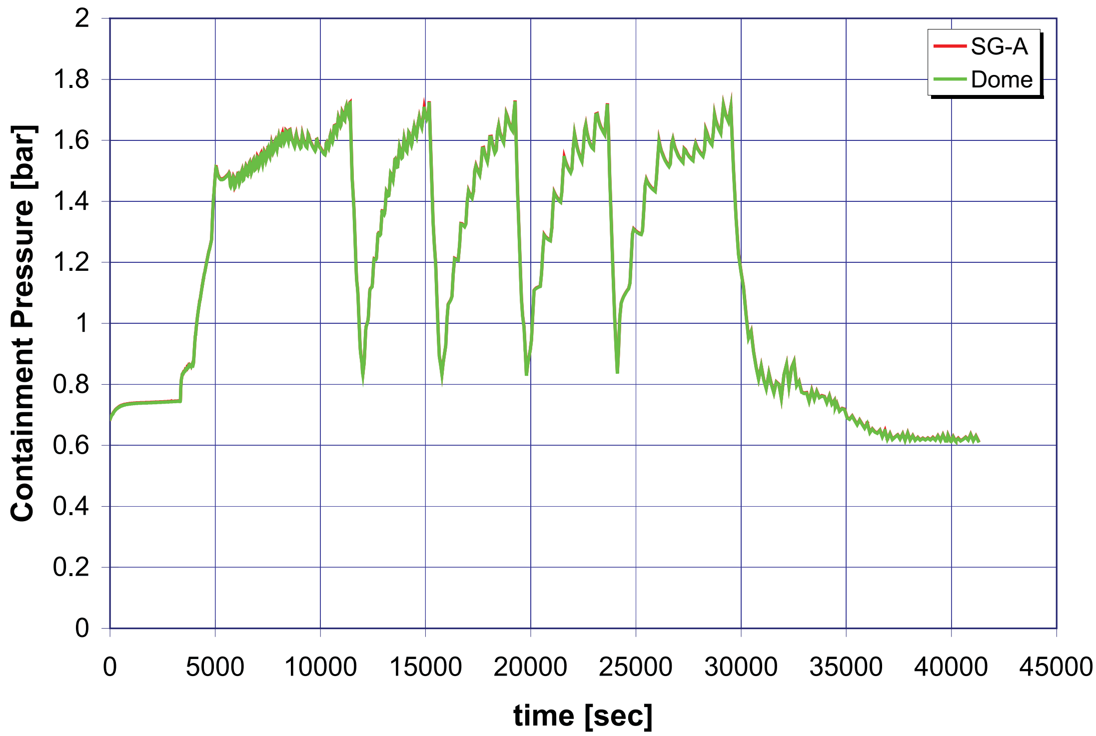
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
0
2
4
6
8
10
12
14
16
18
0
5000
10000
15000
20000
25000
30000
35000
40000
45000

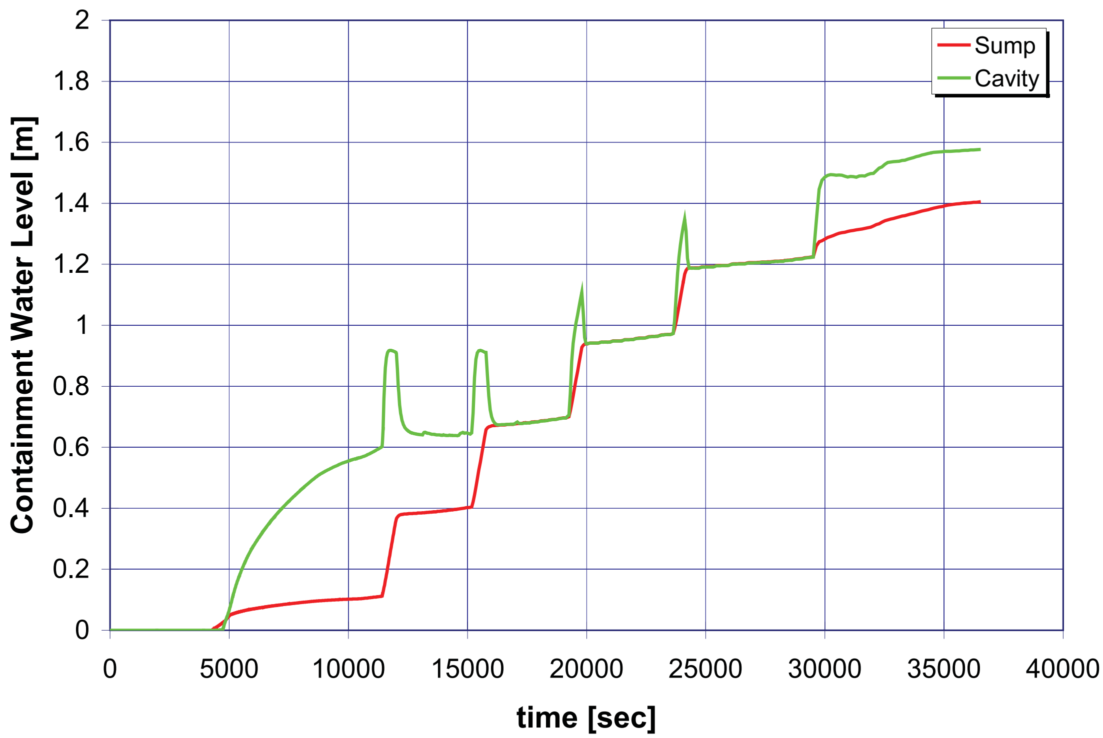


Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
0.0E+00
5.0E+04
1.0E+05
1.5E+05
2.0E+05
2.5E+05
3.0E+05
3.5E+05
4.0E+05
4.5E+05
0
5000
10000
15000
20000
25000
30000
35000
40000
45000


Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
5
10
15
20
25
0
5000
10000
15000
20000
25000
30000
35000
40000
45000


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
500
700
900
1100
1300
1500
1700
0
5000
10000
15000
20000
25000
30000
35000
40000
45000


SG-A
Dome
Containment Pressure [bar]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000
35000
40000
45000


Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000
35000
40000

A.4 Steam Generator Tube Rupture Event Tree Timing

Analysis Summary

For Section A.4.1 through Section A.4.5, operators fail to (1) isolate faulted steam generator (SG), (2) depressurize and cool RCS, and (3) extend ECCS duration by refilling RWST or cross-connection to other unit's RWST. Loop A has the faulted SG. Table 10 and Table 11 provide results for this portion of the analysis.

Table 10 Surry SGTR Results

<table>
<tr><th>Case</th><th>No. Tubes</th><th>HHSI Pumps</th><th>SG PORV Treatment</th><th>TD-AFW</th><th>MD-AFW</th><th>Nominal Break Flow Prior to Loss of HHSI (kg/sec)</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td>1</td><td rowspan="2">3/2</td><td rowspan="3">Does not stick open[1]</td><td colspan="2" rowspan="5">Yes[2]</td><td>30</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>2</td><td>5</td><td>50 – 60</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>3</td><td rowspan="2">1</td><td>3/1</td><td>23</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>4</td><td rowspan="2">3/2</td><td rowspan="2">Sticks open after 119 lifts</td><td>30 – 40</td><td>No[3]</td><td>No[3]</td></tr>
<tr><td>5</td><td>5</td><td>60 - 70</td><td>No[3]</td><td>No[3]</td></tr>
</table>

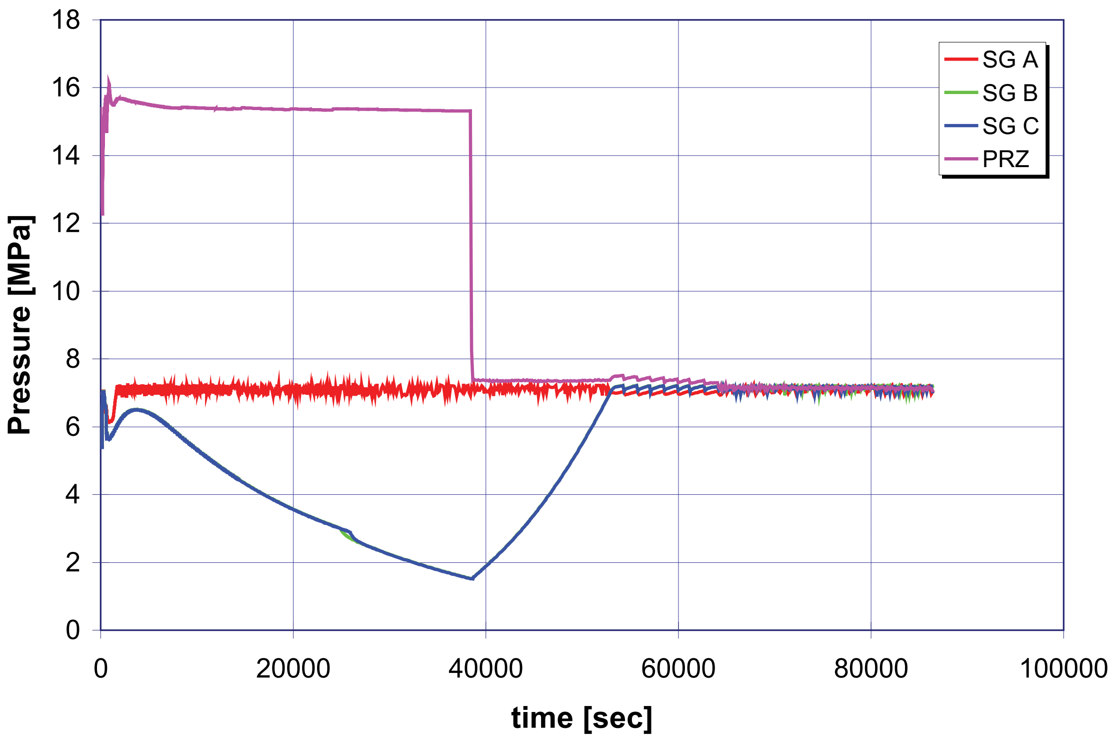
1 Logic was added to address numerical instability (by limiting the flow area to smooth the liquid flow through the faulted SG PORV).
2 TD-AFW is lost within the first hour for all cases due to flooding of the steamline.
3 Based on a 24-hour mission time.

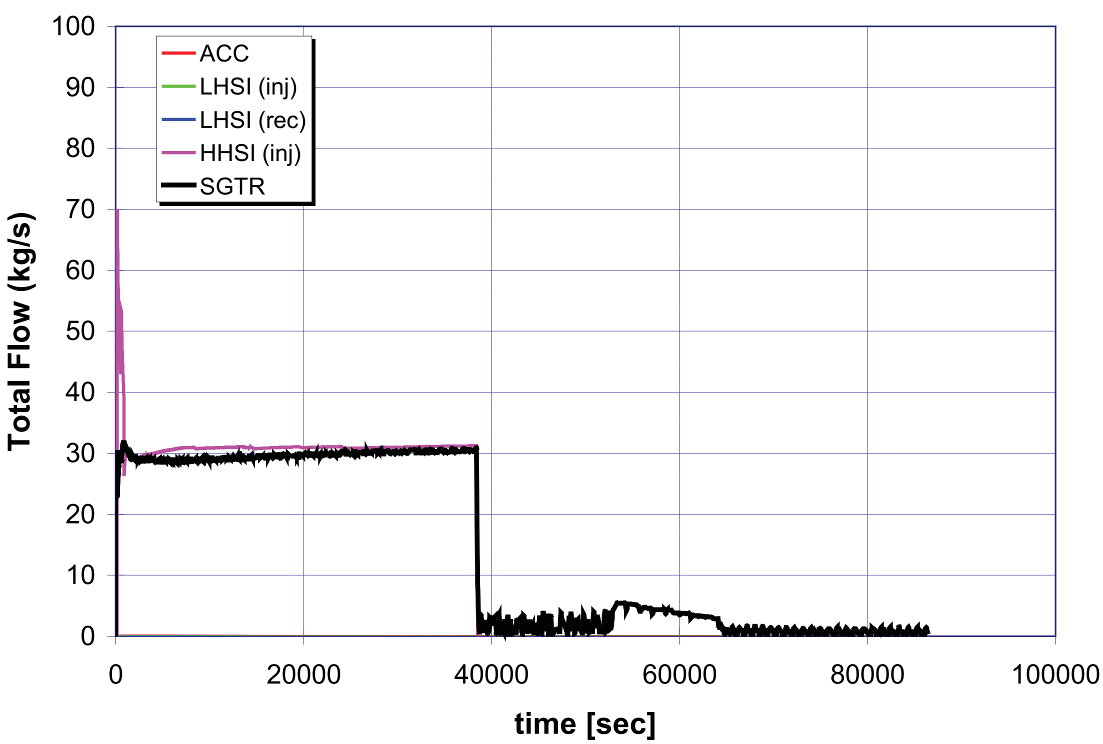
Table 11 Surry SGTR Key Timings

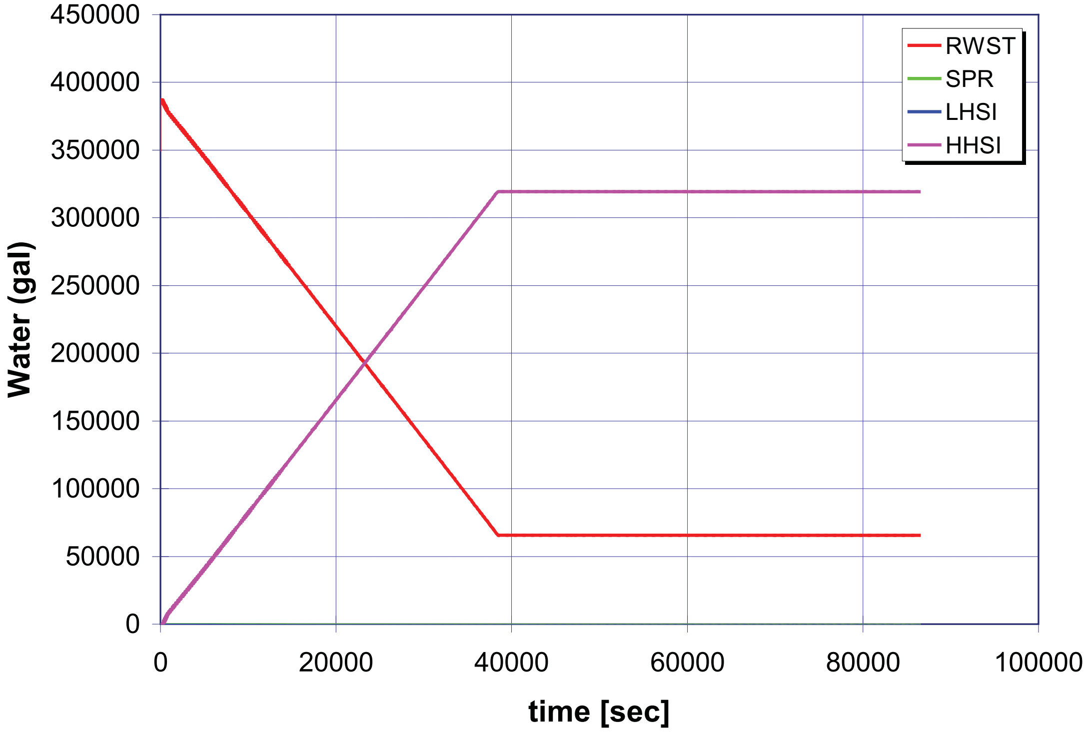
Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)	Case 4 (hr)	Case 5 (hr)
Reactor Trip	0.048	0.012	0.048	0.048	0.012
HHSI initiates (3 pumps)	0.051	0.013	0.051	0.051	0.013
1 of 3 HHSI pumps secured	0.25	0.25	N/A	0.25	0.25
2 of 3 HHSI pumps secured	N/A	N/A	0.25	N/A	N/A
TD-AFW shut down[1]	0.70	0.32	0.75	0.70	0.32
Faulted SG PORV stuck open	N/A	N/A	N/A	0.76	0.35
RWST depletion (<13.5%)[2]	10.68	5.58	14.06	8.41	4.69
Accumulator injection	N/A	N/A	N/A	8.62	0.94
RCP trip (10% void)	17.81	11.71	20.20	12.44	5.02
Emergency CST empty[3]	>24 hours	>24 hours	>24 hours	>24 hours	22.20
Core damage	>24 hours				

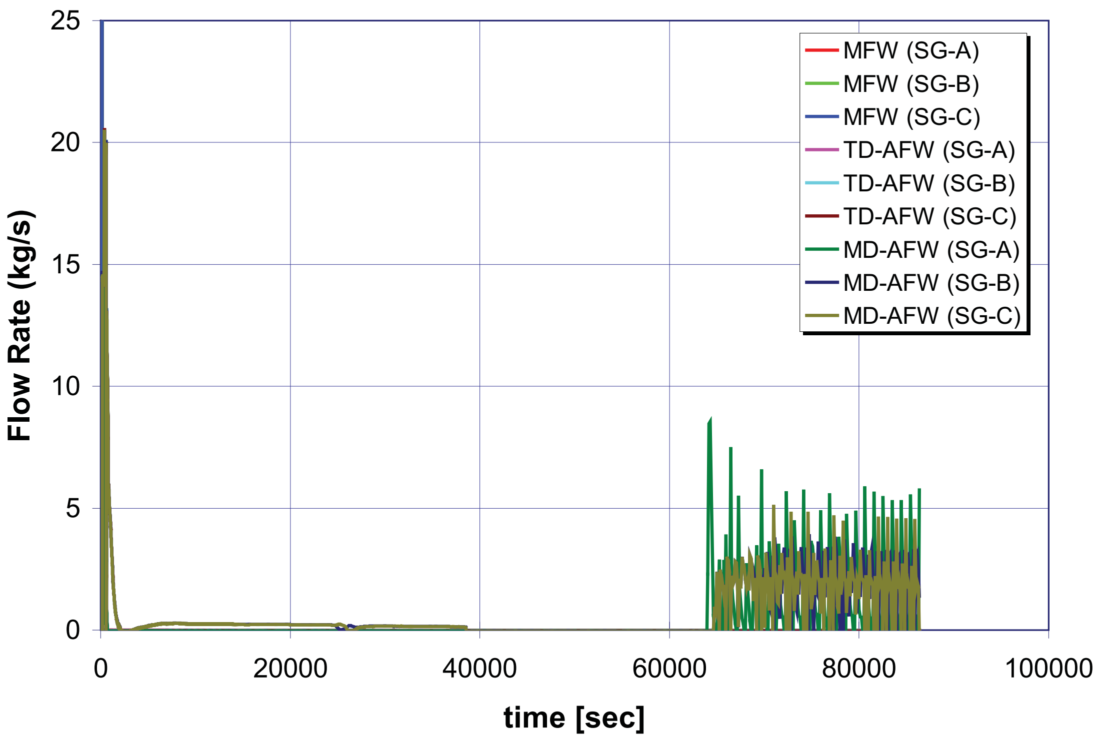
1 TD-AFW shuts down due to filling of the steamline and flooding of the pump.
2 Recall that since the RCS leak location is the ruptured SG tube(s), a substantial amount of water is expelled from the system via the SG relief valves (rather than into containment) and is thus unavailable for containment sump recirculation.
3 Depletion of the emergency condensate storage tank (CST) (96,000 gal (363 m^3)), which is the normal injection source for auxiliary feedwater (AFW), stops MD-AFW.

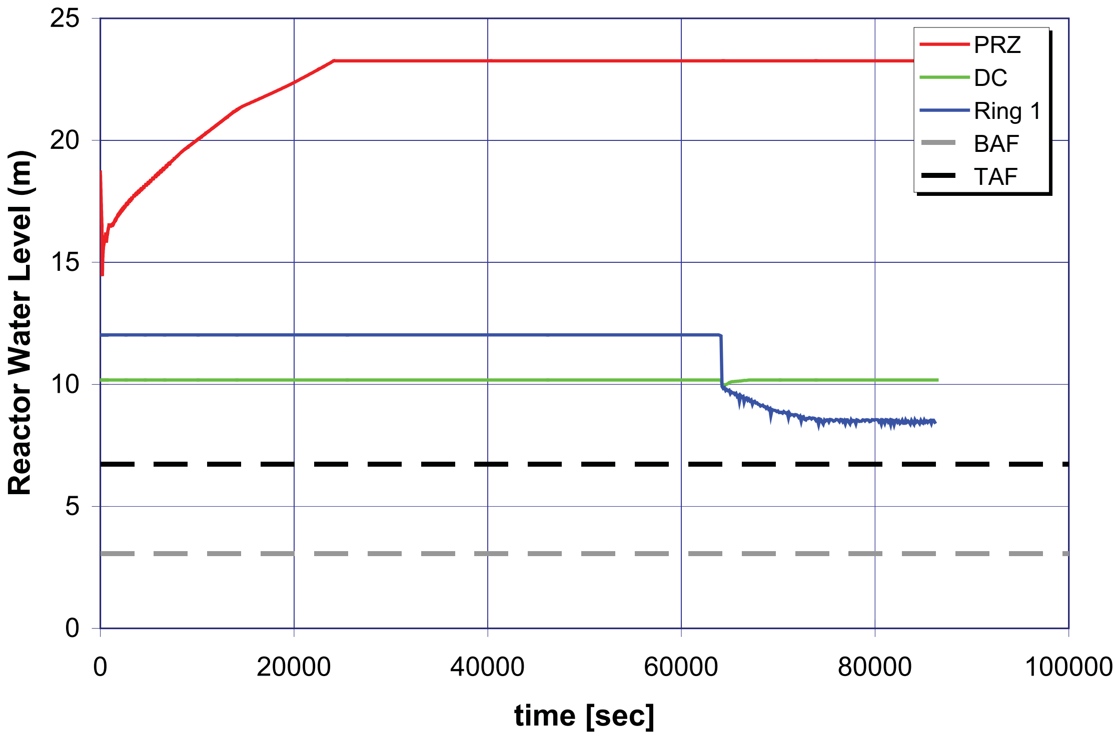
A.4.1 Case 1: One Tube and Secure One HHSI Pump



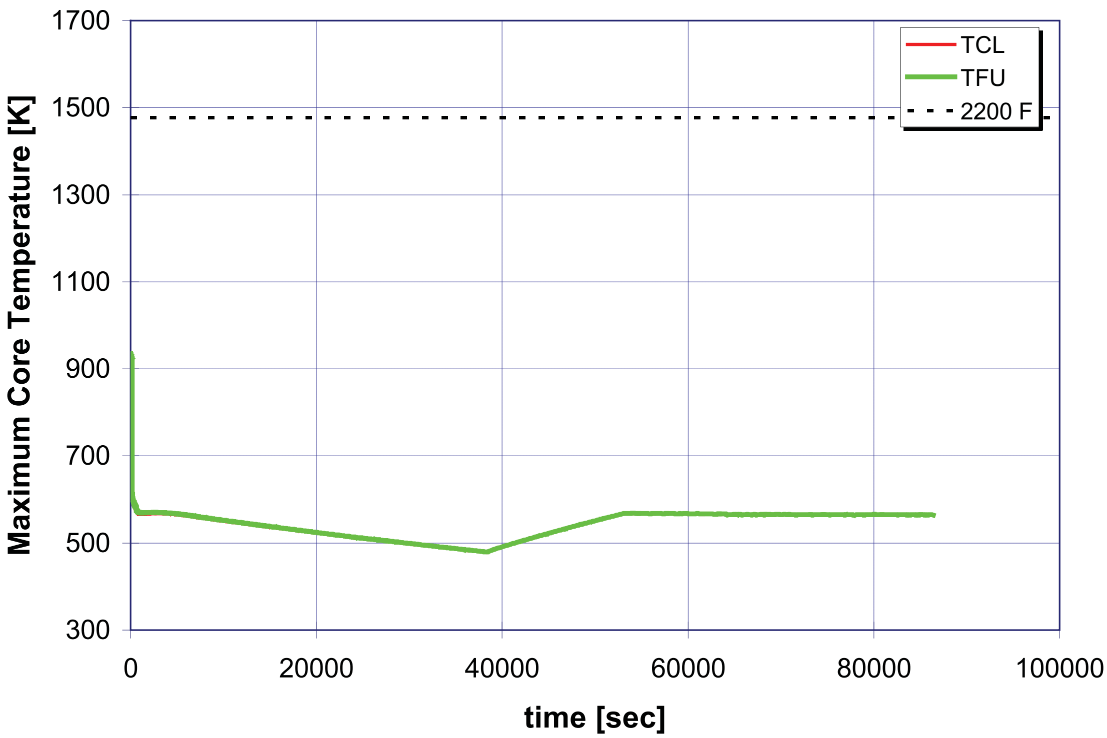



450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
20000
40000
60000
80000
100000
time [sec]
RWST
SPR
LHSI
HHSI


25
20
15
10
5
0
Flow Rate (kg/s)
0
20000
40000
60000
80000
100000
time [sec]
MFW (SG-A)
MFW (SG-B)
MFW (SG-C)
TD-AFW (SG-A)
TD-AFW (SG-B)
TD-AFW (SG-C)
MD-AFW (SG-A)
MD-AFW (SG-B)
MD-AFW (SG-C)

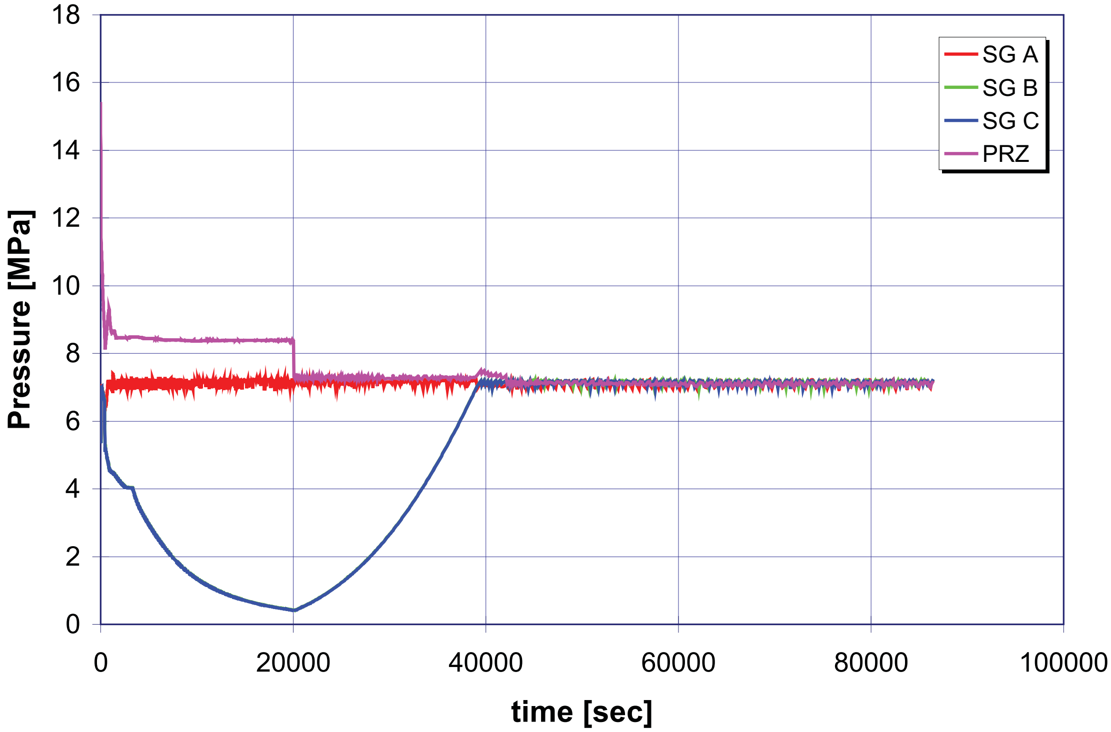


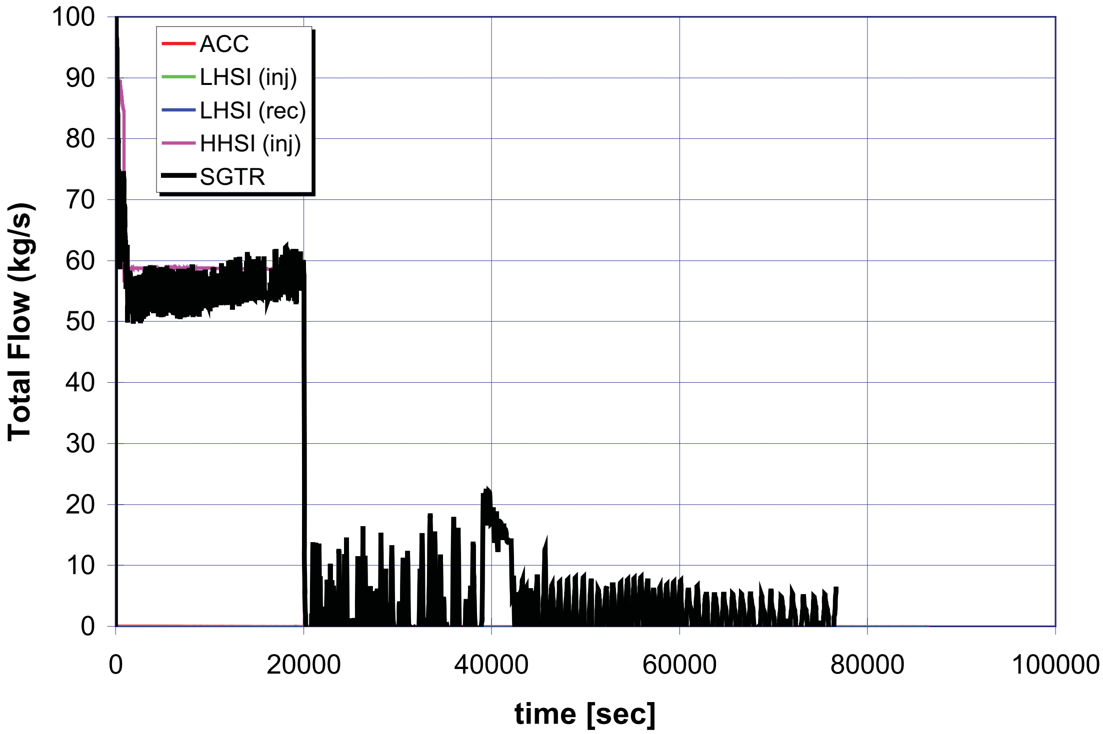
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
25
20
15
10
5
0
0
20000
40000
60000
80000
100000

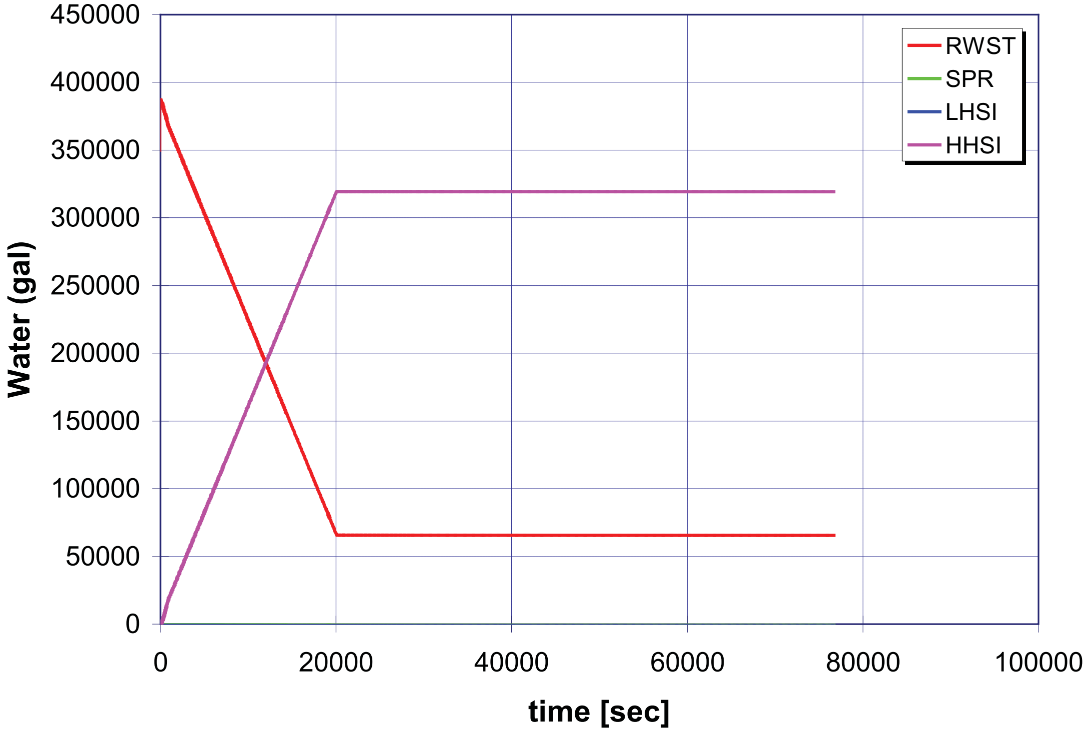


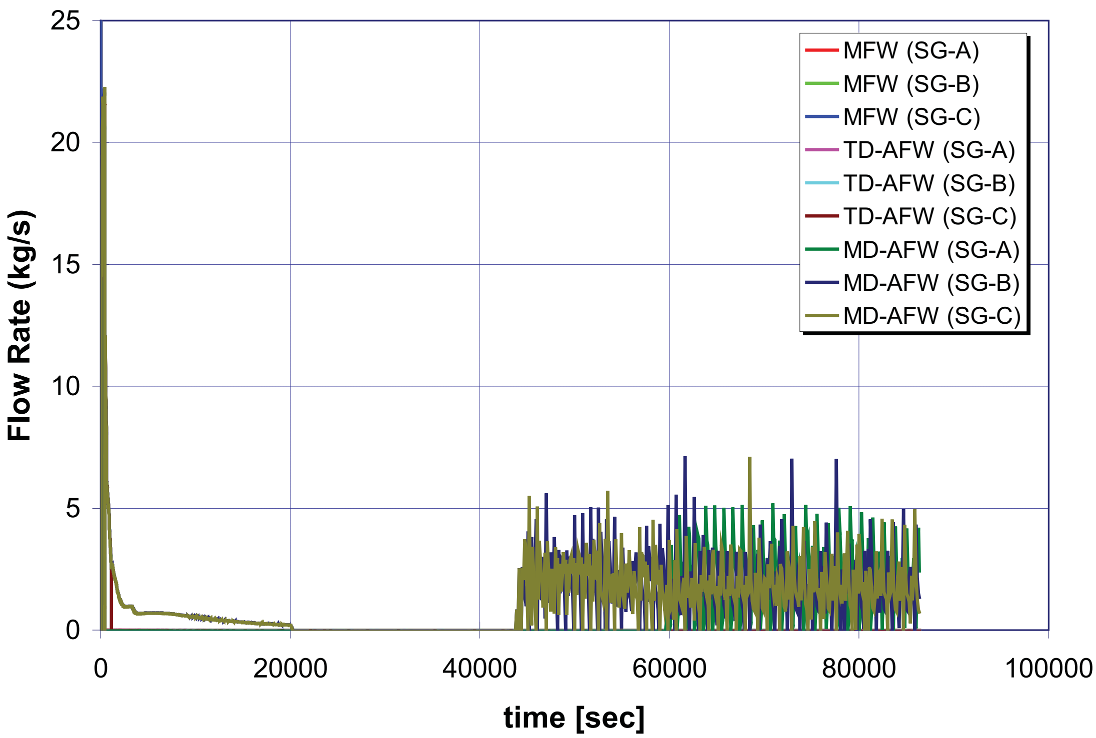
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
1700
1500
1300
1100
900
700
500
300
0
20000
40000
60000
80000
100000

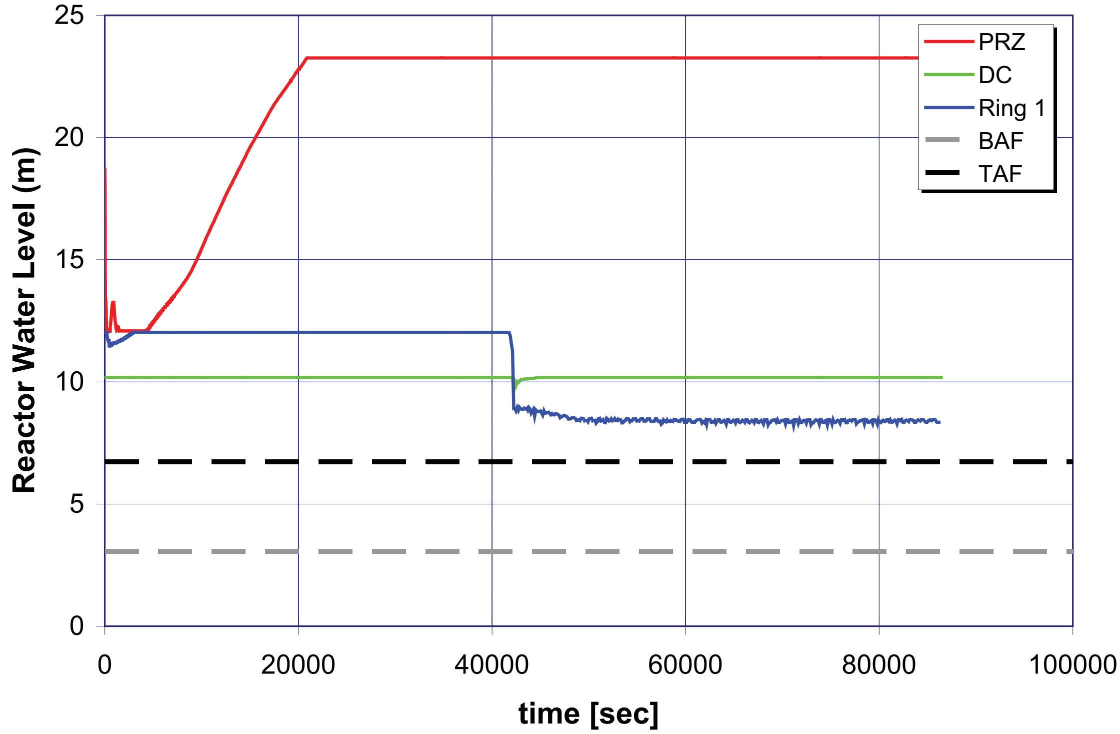
A.4.2 Case 2: Five Tubes and Secure One HHSI Pump



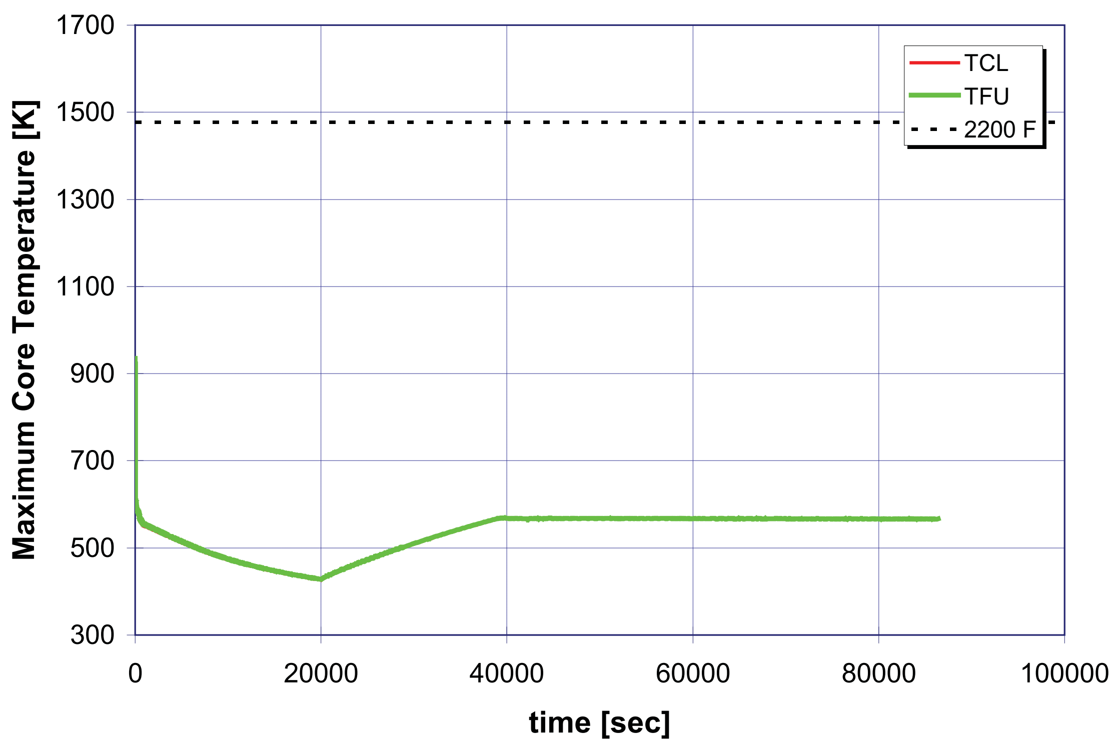



450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
20000
40000
60000
80000
100000
time [sec]
RWST
SPR
LHSI
HHSI


25
20
15
10
5
0
Flow Rate (kg/s)
0
20000
40000
60000
80000
100000
time [sec]
MFW (SG-A)
MFW (SG-B)
MFW (SG-C)
TD-AFW (SG-A)
TD-AFW (SG-B)
TD-AFW (SG-C)
MD-AFW (SG-A)
MD-AFW (SG-B)
MD-AFW (SG-C)

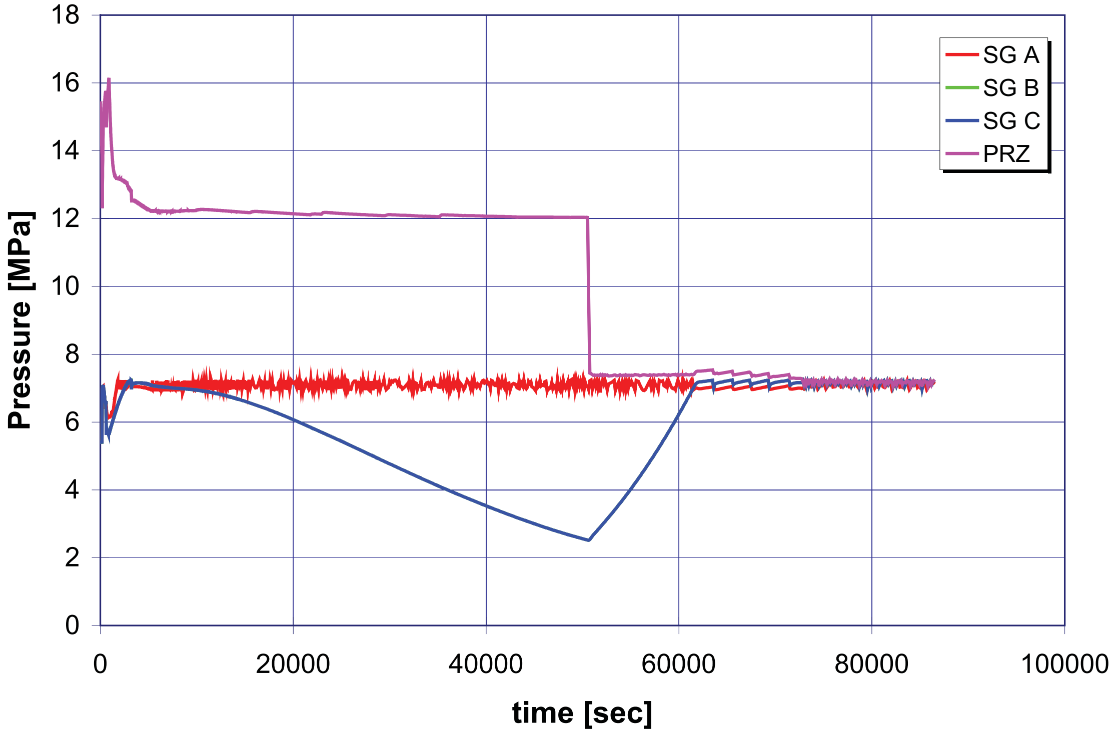


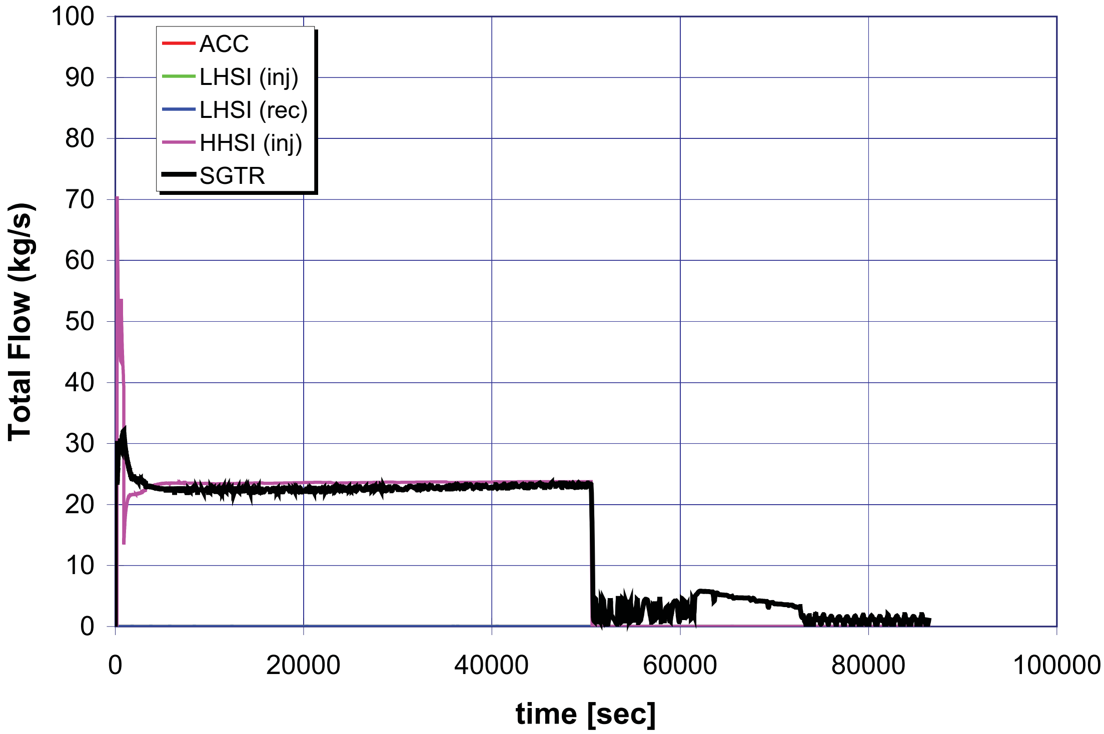
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
5
10
15
20
25
0
20000
40000
60000
80000
100000

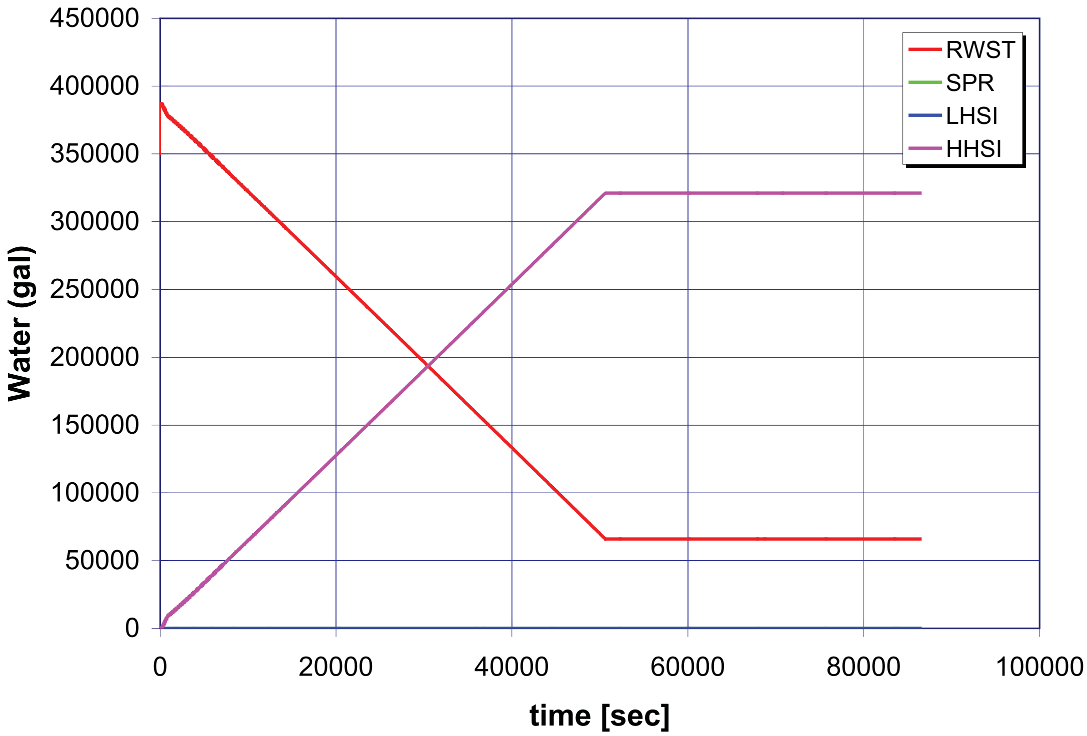


TCL
TFU
2200 F
Maximum Core Temperature [K]
time [sec]
300
500
700
900
1100
1300
1500
1700
0
20000
40000
60000
80000
100000

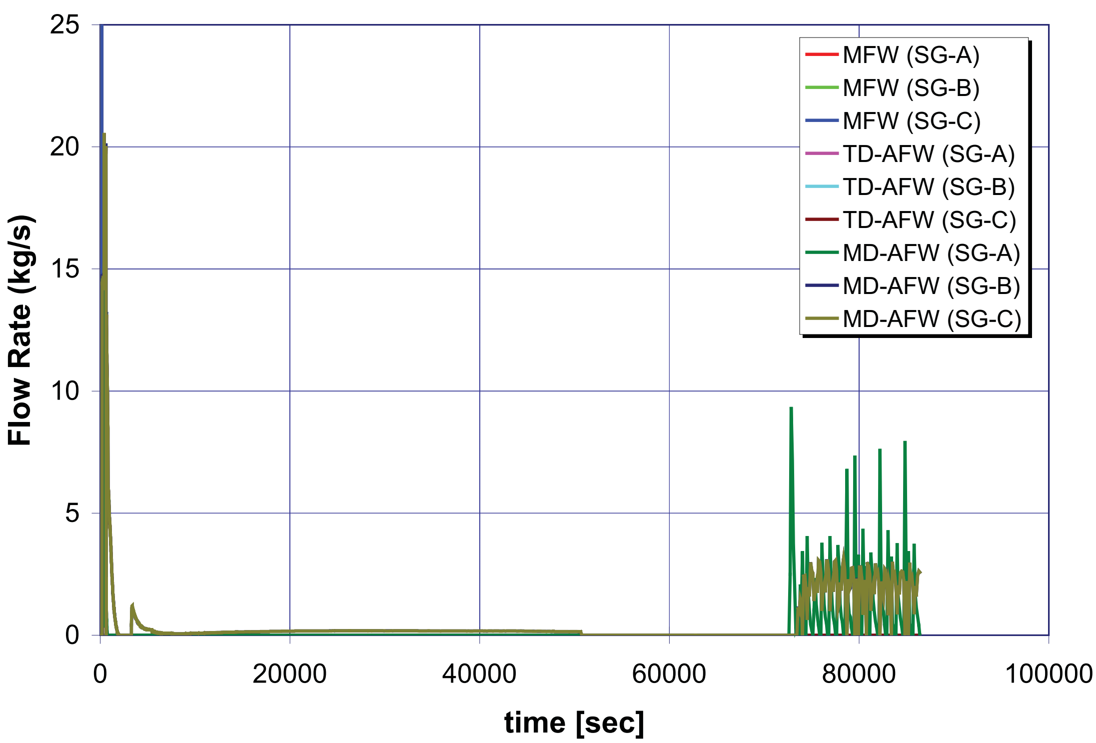
A.4.3 Case 3: One Tube and Secure Two HHSI Pumps



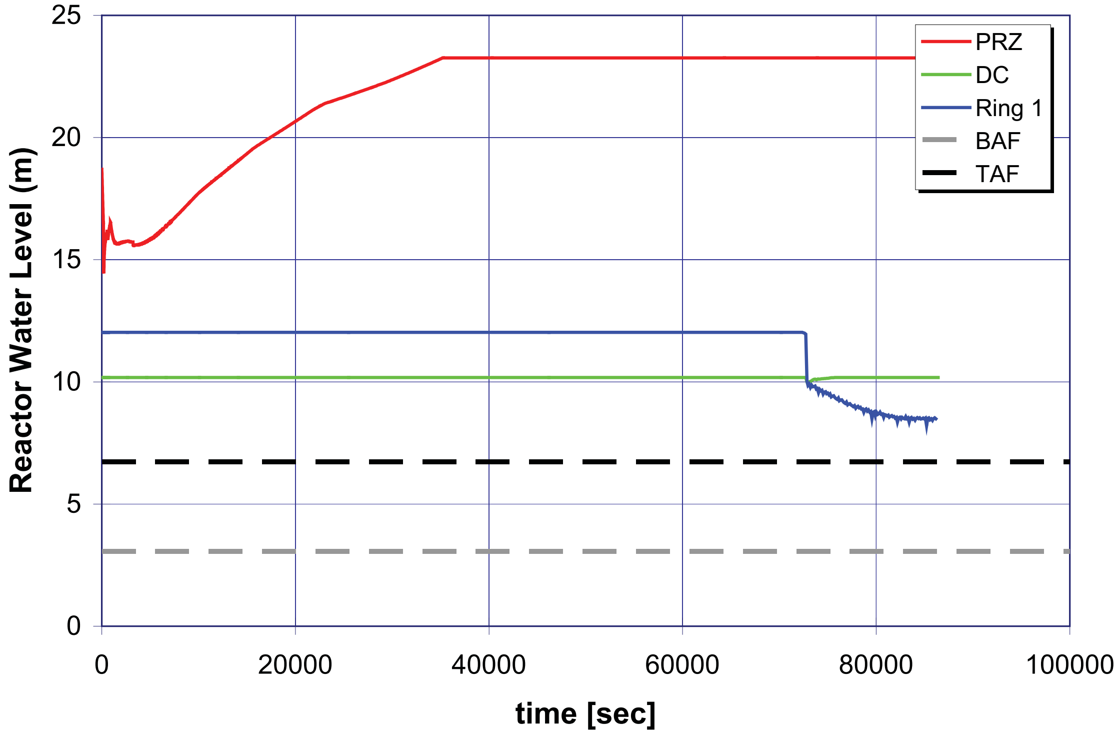


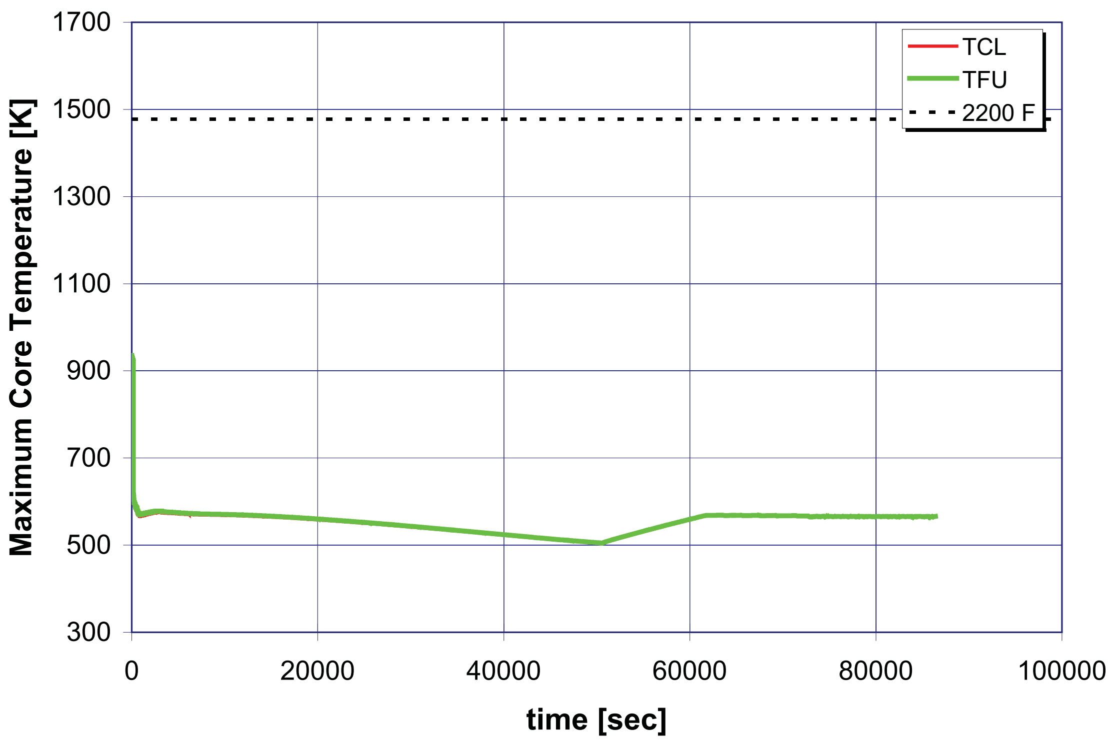


Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
20000
40000
60000
80000
100000

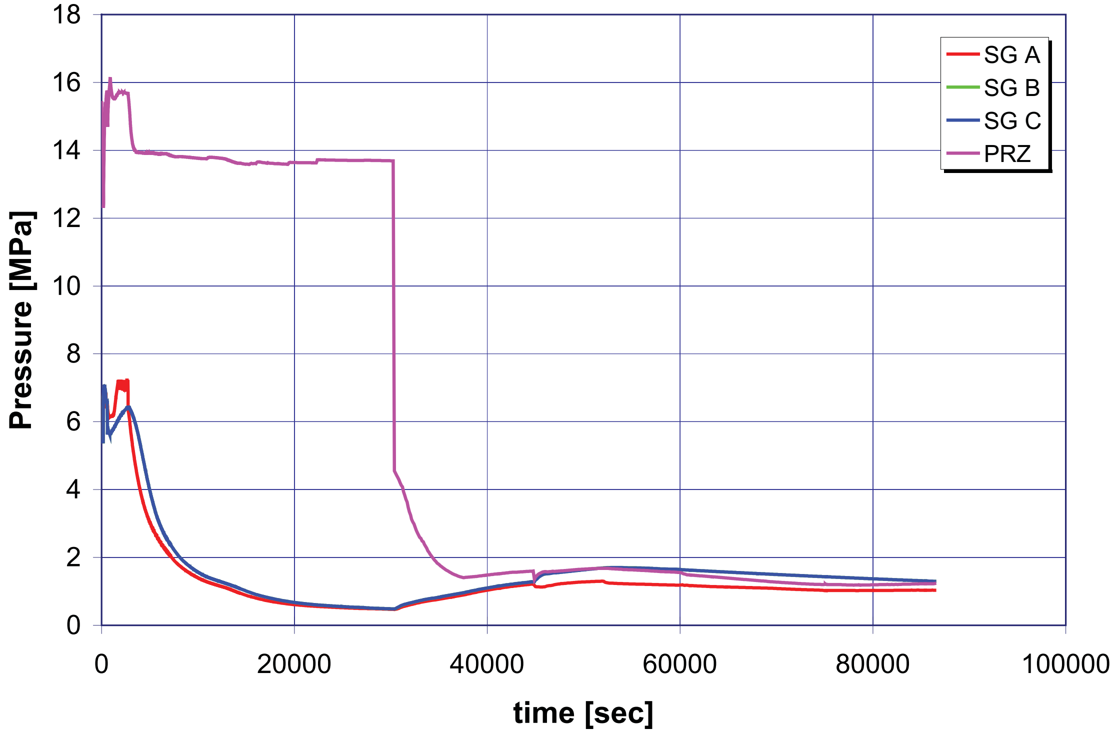


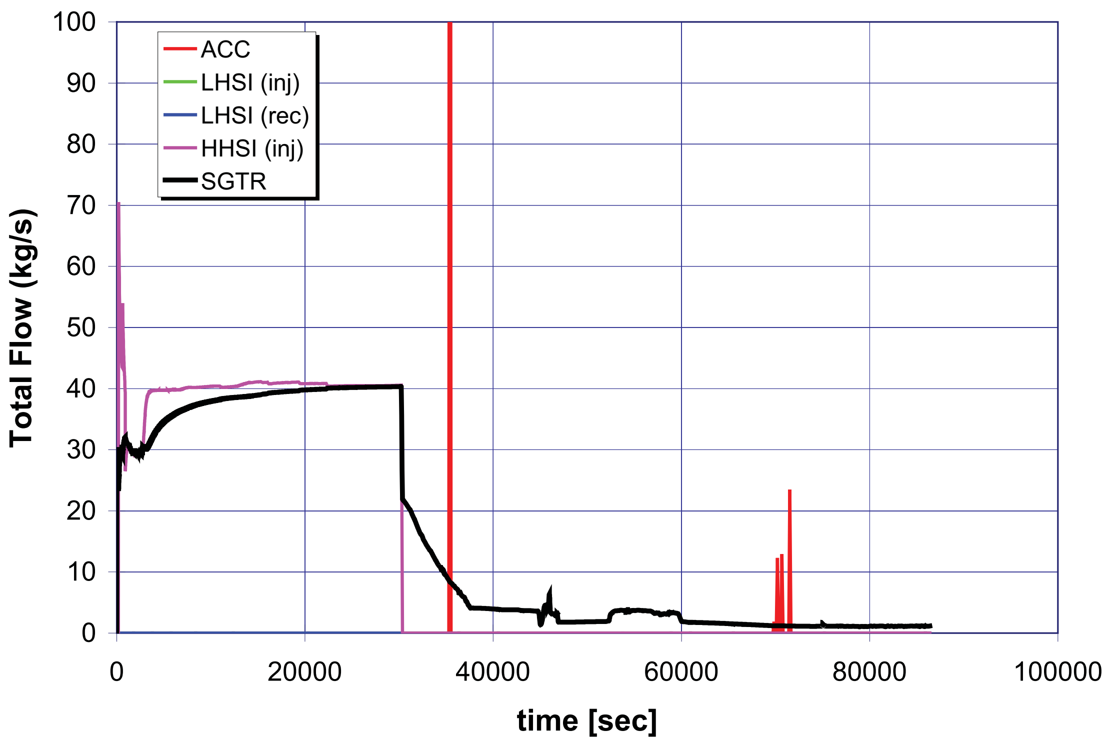
Flow Rate (kg/s)
time [sec]
MFW (SG-A)
MFW (SG-B)
MFW (SG-C)
TD-AFW (SG-A)
TD-AFW (SG-B)
TD-AFW (SG-C)
MD-AFW (SG-A)
MD-AFW (SG-B)
MD-AFW (SG-C)
25
20
15
10
5
0
20000
40000
60000
80000
100000

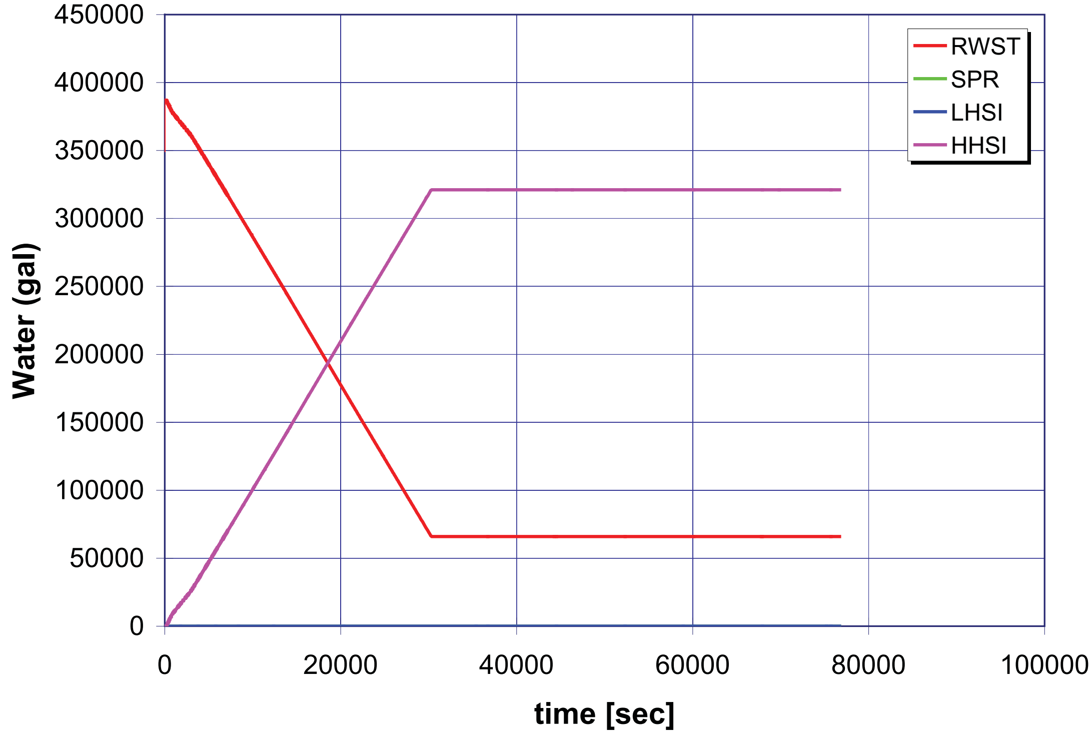

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
5
10
15
20
25
0
20000
40000
60000
80000
100000


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
500
700
900
1100
1300
1500
1700
0
20000
40000
60000
80000
100000

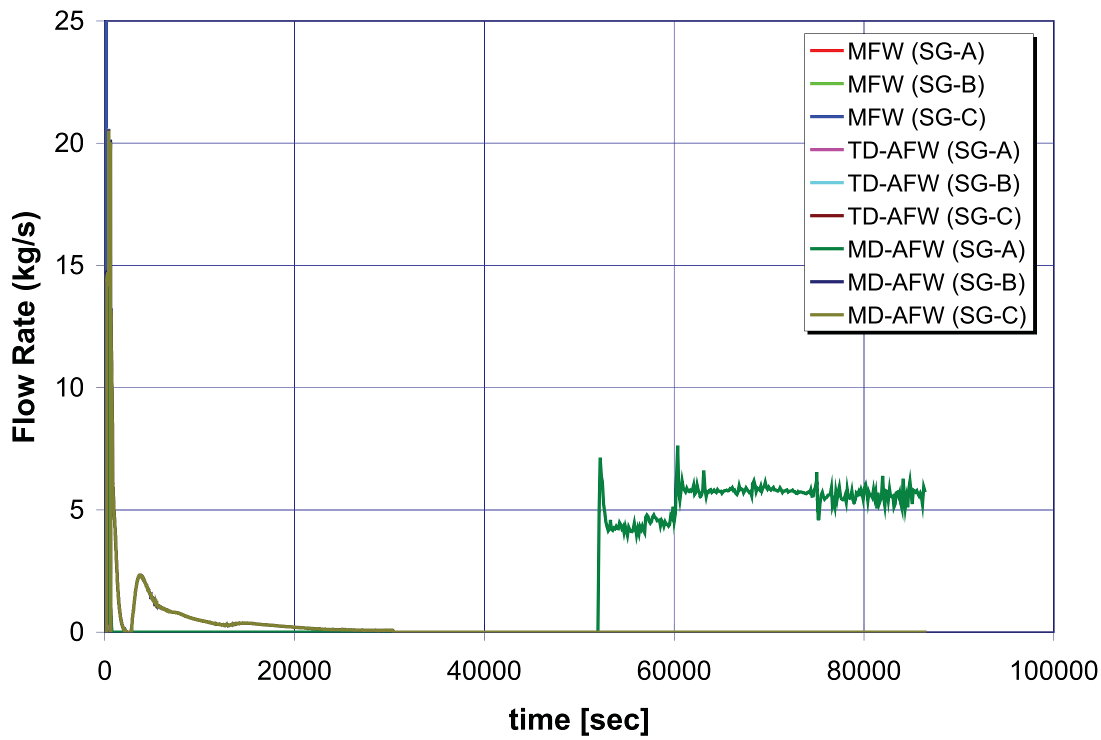
A.4.4 Case 4: One Tube, Secure One HHSI Pump and Stuck-Open SG PORV



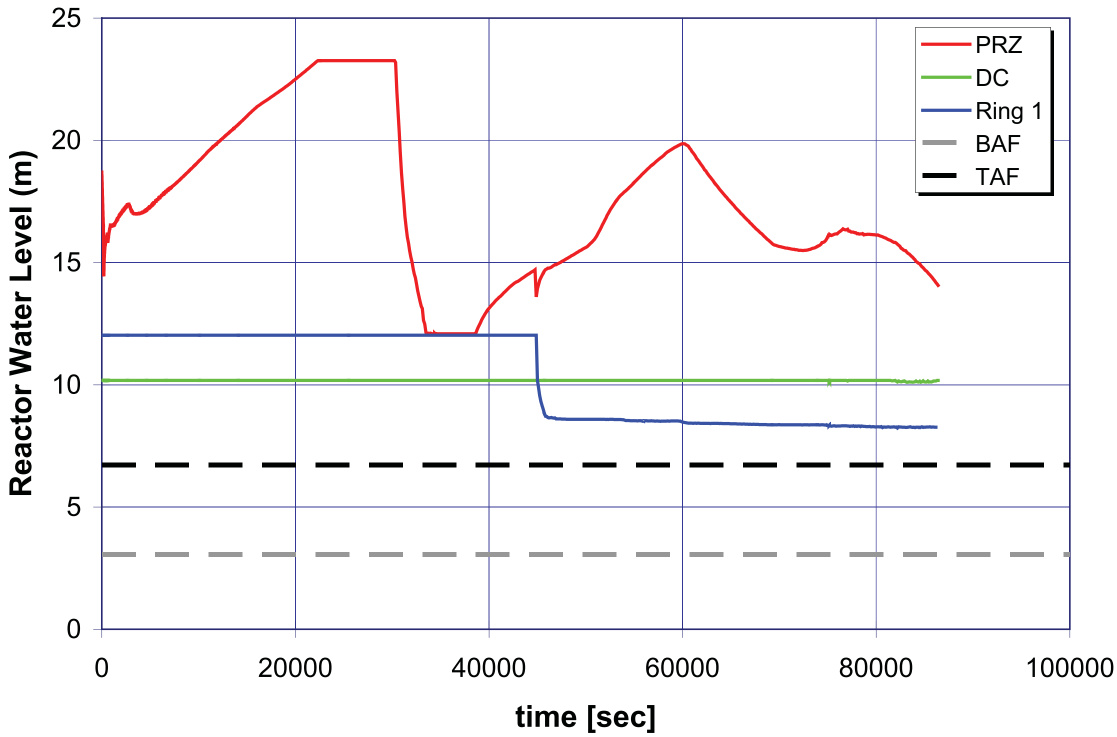


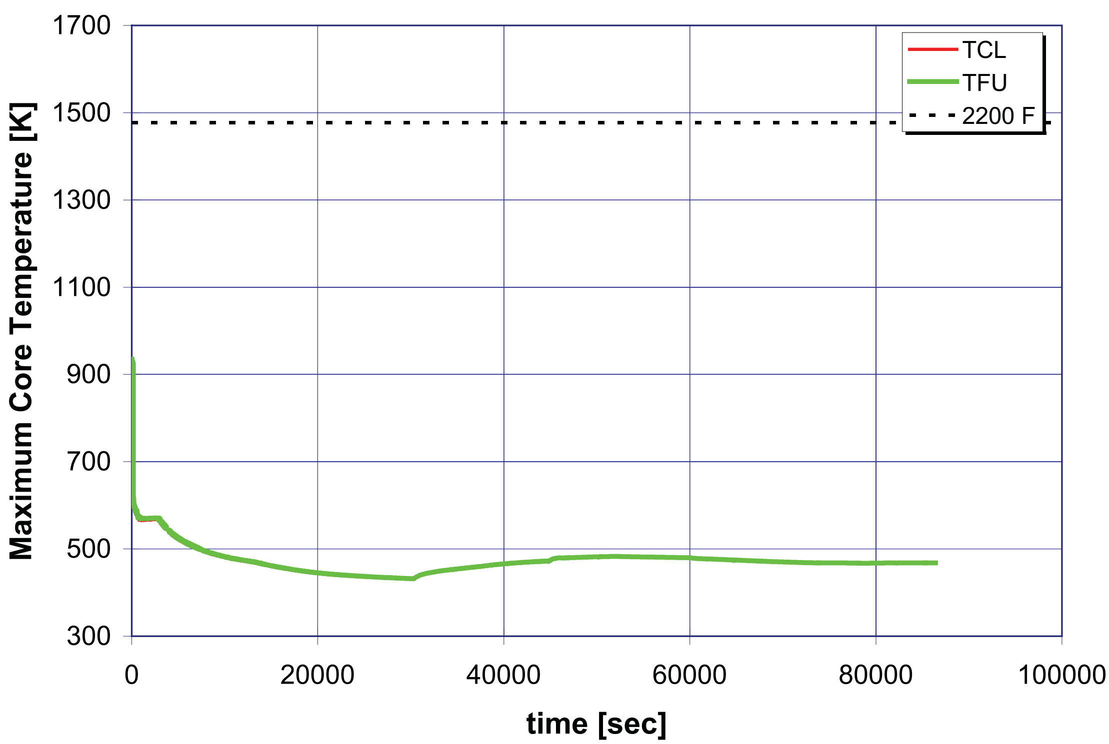


450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
20000
40000
60000
80000
100000
time [sec]
RWST
SPR
LHSI
HHSI

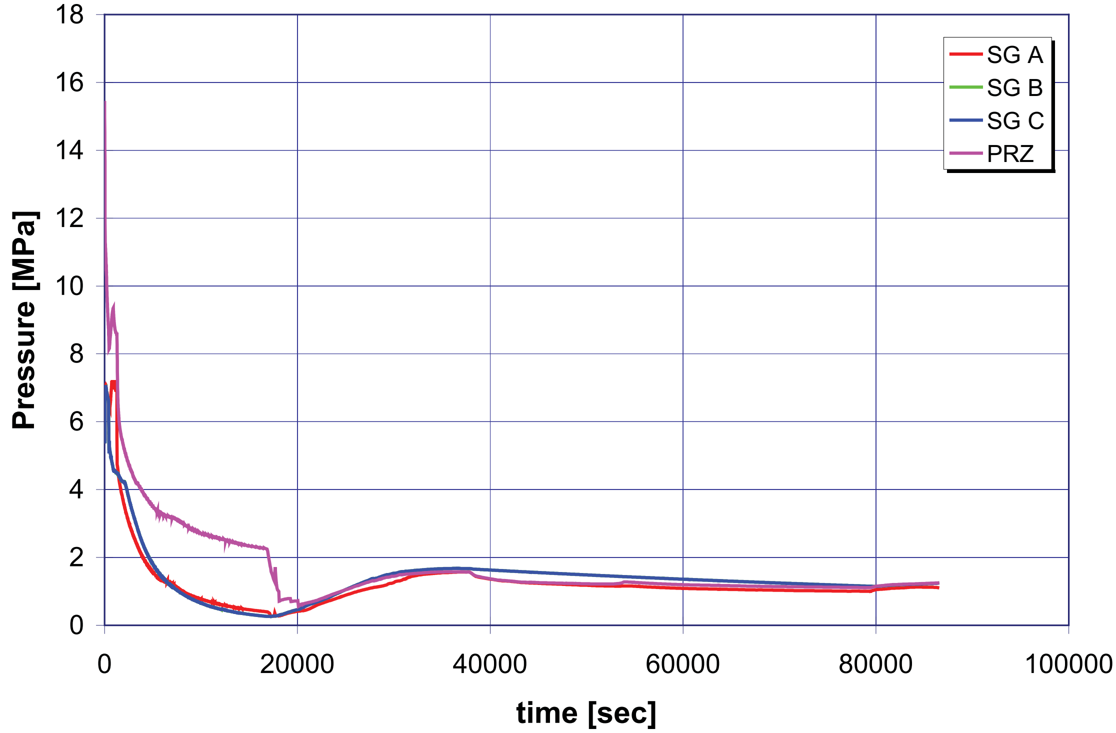


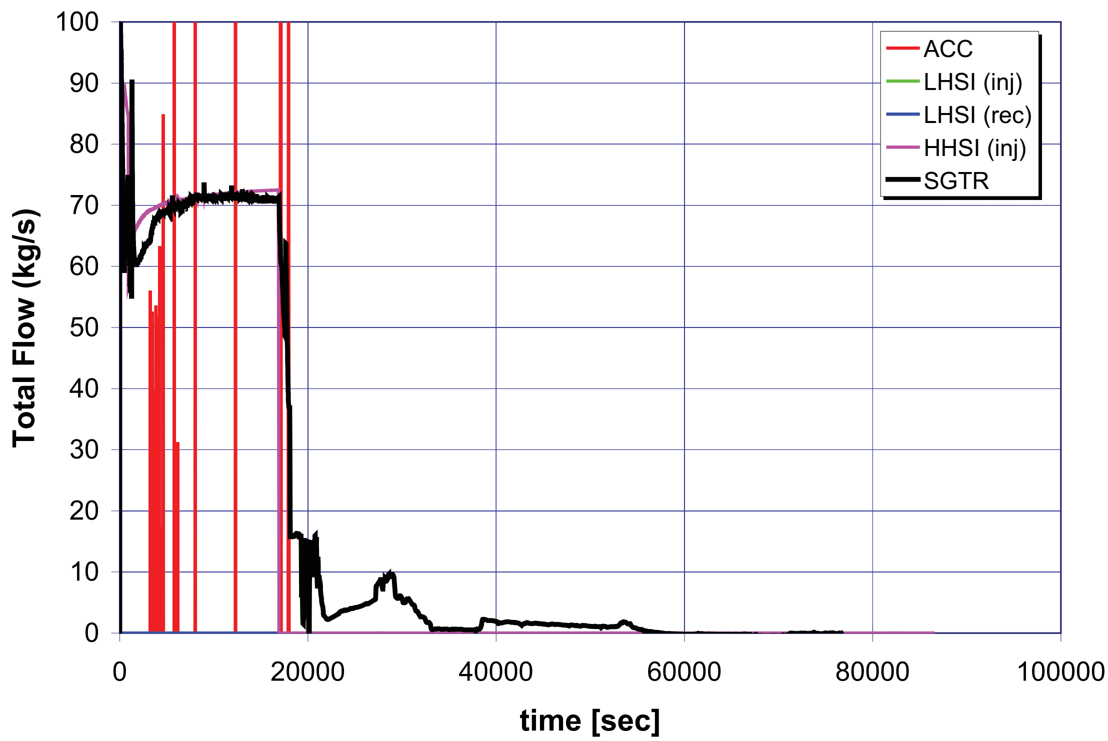
25
20
15
10
5
0
Flow Rate (kg/s)
0
20000
40000
60000
80000
100000
time [sec]
MFW (SG-A)
MFW (SG-B)
MFW (SG-C)
TD-AFW (SG-A)
TD-AFW (SG-B)
TD-AFW (SG-C)
MD-AFW (SG-A)
MD-AFW (SG-B)
MD-AFW (SG-C)

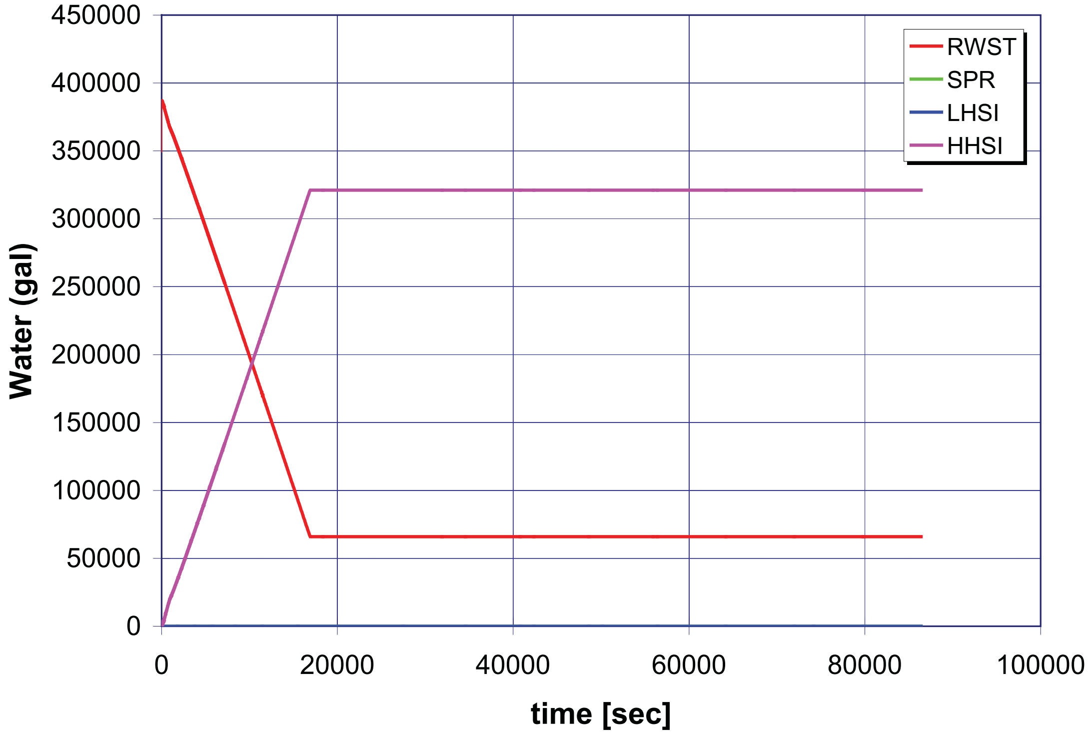

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
5
10
15
20
25
0
20000
40000
60000
80000
100000


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
500
700
900
1100
1300
1500
1700
0
20000
40000
60000
80000
100000

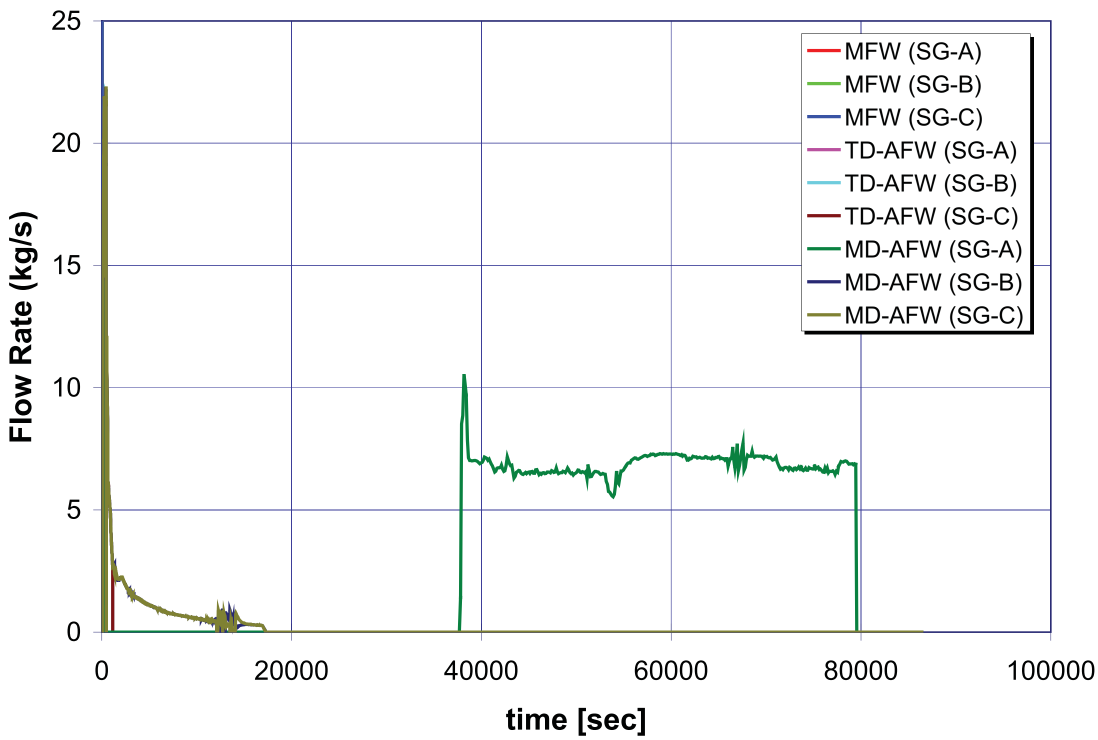
A.4.5 Case 5: Five Tubes, Secure One HHSI Pump, and Stuck-Open SG PORV



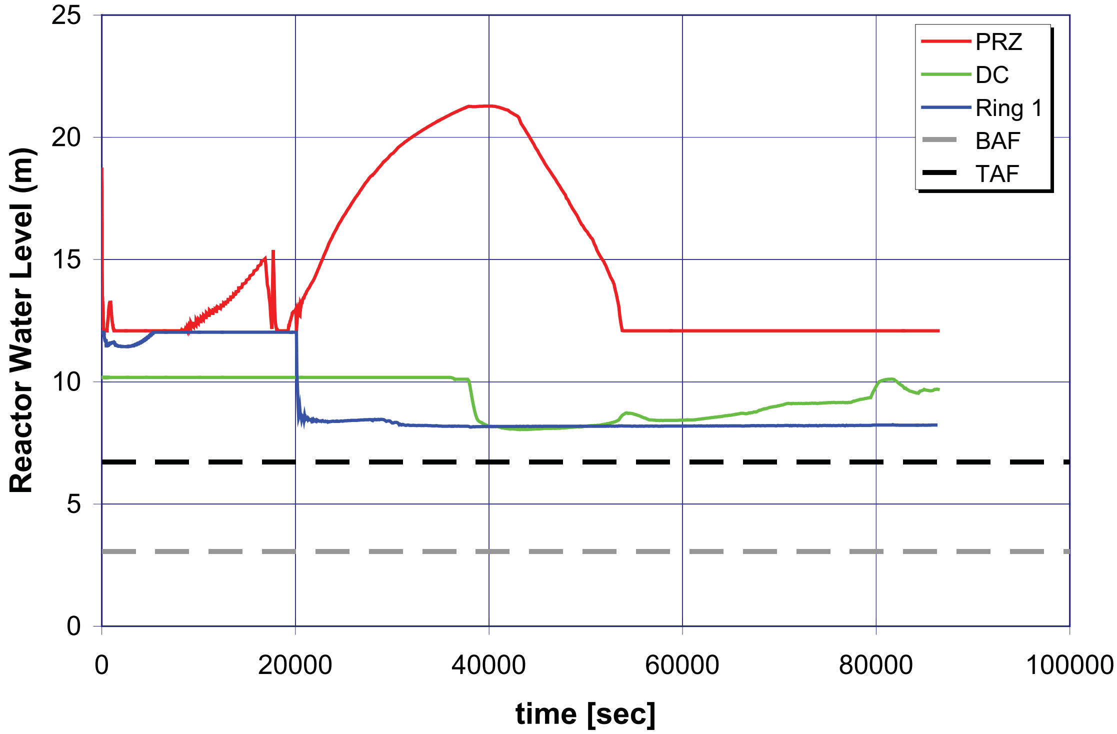


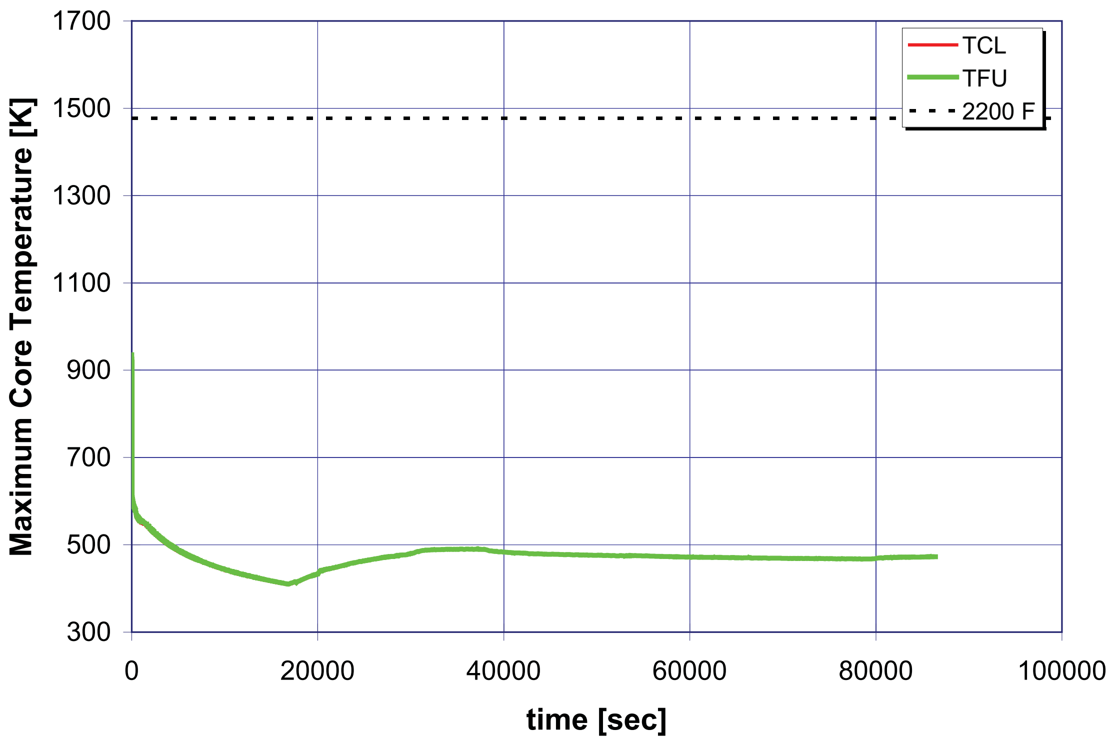


450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
20000
40000
60000
80000
100000
time [sec]
RWST
SPR
LHSI
HHSI



25
20
15
10
5
0
Flow Rate (kg/s)
0
20000
40000
60000
80000
100000
time [sec]
MFW (SG-A)
MFW (SG-B)
MFW (SG-C)
TD-AFW (SG-A)
TD-AFW (SG-B)
TD-AFW (SG-C)
MD-AFW (SG-A)
MD-AFW (SG-B)
MD-AFW (SG-C)


Reactor Water Level (m)
25
20
15
10
5
0
0
20000
40000
60000
80000
100000
time [sec]
PRZ
DC
Ring 1
BAF
TAF


Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
20000
40000
60000
80000
100000
time [sec]
TCL
TFU
2200 F

A.5 Pressurized-Water Reactor Station Blackout

Analysis Summary

The station blackout sequence is similar to the SOARCA analysis. In all cases, there is 21 gpm existing leakage, but in some cases seal failure at 13 minutes leads to either 182 gpm or 500 gpm leakage. Note that this is different from how RCP seal failure is modeled in the SOARCA project. For the modeling of stuck-open pressurizer SRV, there are two choices: (1) SRV sticks open based on number of cycles or (2) the valve does not recluse after the first lift-off. Note that none of the cases reach the 256-lift criterion before core damage.

Section A.5.1 reports two sensitivity calculations: (1) initiation of three HHSI pumps at 2.14 hours (when core damage occurs) and (2) initiation of three HHSI pumps at 1.64 hours (half an hour earlier). Section A.5.1.1 shows that core damage would continue for the former case, while Section A.5.1.2 shows that there is sufficient time and injection flow rate to avert fuel melting and arrest core heatup in the latter case. Table 12 through Table 15 below provides results for this portion of the analysis.

Table 12 Surry Station Blackout Results

<table>
<tr><th>Case</th><th>Seal Leakage Rate[1] after Failure (gpm[3] per pump)</th><th>Seal Failure Time (min)</th><th>SRV Stuck Open</th><th>TD-AFW</th><th>ac/dc</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td rowspan="4">500</td><td rowspan="4">13</td><td rowspan="6">N/A[2]</td><td rowspan="3">Fails to start</td><td>-</td><td>1.4</td><td>2.1</td></tr>
<tr><td>1a</td><td>ac recovery at 2.1 hours</td><td>1.4</td><td>2.1</td></tr>
<tr><td>1b</td><td>ac recovery at 1.6 hours</td><td>1.4</td><td>-</td></tr>
<tr><td>2</td><td>Available</td><td rowspan="7">-</td><td>1.6</td><td>2.3</td></tr>
<tr><td>3</td><td rowspan="4">21</td><td rowspan="4">-</td><td>Fails to start</td><td>2.3</td><td>3.4</td></tr>
<tr><td>4</td><td>Available; successful blind feeding</td><td>13.3</td><td>16.3</td></tr>
<tr><td>5</td><td rowspan="2">1st lift</td><td>Fails to start</td><td>2.1</td><td>2.6</td></tr>
<tr><td>6</td><td>Available; successful blind feeding</td><td>13.0</td><td>13.8</td></tr>
<tr><td>7</td><td rowspan="2">182</td><td rowspan="2">13</td><td rowspan="3">N/A[2]</td><td>Fails to start</td><td>2.0</td><td>3.1</td></tr>
<tr><td>8</td><td>Available</td><td>3.9</td><td>4.8</td></tr>
<tr><td>9</td><td rowspan="2">21</td><td rowspan="2">-</td><td rowspan="2">Available; lost at 4 hours</td><td rowspan="2">dc lost at 4 hours</td><td>8.4</td><td>10.9</td></tr>
<tr><td>10</td><td>1st lift</td><td>8.1</td><td>8.8</td></tr>
</table>

1 The leakage rate provided is the leakage rate at full system pressure. As the system depressurizes, the leak rate decreases.

2 The model is set to stick the valve open after 256 lifts, but the valve does not lift that many times for these calculations.

3 500 gpm = 1.89 m^3/min;182 gpm = 0.689 m^3/min; 21 gpm = 0.076 m^3/min.

Table 13 Surry Station Blackout Key Timings (Cases 1–2)

Event[1]	Case 1 (hr)	Case 1a (hr)	Case 1b (hr)	Case 2 (hr)
Reactor trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0	0	0
Seal leakage (21 gpm/pump)	0	0	0	0
Seal failure (500 gpm/pump)	0.22	0.22	0.22	0.22
Primary-side SG tubes water level starts to decrease	0.52	0.52	0.52	0.52
Primary-side SG tubes dry	0.96	0.96	0.96	0.98
SG dryout	1.16	1.16	1.16	-
Core uncovery	1.40	1.40	1.40	1.63
Gap release	1.92	1.92	-	2.15
Core damage (max. temp. >2,200 °F)	2.14	2.14	-	2.25

[1] 500 gpm = 1.89 m^3/min; 21 gpm = 0.076 m^3/min; 2,200 °F = 1,204 °C.

Table 14 Surry Station Blackout Key Timings (Cases 3–6)

Event[1]	Case 3 (hr)	Case 4 (hr)	Case 5 (hr)	Case 6 (hr)
Reactor trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0	0	0
Seal leakage (21 gpm/pump)	0	0	0	0
Primary-side SG tubes water level starts to decrease	1.92	5.38	1.52	5.42
Emergency CST depleted	-	7.97	-	7.97
Primary-side SG tubes dry	2.03	11.30	1.66	11.30
SG dryout	1.19	11.77	1.19	11.80
SRV sticks open	N/A	N/A	1.45	12.71
Core uncovery	2.28	13.31	2.06	13.03
Gap release	2.96	14.83	2.42	13.60
Core damage (max. temp. >2,200 °F)	3.40	16.33	2.57	13.80

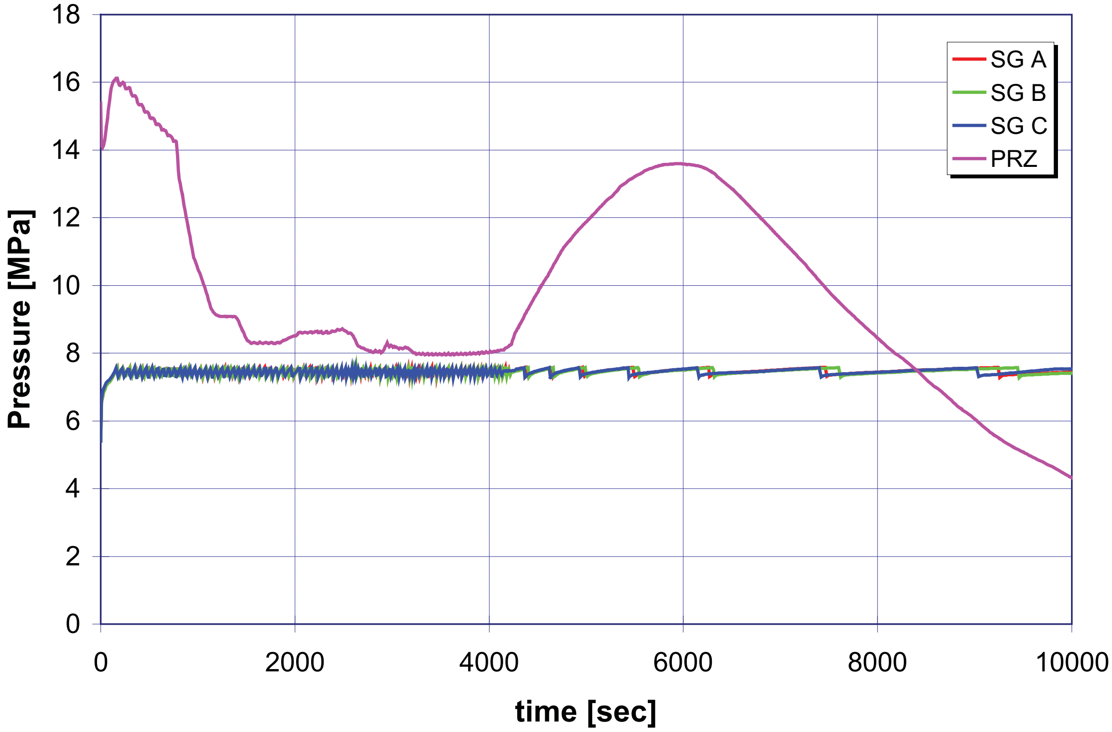
[1] 21 gpm = 0.076 m^3/min; 2,200 °F = 1,204 °C.

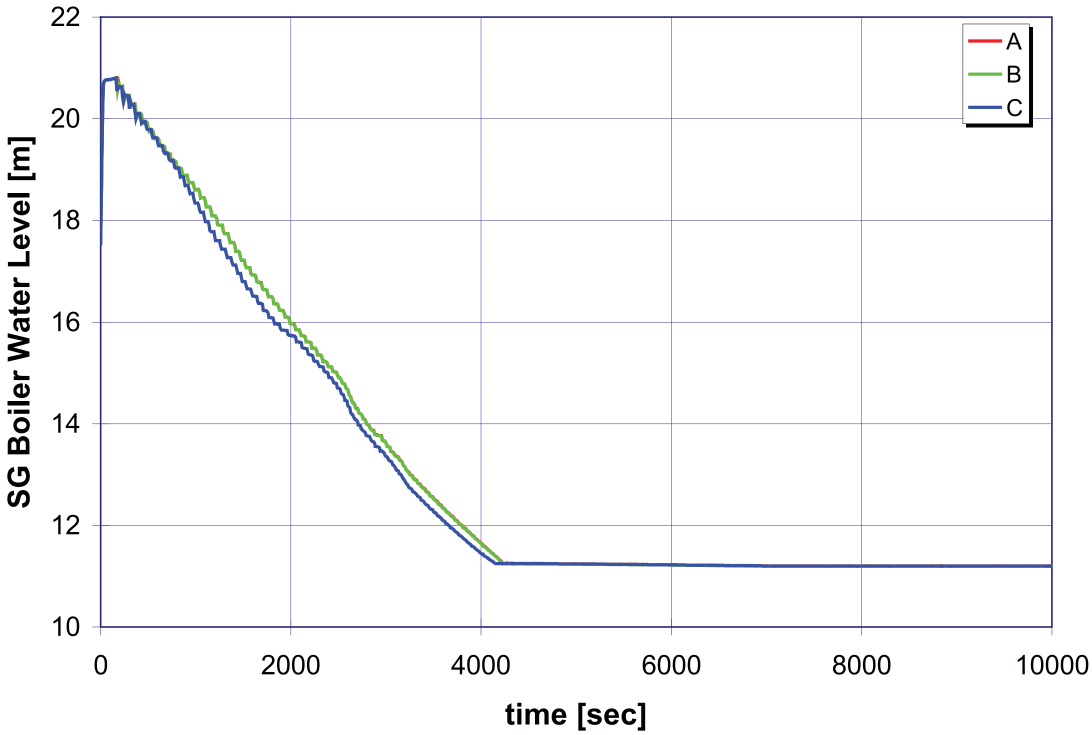
Table 15 Surry Station Blackout Key Timings (Cases 7–10)

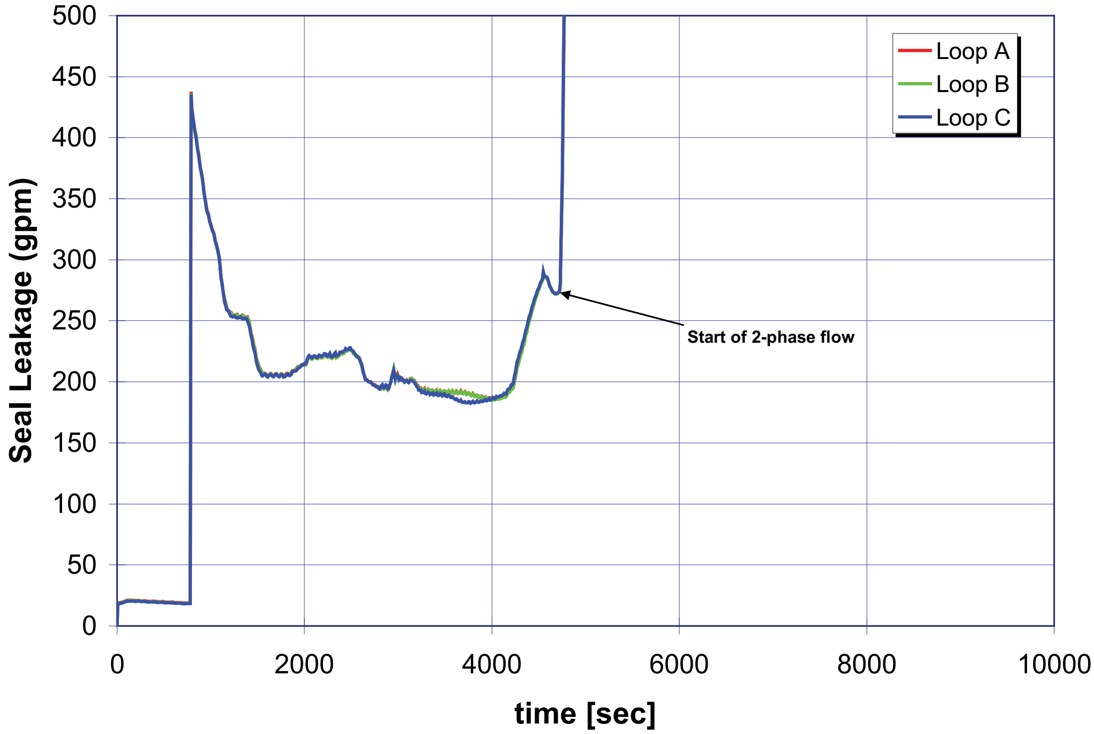
Event[1]	Case 7 (hr)	Case 8 (hr)	Case 9 (hr)	Case 10 (hr)
Reactor trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0	0	0
Seal leakage (21 gpm/pump)	0	0	0	0
Seal failure (182 gpm/pump)	0.22	0.22	-	-
TD-AFW assumed lost at battery depletion	-	-	4	4
Primary-side SG tubes water level starts to decrease	1.04	1.01	5.62	5.63
Primary-side SG tubes dry	1.52	2.22	6.58	6.58
SG dryout	1.22	-	7.13	7.12
SRV sticks open	N/A	N/A	N/A	7.67
Core uncovery	1.98	3.88	8.37	8.10
Gap release	2.63	4.00	9.48	8.59
Core damage (max. temp. >2, 200 °F)	3.09	4.77	10.85	8.77

[1] 182 gpm = 0.689 m^3/min; 21 gpm = 0.076 m^3/min; 2,200 °F = 1,204 °C.

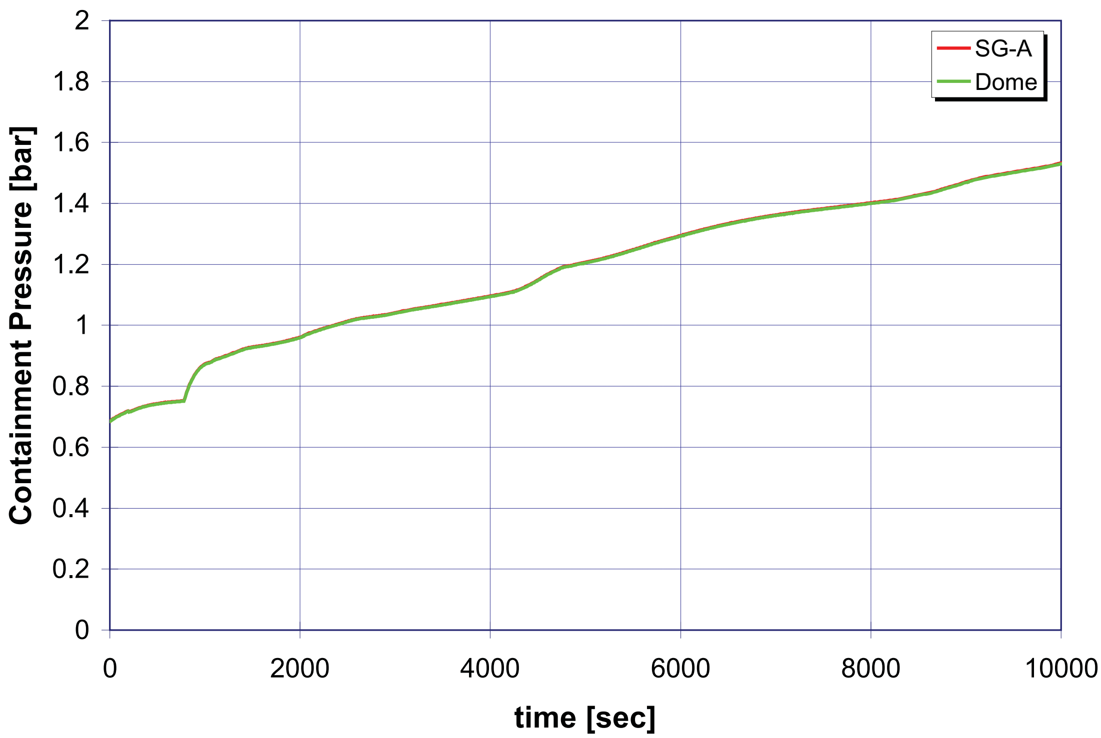
A.5.1 Case 1: Station Blackout without Turbine-Driven Auxiliary Feedwater (500 gpm)



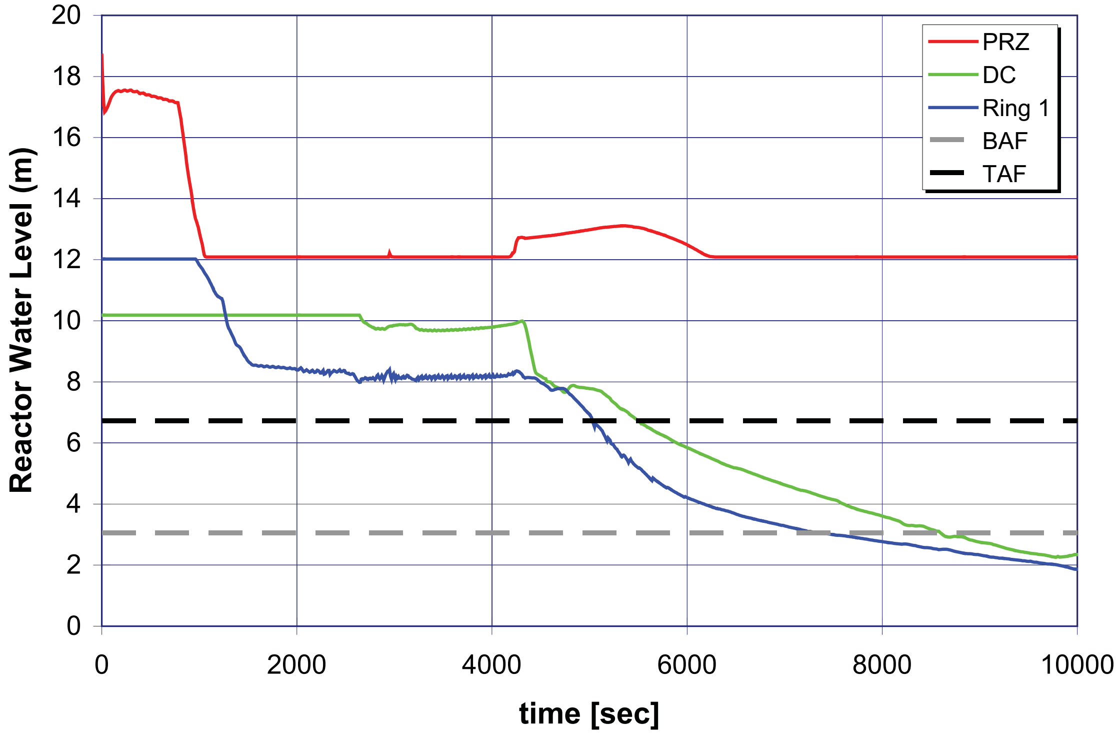




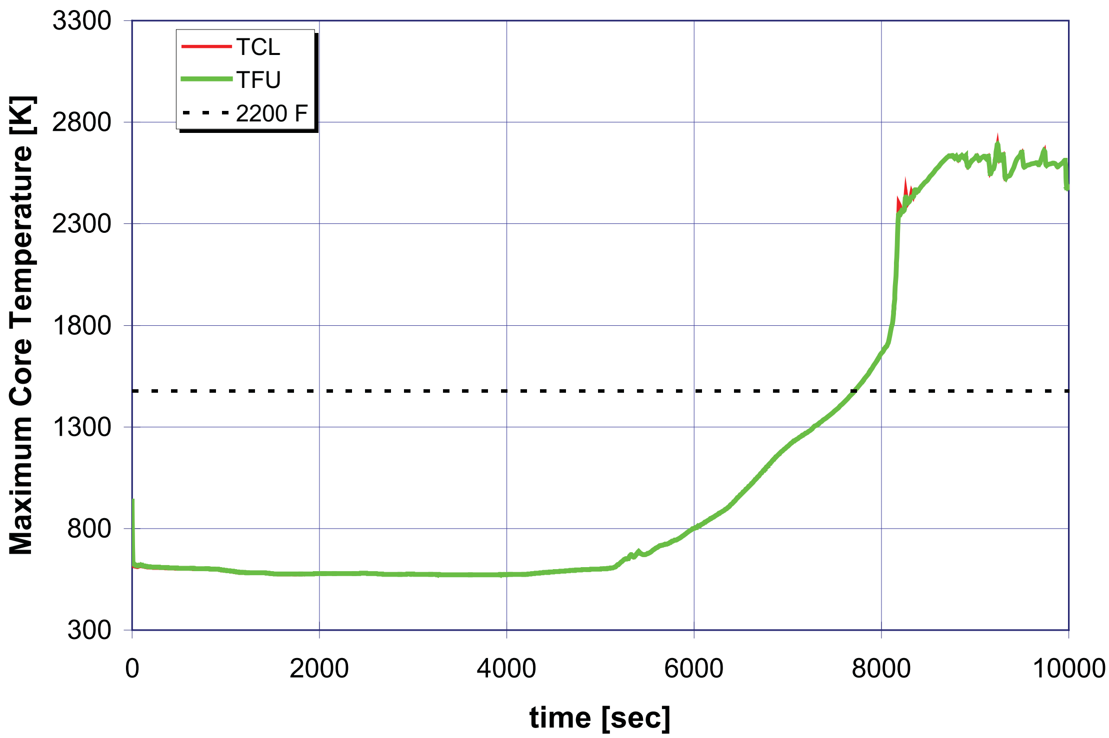
Seal Leakage (gpm)
time [sec]
Loop A
Loop B
Loop C
Start of 2-phase flow
0
50
100
150
200
250
300
350
400
450
500
0
2000
4000
6000
8000
10000



Containment Pressure [bar]
time [sec]
SG-A
Dome
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
2000
4000
6000
8000
10000

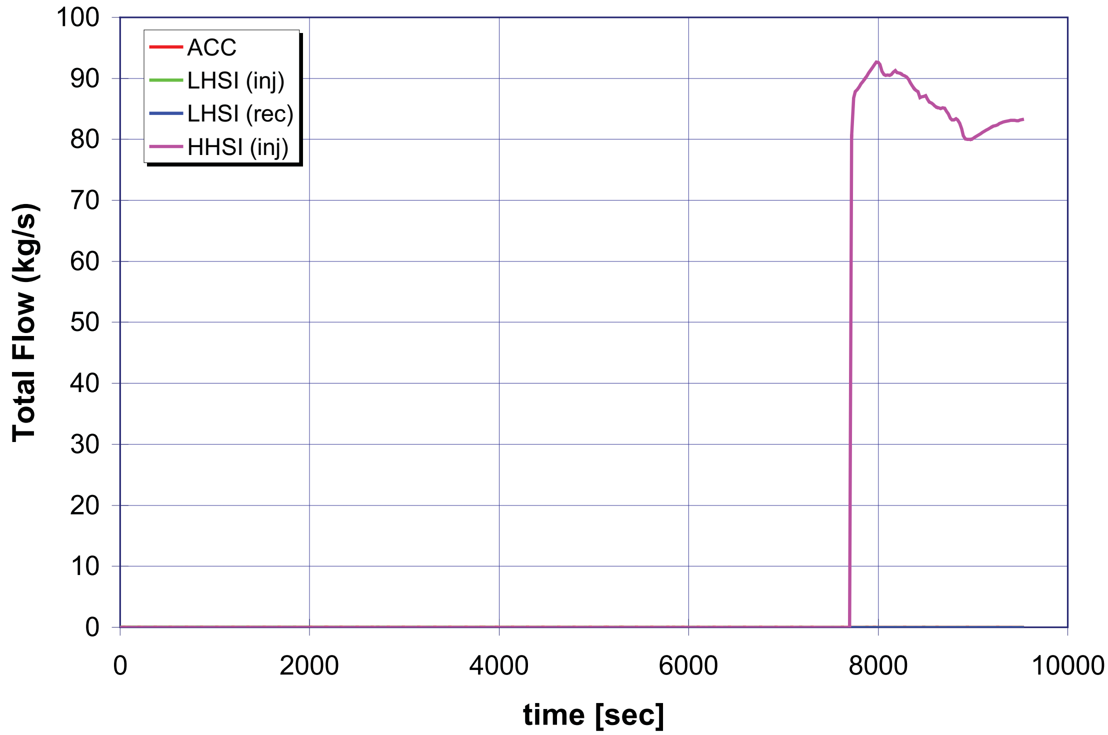


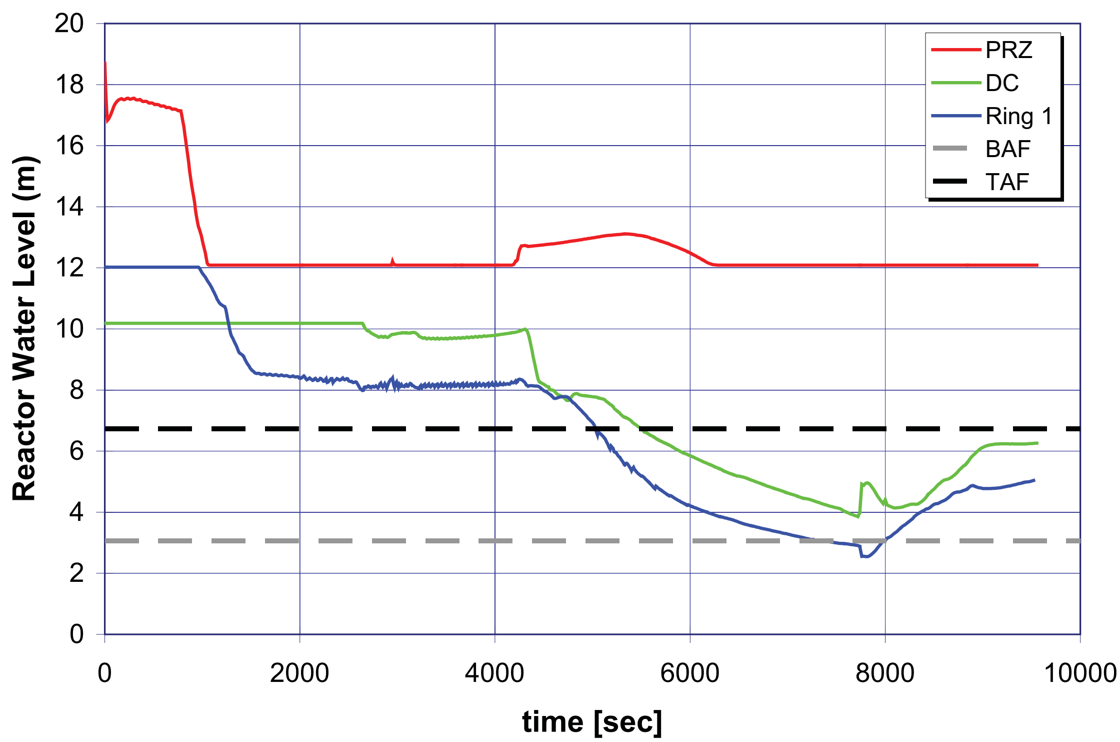
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
2000
4000
6000
8000
10000

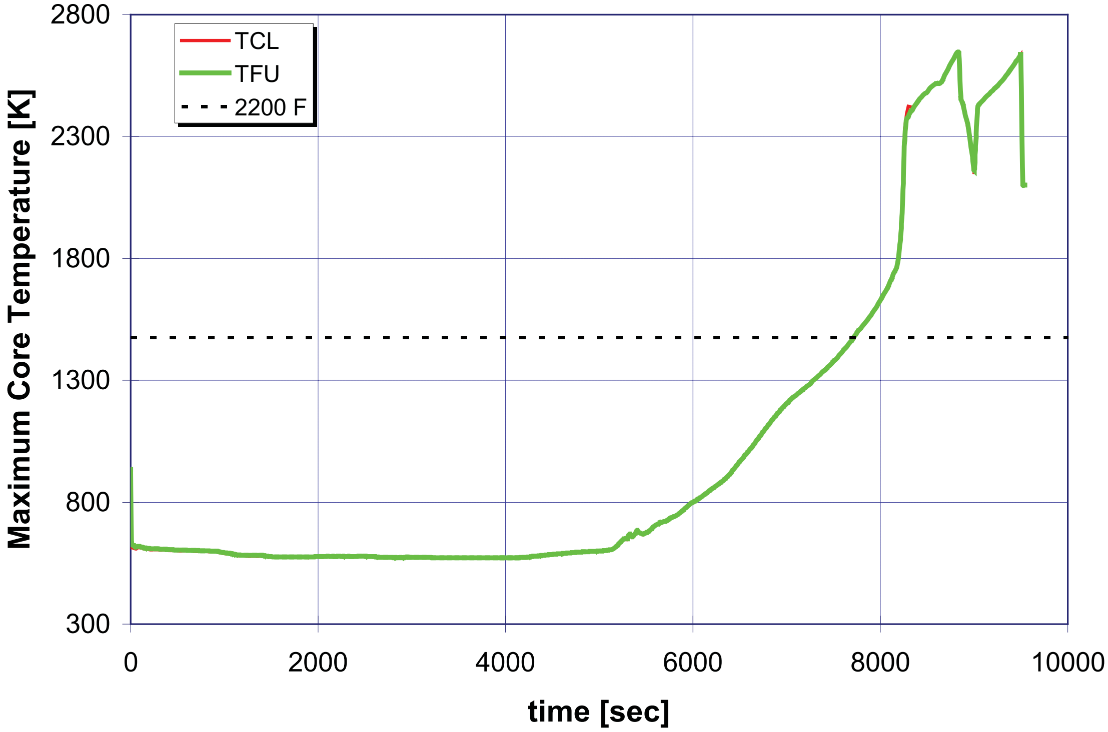


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
800
1300
1800
2300
2800
3300
0
2000
4000
6000
8000
10000

A.5.1.1 Case 1a: Three HHSI Pumps at 2.14 Hours

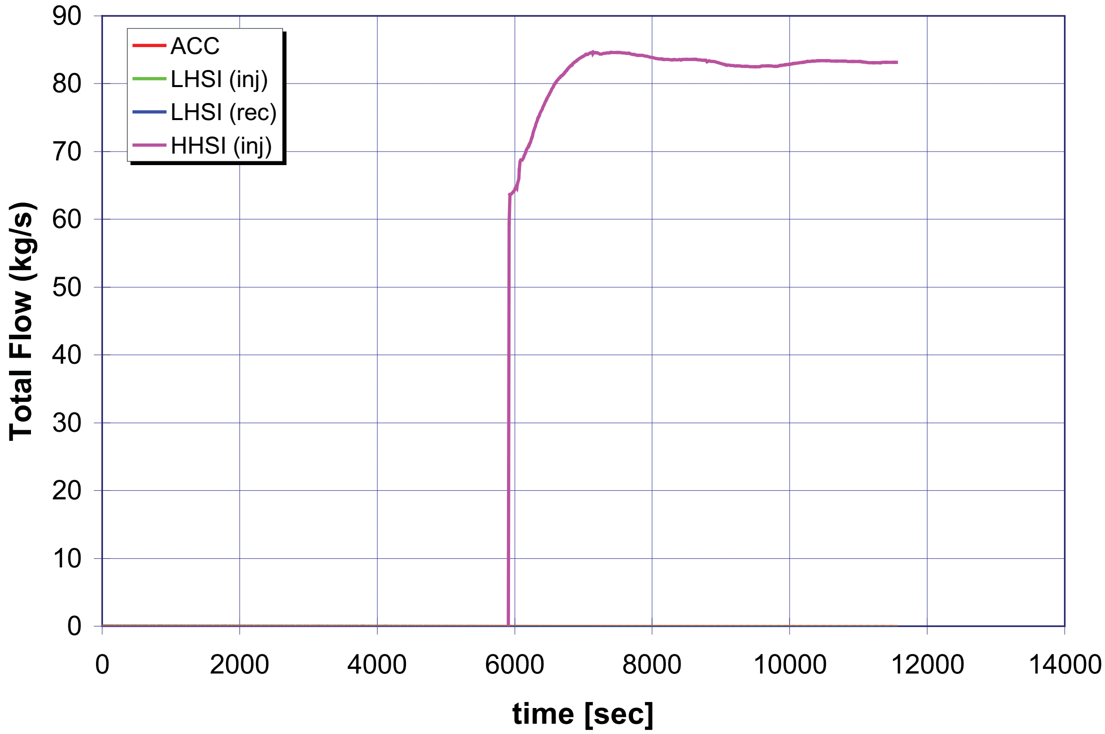


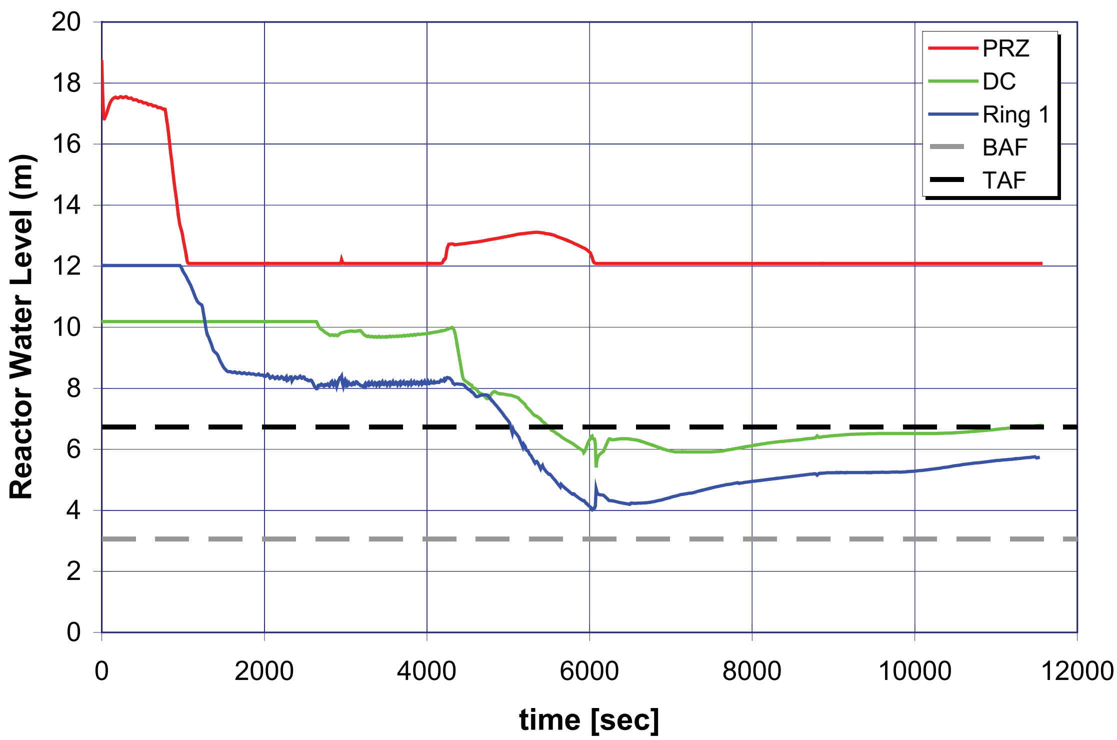


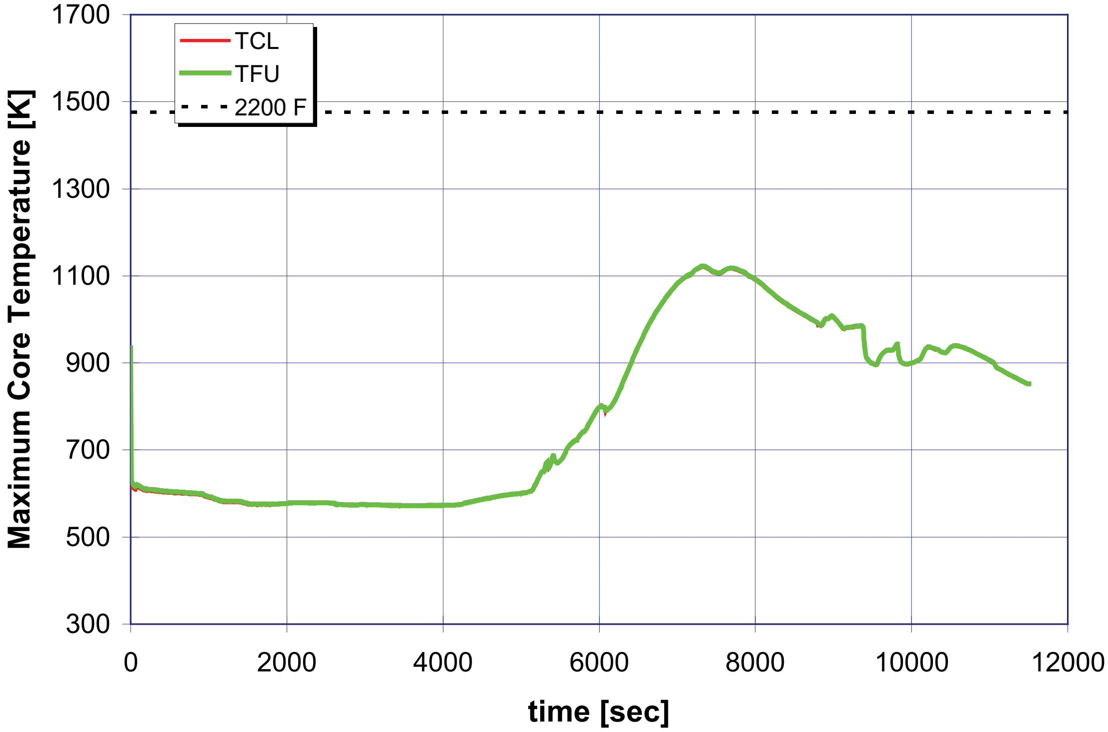


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
2000
4000
6000
8000
10000
time [sec]

A.5.1.2 Case 1b: Three HHSI Pumps at 1.64 Hours

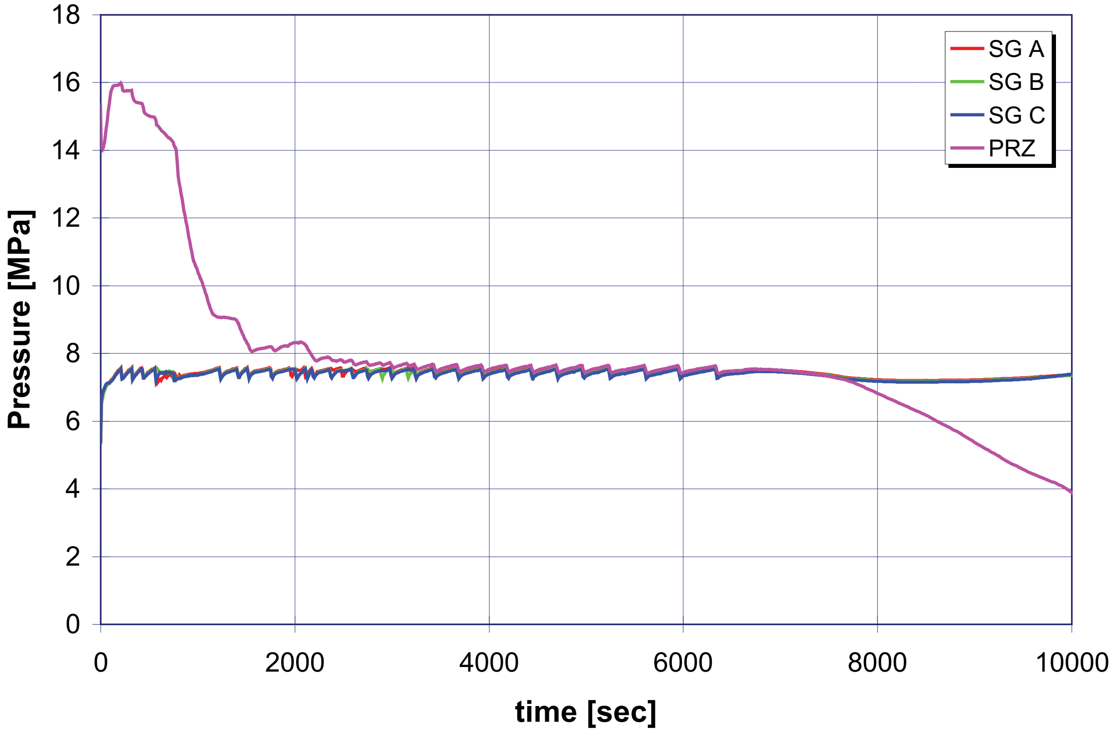


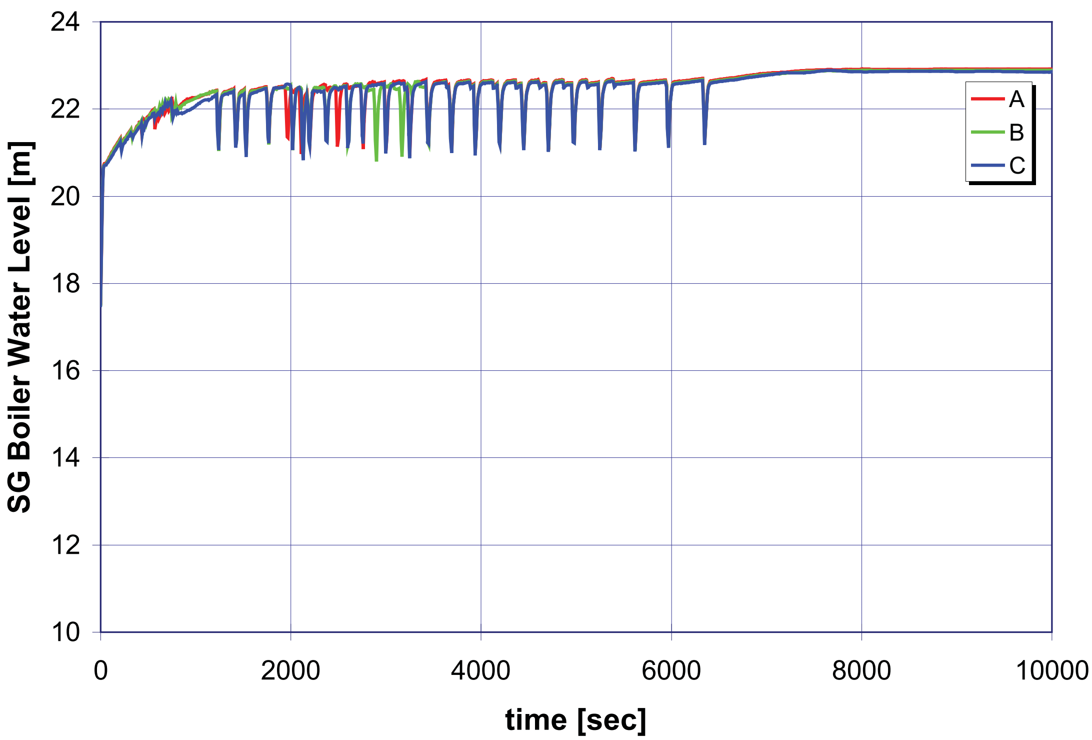


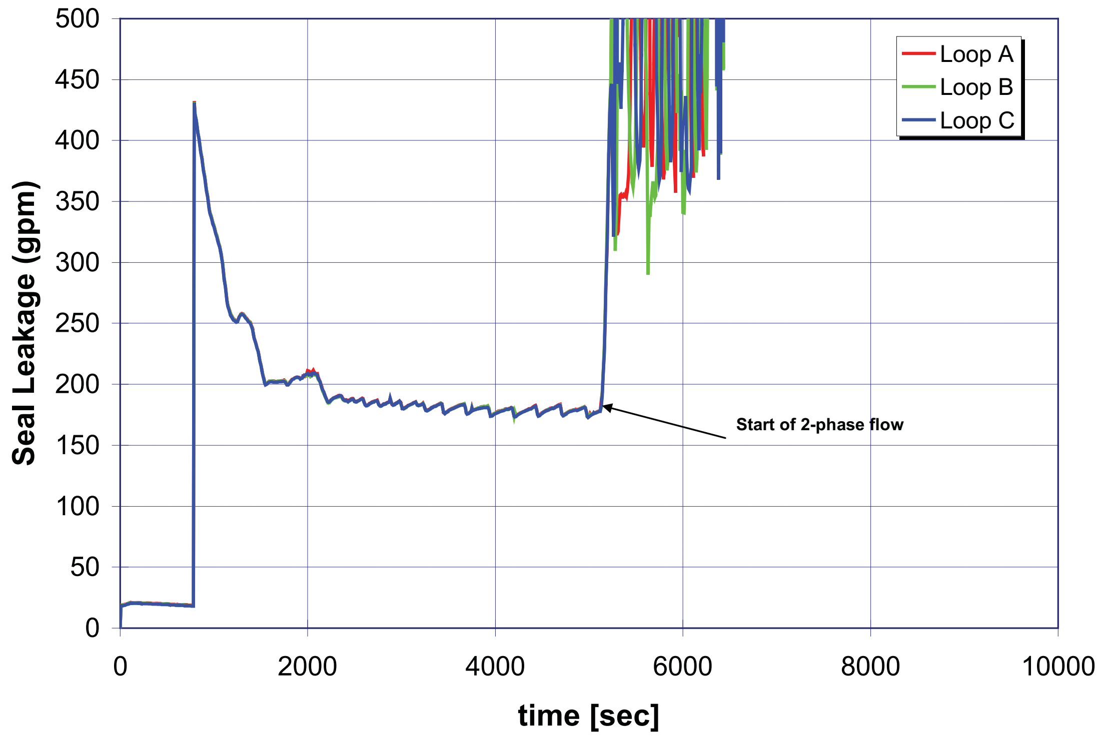


TCL
TFU
2200 F
Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
2000
4000
6000
8000
10000
12000
time [sec]

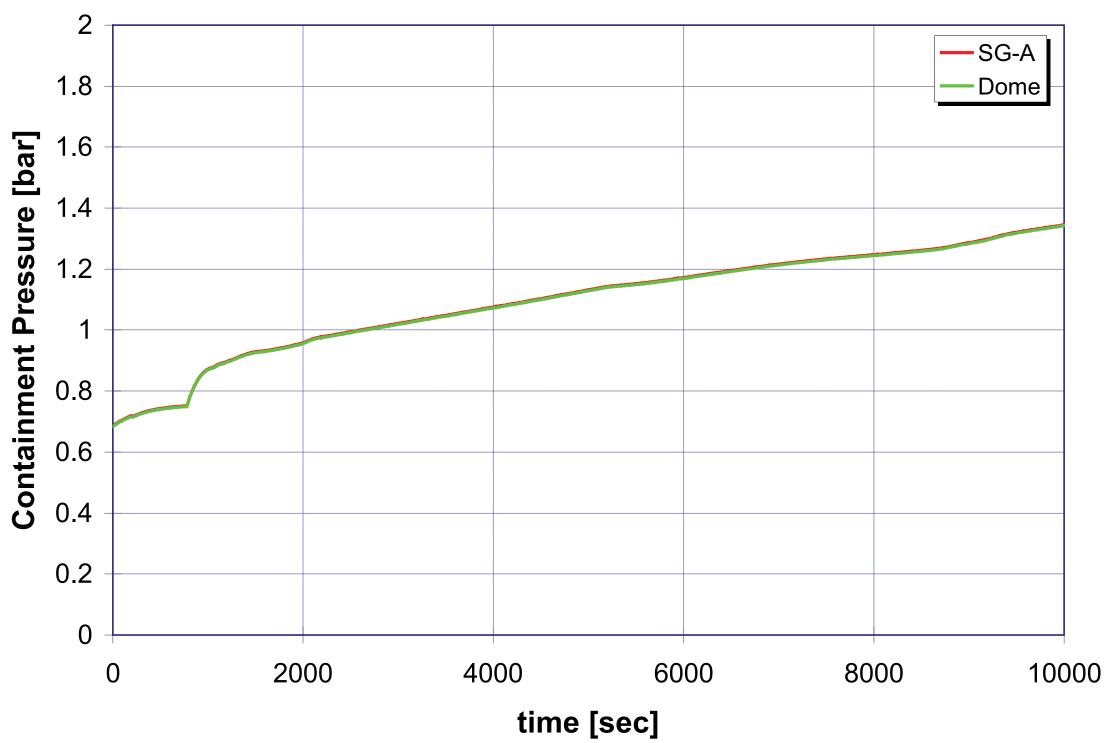
A.5.2 Case 2: Station Blackout with Turbine-Driven Auxiliary Feedwater (500 gpm)



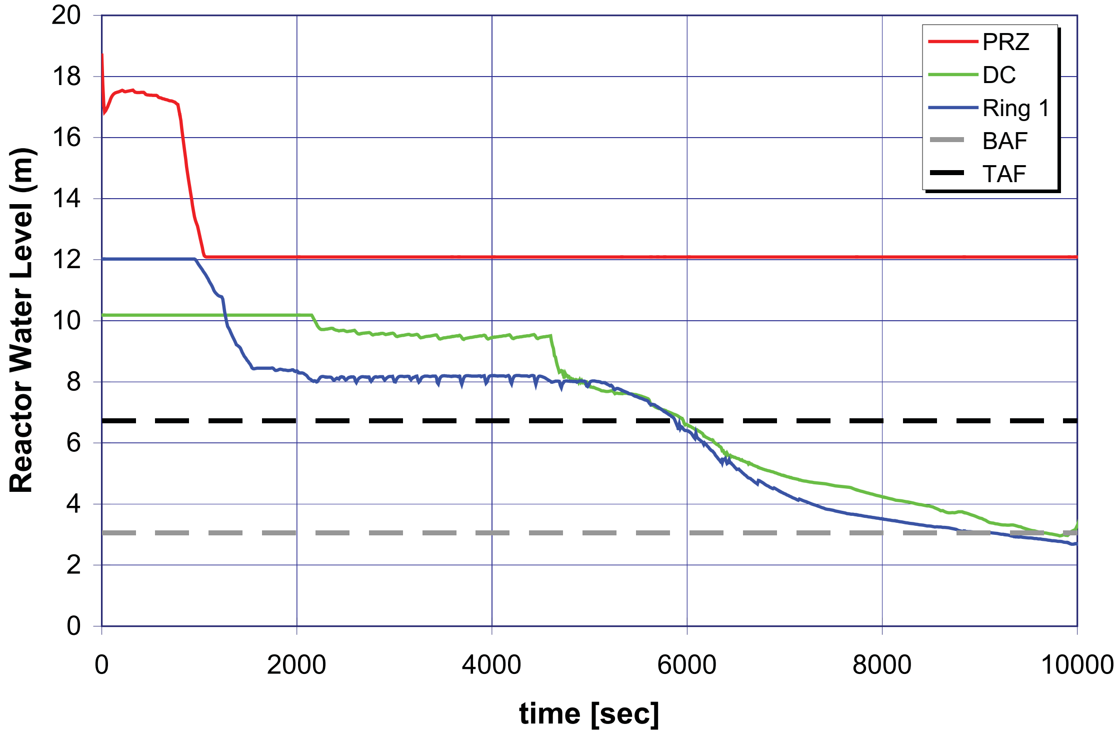




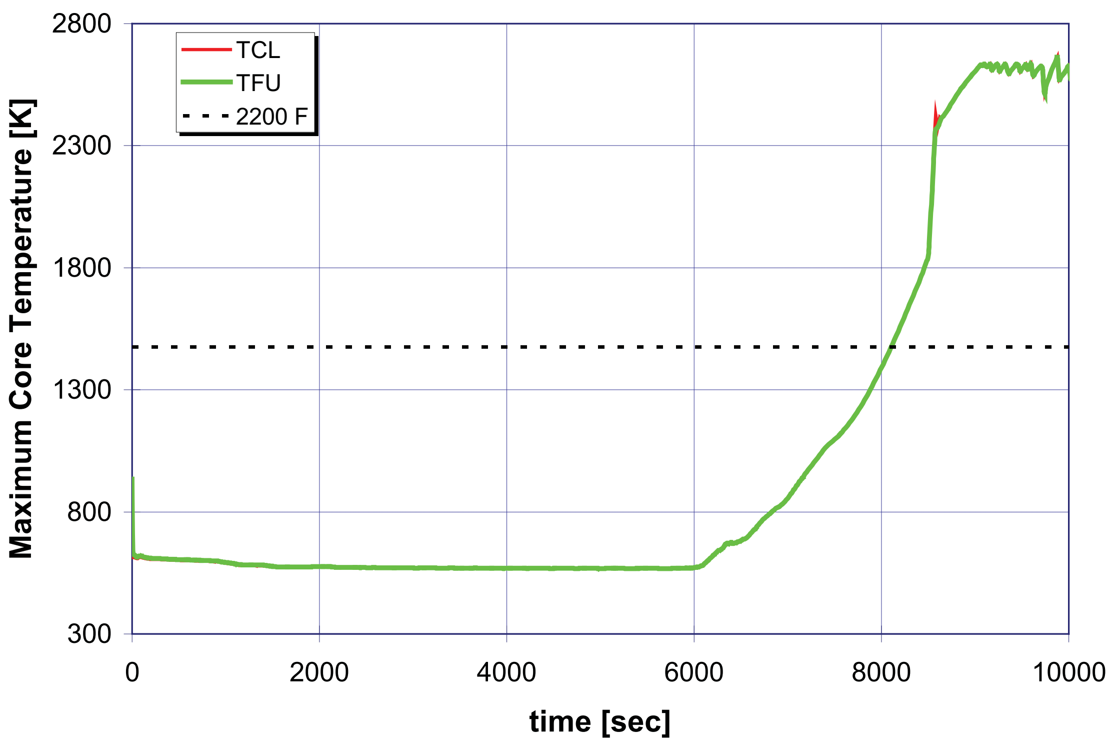
500
450
400
350
300
250
200
150
100
50
0
Seal Leakage (gpm)
0
2000
4000
6000
8000
10000
time [sec]
Loop A
Loop B
Loop C
Start of 2-phase flow



2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
Containment Pressure [bar]
0
2000
4000
6000
8000
10000
time [sec]
SG-A
Dome

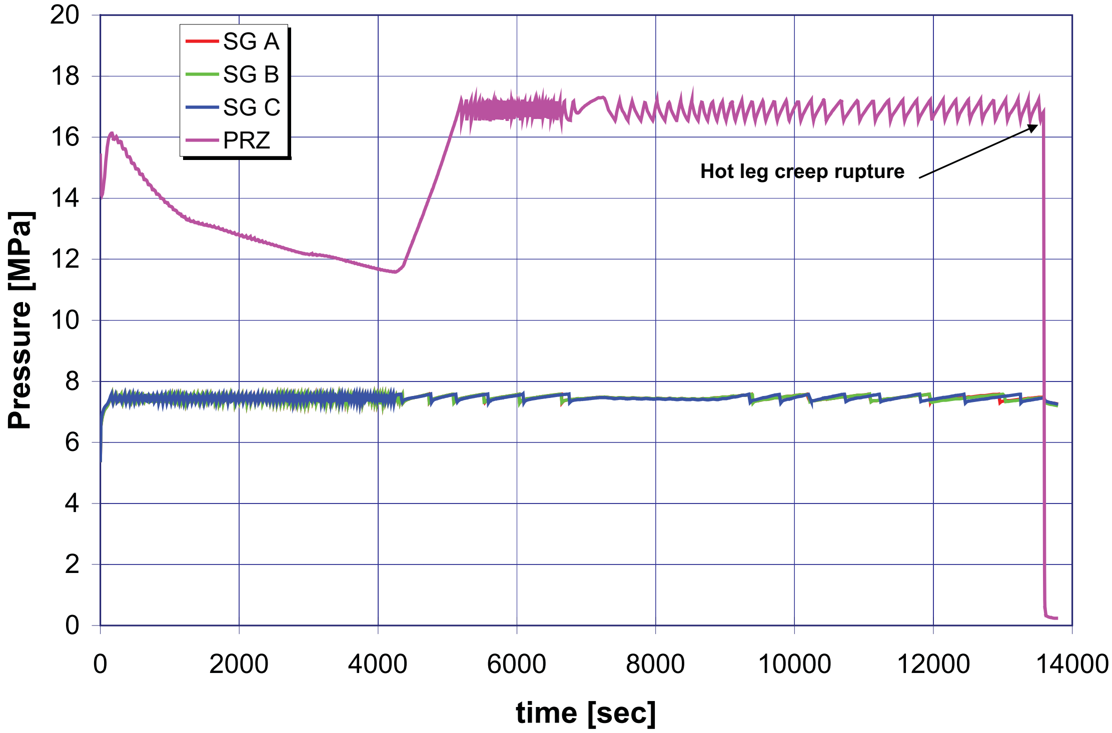


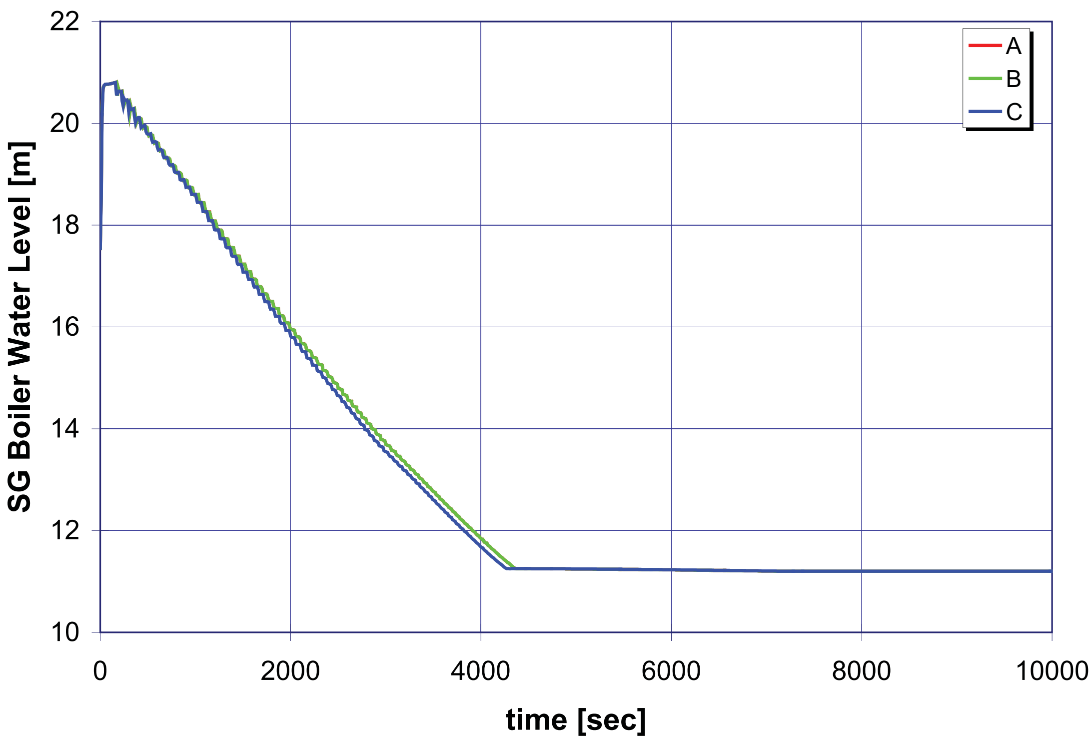
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
20
18
16
14
12
10
8
6
4
2
0
0
2000
4000
6000
8000
10000

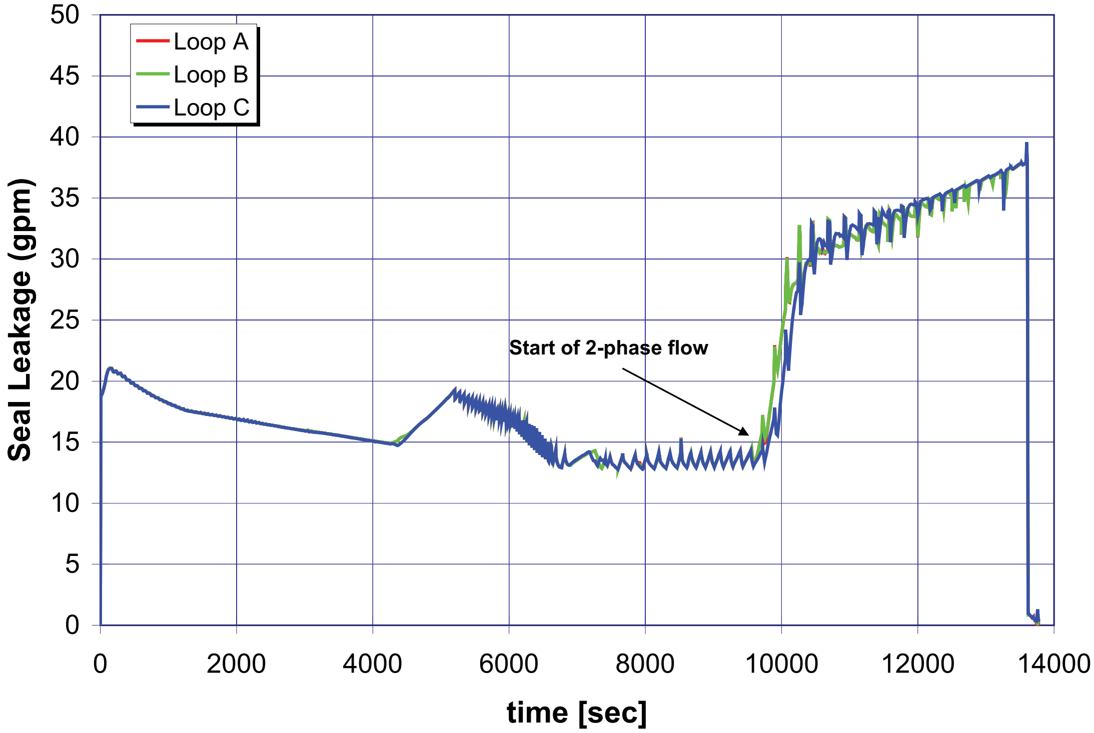


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
2800
2300
1800
1300
800
300
0
2000
4000
6000
8000
10000

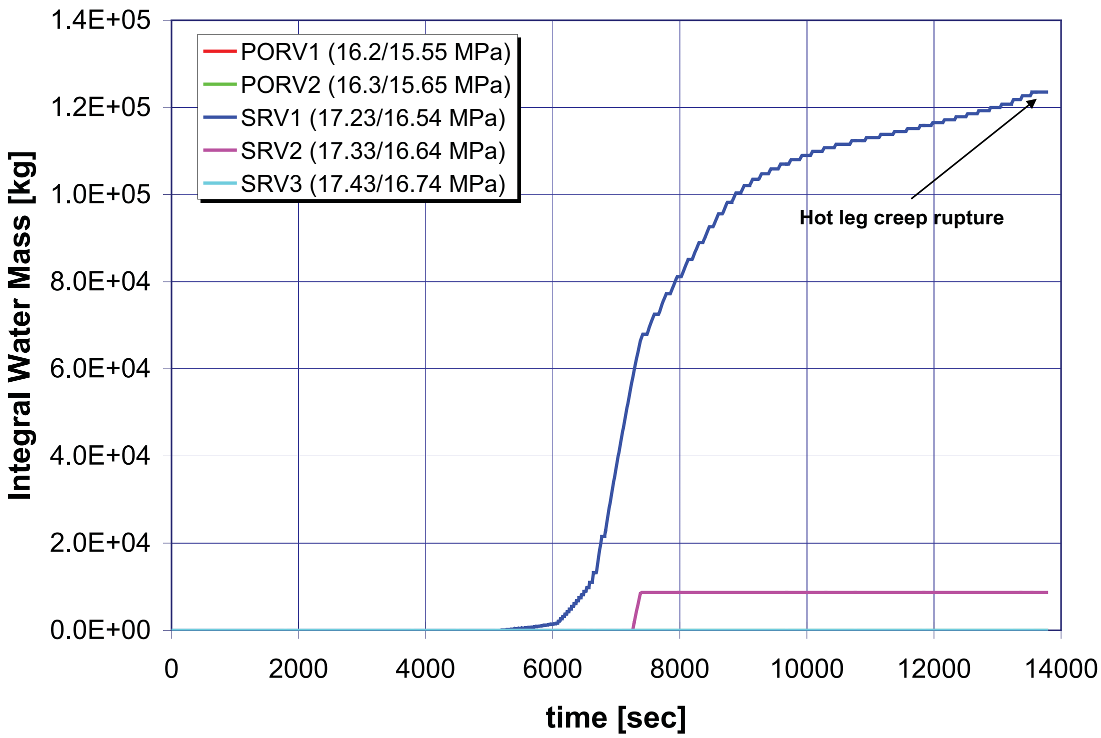
A.5.3 Case 3: Station Blackout without Turbine-Driven Auxiliary Feedwater (21 gpm)



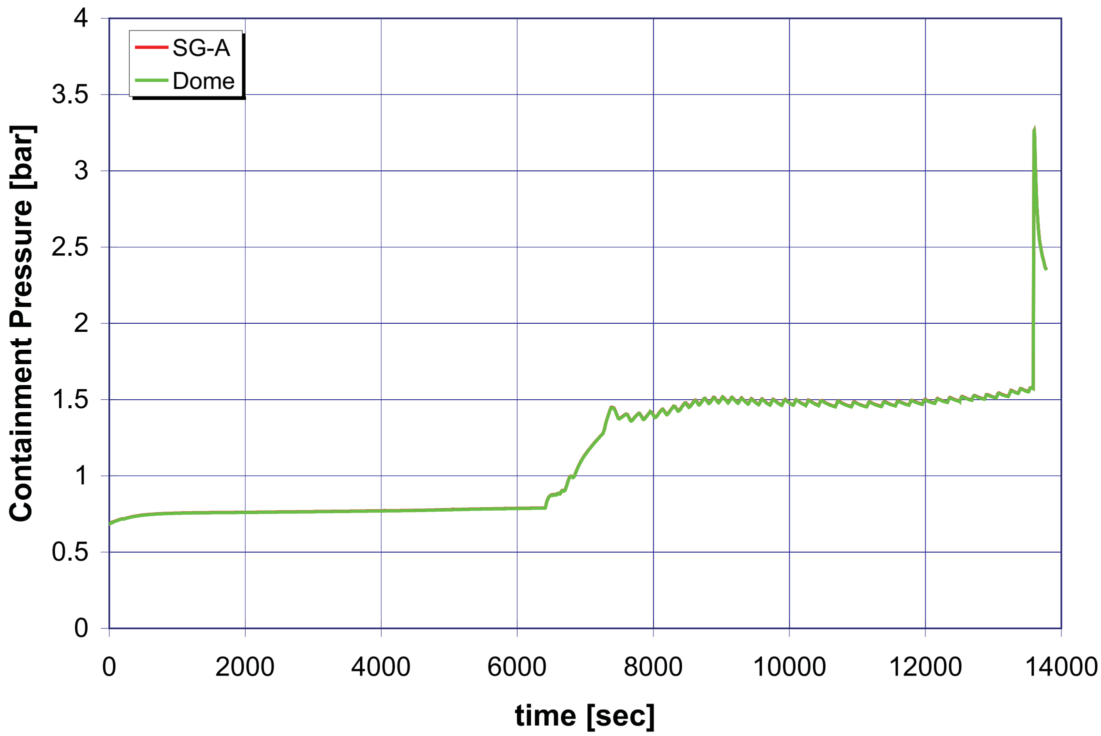




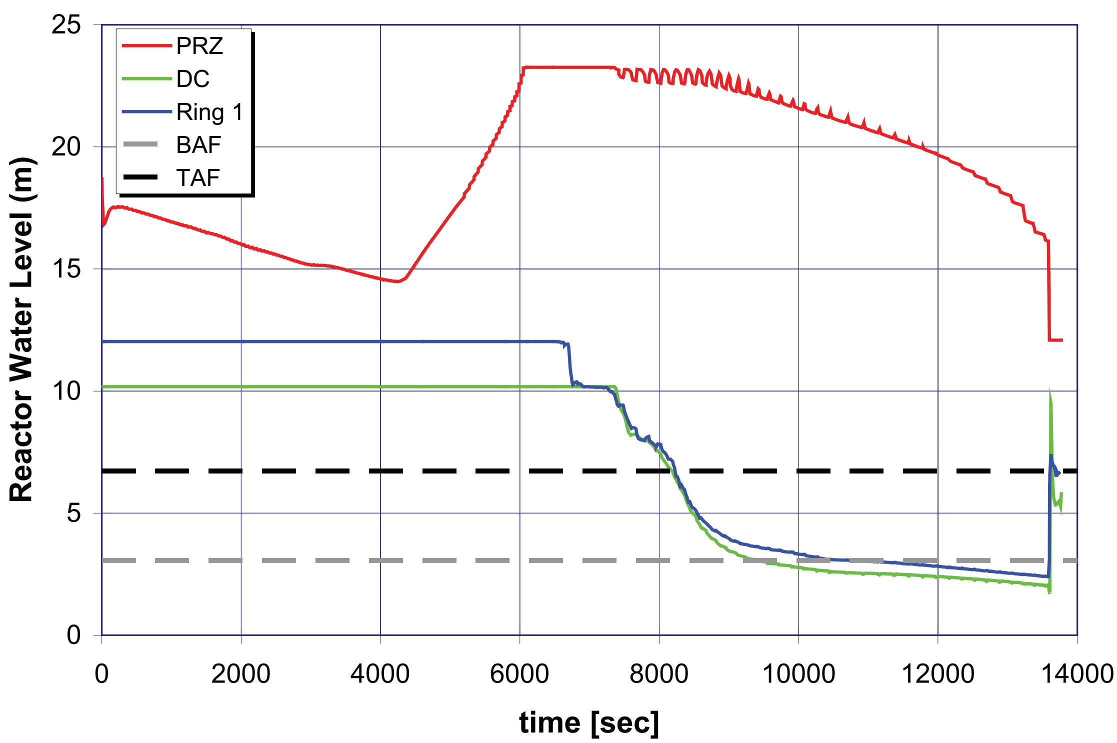
Loop A
Loop B
Loop C
Seal Leakage (gpm)
time [sec]
Start of 2-phase flow
0
5
10
15
20
25
30
35
40
45
50
0
2000
4000
6000
8000
10000
12000
14000



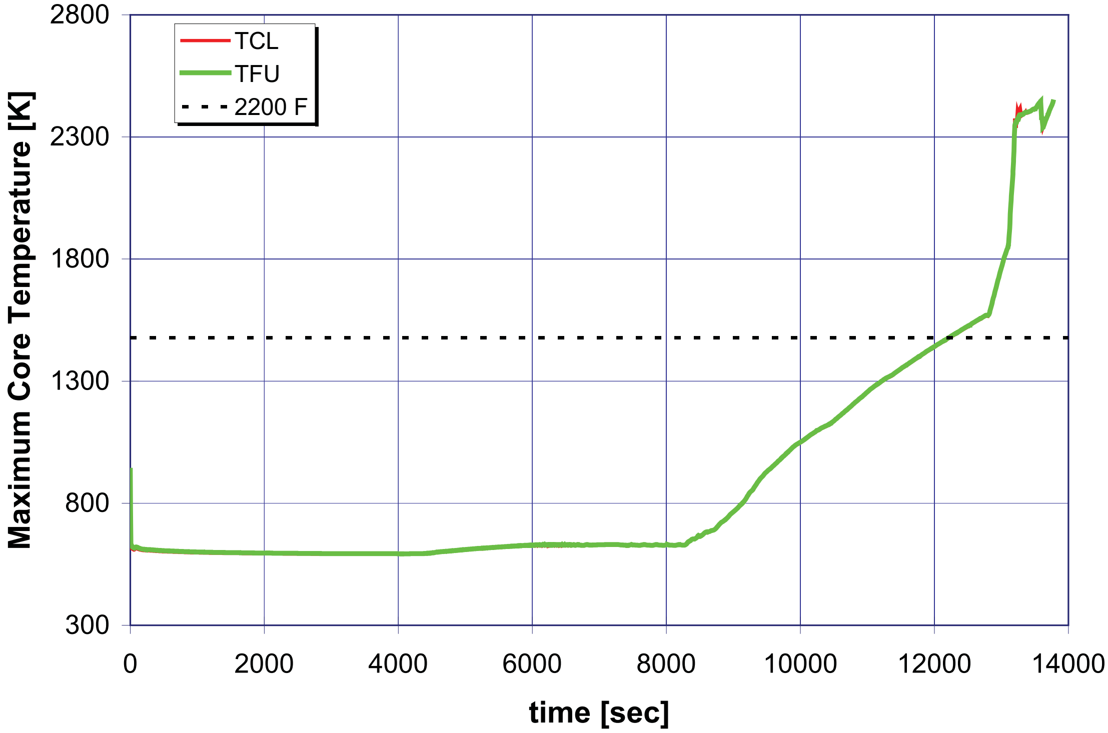
PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
Integral Water Mass [kg]
time [sec]
Hot leg creep rupture
0.0E+00
2.0E+04
4.0E+04
6.0E+04
8.0E+04
1.0E+05
1.2E+05
1.4E+05
0
2000
4000
6000
8000
10000
12000
14000



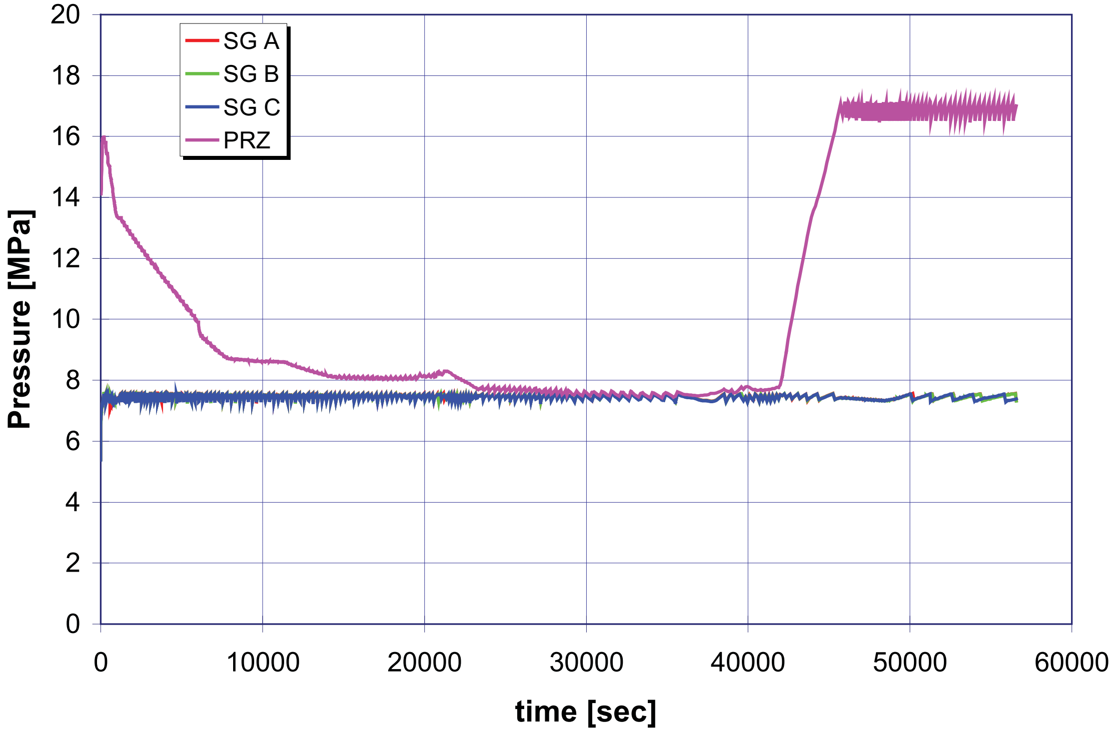
SG-A
Dome
Containment Pressure [bar]
time [sec]
0
0.5
1
1.5
2
2.5
3
3.5
4
0
2000
4000
6000
8000
10000
12000
14000

