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Compendium of Analyses to Investigate Select Level 1 Probabilistic Risk Assessment End-State Definition and Success Criteria Modeling Issues

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ABSTRACT

As part of the routine development, maintenance, and application of the US Nuclear Regulatory Commission's (NRC's) risk tools, assumptions are made with respect to how nuclear power plant accidents will be represented. The NRC conducts research, and performs analyses, to investigate the adequacy of these assumptions. Recently, the Office of Nuclear Regulatory Research has increased its use of the MELCOR computer code for investigating assumptions related to the success criteria, sequence timing, and end-state definition aspects of Level 1 probabilistic risk assessment (PRA). This report augments the existing collection of contemporary Level 1 PRA success criteria analyses and complements other ongoing work related to success criteria analyses for the Byron Station that will be documented in an upcoming NUREG report.

This report focuses on investigations of (a) the effect of modeling assumptions on key figures-of-merit such as the time of depletion of the refueling water storage tank; (b) the relative conservatism or non-conservatism associated with commonly used and potential core damage surrogates; (c) the time required to arrest fuel heatup if mitigation equipment is recovered following a core heatup; and (d) the comparison of MELCOR to the Modular Accident Analysis Program (MAAP) for characterizing the safety margins of selected PRA sequences through an uncertainty quantification process.

FOREWORD

Energy Research, Inc., performed the work documented in this report under sponsorship and direction of the U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research. The work supported ongoing activities related to the NRC's standardized plant analysis risk (SPAR) models, as described below and in the report.

Probabilistic risk assessment (PRA) models for nuclear power plants rely on underlying modeling assumptions known as success criteria and sequence timing assumptions. These criteria and assumptions determine what combination of system and component failures will lead to postulated core damage, as well as the timeframes during which components must operate or operators must take particular actions. This report augments the existing collection of contemporary Level 1 PRA success criteria analyses, and thus improves the defensibility of how core damage surrogate selection (e.g., peak cladding temperature of 1,478 Kelvin (2200 °Fahrenheit)) affects PRA sequence end-state determination, as well as how important modeling assumptions impact accident sequence evolution and associated key figures-of-merit (e.g., the time to initiate an alternate core cooling strategy). As such, this report supports (1) maintaining and enhancing the SPAR models; (2) supporting the NRC's risk analysts when addressing specific issues in the accident sequence precursor (ASP) program and the significance determination process (SDP); and (3) informing other ongoing and planned initiatives.

MELCOR, which is the NRC's Level 1 PRA success criteria and severe accident analysis computer code, was used to perform this work. Relevant assumptions and limitations for each application of the code are described. The results of the work, which are described in greater detail in the abstract and the report, highlight the importance of variations in specific PRA modeling assumptions (e.g., the assumed leakage size for steam generator tube rupture accidents), recommend specific core damage surrogates (e.g., peak cladding temperature of 1,478 Kelvin (2200 °Fahrenheit) for MELCOR success criteria applications at full-power), illustrate timing requirements for restoring injection to arrest fuel heatup, and show good agreement between MELCOR and the Modular Accident Analysis Program Version 4 (MAAP4 – the most widely used tool by the US industry in this arena) for an uncertainty analysis related to loss of all feedwater.

The NRC plans to use the results of this work in its ongoing SPAR development activities, and when appropriate, for particular ASP or SDP applications. The results also will be useful in the agency's efforts to develop a full-scope PRA for Vogtle Units 1 and 2.

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EXECUTIVE SUMMARY

As part of the routine development, maintenance, and application of the US Nuclear Regulatory Commission's (NRC's) risk tools, assumptions are made with respect to how nuclear power plant accidents will be represented in the risk assessment framework. The NRC conducts research, and performs analyses, to investigate the adequacy of these assumptions. Recently, the Office of Nuclear Regulatory Research has increased its use of the MELCOR computer code for investigating assumptions related to the success criteria, sequence timing, and end-state definition aspects of Level 1 probabilistic risk assessment (PRA). Success criteria are used in PRAs for establishing the system or component requirements for satisfying a particular safety function (e.g., to develop system fault tree logic). Sequence timing assumptions are used for establishing time frames for how long a piece of equipment must operate or by what point an operator action must be taken, and are frequently used in human reliability analysis. End-state definition assumptions (e.g., a core damage surrogate of 1478K (2200 °F) peak clad temperature) are used for binning accident sequences into success or failure end-states, such that core damage frequency can be calculated. All of these assumptions are required by a PRA model because it is a Boolean logic representation of plant accident scenarios that captures a multitude (e.g., tens of thousands) of different failure combinations, and therefore cannot reasonably utilize a thermal-hydraulic computer code to simulate each accident sequence.

This report augments the existing collection of contemporary Level 1 PRA success criteria analyses for the purpose of (i) maintaining and enhancing the Standardized Plant Analysis Risk (SPAR) models; (ii) supporting the NRC's risk analysts when addressing specific issues in the Accident Sequence Precursor (ASP) program and the Significance Determination Process (SDP); and (iii) informing other ongoing and planned initiatives. The report also completes some work that was initiated as part of a previous effort, namely the investigation of core damage surrogates described in Section 2 of NUREG-1953, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models – Surry and Peach Bottom," as well as complimenting other ongoing work related to success criteria analyses for the Byron Station that will be documented in an upcoming NUREG report. Figure ES-1 depicts how the current report fits in to this broader set of activities.

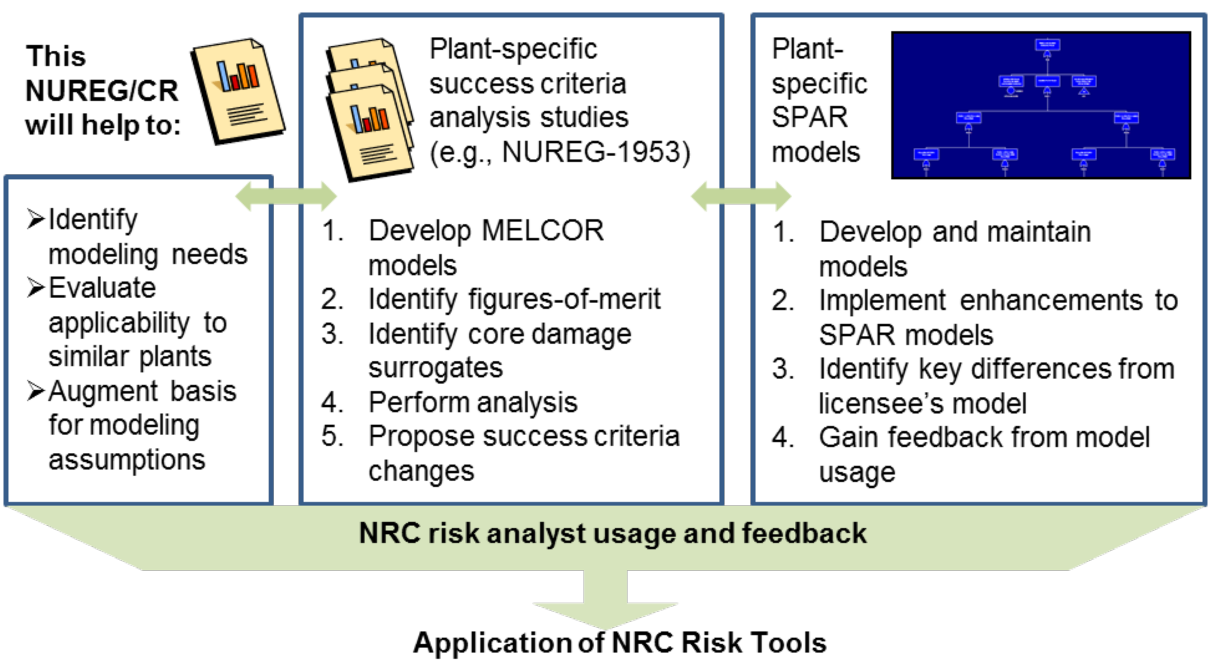

Figure ES-1 Relationship of This Report to Other Activities

This report focuses on investigations of (a) the effect of modeling assumptions on key PRA figures-of-merit such as the time of depletion of the refueling water storage tank; (b) the relative conservatism or non-conservatism associated with commonly used and potential core damage surrogates; (c) the time required to arrest fuel heatup if mitigation equipment is recovered following a core heatup; and (d) the comparison of MELCOR to the Modular Accident Analysis Program (MAAP) for characterizing the safety margins of selected PRA sequences through an uncertainty quantification process.

The report supports the following general conclusions:

- Variations in modeling assumptions (e.g., the effect of reactor coolant system makeup flow prior to a safety injection signal, specification of the hydraulic momentum exchange length) were found to have a modest effect on intermediate figures-of-merit of interest in a Level 1 PRA (e.g., time to steam generator dry out). However, some particular modeling assumptions were found to have a significant effect and should be considered (e.g., by creating an additional initiating event bin) if they are likely to influence a particular end-use. Examples include: (i) break size and location; (ii) number of ruptured steam generator tubes; (iii) reactor power level at the time of trip; (iv) timing of early operator actions; (v) time of battery depletion; (vi) behavior of turbine-driven systems after battery depletion; and (vii) stochastic failure in the open or partially open position of relief valves.

- For selection of core damage surrogates for at-power accidents, a peak cladding temperature in excess of 1,478 K (2,200 °F) is an appropriate choice for a core damage surrogate for MELCOR success criteria applications, because it is not overly-conservative or overly non-conservative. For shutdown conditions, a single metric may not be sufficient for prescribing a realistic surrogate for core damage. Rather, a combination of shutdown core damage surrogates could be used including low reactor pressure vessel water level (one-third of the fuel height); temperature (PCT above 1,478 K); and a Cesium-class release fraction (3 percent released from fuel). In particular, the limited PWR shutdown analyses performed in this study would suggest a one-third active fuel height as a precursor to fuel damage (e.g., gap release) and a PCT above 1,478 K as a precursor to more significant fuel damage. Additional clarifying remarks are provided in the report.

- With regards to the time necessary to arrest fuel heatup (referring to the difference in time between when recovered injection or cooling capabilities resume their function and the time at which the fuel temperature begins to decrease), the present results indicate that a margin of five to ten minutes are generally sufficient for recovery of injection. A much longer time period was required for a case which considered recovery of closed-loop cooling during shutdown. The present analysis did not consider situations where additional recovery actions are happening concurrently (such as automatic depressurization), which might have competing effects on water inventory. Thus, these results should not be applied to those situations. A specific application will also need to consider the additional time required for tasks such as realigning systems and resetting reactor control system permissives.

- For a particular scenario studied (loss of all feedwater with charging unavailable), a MELCOR uncertainty analysis replicating the input assumptions from a recently-published Modular Accident Analysis Program version 4 (MAAP4) uncertainty analysis showed good agreement between the two codes in terms of the fraction of accident simulations predicted to result in core damage.

More details regarding these conclusions are available in associated sections of the report.

ACKNOWLEDGMENTS

The authors wish to acknowledge the contributions of several United States Nuclear Regulatory Commission (NRC) staff including Jeff Mitman, Jeffery Wood, and Selim Sancaktar, who provided input into the selection of boundary conditions for the Byron shutdown analyses; James Corson, who provided recommendations for improving the Byron MELCOR model and some associated analyses; Michelle Flanagan, who consulted on 10 *Code of Federal Regulations* 50.46 aspects; and Laura Kozak, who facilitated a site visit and information gathering for the Byron model.

The authors also wish to acknowledge Kenneth Wagner (dycoda) and Andrew Goldman (Sandia National Laboratories), who performed some of the work on the Surry steam generator tube rupture analysis.

In addition, the authors acknowledge Richard Sherry and Jeff Gabor of ERIN Engineering, who answered questions about an Electric Power Research Institute (EPRI)-sponsored study that is referenced in this report.

Finally, the authors wish to acknowledge Exelon for providing detailed design and operational information about the Byron station, and for supporting a site visit and telephone calls to ensure that boundary conditions were well-informed.

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ACRONYMS

Acronym	Definition
AC	Alternating Current
ADS	Automatic Depressurization System
AFW	Auxiliary Feedwater
B&F	Bleed & Feed
BWR	Boiling-Water Reactor
CCDF	Complementary Cumulative Distribution Function
CCW	Component Cooling Water
CDF	Cumulative Distribution Function
CRD	Control Rod Drive
CST	Condensate Storage Tank
DC	Direct Current
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
FSAR	Final Safety Analysis Report
FW	Feedwater
HHSI	High-Head Safety Injection
HPCI	High-Pressure Coolant Injection
HPI	High-Pressure Injection
LER	Licensee Event Report
LHSI	Low-Head Safety Injection
LHS	Latin Hypercube Sampling
LOCA	Loss-of-Coolant Accident
LoDC	Loss of DC
LoMFW	Loss of Main Feedwater
LPCS	Low-Pressure Core Spray
MAAP	Modular Accident Analysis Program
MPa	megaPascal

Acronym	**Definition**
MPa-g	megaPascal-gauge
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
MSTV	Main Steam Turbine Valve
NPSH	Net Positive Suction Head
NR	Narrow Range
NRC	Nuclear Regulatory Commission
PCC	Partial Correlation Coefficient
PCT	Peak Cladding Temperature
PDF	Probability Distribution Function
PORV	Pilot Operated Relief Valve
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
psig	pounds per square inch gauge
PWR	Pressurized Water Reactor
RCFC	Reactor Containment Fan Cooler
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RWST	Reactor Water Storage Tank
S	Byron terminology for an SI signal
SBO	Station Blackout
SC	Success Criteria
SG	Steam Generator
SGT	Steam Generator Tube
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SNL	Sandia National Laboratories

Acronym	**Definition**
SOARCA	State-of-the-Art Reactor Consequence Analyses
SPAR	Standardized Plant Analysis Risk
SRV	Safety Relief Valve
TAF	Top of Active Fuel
TBV	Turbine Bypass Valve
TCD	Temperature of Core Damage
VCT	Volume Control Tank
WR	Wide Range

1. INTRODUCTION

To date, the success criteria in the United States Nuclear Regulatory Commission's (NRC's) Standardized Plant Analysis Risk (SPAR) models have, for the most part, been extracted from those used in the associated licensee Probabilistic Risk Assessment (PRA) models. The licensees' PRAs 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 were observed to have different success criteria for specific accident scenarios. This issue has been recognized for some time, but the infrastructure was not in place until recently at the NRC to support the refinement of these criteria [1].

The MELCOR computer code has been under development at Sandia National Laboratories (SNL) with NRC support. The code is designed for application to severe accident analyses and PRA studies. NRC recently initiated an activity applying MELCOR to level-1 PRA success criteria analyses, including the evaluation of several options for defining core damage in level-1 PRA studies and for conducting success criteria analyses for (1) a 3-loop Westinghouse pressurized-water reactor (PWR) with a sub-atmospheric containment [1]; (2) a boiling-water reactor (BWR) with a Mark I containment [1]; and (3) a 4-loop Westinghouse PWR with a large, dry containment (NUREG forthcoming).

This report augments the existing collection of contemporary Level 1 PRA success criteria analyses for the purpose of (i) maintaining and enhancing the SPAR models developed by the NRC; (ii) supporting the NRC's risk analysts when addressing specific issues in the Accident Sequence Precursor (ASP) program and the Significance Determination Process (SDP); and (iii) informing other ongoing and planned initiatives. The report also completes some work that was initiated as part of a previous effort, namely the investigation of core damage surrogates described in Section 2 of NUREG-1953 [1], as well as complimenting other ongoing work related to success criteria analyses for the Byron Station that will be documented in an upcoming NUREG report.

Section 2 of this report provides a description of the three plants used in subsequent MELCOR analysis, salient plant design information, and analytical formulations of interest. This sets the stage for the subsequent sections which focus on investigations of (a) the effect of modeling assumptions on key figures-of-merit such as the time of depletion of the refueling water storage tank (Section 3); (b) the relative conservatism or non-conservatism associated with commonly used and potential core damage surrogates (Section 4); (c) the time required to arrest fuel heatup if mitigation equipment is recovered following a core heatup (Section 5); and (d) the comparison of MELCOR to the Modular Accident Analysis Program (MAAP) for characterizing the safety margins of selected PRA sequences through an uncertainty quantification process (Section 6).

A PWR represented by the Byron Unit 1 plant was selected to illustrate the analysis of an accident initiated by a partial loss of onsite power. Sections 3.1.1 and 4.1.1 (and Appendix A) of this report specifically analyze the Byron sequence referred to as the Loss of DC Bus 111 with Bleed and Feed (B&F). The Surry plant was selected to illustrate the analysis of an accident initiated by a steam generator tube rupture (SGTR). Sections 3.2 and 4.2 (and Appendix B) of this report analyze the sequence referred to as SGTR-9. For a case involving a BWR, the Peach Bottom plant is considered. Sections 3.3 and 4.3 (and Appendix C) of this report specifically focus on an analysis of the sequence referred to as a station blackout (SBO) with reactor core isolation cooling (RCIC) until battery depletion. An analysis of a loss of coolant accident for the Byron plant arising during shutdown conditions is described in Sections 3.1.2 and 4.1.2 (and the latter part of Appendix A). In addition, to compare simulations that used the MELCOR versus the MAAP computer code [2], the Byron loss of main feedwater (FW) accident scenario was reanalyzed using a stratified Monte Carlo approach as documented in Section 5.

Applicability of the analysis in this report to new large light water reactors (LWRs) and advanced integral pressurized water reactors (iPWRs) is limited based on consideration of specific design differences that would affect the results being used. Since some of these new designs have significant differences relative to the plants studied in this report, accidents and timings may be significantly different. The relative timing between reaching core damage surrogates and reaching different figures-of-merit would also be susceptible to differences in plant design and operation. The order of reaching core damage surrogates and different figures-of-merit would be the most transferrable to new plant designs, but could be affected by differences in system capacities and core design. The analysis in this report is not expected to have any applicability to significantly different designs beyond the new large LWRs and iPWRs such as the advanced non-light water reactors.

The analyses contained herein are intended to be confirmatory in nature, and while suitable for their intended use in supporting the NRC's risk activities they are not intended to be used by licensees for risk-informed licensing submittals.

2. DESCRIPTIONS OF PLANTS AND MODELS

2.1 Byron Unit 1 Plant and MELCOR Model

Energy Research Inc. prepared the Byron MELCOR model input deck and maintains supporting documentation for this MELCOR deck, including an account of the various model/deck revisions. The model represents Unit 1; Section 2.4 of this report documents some of the differences that exist between Byron Unit 1 and Byron Unit 2. Most of the plant-specific data for the model are derived from the Byron/Braidwood Final Safety Analysis Report (FSAR), although additional data is drawn from many other sources (e.g., plant models for similar plants). High-level design parameters for the plant are listed in Table 2.1.1.

Table 2.1.1 Byron Plant Data

Characteristic	**Value (as modeled)**
Design Type	Four-loop Westinghouse PWR
Containment Type	Large Dry
Power Level	3,586.6 MWt
Number of Centrifugal Charging Pumps (also used for high-head safety injection)	Two
Number of Safety Injection (SI) Pumps	Two
Number of SI Trains	Two
Pressure Corresponding to Charging / SI Dead-Head	17.9 MPa (2,600 psia) / 10.7 MPa (1,550 psia)
Lowest Pressurizer PORV Opening/Closing Setpoint	16.2 MPa (2,350 psia) / 16.067 MPa (2,330 psia)
Number of Cold-Leg Accumulators	One per loop (four total)
Nominal Operating Pressure	15.5 MPa (2,250 psia)
RWST Volume Technical Specification	1,495 cubic meters (395,000 gal)

Most calculations described in this report were made with Revision 5 of the input deck.[1] All of the calculations used the “YV-3272” release of the MELCOR/MELGEN version 1.8.6, except for one sensitivity calculation pertaining to the “Loss of DC Bus 111 with a B&F” scenario that used MELCOR version 2.1.

Figures 2.1.1 through 2.1.3 show the MELCOR model nodalizations. For the purposes of the success criteria calculations, the containment is adequately represented by one node, because this nodalization scheme adequately predicts bulk containment pressure and temperature response. Were the deck to be used for Level 2 PRA purposes, this nodalization would need to be refined. Regarding Figure 2.1.3, the control volumes in the model are designated by a 3-digit number where the first digit (3, 4, 5, or 6) corresponds to loops D, A, B, and C, respectively, in the plant nomenclature.[2]

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[1] The power level used in this report is the power level before the 1.63% power uprate currently undergoing NRC review. The calculations described in this report that pertain to the “Loss of DC Bus 111 with B&F” scenario were made with an enhanced version of the Revision 5 input deck that included the following two changes: (i) correction of the setpoint for scram on a low RCS pressure and (ii) adjustment of the water volumes delivered from the RWST at levels Lo-2 and Lo-3 to be more realistic. Note that the first of these changes is irrelevant to the LoDC Bus 111 B&F scenario, and the second affects only one of the sensitivity cases (Case A.1.2). The enhanced Revision 5 input deck was the starting point for the development of the Shutdown input deck used in the Section 4.1.2 calculations.

[2] This lettering versus numbering scheme is that of Revisions 7 and later (Revision 5 does not introduce any lettering scheme). The various revisions of the input deck differ on their treatment of differences among the four loops. Of these loop-to-loop differences, only the existence of the pressurizer in one loop is expected to have an effect in most scenarios. The enhanced version of Revision 5 introduced a difference by assuming that each ECCS pump (there are two of each kind) injects into pairs of legs. Apart from that (and the pressurizer), the loops were treated identically in

The Byron "Shutdown" model applies to a phase of shutdown operations, when the reactor coolant system (RCS) is closed and 24 hours have elapsed since reactor shutdown. The following bullets summarize the main differences between the Shutdown model and the main model:

- The decay heat is calculated at a shifted time t' = t + 24 hours, where t is the simulation time. (During the pre-transient part of the calculation [i.e., during -5000 s < t < 0 s], the decay heat is constant at 21.29 MW, which is the decay heat generation rate at 24 hours.)

- The closed-loop operation of one residual heat removal (RHR) pump provides the decay heat removal by taking suction from the pressurizer-loop (Loop D) hot leg. The outflow passes through an RHR heat exchanger and is then injected into the cold legs of Loops A and D. The RHR heat exchanger is modeled with a shell-side component cooling water temperature of 314 K (105 °F) and an efficiency of 0.609. The flow rate is controlled by feedback logic to maintain a constant coolant temperature at the core exit.

- Initial water temperatures throughout the RCS (primary and secondary sides) are assumed to be 466.5 K (380 °F). Initial pressures are the corresponding saturation pressure (1.35 megaPascal [MPa]). (A sensitivity study is performed later to demonstrate the effect of subcooling.)

- There is a provision for a break outside of the containment in the RHR lines that feed the cold legs of Loops B and C; the initial flow rate of this flow area is about 1,000 gpm (0.063 m^3/s).

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that revision. Revisions 7 and later credit equal injection of each ECCS pump into all legs, but introduce loop-to-loop differences in the secondary sides (e.g., volumes of main steam lines).

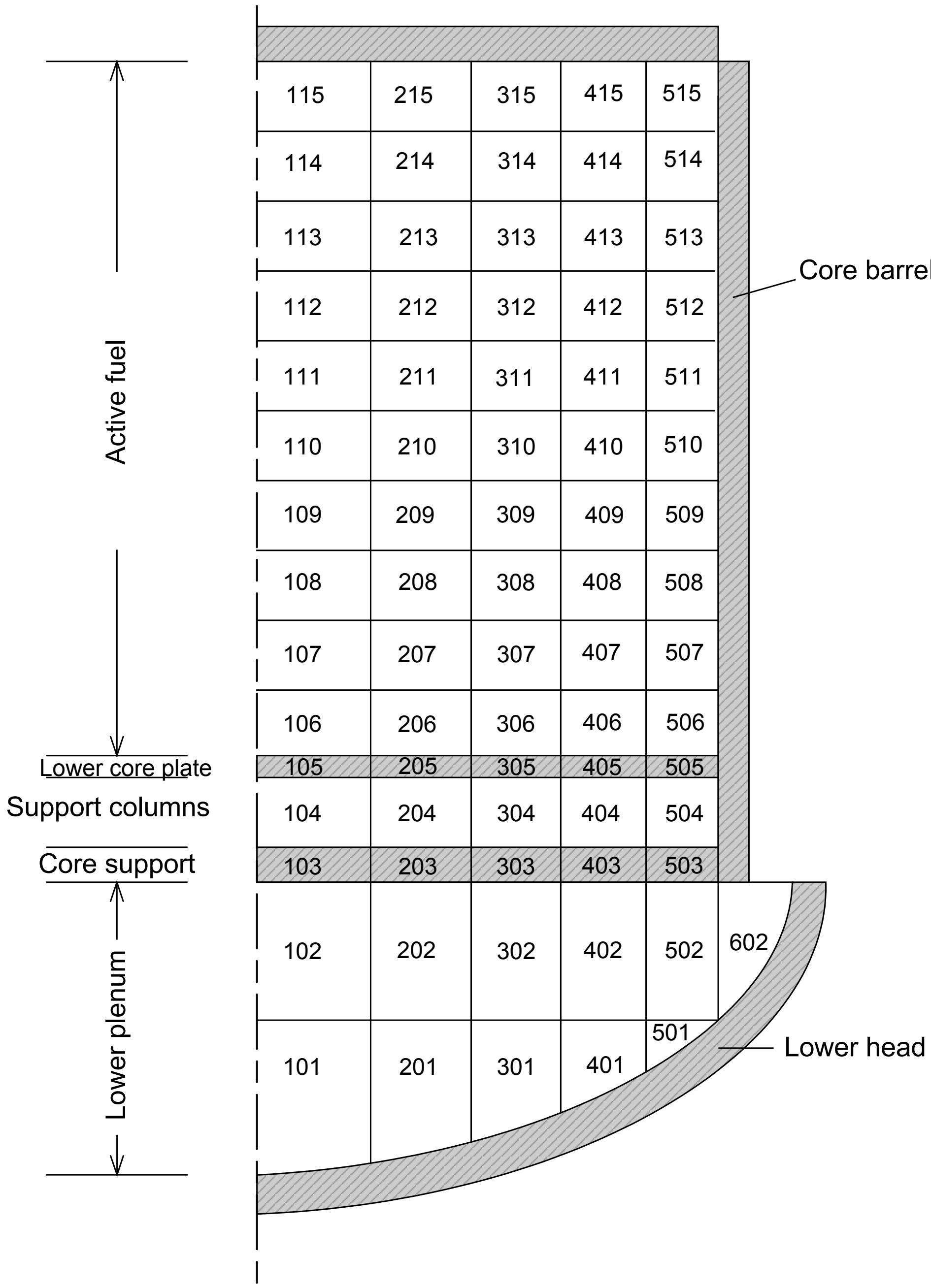


Figure 2.1.1 Reactor Core and Lower Plenum Nodalization

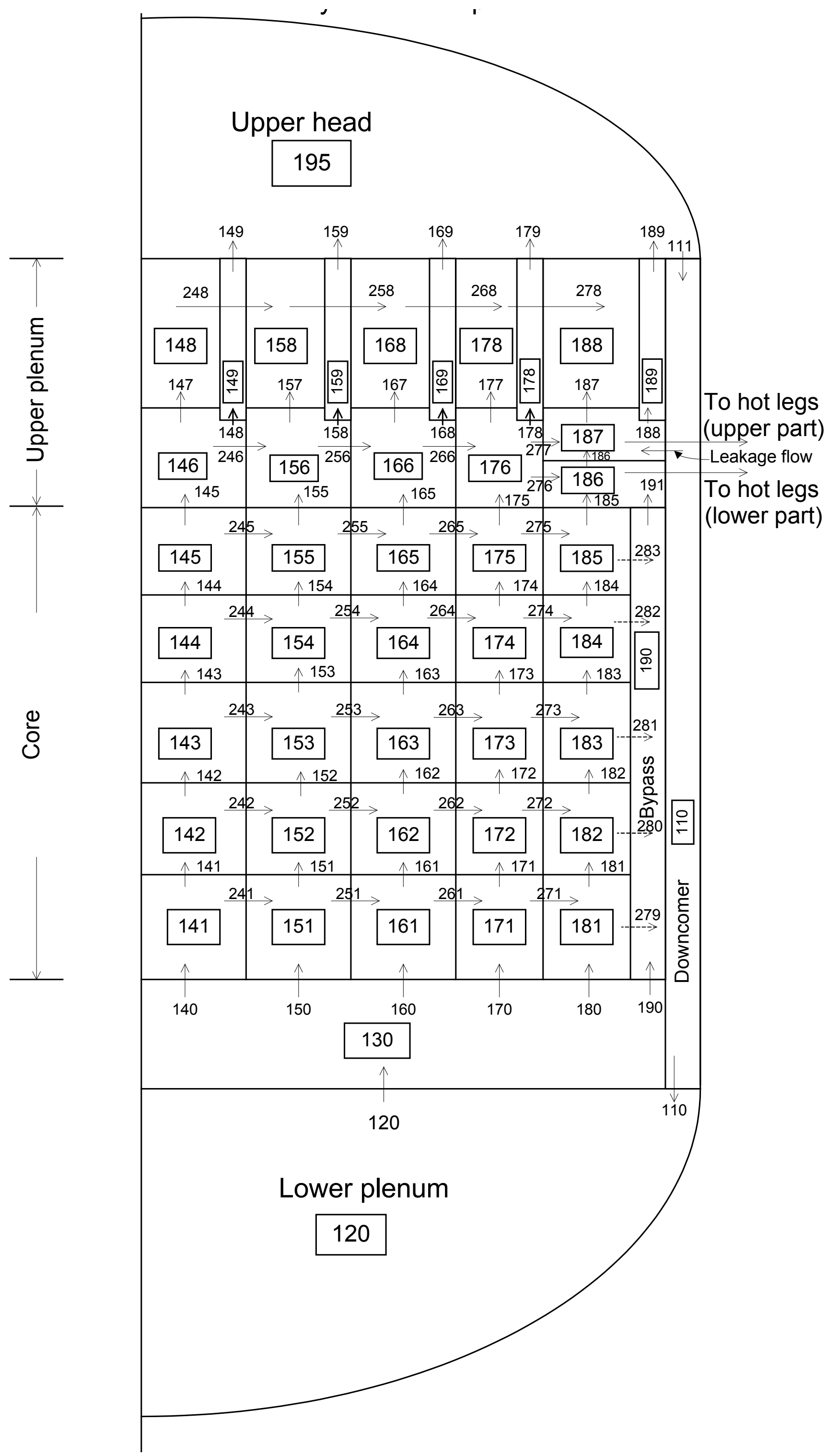
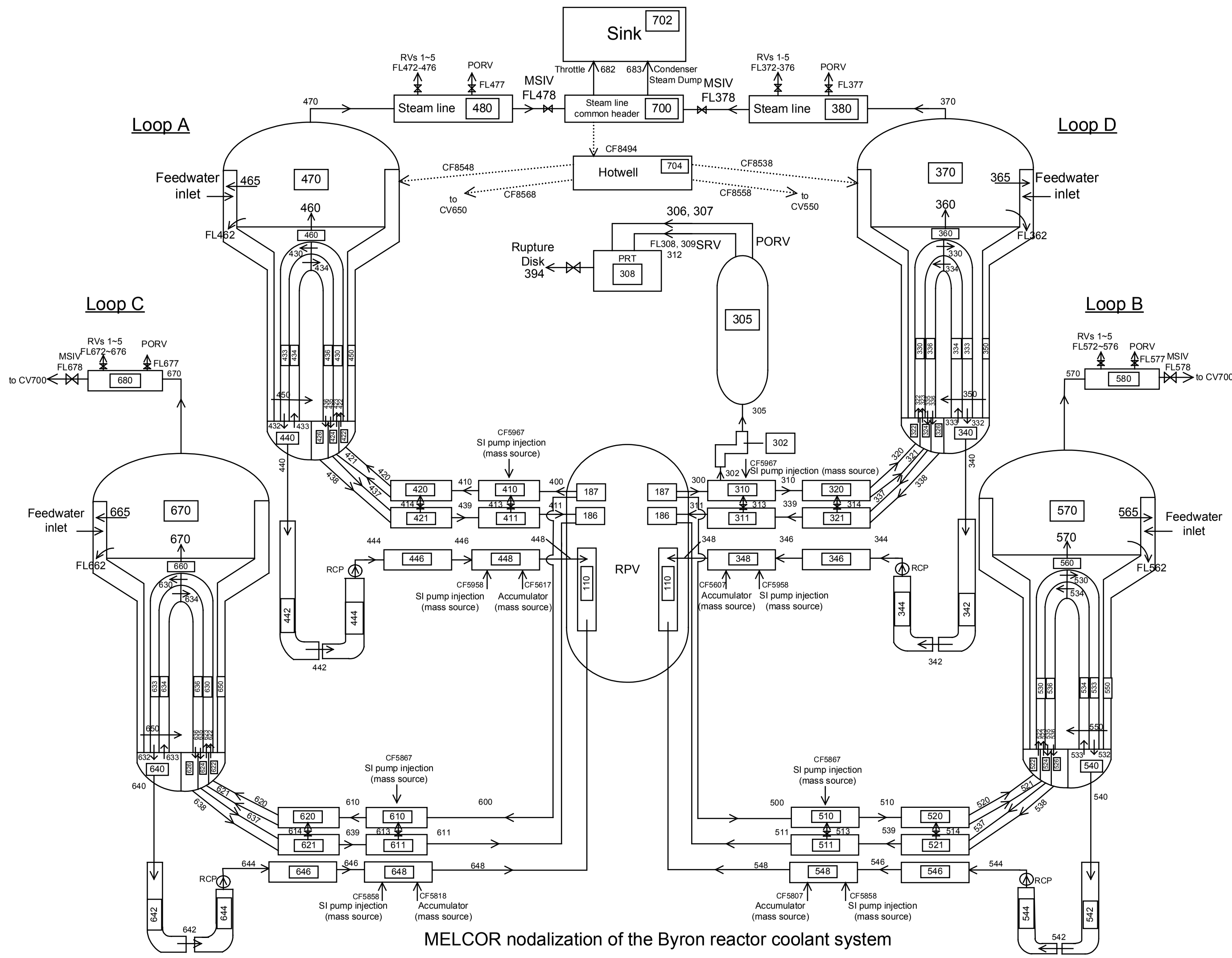


Figure 2.1.2 Reactor Pressure Vessel Control Volume and Flow Path Nodalization



MELCOR nodalization of the Byron reactor coolant system

Figure 2.1.3 Reactor Coolant System Nodalization

2.2 Surry Plant and MELCOR Model

The Surry MELCOR model input deck was prepared at Sandia National Laboratories, under NRC sponsorship, as part of the State-of-the-Art Reactor Consequence Analyses project [3]. High-level design parameters for the plant are listed in Table 2.1.1.

Table 2.2.1 Surry Plant Data

Characteristic	Value (as modeled)
Design Type	Three-loop Westinghouse PWR
Containment Type	Sub-atmospheric
Power Level	2,546 MWt[1]
Number of Centrifugal Charging Pumps (also used for high-head safety injection)	Three
Number of HHSI Trains	Two
Pressure Corresponding to HHSI Dead-Head	17.65 MPa (2,560 psia)[2]
Lowest Pressurizer PORV Opening/Closing Setpoint	16.2 MPa (2,350 psia) / 15.6 MPa (2,260 psia)
Number of Cold-Leg Accumulators	One per loop (three total)
Nominal Operating Pressure	15.4 MPa (2,235 psia)
RWST Volume Technical Specification	1,470 cubic meters (387,100 gal)

1 The power level used in this report is the power level before the October 2010 measurement uncertainty recapture power uprate of 1.6%.

2 Table 6.2-5 of the FSAR provides a design head of 5800 ft (1.8% lower than the 5,905 feet that the stated value corresponds to). The MELCOR input model takes this design pressure in to account when specifying the pump curve, but bases the shutoff conditions on actual test data (Byron Jackson Test T-30705-3).

Figures 2.2.1 through 2.2.5 were reproduced from the SNL Surry model documentation; they show the MELCOR model nodalizations. NRC provided the MELCOR input deck, the raw output for calculations B.1 through B.20, and some incomplete and preliminary documentation to ERI. All of the calculations used the “YV” release of MELCOR/MELGEN version 1.8.6. (SNL performed calculations B.1 through B.20; ERI performed calculation B.21 and all of the analyses.)

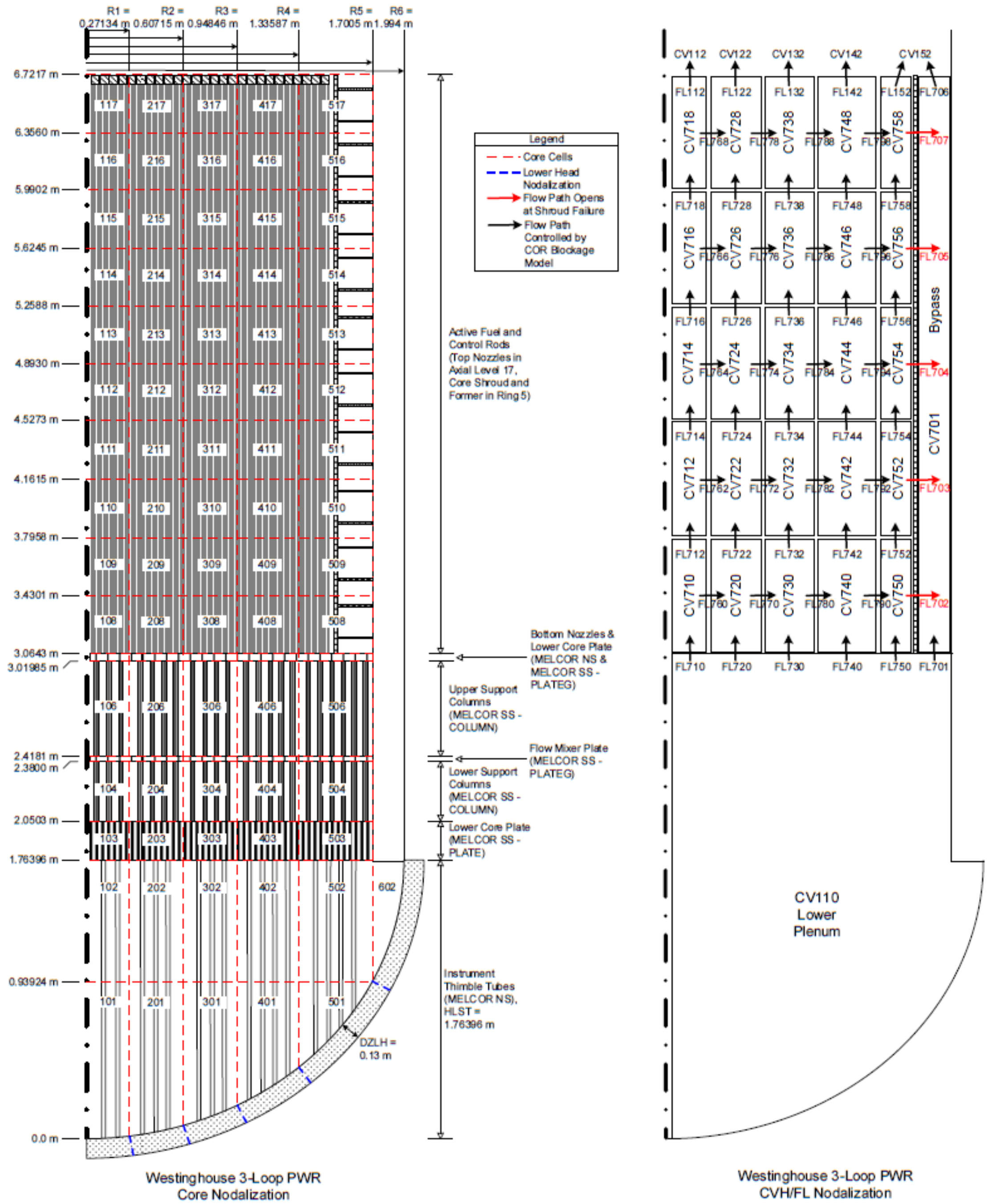


Figure 2.2.1 Nodalizations of the Core and Lower Part of the Vessel

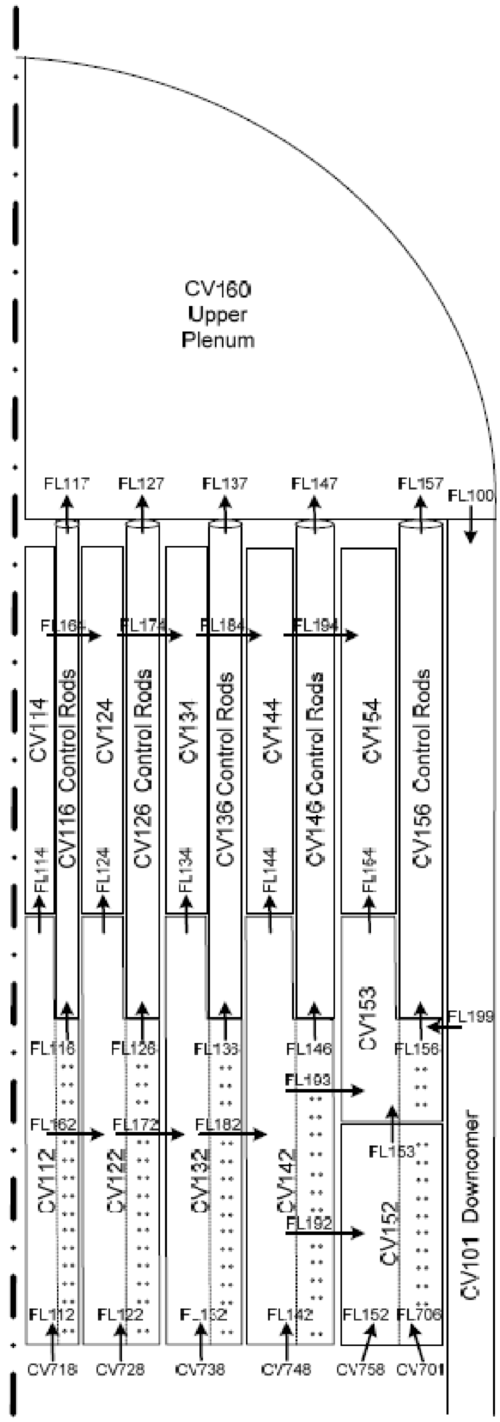


Figure 2.2.2 Nodalization of Upper Part of the Vessel

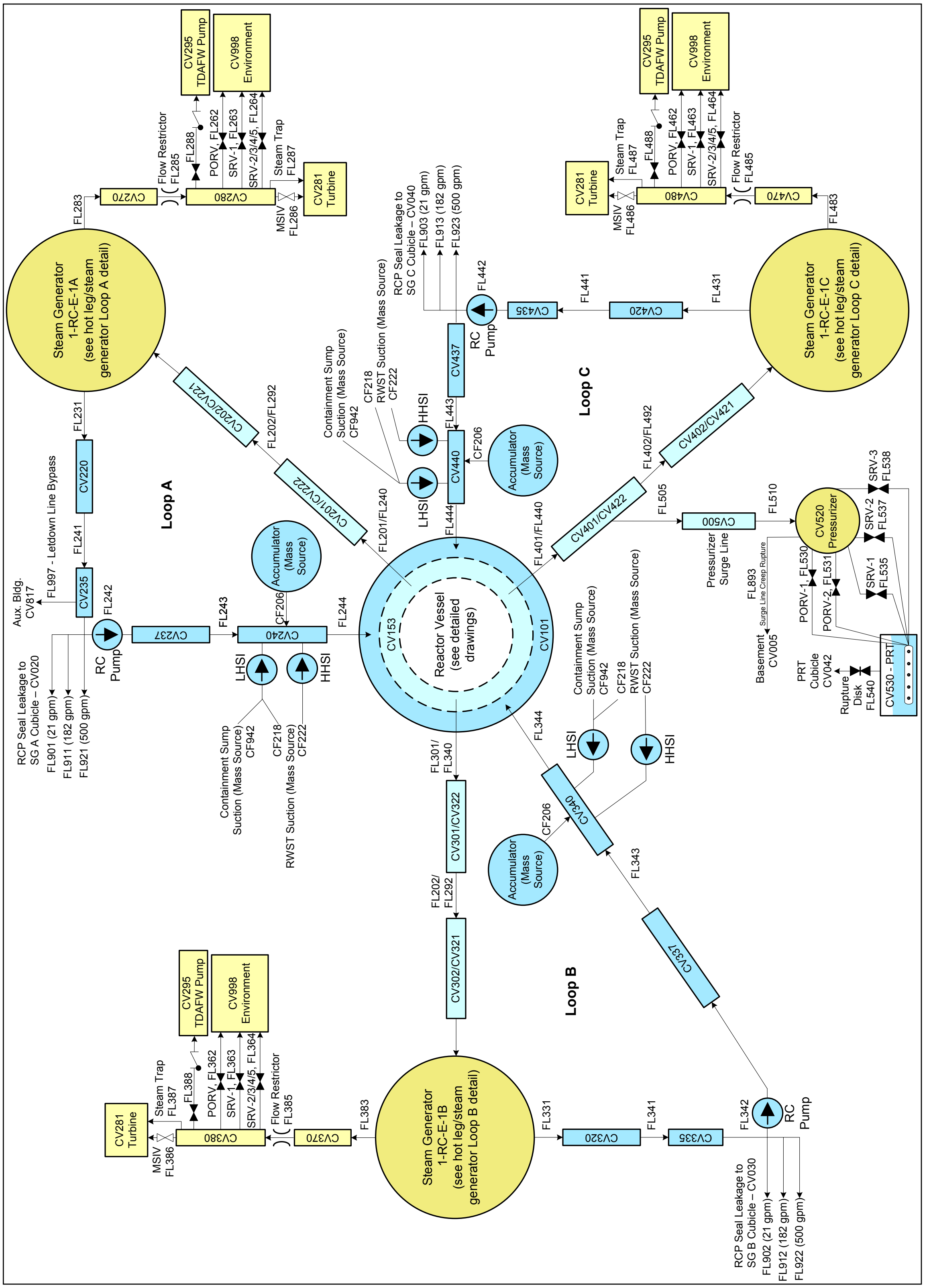


Figure 2.2.3 Nodalization of the Surry RCS

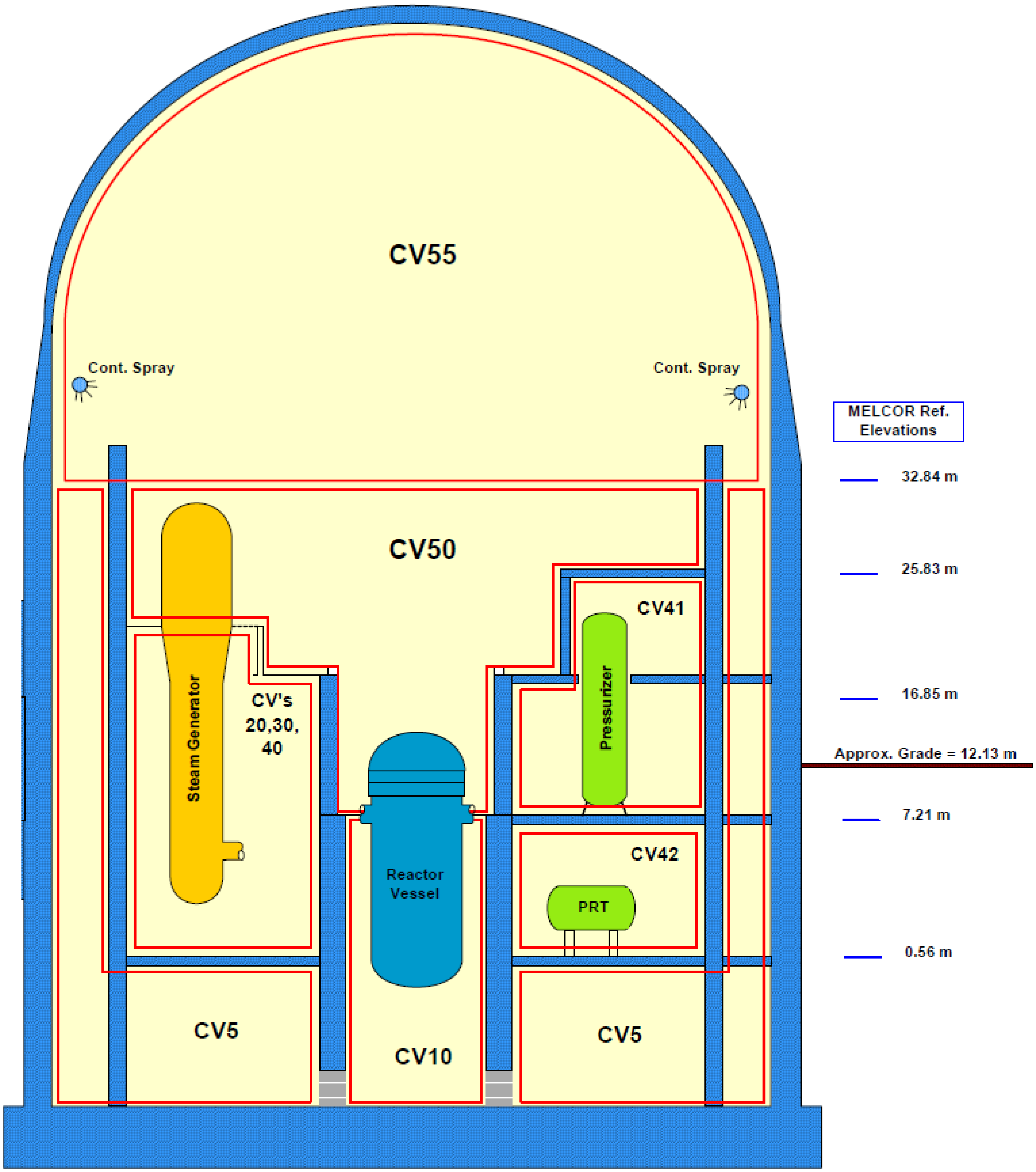


Figure 2.2.4 Nodalization of the Surry Containment

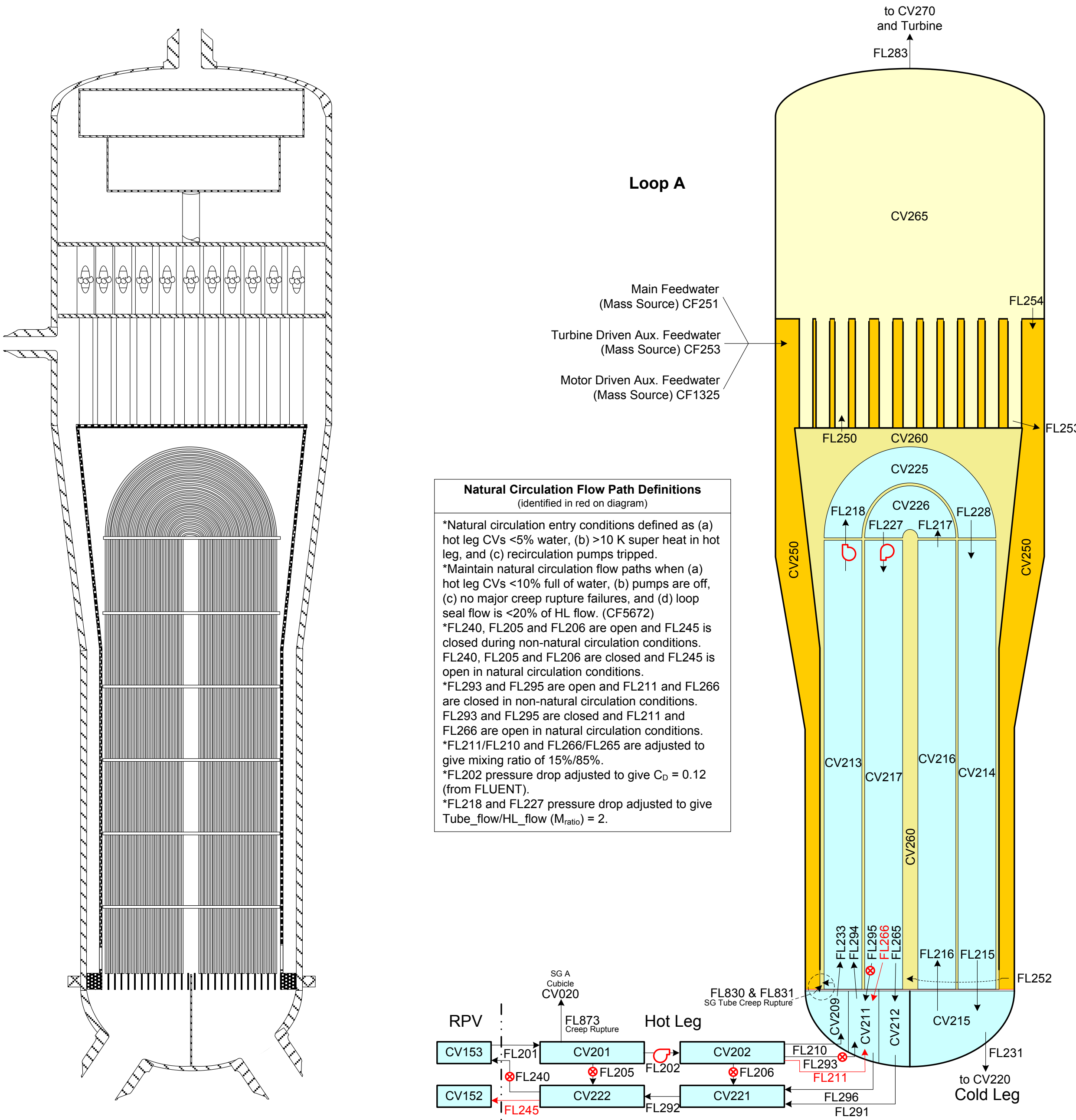


Figure 2.2.5 Drawing of a Steam Generator and Nodalization of Steam Generator A

2.3 Peach Bottom Plant and MELCOR Model

SNL prepared the Peach Bottom MELCOR model, under NRC sponsorship, as part of the State-of-the-Art Reactor Consequence Analyses project [3]. High-level information about the plant are listed in Table 2.3.1.

Table 2.3.1 Peach Bottom Plant Data

Characteristic	**Value (as modeled)**
Design Type	General Electric Type 4 BWR
Containment Type	Mark 1
Power Level	3,514 MWt
RCIC Capacity	2.27 cubic meters/min (600 gpm)
HPCI Capacity	18.9 cubic meters/min (5,000 gpm)
Lowest SRV Opening/Closing Setpoint in Relief Mode	7.81 MPa (1,133.5 psid[3]) / 7.58 MPa (1,099.5 psid)
Nominal Operating Pressure	7.24 MPa (1,050 psia)
Suppression Pool Inventory	3,568 cubic meters (942,300 gal)

All of the calculations used the "YV-3272" release of MELCOR/MELGEN version 1.8.6 and were performed at ERI. Figures 2.3.1 through 2.3.4 were reproduced from the SNL Peach Bottom documentation and show the MELCOR model nodalizations.

The scenario under study depends on the detailed modeling of the turbine-driven RCIC system (because it involves a situation where RCIC is the only available source of injection in to the reactor vessel). The rest of this section describes the RCIC and is closely paraphrased from the SNL Peach Bottom model documentation.

The RCIC system is modeled by control volumes and flow paths including the RCIC turbine (CV611) flow from main steam line C to the RCIC turbine (FL611), the flow from the RCIC turbine to the suppression pool (FL613), the flow from the condensate storage tank (CST) to the FW piping (including RCIC pump FL614), and the flow from the suppression pool to the FW piping (including RCIC pump FL606). Steam flow through the RCIC turbine is modeled to account for the transfer of energy from the steam line to the suppression pool during the RCIC operation.

[3] Pounds per square inch differential (psid) is the differential pressure in psi between the main steamline and the wetwell; the present (2012) Technical Specification value is actually 1135 psid, but this difference is judged negligible for the purposes of these calculations.

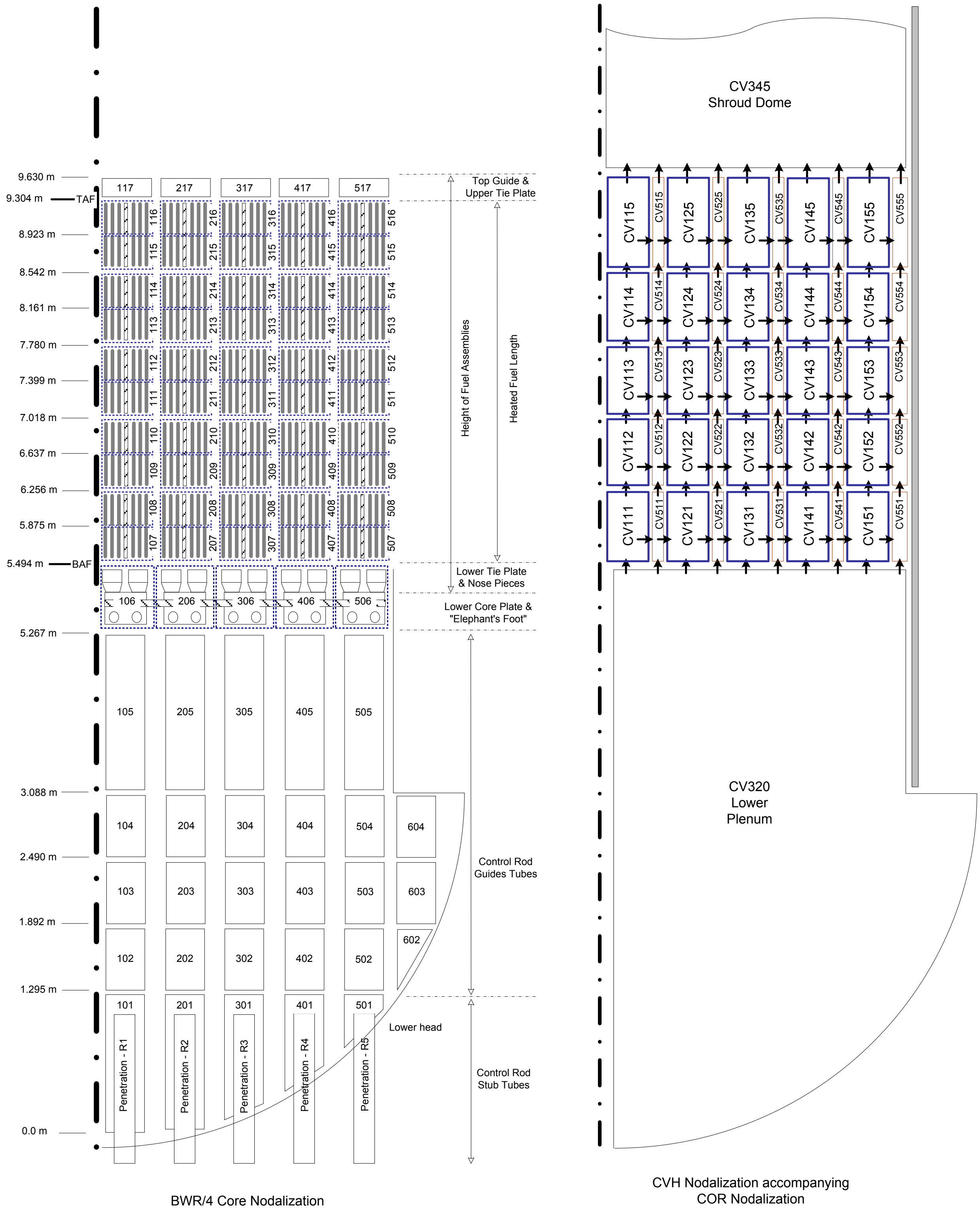


Figure 2.3.1 Nodalization of the Core Region

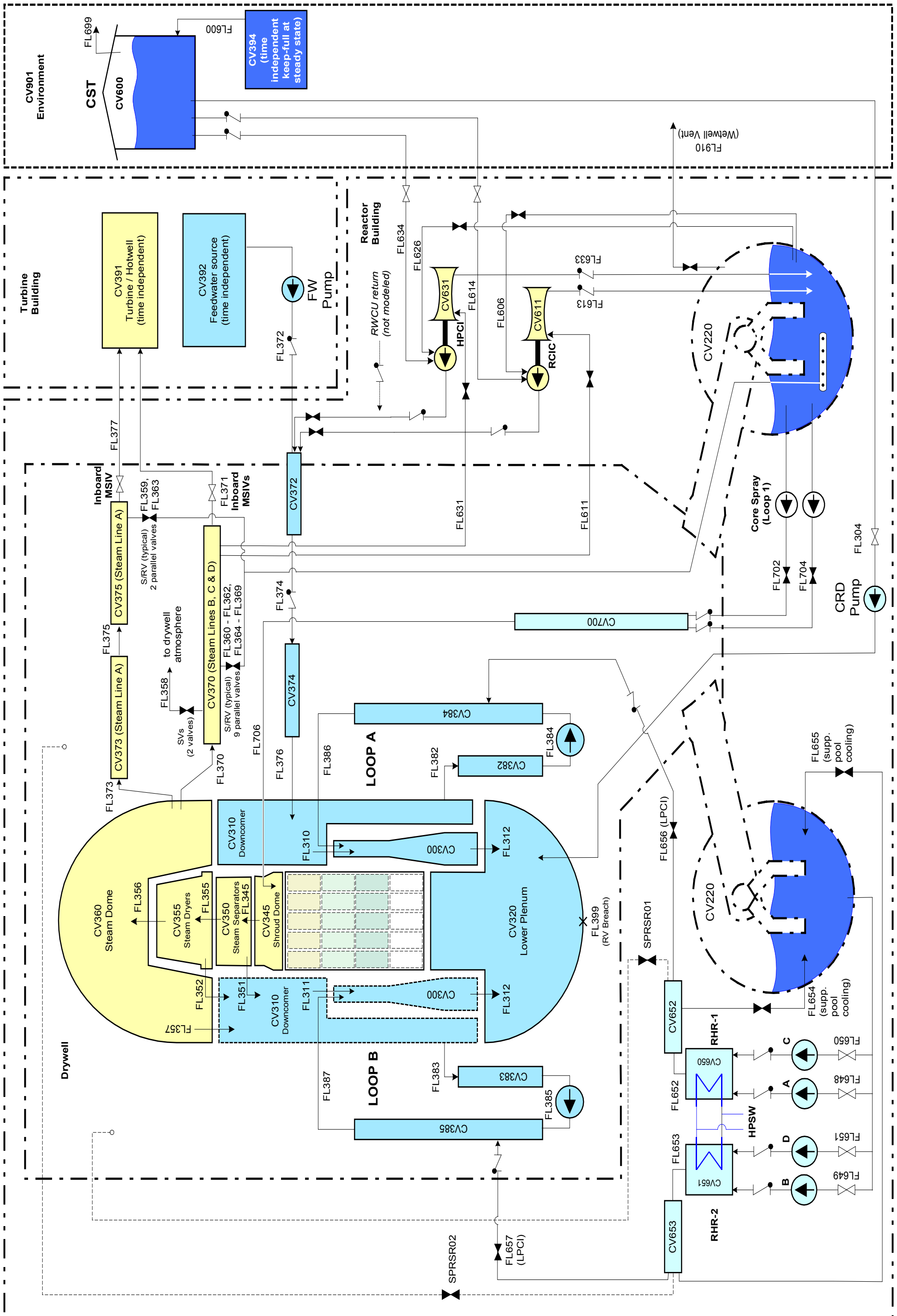


Figure 2.3.2 Nodalization of the Peach Bottom RCS

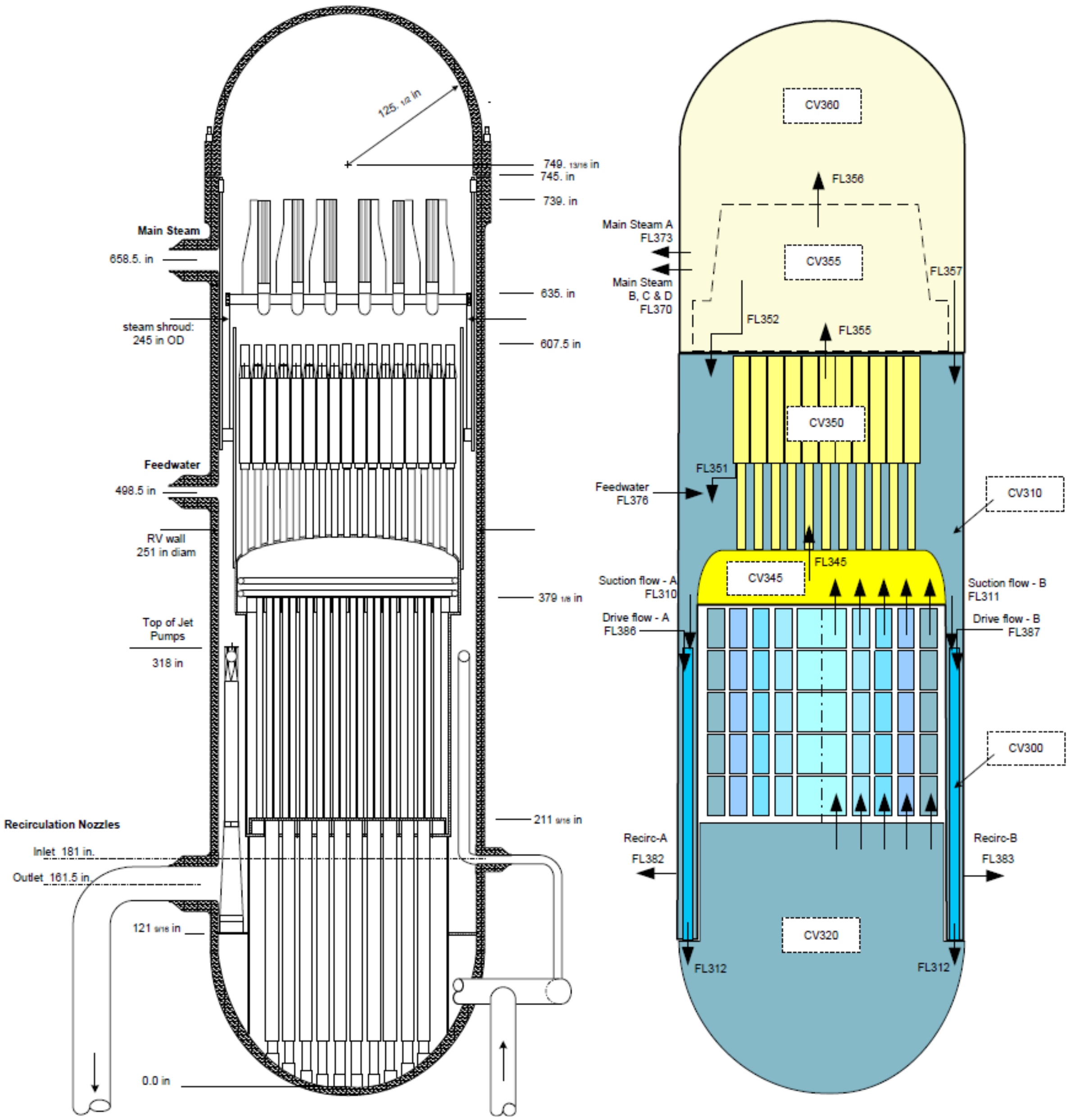


Figure 2.3.3 Nodalization of the Reactor Vessel

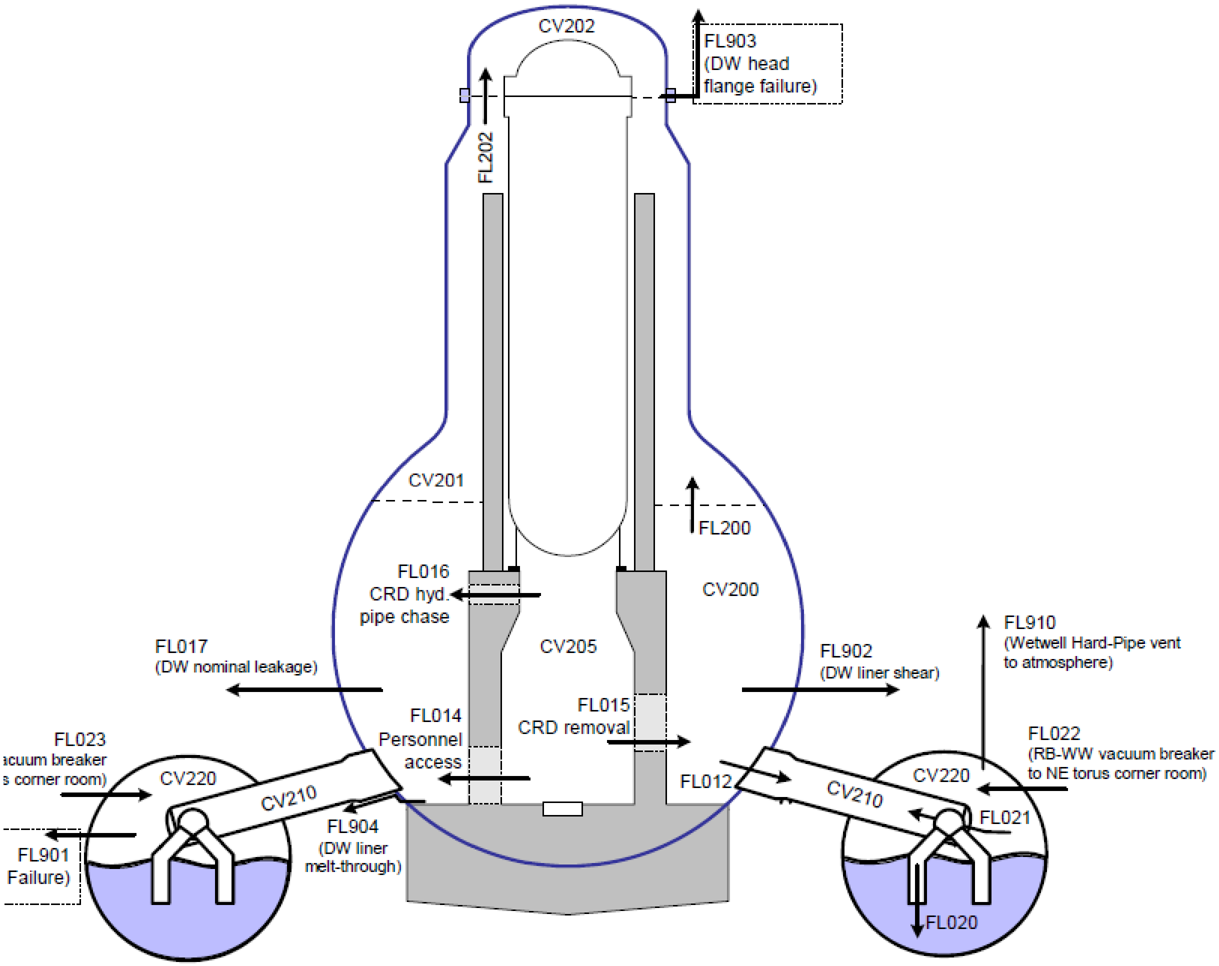


Figure 2.3.4 Nodalization of the Peach Bottom Containment

The RCIC pump delivers 600 gpm (0.038 m^3/s) via the MELCOR option to specify the flow velocity. Suction is initially aligned to the CST. The switchover of pump suction to the suppression pool occurs upon receipt of a low CST water level signal. This event, however, is not attained in any of the present simulations.

RCIC is modeled with automatic initiation and termination criteria. It is initiated upon receipt of a reactor vessel low level-2 signal at 472 inches (12.0 meters [m]). RCIC is terminated upon receipt of a reactor vessel high level-8 signal at 593 inches (15.1 m)[4]. User input (CF937) may also be selected to model manual pump operation, where operators throttle the RCIC turbine/pump to maintain water levels at 30 inches (0.76 m) above instrument zero at 568 inches (14.4 m) after automatic initiation.

[4] It was discovered after completion of the analyses that a value of 583 inches (14.8m) was actually used. The first footnote in Appendix C reports the effect on accident timing due to this setpoint deviation.

In the MELCOR model, the RCIC turbine-driven pump is isolated or tripped under the following conditions:

- loss of alternating current (AC) and direct current (DC) power
- low main steam line C pressure (i.e., main steam line pressure ≤75 pounds per square inch gauge [psig] [0.52 megaPascal-gauge [MPa-g])
- high turbine exhaust pressure (i.e., suppression pool pressure ≥50 psig [0.34 MPa-g])
- suppression pool temperature above pump net positive suction head (NPSH) limits (taken as 183 °F at 0 psig, 207 °F at 5 psig, 225 °F at 10 psig [84 °C at 0 MPa-g, 97 °C at 0.034 MPa-g, 107 °C at 0.069 MPa-g])
- pump bearing cooling fluid temperature limit exceeded (i.e., suppression pool temperature > 220 °F [107 °C])
- suppression pool water level below pump suction vortex limits (taken as 0.67 feet [0.20 m])
- flooding of the main steam line (CV370)

Note that some conditions that could actually fail the RCIC may have been omitted here. For example, effects on the RCIC operation of having no forced ventilation cooling in the RCIC pump room are not modeled. Meanwhile, some of the above trips might not be relevant for situations where the RCIC is being run without DC power.

The time of battery depletion (which results in loss of DC power) is user-specified by control function CF901. When the RCIC is operated manually, user input may specify the ongoing RCIC turbine/pump operation at the (fixed) speed current when the batteries are depleted (CF933).

2.4 Cross-Comparisons of Several 4-Loop Westinghouse PWRs

There are several plants that closely resemble the Byron Unit 1 plant described in Section 2.1. The comparison of these plants is of interest (even though the rest of this report focuses on the three plants discussed in Sections 2.1 through 2.3) because it will aid in an evaluation of the extension of results from a forthcoming and related NUREG exploring Byron success criteria.

2.4.1 Introduction

This section compares the system design data that are important to core damage progression and the containment response for five four-loop PWRs: the Byron, Braidwood, Callaway, Comanche Peak, and Vogtle plants[5]. In addition, simple analytically based estimates of the time of core damage for each plant were compiled and compared for a typical SBO scenario. Section 2.4.2 presents key design data for the plants. Section 2.4.3 presents methods for calculating estimates of core damage and related results.

2.4.2 Comparisons of Plant Design Data Important to Severe Accident Progression

Tables 2.4.1a and 2.4.1b compare key design data for the Byron, Braidwood, Callaway, Comanche Peak, and Vogtle reactors. (These two tables have identical content, except Table 2.4.1b provides the values in English units.) The Byron and Braidwood plants are based on a nearly identical design. Their common design data appear under the heading Byron/Braidwood and are based on the common Byron/Braidwood FSAR. The Byron and Comanche Peak power plants consist of two units with nearly identical nuclear steam supply systems and turbine generators. The steam generators for Byron Unit 1 compared with those for Byron Unit 2 are different; the same is true for the two units at the Comanche Peak site. Table 2.4.1 shows that the RCS volume and heat transfer area in the steam generators for Units 1 and 2 at the Byron and Comanche Peak plants are different. Table 2.4.1 also indicates some missing data as not

[5] There are several other plants of similar design (e.g., Seabrook, Diablo Canyon) but only the 5 identified sites are compared in order to reduce the scope of this comparison effort.

available. This data is subject to variability in the ways that information is presented in the FSAR or other sources (e.g., nominal delivered flow versus theoretical pump flow rate), and should be used with caution.

To compare the system design features important to core damage and containment pressurization among the plants, Table 2.4.2 lists scaling ratios such as system volume to power and hydrogen mass to containment volume. In addition, to estimate containment pressurization from non-condensable gases or from consideration of combustion loads, Table 2.4.2 compares the calculated maximum mass of hydrogen generated by zirconium (Zr) oxidation with the average hydrogen concentration in the containment. Although these issues are not relevant to estimates of time to core damage, they are of interest to level-2 PRAs.

The following noteworthy observations are based on comparisons of the data in Table 2.4.1:

- The ratio of the RCS water volume to reactor power can provide the degree of margin for loss of decay heat removal scenarios. The results of this comparison show that the RCS volume-to-reactor power ratios of Unit 1 at Byron, Callaway, and Comanche Peak are around 10 percent higher than the others. This difference is due to different steam generator designs that result in a bigger total RCS water volume for the former units. With all other conditions remaining the same, the time to boiling for the entire RCS water inventory should be about 10 percent longer for Unit 1 of Byron, Callaway, and Comanche Peak compared to the rest of the plants.

- The containment free-volume-to-reactor power ratio provides an indication of the degree of margin for the buildup of pressure in the containment during severe accidents. The Callaway plant shows the lowest value. On the other hand, the Callaway plant has the highest containment design pressure.

- The ratio of Zircaloy and fuel masses to the containment volume indicates the degree of margin for containment pressurization by non-condensable gases generated during molten core concrete interactions or Zircaloy oxidation. In other words, as the value of this ratio becomes larger, the containment pressurization rate is expected to increase. The Callaway plant has the highest such ratio, so the expectation is that its containment will show the fastest pressurization rate compared to the other plants.

- The uniform maximum hydrogen concentration inside the containment appears to be similar for all of these plants except Callaway, which is around 35 percent higher than the other listed plants. This concentration is consistent with the comparison of the ratio of Zircaloy mass to containment free volumes.

Table 2.4.1a Comparisons of Design Data for Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Reactors (SI units)

Component	Parameters*	Unit	Byron/Braidwood	Callaway	Comanche Peak	Vogtle
Reactor Coolant System						
	Reactor power	MW_{th}	3586.6	3565	3612	3626
	Heated fuel length	m	3.66	3.65	3.65	3.65
	UO_2 weight	kg	92641	92636	93148	92639
	Cladding weight	kg	19675	24422	23024	20827
	Normal primary pressure	MPa-a	15.65	15.41	15.51	15.51
	Normal primary temperature	K	583.9	584.9	584.6	584.3
	Normal core mass flow rate	10^6 kg/hr	62.2	63.2	64.4	63.3
	Total system volume including PRZ and surge line	m^3	386 / 350 (Unit1/Unit 2)	394	379 / 354 (Unit1/Unit 2)	350
	Total system water volume including PRZ and surge line	m^3	375 / 338 (Unit1/Unit 2)	370	362 / 340 (Unit1/Unit 2)	328
	Total reactor vessel water volume below active fuel	m^3	65	65 (1)	65 (1)	65 (1)
	Reactor cooling pump (RCP) coast down duration	s	30	30	30	30
Pressurizer						
PORV (2 valves)	Opening pressures of pressurizer relief valves	MPa-g	16.2 / 16.1 (Unit1/Unit 2)	16.1	16.1	16.2 / 16.1 (Unit1/Unit 2)
	Capacity of pressurizer relief valve for one unit	kg/hr	9.5E4	9.5E4	9.5E4	9.5E4
	Nominal volume of steam in pressurizer	m^3	11	20	23 / 14 (Unit1/Unit 2)	20
SRV (3 valves)	Opening pressure	MPa-g	16.97	16.97	17.13	16.97
	Capacity of valve for one unit	kg/hr	1.90E5	1.90E5	1.90E5	1.90E5
Steam Generators						
	Heat transfer area	m^2	7414 / 4487 (Unit1/Unit 2)	7334	7061 / 4487 (Unit1/Unit 2)	5110
	Nominal secondary pressure	MPa-g	6.97 / 6.47 (Unit1/Unit 2)	7.12	7.04 / 6.86 (Unit1/Unit 2)	6.59

Table 2.4.1a Comparisons of Design Data for Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Reactors (SI units)

Component	Parameters*	Unit	Byron/Braidwood	Callaway	Comanche Peak	Vogtle
	Nominal secondary liquid volume	m^3	71.3/Not Available (Unit 1/Unit 2) (2)	66.8	Not Available	72.4
	Opening pressure of steam generator safety relief valve (5 valves)	MPa-g	8.20 (2) 8.31 (2) 8.41 (2) 8.52 (2) 8.10 (2) (Unit 1 only)	Not Available	Not Available	Not Available
	Total steam flow	10^6 kg/hr	7.27 / 7.23 (Unit 1/Unit2)	7.24	7.01 / 6.97 (Unit1/Unit2)	7.36
	Total capacity of relief valves (% of total steam flow)	%	112 (2)	Not Available	105	Not Available
AFW	Design flow rate: motor-driven / turbine- or diesel-driven, per pump (# and type of pumps)	m^3/s	0.056 (1m) / 0.056 (1d)	0.036 (2m) / 0.072 (1t)	0.036 (2m) / 0.072 (1t)	0.040 (2m) / 0.074 (1t)
Accumulator (4 units)						
	Accumulator pressure	MPa-g	4.41	4.14-4.48	4.16-4.78	4.41
	Liquid temperature	K	311-339	322	294-339	289-322
	Accumulator water volume	m^3	26.5	24.1	23.2	25.5
	Total volume	m^3	38.2	Not Available	38.2	38.2
Emergency Core Cooling System						
Charging pump (2 units)	Maximum flow rate	m^3/s	0.035	0.028(3)	0.035	0.035
	Head at maximum flow rate(4)	MPa-d	5.17	Not Available	4.14	5.03
	Dead head pressure	MPa-d	17.6	Not Available	Not Available	Not Available
Safety injection (SI) pump (2 units)	Maximum flow rate	m^3/s	0.041	0.028 (3)	0.041	0.041
	Head at maximum flow rate(4)	MPa-d	5.59	Not Available	4.88	5.17
	Dead head pressure	MPa-d	10.7	Not Available	Not Available	Not Available
RHR pump (2 units)	Design flow rate	m^3/s	0.189	0.303	0.240	0.189
	Dead head pressure	MPa-d	1.17	Not Available	Not Available	Not Available
	Head at maximum flow rate(4)	MPa-d	1.10	Not Available	1.03	1.10

Table 2.4.1a Comparisons of Design Data for Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Reactors (SI units)

Component	Parameters*	Unit	Byron/Braidwood	Callaway	Comanche Peak	Vogtle
Spray pump	Number of units	-	2	2	2	2
	Approx. deliverable flow per train(5)	m^3/s	0.207	0.195	0.284	0.202
Reactor water storage tank (RWST)	Tank temperature	K	322	311	322	328
	RWST total deliverable water volume	m^3	1238	1237	1471	1897
Containment						
	Free volume	m^3	7.810E+4	7.079E+4	8.580E+4	7.787E+4
	Design pressure	MPa-g	0.34	0.41	0.34	0.36)
	Basemat thickness	m	2.44(6)	3.05	3.66	2.44 (6)
	Basemat concrete composition	-	Zinc paint/carbon steel/concrete	Zinc paint/carbon steel/concrete	Zinc paint/carbon steel/concrete	Zinc paint/carbon steel/concrete
	Reactor cavity flow area	m^2	84.3 (Calculated)	Not Available	Not Available	396.0 (heat transfer area)

* The design parameters are from the Final Safety Analysis Reports. Additionally, in the case of Byron Unit 1, some parameters are based on the MELCOR model input deck, as noted.