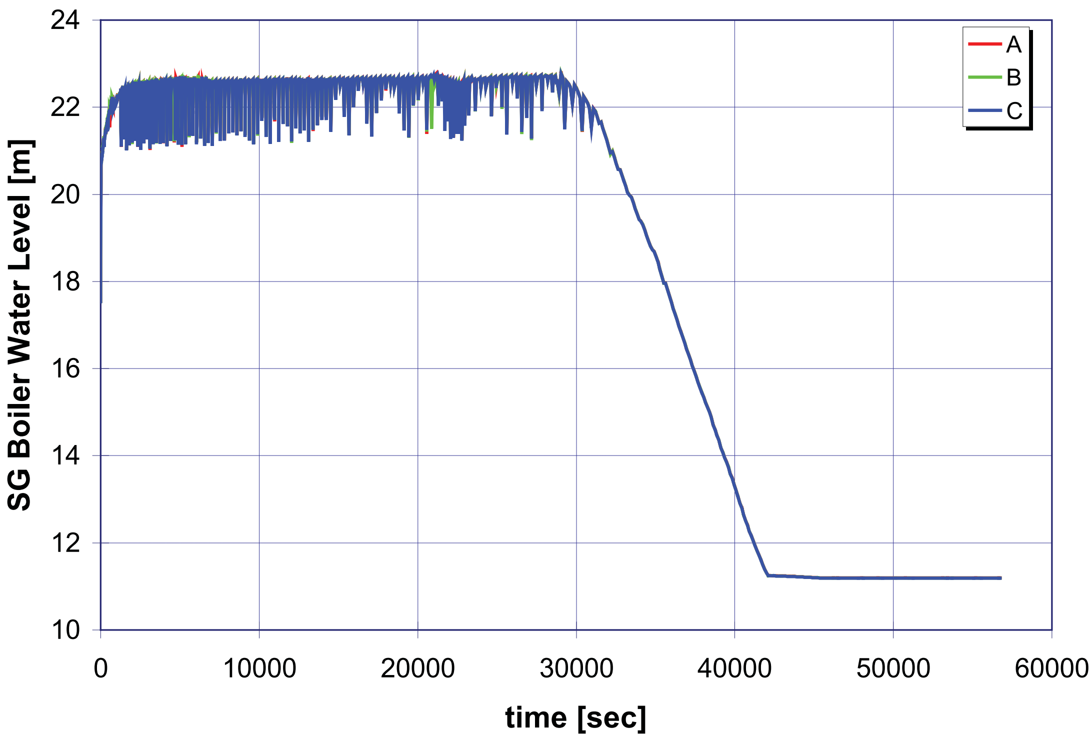


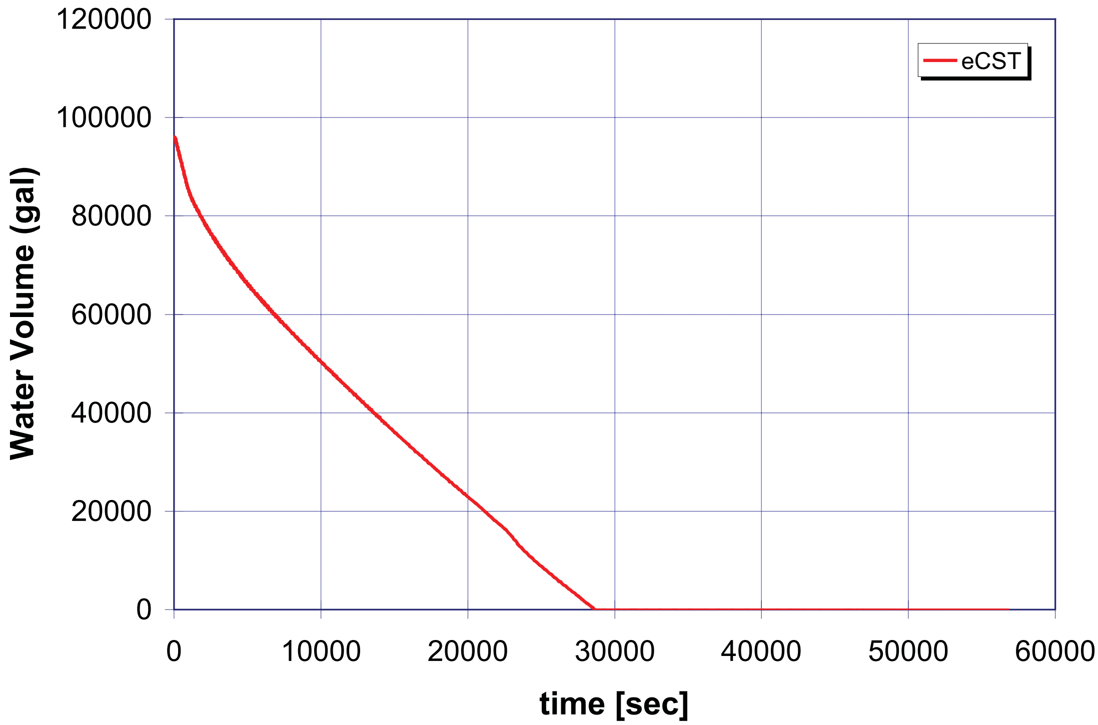
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
5
10
15
20
25
0
2000
4000
6000
8000
10000
12000
14000


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
2000
4000
6000
8000
10000
12000
14000
time [sec]

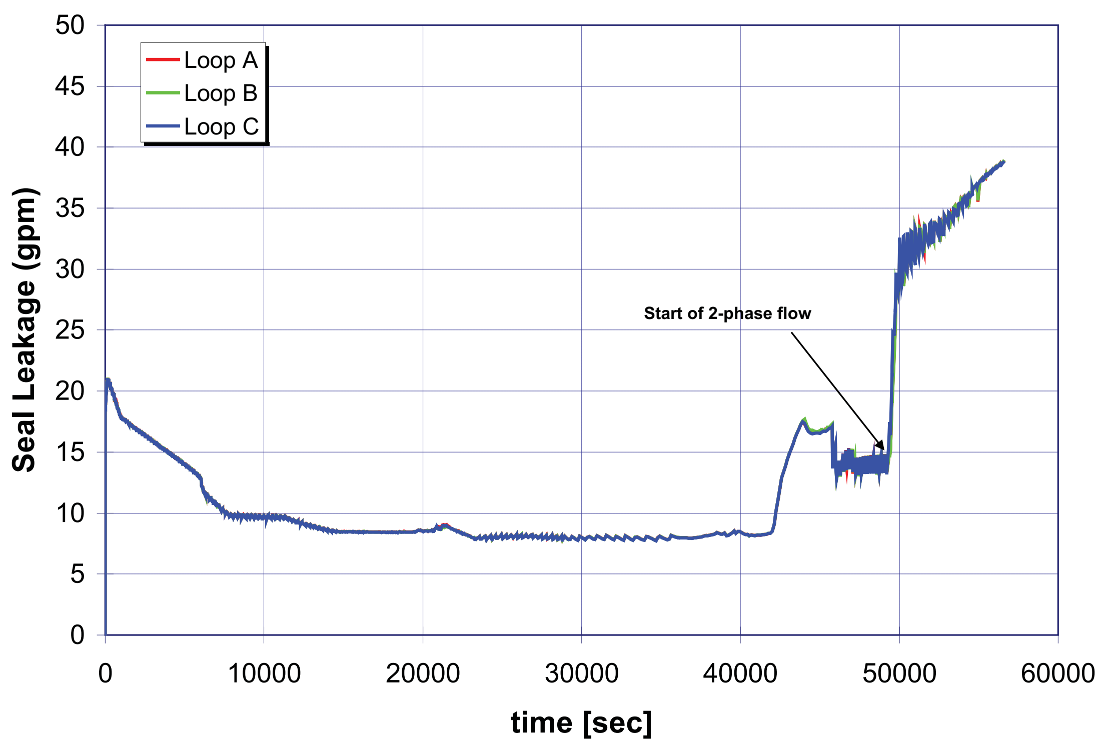
A.5.4 Case 4: Station Blackout with Turbine-Driven Auxiliary Feedwater (21 gpm)



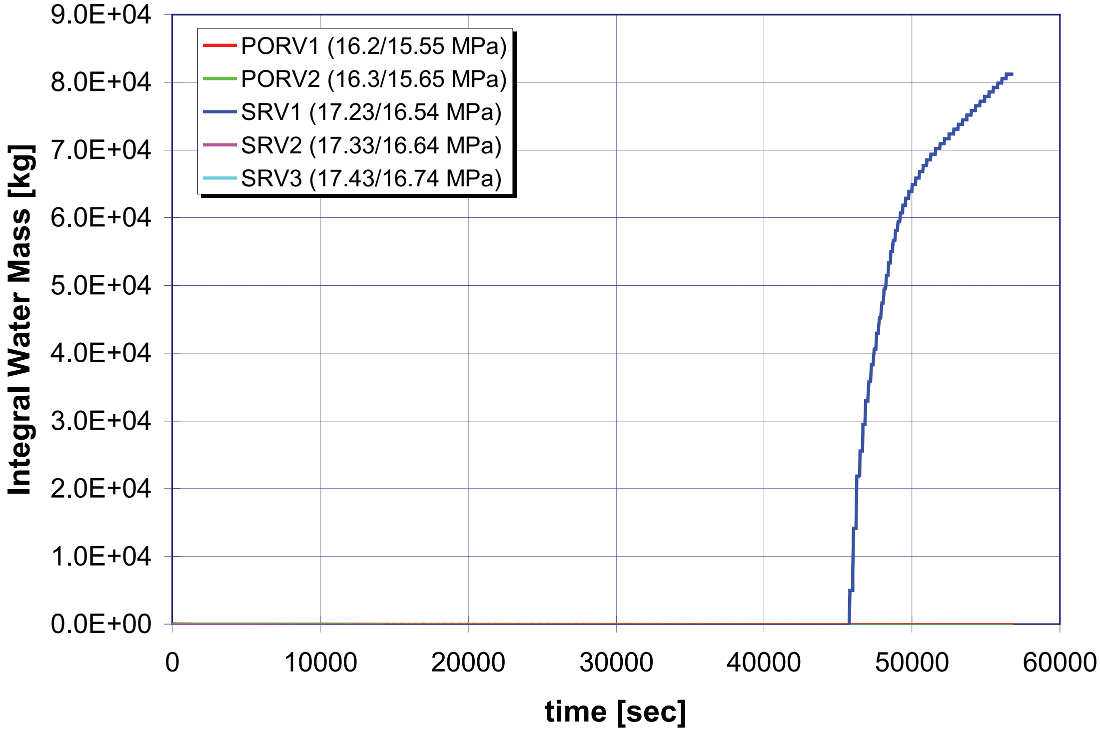




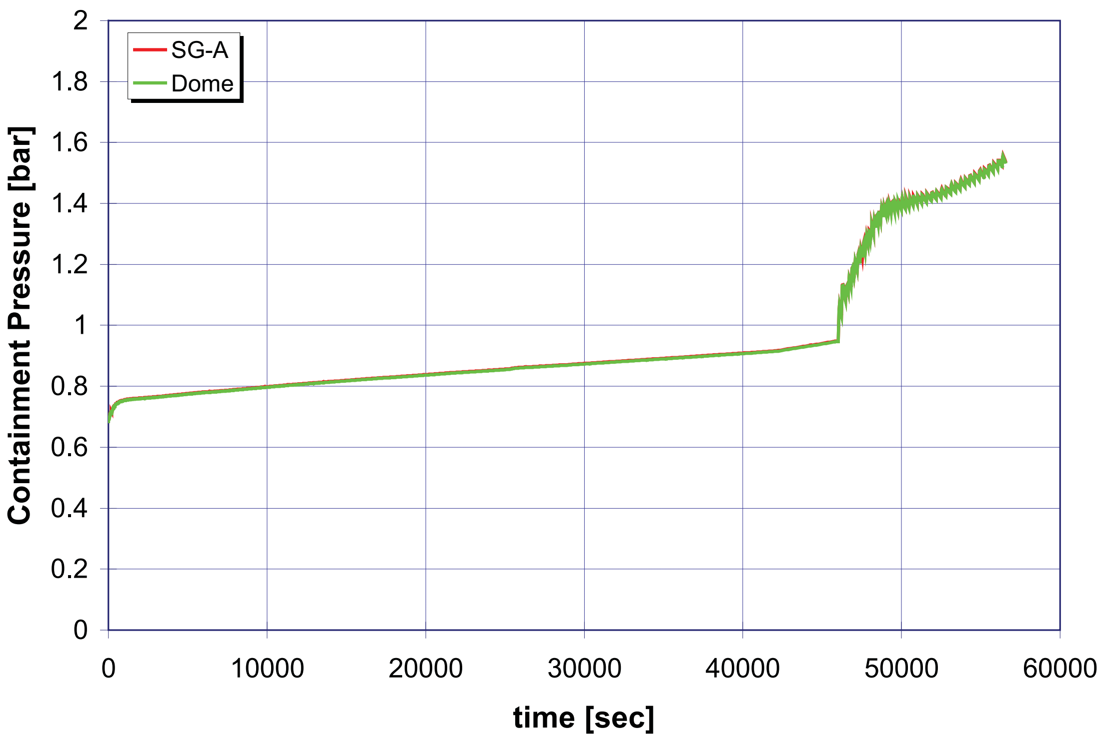
eCST
Water Volume (gal)
time [sec]
120000
100000
80000
60000
40000
20000
0
0
10000
20000
30000
40000
50000
60000



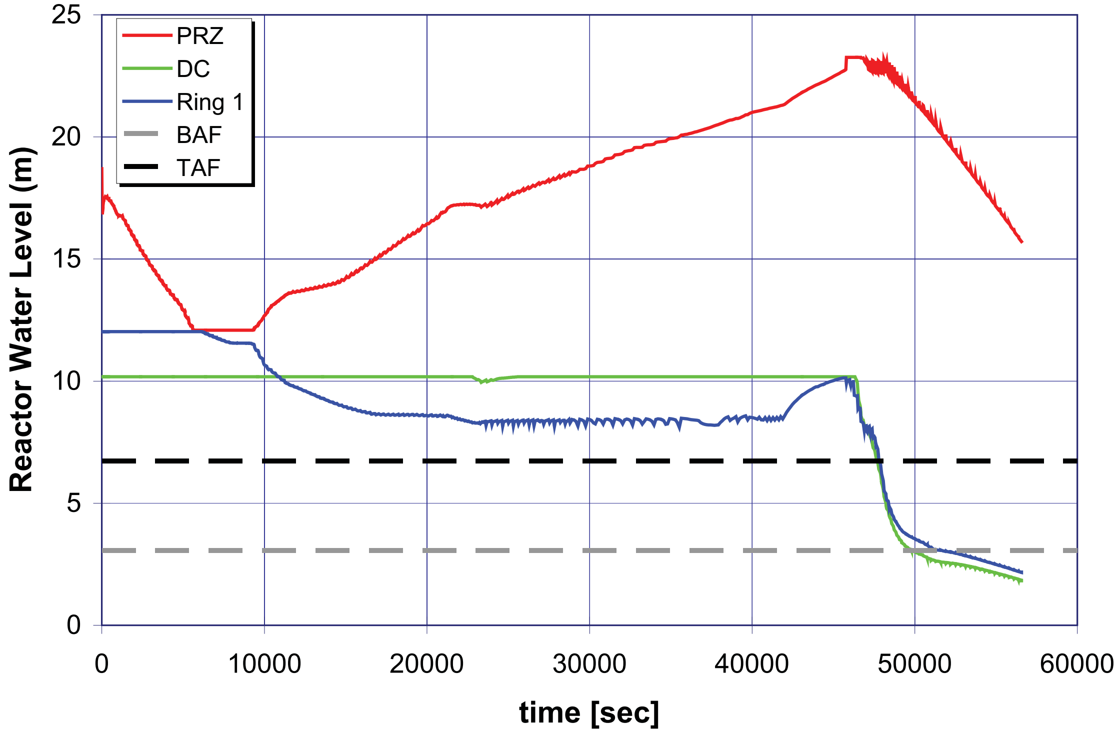
Loop A
Loop B
Loop C
Start of 2-phase flow
Seal Leakage (gpm)
time [sec]
50
45
40
35
30
25
20
15
10
5
0
0
10000
20000
30000
40000
50000
60000



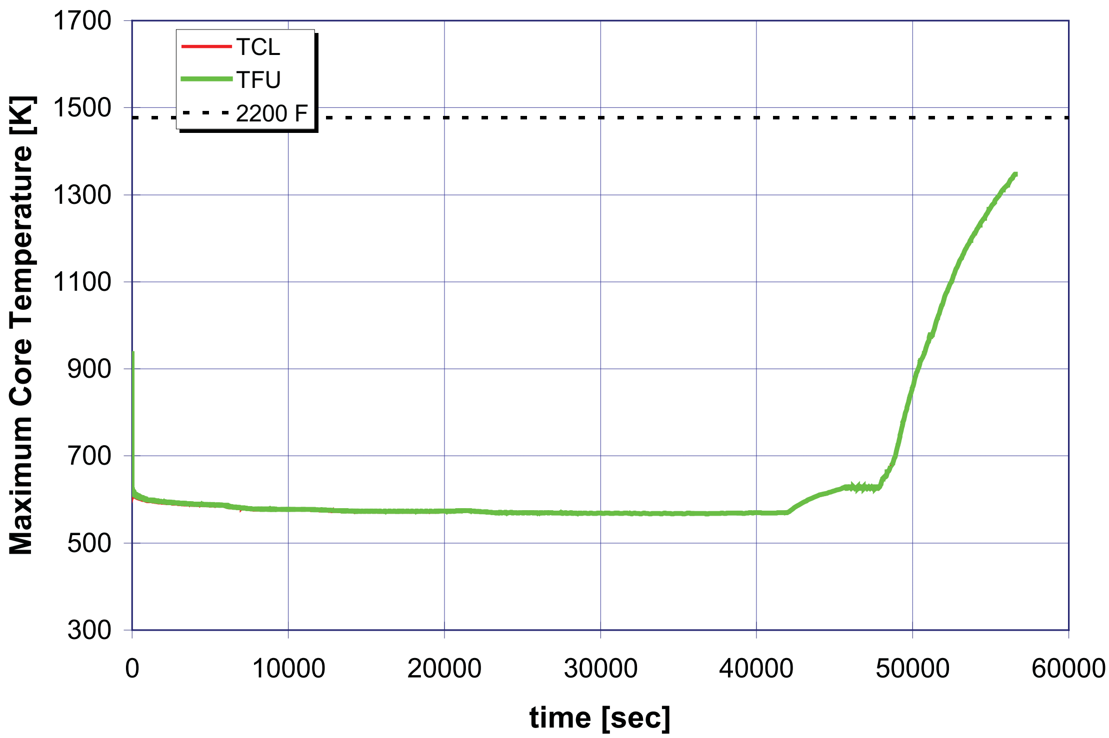
9.0E+04
8.0E+04
7.0E+04
6.0E+04
5.0E+04
4.0E+04
3.0E+04
2.0E+04
1.0E+04
0.0E+00
Integral Water Mass [kg]
PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
0
10000
20000
30000
40000
50000
60000
time [sec]



2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
Containment Pressure [bar]
SG-A
Dome
0
10000
20000
30000
40000
50000
60000
time [sec]

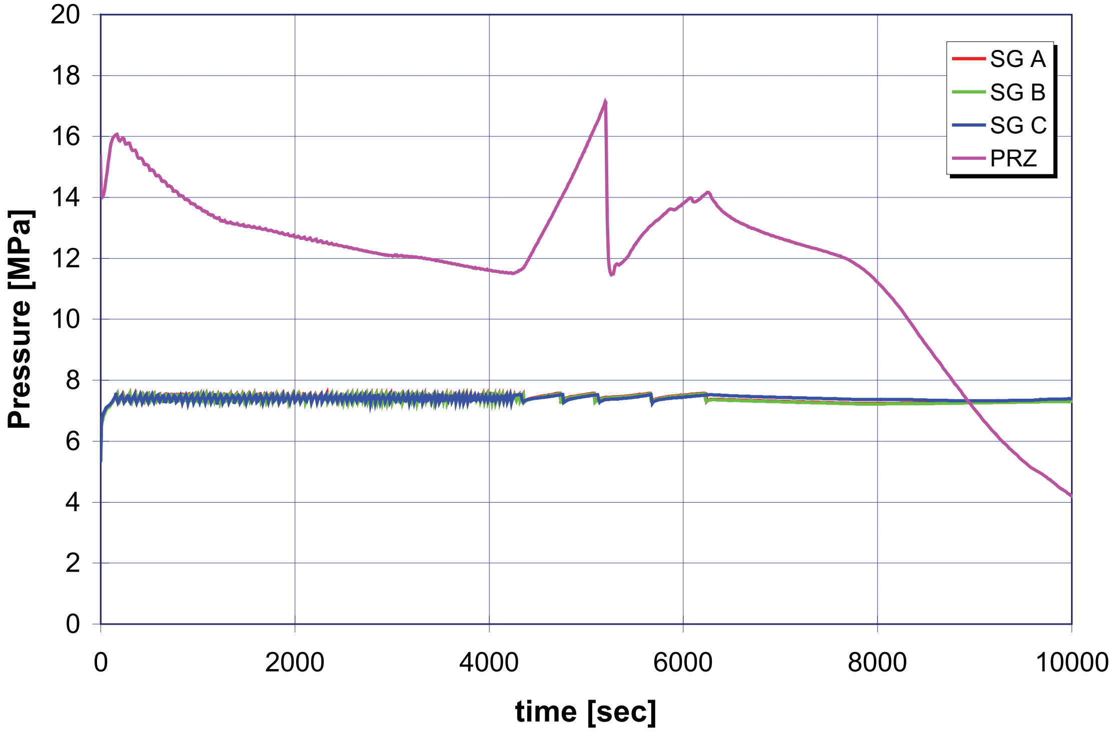


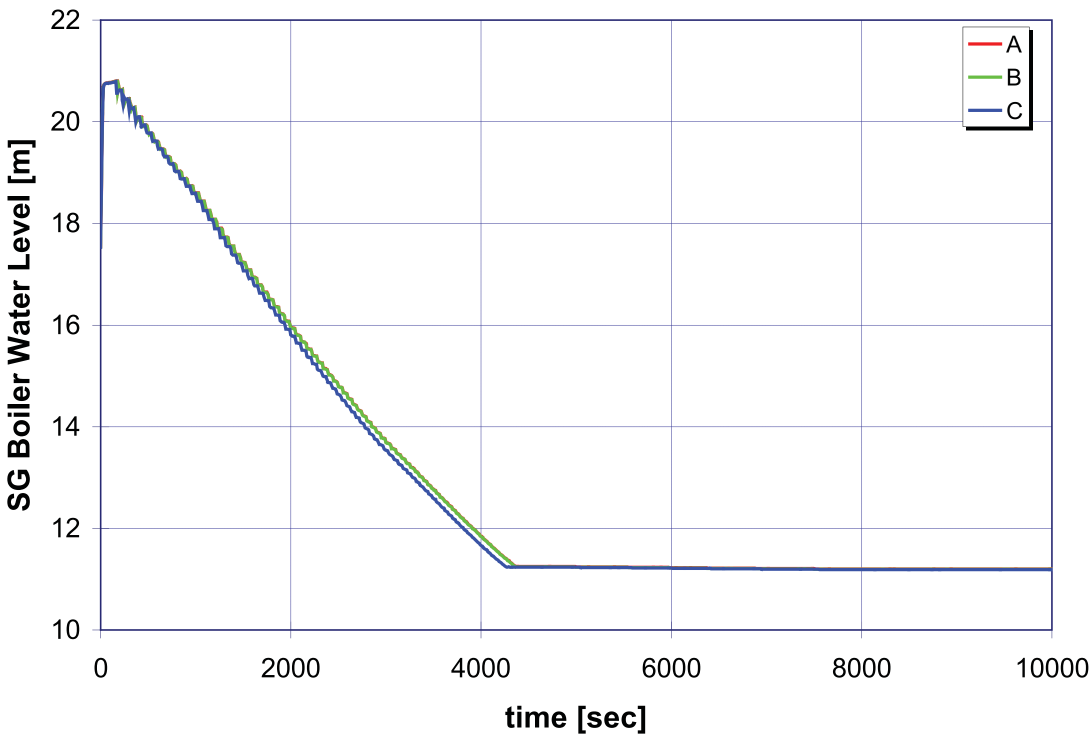
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
5
10
15
20
25
0
10000
20000
30000
40000
50000
60000

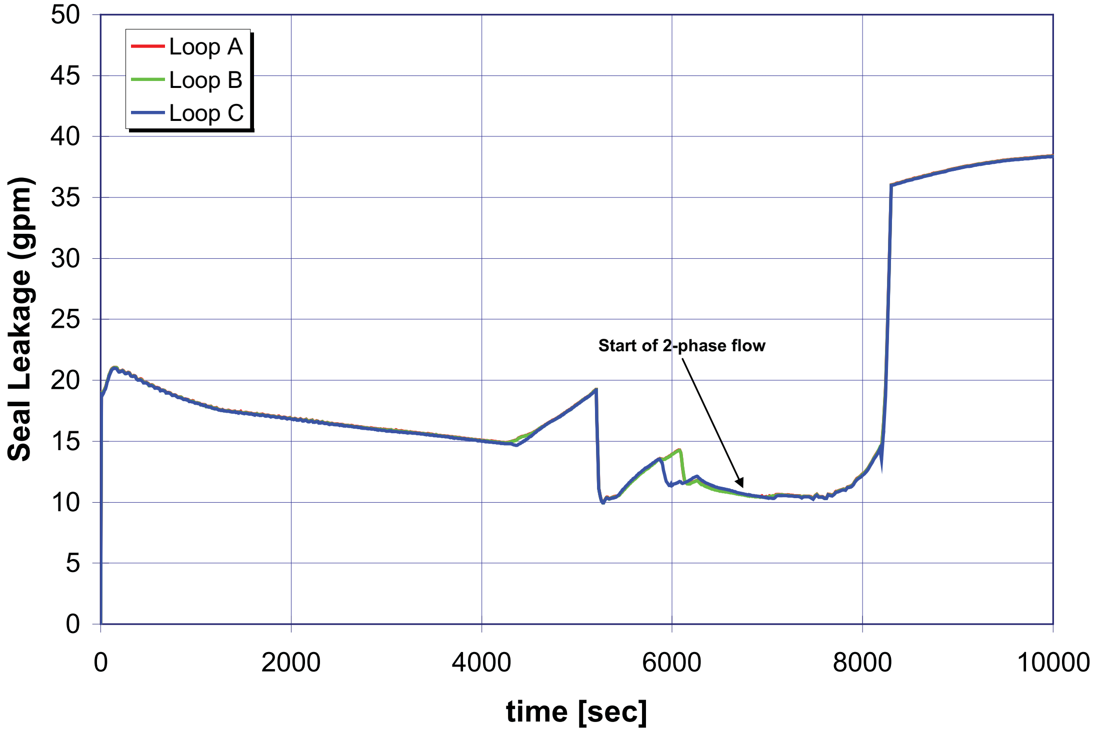


TCL
TFU
2200 F
Maximum Core Temperature [K]
time [sec]
300
500
700
900
1100
1300
1500
1700
0
10000
20000
30000
40000
50000
60000

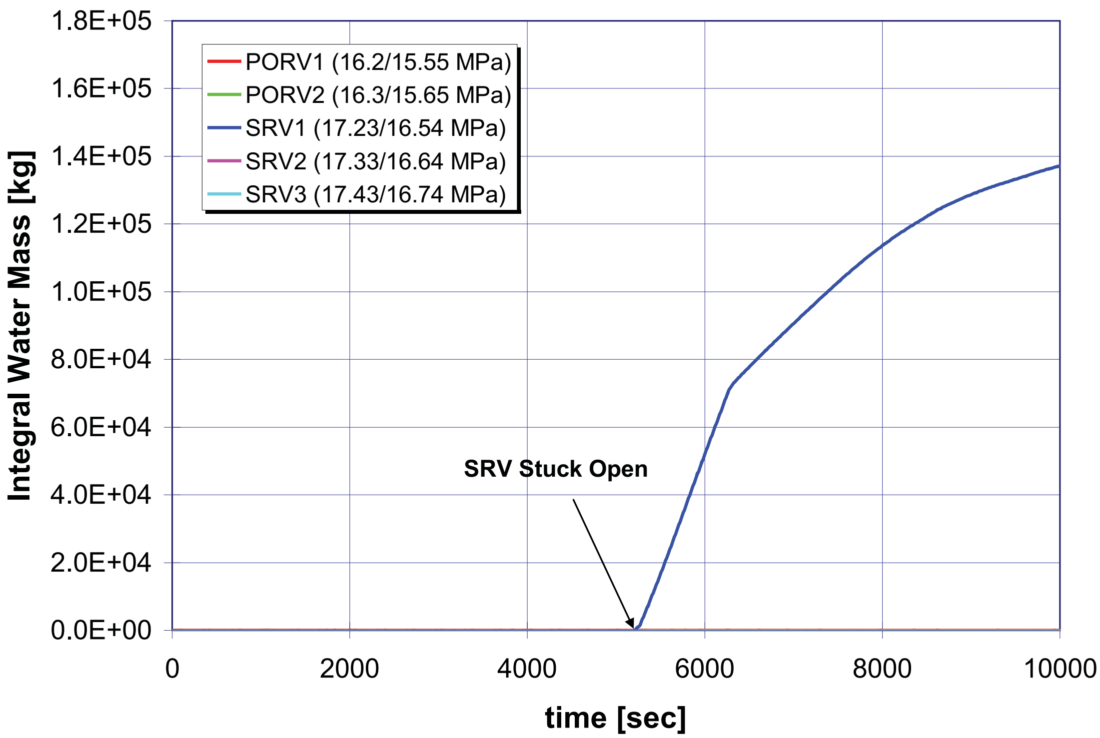
A.5.5 Case 5: Station Blackout without Turbine-Driven Auxiliary Feedwater (21 gpm); Stuck-Open Relief Valve



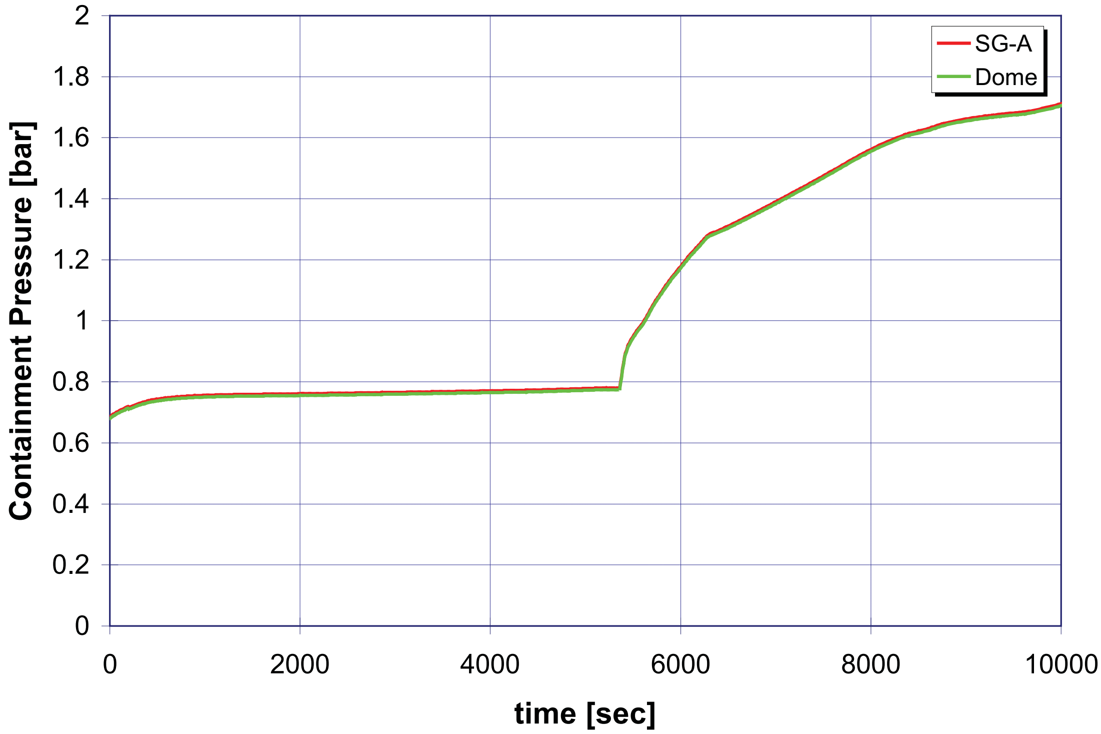




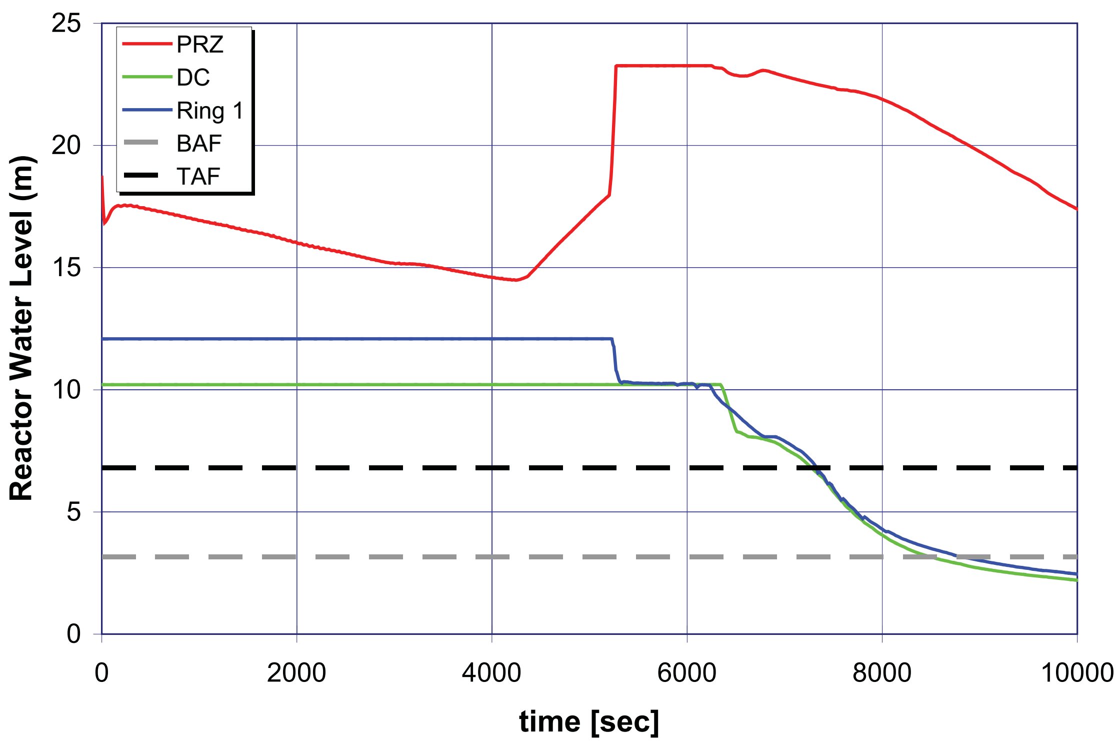
Loop A
Loop B
Loop C
Start of 2-phase flow
Seal Leakage (gpm)
time [sec]



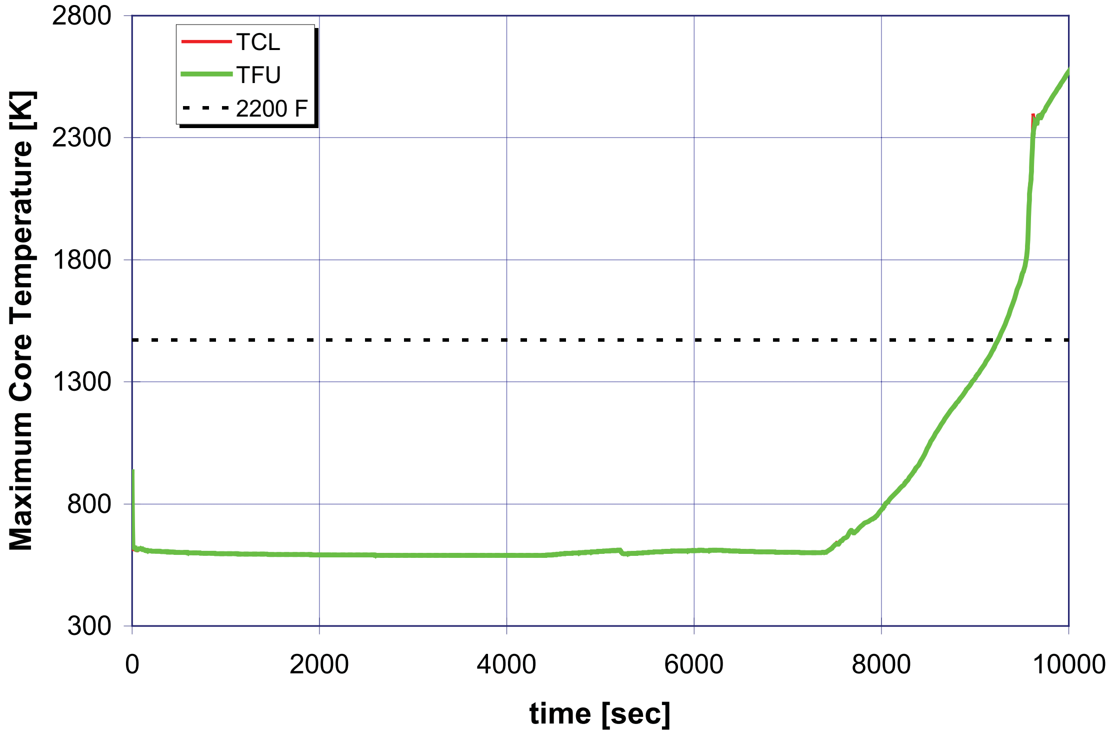
PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
SRV Stuck Open
Integral Water Mass [kg]
time [sec]



SG-A
Dome
Containment Pressure [bar]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
2000
4000
6000
8000
10000

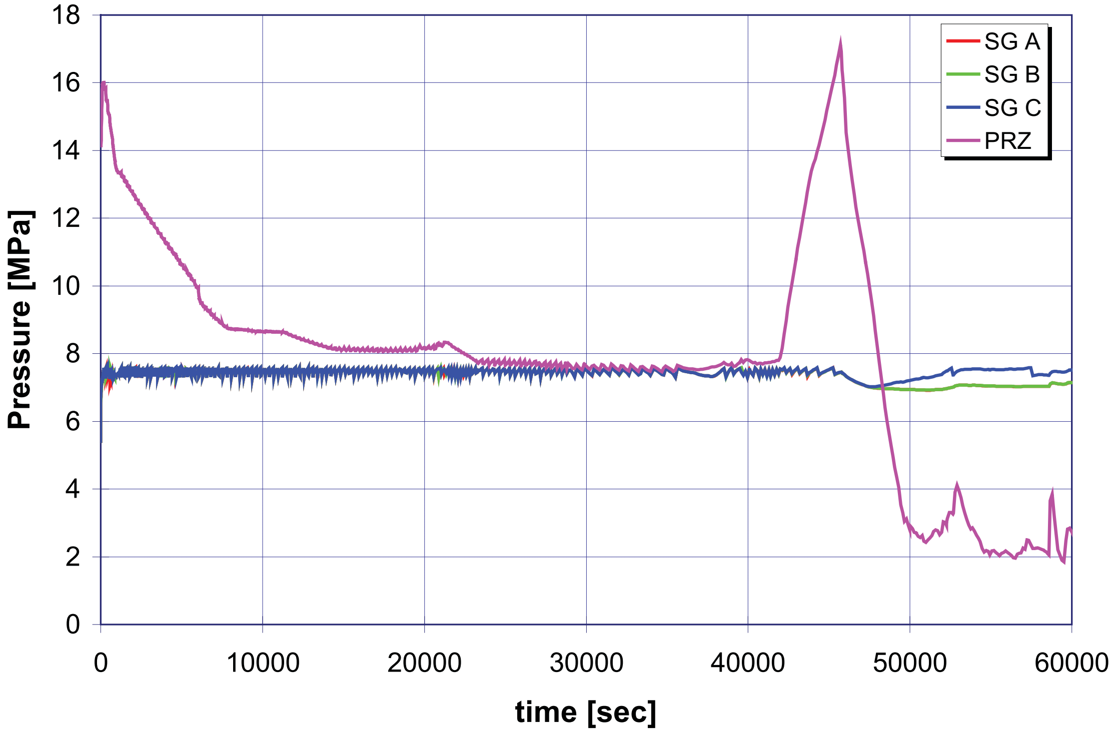


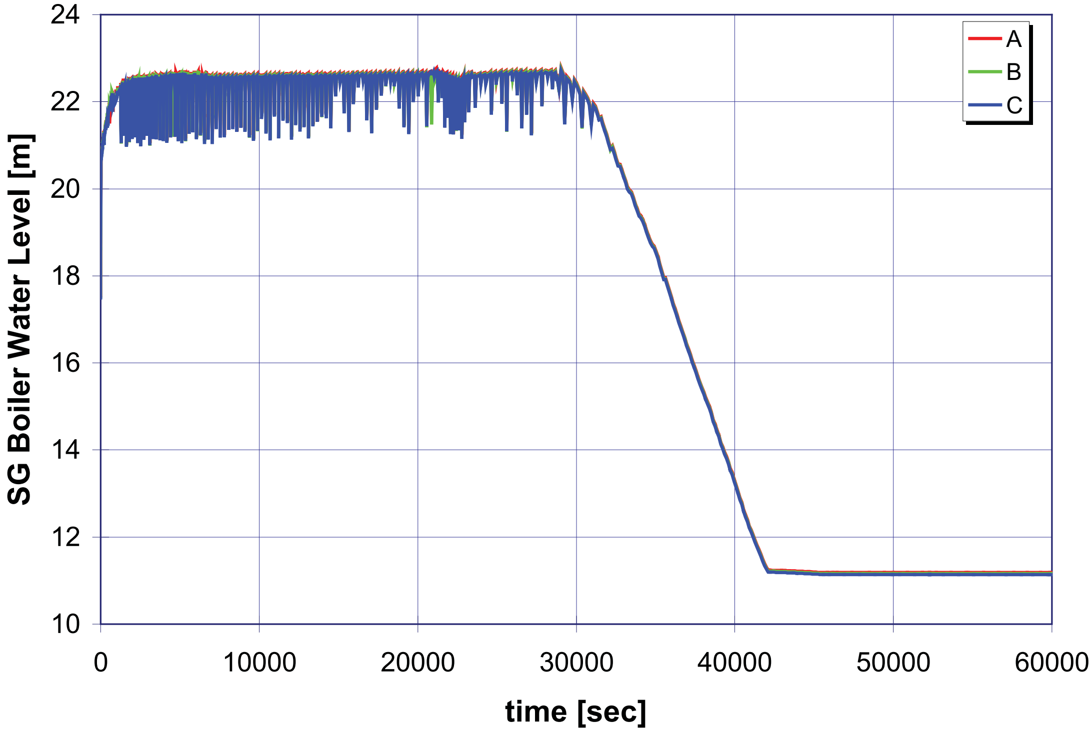
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
5
10
15
20
25
0
2000
4000
6000
8000
10000

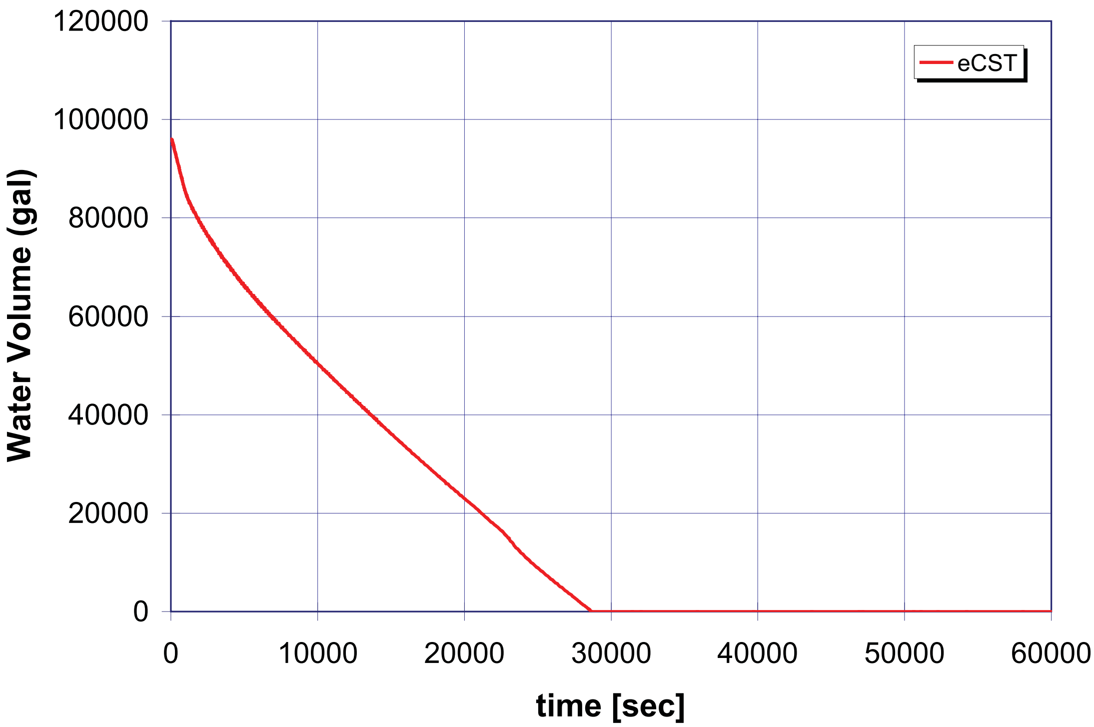


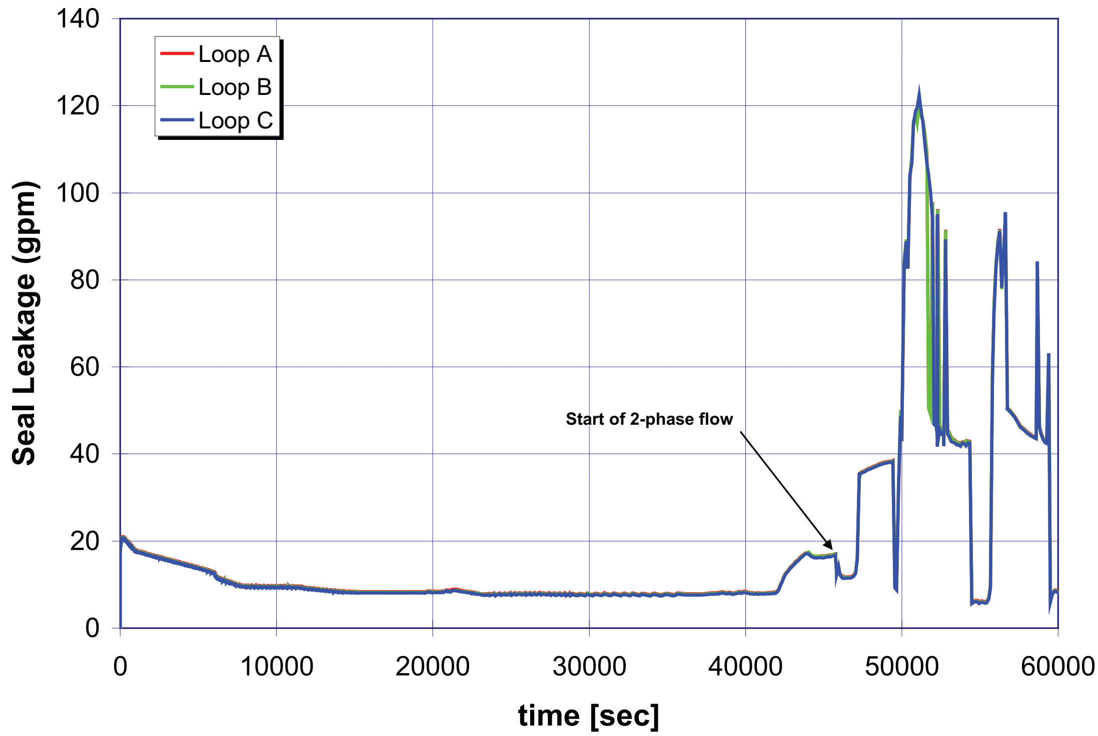
TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
2000
4000
6000
8000
10000
time [sec]

A.5.6 Case 6: Station Blackout with Turbine-Driven Auxiliary Feedwater (21 gpm); Stuck-Open Relief Valve

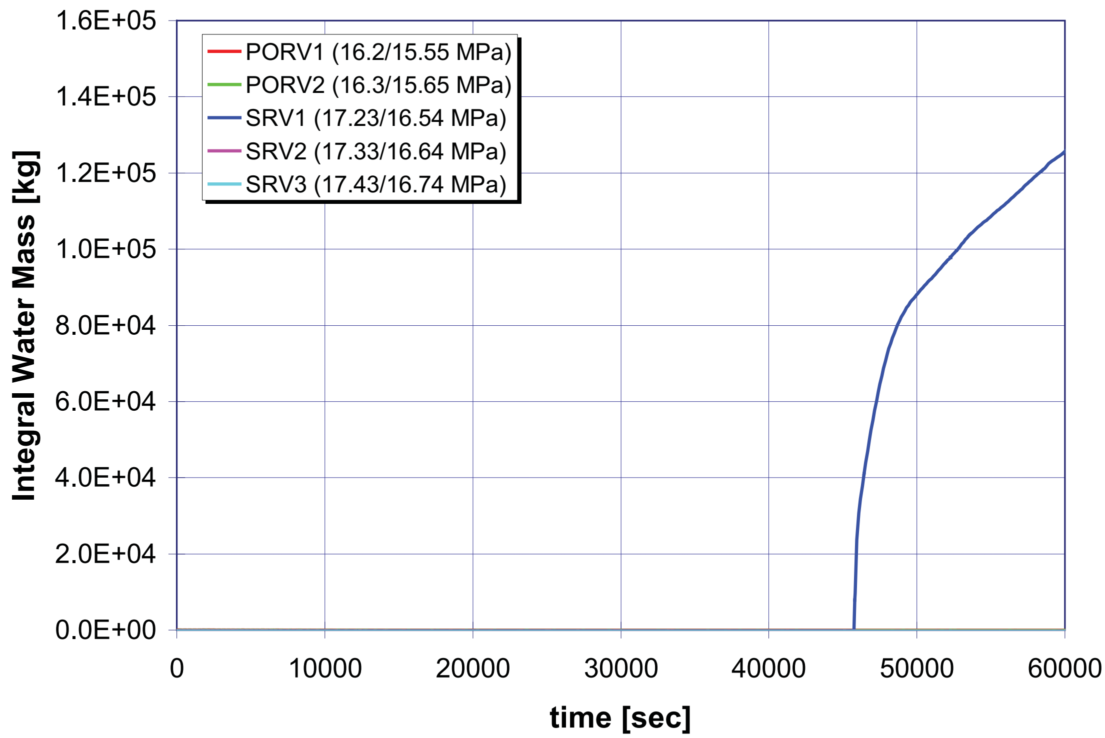



24
22
20
18
16
14
12
10
SG Boiler Water Level [m]
0
10000
20000
30000
40000
50000
60000
time [sec]
A
B
C

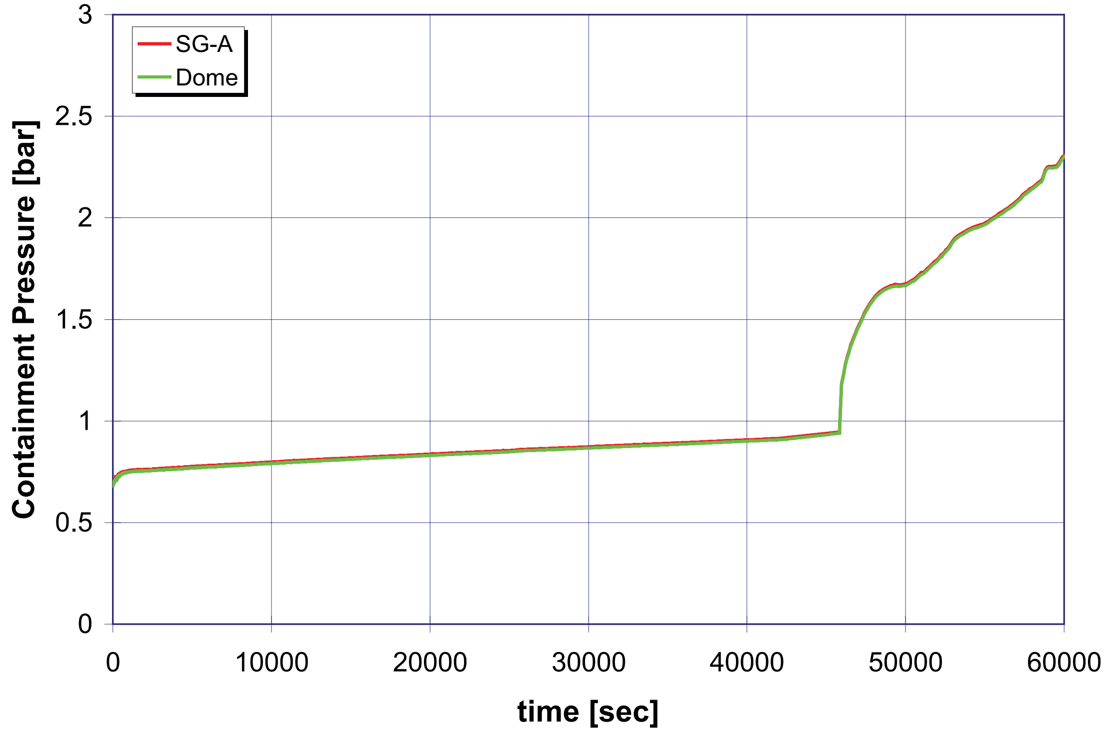

120000
100000
80000
60000
40000
20000
0
Water Volume (gal)
0
10000
20000
30000
40000
50000
60000
time [sec]
eCST



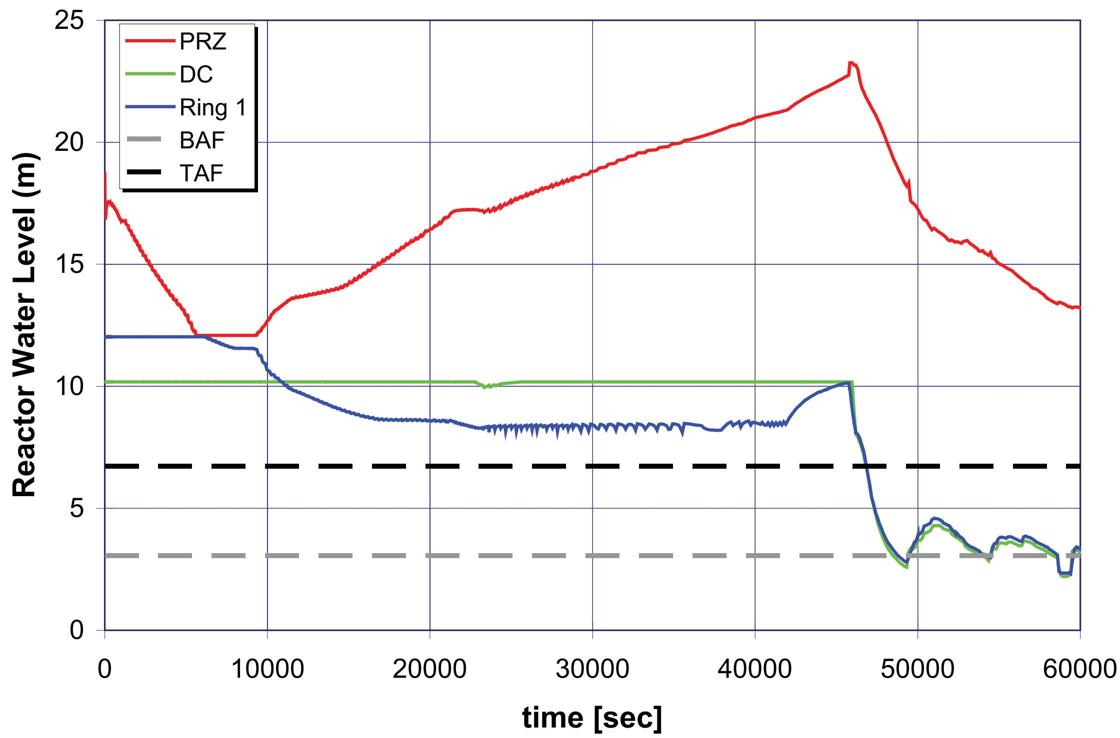
Loop A
Loop B
Loop C
Start of 2-phase flow
Seal Leakage (gpm)
time [sec]



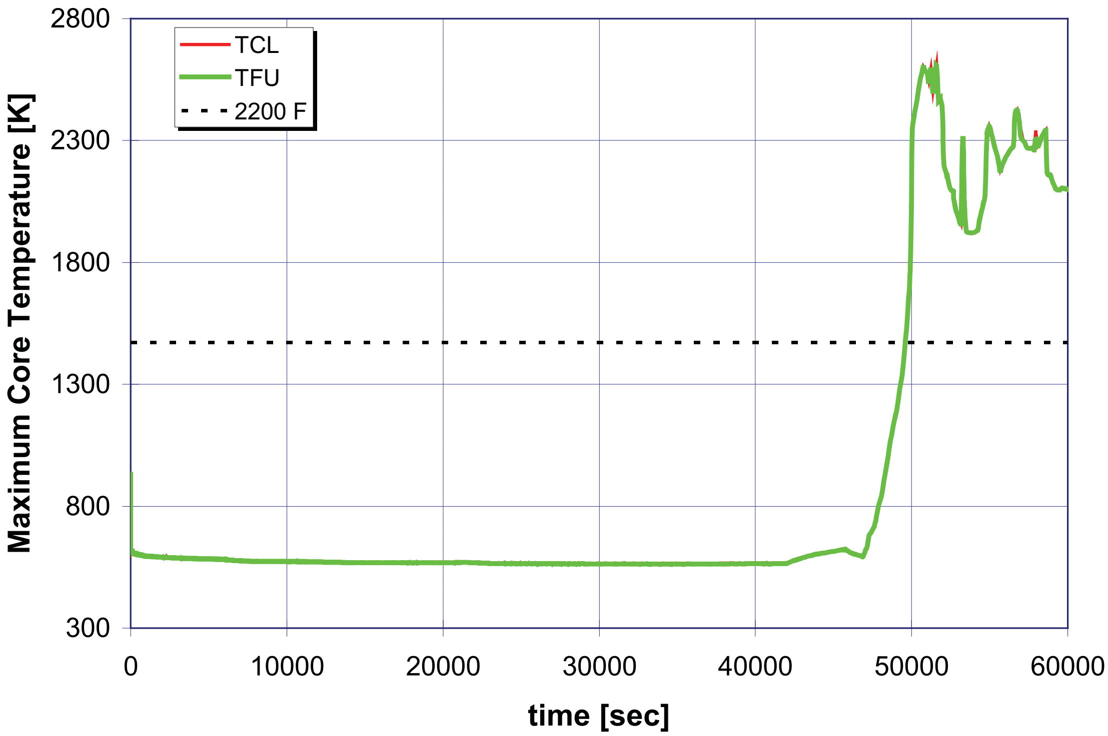
PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
Integral Water Mass [kg]
time [sec]



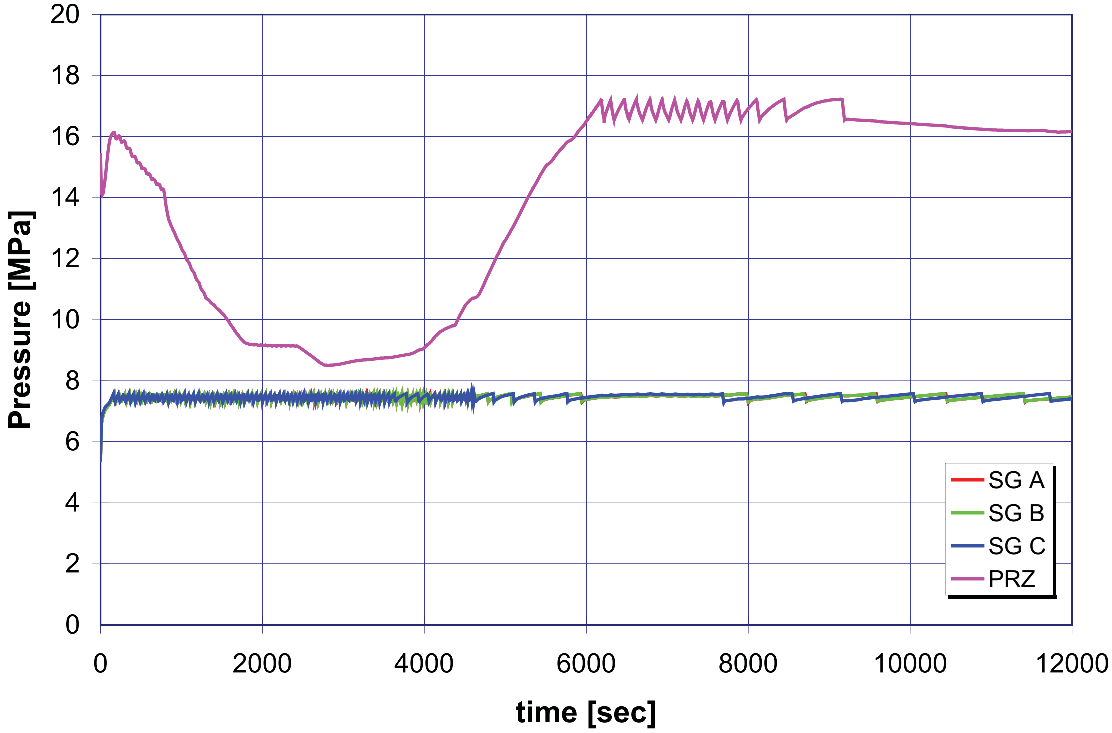
SG-A
Dome
Containment Pressure [bar]
time [sec]
0
0.5
1
1.5
2
2.5
3
0
10000
20000
30000
40000
50000
60000

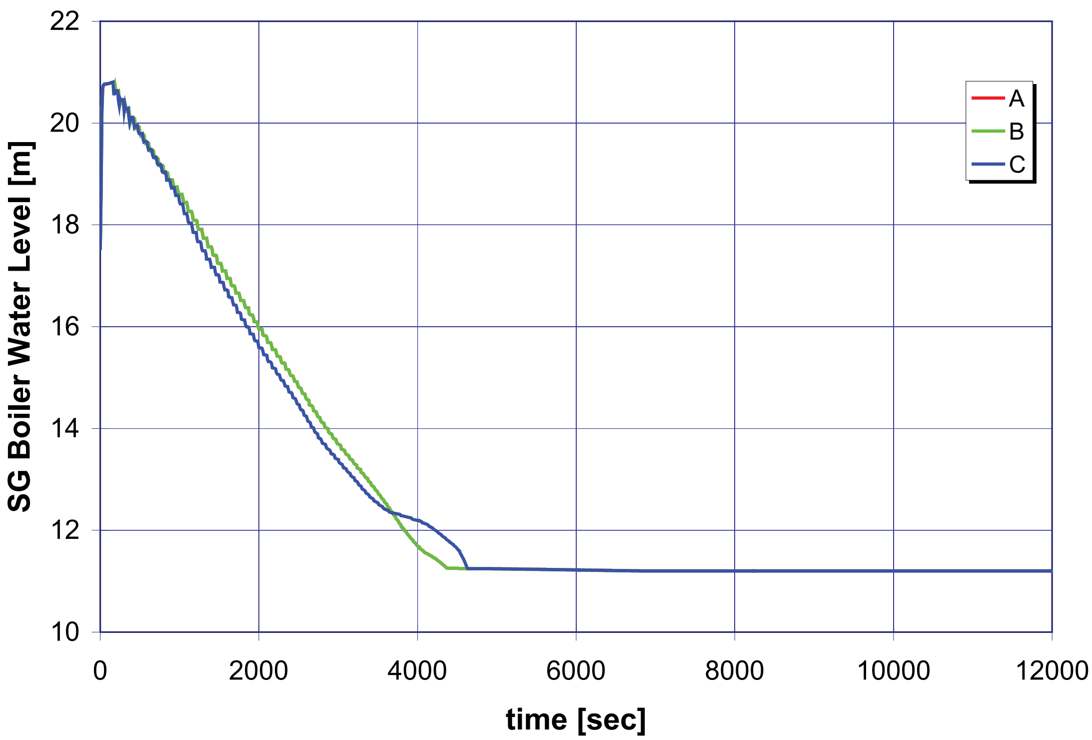


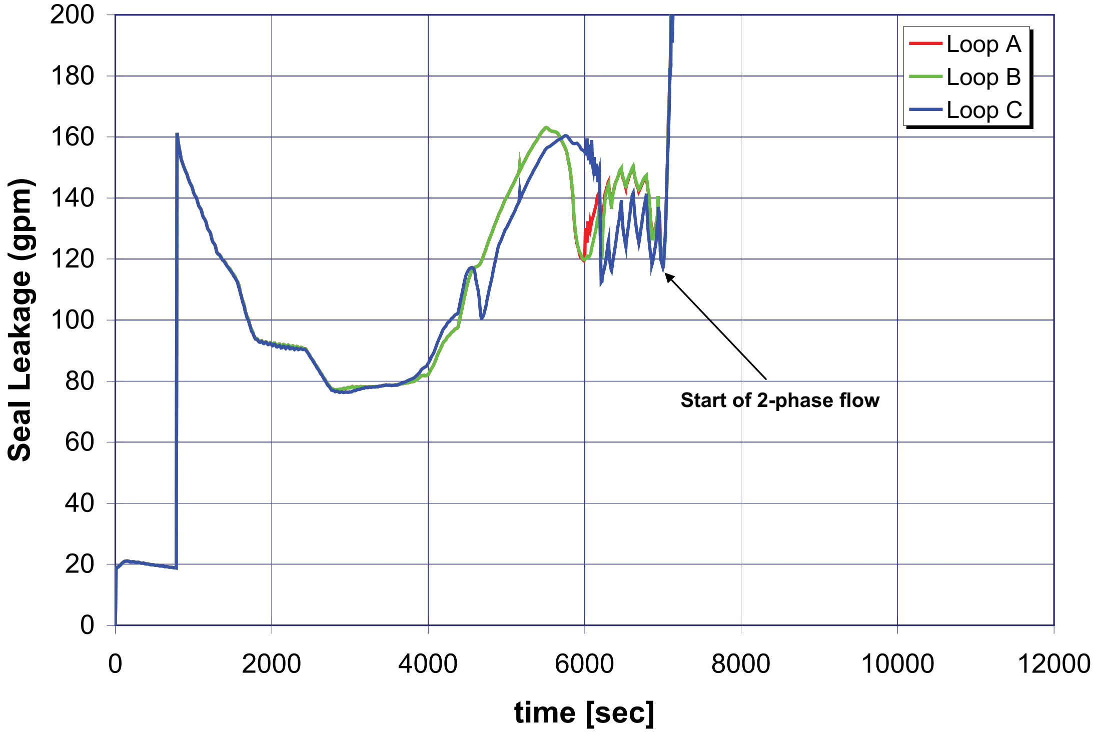
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
5
10
15
20
25
0
10000
20000
30000
40000
50000
60000


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
10000
20000
30000
40000
50000
60000
time [sec]

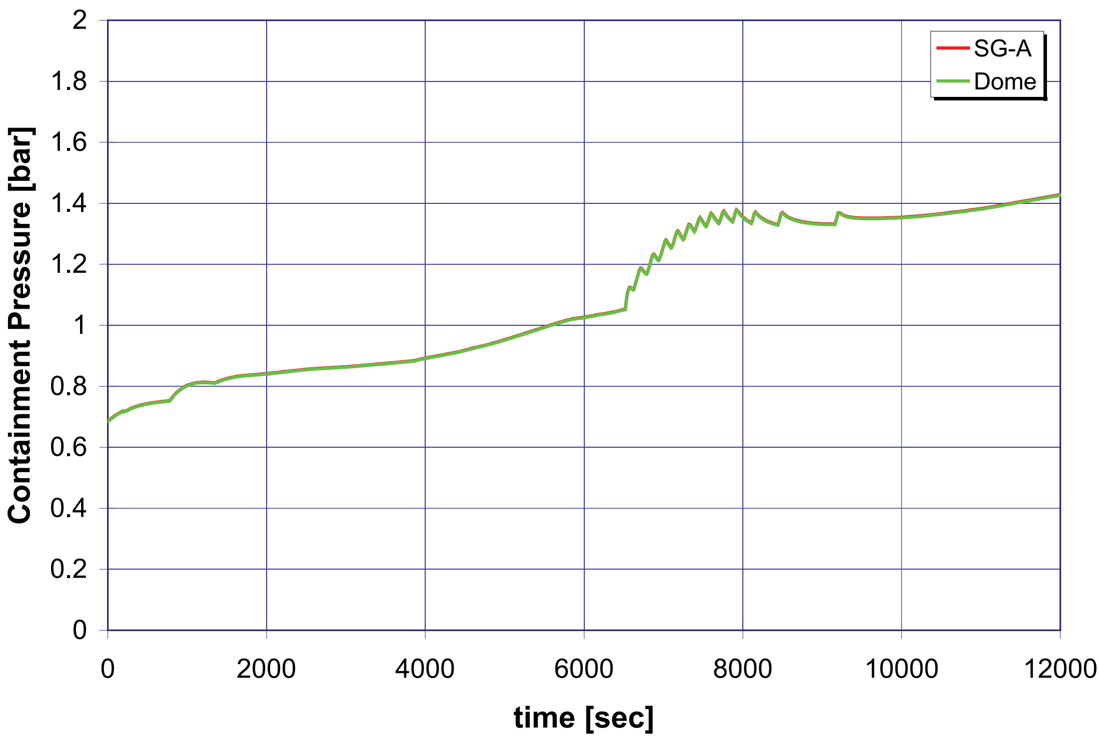
A.5.7 Case 7: Station Blackout without Turbine-Driven Auxiliary Feedwater (182 gpm)



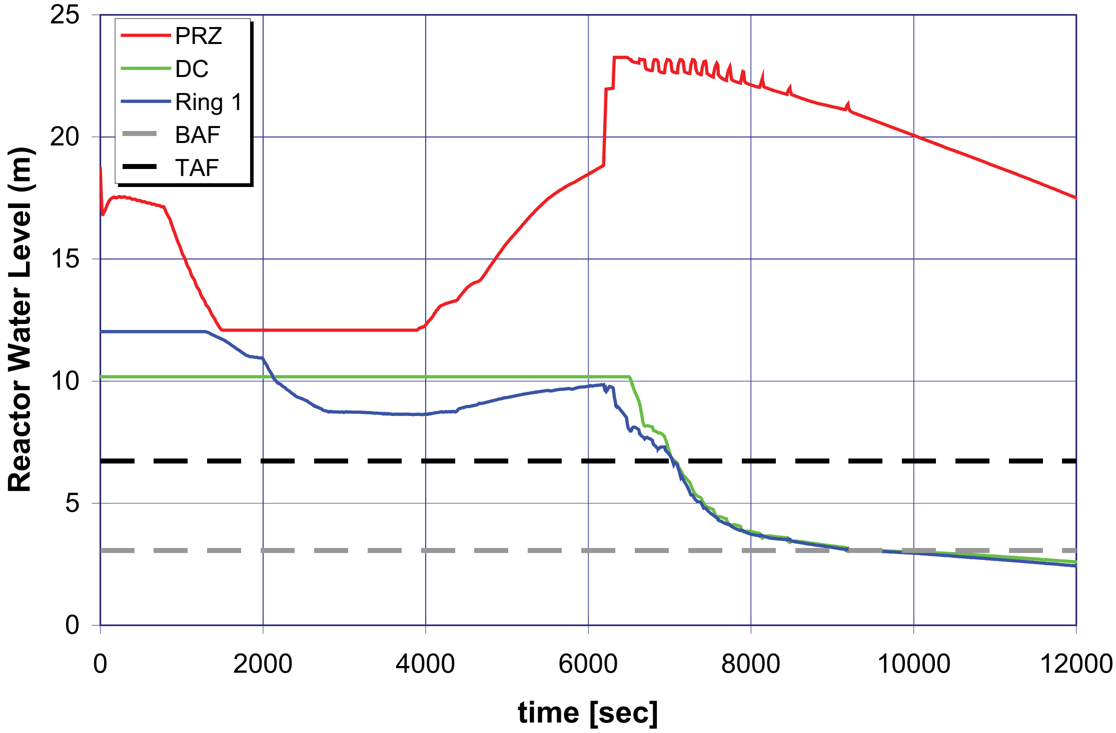




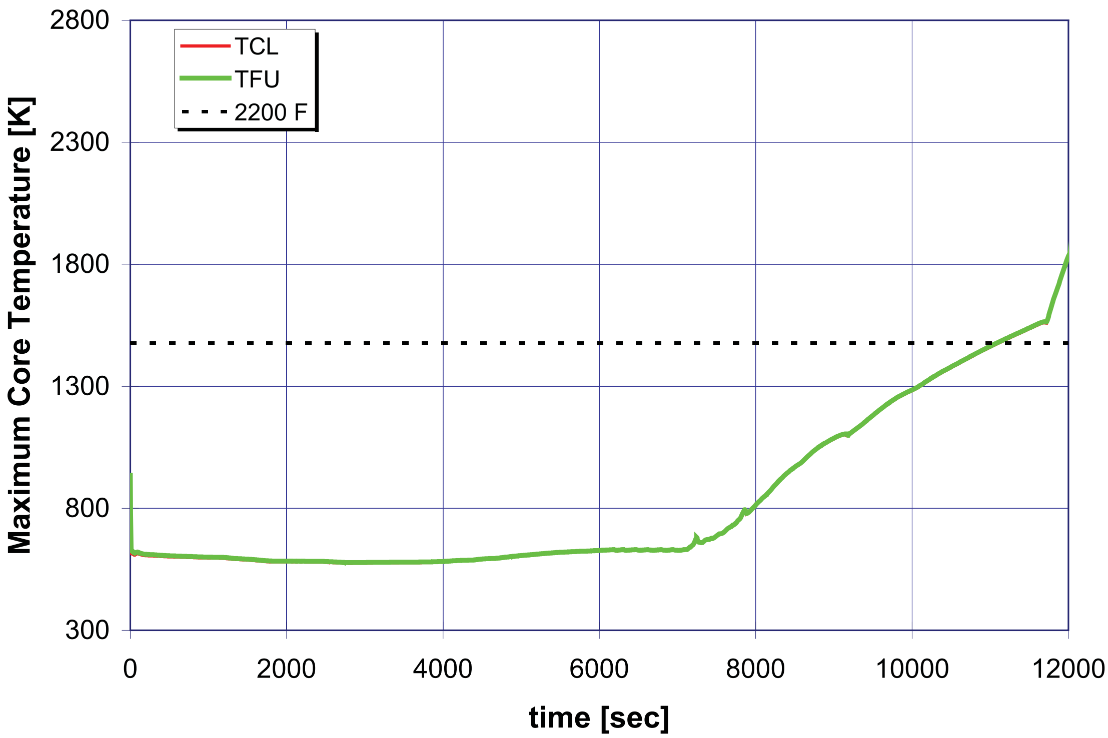
Seal Leakage (gpm)
time [sec]
Loop A
Loop B
Loop C
Start of 2-phase flow



Containment Pressure [bar]
time [sec]
SG-A
Dome

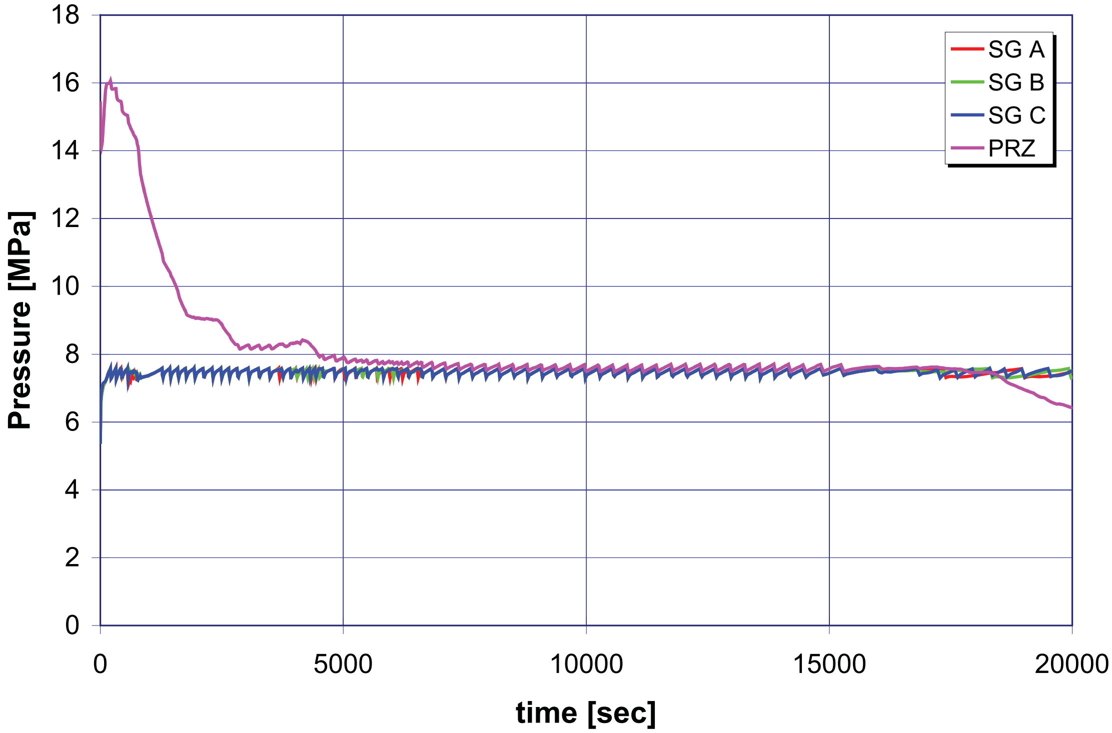


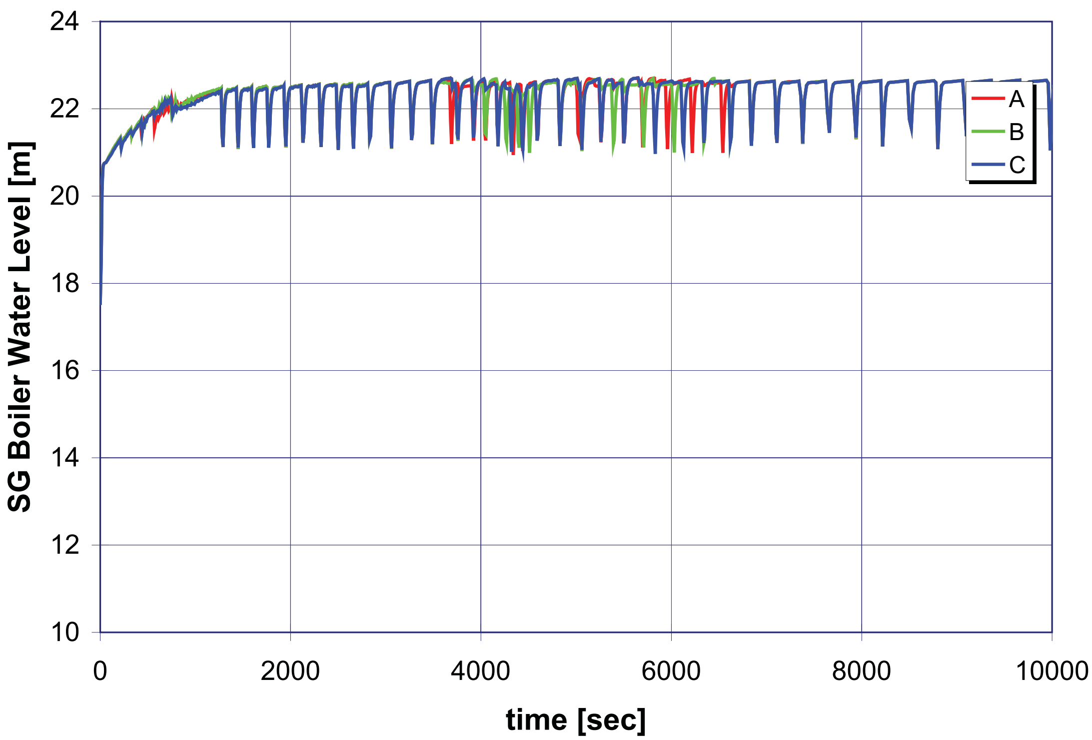
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
25
20
15
10
5
0
0
2000
4000
6000
8000
10000
12000

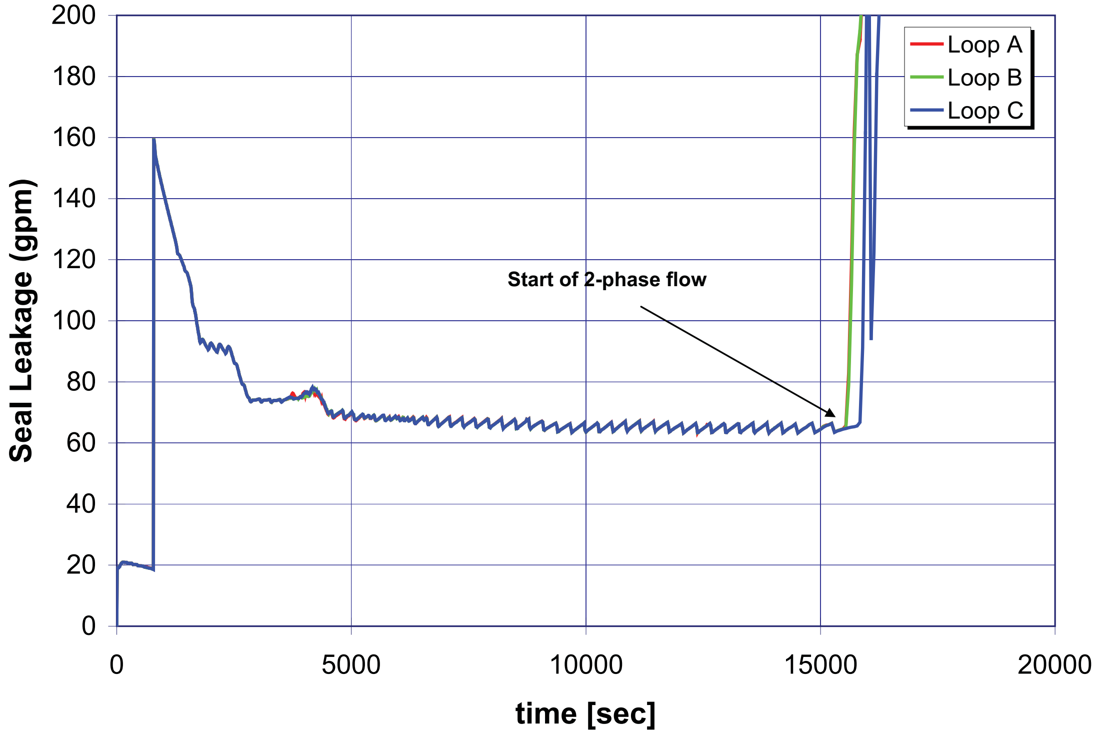


TCL
TFU
2200 F
Maximum Core Temperature [K]
time [sec]
2800
2300
1800
1300
800
300
0
2000
4000
6000
8000
10000
12000

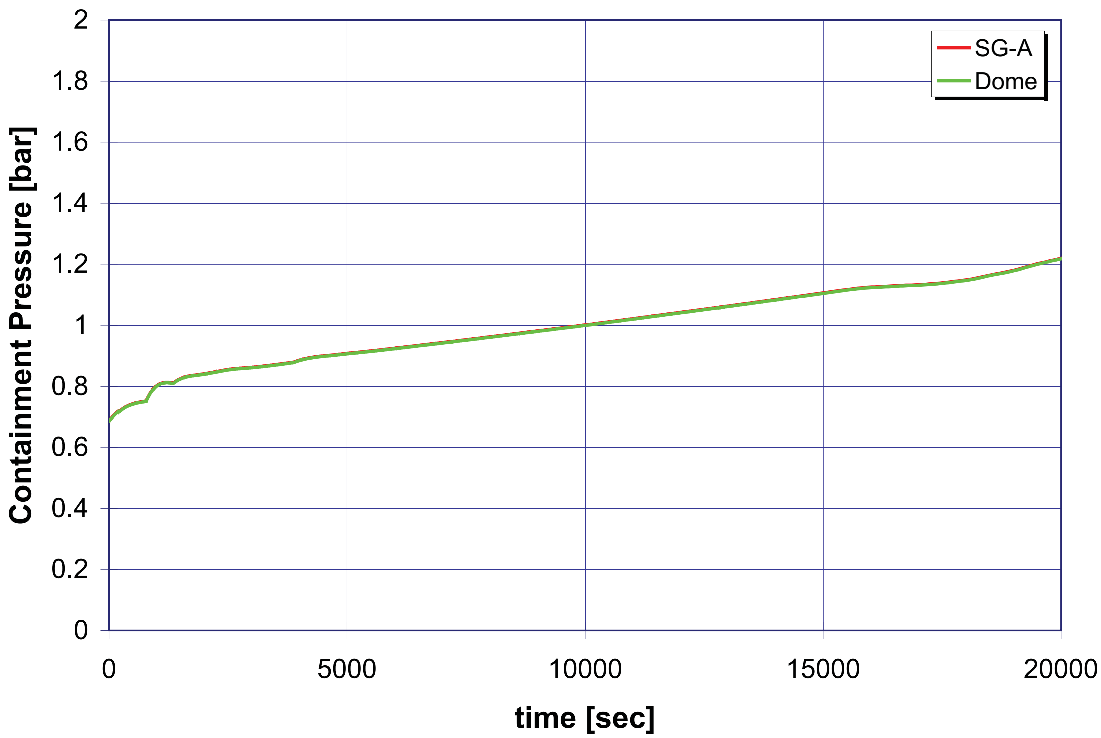
A.5.8 Case 8: Station Blackout with Turbine-Driven Auxiliary Feedwater (182 gpm)



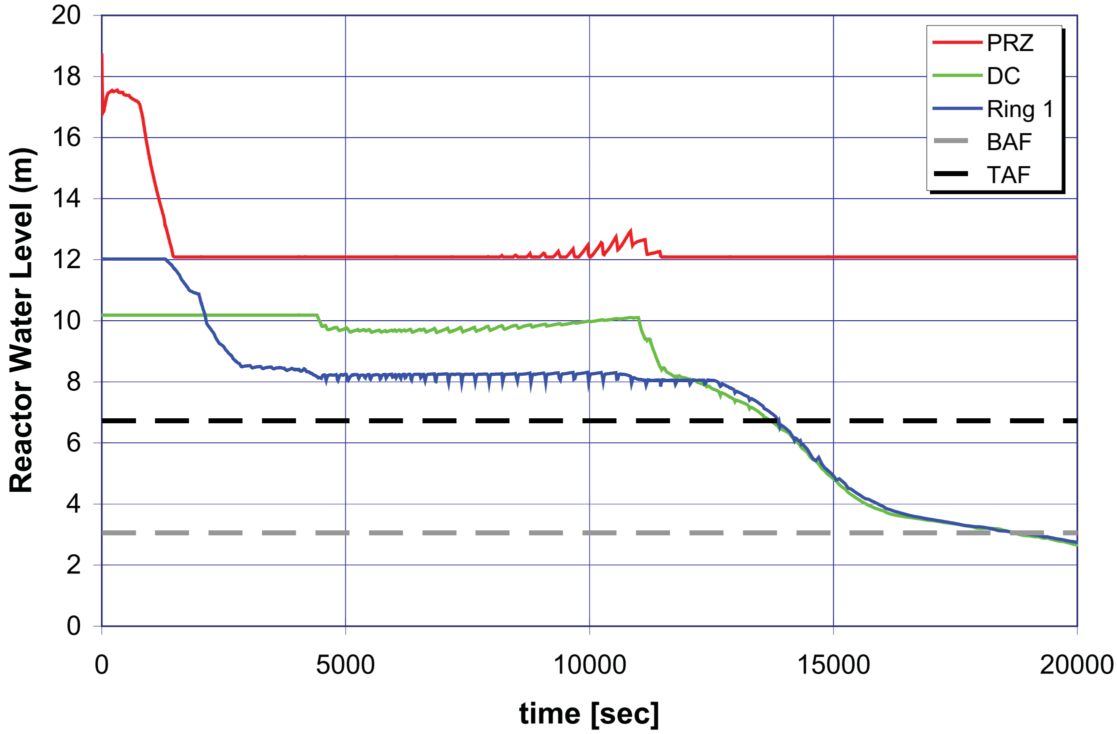




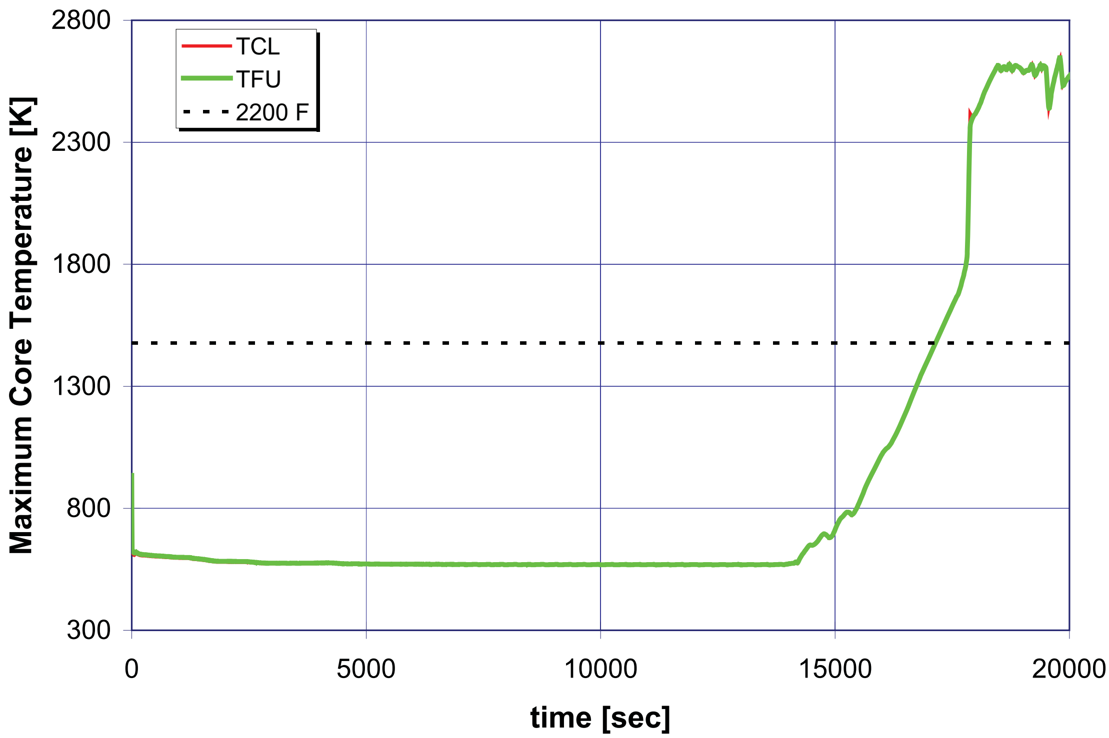
Seal Leakage (gpm)
time [sec]
Loop A
Loop B
Loop C
Start of 2-phase flow



Containment Pressure [bar]
time [sec]
SG-A
Dome

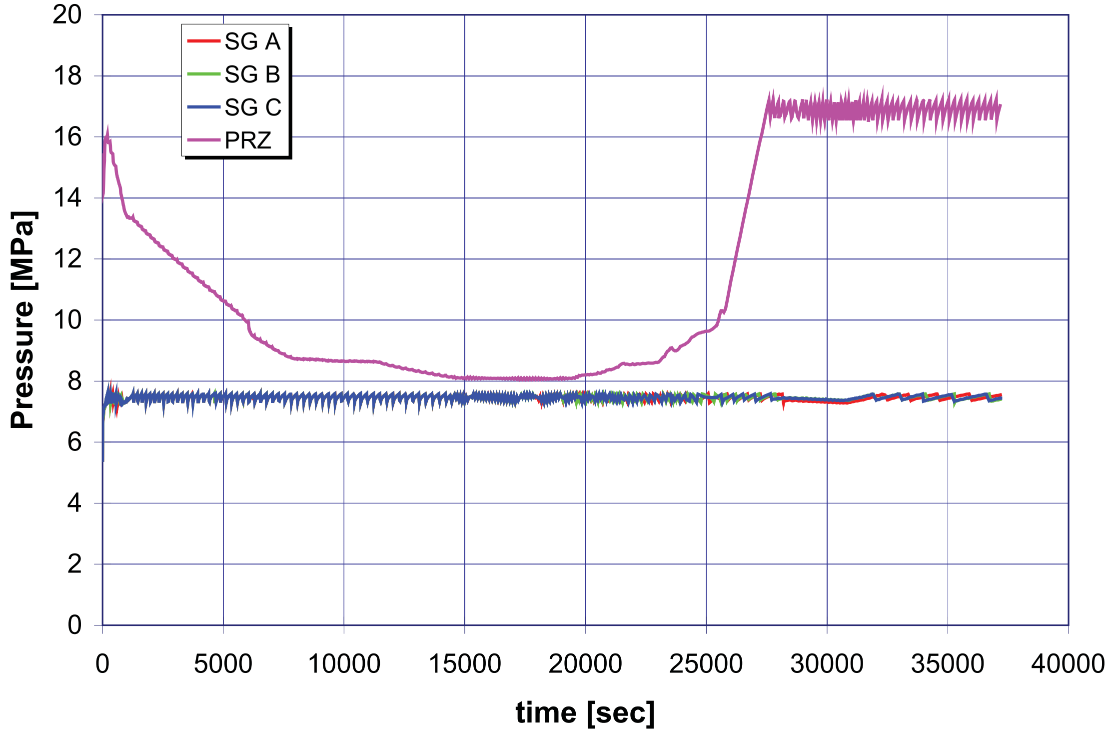


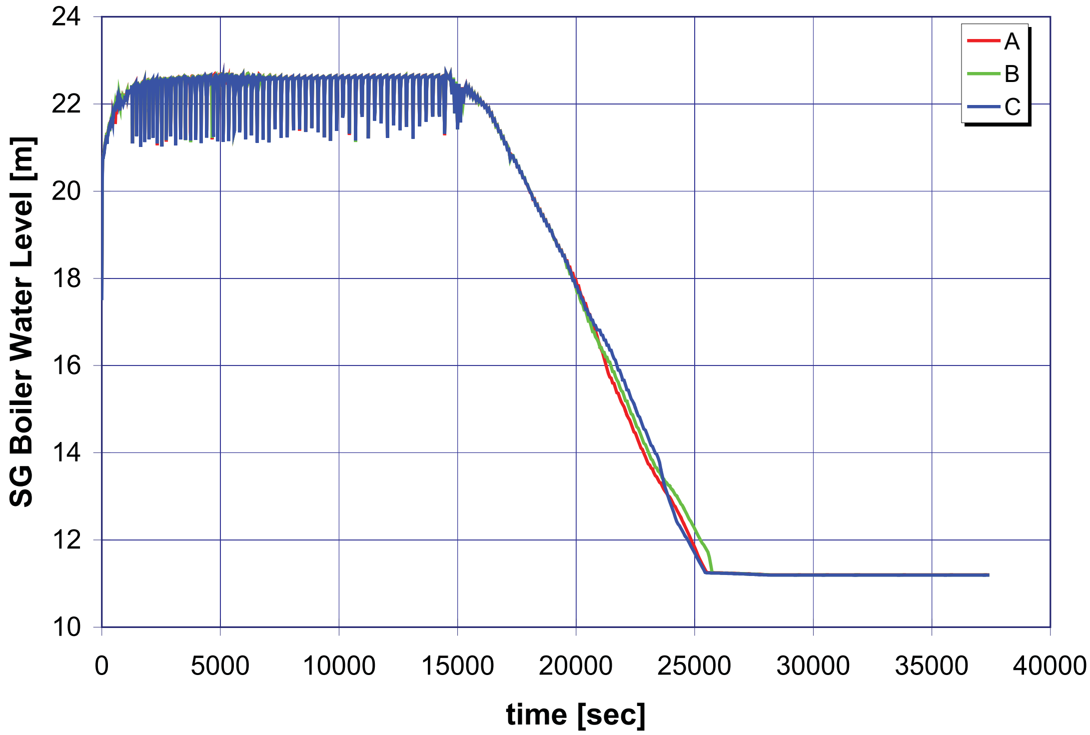
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
5000
10000
15000
20000

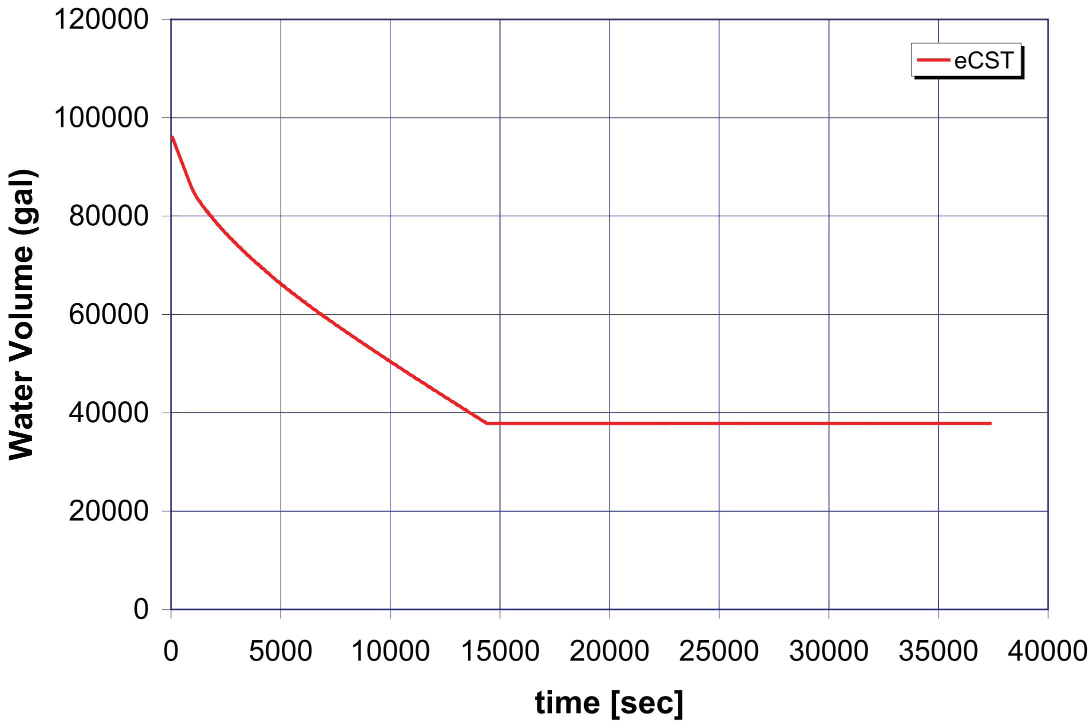


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
800
1300
1800
2300
2800
0
5000
10000
15000
20000

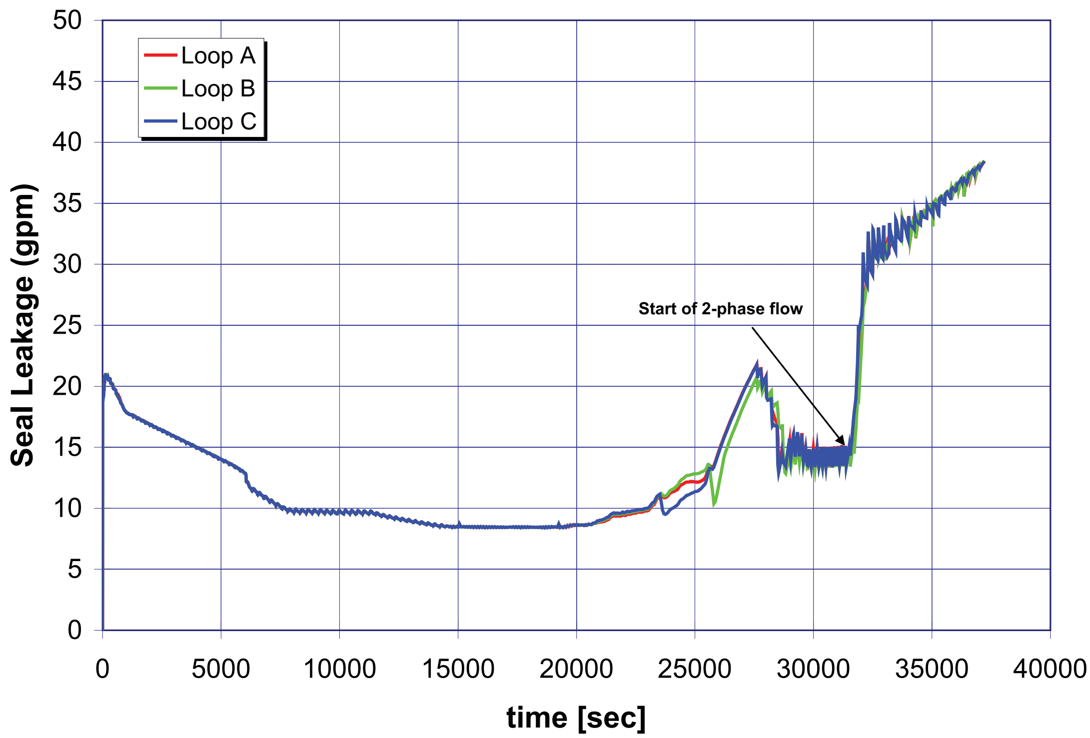
A.5.9 Case 9: Station Blackout with Turbine-Driven Auxiliary Feedwater (21 gpm) and 4-Hour Direct Current



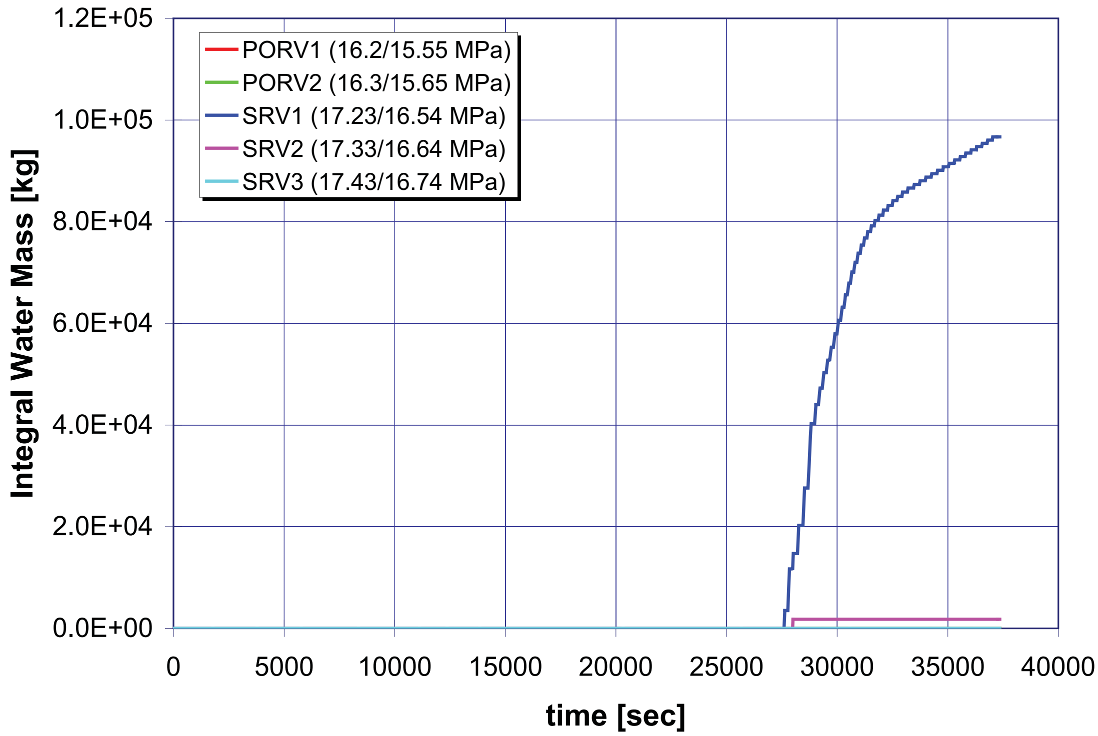




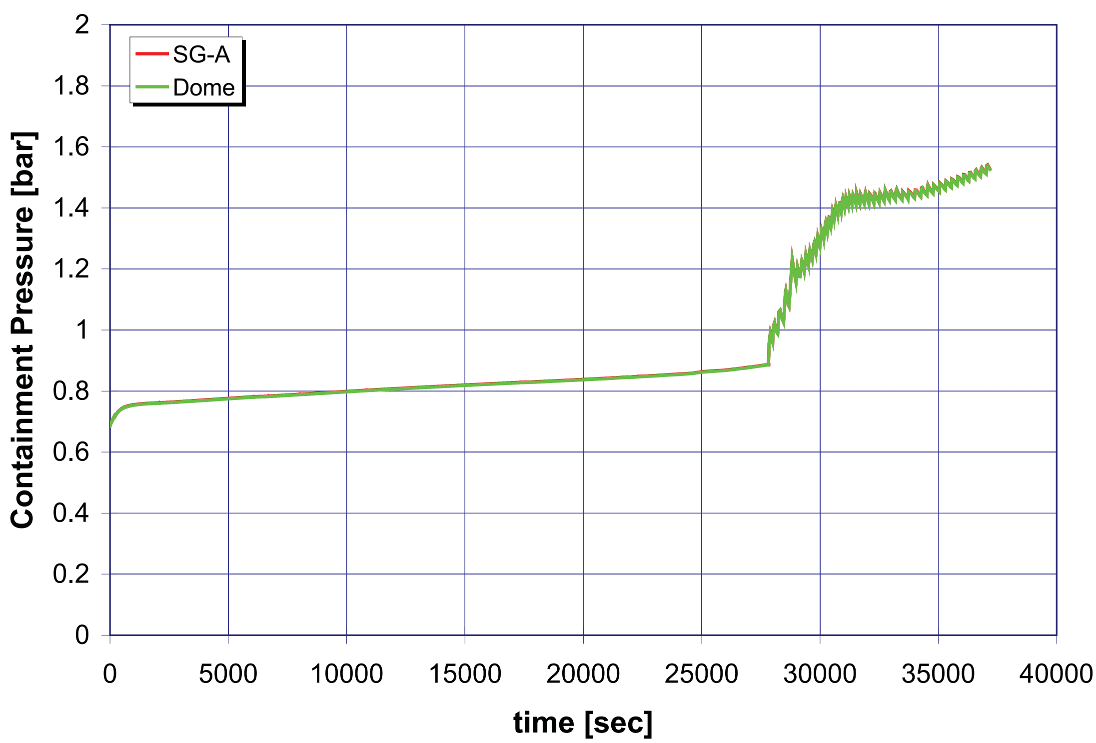
eCST
Water Volume (gal)
time [sec]
120000
100000
80000
60000
40000
20000
0
0
5000
10000
15000
20000
25000
30000
35000
40000



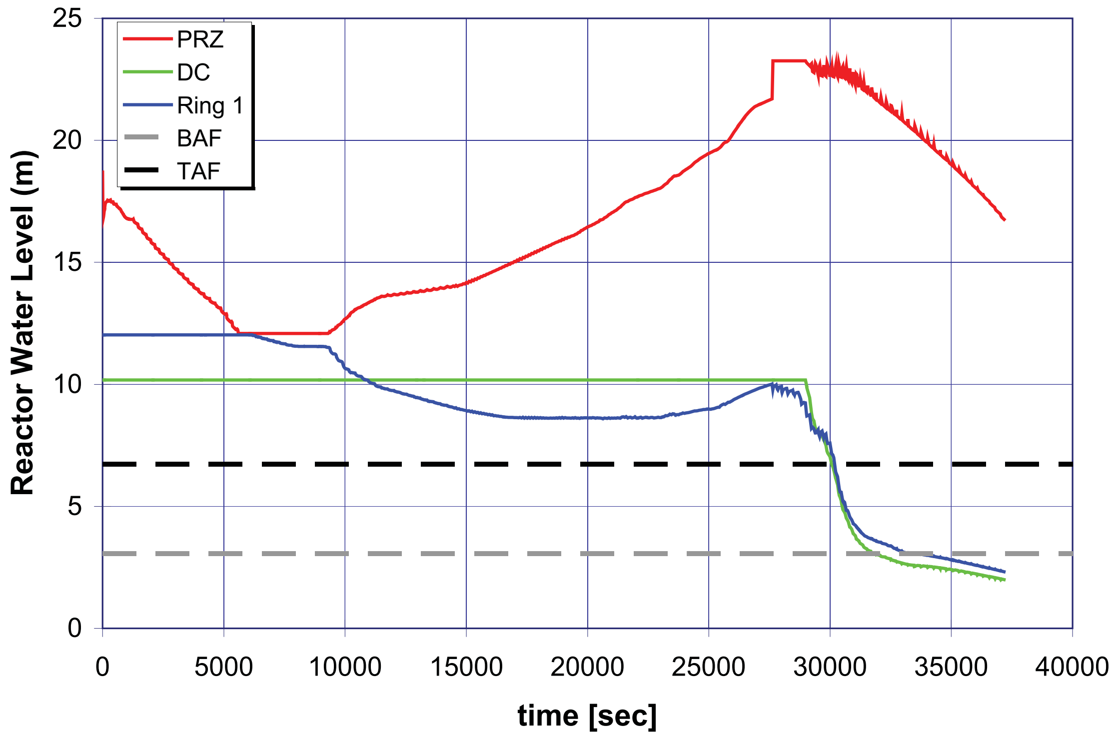
Loop A
Loop B
Loop C
Start of 2-phase flow
Seal Leakage (gpm)
time [sec]
50
45
40
35
30
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
35000
40000



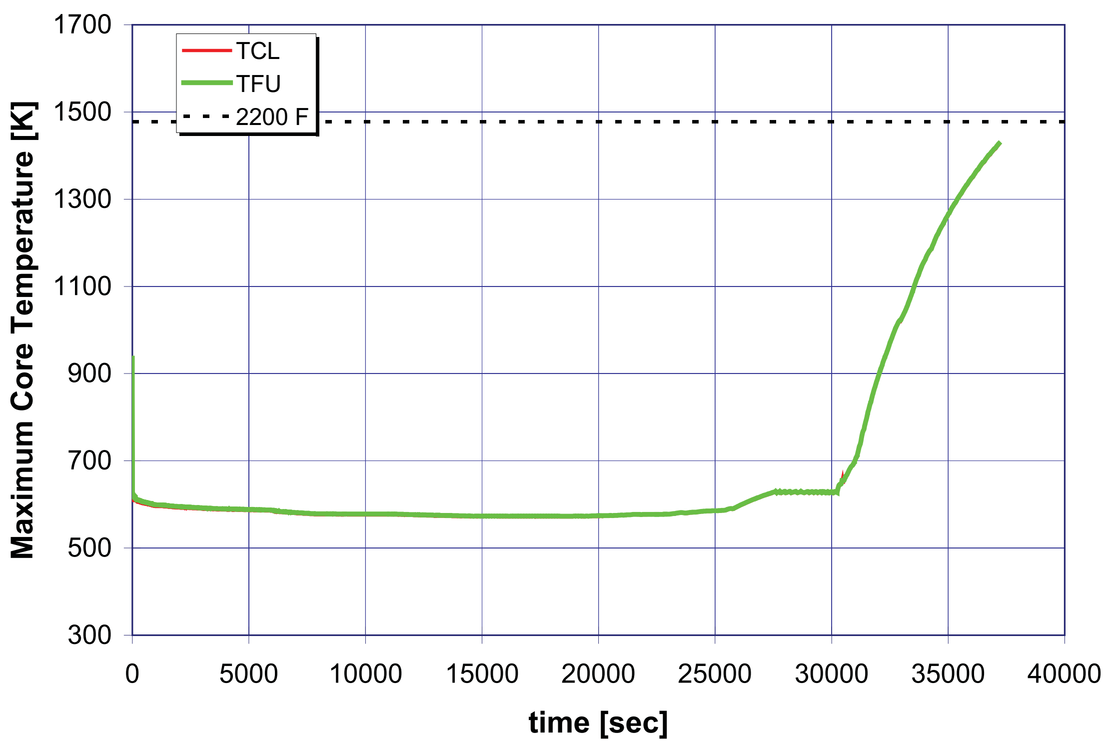
Integral Water Mass [kg]
1.2E+05
1.0E+05
8.0E+04
6.0E+04
4.0E+04
2.0E+04
0.0E+00
PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]



Containment Pressure [bar]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
SG-A
Dome
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]

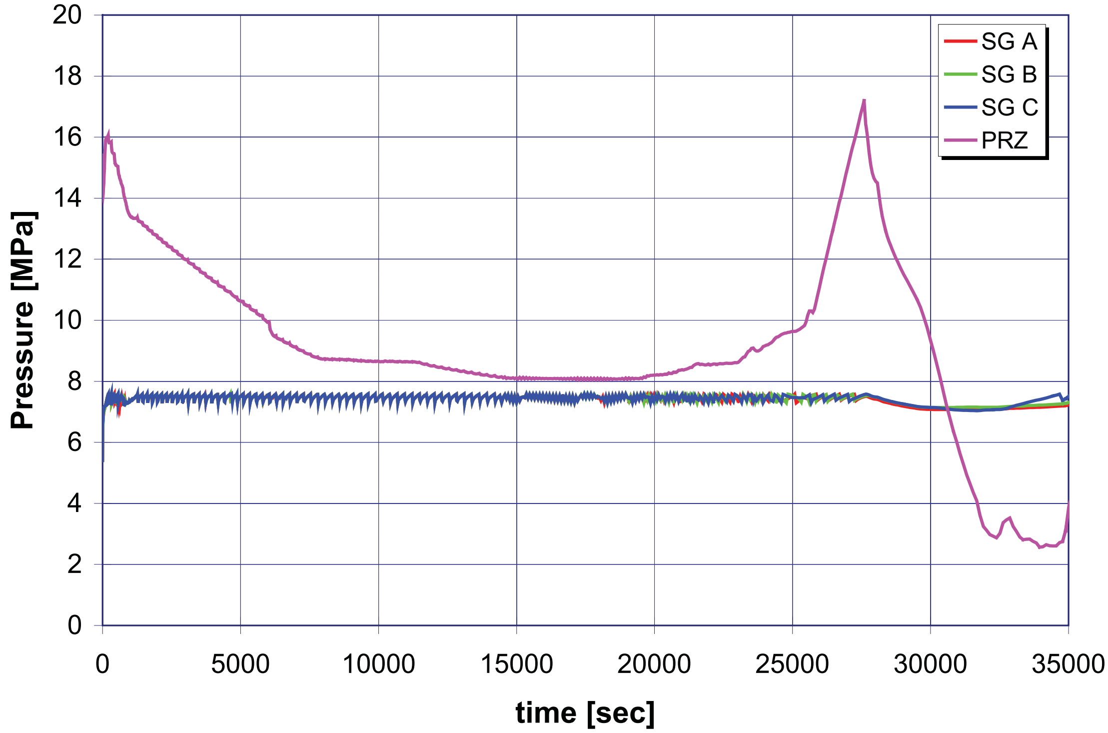


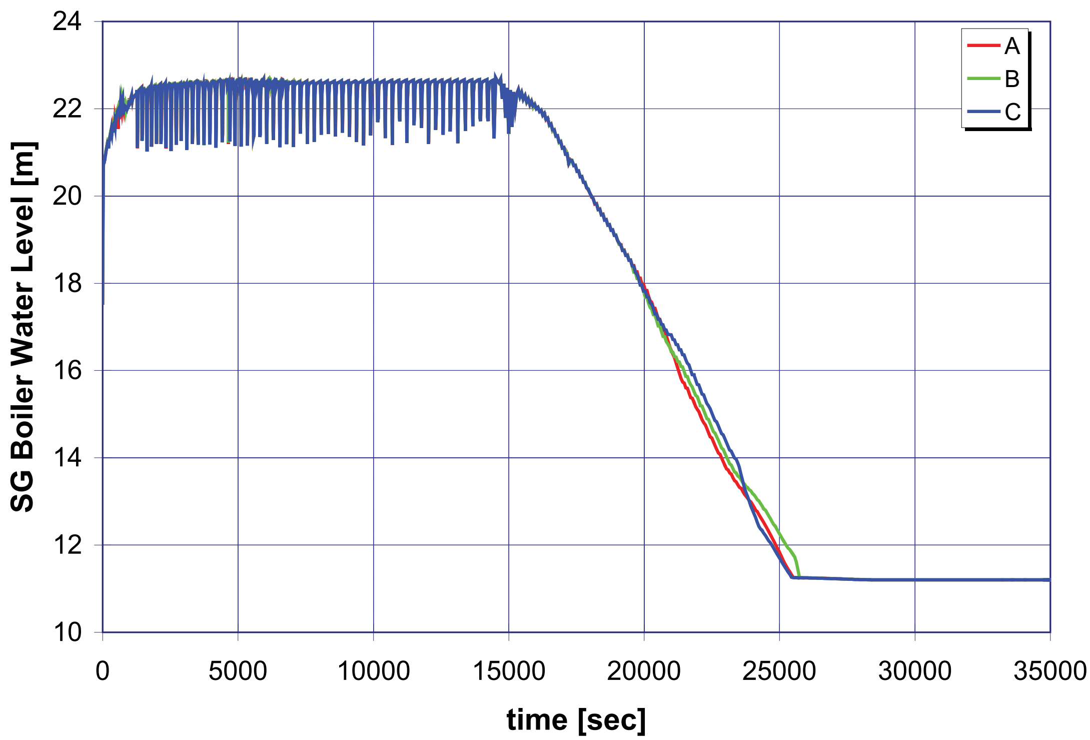
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]

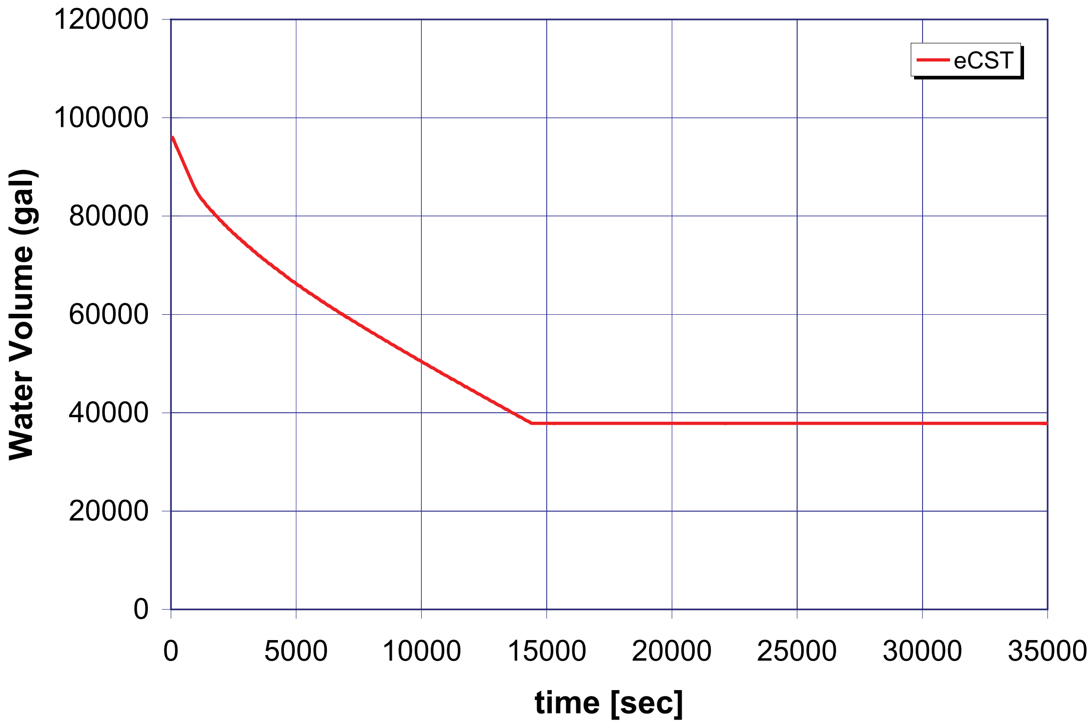


TCL
TFU
2200 F
Maximum Core Temperature [K]
time [sec]

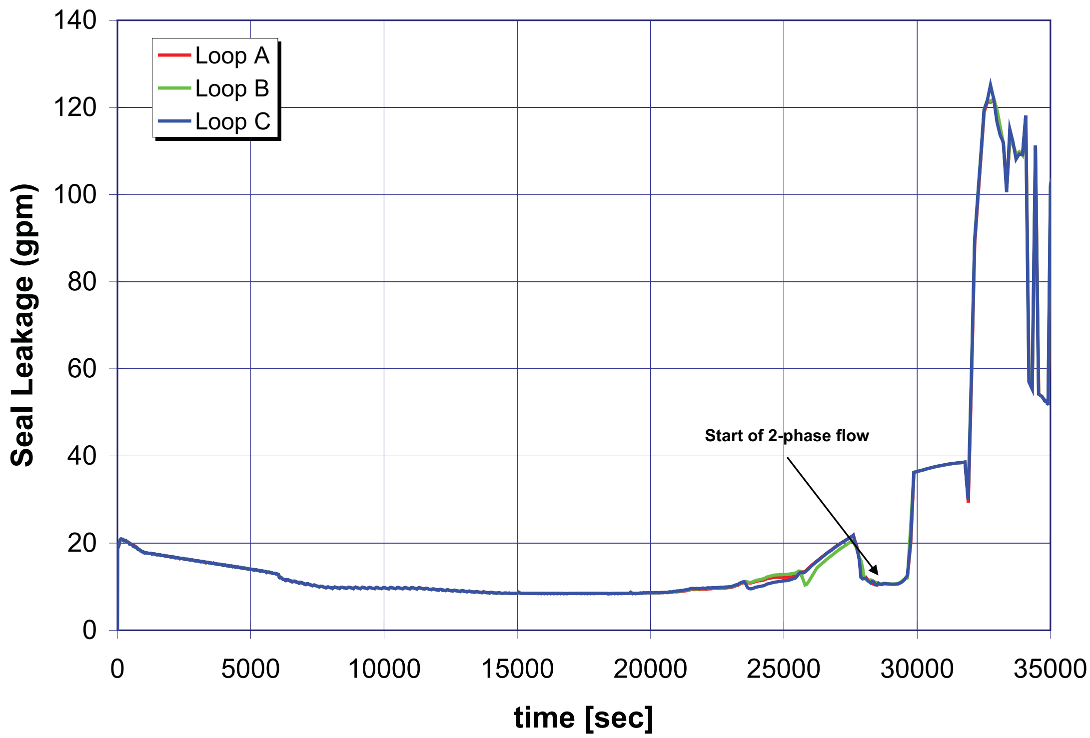
A.5.10 Case 10: Station Blackout with Turbine-Driven Auxiliary Feedwater (21 gpm) and 4-Hour Direct Current; Stuck-Open Relief Valve



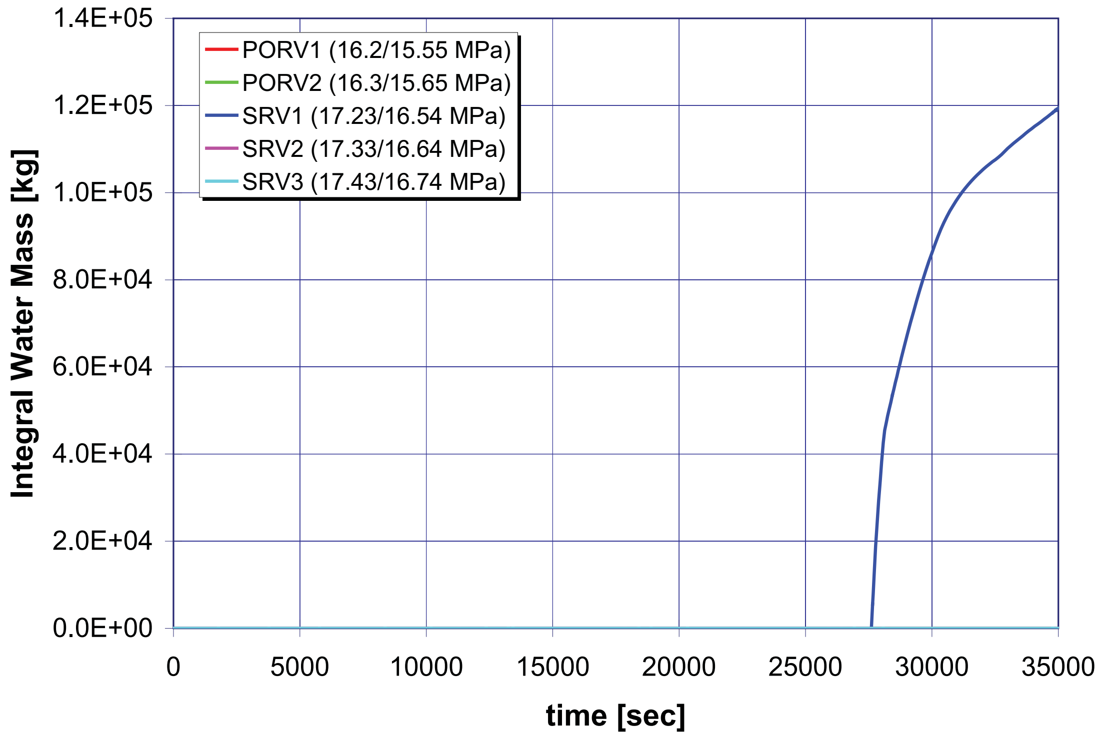


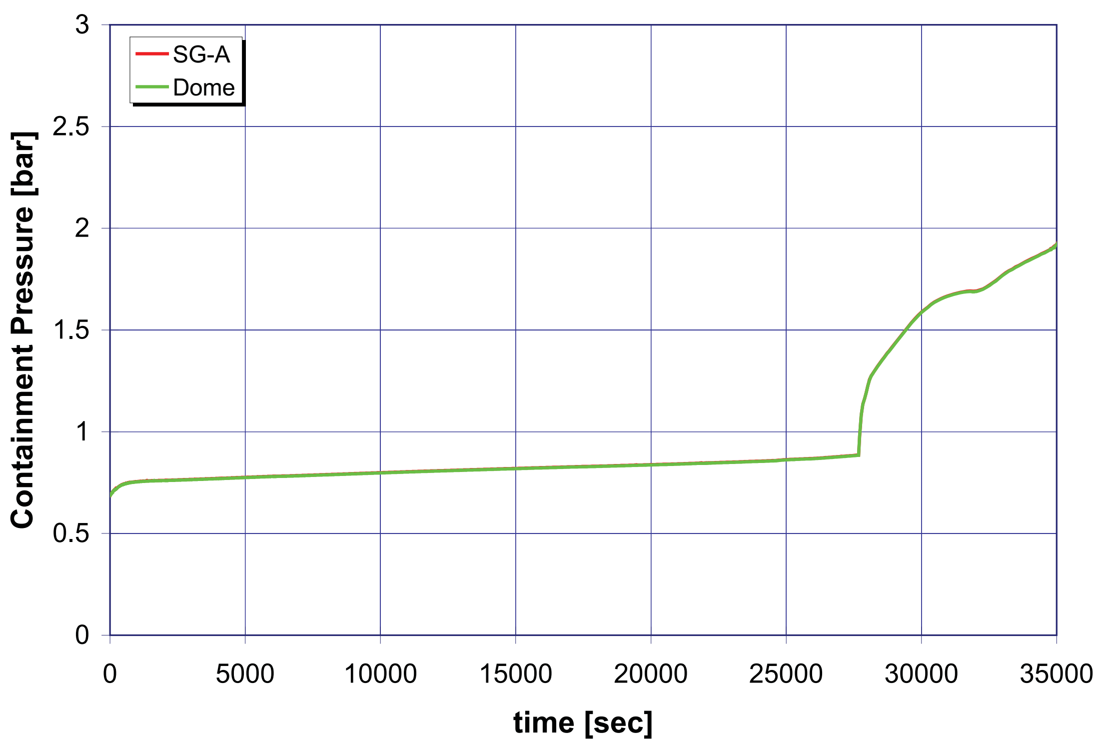


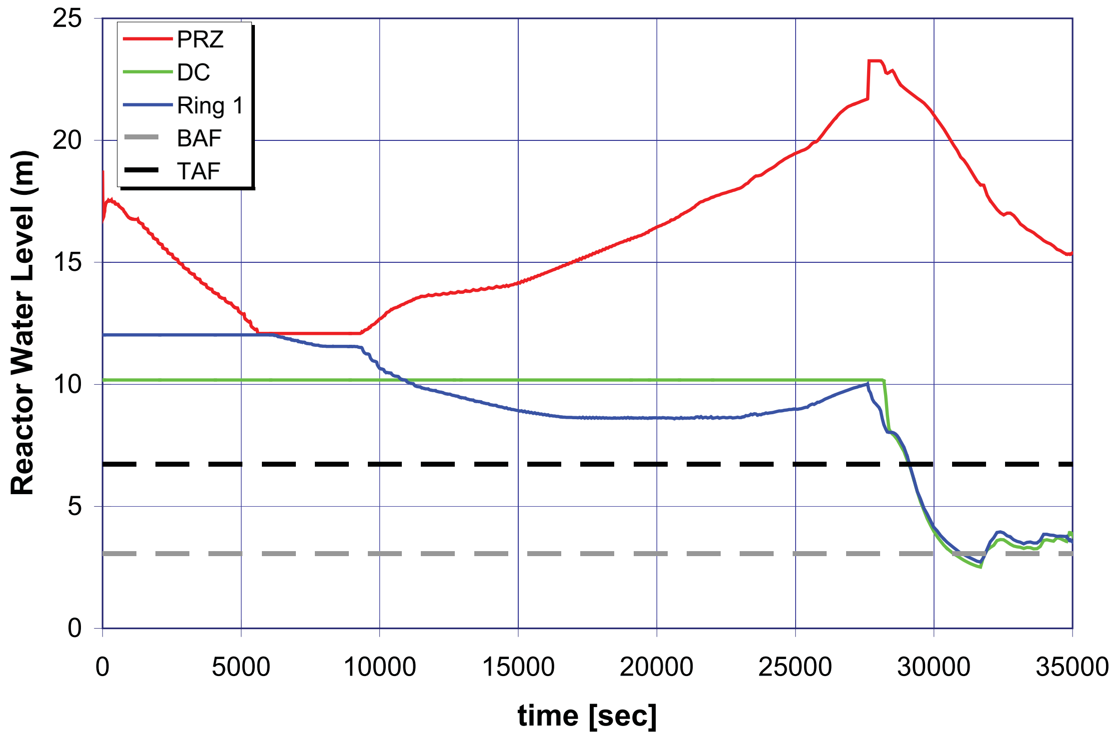
eCST
Water Volume (gal)
120000
100000
80000
60000
40000
20000
0
0
5000
10000
15000
20000
25000
30000
35000
time [sec]

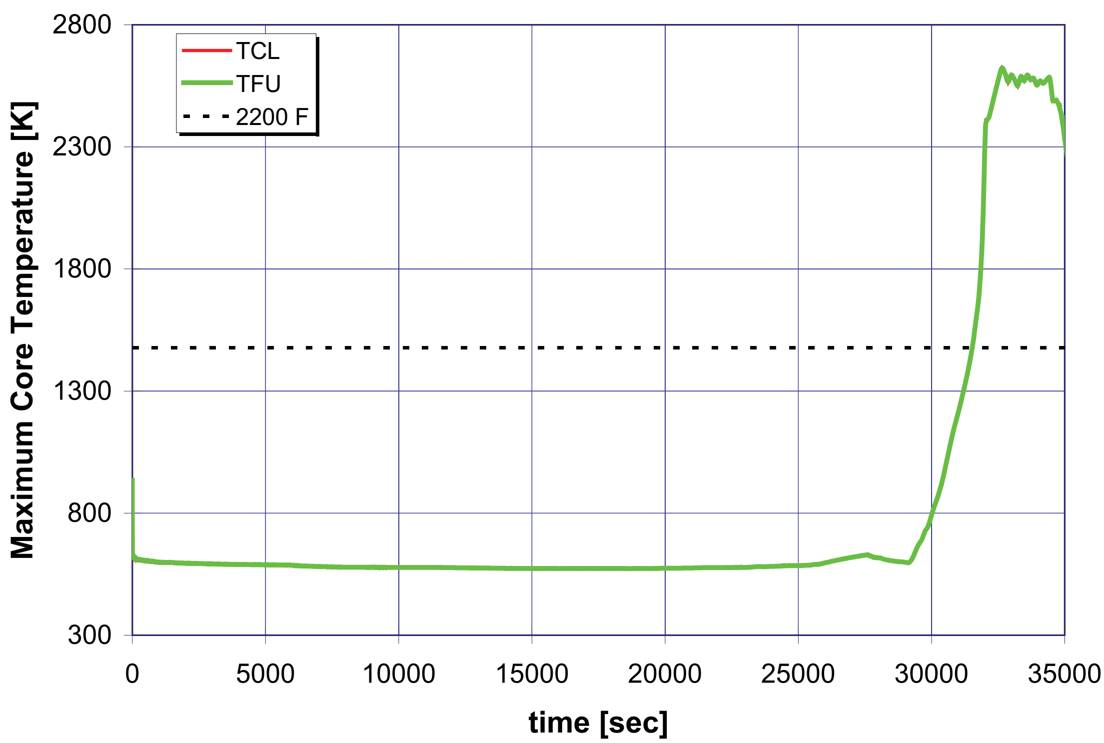


Loop A
Loop B
Loop C
Seal Leakage (gpm)
140
120
100
80
60
40
20
0
Start of 2-phase flow
0
5000
10000
15000
20000
25000
30000
35000
time [sec]


PORV1 (16.2/15.55 MPa)
PORV2 (16.3/15.65 MPa)
SRV1 (17.23/16.54 MPa)
SRV2 (17.33/16.64 MPa)
SRV3 (17.43/16.74 MPa)
Integral Water Mass [kg]
1.4E+05
1.2E+05
1.0E+05
8.0E+04
6.0E+04
4.0E+04
2.0E+04
0.0E+00
0
5000
10000
15000
20000
25000
30000
35000
time [sec]


SG-A
Dome
Containment Pressure [bar]
3
2.5
2
1.5
1
0.5
0
0
5000
10000
15000
20000
25000
30000
35000
time [sec]


PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
25
20
15
10
5
0
0
5000
10000
15000
20000
25000
30000
35000
time [sec]


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
5000
10000
15000
20000
25000
30000
35000
time [sec]

A.6 Pressurized-Water Reactor Medium- and Large-Break LOCA Initial Response

Analysis Summary

For all LOCA scenarios, containment fan coolers and containment sprays are available. The break is assumed to occur in the horizontal part of the cold leg in Loop A (pressurizer loop). Table 16 through Table 23 below provide results for this portion of the analysis.

Table 16 Surry MBLOCA and LBLOCA Results

<table>
<tr><th>Case</th><th>Break Size (inch)[4]</th><th># HHSI Pumps</th><th># Accum.</th><th># LHSI Pumps</th><th>AFW?[1]</th><th>Time of Initial Core Uncovery (hr)</th><th>Core Damage During Injection Phase? (hr)</th></tr>
<tr><td>9</td><td rowspan="6">2</td><td>1</td><td>0</td><td>0</td><td rowspan="4">Yes</td><td>0.42</td><td>No[2]</td></tr>
<tr><td>15</td><td>0</td><td>2</td><td>1</td><td>0.41</td><td>0.73</td></tr>
<tr><td>20</td><td>1</td><td>1</td><td>0</td><td>0.42</td><td>No[2]</td></tr>
<tr><td>21</td><td>1</td><td>0</td><td>1</td><td>0.42</td><td>No[3]</td></tr>
<tr><td>27</td><td>1</td><td>1</td><td>0</td><td rowspan="2">No</td><td>0.38</td><td>No[2]</td></tr>
<tr><td>29</td><td>1</td><td>0</td><td>1</td><td>0.38</td><td>No[3]</td></tr>
<tr><td>1</td><td rowspan="8">4</td><td>1</td><td>0</td><td>1</td><td rowspan="6">Yes</td><td>0.09</td><td>No</td></tr>
<tr><td>11</td><td>1</td><td>0</td><td>0</td><td>0.09</td><td>No[2]</td></tr>
<tr><td>12</td><td>0</td><td>0</td><td>1</td><td>0.10</td><td>0.27</td></tr>
<tr><td>13</td><td>0</td><td>1</td><td>1</td><td>0.10</td><td>0.27</td></tr>
<tr><td>14</td><td>0</td><td>2</td><td>1</td><td>0.10</td><td>No</td></tr>
<tr><td>22</td><td>1</td><td>1</td><td>0</td><td>0.09</td><td>No[2]</td></tr>
<tr><td>25</td><td>1</td><td>0</td><td>1</td><td rowspan="2">No</td><td>0.09</td><td>No</td></tr>
<tr><td>28</td><td>1</td><td>1</td><td>0</td><td>0.09</td><td>No[2]</td></tr>
<tr><td>2</td><td rowspan="8">6</td><td>1</td><td>0</td><td>1</td><td rowspan="5">Yes</td><td>0.04</td><td>No</td></tr>
<tr><td>5</td><td>0</td><td>0</td><td>1</td><td>0.04</td><td>0.16</td></tr>
<tr><td>6</td><td>0</td><td>1</td><td>1</td><td>0.04</td><td>No</td></tr>
<tr><td>7</td><td>1</td><td>0</td><td>0</td><td>0.07</td><td>0.28</td></tr>
<tr><td>8</td><td>1</td><td>1</td><td>0</td><td>0.08</td><td>No[2]</td></tr>
<tr><td>16</td><td>1</td><td>0</td><td>1</td><td rowspan="3">No</td><td>0.04</td><td>No</td></tr>
<tr><td>17</td><td>1</td><td>1</td><td>0</td><td>0.06</td><td>No[2]</td></tr>
<tr><td>26</td><td>0</td><td>1</td><td>1</td><td>0.04</td><td>No</td></tr>
<tr><td>3</td><td rowspan="3">8</td><td>1</td><td>0</td><td>1</td><td rowspan="7">Yes</td><td>0.02</td><td>No</td></tr>
<tr><td>18</td><td>1</td><td>1</td><td>0</td><td>0.01</td><td>No[2]</td></tr>
<tr><td>23</td><td>0</td><td>1</td><td>1</td><td>0.03</td><td>No</td></tr>
<tr><td>4</td><td rowspan="3">10</td><td>1</td><td>0</td><td>1</td><td>0.01</td><td>No</td></tr>
<tr><td>19</td><td>1</td><td>1</td><td>0</td><td>0.01</td><td>No[2]</td></tr>
<tr><td>24</td><td>0</td><td>1</td><td>1</td><td>0.02</td><td>No</td></tr>
<tr><td>10</td><td>Double-ended</td><td>0</td><td>0</td><td>1</td><td>0.02</td><td>No</td></tr>
</table>

1 Conventionally, AFW is not needed for success for an LBLOCA; the break size is large enough to remove decay heat and the system fully depressurizes.

2 Note that core damage eventually occurs (or would occur, in cases in which the calculation was terminated early) because of the inability to go to HHSI recirculation (due to the unavailability of LHSI) or, more directly, from the lack of a low-pressure injection source. Recall that the present calculations are focused only on the injection phase success criteria.

3 For these cases, core damage eventually occurs because HHSI recirculation is not modeled, and the pressure is not sufficiently low prior to core damage to allow for LHSI recirculation.

4 2 in. = 5.1 cm; 4 in. = 10.2 cm; 6 in. = 15.2 cm; 8 in. = 20.3 cm; 10 in. = 25.4 cm.

Table 17 Surry MBLOCA and LBLOCA Key Timings (2-Inch Breaks)

Event	Case 9 (hr)	Case 15 (hr)	Case 20 (hr)	Case 21 (hr)	Case 27 (hr)	Case 29 (hr)
Reactor trip	0.01	0.003	0.01	0.01	0.01	0.01
HHSI injection	0.01	-	0.01	0.01	0.01	0.01
RCP trip (10% void)	0.28	0.07	0.28	0.28	0.18	0.17
First actuation of containment sprays	1.14	-	1.21	1.14	0.94	0.94
Core uncovery (water <TAF)	0.42	0.41	0.42	0.42	0.38	0.38
LHSI injection	-	-	-	6.39	-	6.17
Maximum cladding temperature timing (max. temperature)	0.44 (592 K)	0.73 (1,477 K[1])	0.44 (592 K)	0.44 (592 K)	0.40 (592 K)	0.40 (592 K)
Core covered	0.87	N/A	0.8	0.87	0.75	0.75

[1] Actual peak temperature would be higher; this value corresponds to the surrogate used in this project for core damage, 2,200 °F (1,204 °C).

Table 18 Surry MBLOCA and LBLOCA Key Timings (4-Inch Breaks Group 1)

Event	Case 1 (hr)	Case 11 (hr)	Case 12 (hr)	Case 13 (hr)
Reactor trip	0.003	0.003	0.003	0.003
HHSI injection	0.003	0.004	-	-
RCP trip (10% void)	0.04	0.04	0.04	0.04
First actuation of containment sprays	0.08	0.08	0.07	0.07
Core uncovery (water <TAF)	0.09	0.09	0.10	0.10
LHSI injection	0.29	-	0.33	0.45
Maximum cladding temperature timing (max. temperature)	0.34 (982 K)	0.53 (1,209 K)	0.27 (1,477 K[1])	0.27 (1,477 K[1])
Core covered	0.38	>0.83	N/A	N/A

[1] Actual peak temperature would be higher; this value corresponds to the surrogate used in this project for core damage, 2,200 °F (1,204 °C).

Table 19 Surry MBLOCA and LBLOCA Key Timings (4-Inch Breaks Group 2)

Event	Case 14 (hr)	Case 22 (hr)	Case 25 (hr)	Case 28 (hr)
Reactor trip	0.003	0.003	0.003	0.003
HHSI injection	-	0.004	0.004	0.004
RCP trip (10% void)	0.04	0.04	0.04	0.03
First actuation of containment sprays	0.07	0.08	0.08	0.07
Core uncovery (water <TAF)	0.10	0.09	0.09	0.09
LHSI injection	0.73	-	0.30	-
Maximum cladding temperature timing (max. temperature)	0.73 (1,183 K)	0.21 (807 K)	0.32 (1,054 K)	0.26 (721 K)
Core covered	0.79	0.39	0.39	0.41

Table 20 Surry MBLOCA and LBLOCA Key Timings (6-Inch Breaks Group 1)

Event	Case 2 (hr)	Case 5 (hr)	Case 6 (hr)	Case 7 (hr)
Reactor trip	0.002	0.002	0.002	0.002
HHSI injection	0.002	-	-	0.002
RCP trip (10% void)	0.02	0.02	0.02	0.02
First actuation of containment sprays	0.03	0.03	0.03	0.03
Core uncovery (water <TAF)	0.04	0.04	0.04	0.07
LHSI injection	0.13	0.14	0.18	-
Maximum cladding temperature timing (maximum temperature)	0.15 (774 K)	0.16 (1,477 K[1])	0.16 (990 K)	0.28 (1,477K[1])
Core covered	0.19	N/A	0.20	N/A

[1] Actual peak temperature would be higher; this value corresponds to the surrogate used in this project for core damage, 2,200 °F (1,204 °C).

Table 21 Surry MBLOCA and LBLOCA Key Timings (6-Inch Breaks Group 2)

Event	Case 8 (hr)	Case 16 (hr)	Case 17 (hr)	Case 26 (hr)
Reactor trip	0.002	0.002	0.002	0.002
HHSI injection	0.002	0.002	0.002	-
RCP trip (10% void)	0.02	0.02	0.02	0.02
First actuation of containment sprays	0.03	0.03	0.03	0.03
Core uncovery (water <TAF)	0.08	0.04	0.06	0.04
LHSI injection	-	0.13	-	0.18
Maximum cladding temperature timing (maximum temperature)	0.04 (592 K)	0.152 (775 K)	0.04 (575 K)	0.13 (931 K)
Core covered	0.10	0.19	0.12	0.22

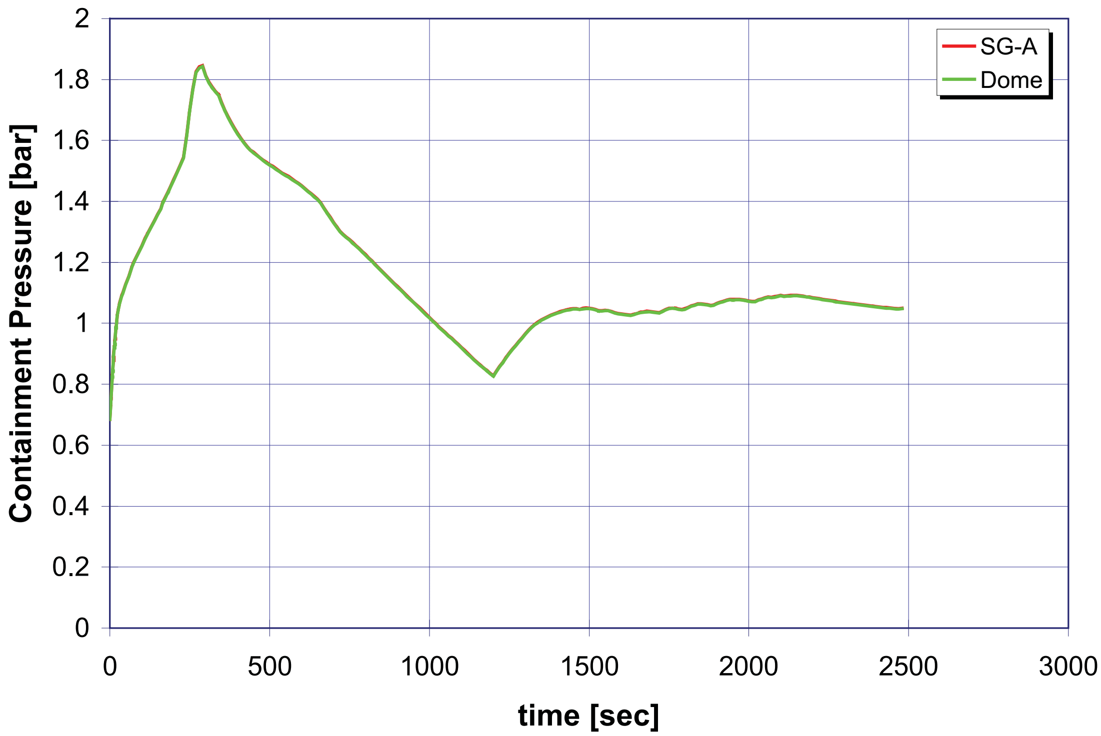
Table 22 Surry MBLOCA and LBLOCA Key Timings (8-Inch Breaks)

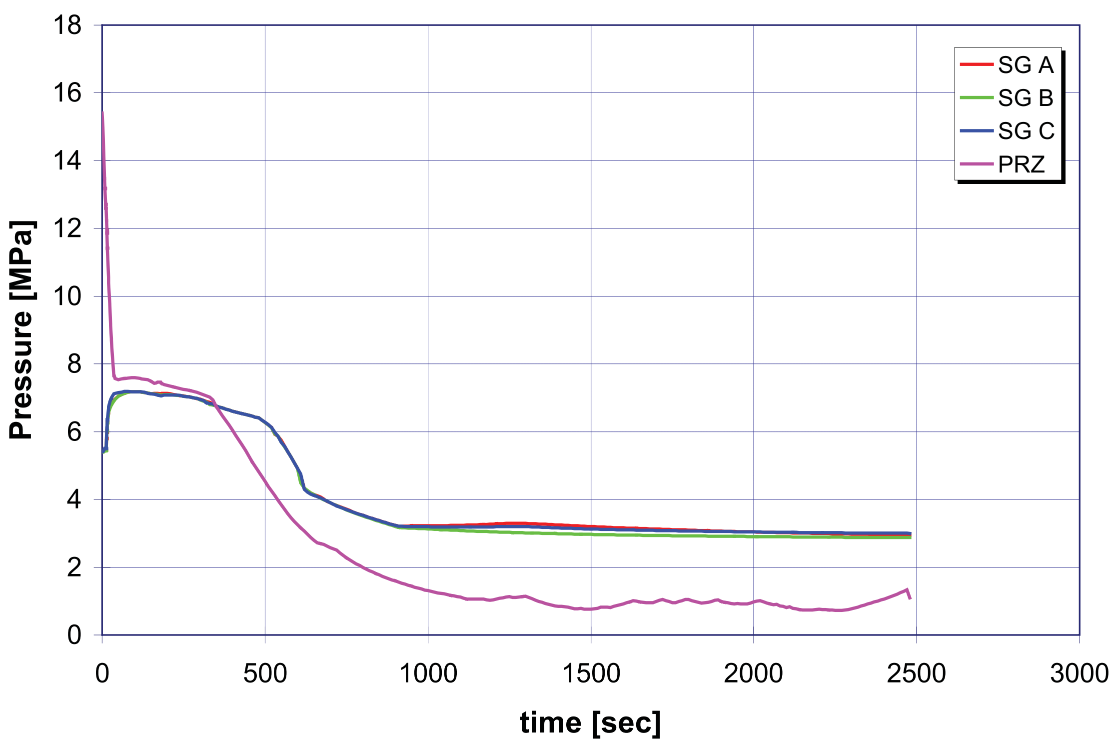
Event	Case 3 (hr)	Case 18 (hr)	Case 23 (hr)
Reactor trip	0.002	0.002	0.002
HHSI injection	0.002	0.002	-
RCP trip (10% void)	0.009	0.009	0.01
First actuation of containment sprays	0.01	0.01	0.01
Core uncovery (water <TAF)	0.02	0.01	0.03
LHSI injection	0.07	-	0.08
Maximum cladding temperature timing (maximum temperature)	0.10 (851 K)	0.40 (1,085 K)	0.07 (792 K)
Core covered	0.14	0.91	0.11

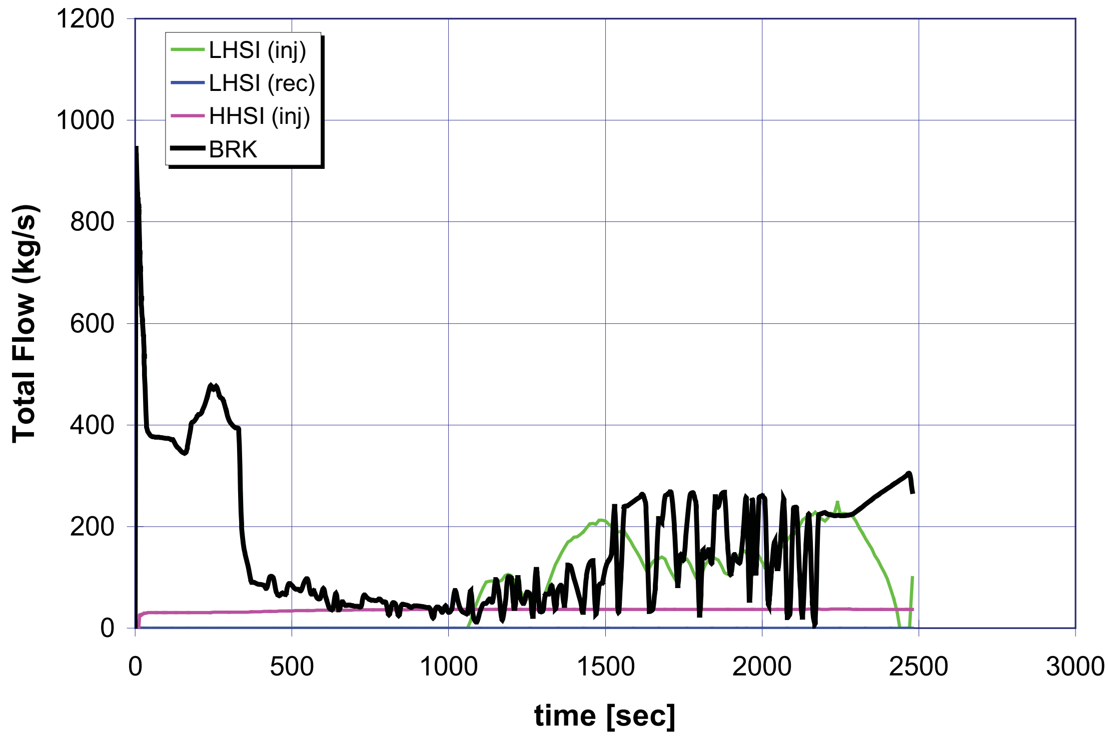
Table 23 Surry MBLOCA and LBLOCA Key Timings (≥10-Inch Breaks)

Event	Case 4 (hr)	Case 19 (hr)	Case 24 (hr)	Case 10 (hr)
Reactor trip	0.001	0.001	0.001	0.001
HHSI injection	0.001	0.001	-	-
RCP trip (10% void)	0.008	0.008	0.006	0.001
First actuation of containment sprays	0.008	0.008	0.008	0.005
Core uncovery (water <TAF)	0.01	0.008	0.02	0.022
LHSI injection	0.04	-	0.05	0.005
Maximum cladding temperature timing (maximum temperature)	0.08 (850 K)	0.30 (835 K)	0.04 (640 K)	0.036 (1,043 K)
Core covered	0.12	0.87	0.06	0.053

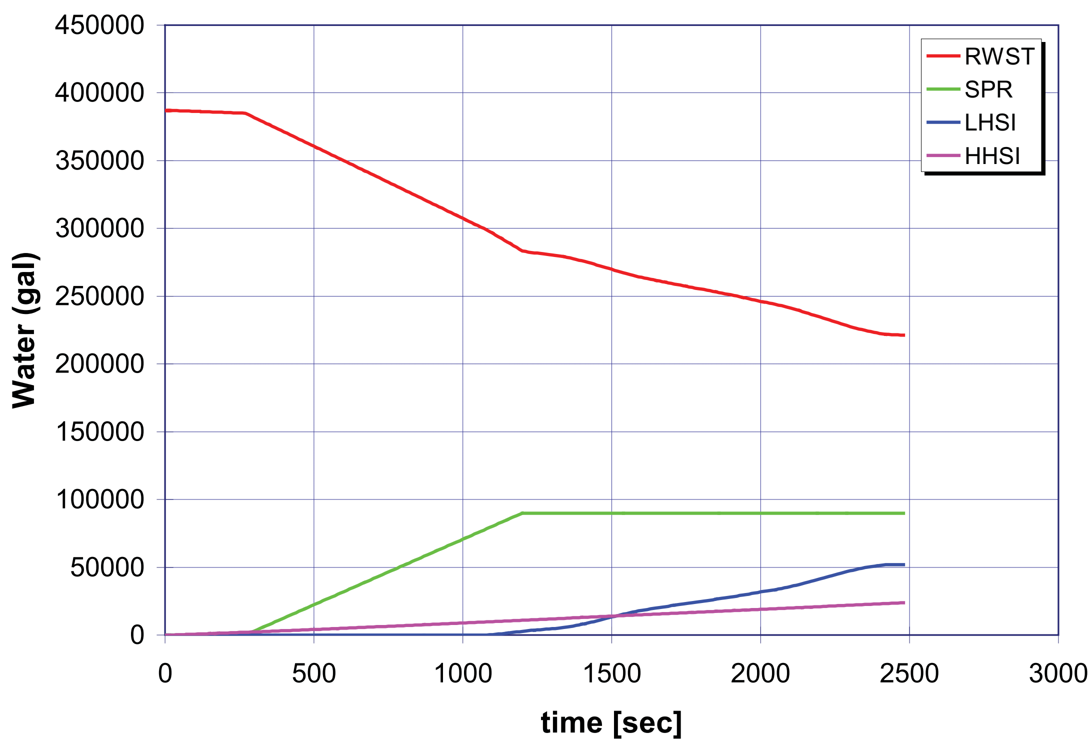
A.6.1 Case 1: 4-Inch Break LOCA, One HHSI, One LHSI, and No ACC



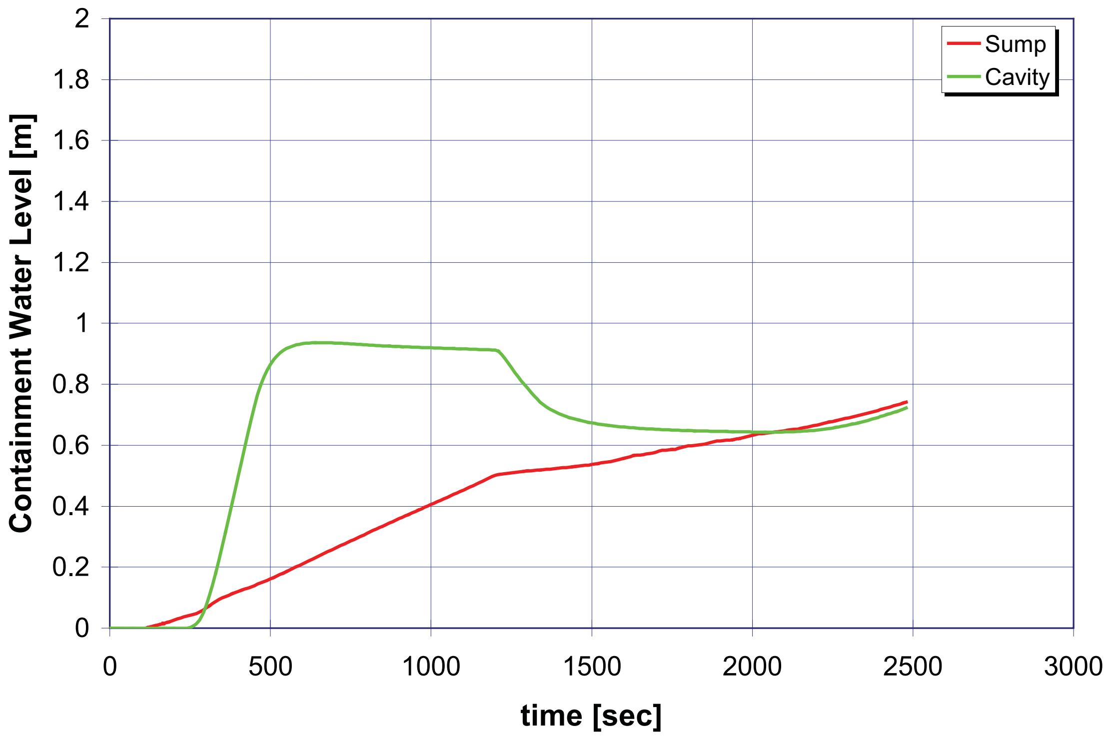


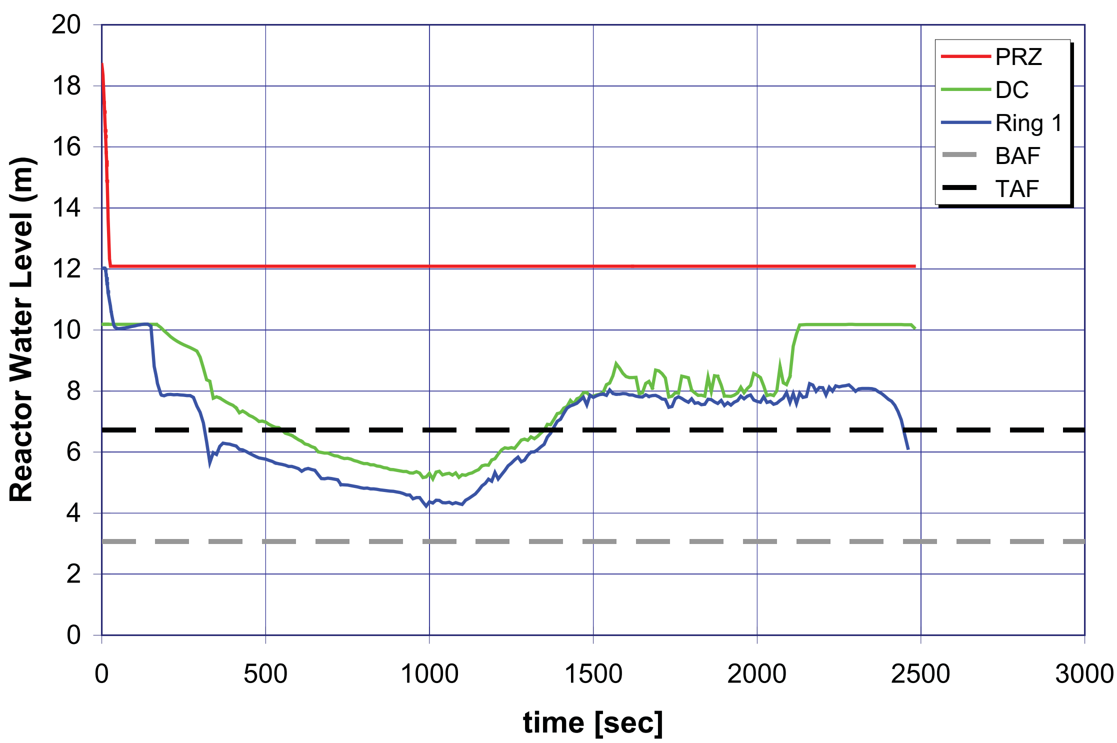


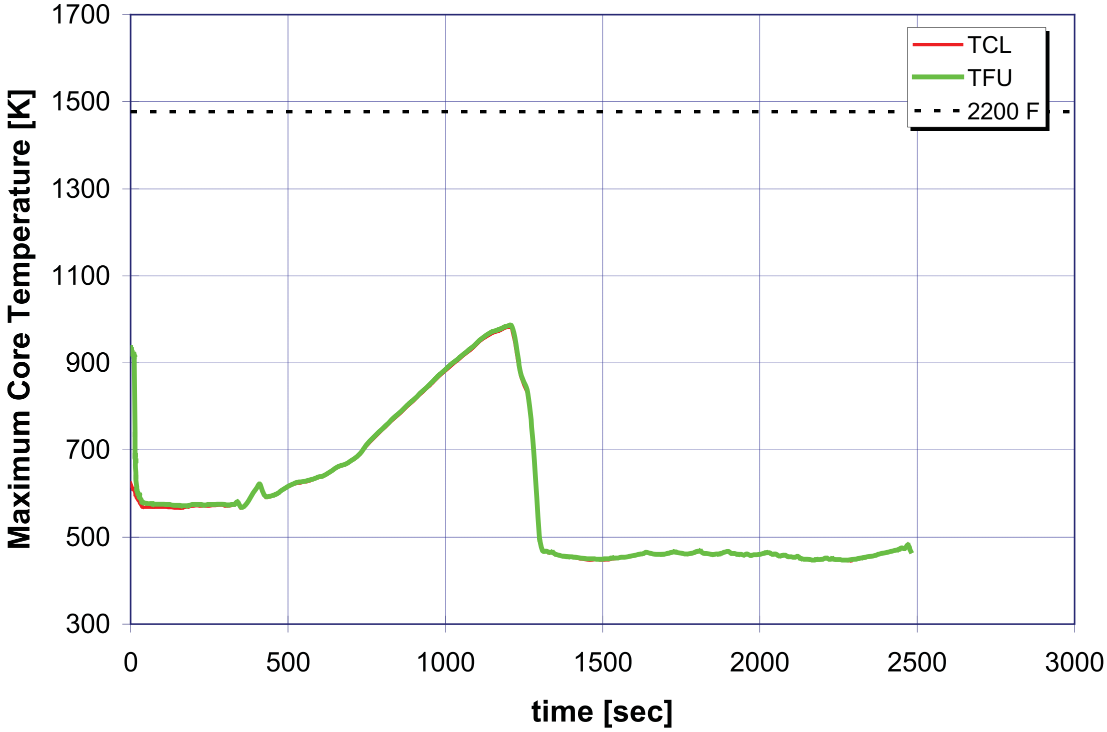
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]



RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]

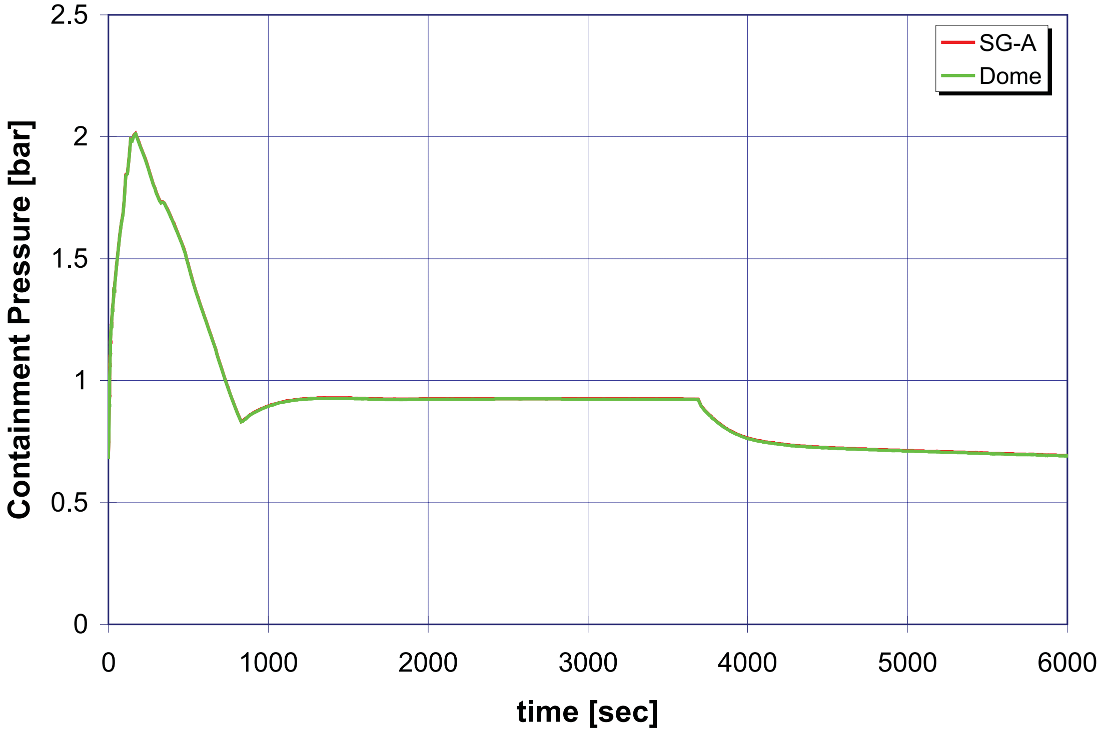

Containment Water Level [m]
time [sec]
Sump
Cavity

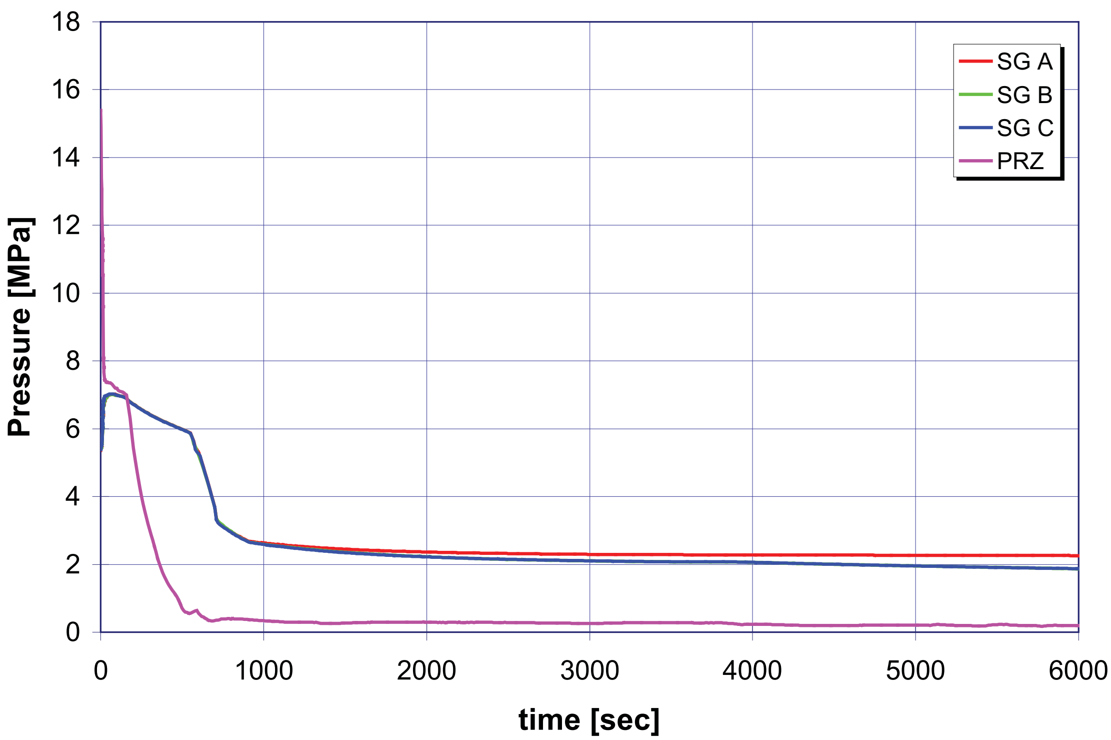

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF

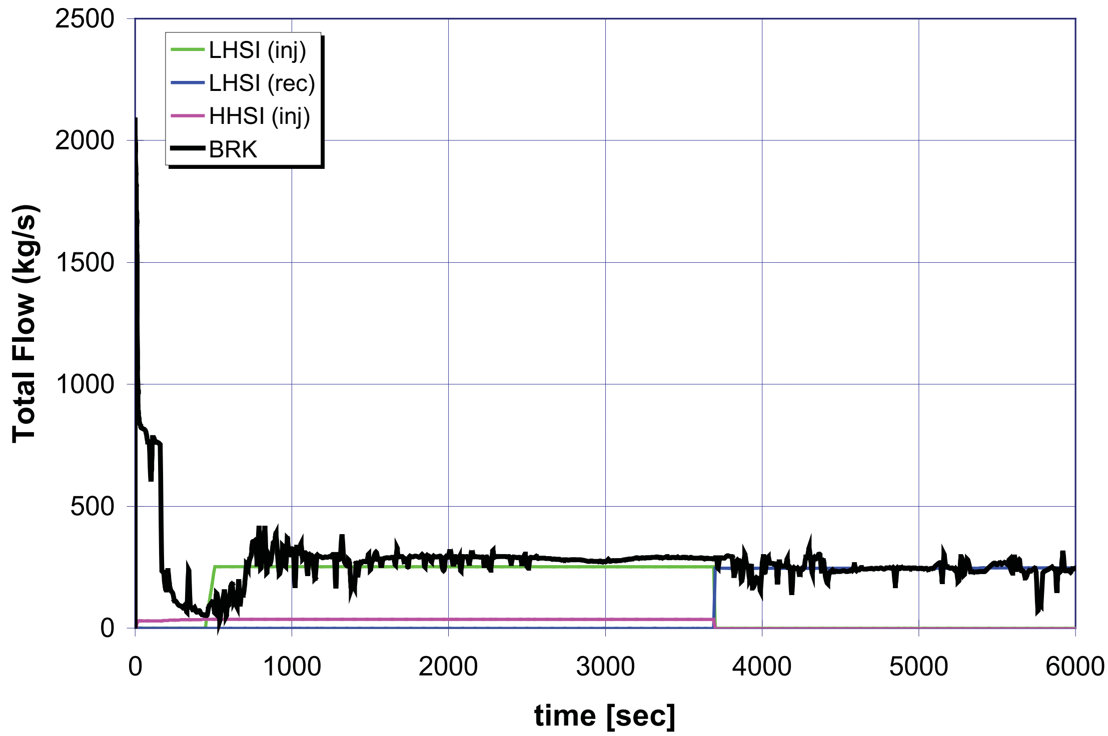


1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F

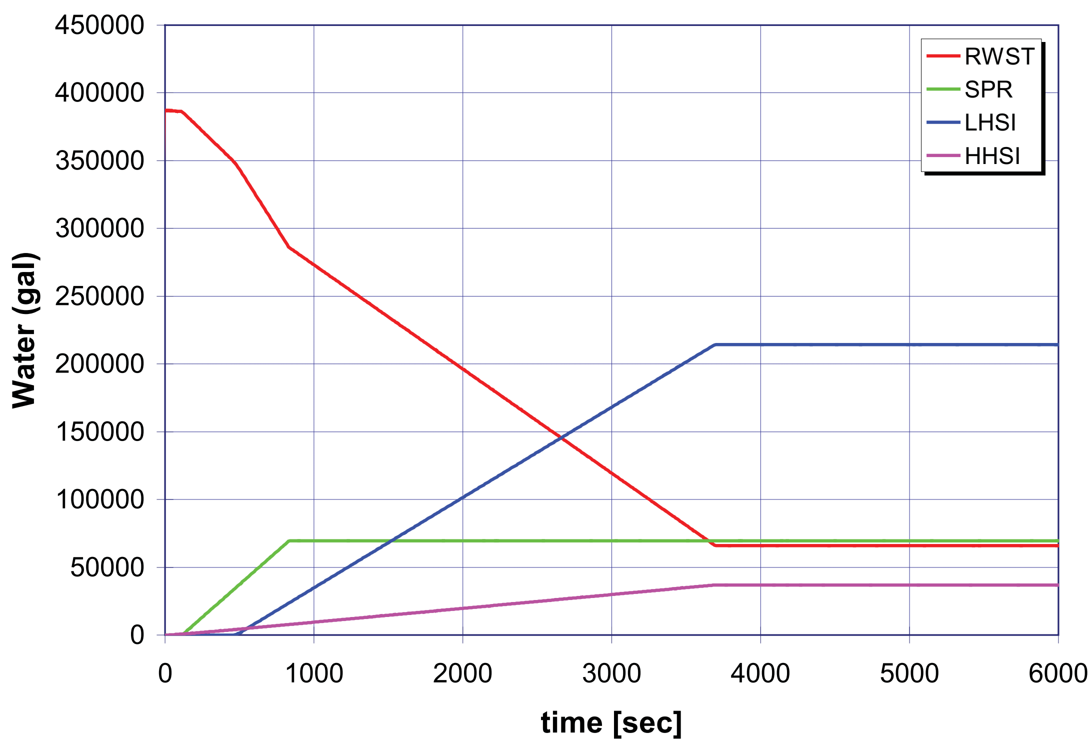
A.6.2 Case 2: 6-Inch Break LOCA, One HHSI, One LHSI, and No ACC



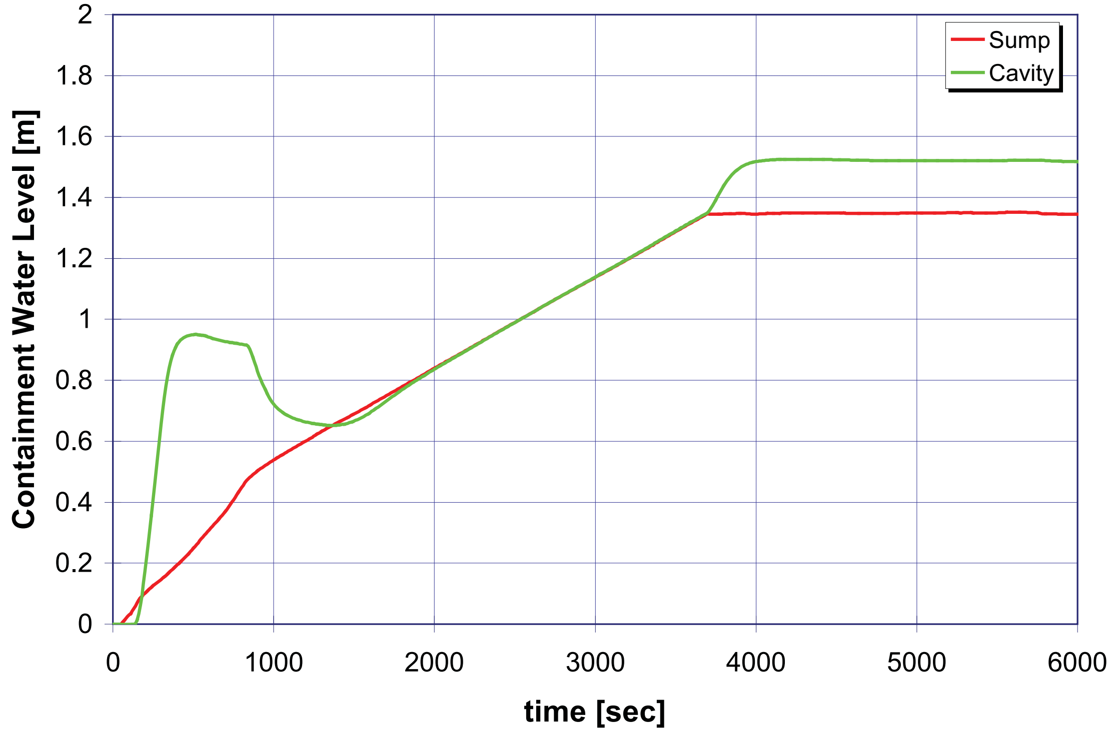




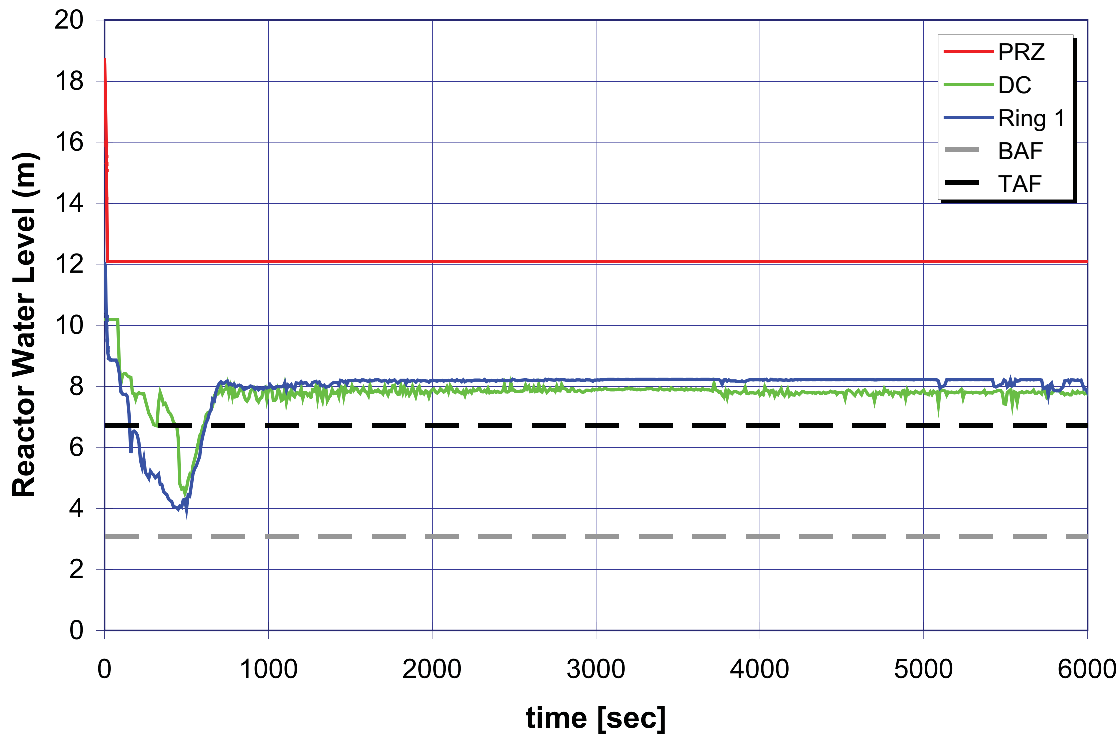
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
2500
2000
1500
1000
500
0
0
1000
2000
3000
4000
5000
6000
time [sec]



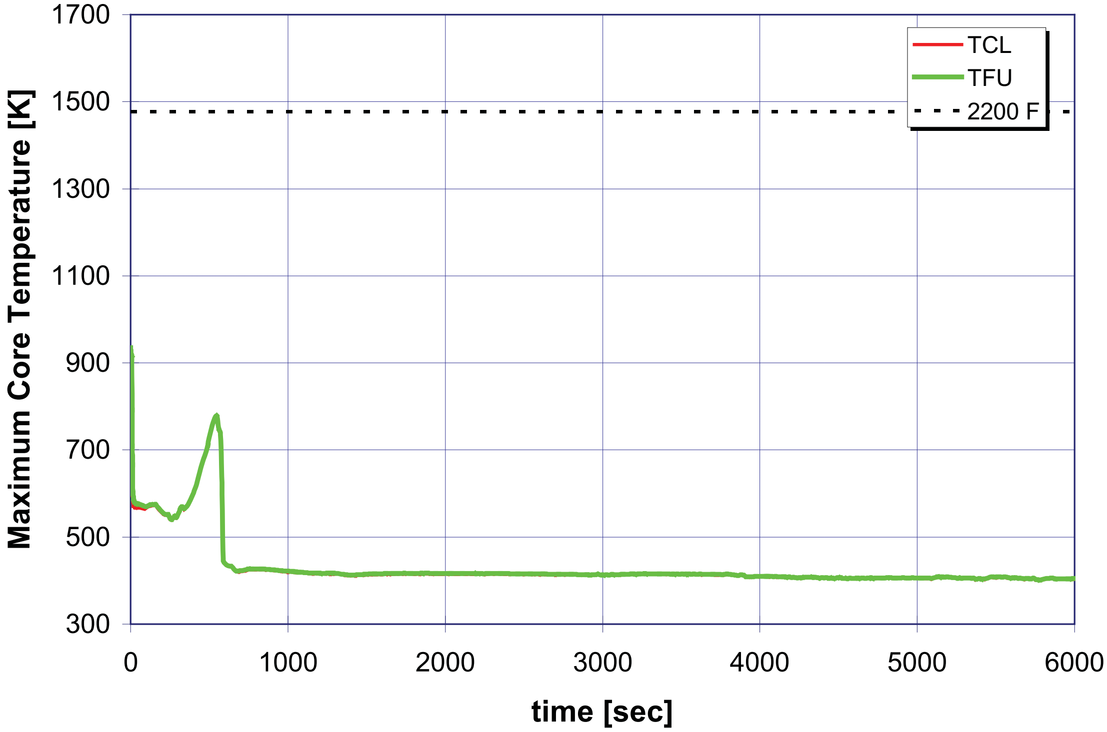
RWST
SPR
LHSI
HHSI
Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
6000
time [sec]



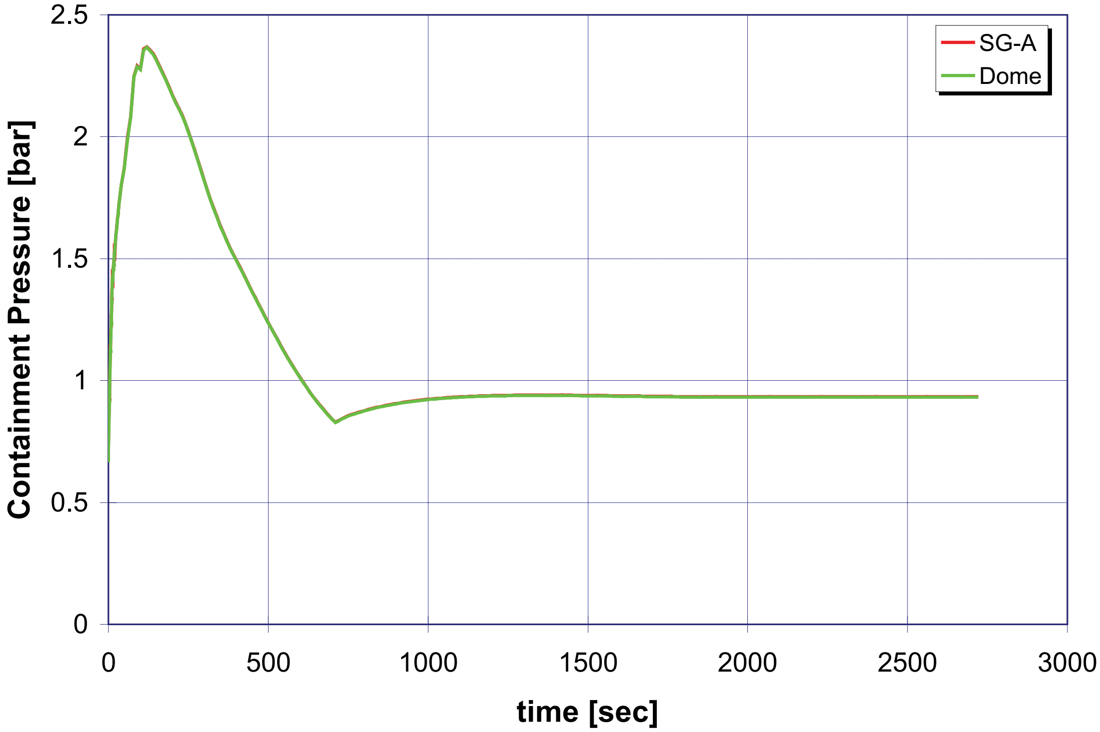
Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
1000
2000
3000
4000
5000
6000

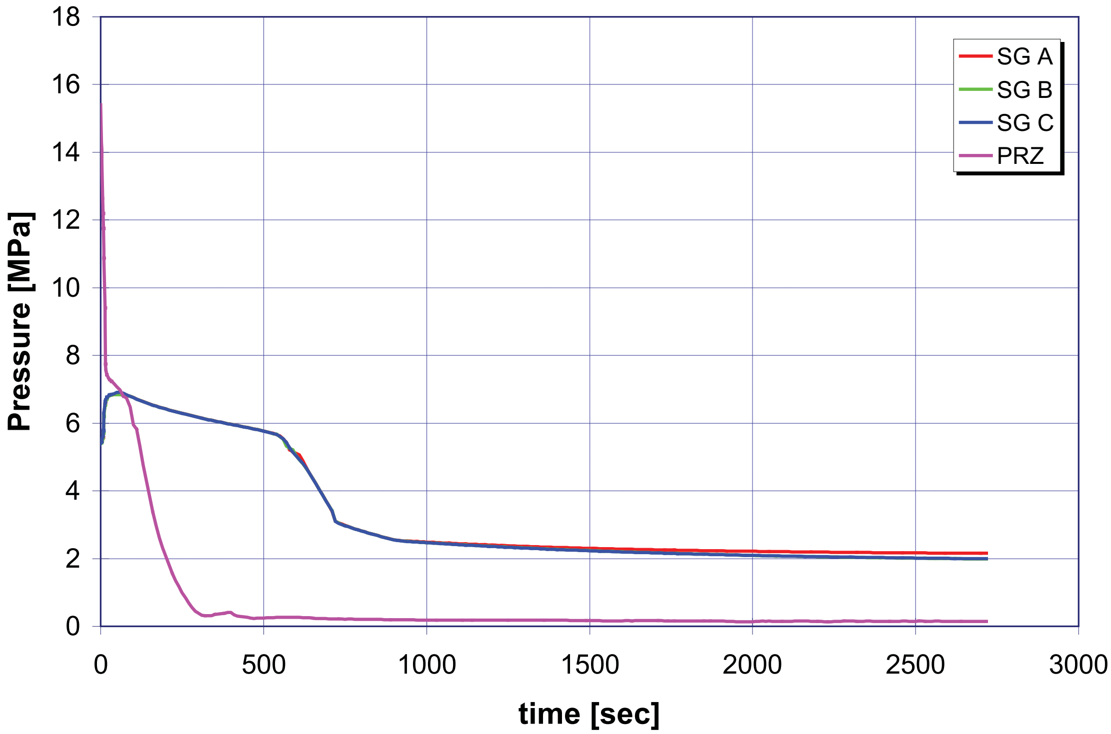


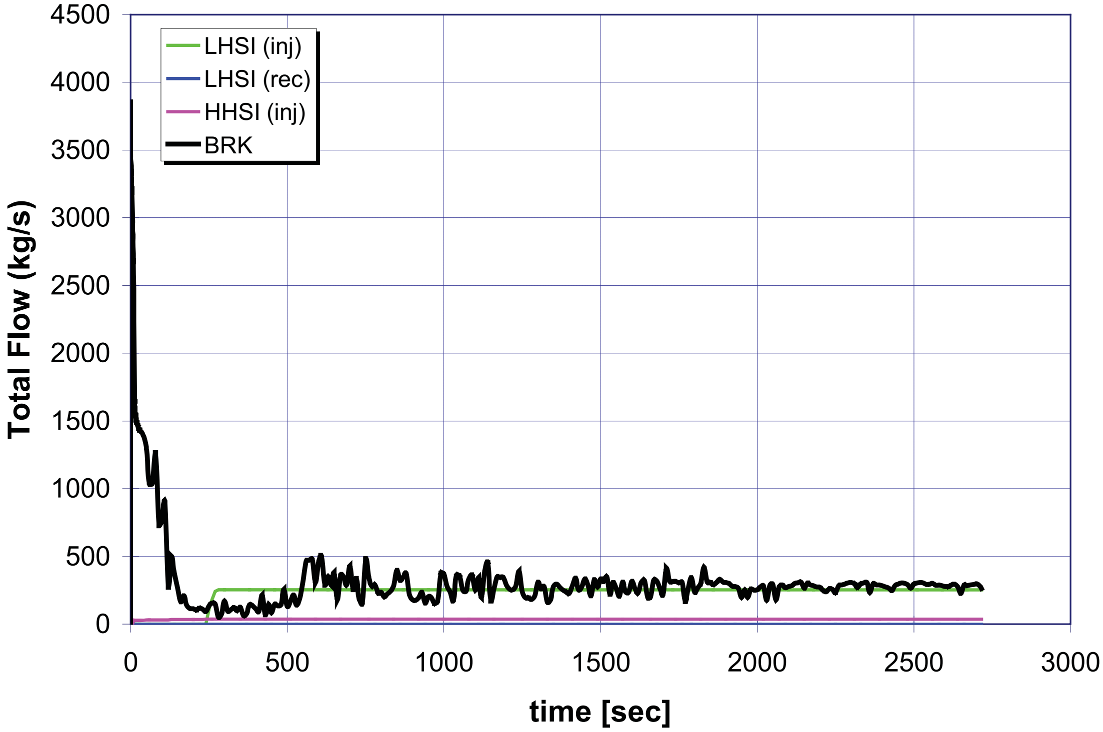
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
1000
2000
3000
4000
5000
6000

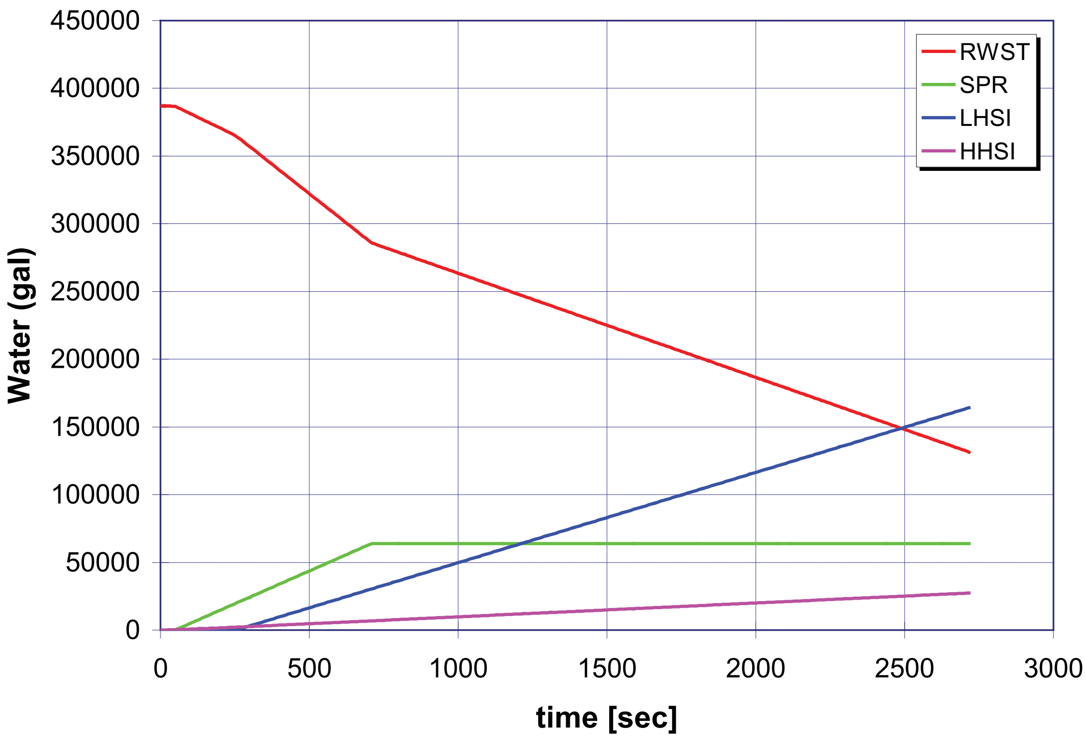

1700
1500
1300
1100
900
700
500
300
Maximum Core Temperature [K]
0
1000
2000
3000
4000
5000
6000
time [sec]
TCL
TFU
2200 F

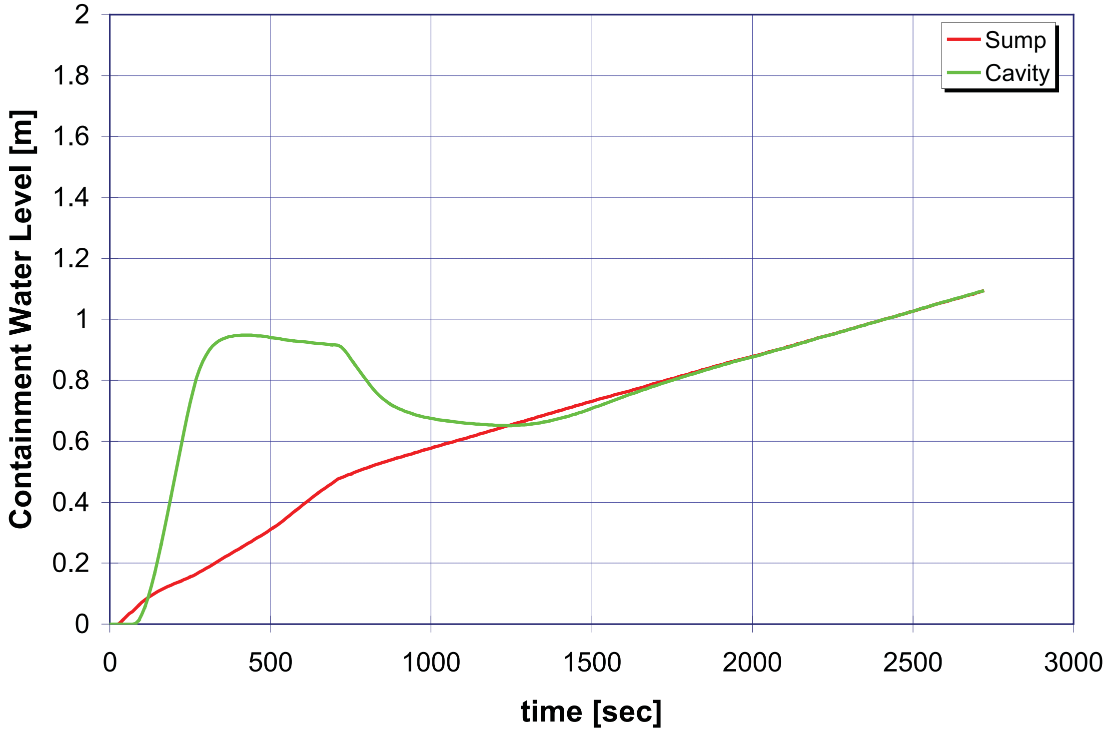
A.6.3 Case 3: 8-Inch Break LOCA, One HHSI, One LHSI, and No ACC

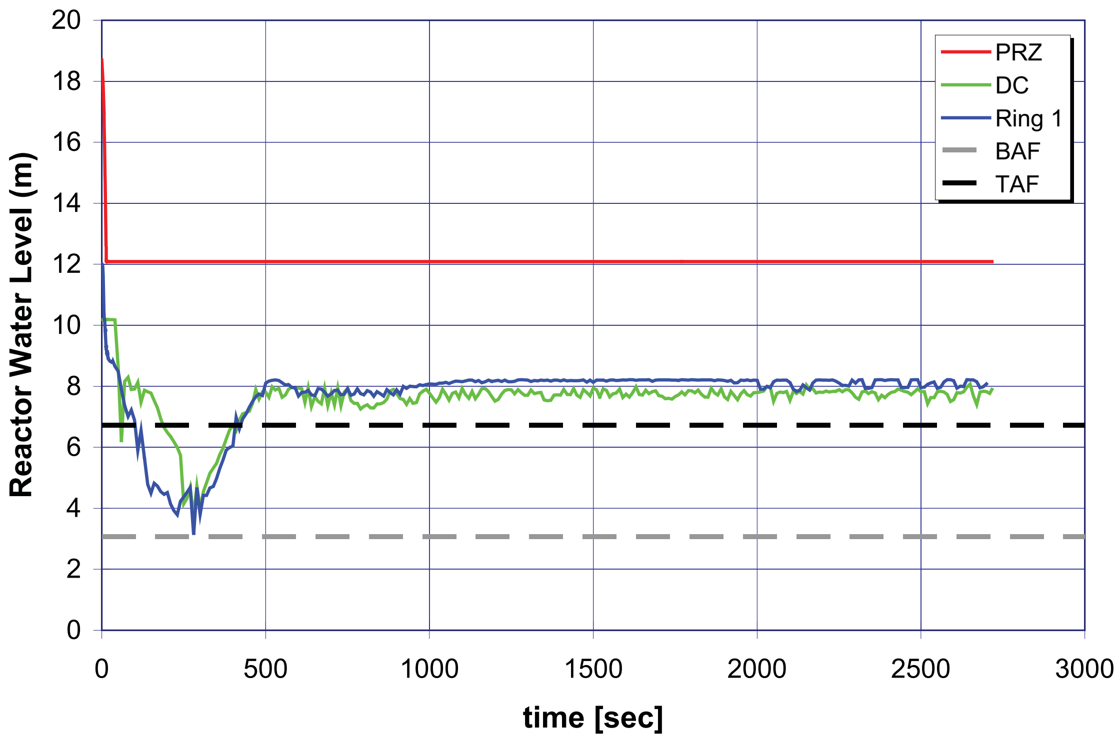


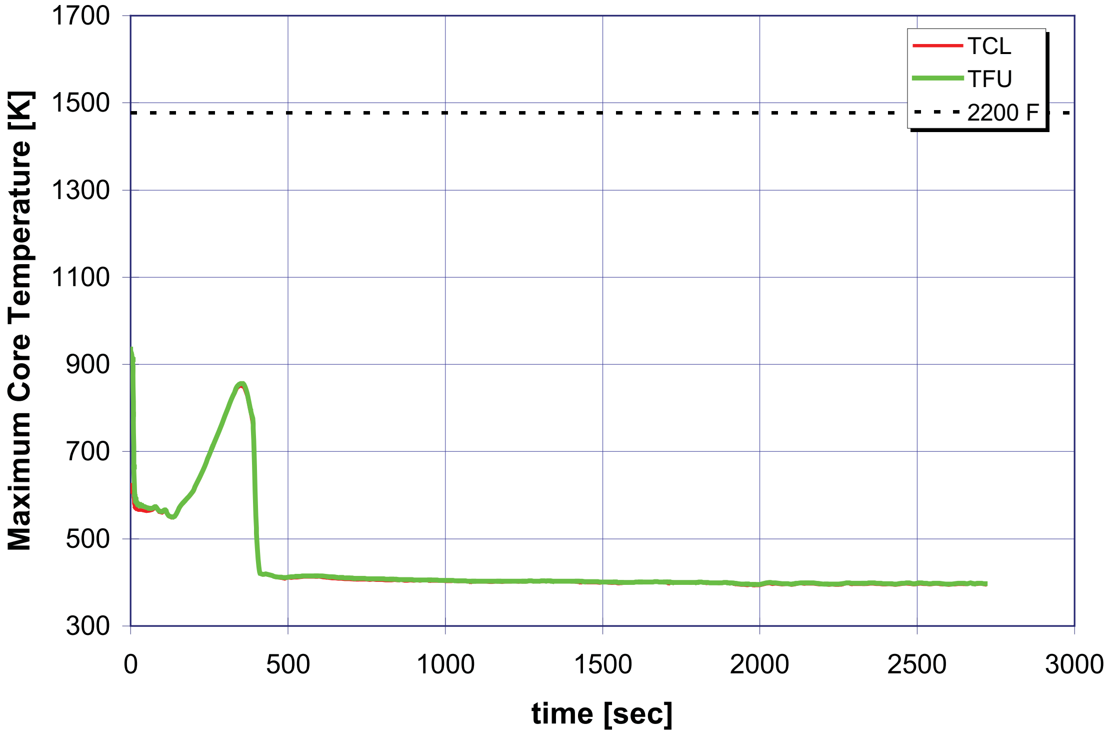



LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
0
500
1000
1500
2000
2500
3000
3500
4000
4500
0
500
1000
1500
2000
2500
3000

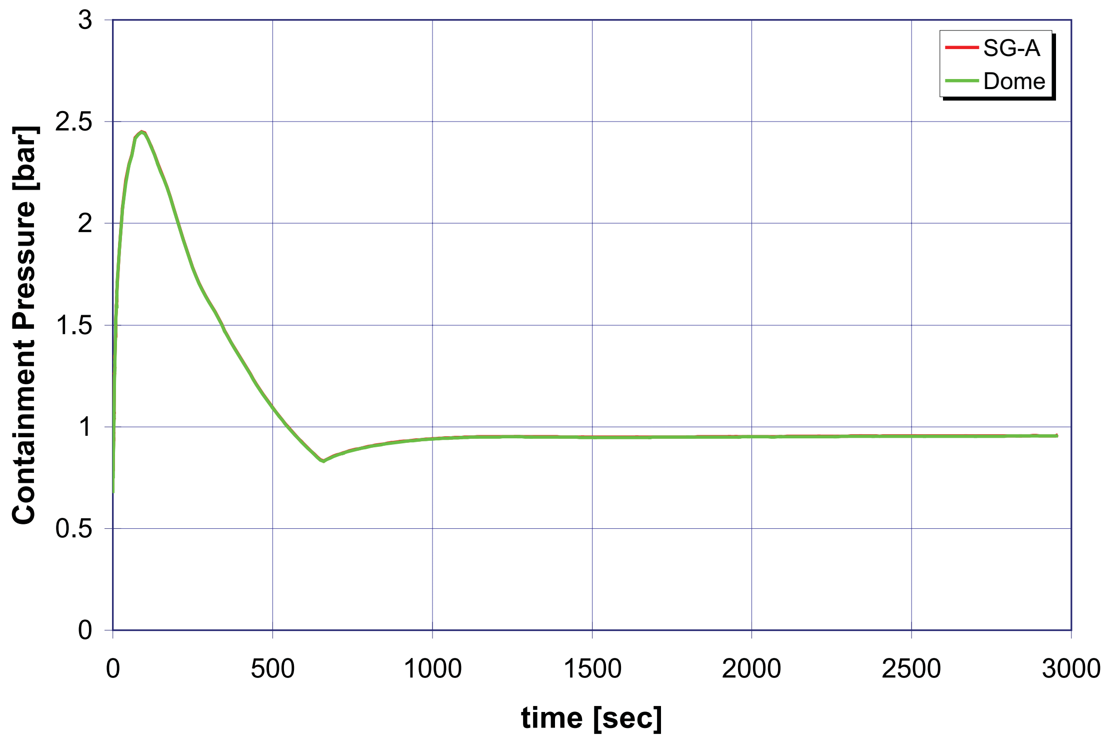

RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
500
1000
1500
2000
2500
3000

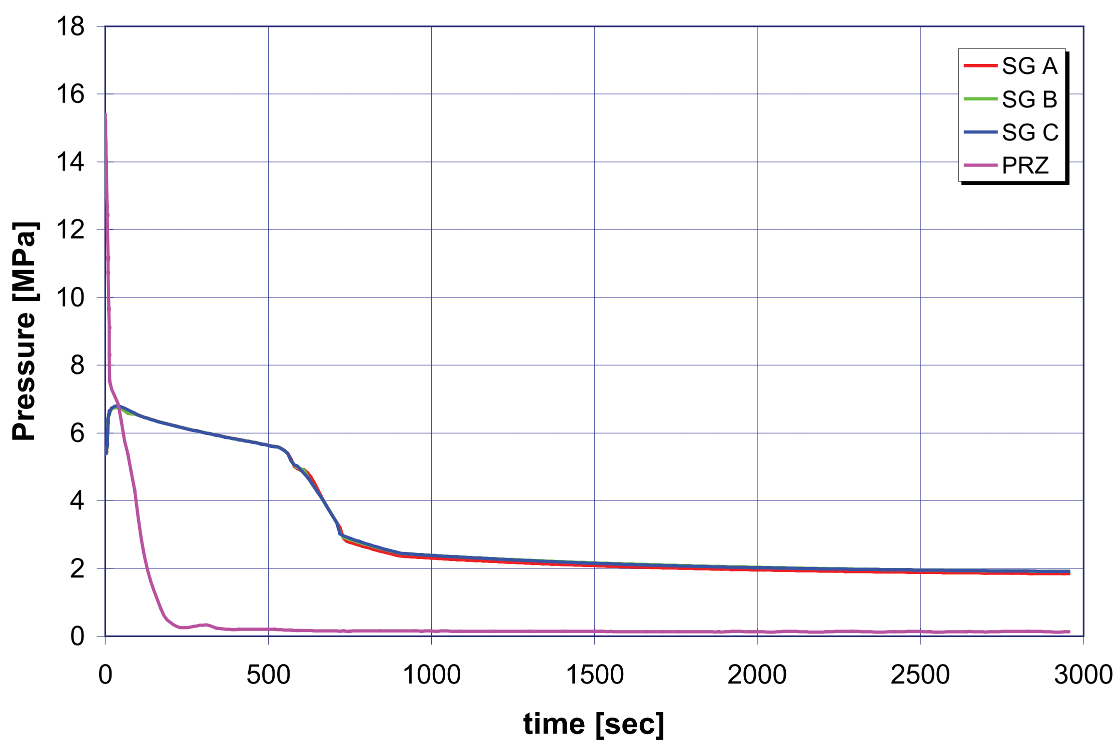

Containment Water Level [m]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
500
1000
1500
2000
2500
3000
time [sec]
Sump
Cavity

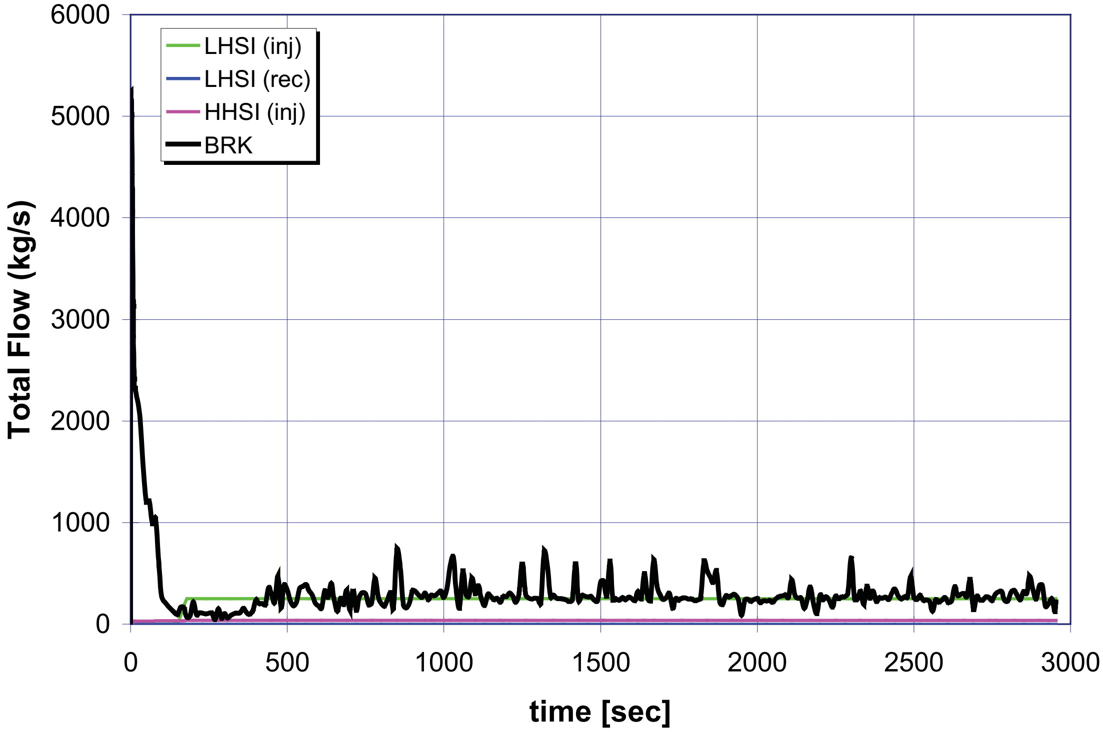

Reactor Water Level (m)
20
18
16
14
12
10
8
6
4
2
0
0
500
1000
1500
2000
2500
3000
time [sec]
PRZ
DC
Ring 1
BAF
TAF


Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000
time [sec]
TCL
TFU
2200 F

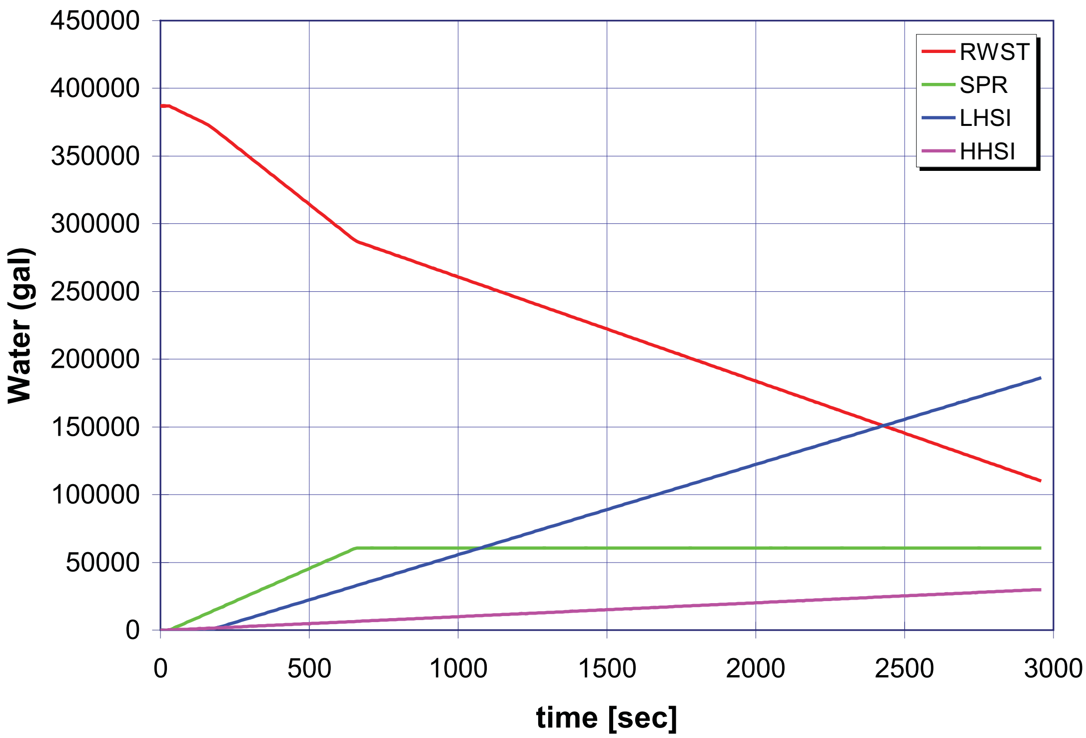
A.6.4 Case 4: 10-Inch Break LOCA, One HHSI, One LHSI, and No ACC



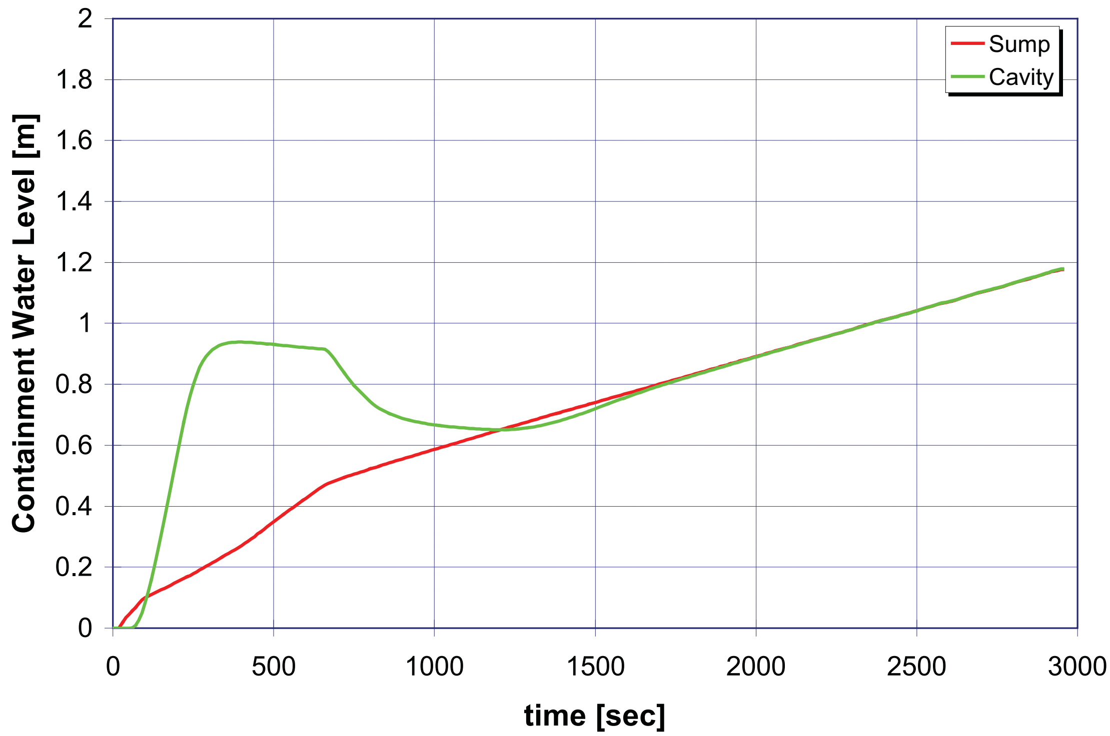


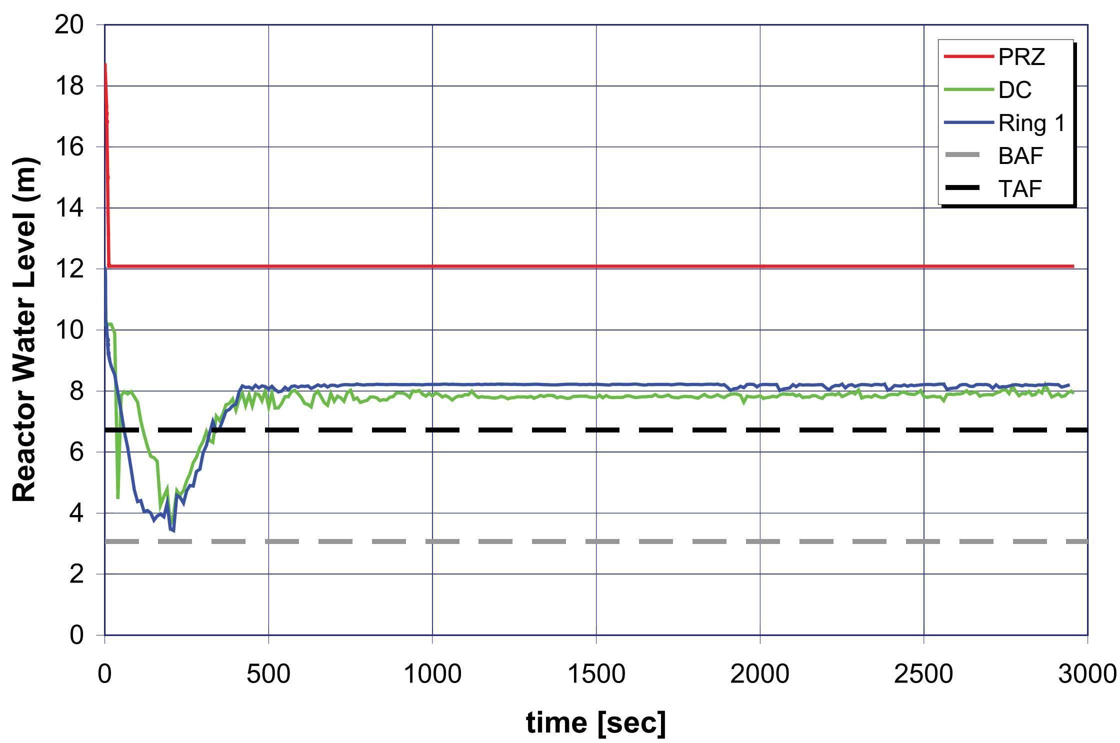


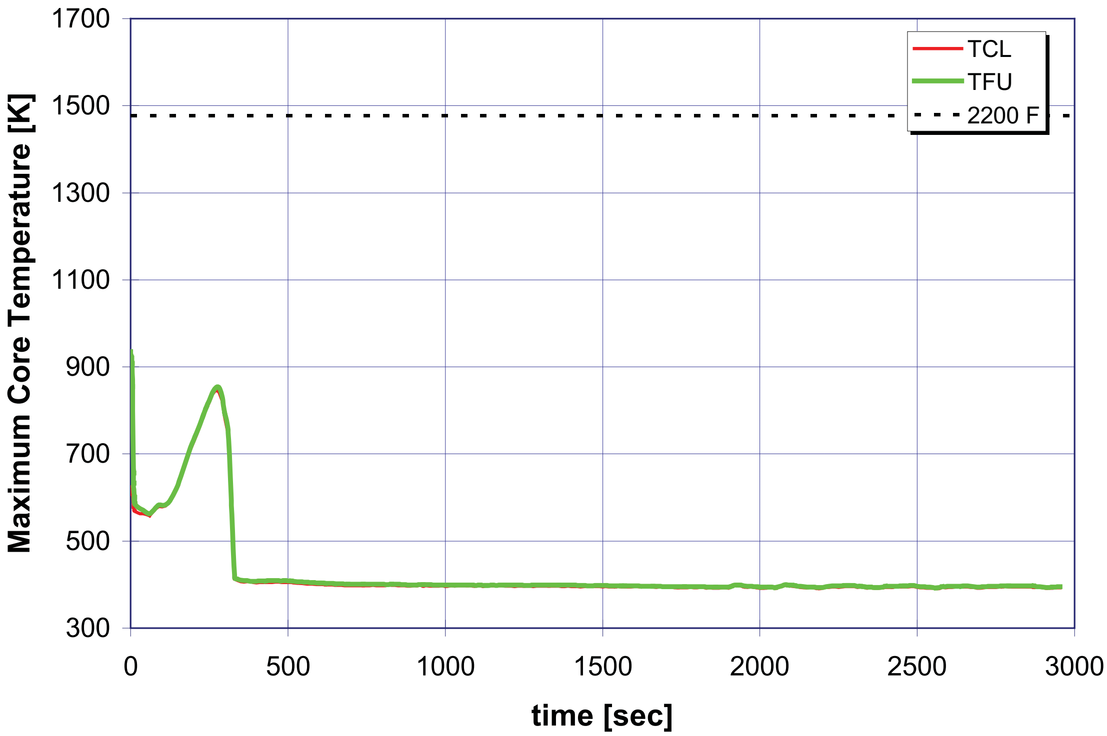
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
0
1000
2000
3000
4000
5000
6000
0
500
1000
1500
2000
2500
3000



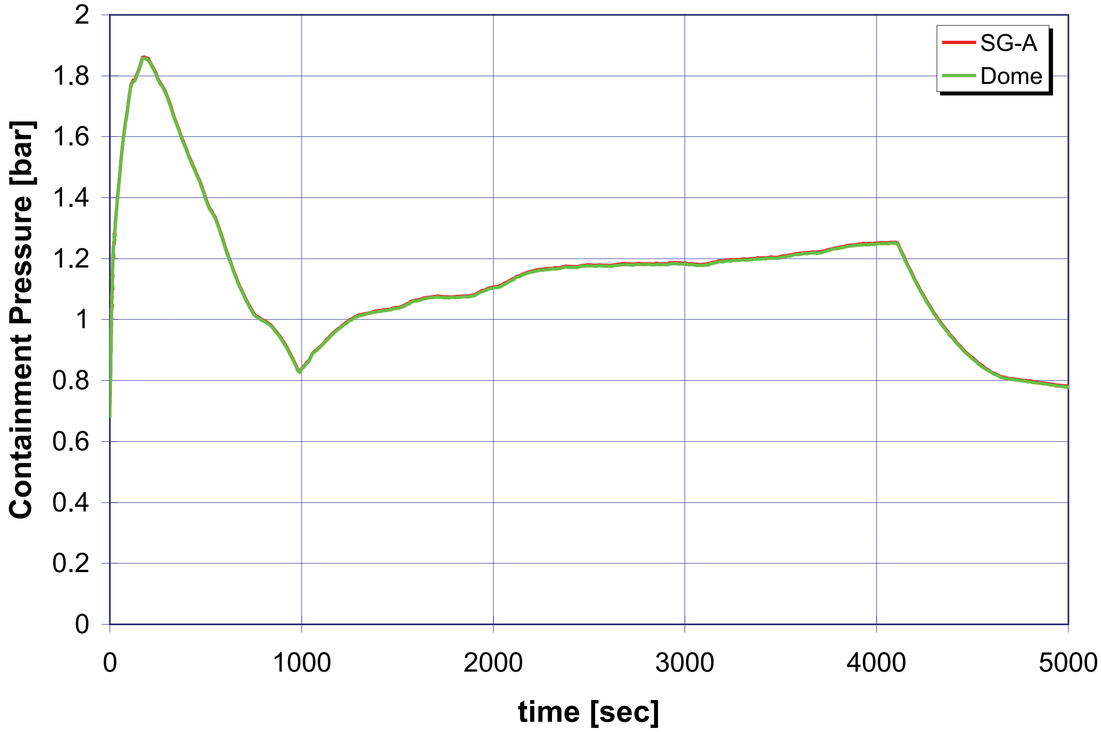
RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
500
1000
1500
2000
2500
3000

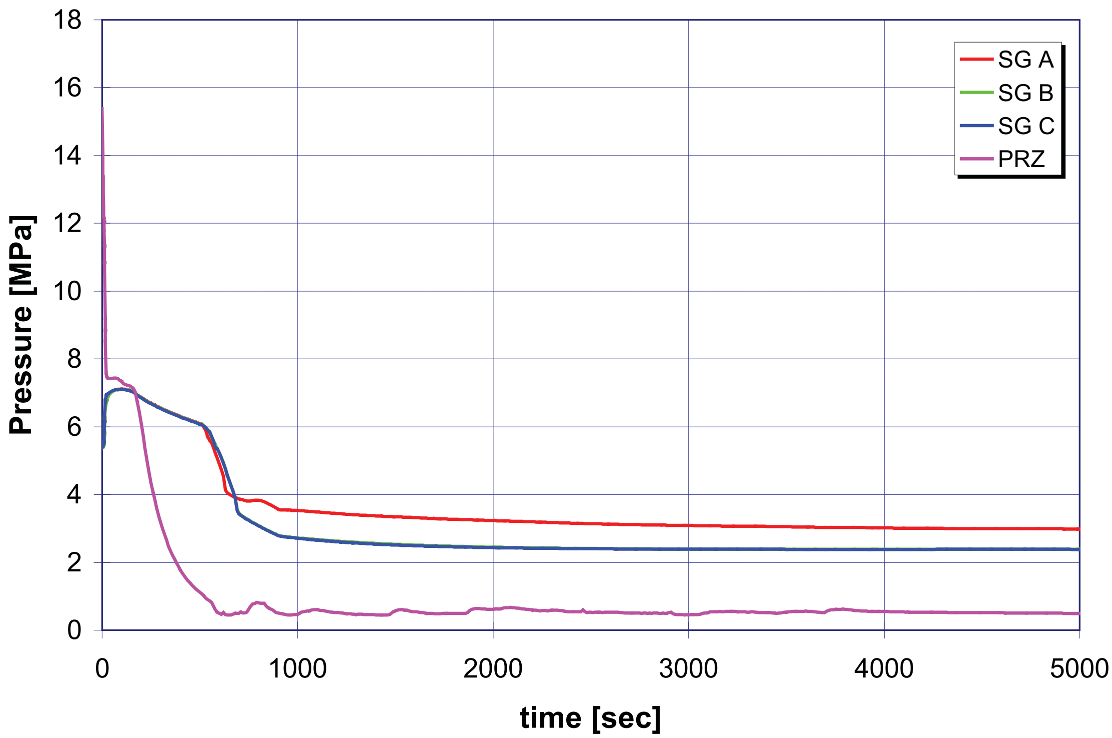

Containment Water Level [m]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
500
1000
1500
2000
2500
3000
time [sec]
Sump
Cavity

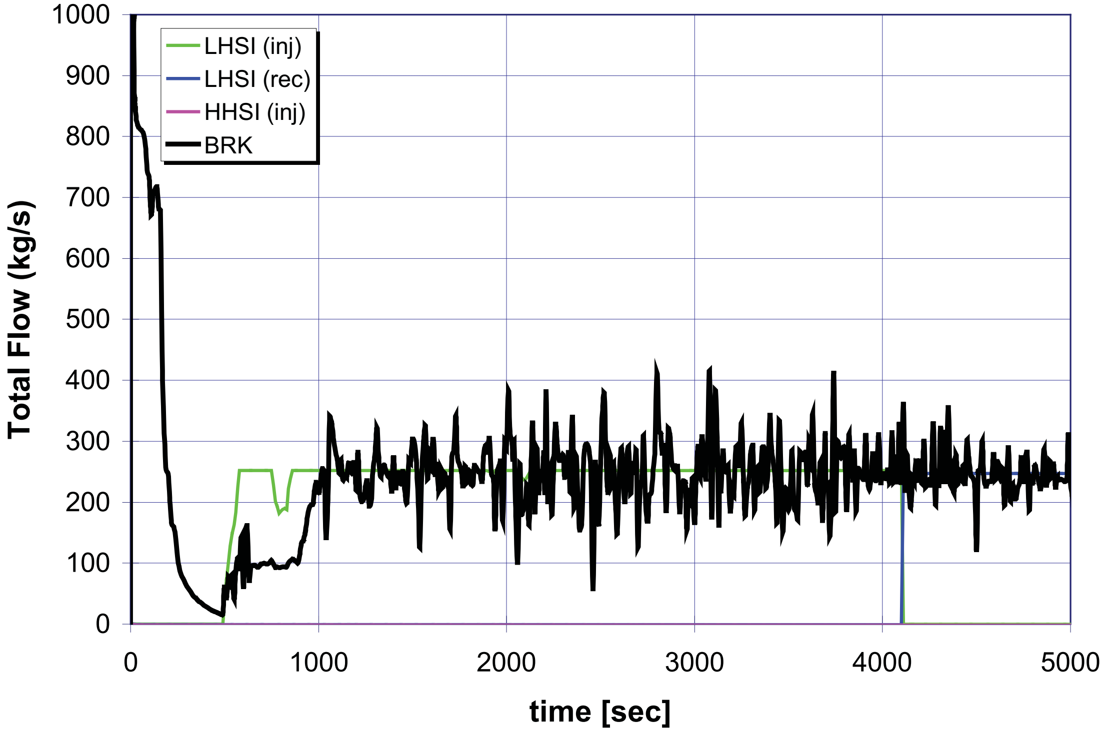

Reactor Water Level (m)
20
18
16
14
12
10
8
6
4
2
0
0
500
1000
1500
2000
2500
3000
time [sec]
PRZ
DC
Ring 1
BAF
TAF


TCL
TFU
2200 F
1700
1500
1300
1100
900
700
500
300
Maximum Core Temperature [K]
0
500
1000
1500
2000
2500
3000
time [sec]

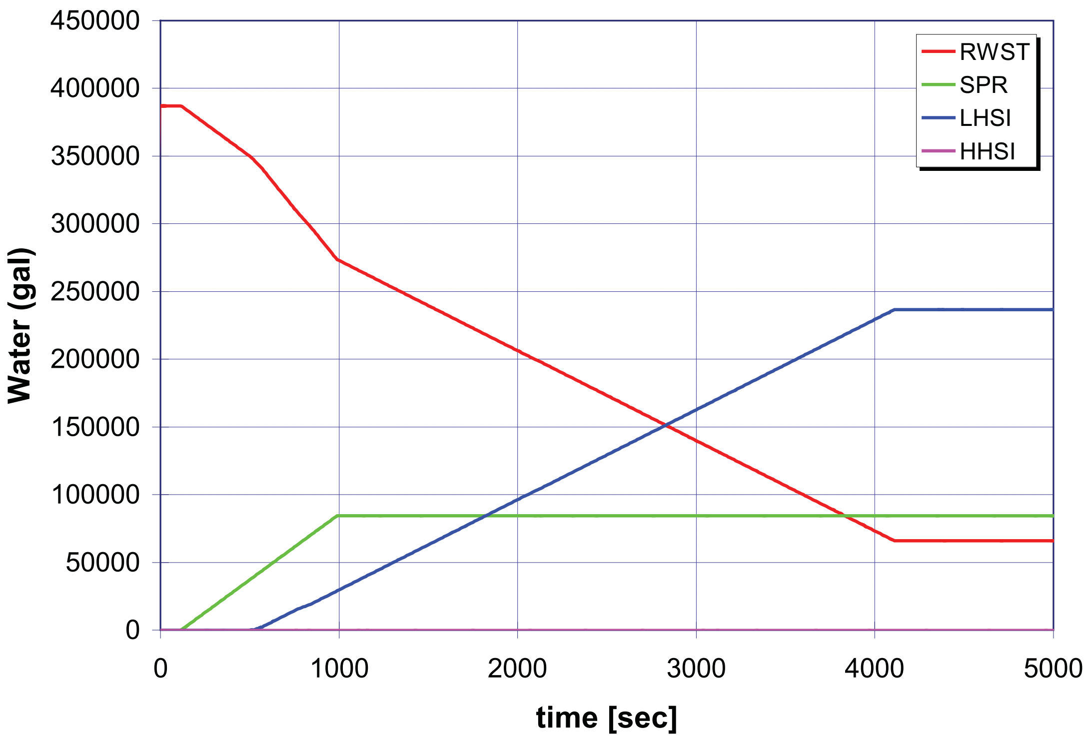
A.6.5 Case 5: 6-Inch Break LOCA, No HHSI, One LHSI, and No ACC



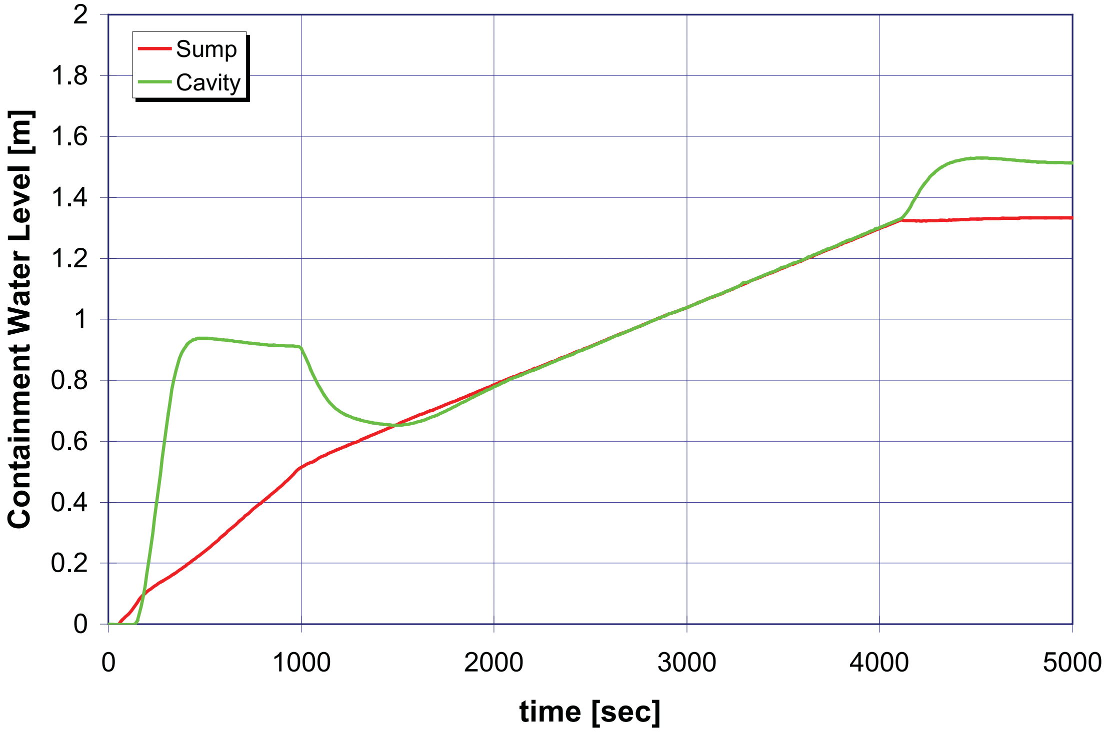




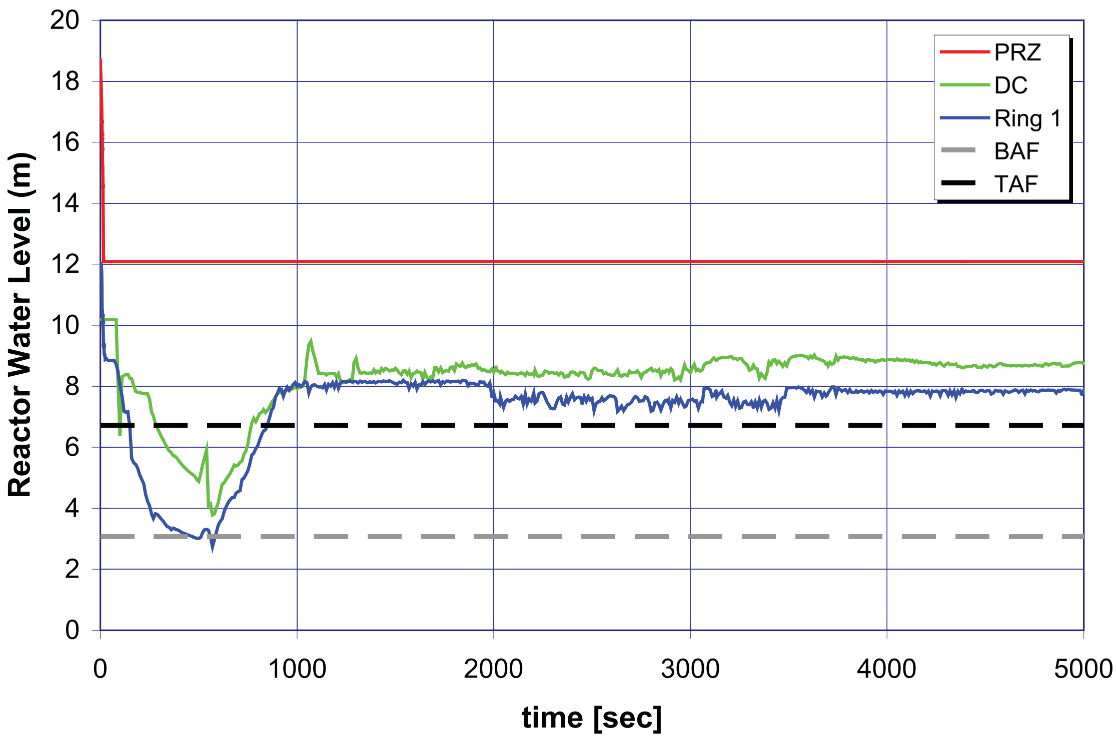
Total Flow (kg/s)
1000
900
800
700
600
500
400
300
200
100
0
0
1000
2000
3000
4000
5000
time [sec]
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



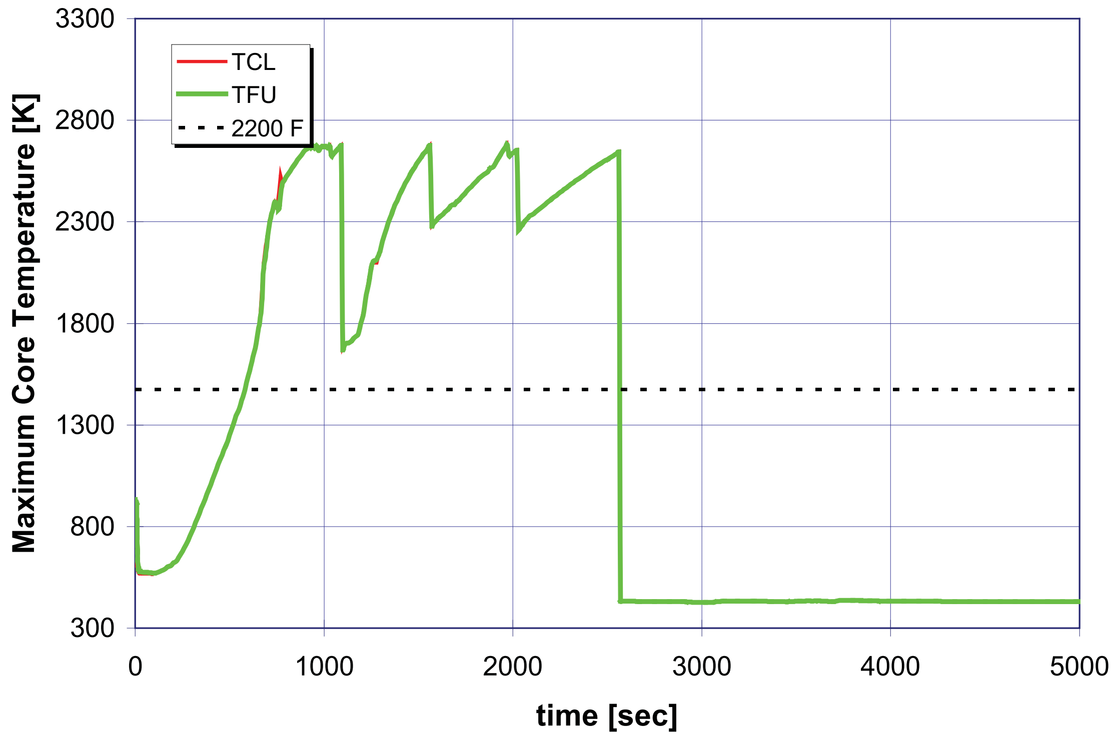
Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
time [sec]
RWST
SPR
LHSI
HHSI



Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
1000
2000
3000
4000
5000

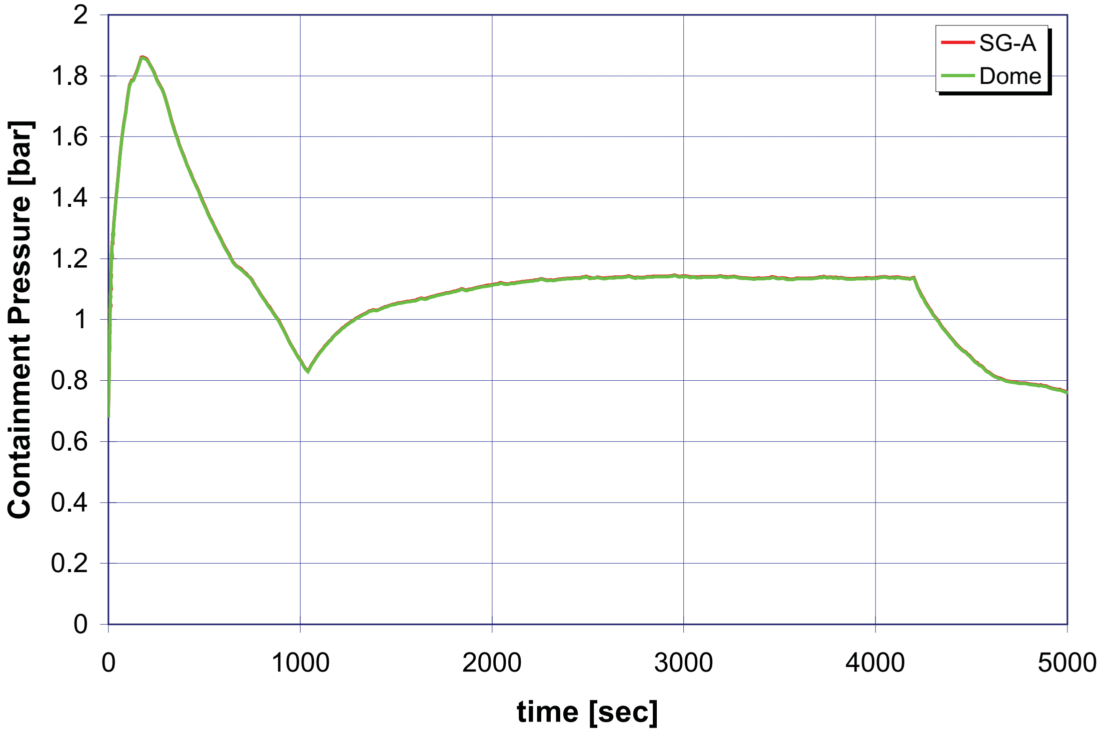


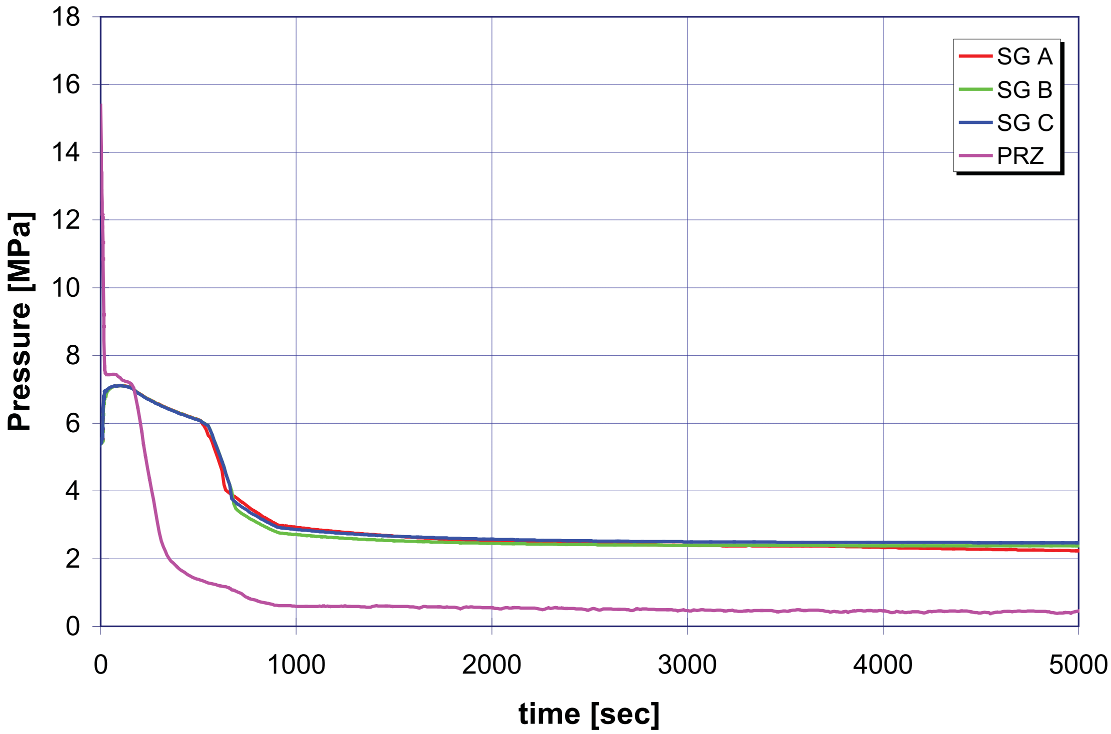
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
1000
2000
3000
4000
5000

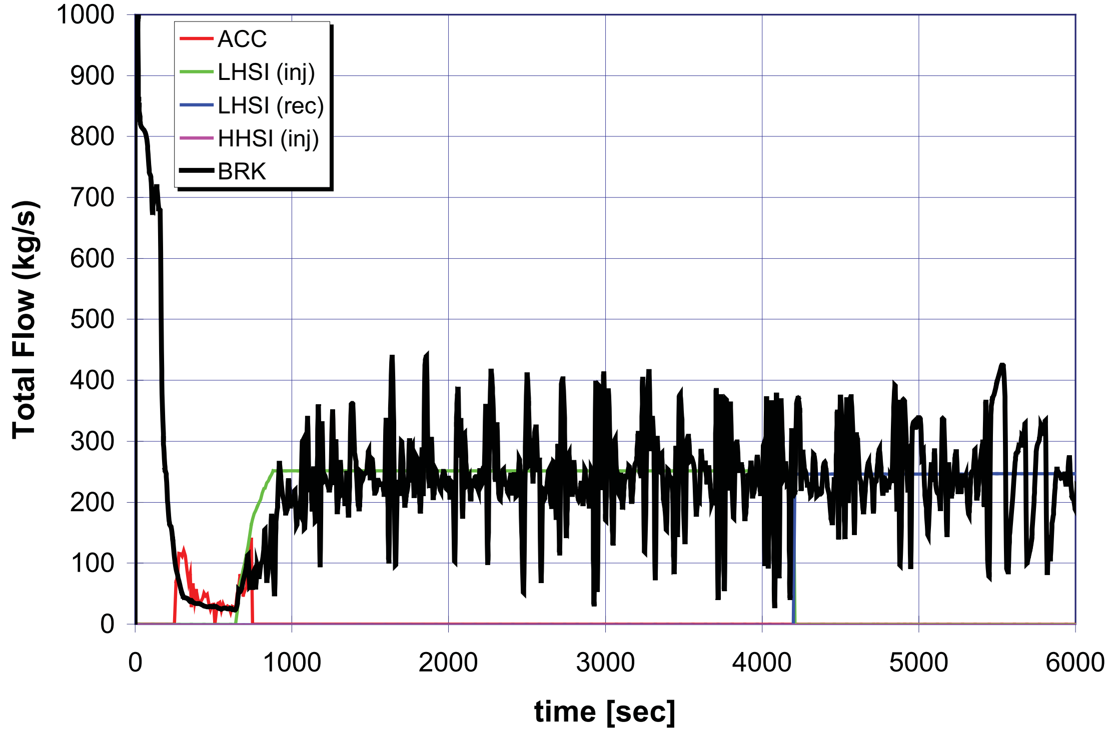


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
3300
2800
2300
1800
1300
800
300
0
1000
2000
3000
4000
5000

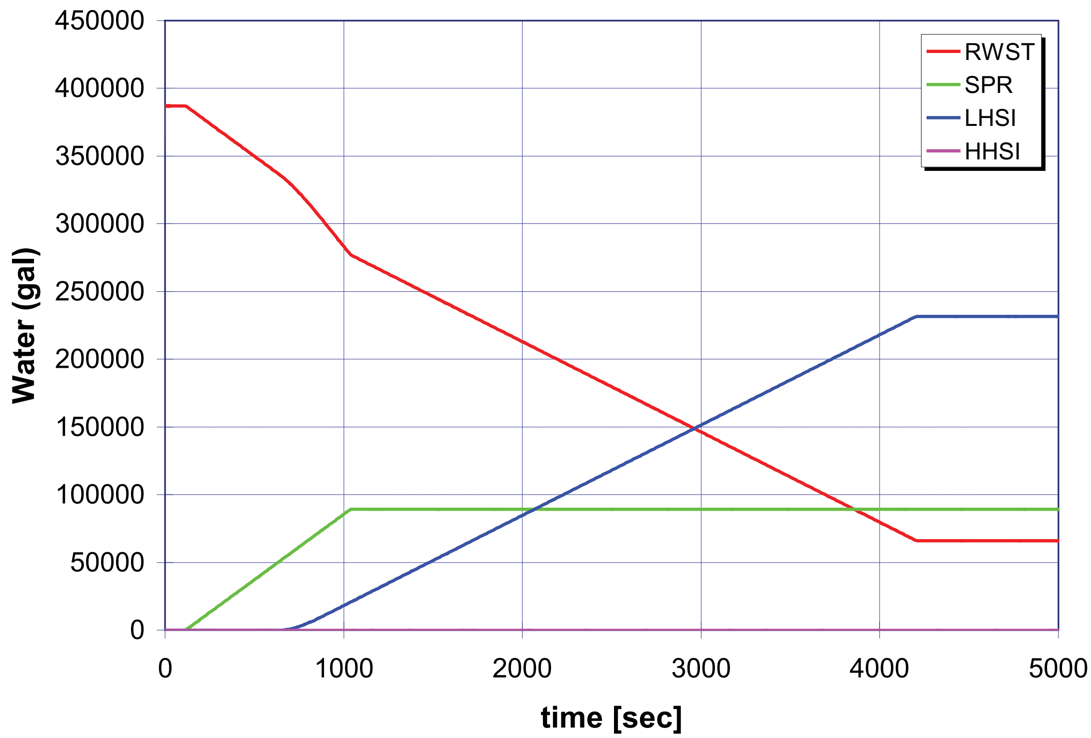
A.6.6 Case 6: 6-Inch Break LOCA, No HHSI, One LHSI, and One ACC



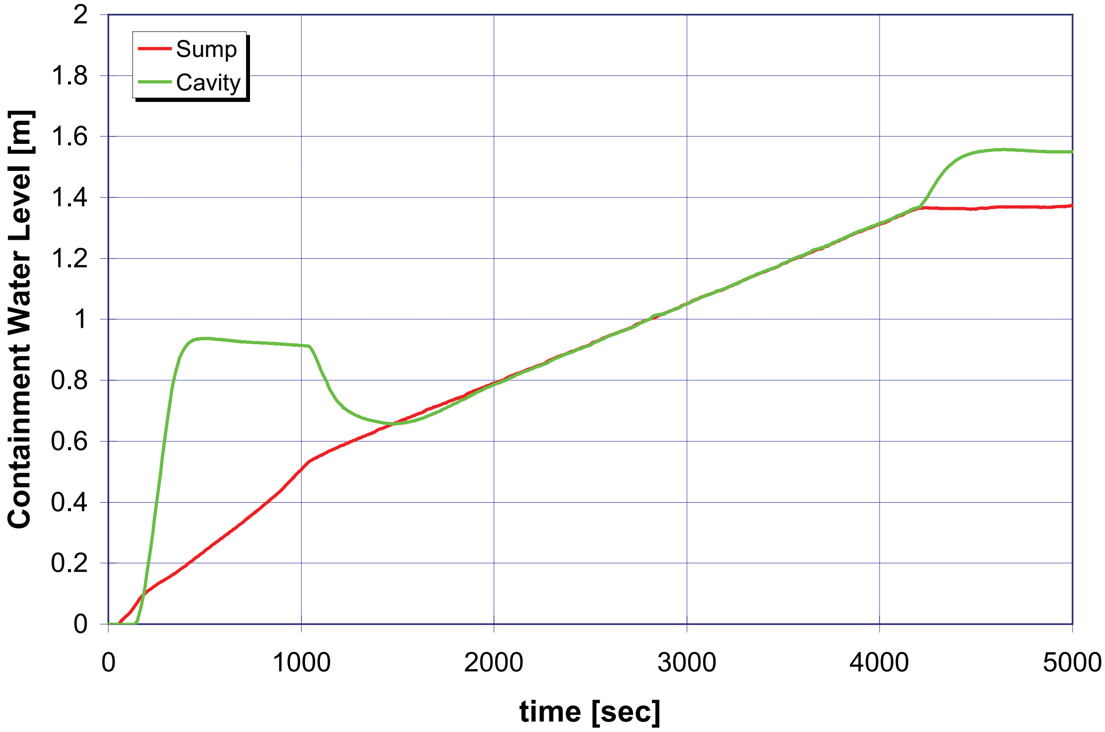




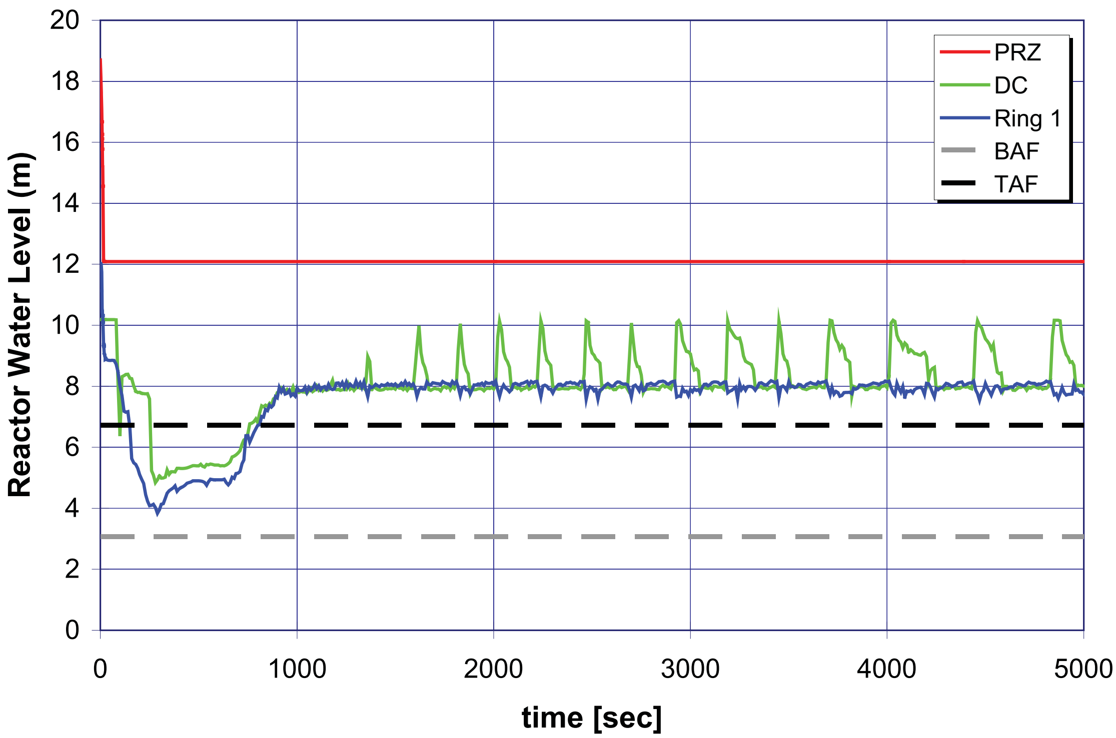
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



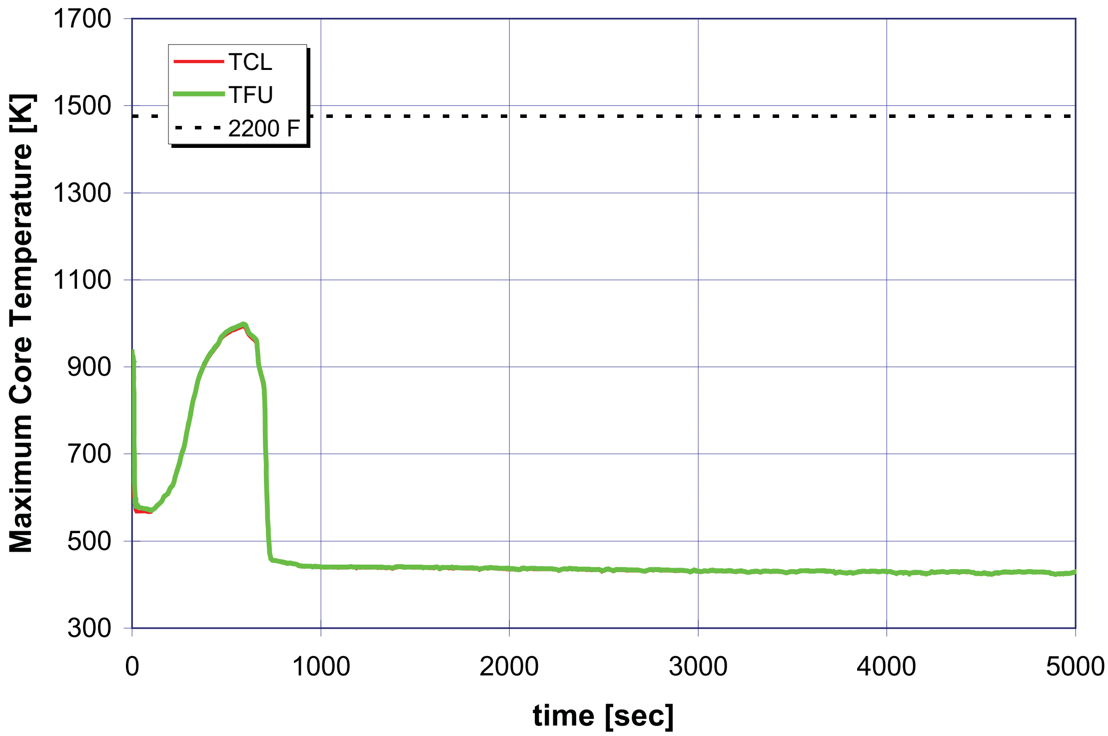
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI



Sump
Cavity
Containment Water Level [m]
time [sec]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
1000
2000
3000
4000
5000

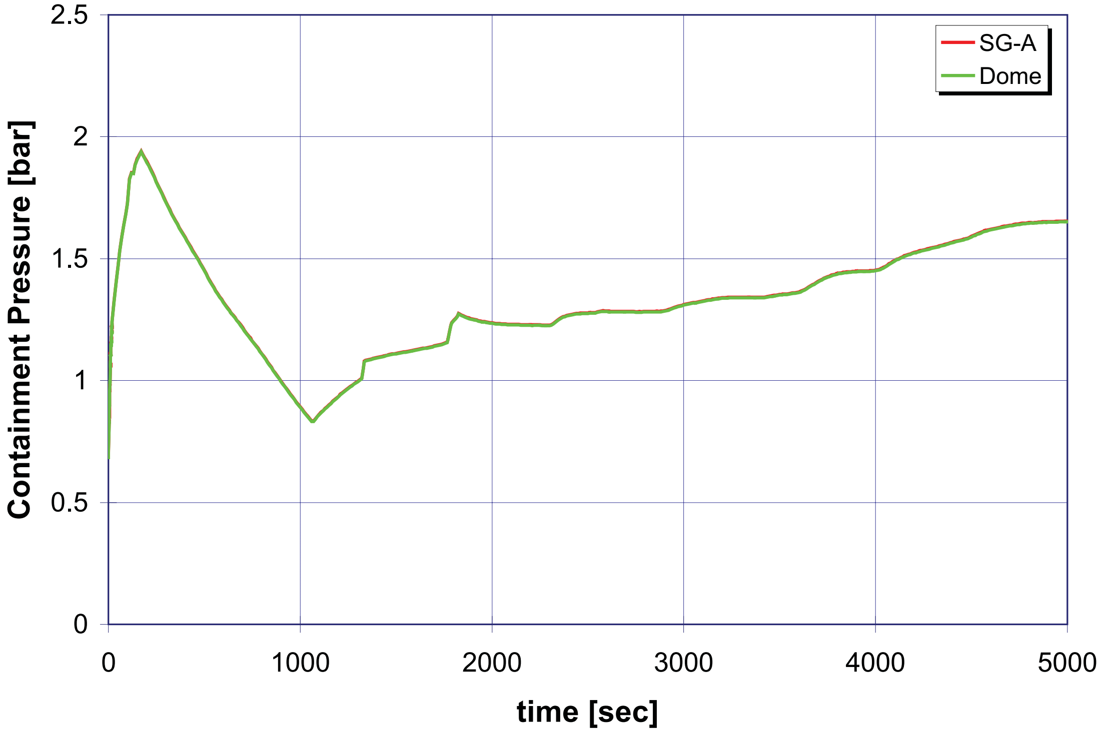


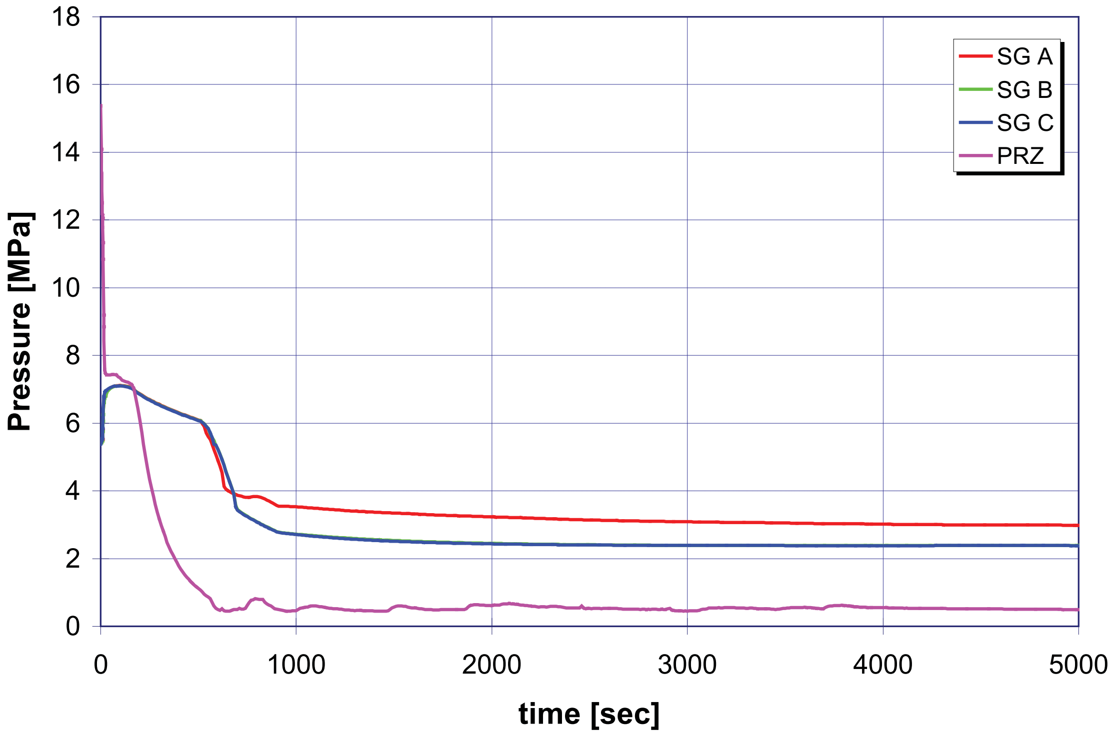
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
20
18
16
14
12
10
8
6
4
2
0
1000
2000
3000
4000
5000

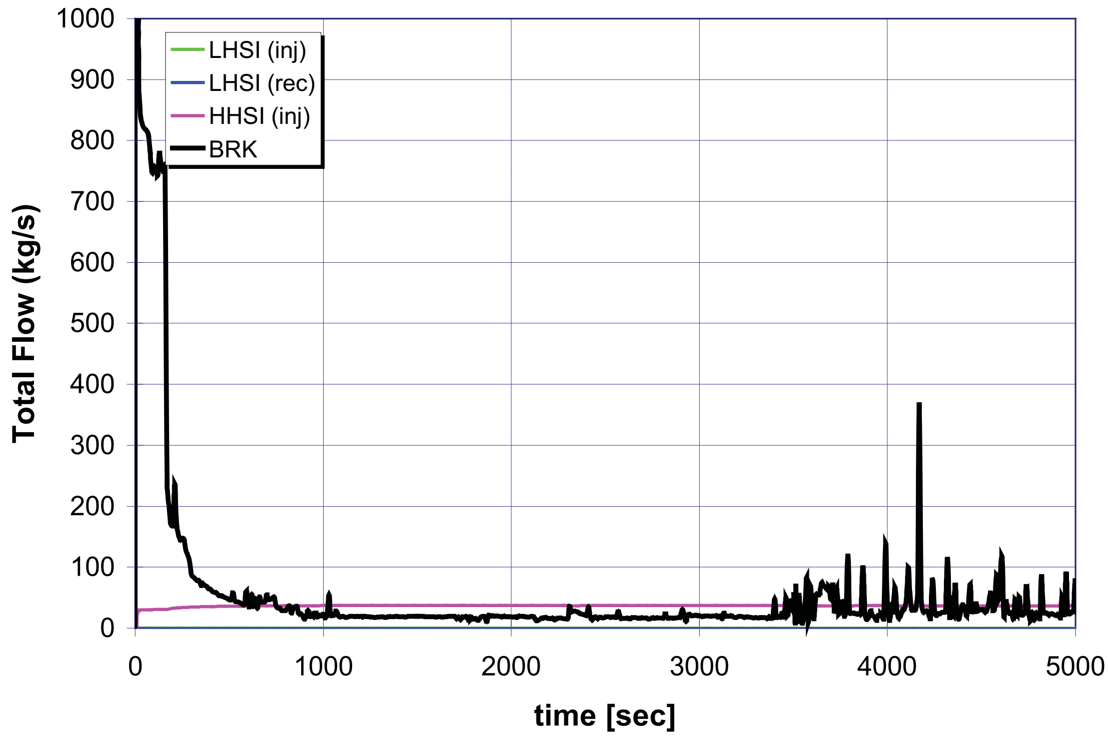


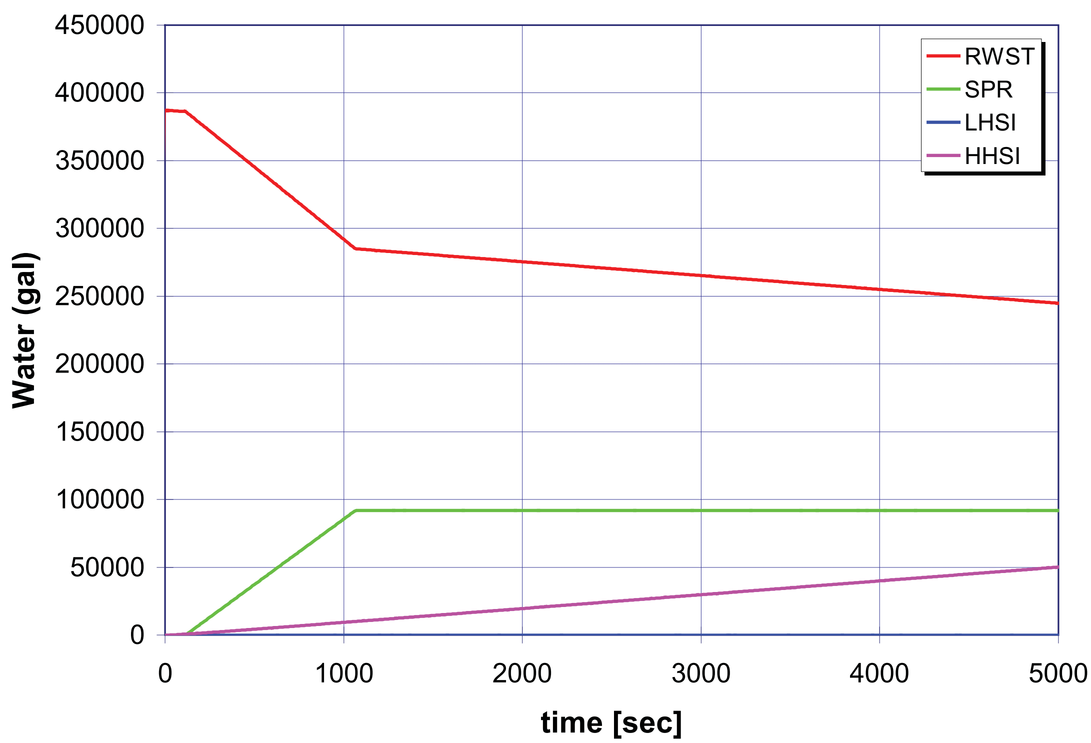
Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
TCL
TFU
2200 F
0
1000
2000
3000
4000
5000
time [sec]

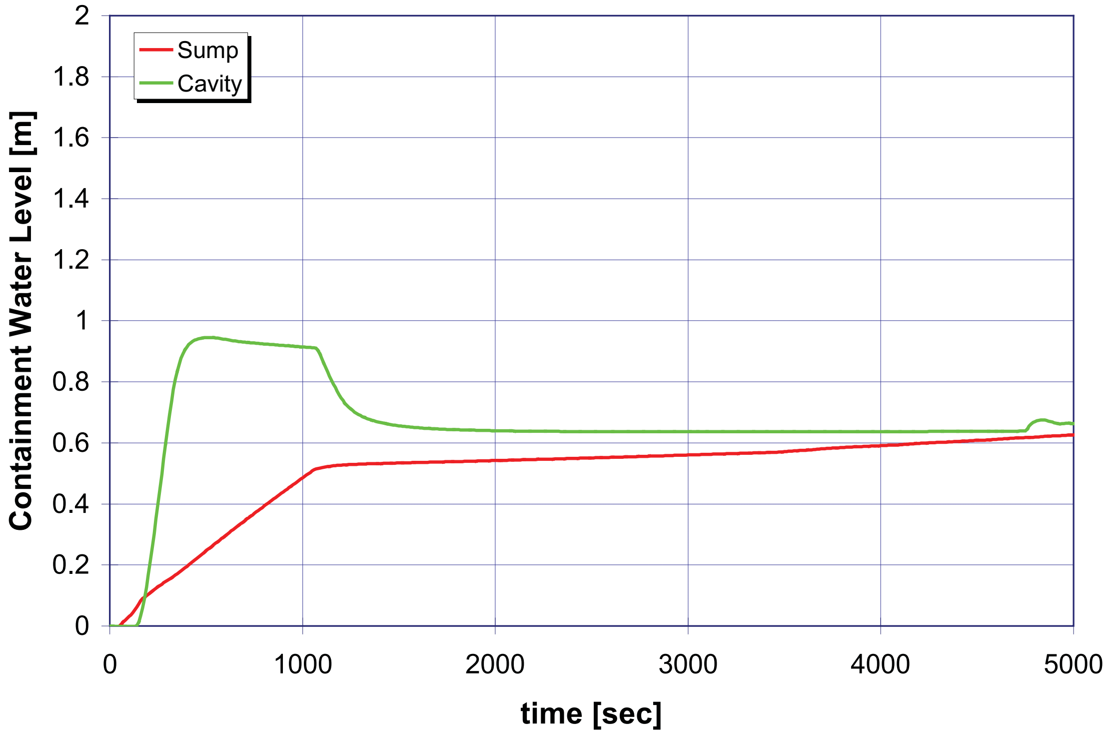
A.6.7 Case 7: 6-Inch Break LOCA, One HHSI, No LHSI, and No ACC



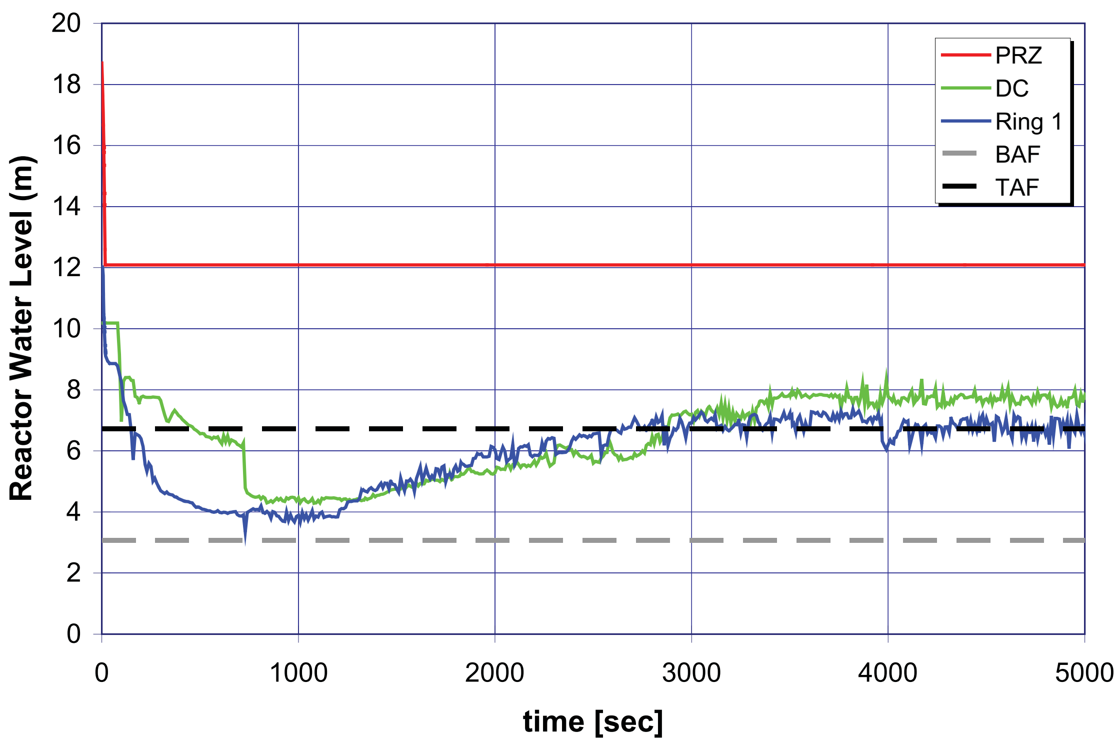



Total Flow (kg/s)
1000
900
800
700
600
500
400
300
200
100
0
0
1000
2000
3000
4000
5000
time [sec]
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK

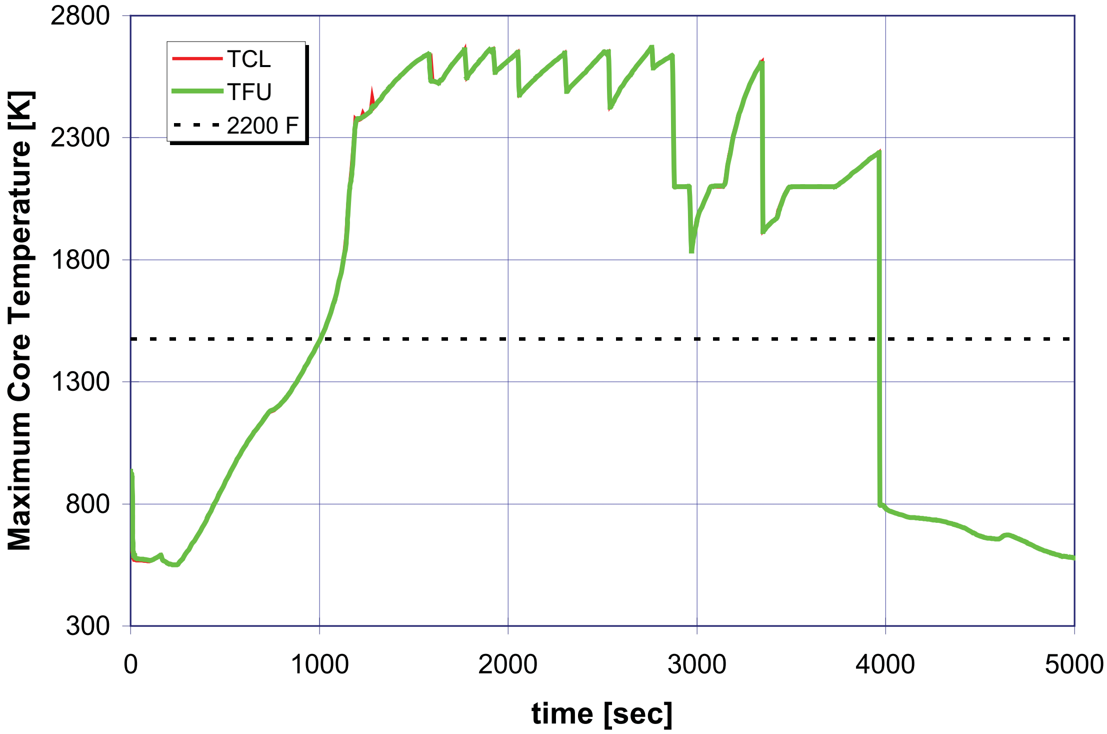

Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
time [sec]
RWST
SPR
LHSI
HHSI



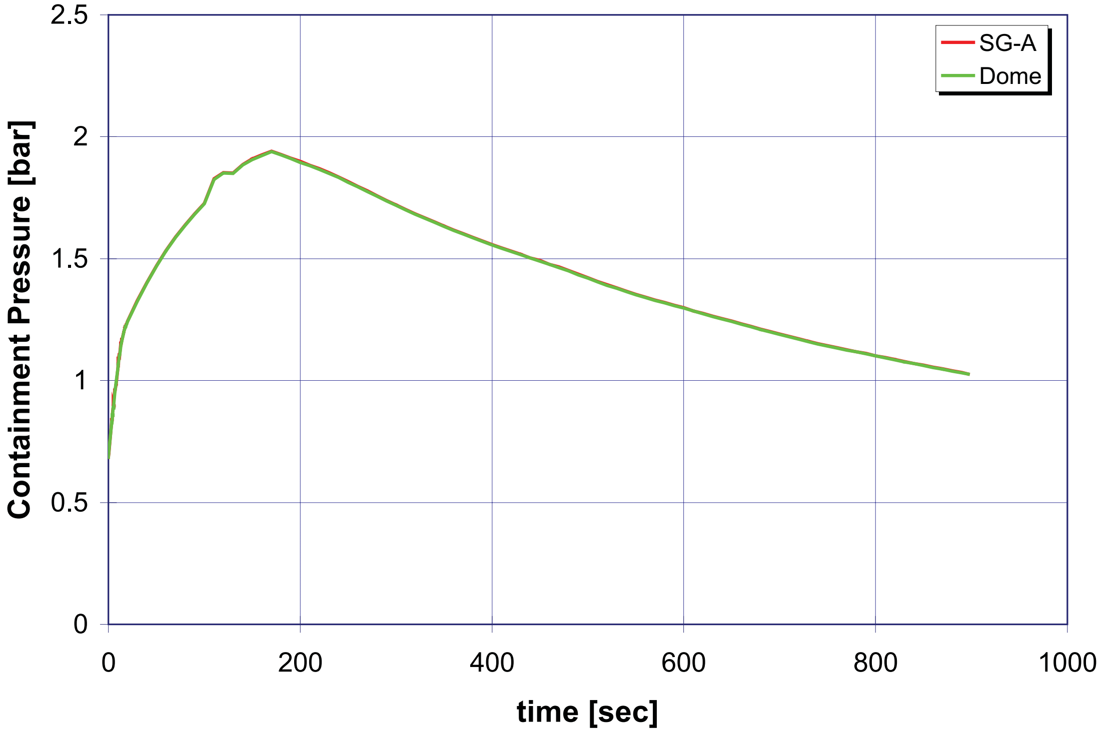
Containment Water Level [m]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
1000
2000
3000
4000
5000
time [sec]

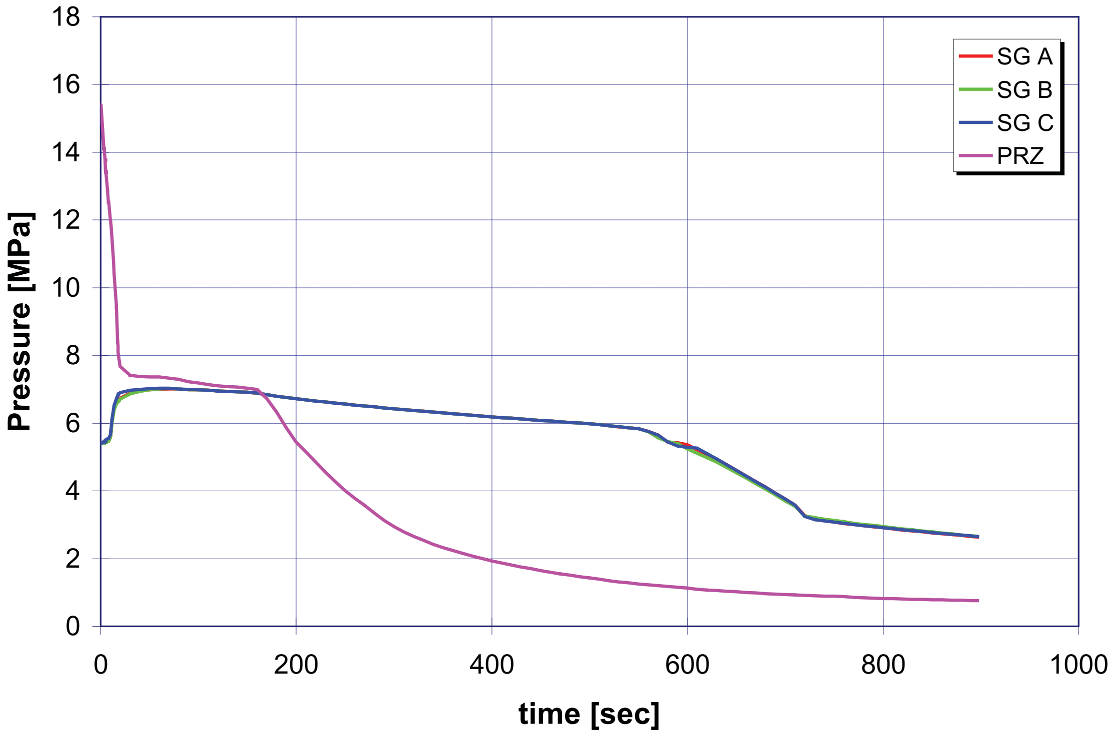


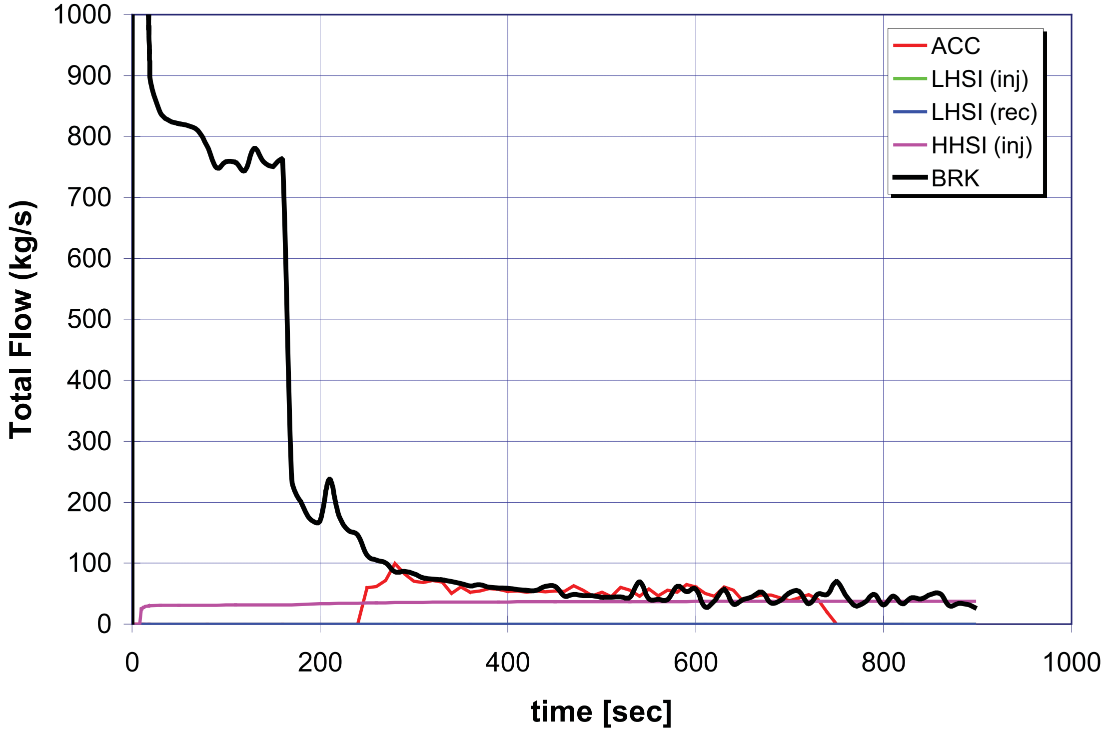
Reactor Water Level (m)
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
1000
2000
3000
4000
5000
time [sec]


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
1000
2000
3000
4000
5000
time [sec]

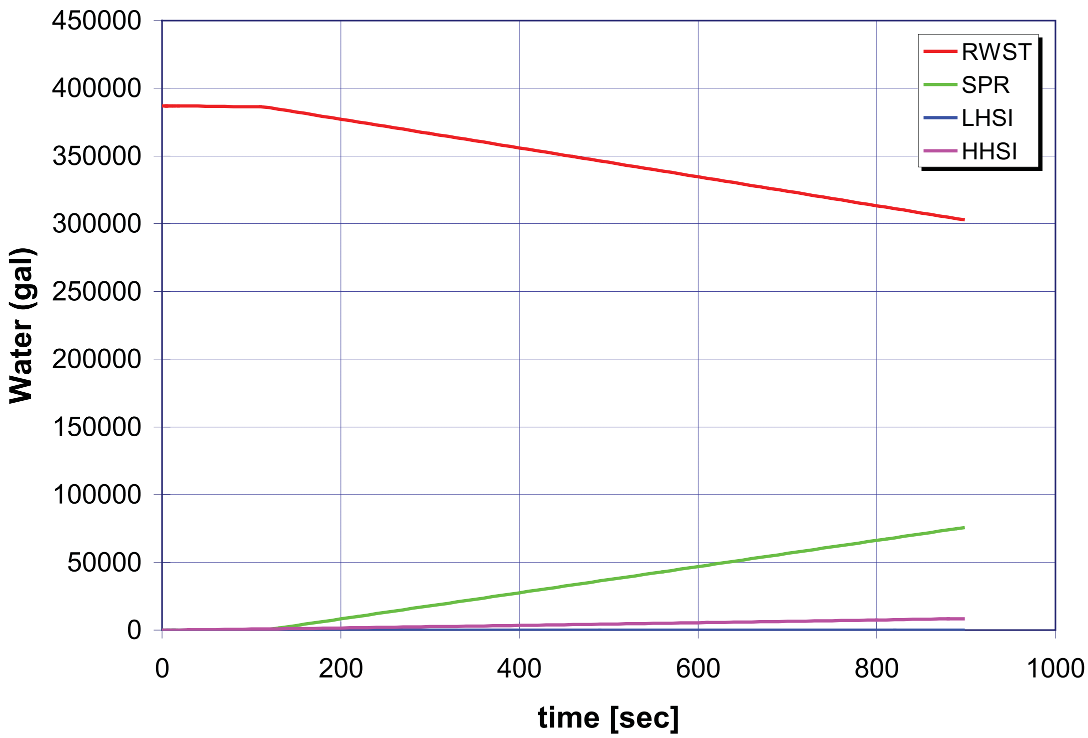
A.6.8 Case 8: 6-Inch Break LOCA, One HHSI, No LHSI, and One ACC



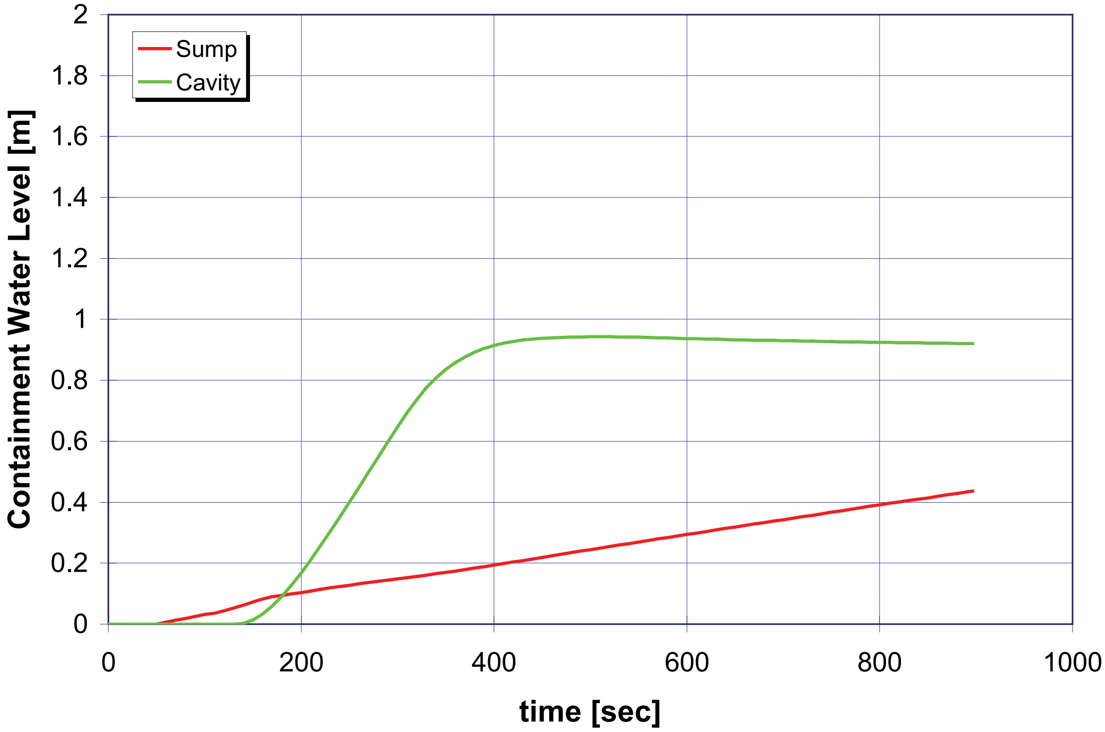


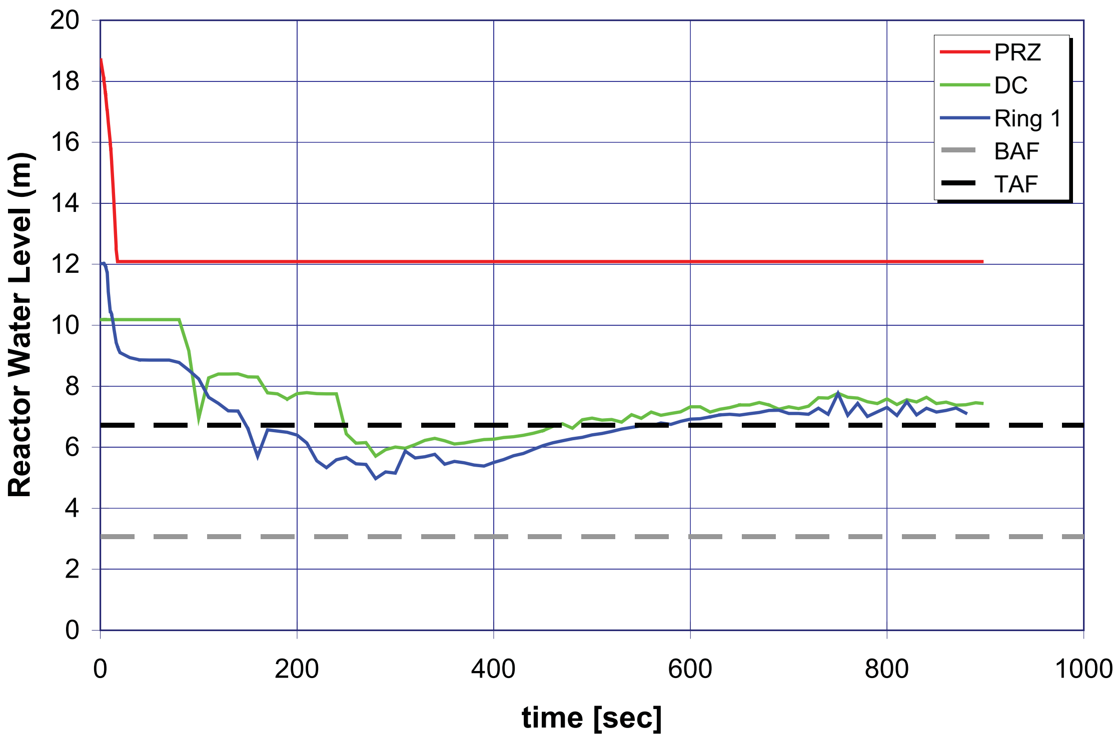


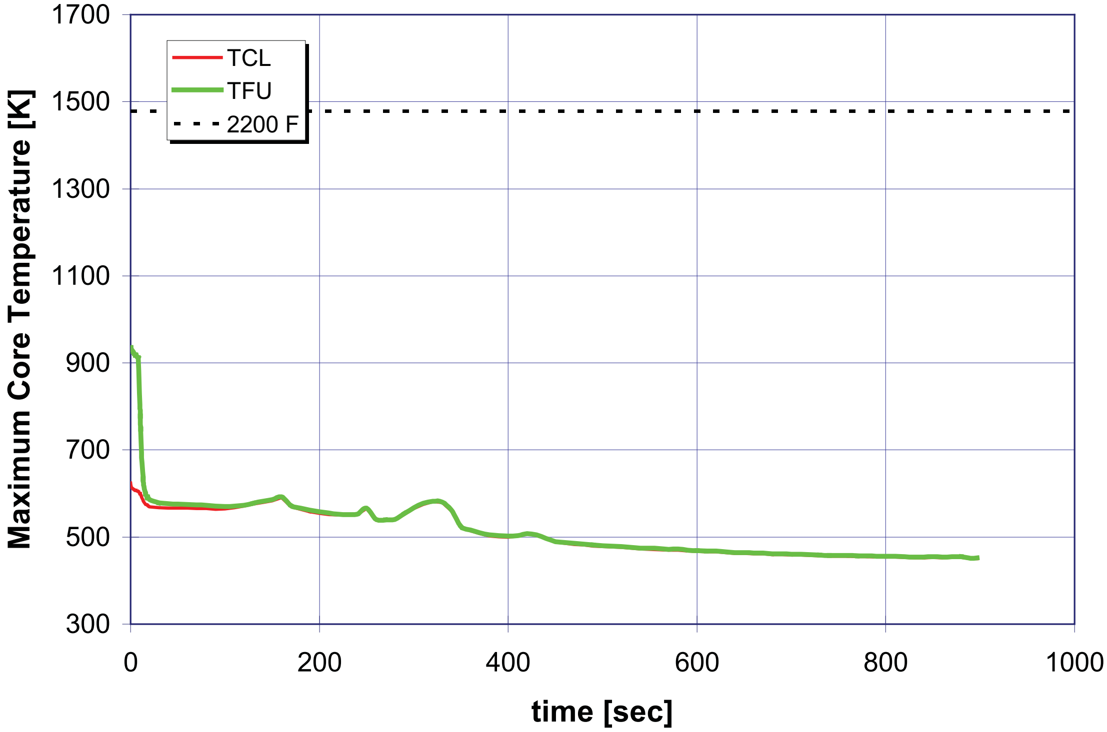
Total Flow (kg/s)
1000
900
800
700
600
500
400
300
200
100
0
0
200
400
600
800
1000
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
200
400
600
800
1000
time [sec]
RWST
SPR
LHSI
HHSI

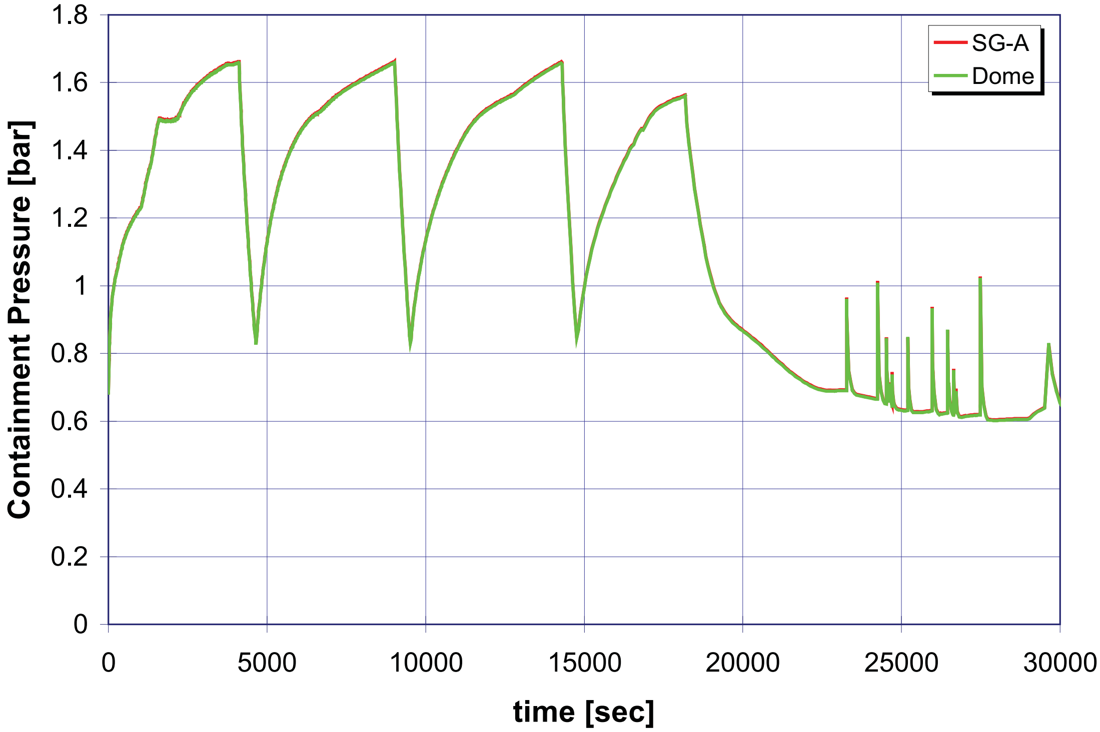

Sump
Cavity
Containment Water Level [m]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
200
400
600
800
1000
time [sec]

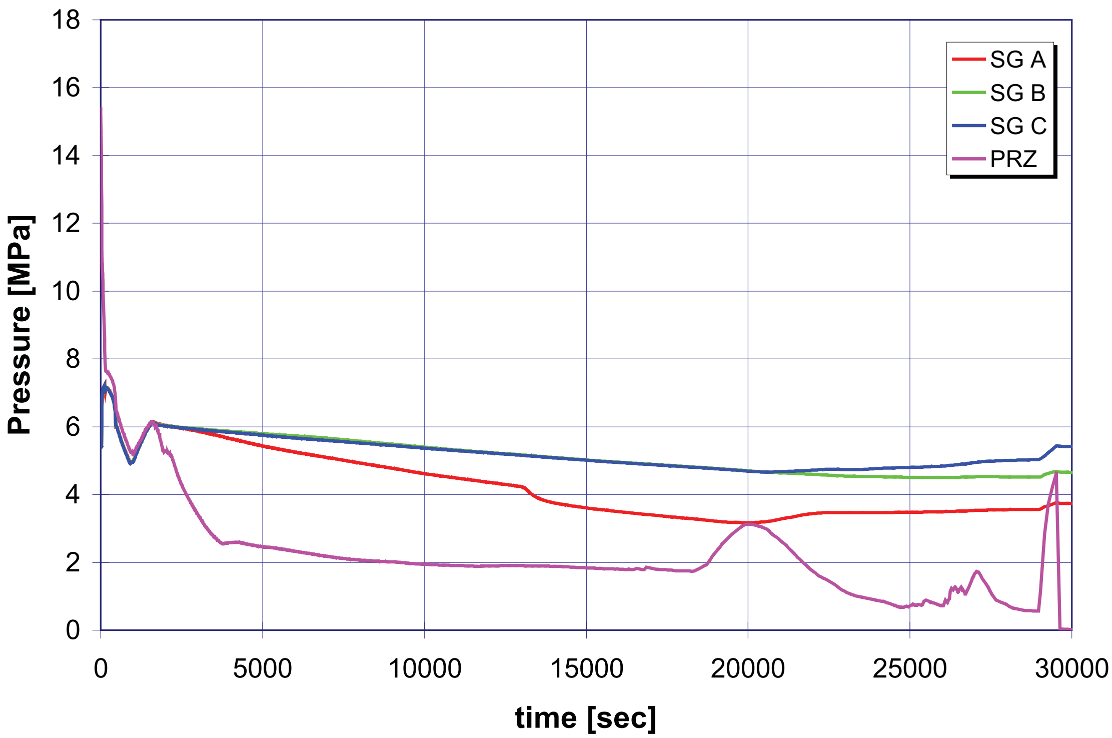

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
20
18
16
14
12
10
8
6
4
2
0
0
200
400
600
800
1000
time [sec]

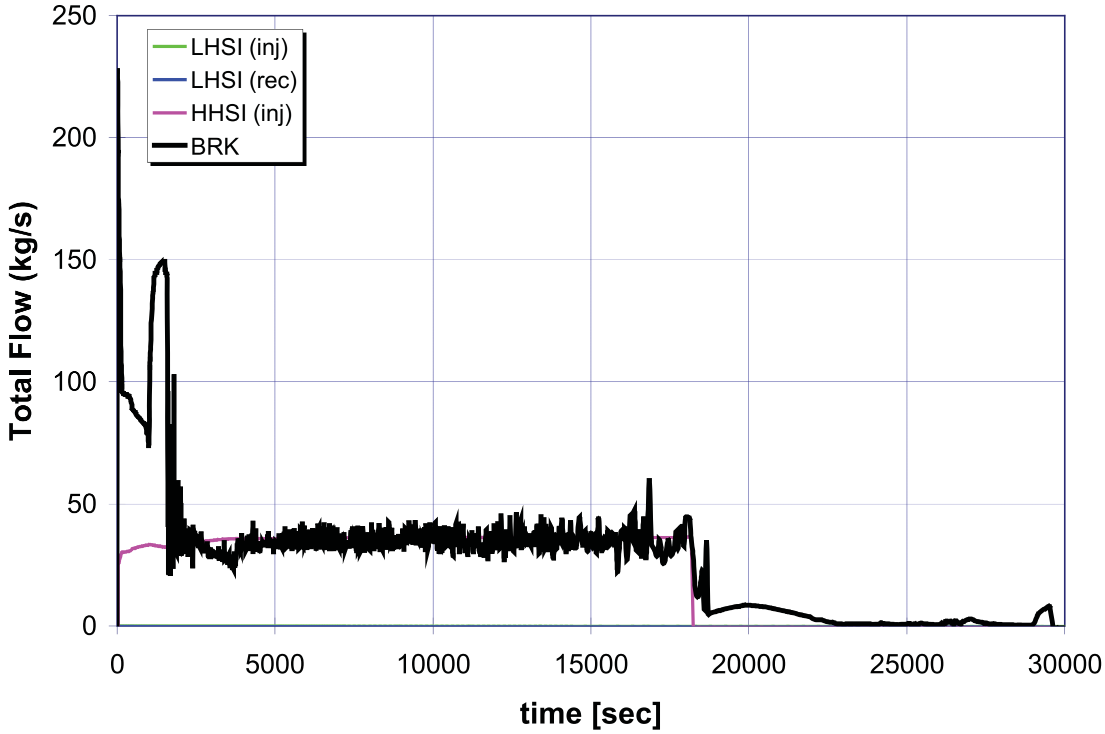


TCL
TFU
2200 F
Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
200
400
600
800
1000
time [sec]

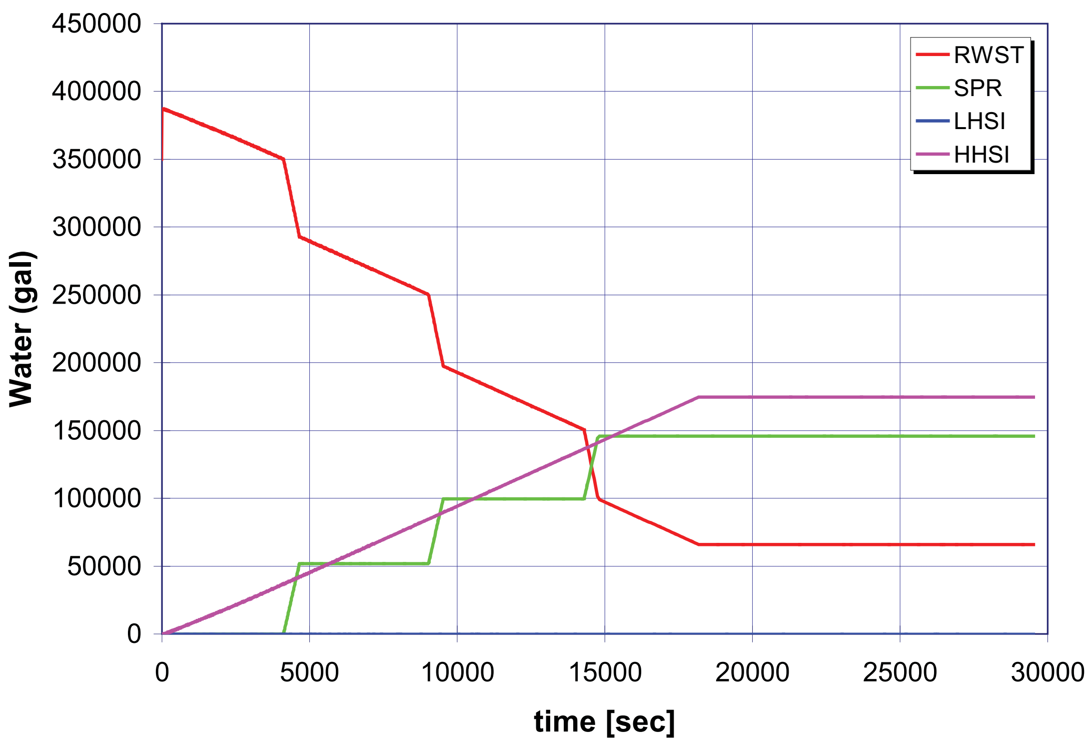
A.6.9 Case 9: 2-Inch Break LOCA, One HHSI, No LHSI, and No ACC



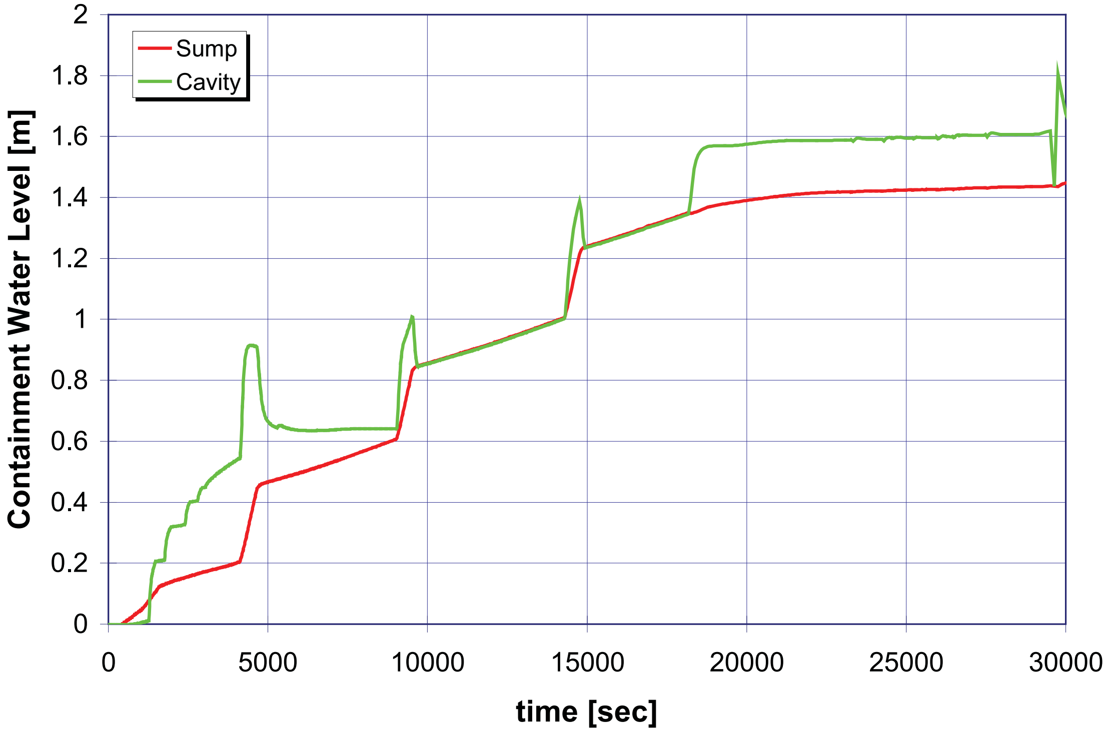


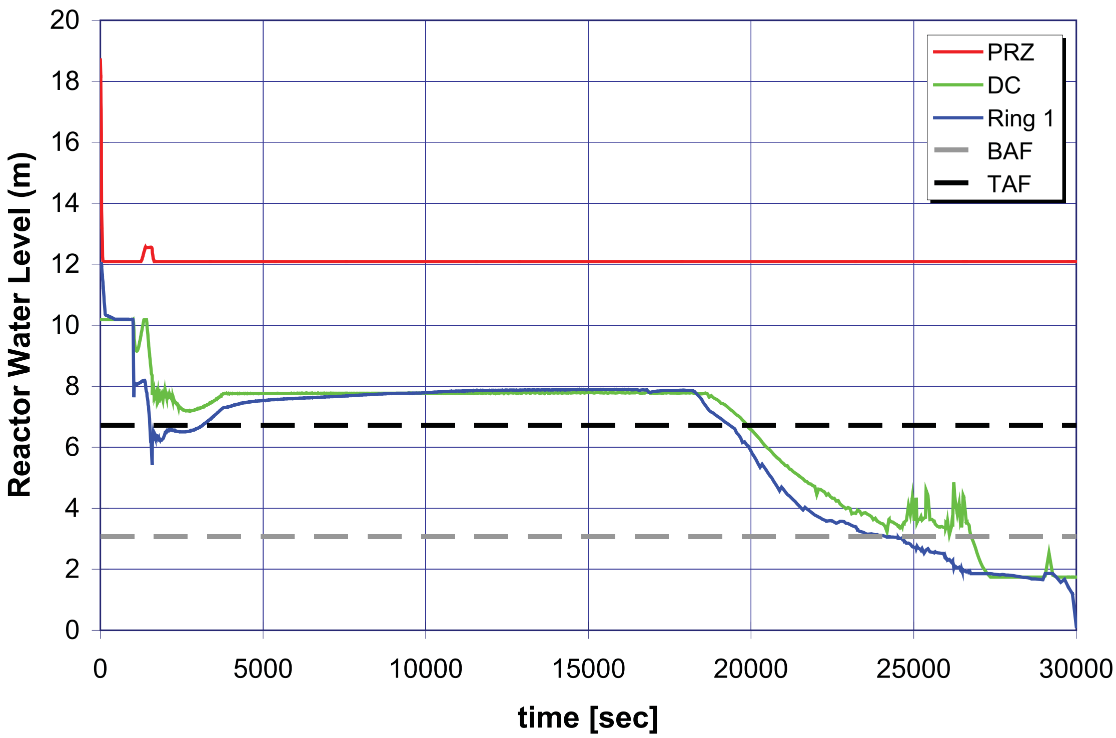


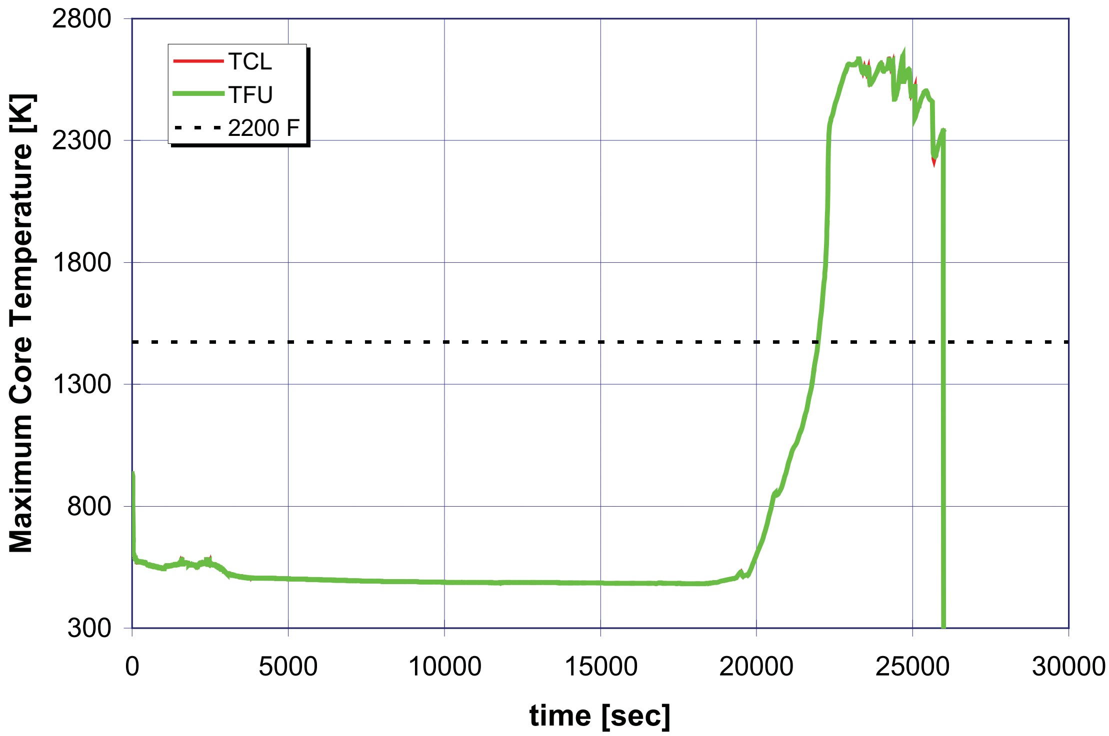
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
250
200
150
100
50
0
0
5000
10000
15000
20000
25000
30000
time [sec]



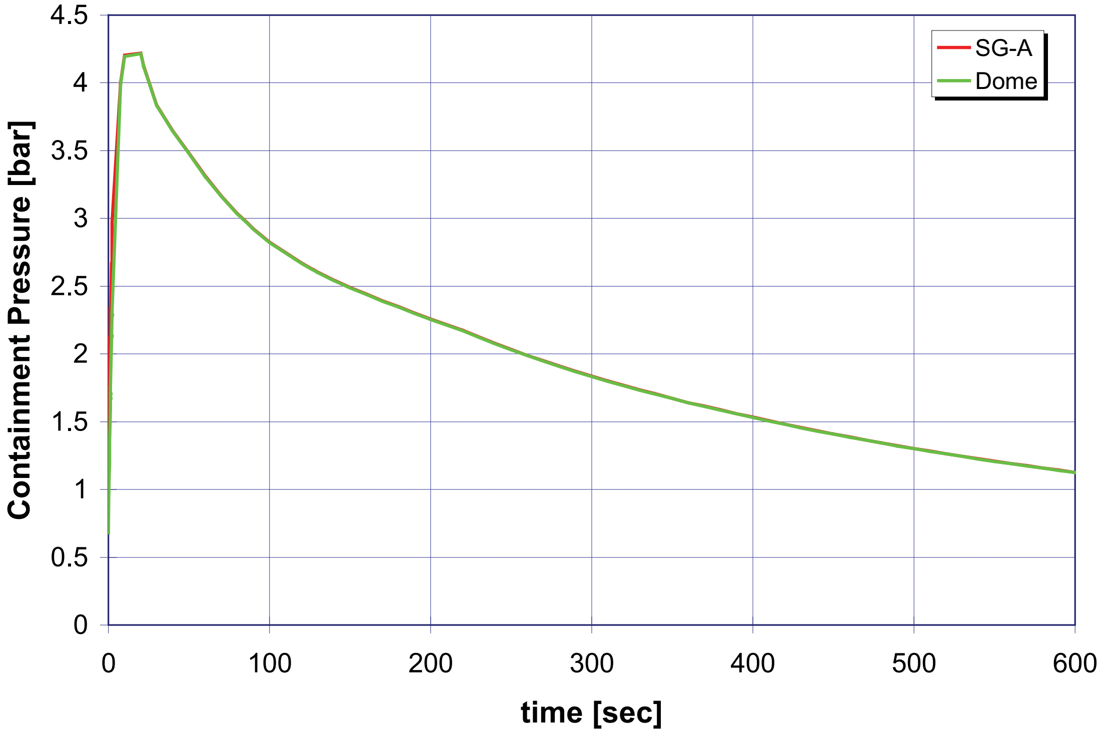
RWST
SPR
LHSI
HHSI
Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
5000
10000
15000
20000
25000
30000
time [sec]

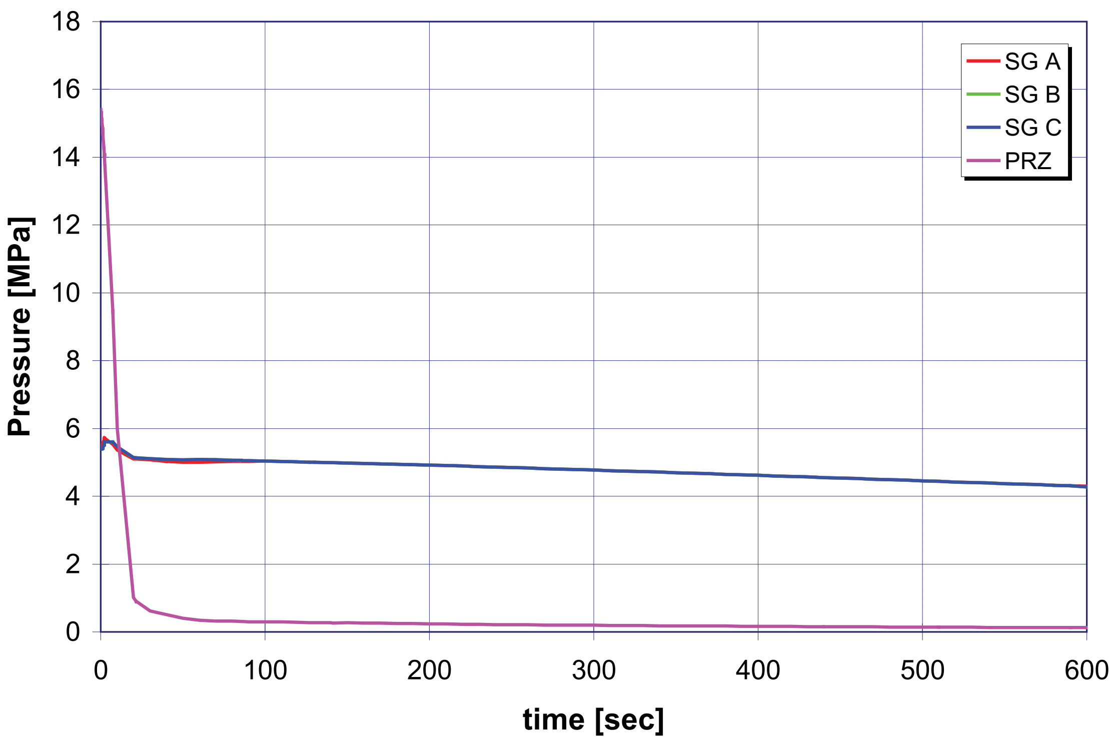

Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000

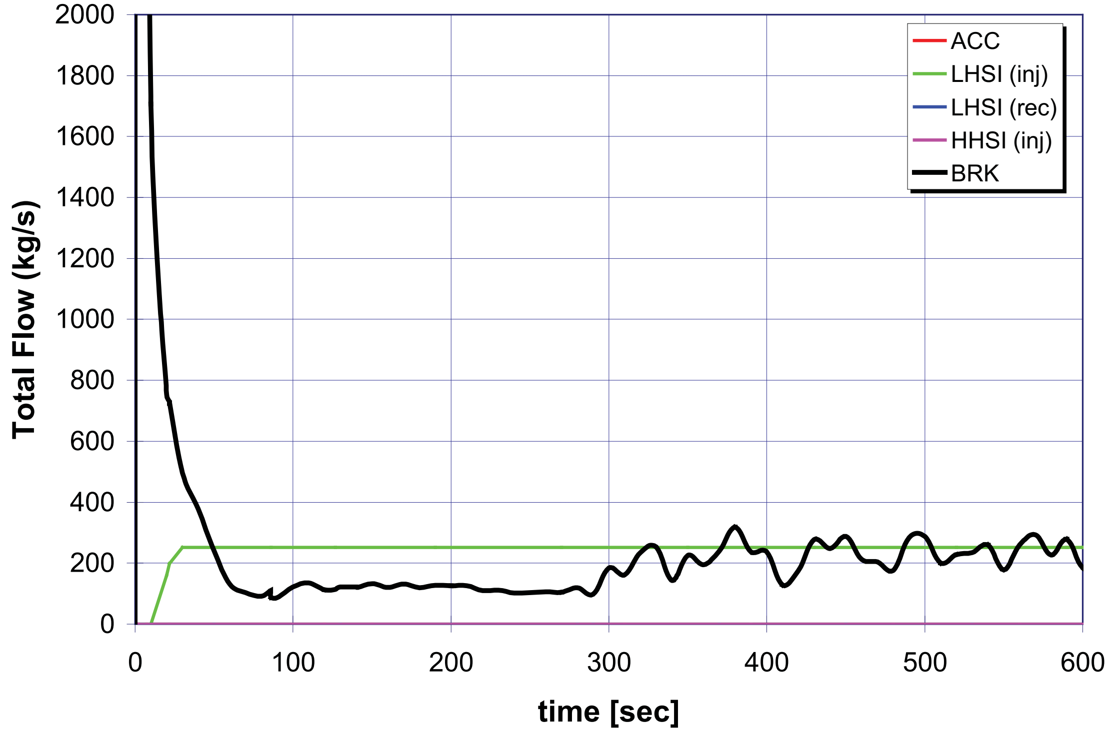

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
5000
10000
15000
20000
25000
30000


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
5000
10000
15000
20000
25000
30000
time [sec]

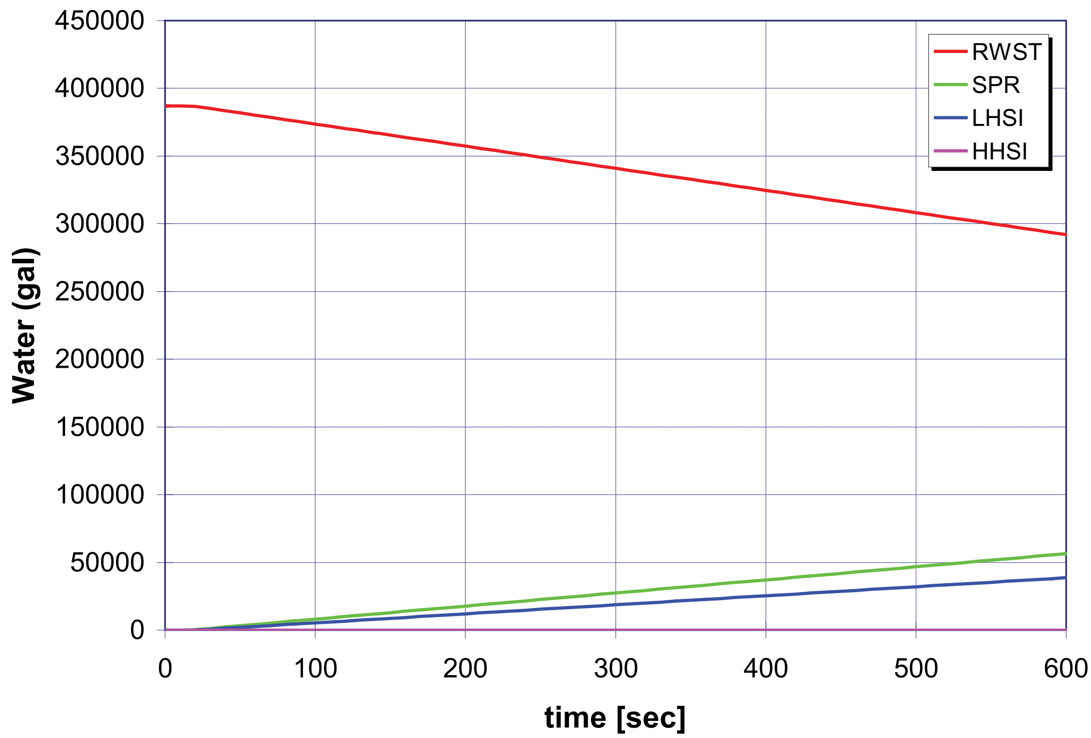
A.6.10 Case 10: Double-Ended Cold-Leg Break LOCA, No HHSI, One LHSI, and No ACC



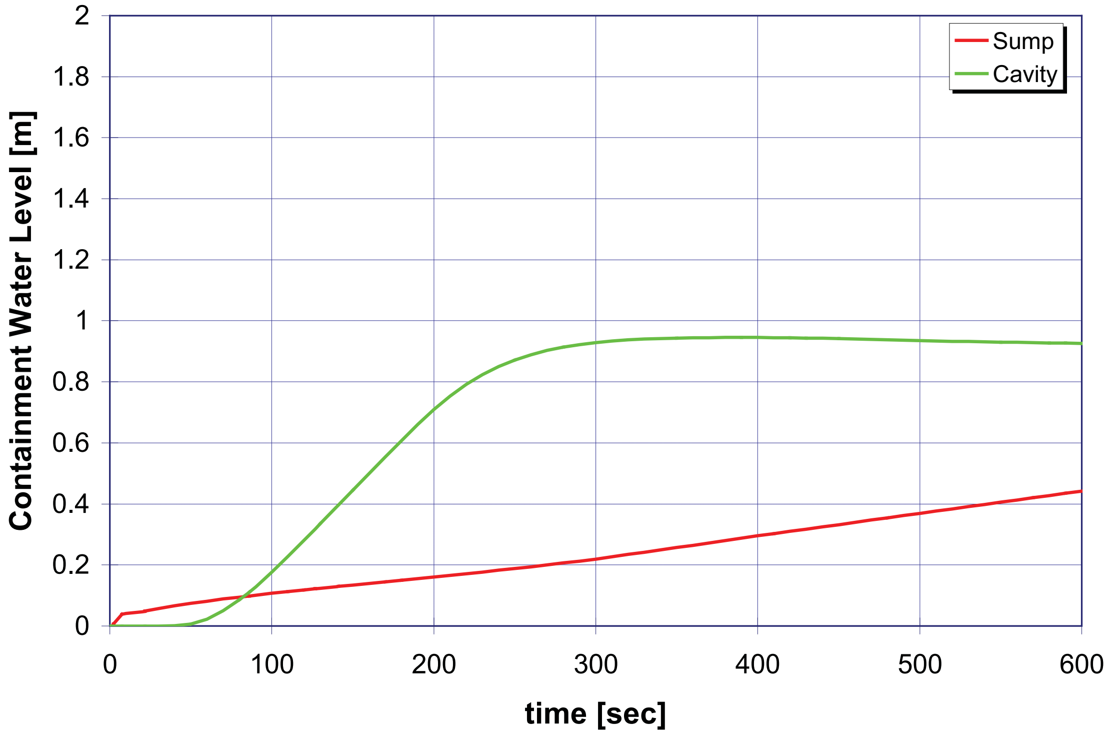


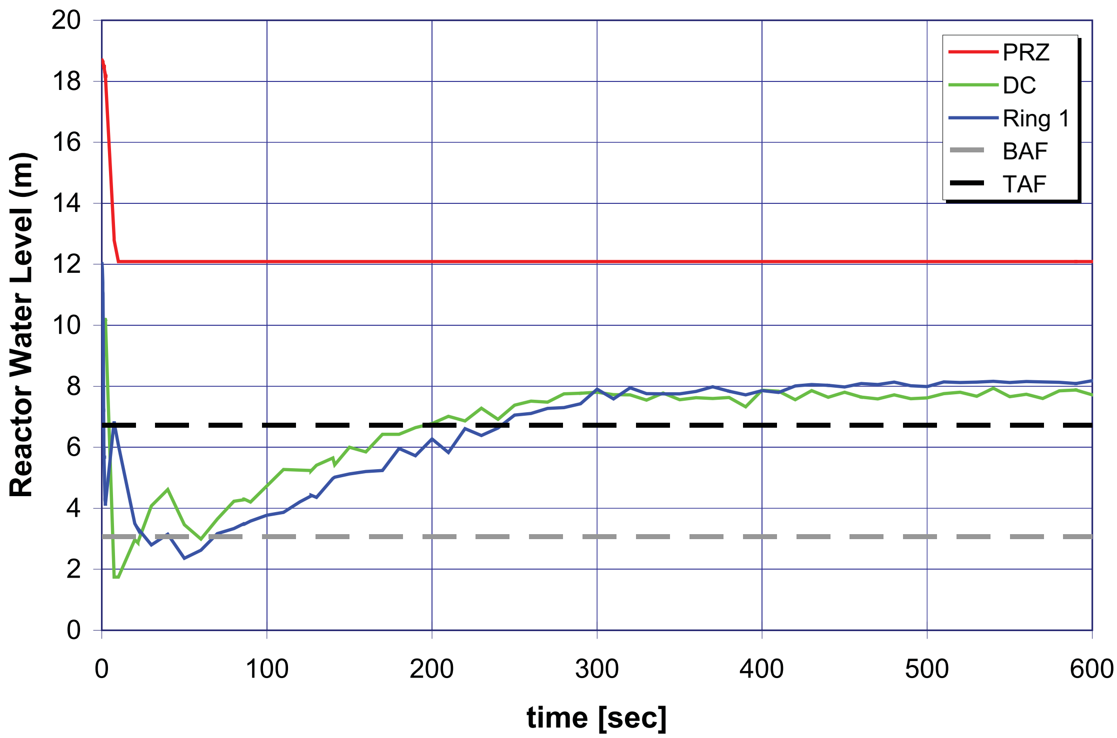


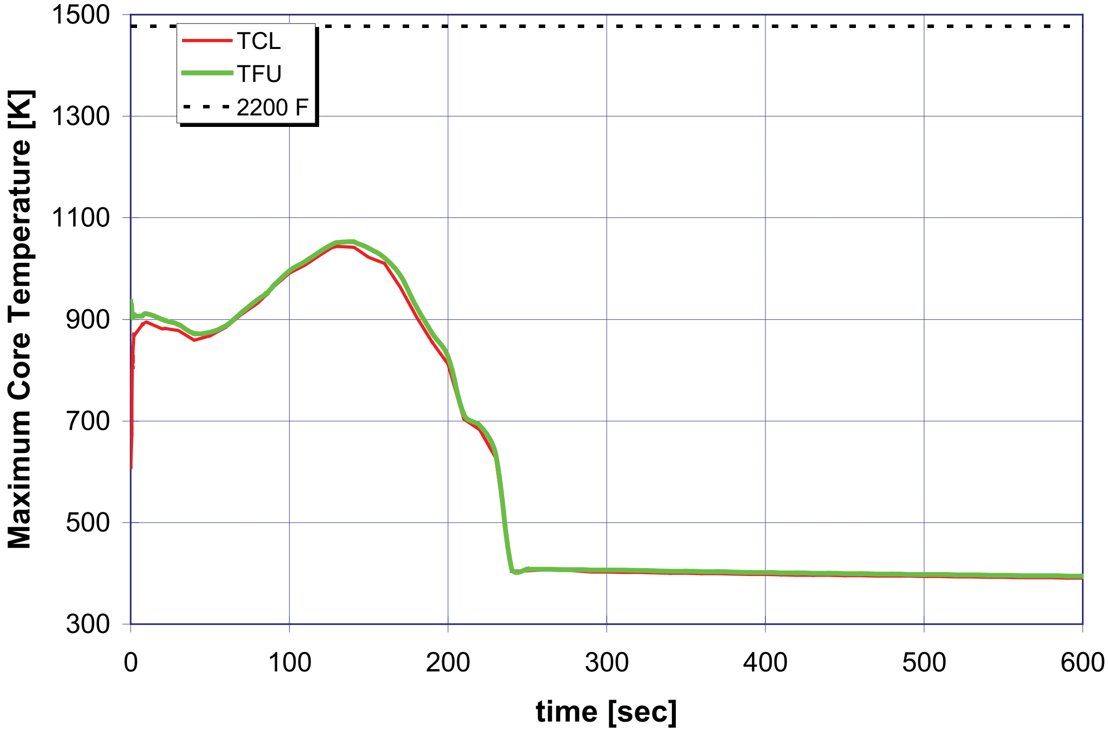
Total Flow (kg/s)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
100
200
300
400
500
600
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
100
200
300
400
500
600
time [sec]
RWST
SPR
LHSI
HHSI

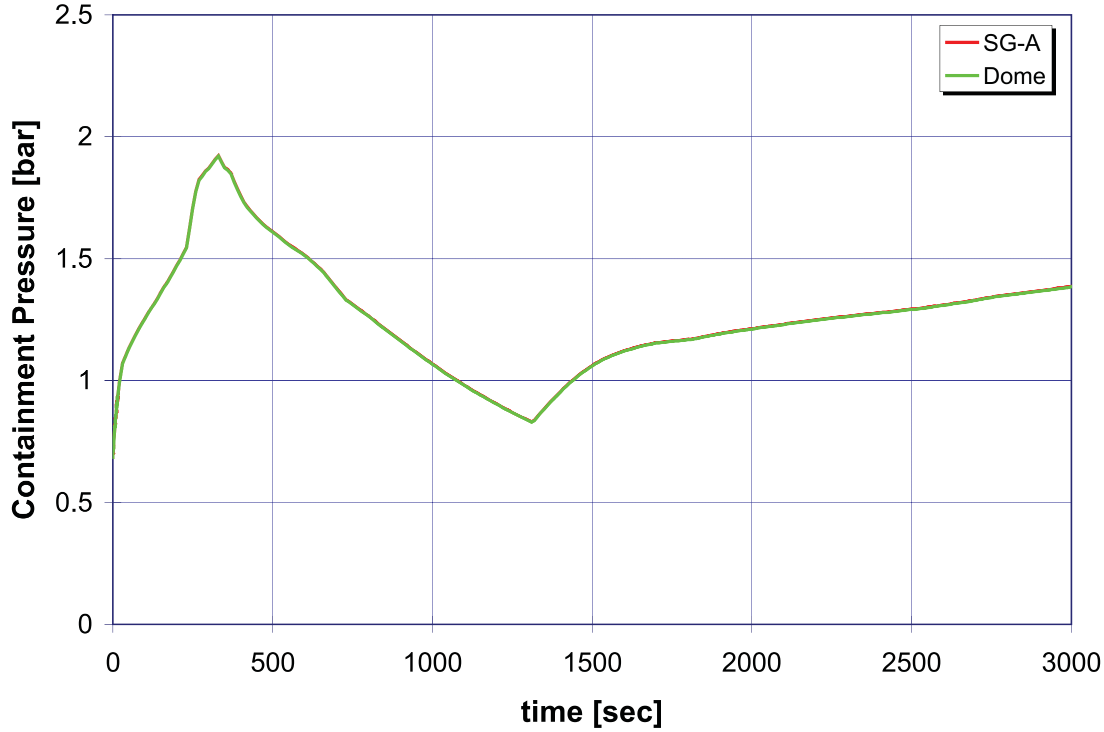

Containment Water Level [m]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
Sump
Cavity
0
100
200
300
400
500
600
time [sec]

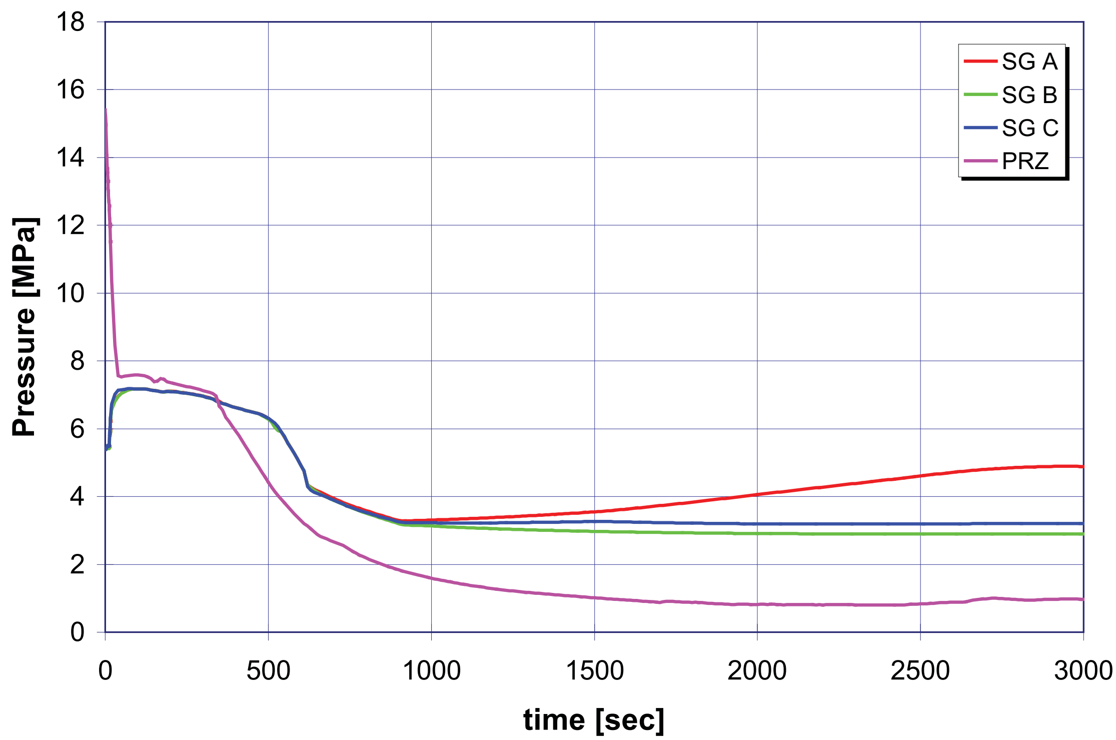

Reactor Water Level (m)
20
18
16
14
12
10
8
6
4
2
0
PRZ
DC
Ring 1
BAF
TAF
0
100
200
300
400
500
600
time [sec]

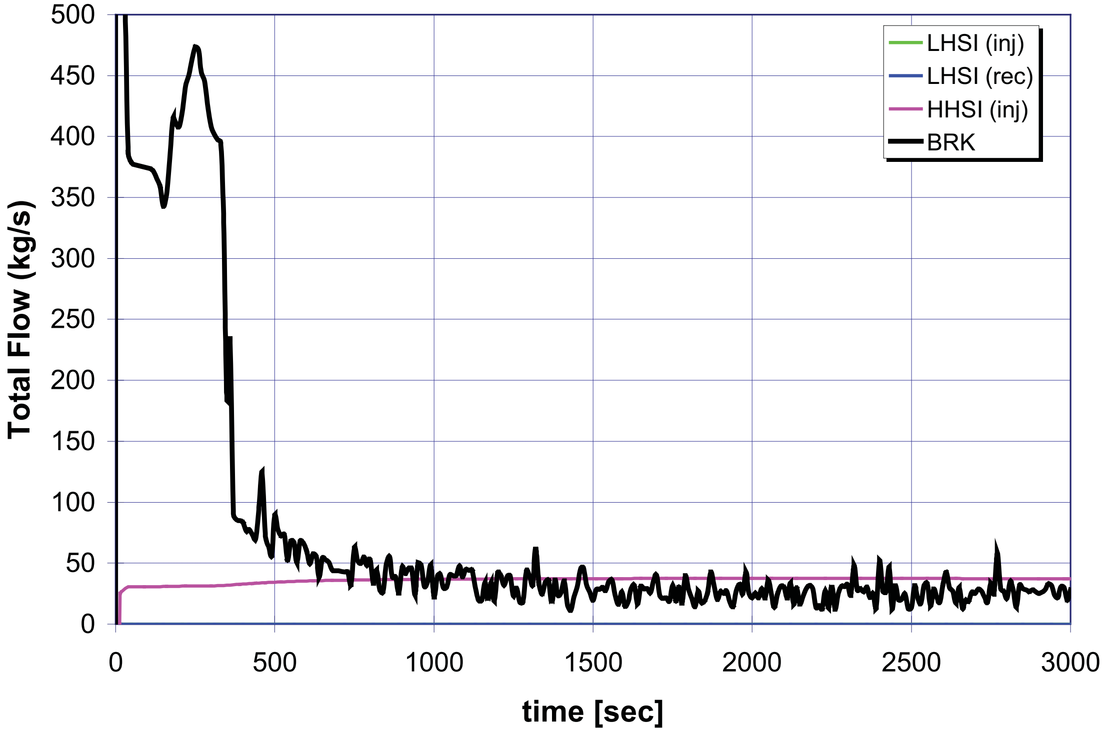


1500
1300
1100
900
700
500
300
Maximum Core Temperature [K]
TCL
TFU
2200 F
0
100
200
300
400
500
600
time [sec]