(1) All reactors are assumed to have the same water volume below the top of the active fuel since documents show that the core diameters and equivalent areas are all equal; the tabulated value derives from the Byron MELCOR deck.

(2) Value from the ERI MELCOR model of Byron Unit 1.

(3) Charging pump capacity from Callaway FSAR Table 6.3-12.

(4) Head at maximum flow rate (MPa-d, a.k.a., MPa-differential) is calculated based on the head in feet at the maximum flow rate.

(5) This is the representative value cited in the containment analysis (Section 6 of the FSAR). In some cases, other values are given in the FSAR for pump capacity.

(6) Basemat thickness at the cavity.

Table 2.4.1b Comparisons of Design Data for Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Reactors (English units)

Component	Parameters*	Unit	Byron/Braidwood	Callaway	Comanche Peak	Vogtle
Reactor Coolant System						
	Reactor power	MW_{th}	3586.6	3565	3612	3626
	Heated fuel length	in.	144	143.7	143.7	143.7
	UO_2 weight	lb	204236	180200	205352	204231
	Cladding weight	lb	43376	53840	50759	45914
	Normal primary pressure	Psia	2270	2235	2250	2250
	Normal primary temperature	F	591.7	593.1	592.8	592.3
	Normal core mass flow rate	10^6 lb/hr	137.2	139.4	142	139.5
	Total system volume including PRZ and surge line	ft^3	13620/12340 (unit1/unit 2)	13910	13400/12500 (unit1/unit 2)	12347
	Total system water volume including PRZ and surge line	ft^3	13230/11950 (unit1/unit 2)	13068	12800/12000 (unit1/unit 2)	11594
	Total reactor vessel water volume below active fuel	ft^3	2293	2293 (1)	2293 (1)	2293 (1)
	Reactor cooling pump (RCP) coast down duration	s	30	30	30	30
Pressurizer						
PORV (2 valves)	Opening pressures of pressurizer relief valves	psig	2345/2335	2335	2335	2345/2335
	Capacity of pressurizer relief valve for one unit	lb/h	2.10E+05	2.10E+05	2.10E+05	2.10E+05
	Nominal volume of steam in pressurizer	ft^3	390/390 (unit1/unit 2)	720	800/500 (unit1/unit 2)	720
SRV (3 valves)	Opening pressure	psig	2460	2460	2485	2460
	Capacity of valve for one unit	lb/hr	4.20E+05	4.20E+05	4.20E+05	4.20E+05
Steam Generators						
	Heat transfer area	ft^2	79800/48300 (unit1/unit 2)	78946	76000/48300 (unit1/unit 2)	55000
	Nominal secondary pressure	psig	1010/938	1033	1021/995	955.7

Table 2.4.1b Comparisons of Design Data for Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Reactors (English units)

Component	Parameters*	Unit	Byron/Braidwood	Callaway	Comanche Peak	Vogtle
			(unit1/unit 2)		(unit1/unit 2)	
	Nominal secondary liquid volume	ft^3	2516/Not Available (Unit 1/Unit 2) (2)	2358	Not Available	2558(calculated)
	Opening pressure of steam generator safety relief valve (5 valves)	psig	1190 (2) 1205 (2) 1220 (2) 1235 (2) 1175 (2) (Unit 1 only)	Not Available	Not Available	Not Available
	Total steam flow	10^6 lb/hr	16.02 / 15.95 (Unit 1/Unit2)	15.96	15.46 / 15.36 (Unit1/Unit2)	16.22
	Total capacity of relief valves (% of total steam flow)	%	112 (2)	Not Available	105	Not Available
AFW	Design flow rate: motor-driven / turbine- or diesel-driven, per pump (# and type of pumps)	gpm	900 (1m) / 900 (1d)	575 (2m) / 1145 (1t)	570 (2m) / 1145 (1t)	630 (2m) / 1175(1t)
Accumulator (4 units)						
	Accumulator pressure	psig	640	600-650	603-693	640
	Liquid temperature	F	100-150	120	70-150	60-120
	Accumulator water volume	ft^3	935	850	820	900
	Total volume	ft^3	1350	Not Available	1350	1350
Emergency Core Cooling System						
Charging pump (2 units)	Maximum flow rate	gpm	550	450(3)	550	550
	Head at maximum flow rate(4)	ft	1750	Not Available	1400	1700
	Dead head pressure	psid	2550	Not Available	Not Available	Not Available
Safety injection (SI) pump (2 units)	Maximum flow rate	gpm	655	450 (3)	650	650
	Head at maximum flow rate(4)	ft	1890	Not Available	1650	1750
	Dead head pressure	psid	1550	Not Available	Not Available	Not Available
RHR pump (2 units)	Design flow rate	gpm	3000	4800	3800	3000
	Dead head pressure	psid	170	Not Available	Not Available	Not Available
	Head at maximum flow rate(4)	ft	375	Not Available	350	375
Spray pump	Number of units	-	2	2	2	2

Table 2.4.1b Comparisons of Design Data for Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Reactors (English units)

Component	Parameters*	Unit	Byron/Braidwood	Callaway	Comanche Peak	Vogtle
	Approx. deliverable flow per train(5)	gpm	3285	3086	4500	3200
Reactor water storage tank (RWST)	Tank temperature	F	120	100	120	130
	RWST total deliverable water volume	gallon	326972	326860	388677	501257
Containment						
	Free volume	ft^3	2.758E+06	2.50E+06	3.03E+06	2.75E+06
	Design pressure	psig	50	60	50	52
	Basemat thickness	ft	8(6)	10	12	8 (6)
	Basemat concrete composition	-	Zinc paint/carbon steel/concrete	Zinc paint/carbon steel/concrete	Zinc paint/carbon steel/concrete	Zinc paint/carbon steel/concrete
	Reactor cavity flow area	ft^2	907.9 (calculated)	Not Available	Not Available	4263 (heat transfer area)

* The design parameters are from the Final Safety Analysis Reports. Additionally, in the case of Byron Unit 1, some parameters are based on the MELCOR model input deck, as noted.

(1) All reactors are assumed to have the same water volume below the top of the active fuel since documents show that the core diameters and equivalent areas are all equal; the tabulated value derives from the Byron MELCOR deck.

(2) Value from the ERI MELCOR model of Byron Unit 1.

(3) Charging pump capacity from Callaway FSAR Table 6.3-12.

(4) Head at maximum flow rate is based on the head in feet at the maximum flow rate.

(5) This is the representative value cited in the containment analysis (Section 6 of the FSAR). In some cases, other values are given in the FSAR for pump capacity.

(6) Basemat thickness at the cavity.

Table 2.4.2 Summary of Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Design Features

Parameters	Unit	Byron/Braidwood (Unit 1/Unit 2)	Callaway	Comanche Peak (Unit 1/Unit 2)	Vogtle
RCS water volume/power	m^3/MW_{th}	0.104/0.094	0.104	0.100/0.094	0.091
Containment free volume/power	m^3/MW_{th}	22	20	24	22
Zr mass/containment free volume	kg/m^3	0.252	0.345	0.268	0.267
Fuel mass/containment free volume	kg/m^3	1.186	1.308	1.086	1.190
Maximum mass of H_2 generated by Zr oxidation*	Kg	863.0	1071.1	1009.8	913.4
Maximum H_2 concentration	10^{-3} moles/m^3	5.52	7.57	5.88	5.87
Containment design pressure	MPa	0.44	0.51	0.44	0.46

* 100% of cladding oxidation was assumed in calculating maximum mass of H_2 generated

2.4.3 Time to Core Damage Estimate

This section formulates and uses simple analytic approaches to estimate the time to core damage for an assumed SBO scenario to illustrate their usefulness when a simplified approach is desired for estimating operator response timings. Section 3 in this report contains fully mechanistic calculations of time to core damage. The assumed initiator of this scenario is a reactor scram and a complete loss of cooling (including the main and auxiliary FW systems). All normal makeup and emergency core cooling systems (ECCS) are assumed to be unavailable. Two criteria are used to define core damage: (i) the time that core uncovery started, and (ii) the time when the peak cladding temperature (PCT) reached 1,478 K (2,200 °F). The first criterion is a more conservative indicator since it is based on the consideration of the water level reaching the top of the active fuel (TAF) only. The second criterion is based on the time required for the PCT to attain 1,478 K, which typically corresponds to a time when the water level is considerably lower than the TAF. Therefore, this criterion provides a more realistic estimate of core damage initiation.

Note that following discussion neglects the effects of RCP seal leakage, and this effect would need to be accounted for in cases where significant leakage is projected.

2.4.3.1 Time to Core Uncovery Estimate

This section estimates the time of core uncovery in the case of an SBO accident scenario. In an SBO, the time of core uncovery is related to the time integral of the decay heat since the reactor shutdown. The SBO scenario is the simplest boil-off scenario where the heat sink is lost and the fluid leaves through the pressurizer (PRZ) relief valves. Core uncovery is assumed to begin when all of the water in the RCS above the TAF is depleted due to the boil-off. Initially, the decay heat is removed by pool boiling in the steam generators (secondary sides). After the secondary sides of the steam generators dry out, the decay heat is removed by boiling off the primary system water, which then passes through the pressurizer PORVs as steam.

The first step for calculating core uncovery time is to estimate the time of steam generator secondary side dryout (see Reference [4]). The time (t_1) at which the secondary system dries out is given approximately by:

$$\int_0^{t_1} Q_d \, dt = \rho h_{fg} V_{sc} + insulation\_losses \qquad (2.4.1)$$

where Q_d is the decay power, Vsc is the volume of liquid on the secondary side of the steam generators, ρ is the density of water, and h_{fg} is the latent heat of vaporization at the steam generator relief valve set point of about 1,190 psia (8.2 MPa)[6]. Note that the insulation heat losses to the containment are assumed to be negligible since the steam generators are well insulated.

For this calculation, a simplified decay heat curve describing typical PWRs is developed. For a specific example, the system data for the Byron Unit 1 reactor are used. Two time periods are defined for more accurate decay heat representation:

$$Q_d = 0.064 Q_0 \times t^{-0.082} \quad 1 < t < 50 \text{ s} \qquad (2.4.2a)$$

$$Q_d = 0.1177 Q_0 \times t^{-0.24} \quad 50 < t < 10^6 \text{ s} \qquad (2.4.2b)$$

Here Q_0 is the initial power level (e.g., 3,586.6 MW(t) in the case of Byron Unit 1). The required energy to vaporize the water mass (right-hand side of Equation (2.4.1) is calculated as:

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[6] Note that using the steam generator PORV setpoint would change h_{fg} by less than 0.5%.

$$\int_0^{t_1} Q_d dt = E = \rho h_{fg} V_{sc} = 3.01 \times 10^5 \text{ MJ}$$

where the mass of water in the steam generator (ρV_{sc} = 740.7 × 71.2 × 4) = 210,950 kg at a pressure of 1,025 psia (7.067 MPa); the latent heat for vaporization, h_{fg}, is 1.428 x 10^6 J/kg at a SRV opening pressure of 1,190 psia (8.2 MPa).

Therefore, Equation (2.4.1) is expressed as follows using the decay heat as approximated by Equation (2.4.2):

$$\int_0^{t_1} Q_d dt = 0.1548 Q_0 (t_1^{0.76} - 50^{0.76}) + 2.459 Q_0 = 3.01 \times 10^5 \text{ MJ}$$

which can be used to calculate t_1 = 4,060 seconds (68 minutes).

As a consequence of the drying out of the steam generators, no major heat sink is subsequently available for the removal of the heat that continues to evolve within the primary system. After the secondary system dries out, the pressure in the primary system rises until it reaches the opening setpoint pressure of the pressurizer PORVs.

The core uncovery time (t_2) can be obtained by balancing the decay heat that results in the evaporation of saturated water mass, m_{taf}, to bring the water level to the TAF:

$$m_{taf} = \int_{t_1}^{t_2} \frac{Q_d(t)}{h_{fg}} dt \tag{2.4.3}$$

where t_1 is the time at which the steam generator secondary side dries out in seconds, t_2 is the time to core uncovery in seconds, and h_{fg} is the heat of vaporization of water in J/kg. Note that the energy required to heat the initially subcooled liquid to saturation is neglected here.

Using the decay heat representation of Equation 2.4.2 that is of the form:

$$Q_d = a Q_0 \times t^{-b} \tag{2.4.4}$$

Substituting Equation (2.4.4) into (2.4.3) and performing the integration yields:

$$m_{taf} h_{fg} = \int_{t_1}^{t_2} Q_d dt = a Q_0 \int_{t_1}^{t_2} t^{-b} dt = a Q_0 \frac{(t_2^{1-b} - t_1^{1-b})}{(1-b)} \tag{2.4.5}$$

which can be rearranged to obtain an expression for the time to core uncovery:

$$t_2 = [(1-b)\frac{m_{taf} h_{fg}}{a Q_0} + t_1^{1-b}]^{1/(1-b)} \tag{2.4.6}$$

In the case of Byron Unit 1, the mass of water in the primary side at the normal operating condition (2,270 psia [15.651 MPa]) is 2.63 × 10^5 kg. The temperature of this water is initially 592 K, being subcooled by 27 K; the corresponding liquid density is 704 kg/m^3. Of this, it is assumed that all primary-side liquid in the RCS at elevations above the active fuel (2.17 × 10^5 kg) must be lost to uncover the core, but not all of it needs to be boiled. Because of the thermal expansion of the initially subcooled water as it heats up and finally becomes saturated at the relief valve opening pressure (2,350 psia [16.203 MPa]; temperature 622 K; density 581 kg/m^3), the liquid expands beyond the capacity of the RCS, and some is spilled through the PORVs. The energy required for this liquid heatup is negligible. The amount of spilled liquid can be estimated as follows:

- The initial liquid volume is 2.63 × 10^5 kg/704 kg/m^3 = 373.6 m^3.
- After expansion, the liquid volume is 2.63 × 10^5 kg/581 kg/m^3 = 452.7 m^3.
- The pressurizer can accommodate 11 m^3 of the increase (initial void).
- Therefore, the spilled volume is 79.1 m^3–11 m^3 = 68.1 m^3.
- Using the simple-average density (642.5 kg/m^3), the spilled mass is 4.4 × 10^4 kg.

Therefore, the required amount of liquid water, m_{taf}, which must be converted to steam to uncover the core is (2.17 × 10^5 - 4.4 × 10^4) = 1.73 ×10^5 kg. With m_{taf} = 1.7 × 10^5 kg and using the latent heat at saturation at 2,350 psia (16.203 MPa] (h_{fg} = 1.428 x 10^6 J/kg), the estimated core uncovery time is determined using Equation (2.4.6) to be t_2 = 6,960 seconds (116 minutes).

These approximate results of 68 minutes for t_1 and 116 minutes for t_2 compare favorably with the MELCOR predictions of 66 and 117 minutes, respectively. Therefore, this approximate calculation provides an adequate physical representation of this sequence during the boil-off stage. Table 2.4.3 compares the times of steam generator secondary side dryout and start of core uncovery among the five plants. The time of core uncovery appears to be similar among the plants, which is consistent with volume-to-power ratio comparison results in Table 2.4.2. Note that the core uncovery time estimates for Byron Unit 2 and the Comanche Peak plants are not listed due to the lack of data for the steam generator secondary side water volume.

Table 2.4.3 Comparisons of Time of Core Uncovery for Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Nuclear Power Plants

Parameters	Unit	Byron/Braidwood	Callaway	Comanche Peak	Vogtle
Core uncovery time	min	116 (Unit 1) (MELCOR, 117)	114	116 (Unit 1)	114
SG dryout time	min	67 (Unit 1) (MELCOR, 66)	61	65*	68

*The steam generator secondary-side water volume of Byron Unit 1 is assumed due to a lack of information.

2.4.3.2 Time to Core Damage Estimate

The time that the peak cladding temperature (PCT) attains any of variously considered limiting temperatures provides a better estimate for the start of core damage than the time that core uncovery starts. The time of core damage is identified with the time when the PCT attains 1,478 K (2,200 °F). To identify this time, the MELCOR calculations for Byron Unit 1 that describe one base and six sensitivity cases of the loss of DC bus with bleed and feed (B&F) scenario (the LoDC Bus scenario referred to here includes independent failure of DD-AFW which results in a total loss of feedwater and the unavailability of 1 PORV; see Section A.1 of Appendix A) and likewise one SBO scenario (see Section 5) are referred to for reference data for the RPV water level at the time that the PCT reaches 1,478 K (see Table 2.4.4).

Table 2.4.4 MELCOR-Predicted Times of Core Damage and Corresponding Water Level (Byron)

Case (See Appendix A for case definitions)	**RPV water level below TAF (hr)**[1]	**PCT>1478K (hr)**	**RPV water level at PCT>1478 K (m)**[2]	**Covered fraction of core**
LoDC Bus (A.1.1)	1.35	1.91	3.72	0.17
LoDC Bus (A.1.3)	1.28	1.83	3.72	0.17
LoDC Bus (A.1.4)	1.28	1.81	3.70	0.17
LoDC Bus (A.1.5)	1.32	1.88	3.74	0.18

Case (See Appendix A for case definitions)	**RPV water level below TAF (hr)**[1]	**PCT>1478K (hr)**	**RPV water level at PCT>1478 K (m)**[2]	**Covered fraction of core**
LoDC Bus (A.1.6)	1.35	1.91	3.73	0.17
LoDC Bus (A.1.7)	1.35	1.91	3.74	0.18
LoDC Bus (A.1.8)	1.35	1.91	3.73	0.17
SBO	1.95	2.57	3.44	0.10

[1] Recall that for the non-SBO scenarios, the time to core uncovery will be different due to the period of time very early in the transient where the reactor is still at full-oower but main feedwater has been lost
[2] This elevation is relative to the inside bottom of the RPV lower head; for reference BAF is just over 3m and TAF is just under 7m; these are collapsed water levels

As shown in Figure 2.4.1, the normalized water level (i.e., fraction of the core that is covered) at the time that the PCT reached 1,478 K for various Byron LoDC Bus cases and one Byron SBO case ranges from 0.1 to 0.18. These levels correspond to RPV levels of 3.44 m and 3.72 m, respectively for LoDC Bus and SBO. Analytic calculations for the PCT time to reach 1,478 K can be performed based on these assumed equivalent water levels. This time is identified when the core water inventories are reduced to these two levels. Two possible values for the equivalent water level are derived from somewhat different scenarios (i.e., they differ in that the LoDC Bus includes B&F), thus providing some indication of the variability in the equivalent water level. The results are intended to apply to the SBO scenarios because the estimated levels apply to that case.

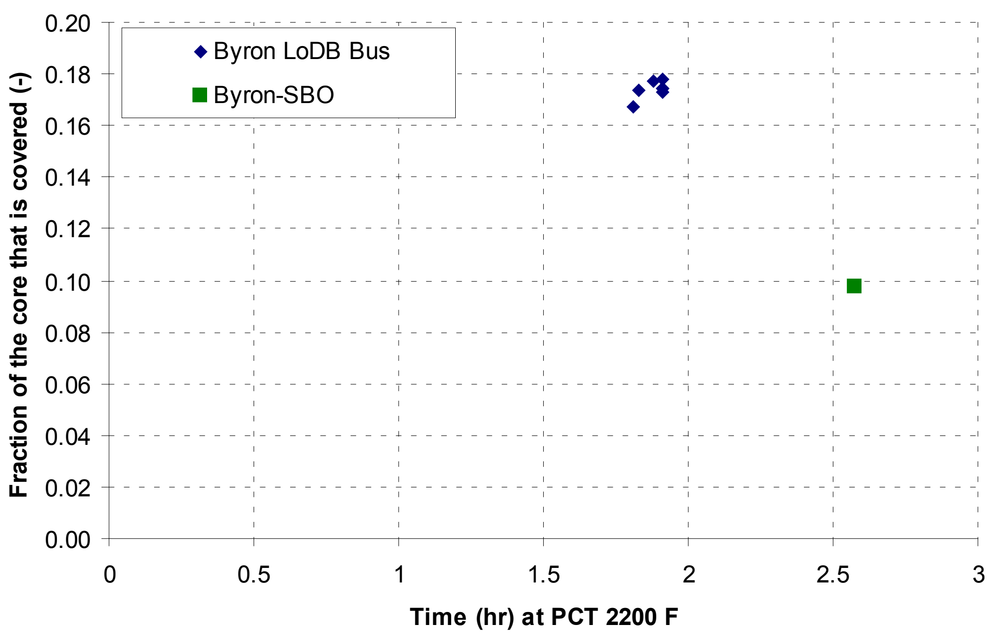

Figure 2.4.1 Normalized RPV Collapsed Water Level at the Time the PCT Reaches 1,478 K (2,200 °F)

The calculation method is similar to that of the starting time for core uncovery, except that the amount of water to be boiled off correspondingly increases by the amount between TAF and the considered equivalent water level in the core region. Additionally, it is assumed that the heat transferred to the water is proportional to the fraction of the core that is covered. As a consequence of this simple approach, the water level approaches the bottom of the active fuel (BAF) asymptotically.

The time (t_3) when PCT reaches 1,478 K (or actually, the time when the water level reaches the assumed equivalent level) can be obtained from the following balance between decay heat into the water and the energy removed by evaporation:

$$h_{fg} \frac{dm}{dt} = -Q_d(t) \frac{m(t)}{M} \tag{2.4.7}$$

where m(t) is the liquid mass currently in the active core region, M is the liquid mass initially in the active core region, h_{fg} is the heat of vaporization of water, and $Q_d(t)$ is the decay heat. Since the water mass m(t) is proportional to the water height z(t) in the core region (reckoned from the BAF), Equation (2.4.7) can be expressed in terms of water height z(t) as

$$h_{fg} \frac{dz}{dt} = -Q_d(t) \frac{z(t)}{M} \tag{2.4.8}$$

The solution of Equation (2.4.8) is

$$z(t) = (L_{TAF} - L_{BAF}) \exp\left(- \frac{\int_{t_2}^{t_3} Q(t)dt}{Mh_{fg}} \right) + L_{BAF} \tag{2.4.9}$$

where L_{TAF} and L_{BAF} are the elevation of the top and bottom of active fuel, respectively, and z(t) denotes the water level reckoned from the usual reference point (i.e., the inside bottom of the vessel); t_2 is the previously evaluated time of the beginning of core uncovery. On setting z(t) to the considered equivalent level (i.e., either 3.44 m or 3.72 m in the case of Byron), Equation (2.4.9) can be rearranged to obtain an expression for the time (t_3) it takes the PCT to reach 1,478 K (2,200 °F):

$$t_3 = \left[\frac{(1-b)Mh_{fg}}{aQ_0} \ln\left(\frac{L_{TAF} - L_{BAF}}{z(t) - L_{BAF}} \right) + t_2^{1-b} \right]^{1/(1-b)} \tag{2.4.10}$$

For the SBO scenario at the various plants, Table 2.4.5 compares the estimated times that core uncovery starts and the PCT attains 1,478 K (2,200 °F), which is implied by the assumed equivalent water levels. A minor variation arises due to the variability in the value of the equivalent water level (estimated as the level range spanned by the values defined in the Byron SBO versus the LoDC Bus scenarios).

Table 2.4.5 Estimated Times of Core Uncovery and PCT Attaining 1,478 K (2200 °F) for SBO Scenarios at the Byron/Braidwood, Callaway, Comanche Peak, and Vogtle Nuclear Power Plants

Parameters	**Unit**	**Byron/Braidwood**	**Callaway**	**Comanche Peak**	**Vogtle**
Time of core uncovery	min	116(Unit 1) (MELCOR, 117)	114	116 (Unit 1)	114
Time of PCT at 1,478 K (SBO equivalent level)	min	149 (MELCOR, 154)	147	148	146
Time of PCT at 1,478 K (LoDC Bus equivalent level)	min	141	139	140	138

The simple analysis under-predicts (in the Byron case) the amount of time when the PCT reaches 1,478 K (2,200 °F) by only a few minutes compared with the MELCOR prediction. The discrepancy may be due to an overestimation in the simple analytical approach of the decay heat available to boil off the water. As the core water level decreases to low levels in the MELCOR calculation, the decay heat results not only in boiling off the water but also in heating up structures such as fuel rods. On the other hand, the simple analysis assumes that all decay heat generated within the estimated covered section of the core goes only into the water. As a result, the analytical estimate shows an earlier prediction of when the PCT will reach 1,478 K (2,200 °F) than the MELCOR estimate does. However, the overall agreement is good. As such, this section has demonstrated that the simplified analytical approach gives useful core damage timing estimates when more detailed analyses are not available.

3. INVESTIGATION OF SEQUENCE VARIABILITY

3.1 Investigation of Sequence Variability—Byron Unit 1

3.1.1 Investigation of Sequence Variability – Byron Unit 1, Loss of DC Bus 111 with B&F

The transient initiator considered in this analysis is loss of DC bus 111. This bus powers the main feedwater regulating valves and their associated bypasses, which close upon the loss of DC power, so the initiator implies a loss of the main FW (MFW). Additional consequences of the loss of the bus include the unavailability of one of the two pressurizer PORVs, and all motor-driven auxiliary feedwater (AFW). Furthermore, the scenario definition assumes that diesel-driven AFW and charging (after an ECCS signal) are not available because of independent failure. The condenser steam dump valves are assumed to be available and are used while the main steam isolation valves (MSIVs) remain open to control the primary-side coolant temperature. (In the present instance, the MSIVs eventually close automatically due to their trip on MELCOR-predicted excessive rapid decrease of steam generator pressure following SG dryout and subsequent automatic condenser steam dump operation: see Section A.2.2.1 for more details.) As stated above, charging pumps are not available for ECCS injection. However, normal-operation injection by one of the charging pumps is accounted for initially after scram. One of two SI pumps and one of two RHR pumps are available[7]. A successful transition to the recirculation mode is assumed, but the RHR heat exchanger is assumed to be unavailable. To test the containment fan cooler effectiveness, only one out of the four containment fan coolers is available. There is no consideration of a possible discharge from the containment into the environment, whether from a building failure or from any operator-initiated venting. The operators are assumed to initiate B&F at 20 minutes. Table 3.1.1 summarizes conditions of the LoDC Bus 111 scenario. (Table A.1 in Appendix A also summarizes the availability of various systems.) This set of conditions usually leads to core damage in Level 1 PRAs.

Table 3.1.1 Conditions of the Loss of DC Bus 111 Scenario

Accident initiator	Loss of DC bus 111
Unavailable systems	FW of any type
	Accumulators (by PRA convention – RCS pressure doesn't permit injection)
	Charging pumps (in the ECCS configuration)[1]
	One pressurizer PORV
	One SI pump and one RHR pump
	Three containment fan coolers
	Containment sprays
Available systems	One charging pump aligned for normal makeup[1]
	One pressurizer PORV
	One SI pump and one RHR pump
	One fan cooler
	SG PORVs and dump valves
	RCPs
Operator actions	B&F[2]

1 One charging pump injects until the S signal is sent or the volume control tank depletes. The failure that causes loss of charging during ECCS alignment is also assumed to prevent automatic switchover from the VCT to the RWST during normal charging.
2 Bleed and feed consists of: tripping the RCP; opening the PRZ PORV; and manually sending the S signal.

[7] The high-pressure ECCS pumps at Byron are usually referred to as SI pumps, and sometimes as HPI pumps. This usage is in addition to the standard abbreviation, SI, for safety injection.

To investigate the variability of the predictions, eight sensitivity cases are defined in addition to the base case. Table A.2 in Appendix A shows how each sensitivity case differs from the base case.

In the base-case scenario (Case A.1.1; see Section A.2.1 of Appendix A) and in all associated sensitivity cases, reactor trip occurs at 40 seconds or earlier. The results show that the RCS pressure drop provided by opening one pressurizer PORV at 20 minutes is sufficient to allow one SI pump to begin to inject soon thereafter (at about 21 minutes). Later, however, in the base case and in all sensitivity cases, RCS pressure increases high enough to deadhead the pump. The return of higher pressures is due to the loss of steam generator cooling after the steam generators dry out. In the base case, the time when the SI pump is deadheaded is from 0.75 to 2.04 hours. Due to the lack of injection, the core partly uncovers and results in gross core damage (e.g., the relocation of fuel and support structures). In the base case, the temperature of the hottest cladding first exceeds 1,478 K (2,200 °F) at 1.91 hours.

Two sensitivity cases (A.1.2 and A.1.9) avoid gross core damage (e.g., material relocation). In Case A.1.2, an earlier reactor trip was prescribed at 11 seconds. The SI pump deadhead and partial core uncovery still occur and the core still heats up, but only such that the temperature of the hottest cladding attains a maximum of 1,123 K. Therefore, MELCOR does not predict any gap releases to occur. In Case A.1.9, the decay heat is uniformly reduced to 80 percent of its base-case value. Again, SI pump deadhead, partial core uncovery, and core heatup still occur, but the cladding temperature remains below 964 K, too low for gap releases or gross core damage to occur.

The other sensitivity cases, all of which result in core damage, investigate the following factors:

- the role of injection by a charging pump before the ECCS signal (Case A.1.3 discounts such injection),
- the influence of manual steam generator depressurization (Case A.1.4 imposes depressurization of all four steam generators at 10 minutes),
- the effect of a stuck-open steam generator relief valve (Case A.1.5),
- the effect of accumulator injection (Case A.1.6 credits all of the accumulators, which are discounted in the base case),
- the effect of additional fan coolers (Case A.1.7 credits all four fan coolers, whereas the base case credits only one),
- uncertainties due to different MELCOR code versions (Case A.1.8 repeats the base case using MELCOR version 2.1 instead of version 1.8.6).

These aforementioned factors entail little variation in the times when core damage occurs. See Section 4.1.1 and Appendix A for additional details.

Among all factors considered, early reactor trip and lower power level have the strongest effect.

3.1.2 Investigation of Sequence Variability - Byron Unit 1, Loss-of-Coolant During Shutdown

Section 2.1 of this report briefly describes a MELCOR model that represents the Byron plant during shutdown operations. The section describes a loss of reactor coolant event near the time of refueling, but while the RCS is still closed. The loop stop valves (which serve the same function as nozzle dams at other plants) are open, the RCS is full of water, and 24 hours have elapsed since the reactor shut down. The scenario considers a break outside of the containment in piping that connects an RHR pump to two cold legs. Initial temperatures around the RCS are 466.5 K (380F). The RHR pump not affected by the break is in use in a closed-loop configuration to remove the decay heat, which is 21.29 MW initially. The break flow area has been assigned so as to result in an initial break discharge rate of about 1,000 gpm (0.063 m^3/s). Table 3.1.2 below summarizes the conditions of the shutdown scenario. Table A.5 in Appendix A summarizes the availability of various systems.

Table 3.1.2 Conditions of the Shutdown Scenario

Mode	Analogous to hot shutdown conditions	
RCS pressure and temperature	1.35 MPa (196 psi); 466.5 K (380°F)	
RCS water level	Vessel, all piping, and steam generator tubes full; pressurizer nearly full (1 m^3 steam bubble)	
Availability of steam generators	SG loops	Filled
	SG secondary side water level/pressure/temperature	~normal operating level/1.35 MPa (196 psi) / 466.5 K (380°F)
	Loop stop valves	Open
	MFW/AFW	Not available
Availability of other relevant structures, systems, and components	Sump	Available (not drained)
	RWST	Available (not drained)
RHR train alignments	1 train aligned for decay heat removal, 1 train aligned for ECCS LPI	
Planned refueling outage versus forced outage	Analogous to a planned outage	
Before versus after refueling	Before	

To investigate the variability of the predictions, thirteen sensitivity cases are defined in addition to the base case. Table A.6 in Appendix A shows how each sensitivity case differs from the base case.

In the base case scenario (Case A.2.1; see Section A.3.1 of Appendix A), no operator actions are credited, which leads to core damage due to the RHR pump's loss of suction as the hot leg is depleted due to the loss of coolant inventory. Core damage is calculated to initiate at 3.34 hours, according to the measure that associates core damage with the temperature of the hottest cladding exceeding 1,478 K (2,200 ° F).

Variants of the base case that also reach core damage consider the following:

- A modified treatment of the behavior of the RHR pump on very low hot leg water levels (Case A.2.4 predicts core damage 0.13 hours earlier on the assumption that the pump trips and cannot be recovered),

- An attempt to bring about more steam generator cooling, contrary to proceduralized actions (Case A.2.6 initiates a slow secondary-side depressurization starting at 0.5 hours and predicts core damage 1.12 hours earlier; it is included here to show the negative effect of this action but it is acknowledged that the accuracy of the simulation is affected by the assumed initial conditions; see Section A.3.2.6),

- Break flows of different size (Case A.2.7a reduces the break flow area to 75 percent of the base-case value and predicts core damage 0.37 hours later; Case A.3.7b reduces the break flow area to 10 percent of the base-case value and predicts core damage at 11.38 hours; Case A.3.7c reduces the break flow area to 5 percent of the base-case value and predicts core damage at 23.92 hours; Case A.3.7d increases the break flow area to 500 percent of the base-case value and predicts core damage at 39.8 minutes),

- Lower decay heat (Case A.2.8 reduces the decay heat to 75 percent of the base-case value[8] and predicts core damage 0.12 hours later),

- Different break location (Case A.2.10 considers a hot leg break and predicts core damage 2.25 hours later),

- Initially sub-cooled RCS (Case A.2.11 starts the transient from water-solid, pressurized primary-side conditions and predicts core damage 0.13 hours later).

In the above bullets, the times to core damage refer to the measure that associates core damage with the temperature of the hottest cladding region exceeding 1,478 K (2,200 °F).

Case A.2.2 avoids gross core damage on the assumption that operators initiate ECCS injections at 0.5 hours, with one charging pump and one SI pump taking suction from the reactor water storage tank (RWST) that is assumed to be refilled as necessary. Core uncovery is not prevented (the time that the top third of the core is first uncovered is 2.14 hours). However, gap releases and the relocations of fuel or structures are predicted not to occur.

Variants of Case A.2.2 that also avoid gross core damage consider the following:

- A later time for the same actions (Case A.2.3 credits the actions to occur at 1.0 hour),

- A modified treatment of the behavior of the RHR pump on very low hot leg water levels (Case A.2.5 applies the same treatment considered in Case A.2.4 to Case A.2.2), and

- Available switchover to the ECCS sump recirculation mode (Case A.2.9 reinterprets the break as located inside the containment, so that water accumulates in the sump).

See Section 4.1.2 and Appendix A for additional details. In summary, break size and location, as well as operator actions, are found to have the largest effect on the results.

Note that the general applicability of these findings is subject to some limitations. In this report, the shutdown modeling is considered a first foray into this subject and consequently, there is a preference that the shutdown model should not depart too much from the main, full-power model[9]. For this reason, a phase in the shutdown operations is considered in which, for example, the head is still on the vessel. Considerations of risk importance might have led to other choices for a shutdown scenario, but risk importance is not central to the goals of this report. In the general time frame under consideration, actual shutdown operations would likely have water-solid conditions in the RCS, with a pressure well above the saturation pressure. Because of anticipated possible numerical difficulties related to water incompressibility, the adopted model is set up with saturated conditions and with a steam bubble volume of 1 m^3 in the pressurizer. The impact of this assumption is diminished by the fact that the considered scenario includes a break, which tends to bring the RCS to saturation quickly, even if it were initially subcooled. This is demonstrated by a sensitivity case that relies on a fictitious cover of pressurized, non-condensable gas to provide pressure compliance before the start of the transient. The base-case break, which connects two cold legs to the environment via a broken RHR injection line, may not be realistic (because of certain check valves inside the containment that the model neglects). A sensitivity case

[8] This initial decay heat is quite similar to the decay heat at 72 hours, which is estimated as 15.85 MW. Case A.2.8 therefore serves as an approximation to a modified scenario in which the transient begins at 72 hours. However, the present treatment with a uniform reduction of the decay heat by a constant factor is not exactly equivalent to evaluating the unmodified decay heat curve at any later times, and the assumed initial conditions (e.g., RPV head still in place) would likely not actually be applicable at 72 hours.

[9] Additional MELCOR shutdown analyses are being performed by NRC for Byron Unit 1, and are planned for Vogtle Units 1 & 2.

where the break is from a hot leg leads to results qualitatively similar to those of the base case, but core damage occurs somewhat later.

Operator actions are considered to be large risk drivers during a shutdown, and knowing how much time the operator has to act is important. The timing of the operator actions considered in these calculations have been set plausibly, but not with any attempt to establish how late these actions can be taken and still lead to the aversion of core damage. Establishing the latest times for successful operator actions for various plants and various scenarios is, in fact, the subject of Section 5 of this report. That section considers a Byron shutdown scenario of the SBO scenario type, not the loss-of-coolant accident (LOCA) scenario considered here. Timing calculations for the scenario considered here could be similarly carried out.

3.2 Investigation of Sequence Variability – Surry, Steam Generator Tube Rupture (SGTR)

The accident sequence (sometimes referred to herein as SGTR-9) is initiated by a steam generator tube rupture (SGTR). The sequence ends in core damage following failures to initiate a primary- and secondary-side cooldown, refill the RWST, and provide long-term heat removal. (See Figure B.1 in Appendix B for an event tree that defines SGTR-9 and other possible sequences that may result from a SGTR accident.) The MELCOR model includes the following:

- An initiating SGTR at time t = 0.
- Scram, automatic MSIV closure, MFW trips, and RCP trips from cavitation at model-predicted times (note that procedurally-directed trip of all RCPs early in the transient is not modeled).
- Available high- and low-head safety injection (HHSI, LHSI) and containment sprays with the RWST as the source.
- Available turbine-driven and motor-driven AFW with the CST as the source.
- Operator actions to isolate the faulted steam generator, occurring at a hard-wired time (17.88 minutes).
- No provision for steam generator cooldown (e.g., by controlled depressurization).
- Recirculation mode unavailable for all systems.
- No refilling of the RWST or CST.

The base-case calculation also assumes no termination or control of HHSI by operators, no B&F, and no RHR. However, some of these actions are considered in the sensitivity cases. Table 3.2.1 summarizes conditions of the SGTR scenario.

Table 3.2.1 Conditions of the SGTR Scenario

Accident initiator	Double-ended rupture of one steam generator tube
Unavailable systems	Refilling of the RWST or CST
	Switchover to the ECCS recirculation mode
Available systems	Turbine-driven and motor-driven AFW with CST as the source
	HHSI, LHSI, and containment sprays with RWST as the source
	RCPs (until tripped by cavitation) – procedurally-directed trip early in the transient not modeled
Operator actions	Isolation of one of three high-head injection pumps (in the base case); B&F and/or water-level-dependent control of SI (in sensitivity cases)[1]

1 Cases for controlling HHSI above TAF (i.e., below the normal pressurizer level) and B&F are counter to EOPs for the prescribed conditions. They are included for illustrative purposes, though they could be viewed as representing errors of commission.

To investigate the variability of the predictions, twenty sensitivity cases are defined in addition to the base case. Table B.2 in Appendix B shows how each sensitivity case differs from the base case.

In the base-case scenario (Case B.1; see Section B.2.1 of Appendix B), core damage occurs at about 38 hours (e.g., 37.90 hours according to the surrogate measure that identifies core damage with the PCT exceeding 2,200 °F [1,478 K]).

Core damage is much earlier (5.1 to 7.8 hours by the PCT >2,200 °F [1,478 K] measure) in the sensitivity cases that lead to a stuck-open or held-open pressurizer PORV. A stuck-open PORV occurs in the half-tube break case (Case B.3) because the small break does not depressurize the vessel, which leads to excessive PORV cycling. Several cases (B.17 through B.20) assume held-open PORVs as one of the conditions that would be established by operators as part of initiating a B&F (which would be an error of commission in this scenario, and are for illustrative purposes only). Whether the PORV is stuck open or is held open, the resulting steam flow to the containment actuates the sprays and consequently, the RWST is depleted more quickly.

The sensitivity cases that result in the actual reduction of the HHSI rate (Cases B.5, B.6, and B.16) lead to longer times to core damage, because depletion of the RWST is postponed. Note that Cases B.17 through B.20 are not included in this category, although water-level-dependent control of the HHSI is credited. The injection rate tends to be higher than in the base case because of the held-open PORV.

The one case that avoids core damage (Case B.16, with no core damage as of 72 hours) assumes that the rate of HHSI is no more than is required to maintain the core water level somewhat above the TAF, and is included for illustrative purposes.

The cases that consider a five-tube break (B.2, B.10, B.11, and B.15) give intermediate times to core damage (around 23 hours) due to lower vessel pressures (i.e., relative to the base case), which lead to higher HHSI rates and earlier RWST depletion.

These results show that specific boundary condition assignments dominate the sources of variability for the analyzed scenario. These assignments include the number of tubes that ruptured, operator action to initiate a B&F, and operator action to maintain the core water level just above TAF to conserve the RWST inventory. The importance of the latter two specific items is dominated by the scenario's assumption that recirculation is unavailable. Therefore, the key conclusion is that specific boundary condition assignments drive the results, and that which specific boundary condition assignments are important depends on the specifics of the scenario. The importance of one or another boundary condition, depending on the scenario, is in contrast to a situation where all boundary condition specifications contribute relatively equally to the variability of the results.

The times to core damage range from 4.20 to 43.03 hours; they depend on the sensitivity case and on the core damage surrogate measure. This range excludes one sensitivity case where no core damage is indicated as of 72 hours (Case B.16, with a successful operator initiation of the water-level-dependent control of HHSI).

With very few exceptions, the order that the various surrogate measures indicate core damage is independent of the sensitivity case. In most cases, the final measure (PCT temperature above 3,000 °F [1,922 K]) is indicated around two hours after the first measure (one-third of the core uncovered). See Section 4.2 and Appendix B for additional details.

3.3 Investigation of Sequence Variability – Peach Bottom, SBO with RCIC until Battery Depletion

NUREG-1953 [1] provides a range of SBO calculations for Peach Bottom, considering several injection sources. Here, cases where RCIC is available are investigated further. The transient initiator is a loss of all onsite power except for batteries, which are credited to support only the operation of the RCIC system. No system is credited to function except the RCIC system, which is available for 8 hours, the assumed time when the batteries are depleted. In particular, the RHR pumps, low-pressure core sprays, and service water are assumed to be unavailable. It is also assumed that FW, recirculation pumps, and the control rod drive (CRD) hydraulic system fail at the start of the transient. (These last three systems do, however, operate during the pre-transient steady state, which has a duration of 120 seconds.) The automatic depressurization system (ADS) is also assumed to be unavailable due to the power loss or, alternatively, because the operators act to inhibit ADS activation so that the RCIC system may be used. It is also assumed that the safety relief valves (SRVs) will operate indefinitely (i.e., it is assumed that the SRVs do not fail due to excessive cycling). Table 3.3.1 summarizes conditions of this SBO/RCIC scenario.

Table 3.3.1 Conditions of the SBO/RCIC Scenario

Accident initiator	SBO
Unavailable systems	All but the RCIC
Available systems	RCIC until the battery is depleted
Operator actions	None

To investigate the variability of the predictions, ten sensitivity cases are defined in addition to the base case. Table C.2 in Appendix C shows how each sensitivity case differs from the base case. In the base case scenario (Case C.1; see Section C.2.1 of Appendix C), the unavailability of the RCIC after 8 hours– the assumed battery depletion time–results in core damage starting at 10.99 hours (according to the criterion of the PCT exceeding 1,478 K [2,200 °F]). Two sensitivity cases (C.4 and C.5) are identical to the base case, except for the assumption that battery depletion occurs at 2 hours and 5 hours, respectively. Core damage (according to the same criterion) occurs in these cases at 4.27 and 8.03 hours, respectively.

One sensitivity case (C.2) considers that the RCIC may continue to run in an uncontrolled mode after 8 hours, until it shuts off due to flooding in the steam lines at 13.87 hours. In this case (according to the same criterion), core damage occurs at 18.57 hours.

In the base case (C.1), the SRV of the lowest opening pressure opens and closes a total of 1,426 times as of the end of the calculation (at 11.9 hours). In a sensitivity case (C.3), this valve remains stuck open after it has opened 187 times. This event occurs at 2.45 hours. According to the same criterion, core damage occurs at 7.15 hours, the RCIC having become unavailable at 3.88 hours as a result of low vessel pressure stemming from the stuck-open valve.

Another sensitivity case (C.7) assumes that the SRV with the lowest opening pressure is failed in the closed position throughout the calculation. The results are similar to those of the base case because the vessel pressure is relieved by the other SRVs whose opening pressures are only slightly higher. In a sensitivity case that uniformly reduces the decay heat to 90 percent of the base-case value (C.6), core damage occurs at 11.58 hours according to the same criterion.

A sensitivity case (C.8) that assumes a reduction of 10 percent in the maximum flow rate of the RCIC pump leads to a delay of about 0.6 hours in the time of core damage. A case that considers a lower initial temperature of the water in the CST (C.11) leads to a delay of about 0.4 hours. Sensitivity cases that consider adjustments of some parameters that affect the flow through the SRVs (C.9) and the correlation for heat transfer coefficients during boiling (C.10) give results not significantly different from those of the base case. See Section 4.3 and Appendix C for additional details.

3.4 Investigation of Sequence Variability – Timing of Key Events

Table 3.4.1 summarizes the timing of key events for the four basic scenarios at the three plants. For each event, the Table provides a range of times. The range is defined by the earliest and latest times of occurrence predicted by the base and sensitivity cases pertaining to the scenario. The ranges for the Surry scenario do not include results from cases 16 through 20. Some events are not relevant to some scenarios; these are marked N/A. Some relevant events do not occur (e.g., core damage in cases where mitigation actions are successful). For such non-occurring but relevant events, a footnote provides the time of the end of the run. See Appendices A, B, and C for additional details.

Table 3.4.1 Summary of the Timing of Key Events for the Base- and Sensitivity-Case Calculations

Event[1]	**Byron LoDC Bus 111**	**Byron Shutdown**	**Surry SGTR** [2]	**Peach Bottom SBO**
First opening of the pressurizer PORV or SRV	0.13 – 0.33 [3]	N/A	0.33 [4]	0.01 [5]
Number of pressurizer or vessel PORV or SRV cycles at time of core damage (or end of run) [6]	1 – 49 [3]	N/A	495 [4]	187 – 1594 [5]
Pressurizer first goes water-solid	0.42 [7]	0.00 [8]	0.57 – 6.10 [9]	N/A
RCIC becomes unavailable	N/A	N/A	N/A	2.00 – 13.87
Steam generator dry out [10]	0.28 – 1.83	N/A	30.8 – 35.4 [9]	N/A
First time water level is below TAF	1.28 – 2.52	0.19 – 16.81	6.65 – 39.30	3.30 – 16.87
First time when SI pump is deadheaded, subsequent to actual injection	0.30 – 2.06	N/A	N/A	N/A
Reduced pumping rate of RHR pump A due to low water level at suction point	N/A	0.18 – 15.52	N/A	N/A
Pressurizer empties	Does not occur	0.34 – 14.59	0.01 – 9.33 [11]	N/A
Zero pumping rate of RHR pump A due to dry out of suction point	N/A	0.36 – 18.92 [12]	N/A	N/A
RWST depletion	7.23 – 7.74 [13]	2.96 – 3.46 [14]	3.56 – 11.37	N/A
RCPs trip due to cavitation	N/A	N/A	4.23 – 14.97	N/A
CST depletion	N/A	N/A	17.97 – 32.12 [11]	N/A
Time of core damage [6]	1.81 – 1.92 [15]	0.67 – 23.92 [12]	7.77 – 42.80	4.27 – 18.57

1 Times in hours. The ranges are defined by the results obtained from all cases. The symbol N/A indicates that the event is irrelevant or not considered.
2 Results from cases 16 through 20 are not considered.
3 In all 9 cases, PORV1 is opened manually at 20 minutes, and remains open afterwards. In two cases only, it also opens and closes automatically earlier.
4 This event occurs in only one of the 16 cases. In this case, the valve sticks open on cycle #495.
5 Opening time (30 seconds) is the same for all 11 cases. The lower bound (187 cycles) occurs in a case that considers that the valve sticks open.
6 Time of core damage according to the surrogate that associates core damage with the PCT in excess of 1,478 K.
7 In all cases, the non-liquid volume fraction falls to below about $1 \cdot 10^{-4}$ at about 25 minutes.
8 This event occurs in only one of the 14 cases. In that case, the condition (solid pressurizer) applies at the start of the transient.
9 The longest-running calculation in which this event does not occur ends at 23 hours.
10 Byron: first time when all steam generators are dry. Surry: first time when the broken steam generator is dry.
11 The longest-running calculation in which this event does not occur ends at 8 hours.
12 The longest-running calculation in which this event does not occur ends at 10 hours.
13 Only three cases were run long enough to attain this event.
14 This event occurs and is possible only in four cases.
15 The longest-running calculation in which this event does not occur ends at 28 hours.

4. INVESTIGATION OF CORE DAMAGE SURROGATES

A number of criteria can be set for determining the time of core damage, which may be convenient for computer simulations versus an actual accident. Some have been defined for regulatory purposes while others have different technical origins. In order to estimate how much variability in the time of core damage may arise from the definition of the criterion, thirteen surrogate criteria are defined for these studies. The definitions of the core damage surrogates listed in Table 4.1 are based on experience with MELCOR models as are informed by regulatory guidance such as NUREG–1465 (Reference [5]) and the current (2012) revision of 10 CFR 50.46 (Reference [6]).

Table 4.1 Core Damage Surrogates

No.	Surrogate	Origin
1.	Core water level drops below the TAF	A traditional surrogate used in some PRAs for PWRs
2.	Core water level drops below 2/3 of active fuel height	
3.	Core water level drops below 1/3 of active fuel height	A traditional surrogate used in some PRAs for BWRs
4.	Peak local cladding temperature exceeds 1,200 ºF (922 K)	A surrogate commonly used if a particular computer code (or code version) is known to under-predict core heatup during core uncovery
5.	Core-exit coolant temperature exceeds 1,200 ºF (922 K)	Core exit temperature reaching 922 K is used as a transition criterion in some Severe Accident Management Guidelines (SAMGs)
6.	Finite release of noble gases from fuel	Equivalent to a PCT of 1,173 K for the MELCOR inputs used in these analyses
7.	Cumulative hydrogen generation exceeds 1% of potential (i.e., amount of hydrogen corresponding to steam-oxidation of all zirconium in the core)	An ECCS acceptance criterion defined in 10 CFR 50.46(b)(3) (as of 2012)
8.	Release of cesium group from the fuel exceeds 3% of the initial inventory	NUREG–1465 assigns 3% as the gap release fraction for PWRs and BWRs when long-term core cooling is maintained (i.e., accidents not involving degraded or molten core conditions)
9.	Peak local cladding temperature exceeds 2,200 ºF (1,478 K)	An ECCS acceptance criterion defined in 10 CFR 50.46(b)(1) (as of 2012)
10.	Maximum thickness of cladding oxide layer exceeds 17% of cladding thickness	An ECCS acceptance criterion defined in 10 CFR 50.46(b)(2) (as of 2012)
11.	Peak local cladding temperature exceeds 2,500 ºF (1,644 K)	
12.	Peak local cladding temperature exceeds 2,876 ºF (1,853 K)	The oxidation kinetics transition temperature based on the Urbanic-Heidrick oxidation model
13.	Peak local cladding temperature exceeds 3,000 ºF (1,922 K)	

A few technicalities, some of which are plant-specific, concerning the evaluation of some of the surrogate measures are noted in the next paragraphs. Those that are plant-specific are discussed below in the sub-section that pertains to the particular plant. The technicalities that do not depend on the plant are discussed here, as follows:

- For metrics stemming from 10 CFR 50.46, it is important to note that the order in which these surrogates are reached differs from the order typically seen for large break LOCA evaluations for three reasons. First, the present analyses use the severe accident code MELCOR, rather than a code specifically designed for a design-basis LOCA analysis, and these classes of tools have different modeling assumptions associated with fluid flow, heat transfer, and oxidation modeling that are a function of their intended end-use. Second, the present analyses use best-estimate modeling inputs, rather than conservative modeling inputs typical of some 10 CFR 50.46 evaluations. Third, the present analyses investigate small LOCAs and transients, which evolve over longer timescales and therefore have different heatup characteristics compared to large break LOCAs. In summary, the results of the present analyses are representative of the characteristics of typical PRA accident sequences of interest, but should not be attributed to 10 CFR 50.46 evaluations.[10]

- Whereas in these MELCOR simulations surrogate #6 (finite release of noble gases from fuel) is exactly equivalent to the PCT attaining 1,173 K (i.e., gap releases occur according to this temperature condition in MELCOR), surrogate #8 (release of radionuclides of the cesium group from the fuel exceeds 3 percent of the initial inventory) depends on the details of MELCOR's modeling of radionuclide releases from fuel and on user-specified, plant-specific inputs. Examples of inputs include the relative inventories in the fuel versus in the gap and the treatment of compound classes such as cesium iodide and cesium molybdate.

- The transition from slow to rapid oxidation based on the Urbanic-Heidrick correlation occurs at a temperature where a change in the zirconium oxide microstructure is expected. This transition occurs at 1,853 K (2,876 ºF). From the point of view of the surrogate, this transition simply provides a characteristic temperature to use as the setpoint for the surrogate.

- The measure for maximum thickness of the cladding oxide layer is actually calculated using the ratio of the mass of zirconium oxide to that of zirconium. This approach is therefore equivalent to a measure based on a layer thickness only while zirconium oxide forms only on the outside of the fuel. (Note that the relevance of this remark comes from a requirement of 10 CFR 50.46, which states that if the cladding ruptures, then the inside surfaces should be included in the oxidation.)

4.1 Investigation of Core Damage Surrogates – Byron Unit 1

A few plant-specific technicalities concerning the evaluation of some of the measures are noted in the next paragraphs.

For the surrogates based on core water level, the level is defined by summing the heights of the collapsed water columns in the lower plenum, core channel and upper plenum (where the second of five computational rings is used), and the dome.

In the current Byron model, more than 3 percent of the initial cesium class inventory is assigned to the fuel rod gap. Consequently, the cesium release and the noble gas class release surrogates are reached simultaneously.

[10] Note that 10 CFR 50.46 focuses on core coolability. Radiological considerations associated with design-basis accidents are addressed elsewhere in the regulations (e.g., 10 CFR 100). Also note that ECCS acceptance criteria defined in 10 CFR 50.46 are currently the subject of rulemaking efforts.

The measure for maximum thickness of the oxide layer on the cladding is actually calculated using the ratio of the mass of zirconium oxide to that of zirconium. Moreover, because this model does not provide the time-dependent masses of the individual components (such as cladding), the net core-region masses of zirconium oxide and zirconium are used instead of the cladding masses specifically. These net masses include contributions from, for example, the guide tubes.[11]

The surrogate referring to hydrogen generated by zirconium oxidation is actually implemented in terms of the mass of hydrogen generated by oxidizing zirconium and steel.

4.1.1 Investigation of Core Damage Surrogates – Byron Unit 1, Loss of DC Bus 111 with B&F

Table 4.1.1 and Figure 4.1.1 show the times to core damage according to various surrogate measures for the Byron LoDC Bus 111 B&F base and sensitivity cases. Refer to Section 3.1.1 (and also Appendix A) for the sensitivity case definitions and to Section 4 for definitions of the surrogate measures. Table 4.1.2a provides statistical data for the time when core uncovery starts. Table 4.1.2b does the same for the time when the PCT first exceeds 1,478 K (2,200 °F). The definitions of the bins are based on timing similarities and not on any logical similarities among the case definitions. See Appendix A for additional details.

4.1.2 Investigation of Core Damage Surrogates – Byron Unit 1, Shutdown

Table 4.1.3 and Figure 4.1.2 show the times to core damage according to various surrogate measures for the Byron Shutdown base and sensitivity cases. Refer to Section 3.1.2 (and also Appendix A) for the sensitivity case definitions and to Section 4 for the definitions of surrogate measures. Table 4.1.4a provides statistical data for the time when core uncovery starts. Table 4.1.4b does the same for the time when the PCT first exceeds 1,478 K (2,200 °F). The definitions of the bins are based on timing similarities and not on any logical similarities among the case definitions. See Appendix A for additional details.

4.2 Investigation of Core Damage Surrogates—Surry, SGTR

Table 4.2.1 and Figure 4.2.1 show the times to core damage according to various surrogate measures for the Surry SGTR base and sensitivity cases. Refer to Section 3.2 (and also Appendix B) for the sensitivity case definitions and to Section 4 for the definitions of surrogate measures. Table 4.2.2a provides statistical data for the time when core uncovery starts. Table 4.2.2b does the same for the time when the PCT first exceeds 1,478 K (2,200 °F). The definitions of bins are based on timing similarities and not on any logical similarities among the case definitions. See Appendix B for additional details.

A few plant-specific technicalities concerning the evaluation of some of the measures are noted in the next paragraphs.

For the surrogates based on core water level, the level is defined by summing the heights of the swollen water columns in the lower plenum, the core channel, and the upper plenum. Each of the five computational rings provides a water level; the minimum level is then used to evaluate the surrogates.[12]

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[11] The base case of the Loss of DC Bus 111 scenario has been repeated with no change except for the definition of a control function equal to this measure and calculated using the masses peculiar to the cladding component. The difference, relative to the standard post-run analysis that uses the lumped masses of zirconium components of all type, is less than 1 minute for the indicated time of core damage (see Figure A.1.1.13 and related discussion in Appendix A).

[12] This approach to defining the level for Surry generally follows the approach suggested by a built-in control function in the SNL-provided deck, especially with regard to the minimization over rings. The built-in control function, however, extends the sum only as high as the top of the channel, and so produces spuriously early indications of the time when the level falls below the TAF. See more discussion in Appendix B.

Table 4.1.1 Times of Core Damage (in Hours) According to Various Surrogate Measures Compared Among Base and Sensitivity Cases of Byron LoDC Bus 111 B&F Scenario

Case	Water Level Below TAF	Water Level Below 2/3 AF	Water Level Below 1/3 AF	PCT > 922K	CET > 922K	Finite Xe Release	Hydrogen Generation > 1% Potential	Cesium Release > 3% Inventory	PCT > 1478K	Maximum Oxide Depth > 17% Clad Thickness	PCT > 1644K	PCT > 1853K	PCT > 1922K
A.1.1 (Base)	1.35	1.47	1.58	1.66	1.68	1.81	1.88	1.81	1.91	1.94	1.92	1.93	1.94
A.1.2	2.52	2.82	3.03	3.03	3.06	–	–	–	–	–	–	–	–
A.1.3	1.28	1.40	1.51	1.59	1.62	1.73	1.81	1.73	1.83	1.87	1.85	1.86	1.87
A.1.4	1.28	1.39	1.50	1.58	1.60	1.72	1.79	1.72	1.81	1.85	1.83	1.84	1.85
A.1.5	1.32	1.44	1.56	1.63	1.66	1.78	1.85	1.78	1.88	1.92	1.90	1.91	1.92
A.1.6	1.35	1.47	1.58	1.66	1.68	1.81	1.88	1.81	1.91	1.94	1.92	1.93	1.94
A.1.7	1.35	1.48	1.57	1.66	1.68	1.81	1.88	1.81	1.91	1.94	1.93	1.94	1.95
A.1.8	1.35	1.46	1.58	1.66	1.68	1.82	1.91	1.82	1.92	1.95	1.93	1.94	1.94
A.1.9	2.12	2.38	2.63	2.65	–	–	–	–	–	–	–	–	–

Table 4.1.2a Bins and Statistics for Times to Start of Core Uncovery, Byron LoDC Bus 111 B&F Scenario

Bin	**Cases**	**Important Deviations from Base Case**	**Range of Time to TAF (hour)**	**Within-Bin Statistics for Time to TAF: Mean ± One Standard Deviation (hour)**
1	All but A.1.2 and A.1.9	Injection by a charging pump before S signal discounted (A.1.3), depressurized steam generators (A.1.4 and A.1.5), accumulators credited (A.1.6), all fan coolers credited (A.1.7), MELCOR version (A.1.8)	1.28 – 1.35	1.33 ± 0.03
2	A.1.2, A.1.9	Early scram (A.1.2), reduced decay heat (A.1.9) (in both cases, there are no relocations or gap releases)	2.12 – 2.52	2.32 ± 0.20

Table 4.1.2b Bins and Statistics for Times for PCT to Exceed 1,478 K (2,200 °F), Byron LoDC Bus 111 B&F Scenario

Bin (defined in Table 4.1.2a)	**Range of Time to 1,478 K (hour)**	**Within-Bin Statistics for Time to 1,478 K: mean ± one standard deviation (hour)**
1	1.81 – 1.92	1.88 ± 0.04
2	N/A (does not occur)	N/A

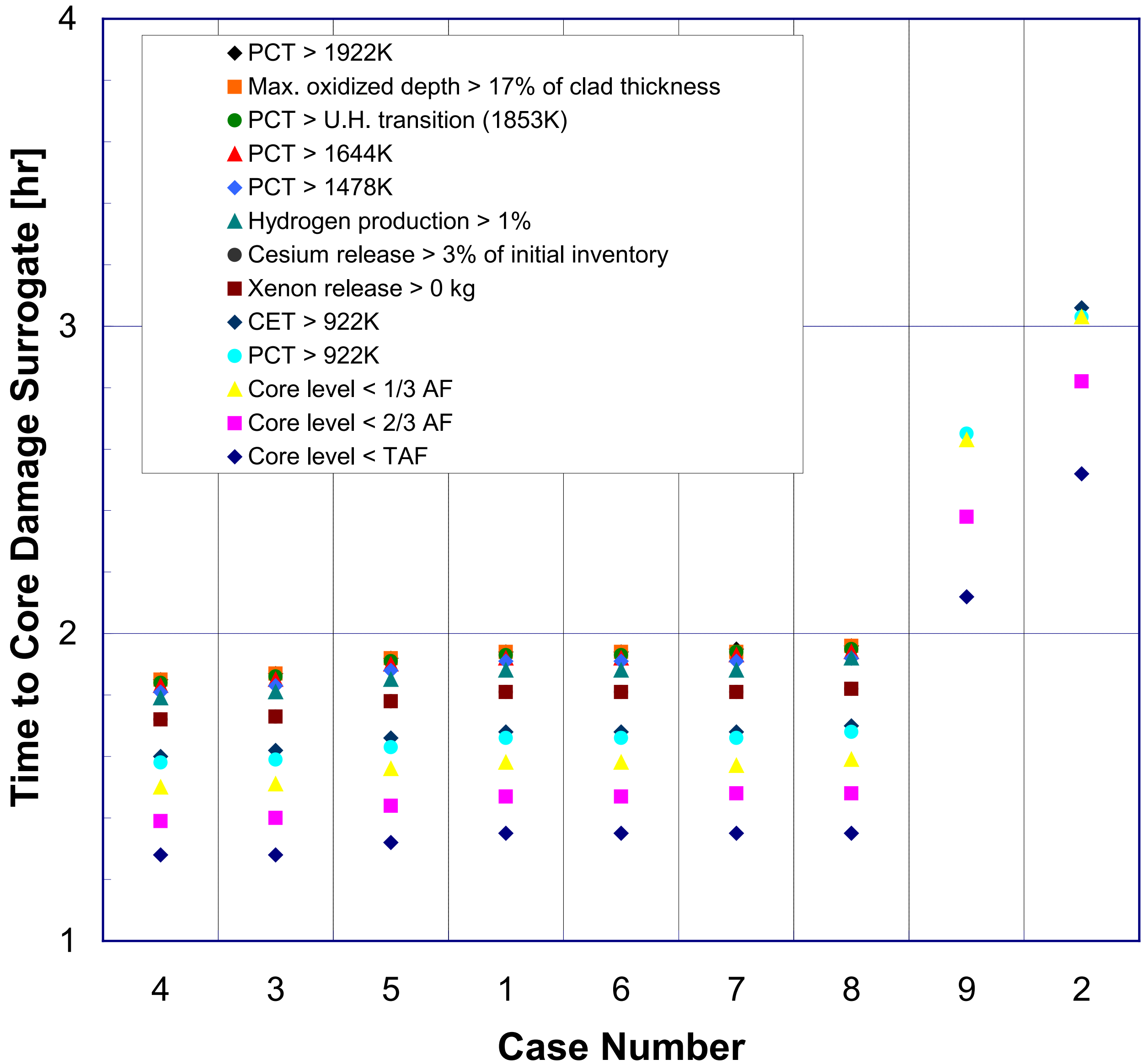

Figure 4.1.1 Times of Core Damage in Byron LoDC Bus 111 B&F Scenario and its Sensitivity Cases, According to Various Surrogate Measures[13]

[13] The case numbers that label the horizontal axis omit the initial prefixes (e.g., "4" in the axis label denotes case A.1.4). Gross core damage is avoided in cases A.1.9 and A.1.2.

Table 4.1.3 Times of Core Damage (in hours) According to Various Surrogate Measures Compared Among Base and Sensitivity Cases of Byron Shutdown Scenario

Case	Water Level Below TAF	Water Level Below 2/3 AF	Water Level Below 1/3 AF	PCT > 922K	CET > 922K	Finite Xe Release	Hydrogen Generation > 1% Potential	Cesium Release > 3% Inventory	PCT > 1478K	Maximum Oxide Depth > 17% Clad Thickness	PCT > 1644K	PCT > 1853K	PCT > 1922K
A.2.1 (Base)	0.75	0.79	3.12	2.87	2.93	3.16	3.32	3.16	3.34	3.55	3.37	3.39	3.40
A.2.2	1.59	2.14	–	–	–	–	–	–	–	–	–	–	–
A.2.3	0.75	0.79	–	–	–	–	–	–	–	–	–	–	–
A.2.4	0.75	0.79	2.95	2.22	2.52	3.08	3.19	3.08	3.21	3.41	3.24	3.26	3.27
A.2.5	1.59	2.52	–	–	–	–	–	–	–	–	–	–	–
A.2.6	0.79	0.86	2.02	1.76	1.83	2.04	2.23	2.04	2.22	2.56	2.26	2.27	2.28
A.2.7a	1.01	1.06	3.46	3.20	3.27	3.56	3.69	3.56	3.71	3.92	3.73	3.75	3.76
A.2.7b	7.87	8.23	9.93	9.38	9.62	10.17	10.36	10.17	11.38	11.36	11.42	11.45	11.46
A.2.7c	16.81	17.43	19.54	19.41	19.50	23.10	20.81	23.10	23.92	23.73	23.96	24.00	24.03
A.2.7d	0.19	0.21	0.36	0.50	0.59	0.59	0.67	0.59	0.67	0.74	0.70	0.74	0.75
A.2.8	0.80	0.85	3.11	2.91	2.98	3.31	3.44	3.31	3.46	3.71	3.49	3.52	3.53
A.2.9	0.76	0.81	–	–	–	–	–	–	–	–	–	–	–
A.2.10	3.61	4.14	5.36	5.06	5.13	5.40	5.56	5.40	5.59	5.84	5.62	5.64	5.65
A.2.11	0.77	0.81	3.24	2.99	3.05	3.29	3.44	3.29	3.47	3.68	3.50	3.52	3.53

Table 4.1.4a Bins and Statistics for Times to Start of Core Uncovery, Byron Shutdown Scenario

Bin	**Cases**	**Important Deviations from Base Case**	**Range of Time to TAF (hour)**	**Within-Bin Statistics for Time to TAF: Mean ± One Standard Deviation (hour)**
1	A.2.1 (base), A.2.4, A.2.6, A.2.7a, A.2.8, A.2.11	Modified treatment of RHR pump as suction is lost (A.2.4), steam dump valve opened (A.2.6), reduced break flow (A.2.7a), reduced decay heat (A.2.8), initially water-solid RCS (A.2.11)	0.75 – 1.01	0.81 ± 0.09
2	A.2.2, A.2.3, A.2.5, A.2.9	Operator recovery of ECCS (no relocations or gap releases)	0.75 – 1.59	1.17 ± 0.42
3	A.2.7b	Reduced break flow	7.87	–
4	A.2.7c	Reduced break flow	16.81	–
5	A.2.7d	Increased break flow	0.19	–
6	A.2.10	Break from a hot leg	4.14	–

Table 4.1.4b Bins and Statistics for Times for PCT to Exceed 1478K (2200 °F), Byron Shutdown Scenario

Bin (defined in Table 4.1.4a)	**Range of time to 1478K (hour)**	**Within-bin statistics for time to 1478K: mean ± one standard deviation (hour)**
1	2.22 – 3.71	3.24 ± 0.48
2	N/A (does not occur)	–
3	11.38	–
4	23.92	–
5	0.67	–
6	5.59	–

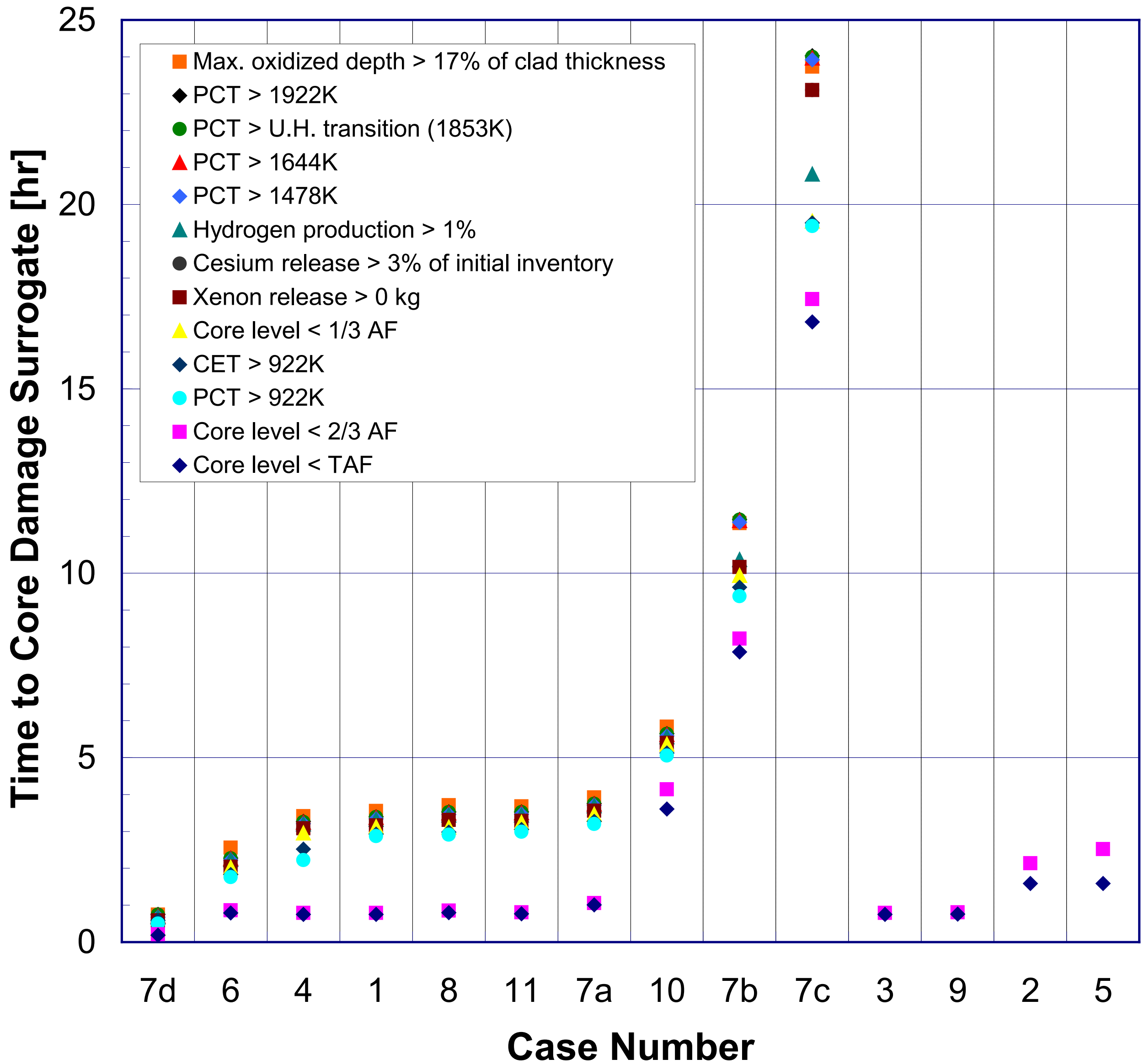

Figure 4.1.2 Times of Core Damage in Byron Shutdown Scenario and its Sensitivity Cases, According to Various Surrogate Measures[14]

[14] The case numbers that label the horizontal axis omit the initial prefixes (e.g., "7d" in the axis label denotes case A.2.7d). Gross core damage is avoided in cases A.2.3, A.2.9, A.2.2, and A.2.5.

Table 4.2.1 Times of Core Damage (in Hours) According to Various Surrogate Measures Compared Among Base and Sensitivity Cases of Surry Scenario SGTR-9

Case	Water Level Below TAF	Water Level Below 2/3 AF	Water Level Below 1/3 AF	PCT > 922K	CET > 922K	Finite Xe release	Hydrogen Generation > 1% Potential	Cesium Release > 3% Inventory	PCT > 1478K	Maximum Oxide Depth > 17% Clad Thickness	PCT > 164K	PCT > 1853K	PCT > 1922K
B.1 (Base)	34.67	36.47	37.00	37.17	37.40	37.57	37.63	37.63	37.90	37.90	38.07	38.13	38.15
B.2	22.00	22.37	22.77	22.67	22.93	22.87	23.07	23.07	23.03	23.10	23.10	23.12	23.12
B.3	6.65	6.92	7.32	7.23	7.40	7.48	7.65	7.67	7.77	7.80	7.83	7.90	7.91
B.4	35.17	36.63	37.53	37.70	37.93	38.10	38.17	38.17	38.43	38.43	38.60	38.67	38.68
B.5	39.30	41.07	41.87	42.00	42.27	42.43	42.50	42.53	42.80	42.80	42.93	43.03	43.03
B.6	34.73	36.10	37.10	37.23	37.47	37.63	37.70	37.73	38.00	38.00	38.13	38.22	38.22
B.7	34.67	36.13	37.03	37.17	37.40	37.57	37.63	37.67	37.93	37.93	38.07	38.16	38.16
B.8	34.67	36.53	37.03	37.17	37.43	37.57	37.67	37.67	37.93	37.93	38.10	38.17	38.18
B.9	34.63	36.23	37.00	37.13	37.37	37.53	37.60	37.63	37.90	37.90	38.03	38.13	38.13
B.10	21.70	22.07	22.47	22.40	22.57	22.60	22.77	22.80	22.73	22.80	22.80	22.82	22.82
B.11	21.80	22.17	22.60	22.50	22.73	22.70	22.87	22.90	22.83	22.90	22.90	22.92	22.92
B.12 – B.14	Same as Case B.1												
B.15	21.67	22.07	22.47	22.37	22.57	22.57	22.73	22.77	22.73	22.77	22.77	22.79	22.79
B.16[1]	No core damage as of 72 hours												
B.17[1]	5.38	5.62	5.98	6.00	6.15	6.30	6.43	6.50	6.65	6.65	6.77	6.85	6.86
B.18[1]	4.20	4.43	4.78	4.70	4.85	4.92	5.10	5.15	5.10	5.15	5.25	5.35	5.36
B.19[1]	Same as Case B.17												
B.20[1]	Same as Case B.18												
B.21	34.67	36.20	37.00	37.17	37.40	37.60	38.00	37.70	38.07	38.13	38.17	38.22	38.22

1 Cases for controlling HHSI above TAF (i.e., below the normal pressurizer level) and B&F are counter to EOPs for the prescribed conditions. They are included for illustrative purposes, though they could be viewed as representing errors of commission.

Table 4.2.2a Bins and Statistics for Times to Start of Core Uncovery, Surry SGTR Scenario

Bin	Cases	Important Deviations from Base Case	Range of Time to TAF (hour)	Within-Bin Statistics for Time to TAF: Mean ± One Standard Deviation (hour)
1	B.3, B.17-B.20[1]	½ tube rupture or B&F initiated[2]	4.2 – 6.7[1]	5.2 ± 0.9
2	B.2, B.10-B.11, B.15	5-tube rupture	21.7 – 22.0	21.8 ± 0.1
3	B.1 (base), B.4-B.9, B.12-B.14, B.21	-	34.6 – 39.3	35.1 ± 1.3
4	B.16[1]	Control of HHSI without B&F	> 72[1]	> 72

1 Cases for controlling HHSI above TAF (i.e., below the normal pressurizer level) and B&F are counter to EOPs for the prescribed conditions. They are included for illustrative purposes, though they could be viewed as representing errors of commission.

2 Both characteristics lead to one or more fully open pressurizer PORVs, increasing primary side inventory loss, which is detrimental due to earlier RWST depletion (with recirculation mode unavailable). A "½ tube rupture" is a break whose flow area is half the cross-sectional area of one SGT.

Table 4.2.2b Bins and Statistics for Times for PCT to Exceed 1478K (2200 °F), Surry SGTR Scenario

Bin (Defined in Table 4.2.2a)	Range of Time to 1,478 K (hour)	Within-Bin Statistics for Time to 1,478 K: mean ± one standard deviation (hour)
1	5.1 – 7.8[1]	6.3 ± 1.0
2	22.7 – 23.0	22.8 ± 0.1
3	37.9 – 42.8	38.4 ± 1.4
4	> 72[1]	> 72

1 Cases for controlling HHSI above TAF (i.e., below the normal pressurizer level) and B&F are counter to EOPs for the prescribed conditions. They are included for illustrative purposes, though they could be viewed as representing errors of commission.