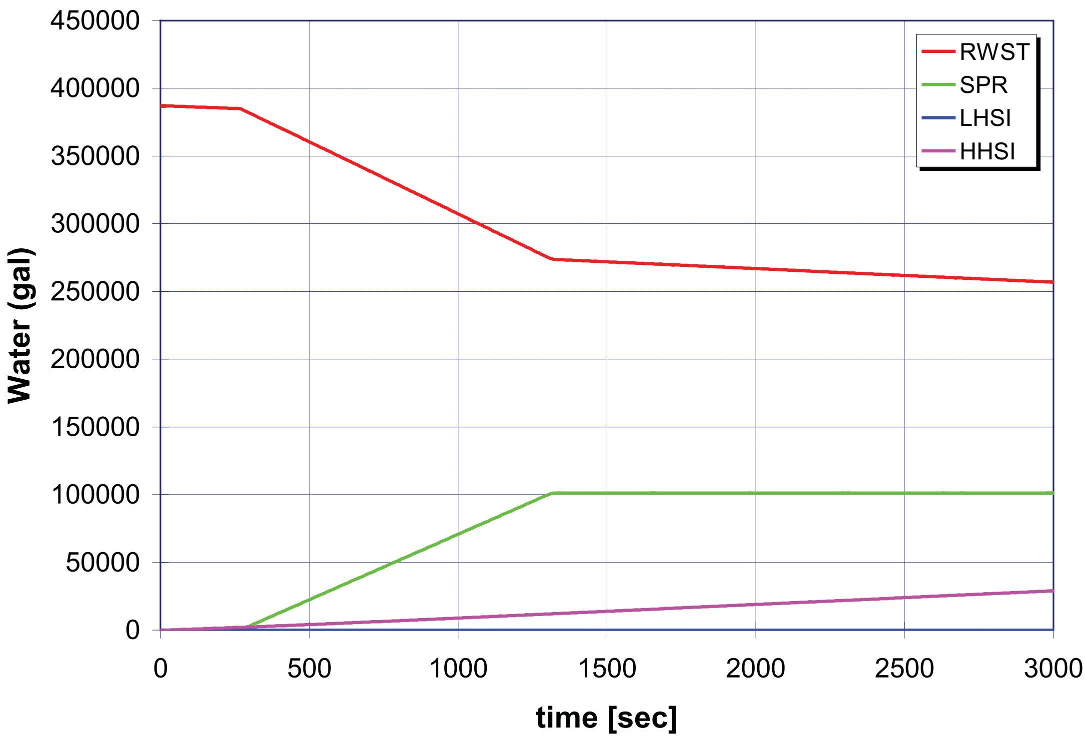
A.6.11 Case 11: 4-Inch Break LOCA, One HHSI, No LHSI, and No ACC



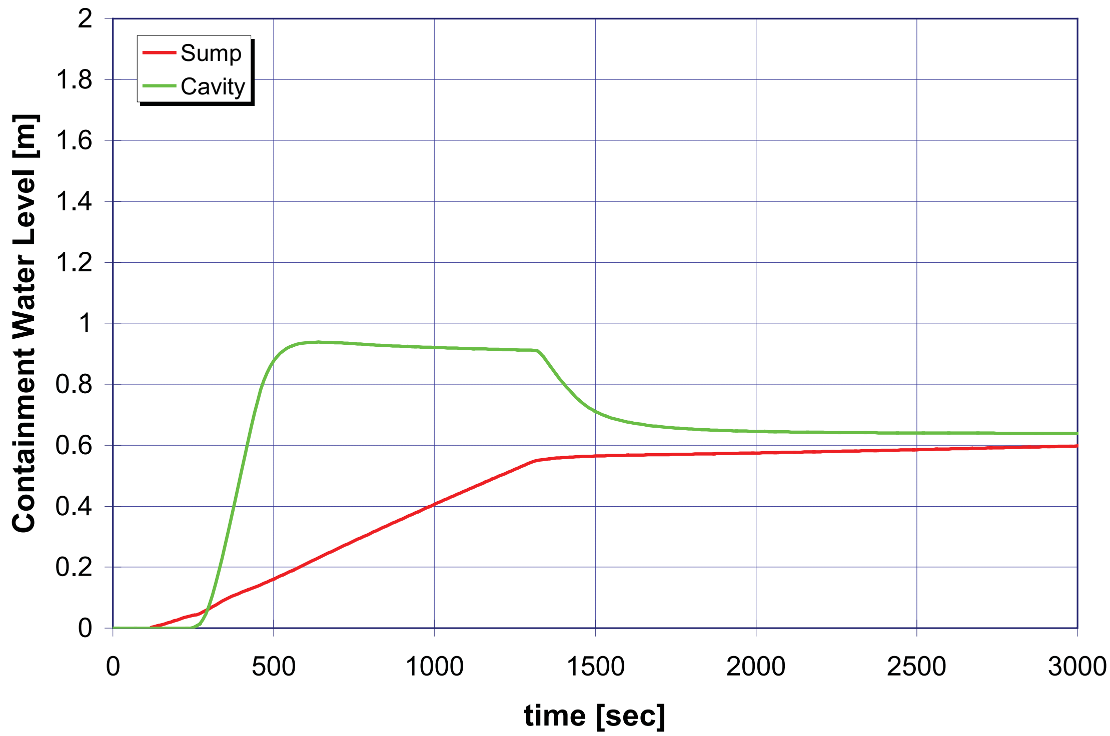


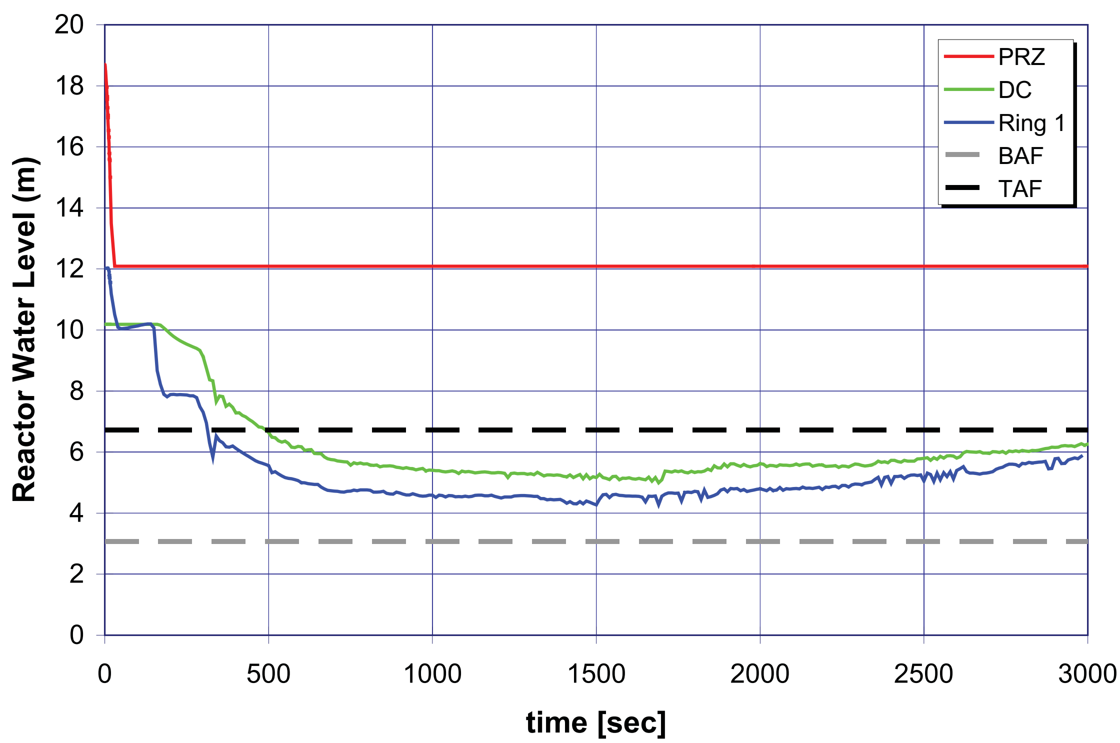


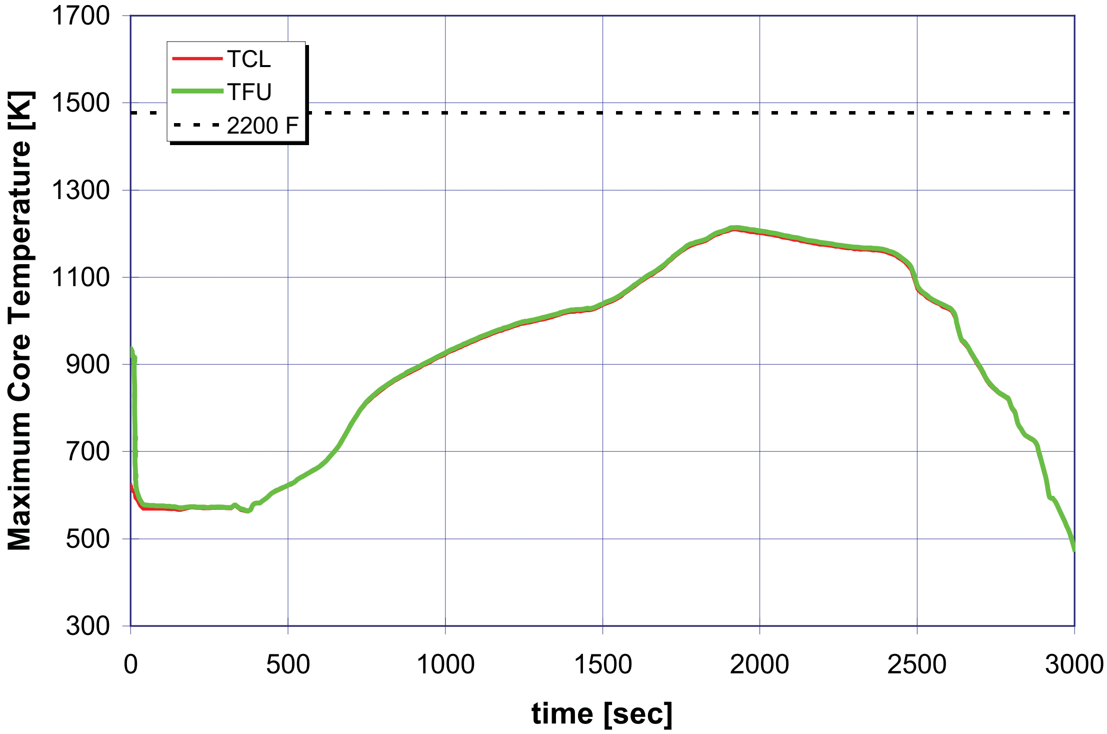
500
450
400
350
300
250
200
150
100
50
0
Total Flow (kg/s)
0
500
1000
1500
2000
2500
3000
time [sec]
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
500
1000
1500
2000
2500
3000
time [sec]
RWST
SPR
LHSI
HHSI

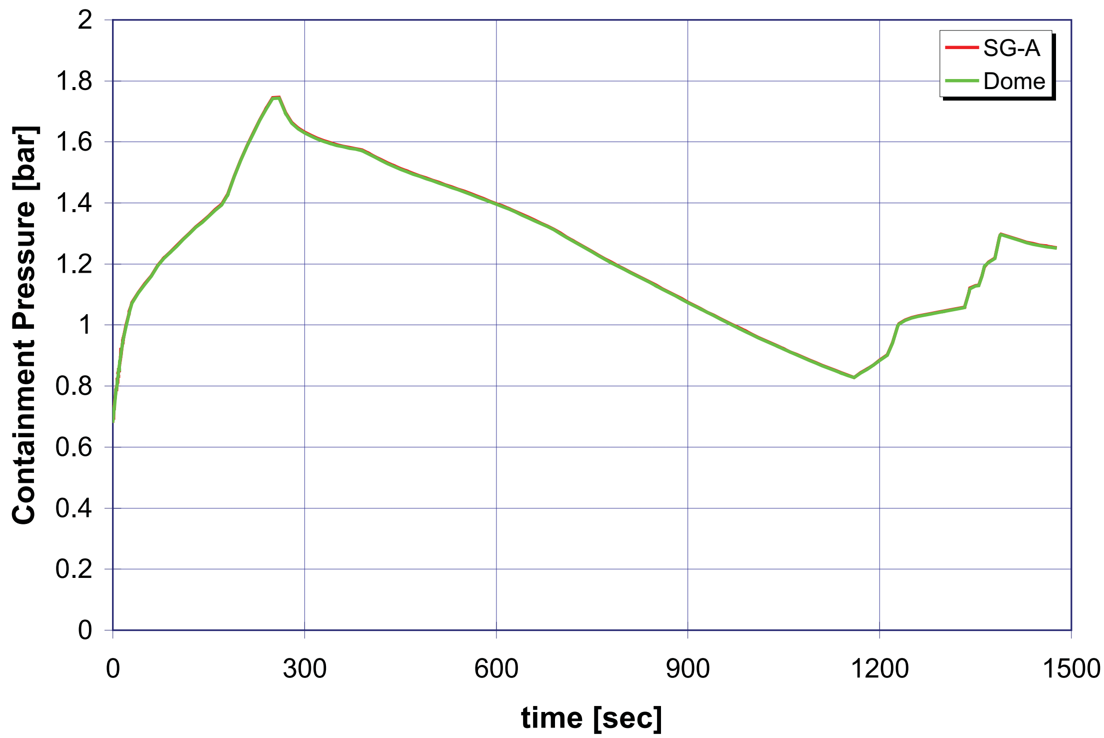

Containment Water Level [m]
time [sec]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000

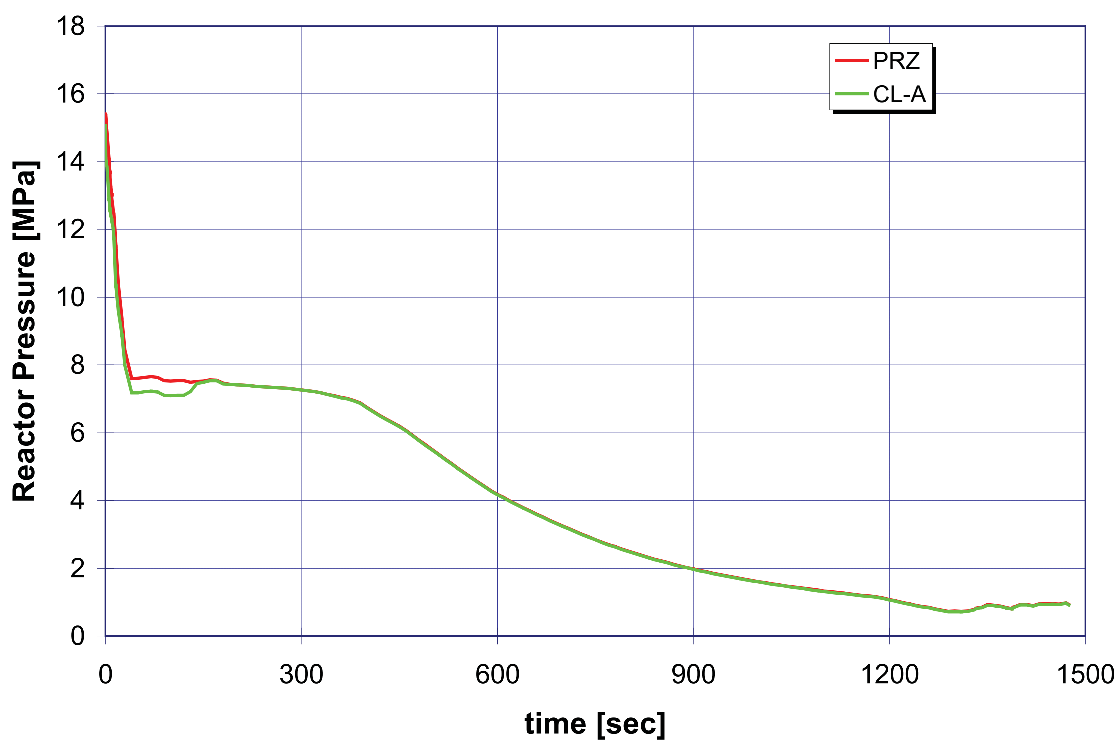

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
500
1000
1500
2000
2500
3000

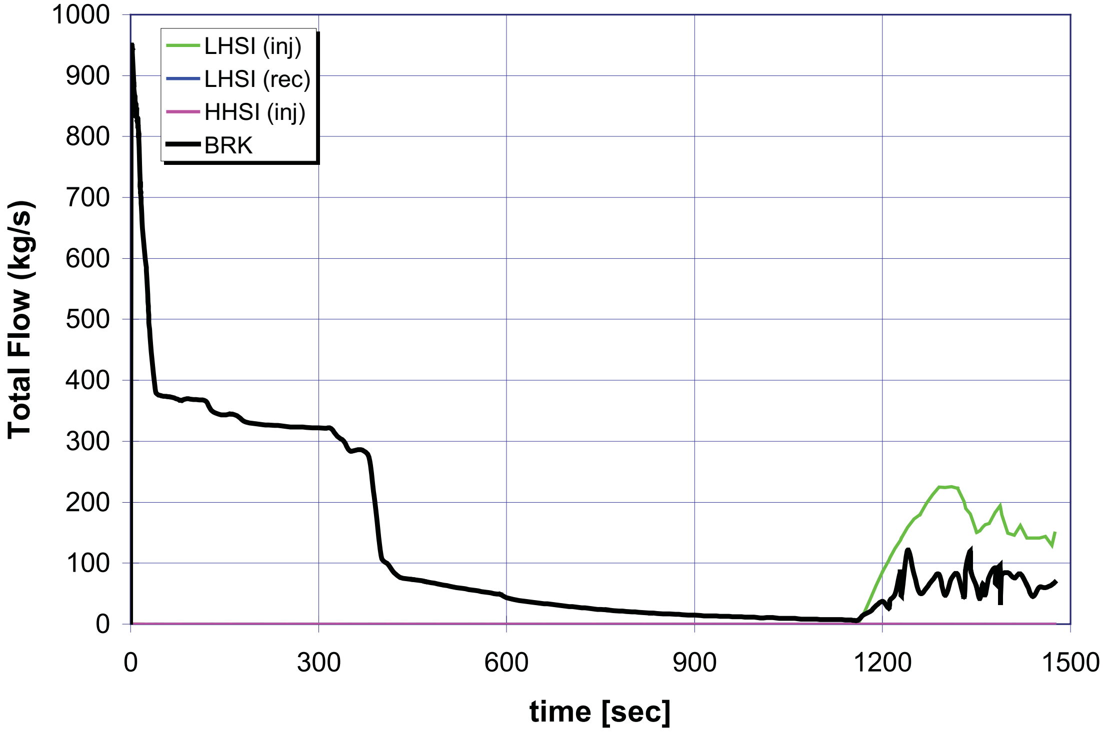


Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
TCL
TFU
2200 F
0
500
1000
1500
2000
2500
3000
time [sec]

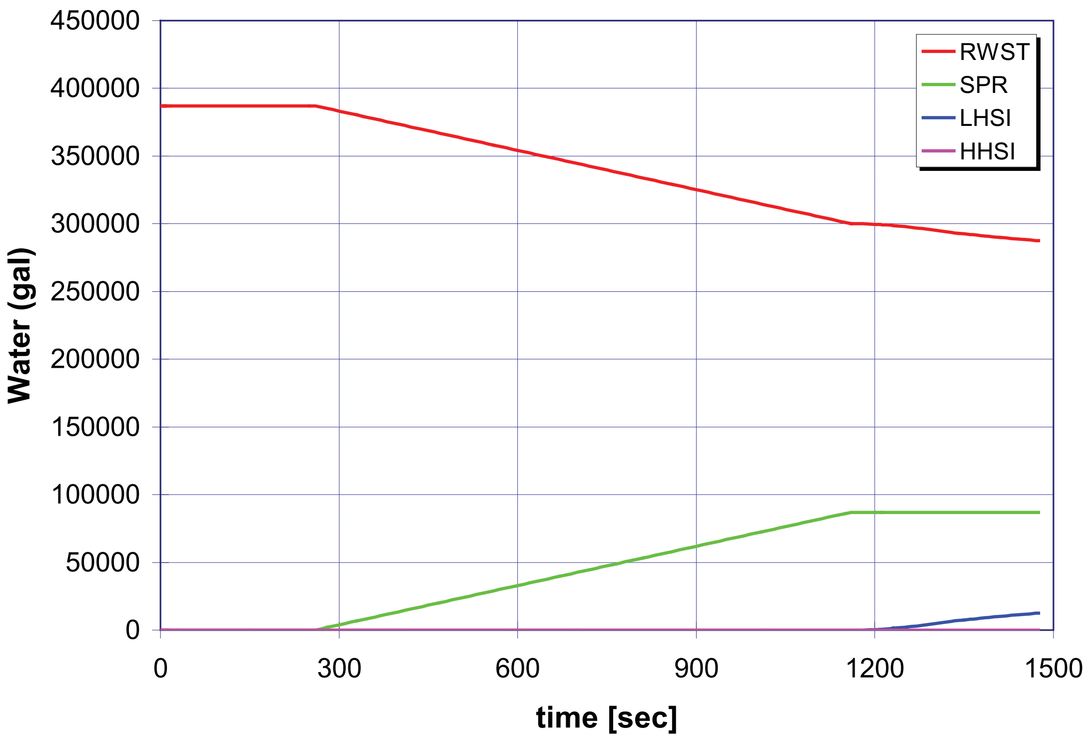
A.6.12 Case 12: 4-Inch Break LOCA, No HHSI, One LHSI, and No ACC



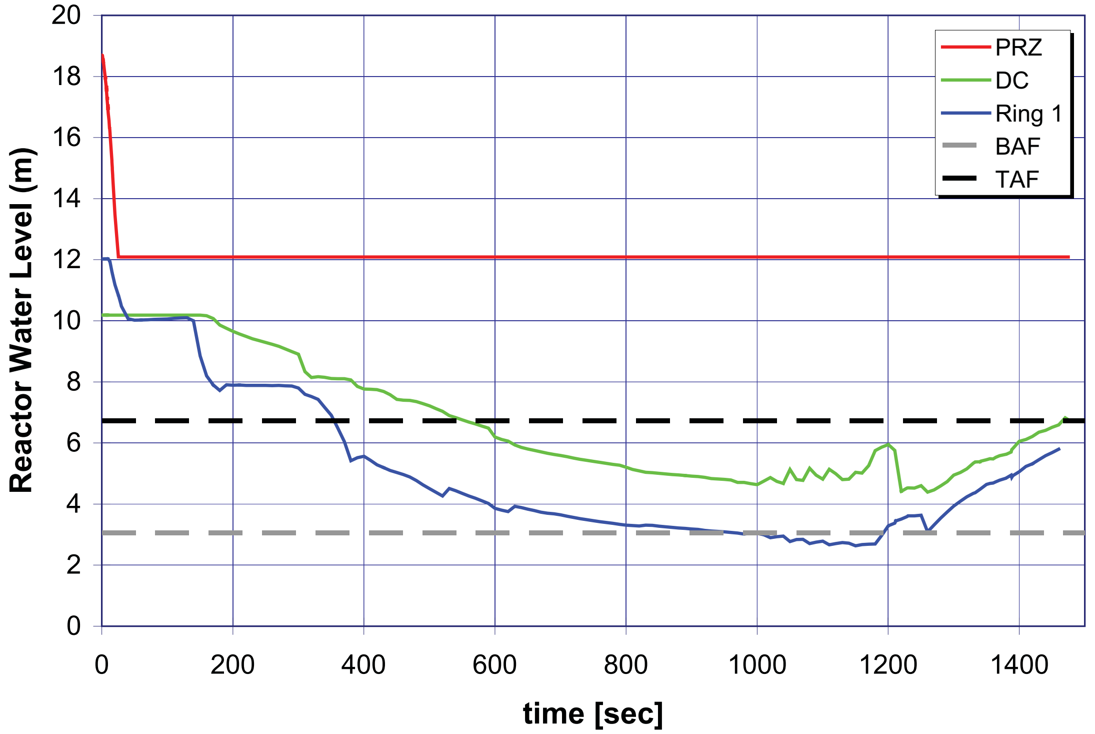




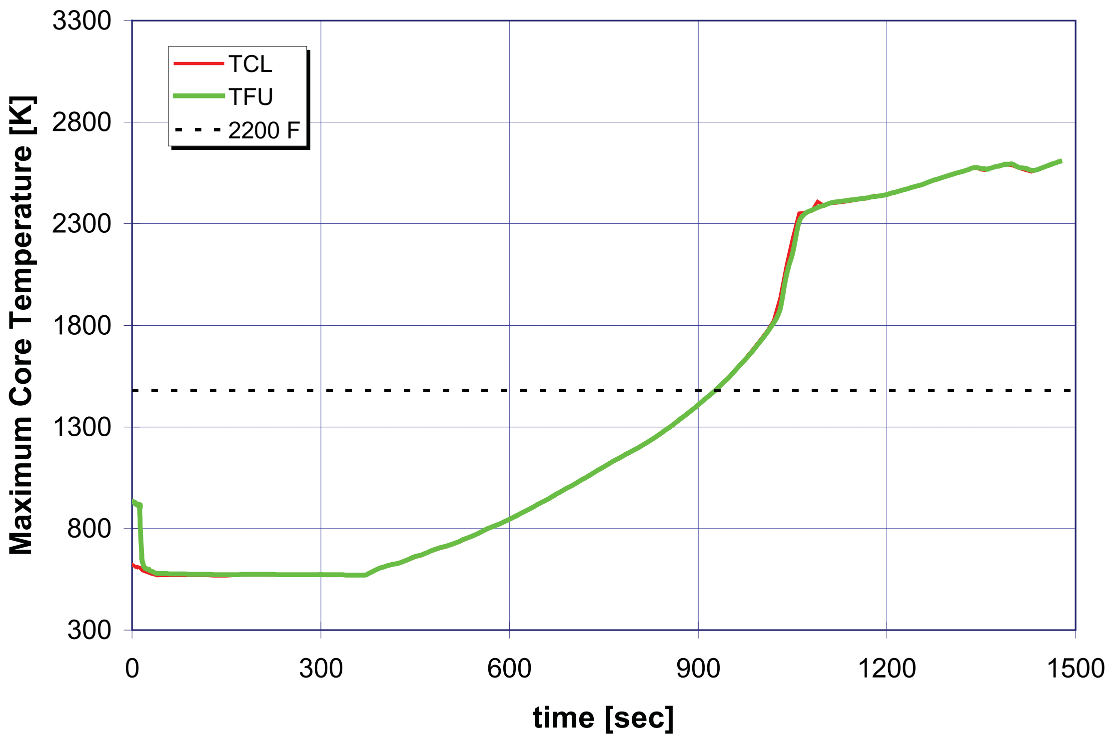
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
1000
900
800
700
600
500
400
300
200
100
0
0
300
600
900
1200
1500



RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
300
600
900
1200
1500

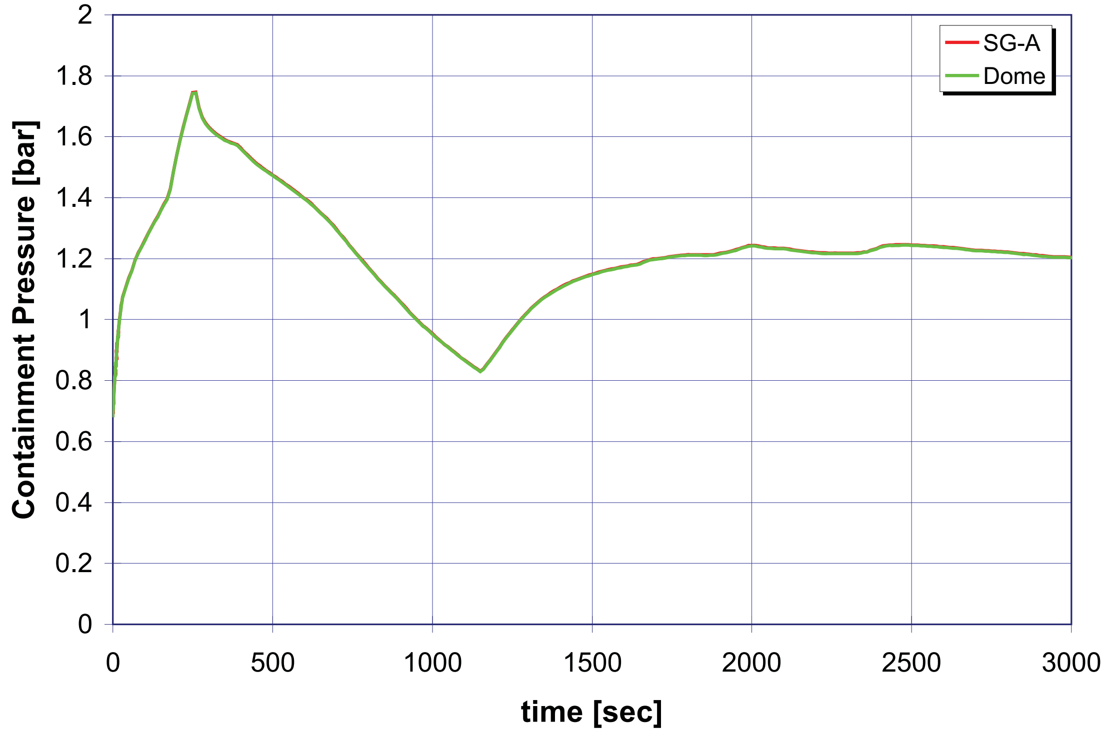


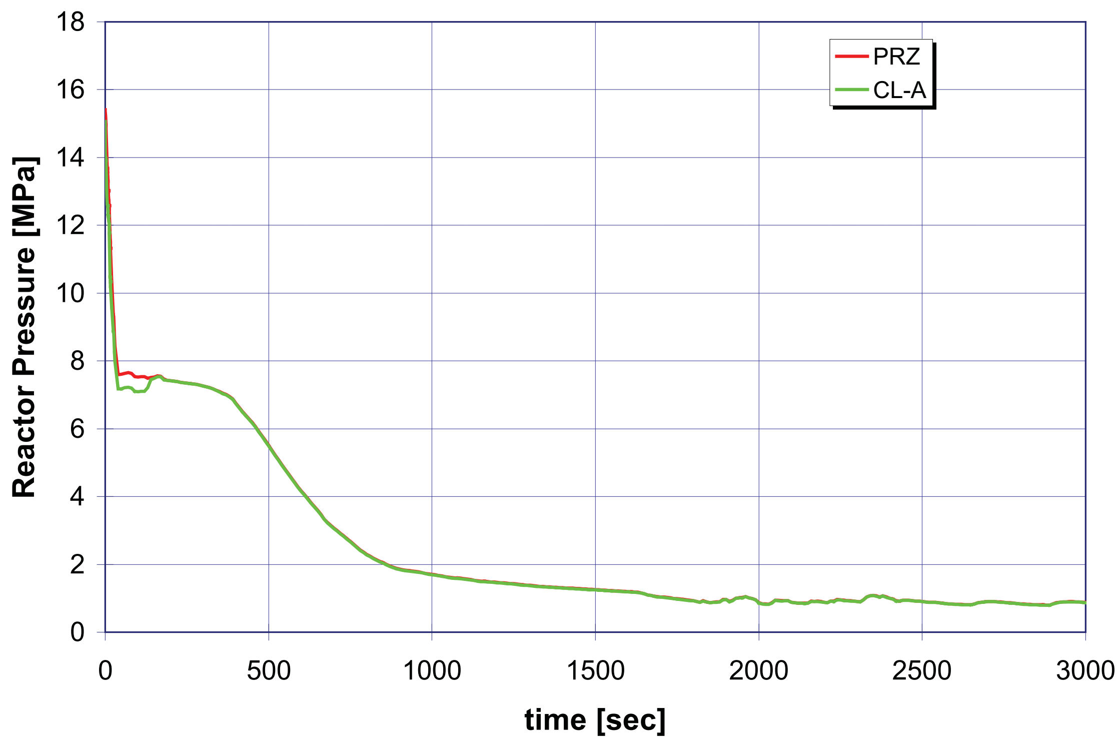
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
200
400
600
800
1000
1200
1400

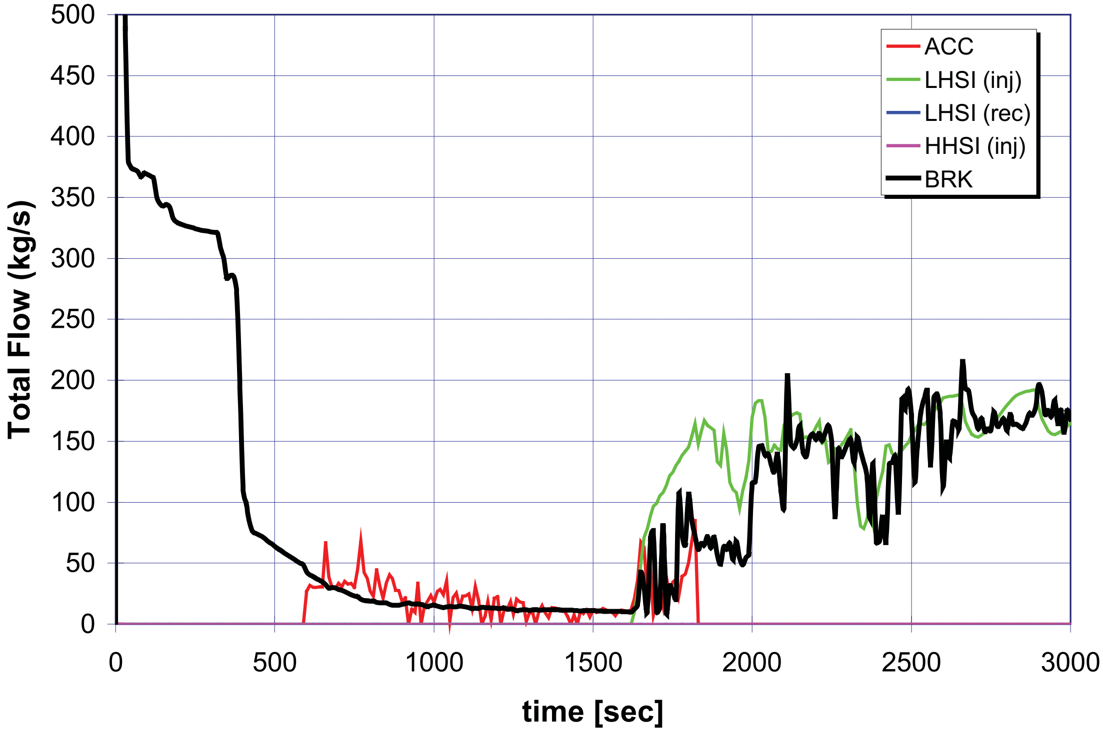


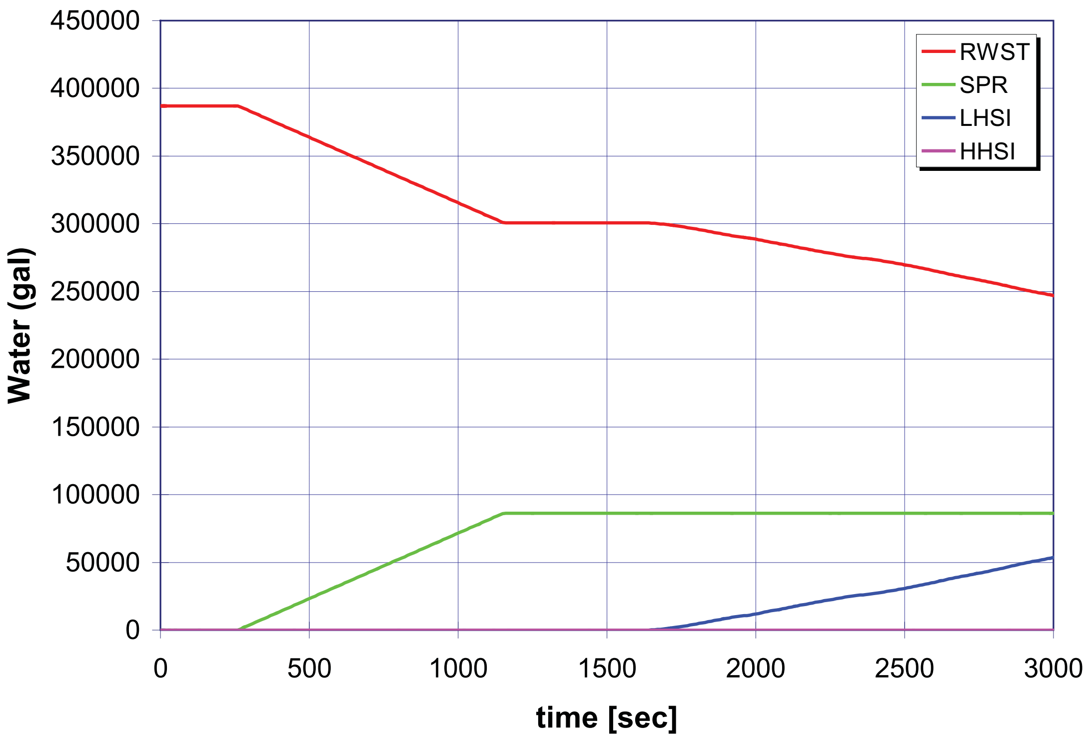
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
300
800
1300
1800
2300
2800
3300
0
300
600
900
1200
1500

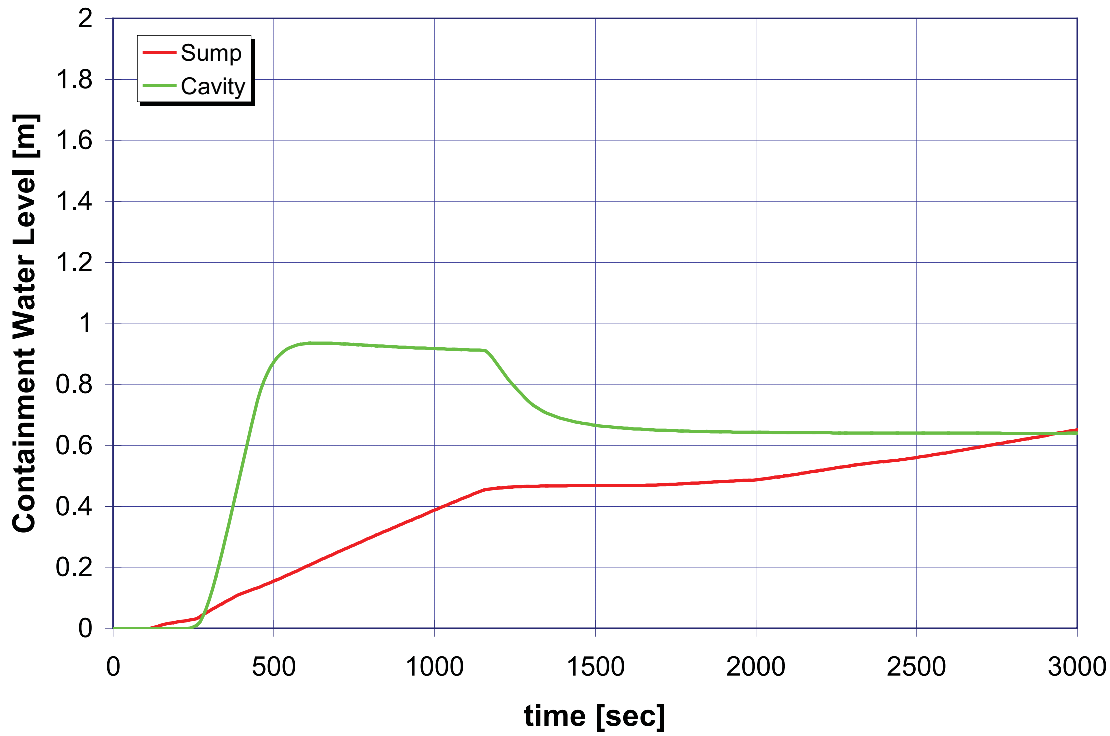
A.6.13 Case 13: 4-Inch Break LOCA, No HHSI, One LHSI, and One ACC



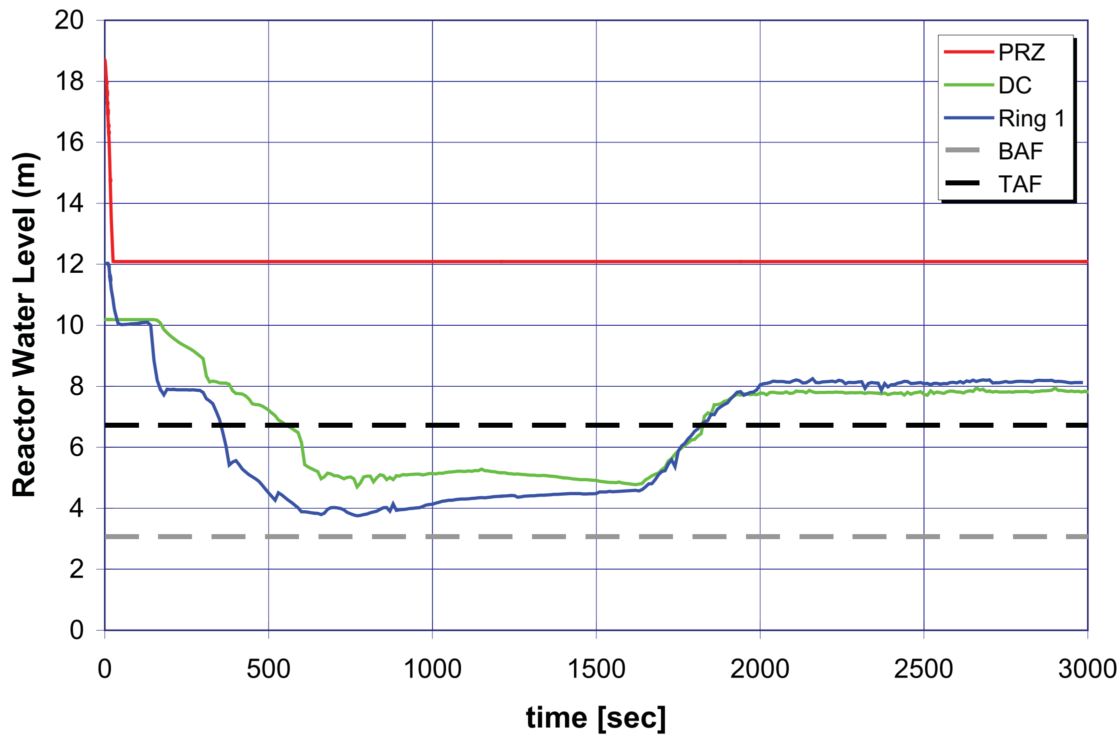



ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
0
50
100
150
200
250
300
350
400
450
500
0
500
1000
1500
2000
2500
3000

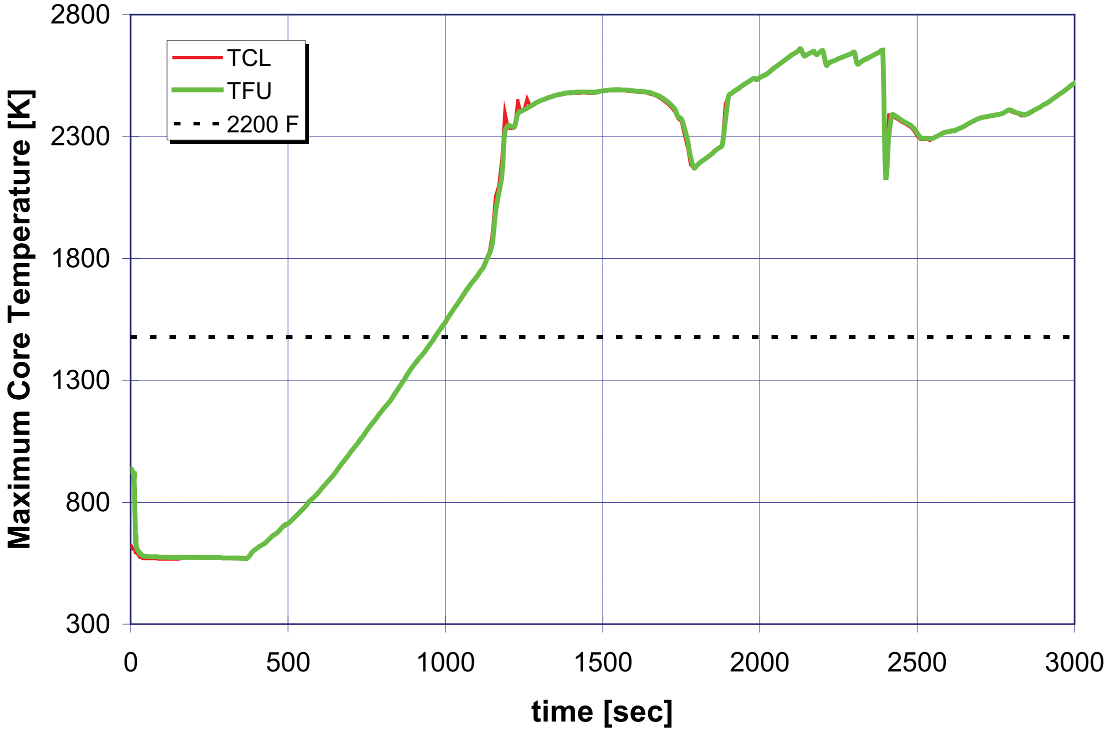

RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
500
1000
1500
2000
2500
3000



Sump
Cavity
Containment Water Level [m]
time [sec]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
500
1000
1500
2000
2500
3000

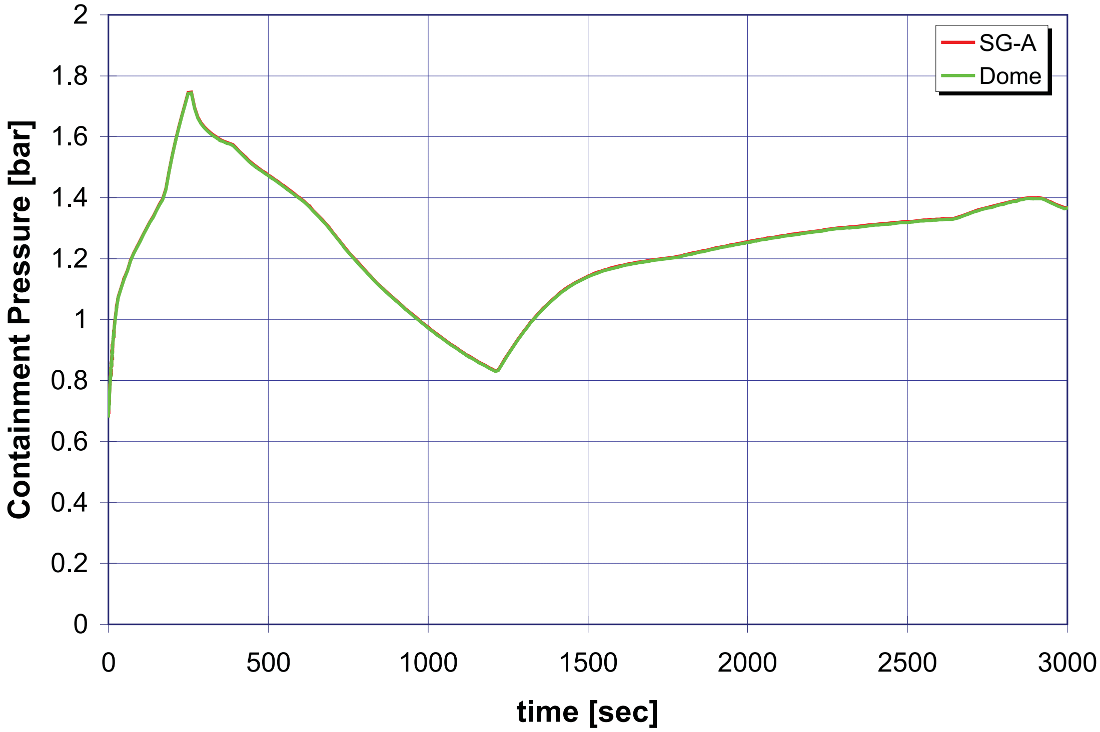


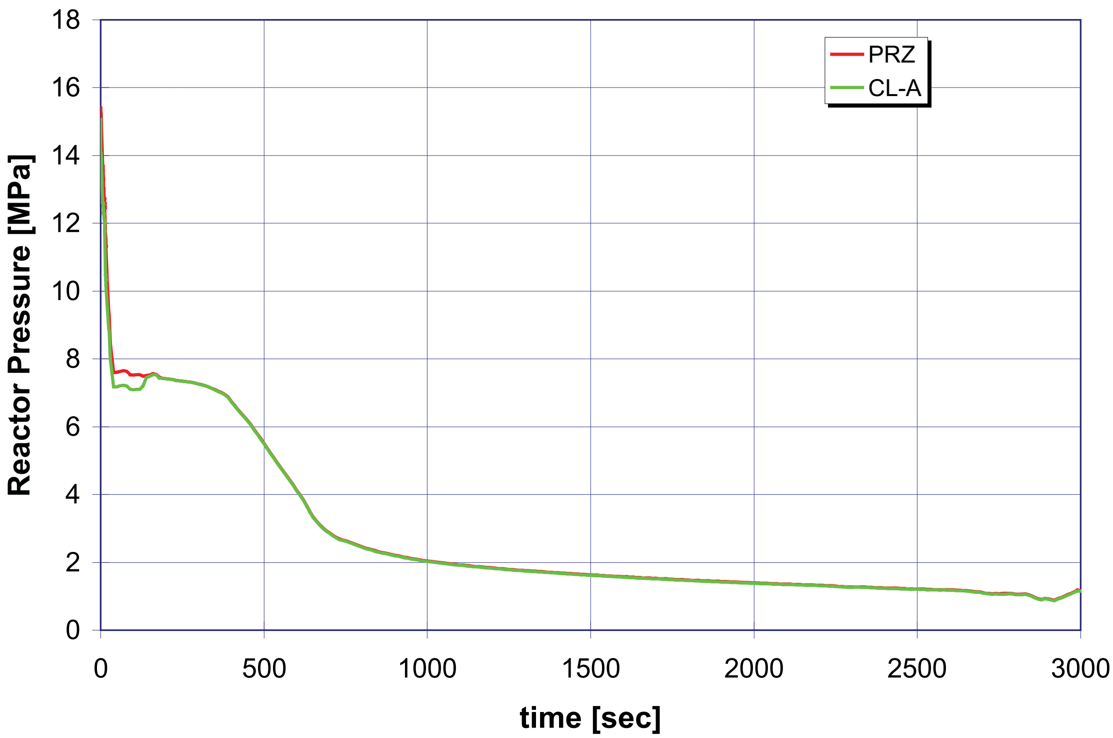
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
20
18
16
14
12
10
8
6
4
2
0
500
1000
1500
2000
2500
3000

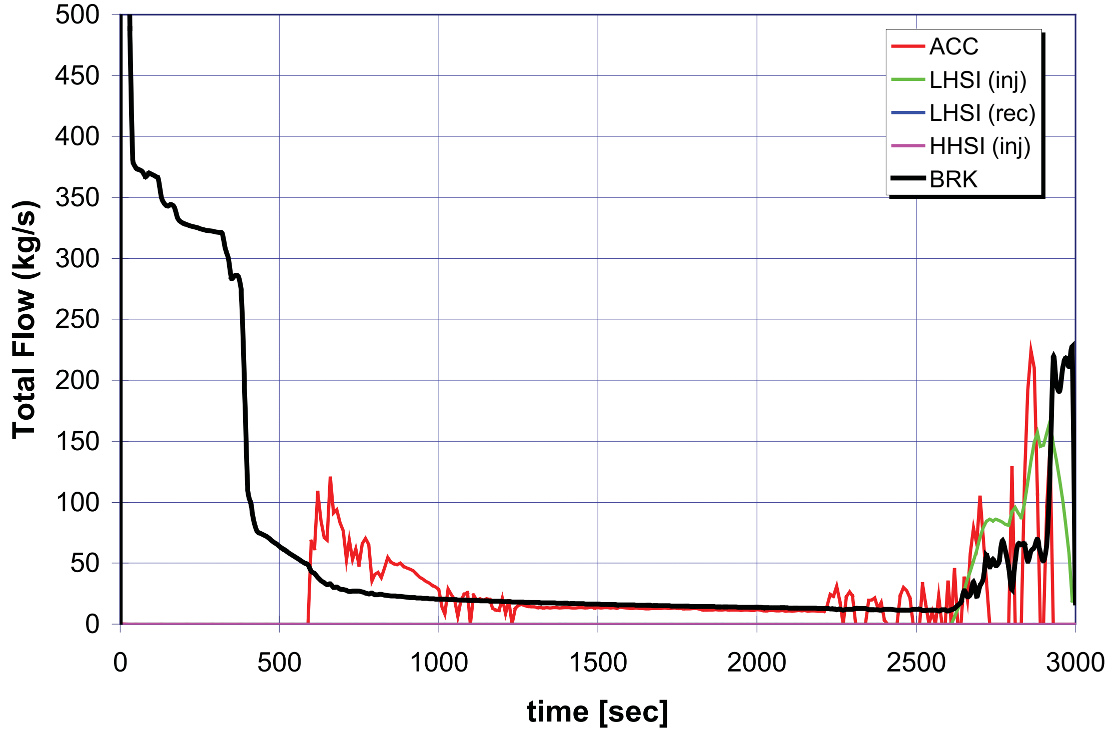


2800
2300
1800
1300
800
300
Maximum Core Temperature [K]
TCL
TFU
2200 F
0
500
1000
1500
2000
2500
3000
time [sec]

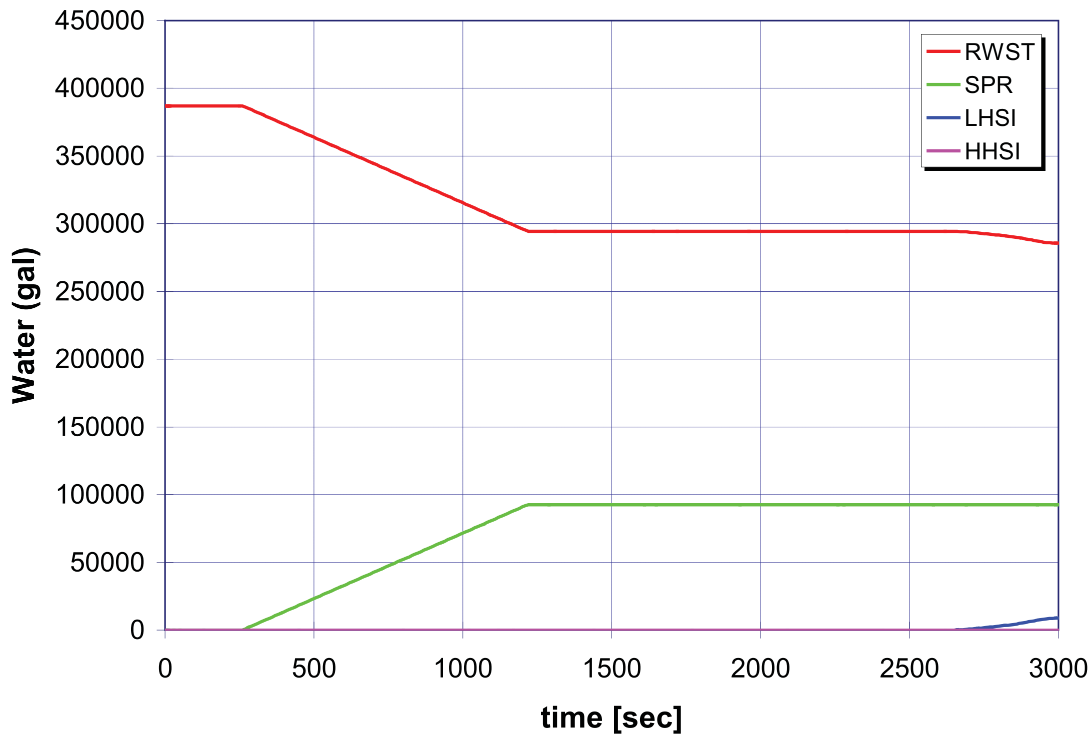
A.6.14 Case 14: 4-Inch Break LOCA, No HHSI, One LHSI, and Two ACC



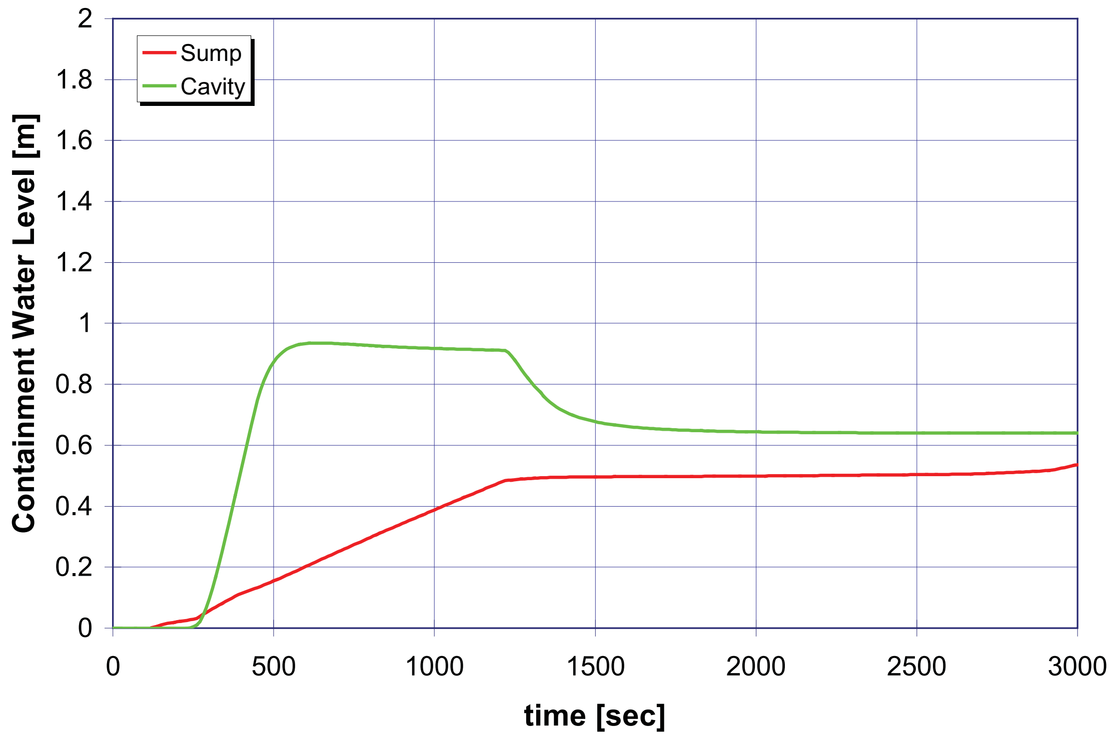


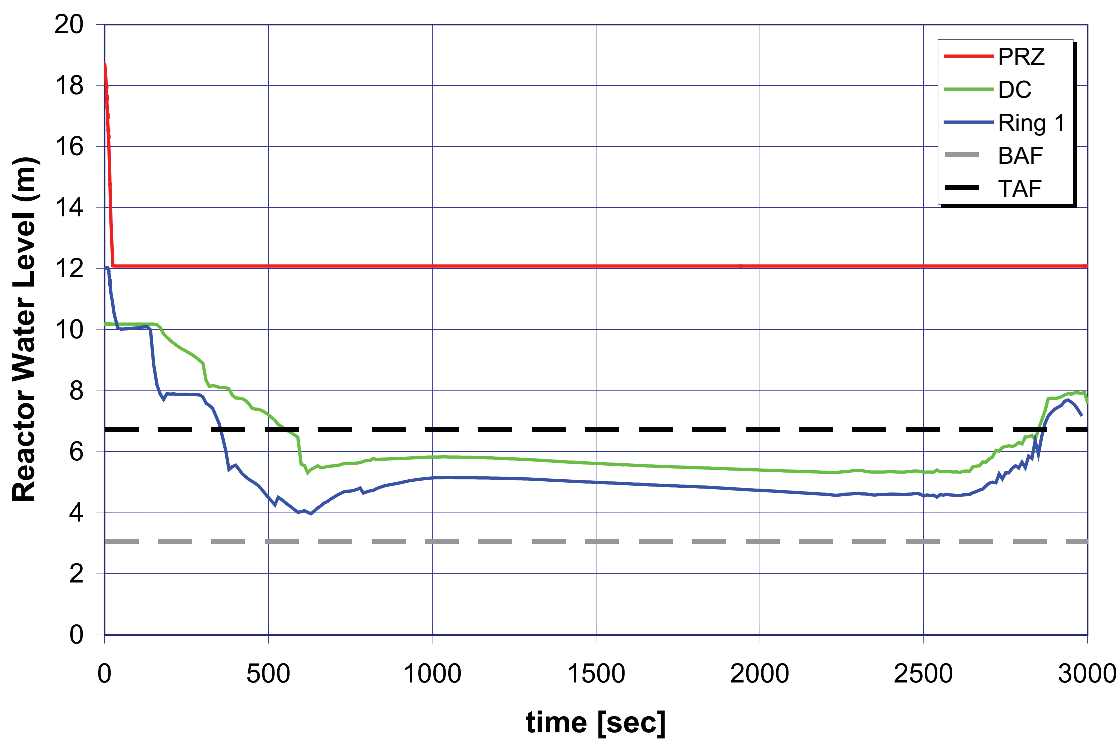


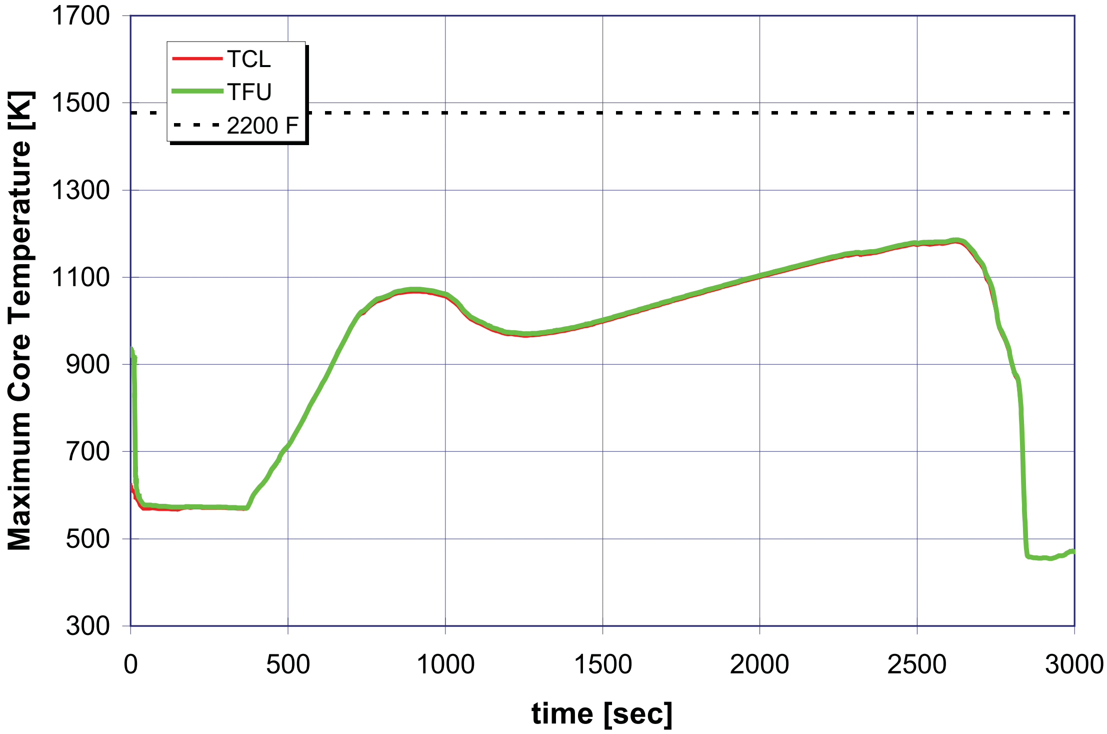
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
0
50
100
150
200
250
300
350
400
450
500
0
500
1000
1500
2000
2500
3000



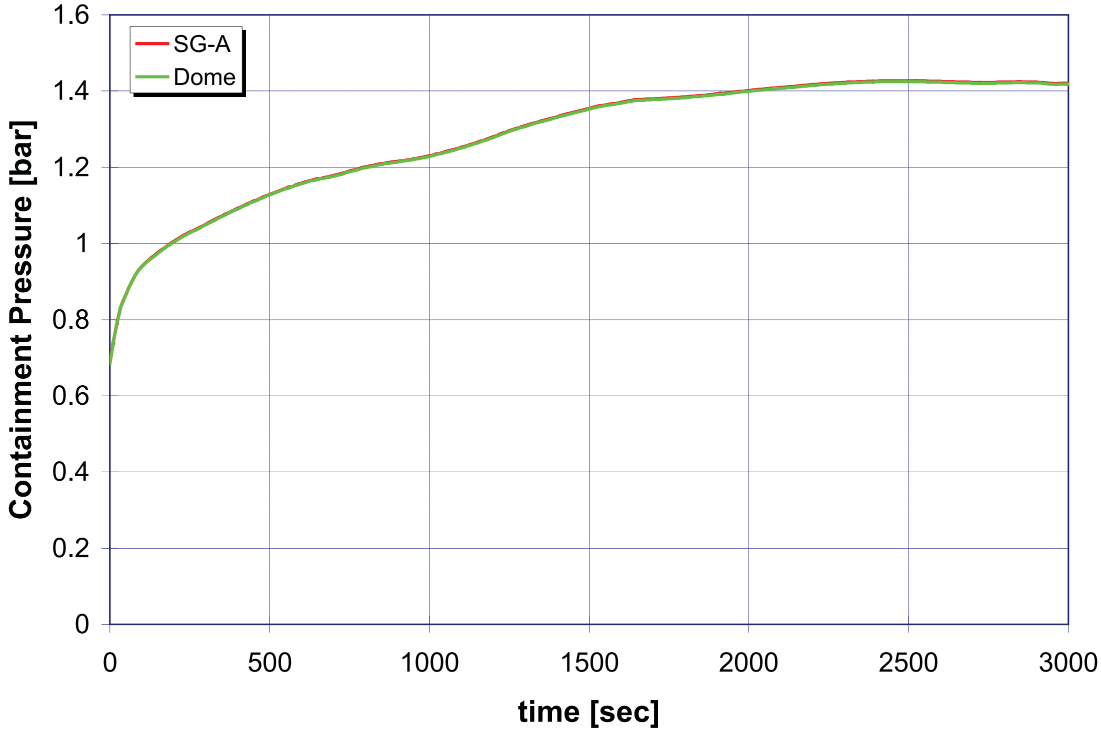
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
500
1000
1500
2000
2500
3000

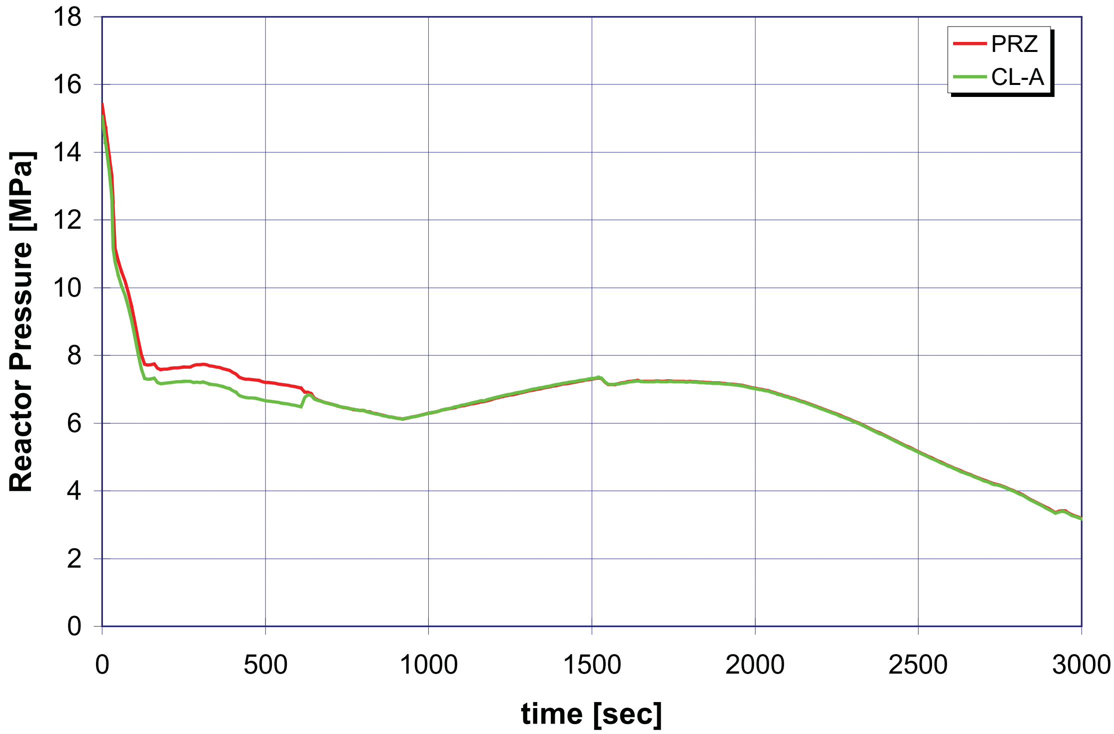

Sump
Cavity
Containment Water Level [m]
time [sec]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
500
1000
1500
2000
2500
3000

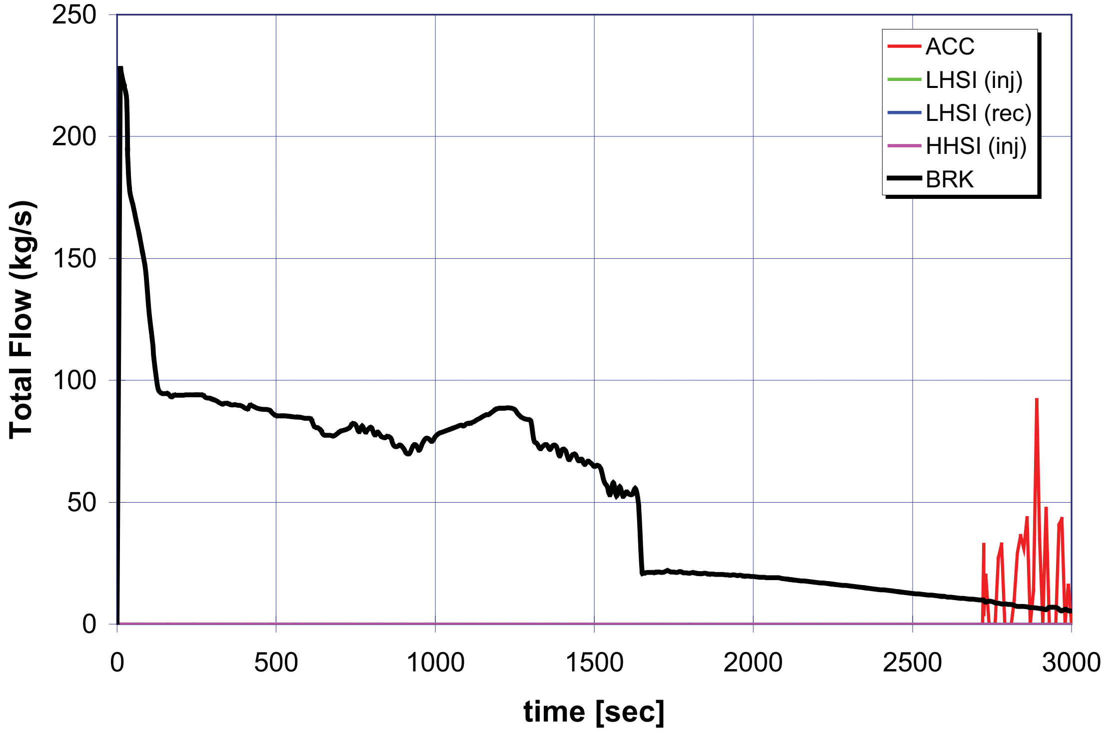

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
20
18
16
14
12
10
8
6
4
2
0
500
1000
1500
2000
2500
3000

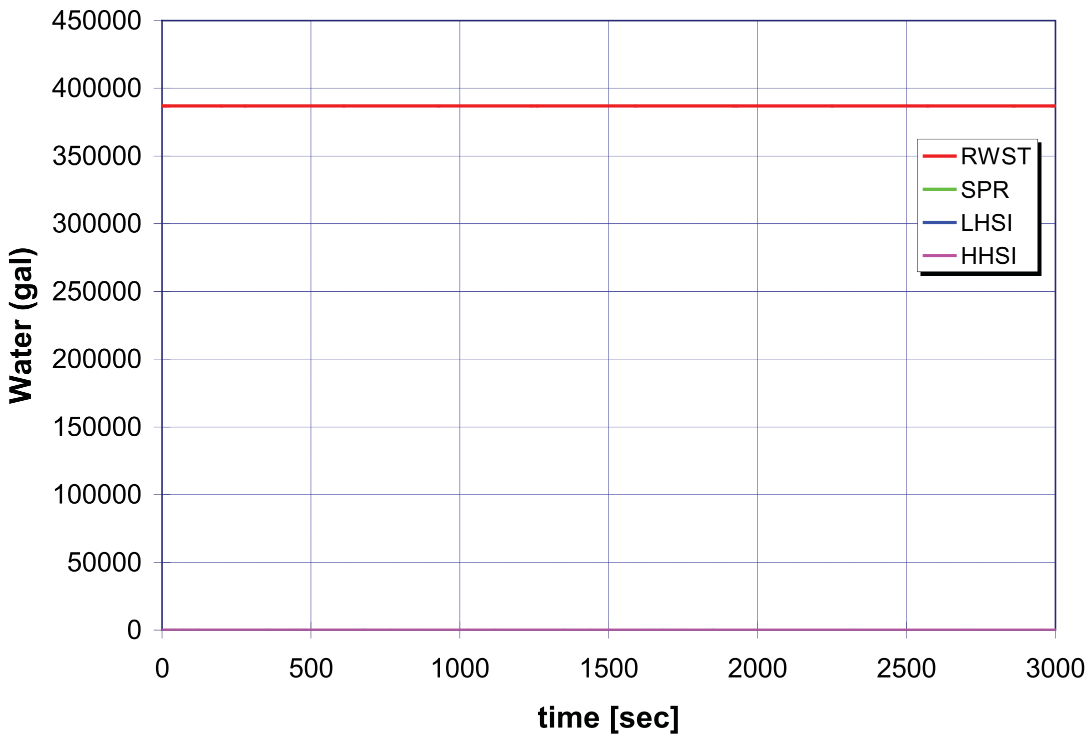

Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000
time [sec]
TCL
TFU
2200 F

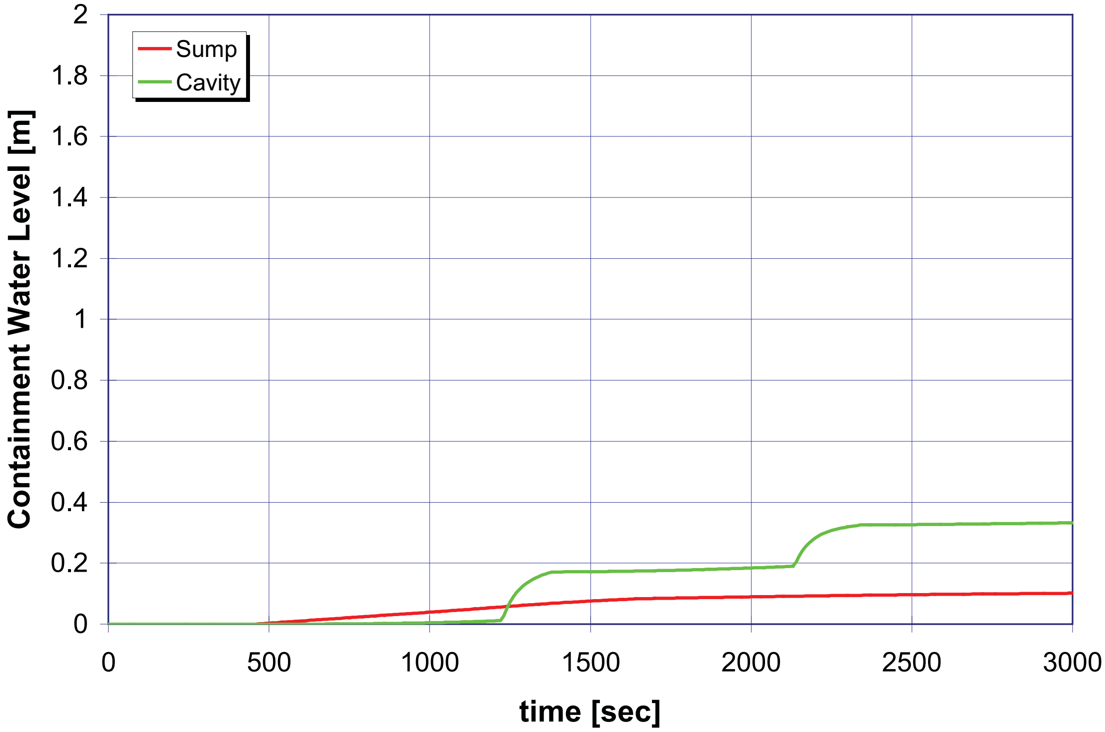
A.6.15 Case 15: 2-Inch Break LOCA, No HHSI, One LHSI, and Two ACC



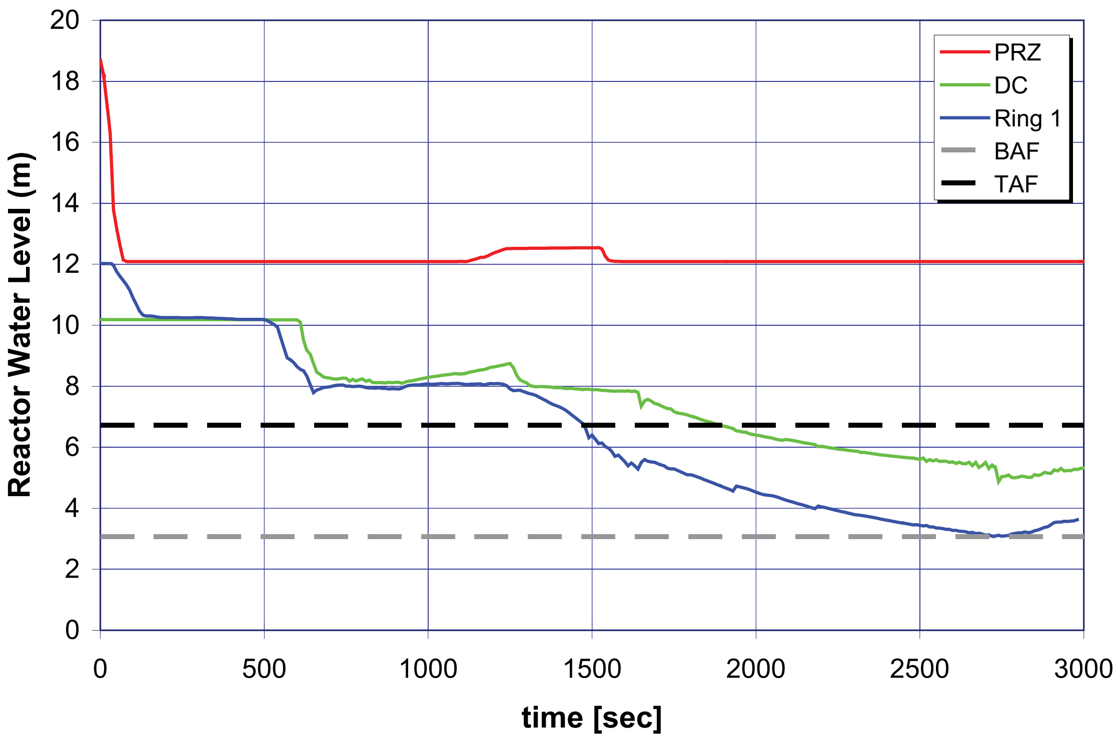



250
200
150
100
50
0
Total Flow (kg/s)
0
500
1000
1500
2000
2500
3000
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK

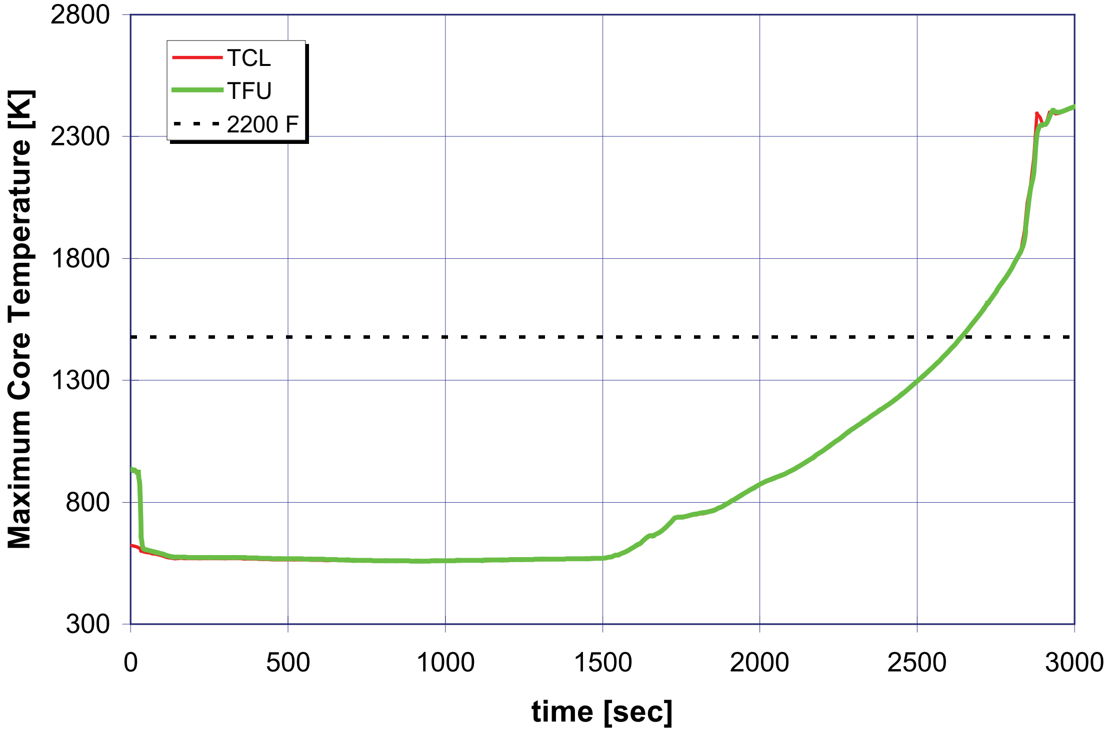

450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
500
1000
1500
2000
2500
3000
time [sec]
RWST
SPR
LHSI
HHSI



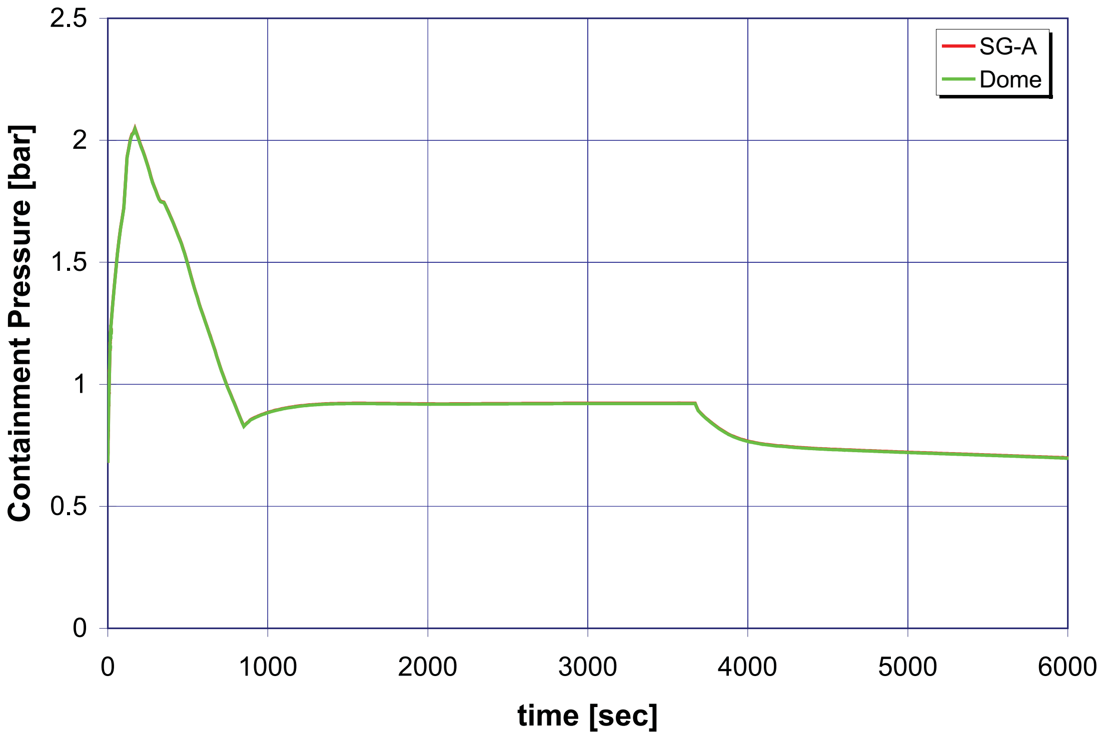
Containment Water Level [m]
Sump
Cavity
time [sec]

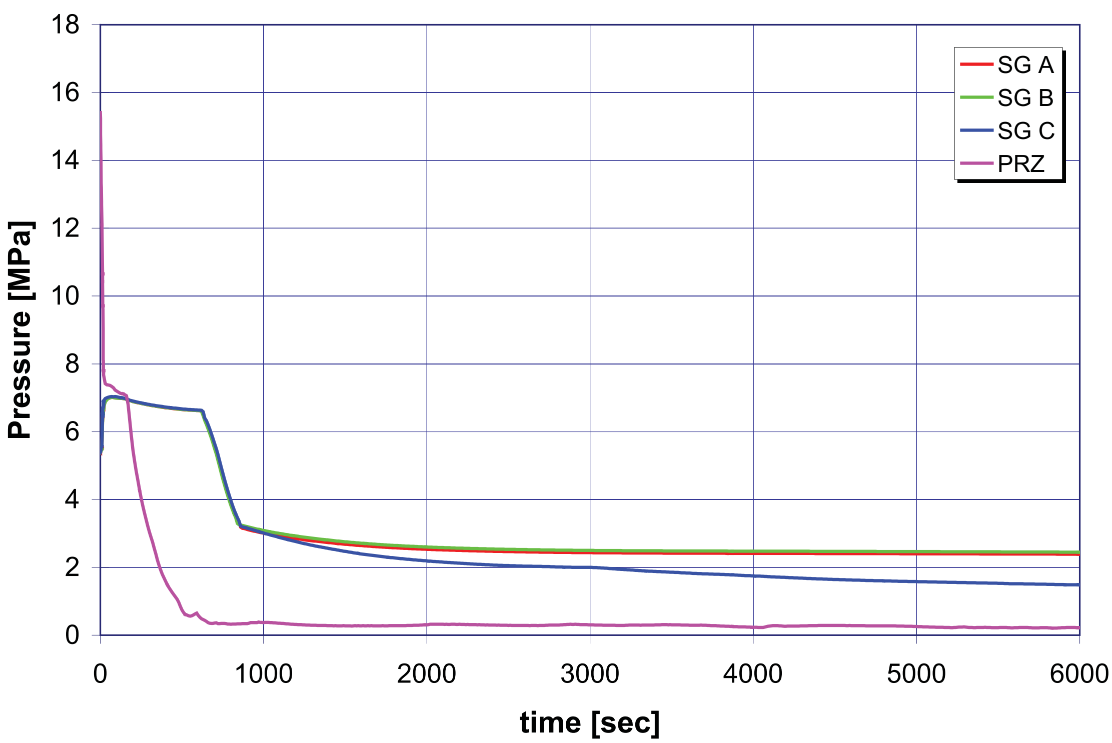


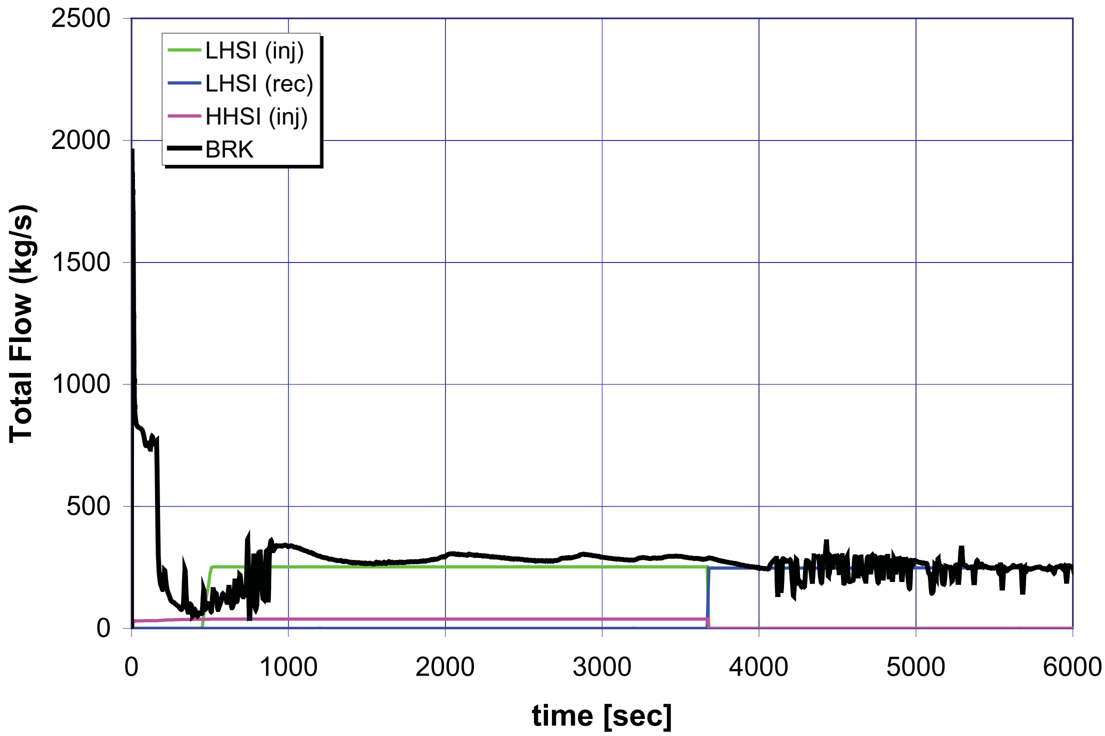
Reactor Water Level (m)
PRZ
DC
Ring 1
BAF
TAF
time [sec]


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
500
1000
1500
2000
2500
3000
time [sec]

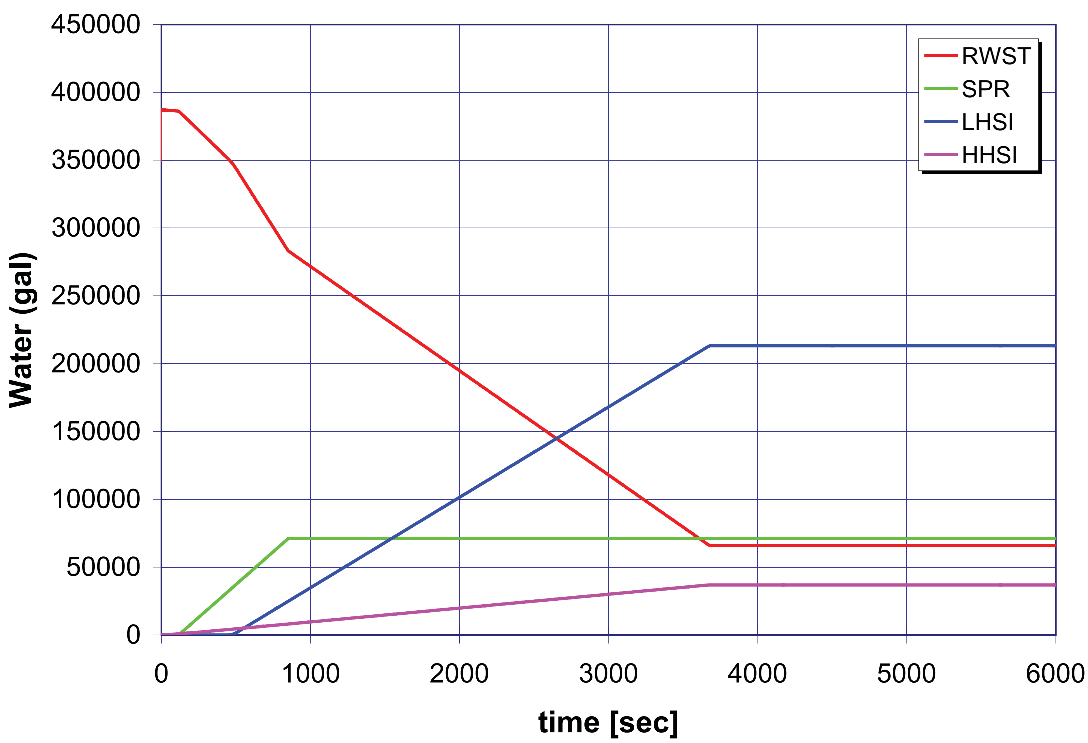
A.6.16 Case 16: 6-Inch Break LOCA, One HHSI, One LHSI, and No ACC, without Auxiliary Feedwater



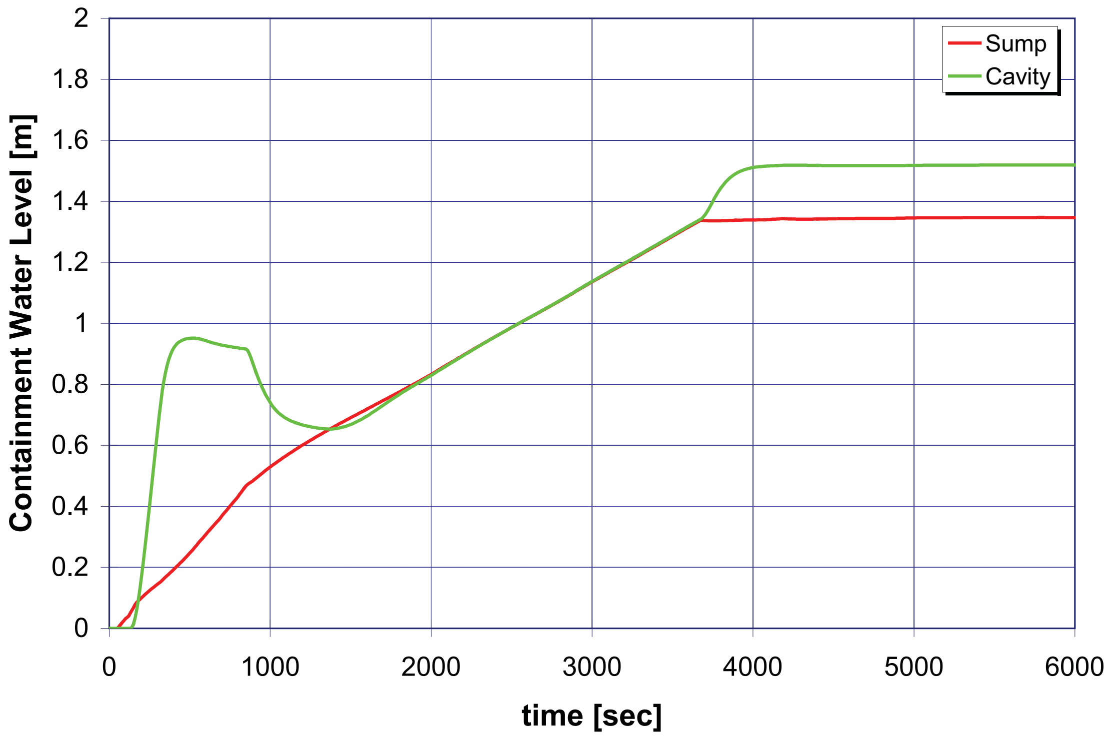


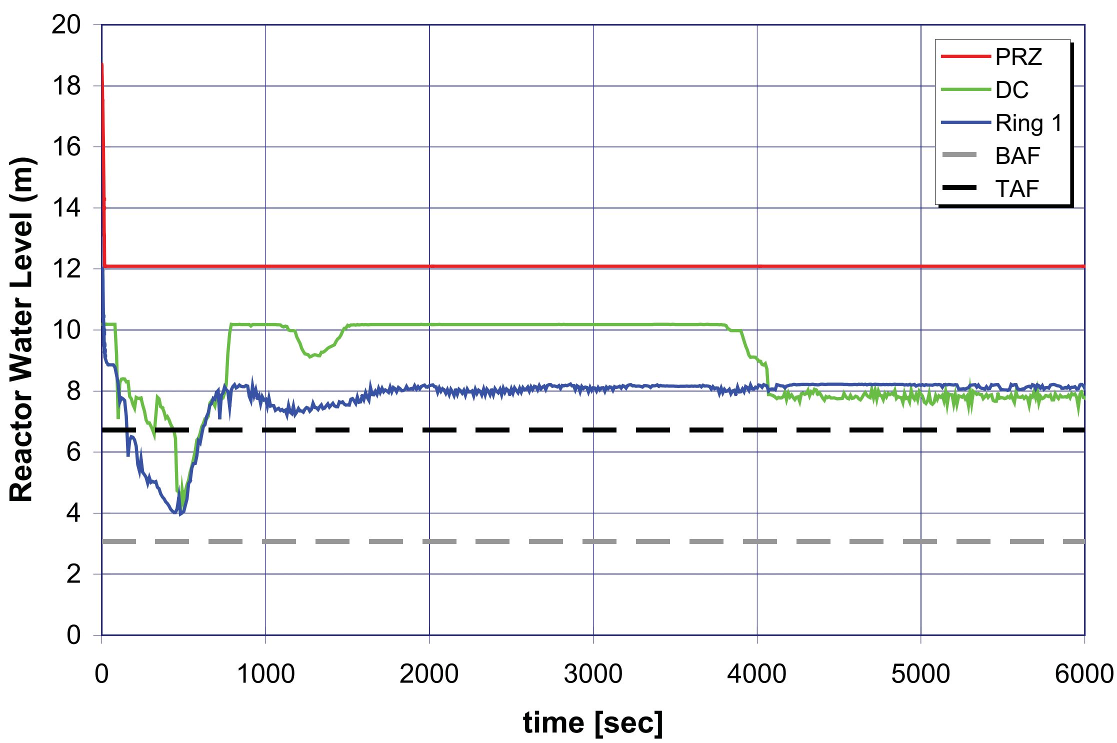


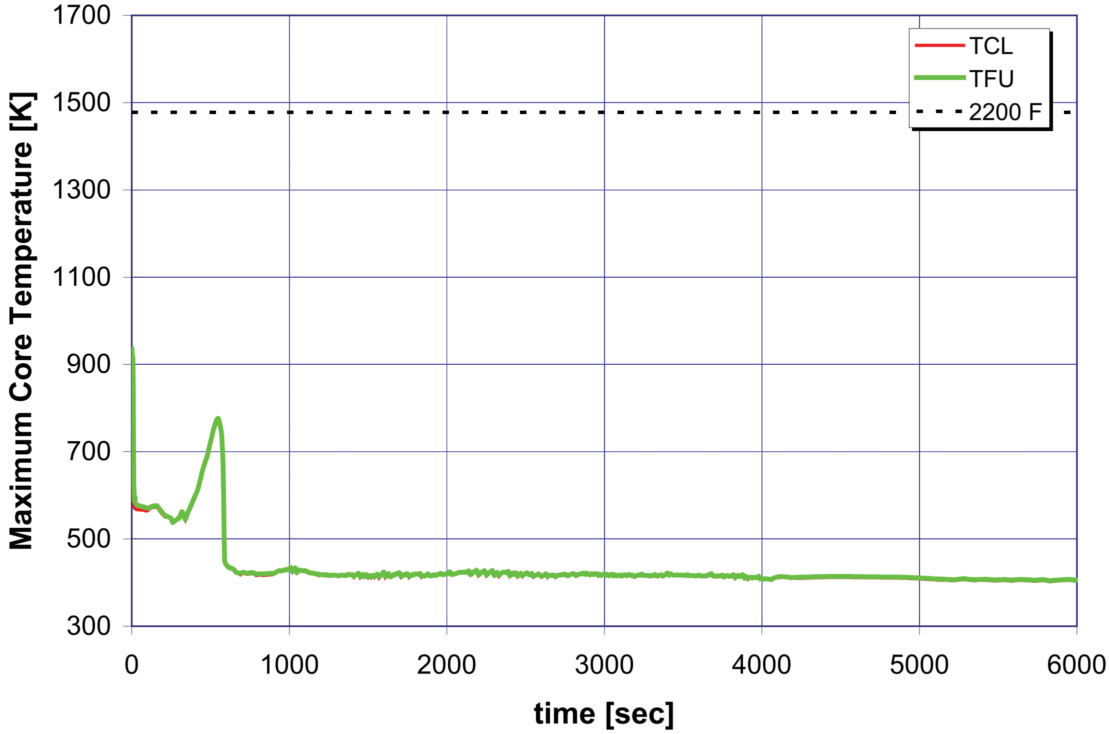
Total Flow (kg/s)
2500
2000
1500
1000
500
0
0
1000
2000
3000
4000
5000
6000
time [sec]
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
6000
time [sec]
RWST
SPR
LHSI
HHSI

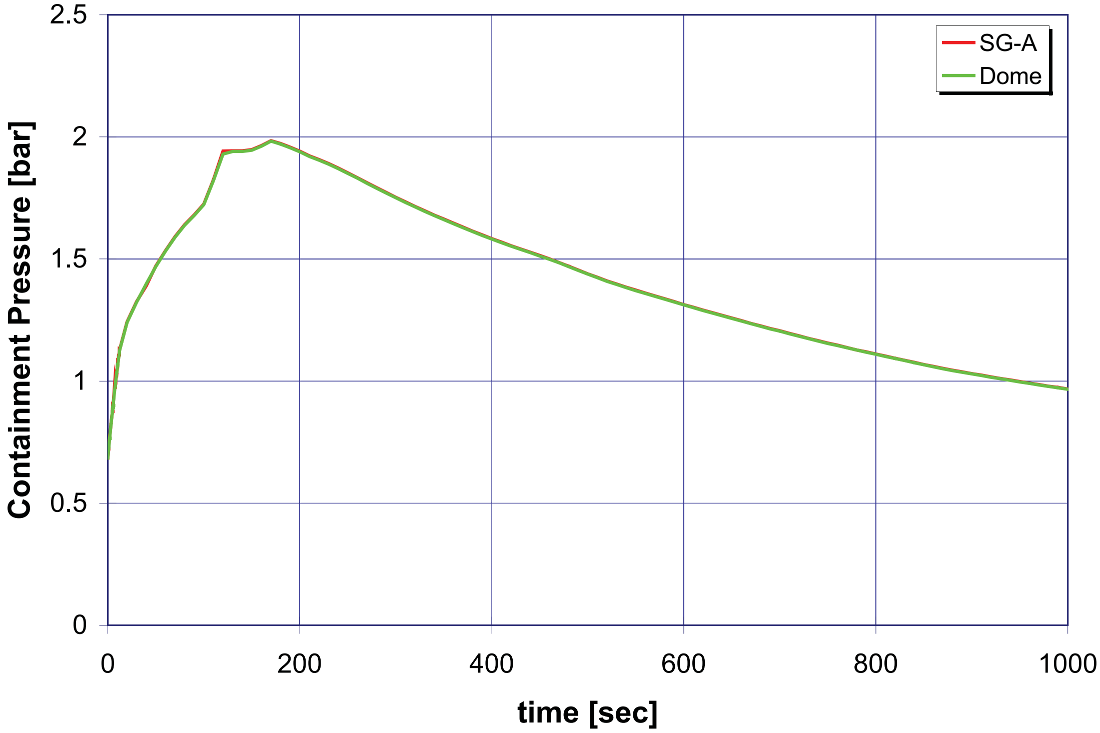

Containment Water Level [m]
time [sec]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
1000
2000
3000
4000
5000
6000

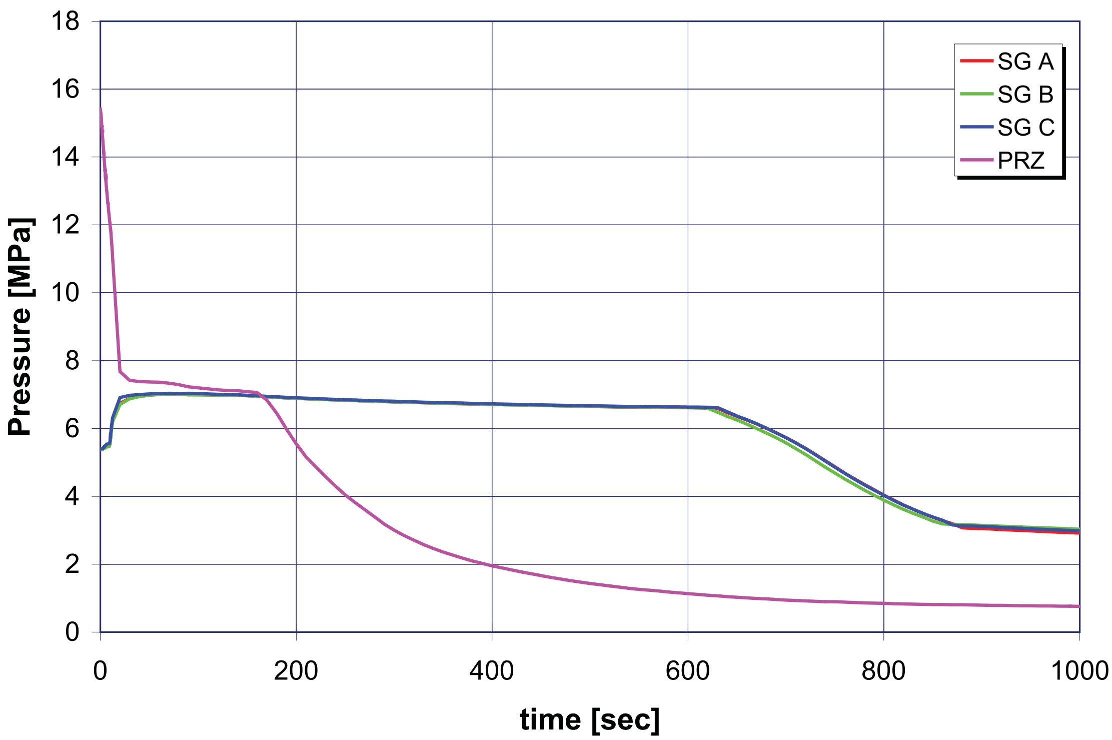

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
1000
2000
3000
4000
5000
6000

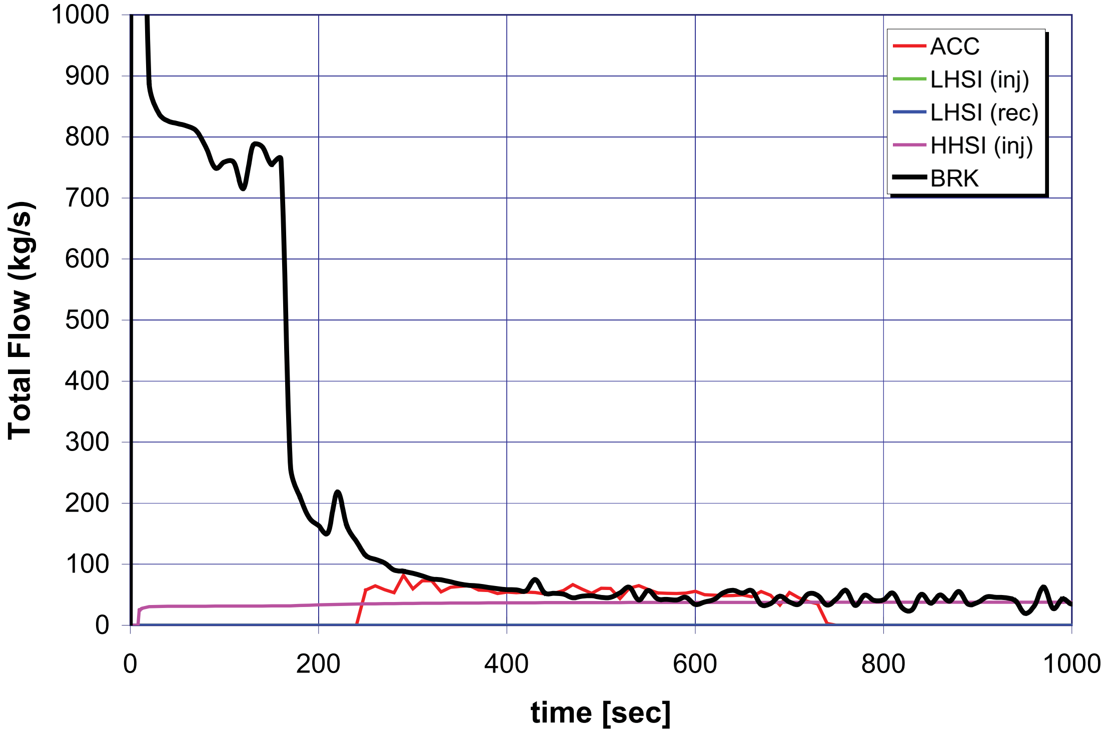


1700
1500
1300
1100
900
700
500
300
0
1000
2000
3000
4000
5000
6000
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F

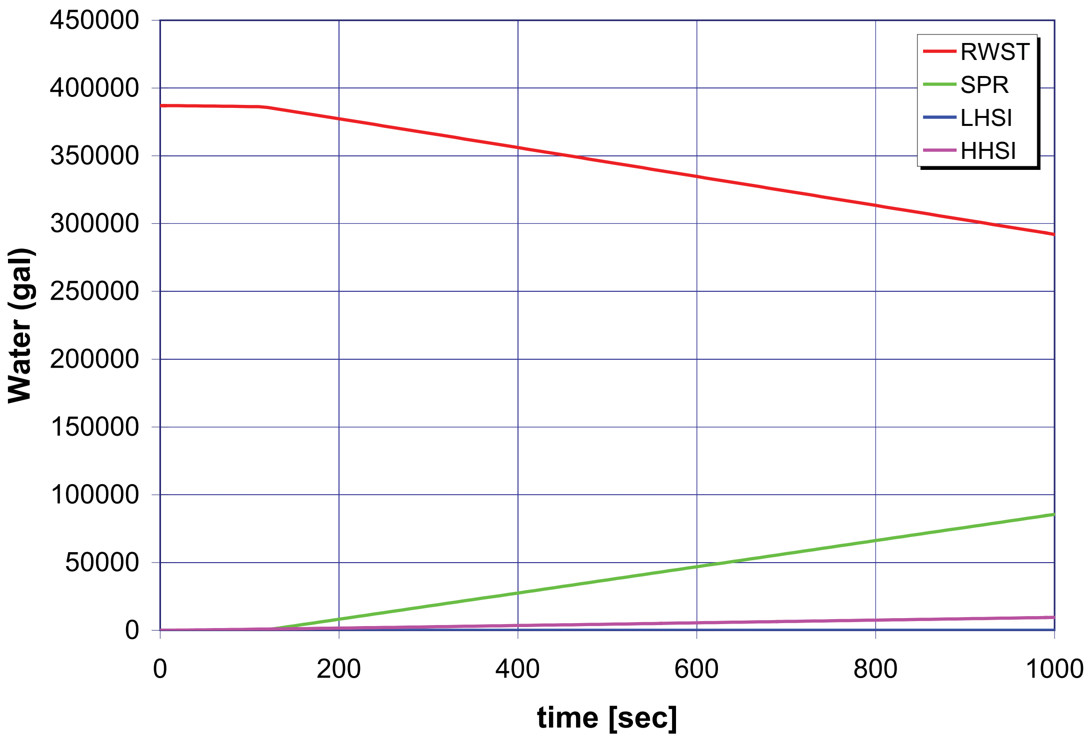
A.6.17 Case 17: 6-Inch Break LOCA, One HHSI, No LHSI, and One ACC, without Auxiliary Feedwater



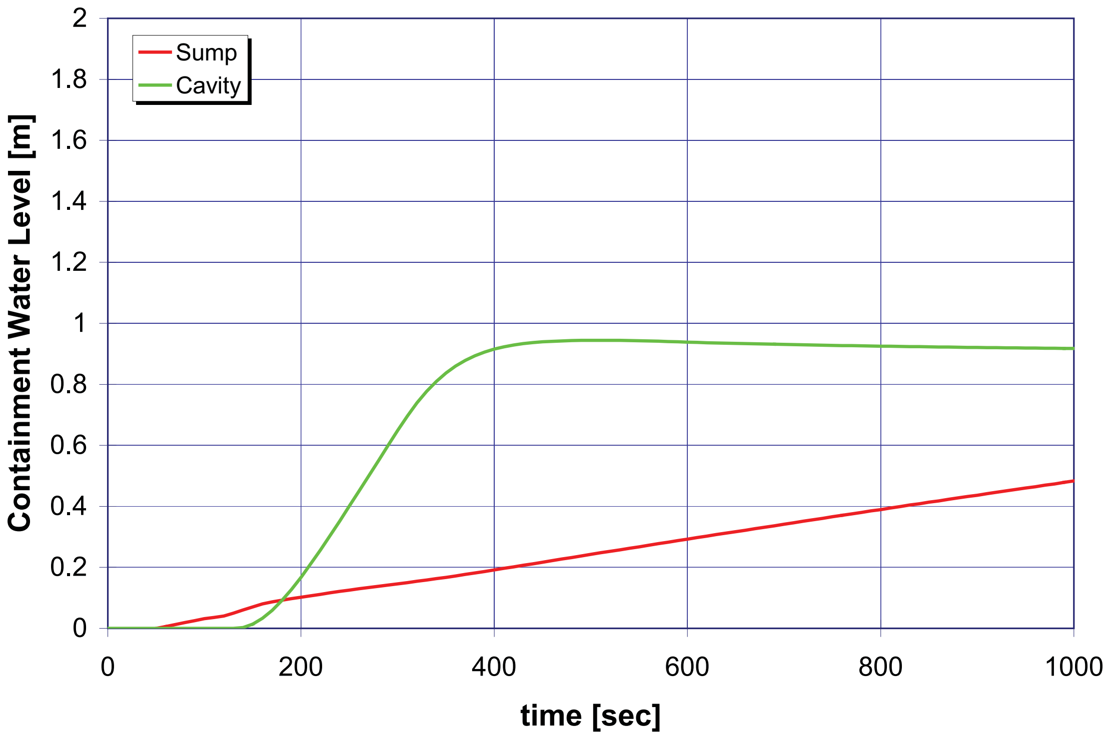




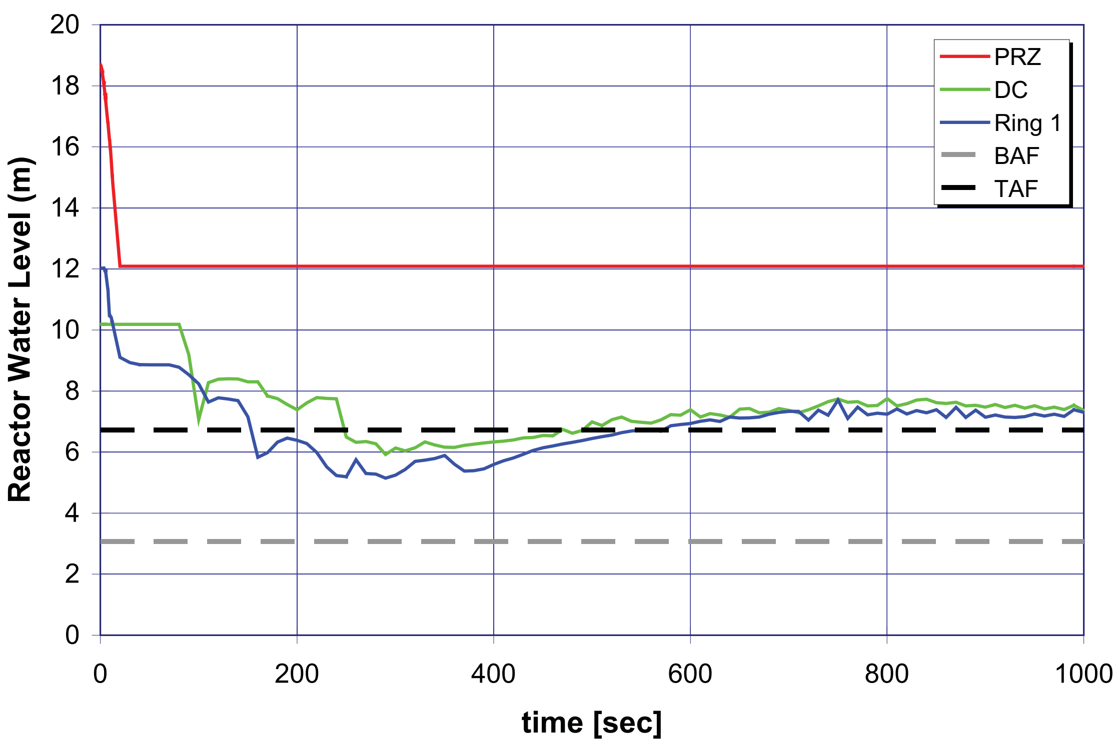
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
0
100
200
300
400
500
600
700
800
900
1000
0
200
400
600
800
1000



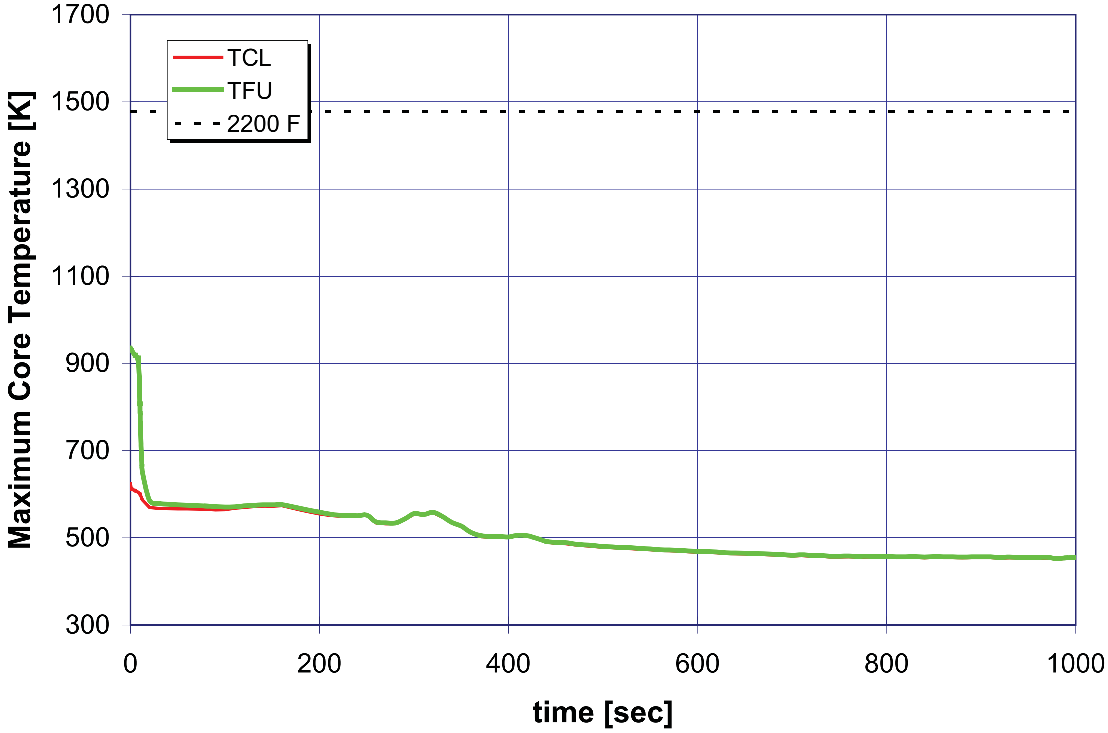
RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
200
400
600
800
1000



Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
200
400
600
800
1000

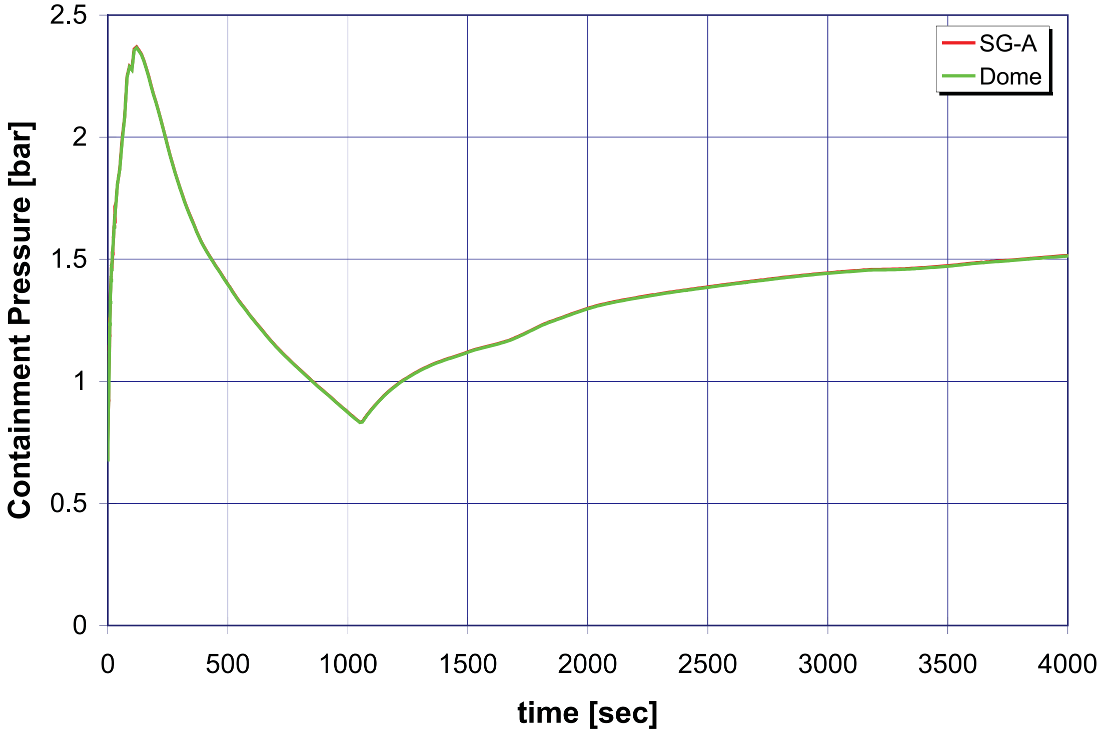


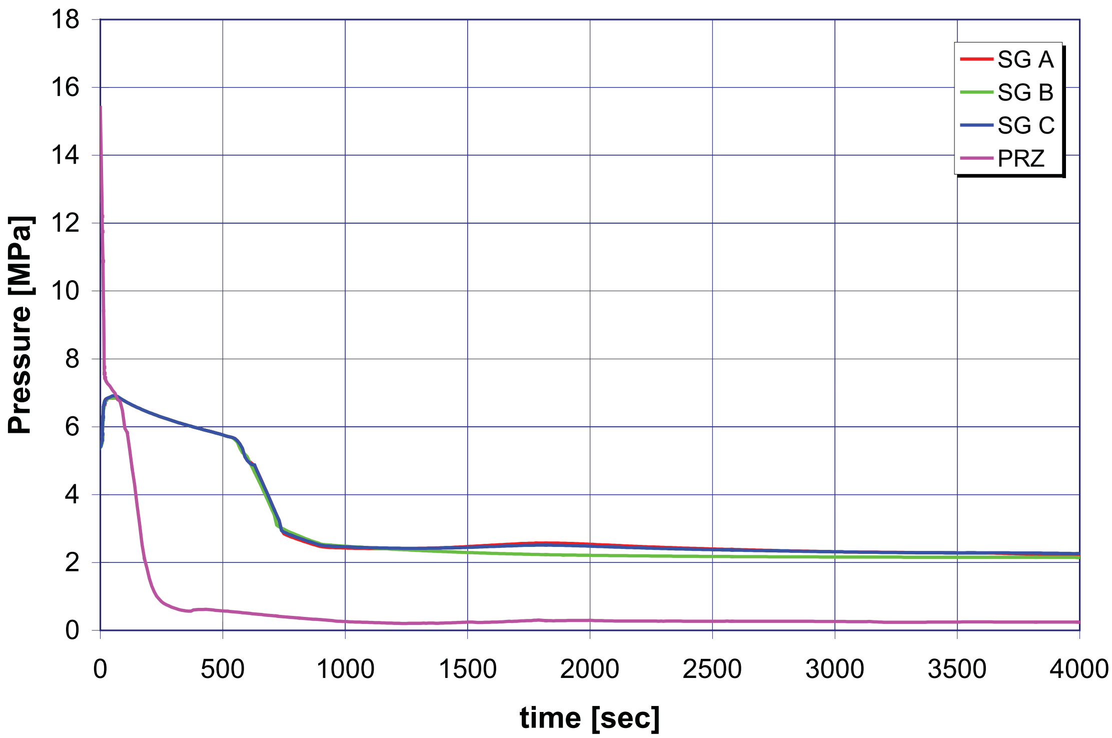
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
200
400
600
800
1000

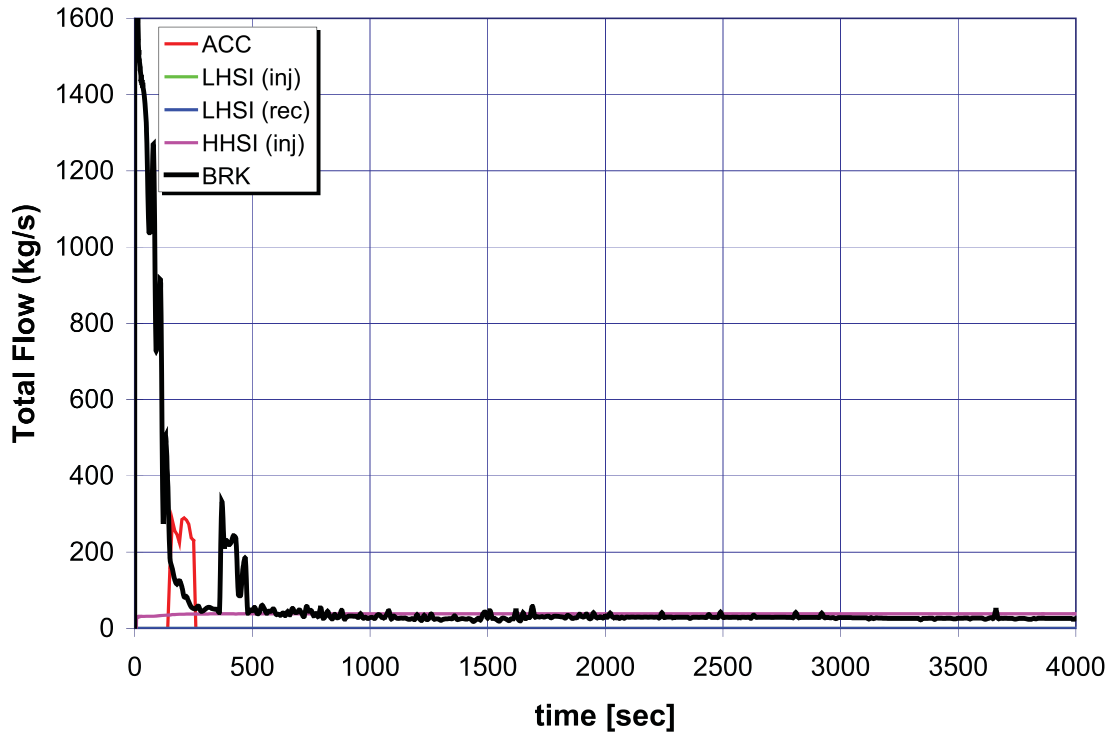


Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
200
400
600
800
1000
time [sec]
TCL
TFU
2200 F

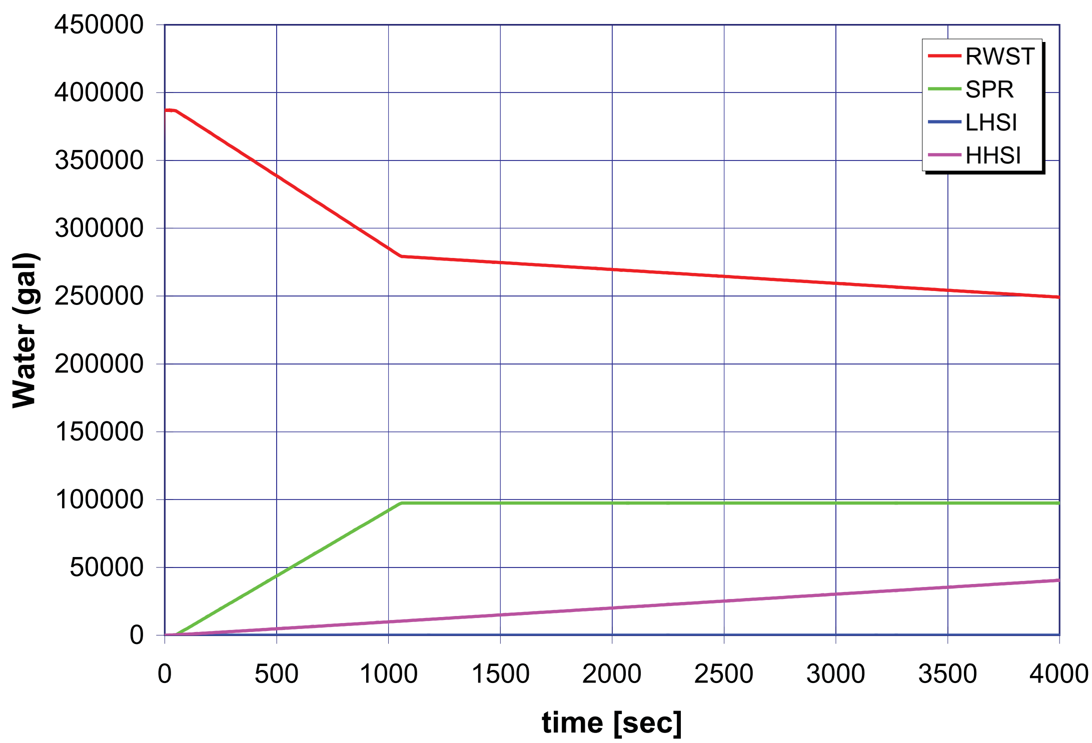
A.6.18 Case 18: 8-Inch Break LOCA, One HHSI, No LHSI, and One ACC



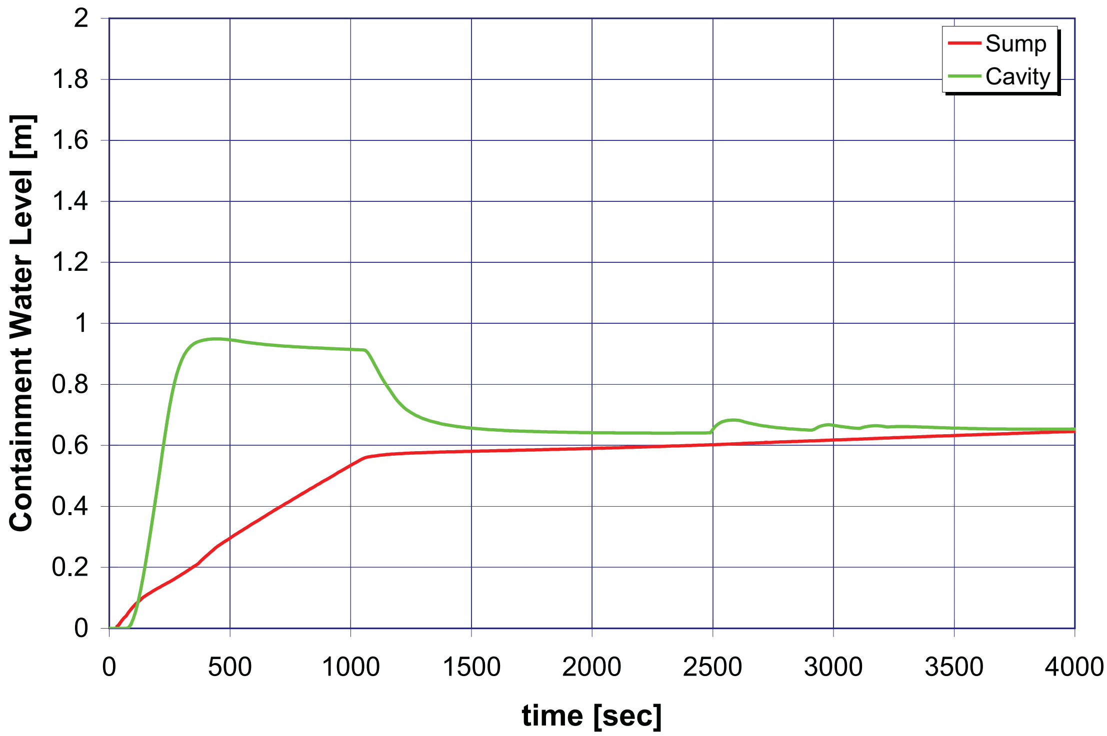




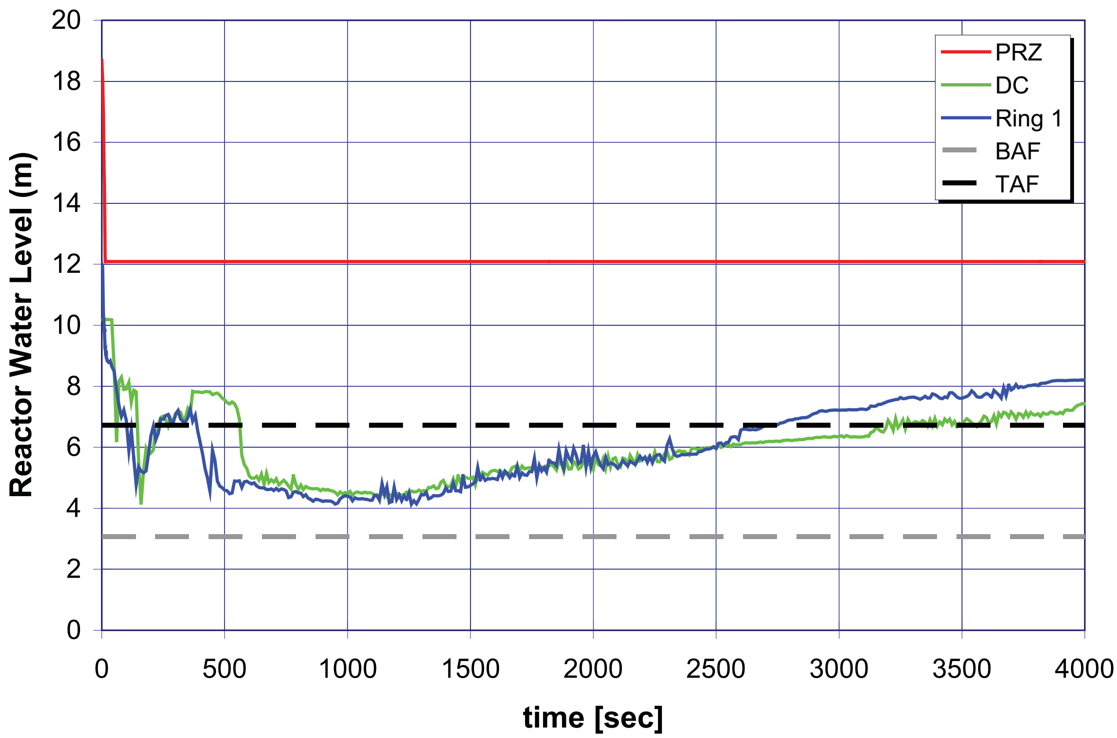
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]



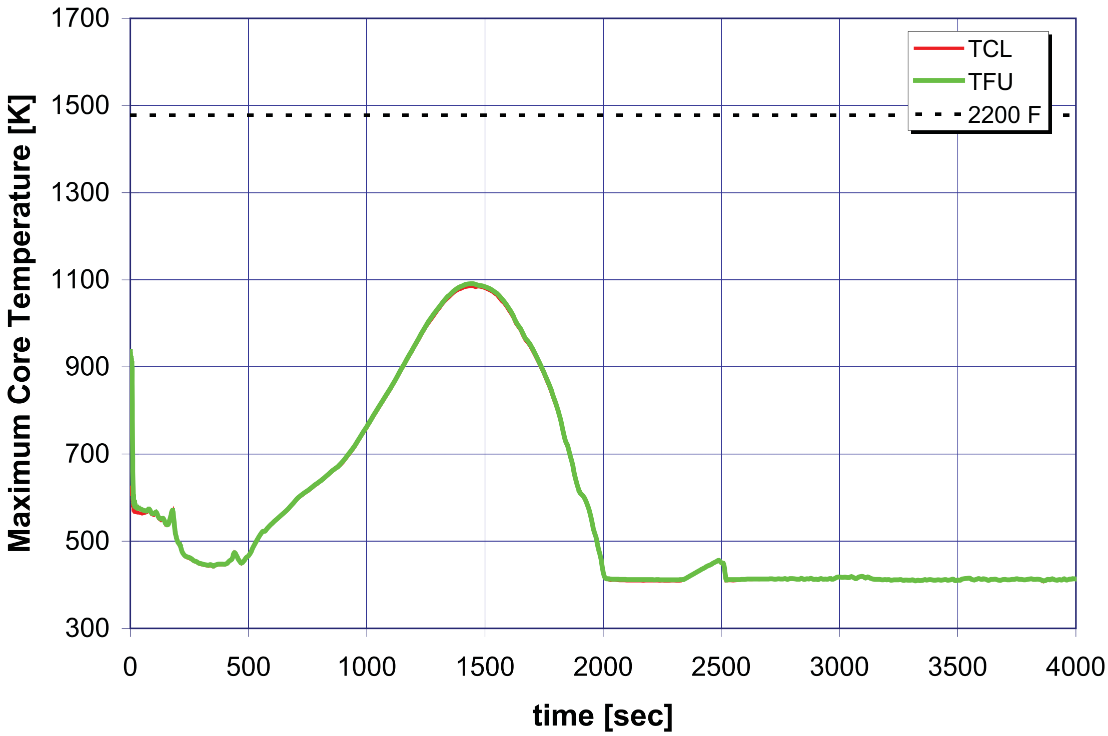
RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]



Containment Water Level [m]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
Sump
Cavity

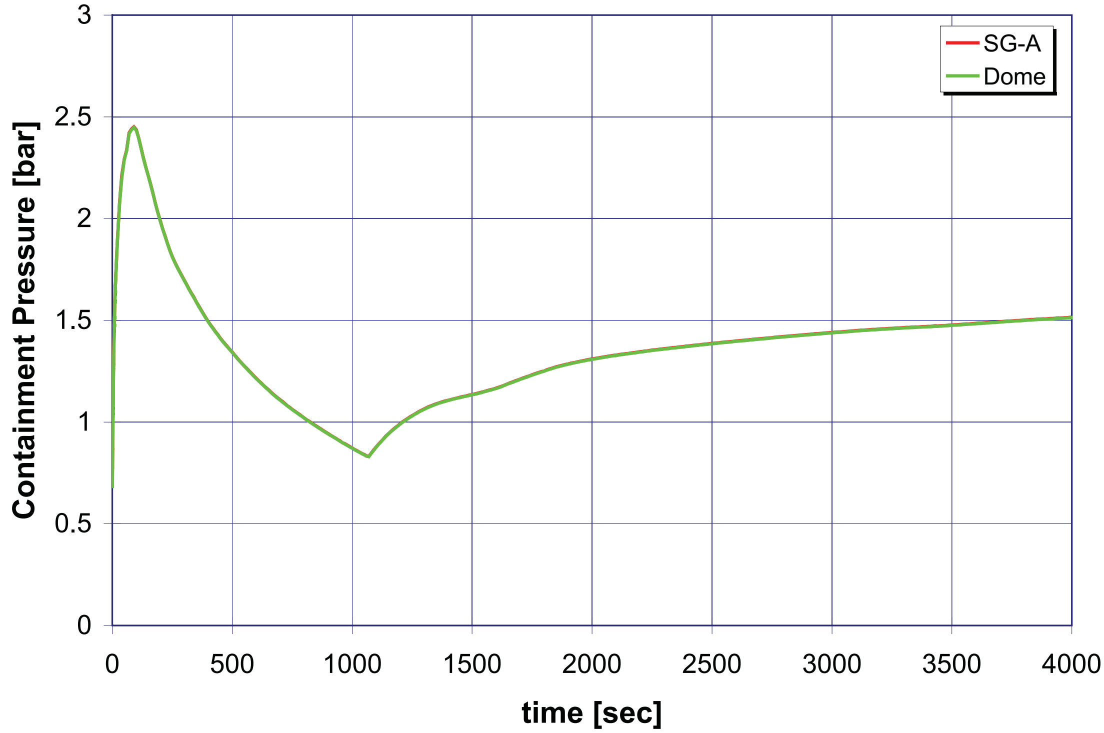


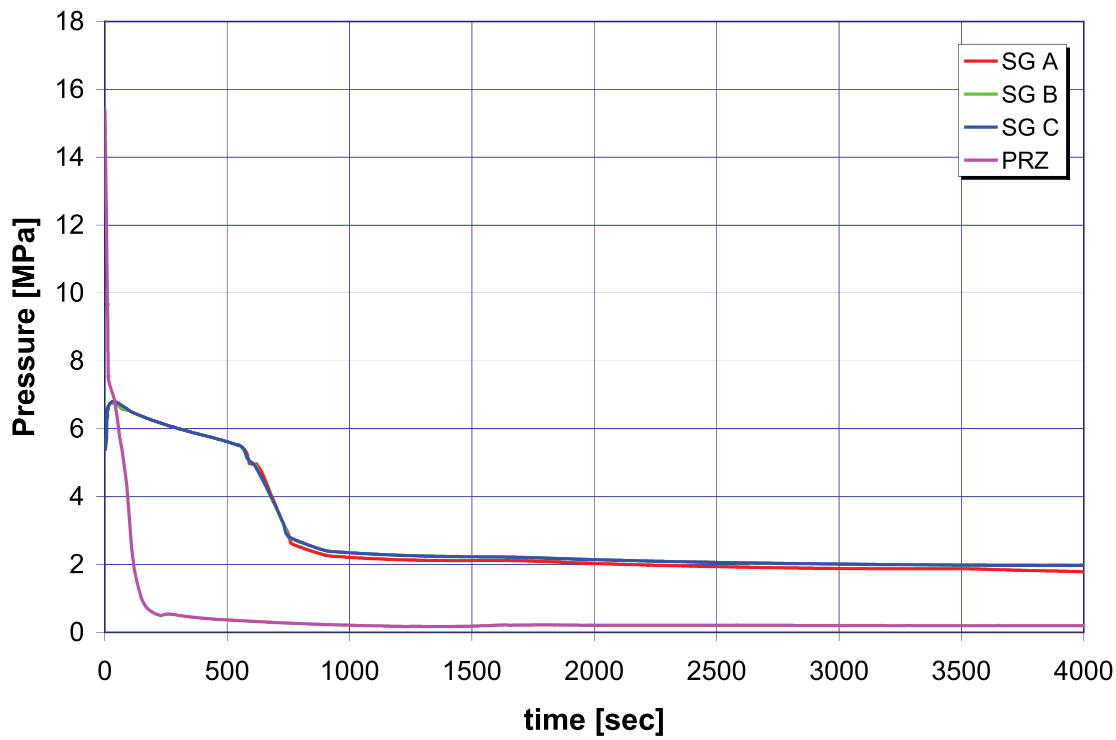
Reactor Water Level (m)
20
18
16
14
12
10
8
6
4
2
0
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
PRZ
DC
Ring 1
BAF
TAF

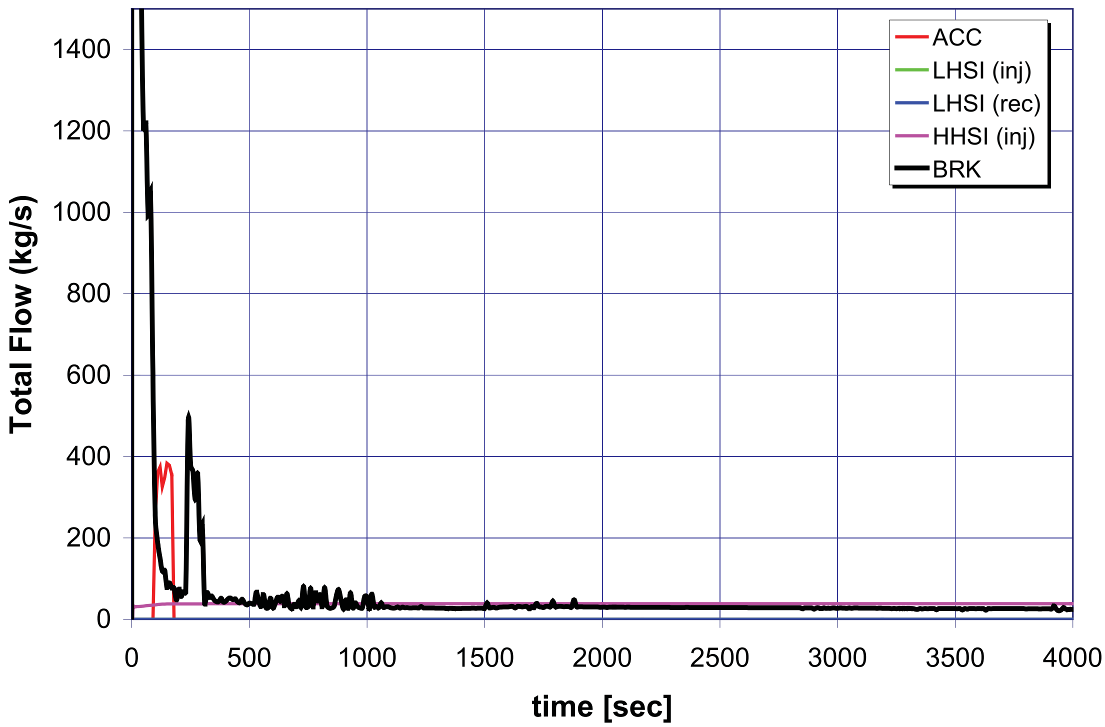


TCL
TFU
2200 F
Maximum Core Temperature [K]
1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]

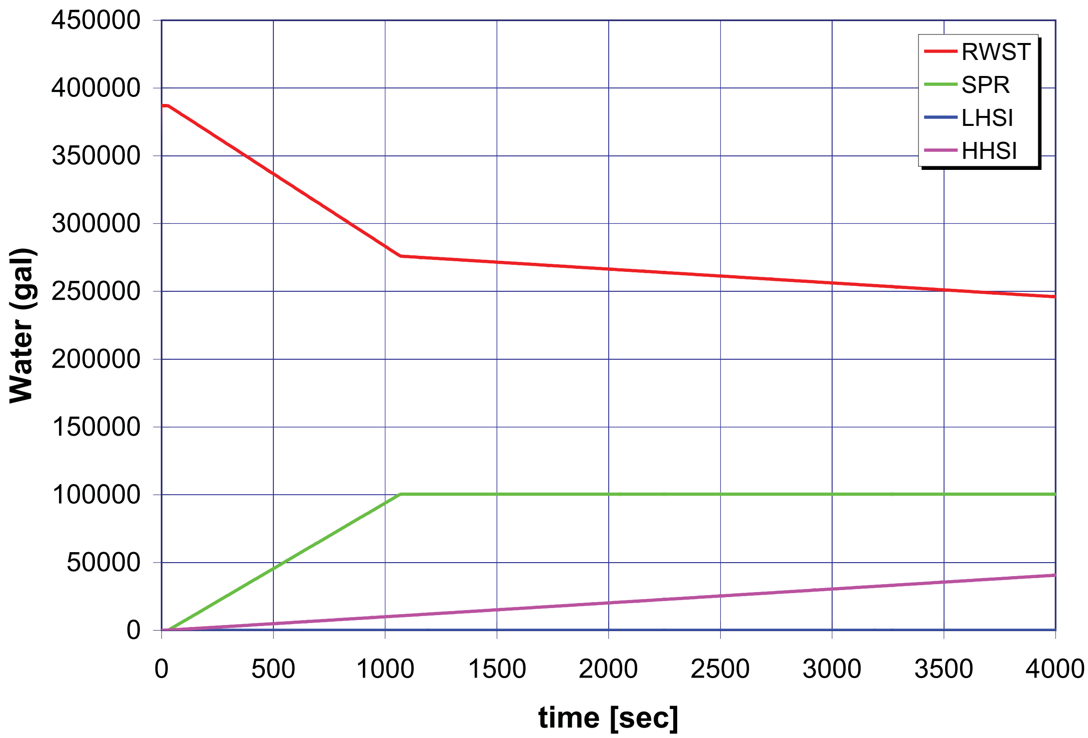
A.6.19 Case 19: 10-Inch Break LOCA, One HHSI, No LHSI, and One ACC



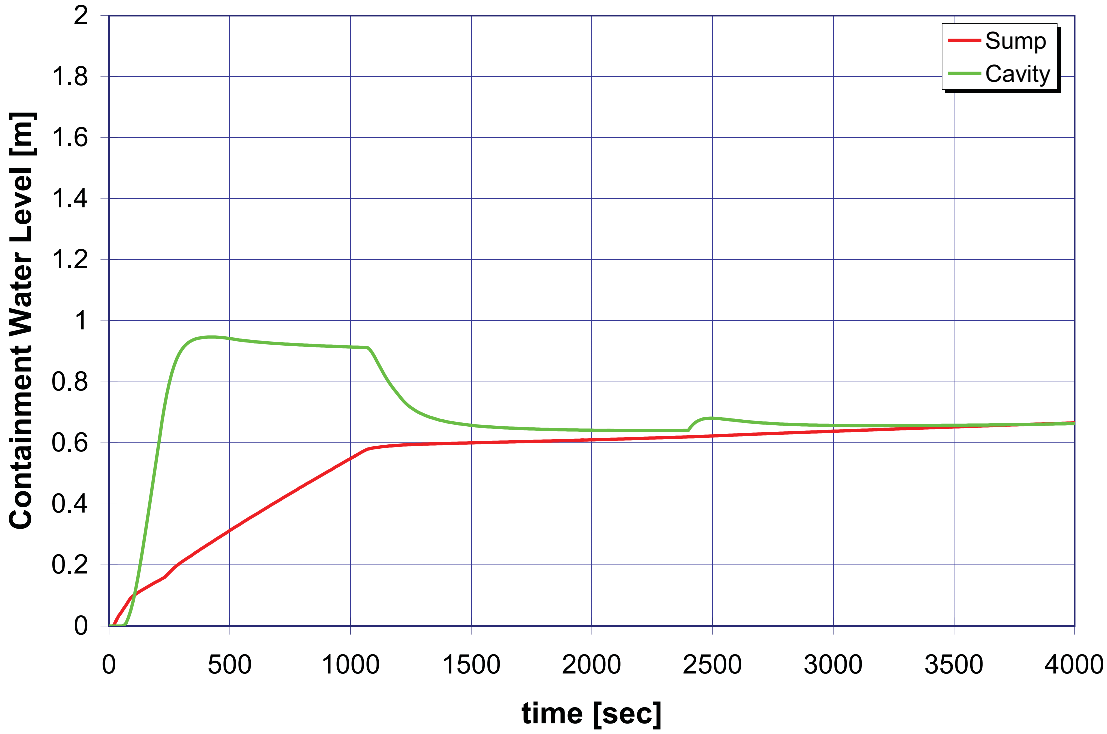


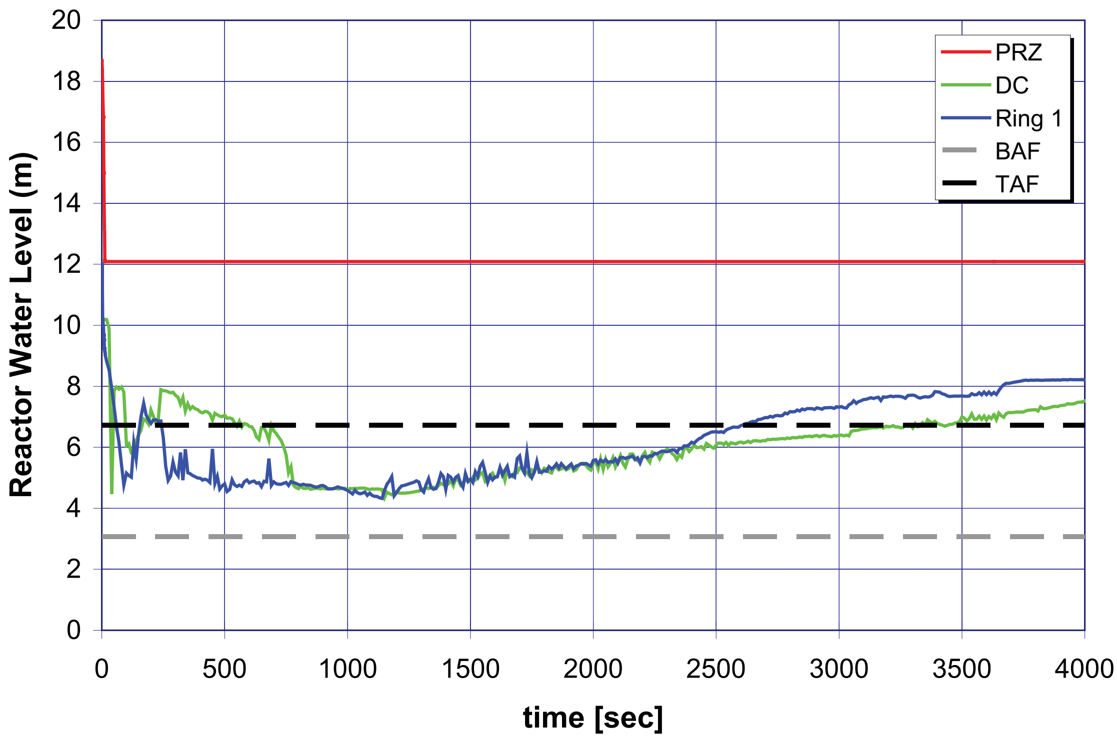


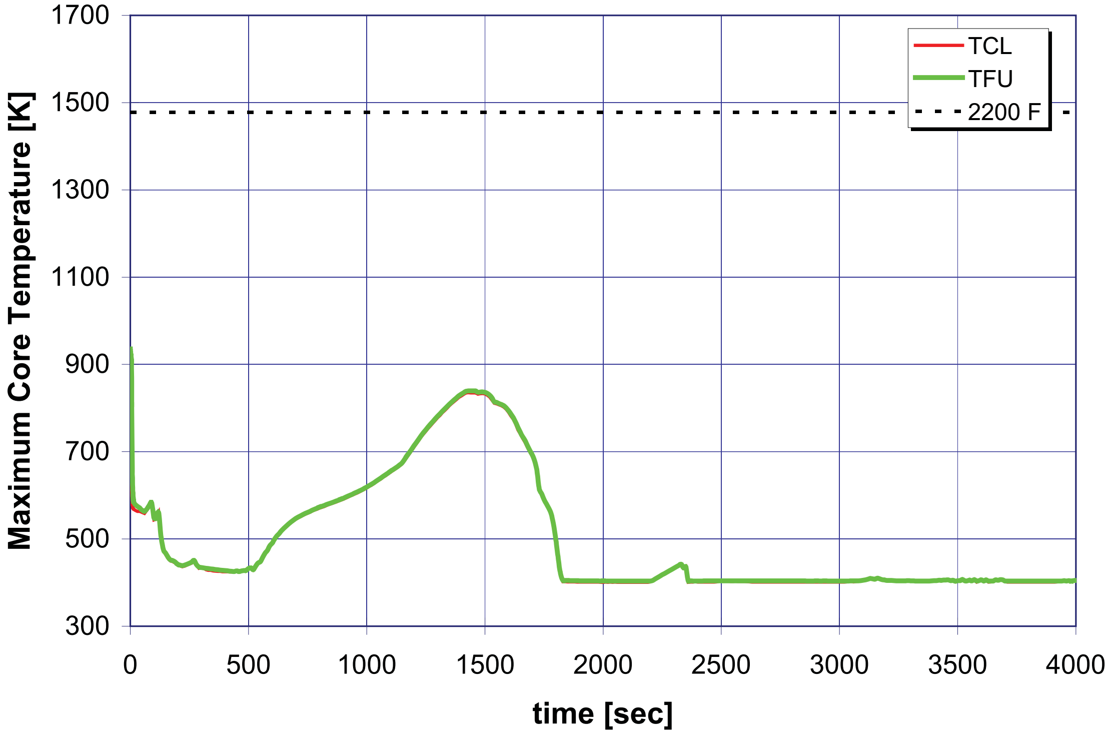
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
0
200
400
600
800
1000
1200
1400
0
500
1000
1500
2000
2500
3000
3500
4000



RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
500
1000
1500
2000
2500
3000
3500
4000

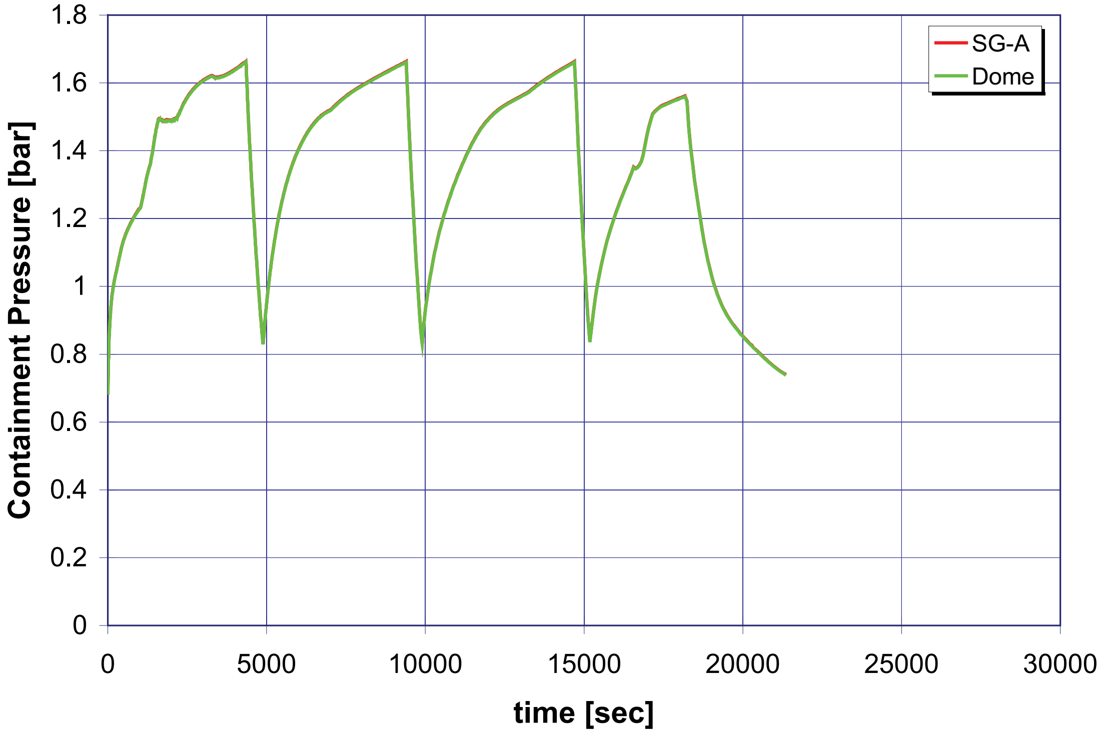

Containment Water Level [m]
time [sec]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000
3500
4000

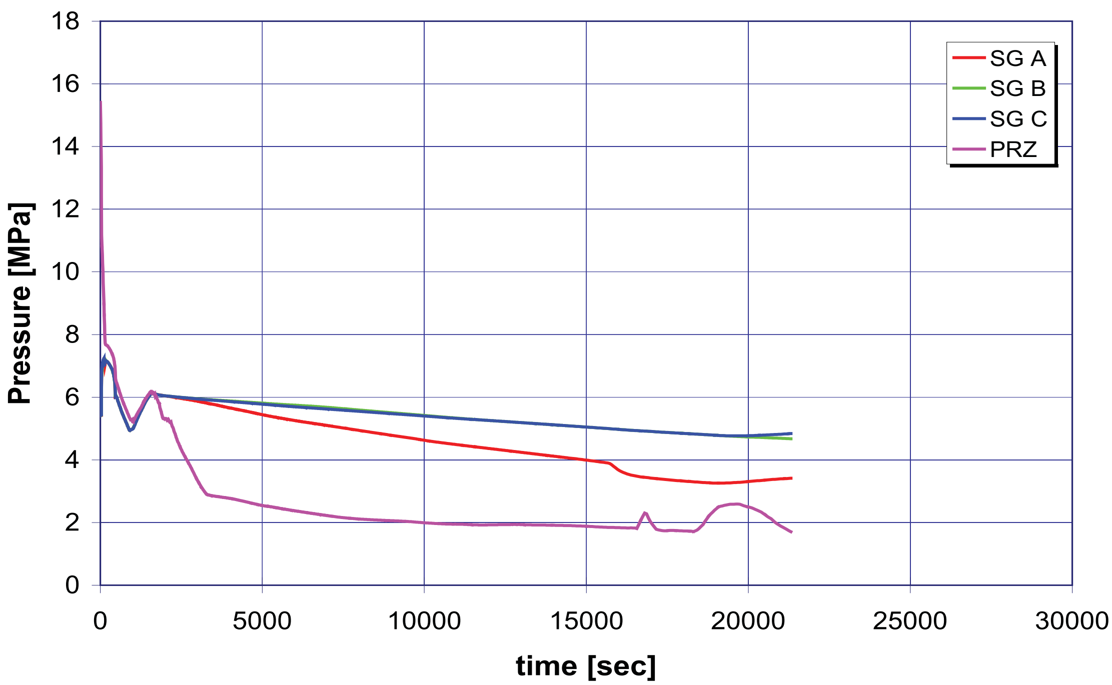

Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
500
1000
1500
2000
2500
3000
3500
4000

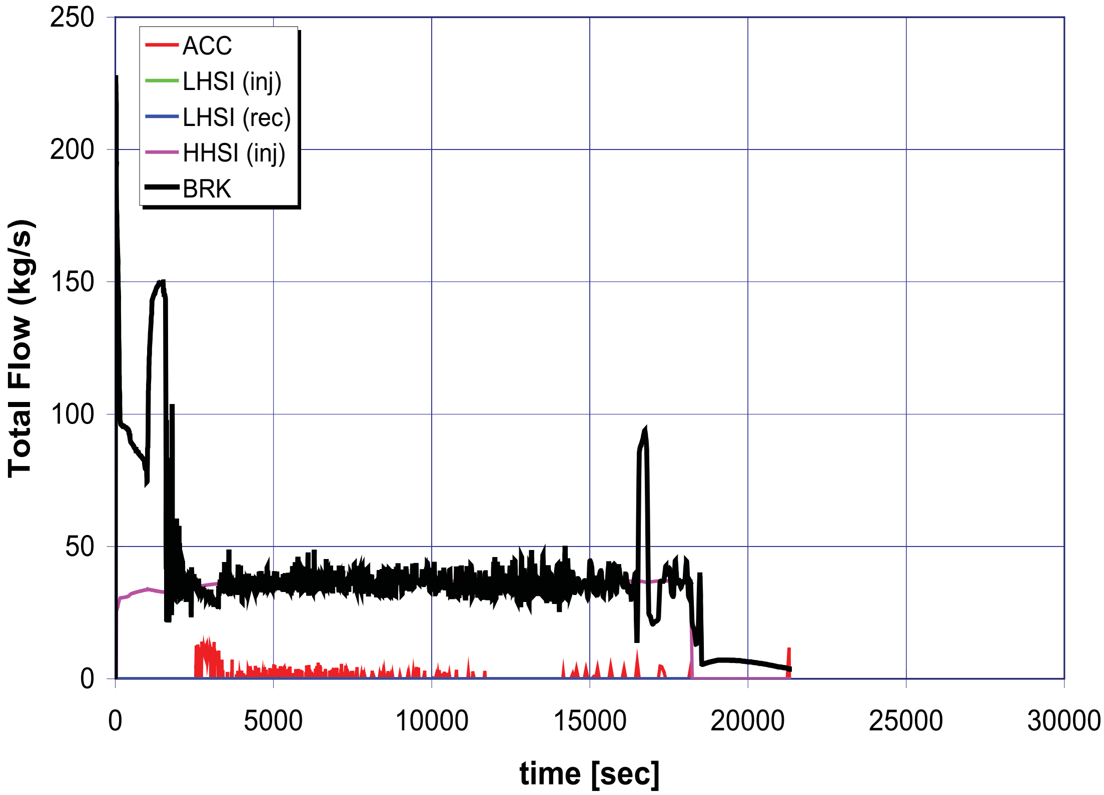


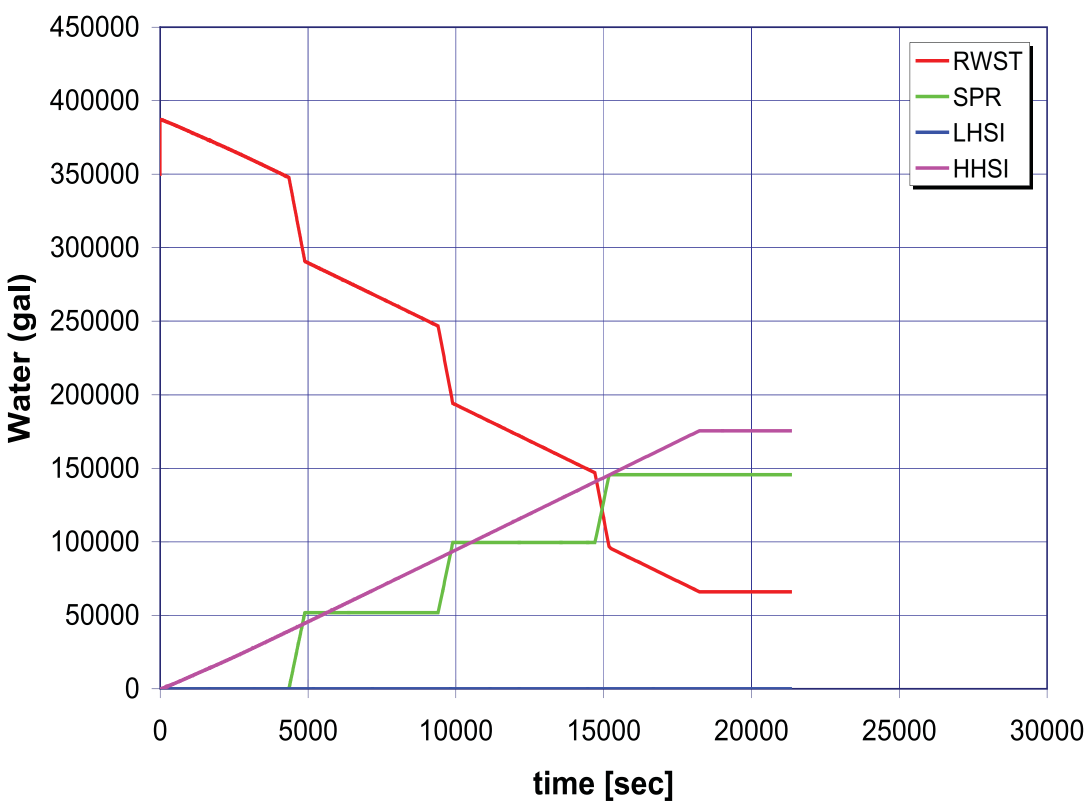
1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000
3500
4000
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F

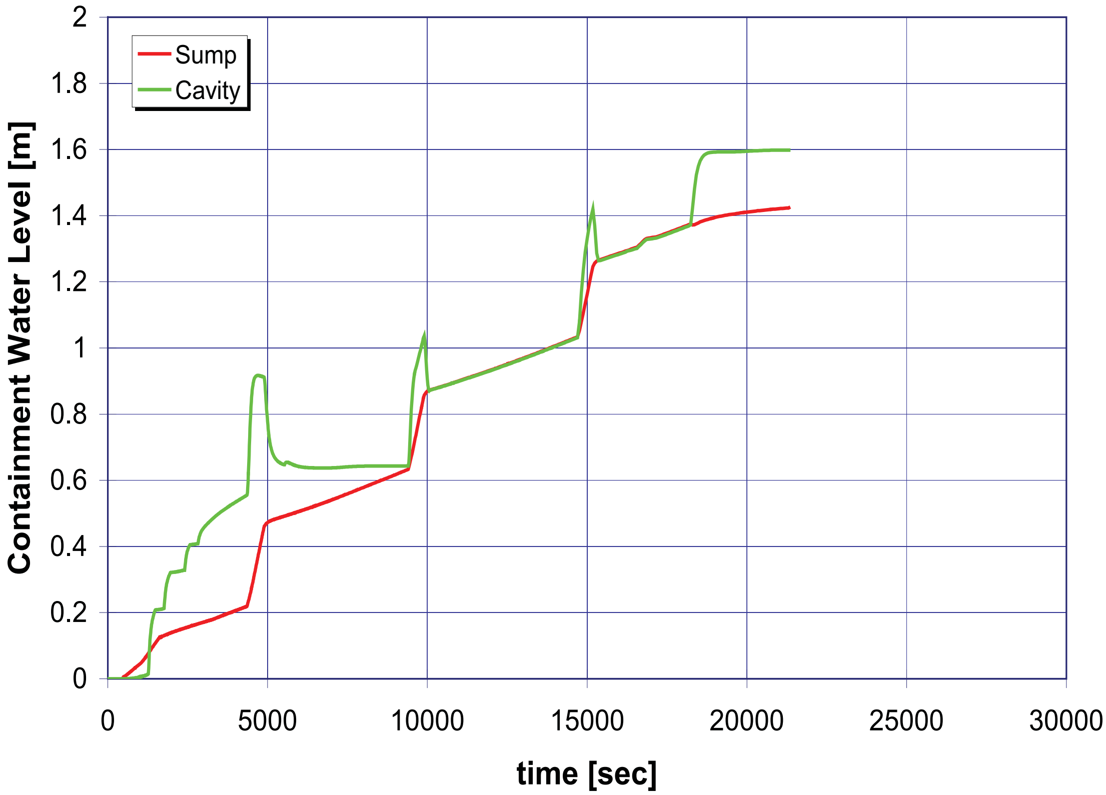
A.6.20 Case 20: 2-Inch Break LOCA, One HHSI, No LHSI, and One ACC



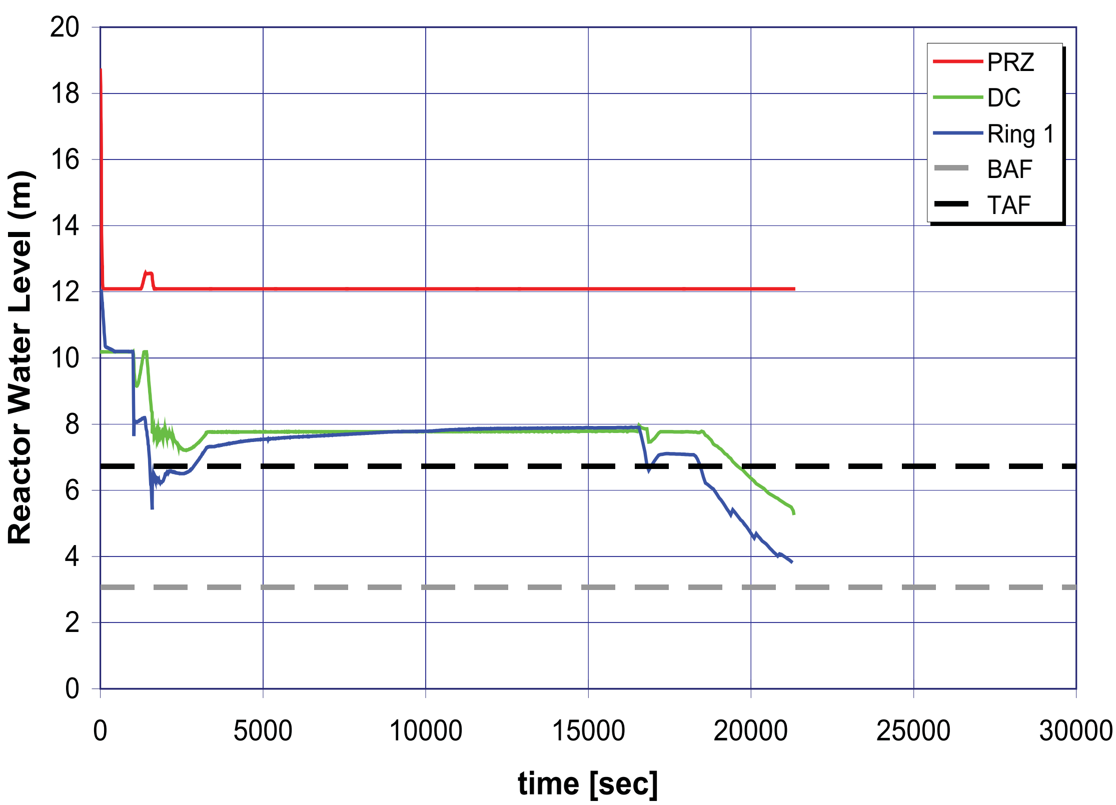



ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
250
200
150
100
50
0
0
5000
10000
15000
20000
25000
30000
time [sec]

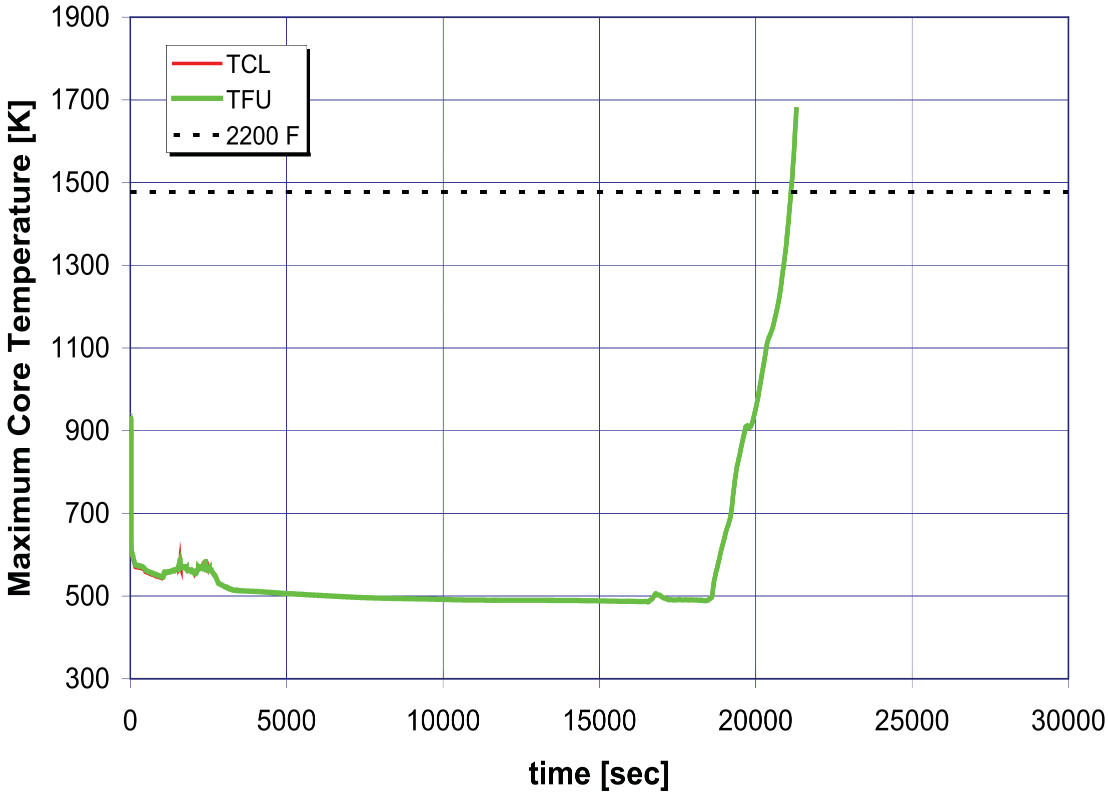

RWST
SPR
LHSI
HHSI
Water (gal)
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
5000
10000
15000
20000
25000
30000
time [sec]



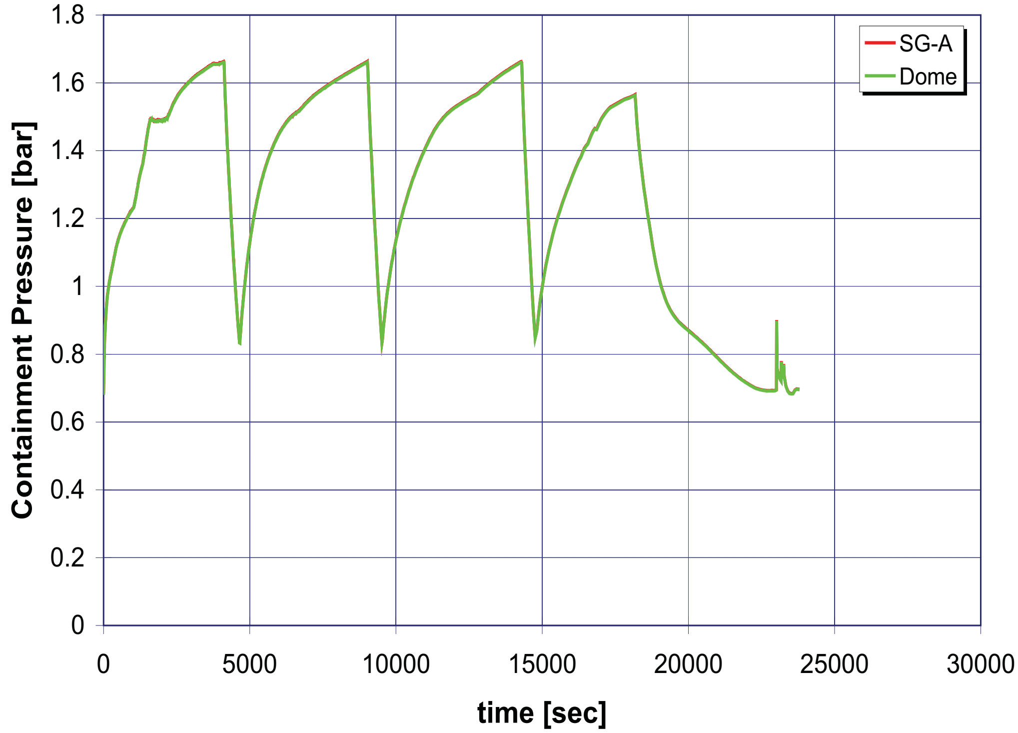
Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000

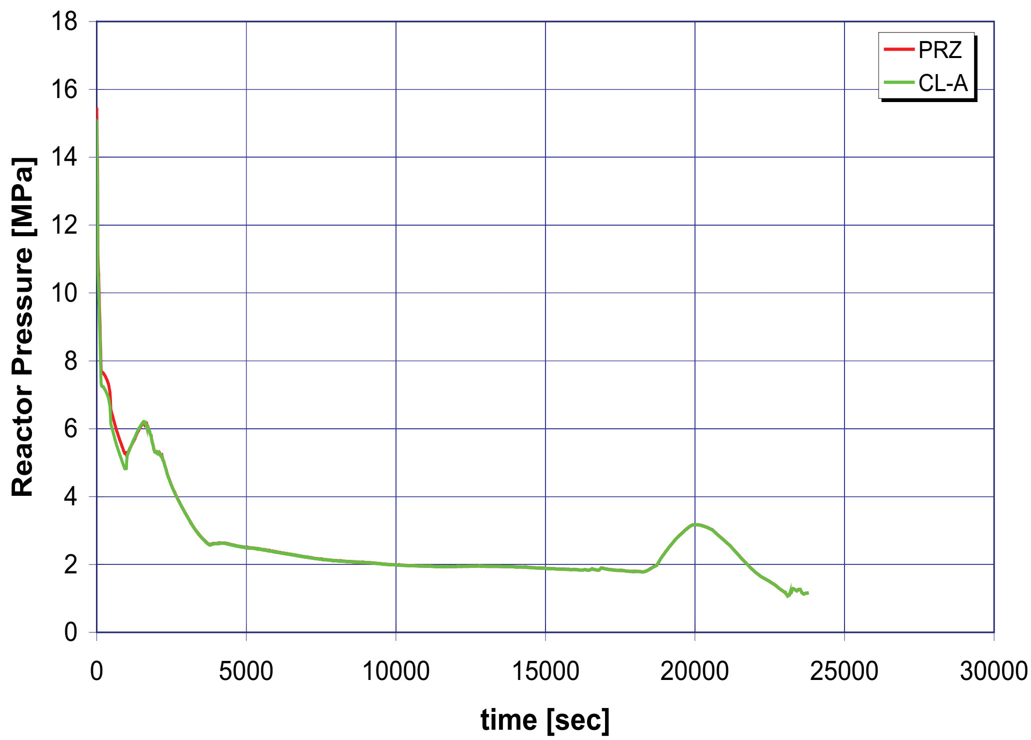


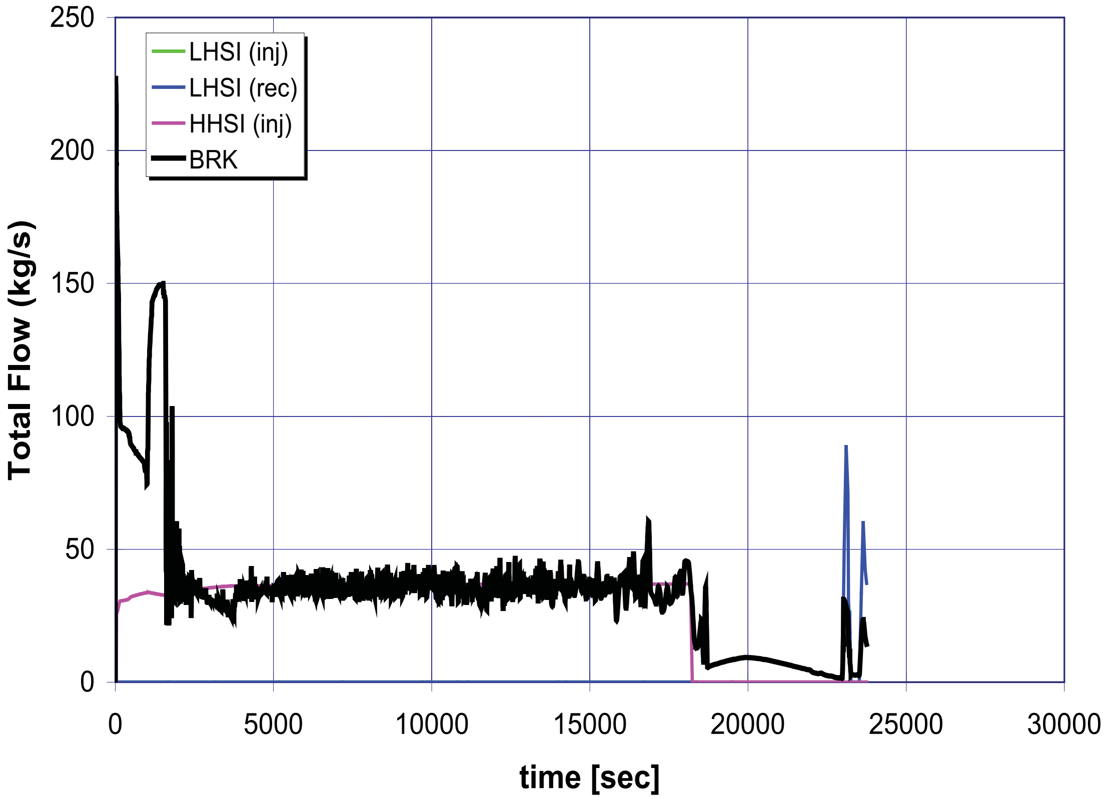
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
5000
10000
15000
20000
25000
30000


TCL
TFU
2200 F
Maximum Core Temperature [K]
time [sec]
1900
1700
1500
1300
1100
900
700
500
300
0
5000
10000
15000
20000
25000
30000

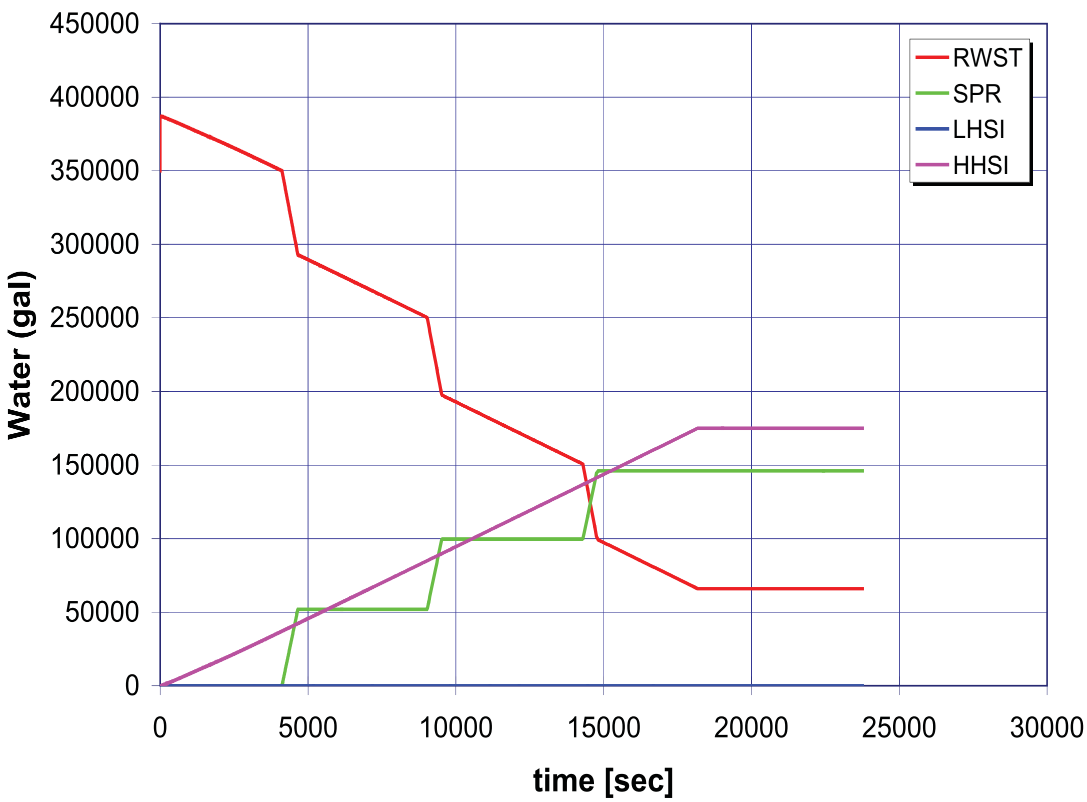
A.6.21 Case 21: 2-Inch Break LOCA, One HHSI, One LHSI, and No ACC



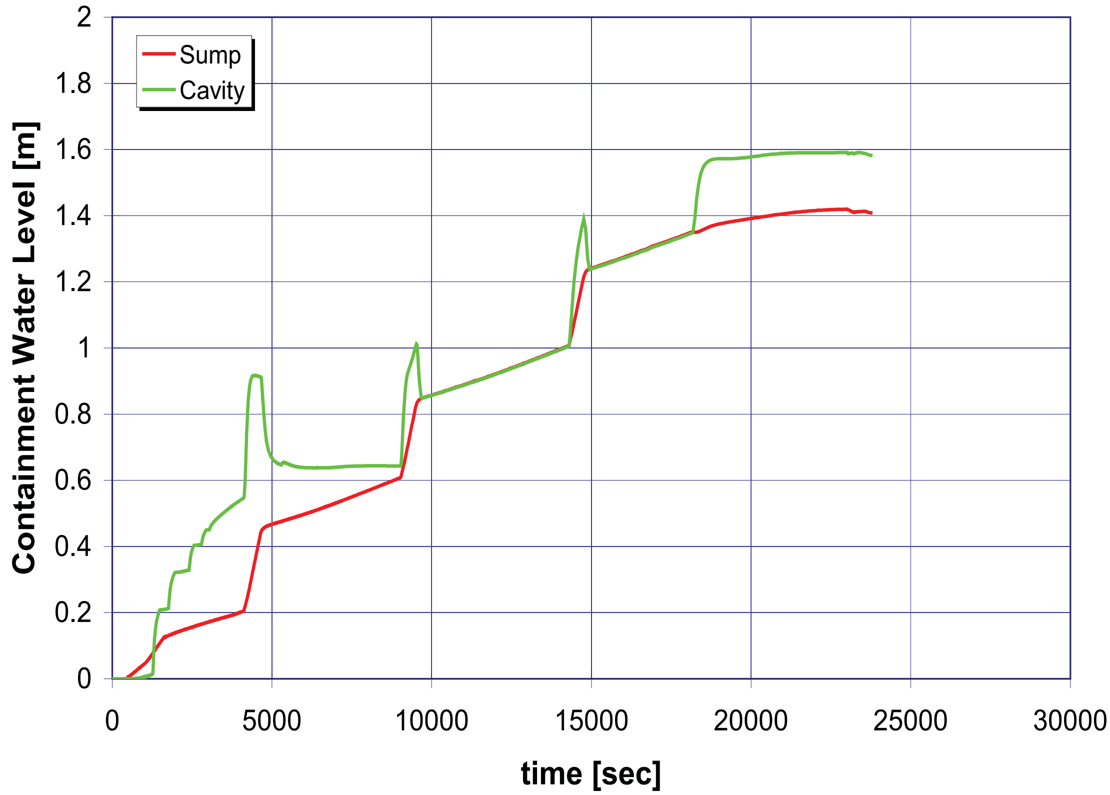




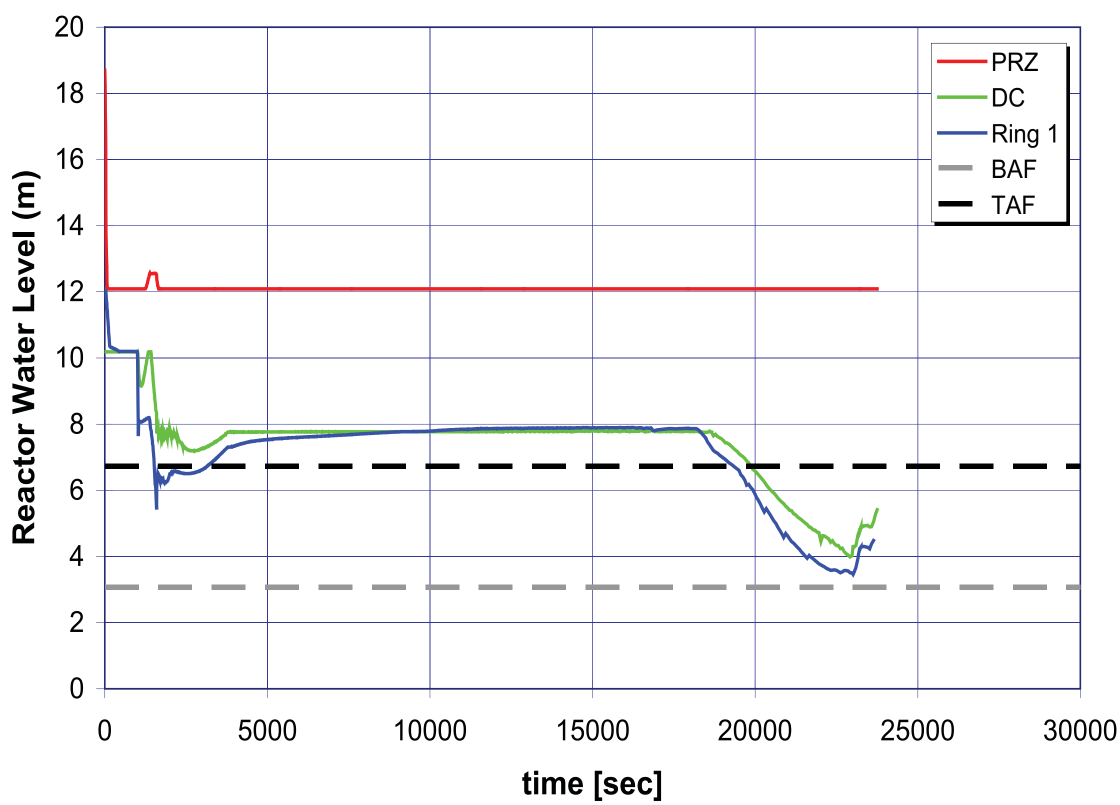
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
250
200
150
100
50
0
0
5000
10000
15000
20000
25000
30000



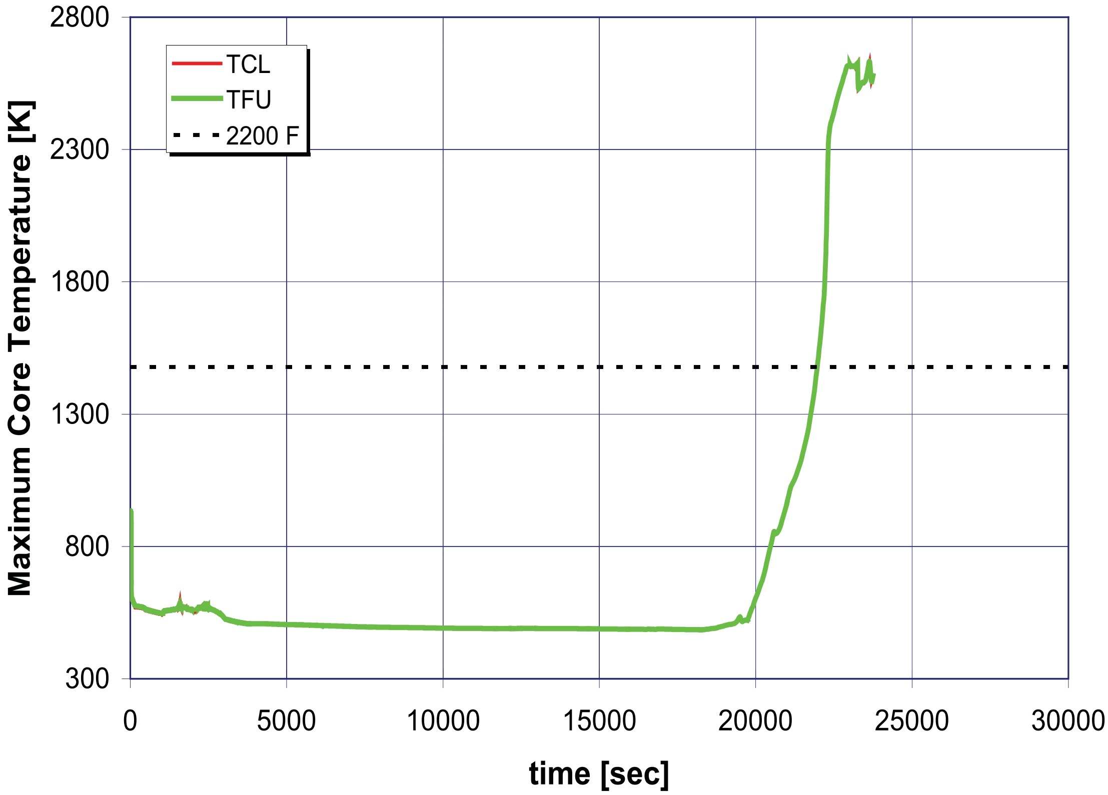
RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
5000
10000
15000
20000
25000
30000



Containment Water Level [m]
time [sec]
Sump
Cavity
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
5000
10000
15000
20000
25000
30000

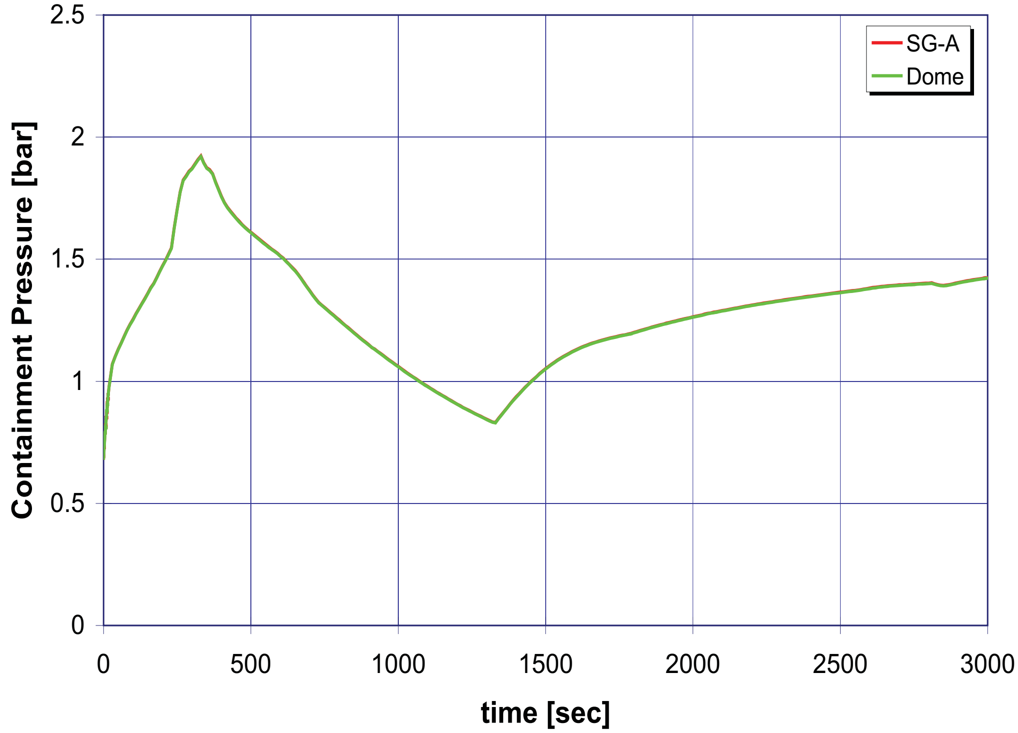


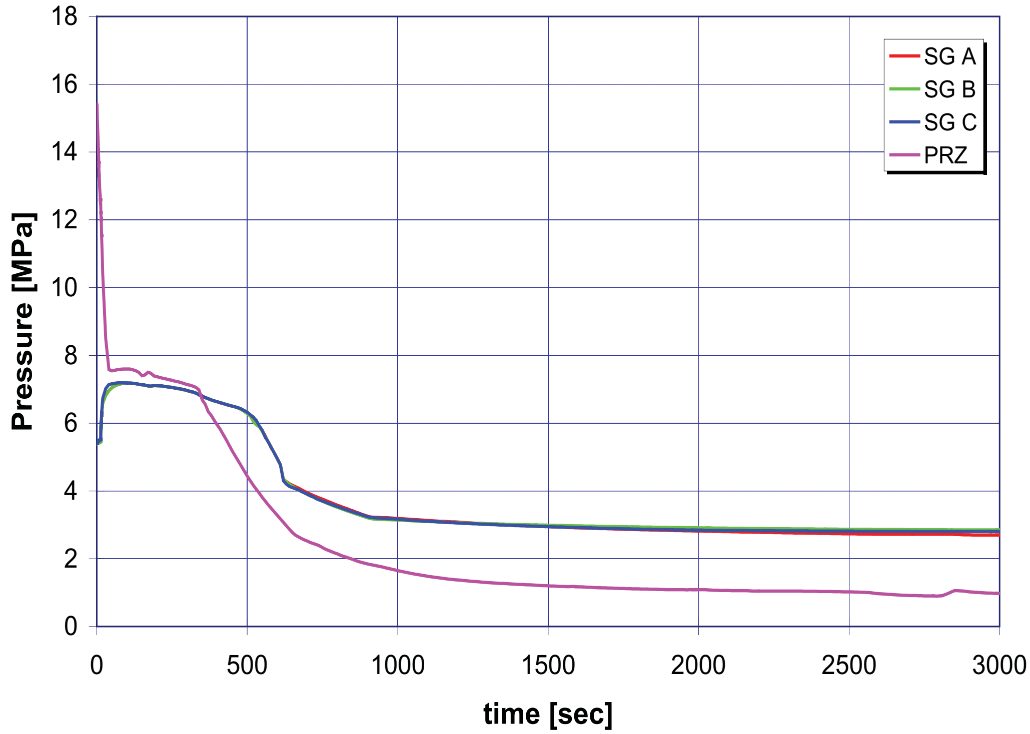
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
20
18
16
14
12
10
8
6
4
2
0
0
5000
10000
15000
20000
25000
30000

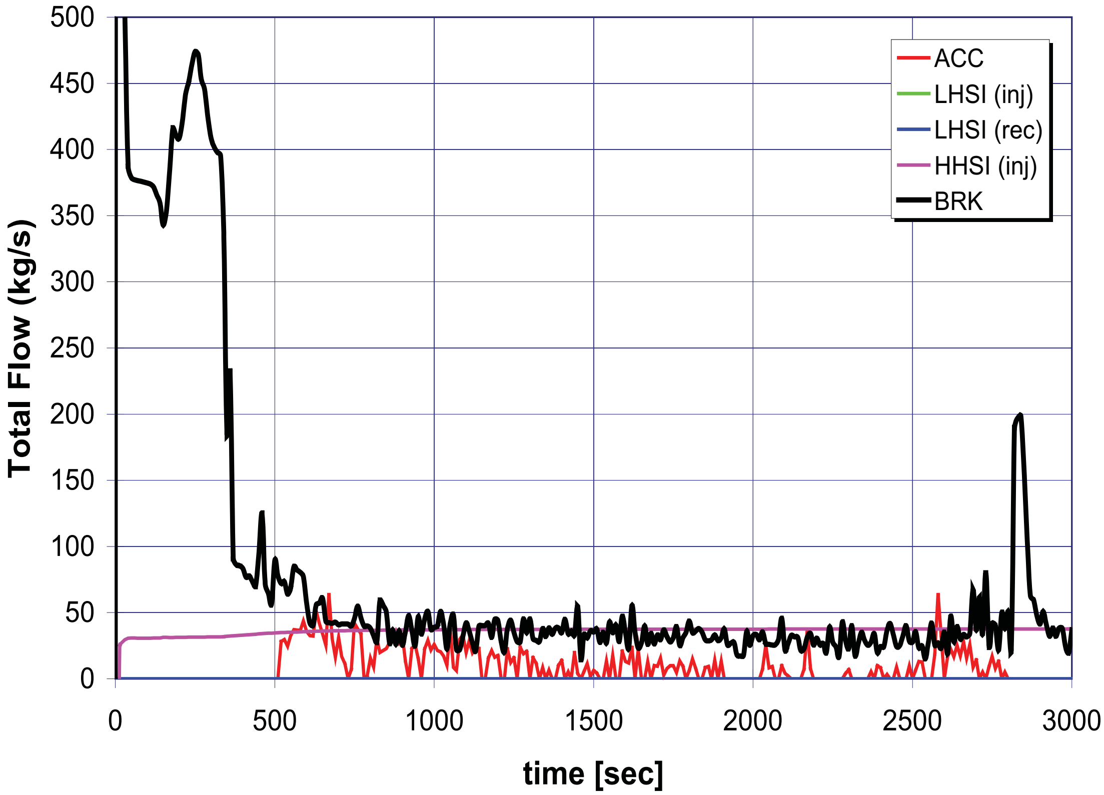


2800
2300
1800
1300
800
300
0
5000
10000
15000
20000
25000
30000
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F

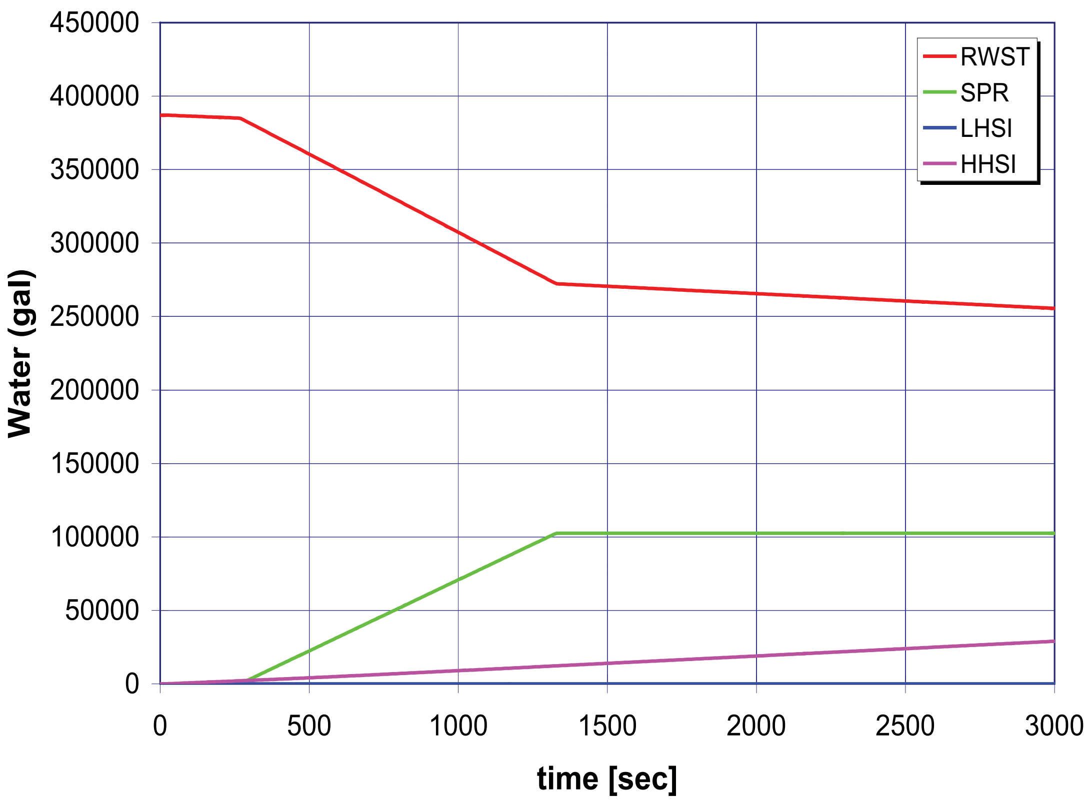
A.6.22 Case 22: 4-Inch Break LOCA, One HHSI, No LHSI, and One ACC



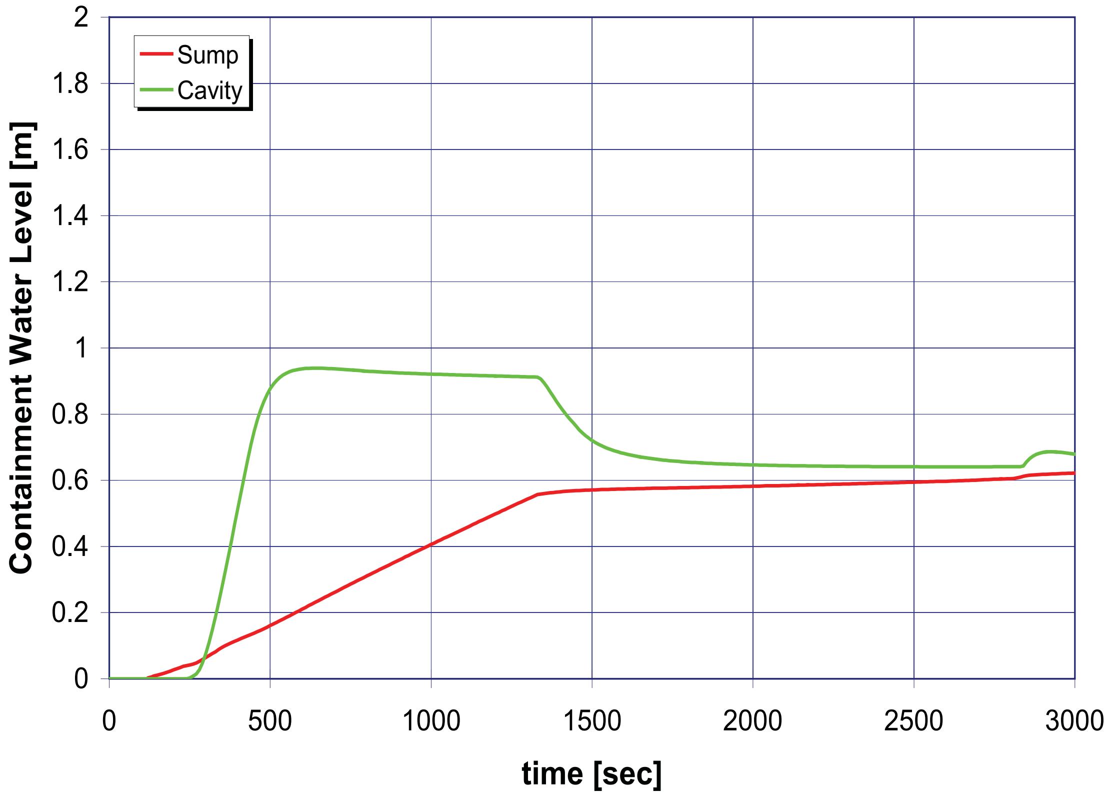


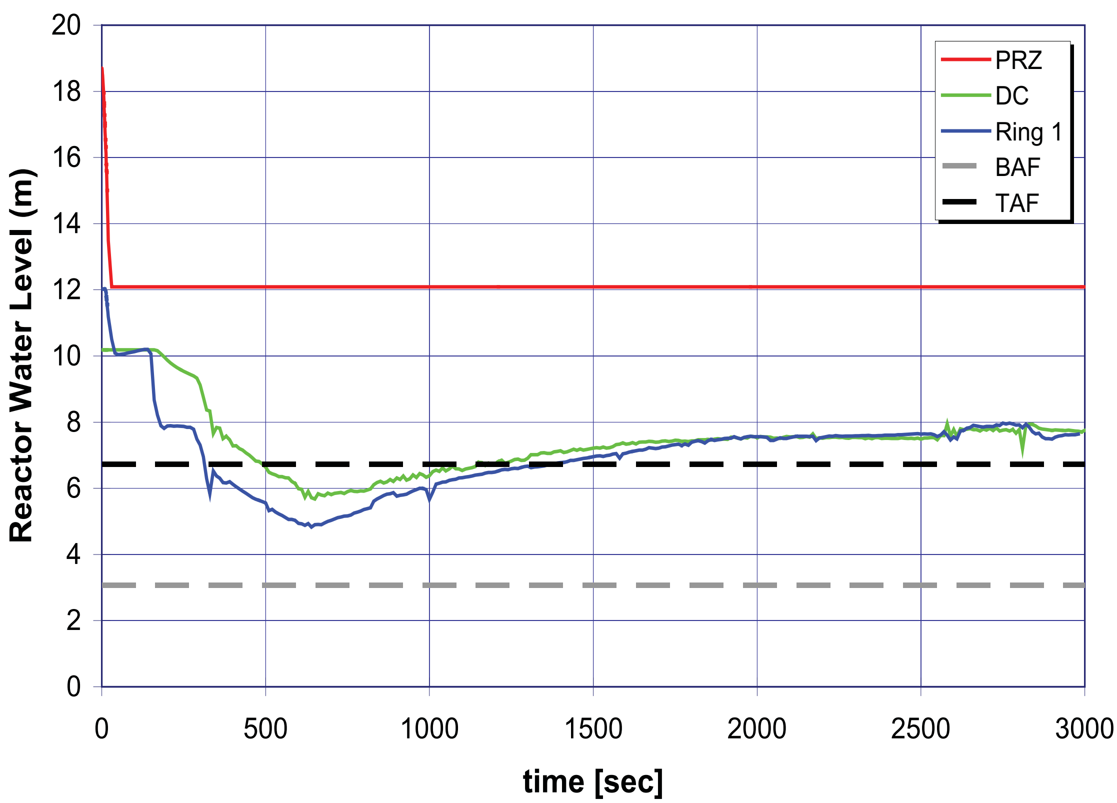


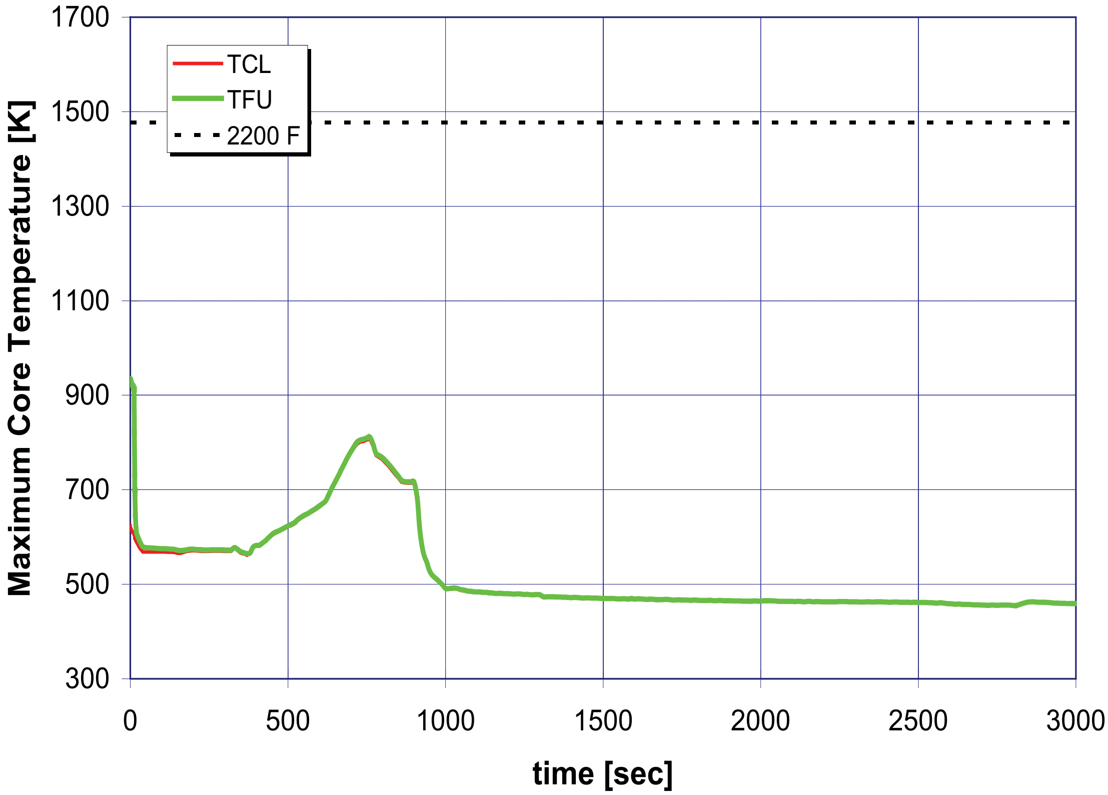
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
500
450
400
350
300
250
200
150
100
50
0
0
500
1000
1500
2000
2500
3000



RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
500
1000
1500
2000
2500
3000

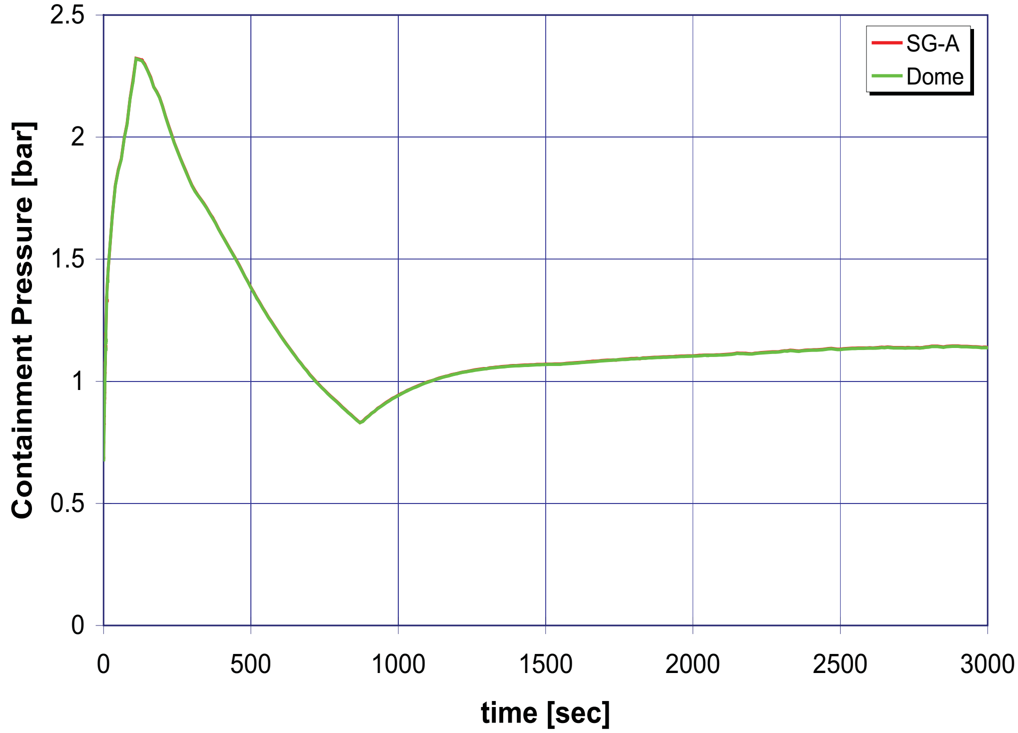

Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000

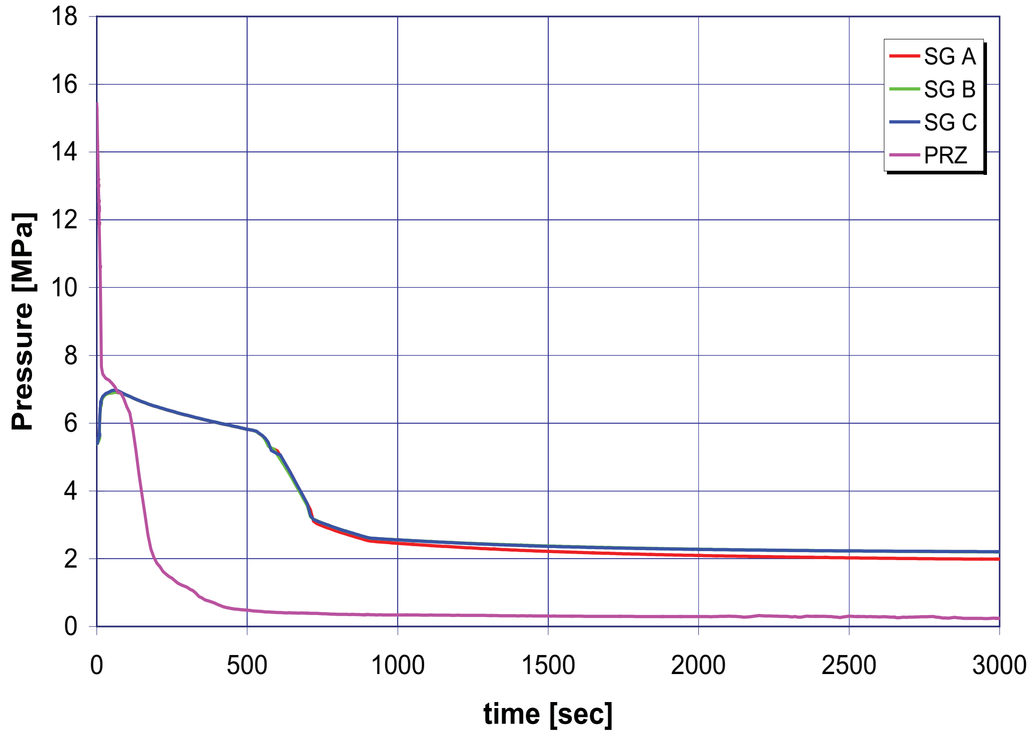

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
500
1000
1500
2000
2500
3000

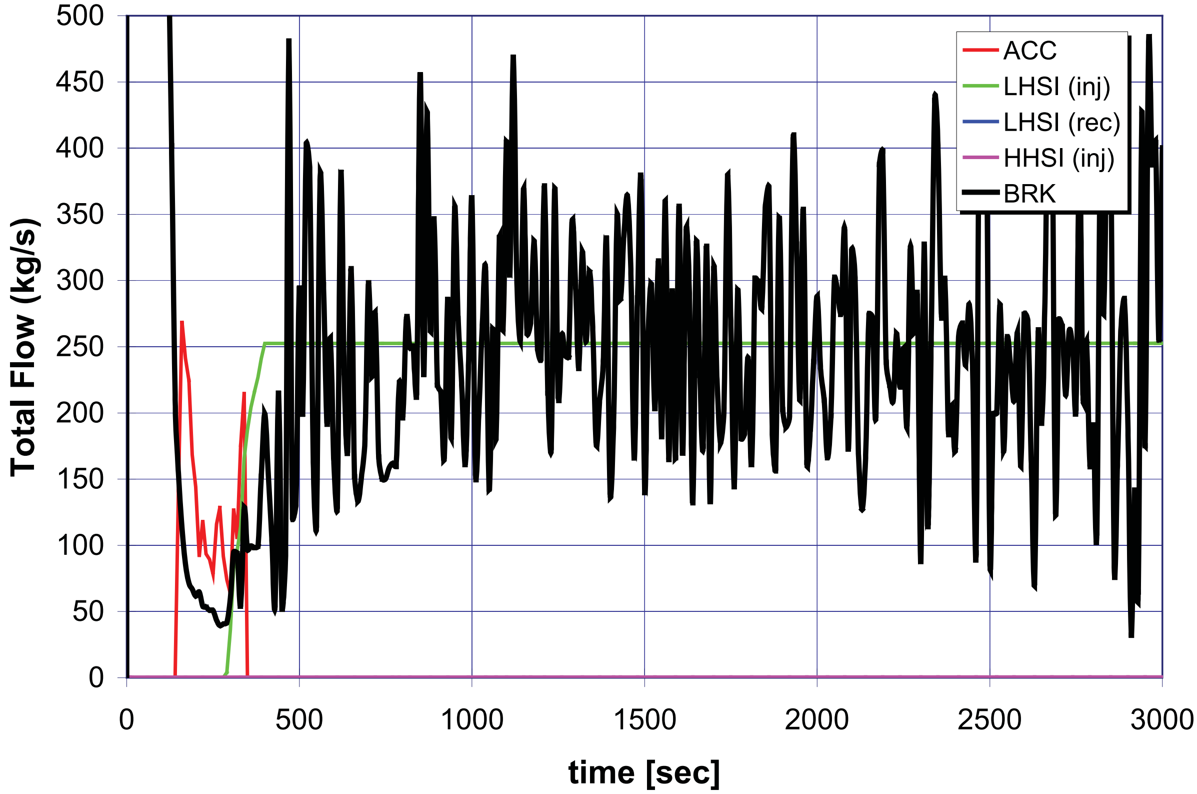


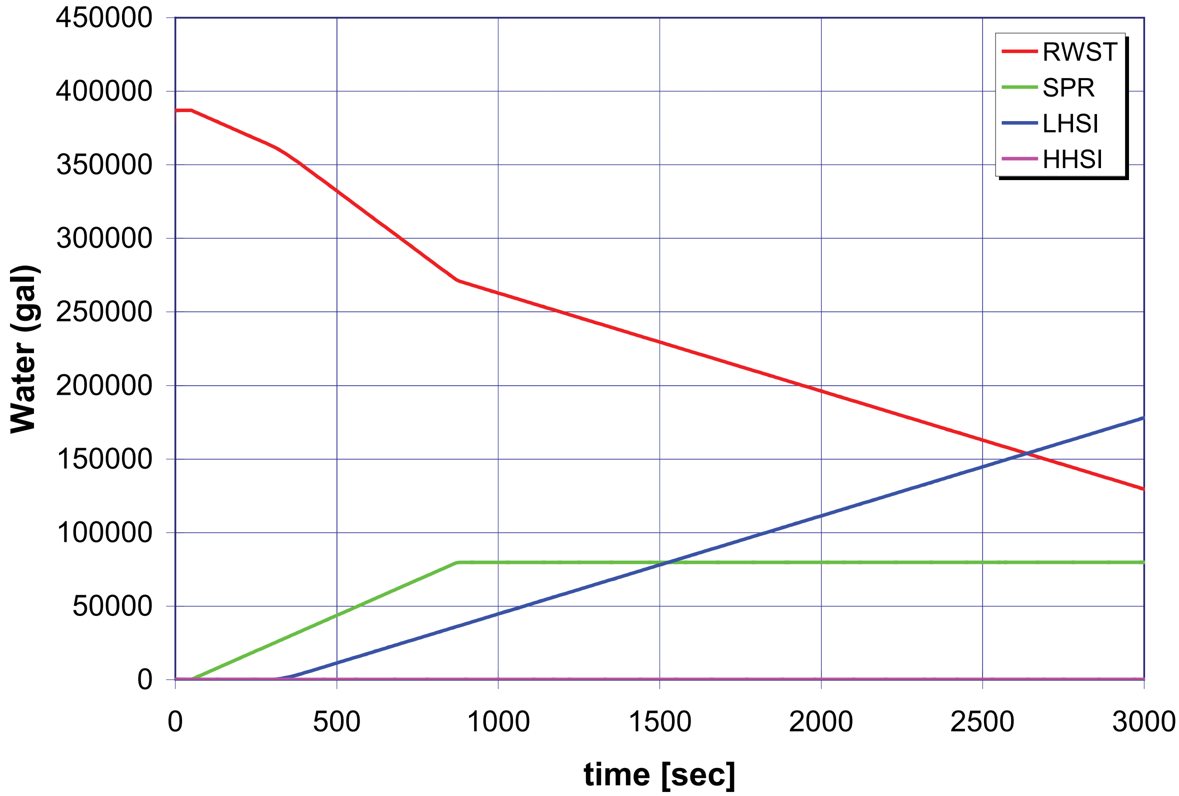
TCL
TFU
2200 F
Maximum Core Temperature [K]
time [sec]
1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000

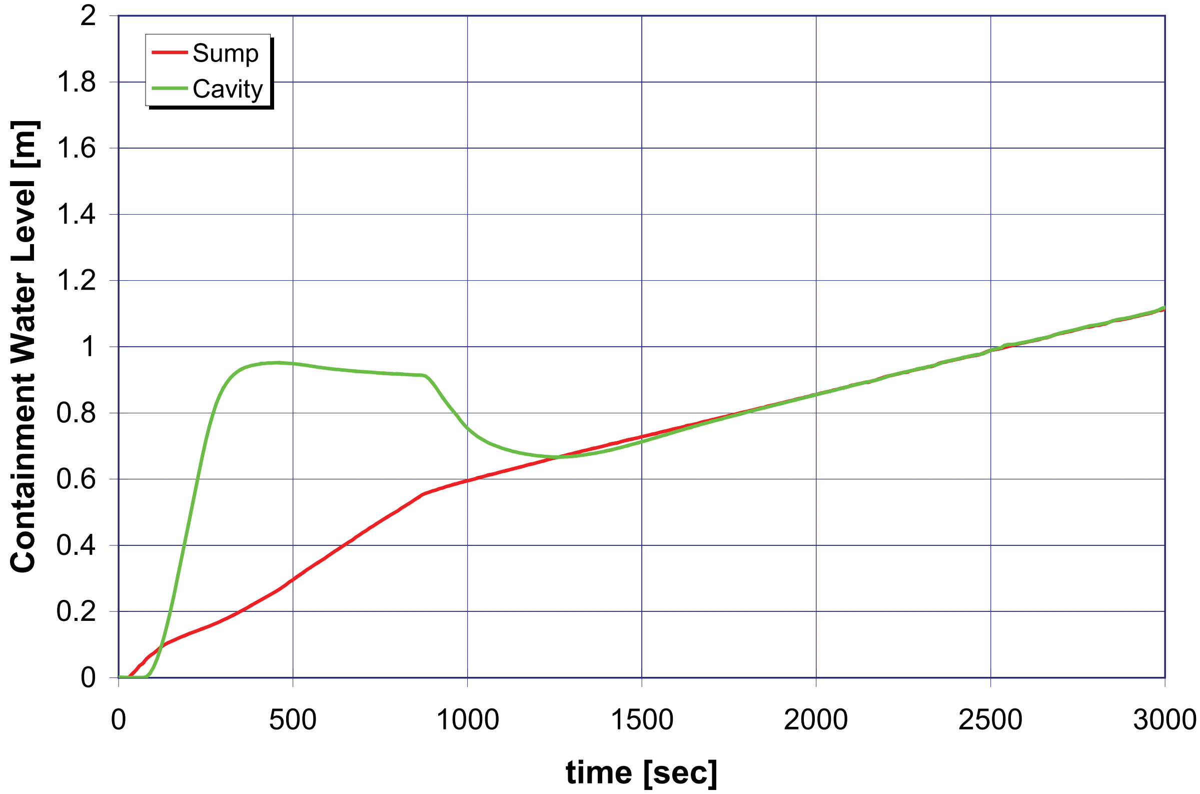
A.6.23 Case 23: 8-Inch Break LOCA, No HHSI, One LHSI, and One ACC



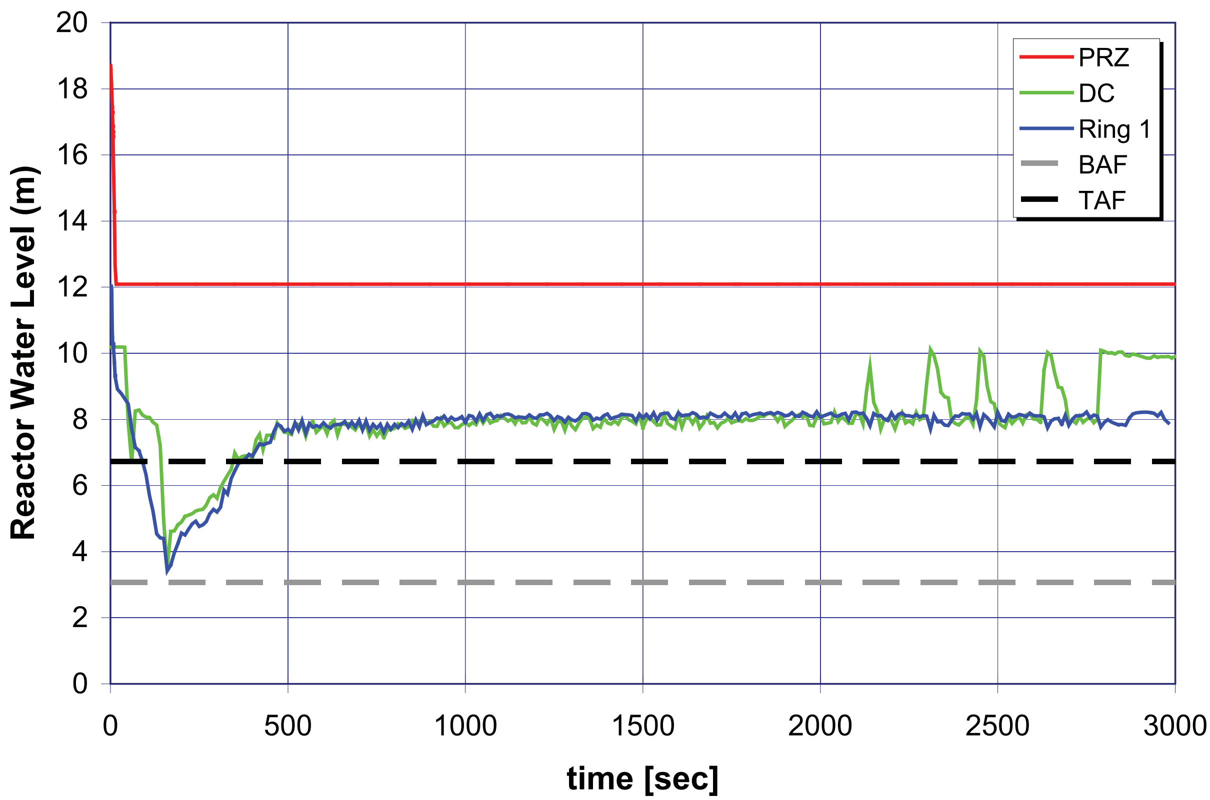



Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
0
50
100
150
200
250
300
350
400
450
500
0
500
1000
1500
2000
2500
3000

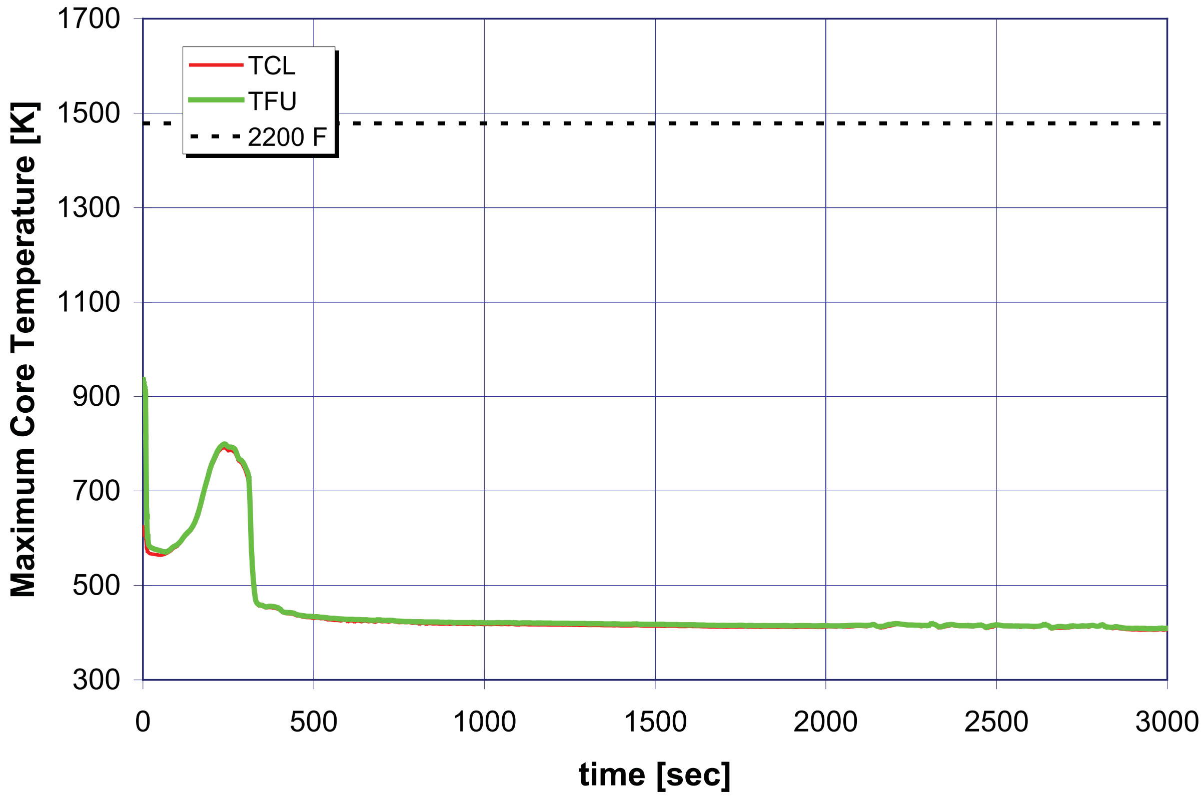

Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
500
1000
1500
2000
2500
3000



Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000

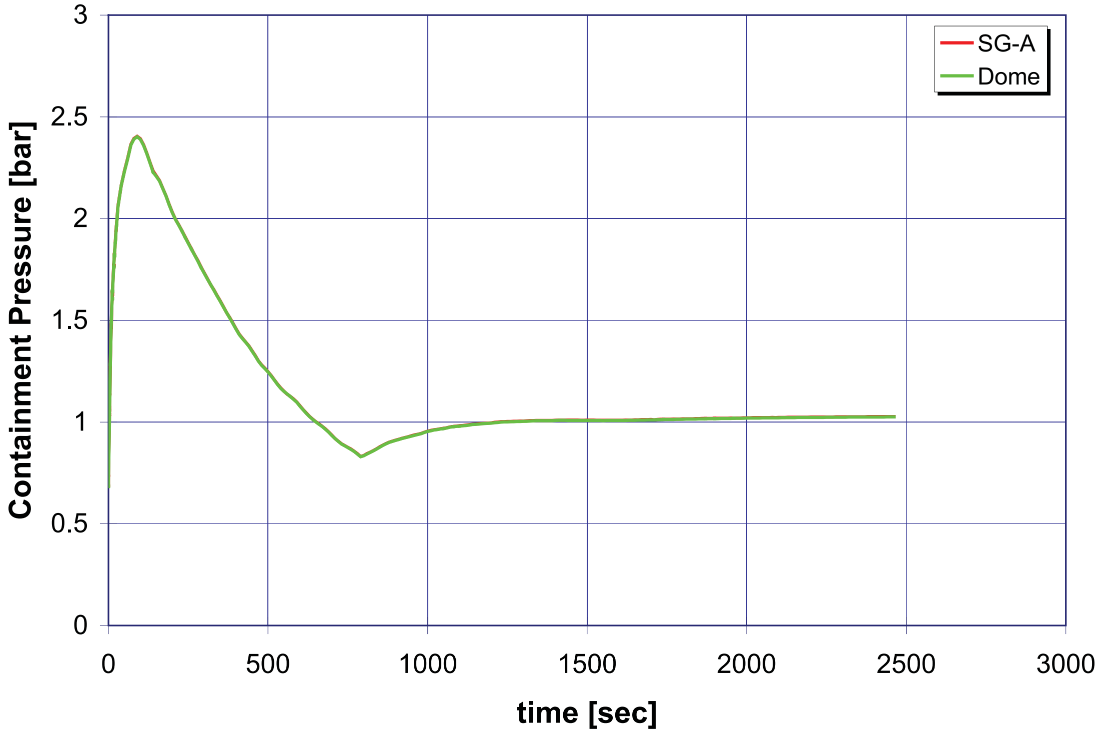


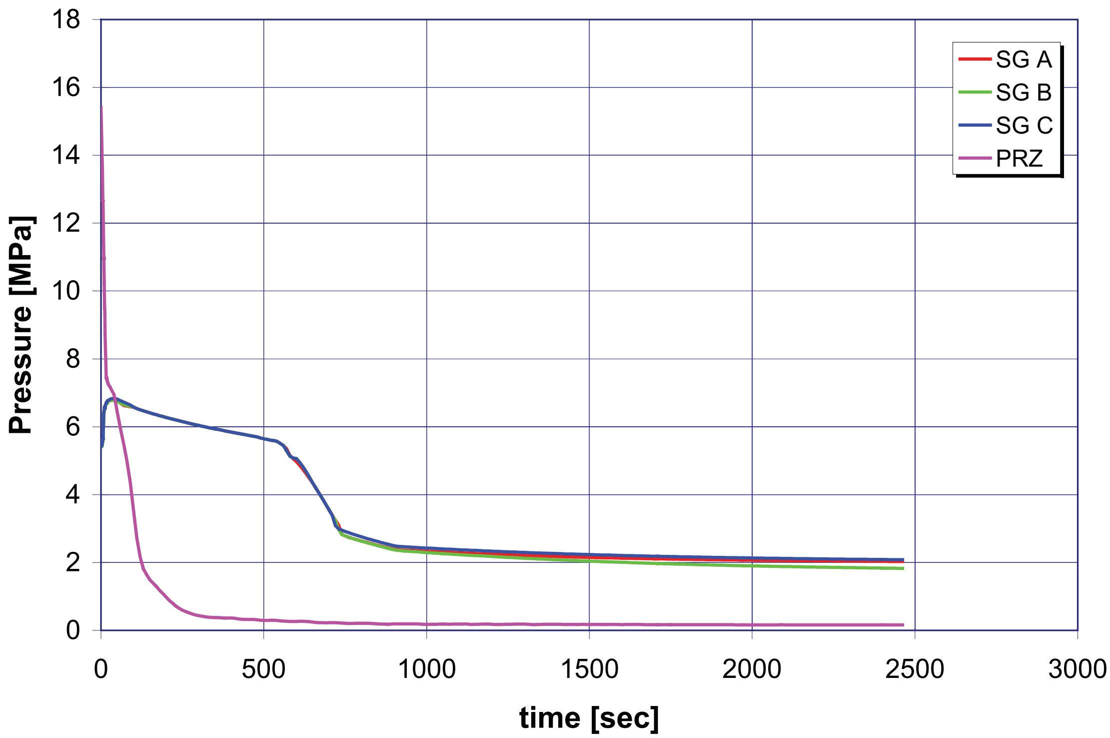
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
500
1000
1500
2000
2500
3000

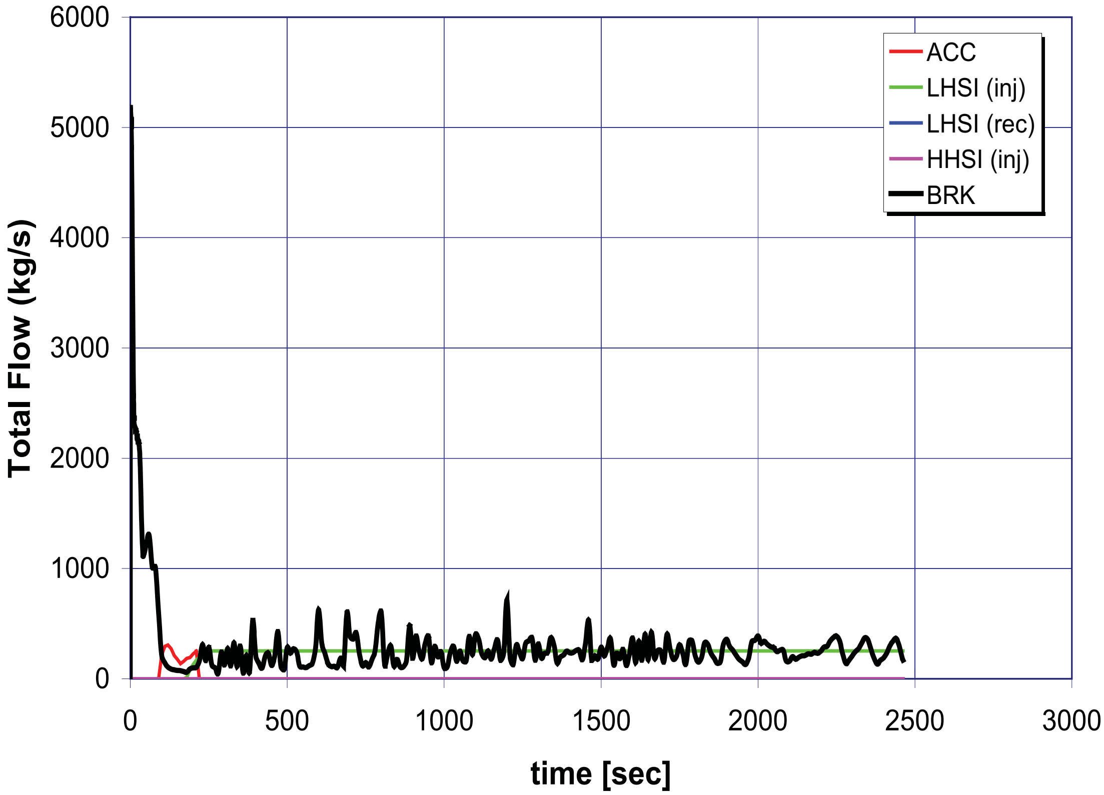


1700
1500
1300
1100
900
700
500
300
Maximum Core Temperature [K]
TCL
TFU
2200 F
0
500
1000
1500
2000
2500
3000
time [sec]

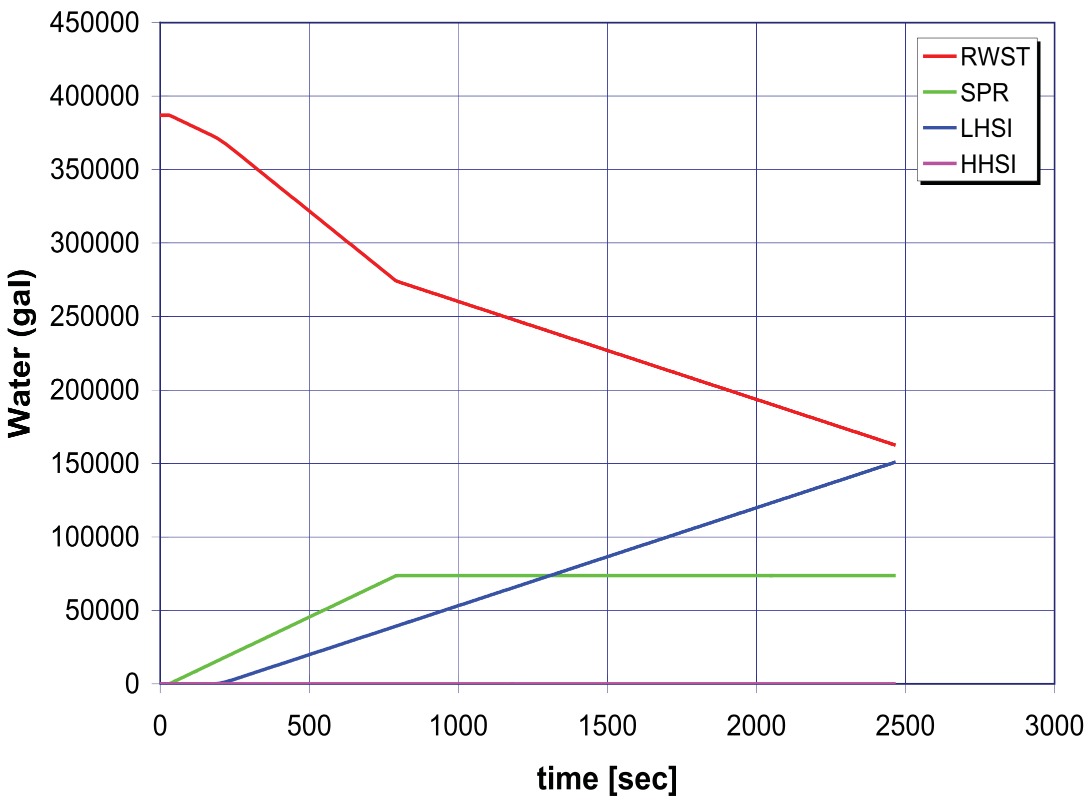
A.6.24 Case 24: 10-Inch Break LOCA, No HHSI, One LHSI, and One ACC



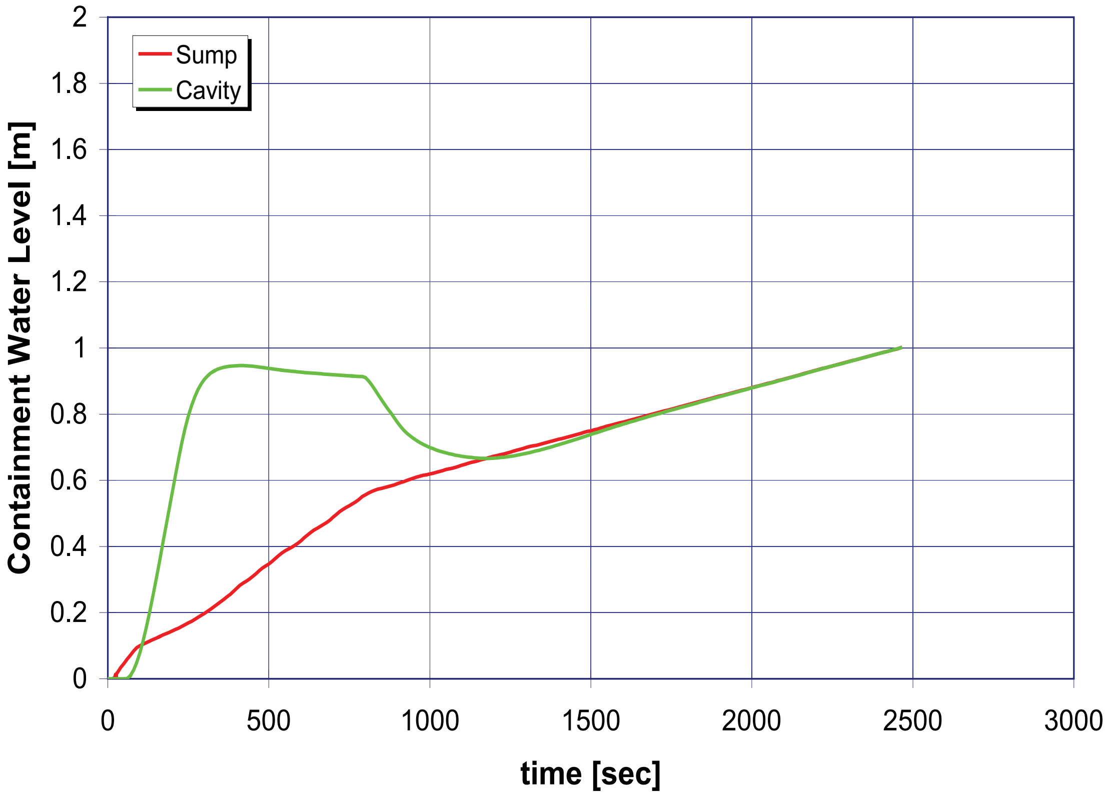


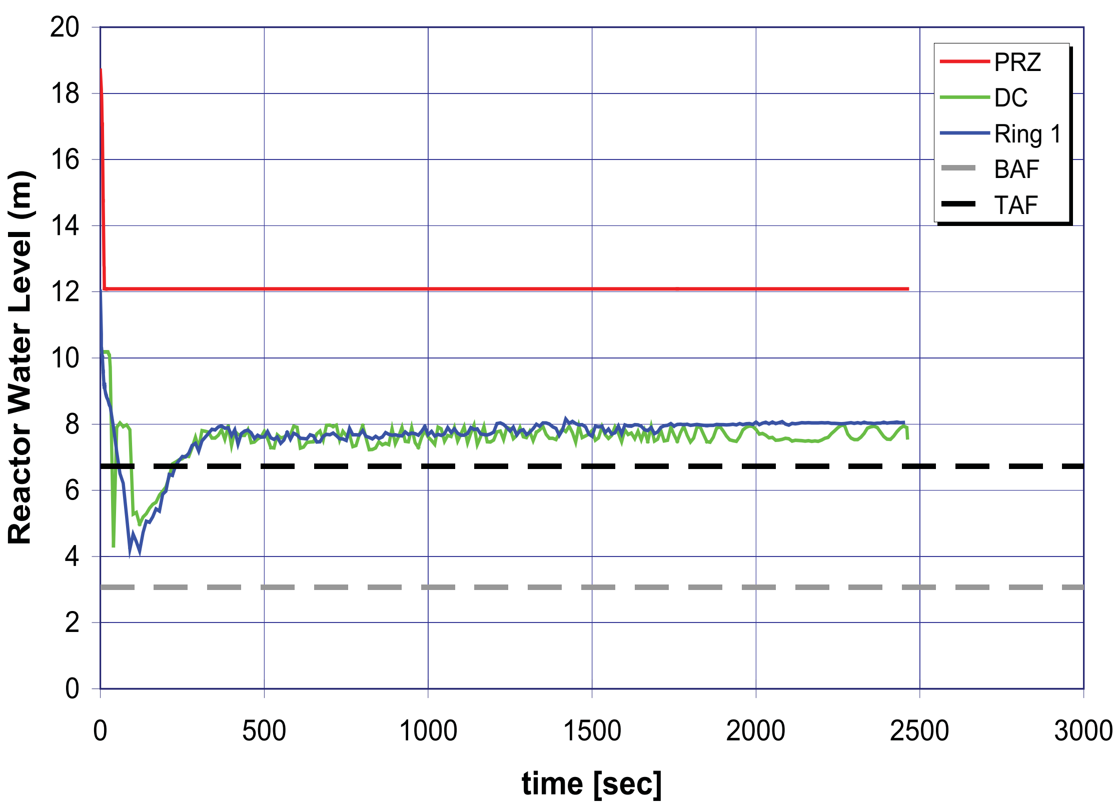


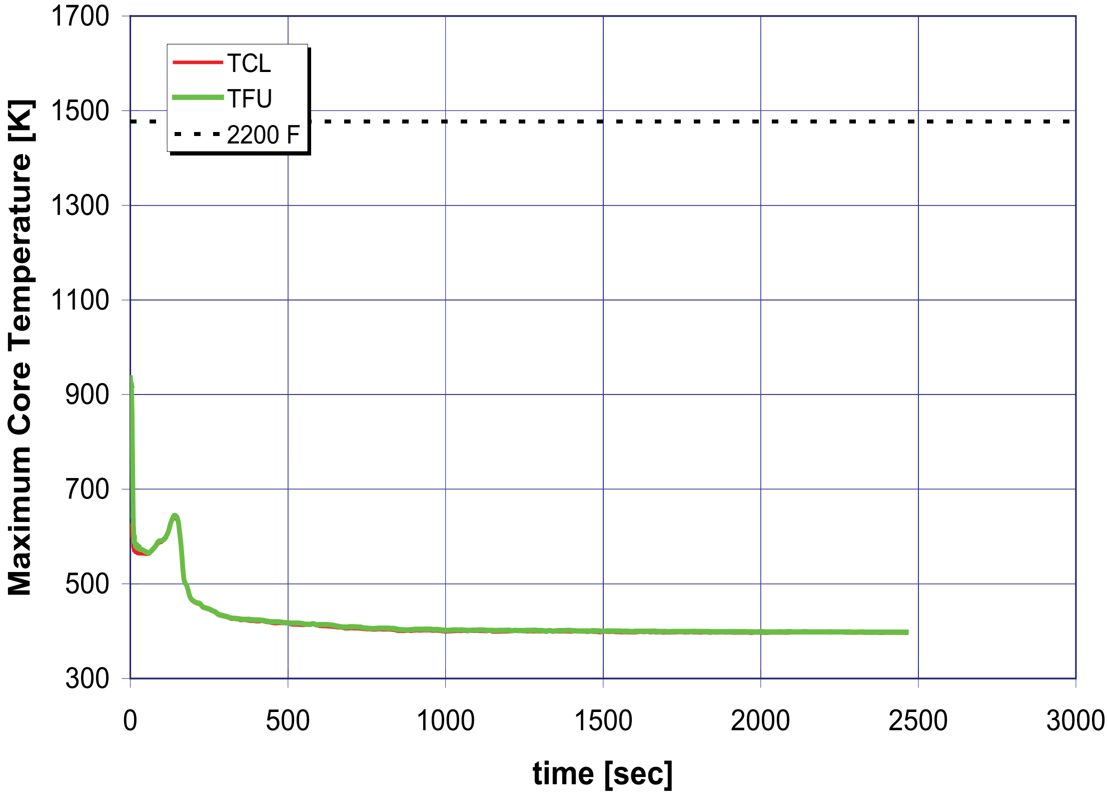
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
0
1000
2000
3000
4000
5000
6000
0
500
1000
1500
2000
2500
3000



Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI
0
50000
100000
150000
200000
250000
300000
350000
400000
450000
0
500
1000
1500
2000
2500
3000

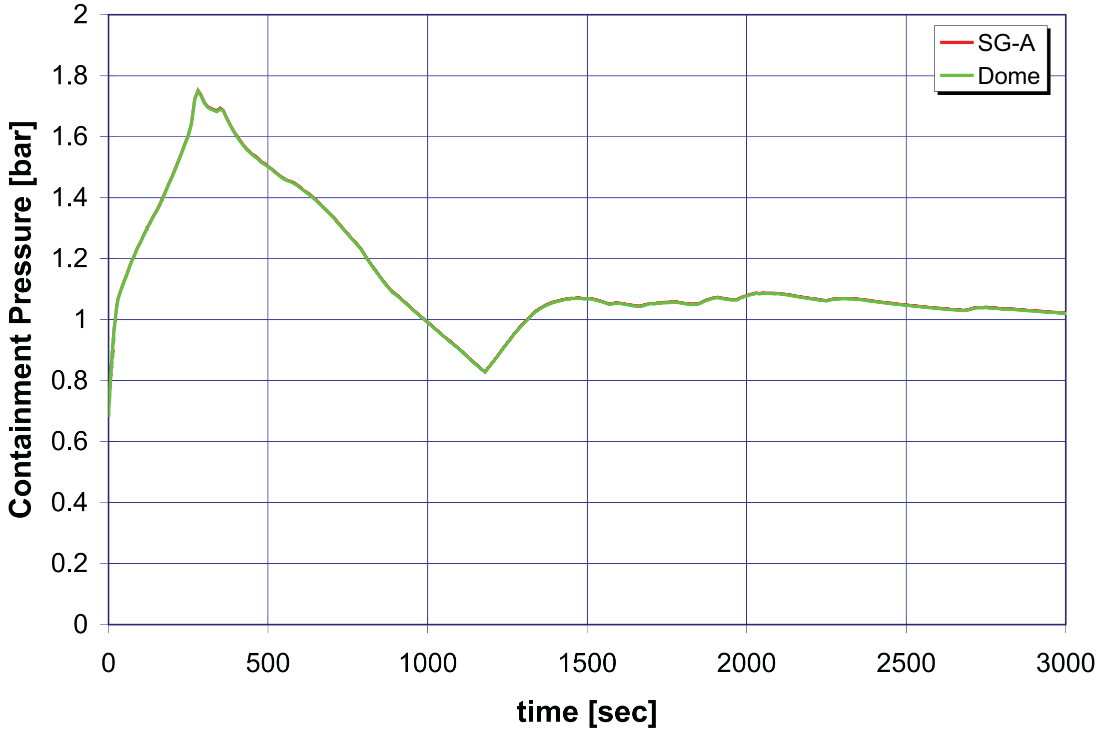

Sump
Cavity
Containment Water Level [m]
time [sec]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
500
1000
1500
2000
2500
3000

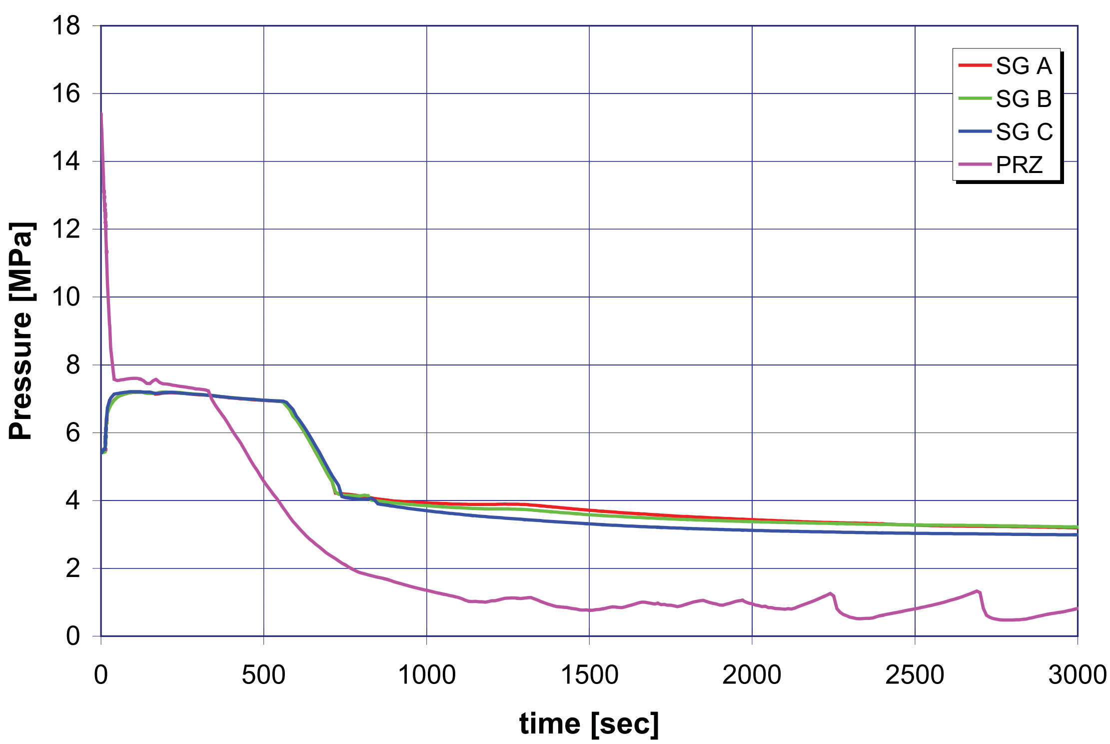

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
20
18
16
14
12
10
8
6
4
2
0
0
500
1000
1500
2000
2500
3000

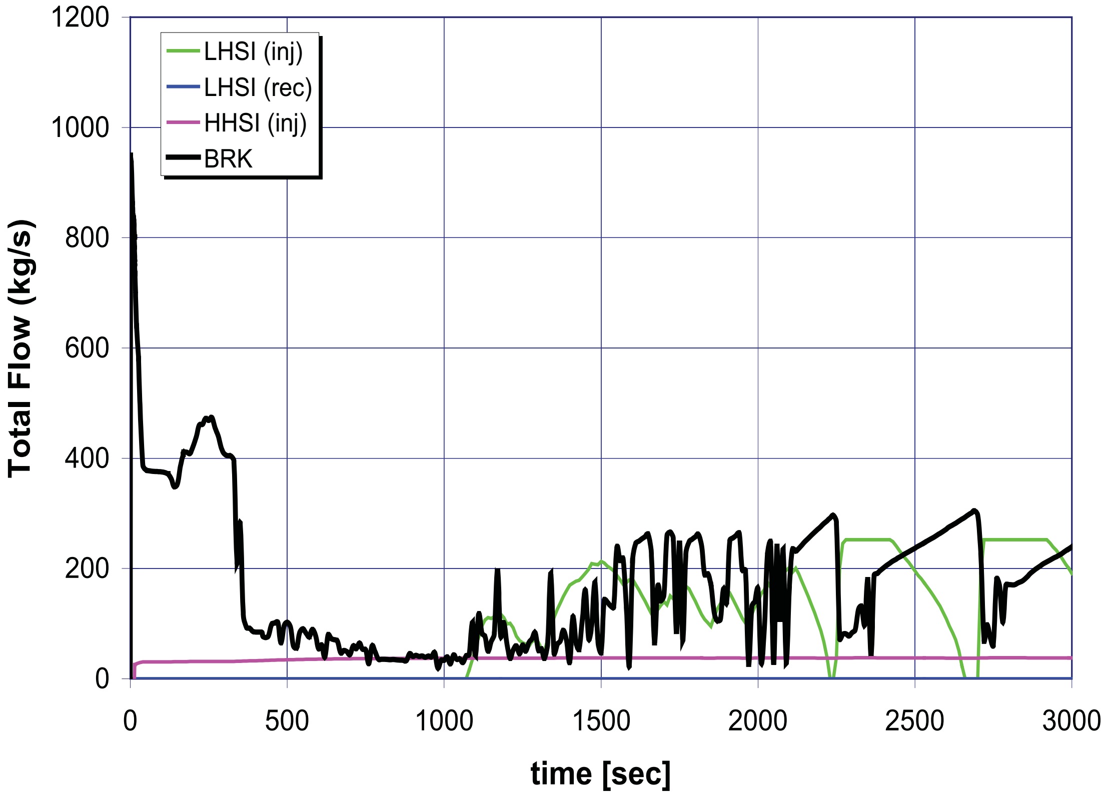


1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F

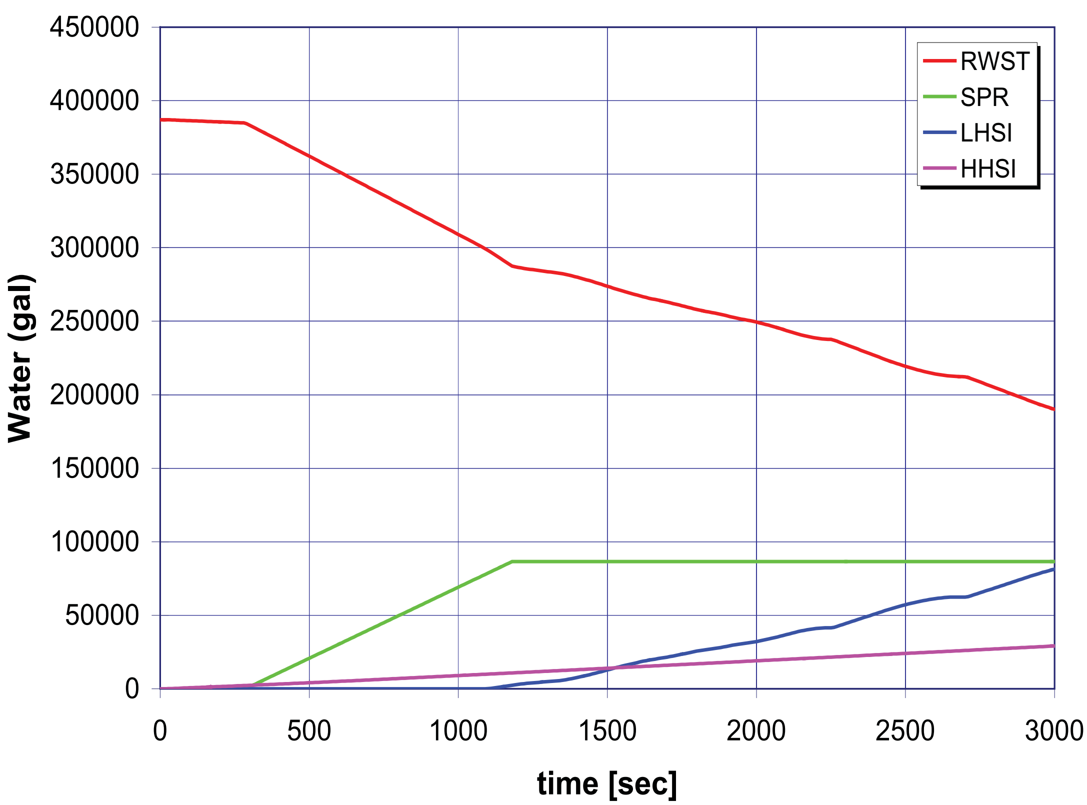
A.6.25 Case 25: 4-Inch Break LOCA, One HHSI, One LHSI, No ACC, without Auxiliary Feedwater



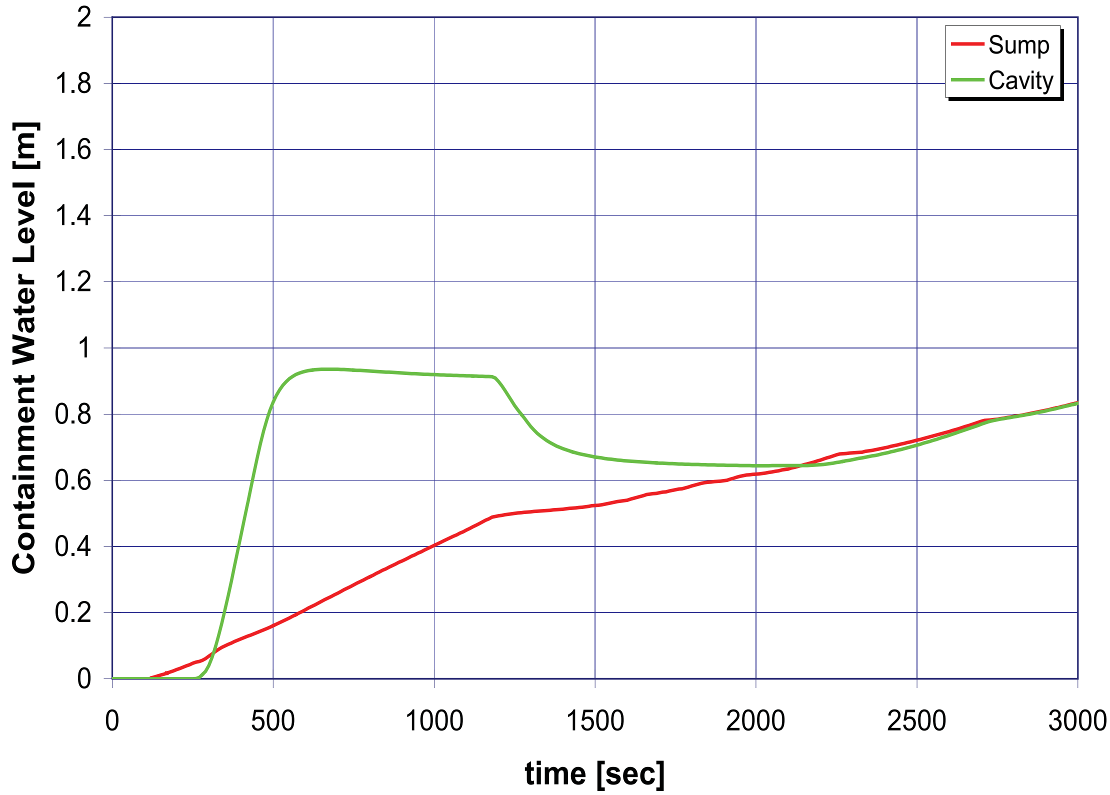




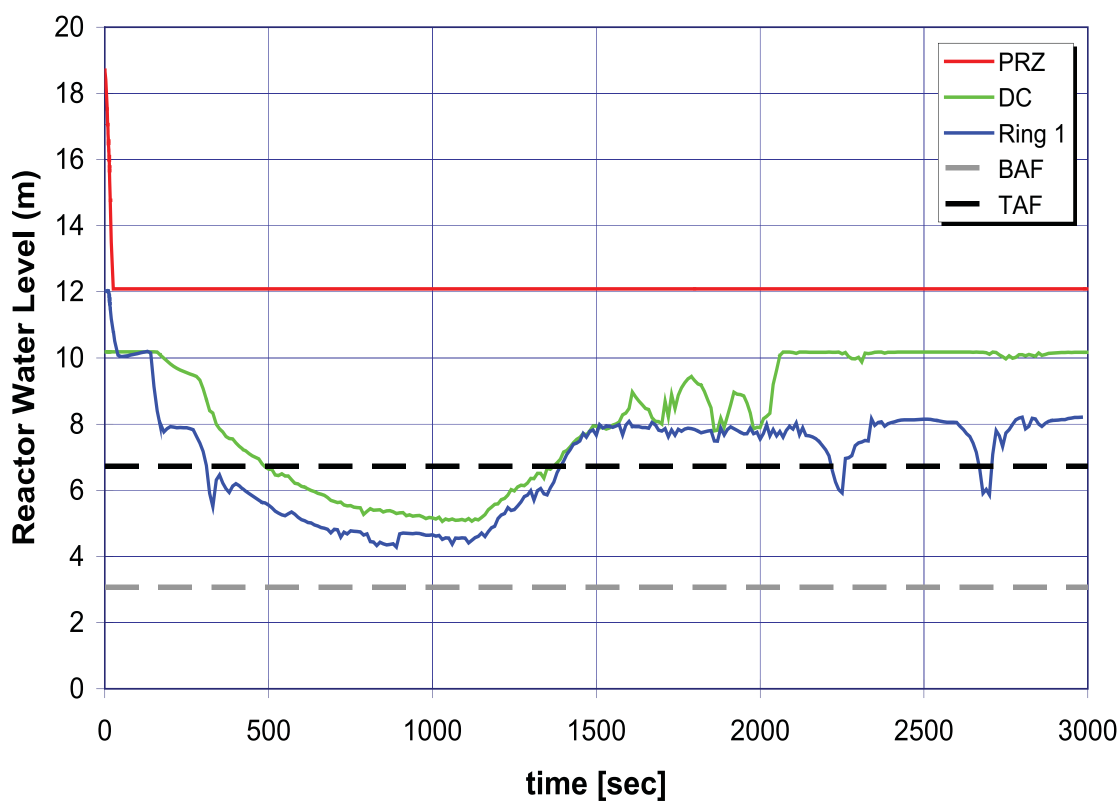
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]



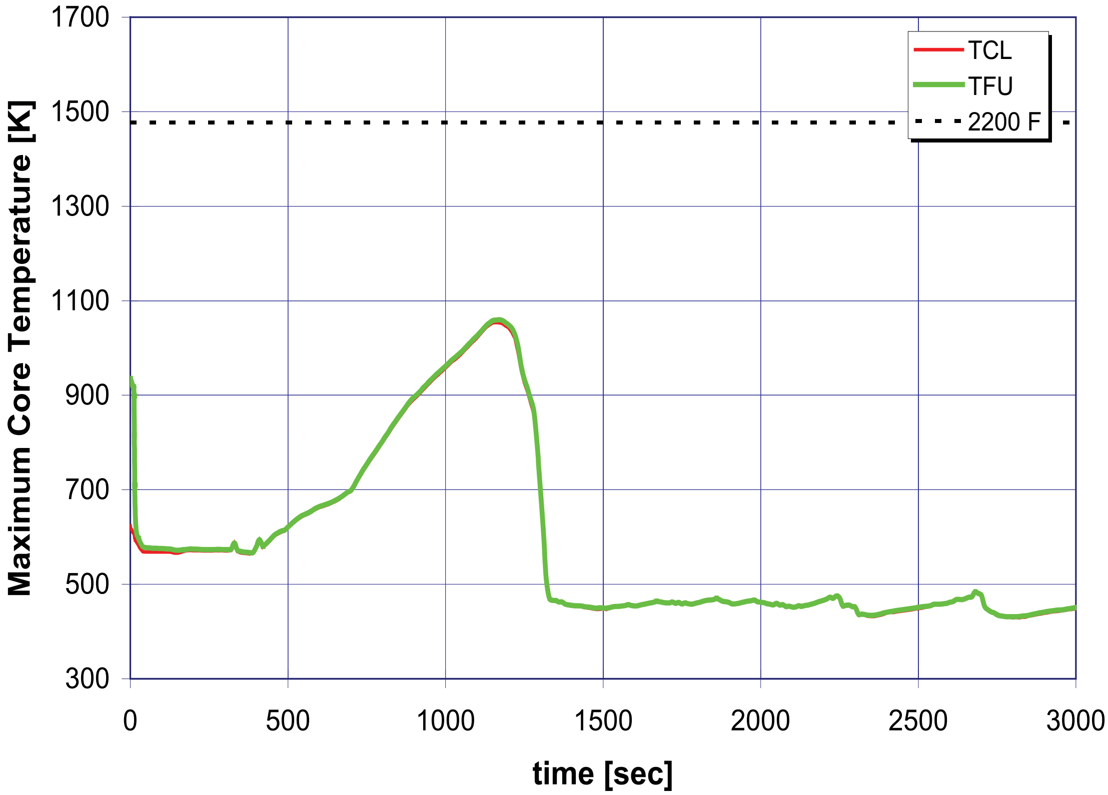
RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]



Containment Water Level [m]
time [sec]
Sump
Cavity
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000

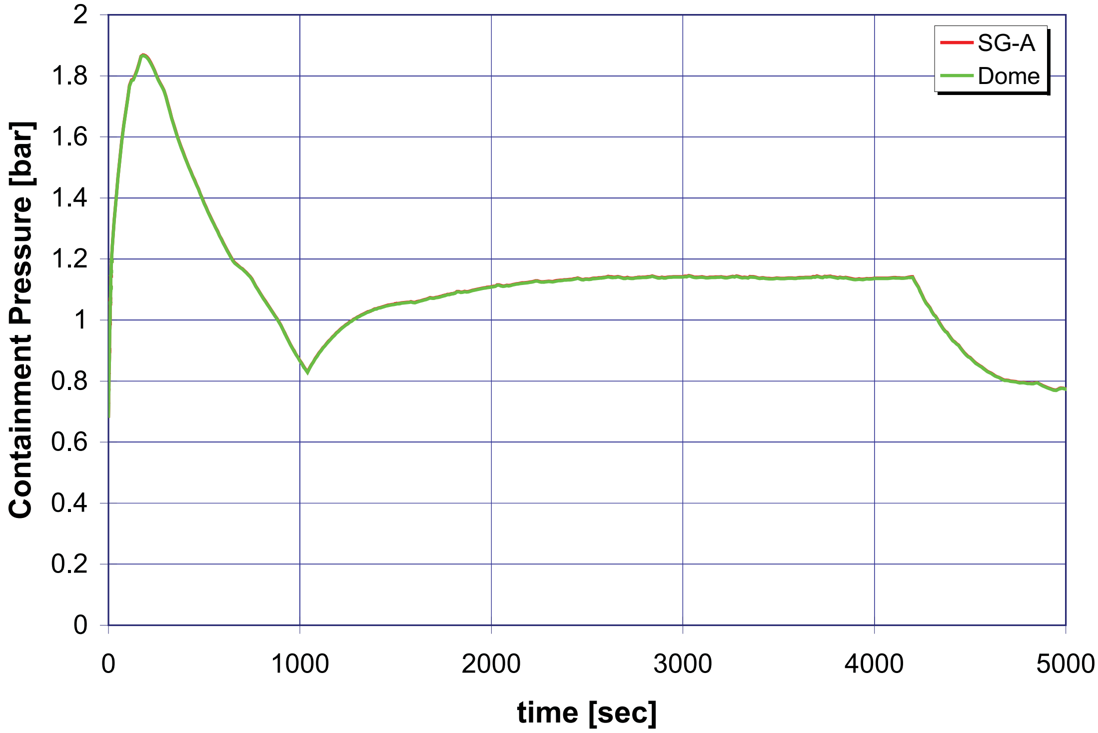


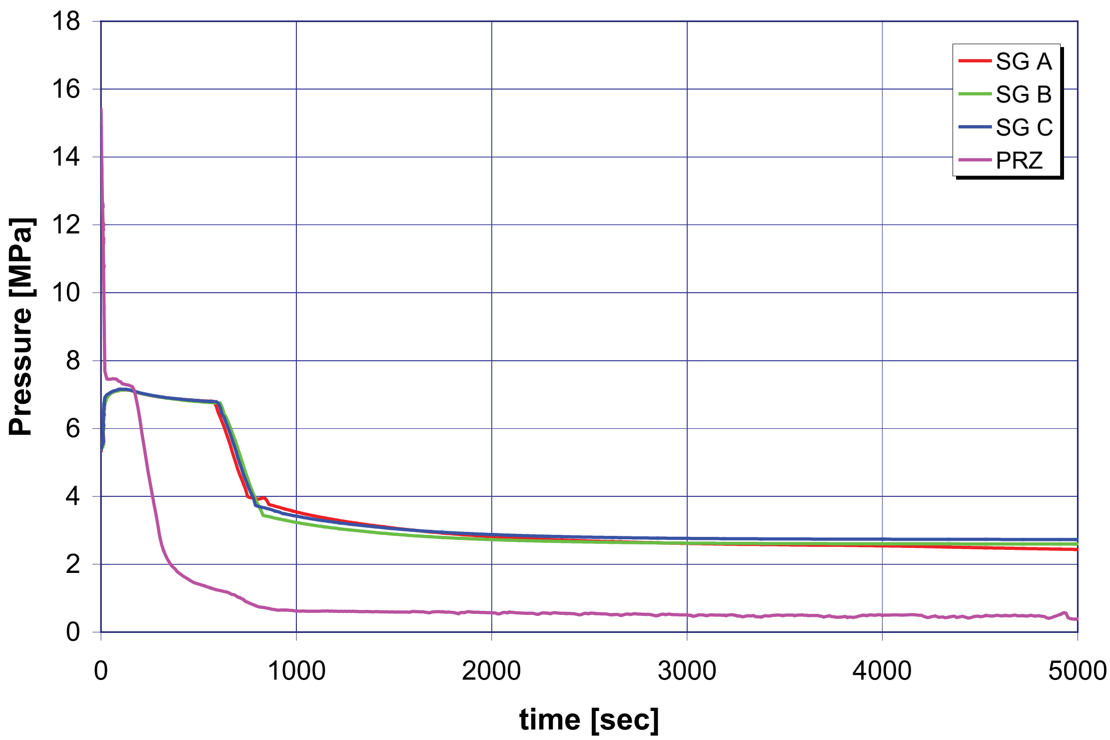
Reactor Water Level (m)
time [sec]
PRZ
DC
Ring 1
BAF
TAF
0
2
4
6
8
10
12
14
16
18
20
0
500
1000
1500
2000
2500
3000

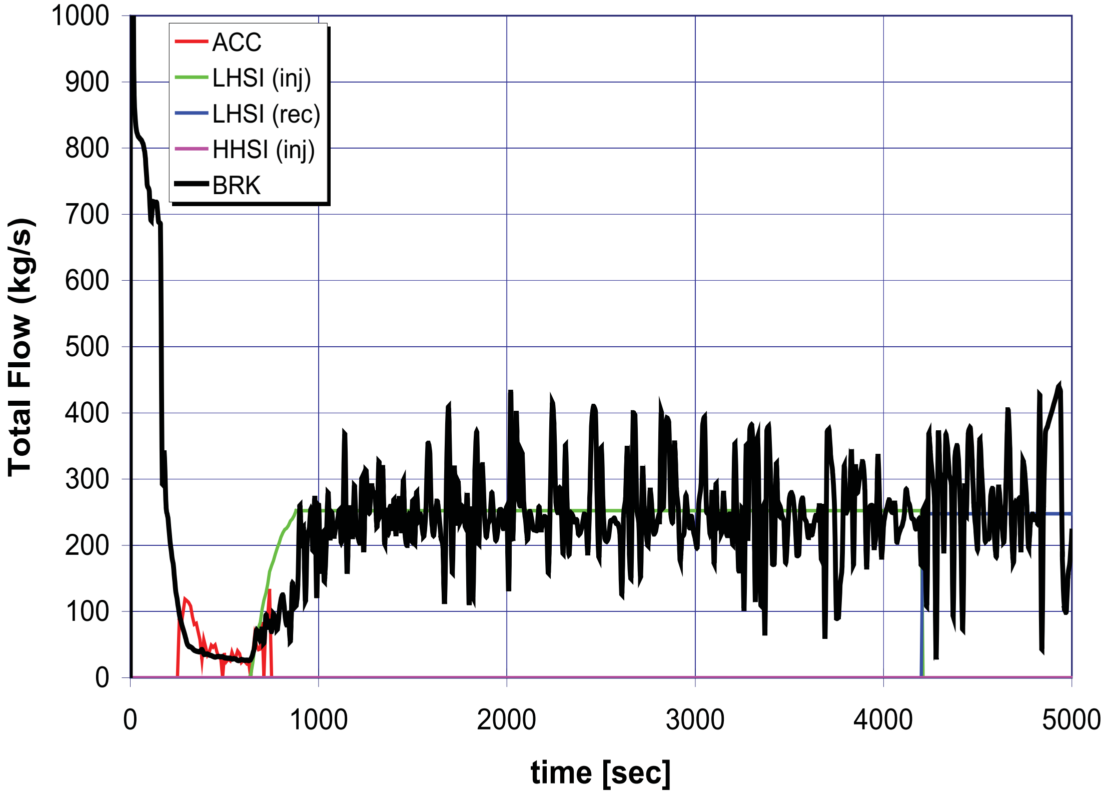


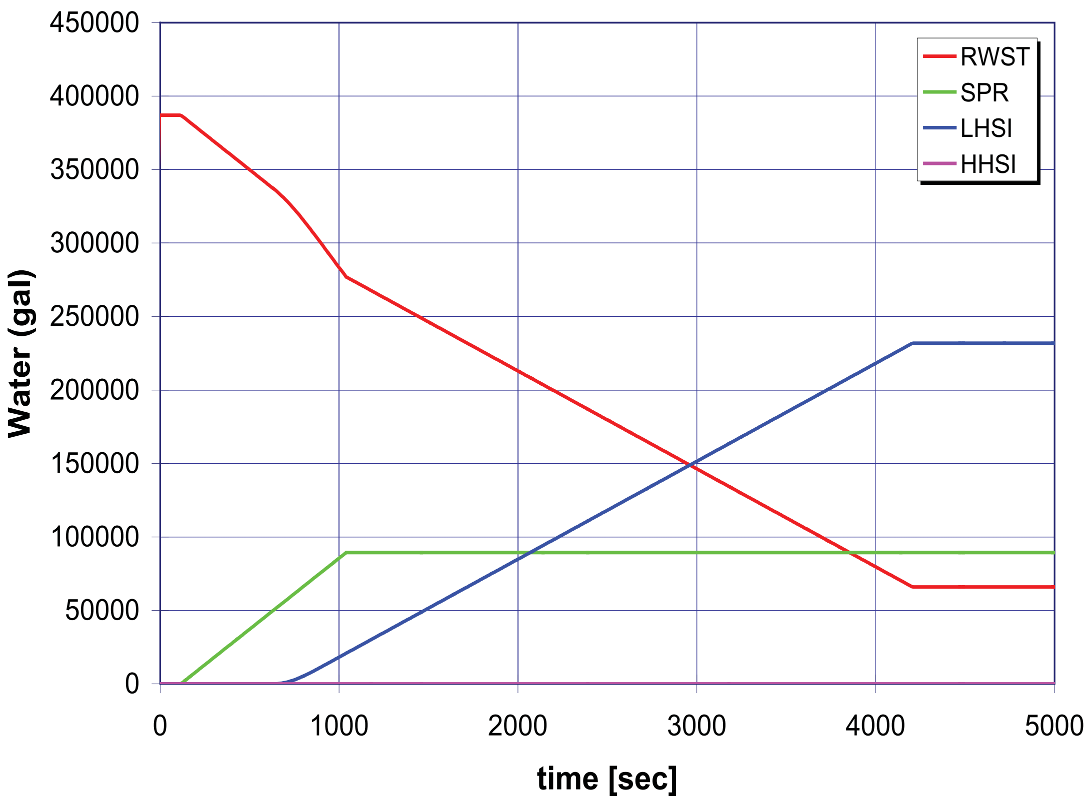
1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000
Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F

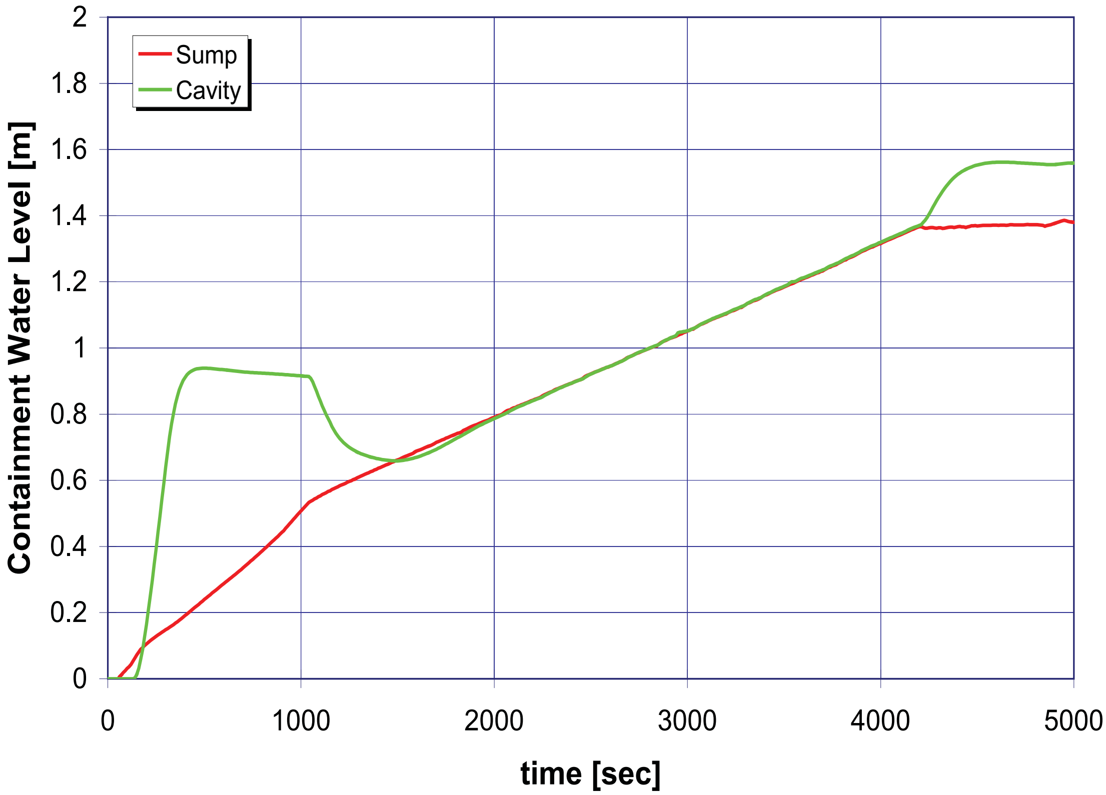
A.6.26 Case 26: 6-Inch Break LOCA, No HHSI, One LHSI, and One ACC, without Auxiliary Feedwater



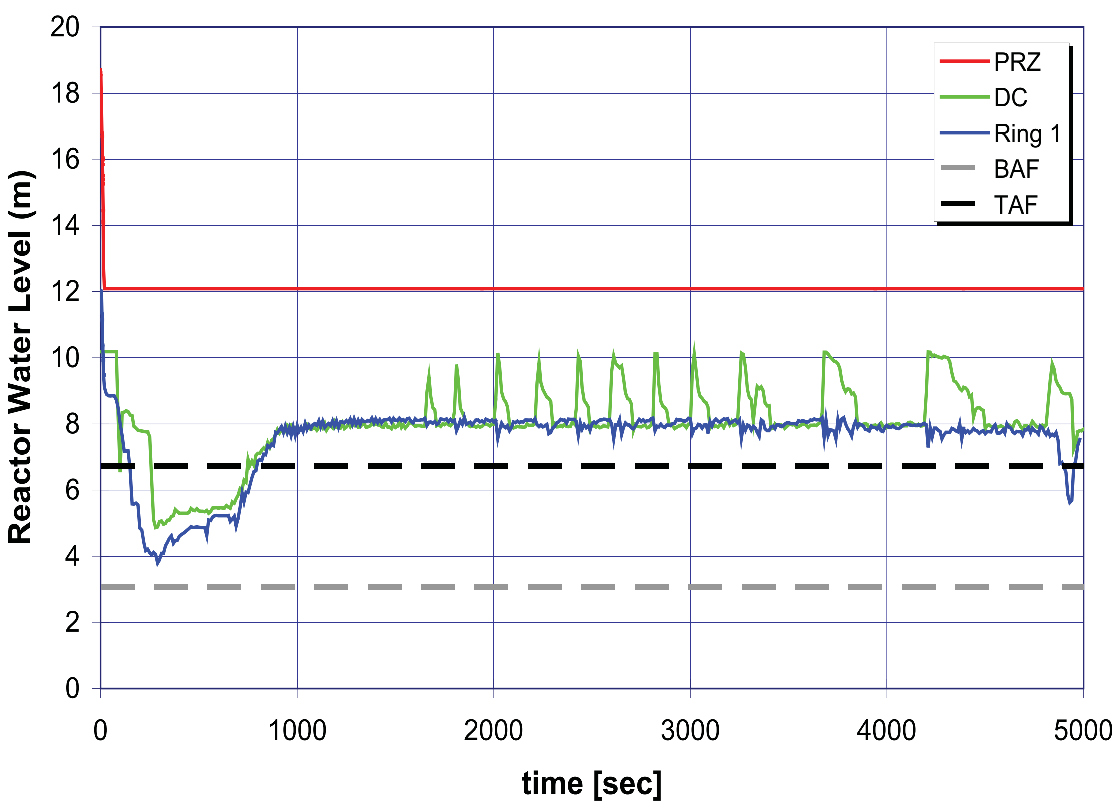



1000
900
800
700
600
500
400
300
200
100
0
0
1000
2000
3000
4000
5000
Total Flow (kg/s)
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK

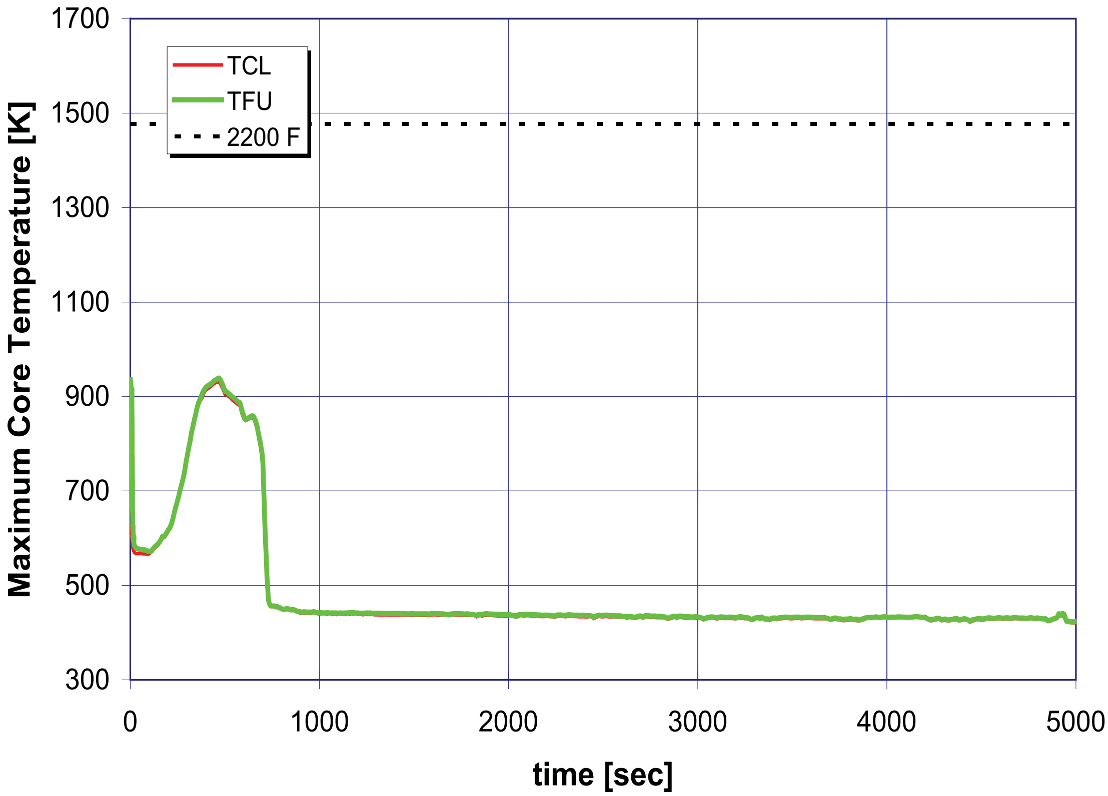

450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
Water (gal)
time [sec]
RWST
SPR
LHSI
HHSI



Containment Water Level [m]
Sump
Cavity
time [sec]
0
1000
2000
3000
4000
5000
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2

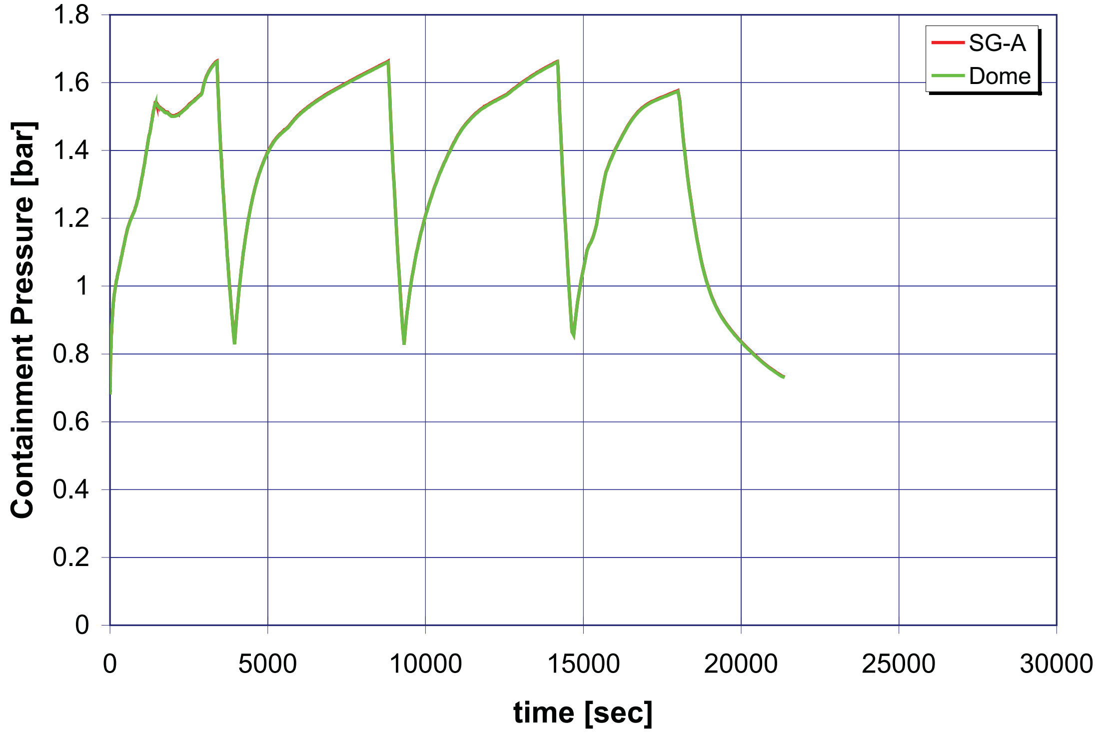


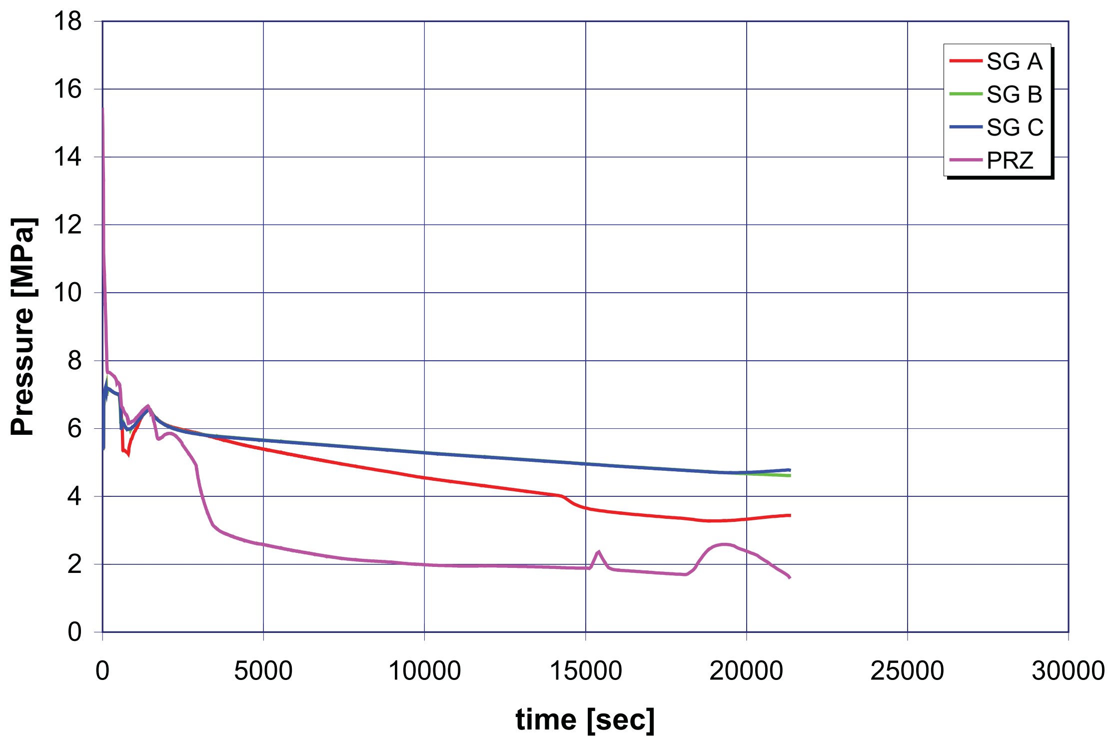
Reactor Water Level (m)
PRZ
DC
Ring 1
BAF
TAF
time [sec]
0
1000
2000
3000
4000
5000
0
2
4
6
8
10
12
14
16
18
20

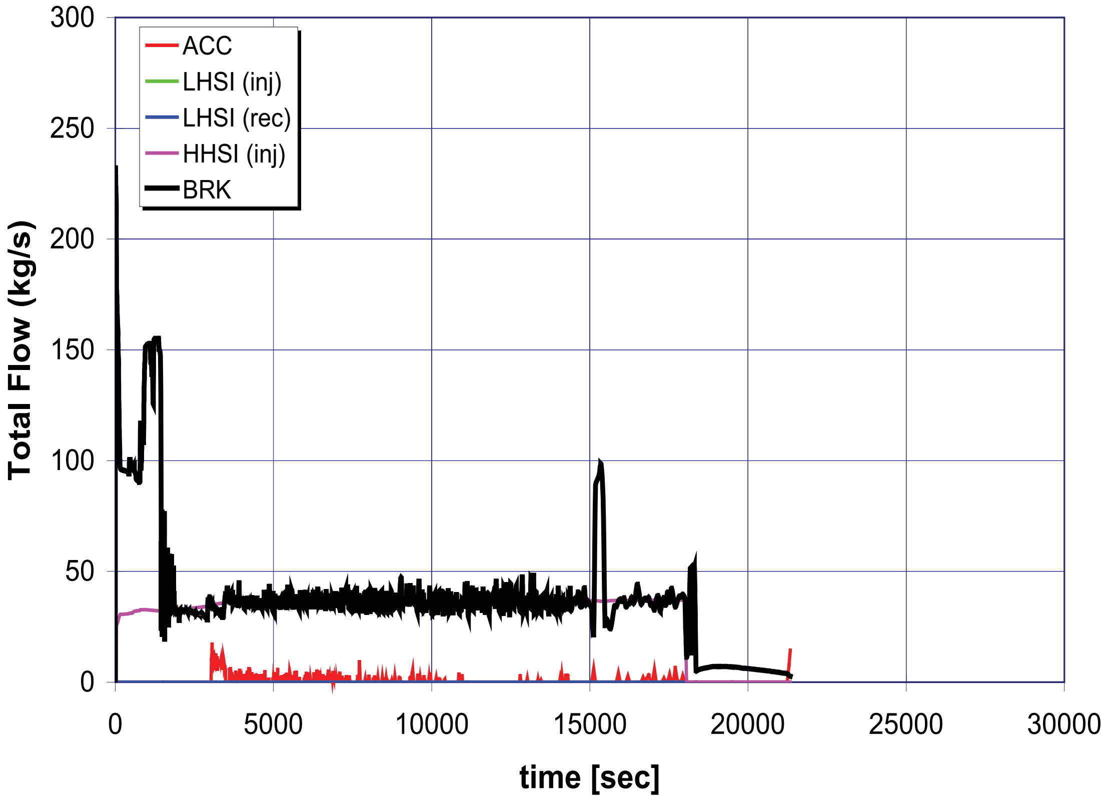


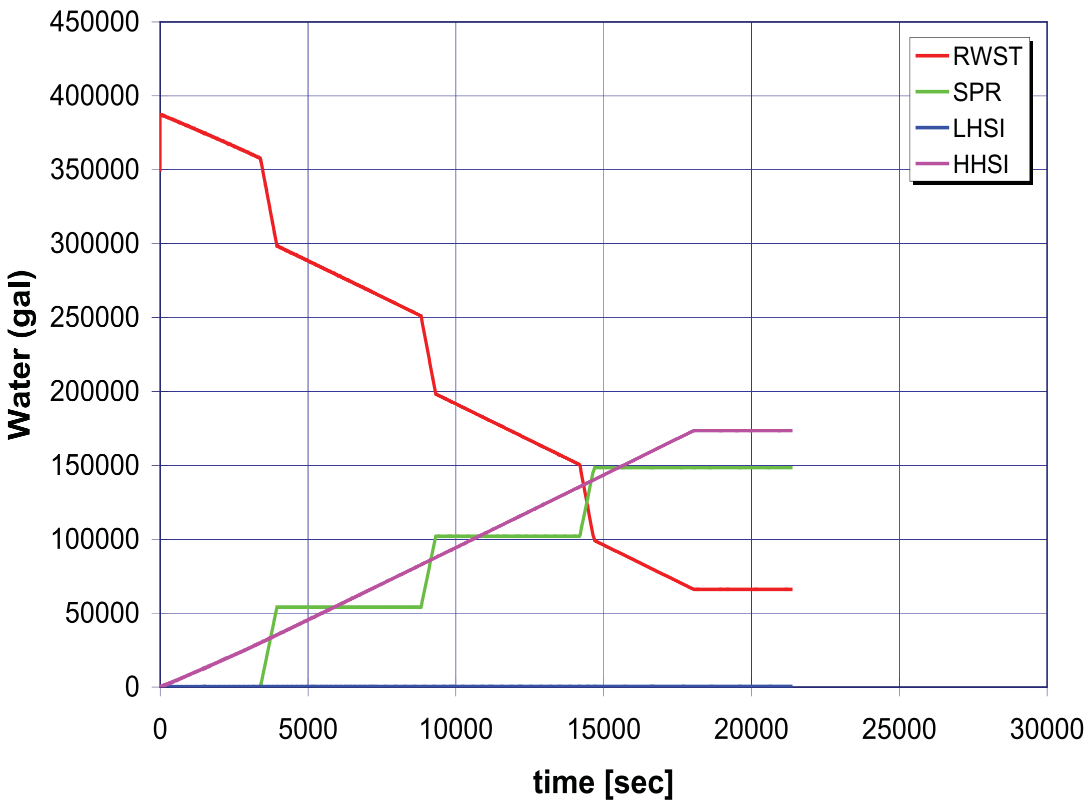
TCL
TFU
2200 F
Maximum Core Temperature [K]
time [sec]
1700
1500
1300
1100
900
700
500
300
0
1000
2000
3000
4000
5000

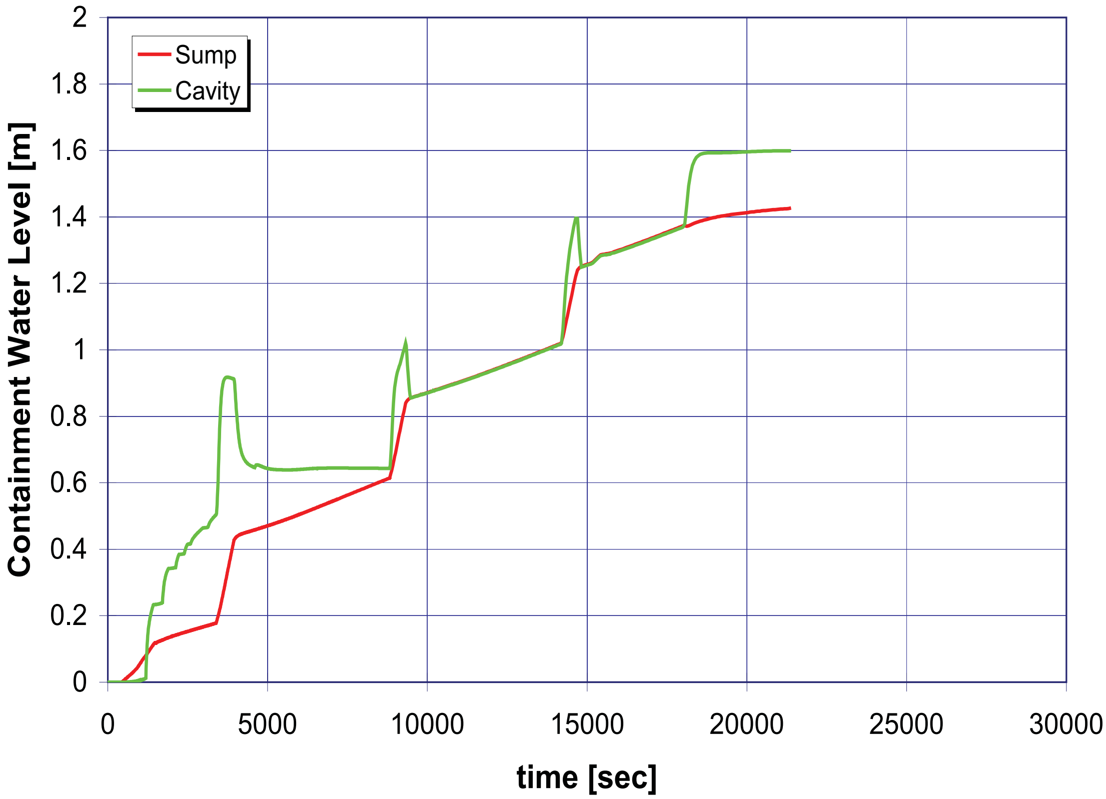
A.6.27 Case 27: 2-Inch Break LOCA, One HHSI, No LHSI, and One ACC, without Auxiliary Feedwater

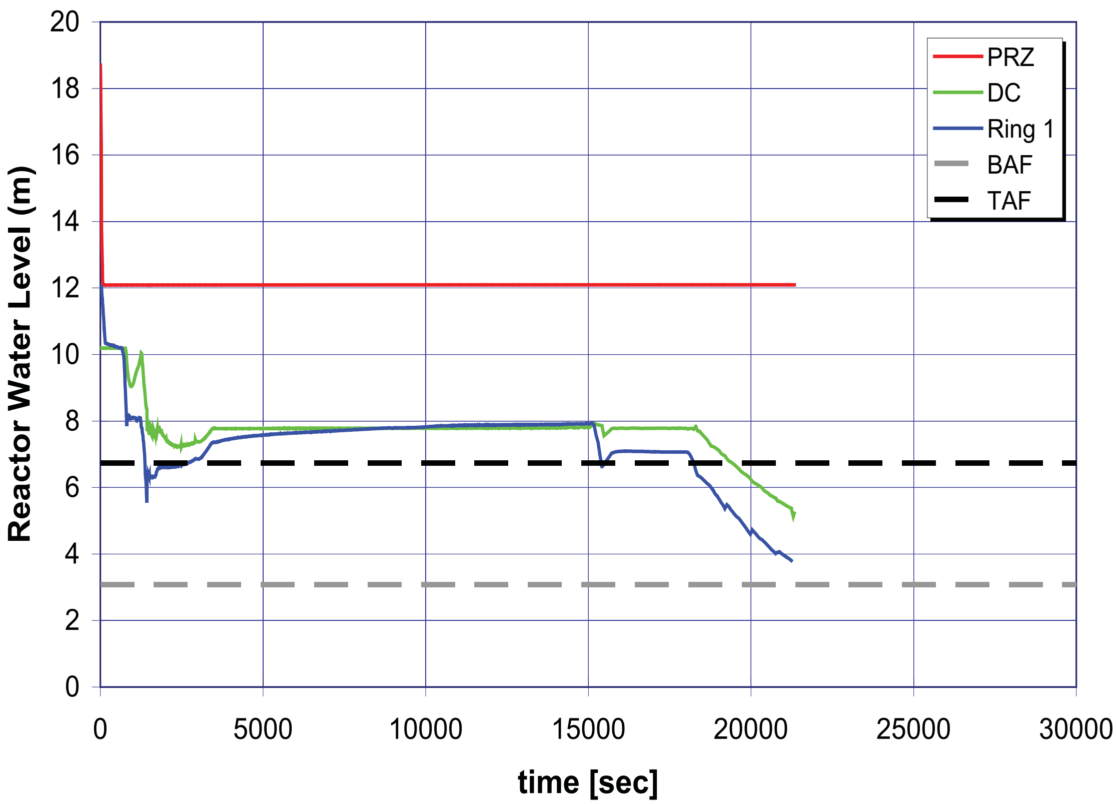


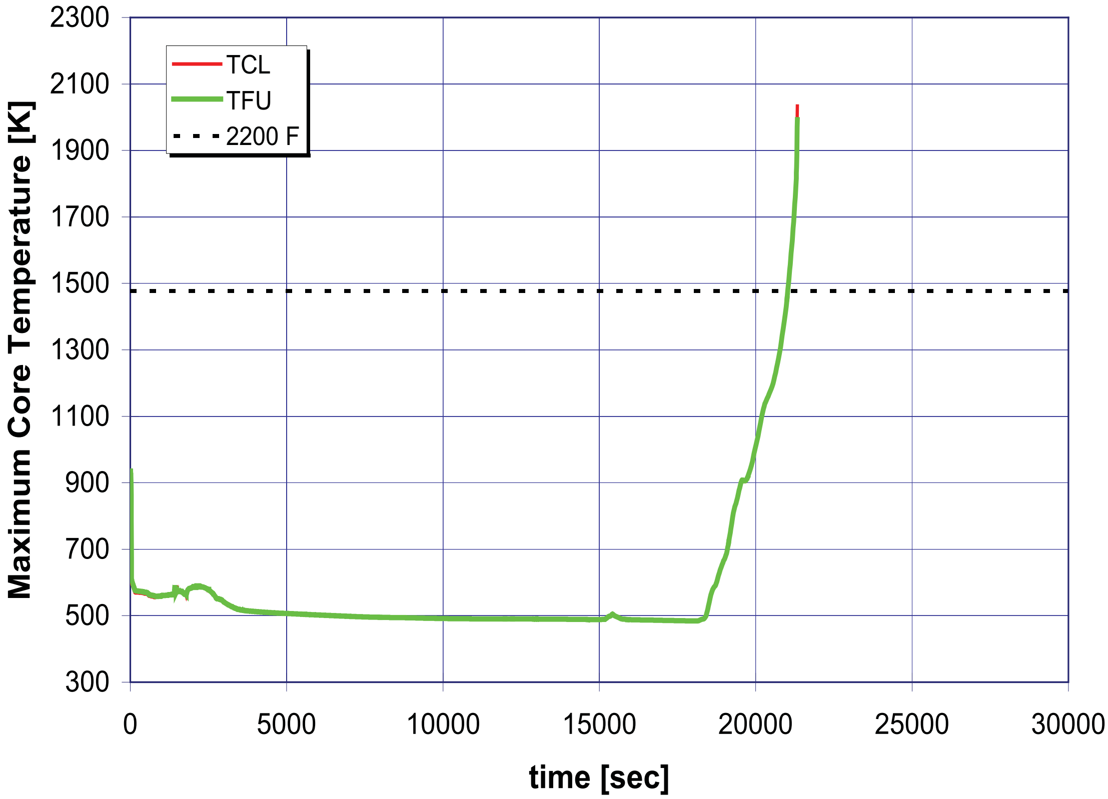



ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]
300
250
200
150
100
50
0
0
5000
10000
15000
20000
25000
30000

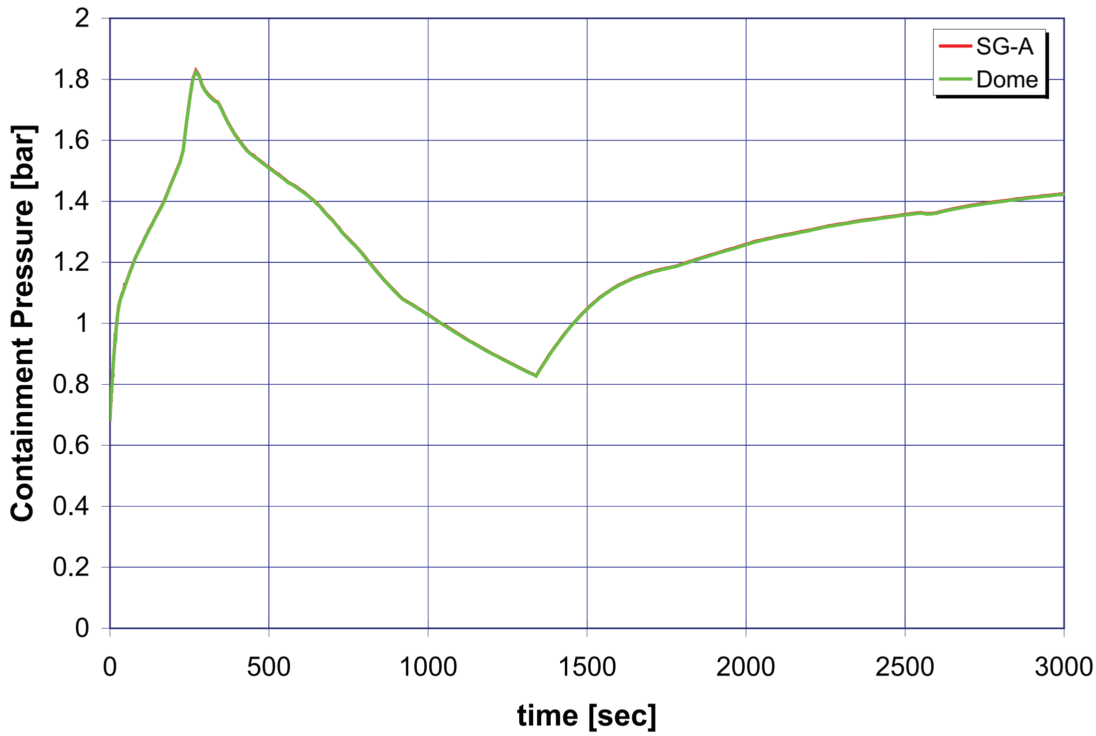

RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
5000
10000
15000
20000
25000
30000

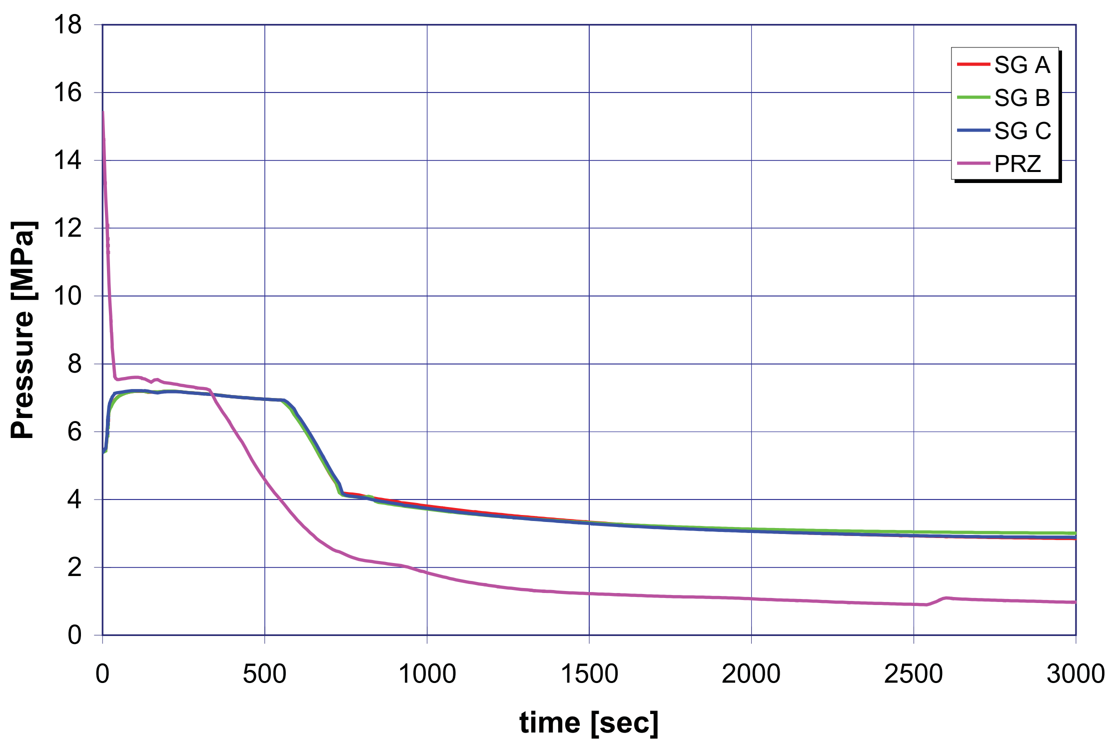

Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000

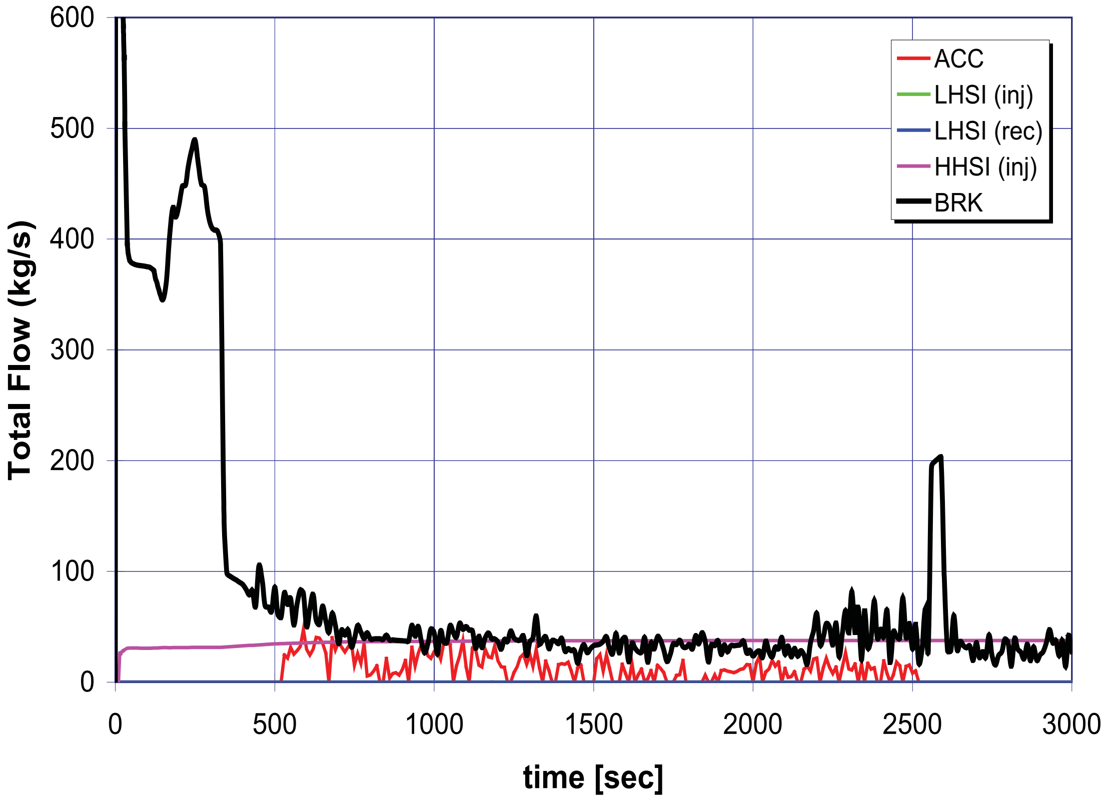

PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
5000
10000
15000
20000
25000
30000


TCL
TFU
2200 F
Maximum Core Temperature [K]
2300
2100
1900
1700
1500
1300
1100
900
700
500
300
0
5000
10000
15000
20000
25000
30000
time [sec]

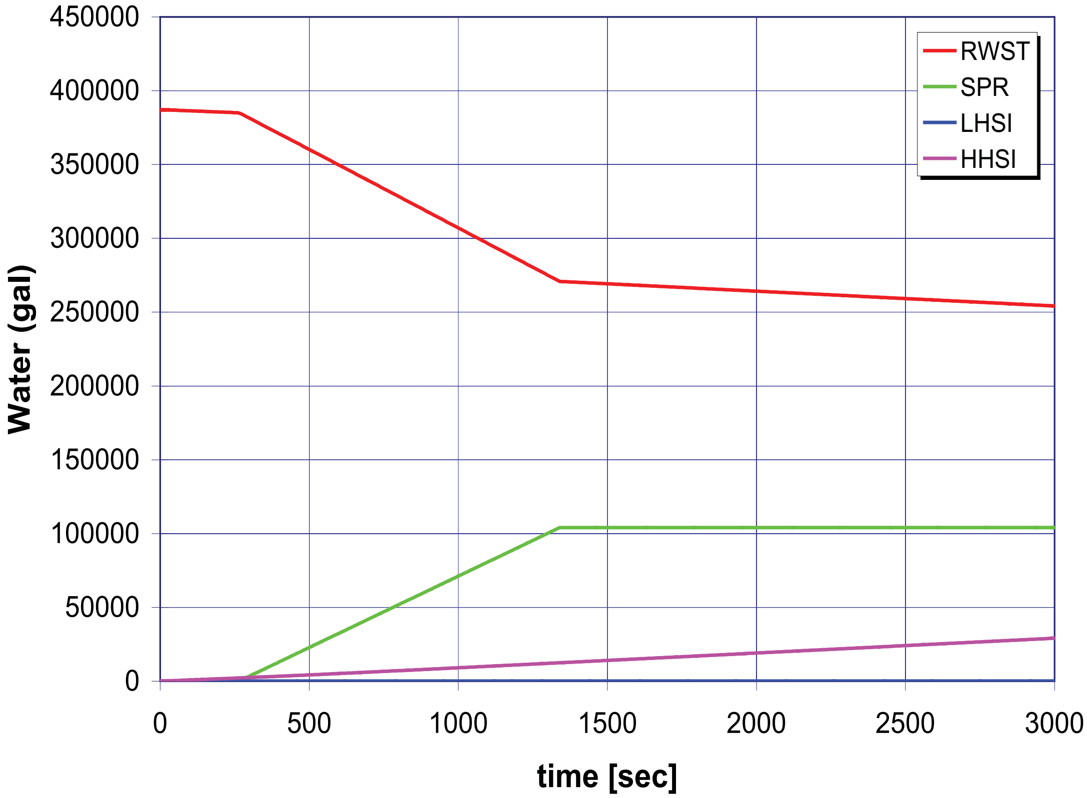
A.6.28 Case 28: 4-Inch Break LOCA, One HHSI, No LHSI, and One ACC, without Auxiliary Feedwater



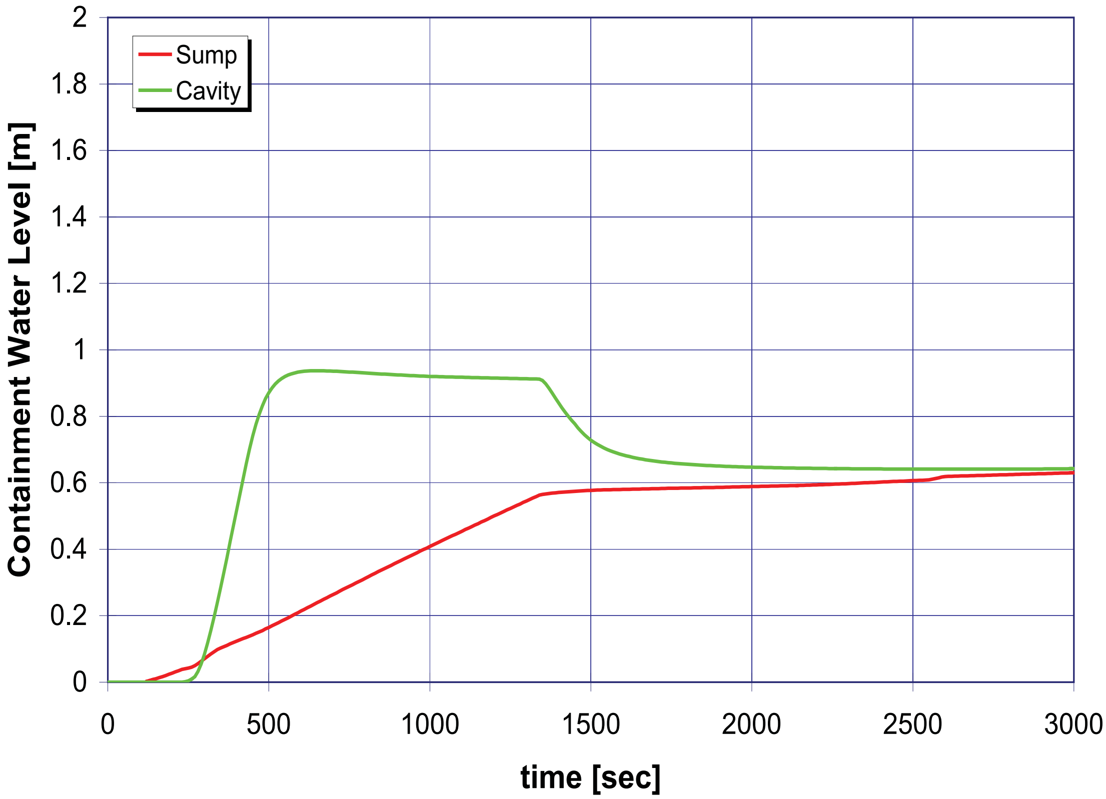




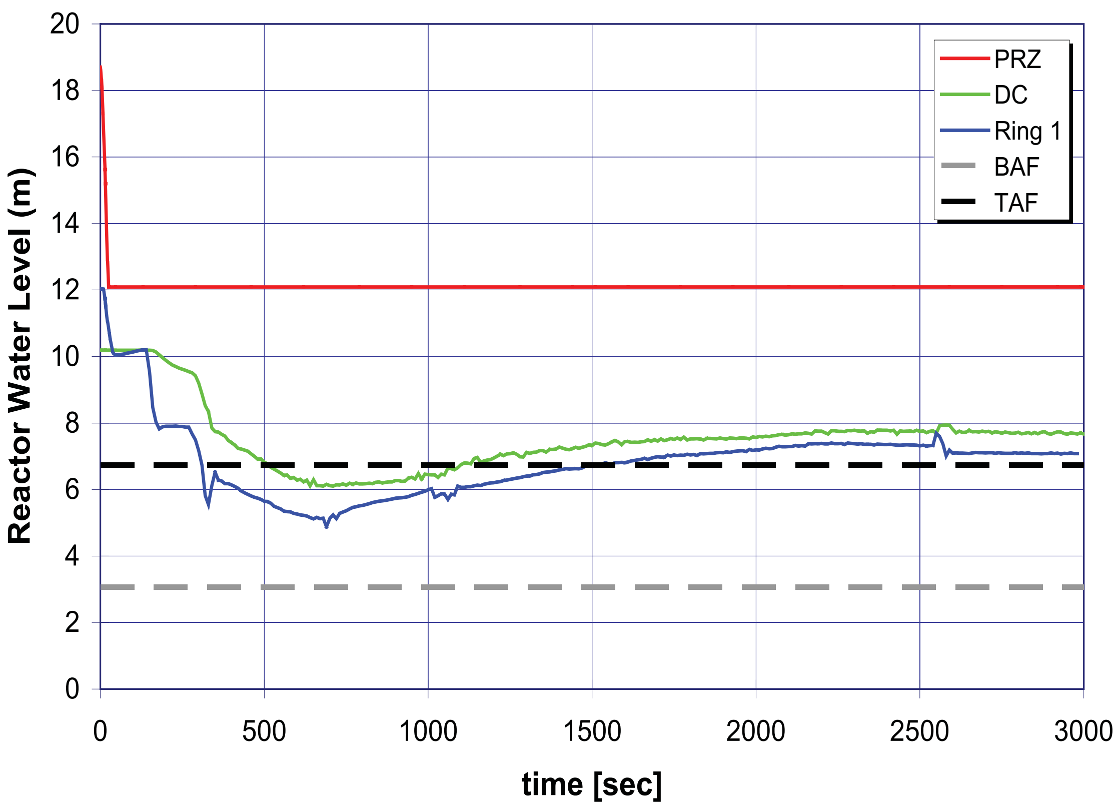
600
500
400
300
200
100
0
Total Flow (kg/s)
0
500
1000
1500
2000
2500
3000
time [sec]
ACC
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK



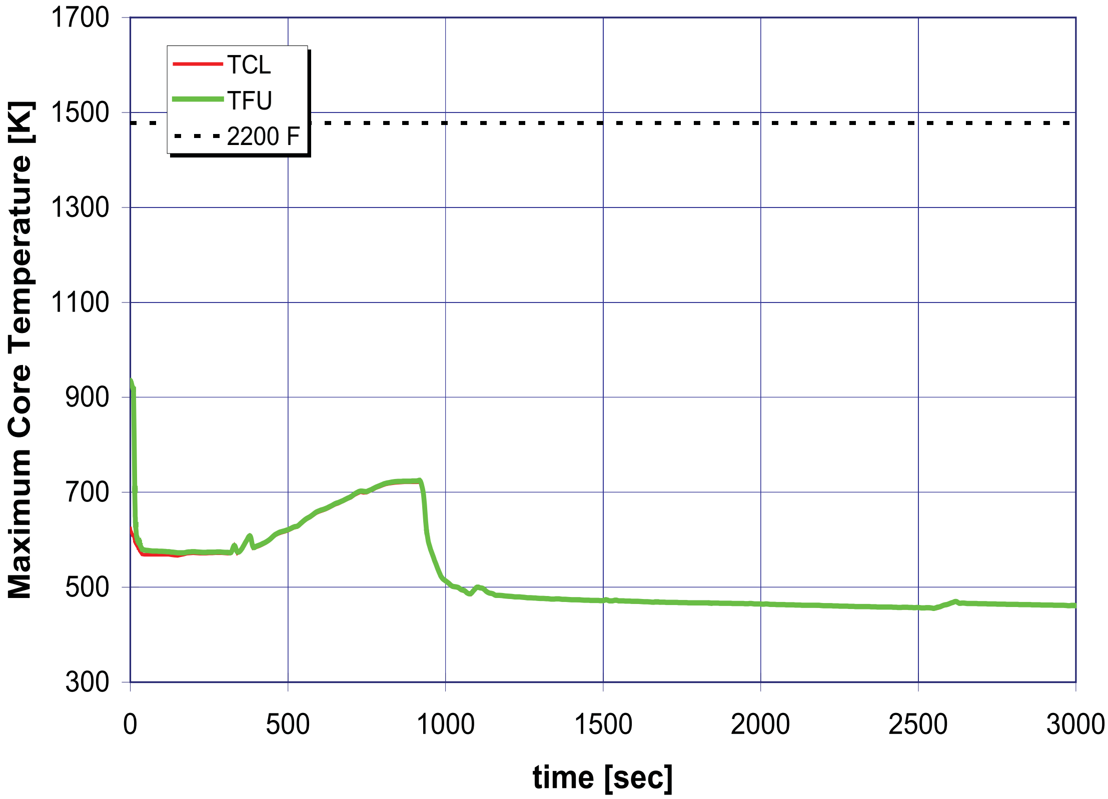
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
Water (gal)
0
500
1000
1500
2000
2500
3000
time [sec]
RWST
SPR
LHSI
HHSI



Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000

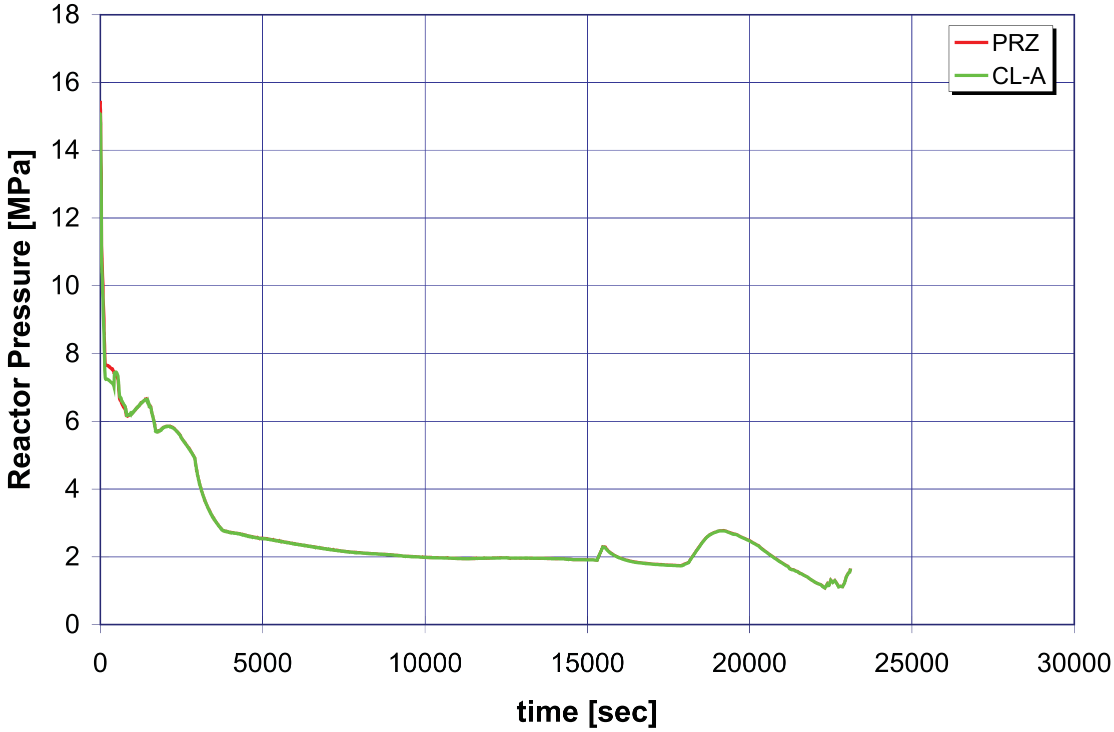


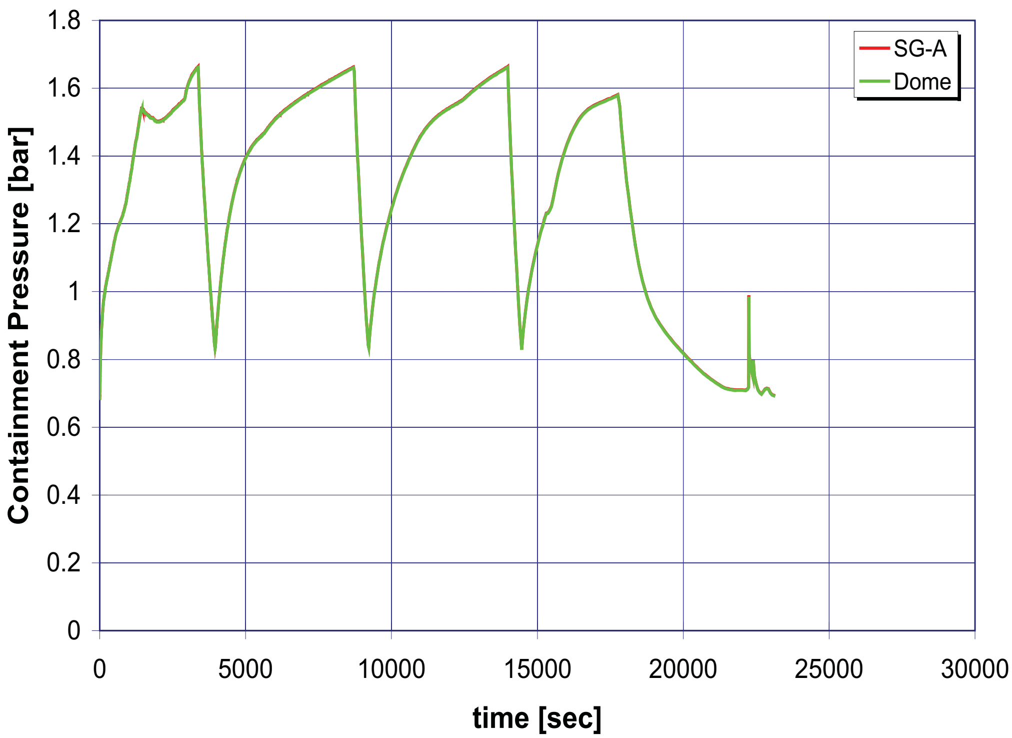
PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
500
1000
1500
2000
2500
3000

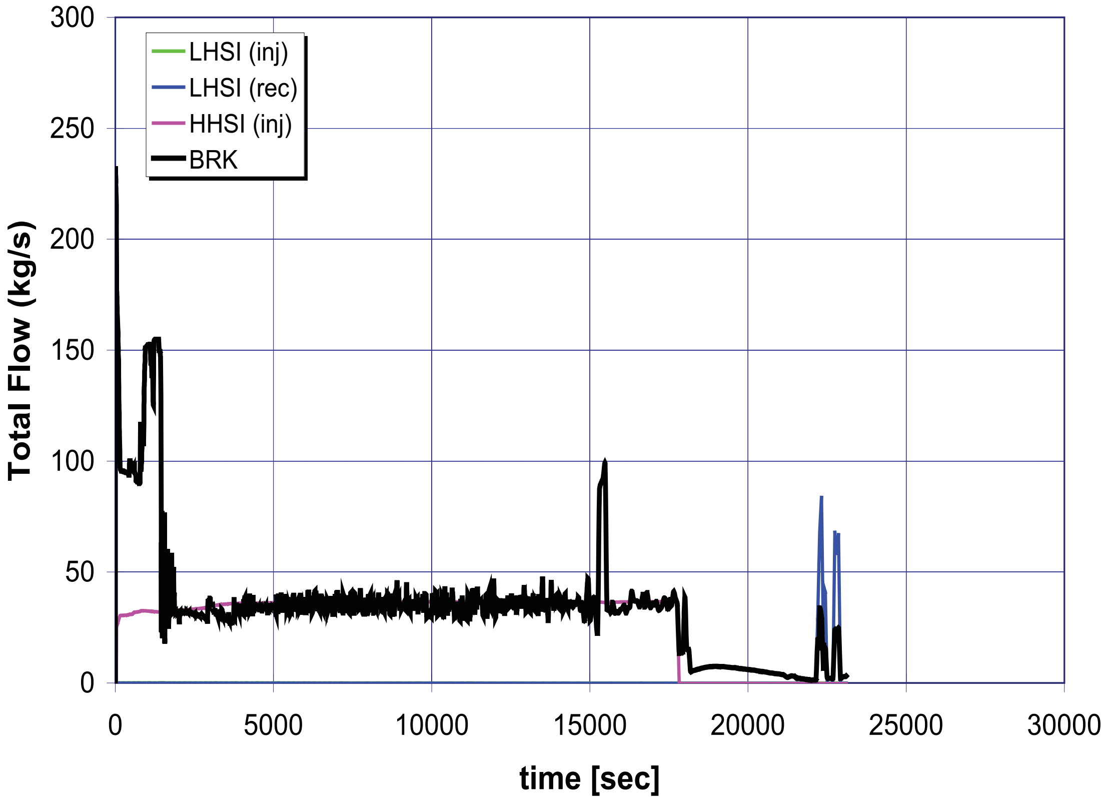


Maximum Core Temperature [K]
time [sec]
TCL
TFU
2200 F
1700
1500
1300
1100
900
700
500
300
0
500
1000
1500
2000
2500
3000

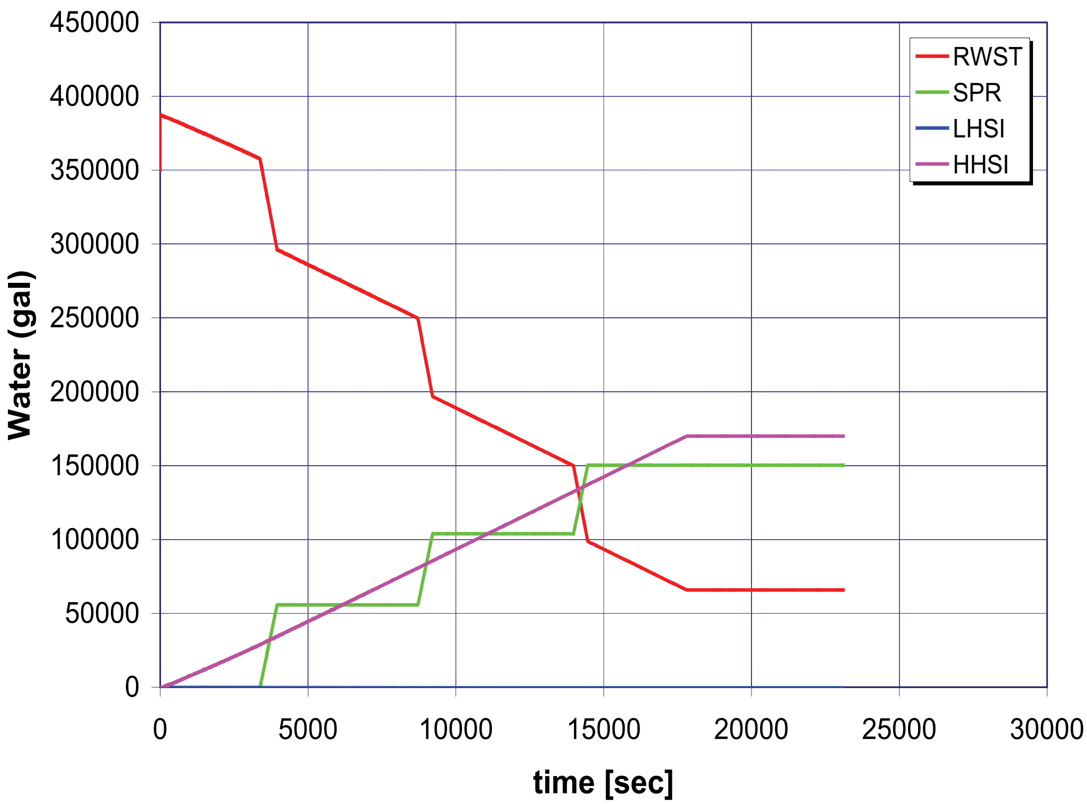
A.6.29 Case 29: 2-Inch Break LOCA, One HHSI, One LHSI, and No ACC, without Auxiliary Feedwater



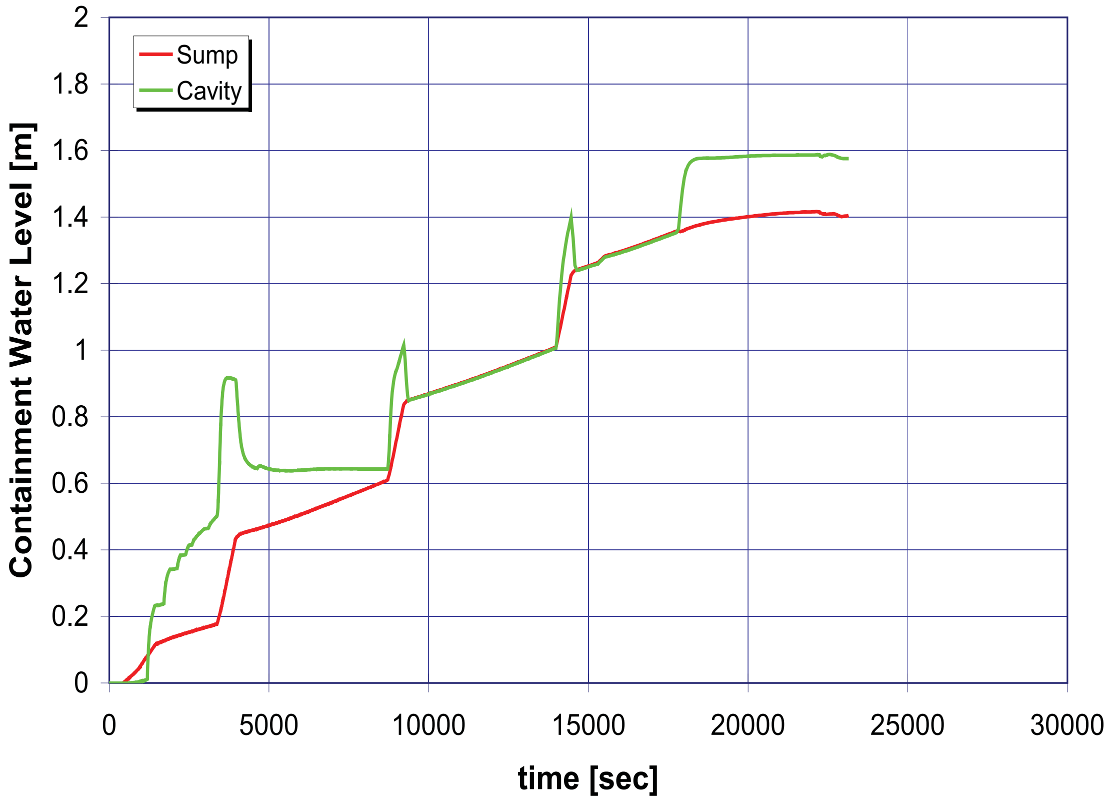


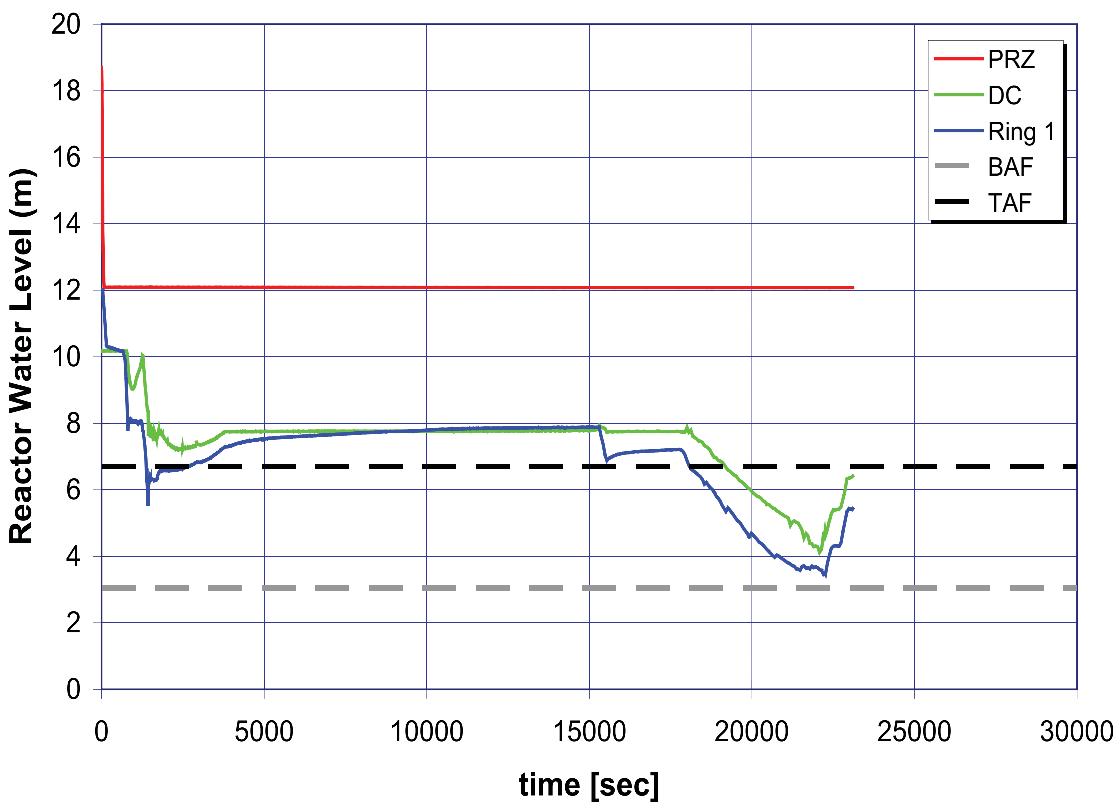


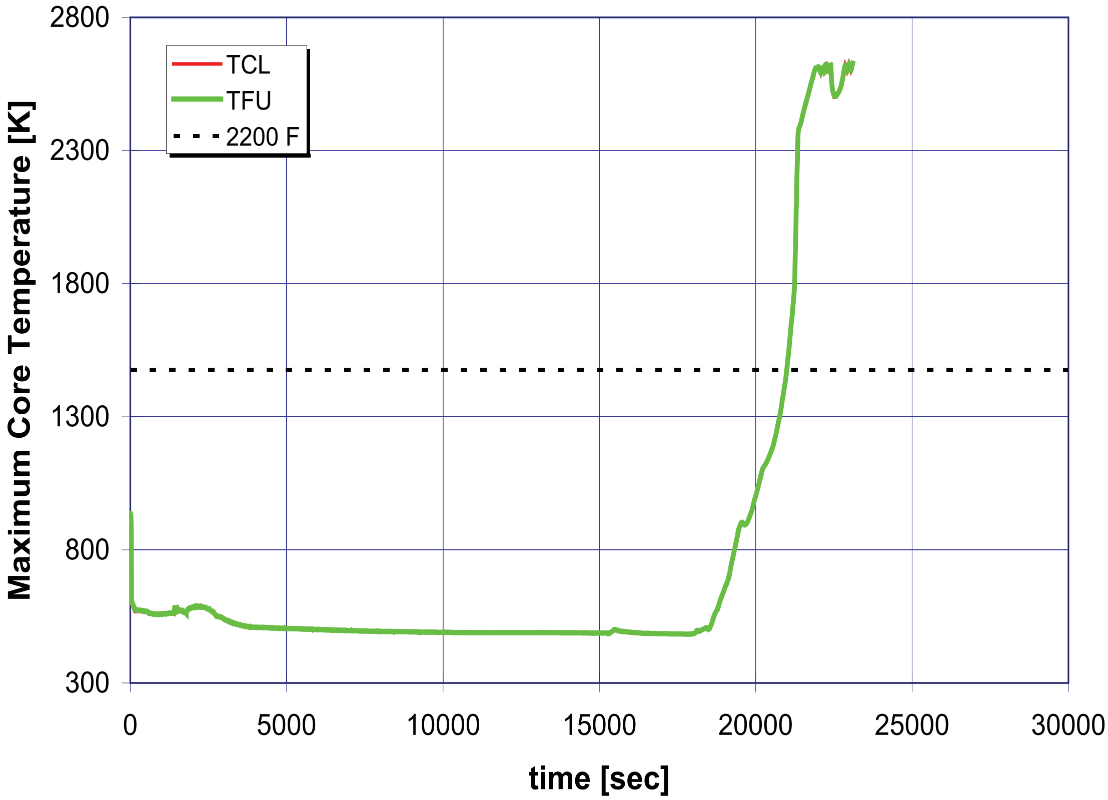
LHSI (inj)
LHSI (rec)
HHSI (inj)
BRK
Total Flow (kg/s)
time [sec]



RWST
SPR
LHSI
HHSI
Water (gal)
time [sec]


Sump
Cavity
Containment Water Level [m]
time [sec]
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
5000
10000
15000
20000
25000
30000


PRZ
DC
Ring 1
BAF
TAF
Reactor Water Level (m)
time [sec]
0
2
4
6
8
10
12
14
16
18
20
0
5000
10000
15000
20000
25000
30000


TCL
TFU
2200 F
Maximum Core Temperature [K]
2800
2300
1800
1300
800
300
0
5000
10000
15000
20000
25000
30000
time [sec]

A.7 REFERENCES

(NRC, 2003) U.S. Nuclear Regulatory Commission, “Virginia Electric and Power Company Docket No. 50-280 Surry Power Station, Unit No. 1 Renewed Facility Operating License,” Appendix A, “Technical Specifications,” March 2003. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML0529103580)

APPENDIX B

PEACH BOTTOM MELCOR ANALYSES

B.1 Inadvertent Open Relief Valve Success Criteria

Analysis Summary

Table 1 and Table 2 below provide results for this portion of the analysis.