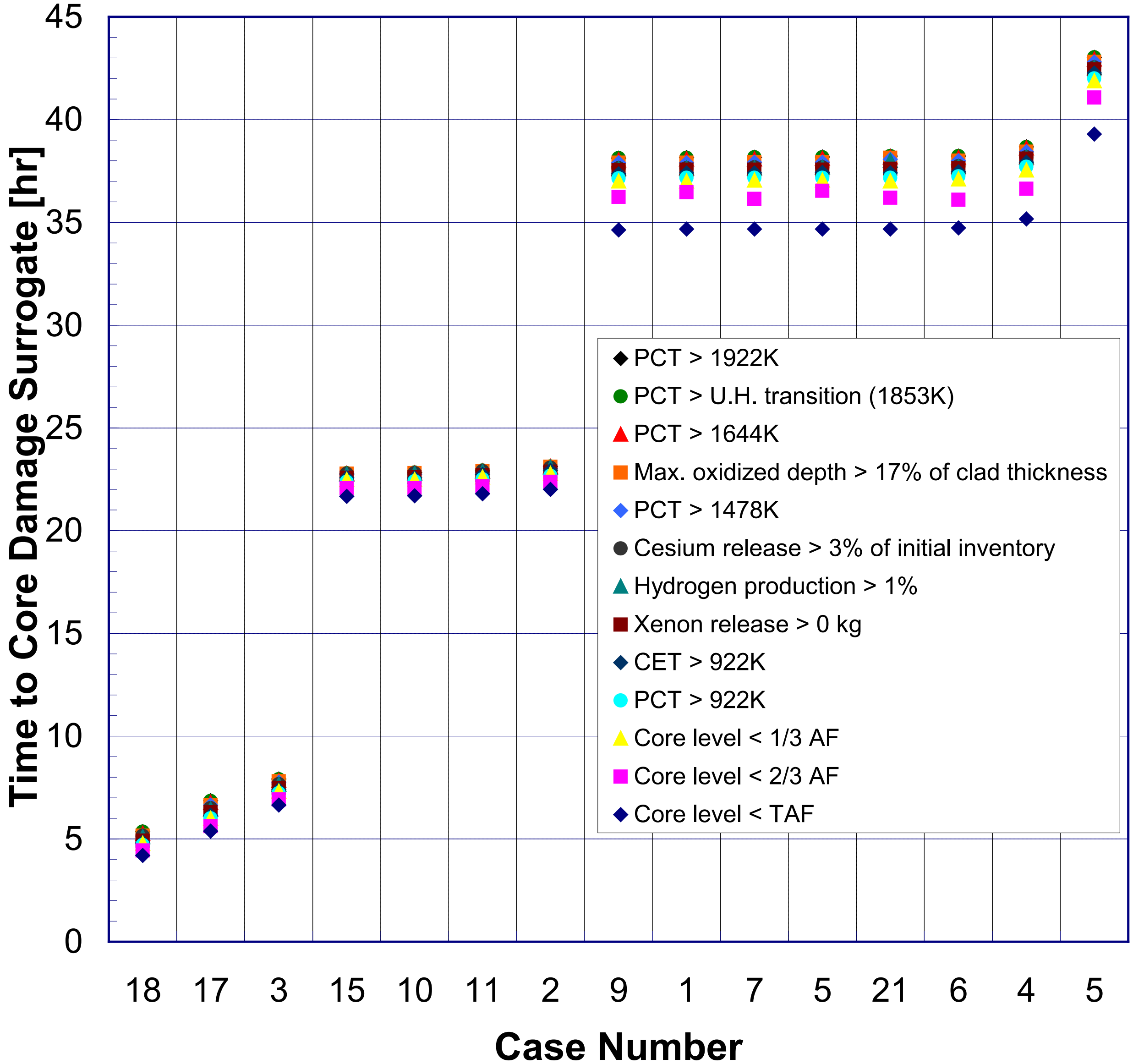

Figure 4.2.1 Times of Core Damage in Surry SGTR Scenario and its Sensitivity Cases, According to Various Surrogate Measures[15]

The measure for maximum thickness of the zirconium oxide layer on the cladding is actually calculated using the ratio of the mass of zirconium oxide to that of zirconium. In the case of Surry, the masses used are in fact specifically those of the cladding, since the required information is available according to the setup of this deck. (For other plants, the masses used necessarily include the contributions from, e.g., guide tubes. The reason is that MELCOR provides the cladding-specific masses only if appropriate control functions are defined before the run is executed, which has been done only in the case of Surry.)

[15] The case numbers that label the horizontal axis omit the initial prefixes (e.g., “18” in the axis label denotes case B.2.18). The Figure omits three cases that duplicate other cases and one case for which no surrogate ever indicates (see Table 4.2.1). Cases 17 and 18 should be viewed with the cautions cited in previous tables in mind.

Indication of the 13th criterion (peak local cladding temperature exceeds 3,000 ºF [1,922 K]) is used to stop the calculations automatically. Consequently, all calculations end with no fuel relocations having occurred. (Applies to Surry model only.)

To evaluate the cesium-class release surrogate predicted by the Surry model, the released mass from class 2 (alkali metals) is added with stoichiometric fractions of the releases of class 16 (cesium iodide) and class 17 (cesium molybdate). The released mass is then compared with the initial class 2 inventory. The model necessarily predicts that the cesium release surrogate indicates core damage later than the noble gas class release surrogate (i.e., because less than 3 percent of the cesium inventory is assigned to the gap), but the precise timing depends on the details of MELCOR's radionuclide release models.

The definition of the surrogate that compares the cladding temperature with the temperature of the Urbanic-Heidrick transition is unchanged in case B.21, even though that case uses a different correlation for the temperature-dependent reaction rate coefficient. (That is., the surrogate is still satisfied when the cladding temperature exceeds 1,853 K, even though the alternative reaction rate correlation may locate the transition at a different temperature.)

The surrogate referring to hydrogen generated by zirconium oxidation is actually implemented in terms of the mass of zirconium oxide.

4.3 Investigation of Core Damage Surrogates – Peach Bottom, SBO with RCIC until Battery Depletion

Table 4.3.1 and Figure 4.3.1 show the times to core damage according to various surrogate measures for the Peach Bottom SBO with RCIC until battery depletion base and sensitivity cases. Refer to Section 3.3 (and also Appendix C) for the sensitivity case definitions and to Section 4 for the definitions of the surrogate measures. Table 4.3.2a provides statistical data for the time when core uncovery starts. Table 4.3.2b does the same for the time when the PCT first exceeds 1,478 K (2,200 °F). Definitions of the bins are based on timing similarities and not on any logical similarities among the case definitions. See Appendix C for additional details.

A few plant-specific technicalities concerning the evaluation of some of the measures are noted in the next paragraphs.

For the surrogates based on core water level, the level is defined by summing the heights of the swollen water columns in the lower plenum, core channel (the third of five computational rings in the model), and the shroud dome.

As discussed in the SNL Peach Bottom model documentation, most of the initial inventory of cesium is in molybdate form (i.e., the MELCOR class 17 radionuclide group is pre-populated), so the 3 percent cesium release surrogate is computed from the cesium molybdate release. The model necessarily predicts that this surrogate indicates later than the noble gas class release surrogate (i.e., because less than 3 percent of the cesium molybdate inventory is assigned to the gap), but the precise timing depends on the details of the MELCOR radionuclide release models.

The measure for maximum thickness of the oxide layer on the cladding is actually calculated using the ratio of the mass of zirconium oxide to that of zirconium. Moreover, because this deck does not provide the time-dependent masses of the individual components (such as cladding), the net core-region masses of zirconium oxide and zirconium are used instead of the cladding masses specifically. These net masses include contributions from the guide tubes, canister, and control rod blade. (See the footnote to the corresponding paragraph in Section 4.1.)

Table 4.3.1 Times of Core Damage (in Hours) According to Various Surrogate Measures Compared Among Base and Sensitivity Cases of Peach Bottom SBO with RCIC until the Battery Depletion Scenario

Case	Water Level Below TAF	Water Level Below 2/3 AF	Water Level Below 1/3 AF	PCT > 922K	CET > 922K	Finite Xe Release	Hydrogen Generation > 1% Potential	Cesium Release > 3% Inventory	PCT > 1478K	Maximum Oxide Depth > 17% Clad Thickness	PCT > 1644K	PCT > 1853K	PCT > 1922K
C.1.1 (Base)	9.60	9.97	10.38	10.36	10.44	10.69	10.83	11.14	10.99	11.07	11.05	11.09	11.10
C.1.2	16.87	17.33	17.82	17.82	17.92	18.20	18.35	18.72	18.57	18.63	18.62	18.67	18.68
C.1.3	5.93	6.35	6.78	6.72	6.80	6.97	7.13	7.28	7.15	7.22	7.20	7.24	7.25
C.1.4	3.30	3.58	3.87	3.85	3.90	4.07	4.20	4.42	4.27	4.35	4.33	4.37	4.38
C.1.5	6.85	7.17	7.53	7.52	7.58	7.78	7.93	8.19	8.03	8.12	8.10	8.14	8.15
C.1.6	10.00	10.42	10.87	10.86	10.95	11.23	11.37	11.74	11.58	11.65	11.64	11.69	11.70
C.1.7	9.62	10.00	10.39	10.39	10.47	10.71	10.85	11.17	11.01	11.09	11.08	11.12	11.13
C.1.8	10.23	10.62	11.04	11.03	11.11	11.35	11.50	11.83	11.67	11.74	11.74	11.78	11.79
C.1.9	9.55	9.94	10.32	10.33	10.40	10.64	10.79	11.09	10.94	11.03	11.01	11.05	11.06
C.1.10	9.60	9.97	10.38	10.36	10.44	10.69	10.83	11.14	10.99	11.07	11.05	11.09	11.10
C.1.11	9.97	10.34	10.76	10.75	10.84	11.08	11.22	11.54	11.39	11.46	11.45	11.49	11.50

Table 4.3.2a Bins and Statistics for Times to Start of Core Uncovery, Peach Bottom SBO with RCIC Scenario

Bin	Cases	Important Deviations from Base Case	Range of Time to TAF (hour)	Within-Bin Statistics for Time to TAF: Mean ± One Standard Deviation (hour)
1	C.3 – C.5	SRV stuck-open (C.1.3); early battery depletion (C.1.4 and C.1.5)	3.30 – 6.85	5.36 ± 1.50
2	C.1, C.6 – C.11	Base case (C.1.1); reduced decay heat (C.2.6); one unavailable SRV (C.1.7); reduced RCIC flow rate (C.1.8); SRV parameters (C.1.9); heat transfer coefficients for boiling (C.1.10); CST liquid temperature (C.1.11)	9.55 – 10.23	9.80 ± 0.25
3	C.2	RCIC in manual mode (C.1.2)	16.87	–

Table 4.3.2b Bins and Statistics for Times for PCT to Exceed 1,478 K (2,200 °F), Peach Bottom SBO with RCIC Scenario

Bin (Defined in Table 4.3.2a)	Range of Time to 1,478 K (hour)	Within-Bin Statistics for Time to 1,478 K: Mean ± One Standard Deviation (hour)
1	4.27 – 8.03	6.48 ± 1.61
2	10.94 – 11.67	11.22 ± 0.29
3	18.57	–

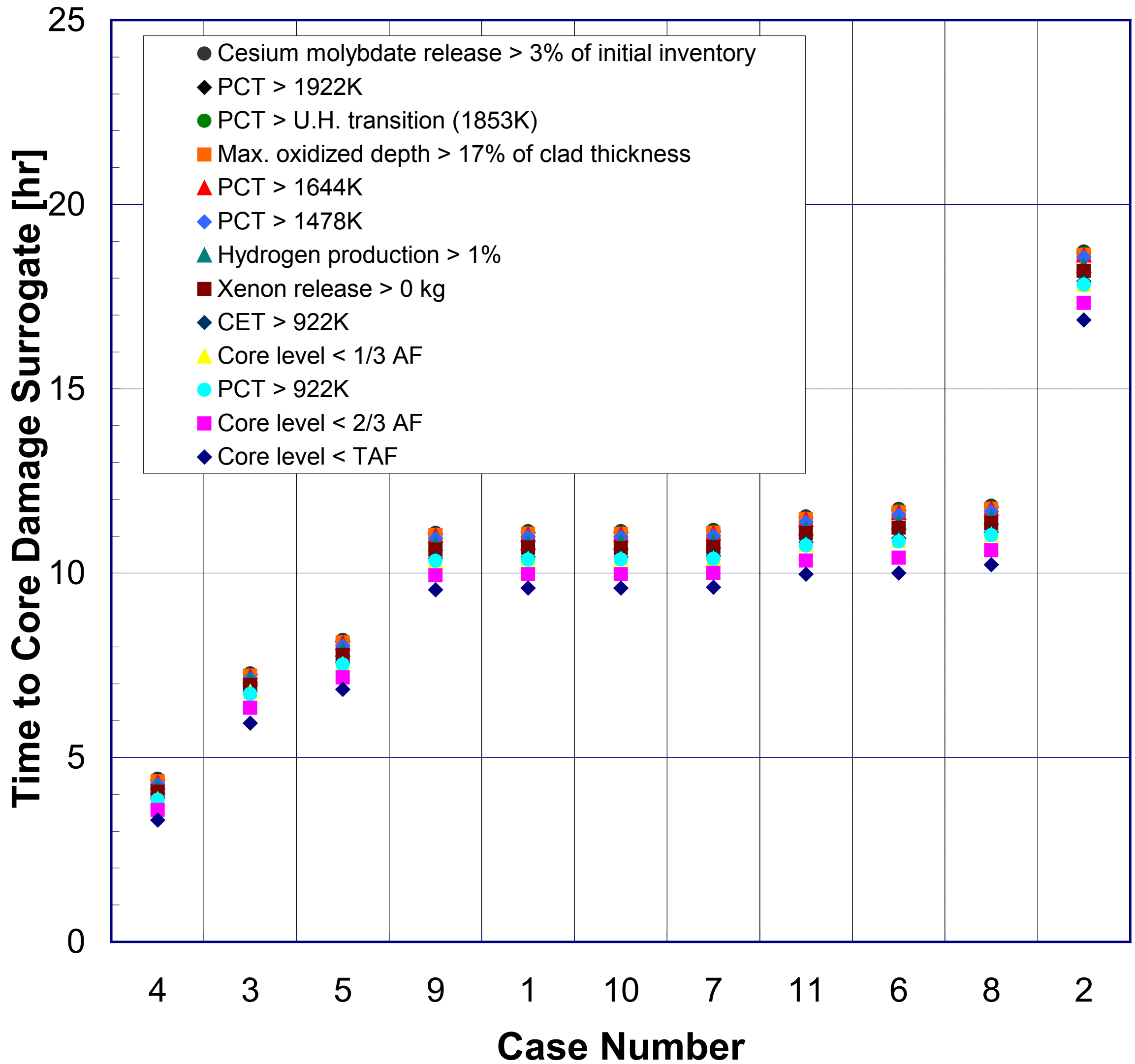

Figure 4.3.1 Times of Core Damage in Peach Bottom SBO with RCIC until Battery Depletion Scenario and its Sensitivity Cases, According to Various Surrogate Measures[16]

The surrogate referring to hydrogen generated by zirconium oxidation is actually implemented in terms of the mass of hydrogen generated by oxidizing zirconium and steel.

4.4 Investigation of Core Damage Surrogates – Observations

Table 4.4.1 summarizes core damage surrogate results for the four base cases. The first surrogate measure to indicate is always water level at the TAF, while the last measure to indicate is plant-dependent. The order within the legends in Figures 4.1.1, 4.1.2, 4.2.1, and 4.3.1 depicts the base-case order in which the surrogate measures indicate. The spread is plant- and scenario-dependent. The findings suggest that the surrogate measure associating core damage with the spatially maximized cladding temperature exceeding 1,478 K [2,200 °F]) is roughly equivalent to an uncovery of 86 percent of the core (i.e., the simple average of the fractions from the four base cases), but there is considerable

[16] The case numbers that label the horizontal axis omit the initial prefixes (e.g., “4” in the axis label denotes case C.1.4).

uncertainty. Note that in the case of the Surry calculation, the water level is actually below the BAF (by about 8 cm) when this measure indicates.

Table 4.4.1 Summary of Core Damage Timings for Base-Case Calculations

Plant / Scenario	Time of 1st Core Damage Surrogate Reached (hr) [Surrogate]	PCT > 1,478 K (hr)	Time of Last Core Damage Surrogate Reached (hour) [Surrogate]	Δt Between 1st and Last Surrogate Reached (hr)	% of Core Uncovery at Time of PCT > 1,478 K
Byron / Loss of DC Bus 111	1.35 [Water level at TAF]	1.91	1.94 [PCT > 1,922 K (3,000 °F); Oxide layer > 17% of clad thickness]	0.59	83.5
Byron / Loss-of-coolant during shutdown	0.75 [Water level at TAF]	3.34	3.55 [Oxide layer > 17% of clad thickness]	2.80	75.4
Surry / SGTR	34.7 [Water level at TAF]	37.9	38.2 [PCT > 1,922 K (3,000 °F)]	3.48	100
Peach Bottom / Station blackout	9.6 [Water level at TAF]	11.0	11.1 [Cesium release from fuel > 3%]	1.54	84.2

There are two conclusions drawn from these results:

(1) For at-power accidents, a PCT in excess of 1,478 K (2,200 °F) is a good choice for a core damage surrogate for MELCOR success criteria applications, because it is not overly-conservative or overly non-conservative.

(2) For shutdown conditions, a single metric may not be sufficient for prescribing a realistic surrogate. Rather, a combination could be used including low water level (one-third of the fuel height); temperature (PCT above 1,478 K); and a Cesium class release fraction (3 percent released from fuel). If only one metric is used, the current limited PWR analyses would suggest that water level at one-third active fuel height is a precursor to fuel damage and a PCT above 1,478 K is a precursor to significant fuel damage.

The definition of the core damage surrogate is tied to the tool that is being used to analyze the accident, as well as the context of the analysis. For instance, if using an Excel calculator for shutdown operator timings that tracks water level, peak clad temperature would not be an available output. Meanwhile, if activation of the accident management infrastructure (as opposed to a PRA interface definition) is the motivation of the analysis, a different criterion might be more appropriate. Furthermore, the above guidance is not intended to suggest that no localized fuel damage will have occurred prior to these conditions. But, given the use of a tool that models core behavior and nodalizes the core similar to contemporary MELCOR input models, and the intent to use a realistic indicator of when an accident has transitioned from a pre-core damage to a post-core damage state, the criteria are reasonable.

5. INVESTIGATION OF TIME NEEDED TO ARREST FUEL HEATUP

This section considers how much time injection systems need to operate in order to arrest fuel heatup to the point of core damage, in the case that such systems, initially unavailable, are recovered prior to core damage. Note that this discussion does not consider the time required to realign these systems or to reset interlocks, etc., because these steps require an additional focus on given applications. Further, the present analysis does not consider situations where additional recovery actions are happening concurrently (such as automatic depressurization), which might have competing effects on water inventory. Thus, these results should not be applied to those situations.

This section highlights six SBO scenarios. Two pertain to the Peach Bottom plant and four pertain to the Byron Unit 1 plant. The discussion considers the effectiveness of the plants' systems (e.g., ECCS) when they are recovered in the course of the SBO. Specifically, it is of interest to determine how much earlier than the time of core damage (i.e., in the absence of that recovery) specific systems must be recovered in order to prevent core damage. Table 5.1 summarizes the definitions of the six selected scenarios. The various assumptions in the Byron scenarios regarding how primary and secondary relief valves may stick open have been made in order to ensure that both high- and low-pressure situations are considered.

Table 5.1 Matrix of Scenarios for Calculating Recovery Actions

<table>
<tr><th colspan="3">Scenario</th><th colspan="2">Recovery Action</th></tr>
<tr><td rowspan="2">Peach Bottom SBO</td><td>A</td><td>RCIC operates until batteries are depleted at 8 hours (i.e., the scenario treated in Appendix C)</td><td colspan="2" rowspan="2">Recover all modeled ECCS[1] (i.e., HPCI and LPCS)</td></tr>
<tr><td>B</td><td>No RCIC</td></tr>
<tr><td rowspan="2">Byron full-power SBO</td><td>A</td><td>No Auxiliary FW; no depressurization of primary or secondary sides of the RCS</td><td colspan="2">Recover 1 train (out of 2) of all modeled ECCS[1]</td></tr>
<tr><td>B</td><td>AFW is available until batteries are depleted (5 hours); secondary-side depressurization due to one stuck relief valve of one steam generator (the valve gets stuck on cycle 239[2]); primary-side depressurization due to one stuck pressurizer relief valve (it gets stuck on the first cycle)</td><td colspan="2">Recover both trains of all modeled ECCS[1]</td></tr>
<tr><td colspan="2" rowspan="2">Byron shutdown SBO</td><td rowspan="2">Similar to the base case of the scenario treated in Appendix A, except there is no break and the RHR pump stops at t = 0; there is no secondary-side depressurization; there is a primary-side depressurization due to one stuck pressurizer relief valve (it gets stuck on cycle 513)</td><td>A</td><td>Recover closed-loop RHR</td></tr>
<tr><td>B</td><td>Recover one train of all modeled ECCS[1,3]</td></tr>
</table>

[1] All pumps are available, but current pressure and flow versus head curves determine whether low-pressure pumps actually inject.
[2] Note that this valve failure mechanism is actually less relevant for Byron, which uses a hydraulically-operated modulating valve type.
[3] The recovered train includes the RHR pump that is not in a closed-loop operation.

The first of the two Peach Bottom SBO scenarios is the one that Appendix C analyzes. In that scenario, the RCIC is operational until the batteries are depleted at 8 hours. Core damage occurs at 10.99 hours. (This section identifies the time of core damage with the time when the spatially-maximized cladding temperature attains 1,478 K [2,200 °F]). The second Peach Bottom scenario is the same, except that the RCIC is not available at any time. Core damage occurs at 1.17 hours in this case. Both cases include recovery of the high-pressure coolant injection (HPCI) and low-pressure core spray (LPCS) systems.

The first two of the four Byron SBO scenarios are full-power scenarios[17]. In the first scenario, no credit is given to AFW, and all relief valves (i.e., all pressurizer relief valves and all steam generator relief valves) are credited to cycle indefinitely as demanded. This scenario results in core damage at 2.58 hours. The second Byron full-power SBO credits AFW until the time of battery depletion, which is assumed to occur at 5 hours. It is also assumed that the first steam generator relief valve to reach a count of 239 open-and-close cycles will remain stuck open (see Note 2 from Table 5.1). That number of cycles has a cumulative probability of 0.50 for a stuck-open valve, given a failure probability of 2.896 x 10^{-3} per-demand. Moreover, should a pressurizer relief valve open even once, the assumption is that it will remain stuck in the open position (i.e., during the first opening cycle). This scenario results in core damage at 10.43 hours. In the first case, the assumption is that only one ECCS train consisting of one charging pump, one high-pressure pump, and one low-pressure (i.e., RHR) pump will be recovered. The second case considers the recovery of the complete Byron ECC system (two trains with a total of six pumps). In both cases, available pumps initially take suction from the RWST.

The last two of the four Byron SBO cases pertain to a shutdown scenario[18]. The two cases are distinguished by the assumed recovery actions. The scenario is most easily defined in terms of the scenario analyzed in the second half of Appendix A. Briefly, the differences between the present scenario and the one discussed in Appendix A are (a) the absence of a break in the present case; and (b) the unavailability in the present case of the RHR pump at the start of the transient. In a shutdown configuration, the RHR pump operates in a closed-loop mode, taking suction from a hot leg. One pressurizer relief valve is assumed to stick open when the valve has cycled 513 times. For that number of cycles, the cumulative probability of having a stuck-open valve is 0.50 given a failure probability of 1.35 x 10^{-3} per demand. With no recovery action, this scenario attains core damage at 15.87 hours. The first shutdown case supposes recovery of the RHR pump operating in the closed-loop shutdown mode. The second case analyzes the recovery of one complete train of the Byron ECCS. That is to say, one each of the charging, SI, and RHR pumps is available to inject by taking suction initially from the RWST. Note that this second case does not consider recovery of the RHR pump that had been operating in closed-loop configuration. In the second case, the RCS pressure determines the actual injection rate of the RHR and other available pumps (which draw from the RWST). This statement also applies to all credited injection systems considered in the six cases, other than the case that considers the closed-loop configuration. (In closed-loop configuration, an RHR pump may operate at high RCS pressure, since the pressure exists on both sides of the pump.)

Table 5.2 lists a summary of the MELCOR-calculated results. The latest successful recovery times are approximate (i.e., the resolution is about five minutes). According to these calculations, action at the tabulated time averts core damage in all cases. With one exception, a time of five to seven minutes before the time of core damage is sufficiently early to actuate the recovered injection system(s) to avert core damage. The exception occurs in the Byron shutdown scenario, in the case of recovery of the RHR pump operating in a closed-loop configuration. In this case, the recovery action has to precede core damage by 119 minutes. In the absence of a recovery action, the Byron shutdown scenario leads to a stuck-open pressurizer relief valve at 14.0 hours, when the valve cycles for the 513th time. The open valve results in a rapid loss of coolant; therefore, if it is recovered only after the sticking of the valve, the RHR pump eventually becomes inoperable again. (The reason is that this pump takes its suction from a hot leg.) To avert core damage, the recovery of the RHR pump must precede the 513th cycle of the relief valve. The recovery time tabulated in Table 5.2 is in fact seven minutes earlier than when the valve becomes stuck, in the absence of any recovery action. The tabulated time is early enough for the RHR system to reduce and maintain the RCS pressure below the relief valve setpoint before the 513th cycle occurs. In contrast, if the RHR pump is recovered five minutes after the relief valve sticks open, core damage is not averted, although the RHR pump is able to operate for about one hour, with the result that core damage is postponed by about 1.3 hours, relative to the case of no action. Note that the requirement

[17] These full-power Byron SBO simulations use Revision 6 of the Byron model.
[18] This Byron shutdown SBO scenario is simulated with the Byron Shutdown model described in Section 3.1.2.

of notably early recovery of closed-loop RHR injection depends critically on the assumption about valve sticking.

A conclusion to draw from these results regarding the time to arrest a fuel heatup is that five to ten minutes are sufficient for the SBO transients investigated, if capability to inject from the RWST (as opposed to closed-loop cooling) is recovered. A specific application will also need to consider the additional time required for tasks such as realigning systems and resetting permissive, as well as any concurrent recovery actions (e.g., system depressurization) that might have competing effects on water inventory.

Table 5.2 Summary of Calculated Time for Successful Recovery Actions (Peach Bottom and Byron Plants)

<table>
<tr><th colspan="2">Scenario</th><th>Noteworthy Event(s)</th><th>Hour</th><th>t_{CD}* (hour)</th><th colspan="2">t_{RC}** (hour)</th></tr>
<tr><td rowspan="2">Peach Bottom SBO</td><td>A</td><td>Last RCIC duty ends</td><td>7.1</td><td>10.99</td><td colspan="2">t_{CD} – 5 minutes</td></tr>
<tr><td>B</td><td>–</td><td></td><td>1.17</td><td colspan="2">t_{CD} – 5 minutes</td></tr>
<tr><td rowspan="4">Byron full-power SBO</td><td>A</td><td>Steam generators dry out</td><td>1.1</td><td>2.58</td><td colspan="2">t_{CD} – 7 minutes</td></tr>
<tr><td rowspan="3">B</td><td>One steam generator relief valve becomes stuck in the open position</td><td>8.5</td><td rowspan="3">10.43</td><td colspan="2" rowspan="3">t_{CD} – 5 minutes</td></tr>
<tr><td>Steam generators dry out</td><td>8.6</td></tr>
<tr><td>One pressurizer relief valve becomes stuck in the open position</td><td>9.0</td></tr>
<tr><td colspan="2" rowspan="2">Byron shutdown SBO</td><td>Steam generators dry out</td><td>13.9</td><td rowspan="2">15.87</td><td>A</td><td>t_{CD} – 119 minutes</td></tr>
<tr><td>One pressurizer relief valve becomes stuck in the open position</td><td>14.0</td><td>B</td><td>t_{CD} – 7 minutes</td></tr>
</table>

* Time of core damage without the recovery action.
** Approximate latest time that the recover action is successful.

6. UNCERTAINTY ANALYSIS

This section documents the results of the uncertainty analysis study that was performed to determine the probability of core damage under the loss of main feedwater (LoMFW) event in the Byron Unit 1 nuclear power plant. The analyses performed as part of this study follow those performed in Reference [7] by the Electric Power Research Institute (EPRI) using the Modular Accident Analysis Program, version 4.0.6, (MAAP4). The major objectives of this study are:

- To quantify in the form of probability density functions the uncertainties associated with key input parameters, and to propagate these uncertainties through selected configurations of the LoMFW scenario using the MELCOR model and a Latin Hypercube Sampling (LHS) process for Byron Unit 1.

- To calculate the probability associated with the occurrence of core damage as defined and using the PCT as the relevant figure-of-merit.

- To assess the most influential input parameters that affect core damage as defined with the PCT as the key figure-of-merit and using a linear regression approach (i.e., the PCC/SRC computer code [8]).

- To determine the sensitivity of results (e.g., probability of core damage) to the sample size.

- To compare the MELCOR results of the analysis to the EPRI-MAAP study results.

6.1 Approach

6.1.1 Selected Accident Scenario

The scenario considered in the EPRI-MAAP analysis was the LoMFW event with a subsequent failure of all AFW and a failure to recover any form of FW (e.g., AFW, MFW, or cross-tie to the other unit). The EPRI-MAAP analysis selected a generic Westinghouse 4-loop PWR design with Model D5 steam generators and both motor- and turbine-driven AFW. Similar to Byron Unit 1, this plant design included high-head centrifugal charging pumps and high-pressure safety injection (HPSI) pumps. The EPRI-MAAP study analyzed eleven different configurations of the LoMFW scenario. These configurations differ in the availability of the RCP trip, the number of PORVs, and the number of HPSI and charging pump trains.

This study of Byron Unit 1 used the MELCOR input deck and analyzed two different configurations of the LoMFW scenario[19]. The assumptions for these configurations are similar to two of the eleven configurations that the EPRI-MAAP study analyzed. The selected configurations differ in the number of PORVs available for a B&F operation. In the first configuration (Configuration 1), there is one available PORV. The second configuration (Configuration 2) considered two available PORVs. The current analysis assumes the availability of one HPSI pump train; the analysis assumes that no charging pump is available. The RCPs are allowed to trip when specific trip conditions are reached. These assumptions are also similar to those of the corresponding two configurations that the EPRI-MAAP study analyzed.

The above selected LoMFW scenario is also similar to the Loss of DC Bus 111 scenario analyzed in Section 3.1.1 of this report (the loss of DC bus 111 also results in a loss of the MFW), except for the following difference:

- In the Section 3.1.1 analysis, one charging pump injects coolant until the S signal is generated or the volume control tank is depleted. However, for the current uncertainty propagation calculations, the charging pumps are assumed to be unavailable from the initiation of event (for consistency with the EPRI assumptions).

[19] Revision 7 of the Byron MELCOR model was slightly modified to (i) facilitate execution in batch mode and (ii) correct an error concerning the heat input by the RCPs.

6.1.2 Selected Input Parameters

The input parameters and their associated uncertainty distributions that were selected for the current uncertainty analysis are similar to those of the EPRI-MAAP analysis [7]. Specifically, the EPRI-MAAP analysis considered uncertainty distributions in the following parameters:

- the power level at the start of an incident
- three steam generator water level setpoints (they account for uncertainties in steam generator-level measurements)
- time of the reactor trip
- number of PORVs opened for the B&F
- number of available HPSI and charging pump trains
- HPSI pump flow characteristics near the shutoff head
- pressurizer PORV flow characteristics
- time of AFW failure
- reactor coolant pump trip
- time of feed initiation (HPSI actuation)
- temperature of core damage (TCD)

The uncertainties associated with three parameters, specifically the number of available HPSI and charging pump trains, the number of PORVs that are opened for B&F, and the tripping of the RCPs, were treated explicitly in the EPRI-MAAP analysis, which resulted in 11 different configurations. Reference [7] notes that the selection of the probability distributions representing the uncertainties in various input parameters were based on reviews of prior MAAP-based thermal/hydraulic and severe accident simulations, emergency operating procedures, and PRA LoMFW accident sequence event tree analyses. The input parameters and the associated uncertainty distributions selected for the current analysis are discussed below.

Power Level at the Start of an Accident – This analysis assumes three discrete estimates representing the probability distribution for reactor power level at the start of the accident that are identical to those defined as part of the EPRI-MAAP study (see Figure 6.1 and Table 6.1). Reference [7] notes that this distribution is based on reviews of the data from the Licensee Event Reports (LERs) for the LoMFW initiating events [9].

Steam Generator Water Level Setpoints – In the EPRI-MAAP analysis, the uncertainty associated with the steam generator level measurement was modeled by assigning distributions to the following three steam generator level setpoints:

- 36.3 percent narrow range (NR) for the automatic reactor trip

- 14 percent NR for entry to Emergency Operating Procedure (EOP) FR-H.1 (used as a trigger to start charging)[20]

- 25 percent wide range (WR) for initiating the B&F

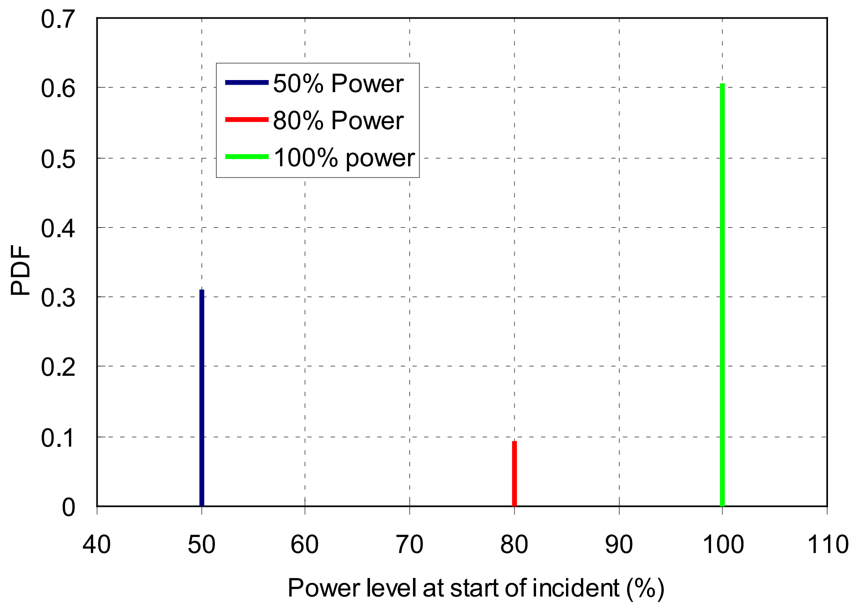

Figure 6.1 Uncertainty Distribution for Power Level at Start of LoMFW Event Initiation

Table 6.1 Uncertainty Distribution for Power Level at Start of LoMFW Event Initiation

Power Level [%]	**Probability**
100	0.602
80	0.091
50	0.307

In defining samples, these setpoints were assumed to be 100 percent correlated. The EPRI-MAAP study considered the triangular distributions corresponding to each steam generator water level setpoint, re-expressed in mass instead of in height and centered at the nominal mass [7]. The upper and lower bounds of the triangular distributions were assumed to be at ±15 percent of the nominal value.

The current MELCOR uncertainty study only considers two steam generator water level setpoints:

- 18.0 percent NR for an automatic reactor trip

- 25 percent WR for the initiation of a B&F

Note that 36.3 percent of the NR water level modeled in the EPRI-MAAP study is a reactor trip setpoint for the Westinghouse 4-loop plant with a Model D5 steam generator. For the Byron Unit 1 plant that uses Babcock and Wilcox steam generators, this setpoint is at 18 percent of the steam generator NR water level. The second steam generator setpoint from the EPRI-MAAP analysis, which defines the entry to

[20] Reference [7] lists 10 percent and not 14 percent. However, this amount was changed to the actual value of 14 percent used in the analyses and is based on a personal communication with the authors.

EOP FR-H.1 (10 percent of the NR for Byron Unit 1), is not used in the MELCOR analysis since charging is unavailable.

Figure 6.2 shows the selected uncertainty distribution in the steam generator water level setpoints. Note that the uncertainty distribution is applied to the actual steam generator NR and WR water level setpoints expressed in height instead of in mass.

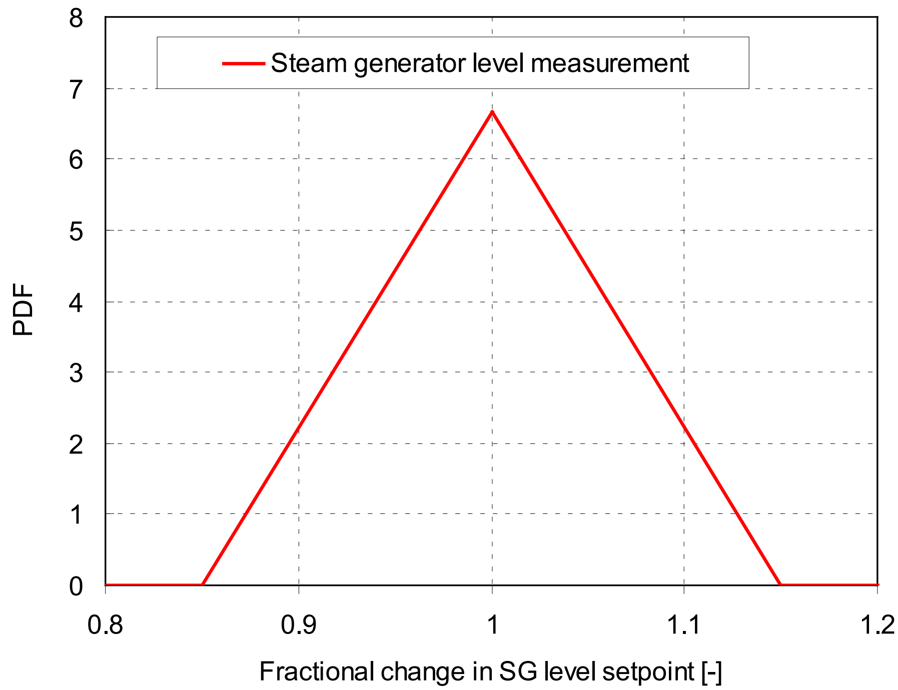

Figure 6.2 Uncertainty Distribution for Steam Generator Water Level Setpoints

<u>Time of Reactor Trip</u> – Figure 6.3 and Table 6.2 show the uncertainty distribution selected for the reactor trip condition. A four-state discrete distribution based on the EPRI-MAAP study is assumed. According to Reference [7], the data from the review of the LERs associated with the “Rate of Initiating Events at U.S. Nuclear Power Plants 1988-2008” [9] for LoMFW-initiating events was used to arrive at the selected uncertainty distribution.

Reactor trip condition ‘0’ models the reactor trip by an early operator action or by an assumed early automatic trip and occurs at the initiation of the event (i.e., at problem time *t* = 0 minutes, simultaneous with the loss of the MFW). Reactor trip condition ‘1’ simulates the trip by manual operator action and occurs at *t* = 1 minute. Reactor trip condition ‘2’ models operator action at a code-predicted time related to the steam generator water level, but this condition occurs at a higher level than the normal automatic low-level reactor trip setpoint (whose setpoint is 18 percent of the steam generator NR). This operator action is assumed to take place as soon as the steam generator water level drops by approximately 50 percent of the total level drop required to reach the 18 percent steam generator low level setpoint. Reactor trip condition ‘3’ simulates an automatic reactor trip at the 18 percent steam generator low level setpoint. In a given calculation, the mode that actually applies is fixed by the sample. For example, if the sample dictates condition ‘1,’ then the reactor trip occurs at 1 minute irrespective of automatic trip conditions '2' and/or '3.'

<u>HPSI Pump Flow Characteristics Near the Shutoff Head</u> – This parameter assumes a triangular distribution identical to that of the EPRI-MAAP study. The nominal HPSI flow is used as the mode of the

triangular distribution, and the upper and lower bounds are assumed at ± 10 percent of the nominal value. Figure 6.4 shows the selected uncertainty distribution, which is applied to the HPSI pump flow curve.