Table 1 Peach Bottom Inadvertent Open SRV Results

<table>
<tr><th>Case</th><th>RCIC</th><th>HPCI</th><th>CRD</th><th>LPCI</th><th>LPCS</th><th>ac/dc</th><th>FW, SPC, ADS</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td>Yes</td><td>No</td><td rowspan="2">No</td><td rowspan="7">Yes</td><td rowspan="7">No</td><td rowspan="7">ac/dc</td><td rowspan="5">No</td><td>No</td><td>No</td></tr>
<tr><td>2</td><td rowspan="6">No</td><td>Yes</td><td>No</td><td>No</td></tr>
<tr><td>3</td><td rowspan="5">No</td><td>1 at t = 0 and 2 at t = 10 min</td><td>0.41</td><td>No</td></tr>
<tr><td>4</td><td rowspan="3">1 at t = 0 and 2 at 20 min after SCRAM</td><td>0.37</td><td>No</td></tr>
<tr><td>4a[1]</td><td>0.29</td><td>No</td></tr>
<tr><td>4b[1]</td><td>FW</td><td>No</td><td>No</td></tr>
<tr><td>5</td><td>No</td><td>No</td><td>0.32</td><td>No</td></tr>
</table>

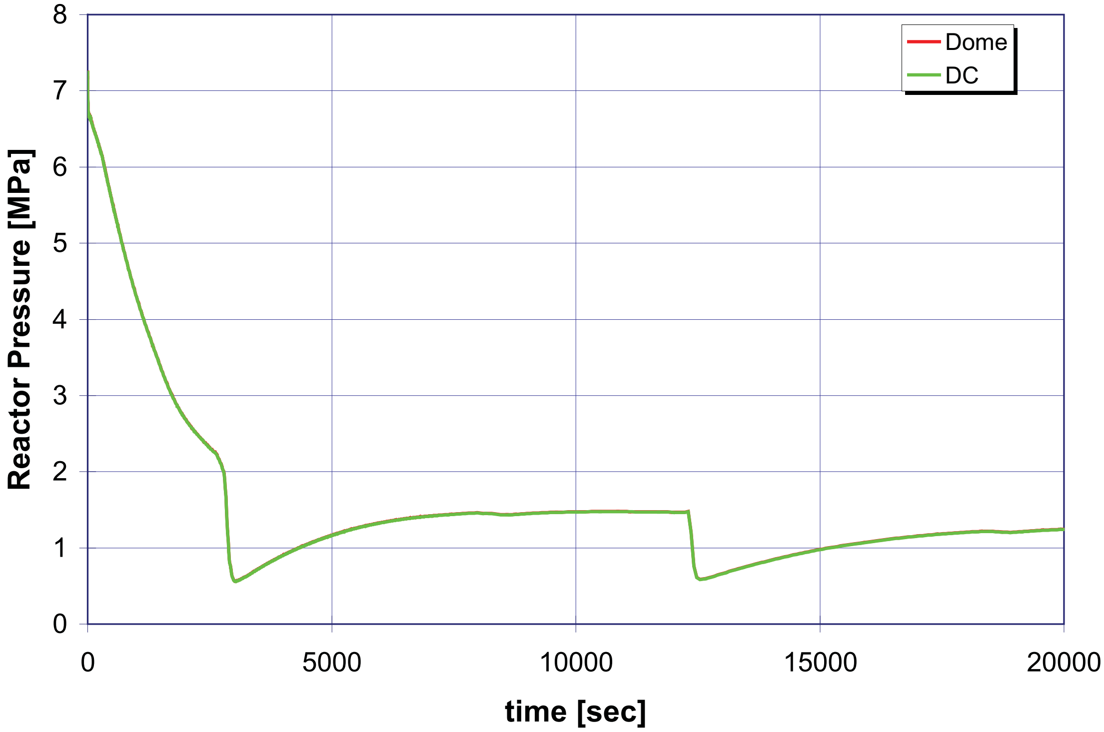
1 For this case, the reactor was allowed to scram based on a reactor protection system trip signal, rather than at time t = 0.

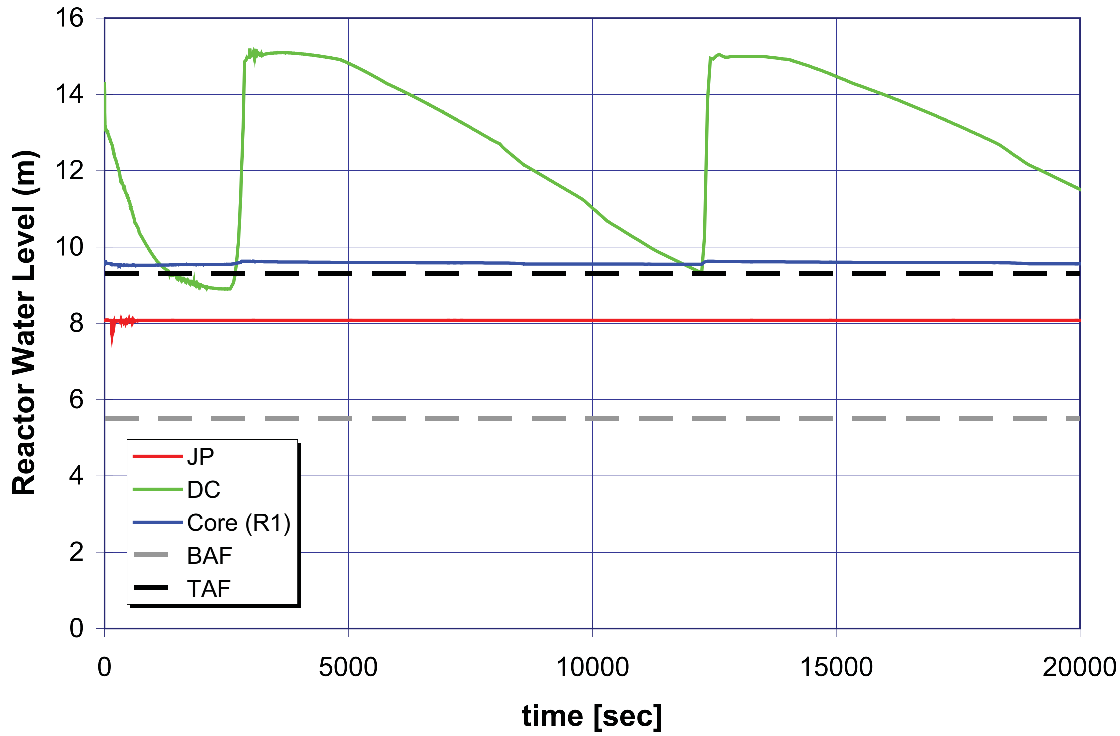
Table 2 Peach Bottom Inadvertent Open SRV Key Timings (Cases 1–5)

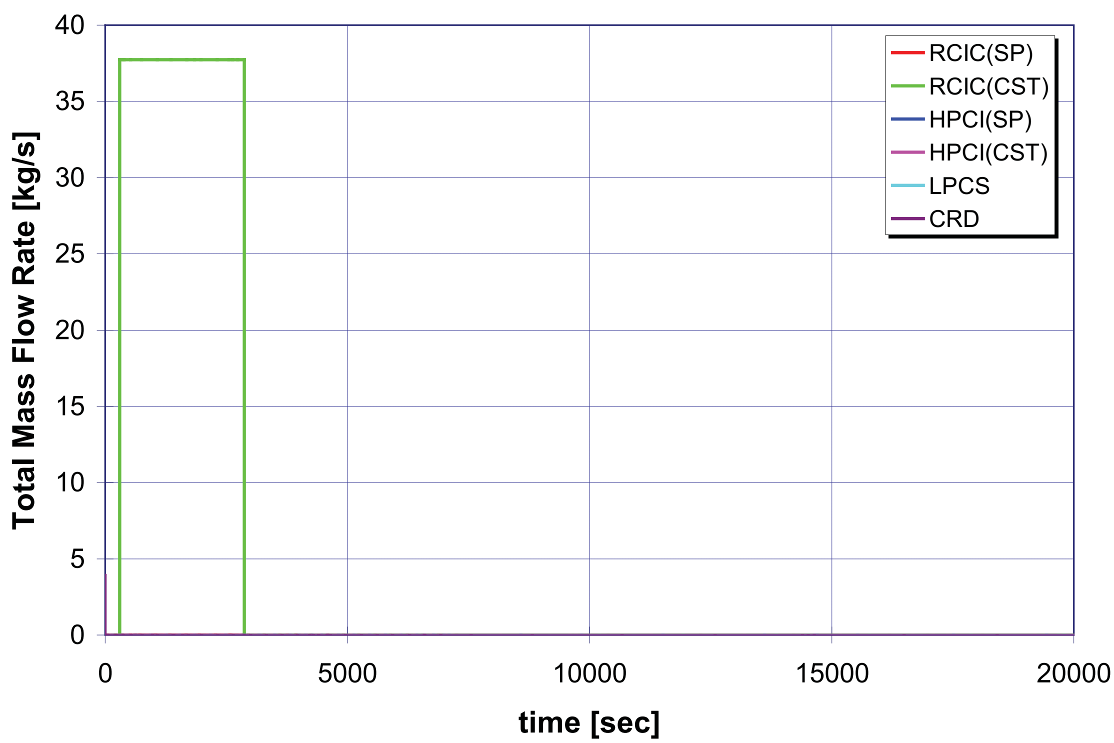
Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)	Case 4 (hr)	Case 4a (hr)	Case 4b (hr)	Case 5 (hr)
SRV 1 open	0	0	0	0	0	0	0
Reactor trip	0	0	0	0	< 0.01[1]	0.76	0
MSIVs close	< 0.01	< 0.01	< 0.01	< 0.01	< 0.01	0.79	0
Downcomer level first reaches L2	0.07	0.07	0.07	0.07	0.03	N/A	0.07
RCIC/HPCI first started (CST injection mode)	0.08	0.08	-	-	-	-	-
2nd CRD pump started	-	-	0.17	0.33	0.33	1.09	-
Downcomer level reaches L1	0.37	8.93	0.32	0.32	0.24	N/A	0.26
Downcomer level below TAF	0.37	8.93	0.35	0.33	0.25	N/A	0.28
Suppression pool temp. >110 °F[3]	0.40	0.61	0.42	0.42	0.41	0.30	0.40
LPCI first started	0.51	8.93	0.59	0.58	0.53	N/A	0.57
RCIC/HPCI pump isolation: low steamline pressure <0.52 MPa (75 psig)	0.82	5.59	-	-	-	-	-
HCTL limit reached[3] (no action taken)	4.5	4.0	> 1[2]	> 1[2]	5.0	0.57	> 1[2]
RHR pump isolation - NPSH	9.6	11.1	> 1[2]	> 1[2]	> 10[2]	5.4	> 1[2]
Maximum cladding temperature timing (max. temperature)	No heatup	No heatup	0.78 (786 K)	0.76 (830 K)	0.67 (941 K)	No heatup	0.75 (1,212 K)

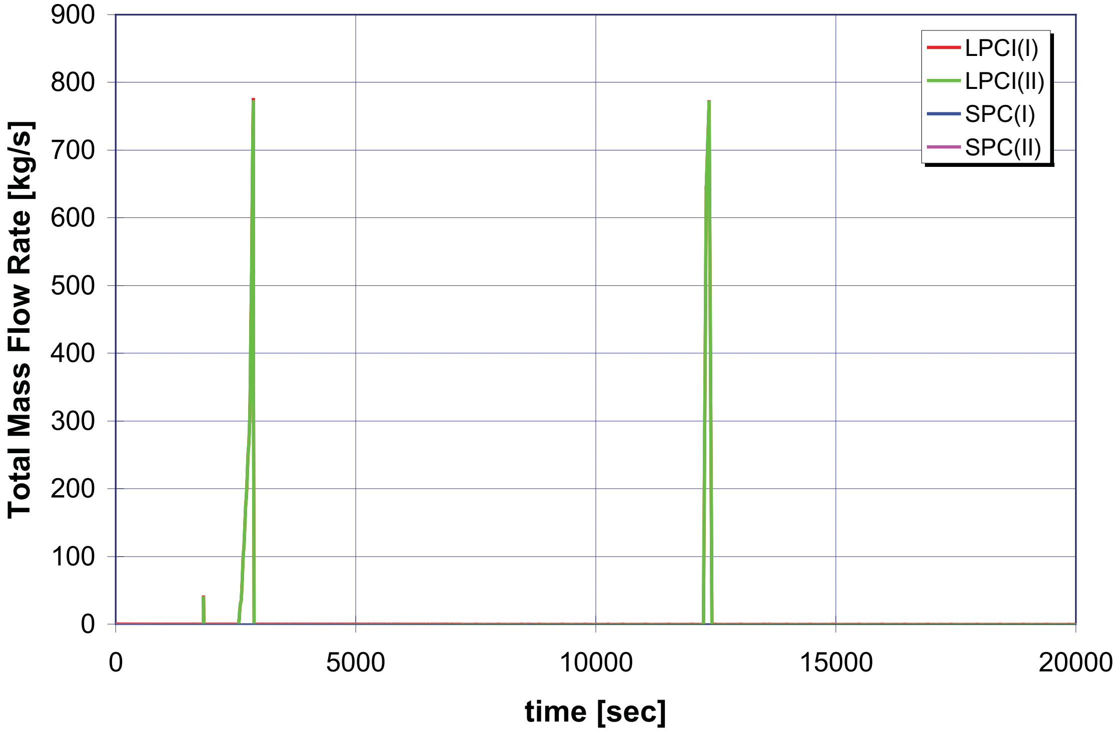
1 Reactor trips at 8 seconds on low RPV level.
2 The simulation was stopped before reaching this condition.
3 The HCTL limit is based on suppression pool temperature, suppression pool level, and RPV pressure.

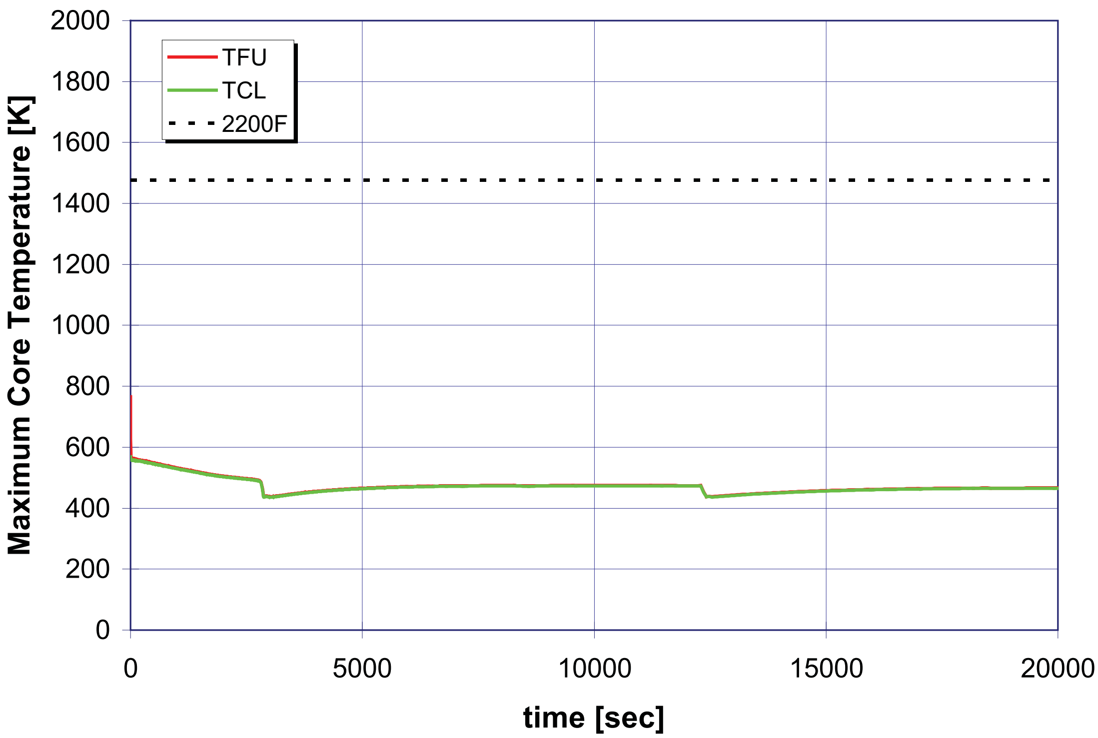
B.1.1 Case 1: Safety Relief Valve Opening and Reactor Core Isolation Cooling

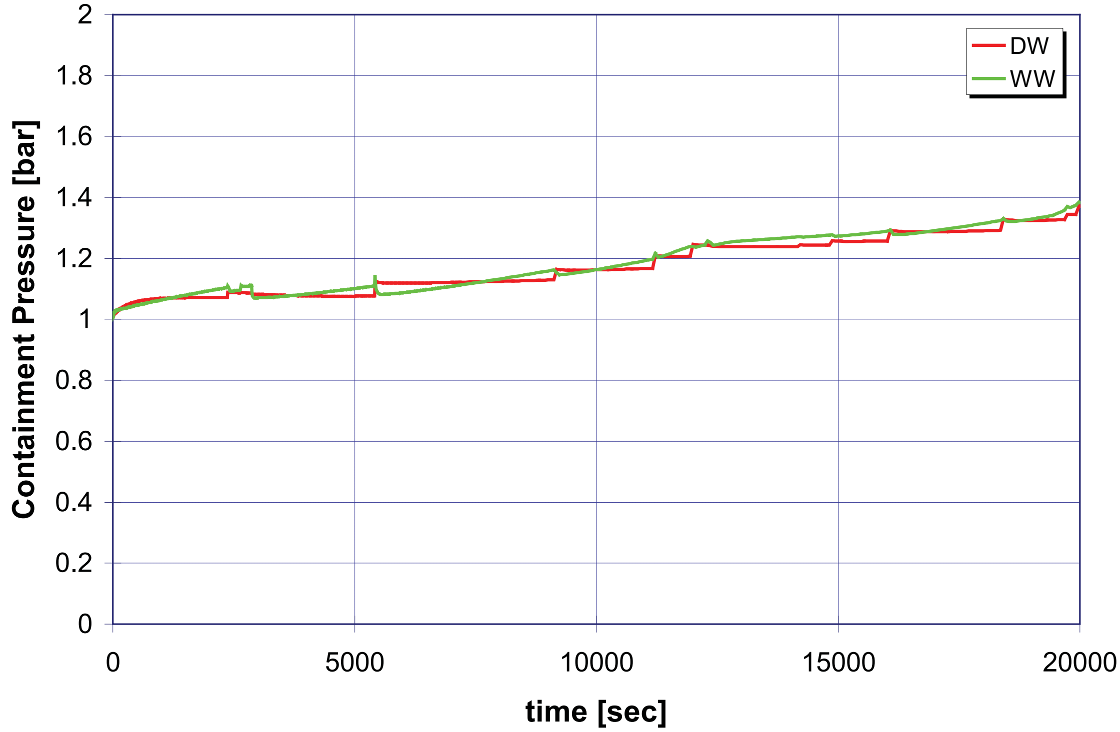



Reactor Water Level (m)
16
14
12
10
8
6
4
2
0
0
5000
10000
15000
20000
time [sec]
JP
DC
Core (R1)
BAF
TAF

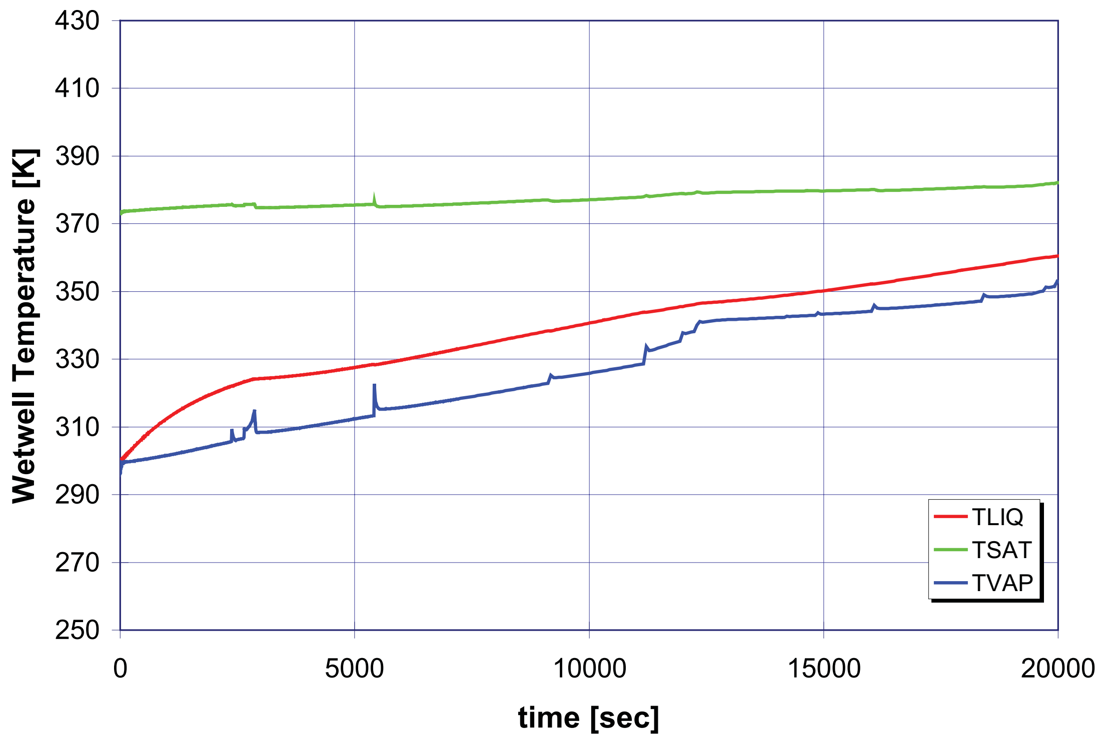

Total Mass Flow Rate [kg/s]
40
35
30
25
20
15
10
5
0
0
5000
10000
15000
20000
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD


Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
100
200
300
400
500
600
700
800
900
0
5000
10000
15000
20000

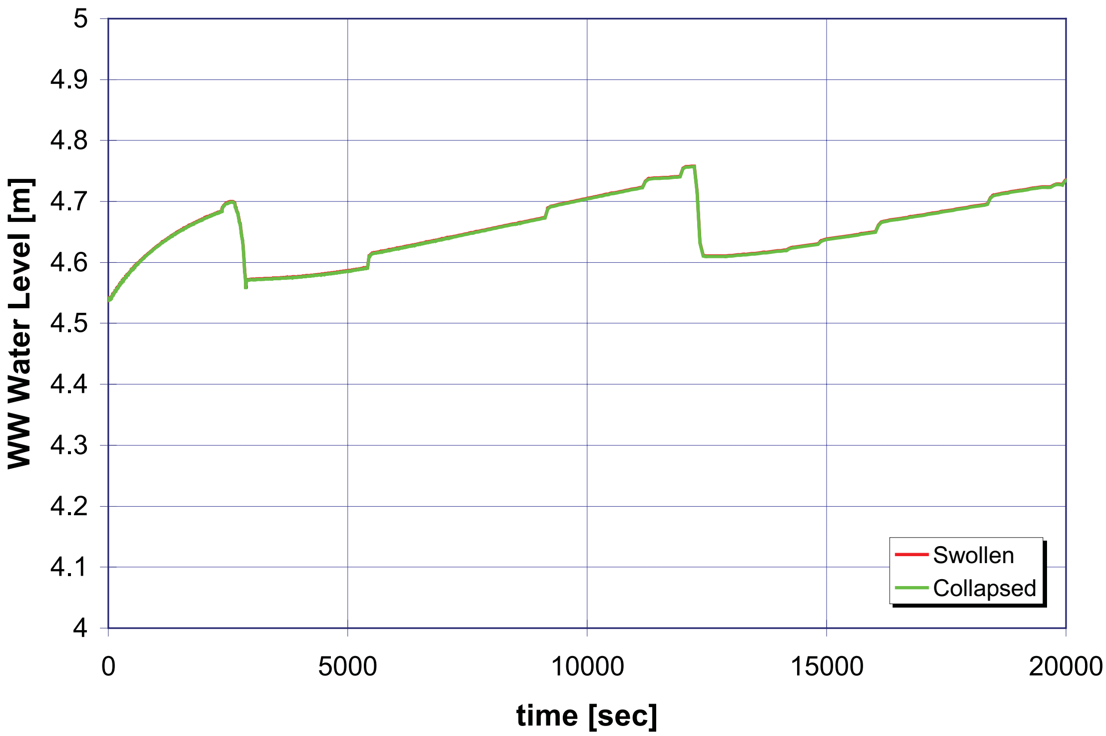

Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
5000
10000
15000
20000



Containment Pressure [bar]
time [sec]
DW
WW
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
5000
10000
15000
20000

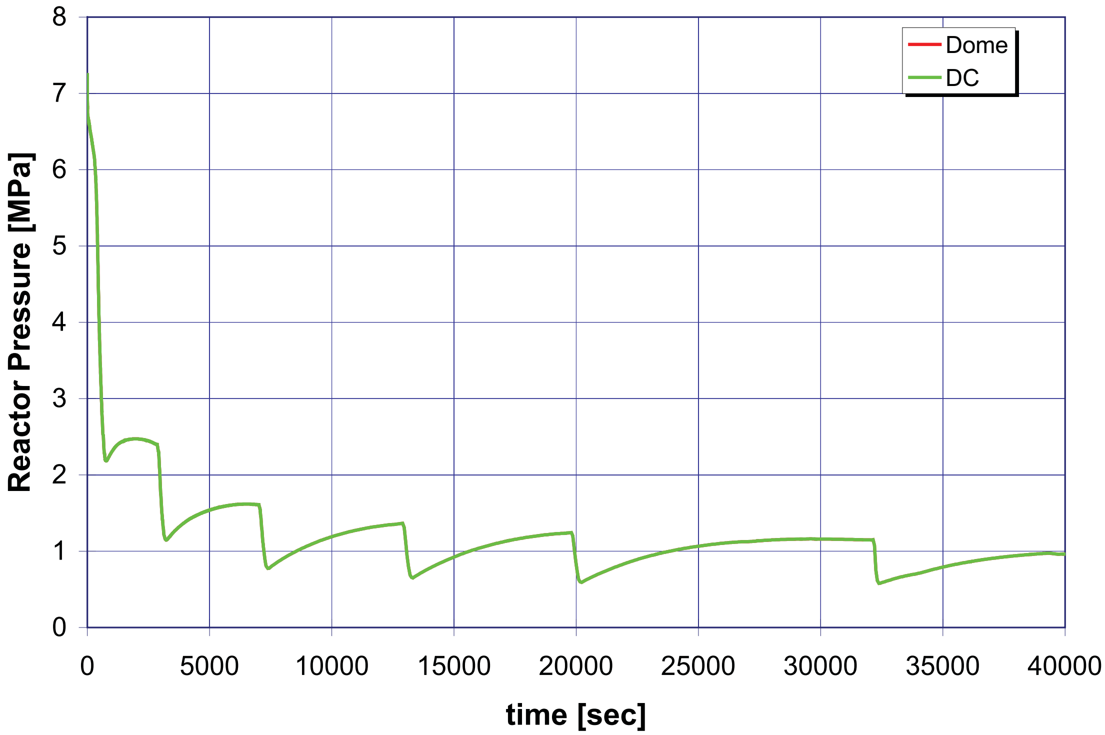


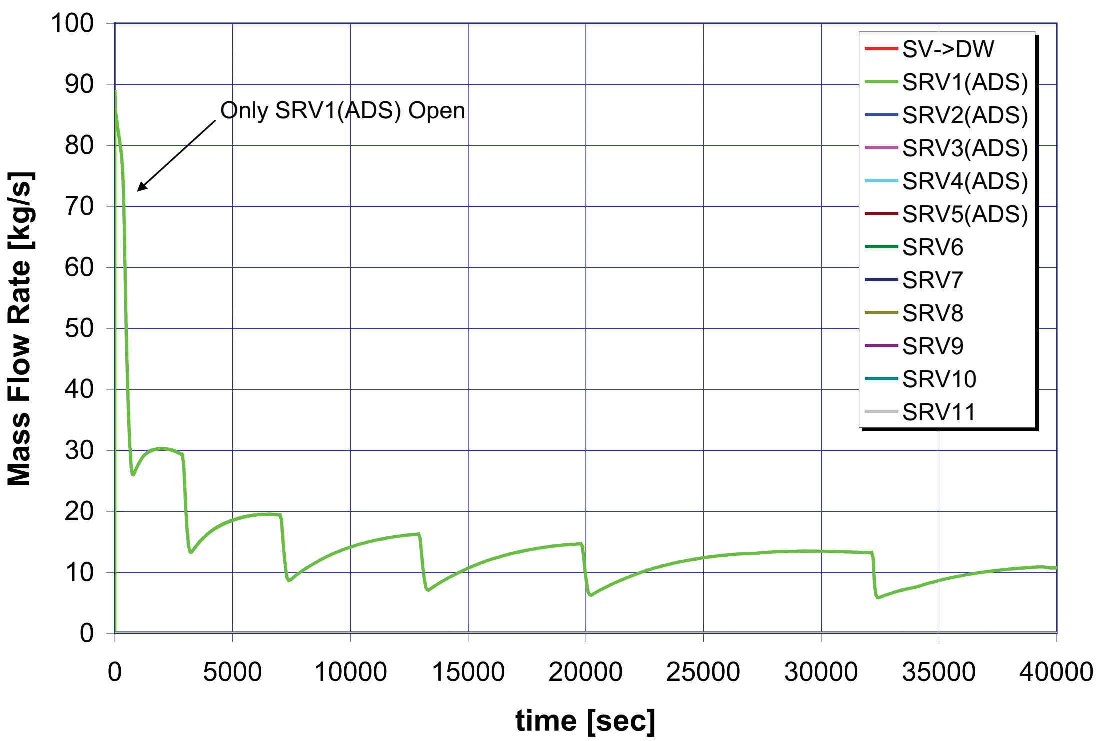
Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP
430
410
390
370
350
330
310
290
270
250
0
5000
10000
15000
20000

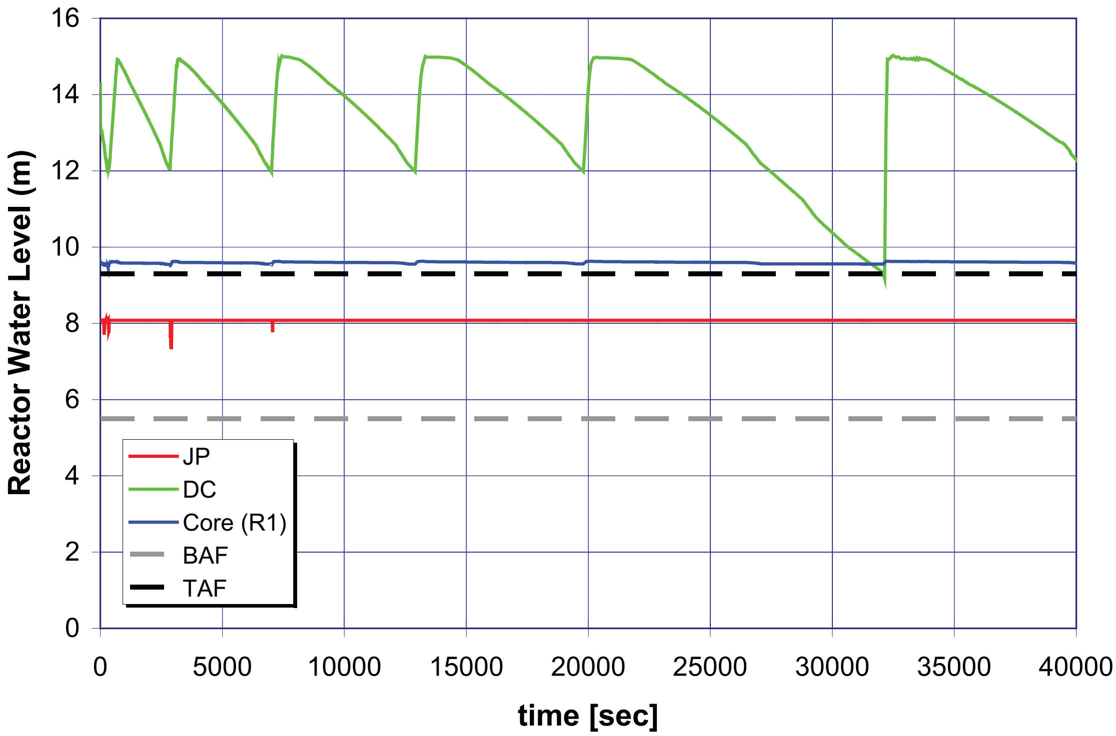


WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
5000
10000
15000
20000
time [sec]
Swollen
Collapsed

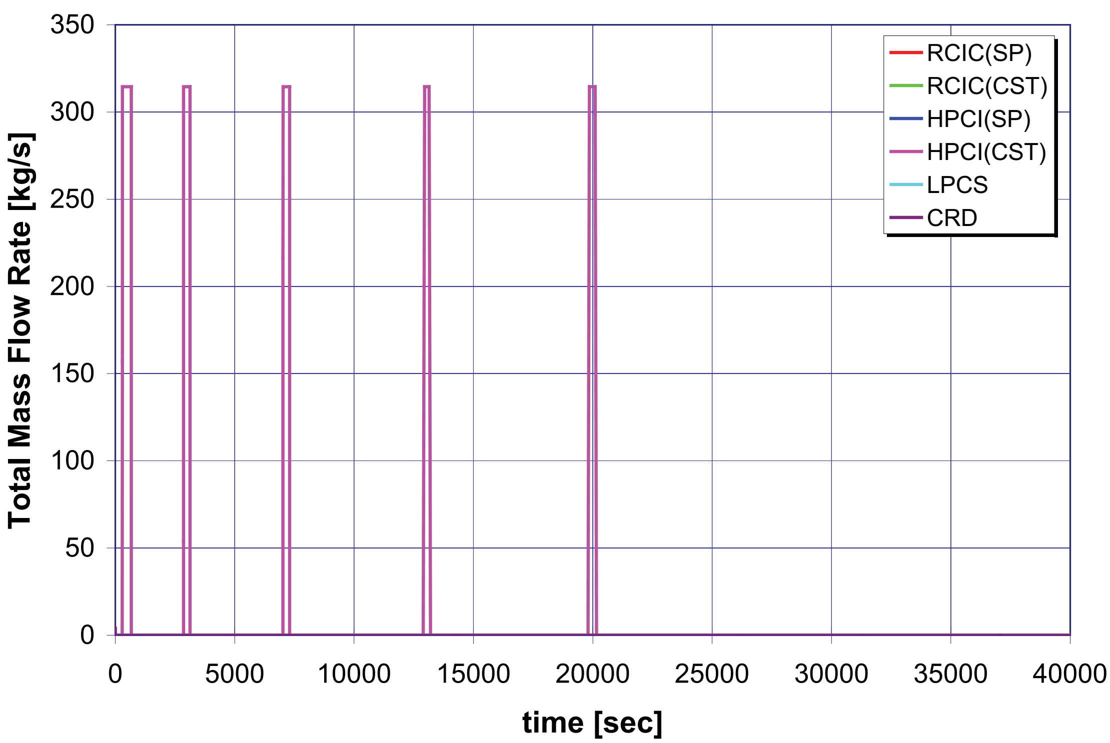
B.1.2 Case 2: Safety Relief Valve Opening and High-Pressure Coolant Injection



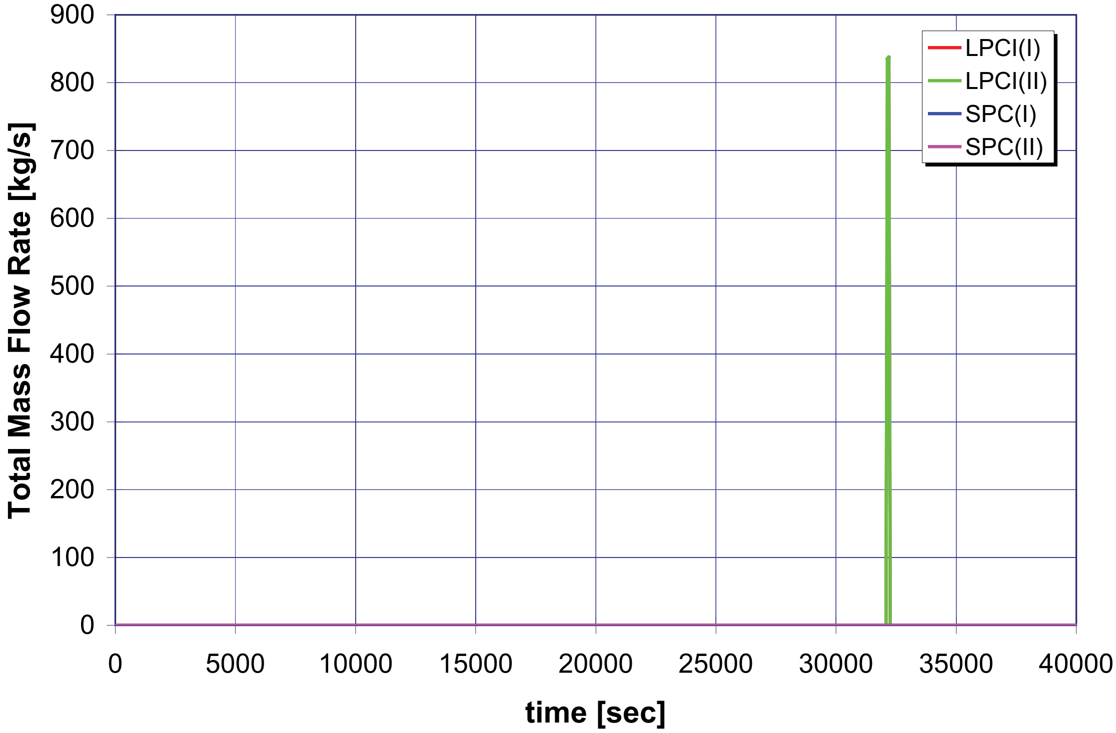




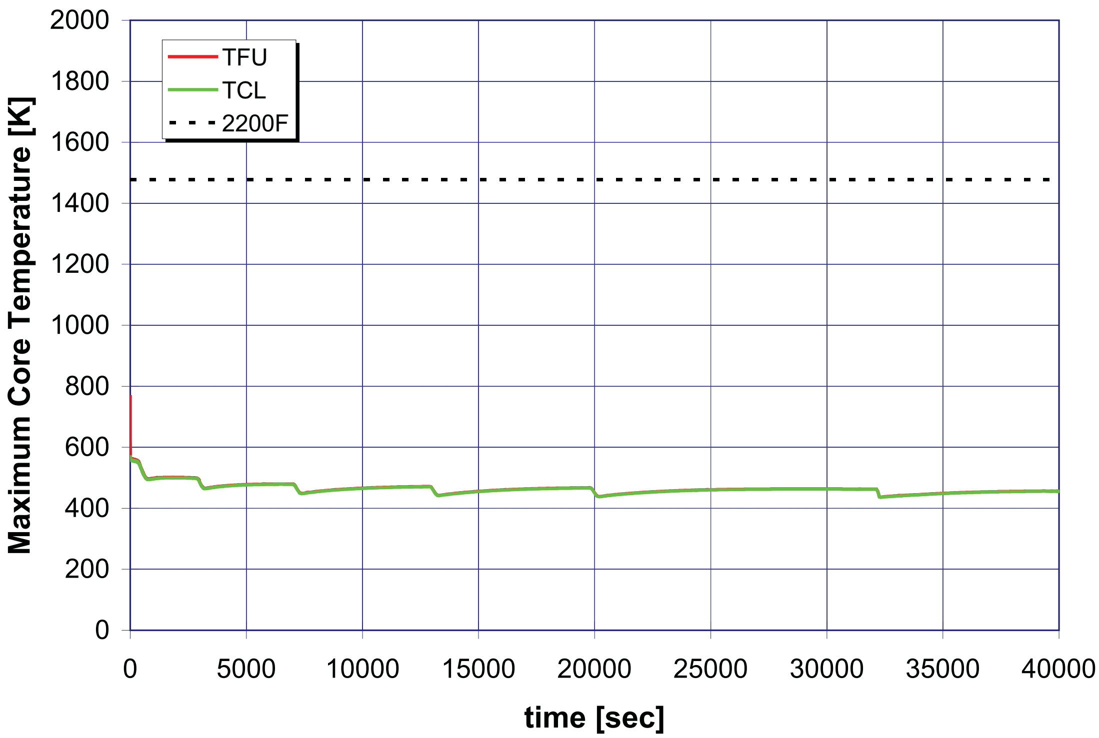
Reactor Water Level (m)
time [sec]
JP
DC
Core (R1)
BAF
TAF



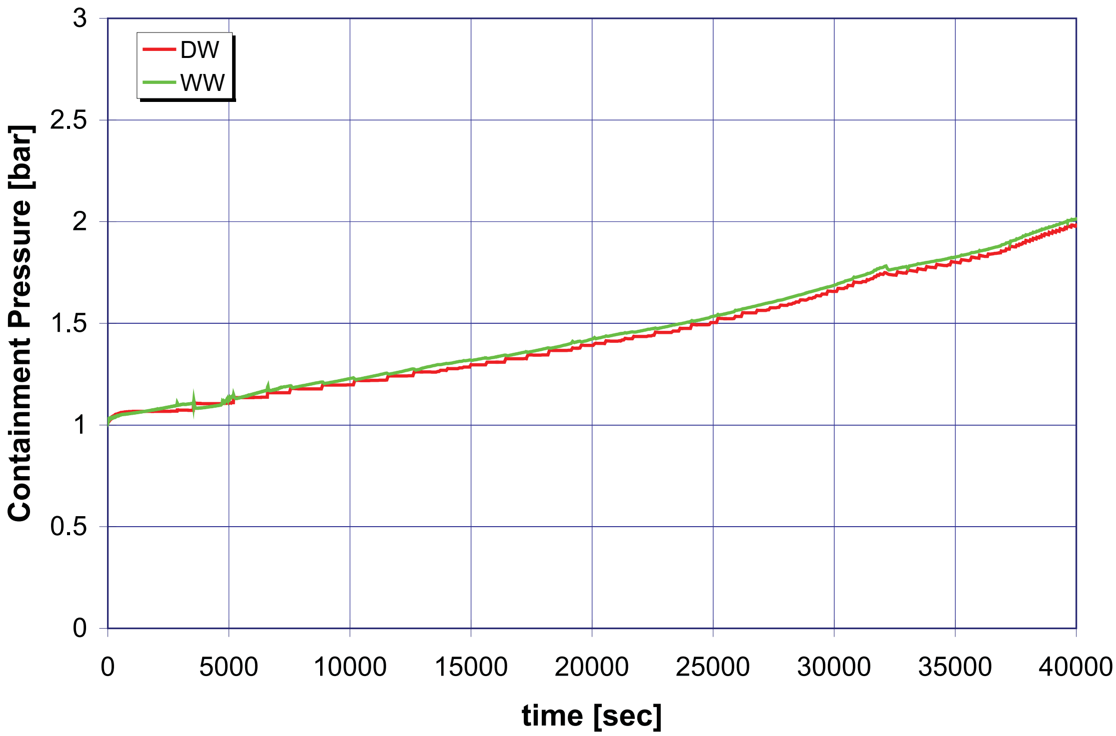
Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD

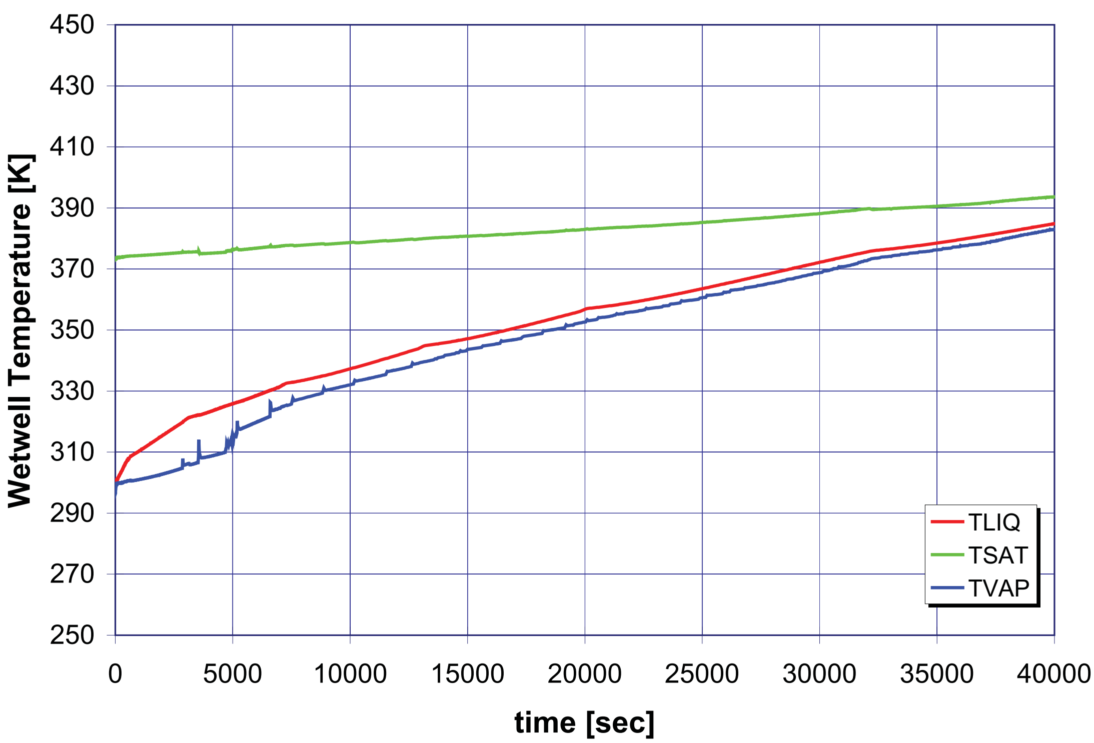


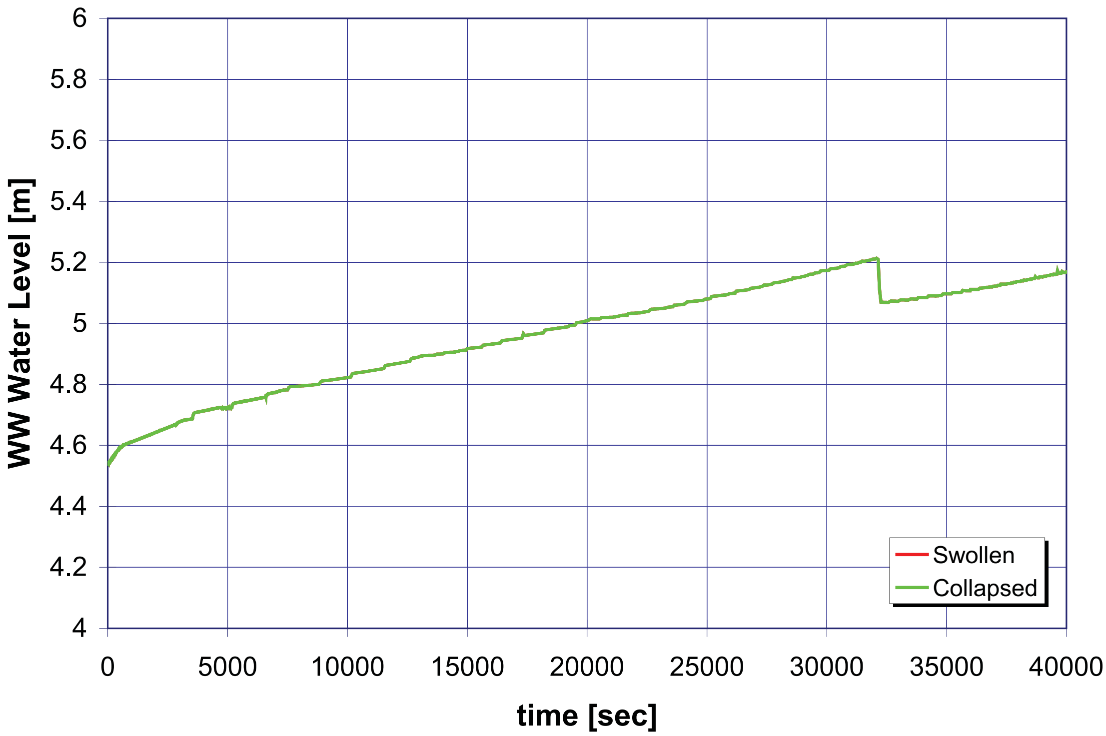
Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
100
200
300
400
500
600
700
800
900
0
5000
10000
15000
20000
25000
30000
35000
40000



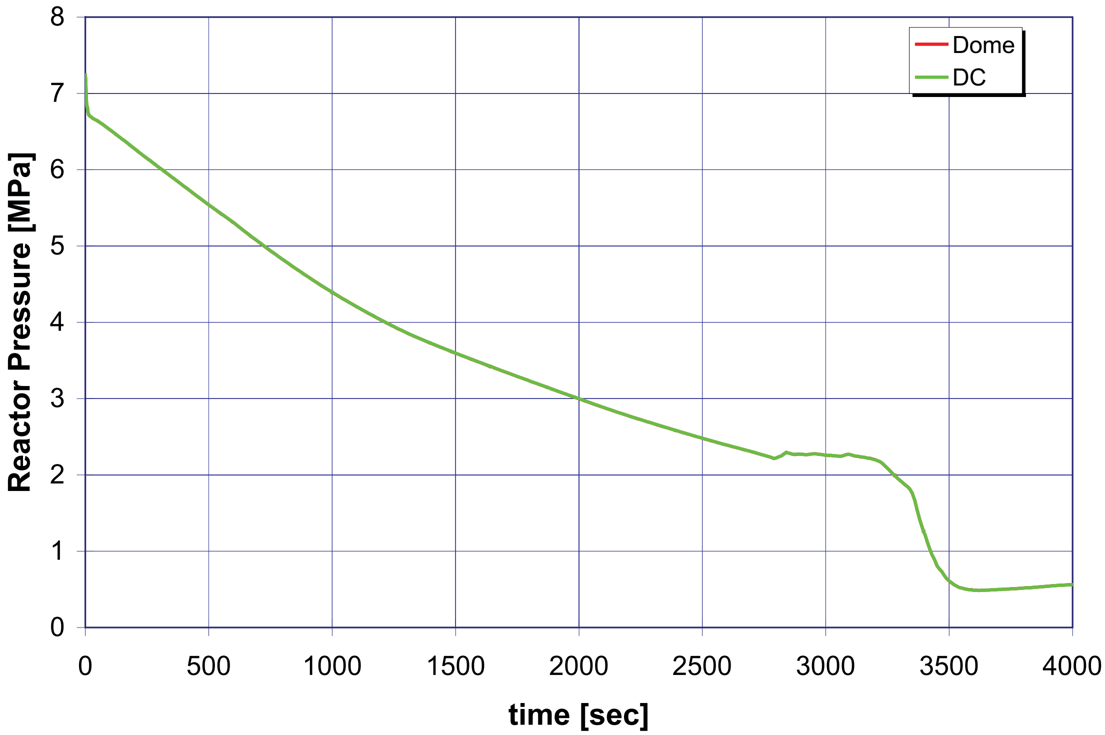
Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
5000
10000
15000
20000
25000
30000
35000
40000

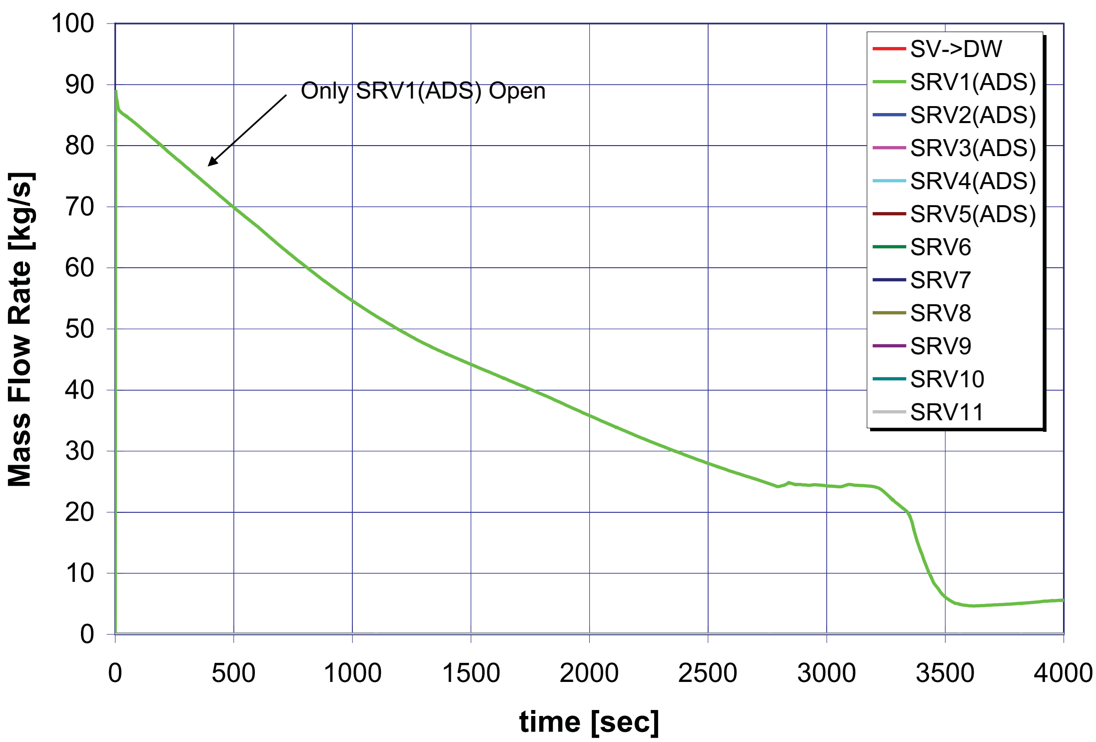

DW
WW
Containment Pressure [bar]
time [sec]
0
0.5
1
1.5
2
2.5
3
0
5000
10000
15000
20000
25000
30000
35000
40000

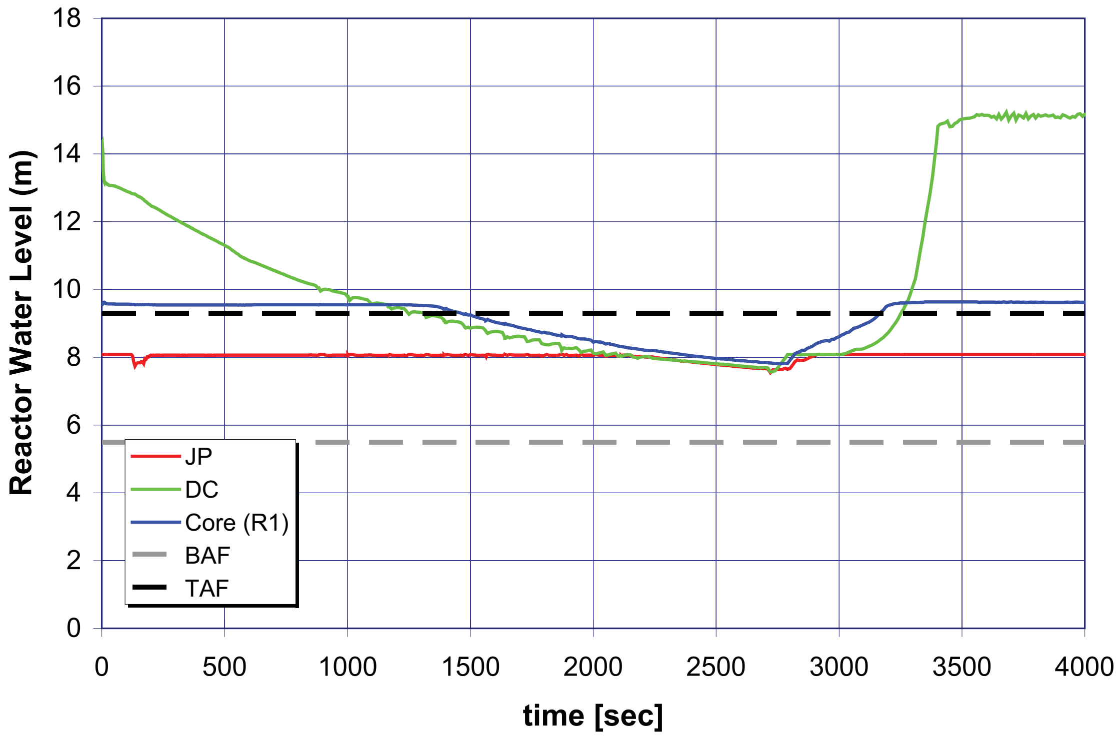

TLIQ
TSAT
TVAP
Wetwell Temperature [K]
time [sec]
250
270
290
310
330
350
370
390
410
430
450
0
5000
10000
15000
20000
25000
30000
35000
40000

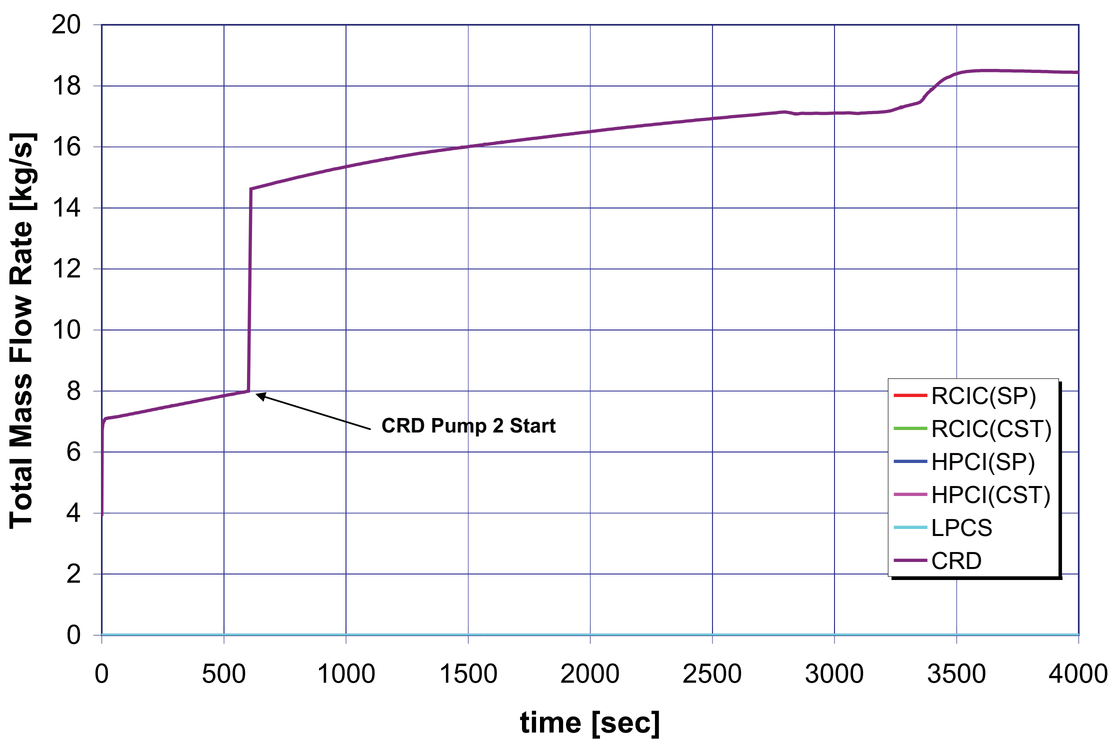

6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
WW Water Level [m]
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
Swollen
Collapsed

B.1.3 Case 3: Safety Relief Valve Opening and CRD 2 at 10 Minutes

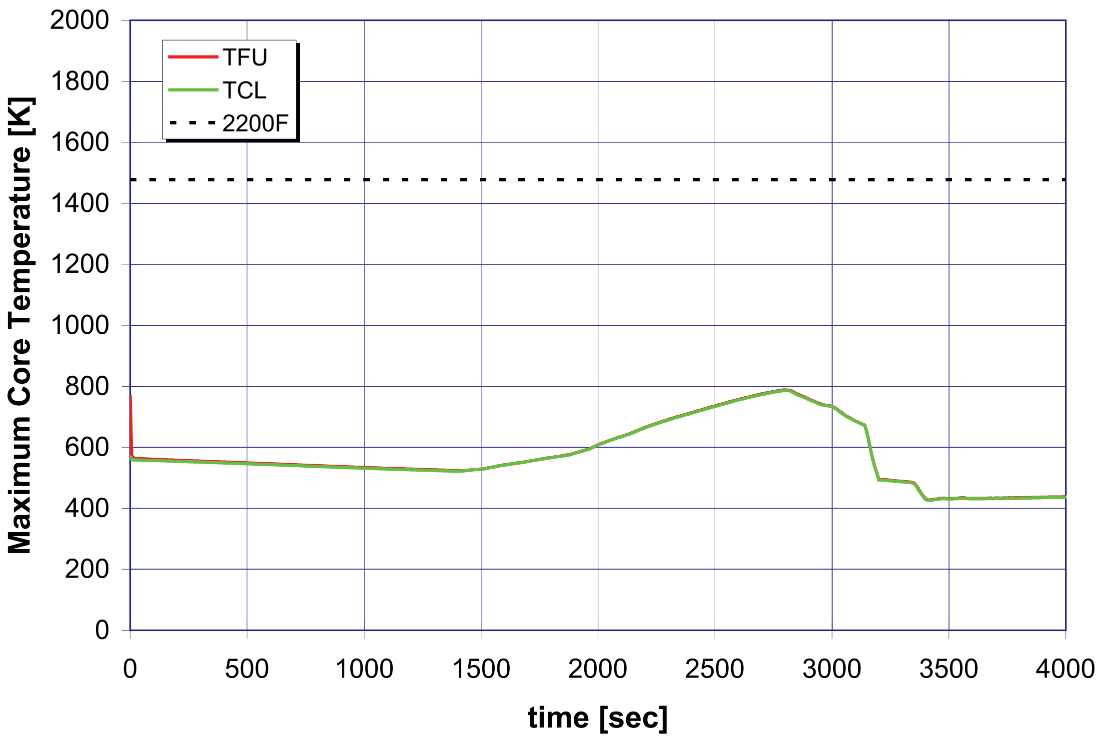




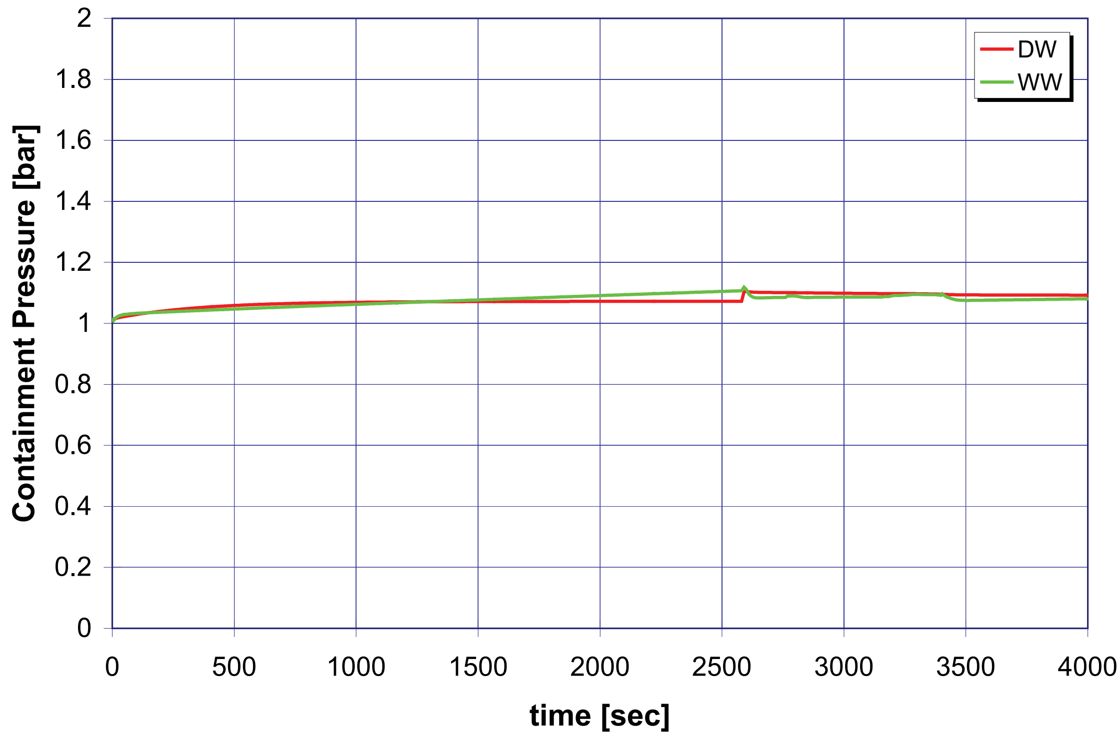

Reactor Water Level (m)
time [sec]
JP
DC
Core (R1)
BAF
TAF


Total Mass Flow Rate [kg/s]
time [sec]
CRD Pump 2 Start
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD

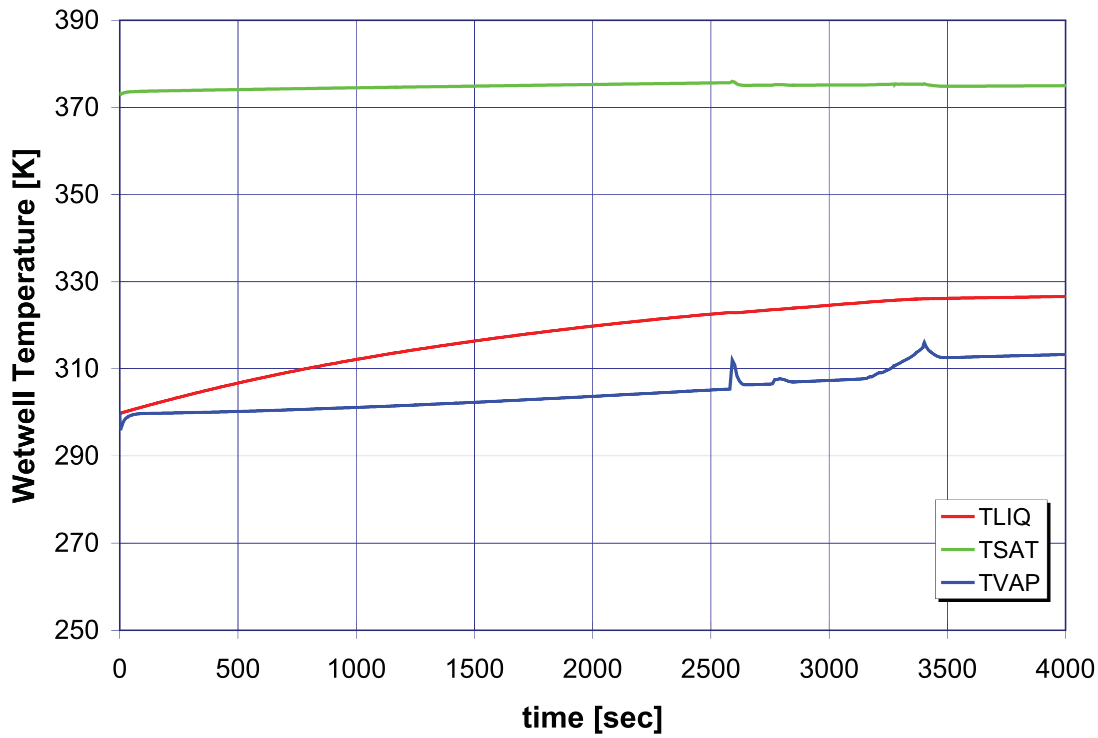
900
800
700
600
500
400
300
200
100
0
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
Total Mass Flow Rate [kg/s]
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]



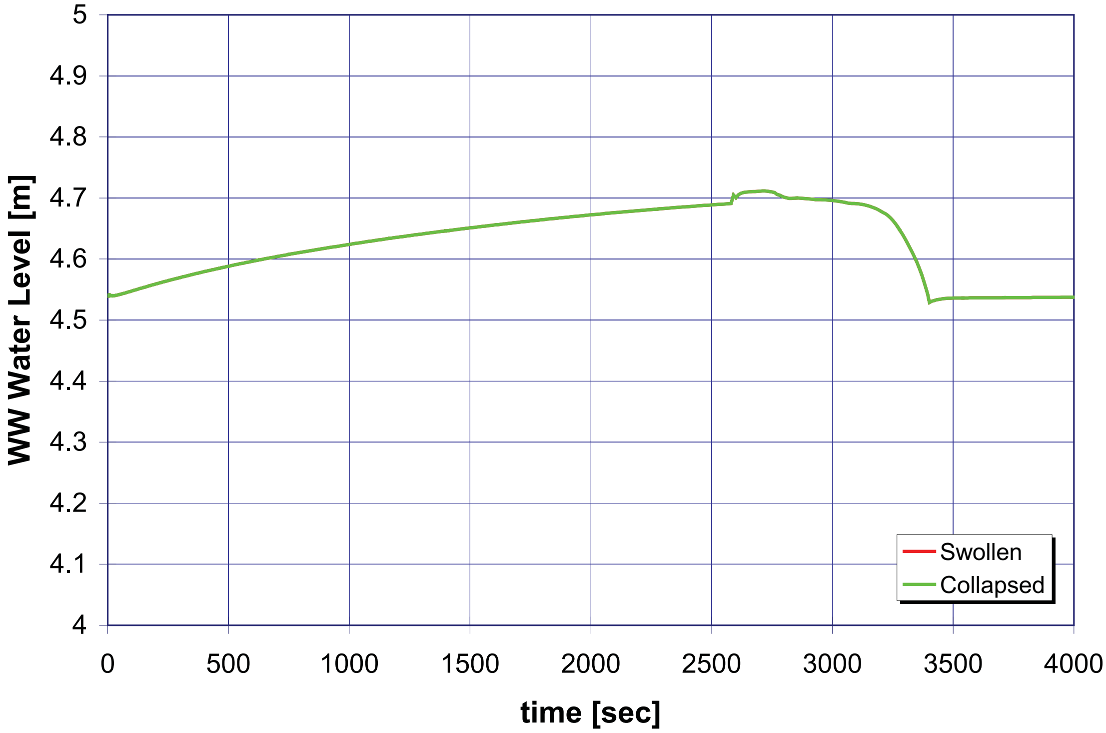
2000
1800
1600
1400
1200
1000
800
600
400
200
0
TFU
TCL
2200F
Maximum Core Temperature [K]
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]



Containment Pressure [bar]
time [sec]
DW
WW
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000
3500
4000

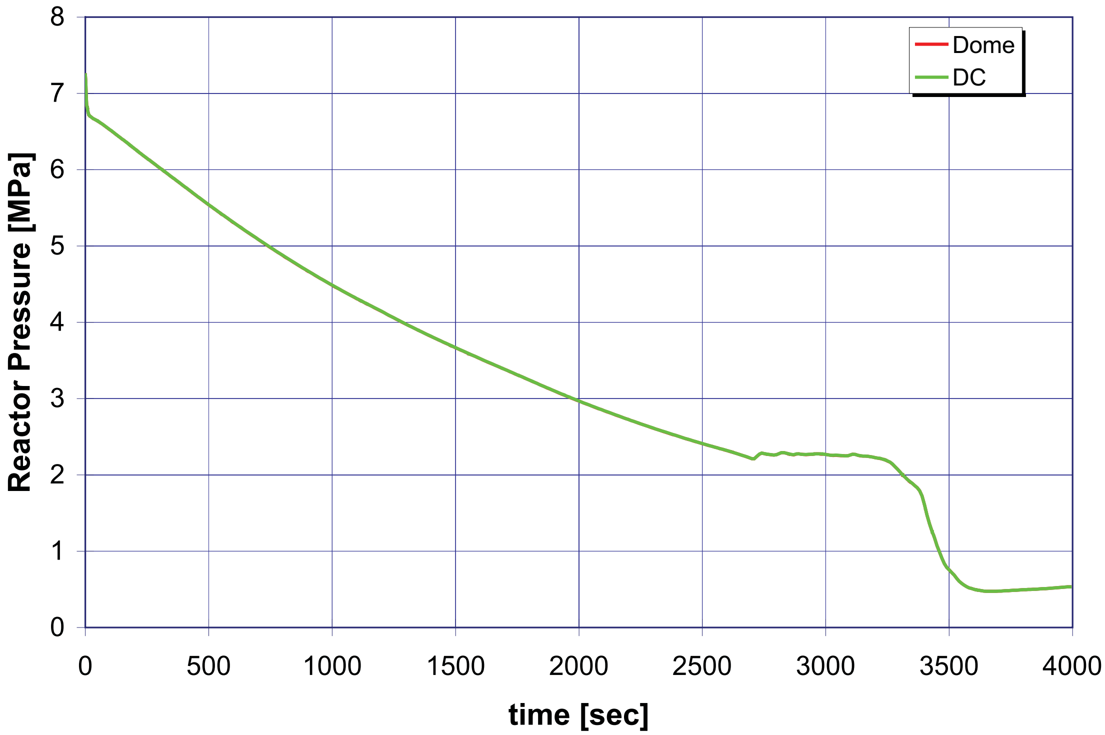


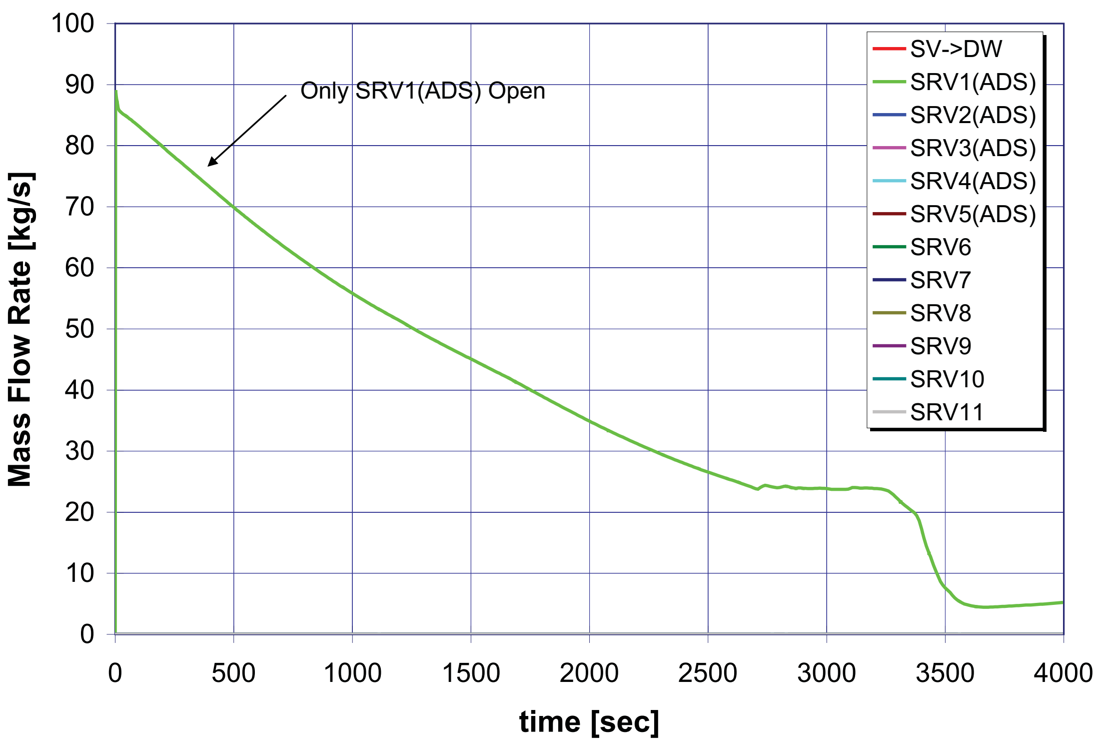
Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP
250
270
290
310
330
350
370
390
0
500
1000
1500
2000
2500
3000
3500
4000

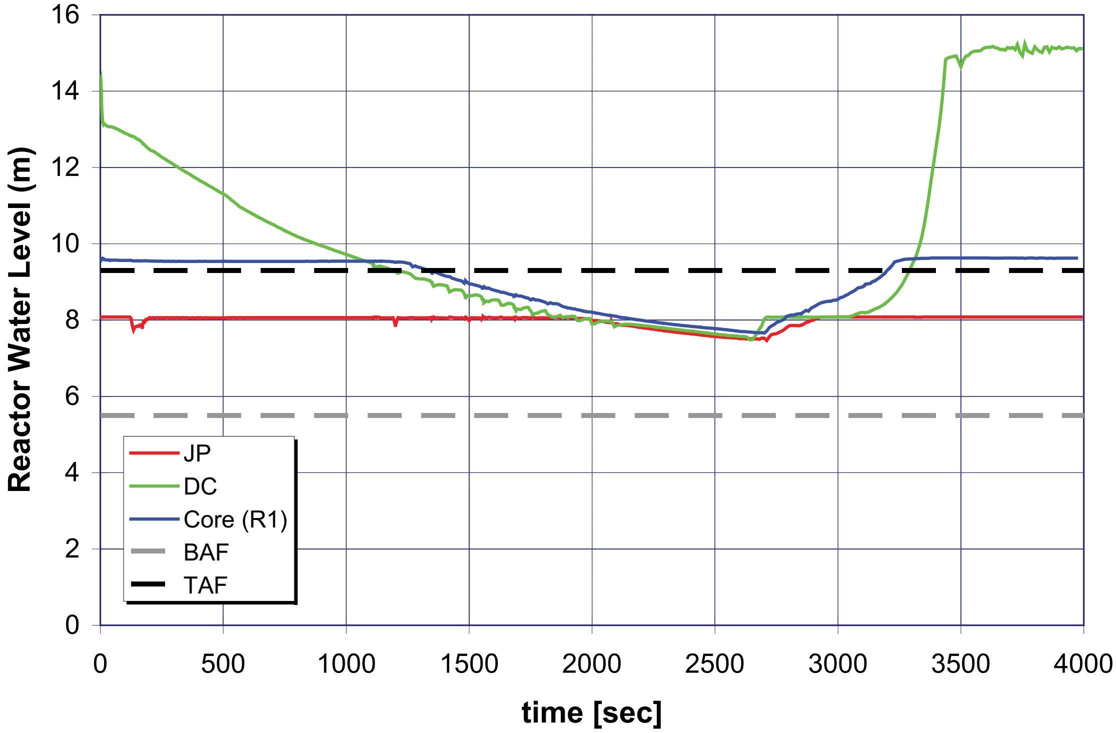


WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
Swollen
Collapsed

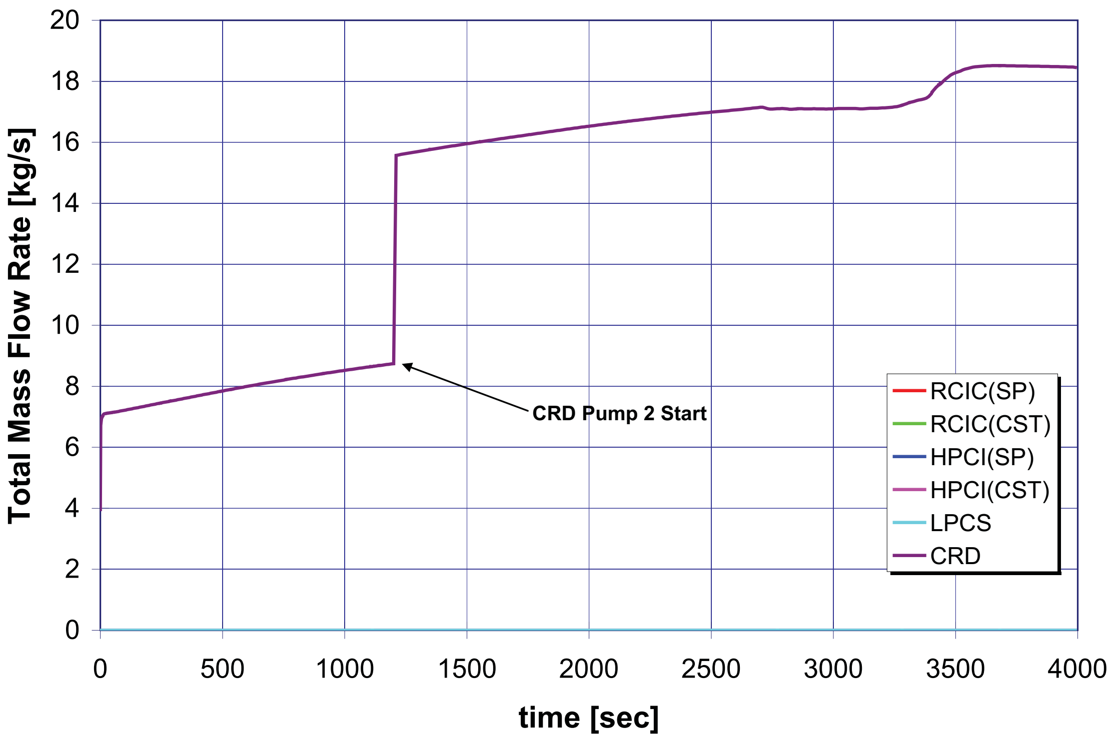
B.1.4 Case 4: Safety Relief Valve Opening and CRD 2 at 20 Minutes



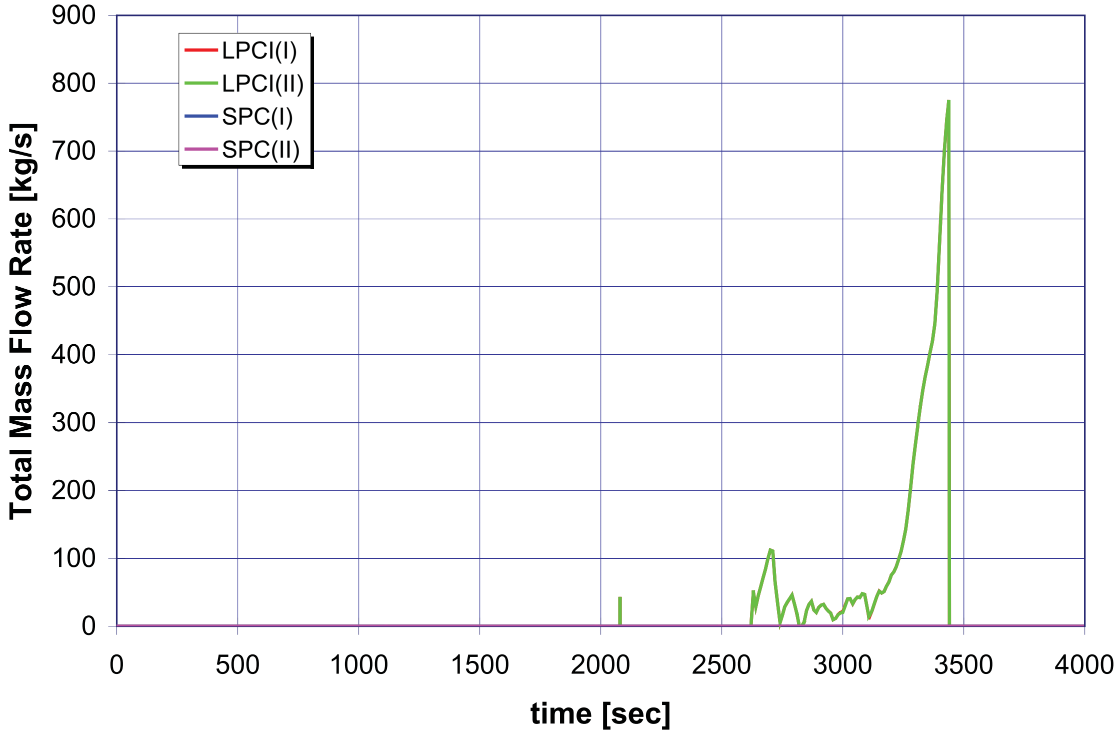




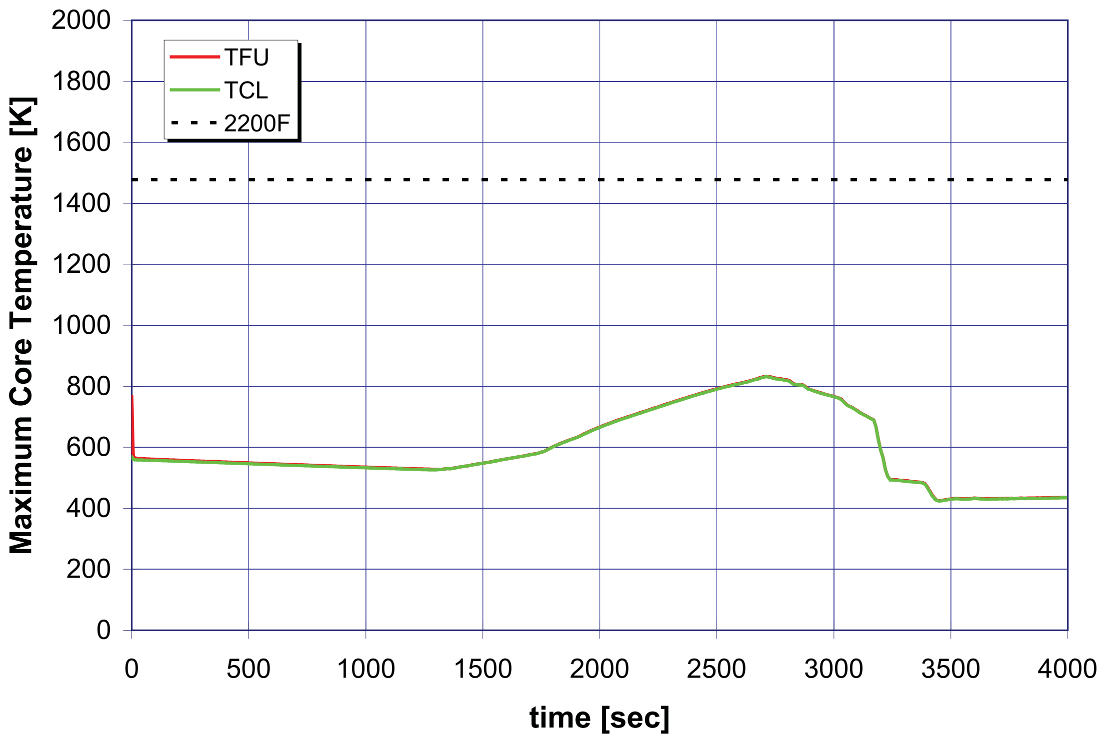
Reactor Water Level (m)
time [sec]
JP
DC
Core (R1)
BAF
TAF



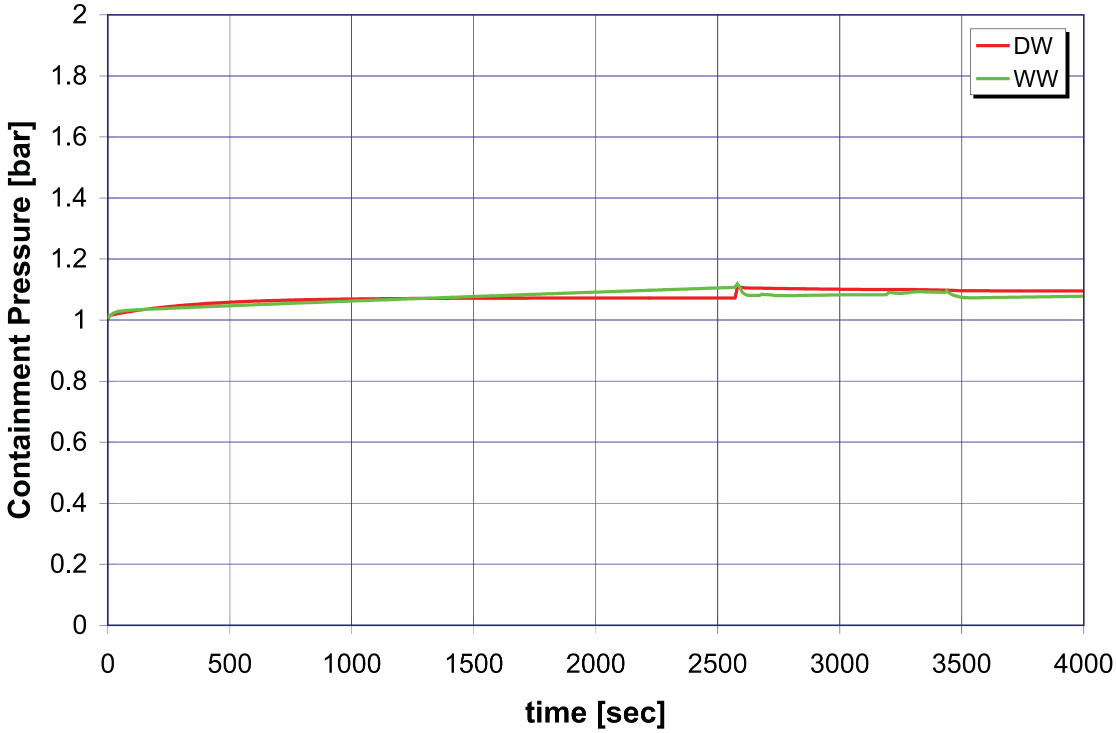
Total Mass Flow Rate [kg/s]
time [sec]
CRD Pump 2 Start
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD



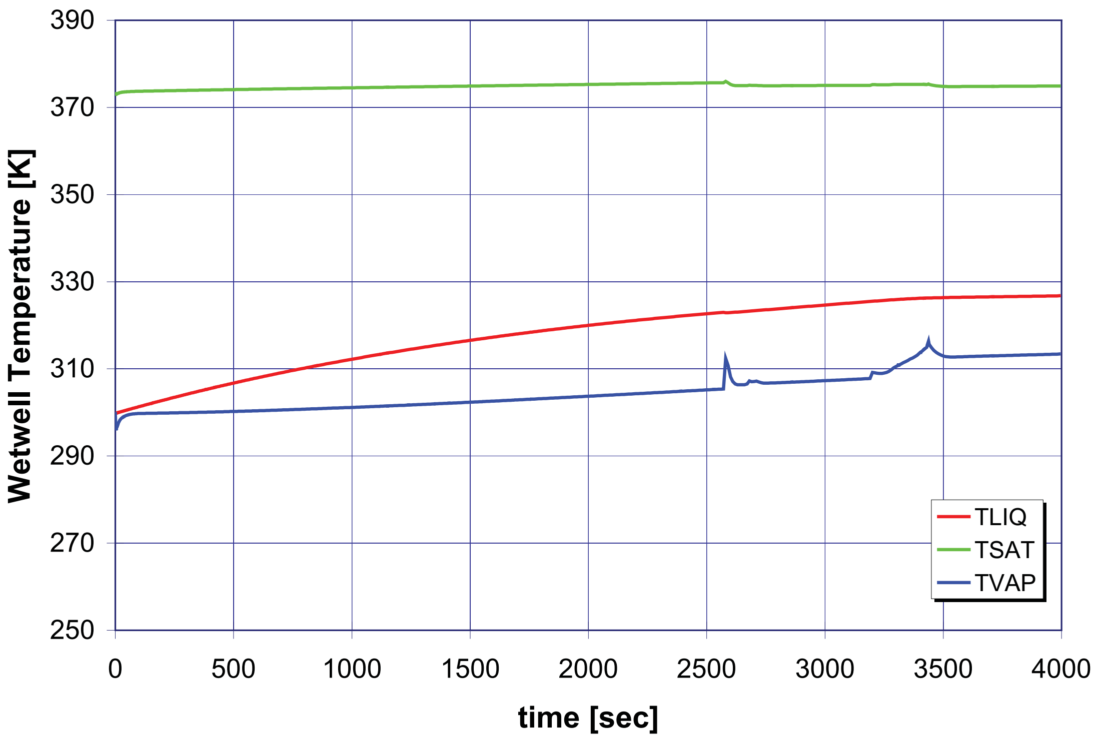
Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
100
200
300
400
500
600
700
800
900
0
500
1000
1500
2000
2500
3000
3500
4000



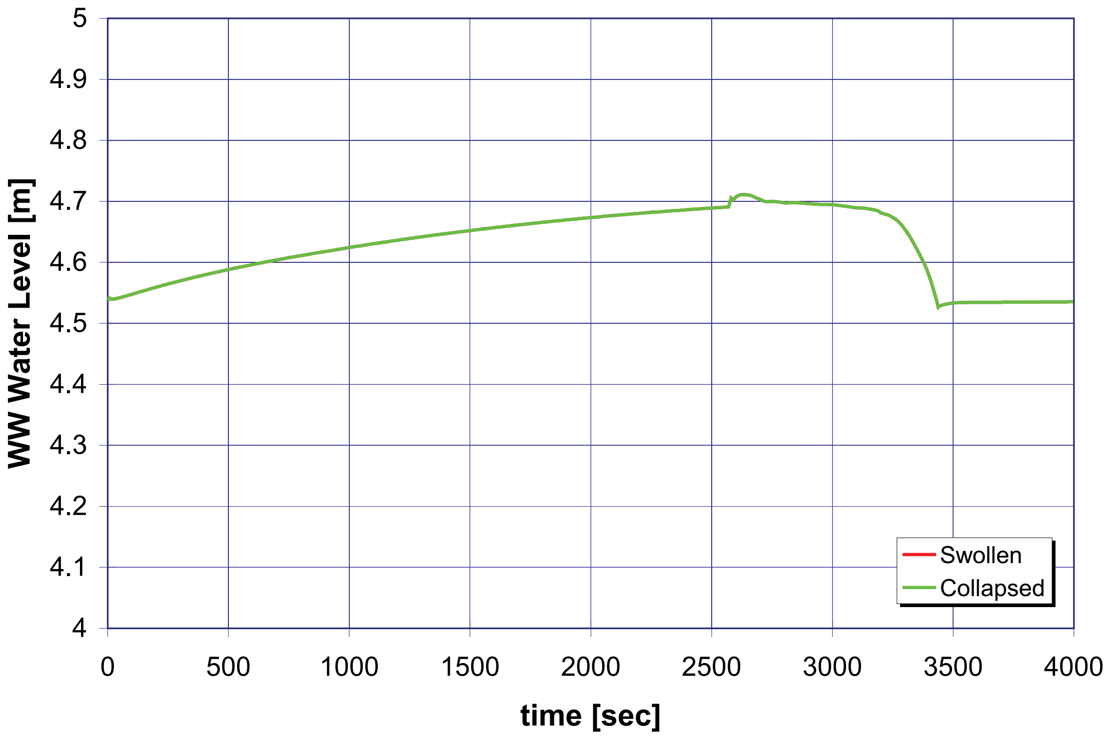
Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
500
1000
1500
2000
2500
3000
3500
4000



DW
WW
Containment Pressure [bar]
time [sec]

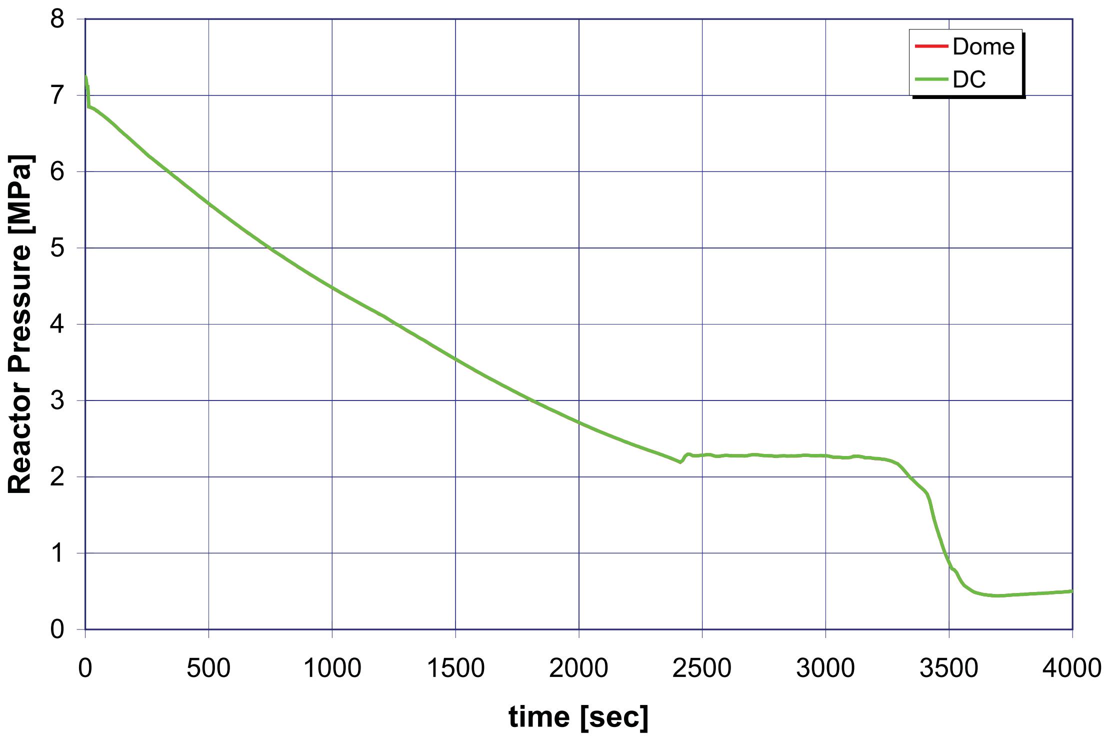


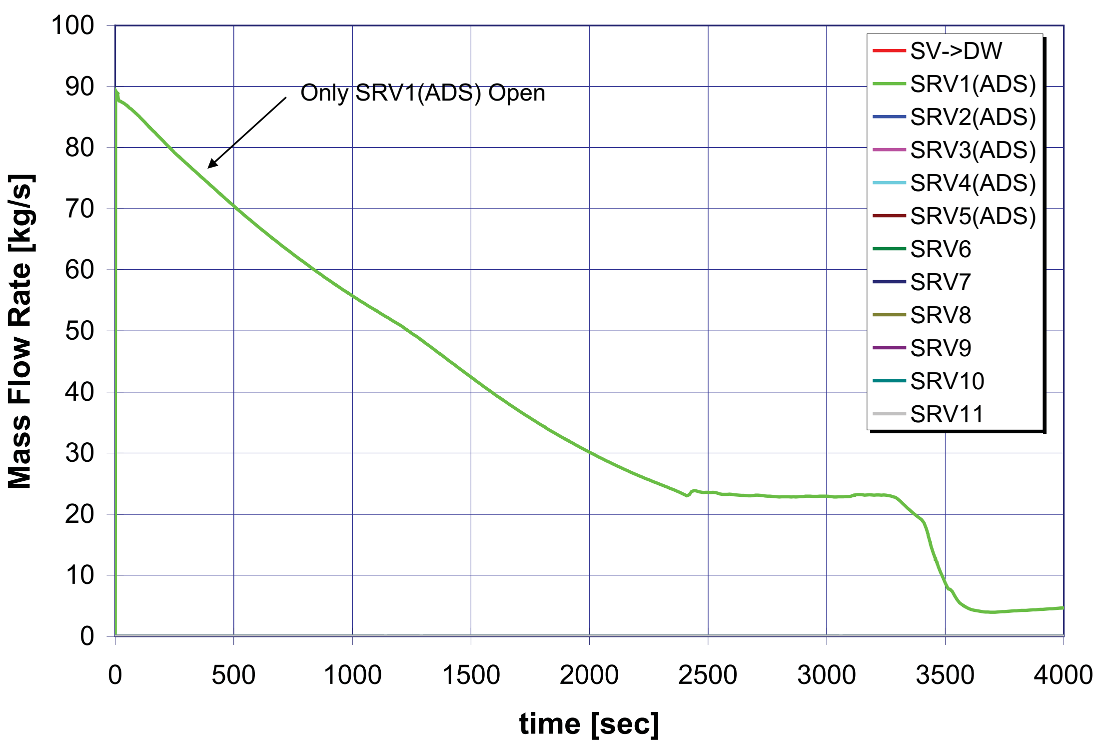
TLIQ
TSAT
TVAP
Wetwell Temperature [K]
time [sec]

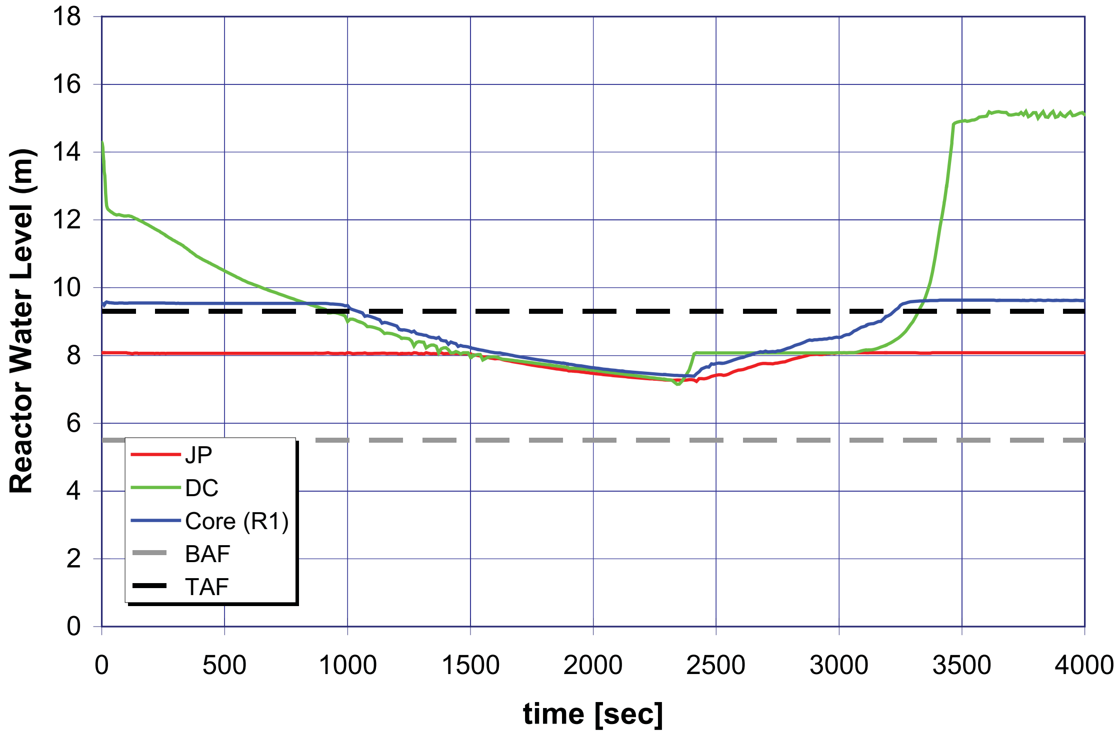


WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
Swollen
Collapsed

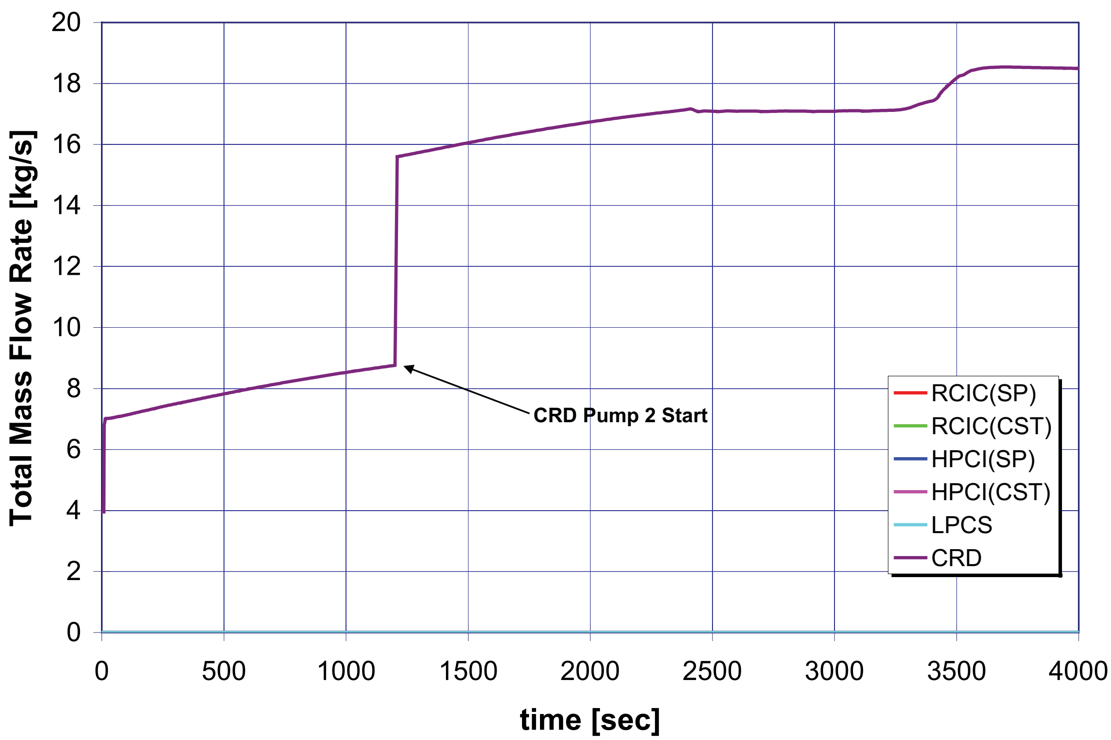
B.1.4.1 Case 4a: Safety Relief Valve Opening, CRD 2 at 20 Minutes, and No Reactor Trip at t = 0



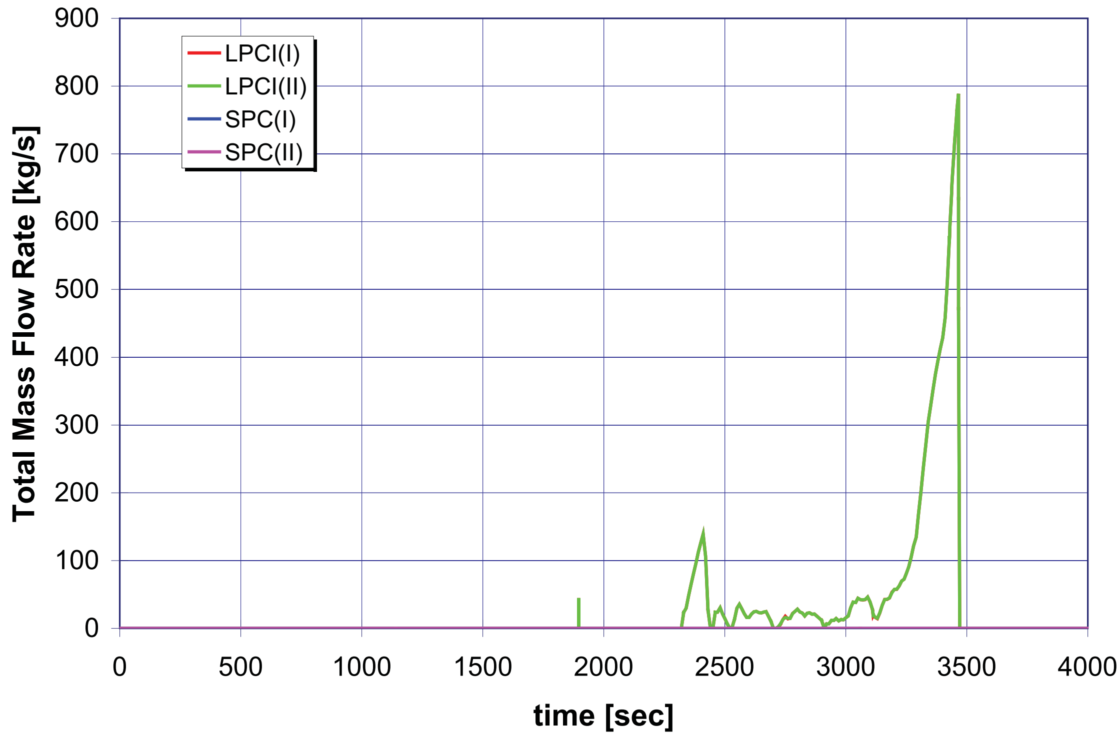




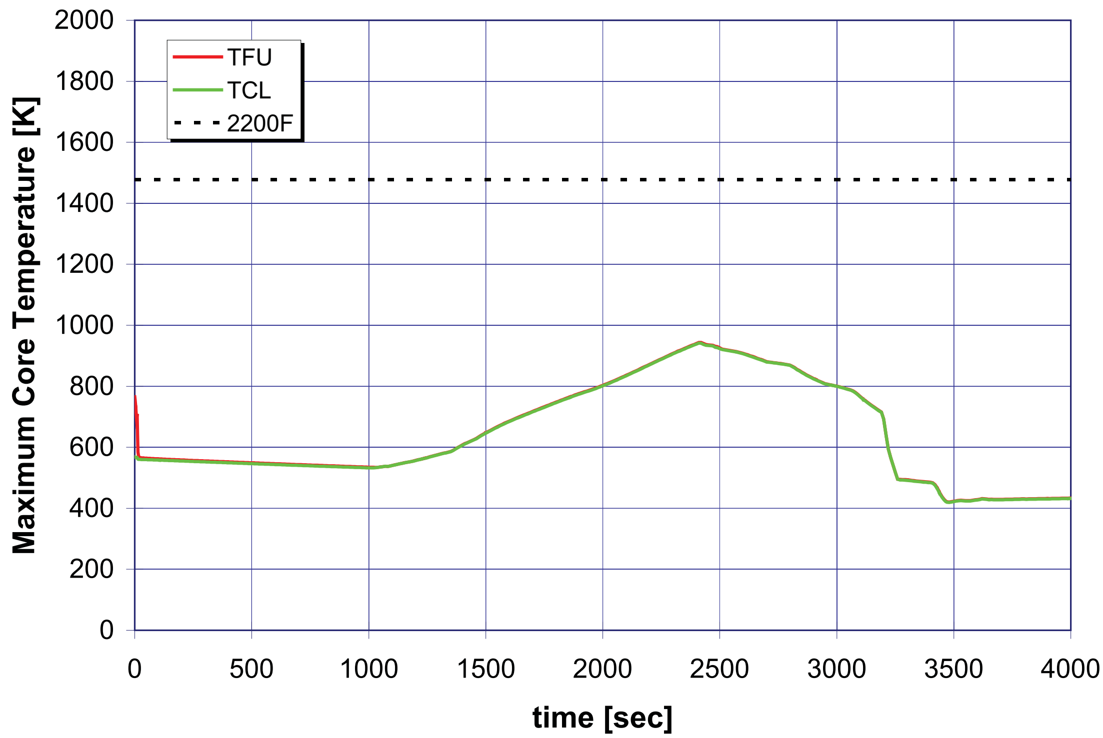
Reactor Water Level (m)
time [sec]
JP
DC
Core (R1)
BAF
TAF



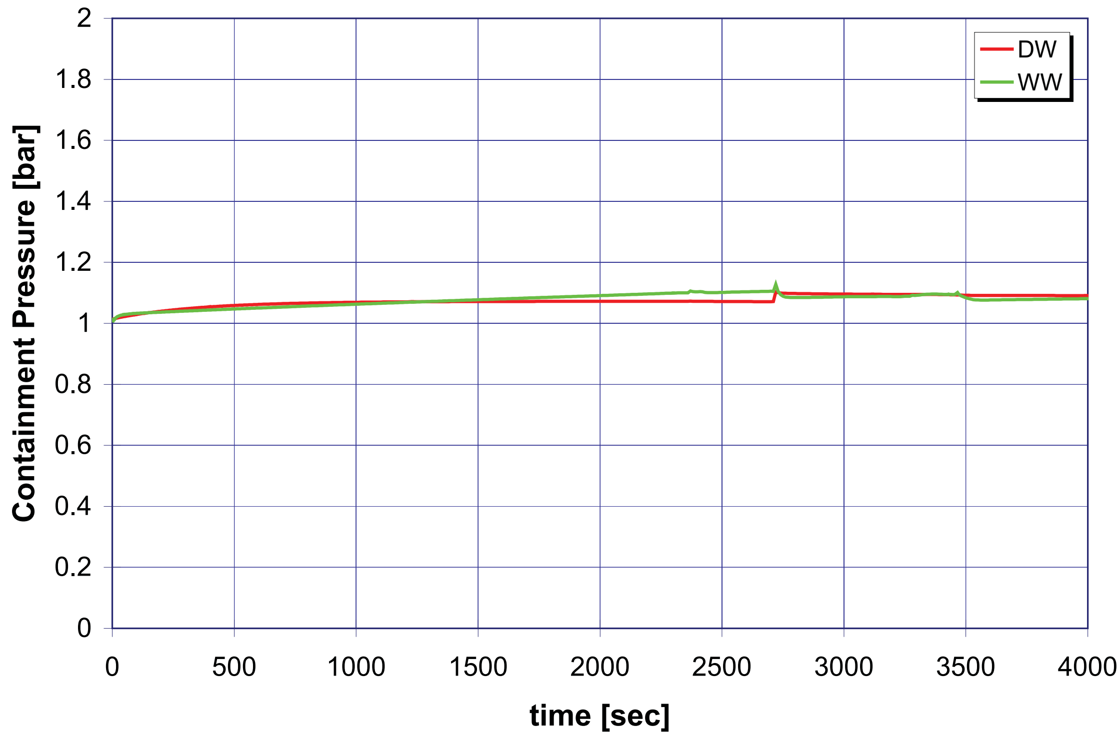
Total Mass Flow Rate [kg/s]
time [sec]
CRD Pump 2 Start
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD



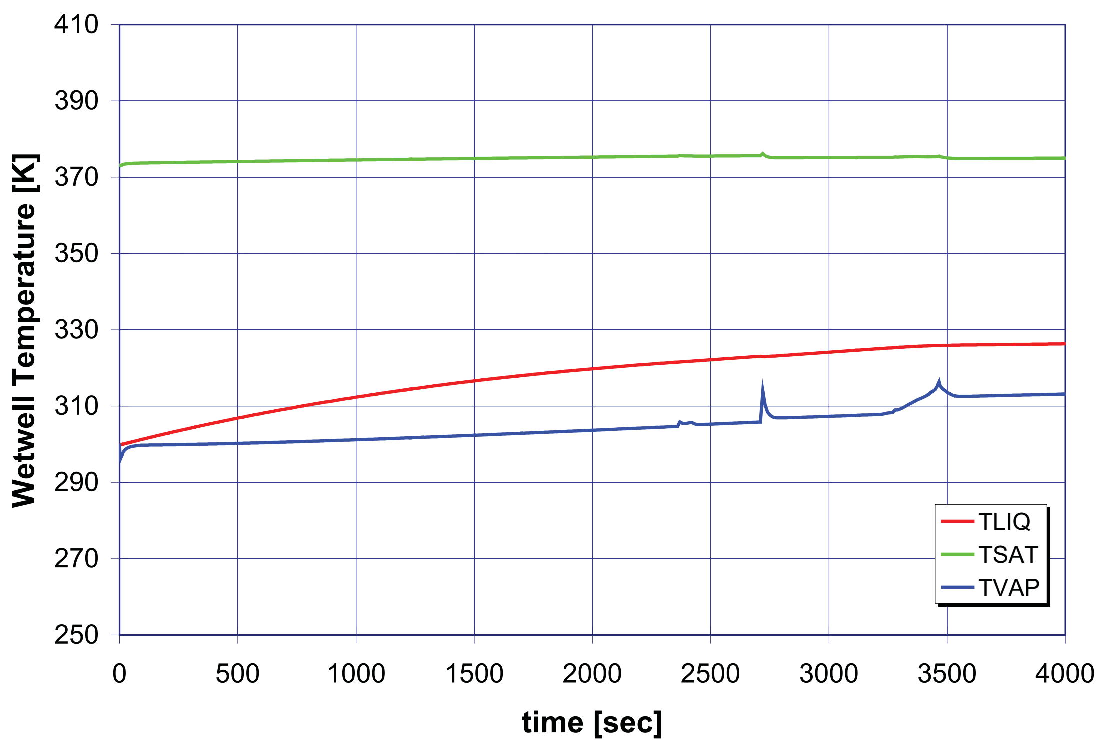
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
Total Mass Flow Rate [kg/s]
time [sec]
0
100
200
300
400
500
600
700
800
900
0
500
1000
1500
2000
2500
3000
3500
4000



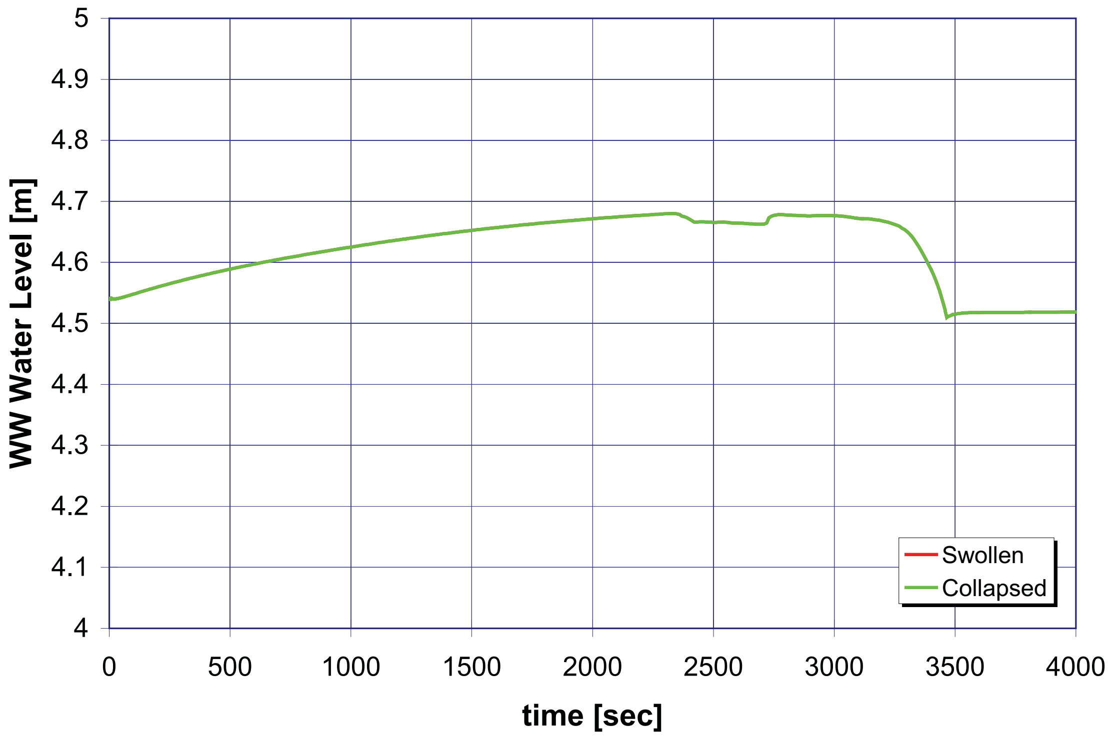
TFU
TCL
2200F
Maximum Core Temperature [K]
time [sec]
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
500
1000
1500
2000
2500
3000
3500
4000



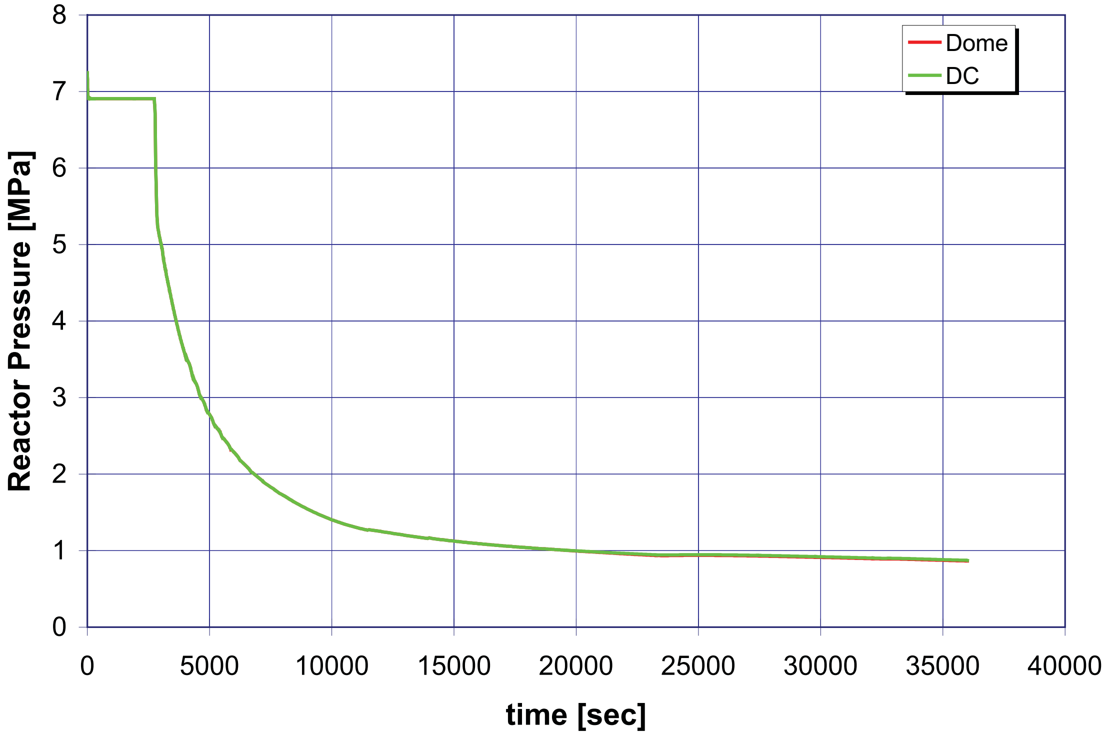
Containment Pressure [bar]
time [sec]
DW
WW
0
0.2
0.4
0.6
0.8
1
1.2
1.4
1.6
1.8
2
0
500
1000
1500
2000
2500
3000
3500
4000

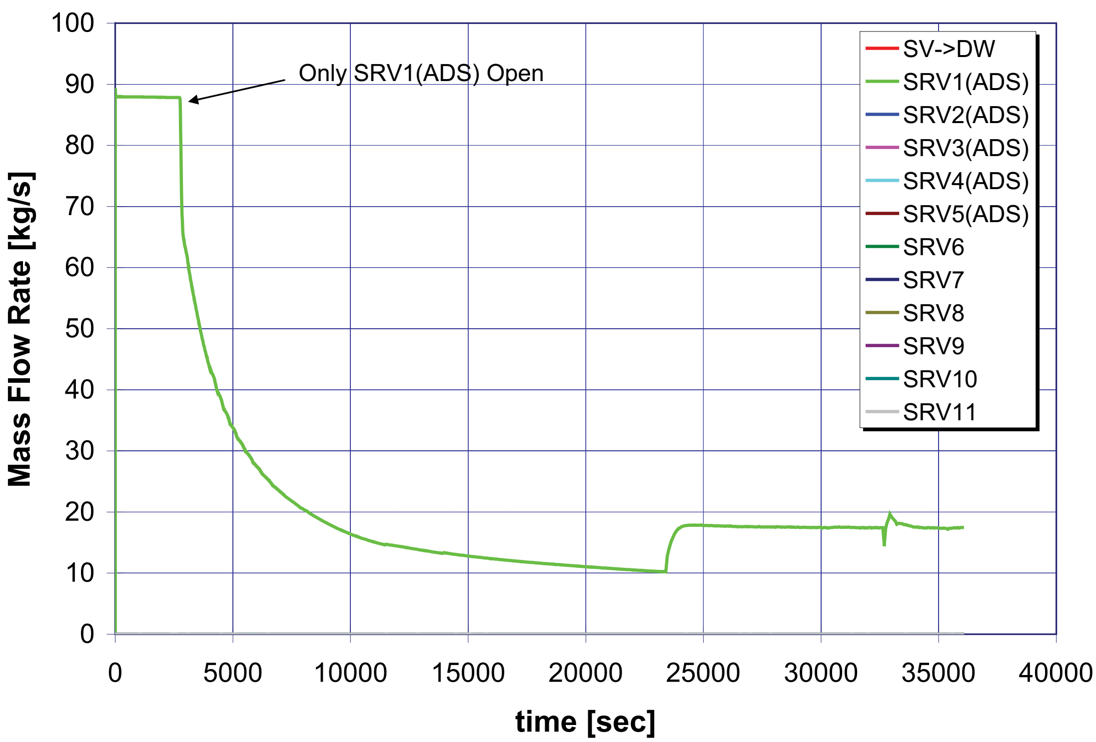


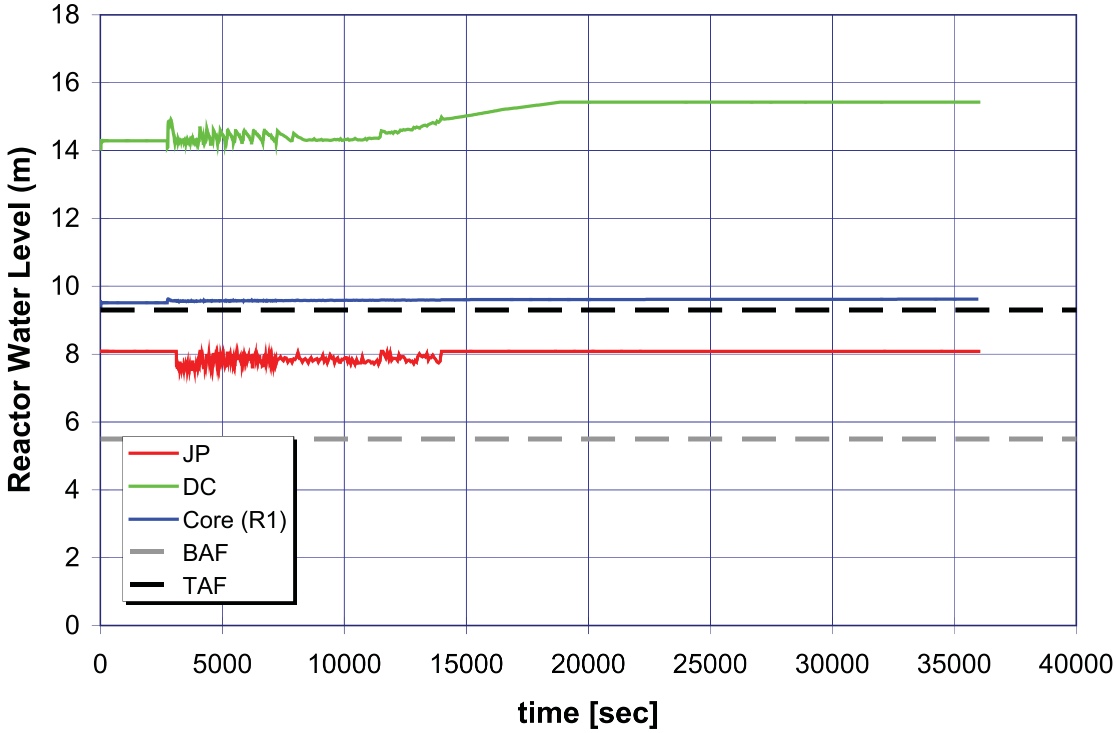
Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP
250
270
290
310
330
350
370
390
410
0
500
1000
1500
2000
2500
3000
3500
4000


WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
Swollen
Collapsed

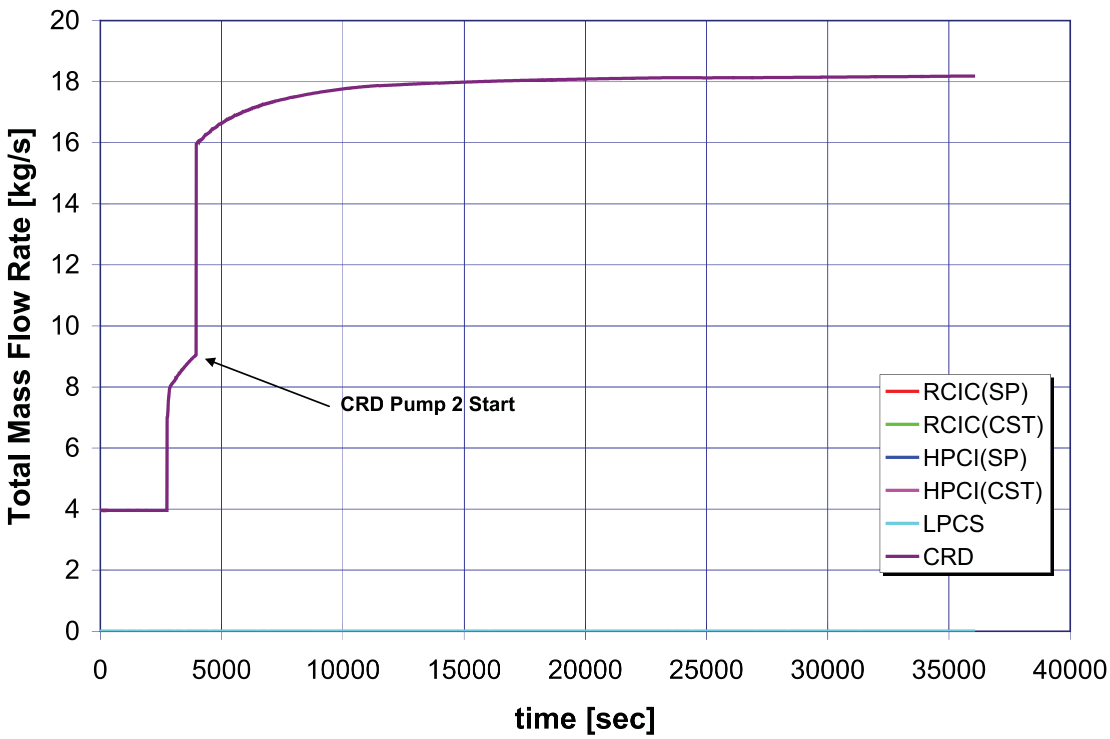
B.1.4.2 Case 4b: Safety Relief Valve Opening, CRD 2 at 20 Minutes, and No Reactor or Feedwater Trip at t = 0



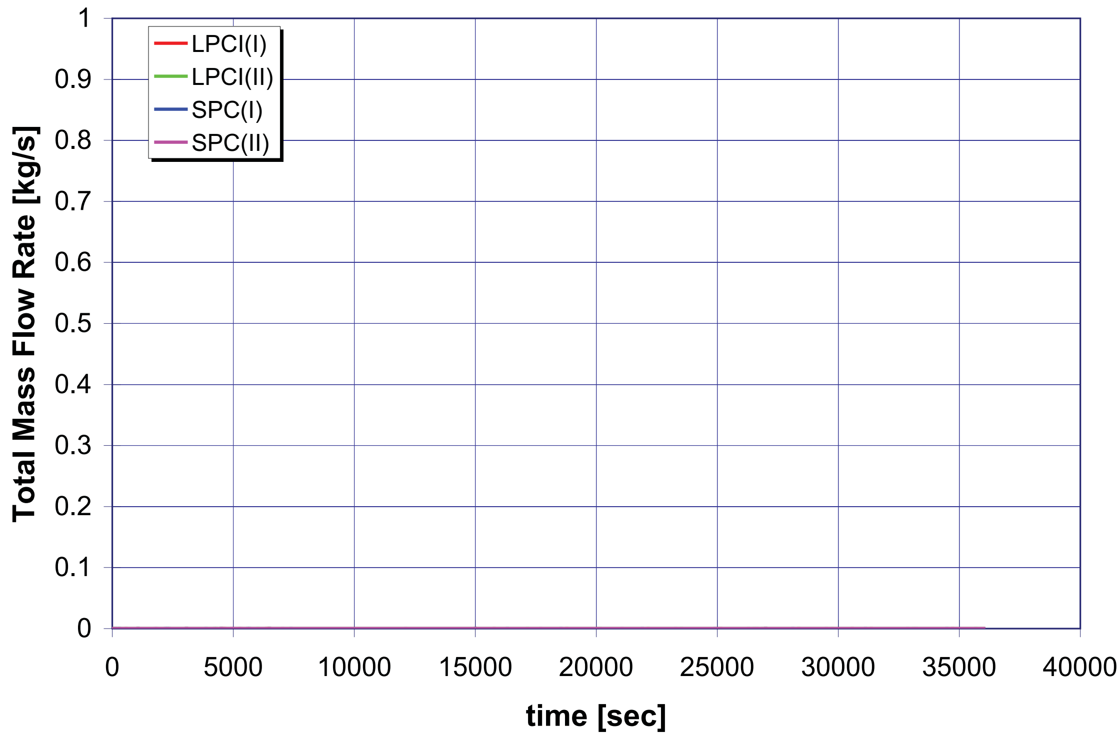




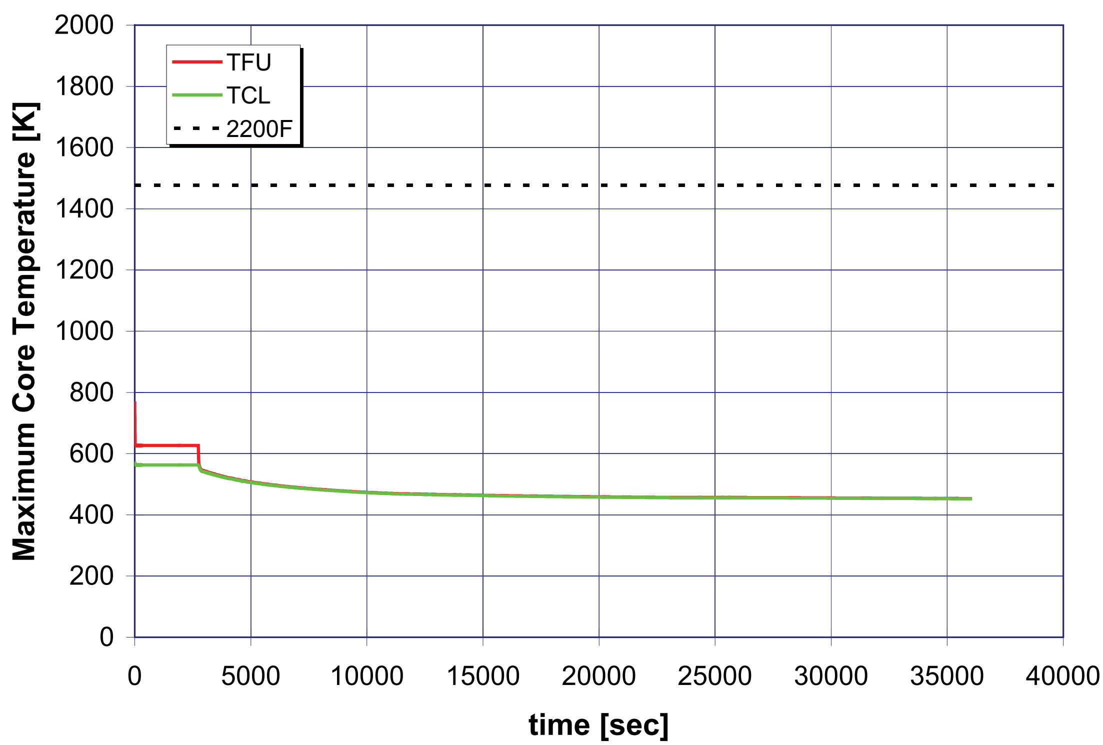
Reactor Water Level (m)
time [sec]
JP
DC
Core (R1)
BAF
TAF



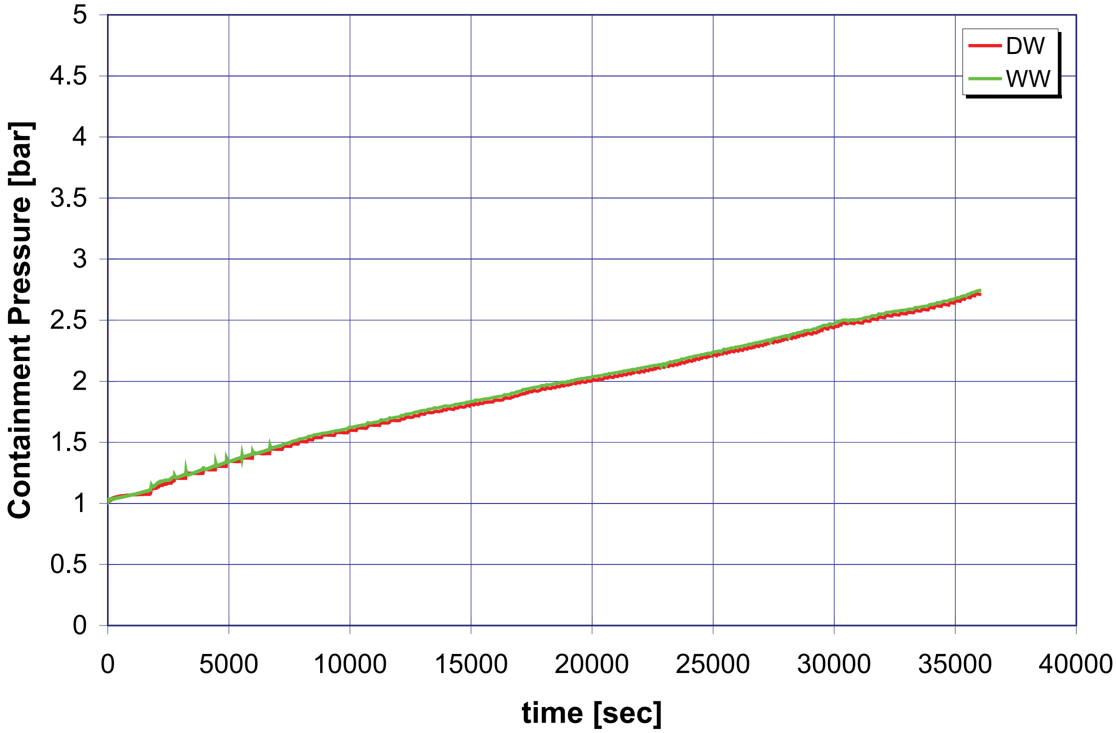
Total Mass Flow Rate [kg/s]
time [sec]
CRD Pump 2 Start
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD

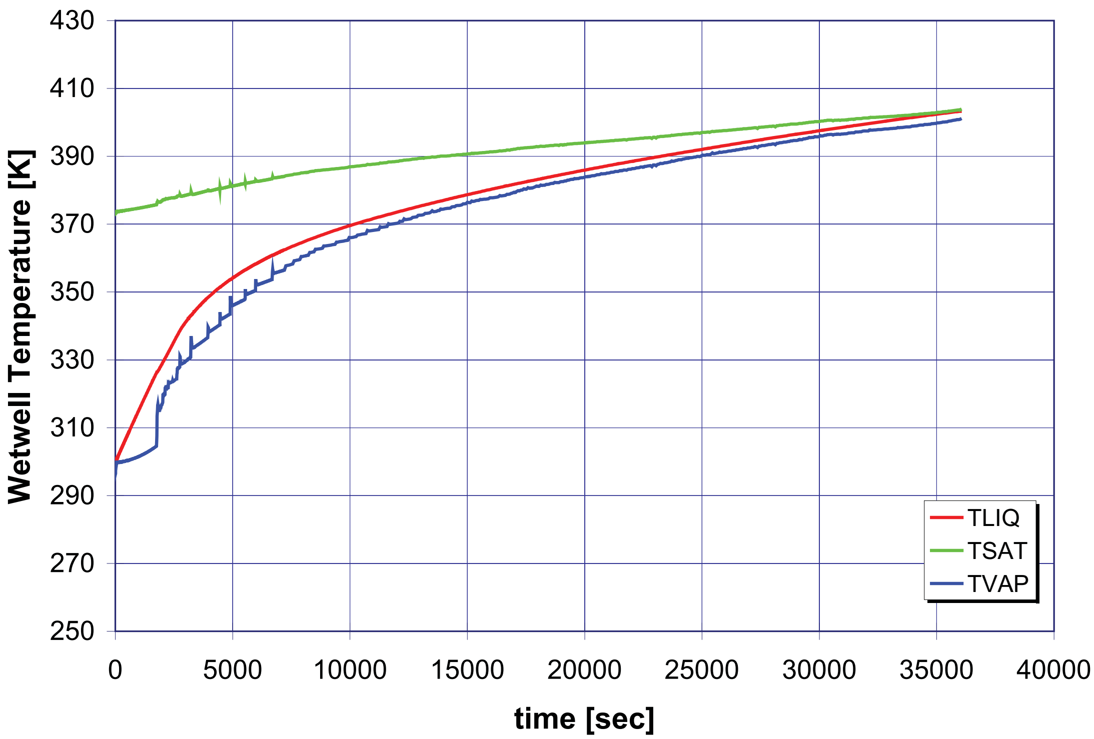


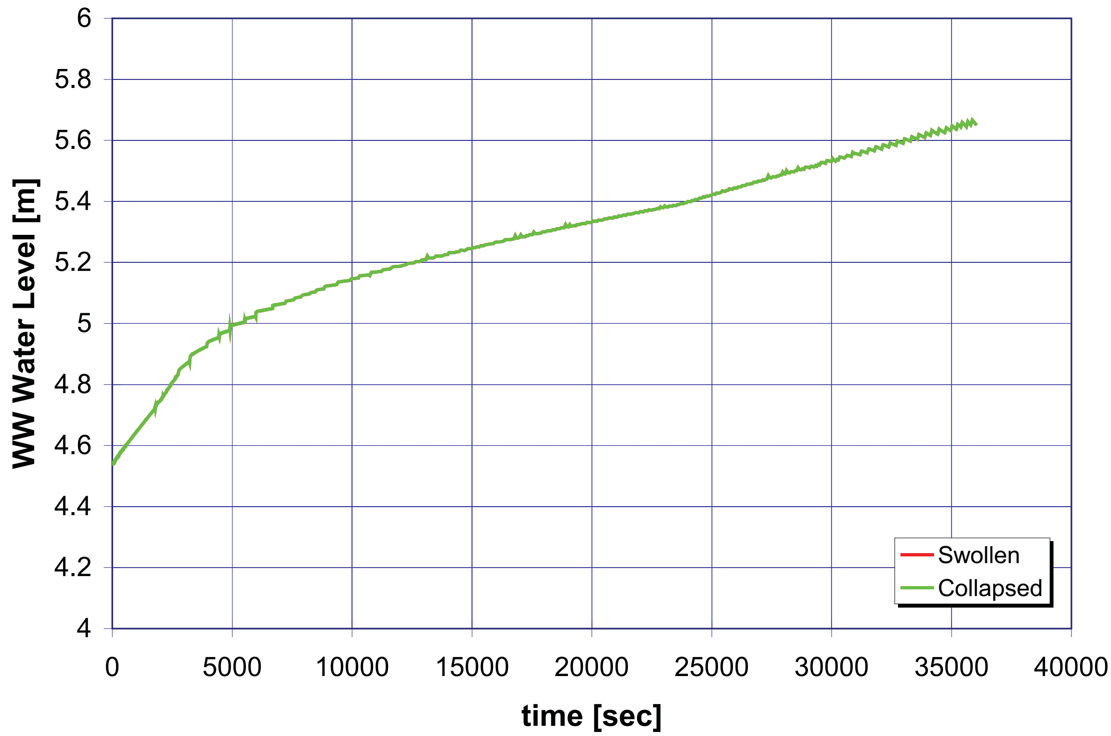
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
Total Mass Flow Rate [kg/s]
time [sec]



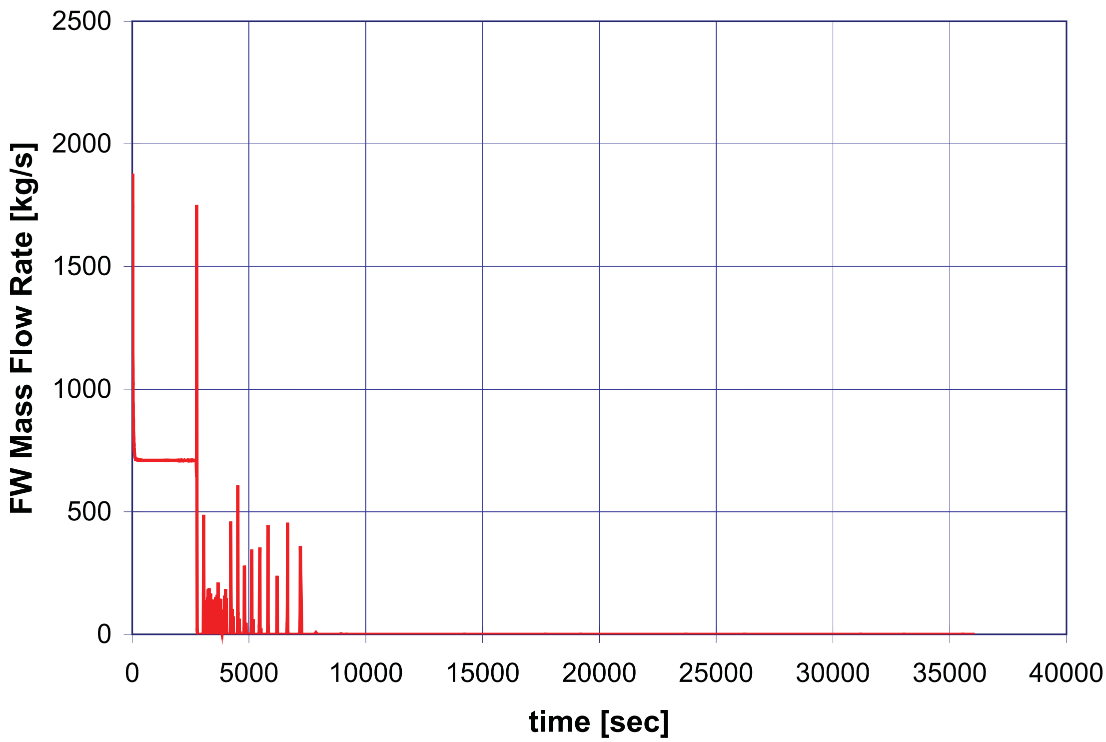
TFU
TCL
2200F
Maximum Core Temperature [K]
time [sec]


Containment Pressure [bar]
time [sec]
DW
WW
5
4.5
4
3.5
3
2.5
2
1.5
1
0.5
0
0
5000
10000
15000
20000
25000
30000
35000
40000


Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP
430
410
390
370
350
330
310
290
270
250
0
5000
10000
15000
20000
25000
30000
35000
40000

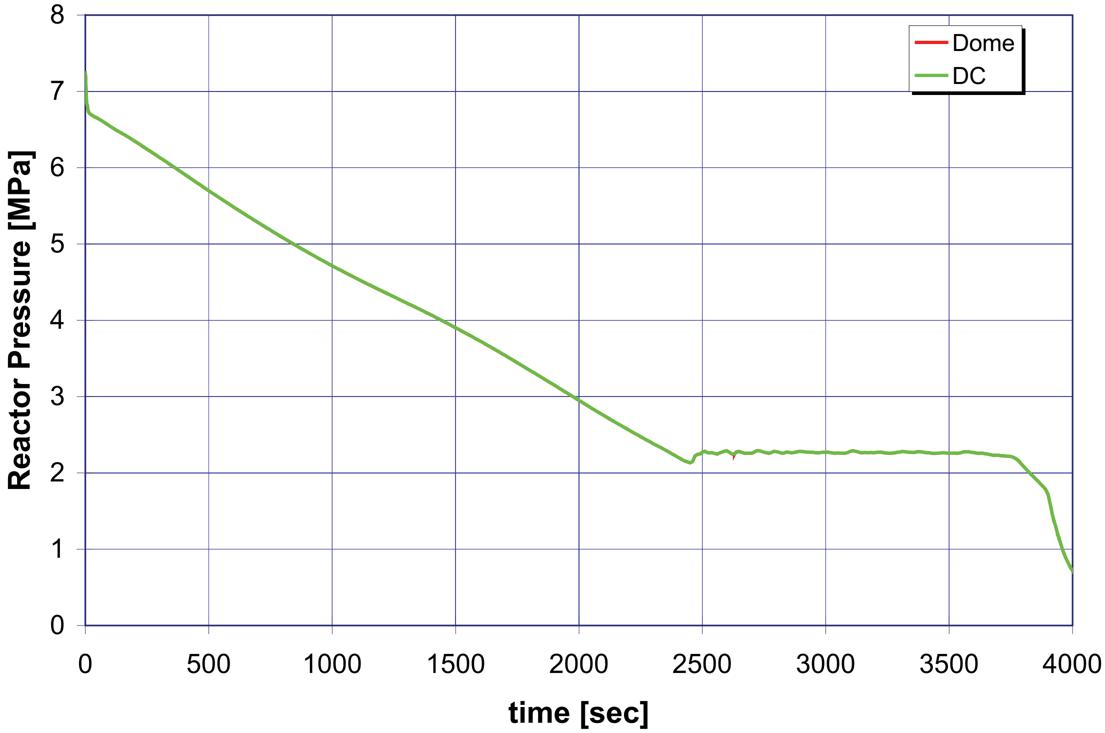


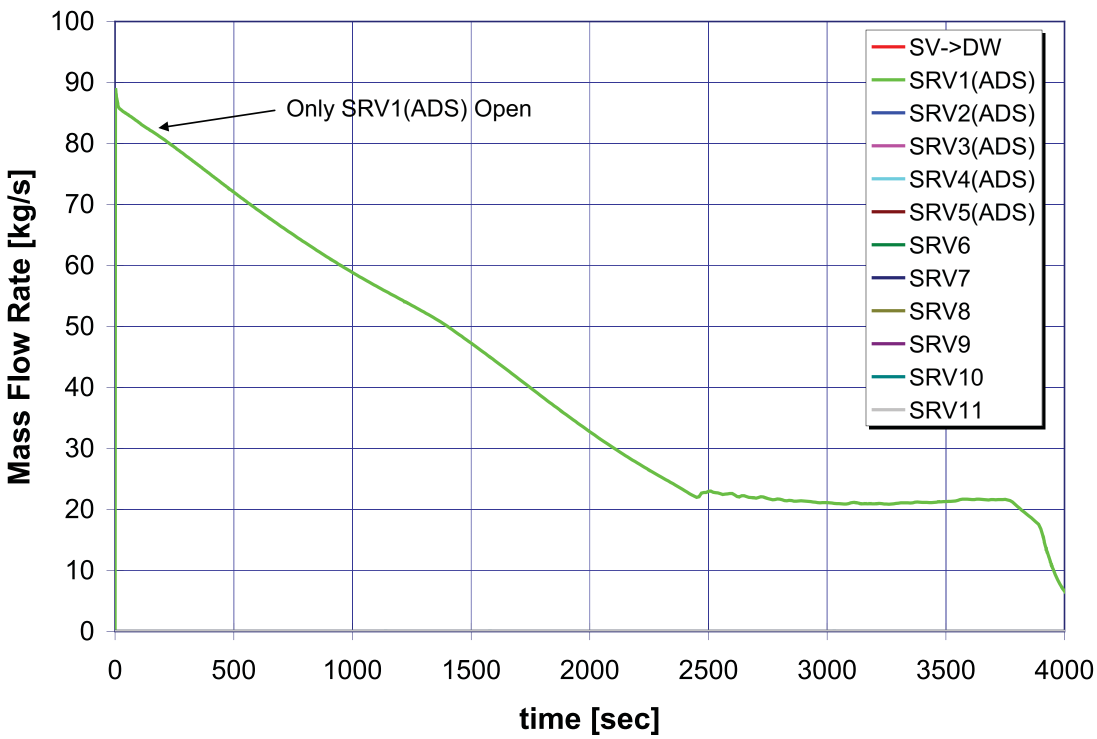
WW Water Level [m]
time [sec]
Swollen
Collapsed

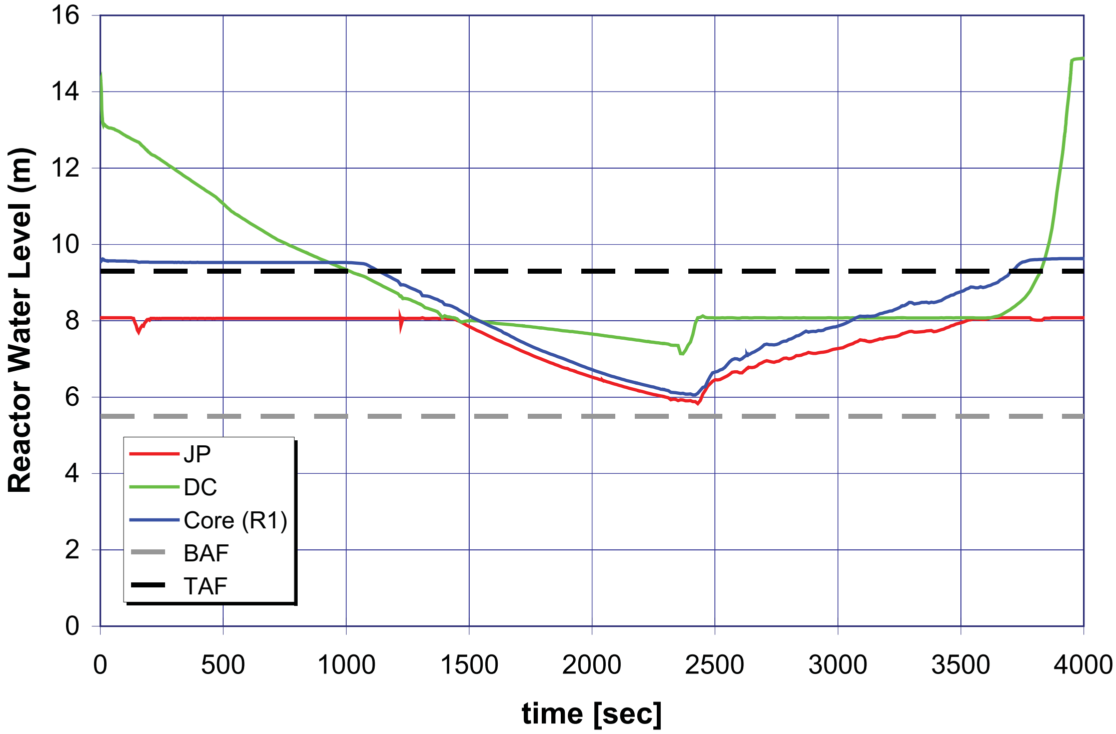


FW Mass Flow Rate [kg/s]
time [sec]

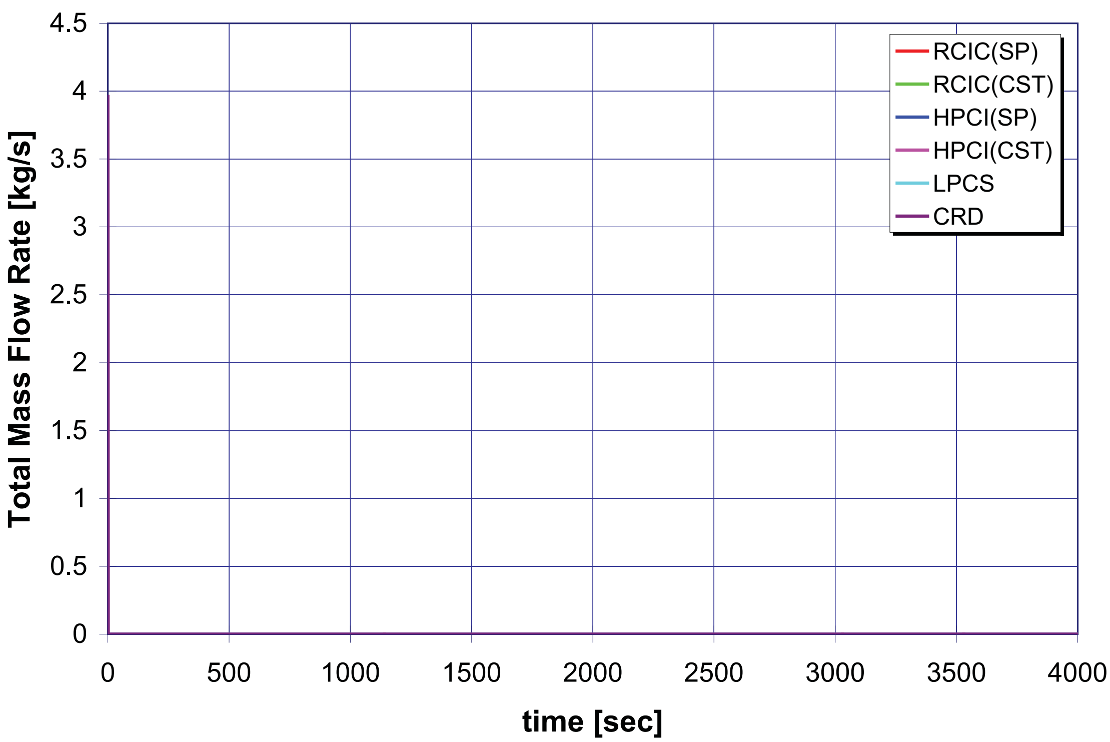
B.1.5 Case 5: Safety Relief Valve Opening



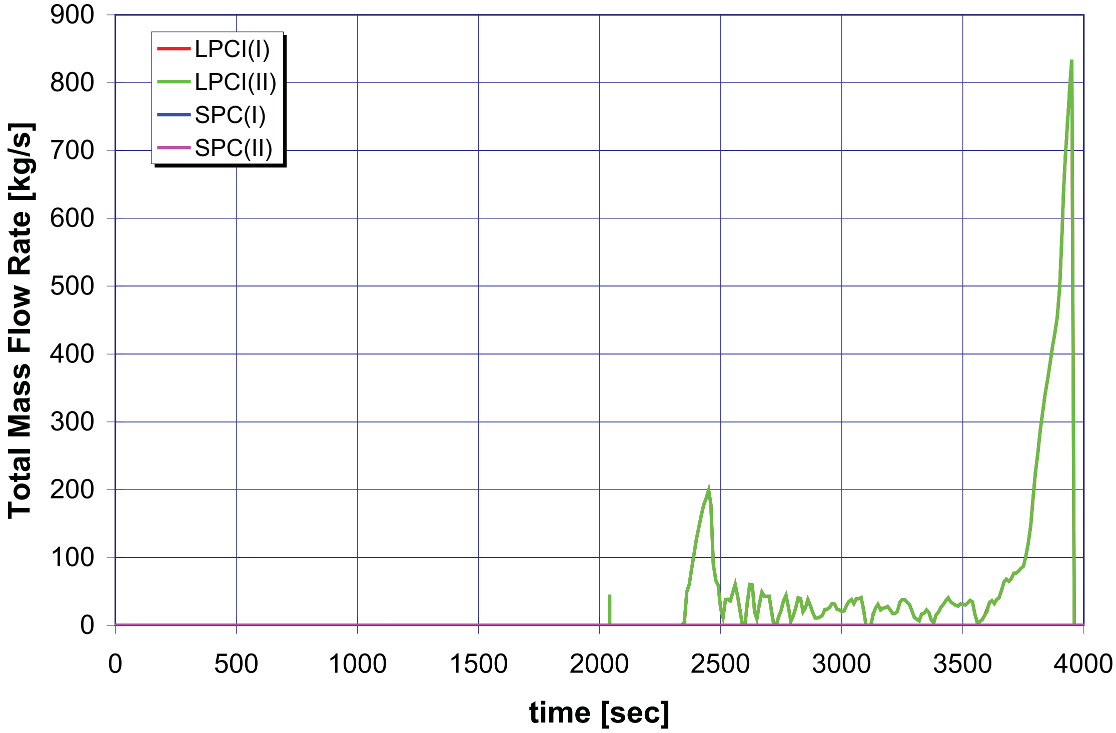


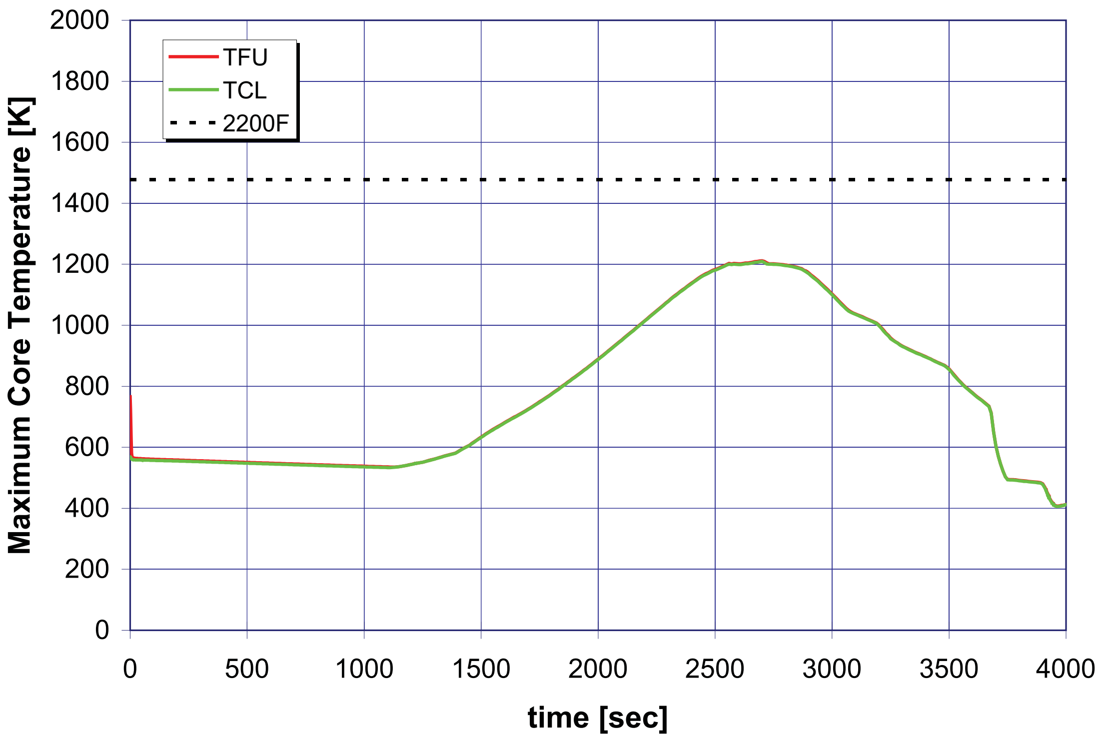


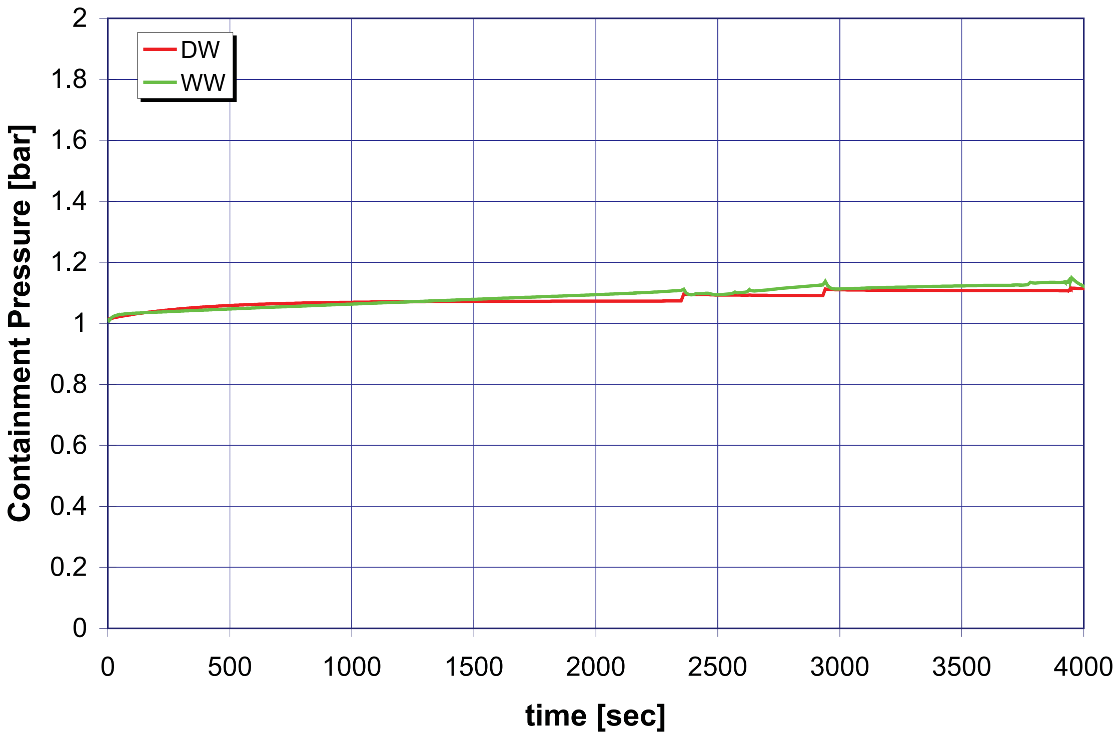
16
14
12
10
8
6
4
2
0
Reactor Water Level (m)
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
JP
DC
Core (R1)
BAF
TAF



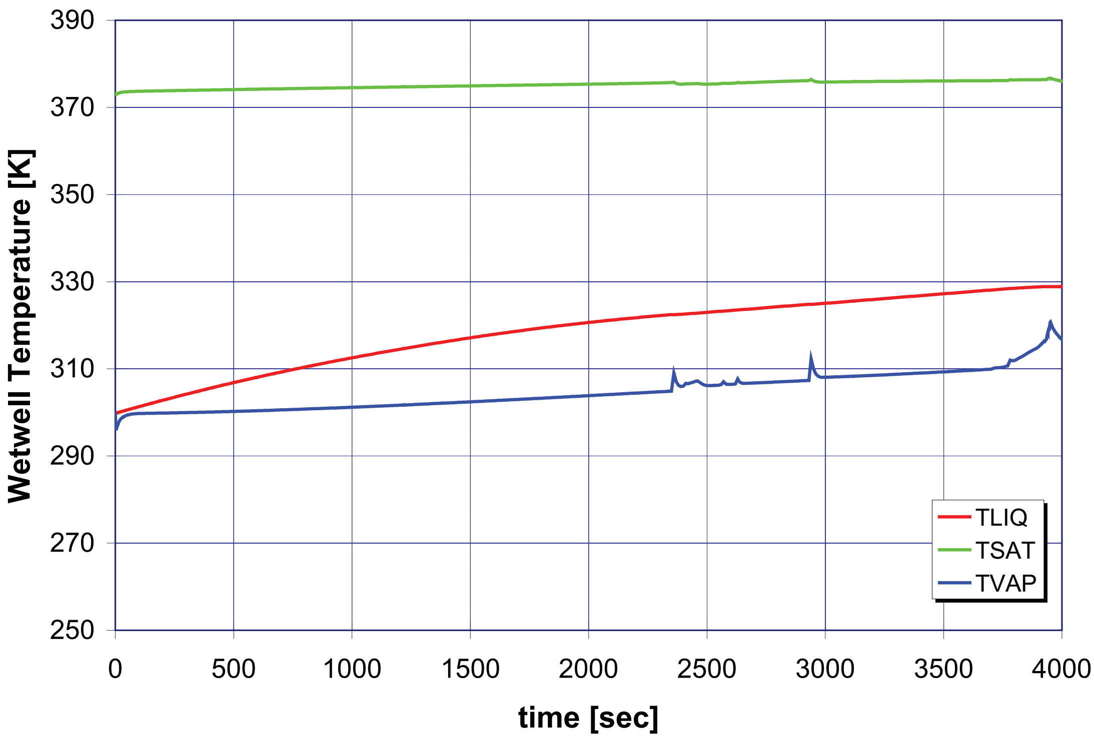
4.5
4
3.5
3
2.5
2
1.5
1
0.5
0
Total Mass Flow Rate [kg/s]
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD


Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
100
200
300
400
500
600
700
800
900
0
500
1000
1500
2000
2500
3000
3500
4000

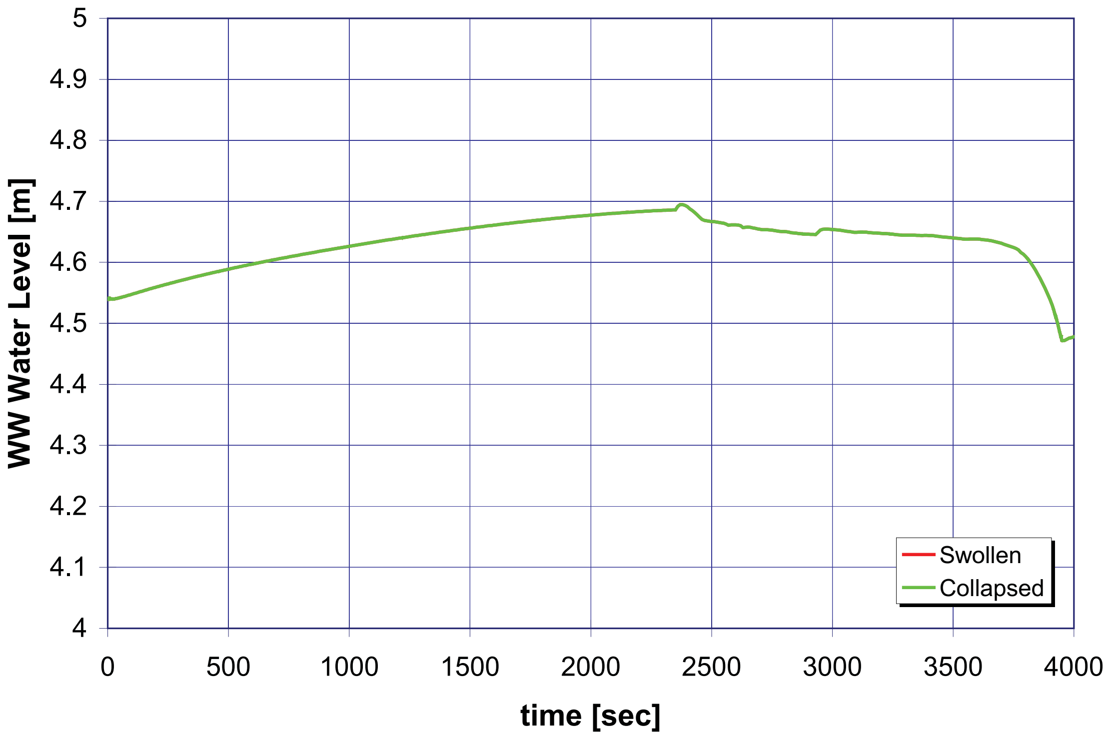

Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
500
1000
1500
2000
2500
3000
3500
4000



Containment Pressure [bar]
time [sec]
DW
WW
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
500
1000
1500
2000
2500
3000
3500
4000



Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP
390
370
350
330
310
290
270
250
0
500
1000
1500
2000
2500
3000
3500
4000



5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
WW Water Level [m]
0
500
1000
1500
2000
2500
3000
3500
4000
time [sec]
Swollen
Collapsed

B.2 Boiling-Water Reactor Station Blackout

Analysis Summary

Table 3 through Table 6 below provide results for this portion of the analysis.

Table 3 Peach Bottom Station Blackout Results

<table>
<tr><th>Case</th><th>RCIC</th><th>HPCI</th><th>ac/dc</th><th>SRV Sticks Open?</th><th>HCTL Depress?</th><th>Core Uncovery (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td rowspan="3">No</td><td rowspan="6">No</td><td>-</td><td>No[1]</td><td rowspan="4">No</td><td>0.5</td><td>1.2</td></tr>
<tr><td>1a</td><td>ac recovery at 1.2 hr</td><td>No</td><td>0.5</td><td>1.2[2]</td></tr>
<tr><td>2</td><td>-</td><td>At t = 0</td><td>0.3</td><td>0.8</td></tr>
<tr><td>3</td><td rowspan="4">Yes</td><td rowspan="2">dc is always available</td><td rowspan="3">No</td><td>17.7</td><td>19.4</td></tr>
<tr><td>4</td><td>Yes</td><td>6.0</td><td>7.2</td></tr>
<tr><td>5</td><td>2 hr of dc</td><td rowspan="3">No</td><td>3.3</td><td>4.3</td></tr>
<tr><td>6</td><td rowspan="3">dc is always available</td><td>At 187 lifts</td><td>6.0</td><td>7.2</td></tr>
<tr><td>7</td><td rowspan="4">No</td><td rowspan="4">Yes</td><td rowspan="3">No</td><td>17.5</td><td>19.3</td></tr>
<tr><td>8</td><td>Yes</td><td>9.3</td><td>10.8</td></tr>
<tr><td>9</td><td>2 hr of dc</td><td rowspan="2">No</td><td>3.8</td><td>4.9</td></tr>
<tr><td>10</td><td>dc is always available</td><td>At 187 lifts</td><td>9.2</td><td>10.7</td></tr>
</table>

[1] For this case, the SRV does not stick open until after core damage, so this assumption does not affect the outcome.

[2] Recovery of injection upon reaching 2,200 °F (1,204 °C) quickly arrests further heatup.

Table 4 Peach Bottom Station Blackout Key Timings (Cases 1, 1a, and 2)

Event	Case 1 (hr)	Case 1a (hr)	Case 2 (hr)
Reactor trip, MSIV closure	0	0	0
Downcomer level reaches L2	0.16	0.16	0.16
Downcomer level reaches L1	0.50	0.50	0.27
Downcomer level below TAF	0.50	0.50	0.27
Gap release: 900 °C (1,652 °F)	1.02	1.02	0.69
Core damage: max. temp. >1,204 °C (2,200 °F)	1.17	1.17	0.79
HPCI, RCIC, CRD injection start	-	1.17	-
ADS actuated	-	1.24	-
Downcomer level recovers above TAF	-	1.27	-
SRV sticks open due to high # of cycles	1.75	-	-

Table 5 Peach Bottom Station Blackout Key Timings (Cases 3–6)

Event	Case 3 (hr)	Case 4 (hr)	Case 5 (hr)	Case 6 (hr)
Reactor trip, MSIV closure	0	0	0	0
Downcomer level first reaches L2	0.16	0.16	0.16	0.16
RCIC started (CST injection mode)	0.17	0.17	0.17	0.17
RCIC fails due to loss of dc	-	-	2.00	-
HCTL limit reached	2.46 (no action taken)	2.46	2.46 (no action taken)	2.46 (no action taken)
SRV sticks open due to high # of cycles	-	-	-	2.47
RCIC NPSH limit exceeded[1]	12.67	-	-	-
RCIC pump isolation: low steamline pressure <0.52 MPa (75 psig)	-	3.90	-	3.92
RCIC injection ends due to CST level <5 ft (1.5 m)	14.43	-	-	-
Downcomer level reaches L1	17.68	5.61	3.25	5.62
Downcomer level below TAF	17.68	5.61	3.25	5.62
Gap release: 900 °C (1,652 °F)	19.06	6.99	4.04	7.00
Core damage max. temp. >1,204 °C (2,200 °F)	19.42	7.17	4.25	7.18
Exhaust pressure exceeded: 0.35 MPa (50 psig)	20.14	-	-	-

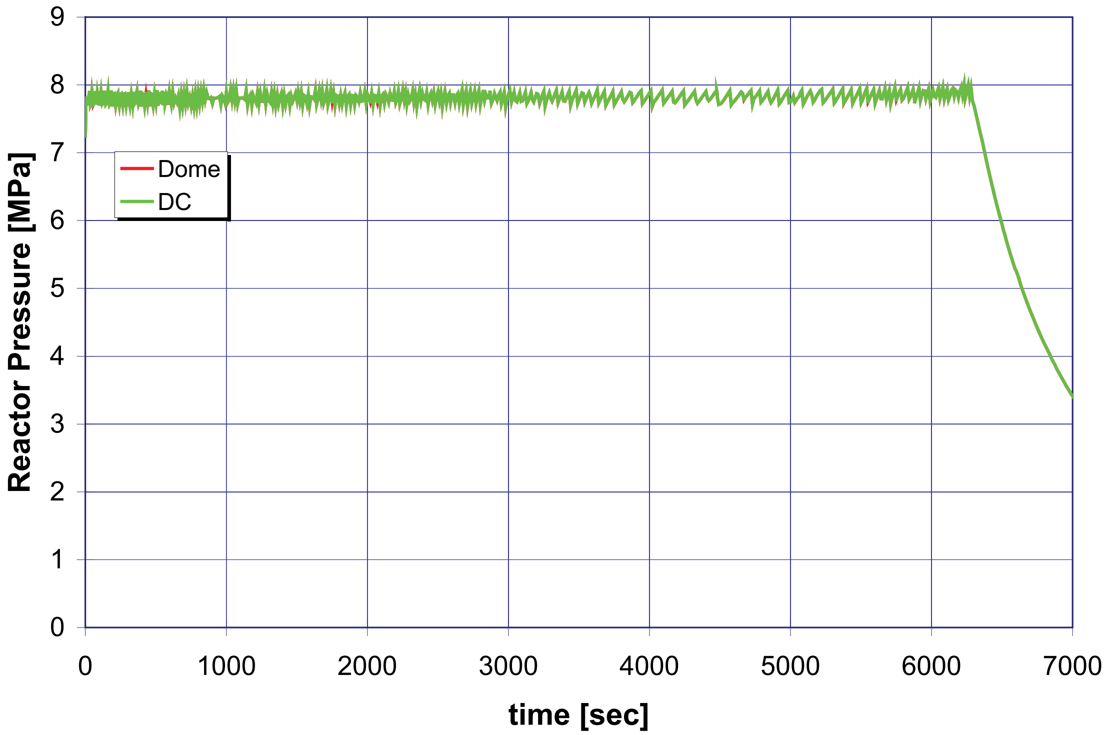
[1] Switchover to the suppression pool is not permitted after this point.

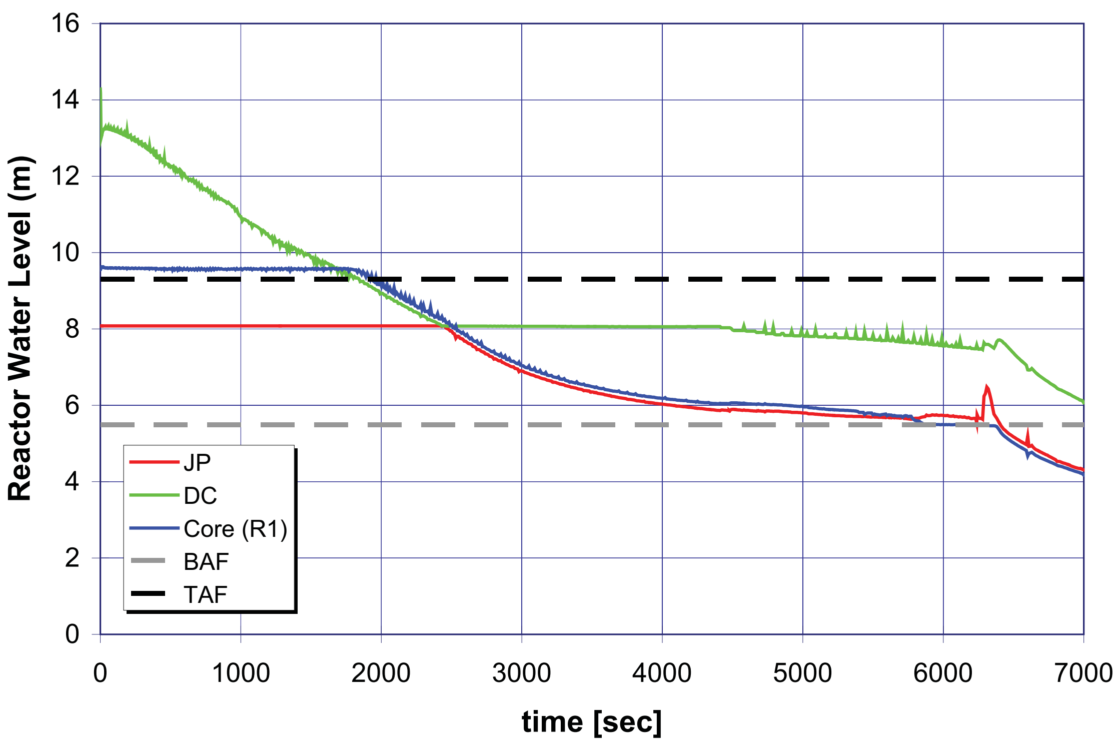
Table 6 Peach Bottom Station Blackout Key Timings (Cases 7–10)

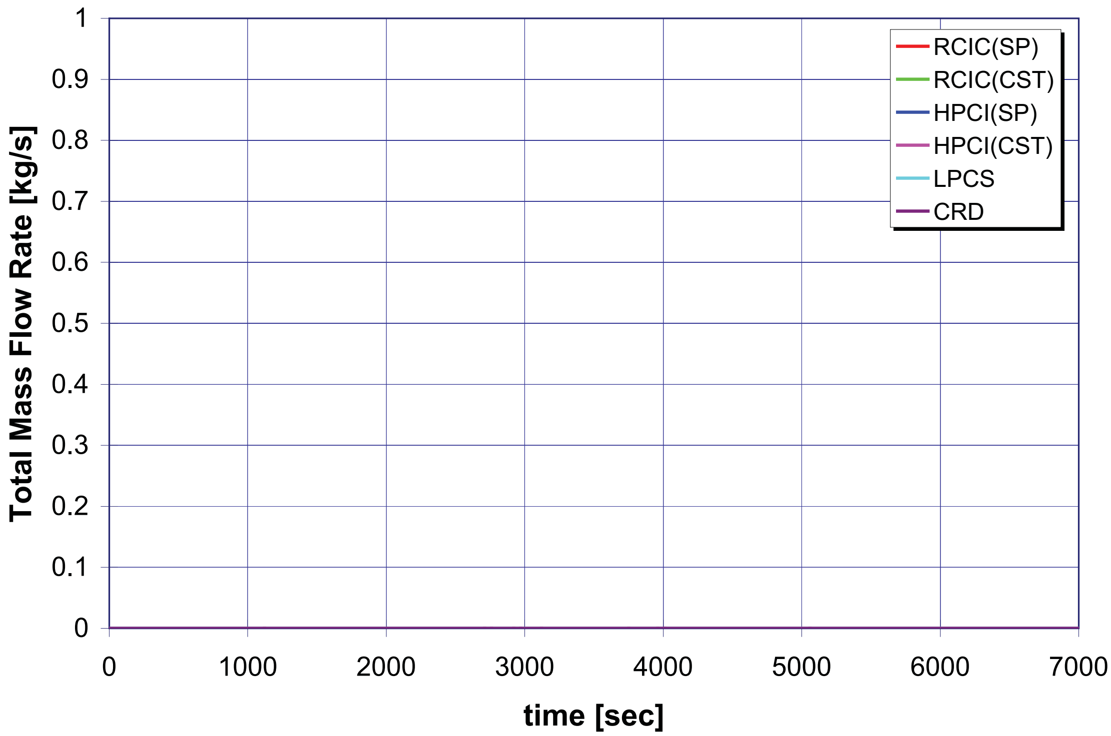
Event	Case 7 (hr)	Case 8 (hr)	Case 9 (hr)	Case 10 (hr)
Reactor trip, MSIV closure	0	0	0	0
Downcomer level first reaches L2	0.16	0.16	0.16	0.16
HPCI started (CST injection mode)	0.17	0.17	0.17	0.17
HPCI fails due to loss of dc	-	-	2.00	-
SRV sticks open due to high # of cycles	-	-	-	2.53
HCTL limit reached	2.67 (no action taken)	2.67	2.67 (no action taken)	2.67 (no action taken)
HPCI NPSH limit exceeded[1]	12.07	-	-	-
HPCI pump isolation: low steamline pressure <0.52 MPa (75 psig)	-	5.72	-	5.61
HPCI injection ends due to CST level <5 ft (1.5 m)	16.05	-	-	-
Downcomer level reaches L1	17.53	8.97	3.82	8.94
Downcomer level below TAF	17.53	9.06	3.82	8.94
Gap release: 900 °C (1,652 °F)	18.96	10.59	4.63	10.46
Core damage max. temp. >1,204 °C (2,200 °F)	19.31	10.8	4.85	10.68
Exhaust pressure exceeded: 1.04 MPa (150 psig)	-	-	-	-

[1] Switchover to the suppression pool is not permitted after this point.

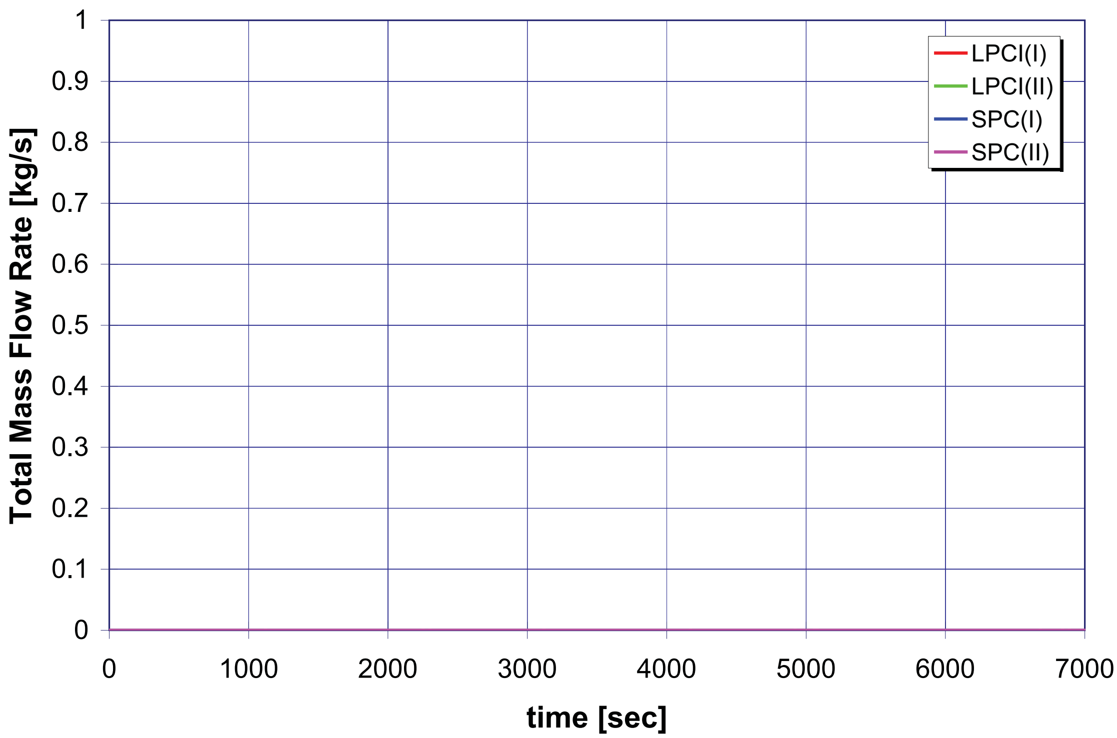
B.2.1 Case 1: Station Blackout with No Injection



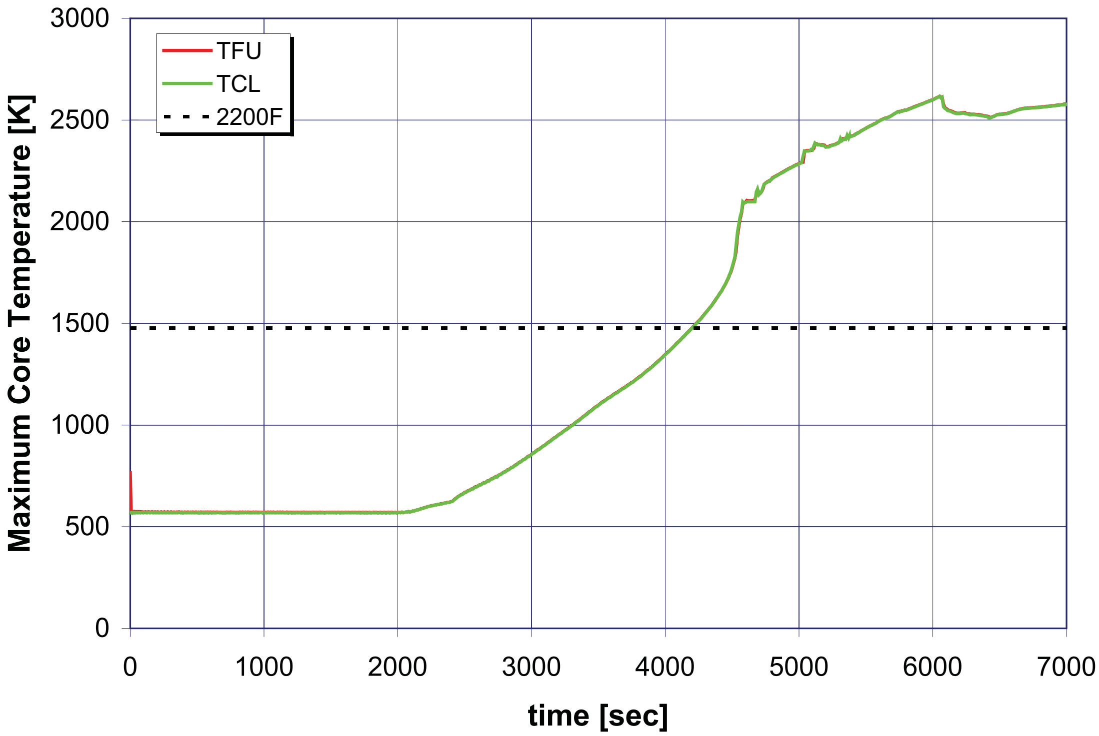




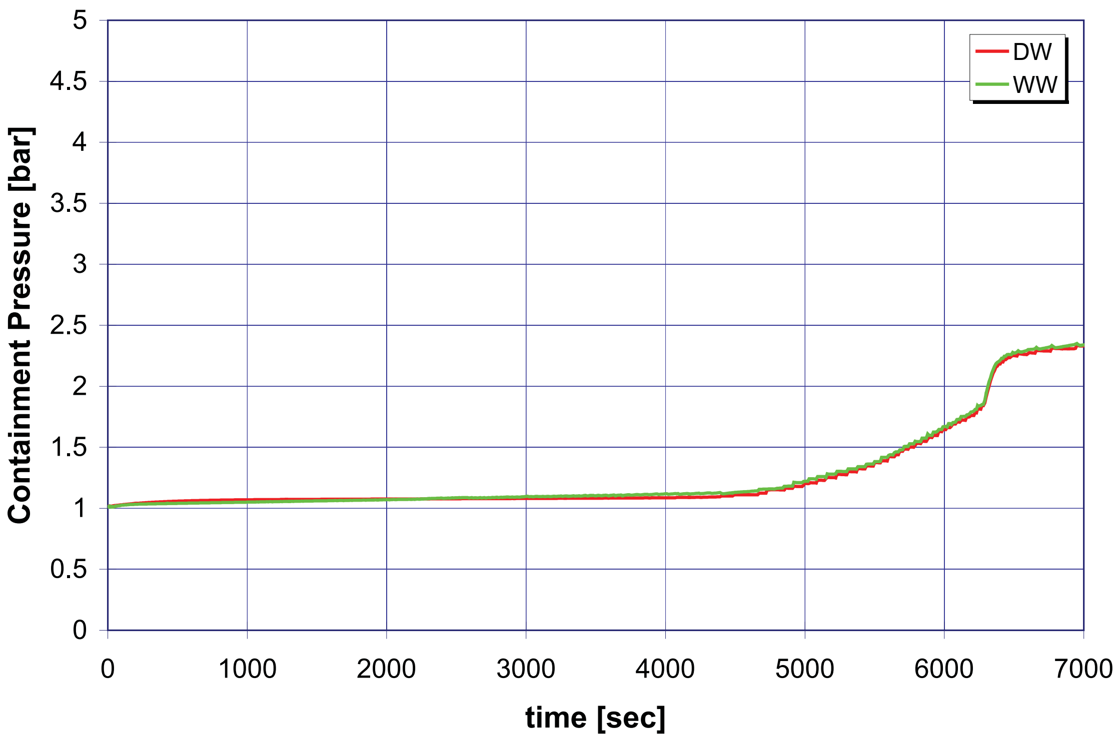
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD
Total Mass Flow Rate [kg/s]
time [sec]



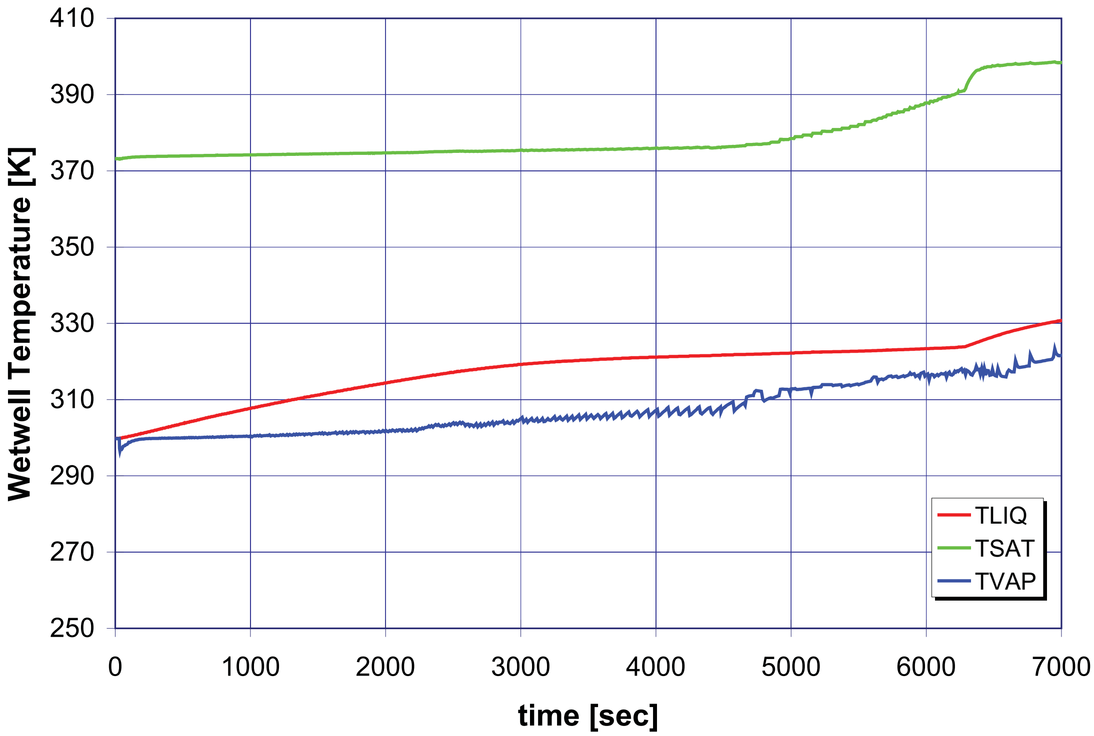
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
Total Mass Flow Rate [kg/s]
time [sec]



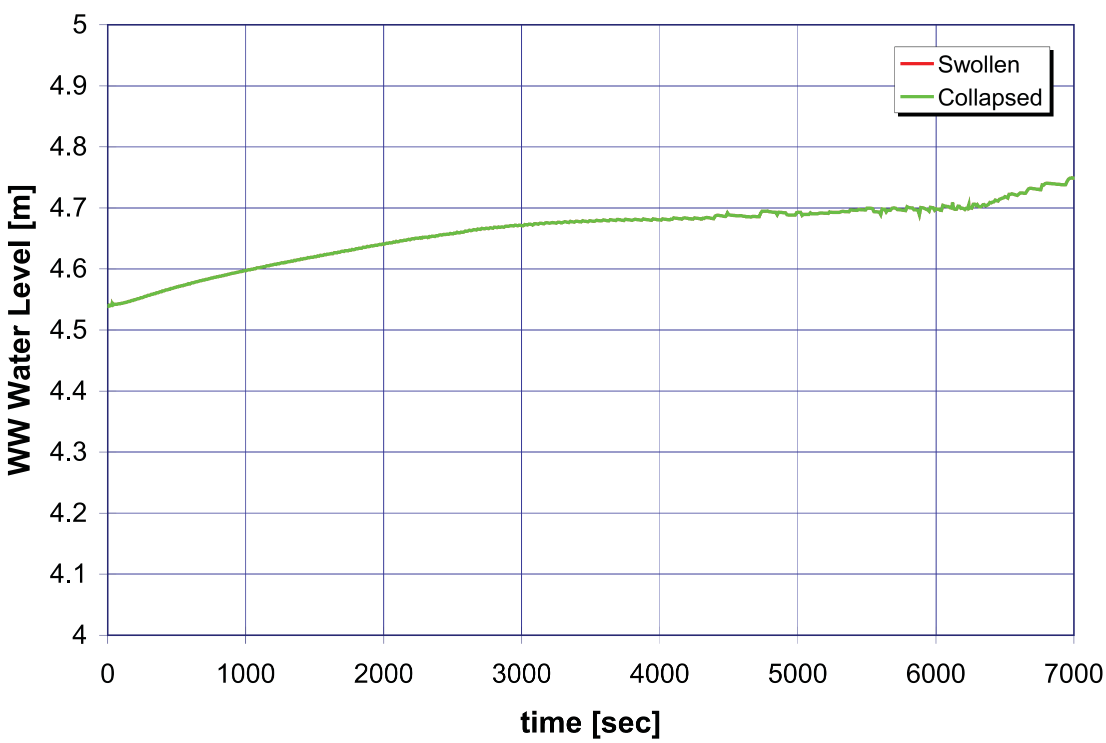
Maximum Core Temperature [K]
TFU
TCL
2200F
0
500
1000
1500
2000
2500
3000
0
1000
2000
3000
4000
5000
6000
7000
time [sec]



Containment Pressure [bar]
DW
WW
0
0.5
1
1.5
2
2.5
3
3.5
4
4.5
5
0
1000
2000
3000
4000
5000
6000
7000
time [sec]

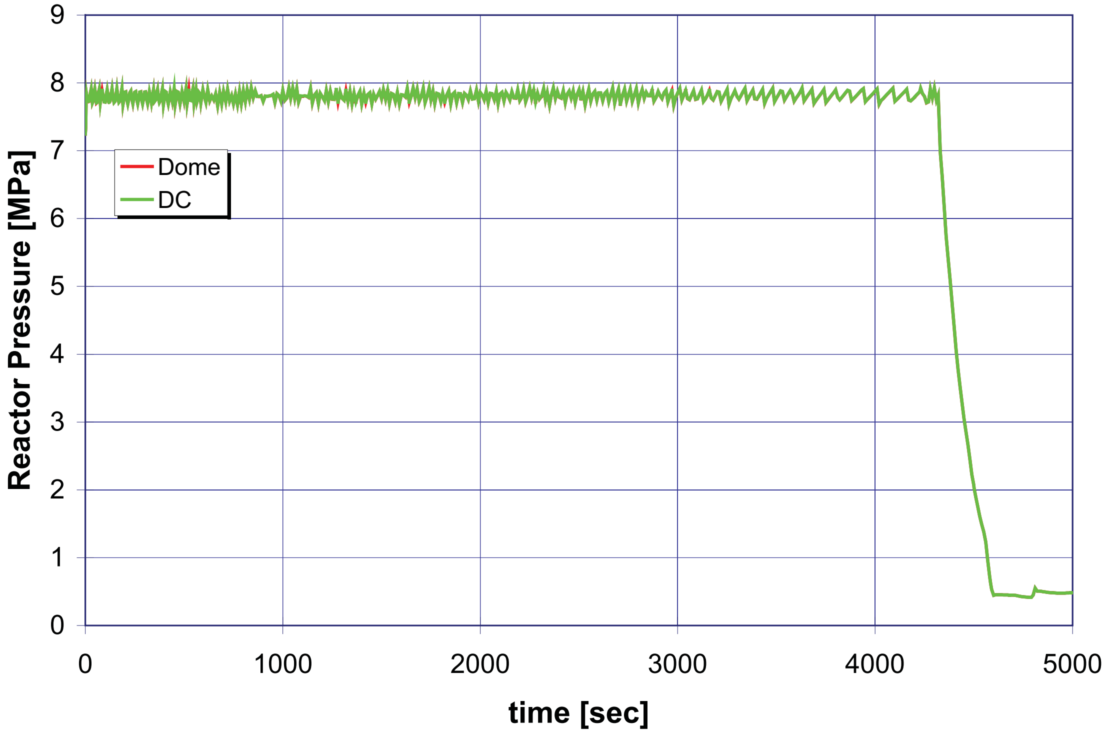


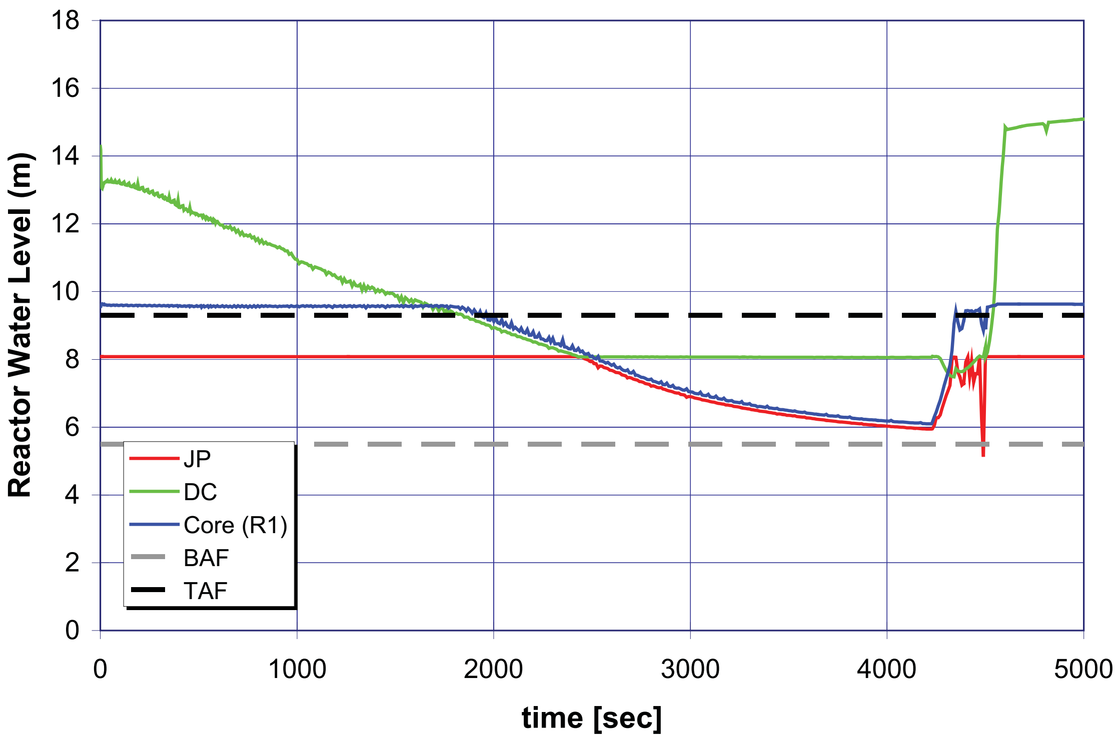
Wetwell Temperature [K]
410
390
370
350
330
310
290
270
250
0
1000
2000
3000
4000
5000
6000
7000
time [sec]
TLIQ
TSAT
TVAP

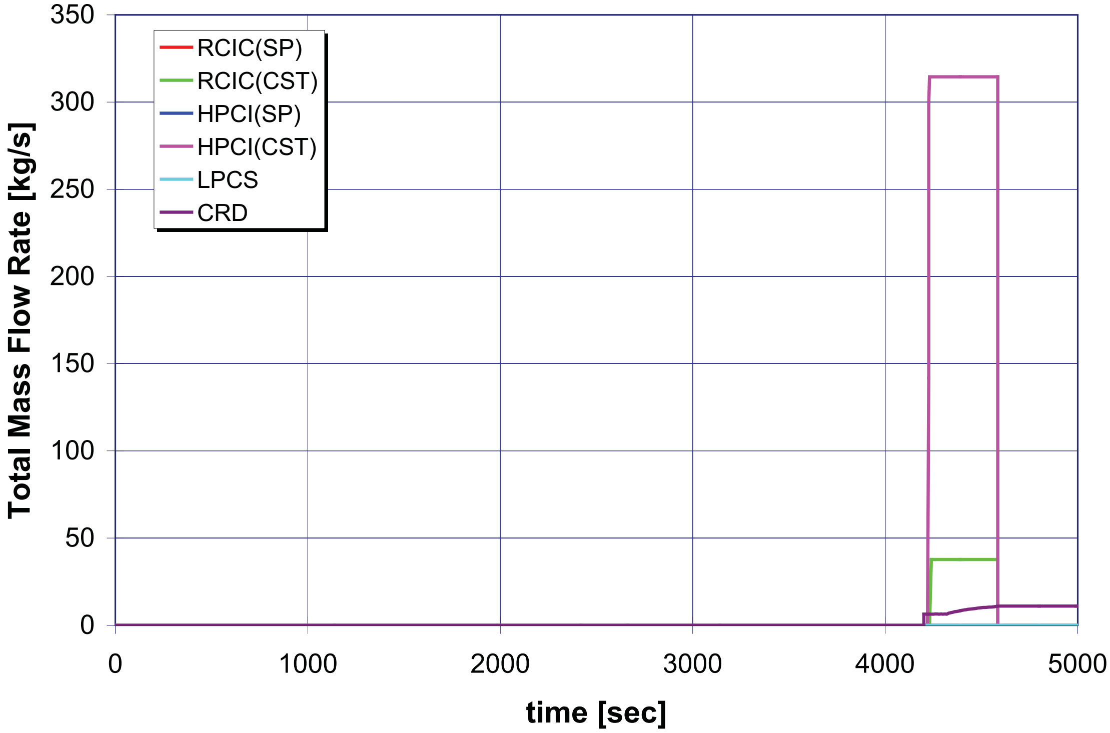


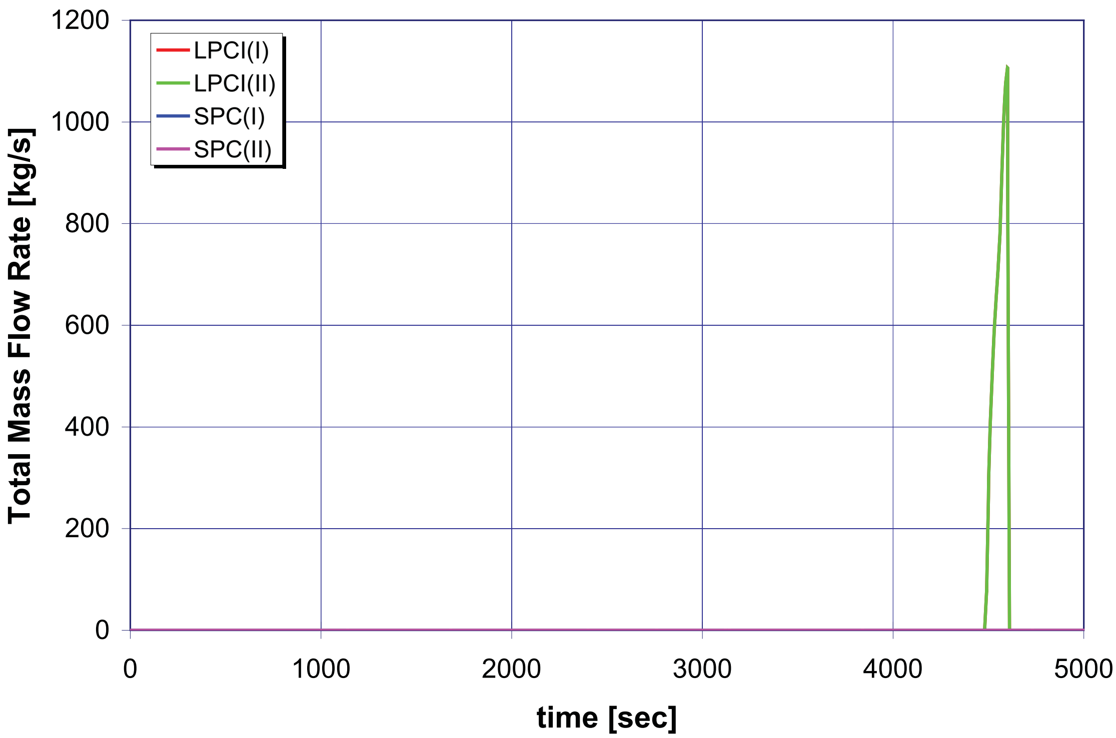
WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
1000
2000
3000
4000
5000
6000
7000
time [sec]
Swollen
Collapsed

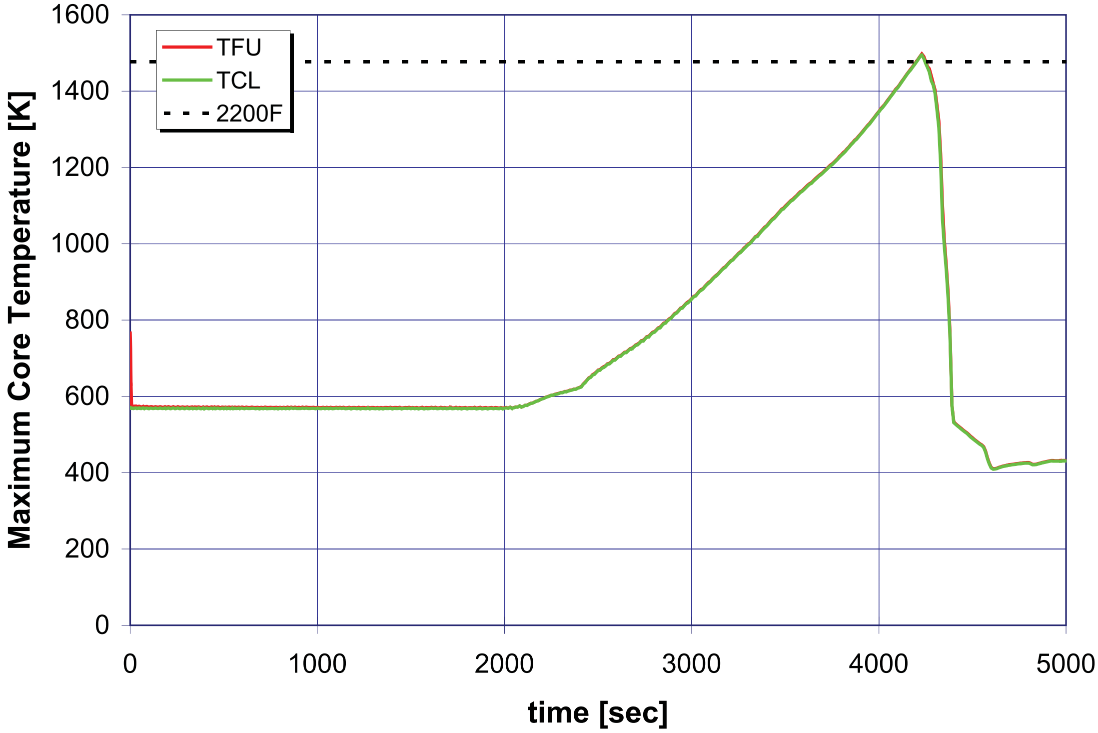
B.2.1.1 Case 1a: Station Blackout with No Injection and Power Recovery at Core Damage

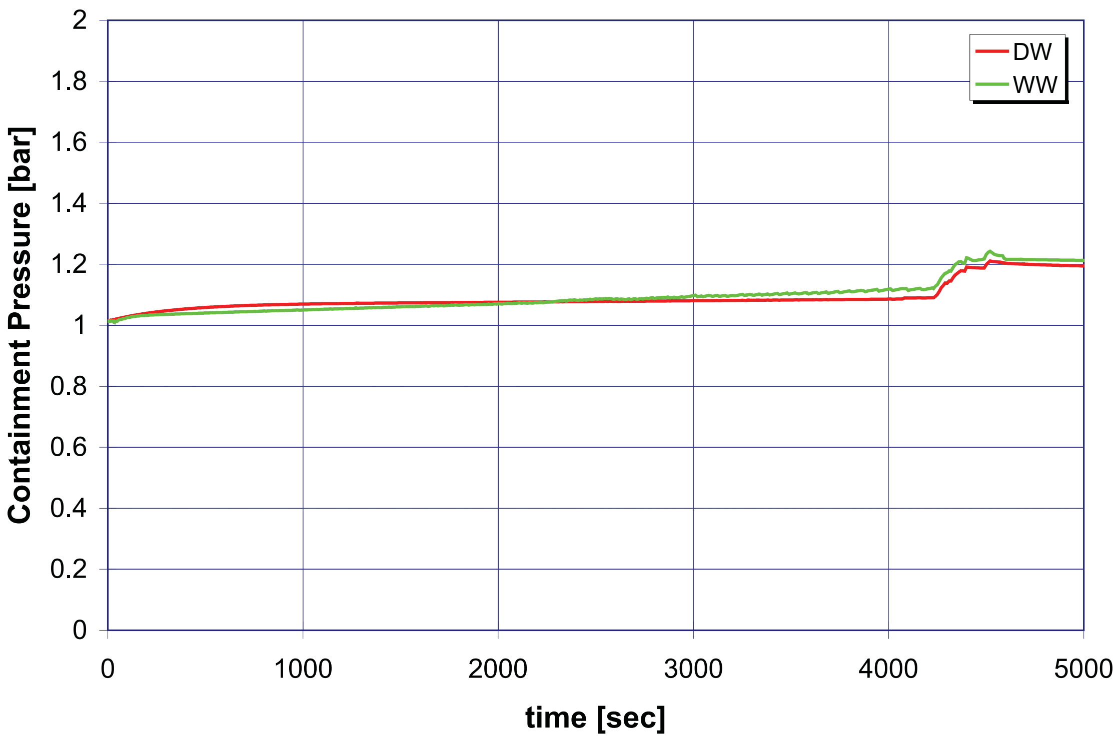


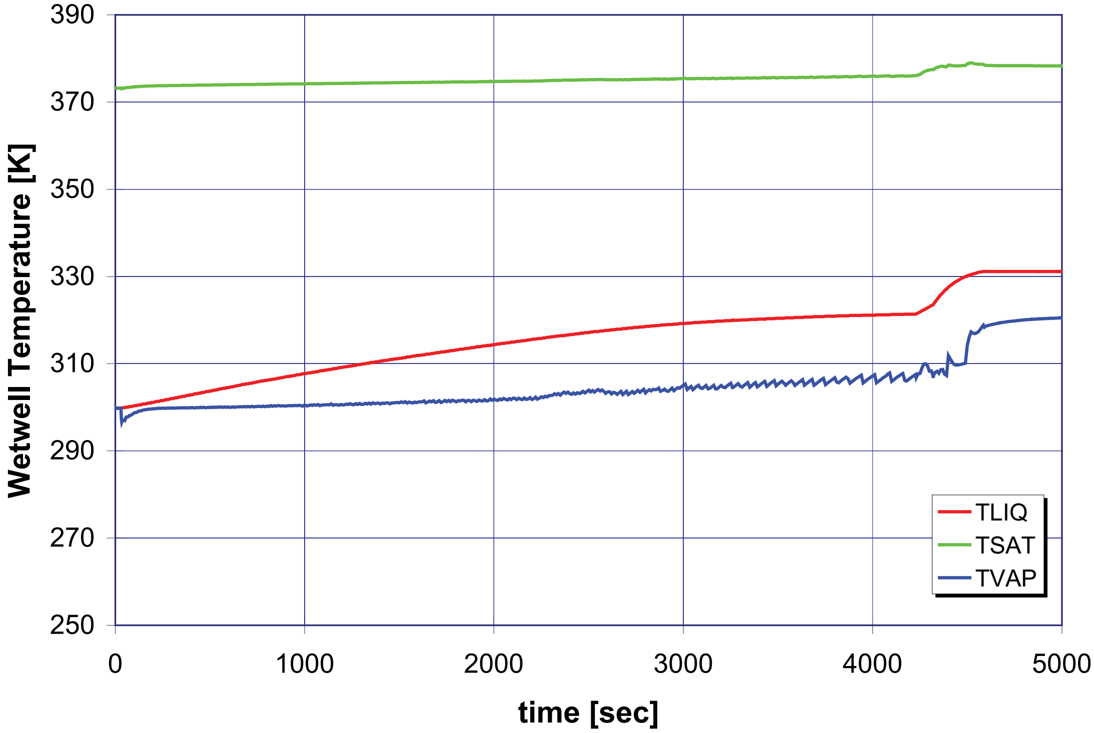


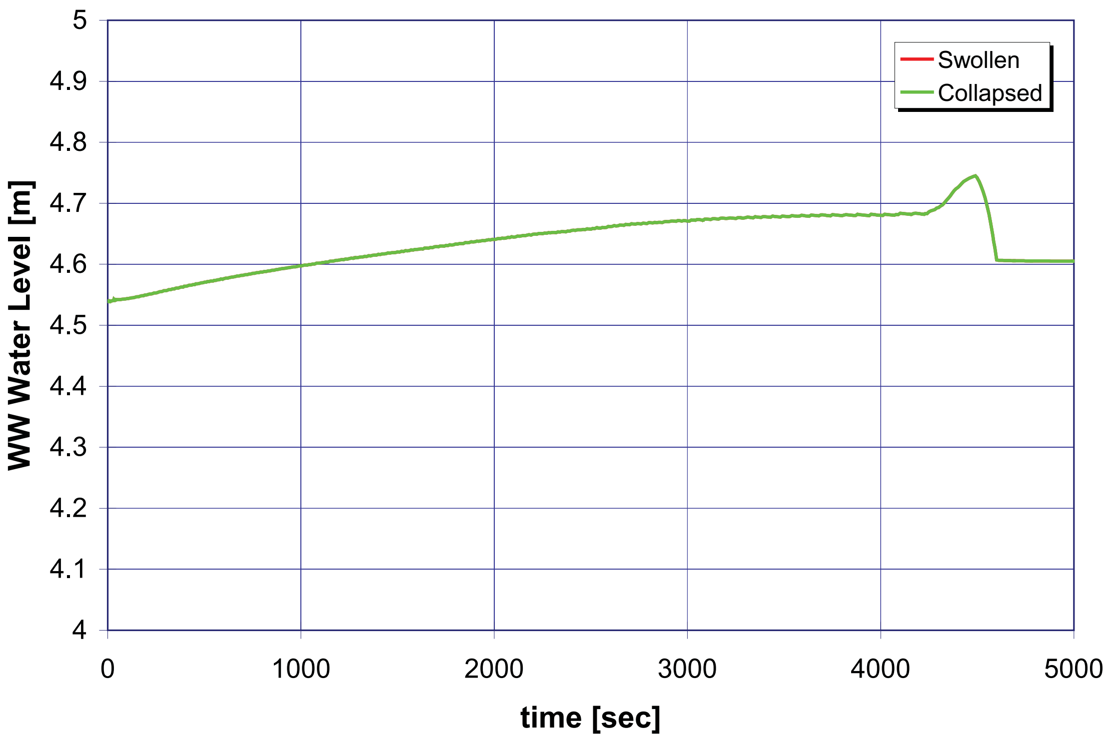

Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD
0
50
100
150
200
250
300
350
0
1000
2000
3000
4000
5000


Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
200
400
600
800
1000
1200
0
1000
2000
3000
4000
5000

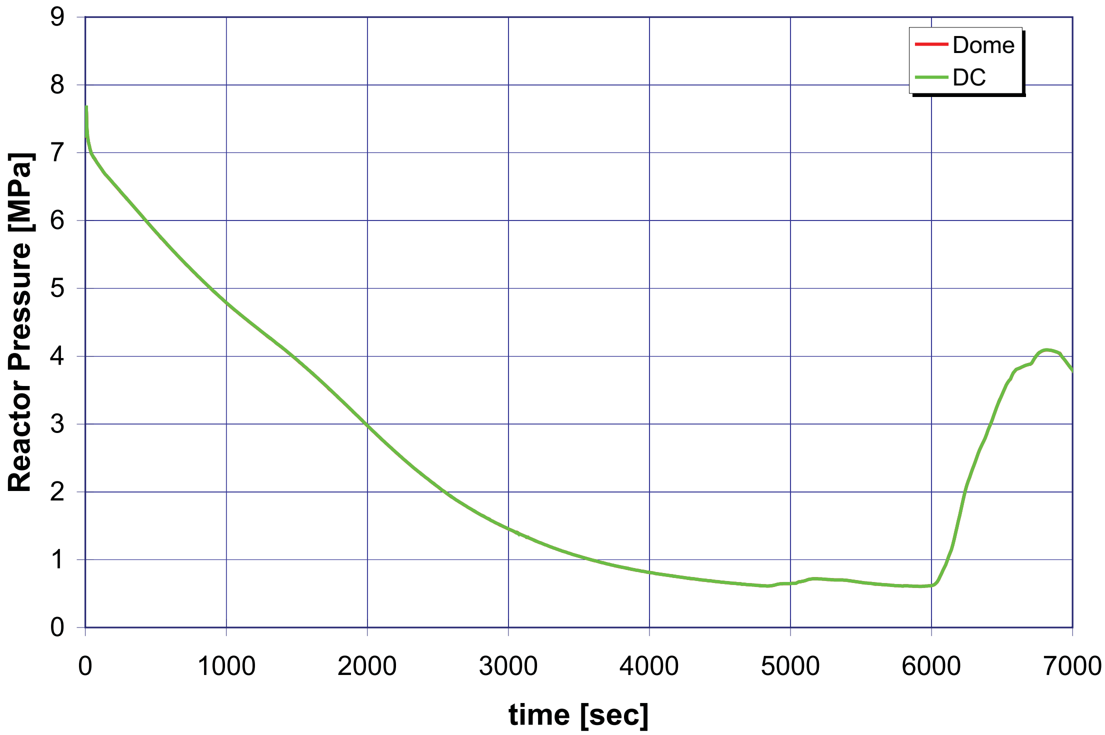

Maximum Core Temperature [K]
1600
1400
1200
1000
800
600
400
200
0
TFU
TCL
2200F
0
1000
2000
3000
4000
5000
time [sec]

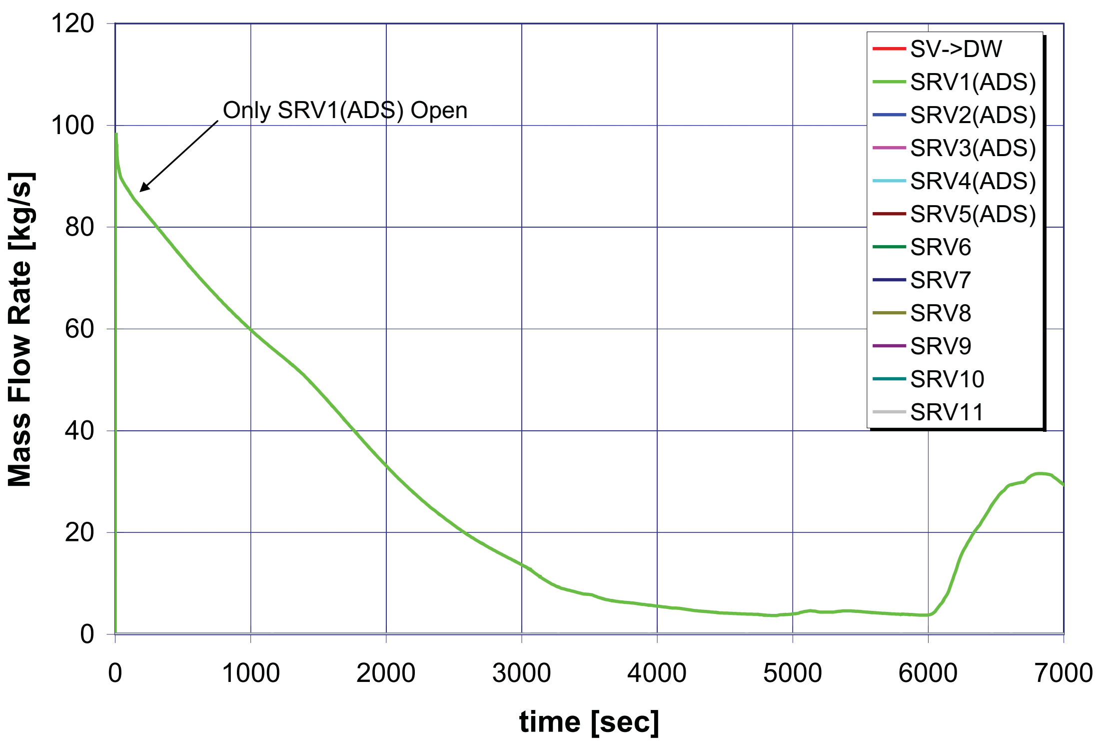

Containment Pressure [bar]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
DW
WW
0
1000
2000
3000
4000
5000
time [sec]

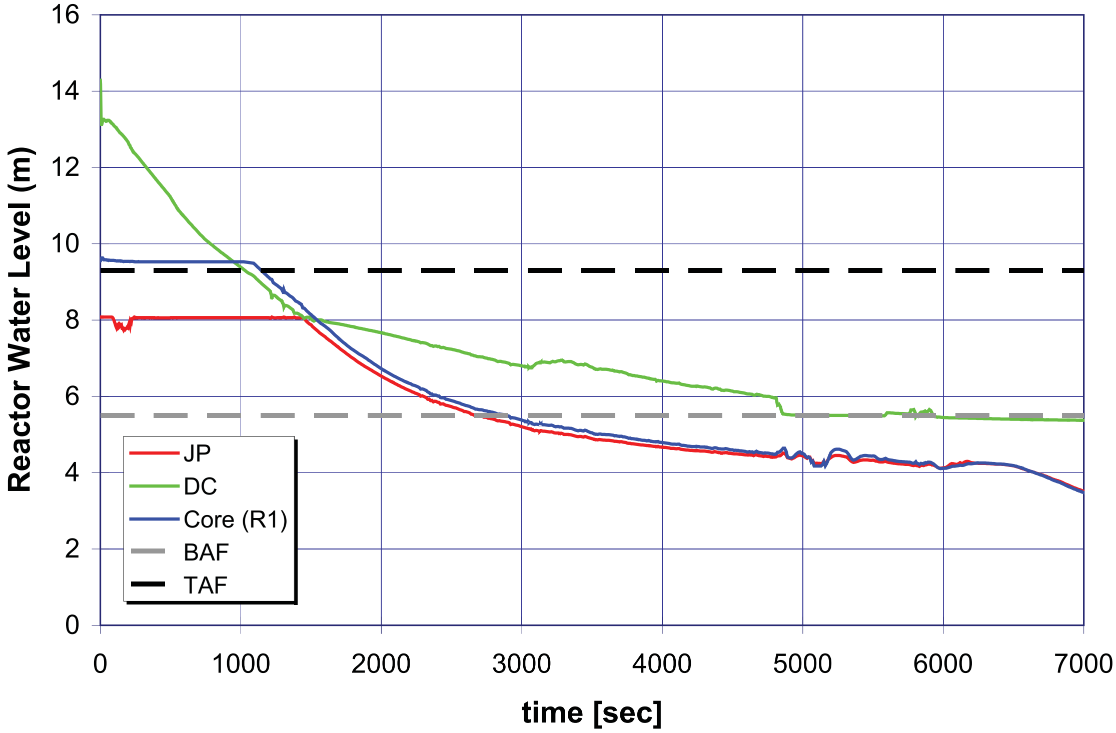

Wetwell Temperature [K]
390
370
350
330
310
290
270
250
0
1000
2000
3000
4000
5000
time [sec]
TLIQ
TSAT
TVAP


WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
1000
2000
3000
4000
5000
time [sec]
Swollen
Collapsed

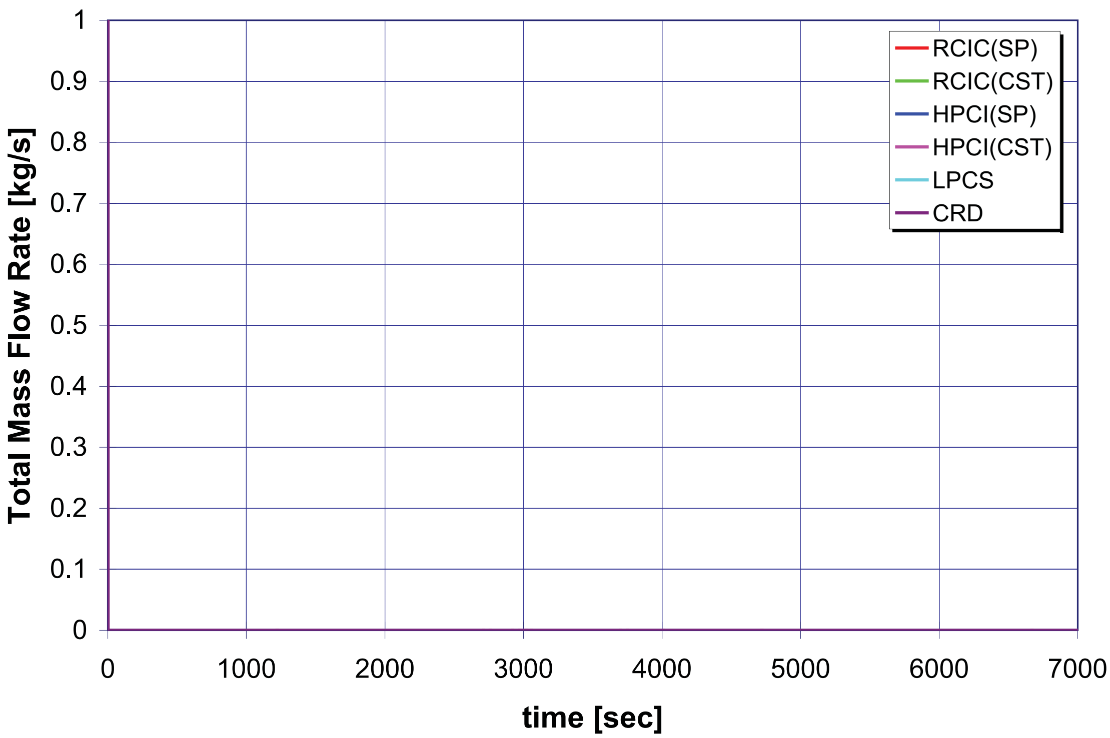
B.2.2 Case 2: Station Blackout and Safety Relief Valve Open at t = 0



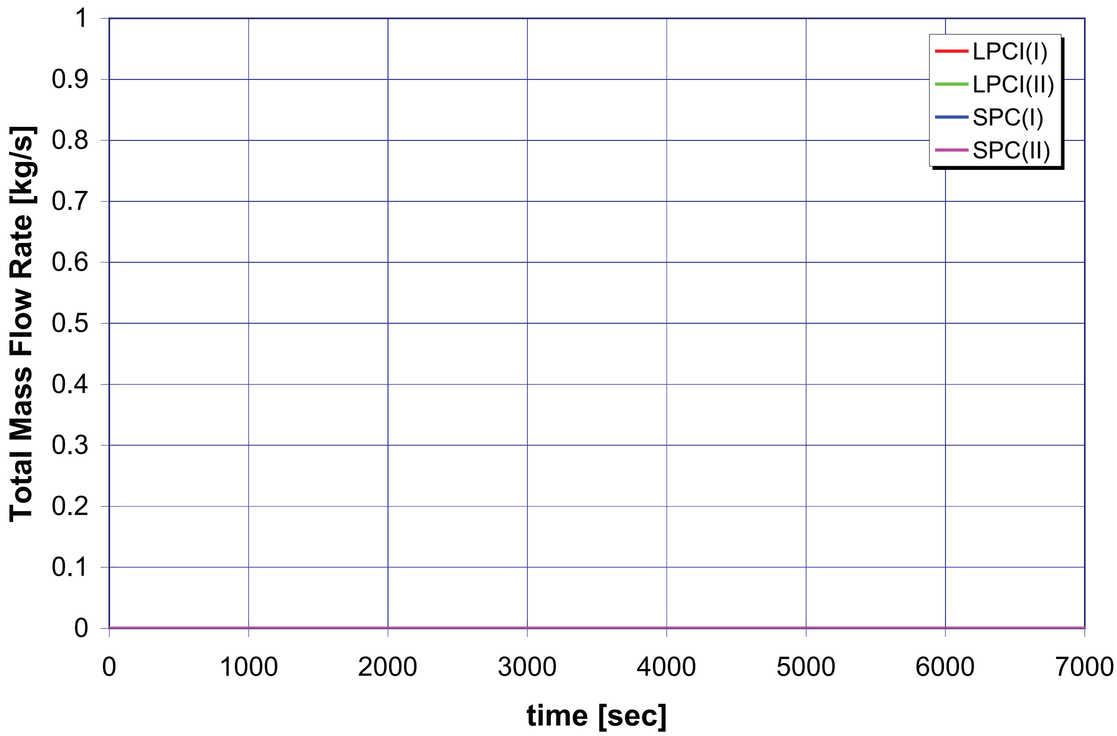




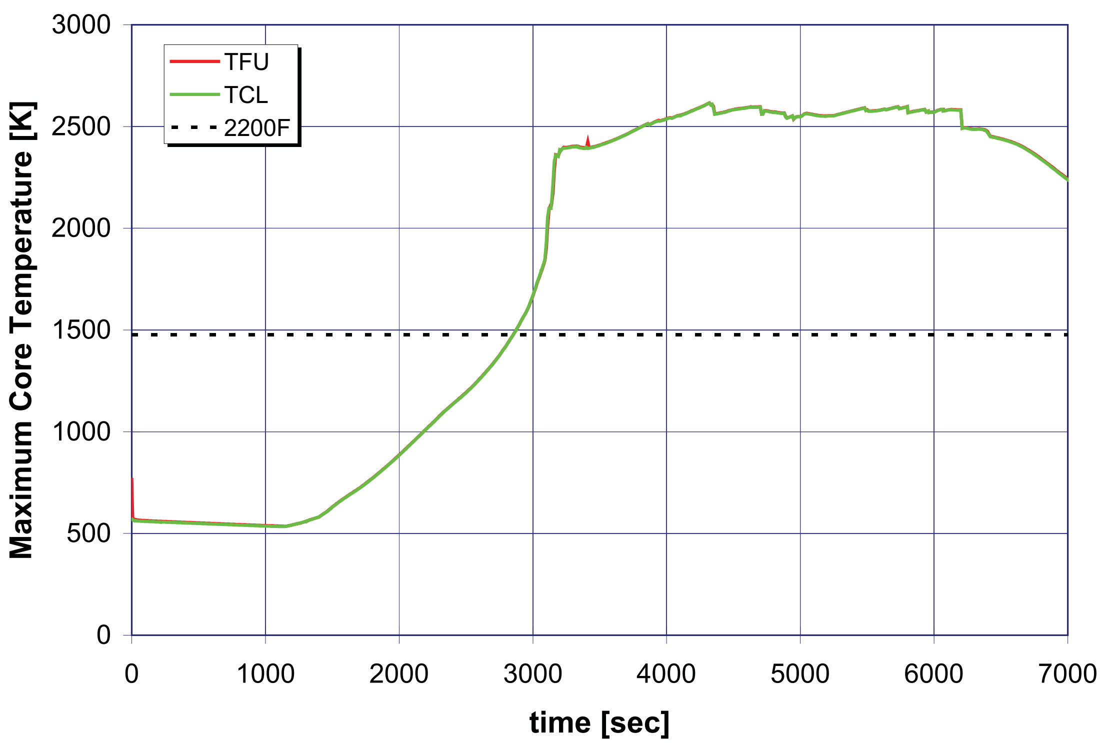
Reactor Water Level (m)
16
14
12
10
8
6
4
2
0
0
1000
2000
3000
4000
5000
6000
7000
time [sec]
JP
DC
Core (R1)
BAF
TAF



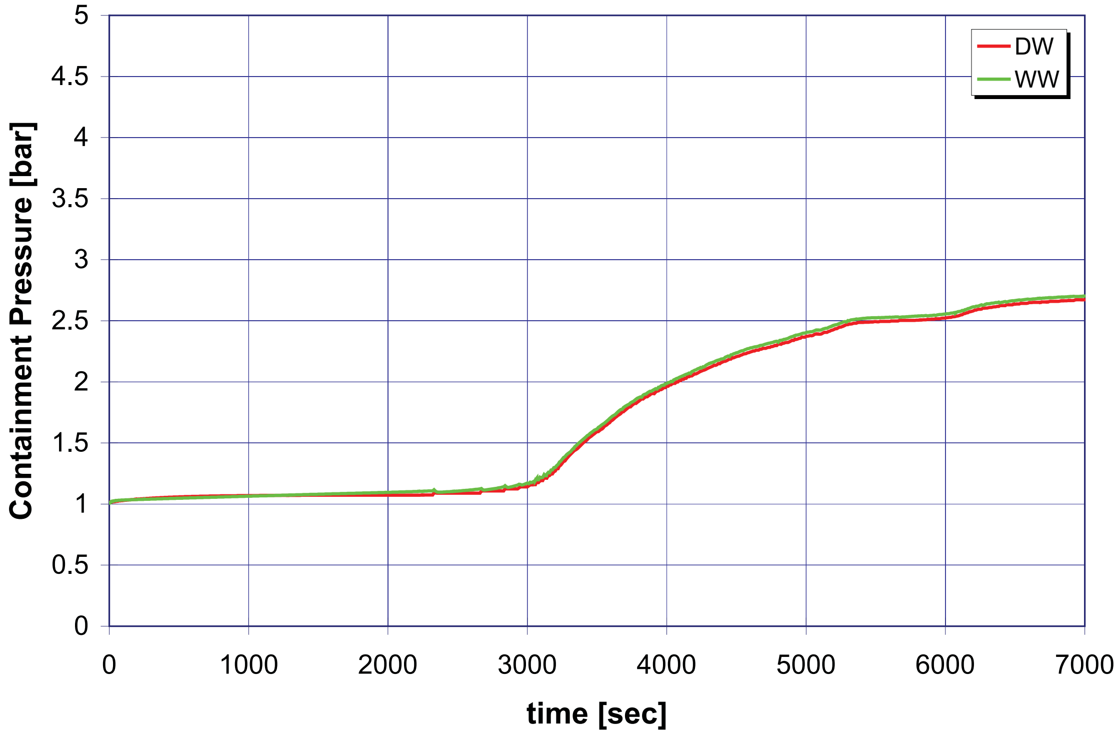
Total Mass Flow Rate [kg/s]
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
1000
2000
3000
4000
5000
6000
7000
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD



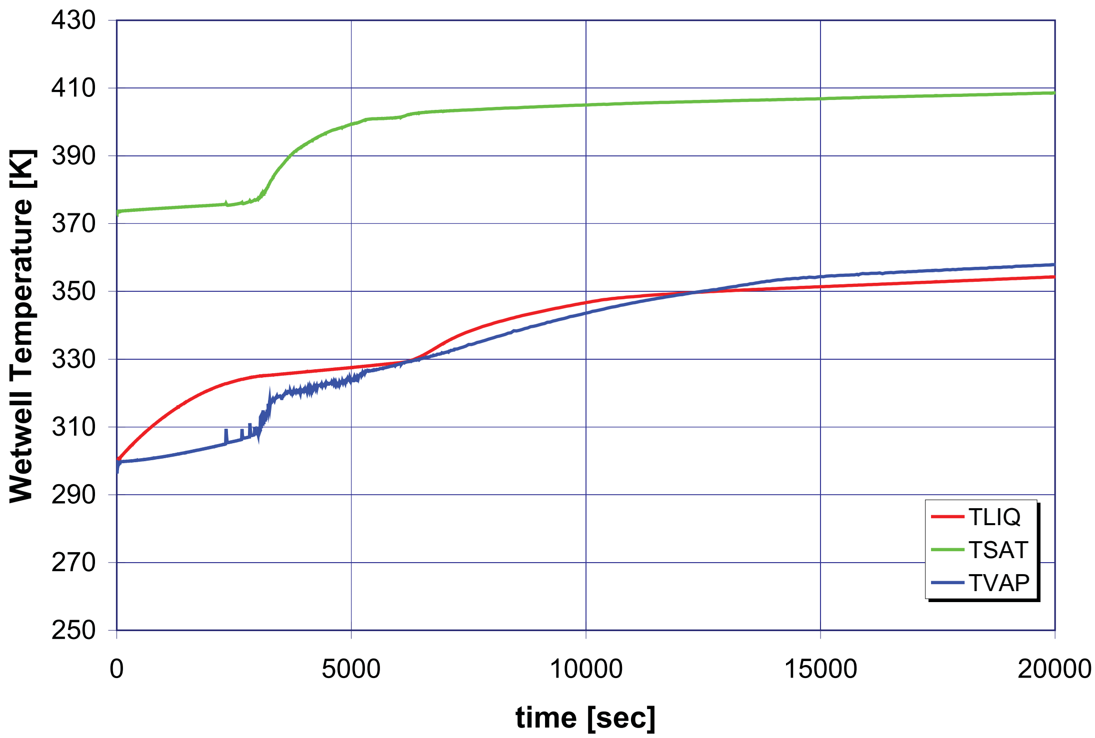
Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)



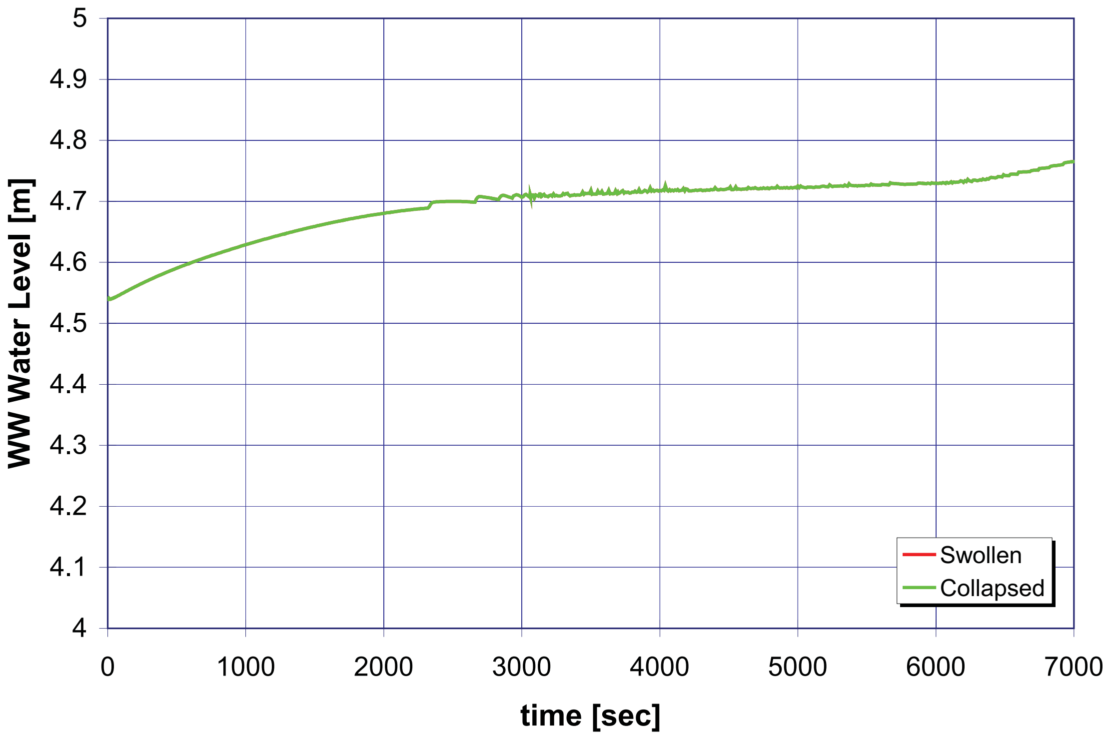
Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F



DW
WW
Containment Pressure [bar]
time [sec]
0
0.5
1
1.5
2
2.5
3
3.5
4
4.5
5
0
1000
2000
3000
4000
5000
6000
7000



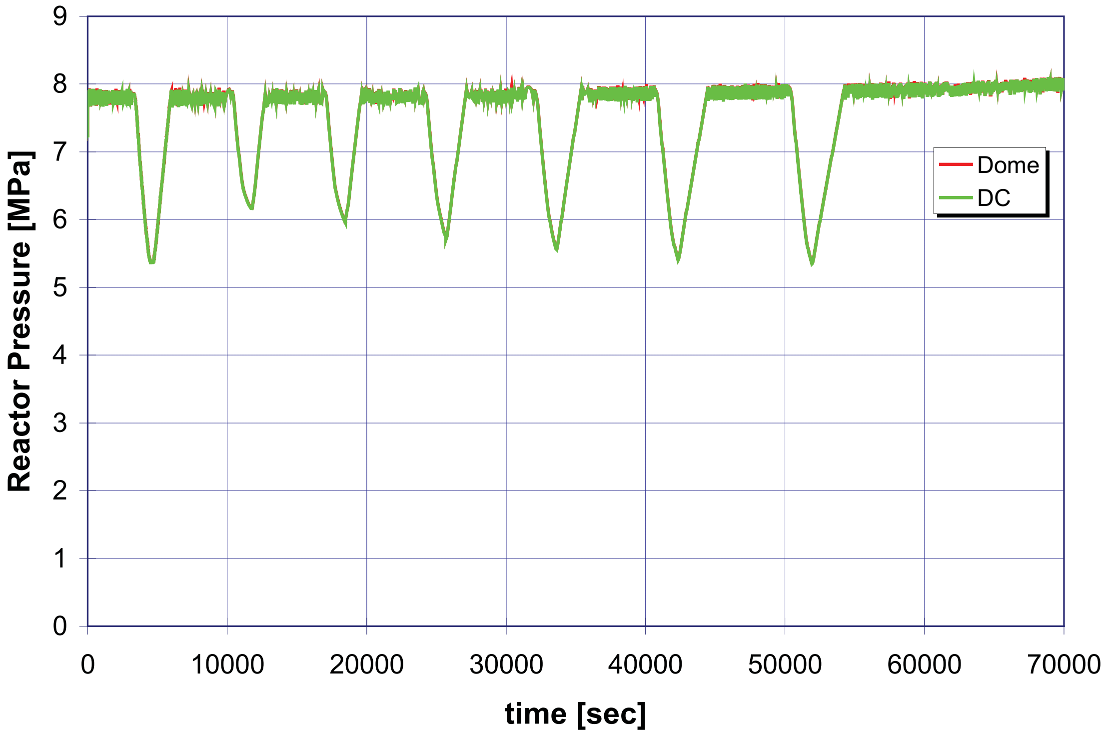
TLIQ
TSAT
TVAP
Wetwell Temperature [K]
time [sec]
250
270
290
310
330
350
370
390
410
430
0
5000
10000
15000
20000

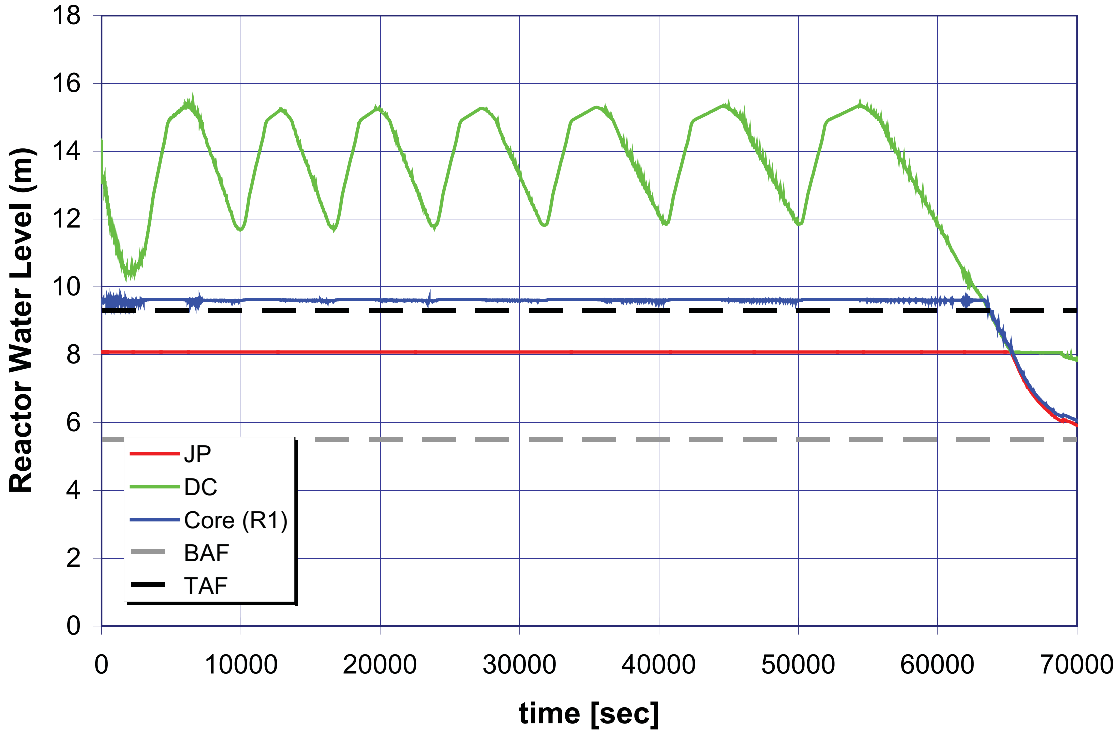


WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
1000
2000
3000
4000
5000
6000
7000
time [sec]
Swollen
Collapsed

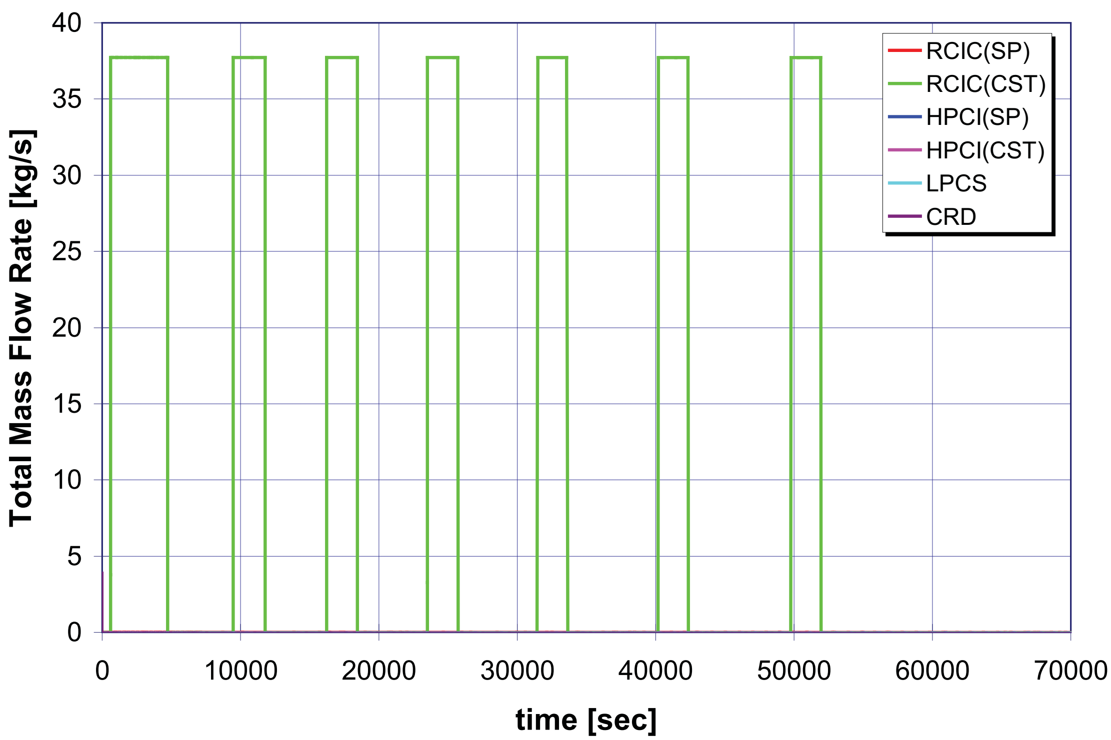
B.2.3 Case 3: Station Blackout and Reactor Core Isolation Cooling

Note: By the time reactor core isolation cooling (RCIC) injection stops from condensate storage tank (CST) depletion at 14.4 hours, the RCIC pump net positive suction head (NPSH) limit has already been exceeded at 11.6 hours.

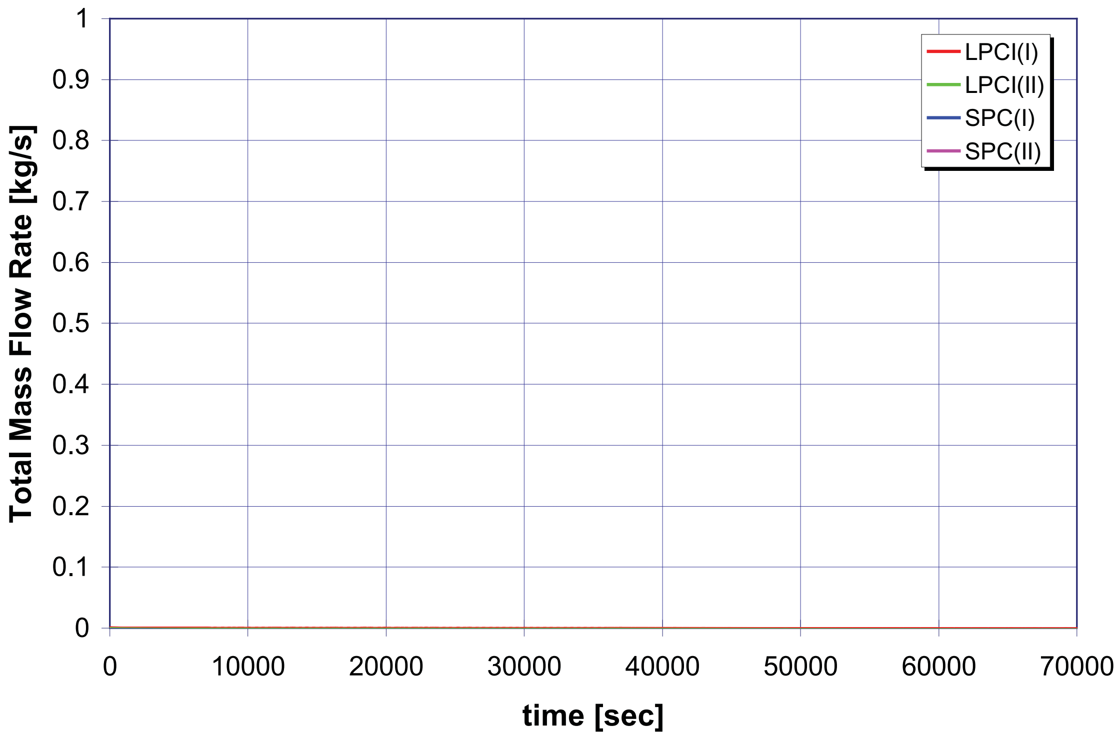




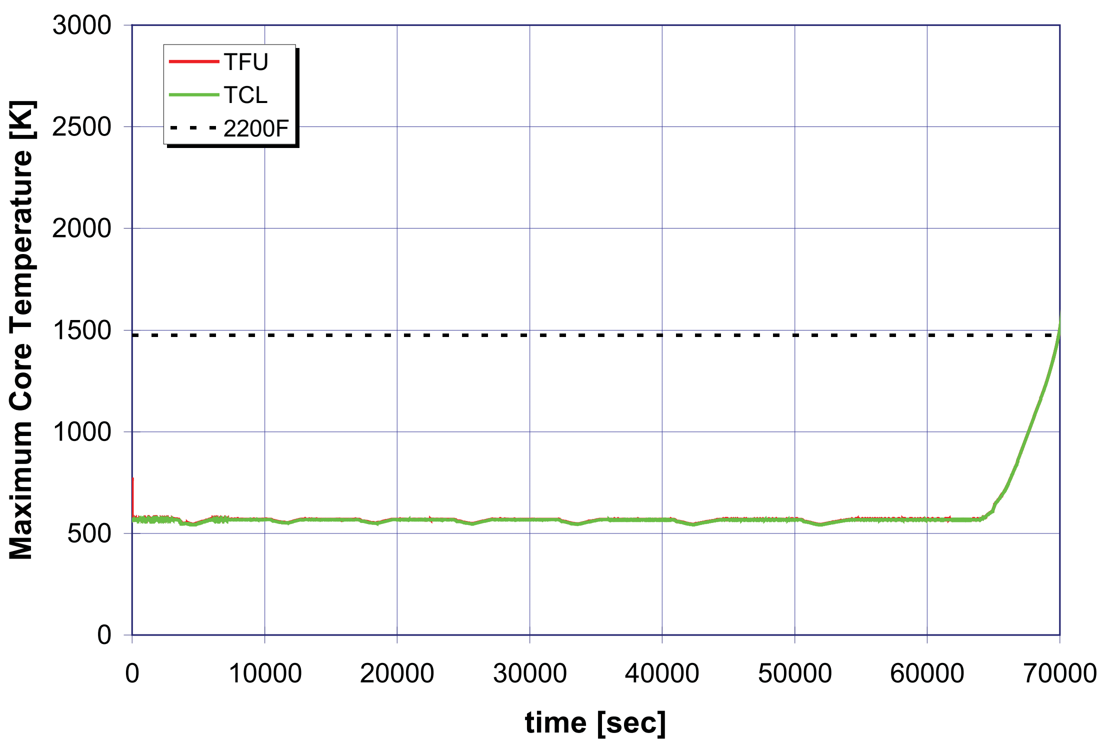
Reactor Water Level (m)
18
16
14
12
10
8
6
4
2
0
0
10000
20000
30000
40000
50000
60000
70000
time [sec]
JP
DC
Core (R1)
BAF
TAF



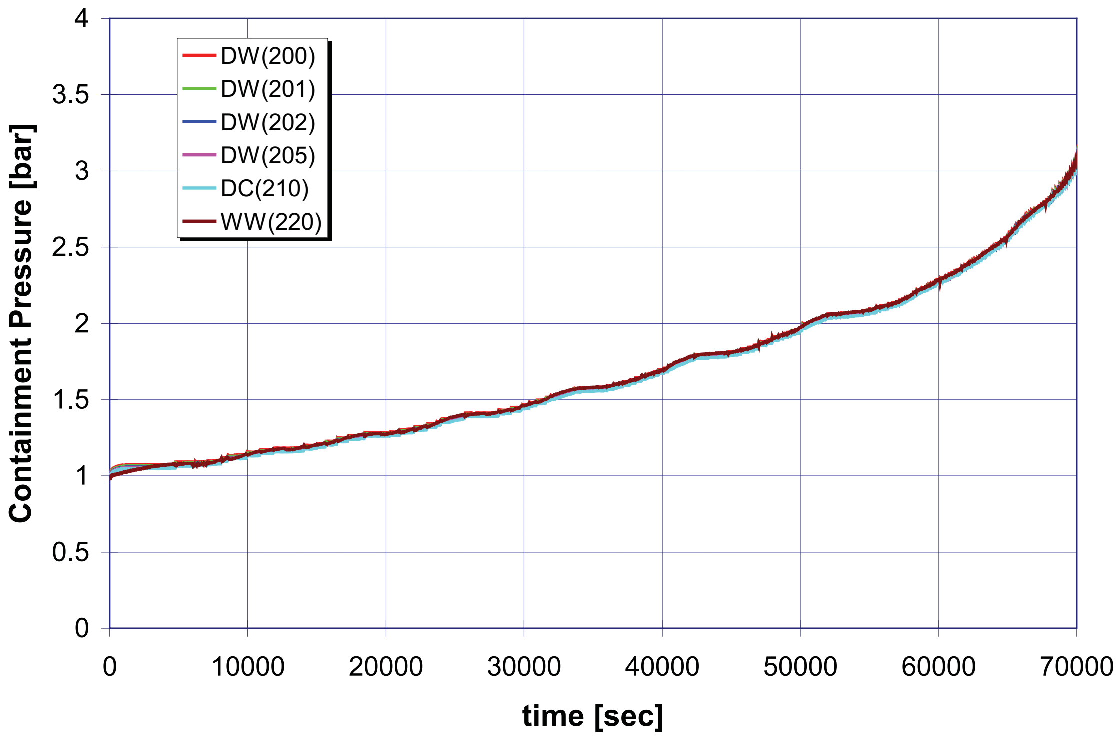
Total Mass Flow Rate [kg/s]
40
35
30
25
20
15
10
5
0
0
10000
20000
30000
40000
50000
60000
70000
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD



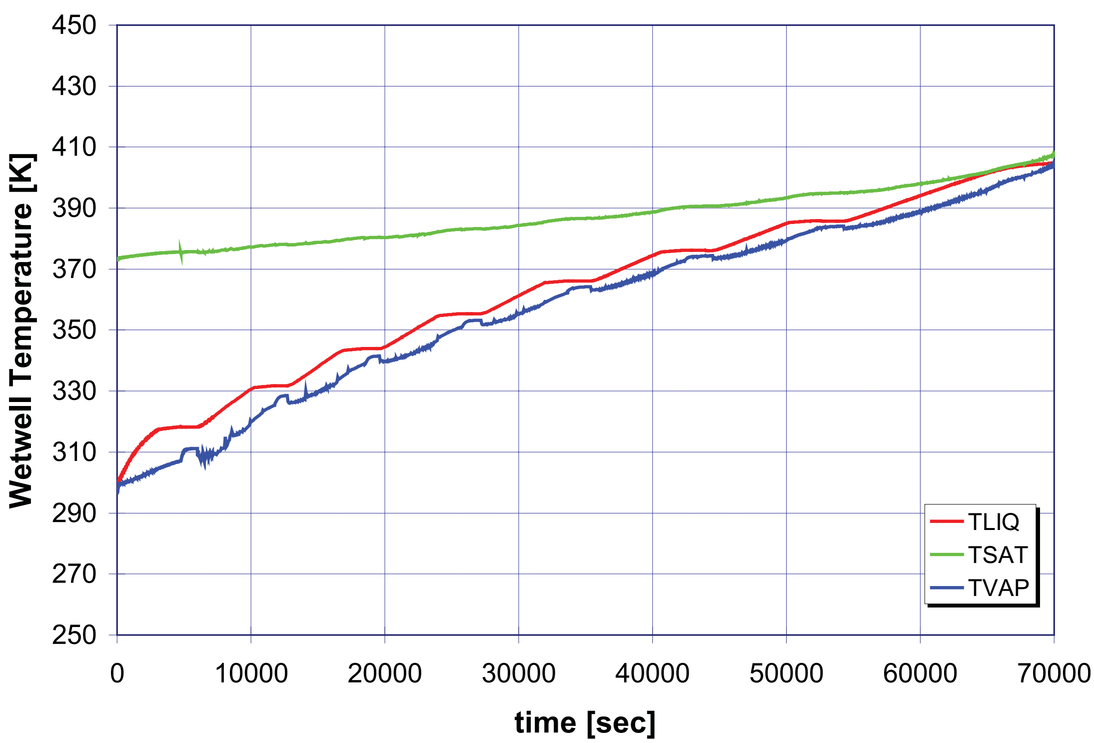
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
Total Mass Flow Rate [kg/s]
time [sec]



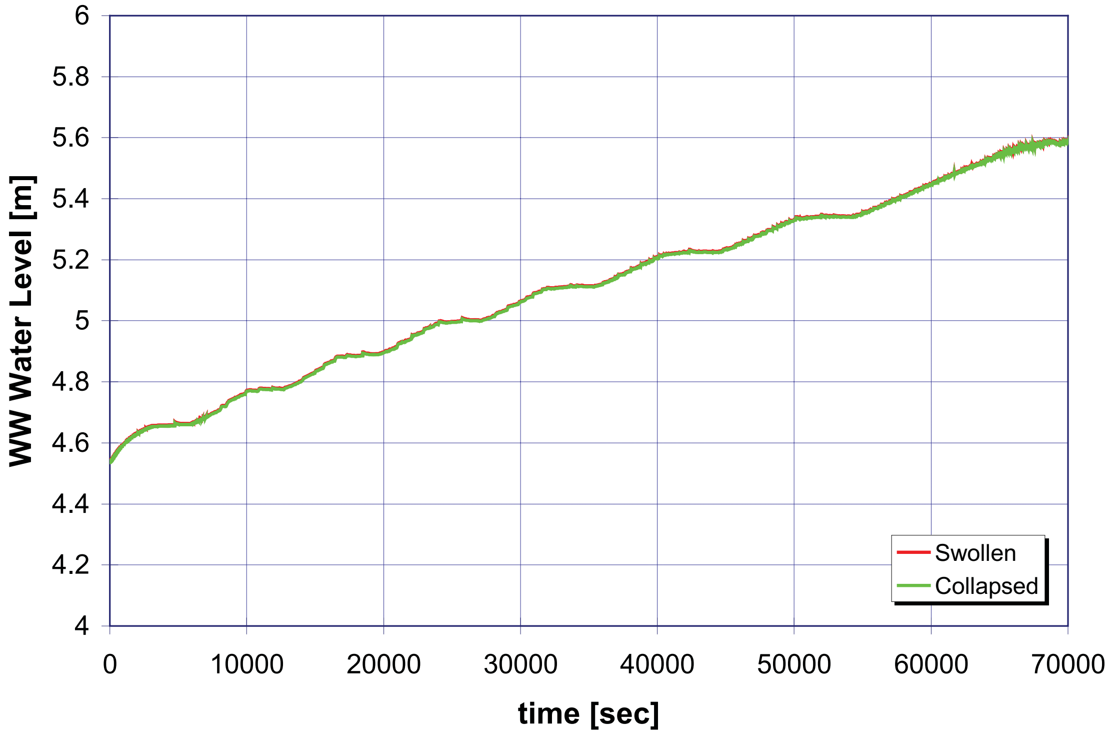
TFU
TCL
2200F
Maximum Core Temperature [K]
time [sec]



DW(200)
DW(201)
DW(202)
DW(205)
DC(210)
WW(220)
Containment Pressure [bar]
time [sec]

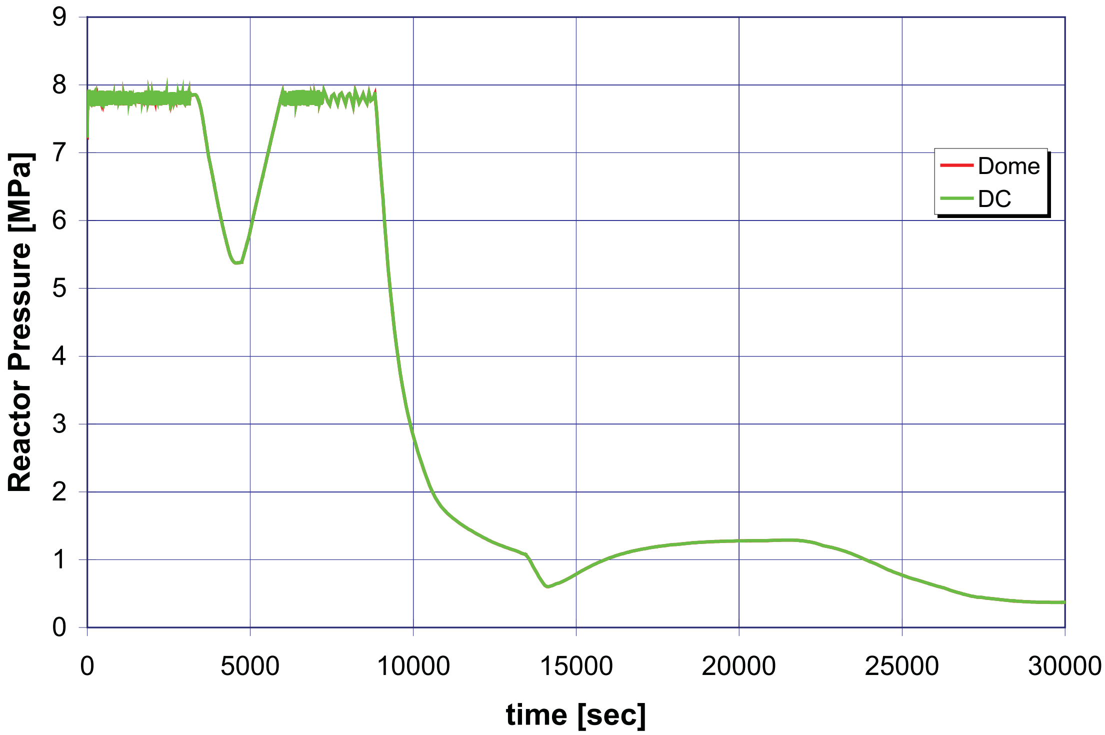


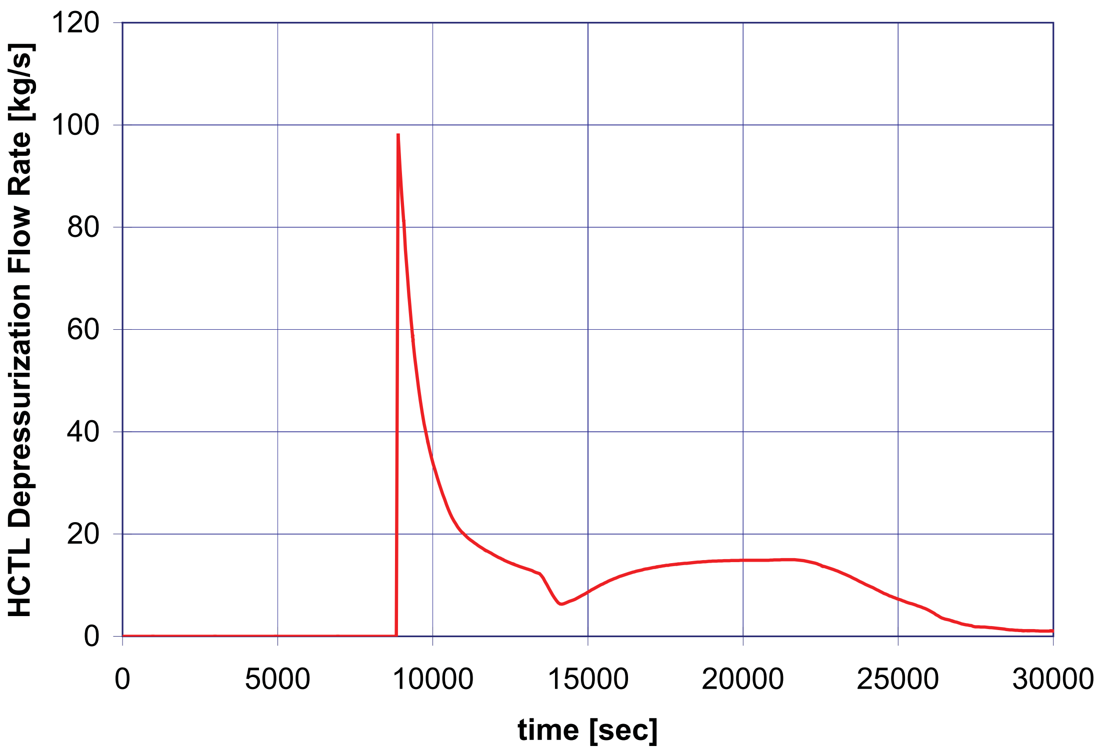
TLIQ
TSAT
TVAP
Wetwell Temperature [K]
time [sec]

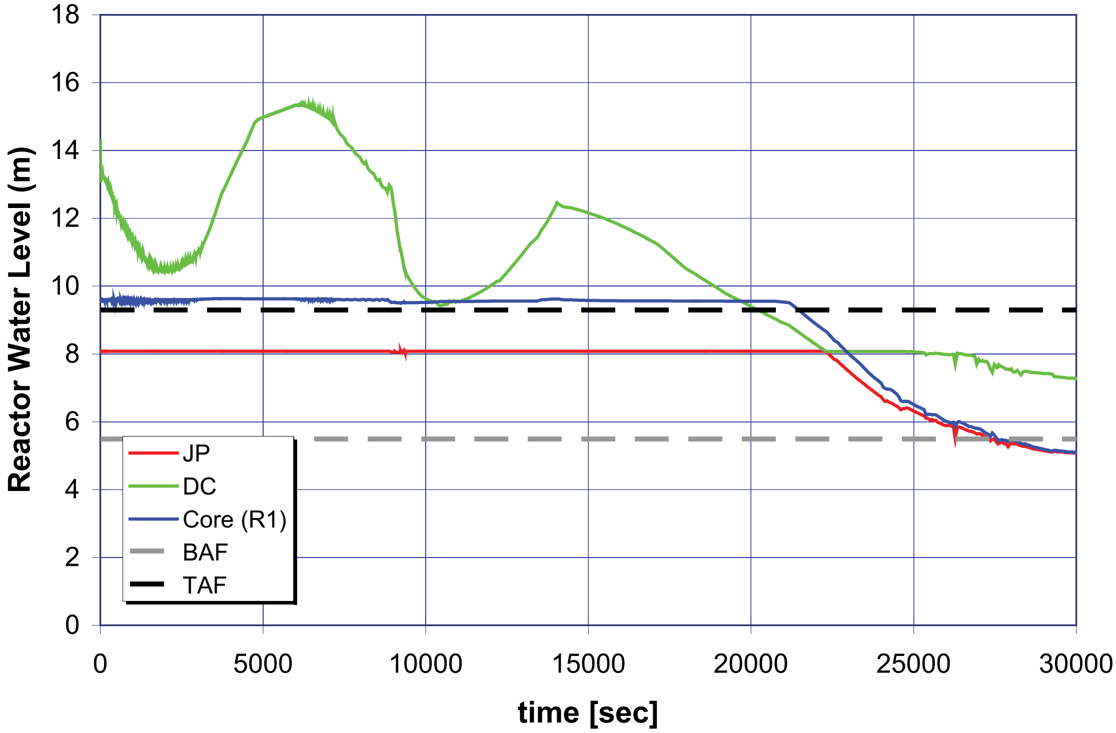


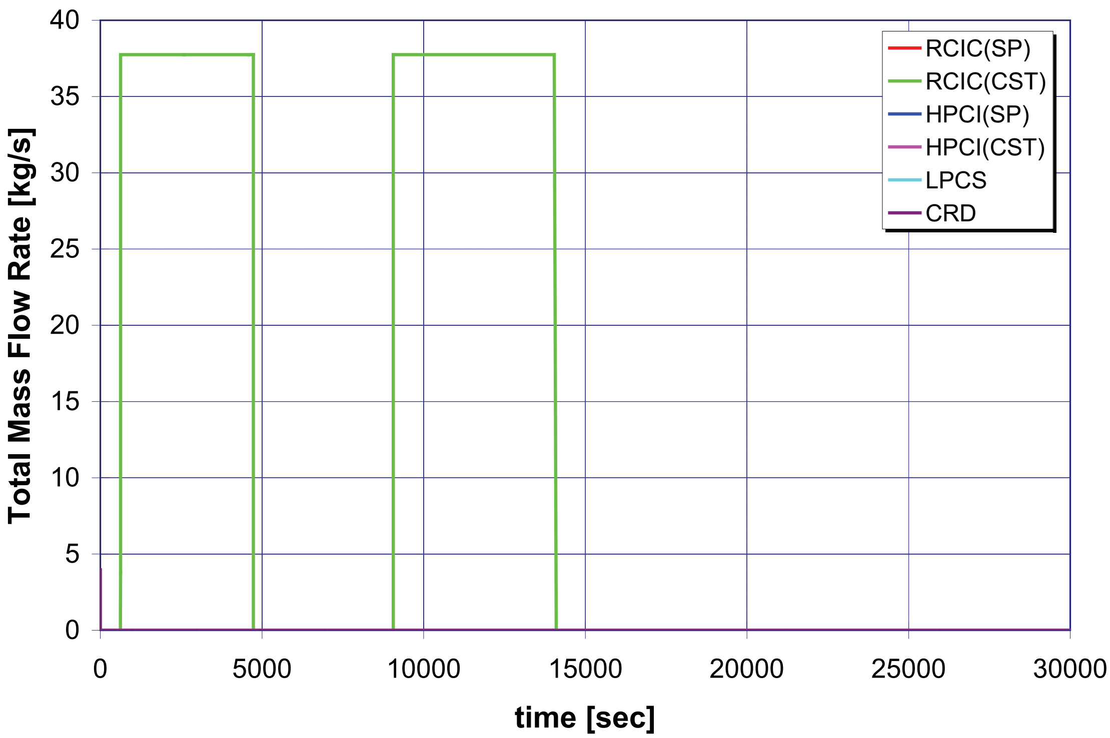
6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
WW Water Level [m]
0
10000
20000
30000
40000
50000
60000
70000
time [sec]
Swollen
Collapsed

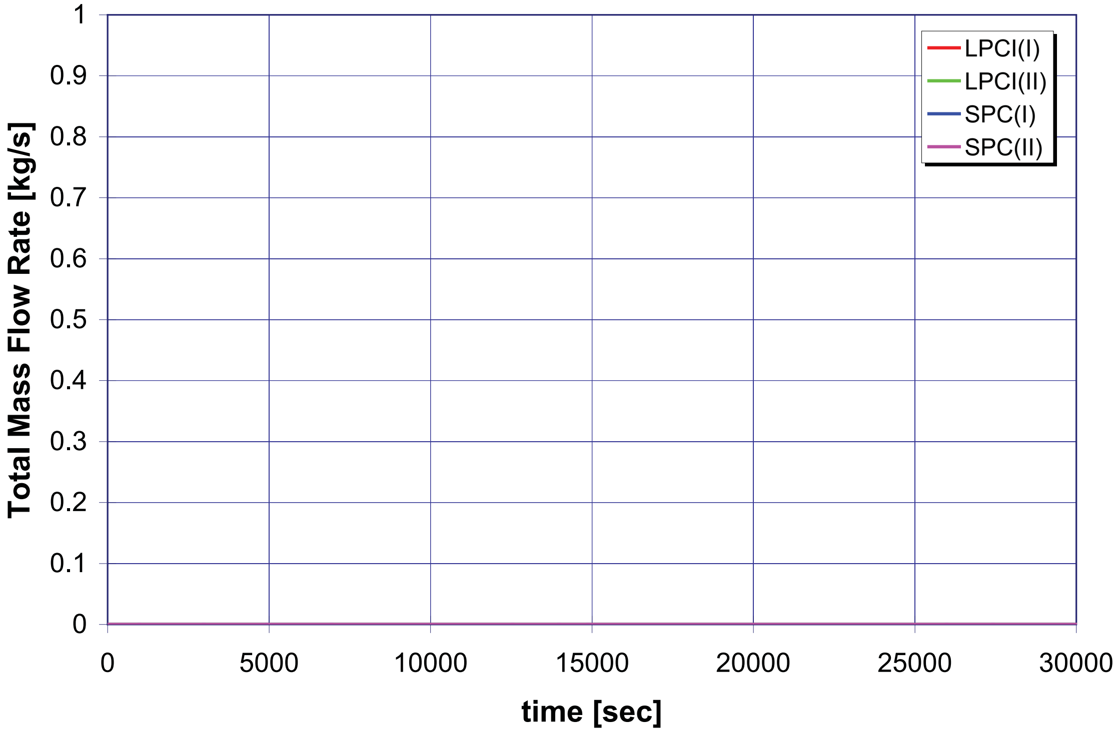
B.2.4 Case 4: Station Blackout and RCIC and HCTL Depressurization



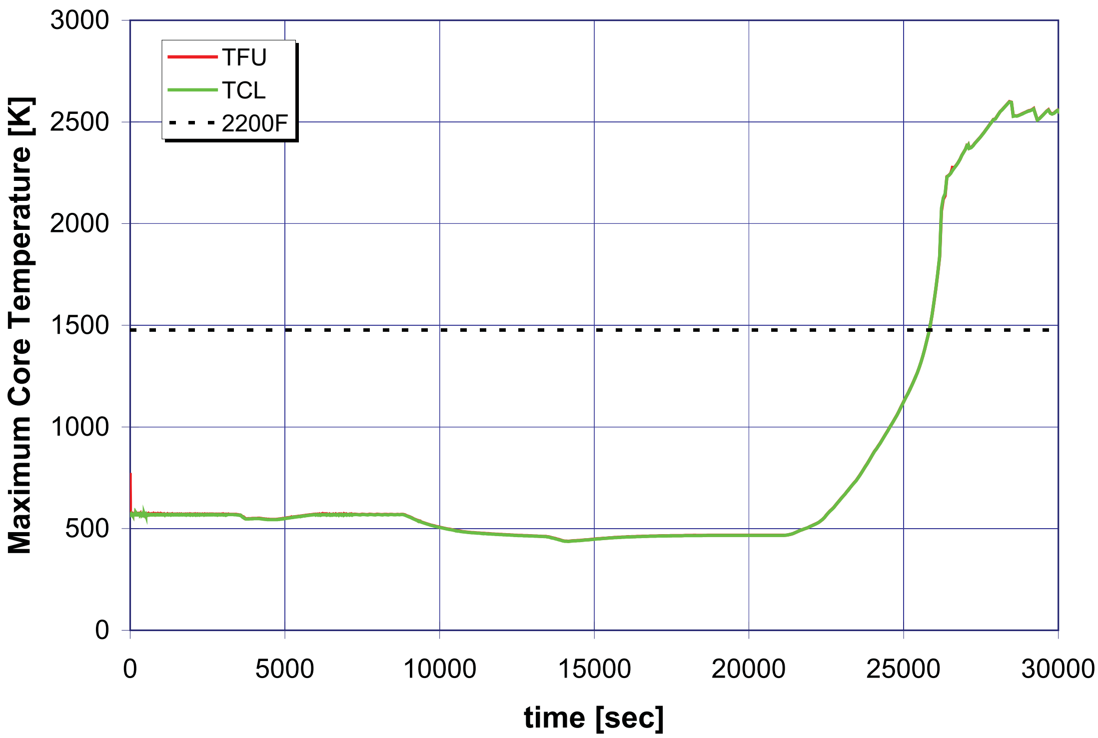



Reactor Water Level (m)
time [sec]
JP
DC
Core (R1)
BAF
TAF
0
2
4
6
8
10
12
14
16
18
0
5000
10000
15000
20000
25000
30000

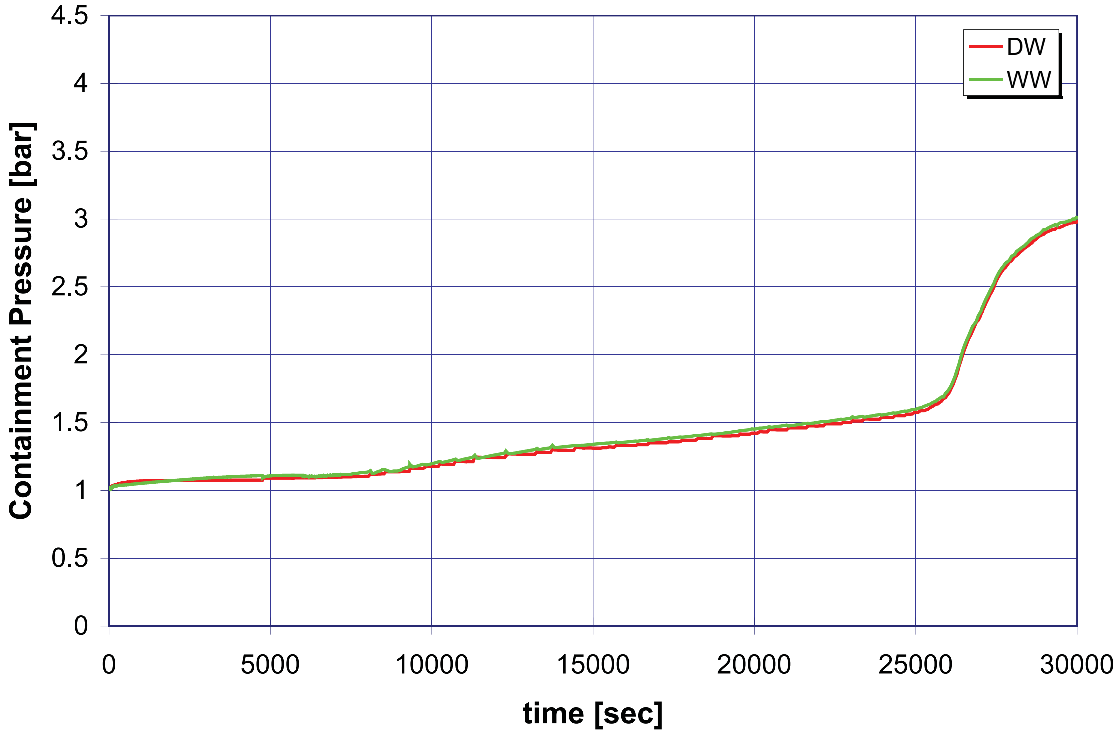

Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD
0
5
10
15
20
25
30
35
40
0
5000
10000
15000
20000
25000
30000



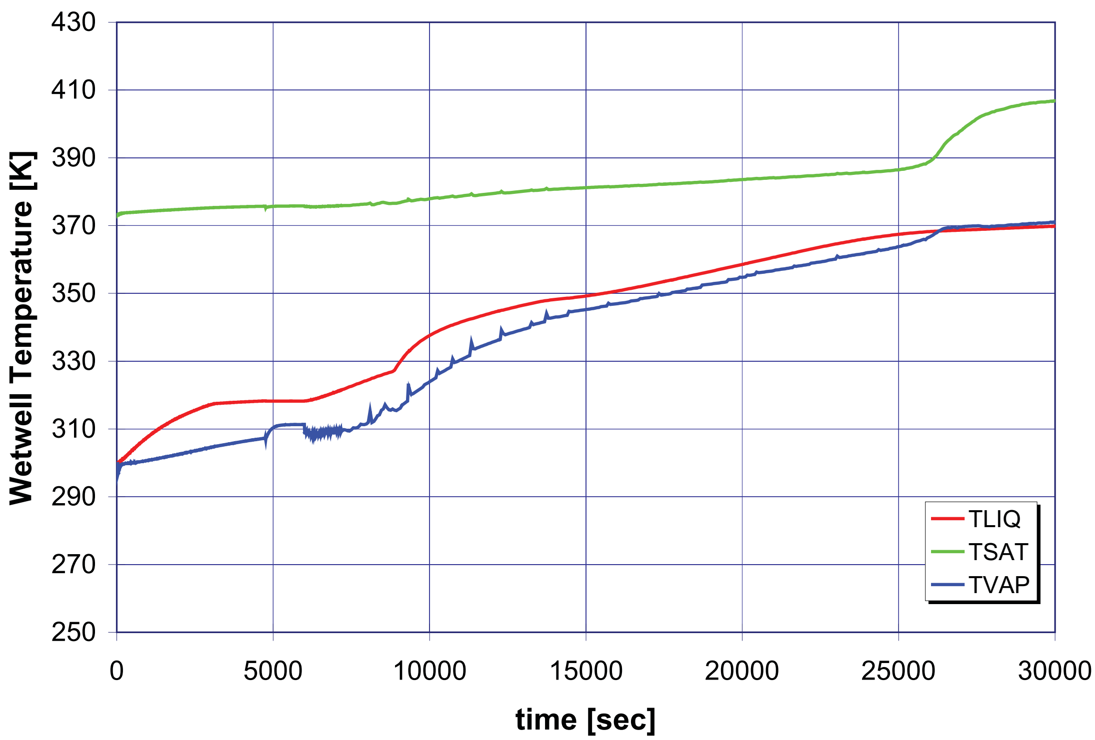
Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
5000
10000
15000
20000
25000
30000



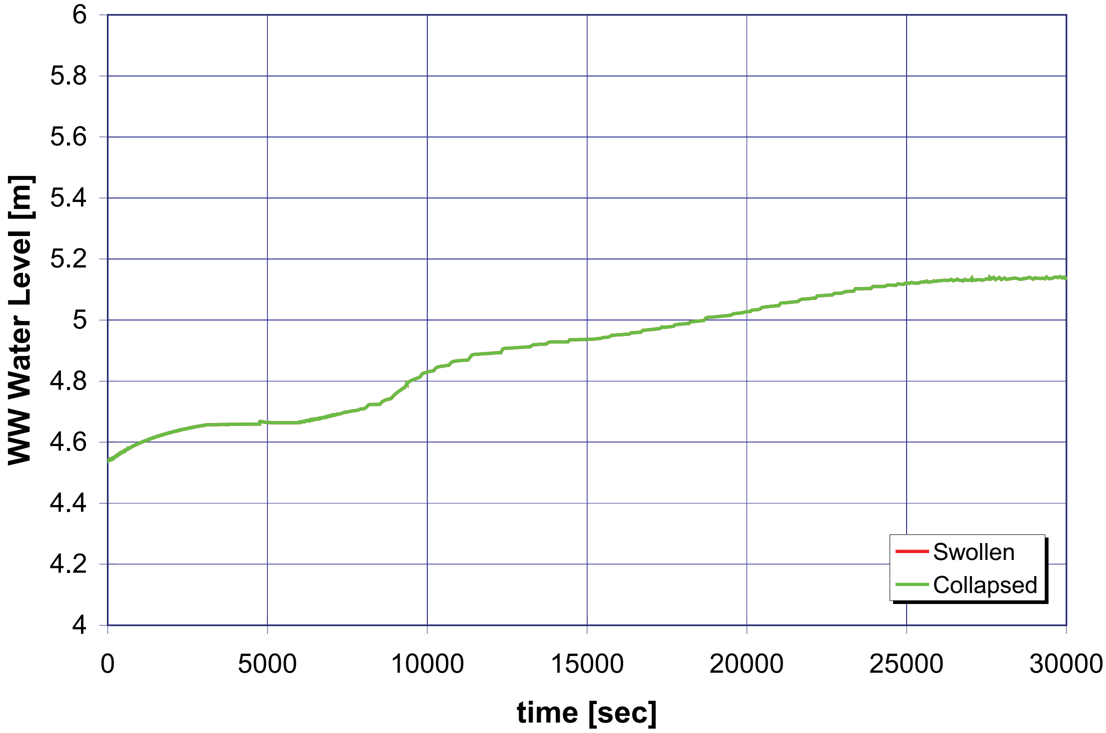
Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F
3000
2500
2000
1500
1000
500
0
0
5000
10000
15000
20000
25000
30000



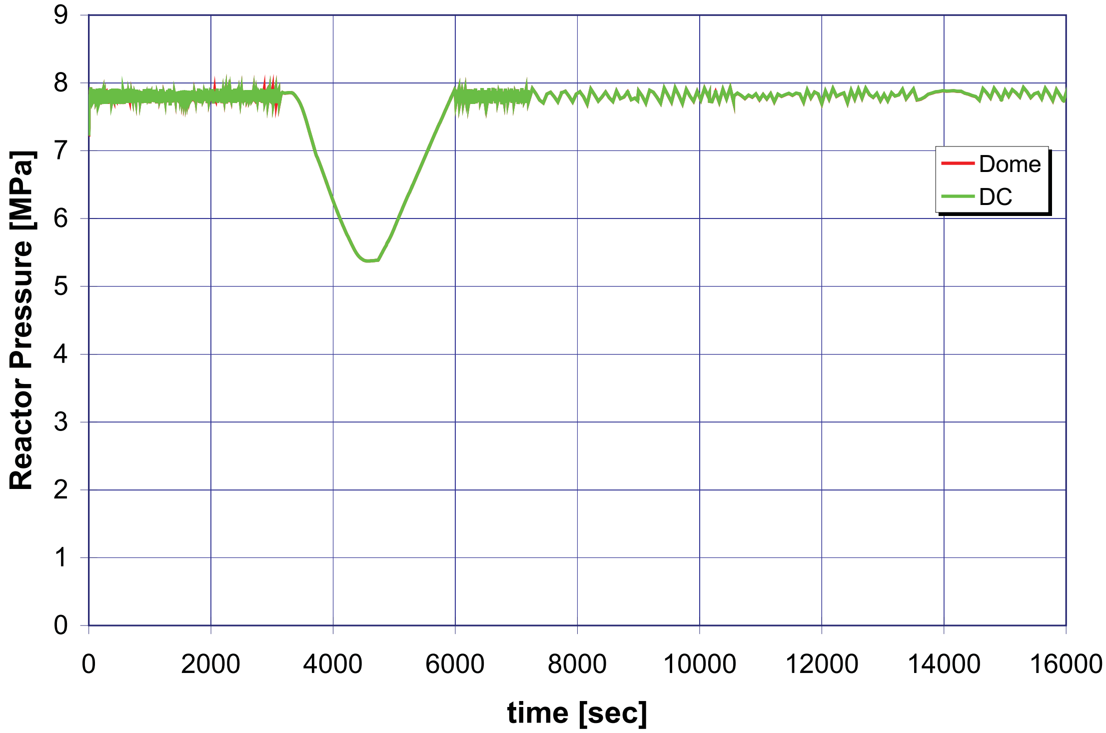
Containment Pressure [bar]
time [sec]
DW
WW

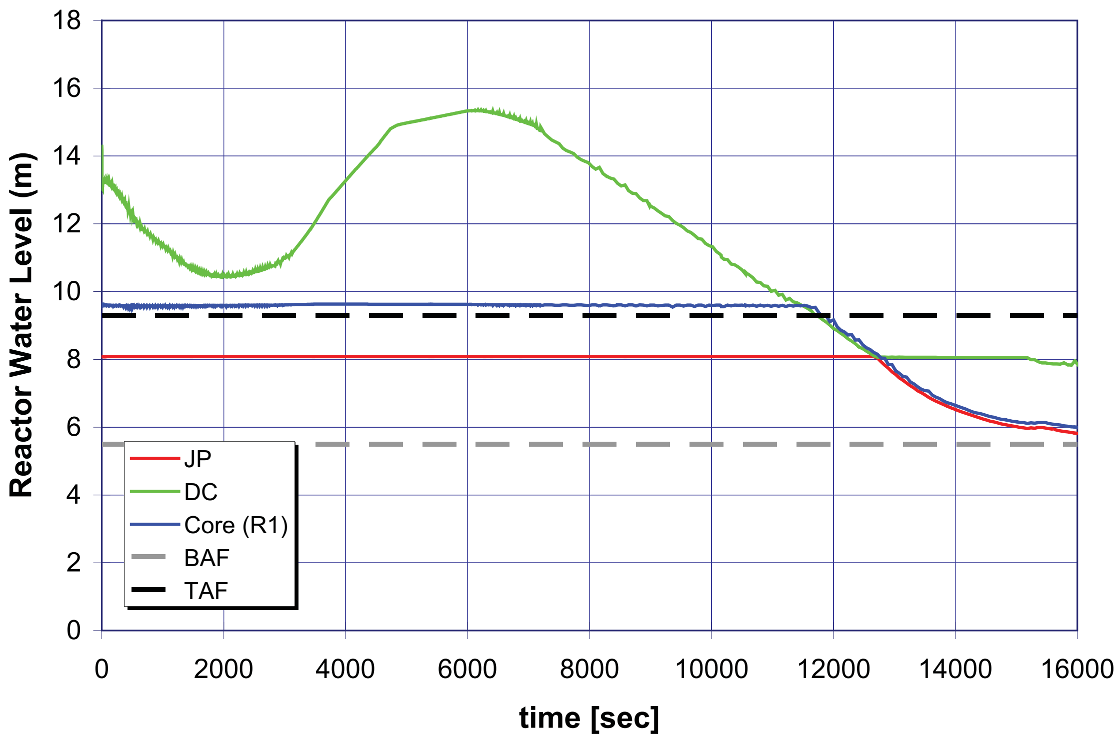


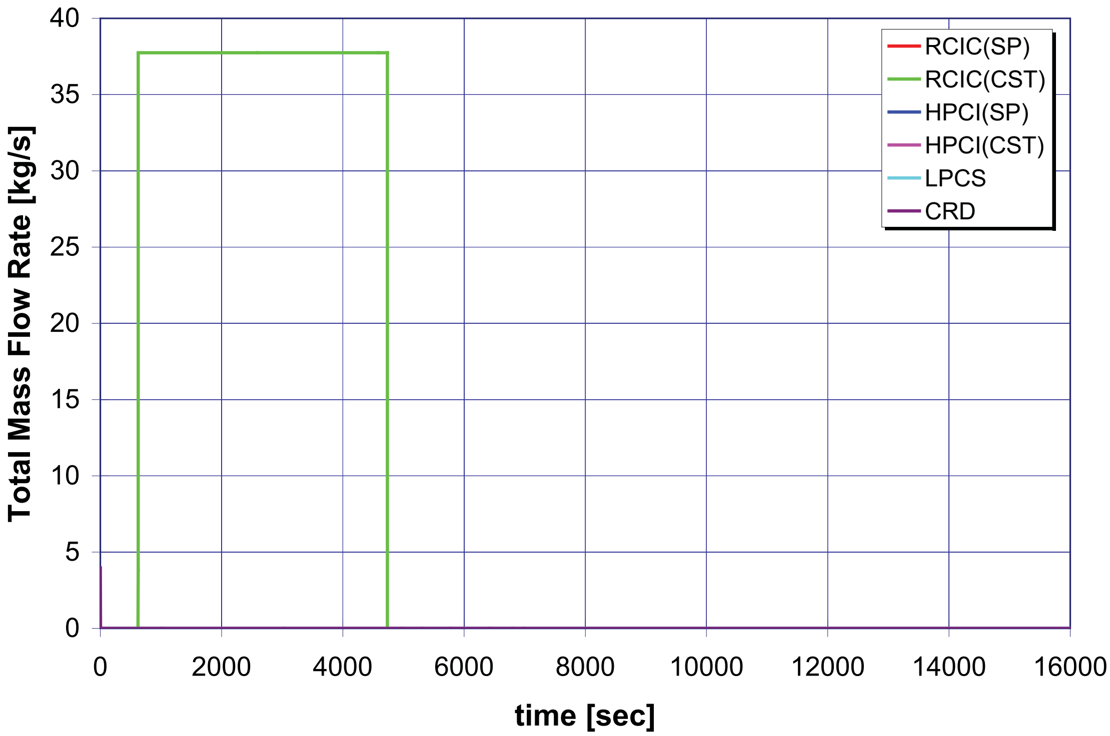
Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP


6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
WW Water Level [m]
0
5000
10000
15000
20000
25000
30000
time [sec]
Swollen
Collapsed

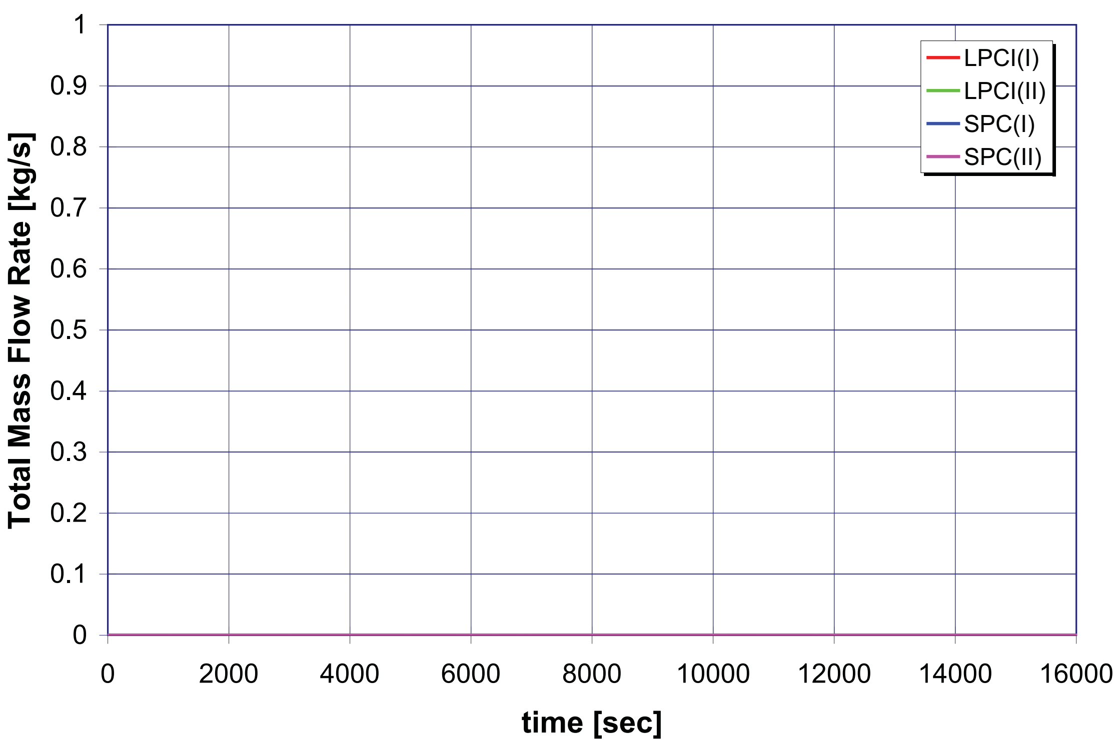
B.2.5 Case 5: Station Blackout and RCIC and 2-Hour Direct Current



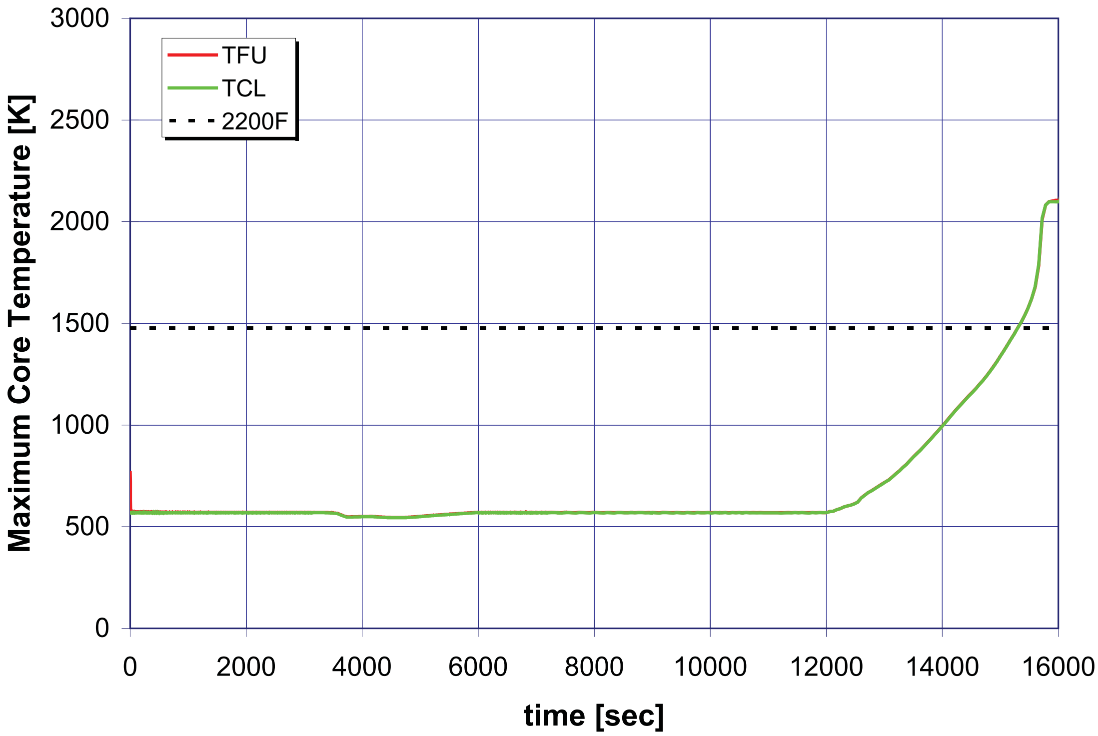




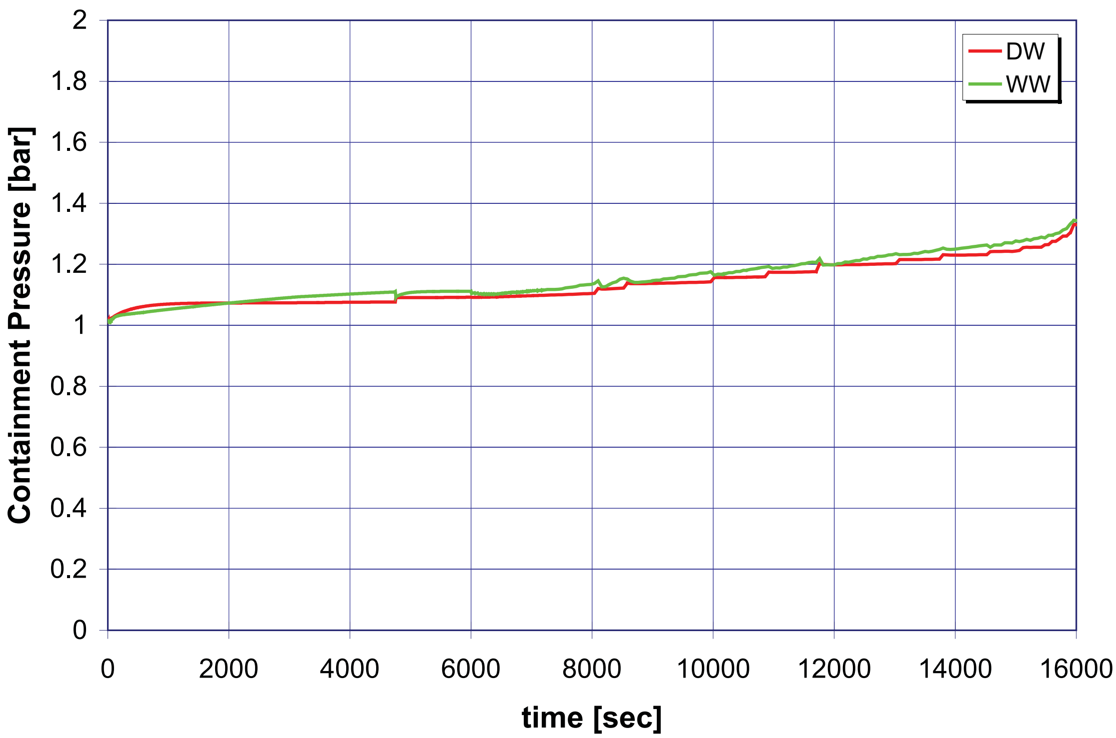
Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD



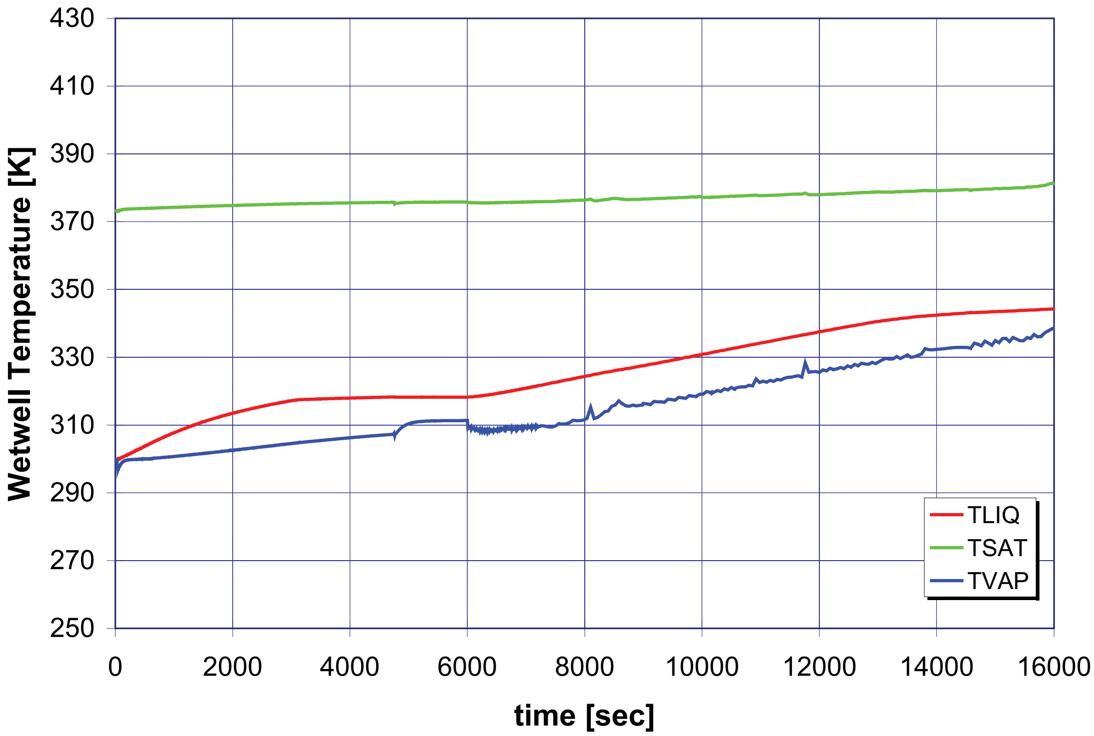
Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)

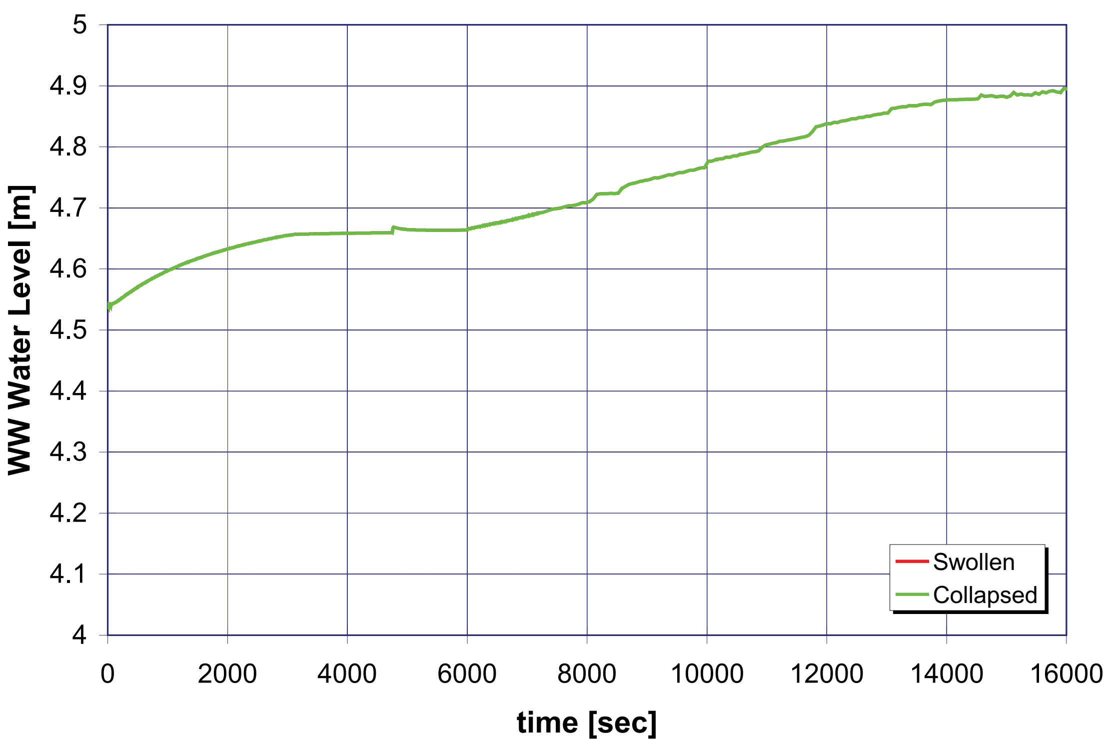


TFU
TCL
2200F
Maximum Core Temperature [K]
3000
2500
2000
1500
1000
500
0
0
2000
4000
6000
8000
10000
12000
14000
16000
time [sec]

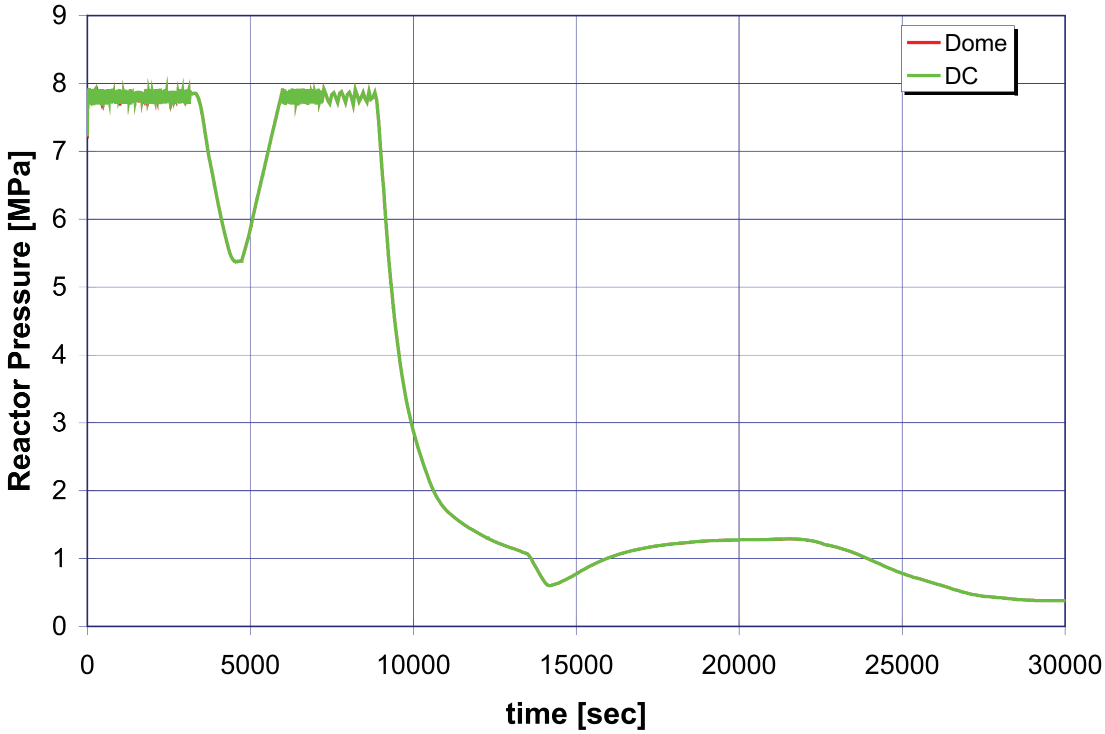


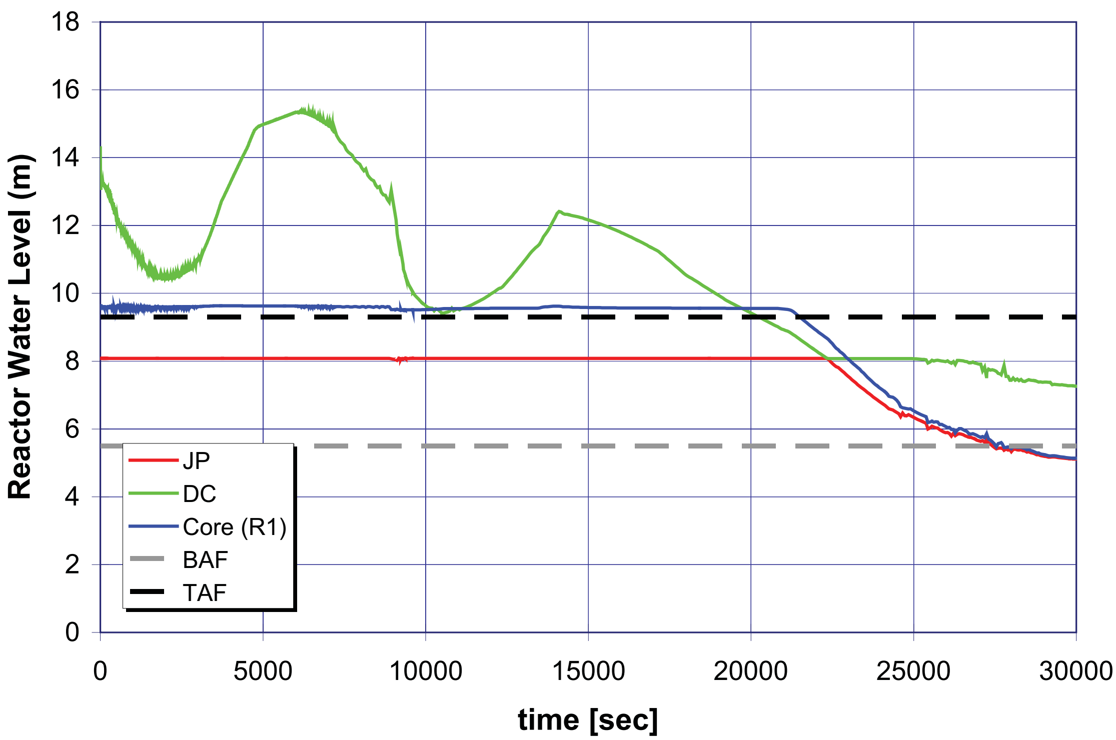
DW
WW
Containment Pressure [bar]
2
1.8
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
2000
4000
6000
8000
10000
12000
14000
16000
time [sec]

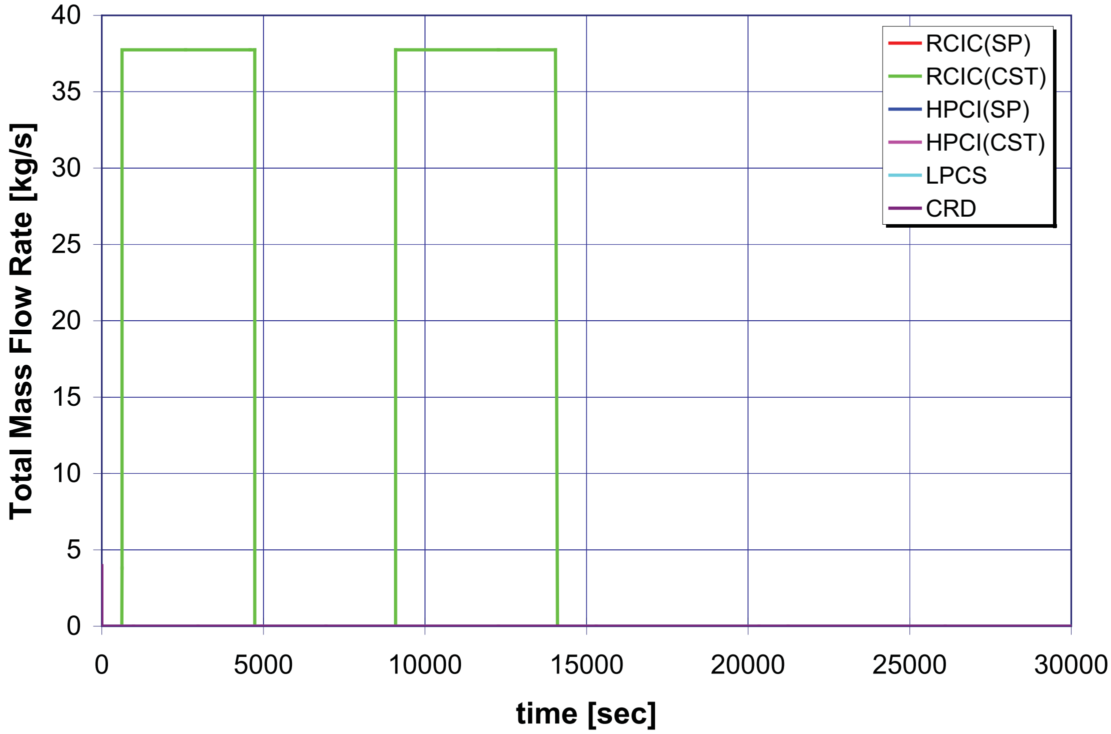

Wetwell Temperature [K]
430
410
390
370
350
330
310
290
270
250
0
2000
4000
6000
8000
10000
12000
14000
16000
time [sec]
TLIQ
TSAT
TVAP