Pressurizer PORV Flow Characteristics – The EPRI-MAAP study assumed a triangular distribution with the nominal PORV flow as the mode and the upper and lower bounds at ± 20% of the nominal value. Figure 6.5 shows that in the current study, the same triangular distribution is applied to the PORV flow area. Configuration 2 has two available PORVs (see Section 6.1.1), and the flow areas of the two PORVs are sampled independently from the triangular distribution in Figure 6.5.

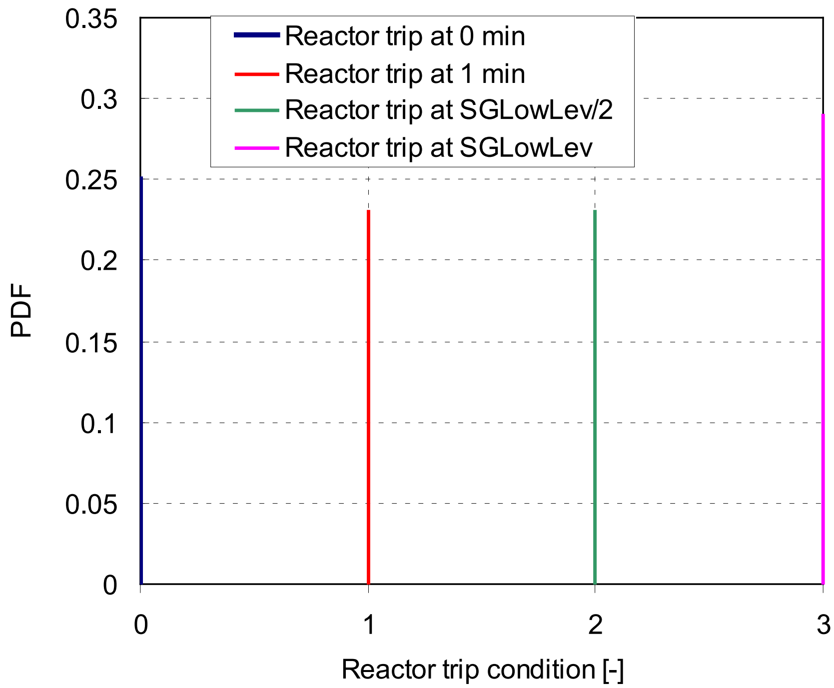

Figure 6.3 Uncertainty Distribution for Time of Reactor Trip

Table 6.2 Uncertainty Distribution for Time of Reactor Trip

Reactor Trip (RT) Condition	Description	Time [min]	Probability [%]
0	Early automatic/manual RT	0	25
1	Early manual RT	1.0	23
2	Intermediate manual RT	$t_{SG\ Low\ Lev}$ / 2	23
3	Late automatic trip	$t_{SG\ Low\ Lev}$	29

Time of AFW Failure – A two-state discrete distribution identical to that of the EPRI-MAAP study is assumed for the time of AFW failure. Figure 6.6 and Table 6.3 show the selected distribution for the time of AFW failure. Reference [7] notes that this distribution was selected using the data from NUREG/CR-6928 [10] for a failure to start and a failure to run (during the first hour) for motor-driven and turbine-driven AFW systems. AFW is activated on a low SG water level setpoint (i.e., 18 percent of the steam generator NR level).

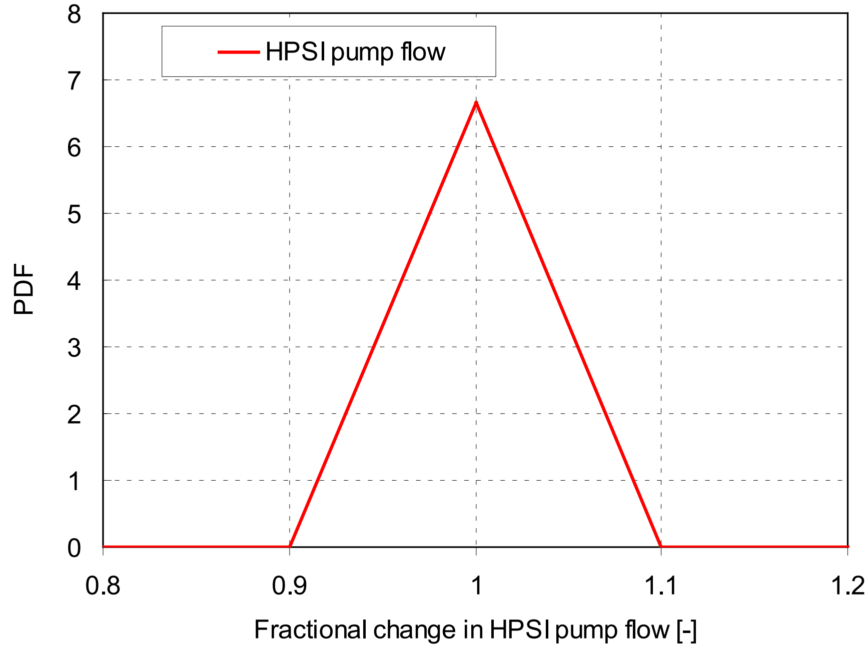

Figure 6.4 Uncertainty Distribution for the HPSI Flow Rate

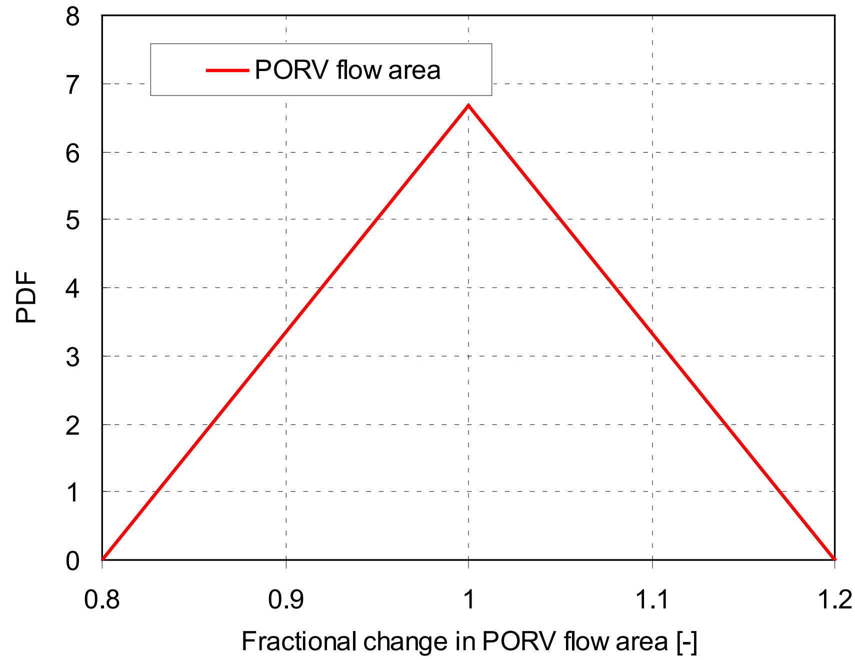

Figure 6.5 Uncertainty Distribution for the PORV Flow Area

Time of Feed Initiation (HPSI Actuation) – This input parameter models the uncertainty in time required by operators to initiate a B&F after the steam generator water level reaches 25 percent of the WR value. A log-normal distribution shown in Figure 6.7 is similar to the EPRI-MAAP study distribution and is assumed for this parameter. Reference [7] notes that this distribution represents the human error probability of ~1.4×10^{-3} for the failure to perform the action within 21 minutes of the actuation cue (i.e., 25 percent of the WR level).

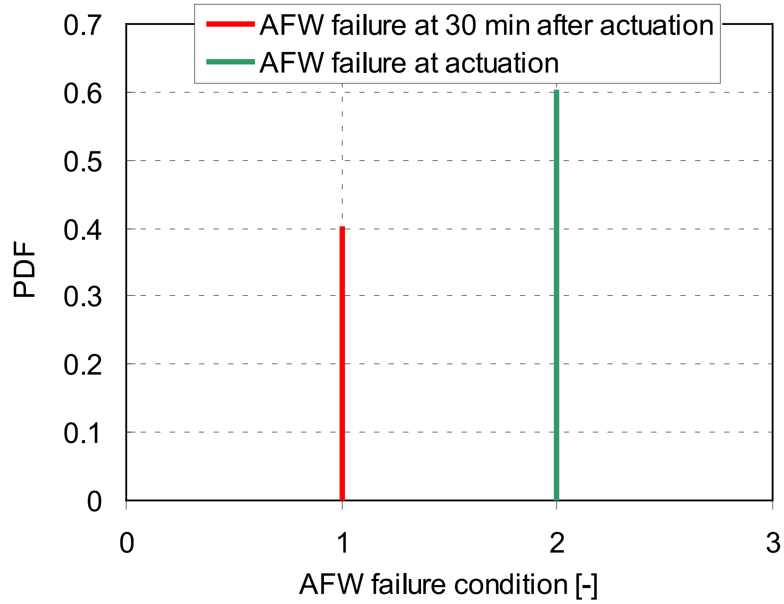

Figure 6.6 Uncertainty Distribution for Time of AFW Failure

Table 6.3 Uncertainty Distribution in Time of AFW Failure

AFW Failure Condition	Description	Time [min]	Probability [%]
1	Delayed failure	30 min after AFW actuation signal	40
2	Early failure	At AFW actuation signal	60

Temperature Used to Define Core Damage – The TCD represents the temperature threshold (treated as the temperature strength in a stress-strength interference type concept). The uncertainty in this parameter (see Figure 6.8) is represented by a triangular distribution, which is similar to the EPRI-MAAP study and is centered at 2,200 °F (1,478 K), with upper and lower bounds anchored at 2,600 °F (1,700 K) (Urbanic-Heidrick transition temperature) and 1,800 °F (1255 K) (representative PRA success criteria), respectively. Furthermore, similar to the EPRI-MAAP study, this parameter is sampled randomly along with the input parameters described above. The probability of core damage is determined based on the number of samples for which the calculated PCT (i.e., load) exceeded the sampled TCD. In the current analysis, Section 6.2 shows that the probability of core damage is also determined based on comparisons of the complementary cumulative distribution functions (CCDF) for the calculated PCT (i.e., load) and the cumulative distribution functions (CDF) of the TCD (i.e., capacity CDF calculated from the triangular distribution in Figure 6.8). Conceptually, these two approaches should produce the same results.

6.2 Results

The eight input variables (including two steam generator water level setpoints) and the TCD are sampled using the constrained LHS technique embedded in the LHS77 computer code [11]. The sample size selected for the current analysis is 100, which is the same as the sample size in the EPRI-MAAP study. A sensitivity study was performed to determine the impact of sample size on the estimated probability of core damage. Table 6.4 shows the sampled values of all eight input parameters and the TCD for the 100 samples. Table 6.4 also shows the calculated PCT for Configurations 1 and 2.

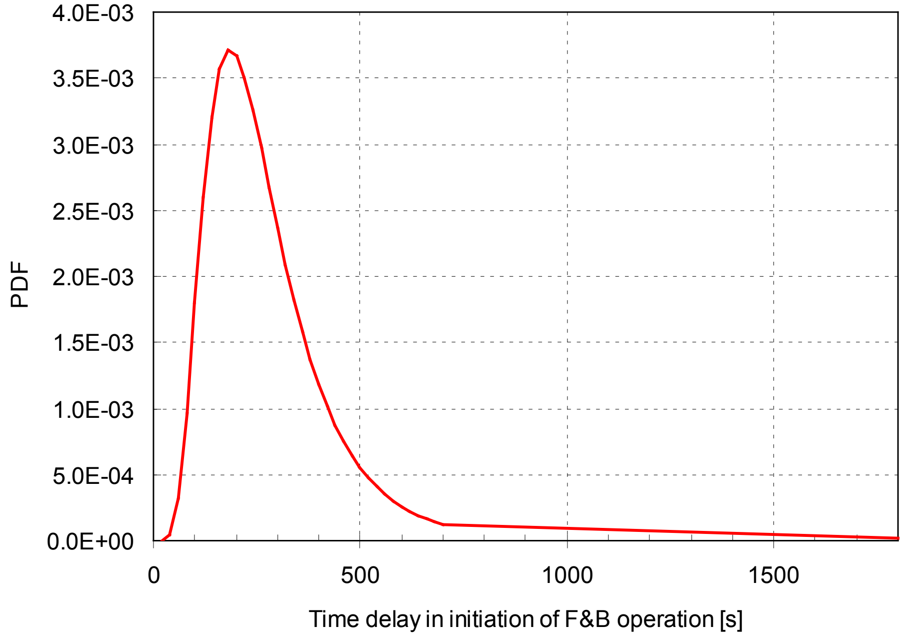

Figure 6.7 Uncertainty Distribution for Time of Feed Initiation

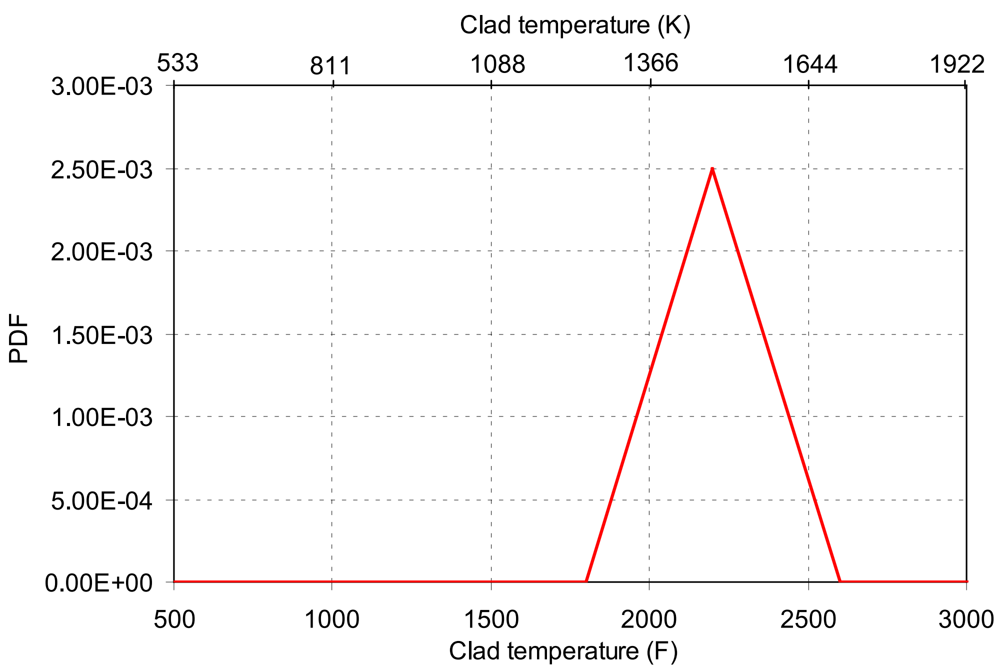

Figure 6.8 Uncertainty Distribution for the TCD

Table 6.4 Sampled Values of All Input Parameters for 100 Samples

Run	Power	SG Setpoint	RT	HPSI Flow	PORV-1 Flow	PORV-2 Flow	AFW Failure	Time of SI Injection	TCD	Configuration 1 PCT	Configuration 2 PCT
	[%]	[-]	[-]	[-]	[-]	[-]	[-]	[s]	[K]	[K]	[K]
1	50	1.07	1	1.06	0.84	0.95	1	306	1350	576	575
2	100	1.06	1	1.06	0.98	0.93	2	135	1322	2800	2800
3	100	1.08	2	0.99	1.16	1.01	1	223	1555	591	590
4	100	0.93	0	1.01	1.01	1.06	1	167	1589	2800	590
5	100	0.87	3	1.02	0.93	1.07	2	395	1439	2800	2800
6	50	0.99	2	0.99	1.04	1.03	2	283	1428	579	579
7	100	0.92	1	0.96	1.12	1.14	1	159	1383	2800	596
8	50	1.01	1	0.97	1.04	1.03	2	115	1466	580	580
9	50	0.87	2	1.05	0.95	0.96	2	74	1550	586	582
10	100	0.97	1	0.95	1.00	1.08	2	362	1450	2800	2800
11	100	1.06	2	0.99	1.06	1.12	2	317	1494	2800	597
12	50	0.97	1	0.98	1.09	1.08	1	240	1405	577	575
13	100	1.12	0	0.97	1.00	1.10	2	92	1600	2800	598
14	50	0.99	0	1.05	1.05	1.03	2	346	1455	577	578
15	100	0.93	3	1.06	1.08	0.97	1	364	1450	2800	592
16	100	0.90	1	0.98	1.02	0.87	2	291	1655	2800	2800
17	100	1.10	3	0.96	1.12	1.04	2	151	1505	2800	598
18	100	0.99	1	1.08	1.03	1.00	2	296	1522	2800	2800
19	100	0.95	2	0.95	1.06	1.15	1	701	1472	2800	594
20	50	0.96	1	1.04	0.98	0.94	2	190	1389	580	580
21	100	1.08	0	1.01	0.93	1.03	2	341	1444	2800	596
22	100	0.96	1	1.03	0.91	1.11	1	110	1528	2800	597
23	100	0.92	1	1.04	1.06	0.88	1	130	1578	2800	602
24	100	1.00	3	1.01	1.10	0.96	2	260	1544	2800	983
25	50	1.02	1	0.99	1.07	0.91	1	177	1500	577	575
26	100	1.00	0	1.04	0.97	0.83	1	180	1672	2800	588
27	50	1.04	0	1.04	1.10	1.03	2	235	1339	576	577

Table 6.4 Sampled Values of All Input Parameters for 100 Samples

Run	Power	SG Setpoint	RT	HPSI Flow	PORV-1 Flow	PORV-2 Flow	AFW Failure	Time of SI Injection	TCD	Configuration 1 PCT	Configuration 2 PCT
	[%]	[-]	[-]	[-]	[-]	[-]	[-]	[s]	[K]	[K]	[K]
28	80	0.99	0	0.92	0.98	0.92	2	198	1516	629	588
29	50	0.94	3	1.01	0.90	1.00	2	206	1611	586	583
30	50	0.94	1	1.02	0.99	0.91	1	559	1466	577	577
31	80	1.03	3	1.05	1.11	0.90	1	381	1461	585	586
32	80	1.06	2	0.94	0.92	0.92	2	76	1444	590	590
33	50	1.05	2	1.02	1.17	0.97	2	219	1422	578	578
34	100	1.13	2	1.02	1.06	1.04	1	136	1633	585	591
35	100	0.91	2	0.93	0.97	0.82	2	145	1289	2800	2800
36	100	0.89	1	0.97	1.07	1.05	1	448	1583	2800	890
37	50	0.91	3	1.00	1.01	0.92	2	424	1483	587	587
38	50	1	3	1.03	0.94	1.15	2	97	1333	583	584
39	100	0.97	0	0.95	1.00	1.11	1	185	1339	2800	590
40	50	1.03	2	0.93	0.96	1.02	2	186	1439	578	579
41	50	1	1	0.94	0.87	0.98	1	504	1566	577	577
42	100	0.98	2	0.90	1.14	0.89	1	277	1361	2800	593
43	100	0.97	2	1.00	0.96	1.09	2	339	1505	2800	975
44	100	1.08	0	0.97	0.86	0.95	2	280	1533	2800	598
45	50	1.07	3	1.00	1.06	1.10	2	374	1366	579	579
46	50	0.94	2	0.99	0.91	1.06	2	154	1428	584	585
47	50	1.01	0	1.00	1.01	0.87	2	267	1494	578	577
48	100	0.95	0	1.05	1.03	1.17	1	403	1300	2800	592
49	100	0.88	0	0.96	1.03	1.01	2	200	1416	2800	610
50	50	1.1	2	0.99	1.02	1.09	1	128	1355	575	574
51	80	1.09	2	0.98	0.99	0.92	2	526	1378	589	589
52	50	0.91	0	0.97	1.02	1.02	1	437	1394	571	575
53	100	0.99	1	1.09	1.04	0.93	2	470	1405	2800	2800
54	80	1.03	0	0.93	1.14	1.06	2	209	1528	589	589

Table 6.4 Sampled Values of All Input Parameters for 100 Samples

Run	Power	SG Setpoint	RT	HPSI Flow	PORV-1 Flow	PORV-2 Flow	AFW Failure	Time of SI Injection	TCD	Configuration 1 PCT	Configuration 2 PCT
	[%]	[-]	[-]	[-]	[-]	[-]	[-]	[s]	[K]	[K]	[K]
55	50	0.93	3	1.07	0.81	0.90	1	147	1455	578	578
56	80	1.02	3	0.95	1.07	1.00	2	521	1655	1163	591
57	50	1	0	1.06	0.98	1.17	2	268	1594	578	578
58	50	0.96	3	0.98	1.00	0.97	1	243	1578	578	578
59	50	0.98	1	0.95	0.96	1.05	1	141	1522	575	570
60	100	0.98	0	1.01	0.89	0.99	2	356	1411	2800	597
61	100	1.05	0	1.02	0.95	1.05	2	246	1600	2800	597
62	100	1.01	1	0.99	1.04	1.01	2	252	1466	2800	2800
63	100	0.98	0	0.97	0.99	1.01	1	236	1416	2800	591
64	100	1	1	1.00	1.02	0.86	1	119	1511	2800	598
65	100	1.01	1	1.03	0.88	0.94	2	156	1372	2800	2800
66	100	1.03	3	0.96	1.01	0.85	2	460	1539	2800	954
67	100	0.95	2	1.01	0.94	0.97	2	411	1505	2800	2800
68	100	1.02	3	0.99	0.91	1.04	1	172	1433	2800	593
69	100	1	3	1.03	1.01	1.08	1	123	1622	2800	593
70	100	1.02	3	1.04	1.08	1.10	2	386	1489	2800	603
71	50	0.98	3	1.00	0.93	1.02	1	480	1611	579	571
72	100	1.09	2	1.03	0.85	0.94	1	842	1455	2800	592
73	50	1.1	0	1.03	1.08	0.94	1	570	1389	576	575
74	100	1.01	3	1.00	0.90	0.99	2	323	1266	2800	1042
75	100	1.03	0	0.94	0.88	0.98	2	164	1566	2800	597
76	100	1.05	0	1.01	0.95	0.99	2	328	1478	2800	597
77	80	1.06	3	1.07	1.10	0.96	2	196	1516	587	590
78	50	1.11	0	0.92	0.97	1.00	2	415	1561	579	579
79	100	0.98	0	1.03	1.15	0.95	2	302	1350	2800	596
80	100	0.92	3	1.00	1.13	1.04	2	257	1483	2800	2800
81	100	1.04	2	1.02	1.15	1.00	1	249	1628	590	591

Table 6.4 Sampled Values of All Input Parameters for 100 Samples

Run	Power	SG Setpoint	RT	HPSI Flow	PORV-1 Flow	PORV-2 Flow	AFW Failure	Time of SI Injection	TCD	Configuration 1 PCT	Configuration 2 PCT
	[%]	[-]	[-]	[-]	[-]	[-]	[-]	[s]	[K]	[K]	[K]
82	100	1.02	3	1.02	0.94	1.07	1	102	1561	2800	591
83	100	0.9	3	0.98	1.05	0.88	2	193	1489	2800	2800
84	100	0.96	3	0.98	1.11	0.99	2	228	1539	2800	2800
85	100	1.07	1	1.07	1.05	0.90	2	224	1461	2800	2800
86	50	1.04	3	0.97	0.85	1.13	1	862	1533	574	577
87	100	1.01	3	1.00	0.90	1.02	2	203	1572	2800	682
88	100	0.96	3	1.01	1.00	1.13	1	332	1478	2800	596
89	100	1.04	3	1.01	1.03	0.85	1	289	1433	2800	587
90	80	1.02	3	1.00	0.87	0.99	1	600	1550	586	586
91	100	0.97	2	0.98	0.89	1.07	2	213	1422	2800	1038
92	50	1.04	3	0.98	1.01	0.89	2	311	1500	580	580
93	80	0.93	0	1.04	0.93	1.12	2	231	1516	588	589
94	50	0.95	1	0.94	0.97	0.96	1	312	1400	577	577
95	100	0.97	2	0.95	0.97	0.95	1	273	1378	2800	593
96	100	1.03	1	1.03	1.09	0.98	2	215	1316	2800	2800
97	100	0.94	0	1.05	0.99	1.06	2	637	1644	2800	608
98	100	1.09	2	0.96	0.94	1.05	1	175	1400	2800	591
99	100	1.05	2	1.08	0.96	0.98	2	168	1478	2800	598
100	100	1.14	2	0.99	0.92	0.98	2	262	1472	2800	598

Table 6.5 shows the results of the MELCOR calculations for Configurations 1 and 2. The number of simulations for each configuration where the calculated PCT (load) exceeds the indicated core damage temperature thresholds is also listed in Table 6.5. As noted earlier, the TCD is a temperature sampled from the triangular distribution shown in Figure 6.8. Since there are a total of 100 samples for each configuration, the probability that the PCT will exceed the TCD is about 0.57 (i.e., 57 out of 100) for Configuration 1 and about 0.15 (i.e., 15 out of 100) for Configuration 2.

Table 6.5 MELCOR Analysis Results for Configurations 1 and 2

Configuration	**>TCD**	**> 1,800 °F (1255 K)**	**> 2,200 °F (1,478 K)**	**> 2,600 °F (1700 K)**
1	57	57	57	57
2	15	15	15	15

Figure 6.9 shows the CCDF of the calculated PCT (load) and the CDF of the TCD (capacity). The CDF of the TCD is calculated from the probability distribution function (PDF) shown in Figure 6.8. The probability that the PCT exceeds the temperature at which core damage is assumed to occur is about 0.57 (57 percent) for Configuration 1 and about 0.15 (15 percent) for Configuration 2.

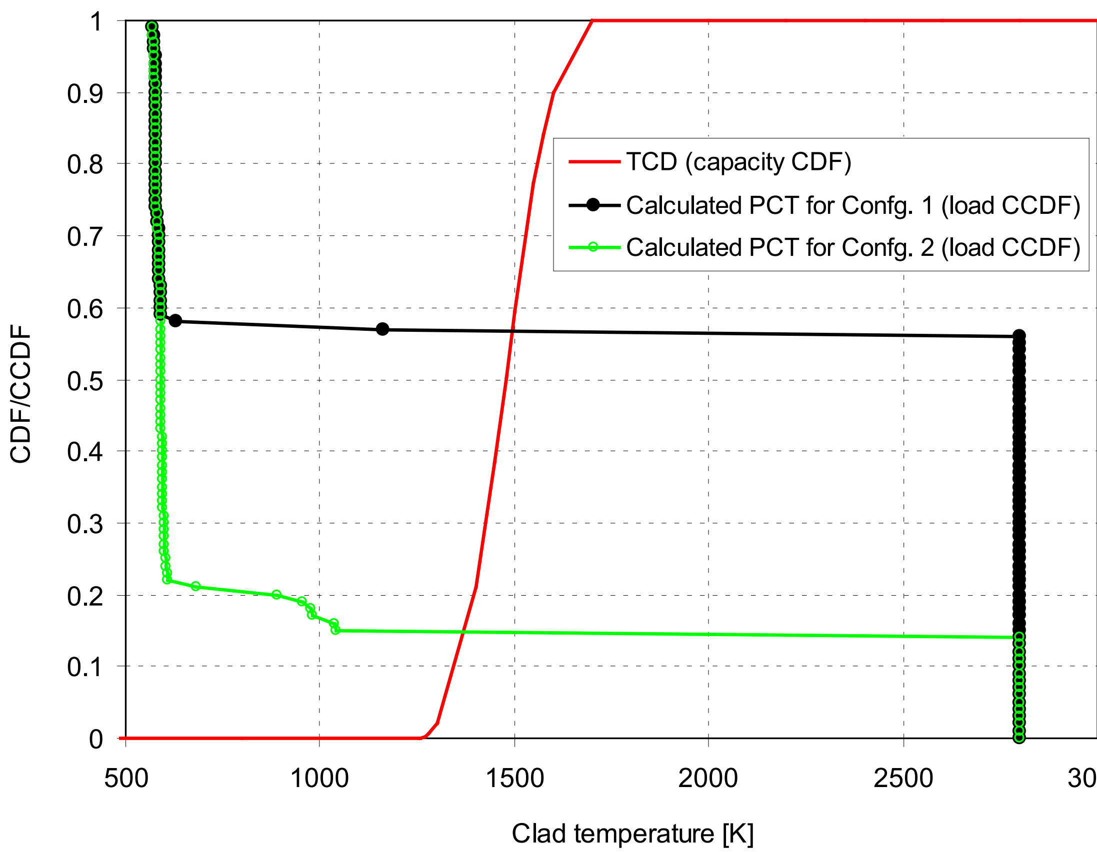

Figure 6.9 Comparison of the CCDF of the Calculated PCT (Load) and the CDF of the Temperature at which Core Damage is Assumed (Capacity)

Figure 6.10a and 6.10b show the PDF of the calculated PCT (load) and the TCD (capacity) for Configurations 1 and 2, respectively. Figure 6.10 shows that the discrete distribution is selected for the calculated PCT and PDF. The PCT for most of the samples (or runs) that avoid core damage remains between 570 K and 590 K. The PCT for the samples where core damage is observed is 2,800 K for all similar samples, because this particular temperature is the user-specified cladding relocation temperature. Once cladding attains this temperature, MELCOR reclassifies it as debris and no longer tracks that temperature as representative of cladding.

The most influential input parameters were identified using the PCC/SRC computer code, which calculates the partial correlation coefficients (PCC) [8]. These coefficients provide measures of the relative contribution (importance) of each input variable to the observed variations in the output figure-of-merit. Table 6.6 shows the results of the importance analyses; the uncertainty in the power level is by far the most important contributor affecting the calculated probability of core damage as estimated. The estimates are based on the present MELCOR analyses. The time of the AFW failure is shown to have a higher importance in Configuration 2. The uncertainties in all other parameters are only marginally significant (i.e., the importance of those with R^2 values less than ~0.70 is not very meaningful).

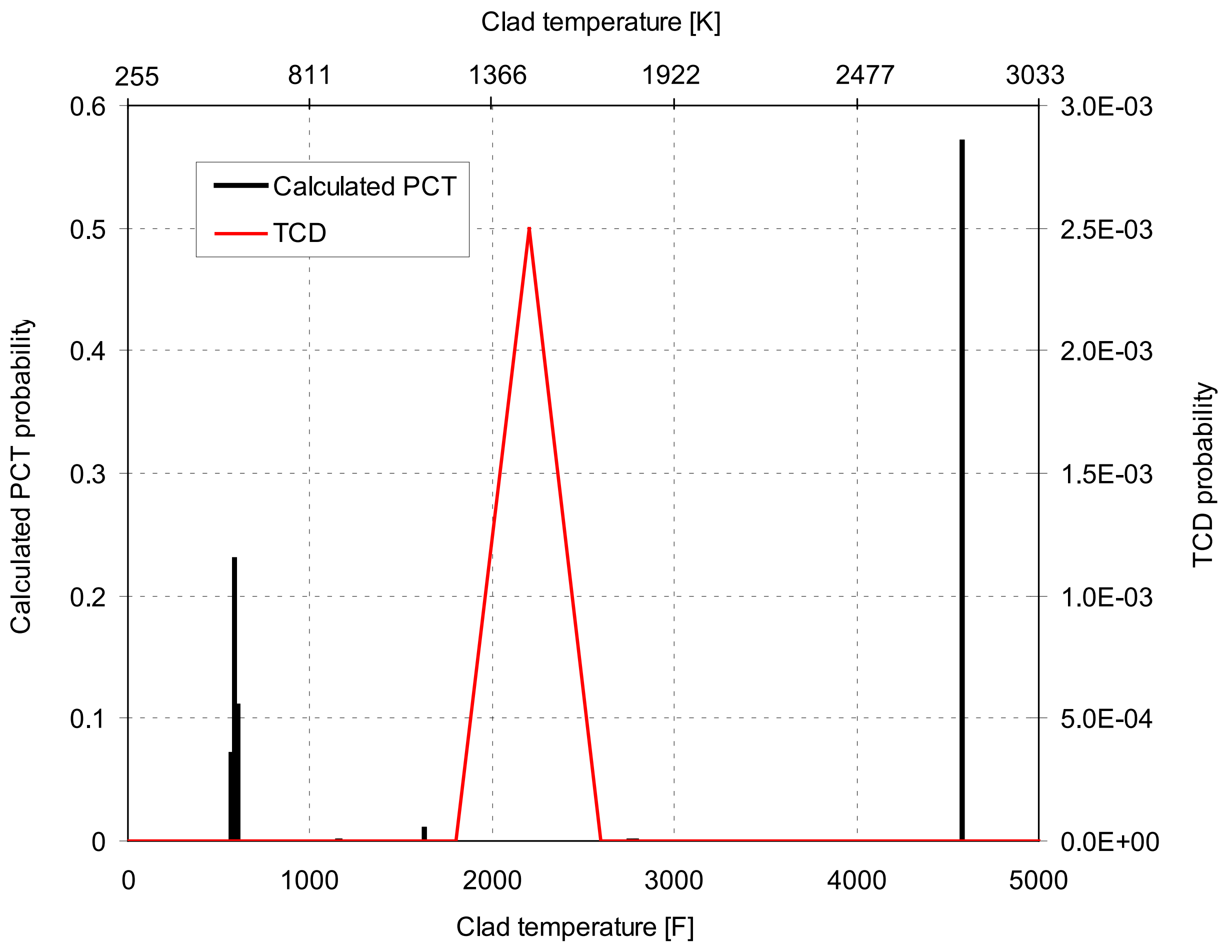

Figure 6.10a The PDF of the Calculated PCT (Load) and Temperature at which Core Damage Is Assumed (Capacity) for Configuration 1

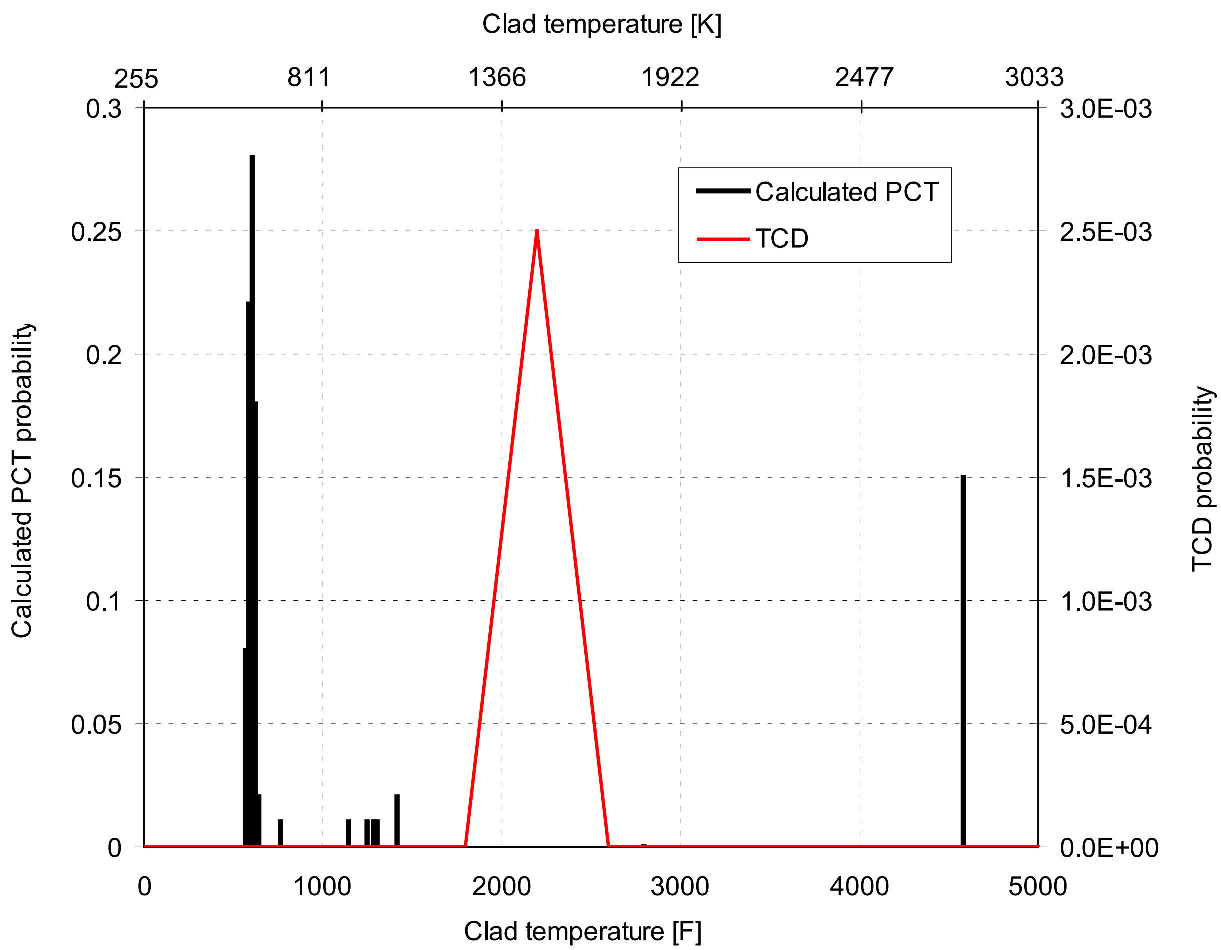

Figure 6.10b PDF of Calculated PCT (Load) and TEMPERATURE at which Core Damage is Assumed (Capacity) for Configuration 2

Table 6.6 Summary of the Rank Regression Results

Parameter	Case 1		Case 2	
	R^2	Importance Ranking	R^2	Importance Ranking
Power	0.97	1	0.94	1
SG water level setpoints	-0.40	3	-0.46	3
Time of reactor trip	0.17	5	-0.33	4
HPSI flow	-0.06	6	0.01	8
PORV 1 flow	-0.26	4	-0.02	7
PORV 2 flow	NA	NA	-0.19	5
Time of AFW failure	0.52	2	0.80	2
Time of B&F initiation	-0.05	7	0.05	6

The effect of the sample size is assessed by performing a sensitivity study for Configuration 1. Fifty (50) samples were selected for this sensitivity study. Table 6.7 shows the comparison of results based on the sample size of one hundred (100) and fifty (50). Furthermore, Figure 6.11 shows the comparison of CCDF of the calculated PCT (load) for the one hundred (100) and fifty (50) sample simulations. The CDF of the TCD (capacity) is also shown in Figure 6.11. The estimated probability of core damage using 50 samples is 0.60, which is slightly higher than the estimated probability of 0.57 based on 100 samples.

Table 6.7 Sensitivity to Sample Size

Sample Size	**>TCD**	**> 1,800 °F (1255 K)**	**> 2,200 °F (1,478 K)**	**> 2,600 °F (1,700 K)**
100	57	57	57	57
50	30	31	30	30

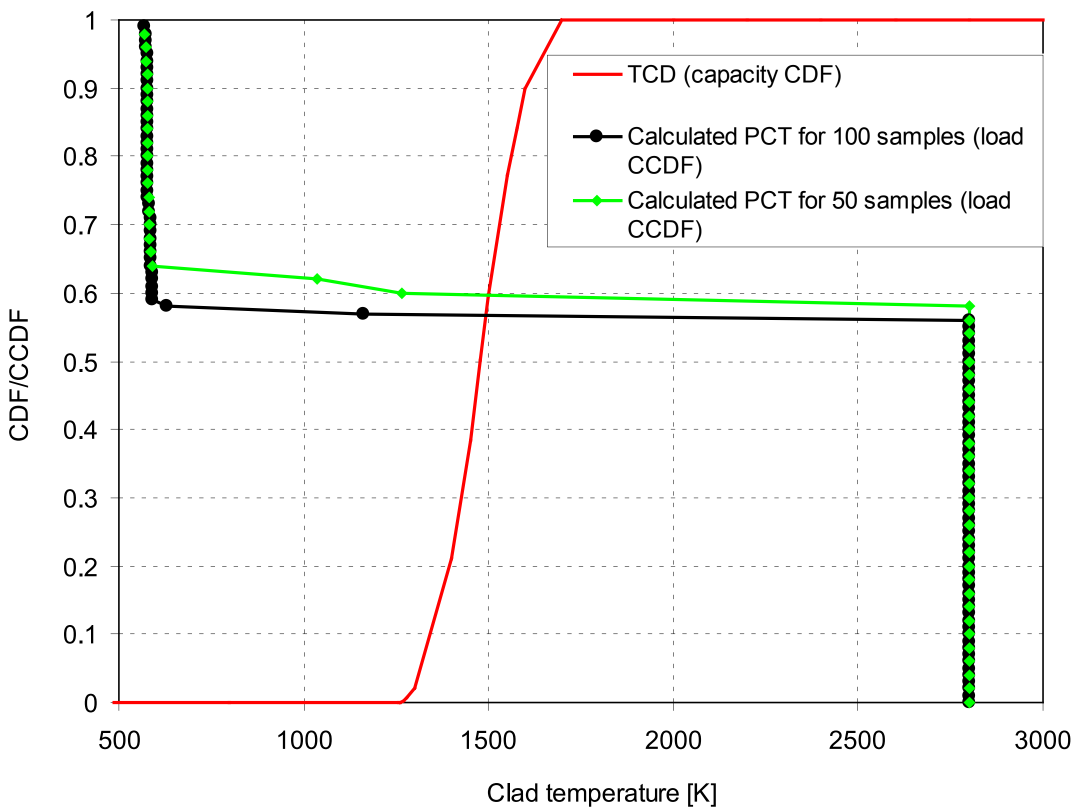

Figure 6.11 Comparison of the CCDF of the Calculated PCT (Load) for One Hundred (100) and Fifty (50) Sample Simulations

6.3 Comparison with the EPRI-MAAP Study

Table 6.8 shows the results of the comparison between the present study and the EPRI-MAAP study. The probability of core damage estimated in the current analysis for Configuration 1 (0.57) is in good agreement with the EPRI-MAAP analysis estimates (0.61). The MELCOR estimated core damage probability of 0.15 for Configuration 2 is also lower than the EPRI-MAAP estimated probability of 0.22.

Table 6.8 Comparison of the EPRI-MAAP Analysis and the Current MELCOR Analysis

Study	Configuration	>TCD	> 1,800 °F (1255 K)	> 2,200 °F (1,478 K)	> 2,600 °F (1,700 K)
NRC-MELCOR Study	1	57	57	57	57
	2	15	15	15	15
EPRI-MAAP Study [7]	1	61	63	61	60
	2	22	26	22	19

Table 6.8 also shows the comparison of the number of samples showing a PCT higher than 1255 K (i.e., the lower bound of the core damage temperature uncertainty distribution), 1478 K (center of the core damage temperature uncertainty distribution), and 1,700 K (upper bound of the core damage temperature uncertainty distribution) between the EPRI-MAAP study and the present study. The estimated probability of core damage is in reasonable agreement in the case of both configurations.

7. REFERENCES

1. H. Esmaili, D. Helton, D. Marksberry, R. Sherry, P. Appignani, D. Dube, M. Tobin, R. Buell, T. Koonce, and J. Schroeder, “Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models-Surry and Peach Bottom,” United States Nuclear Regulatory Commission, NUREG-1953, (September 2011).

2. M. Khatib-Rahbar, et al., "A Probabilistic Approach to Quantifying Uncertainties in the Progression of Severe Accidents", Nuclear Science & Engineering, Volume 102, 219 (1989).

3. R. Chang, J. Schaperow, T. Ghosh, J. Barr, C. Tinkler, and M. Stutzke, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,” U. S. Nuclear Regulatory Commission, NUREG–1935 (January 2012).

4. R. Wilson, et al., “Reviews of Modern Physics” Volume 57, Number 3, Part II (July 1985).

5. L. Soffer, S.B. Burson, C.M. Ferrell, R.Y. Lee, and J.N. Ridgely, “Accident Source Terms for Light-Water Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, NUREG-1465, (February 1995).

6. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, “Domestic Licensing of Production and Utilization Facilities,” Subpart 46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors” (2007).

7. J. Gabor, R. Sherry, and D. True, “Technical Framework for Management of Safety Margins-Loss of Main Feedwater Pilot Application,” EPRI Technical Report, EPRI-TR 1023032 (November 2011).

8. R. L. Iman et al., “A FORTRAN 77 Program and User’s Guide for the Calculation of Partial Correlation and Standardized Regression Coefficients,” NUREG/CR-4122, SAND85-0044, Sandia National Laboratories (June 1985).

9. "Rates of Initiating Events at U.S. Nuclear Power Plants 1988–2008, Reactor Operational Experience Results and Databases," http://nrcoe.inel.gov/results/.

10. "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG/CR-6928 (2007).

11. R.L. Iman and M.J. Shortencarier, “A FORTRAN 77 Program and User’s Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models,” U.S. Nuclear Regulatory Commission, NUREG/CR-3624 (1984).

APPENDIX A: DETAILS OF BYRON LOSS OF DC BUS 111 AND SHUTDOWN ANALYSES

A.1. Introduction

A.1.1 Background

Section 2.1 of the main report describes the Byron Unit 1 plant and the MELCOR models of the plant in full-power and Shutdown configurations. This appendix provides the details of the models and analyses for the Byron Unit 1 nuclear power plant.

A.1.2 Objectives and Scope

The objectives of this appendix are to document the calculations specific to the Byron Unit 1 plant that encompass detailed results for two basic scenarios and their sensitivity cases. These simulations, and in part the model, are limited in scope to considerations important for determining success criteria and sequence timing.

A.1.3 Outline

Sections A.2 and A.3 respectively focus on results that describe two basic scenarios: “Loss of DC Bus 111 with (B&F)” and “Shutdown.” The two scenarios and their sensitivity cases are defined in Sections A.2.1 and A.3.1, respectively, for “Loss of DC Bus 111” and “Shutdown.” Detailed results specific to the base and sensitivity cases appear in subsections of Section A.2.2 (in the case of “Loss of DC Bus 111”) and Section A.3.2 (in the case of “Shutdown”). Summary results pertaining to all cases appear in Section A.2.3 (“Loss of DC Bus 111”) and Section A.3.3 (“Shutdown”). These summary results also appear in Sections 3.1.1 (“Loss of DC Bus 111”) and 3.1.2 (“Shutdown”) of the main report.

All detailed results of the Loss of DC Bus 111 scenario appear in the subsections of Section A.2.2, with each subsection pertaining to a particular (base or sensitivity) case. Likewise, detailed results on the Shutdown scenario appear in the subsections of Section A.3.2. To simplify the Figure numbering, a four-character code is used within these subsections. After “A”, which designates Byron, the next character numbers the scenario. The character after that numbers the sensitivity case (the base case is counted as the first case). The last character in the string is the count among the Figures that apply to a given sensitivity case of a given scenario. For example, the first figure in the second sensitivity case that pertains to the Loss of DC Bus 111 scenario is Figure A.1.2.1. Figures not pertaining to a specific scenario and sensitivity case are not numerous, and are designated by a two-character code whose first character denotes Byron and whose second character is a count over the Appendix. A similar two-character code is used for all the Tables.

A.2 Byron Scenarios and Results: Loss of DC Bus 111 with Bleed and Feed

A.2.1 Loss of DC Bus 111 with Bleed and Feed

The transient initiator considered in this analysis is loss of DC bus 111. The FW regulating valves and their associated bypasses close when there is a loss of DC power, so the initiator implies a loss of MFW. Additional consequences of the loss of the bus are that one of the two pressurizer pilot operated relief valves (PORVs) is unavailable, and all motor-driven (MD) auxiliary feedwater (AFW) is unavailable. Additionally, the scenario definition assumes diesel-driven (DD) AFW and charging (after an ECCS signal) are unavailable. The condenser steam dump valves are assumed to be available and are used until the automatic closure of the main steam isolation valves (MSIVs) occurs. As stated above, charging pumps are not available for ECCS injection. However, normal-operation injection by one charging pump is accounted for initially after scram. One of two SI pumps and one of two RHR pumps are available. A successful transition to the recirculation mode is assumed, but the RHR heat exchanger is not available.

One of four containment fan coolers is available. Containment depressurization, whether by operator action or building failure, is not modeled.[21] Table A.1 summarizes the availability of various systems. The operators are assumed to initiate B&F at 20 minutes.

Table A.1 Loss of Bus 111 with the B&F: System, Phenomenological, and Operator Action Assumptions

Primary side	RCPs tripped if: (i) cold leg void fraction > 0.1 (a surrogate for cavitation); OR (ii) containment pressure > 20 psig; OR (iii) RCS pressure < 1,425 psig AND SI > 100 gpm (0.0063 m^3/s); OR (iv) time to initiation of B&F is less than 2 minutes (assumed based on FR-H.1 guidance to trip RCPs before initiating SI). Only 1 PORV is available (due to initiator) – it is the lower setpoint PORV (FL306)
Secondary side	0 of 1 motor-driven (MD)-AFW trains available (due to initiator); 0 of 1 diesel-driven (DD)-AFW trains available. Injection into steam generators of water in the hotwell by a booster pump is assumed to be unavailable. Cross-tie of the other unit's MD-AFW is not attempted.(1) Condenser steam dump valves and steam generator PORVs are assumed to be available. Atmospheric relief valves are allowed to cycle indefinitely without failing.
ECCS	0 of 4 accumulators are available (by probabilistic risk assessment (PRA) modeling convention). Neither containment spray train is available. 1 of 2 RHR pumps, 1 of 2 SI pumps, 0 of 2 charging pumps are available for ECCS injection mode. 1 of 1 RHR pump, 1 of 2 SI pumps, 0 of 2 charging pumps are available for ECCS recirculation mode. 1 of 4 reactor containment fan cooler (RCFC) units is available.
Operator actions	Available ECCS pumps are secured during dead-head phase, available upon demand. No RWST/CST refill is credited.
Other	No component cooling water (CCW) to RHR heat exchanger is credited. Containment depressurization (i.e., by any gas flow into the environment) is not modeled.

[1] As of this writing, the capability to cross-tie the other unit's MD-AFW was removed from service awaiting NRC approval of a license amendment request.

[21] That is, the model lacks any flow path from the containment into the environment that describes either filtered (or otherwise controlled) venting or overpressure failure.

To investigate the variability of the predictions, eight sensitivity cases are defined in addition to the base case. The highlighted entries in Table A.2 show how each sensitivity case differs from the base case.

- Case A.1.1 is the base case.
- Case A.1.2 assumes a higher level for the setpoint for the scram due to low steam generator water levels, leading to an earlier time for scram.
- Case A.1.3 discounts the normal-operation injection by one charging pump, which explores a situation where charging isolates due to a cascading electrical distribution failure or other cause.
- Case A.1.4 imposes depressurization of all four steam generators at 10 minutes, which explores a situation where the operators attempt to depressurize the steam generators early to allow for a FW injection from a low-pressure source. (No FW injection occurs, however.)
- Case A.1.5 assumes that the condenser steam dump valves and the steam generator PORVs are not available, and a safety valve of one of the steam generators becomes stuck in the open position after excessive cycling.
- Case A.1.6 credits all ECCS cold-leg accumulators, which are not credited in the base case due to a PRA modeling convention not to credit systems that would not be expected to participate, because their failure is not included in the logic modeling. (Note that this case is identical to the base case through the time of core damage).
- Case A.1.7 credits all fan coolers.
- Case A.1.8 is the same as the base case, except that the calculations use MELCOR version 2.1 instead of version 1.8.6.
- Case A.1.9 is the same as the base case, except that the decay heat is uniformly reduced by a factor of 0.8 relative to the base case.

The base-case calculation for the Byron Loss of Bus 111 with B&F scenario is described in detail in Section A.2.2.1. Sections A.2.2.2 through A.2.2.9 typically describe the sensitivity calculations in less detail. A summary discussion pertaining to this scenario appears in Section A.2.3, which also appears in the main report.

A.2.2 Results

A.2.2.1 Results of Case A.1.1 (Base Case)

The scenario progression is outlined in Table A.3 and illustrated in Figures A.1.1.1 to A.1.1.11. Loss of FW at t = 0 seconds (after a pre-transient steady-state period lasting 1,000 seconds) initiates the transient. No form of AFW is available.

Before turning to the MELCOR-predicted results, it is useful to make some estimates. The scenario definition calls for the bleed and feed operation to begin at 20 minutes. At 20 minutes, the decay heat is 87 MW. The maximum flow rate of one SI pump is 43.7 kg/s, which would be delivered only at an RCS pressure lower than about 5.1 MPa. The rate of heat transfer required to raise the temperature of the SI-injected liquid (311K) up to its saturation temperature (618K at 15.51 MPa) would be about 90 MW, if the water were to be injected at the maximum rate. Here the heat to vaporize the injected liquid is not considered. Thus one SI pump is only marginally capable of keeping the core subcooled, and can do so

Table A.2 Sensitivity Cases for Byron Loss of Bus 111 with Bleed and Feed Scenario

Case	Scram	Pre-S signal injection by one charging pump after scram(2)	Number of Steam Generators depressurized	SG PORVs and Condenser Dump Valves Available(3)	Number of Credited Accumulators	Number of Credited Fan Coolers	MELCOR Version	Decay Heat Factor
A.1.1 (Base)	Automatic(1)	Yes	0	Yes	0	1	1.8.6	1.0
A.1.2	Early(1)	Yes	0	Yes	0	1	1.8.6	1.0
A.1.3	Automatic(1)	No	0	Yes	0	1	1.8.6	1.0
A.1.4	Automatic(1)	Yes	4 (at 10 minutes)(4)	Yes	0	1	1.8.6	1.0
A.1.5	Automatic(1)	Yes	0	No(5)	0	1	1.8.6	1.0
A.1.6	Automatic(1)	Yes	0	Yes	4	1	1.8.6	1.0
A.1.7	Automatic(1)	Yes	0	Yes	0	4	1.8.6	1.0
A.1.8	Automatic(1)	Yes	0	Yes	0	1	2.1	1.0
A.1.9	Automatic(1)	Yes	0	Yes	0	1	1.8.6	0.8

(1) Automatic: all modeled scram trips operate; scram occurs at 40 s due to low SG water level. Early: manual scram occurs at 11 s representing a case where degrading conditions forewarned the operators that loss of the bus was imminent, and they were thus able to take immediate action.

(2) This normal-operation injection by a charging pump continues until VCT depletes or the S signal is sent. No charging pumps are available for ECCS purposes. The failure that causes loss of charging injection during ECCS is assumed to also prevent automatic realignment of normal charging from the VCT to the RWST.

(3) If available, dump valves actuate at scram and act to control legs coolant temperature to 564.8 K.

(4) Four steam generators are depressurized by MSIV and steam dump valve closure followed by a full opening of all steam generator PORVs.

(5) Additionally, the secondary safety valve of the lowest opening pressure of the pressurizer loop (i.e., the valve represented by FL376 opening at 8.205 MPa) is assumed to be stuck open on the 25th demand to close.

Table A.3 Timing of Key Events Predicted for the MELCOR Model Loss of Bus 111 with Bleed and Feed Base-Case Calculation (Times in Hours Unless Stated Otherwise)

Event Description	**Time**
Loss of all FW following loss of Bus 111	0 seconds
Scram (because of low water levels in steam generators)	40 seconds
Injection by one charging pump (in excess of letdown) begins	40 seconds
Condenser steam dump valves open	40 seconds
Injection by one charging pump ends (due to VCT depletion)(1)	10.9 minutes
Manual trip of all RCPs	18.0 minutes
Manual opening of pressurizer PORV-1; manual transmission of S signal	20.0 minutes
Injection by one SI pump drawing from RWST begins	20.7 minutes
MSIVs close (because of excessive rate of main steam line depressurization); condenser steam dump valves close	33.0 minutes
Steam generators dryout	33 minutes
SI pump deadheaded	0.75 to 2.04 ; 2.72 to 2.94
Core partially uncovered	1.35 to 3.5
Coolant temperature at core exit reaches 922 K (1,200 °F)	1.68
First gap release(2)	1.80
Peak clad temperature reaches 1,478 K (2,200 °F)	1.91
First failure of the core support plate	2.27
Calculation end	4.49
Accumulators inject	Not available
Containment sprays actuate	Not available
SG PORVs open	Does not occur
Steam generator safety valves open	Does not occur
Switch to ECCS recirculation mode	Does not occur
Sump water becomes saturated	Does not occur

1 The failure that causes loss of charging injection during ECCS is assumed to also prevent automatic realignment of normal charging from the VCT to the RWST.

2 In MELCOR, the volatile radionuclides in the fuel gaps are released by rings at the time when the cladding temperature, spatially maximized over a given computational ring, reaches 1,173 K. In the present case, the core's entire gap inventory is released after 2.02 hours.

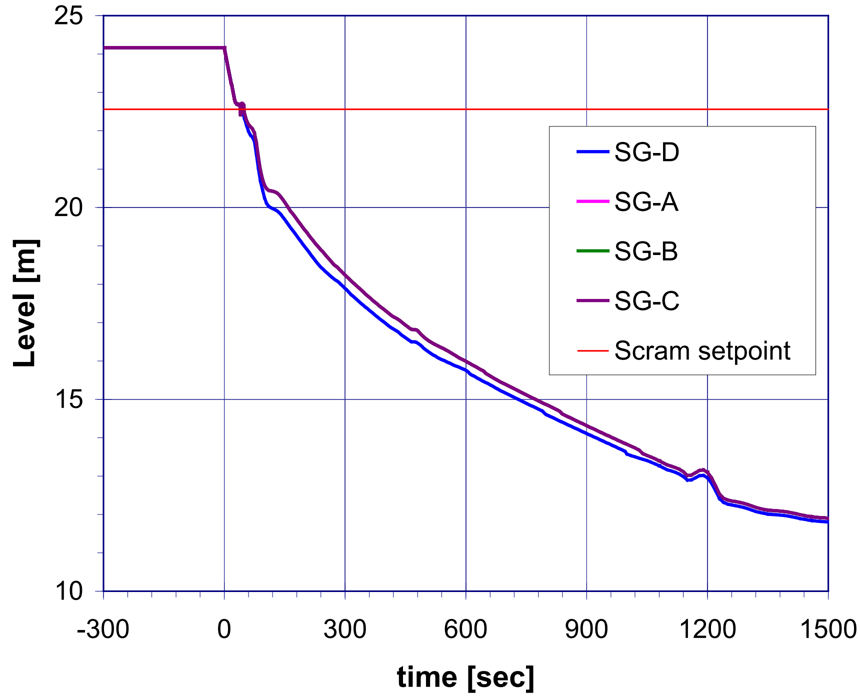

Figure A.1.1.1a Water Level in the Steam Generators (Case A.1.1 short-term)

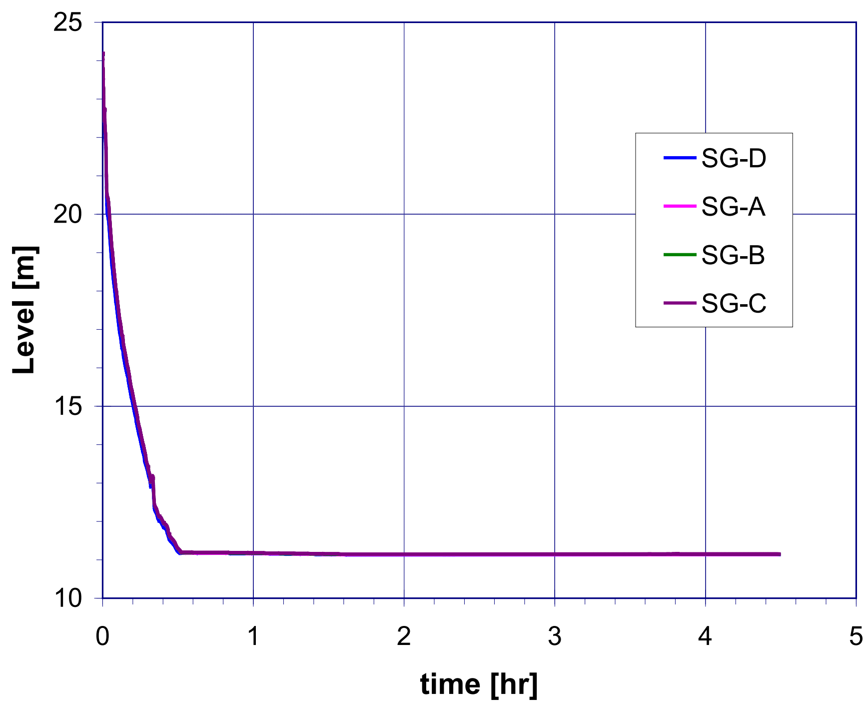

Figure A.1.1.1b Water Level in the Steam Generators (Case A.1.1 longer-term)

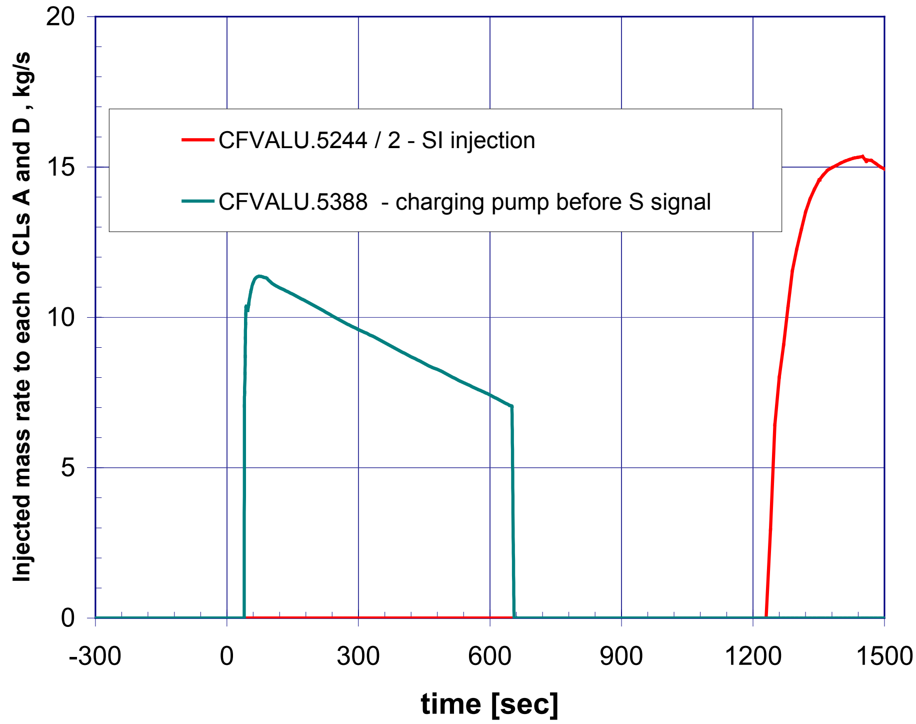

Figure A.1.1.2 Coolant Injection Rates before and around the Time of S Signal (Case A.1.1)

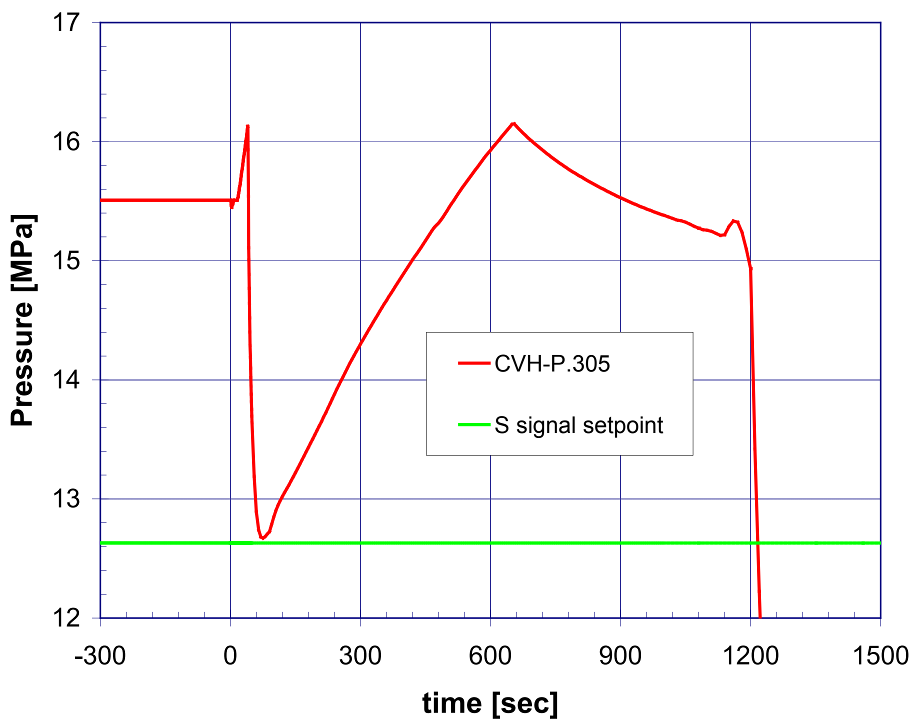

Figure A.1.1.3a Pressure in the Pressurizer (Case A.1.1 short-term)

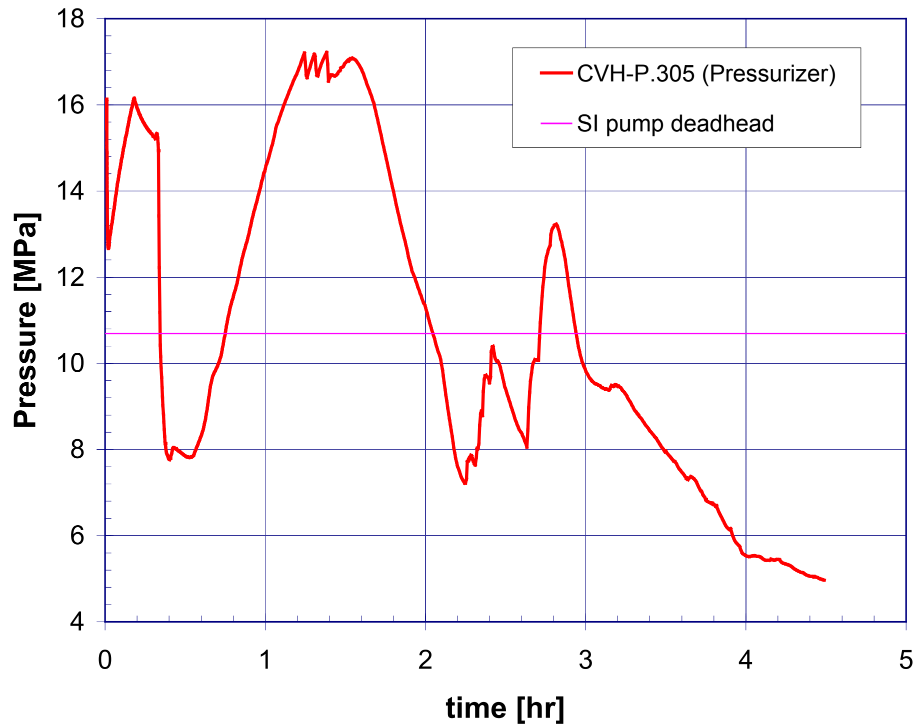

Figure A.1.1.3b Pressure in the Pressurizer (Case A.1.1 longer-term)

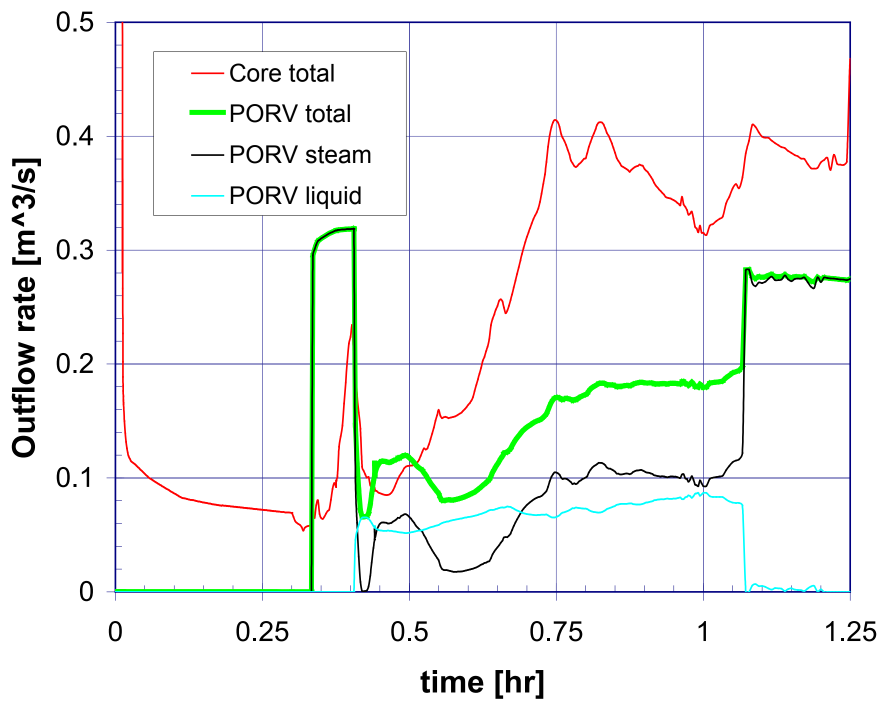

Figure A.1.1.4a PORV and Net Core Volumetric Outflow Rates (Case A.1.1 short-term)

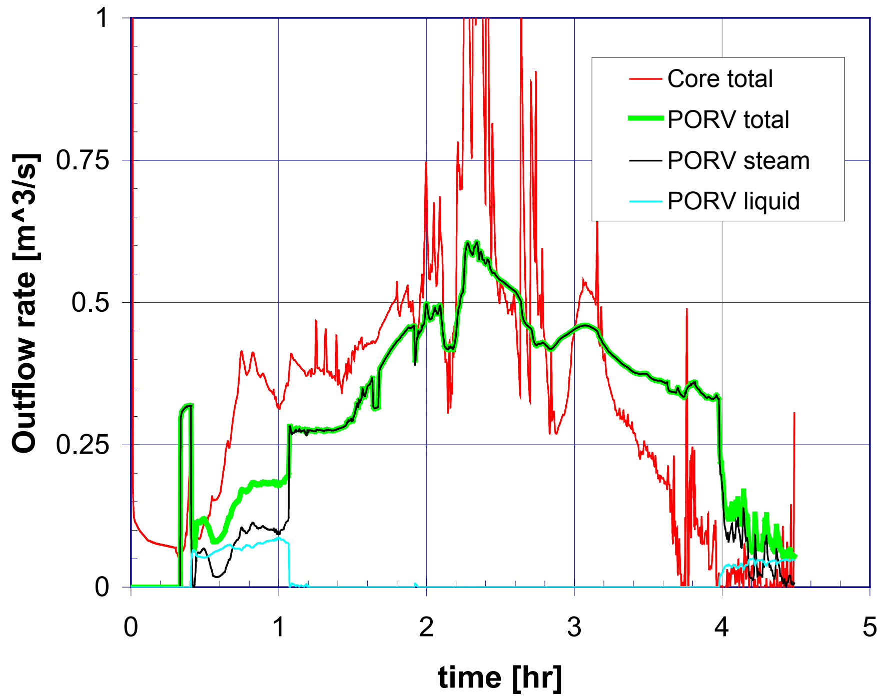

Figure A.1.1.4b PORV and Net Core Volumetric Outflow Rates (Case A.1.1 longer-term)

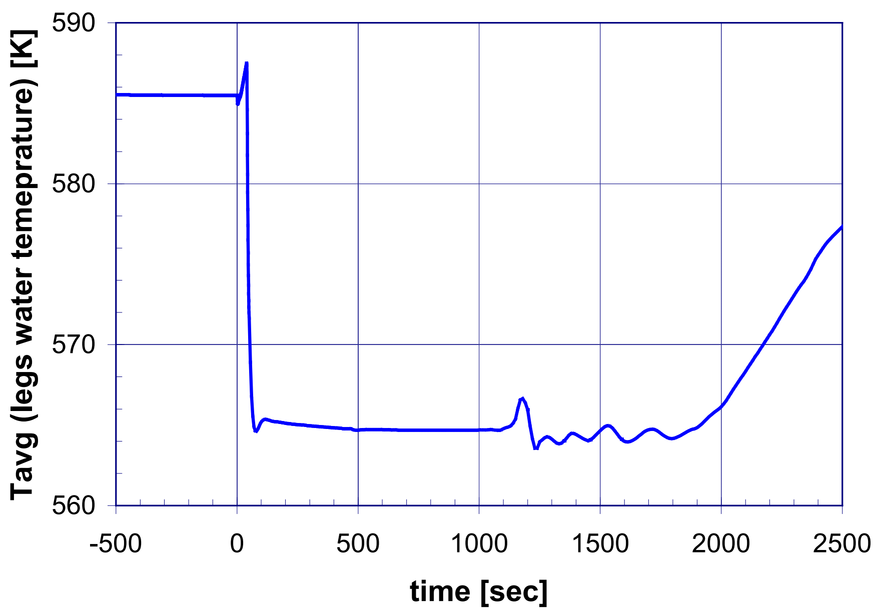

Figure A.1.1.5a Average Liquid Temperatures in the Hot and Cold Legs (Case A.1.1 short-term)

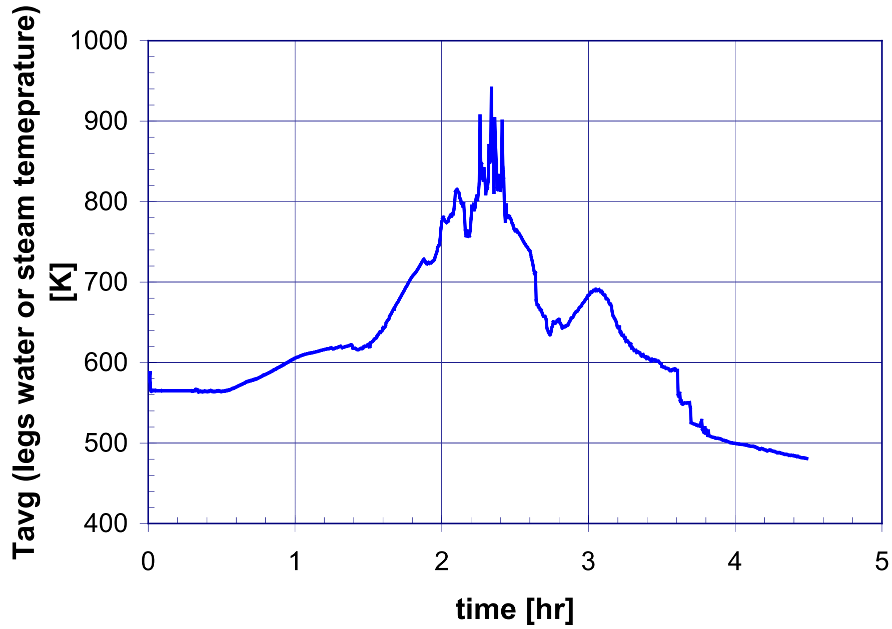

Figure A.1.1.5b Average Fluid Temperatures in the Hot and Cold Legs (Case A.1.1 longer-term)

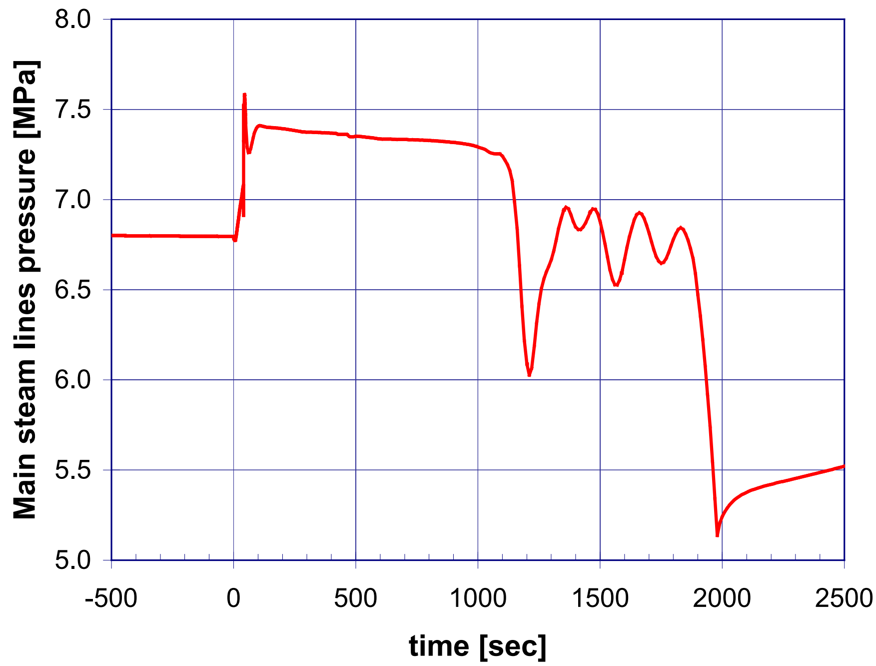

Figure A.1.1.6 Average Main Steam Line Pressure (Case A.1.1)

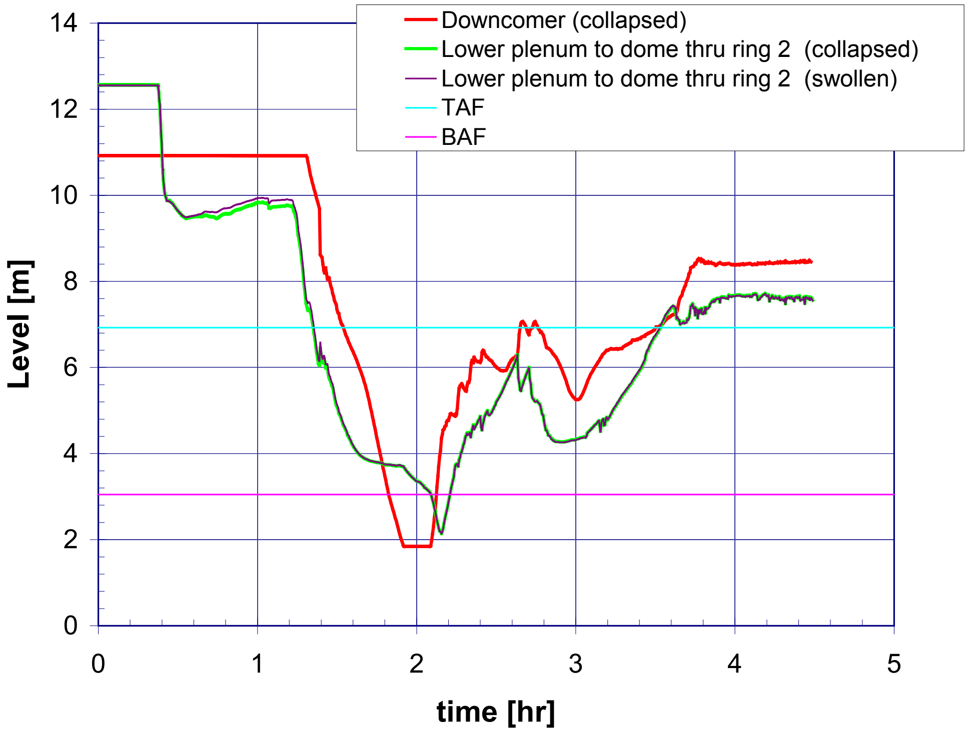

Figure A.1.1.7a Water Levels in the Vessel (Case A.1.1)

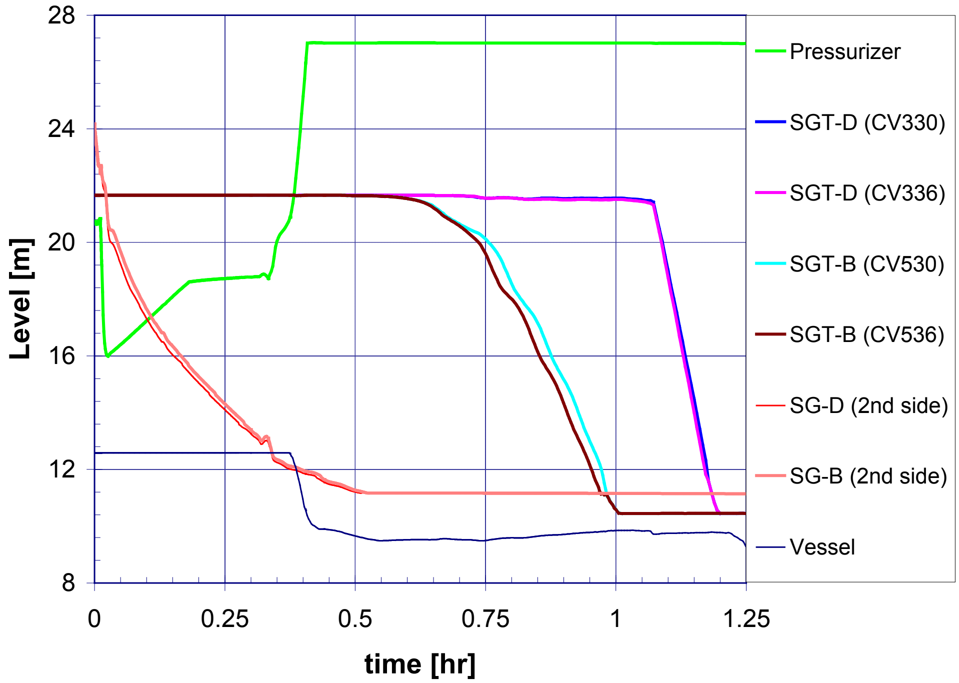

Figure A.1.1.7b Water Levels throughout the Plant (Case A.1.1)