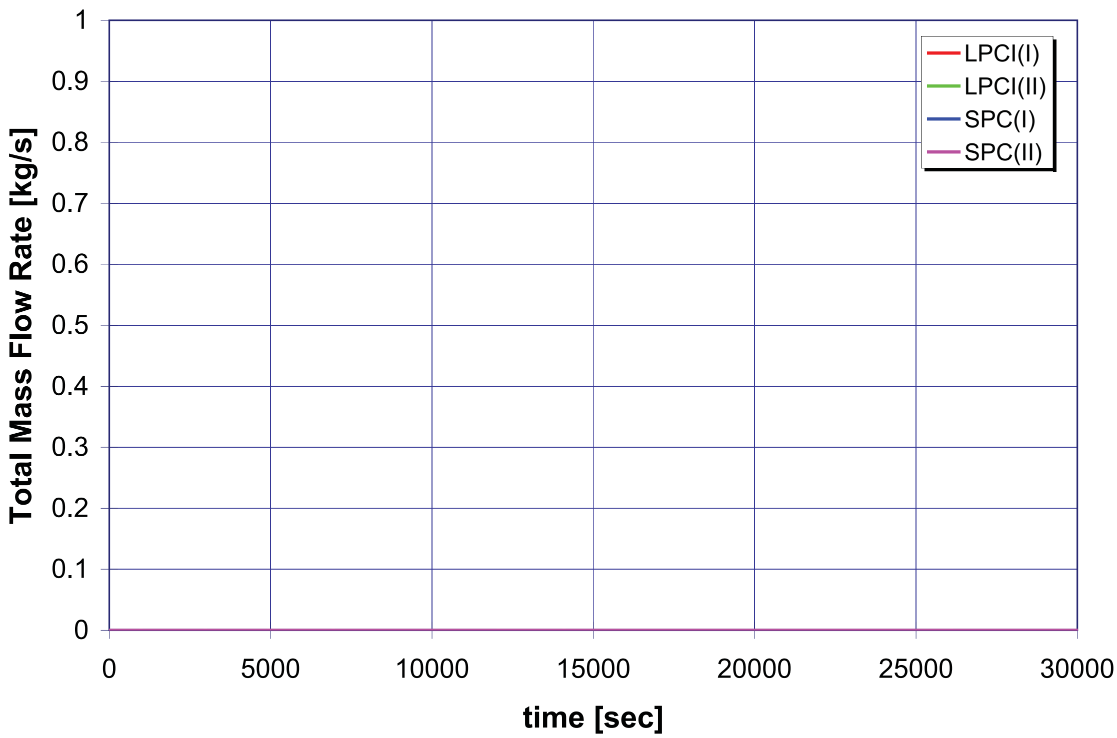

WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
2000
4000
6000
8000
10000
12000
14000
16000
time [sec]
Swollen
Collapsed

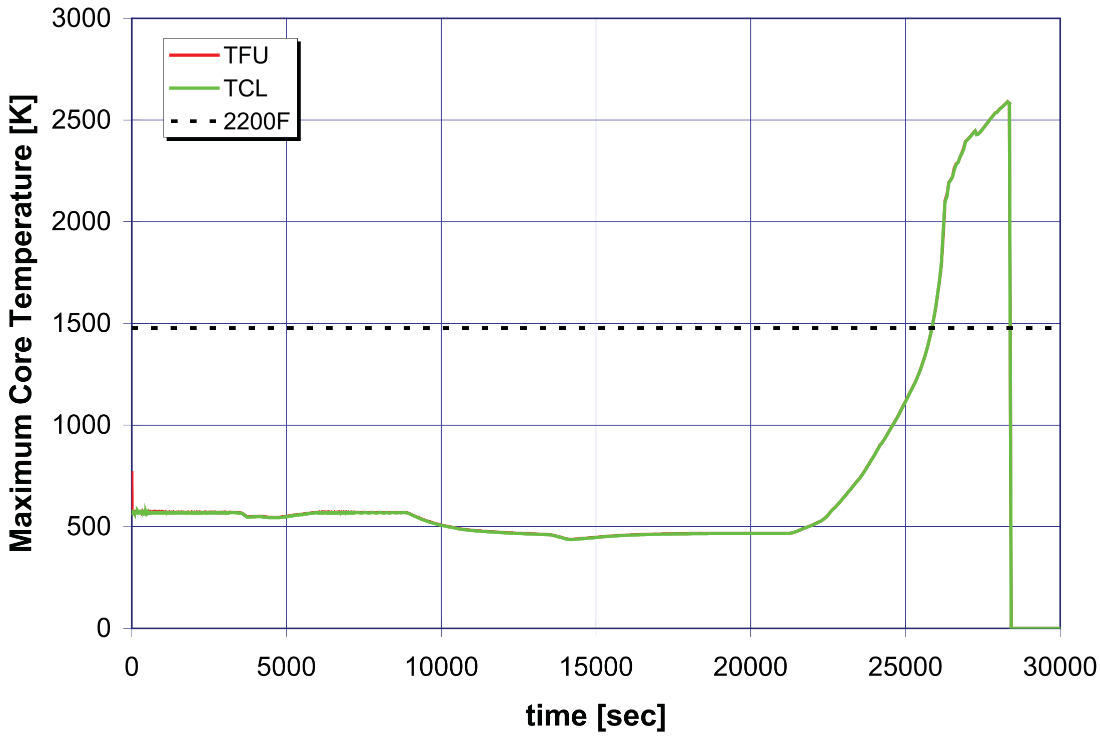
B.2.6 Case 6: Station Blackout and RCIC and SRV Stuck Open

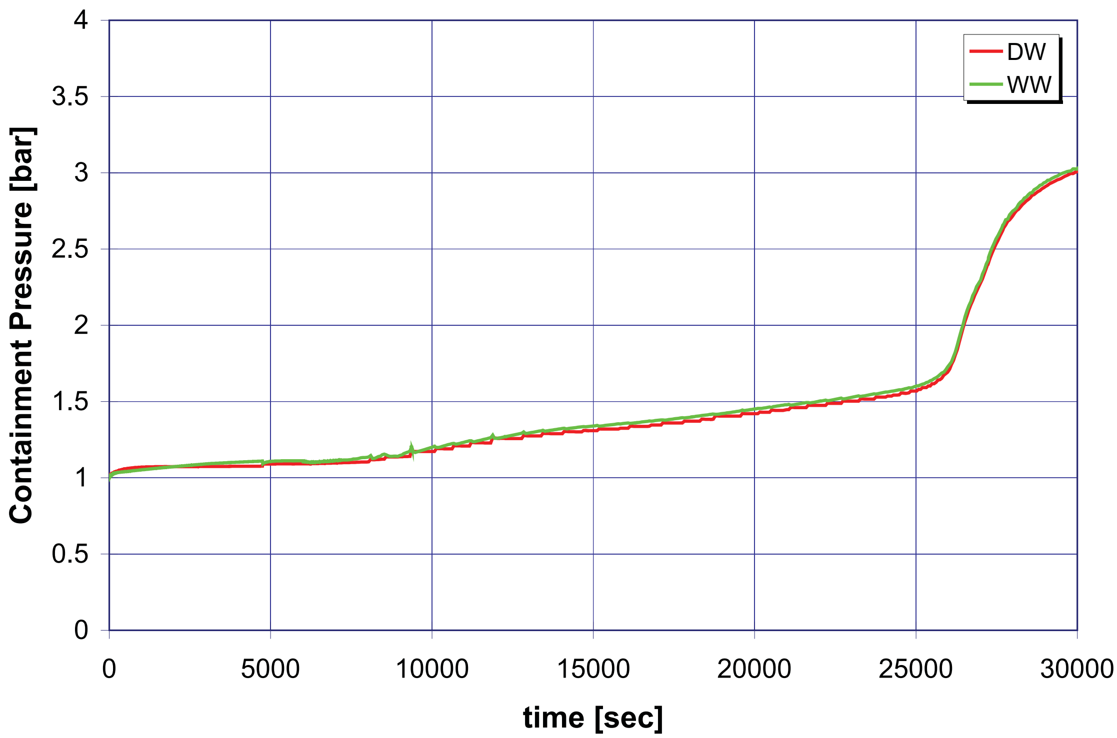


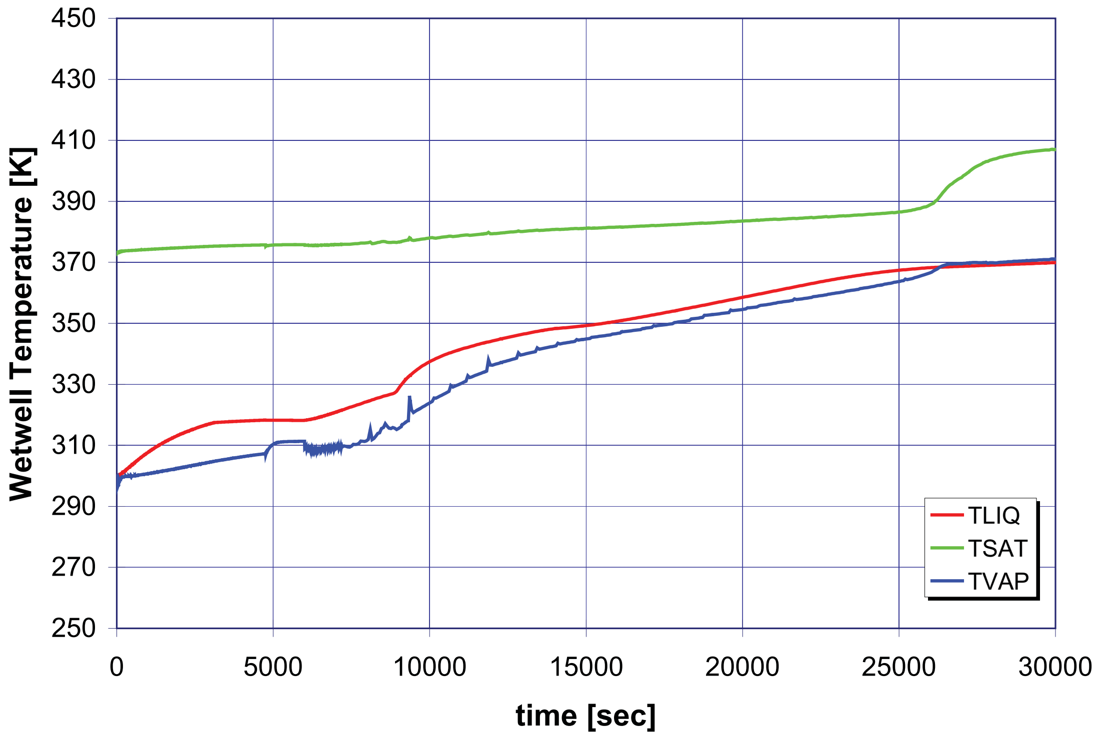



Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD
0
5
10
15
20
25
30
35
40
0
5000
10000
15000
20000
25000
30000

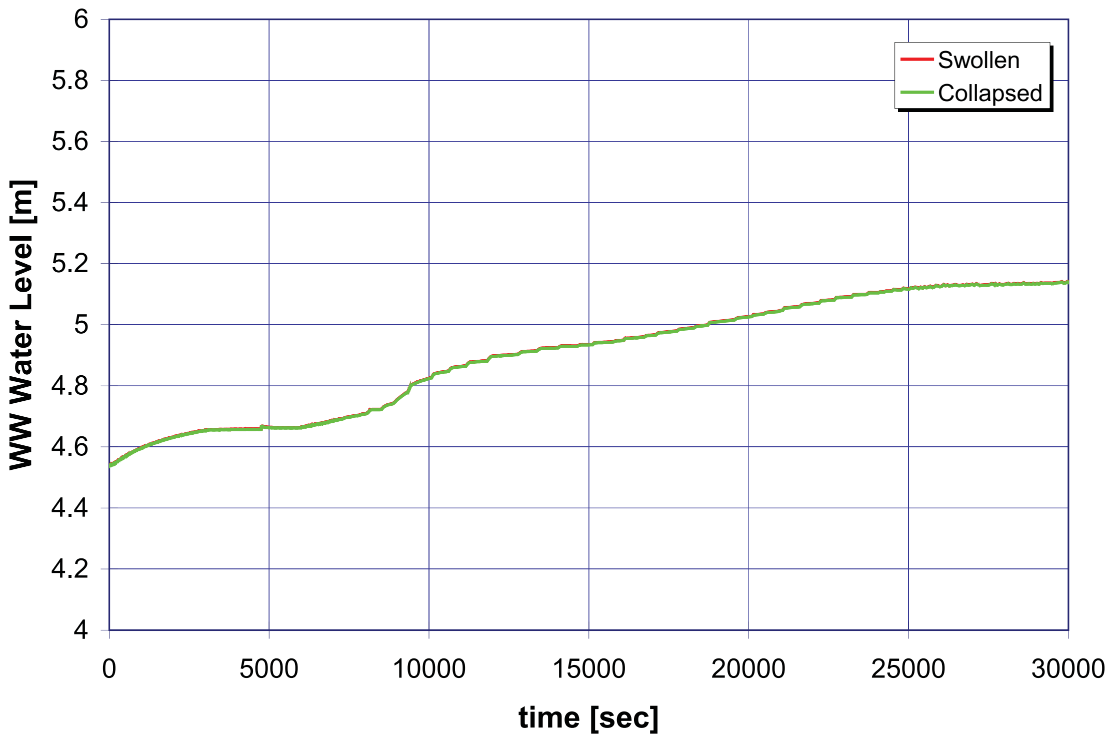

Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
0
5000
10000
15000
20000
25000
30000


3000
2500
2000
1500
1000
500
0
Maximum Core Temperature [K]
TFU
TCL
2200F
0
5000
10000
15000
20000
25000
30000
time [sec]


4
3.5
3
2.5
2
1.5
1
0.5
0
Containment Pressure [bar]
DW
WW
0
5000
10000
15000
20000
25000
30000
time [sec]



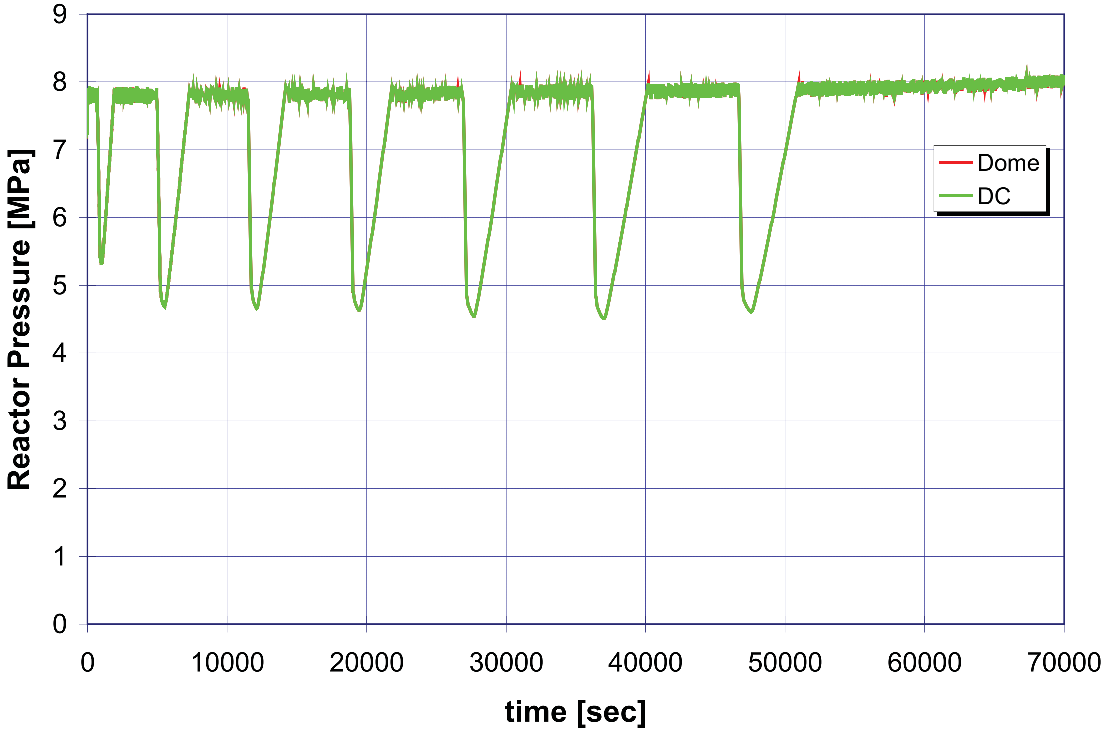
Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP
450
430
410
390
370
350
330
310
290
270
250
0
5000
10000
15000
20000
25000
30000

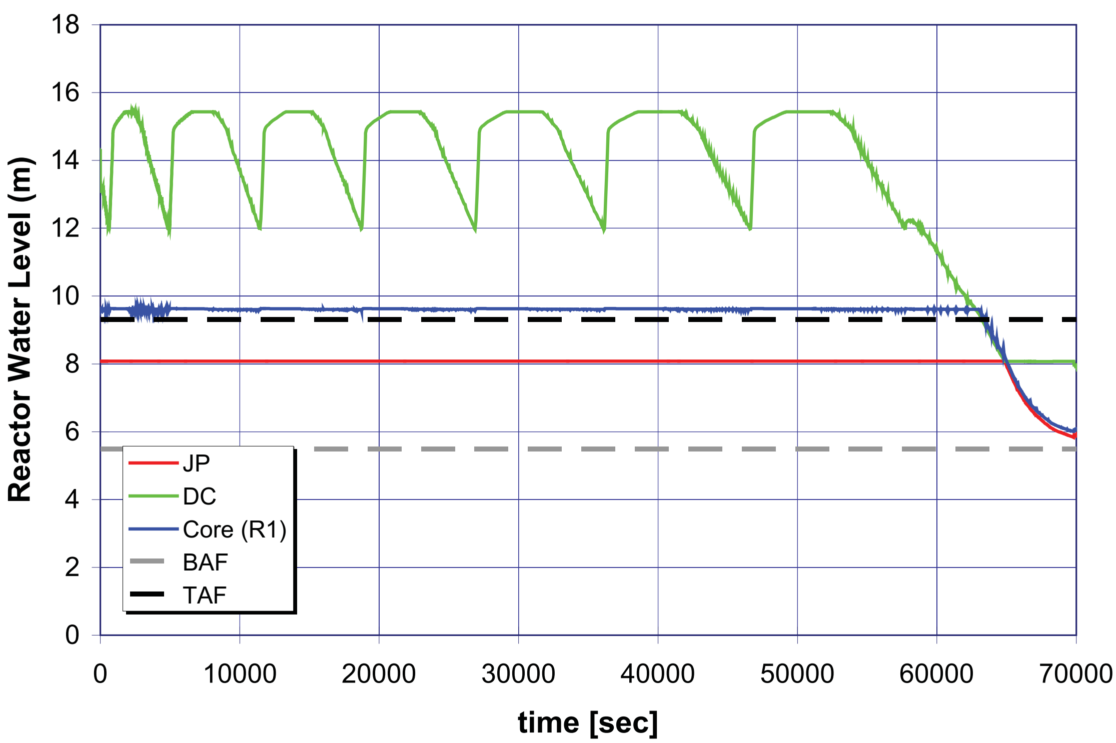


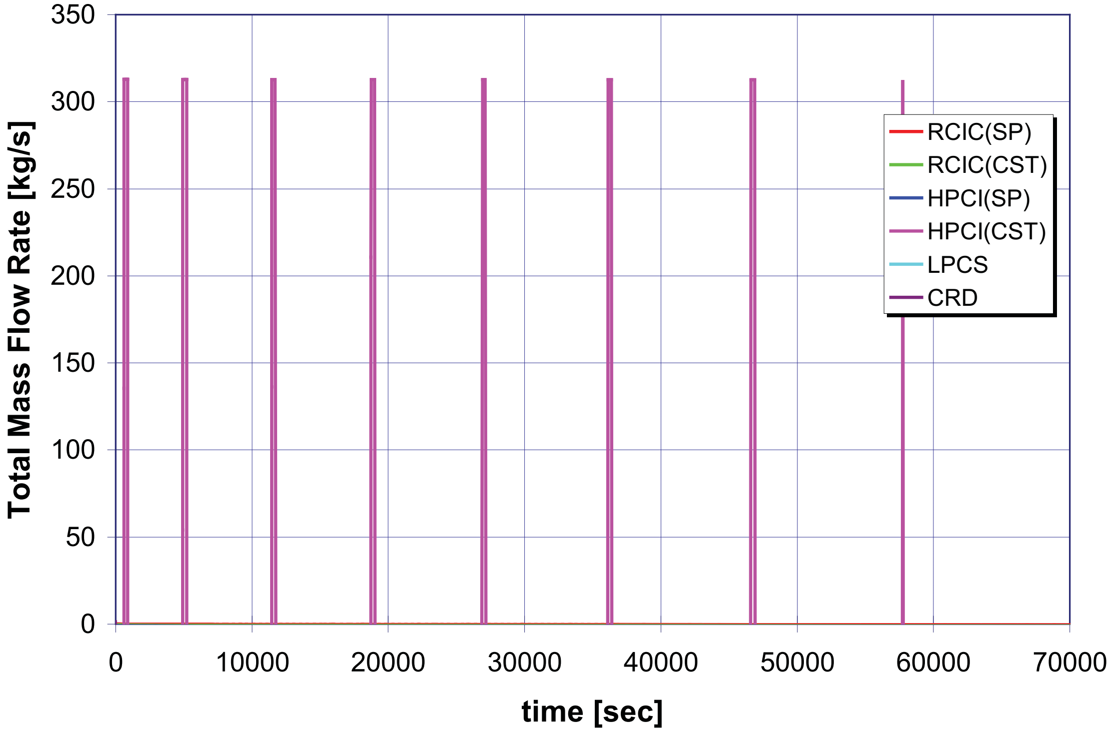
WW Water Level [m]
time [sec]
Swollen
Collapsed
6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
0
5000
10000
15000
20000
25000
30000

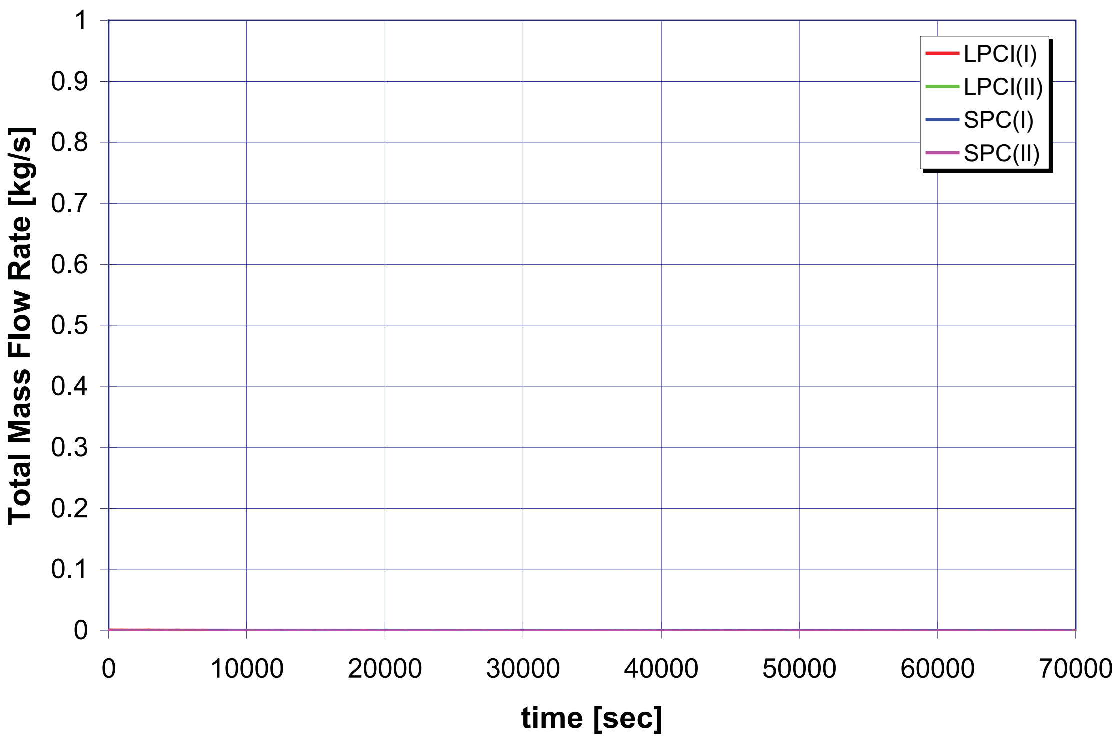
B.2.7 Case 7: Station Blackout and High-Pressure Coolant Injection

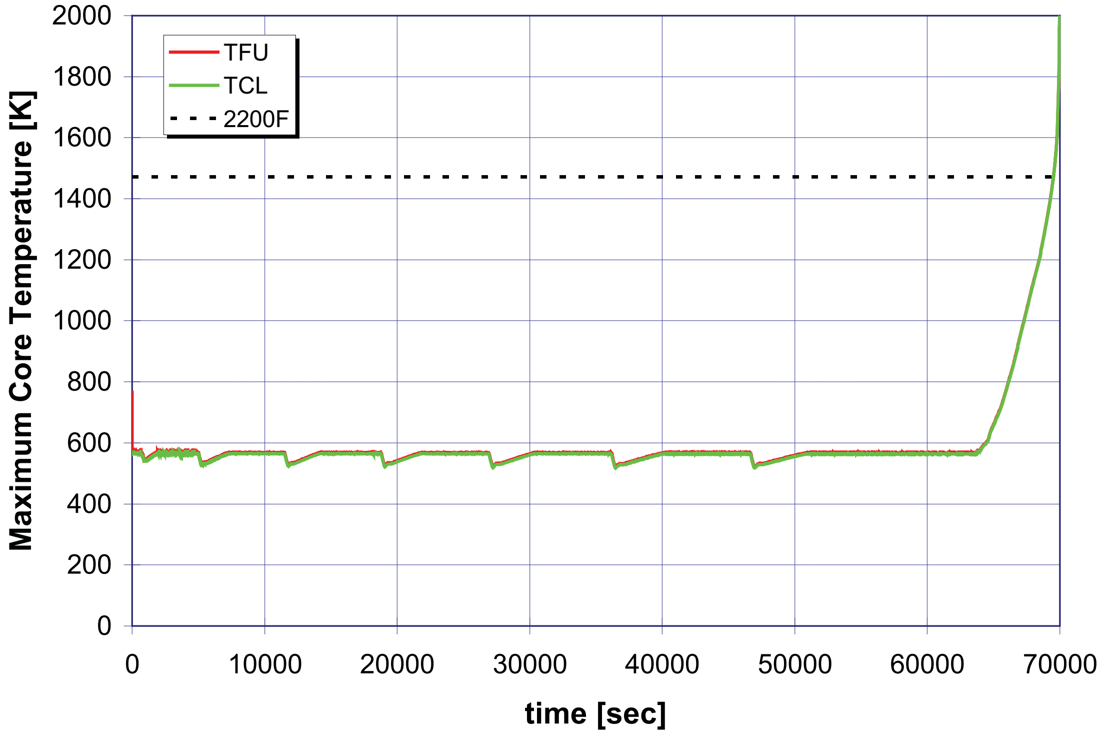
Note: By the time high-pressure coolant injection (HPCI) injection stops from the CST at 16.05 hours, the HPCI pump NPSH limit has already been exceeded at 12.07 hours.

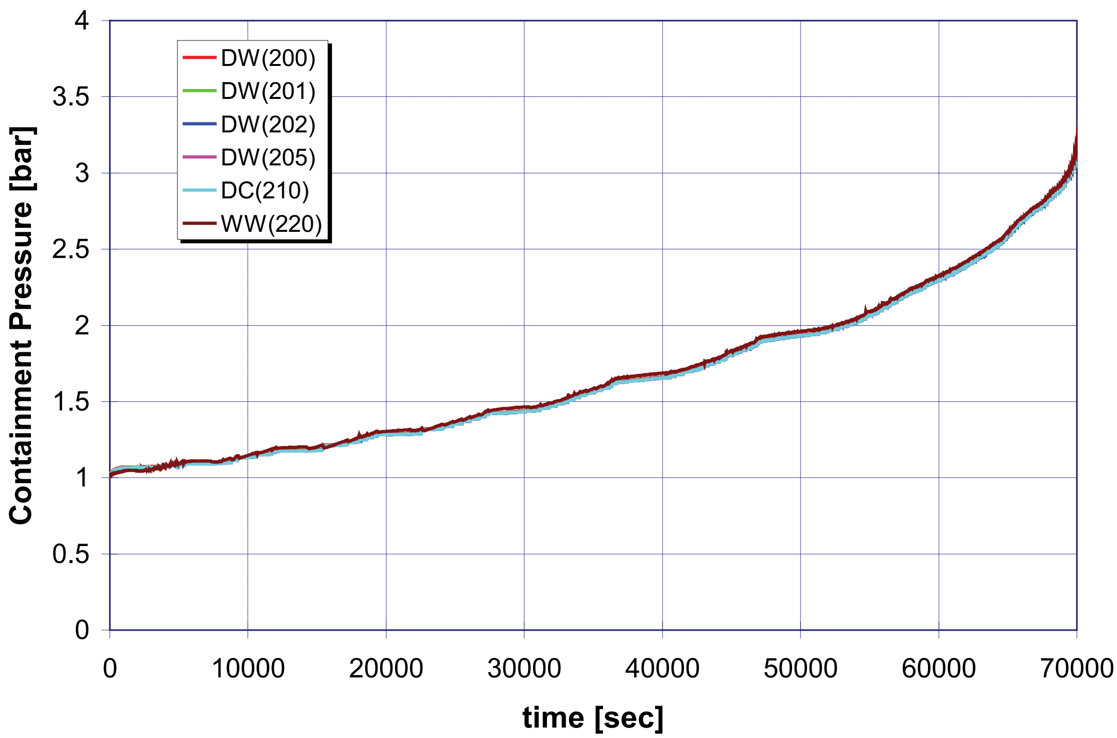


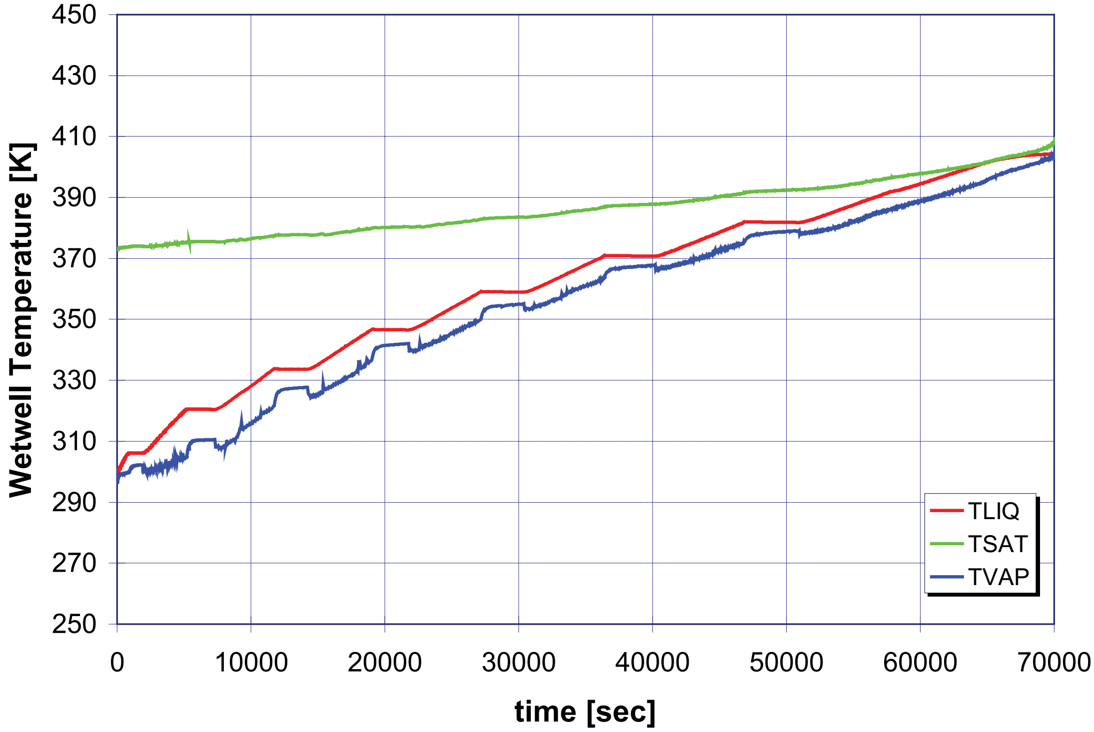



Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD
0
50
100
150
200
250
300
350
0
10000
20000
30000
40000
50000
60000
70000

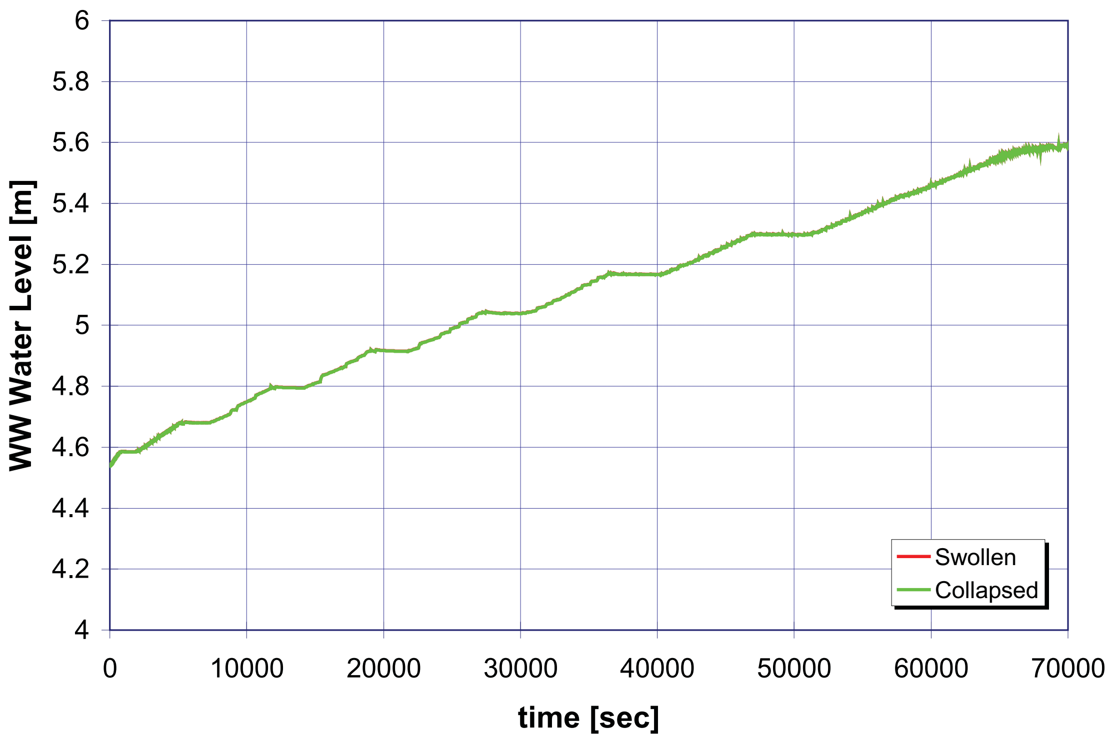

Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
0
10000
20000
30000
40000
50000
60000
70000


Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
10000
20000
30000
40000
50000
60000
70000


Containment Pressure [bar]
time [sec]
DW(200)
DW(201)
DW(202)
DW(205)
DC(210)
WW(220)
0
0.5
1
1.5
2
2.5
3
3.5
4
0
10000
20000
30000
40000
50000
60000
70000

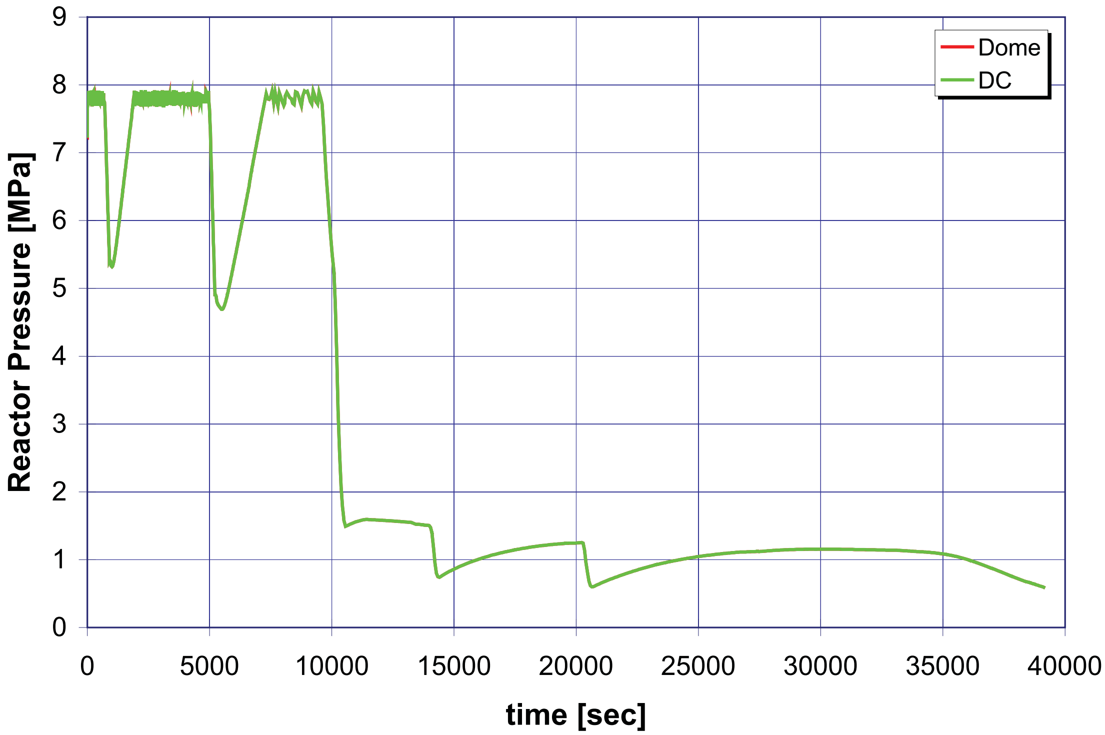


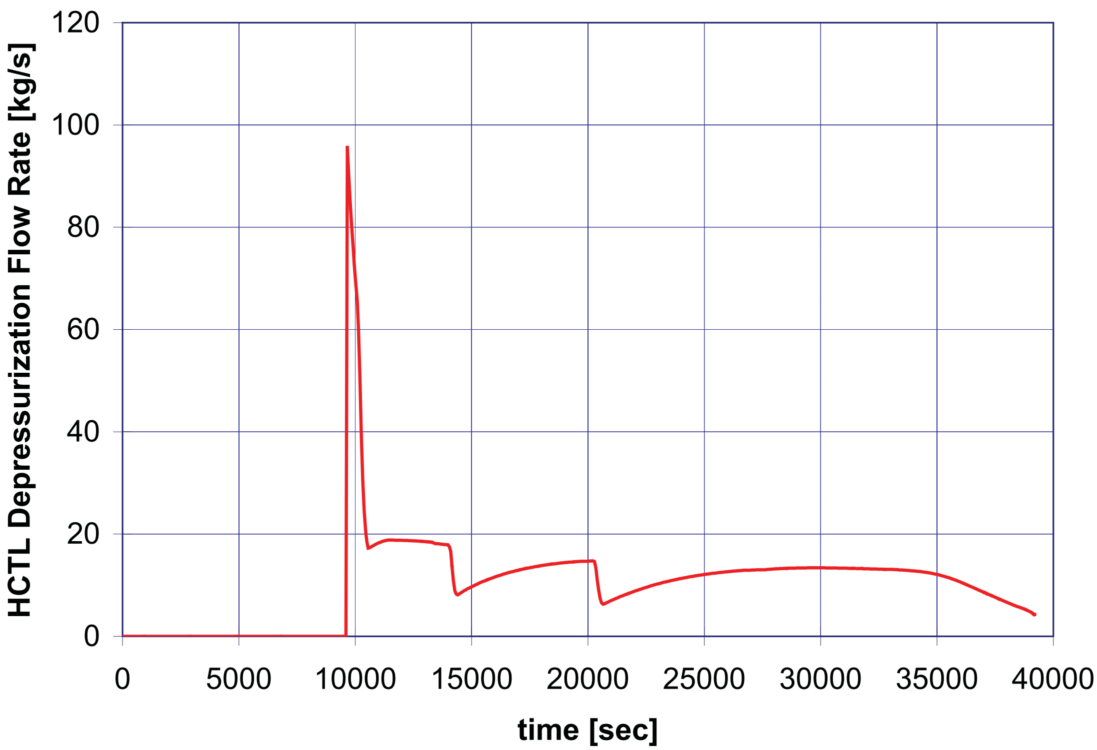
Wetwell Temperature [K]
450
430
410
390
370
350
330
310
290
270
250
0
10000
20000
30000
40000
50000
60000
70000
time [sec]
TLIQ
TSAT
TVAP

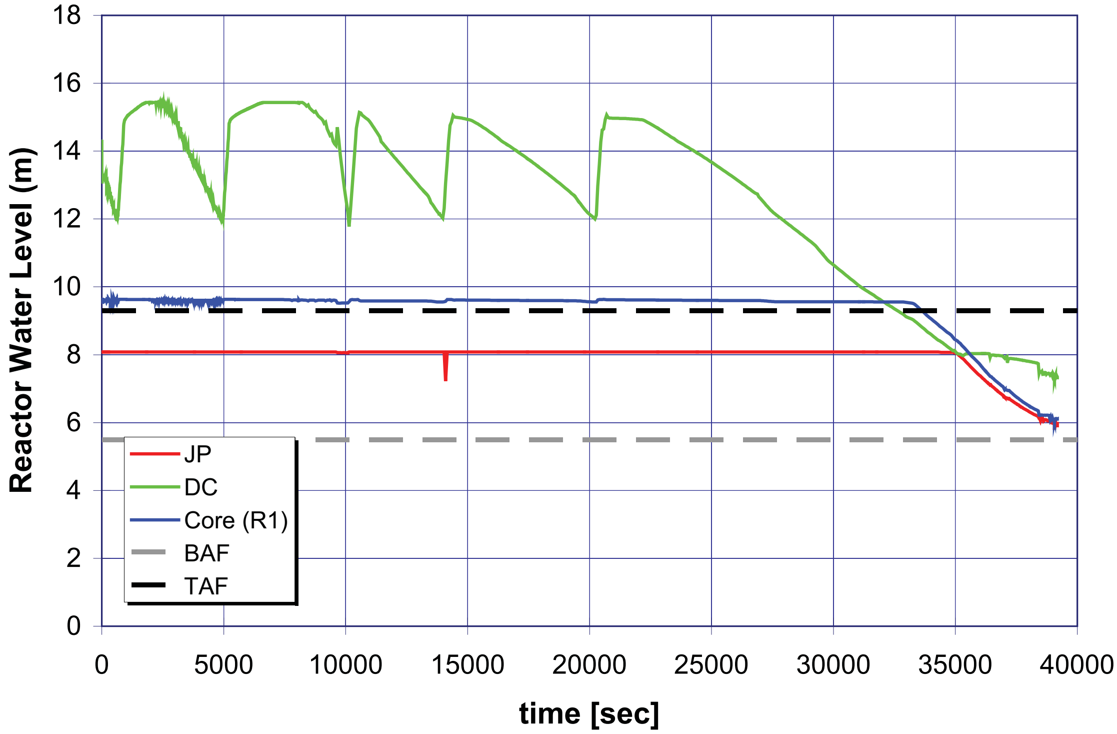


WW Water Level [m]
6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
0
10000
20000
30000
40000
50000
60000
70000
time [sec]
Swollen
Collapsed

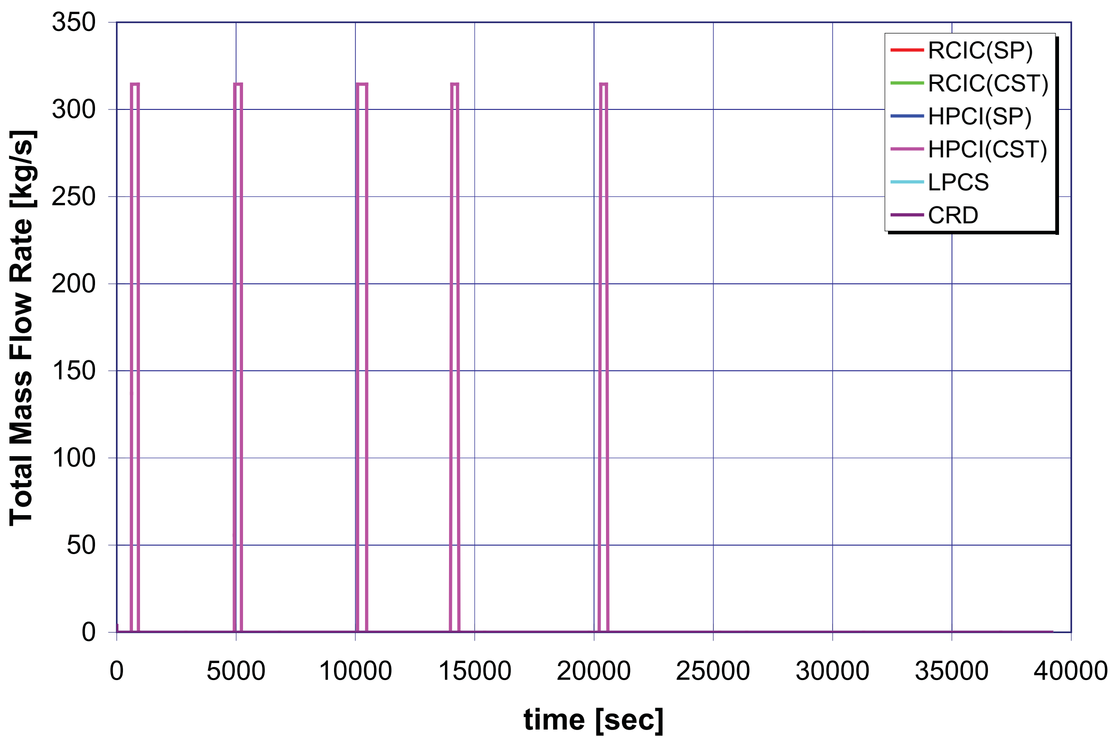
B.2.8 Case 8: Station Blackout and HPCI and HCTL Depressurization



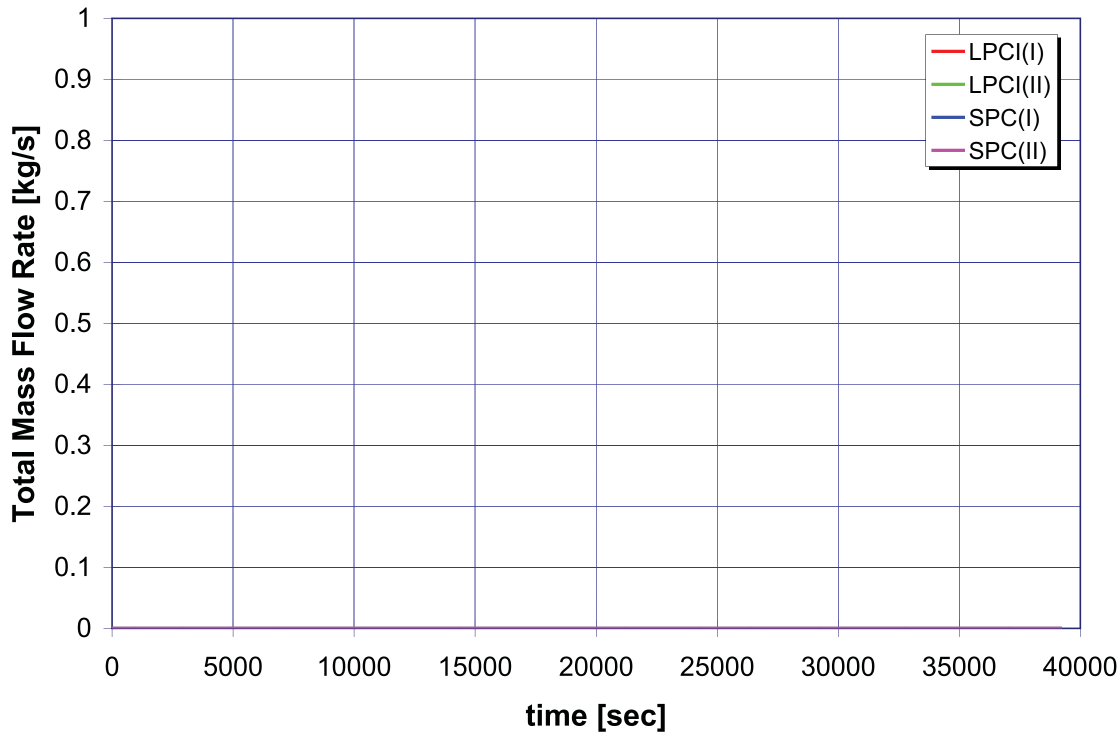


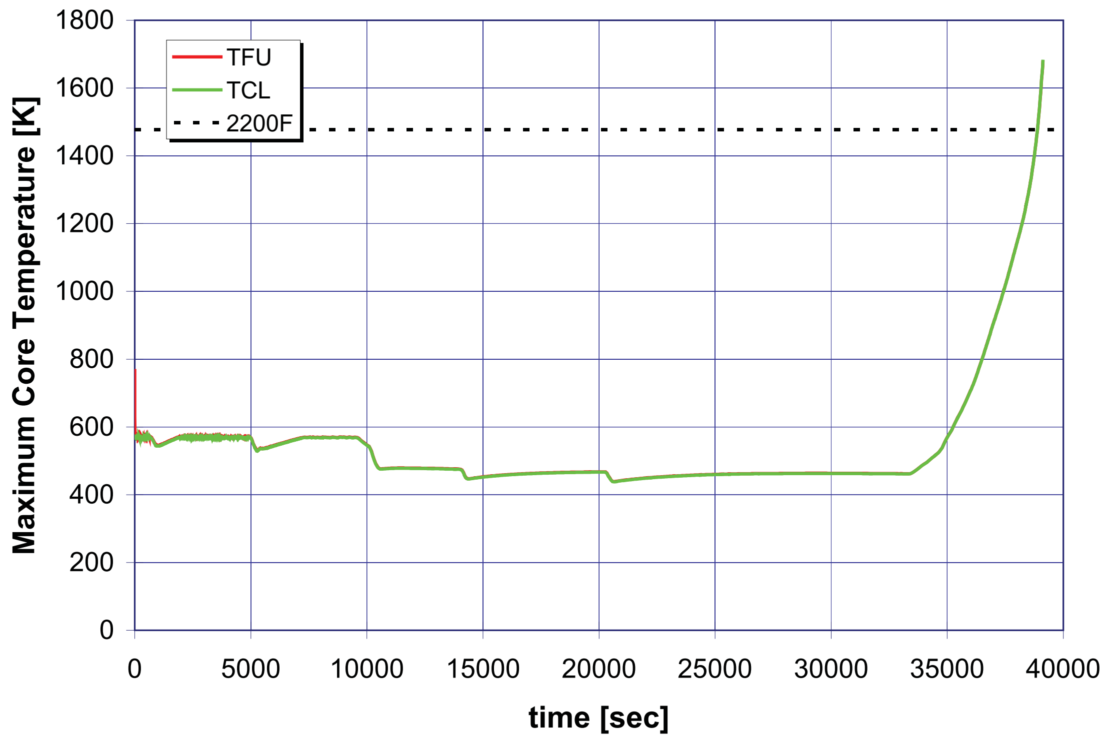


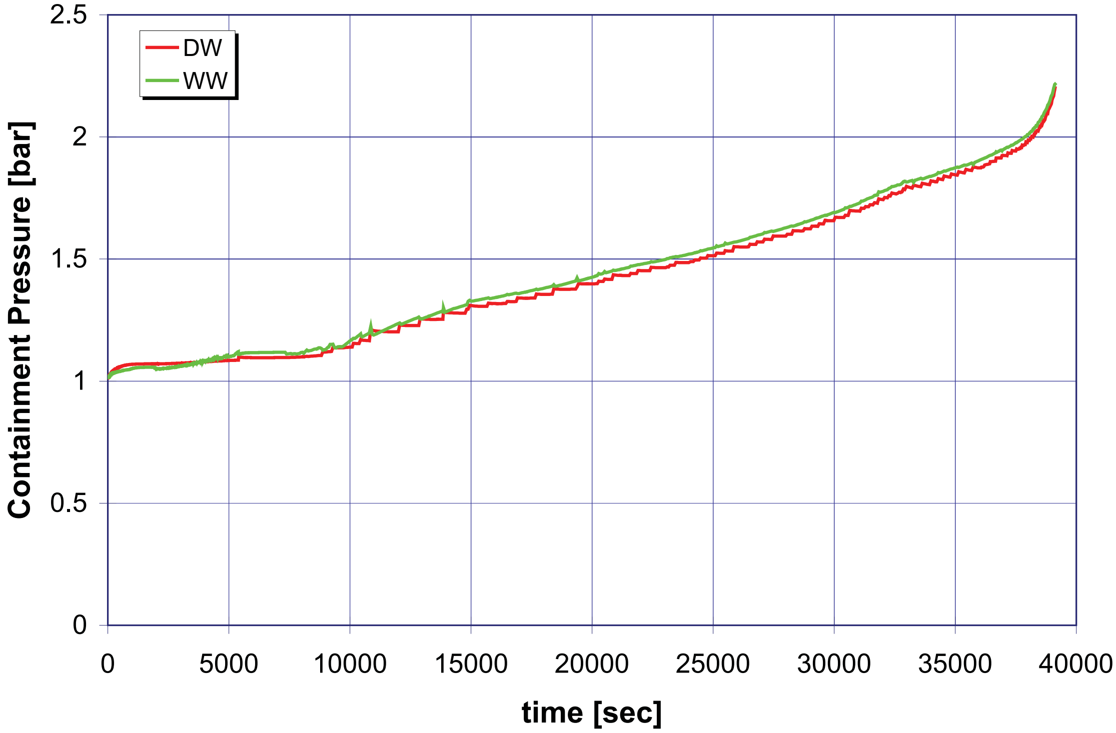
Reactor Water Level (m)
time [sec]
JP
DC
Core (R1)
BAF
TAF
0
2
4
6
8
10
12
14
16
18
0
5000
10000
15000
20000
25000
30000
35000
40000



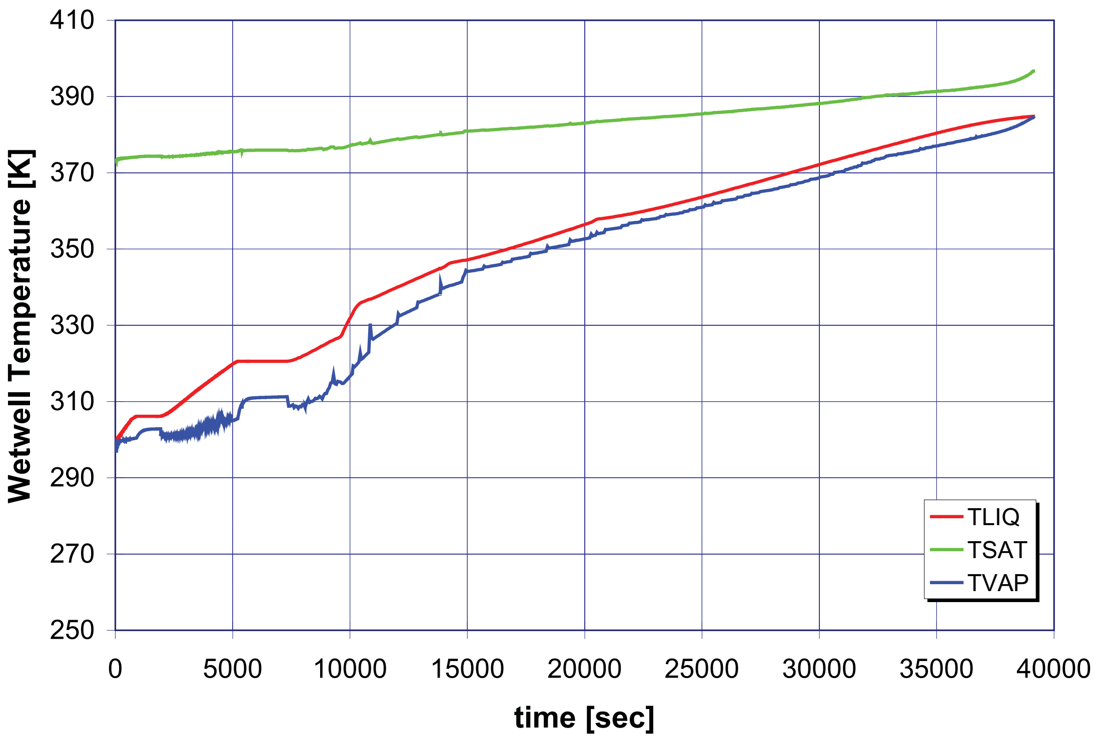
Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD
0
50
100
150
200
250
300
350
0
5000
10000
15000
20000
25000
30000
35000
40000


Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
0
5000
10000
15000
20000
25000
30000
35000
40000

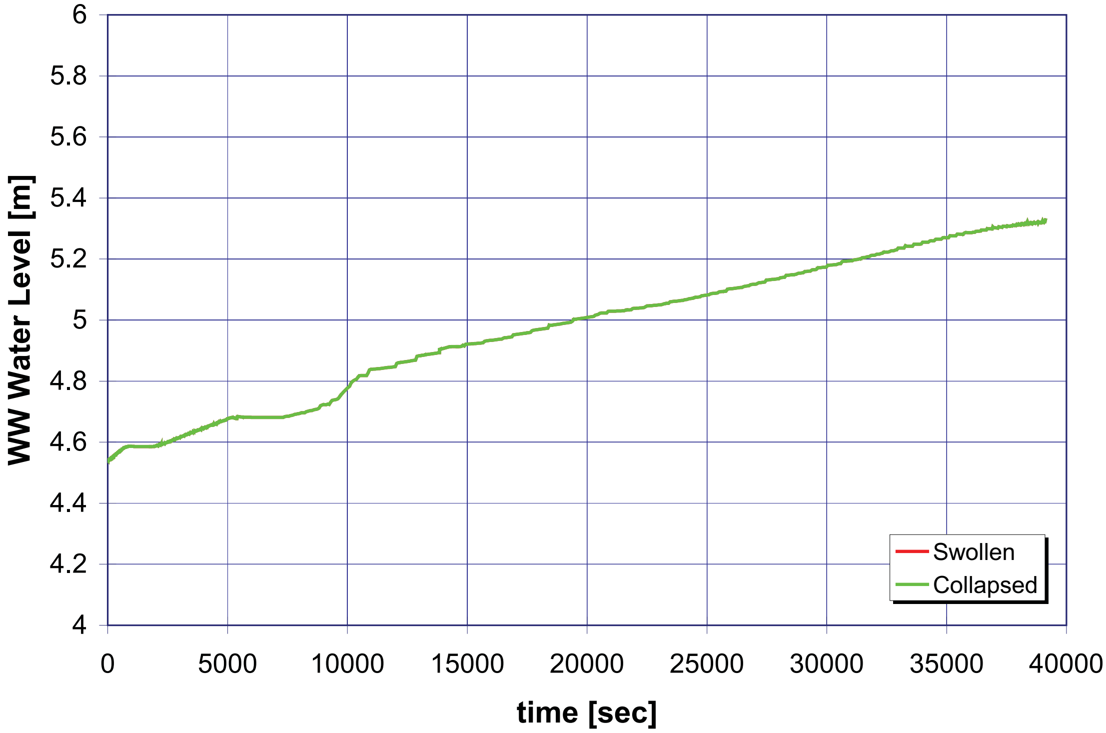

Maximum Core Temperature [K]
time [sec]
TFU
TCL
2200F
0
200
400
600
800
1000
1200
1400
1600
1800
0
5000
10000
15000
20000
25000
30000
35000
40000



Containment Pressure [bar]
time [sec]
DW
WW
0
0.5
1
1.5
2
2.5
0
5000
10000
15000
20000
25000
30000
35000
40000

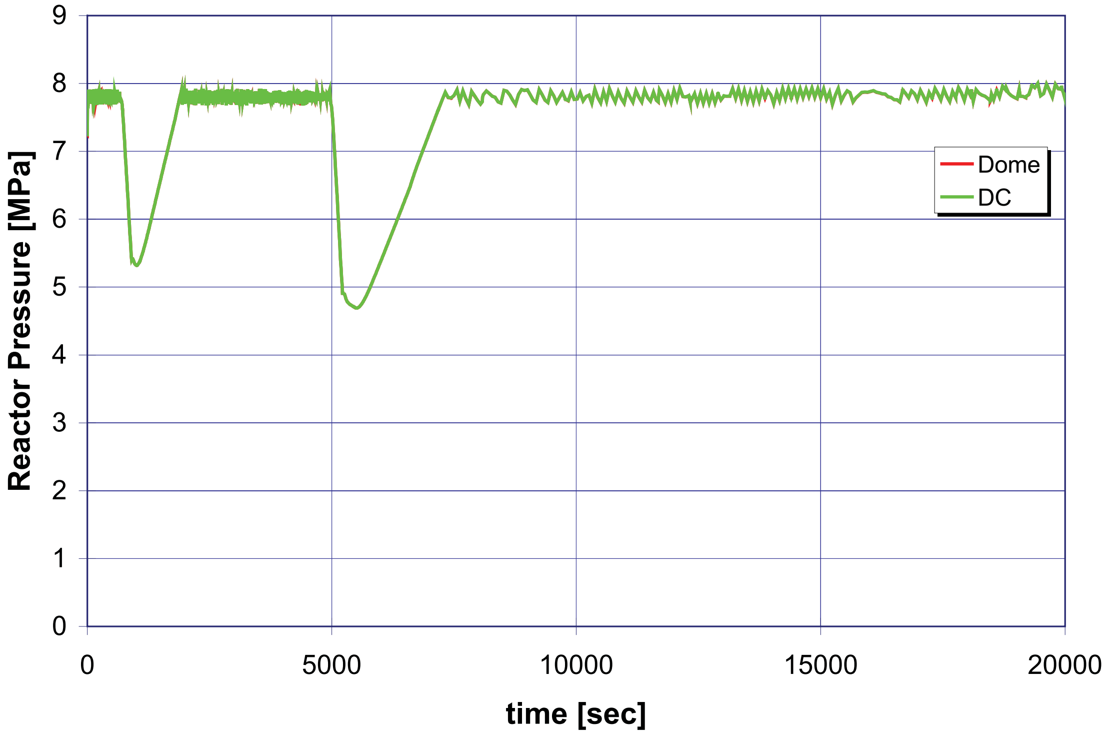


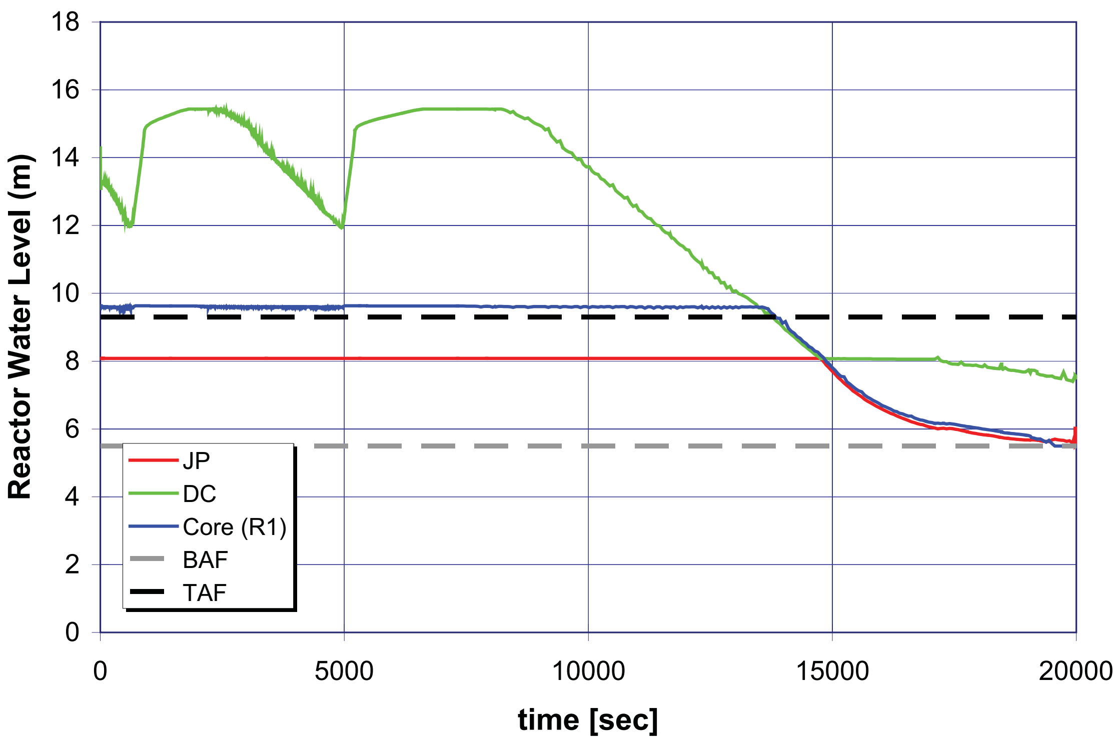
Wetwell Temperature [K]
time [sec]
TLIQ
TSAT
TVAP
250
270
290
310
330
350
370
390
410
0
5000
10000
15000
20000
25000
30000
35000
40000

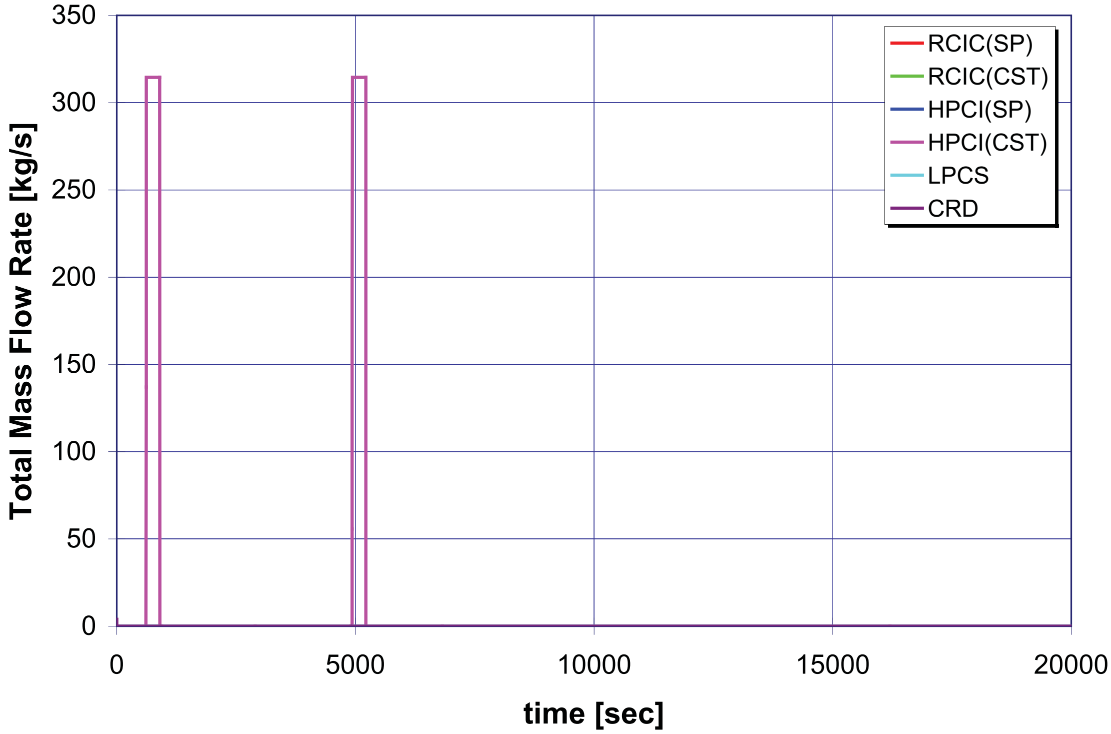


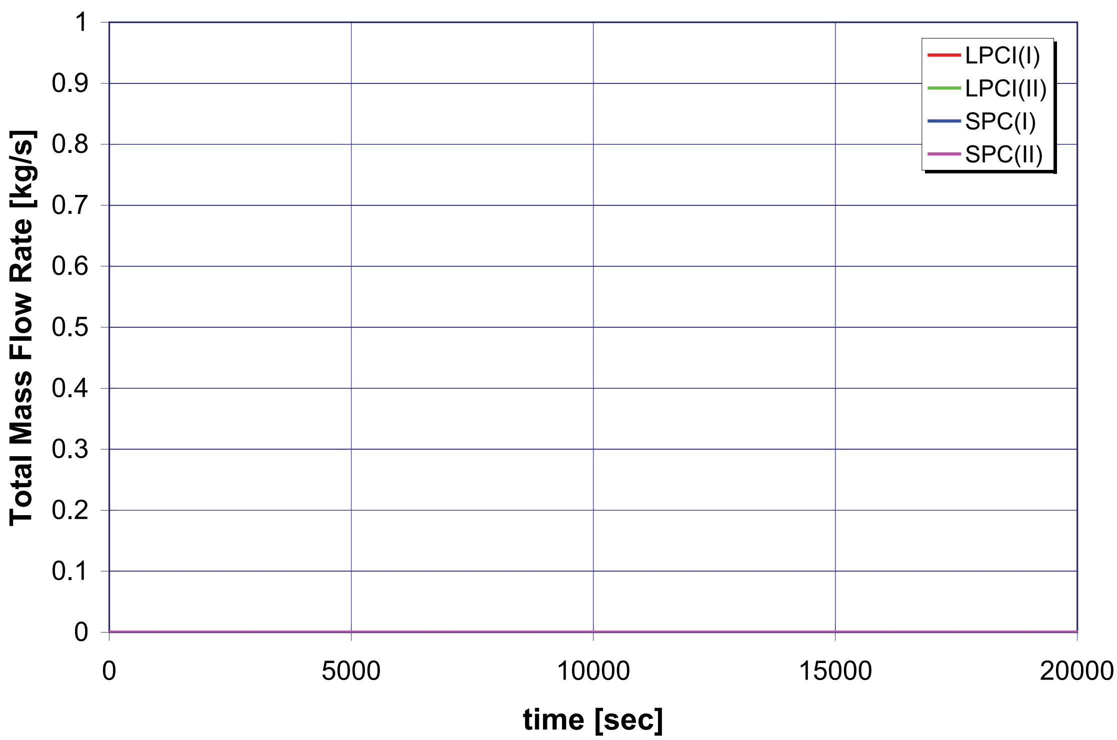
WW Water Level [m]
6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
0
5000
10000
15000
20000
25000
30000
35000
40000
time [sec]
Swollen
Collapsed

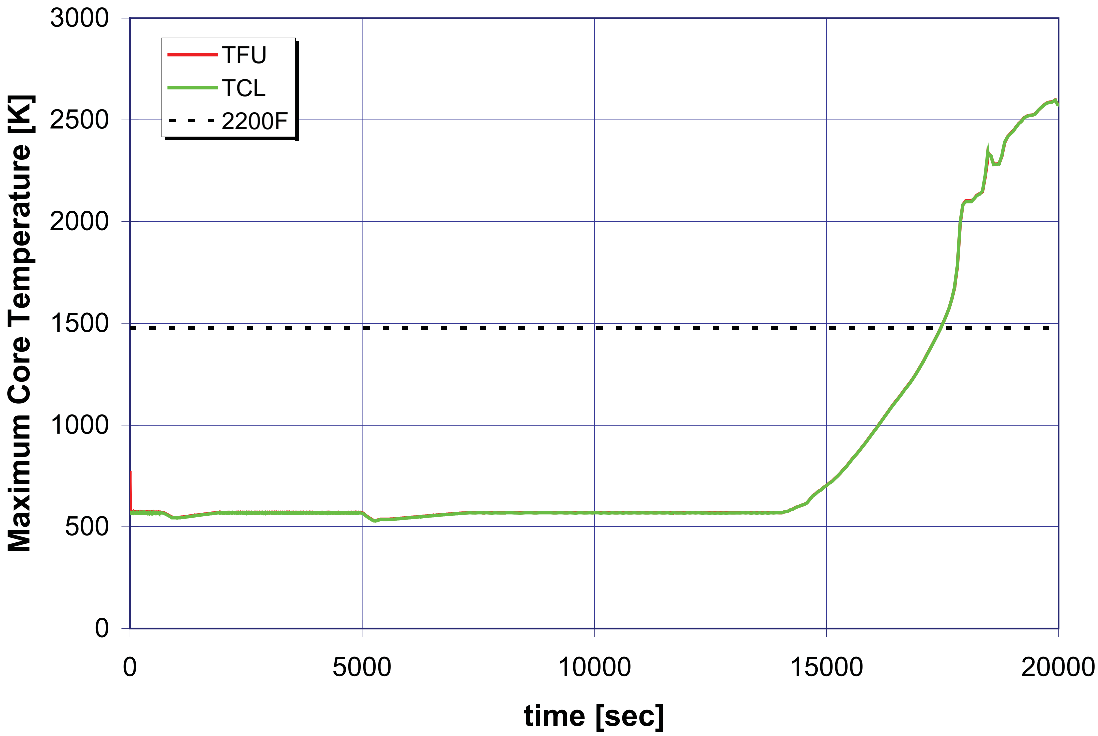
B.2.9 Case 9: Station Blackout and HPCI and 2-Hour Direct Current



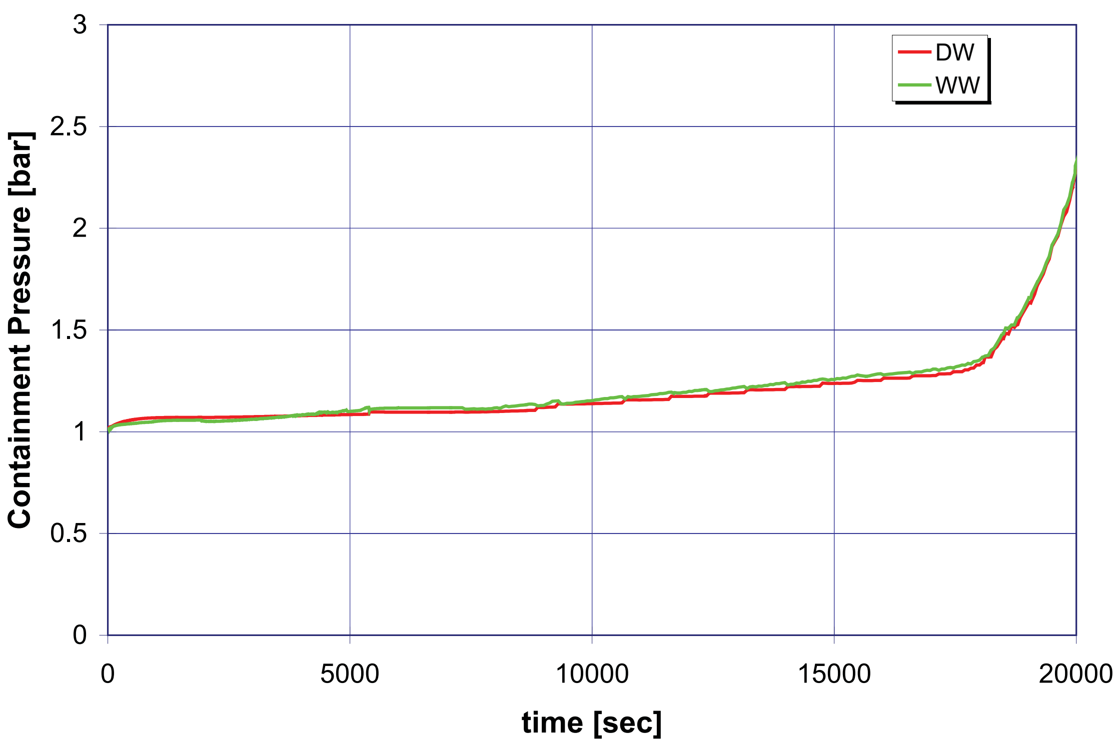



350
300
250
200
150
100
50
0
Total Mass Flow Rate [kg/s]
0
5000
10000
15000
20000
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD

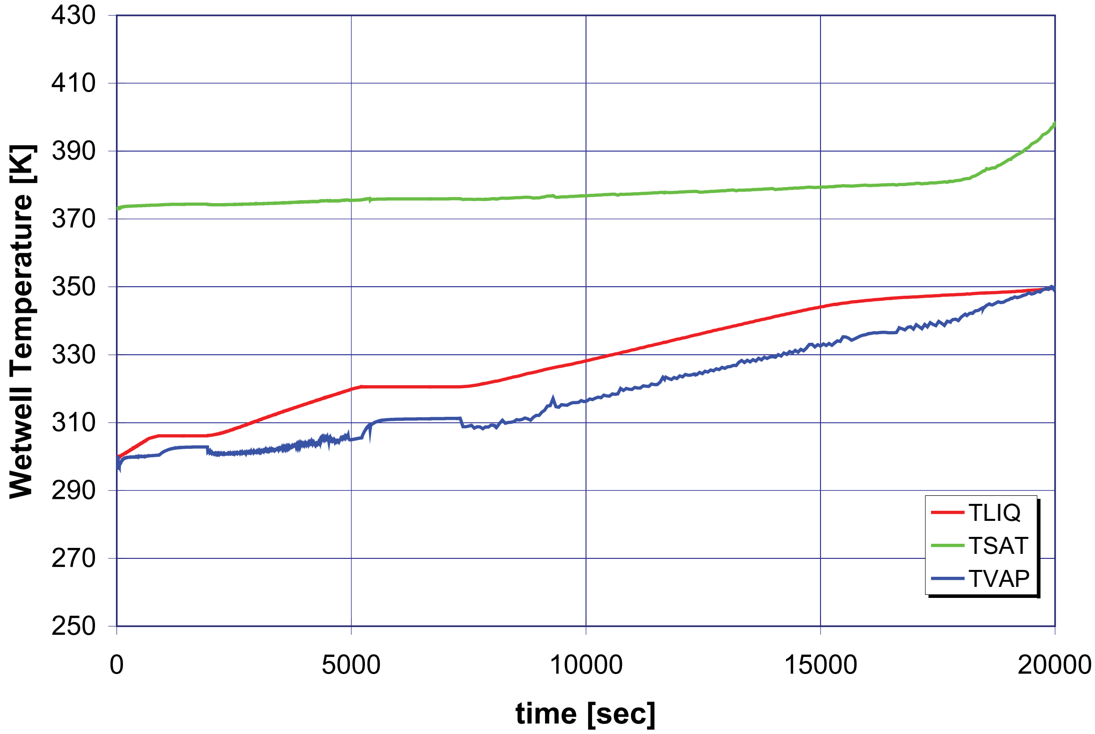

1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
Total Mass Flow Rate [kg/s]
0
5000
10000
15000
20000
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)

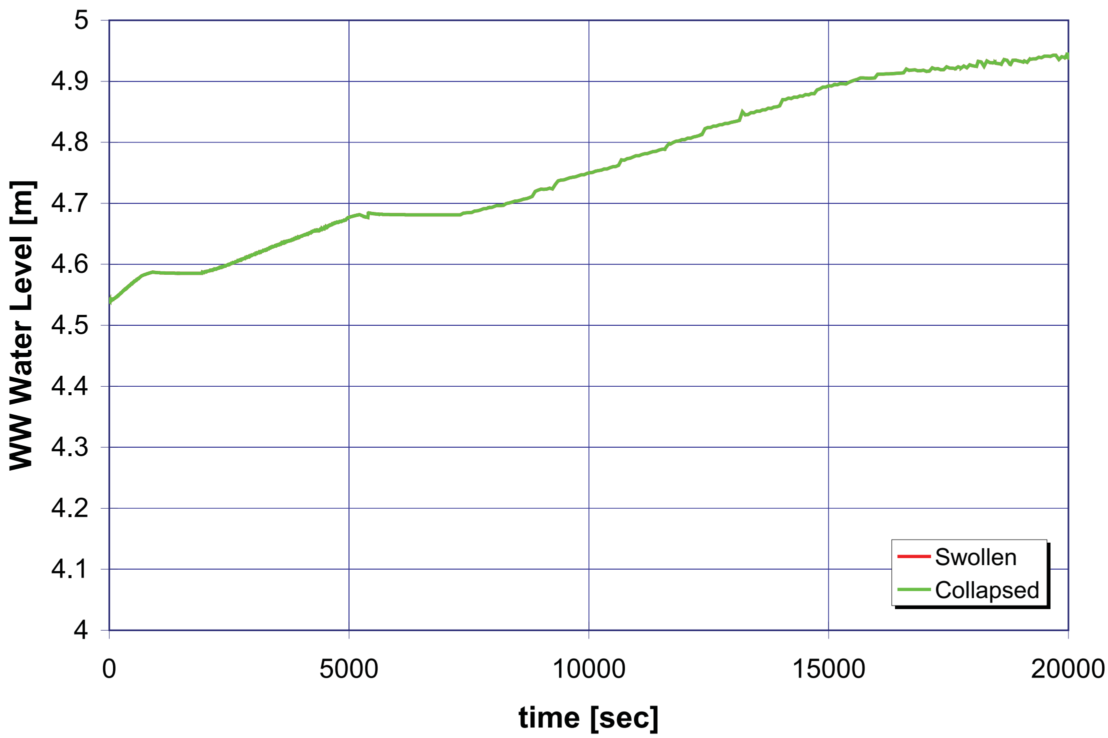


TFU
TCL
2200F
Maximum Core Temperature [K]
3000
2500
2000
1500
1000
500
0
0
5000
10000
15000
20000
time [sec]

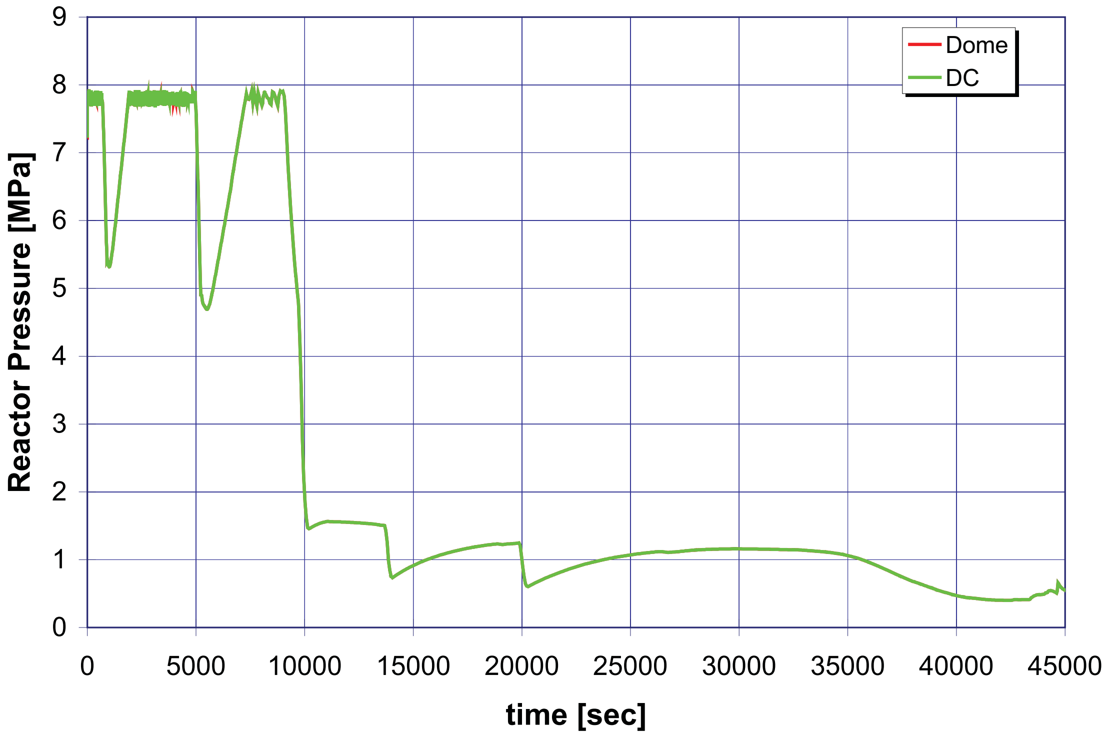


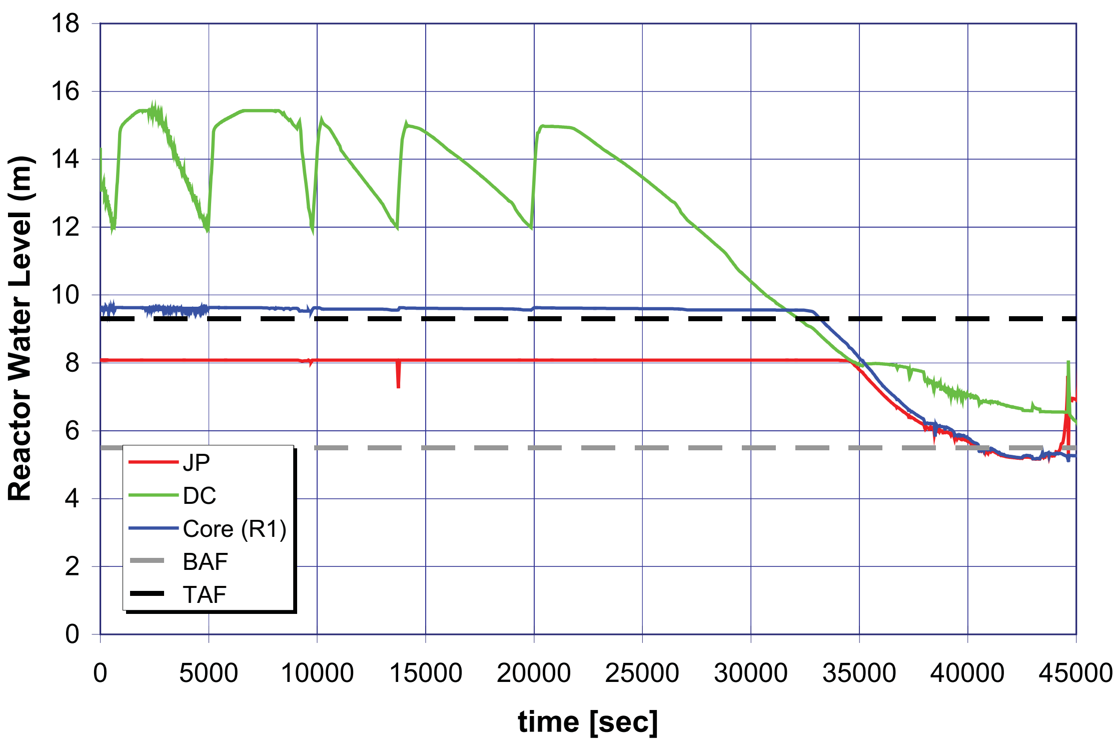
DW
WW
Containment Pressure [bar]
3
2.5
2
1.5
1
0.5
0
0
5000
10000
15000
20000
time [sec]

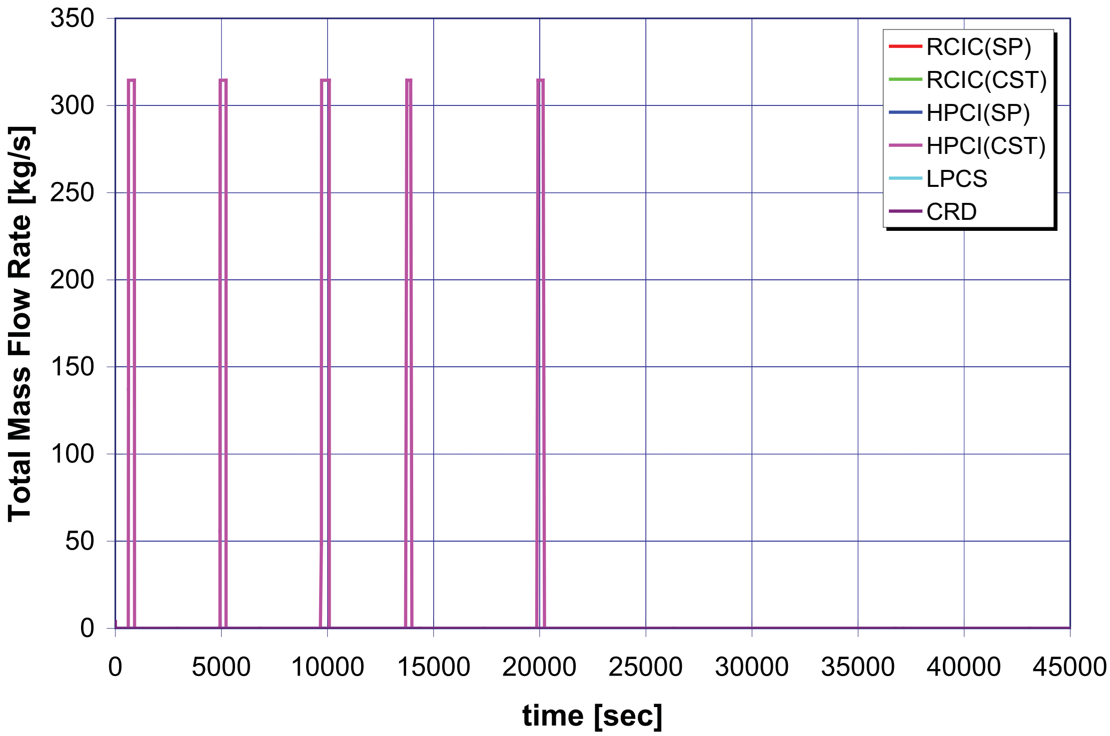

Wetwell Temperature [K]
430
410
390
370
350
330
310
290
270
250
0
5000
10000
15000
20000
time [sec]
TLIQ
TSAT
TVAP


WW Water Level [m]
5
4.9
4.8
4.7
4.6
4.5
4.4
4.3
4.2
4.1
4
0
5000
10000
15000
20000
time [sec]
Swollen
Collapsed

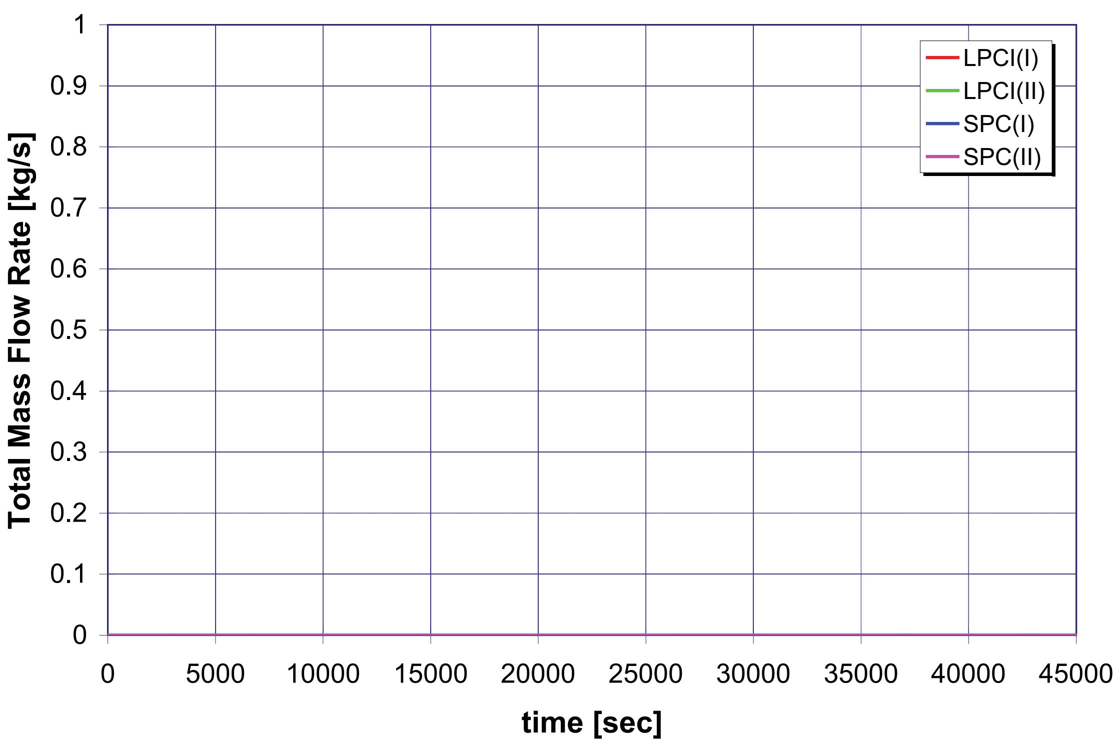
B.2.10 Case 10: Station Blackout and HPCI and SRV Stuck Open



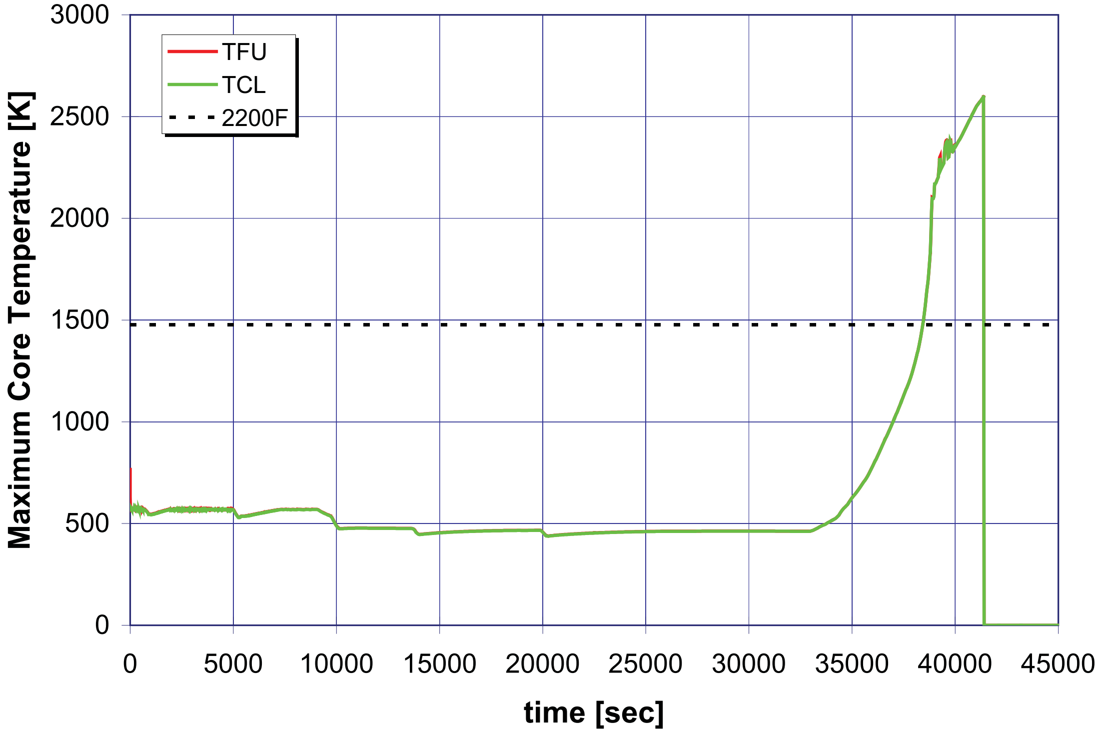


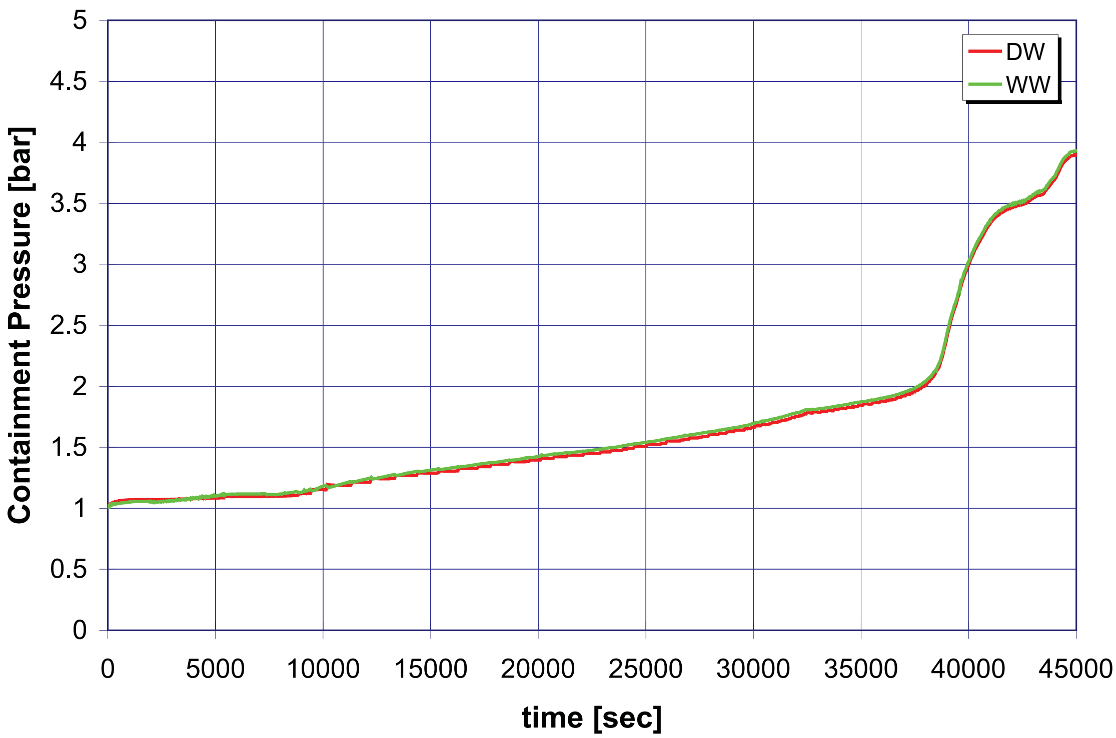


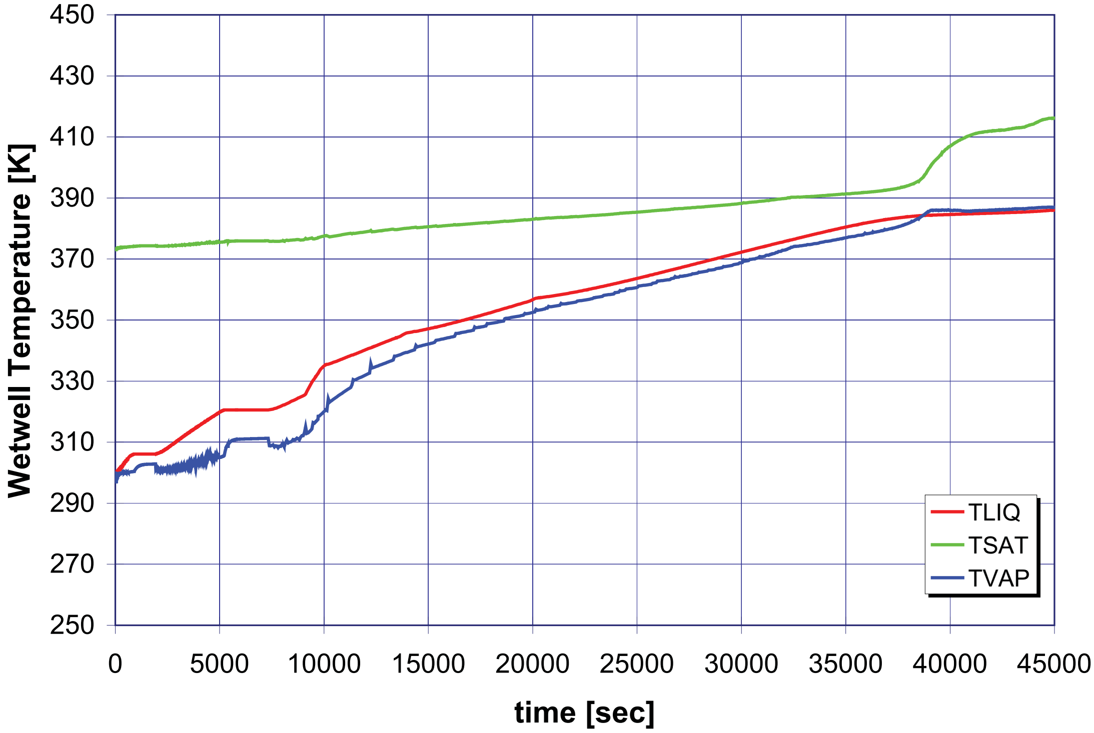
Total Mass Flow Rate [kg/s]
time [sec]
RCIC(SP)
RCIC(CST)
HPCI(SP)
HPCI(CST)
LPCS
CRD
0
50
100
150
200
250
300
350
0
5000
10000
15000
20000
25000
30000
35000
40000
45000



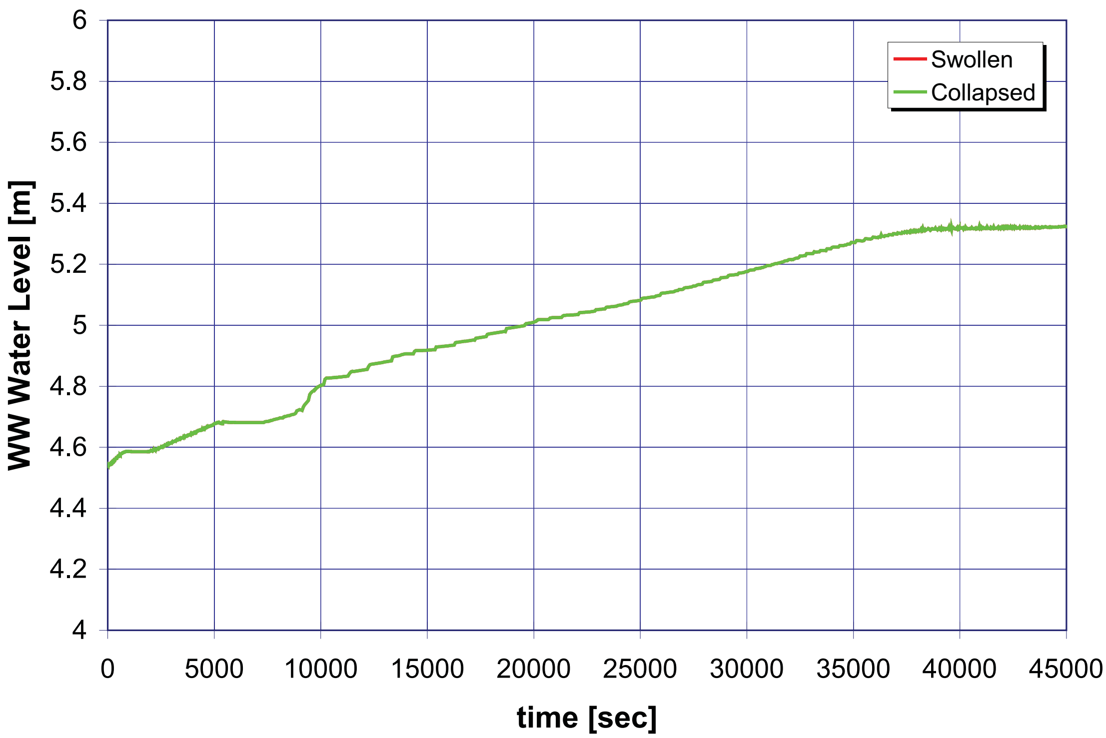
Total Mass Flow Rate [kg/s]
time [sec]
LPCI(I)
LPCI(II)
SPC(I)
SPC(II)
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
0
5000
10000
15000
20000
25000
30000
35000
40000
45000


TFU
TCL
2200F
Maximum Core Temperature [K]
3000
2500
2000
1500
1000
500
0
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
time [sec]


DW
WW
Containment Pressure [bar]
5
4.5
4
3.5
3
2.5
2
1.5
1
0.5
0
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
time [sec]



Wetwell Temperature [K]
450
430
410
390
370
350
330
310
290
270
250
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
time [sec]
TLIQ
TSAT
TVAP



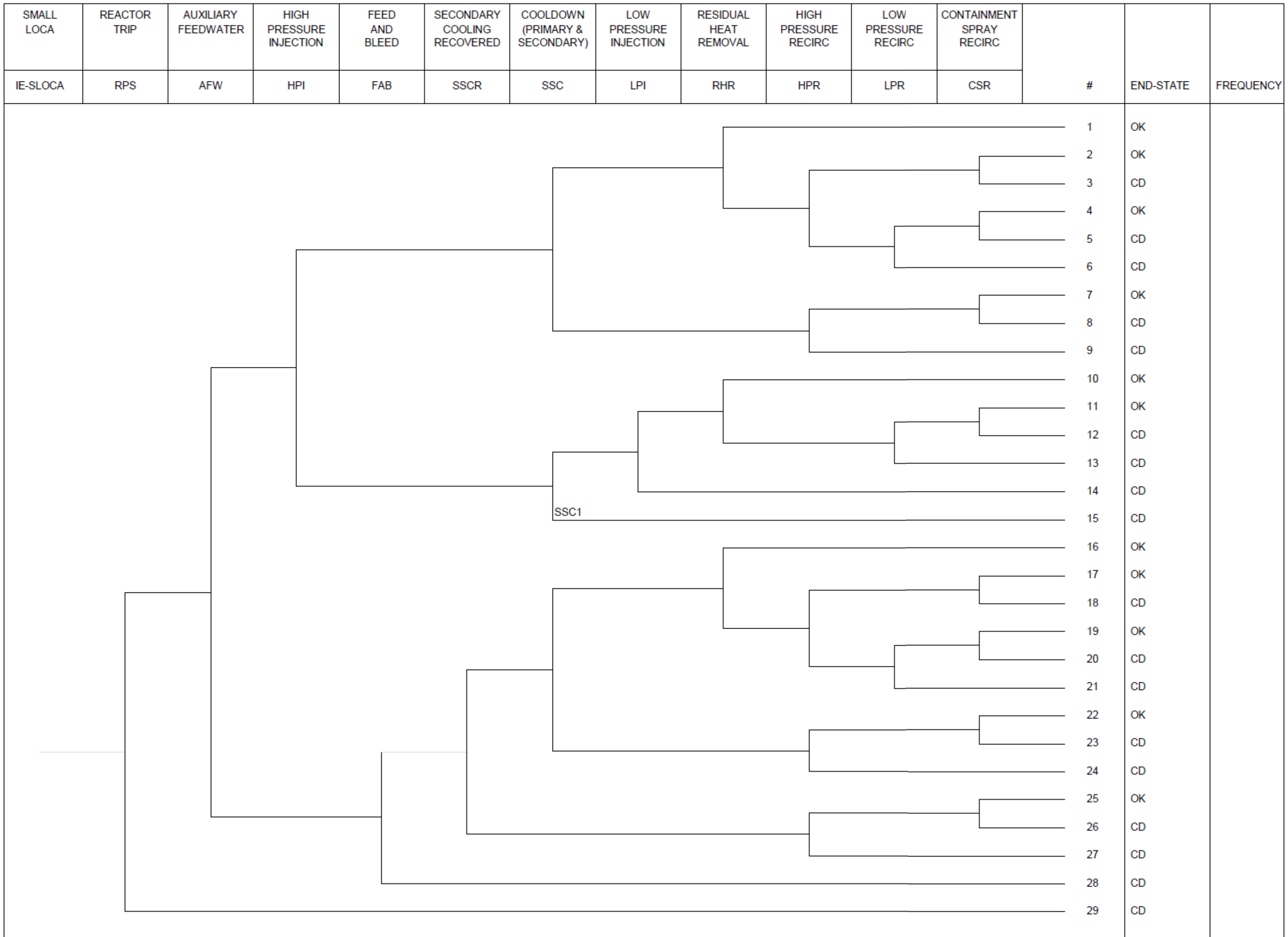
WW Water Level [m]
6
5.8
5.6
5.4
5.2
5
4.8
4.6
4.4
4.2
4
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
time [sec]
Swollen
Collapsed

APPENDIX C

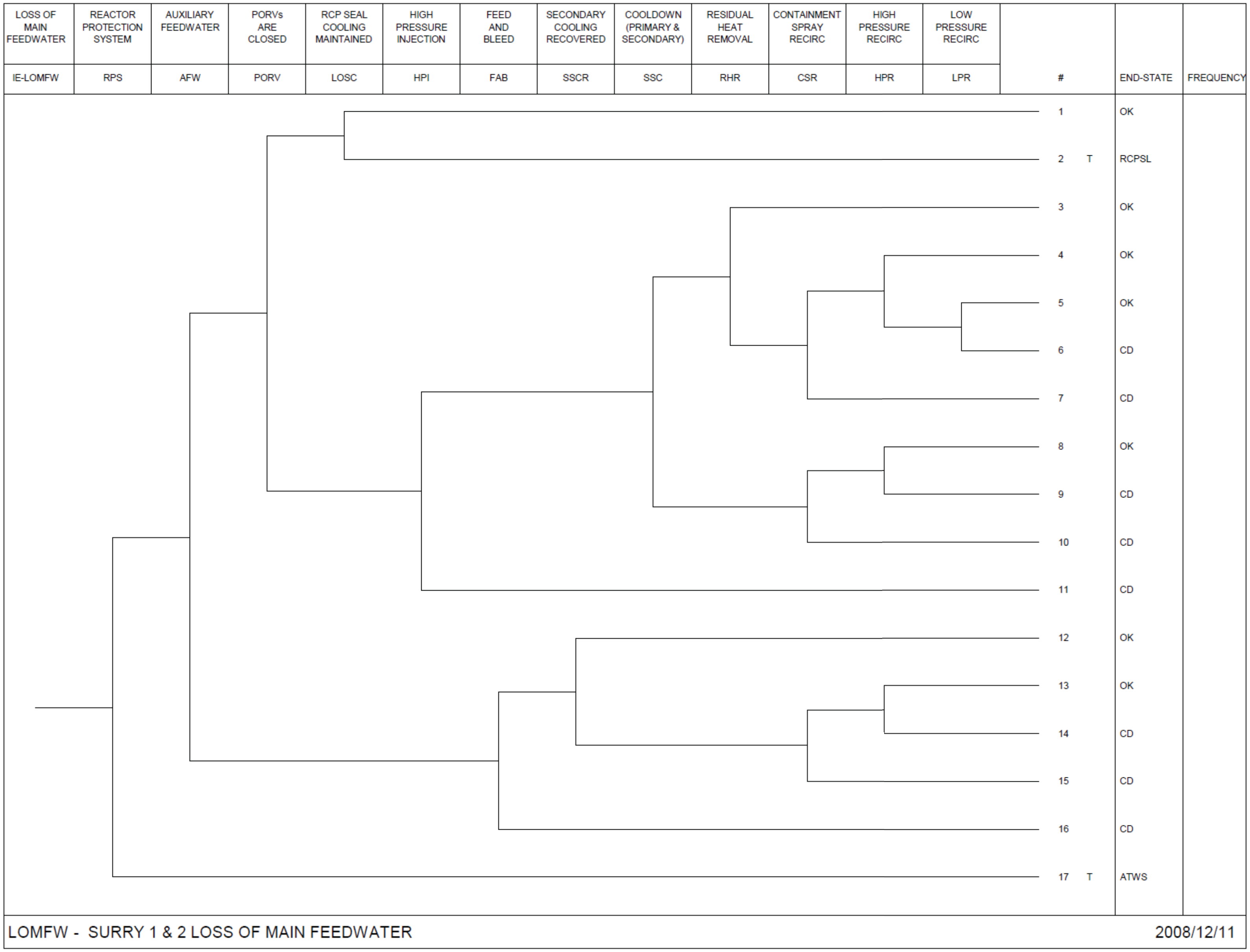
EVENT TREE MODELS FOR SURRY AND PEACH BOTTOM

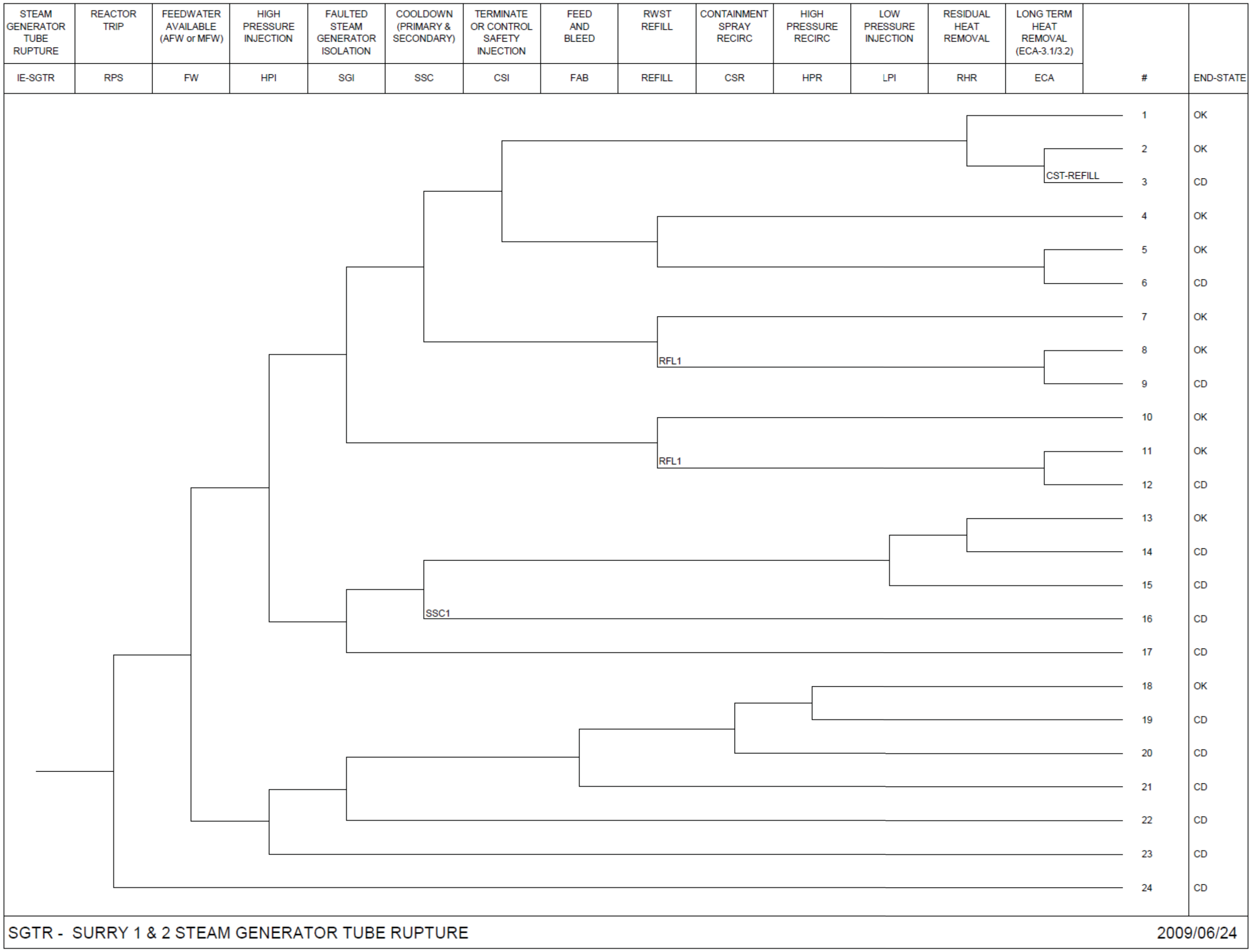
C.1 Surry Event Trees

This section provides the relevant event trees from the Surry (v3.52) Standardized Plant Analysis Risk model dated November 2009. These event trees show the sequences described in the main report.

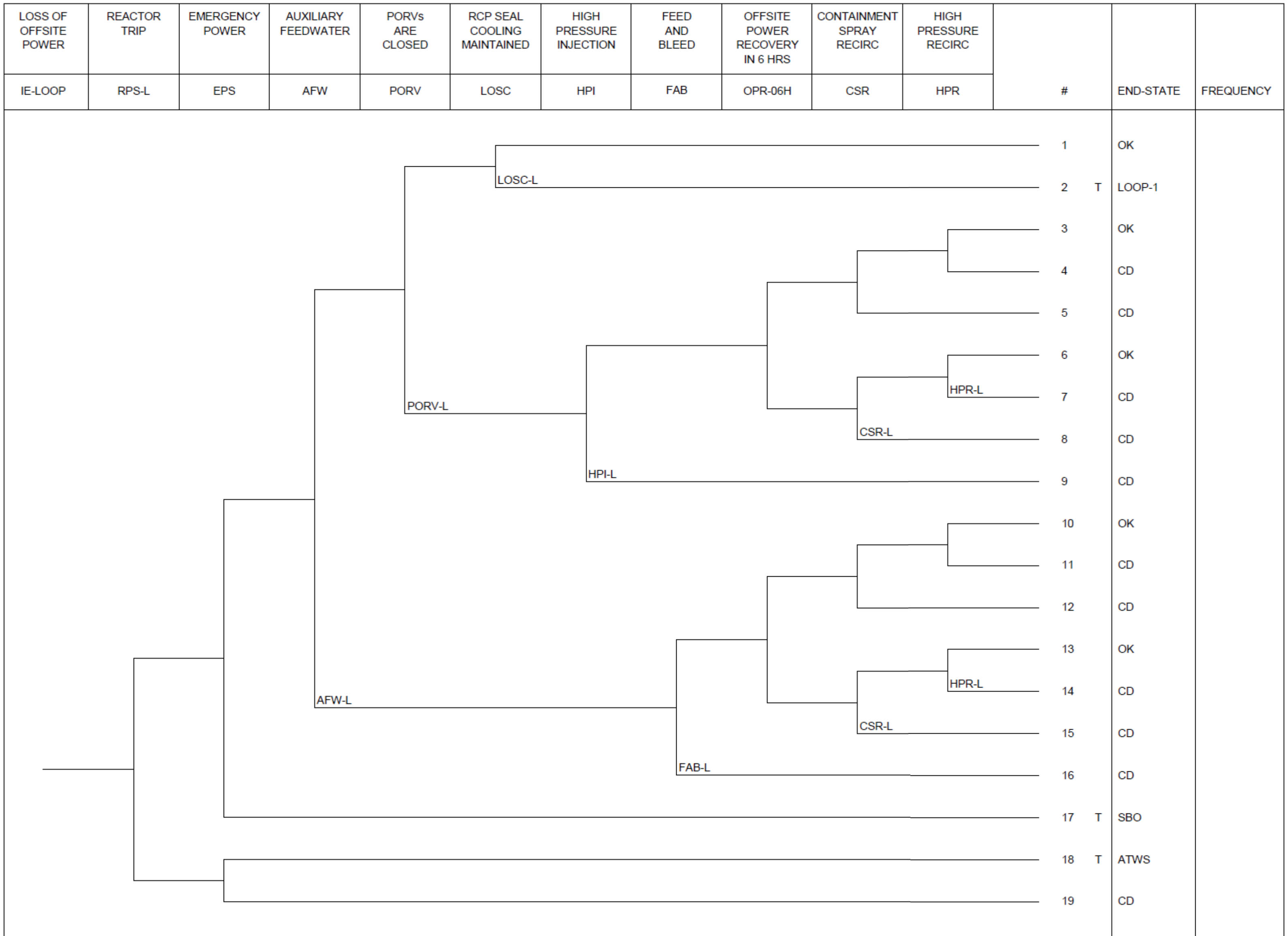


SLOCA - SURRY 1 & 2 SMALL LOSS-OF-COOLANT ACCIDENT 2009/07/09

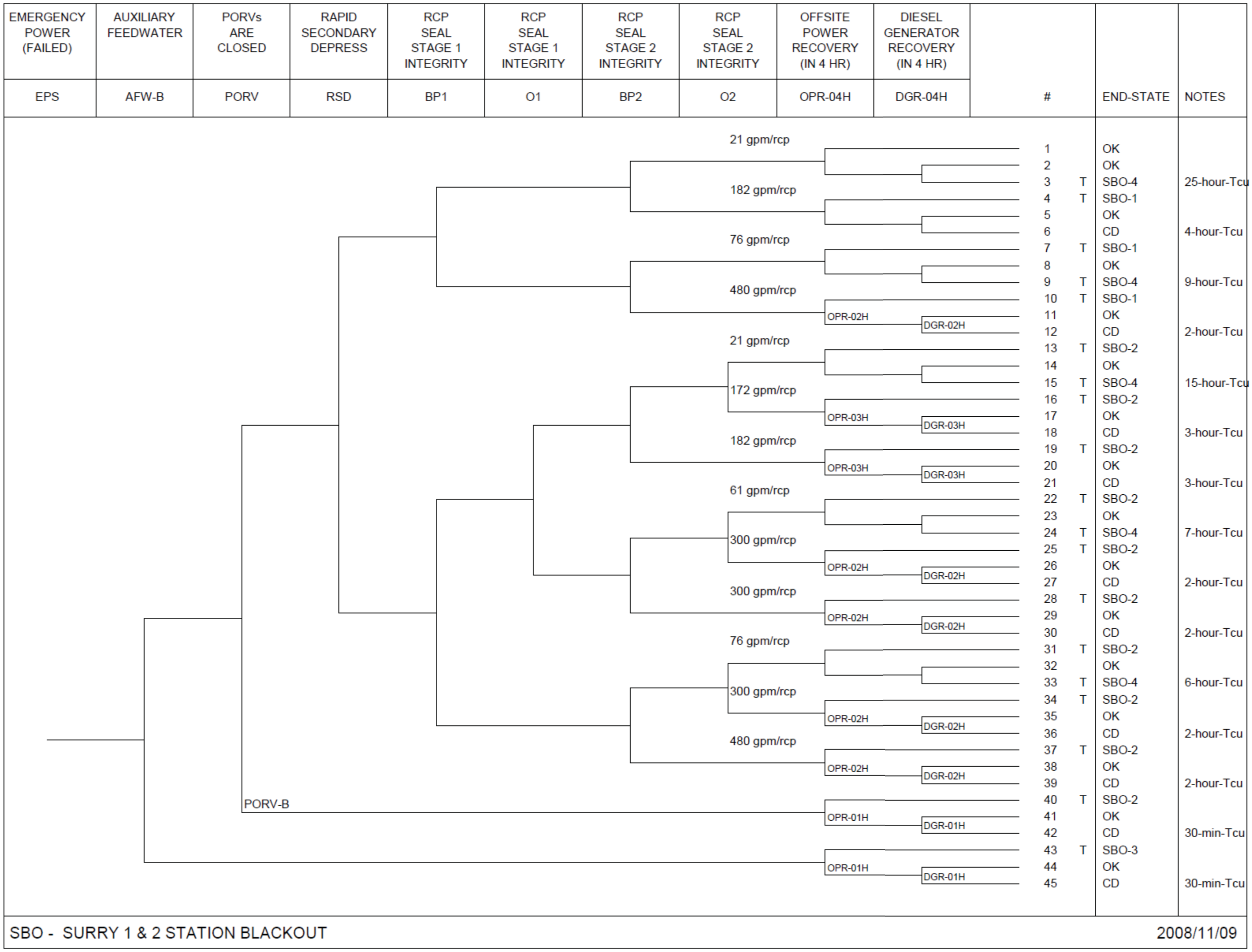

LOSS OF MAIN FEEDWATER
IE-LOMFW
REACTOR PROTECTION SYSTEM
RPS
AUXILIARY FEEDWATER
AFW
PORVs ARE CLOSED
PORV
RCP SEAL COOLING MAINTAINED
LOSC
HIGH PRESSURE INJECTION
HPI
FEED AND BLEED
FAB
SECONDARY COOLING RECOVERED
SSCR
COOLDOWN (PRIMARY & SECONDARY)
SSC
RESIDUAL HEAT REMOVAL
RHR
CONTAINMENT SPRAY RECIRC
CSR
HIGH PRESSURE RECIRC
HPR
LOW PRESSURE RECIRC
LPR
#
END-STATE
FREQUENCY
1 OK
2 T RCPSL
3 OK
4 OK
5 OK
6 CD
7 CD
8 OK
9 CD
10 CD
11 CD
12 OK
13 OK
14 CD
15 CD
16 CD
17 T ATWS
LOMFW - SURRY 1 & 2 LOSS OF MAIN FEEDWATER
2008/12/11

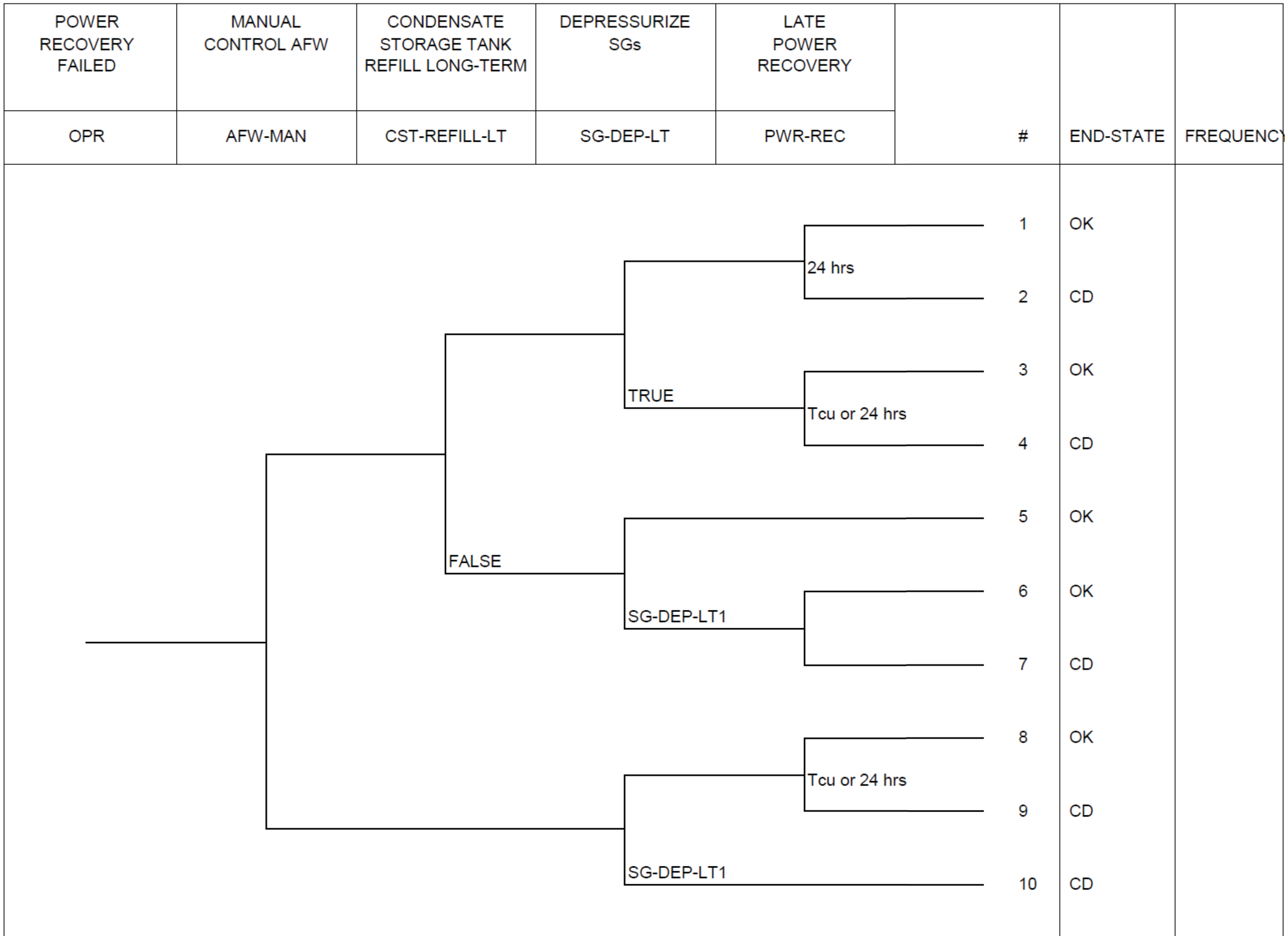


STEAM GENERATOR TUBE RUPTURE
IE-SGTR
REACTOR TRIP
RPS
FEEDWATER AVAILABLE (AFW or MFW)
FW
HIGH PRESSURE INJECTION
HPI
FAULTED STEAM GENERATOR ISOLATION
SGI
COOLDOWN (PRIMARY & SECONDARY)
SSC
TERMINATE OR CONTROL SAFETY INJECTION
CSI
FEED AND BLEED
FAB
RWST REFILL
REFILL
CONTAINMENT SPRAY RECIRC
CSR
HIGH PRESSURE RECIRC
HPR
LOW PRESSURE INJECTION
LPI
RESIDUAL HEAT REMOVAL
RHR
LONG TERM HEAT REMOVAL (ECA-3.1/3.2)
ECA
#
END-STATE
CST-REFILL
RFL1
RFL1
SSC1
1 OK
2 OK
3 CD
4 OK
5 OK
6 CD
7 OK
8 OK
9 CD
10 OK
11 OK
12 CD
13 OK
14 CD
15 CD
16 CD
17 CD
18 OK
19 CD
20 CD
21 CD
22 CD
23 CD
24 CD
SGTR - SURRY 1 & 2 STEAM GENERATOR TUBE RUPTURE
2009/06/24



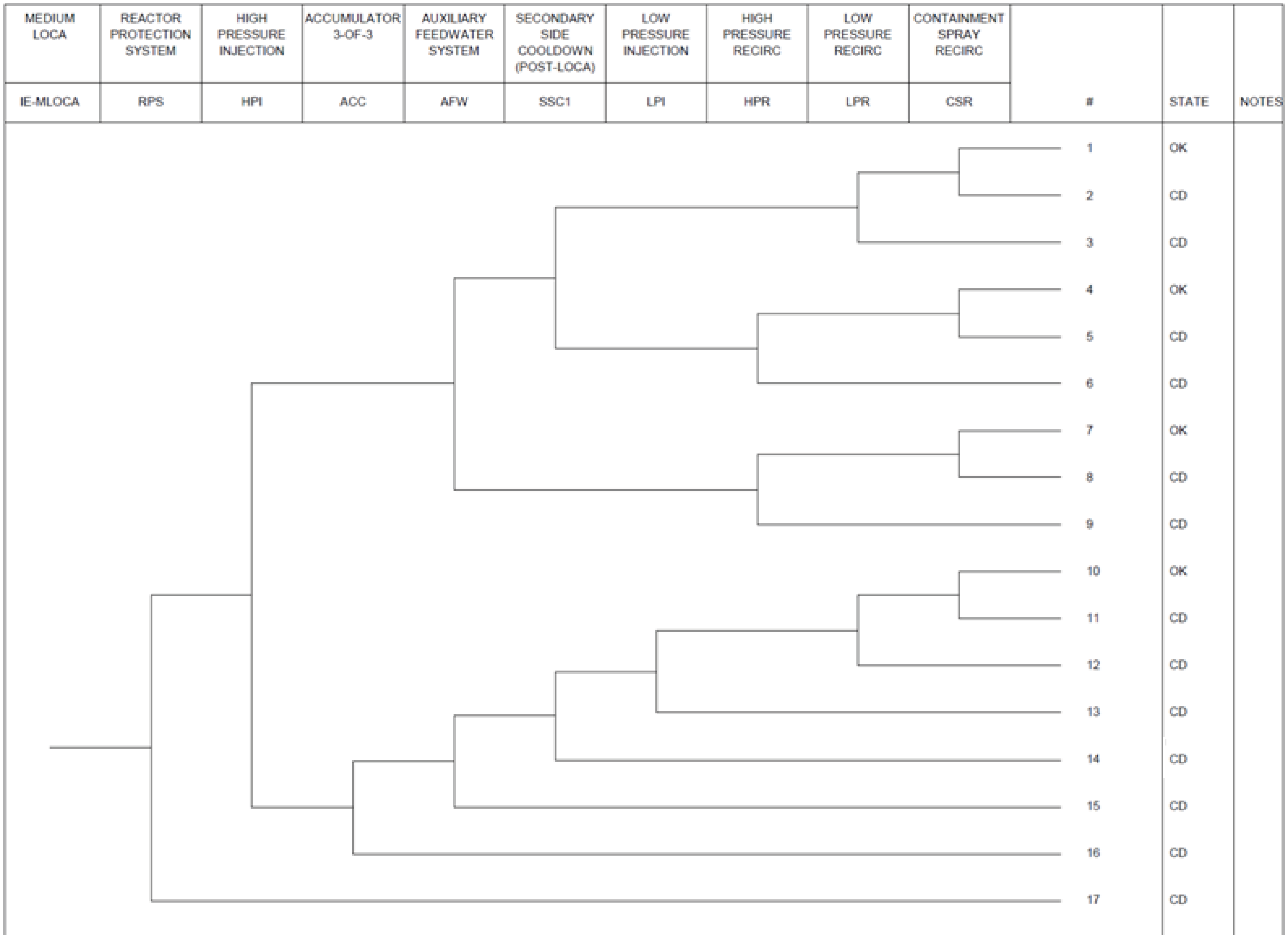
LOOPAV - SURRY 1 & 2 LOSS OF OFFSITE POWER: Average Value 2009/07/01


EMERGENCY POWER (FAILED)
EPS
AUXILIARY FEEDWATER
AFW-B
PORVs ARE CLOSED
PORV
RAPID SECONDARY DEPRESS
RSD
RCP SEAL STAGE 1 INTEGRITY
BP1
RCP SEAL STAGE 1 INTEGRITY
O1
RCP SEAL STAGE 2 INTEGRITY
BP2
RCP SEAL STAGE 2 INTEGRITY
O2
OFFSITE POWER RECOVERY (IN 4 HR)
OPR-04H
DIESEL GENERATOR RECOVERY (IN 4 HR)
DGR-04H
#
END-STATE
NOTES
21 gpm/rcp
182 gpm/rcp
76 gpm/rcp
480 gpm/rcp
21 gpm/rcp
172 gpm/rcp
182 gpm/rcp
61 gpm/rcp
300 gpm/rcp
300 gpm/rcp
76 gpm/rcp
300 gpm/rcp
480 gpm/rcp
PORV-B
OPR-02H
DGR-02H
OPR-03H
DGR-03H
OPR-03H
DGR-03H
OPR-02H
DGR-02H
OPR-02H
DGR-02H
OPR-02H
DGR-02H
OPR-02H
DGR-02H
OPR-01H
DGR-01H
OPR-01H
DGR-01H
1 OK
2 OK
3 T SBO-4 25-hour-Tcu
4 T SBO-1
5 OK
6 CD 4-hour-Tcu
7 T SBO-1
8 OK
9 T SBO-4 9-hour-Tcu
10 T SBO-1
11 OK
12 CD 2-hour-Tcu
13 T SBO-2
14 OK
15 T SBO-4 15-hour-Tcu
16 T SBO-2
17 OK
18 CD 3-hour-Tcu
19 T SBO-2
20 OK
21 CD 3-hour-Tcu
22 T SBO-2
23 OK
24 T SBO-4 7-hour-Tcu
25 T SBO-2
26 OK
27 CD 2-hour-Tcu
28 T SBO-2
29 OK
30 CD 2-hour-Tcu
31 T SBO-2
32 OK
33 T SBO-4 6-hour-Tcu
34 T SBO-2
35 OK
36 CD 2-hour-Tcu
37 T SBO-2
38 OK
39 CD 2-hour-Tcu
40 T SBO-2
41 OK
42 CD 30-min-Tcu
43 T SBO-3
44 OK
45 CD 30-min-Tcu
SBO - SURRY 1 & 2 STATION BLACKOUT
2008/11/09

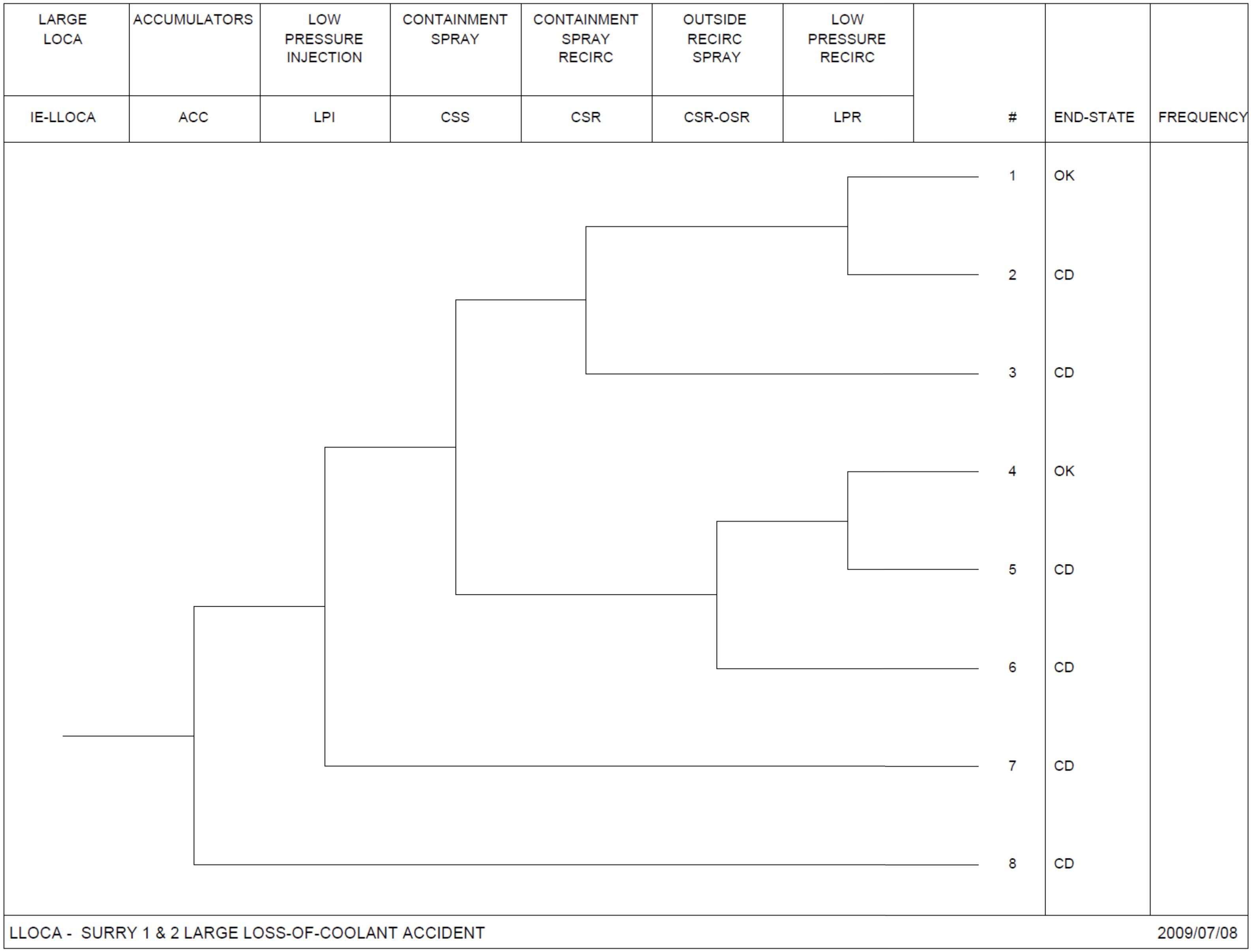


SBO-4 - SURRY 1 & 2 STATION BLACKOUT - CONTINUED

2008/12/19

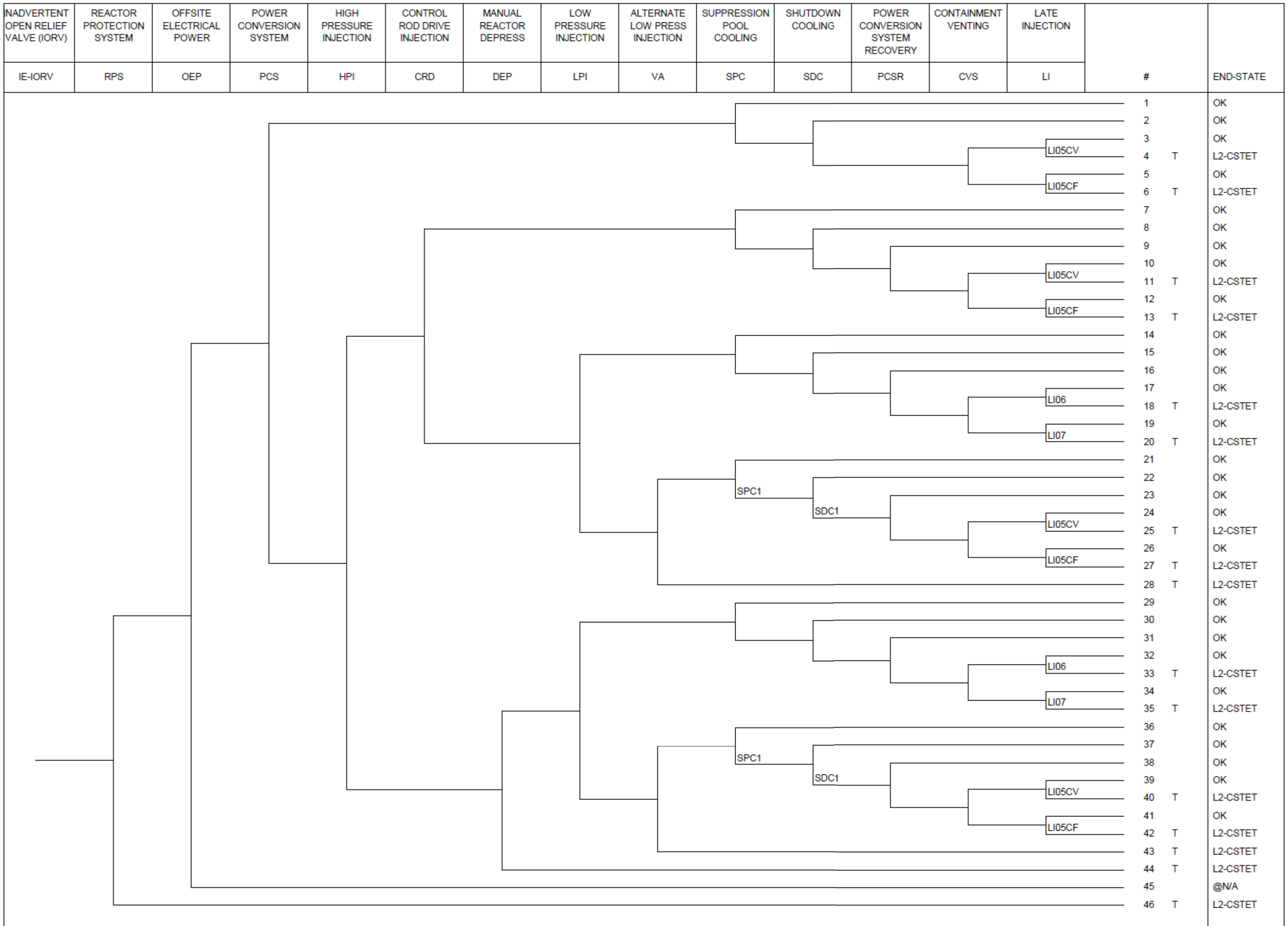


MLOCA - SURRY 1 & 2 MEDIUM LOSS-OF-COOLANT ACCIDENT 2009/07/12

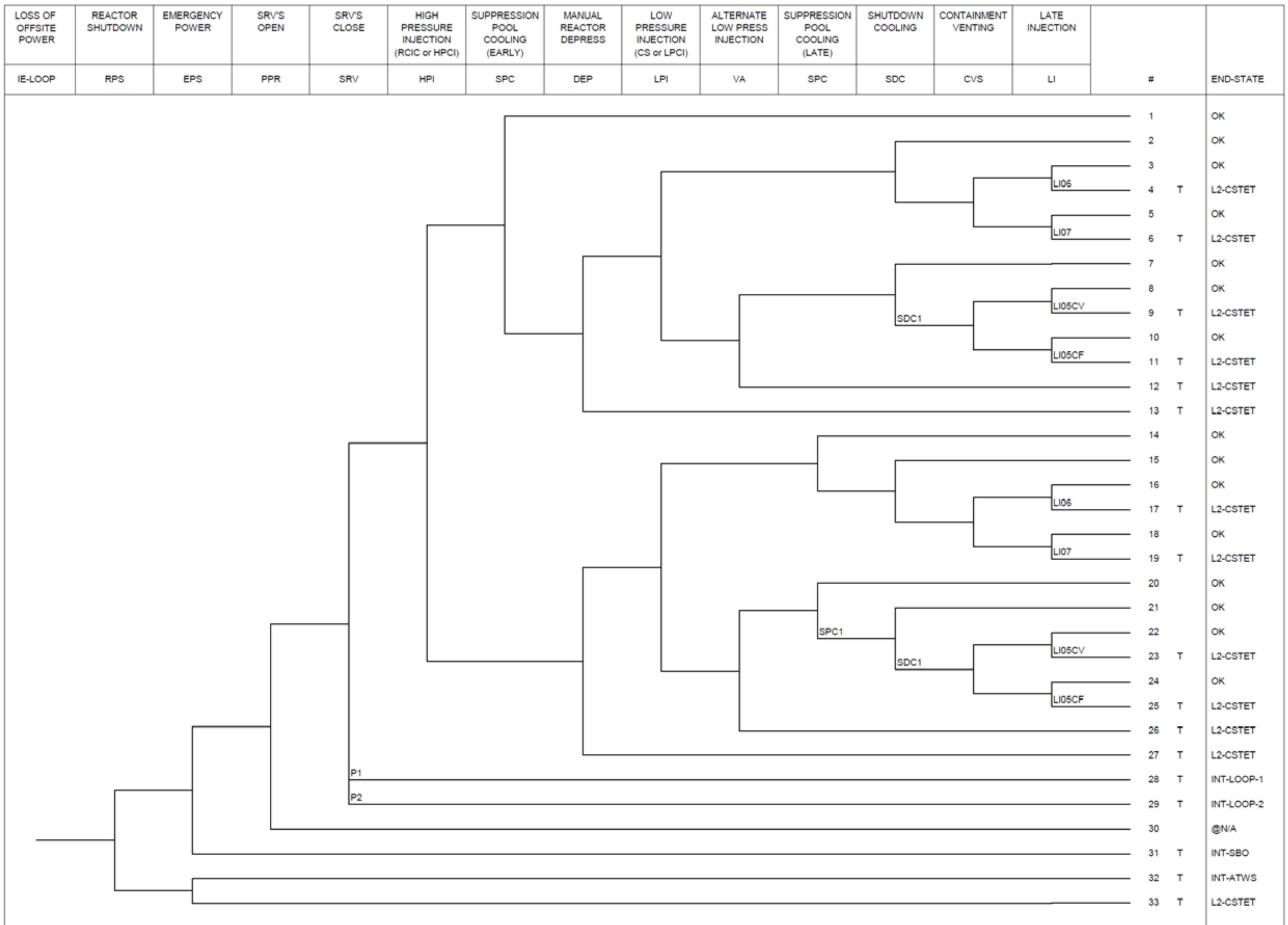

LARGE LOCA
ACCUMULATORS
LOW PRESSURE INJECTION
CONTAINMENT SPRAY
CONTAINMENT SPRAY RECIRC
OUTSIDE RECIRC SPRAY
LOW PRESSURE RECIRC
IE-LLOCA
ACC
LPI
CSS
CSR
CSR-OSR
LPR
#
END-STATE
FREQUENCY
1 OK
2 CD
3 CD
4 OK
5 CD
6 CD
7 CD
8 CD
LLOCA - SURRY 1 & 2 LARGE LOSS-OF-COOLANT ACCIDENT
2009/07/08

C.2 Peach Bottom Event Trees

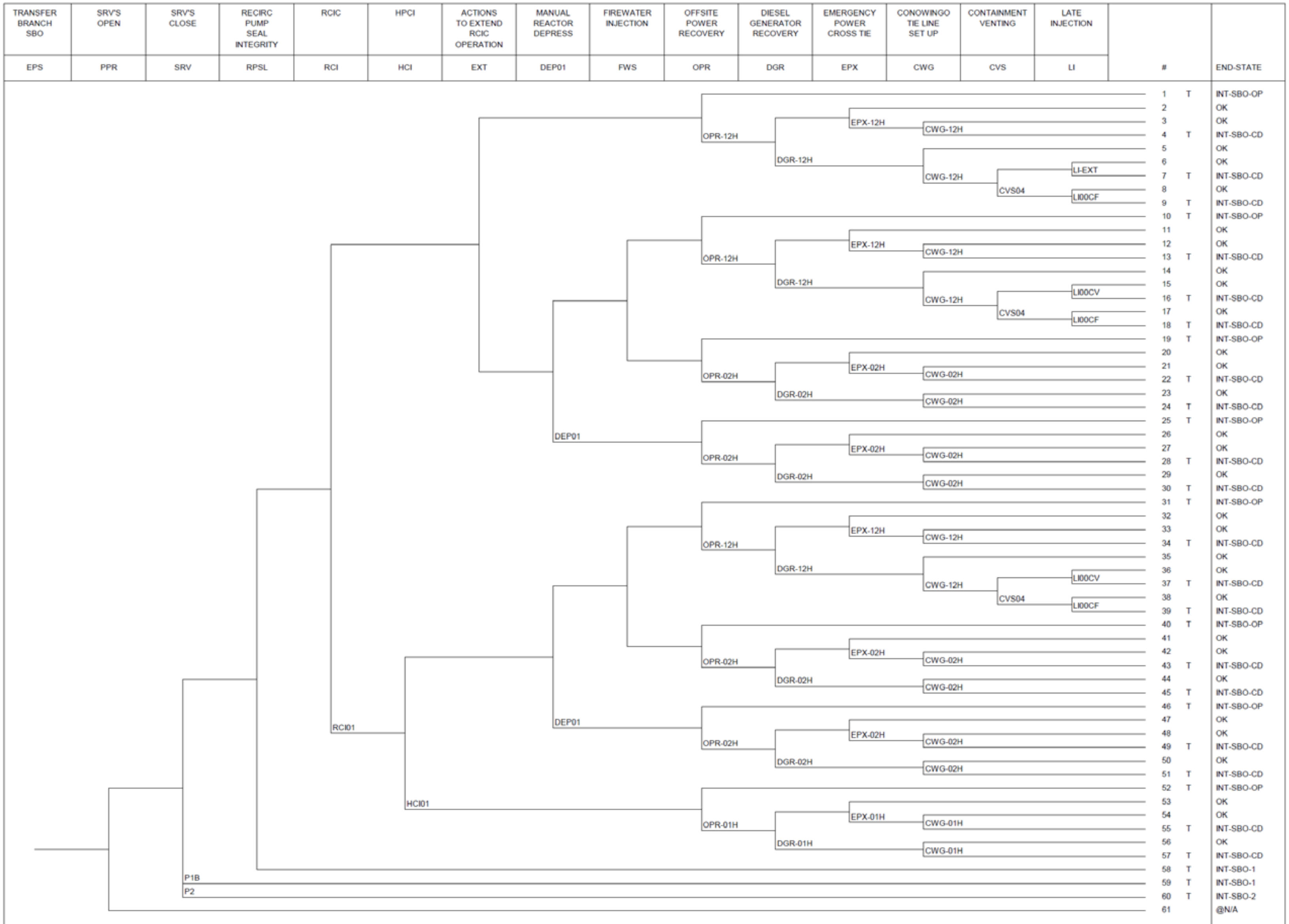
This section provides the relevant event trees from the Peach Bottom (v3.50) Standardized Plant Analysis Risk model dated October 2009. These event trees show the sequences described in the main report.



INT-IORV - PEACH BOTTOM INADVERTENT OPEN RELIEF VALVE (IORV) 2009/07/01

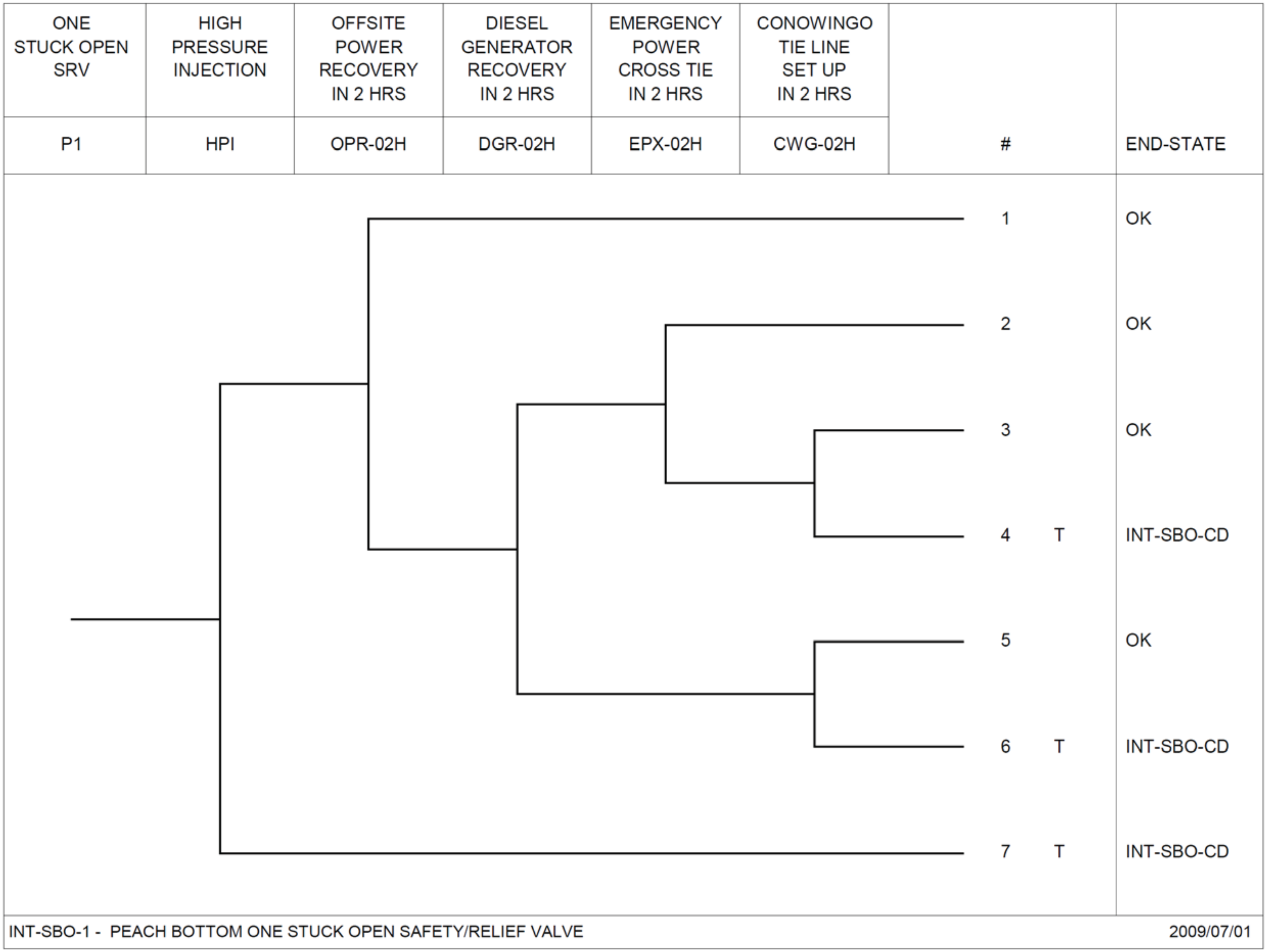


INT-LOOP - PEACH BOTTOM LOSS OF OFFSITE POWER 2009/07/02



INT-SBO - PEACH BOTTOM STATION BLACKOUT

2009/07/02


ONE STUCK OPEN SRV
HIGH PRESSURE INJECTION
OFFSITE POWER RECOVERY IN 2 HRS
DIESEL GENERATOR RECOVERY IN 2 HRS
EMERGENCY POWER CROSS TIE IN 2 HRS
CONOWINGO TIE LINE SET UP IN 2 HRS
P1
HPI
OPR-02H
DGR-02H
EPX-02H
CWG-02H
#
END-STATE
1
OK
2
OK
3
OK
4
T
INT-SBO-CD
5
OK
6
T
INT-SBO-CD
7
T
INT-SBO-CD
INT-SBO-1 - PEACH BOTTOM ONE STUCK OPEN SAFETY/RELIEF VALVE
2009/07/01

APPENDIX D

RESPONSE TO PUBLIC COMMENTS

D.1 Introduction

In November 2010, the U.S. Nuclear Regulatory Commission (NRC) issued a draft of NUREG-1953 for a 30-day public comment period. In December 2010, the NRC extended the comment period to February 28, 2011, in response to an external request. Exelon Nuclear (Exelon, 2010) and the Nuclear Energy Institute (NEI, 2011) submitted comments to the agency. The following sections respond to each comment received, and NUREG-1953 has been modified as necessary to address some concerns. The comments were helpful not only for improving the utility of this report, but also for providing insights for future analyses.

D.2 Comments from Exelon Nuclear Dated December 15, 2010

D.2.1 Plant Representation in MELCOR

Section 5.1 of the draft NUREG states that the core nodalization assumed 10 axial by 5 radial regions. Further clarification on this investigation to the sensitivity of this assumed nodalization scheme would help demonstrate the impact of this assumption.

NRC Response

The nodalization used in the Surry model follows a well-established nodalization convention for the use of MELCOR in reactor applications. Past sensitivity studies have shown this nodalization to reproduce the necessary physics for the types of accidents being considered in this report.

D.2.2 Stuck Open Safety Relief Valves

The State-of-the-Art Reactor Consequence Analyses (SOARCA) project identified a significant sensitivity to Safety Relief Valves (SRVs) sticking open due to elevated gas temperatures. Further clarification on how this impacts the current success criteria analysis would be beneficial.

NRC Response

Regarding the issue of relief valves sticking open due to elevated gas temperatures, the elevated temperatures at the valve necessary to prompt this concern were not seen until the time of core damage. Since the present study only considers the phase of the accident up to the start of core damage, this valve failure mechanism is not believed to be relevant here. Section 5.1 of the main report now addresses this issue.

D.2.3 Reactor Coolant Pump Manual Trip

It appears that a credit was not assumed for operator actions to trip the Reactor Coolant Pumps (RCPs) manually. It is recommended that a sensitivity be included to demonstrate the impact of manual actions to trip the RCPs in accordance with the existing guidance.

NRC Response

The staff agrees with the need to assess this impact. To that end, the staff revisited this issue for each of the initiators considered for Surry and developed the following simplified criteria:

- For small-break loss-of-coolant accidents (SBLOCAs), manual RCP trip is covered in E-0, “Reactor Trip or Safety Injection” and E-1, Loss of Reactor or Secondary Cooling”. For the Surry procedures used in this study, the RCPs will be tripped when reactor coolant system subcooling reaches 30 degrees Fahrenheit (F) (16.7 degrees Celsius (C)) if at least one charging pump is running. The situation for depressurization and cooldown cases would be more complicated. Two cases from the main report were re-run using the above criteria to assess the effect. Table 1 and Table 2 provide the results of these calculations.

- E-0, “Reactor Trip or Safety Injection” and FR-H.1, “Response to Loss of Secondary Heat Sink” cover manual RCP trip for loss-of-all feedwater events. We assume that the RCPs will be tripped between 5 to 15 minutes following reactor trip, based on data from a recent Halden study that investigated crew response to this initiator as presented in (Coyne 2009). For simplicity, 10 minutes is used for the trip criteria. One of the cases from the main report was re-run, and results are provided in Table 3.

- Manual RCP trip for steam generator tube rupture (SGTR) is covered in E-0, “Reactor Trip or Safety Injection” and early in E-3, “Steam Generator Tube Rupture”.[1] We assume that the RCP trip criteria would not be reached during the E-0 and early E-3 execution, and it is not a continuous action in E-3. Based on this, we do not assume that operators trip the RCPs, even as a sensitivity.

- Manual RCP trip for station blackout is not relevant since the RCPs require alternating current (ac) power.

- E-0, “Reactor Trip or Safety Injection” and E-1, “Loss of Reactor or Secondary Coolant” cover manual RCP trip for medium-break loss-of-coolant accidents (MBLOCAs) and large-break loss-of-coolant accidents (LBLOCAs). The RCPs will be tripped when subcooling reaches 30 degrees F (16.7 degrees C) if at least one charging pump is running. However, for these cases, the 10-percent void criteria assumed in the MELCOR model is reached early (latest case is approximately 17 minutes). For the purposes of these calculations, it is judged that additional sensitivities are not required.

As cited above, the following sensitivity studies were completed to demonstrate the impact of manual actions to trip the RCPs in accordance with the existing guidance. The impact on the time to key events, including the time to core damage, was very small.

Table 1 SBLOCA Case 2 with RCP Trip at 30 °F (16.7 °C) Subcooling

Event	Time (hr)	
	2	**2c**
Reactor trip	0.03	0.03
HHSI injection	0.03	0.03
First actuation of containment sprays (containment pressure >1.72 bars)	2.65	2.65
RWST depletion (<13.5%)	4.30	4.30
Spray recirculation	4.30	4.30
Accumulator start to inject	4.52	4.52
RCP trip (30 °F (16.7 °C) subcooled)	-	5.03
RCP trip (10-percent void)	5.76	-
Core uncovery	7.32	7.40
Core damage (max. temp. >2,200 °F)	9.93	10.1

[1] 1.72 bars = 0.172 MPa; 2,200 °F = 1,204 °C.

[1] Note that the Surry E-3 procedure specifically directs operators to disregard RCP trip criteria once depressurization and cooldown has been initiated.

Table 2 SBLOCA Case 8 with RCP Trip at 30 °F (16.7 °C) Subcooling

Event	Time (hr)	
	8	**8a**
Reactor trip	0.01	0.01
HHSI injection	0.01	0.01
First actuation of containment sprays (containment pressure >1.72 bars)	3.23	3.23
RWST depletion (<13.5%)	5.52	5.52
Spray recirculation	5.53	5.53
Accumulator start to inject	5.65	5.65
RCP trip (30 °F (16.7 °C) subcooled)	-	6.35
RCP trip (10% void)	10.3	-
Core uncovery	14.4	14.3
Core damage (max. temp. >2,200 °F)	21.4	21.4

1 1.72 bars = 0.172 MPa; 2,200 °F = 1,204 °C.

Table 3 LOMFW Case 2 with RCP Trip at 10 Minutes

Event	Time (hr)	
	2	**2a**
MFW, MD-AFW, TD-AFW unavailable	0	0
Reactor trip	0.008 (29 sec)	0.008 (29 sec)
MCP trip (10 min)	-	0.17
SG dryout	0.63	0.65
PRT rupture disk open	0.97	0.96
SI signal (containment pressure >1.22 bars)	1.36	1.31
RCP trip (10% void)	1.43	-
First actuation of containment sprays (containment pressure >1.72 bars)	3.24	3.10
RWST depletion (<13.5%)	8.35	8.10
Core uncovery	1.65/9.54[2]	9.26
Core damage (max. temp. >2,200 °F)	11.80	11.47

1 1.22 bars = 0.122 MPa; 1.72 bars = 0.172 MPa; 2,200 °F = 1,204 °C.

2 For Case 2, the core uncovers early in the accident, recovers prior to significant heatup, and later uncovers again (due to the loss of HHSI).

D.2.4 General Comments

The draft NUREG summarizes best estimate analyses for Surry and Peach Bottom success criteria. The NUREG provides adequate details to describe the sequences being investigated and provides a clear summary of the results. In addition, the results are summarized in terms of the proposed changes to the current SPAR model assumptions.

A detailed comparison of these results with those from the Modular Accident Analysis Program (MAPP4) code is currently underway as an Electric Power Research Institute (EPRI) sponsored project. That comparison effort may reveal additional insights, the results of which are expected to be communicated to the NRC when they are completed.

NRC Response
We acknowledge this activity and will consider the results when they are made available.

D.3 Comments from the Nuclear Energy Institute Dated February 23, 2011

D.3.1 Description of Major Plant Characteristics

In the description of the major plant characteristics in Section 4.1, it is suggested that for Surry, it be noted that successful sump recirculation function requires containment heat removal through the recirculation spray system.

NRC Response

A note to this effect has been added in Section 4.1 of the main report.

D.3.2 Plant Representation in MELCOR

In describing the plant representation used for the study, it is stated that the core nodalization assumed 10 axial and 5 radial regions. Clarification of the sensitivity of this nodalization assumption would be helpful in illustrating its impact.

NRC Response

Exelon Nuclear also submitted this comment. Please see the NRC's response in Section D.2.1.

D.3.3 Small-Break Loss-of-Coolant Accident Case Assumptions

It appears that accumulator injection was credited for all SLOCA cases discussed in this report; however, probabilistic risk assessments normally do not credit accumulator injection for SLOCA mitigation. The impact of this should be explored before issuance of the final NUREG.

NRC Response

An examination of the various cases run for SBLOCA determined that the crediting of the accumulators was not expected to affect whether the simulation went to core damage or to a stable end state. However, in some cases the assumption may have affected other aspects of the results. Specifically, in cases 1, 2, 5, 6, and 6a, the accumulators injected after refueling water storage tank (RWST) depletion and, therefore, may have affected the time between RWST depletion and core damage. In cases 2b, 3, 4, and 6b, the accumulators injected before RWST depletion and, therefore, may have had an effect on the time to RWST depletion. Crediting the accumulators is not expected to have made a significant difference in even the intermediate results for cases 2a, 7, and 8. Since no changes were made to the SPAR models based on the SBLOCA results (see Section 7 of the main report), sensitivity calculations were not performed to assess the effect of this assumption. However, cautionary statements now appear in multiple places in the main report to highlight the potential effect the accumulators might have on some of the intermediate results.

D.3.4 Additional Sensitivities to Consider

While the work described in the draft NUREG involved extensive analysis evaluating sensitivities, the industry suggests two other sensitivities to consider. The first is the impact of crediting manual actions to trip the Reactor Coolant Pumps in accordance with existing guidance, as such credit was not assumed in the analysis. The second suggested sensitivity that the industry suggests evaluating is the impact of the Safety Relief Valves at Peach Bottom sticking open due to elevated gas temperatures, as the State-of-the-Art Reactor Consequence Analysis identified this as a significant sensitivity.

NRC Response

Exelon Nuclear also submitted these comments. Please see Sections D.2.2 and D.2.3 for the NRC's response.

D.4 REFERENCES

(Coyne, 2009) Coyne, Kevin A., "A Predictive Model of Nuclear Power Plant Crew Decision-Making and Performance in a Dynamic Simulation Environment," pg. 333–334, Doctoral dissertation, University of Maryland, 2009.

(Exelon, 2010) Exelon Generation Company, LLC, "Comments Concerning Draft NUREG-1953," December 15, 2010. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML103510341)

(NEI, 2011) Nuclear Energy Institute, "Industry Comments on Draft NUREG-1953," February 23, 2011. (ADAMS Accession No. ML110680444)

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	4. FIN OR GRANT NUMBER
5. AUTHOR(S) **H. Esmaili (NRC), D. Helton (NRC), D. Marksberry (NRC), R. Sherry (NRC, retired), P. Appignani (NRC), D. Dube (NRC), M. Tobin (NRC)R. Buell (Idaho National Laboratory), T. Koonce (Idaho National Laboratory), J. Schroeder (Idaho National Laboratory)**	6. TYPE OF REPORT **technical**
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Division of Risk Analysis, Office of Nuclear Regulatory Research, US Nuclear Regulatory Commission, MS CSB4-A7M, Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report describes MELCOR analyses performed to augment the technical basis for supporting or modifying SPAR success criteria. The analyses are intended to be confirmatory in nature, and while suitable for their intended use in supporting the SPAR models they are not intended to be licensing-quality analyses. This report first provides a basis for using a core damage surrogate of 2,200 degrees Fahrenheit (1,204 degrees Celsius) peak cladding temperature, based on MELCOR analyses for representative sequences and a consideration of historical core damage surrogates. Following this discussion are descriptions of the major plant characteristics for the two plants used for this analysis (Surry Power Station and Peach Bottom Atomic Power Station) and the MELCOR models used to represent these plants. Finally, the report presents the results of many MELCOR calculations and compares these results to the corresponding sequences and success criteria in the SPAR models for Surry and Peach Bottom.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
SPAR, Surry, Peach Bottom,MELCOR, success criteria, loss-of-coolant accident, steam generator tube rupture, station blackout, inadvertent open relief valve	unlimited
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	15. NUMBER OF PAGES
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September 2011