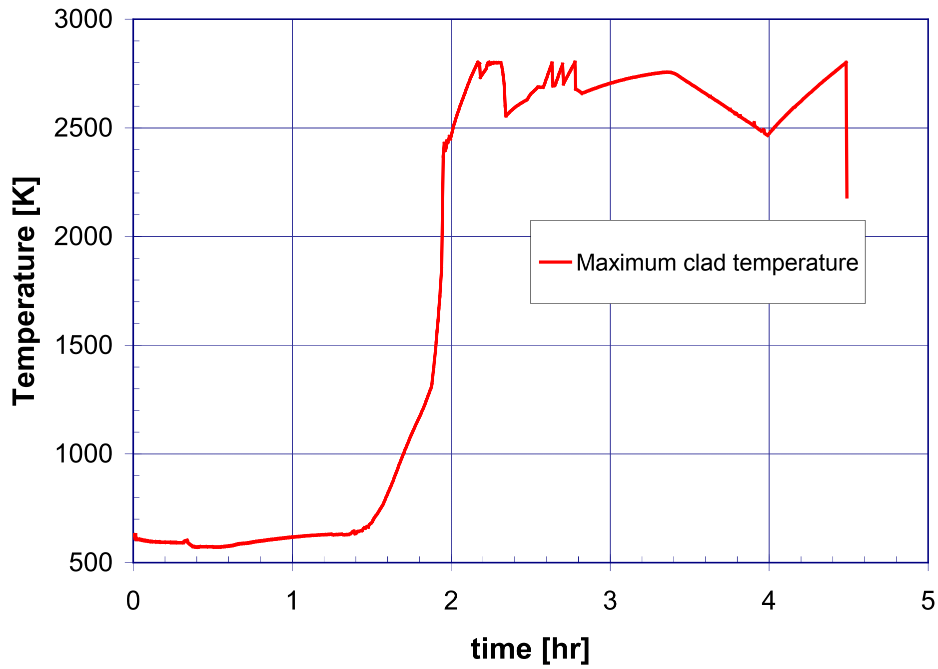

Figure A.1.1.8 Peak Cladding Temperature (Case A.1.1)

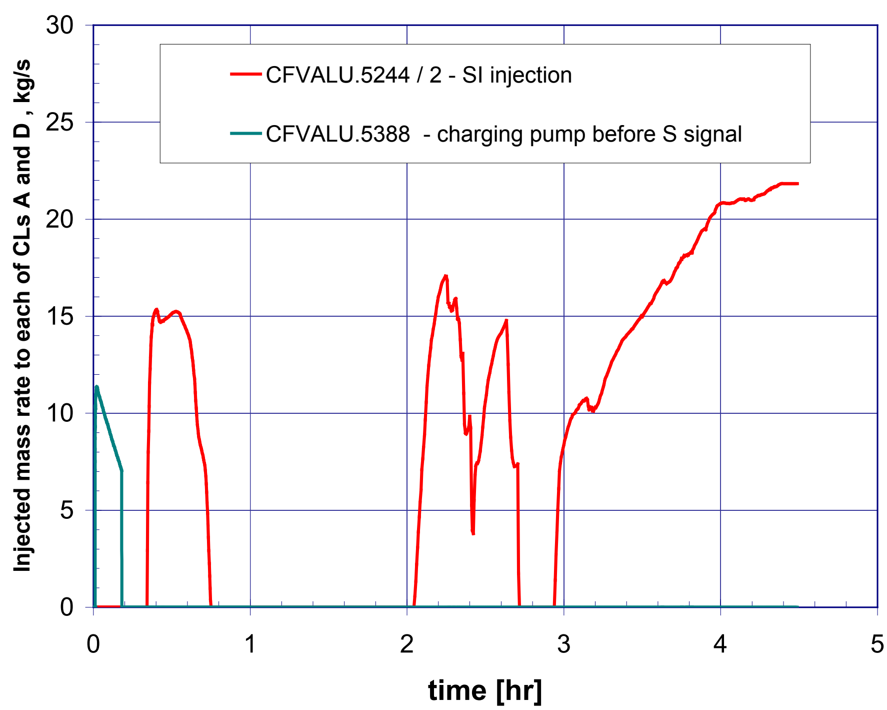

Figure A.1.1.9 ECCS Injection Rates (Case A.1.1)

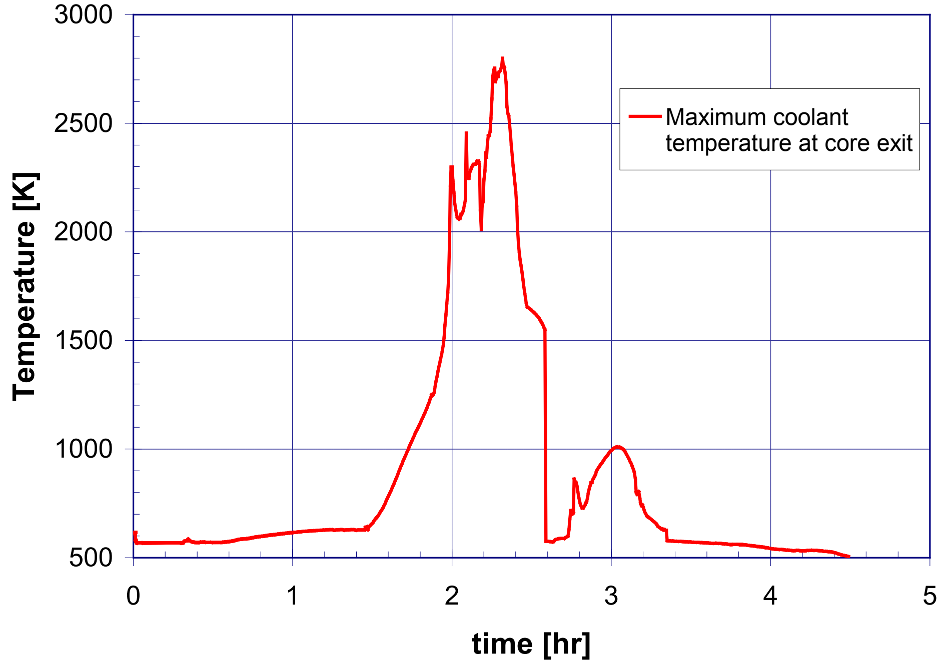

Figure A.1.1.10 Core Exit Temperature (Case A.1.1)

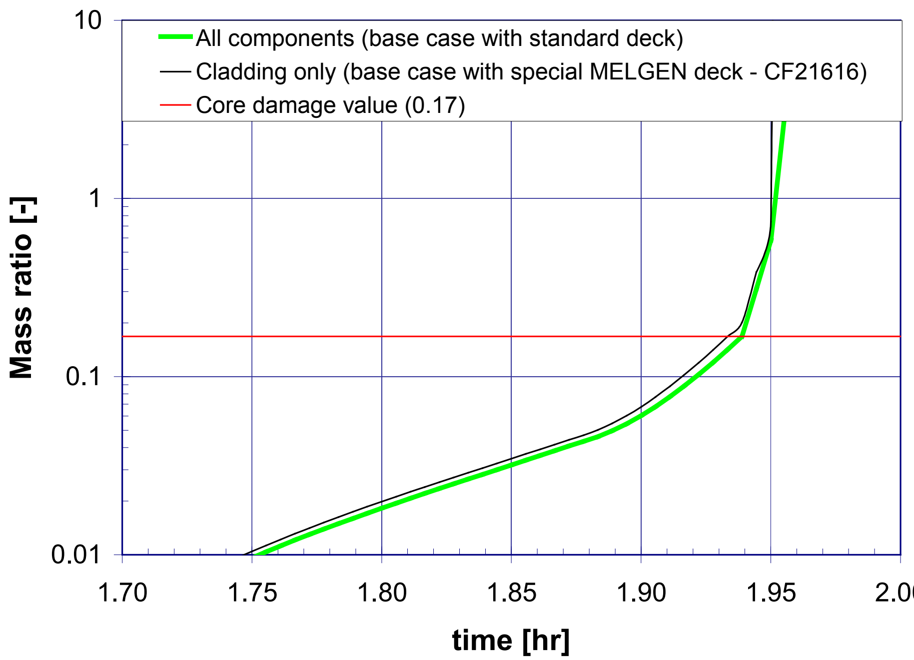

Figure A.1.1.11 Spatially Maximized Mass Ratio, Zirconium Oxide to Zirconium (Case A.1.1)

only if the open PORV can depressurize the RCS to less than about 5.1 MPa. Taking now a result from the calculation, the pressure that the RPV quickly falls to after the PORV is opened is about 8 MPa. At that pressure, one SI pump injects at only 29.7 kg/s, and the heat transfer rate required to bring the liquid up to saturation, boiling being discounted, is 61 MW. Since this is less than the decay heat, boiling of the coolant will occur and tend to raise its pressure and temperature further, if other forms of cooling are not available. The SI pump deadheads at 10.7 MPa. It can thus be anticipated from these simple estimates that the ability of a single SI pump to sustain injection is in doubt.

A detailed account of the predicted transient is as follows. Scram due to low steam generator water levels occurs at 40 seconds (see Figure A.1.1.1a). Beginning at this time, normal-operation injection by one of the two charging pumps is accounted for. At earlier times, the injection by one charging pump that occurs during normal plant operation is assumed in the model to be exactly canceled by normal letdown. The letdown is not included in the model, and it is supposed to end actually at scram while injection by a charging pump is supposed to continue after scram. This injection by one charging pump ends at 10.9 minutes upon depletion of the volume control tank (VCT). Figure A.1.1.2 depicts early injection into the RCS.

Scram at 40 seconds results in a closure of the turbine throttle valve. It is assumed that the previously shut condenser steam dump valves become active at this time, immediately coming under automatic control with the attempt to keep the T_{avg} constant. Here, the T_{avg} is the simple four-loop average of the liquid temperatures in the hot and cold legs; in the MELCOR model, the hot and cold leg nodes nearest the steam generators are used. The assumed setpoint for T_{avg} is 564.8 K (557 °F). This action by the dump valves keeps main steam line pressures below the opening pressures of the steam generator PORVs (one per loop) and the steam generator safety valves (five per loop), which are all individually modeled. (The steam generator safety valves with the lowest opening pressure (8.205 MPa) are modeled by FL476, FL576, FL676, and FL376 in loops A through D[22], respectively. The steam generator PORVs, which open at 7.79 MPa, are modeled by flow paths FL477, FL577, FL677, and FL377.)

At 18 minutes, the operators trip all RCPs in preparation for initiating the B&F. At 20 minutes, the operators fully open one of the two pressurizer PORVs and manually trigger the "S" signal that initiates the ECCS. Figure A.1.1.3a shows the pressure in the pressurizer at early times. The events and processes that influence this pressure are: FW loss (RCS pressure rises); scram and a period of about 30 seconds duration during which the dump valves open widely to bring the T_{avg} to setpoint (RCS pressure drops); injection by one charging pump into the RCS prior to depressurization with the PORV (RCS pressure rises); end of this injection due to VCT depletion (RCS pressure drops); RCP pump trip (RCS pressure rises); and the opening of one PORV (RCS pressure drops). Note that at 80 seconds, the low RCS pressure setpoint for an automatic initiation of the S signal is narrowly missed. Had it been reached, the injection by the charging pump would have stopped. This charging pump action is premised on the assumption that the realignment of its suction to the RWST taking place upon receipt of the S signal would fail, which is consistent with the scenario definition that does not credit charging pumps for ECCS purposes. Also, were there to be an S signal at this early time, actual injection by the SI pump would not have started then because of the high RCS pressure. Figure A.1.1.3b shows the full history of the pressurizer pressure.

The manually initiated S signal activates the ECCS at 20 minutes. Credit is given to one of two SI pumps and one of two RHR pumps. Each pump, as well as the charging pump in use between the times of scram and the S signal, is configured to divide its discharge equally between the cold legs of Loops A and D. (A footnote in Section 2.1 describes later correction of this treatment, but it is of little consequence for this scenario since all loops are intact.) Both charging pumps are assumed to be unavailable. (Although a charging pump had been in use earlier, it is assumed that the necessary realignment of the suction point of the charging pumps from the VCT to the RWST fails.) Actual injection rates by the available and activated SI and RHR pumps are calculated in accordance with their flow rate versus head curves. In fact, at the time of the signal, the primary pressure is too high for either pump to inject. As the pressure falls due to the open PORV, the SI pump begins to inject at 20.7 minutes. During the calculation (ending at 4.49 hours), the primary pressure is always too high to permit injection by the RHR pump. Because of the short calculation time, ECCS recirculation mode is not attained. (In the recirculation mode, one RHR

[22] The model's control volumes and flow paths are designated CV*ixy* and FL*ixy* with *i* = 3, 4, 5, and 6 corresponding respectively to loops D, A, B, and C in plant nomenclature.

pump takes suction from the sump, and the other credited ECCS pumps then take their suction from that RHR pump.)

As has been pointed out in the opening paragraph, SI injection at any rate significantly less than one pump's maximum rate supplies insufficient cooling to equal the decay heat by sensible heatup of the injected water alone. Boiling will occur unless there is additional cooling; boiling will result in re-pressurization unless the PORV outflow is sufficiently great. Initially, additional cooling is provided by the steam generators, resulting in a period of approximately steady RCS pressure lasting from about 0.4 to 0.5 hours (about 8 MPa; see Figure A.1.1.3b). Steam generator cooling remains appreciable prior to the time of steam generator dryout (0.55 hours, as Figure A.1.1.1b shows). Later, there is no additional cooling, and re-pressurization depends on the PORV outflow. In general, the outflow through the PORV is a mixture of liquid and steam. Figures A.1.1.4a and A.1.1.4b show the liquid, steam, and total PORV volumetric outflow rates. Also shown is the total volumetric rate at which liquid and steam leave the core. (In the case of the core, the rate of inflows is subtracted from the rate of outflows to define the net outflow.) While the steam generators are not dried out, the core outflow almost always remains below the PORV outflow, and the pressure does not rise. After steam generator dryout, the core outflow increasingly exceeds the PORV outflow during the times shown in Figure A.1.1.4a. This imbalance results in re-pressurization of the RCS (Figure A.1.1.3b) such that the SI pump is deadheaded at 0.75 hours. The pressurization continues until SRV outflows begin at about 1.25 hours.

Due to steam generator dryout and consequent loss of cooling starting at 0.55 hours (33 minutes), the dump valves react according to their control logic and open widely in an unsuccessful attempt to regulate T_{avg} (Figure A.1.1.5a) to its setpoint value. This opening results in a rapid fall of the main steam lines pressure (see Figure A.1.1.6), and at 33.0 minutes the MSIVs close because of a trip on excessive rate of the main steam line pressure drop (i.e., the pressure falls faster than 2.23 psi per second (15.4 kPa/s)). This closure effectively seals off the steam generators and makes the dump valves inoperable. Coolant temperature over the long term is shown by Figure A.1.1.5b.

Figure A.1.1.7a shows the water level in the vessel, which is defined by summing the heights of the columns of water in a stack of control volumes running from the lower plenum to the dome, by way of the second computational ring in the regions of radial nodalization. (See Figure 2.1.2 for the vessel nodalization. The difference between levels defined by swollen versus collapsed column heights is not significant in this case.) Figure A.1.1.7b co-plots the vessel water level with other water levels throughout the plant. The core starts to uncover at about 1.35 hours. It is partially uncovered during 1.35 to 3.5 hours and fully uncovered during a ten-minute period around 2.2 hours.

A period of falling pressure begins at 1.56 hours, shortly after the core begins to uncover. As the fuel uncovers, a significant portion of the decay heat goes to heatup of the fuel (as shown by the climbing peak clad temperatures in Figure A.1.1.8). The SI pump is able to inject again at 2.04 hours. Except for a short period around 2.8 hours when it is deadheaded again, the SI pump is able to inject uninterruptedly for the rest of the calculation. Figure A.1.1.9 shows the ECCS injection rate to each of the cold legs of Loops A and D, due to the one credited SI pump (and, before the S signal at 20 minutes, one charging pump).

During the period from 1.80 to 2.02 hours, gap release is predicted to occur in all five of the computational rings of the core. (In MELCOR, gap release is modeled in a given ring when its axial maximum of cladding temperature reaches 1173 K.) Relocation of both cladding and fuel (modeled at a threshold of 2800 K) is predicted. The lower core support plate also fails starting at 2.27 hours. Figure A.1.1.10 shows the maximum core exit temperature, which exceeds 922 K (1200 °F) at 1.68 hours. Figure A.1.1.8 shows the peak clad temperature (maximum over the entire core). This temperature first exceeds 1478 K (2200 °F) at 1.91 hours. The calculation is terminated at 4.49 hours, because the damage to the core is already extensive by all measures.

As already discussed in the main part of this report, throughout this document the core damage surrogate measure based on maximum thickness of the cladding oxide layer is actually calculated using the ratio of the mass of zirconium oxide to that of zirconium. Moreover, except for the calculations made for Surry, the net core region masses of zirconium oxide and zirconium are used instead of the specific masses of cladding. The net masses include those of other components such as the control rod guide tubes as well as the canister and blade in the case of a BWR. These component-specific masses are not available in the default MELCOR output files. The base case of the present scenario, however, has been repeated with no change except for the definition of a control function equal to this measure, thus making it available in the output. The new control function uses the masses of the cladding component. This special deck is not described or documented anywhere else. As Figure A.1.1.11 shows, the difference relative to the standard post-run analysis that uses the lumped masses of zirconium components of all types is less than one minute in the indicated time of core damage.

A.2.2.2 Results of Case A.1.2 (Early Reactor Trip)

Case A.1.2 addresses uncertainty in the time of scram. In the base case, scram occurs automatically at about 40 seconds due to low water levels in the steam generators. The predicted steady-state SG water level is 24.165 m (with respect to the inside bottom of the reactor pressure vessel [RPV]), to compare with the estimated plant value of 24.30 m. The estimated plant value for the scram setpoint is 22.69 m. (The model controls the steady-state mass of liquid to a known plant value of 53,347 kg per steam generator instead of controlling water level.) The base case uses 22.557 m as the scram setpoint, which preserves the difference between the plant's normal operation value and the plant value for the scram setpoint. This sensitivity case arbitrarily assumes a higher scram setpoint of 23.600 m (to mimic manual scram based on operator action in response to degrading conditions preceding the bus loss), which leads to an earlier scram. Otherwise, this sensitivity case is the same as the base case.

The scenario progression is outlined in Table A.4 and is illustrated in Figures A.1.2.1 to A.1.2.10. Scram stemming from low SG water levels at the assumed higher setpoint occurs at 11 seconds. (Figure A.1.2.1 shows the steam generator levels.) Beginning at this time, the normal-operation injection by one of the two charging pumps is accounted for. This injection ends at 10.6 minutes, when the volume control tank (VCT) is depleted.

Table A.4 Timing of Key Events Predicted for the MELCOR Model Bus Loss with Bleed and Feed Sensitivity Case 2 Calculation (Times in Hours Unless Stated Otherwise)

Event Description	Time
Loss of all FW because of loss of Bus 111	0 seconds
Scram (due to low water levels in steam generators)	11 seconds
Injection by one charging pump (in excess of the letdown) begins	11 seconds
Condenser steam dump valves open	11 seconds
Injection by one charging pump ends (because of VCT depletion)	10.6 minutes
Manual trip of all RCPs	18.0 minutes
Manual opening of pressurizer PORV-1 ; manual send of S signal	20.0 minutes
Injection by one SI pump drawing from the RWST begins	20.7 minutes
MSIVs close (as a result of high containment pressure); condenser steam dump valves close	1.48
SG PORVs open	1.72 to 3.38(1)
SG s dry out	1.8(2)
SI pump is deadheaded	2.06 to 3.04
Core is partially uncovered	2.5 to 4.9
Coolant temperature at the core exit reaches 922 K (1,200 °F)	3.06
Switch to ECCS recirculation mode (RHR heat exchanger not available)	7.74
Calculation end	27.8
Accumulators inject	Not available
Containment sprays actuate	Not available
SG safety valves open	Does not occur
Gap release	Does not occur
Sump water becomes saturated	Does not occur
Peak clad temperature reaches 1,478 K (2,200 °F)	Does not occur

1 Loops B, C, and A. Loop D (pressurizer loop) SG-PORV does not open.
2 Loops B, C, and A. Dryout in Loop D (pressurizer loop) occurs at 1.1 hours.

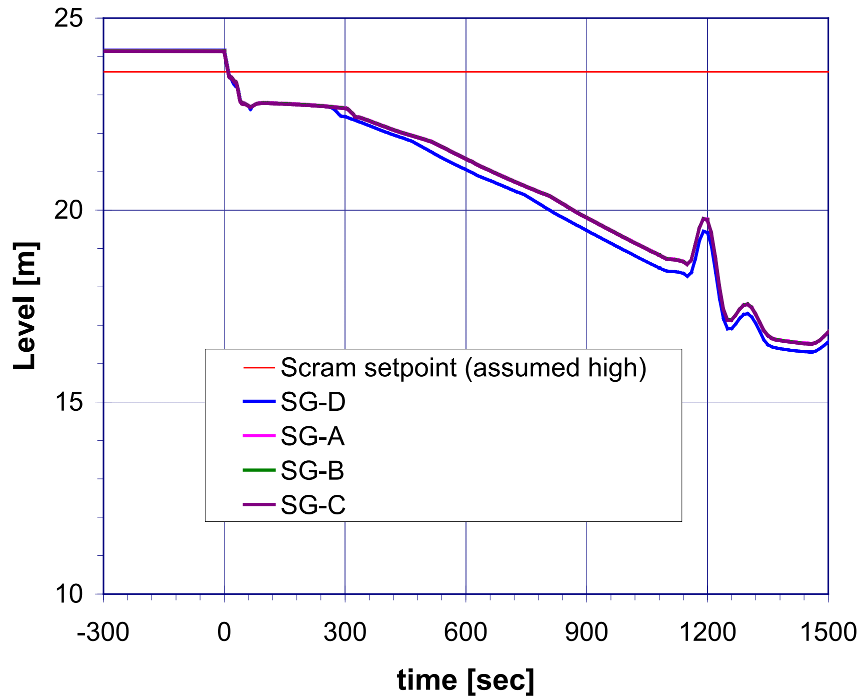

Figure A.1.2.1a Water Level in the Steam Generators (Case A.1.2 short-term)

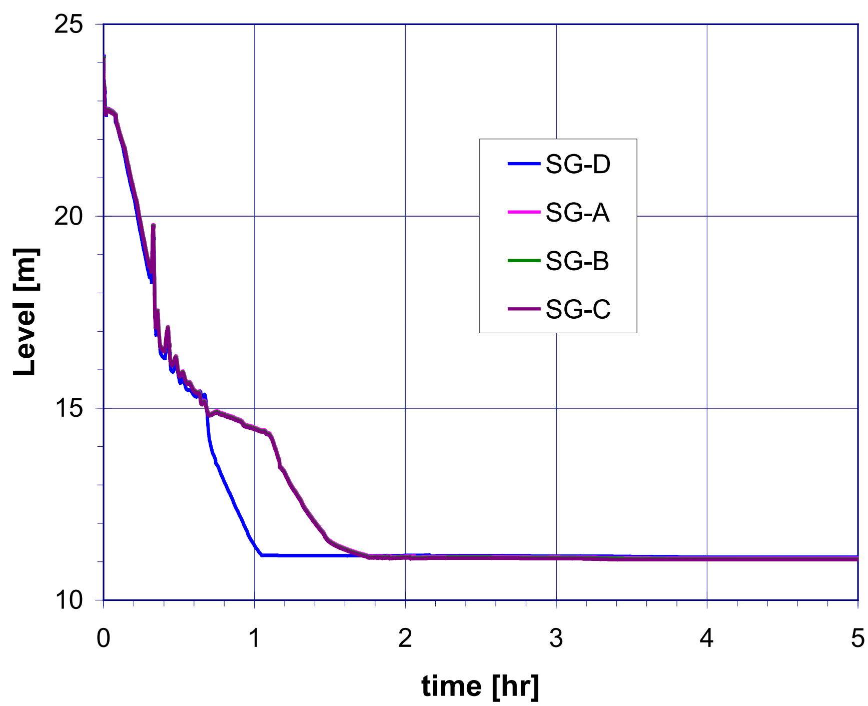

Figure A.1.2.1b Water Level in the Steam Generators (Case A.1.2 longer-term)

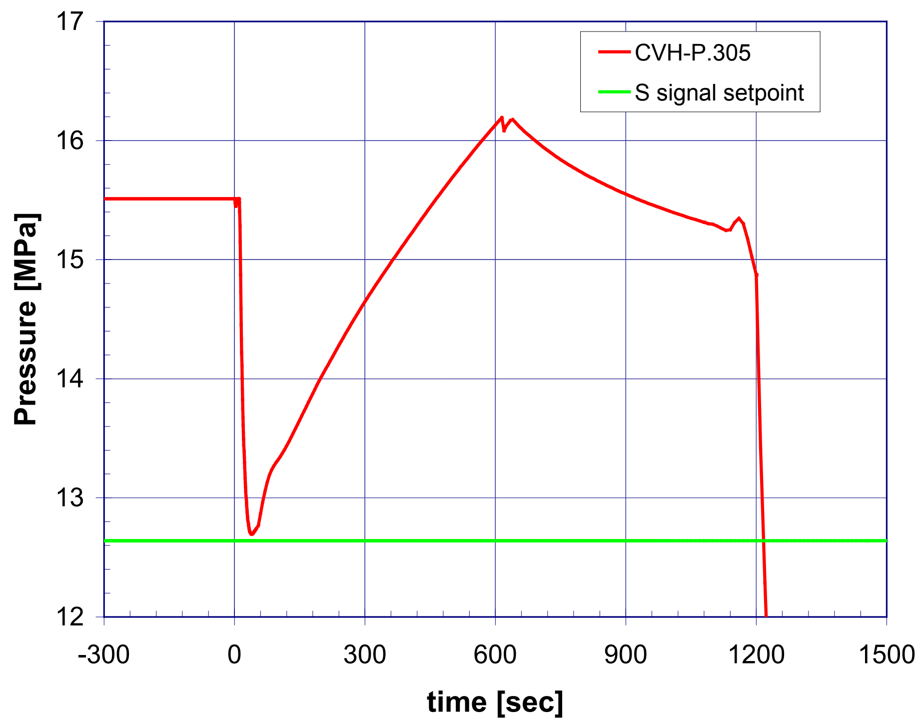

Figure A.1.2.2a Pressure in the Pressurizer (Case A.1.2 short-term)

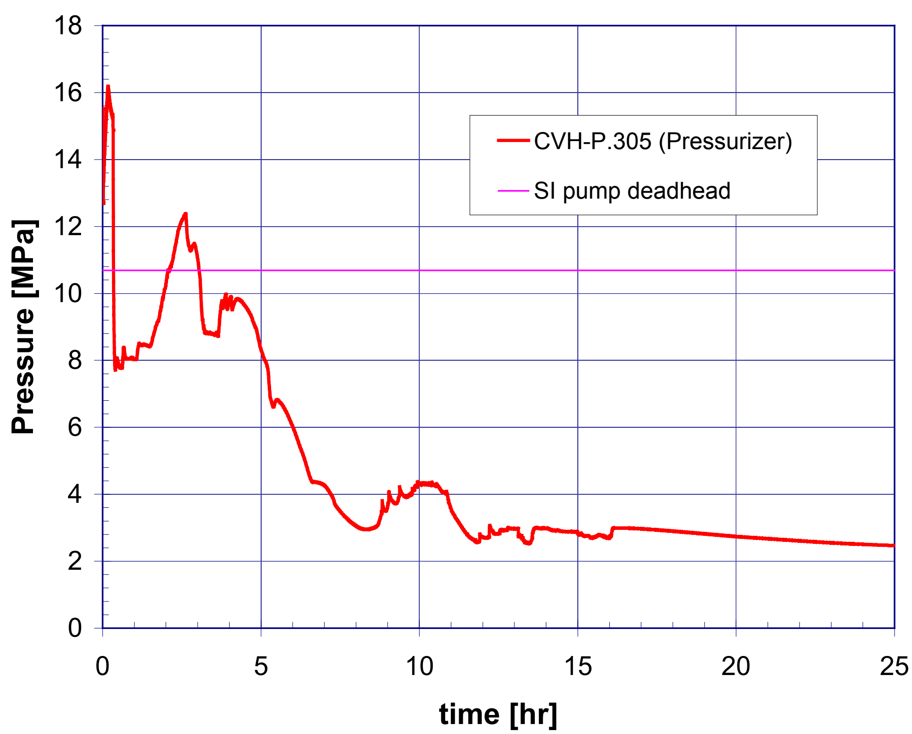

Figure A.1.2.2b Pressure in the Pressurizer (Case A.1.2 longer-term)

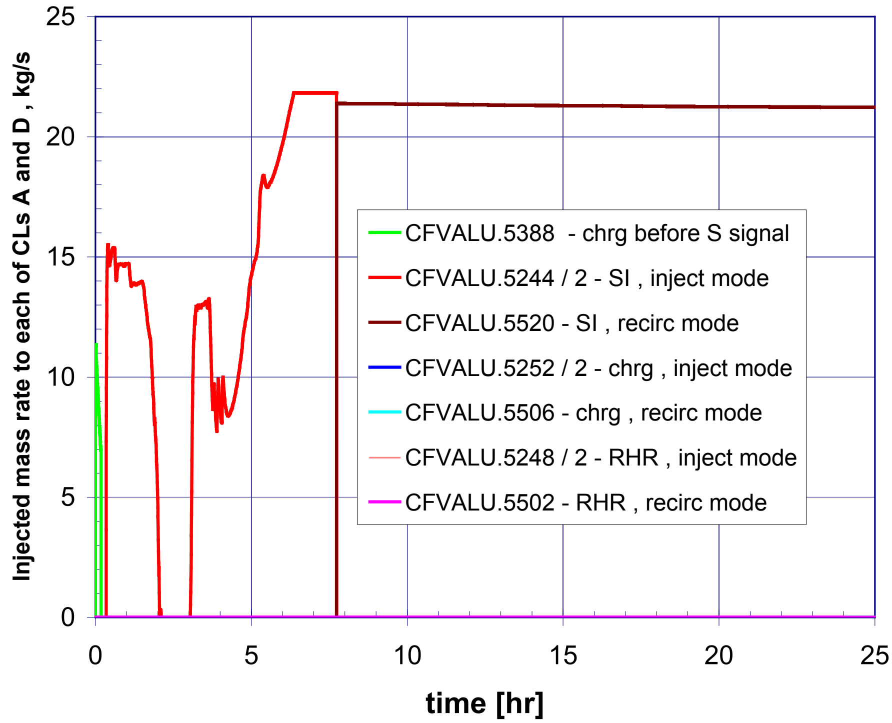

Figure A.1.2.3 ECCS Injection (Case A.1.2)

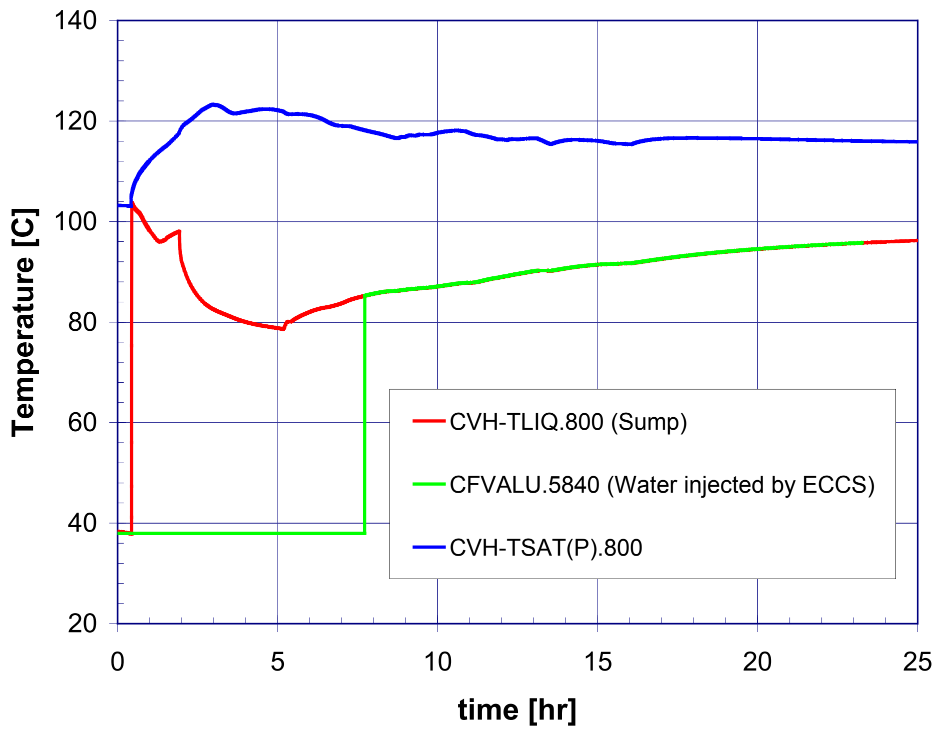

Figure A.1.2.4 Saturation and Sump Water Temperatures (Case A.1.2)

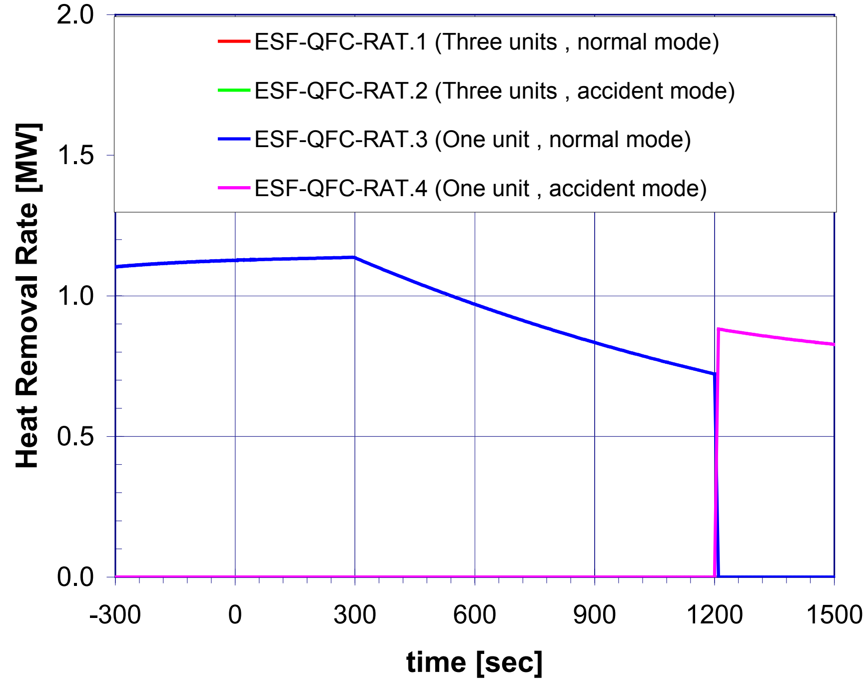

Figure A.1.2.5a Fan Cooler Cooling Rate (Case A.1.2 short-term)

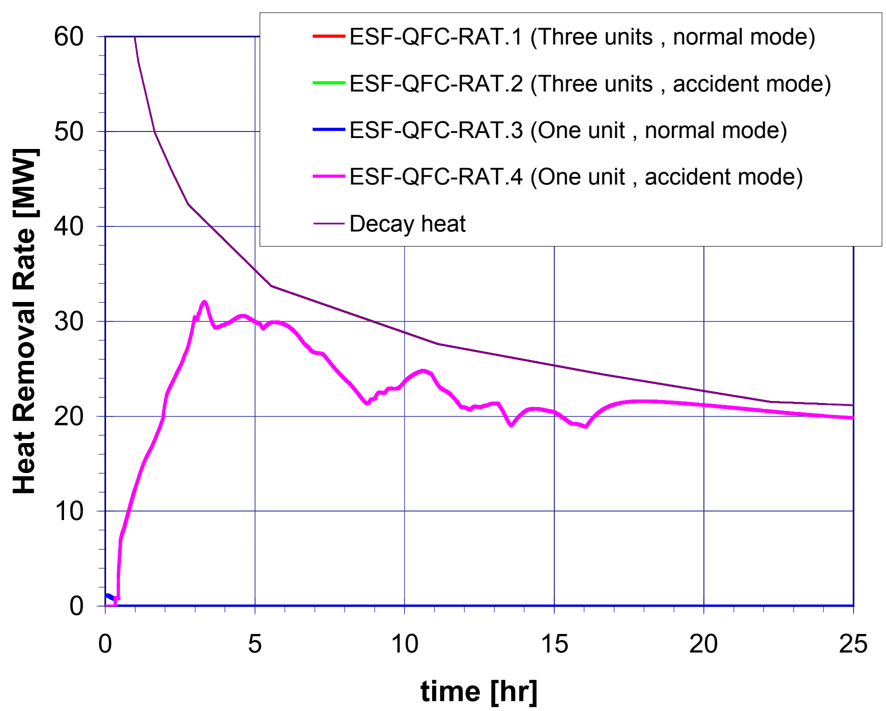

Figure A.1.2.5b Fan Cooler Cooling Rate (Case A.1.2 longer-term)

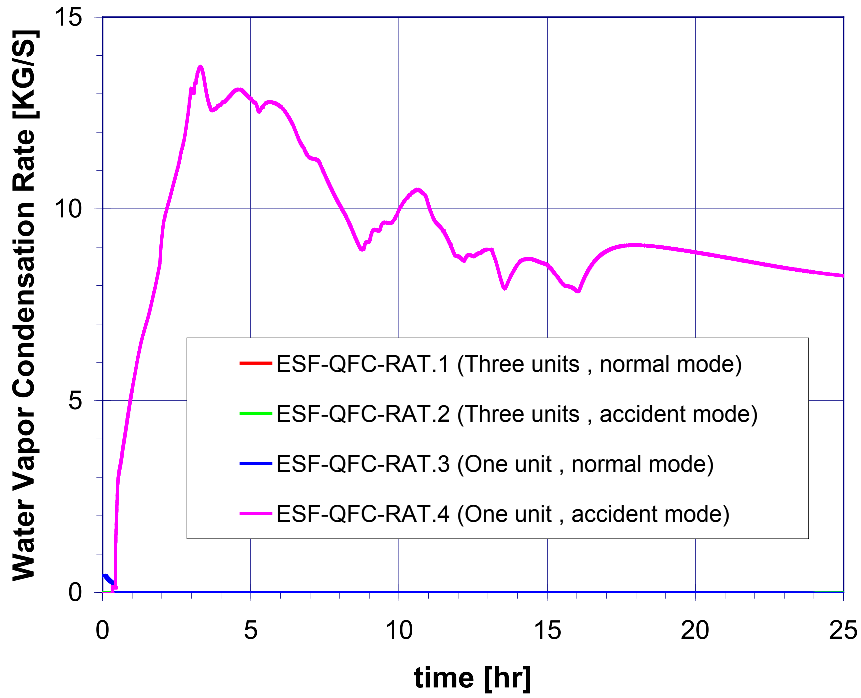

Figure A.1.2.6 Fan Cooler Condensation Rate (Case A.1.2)

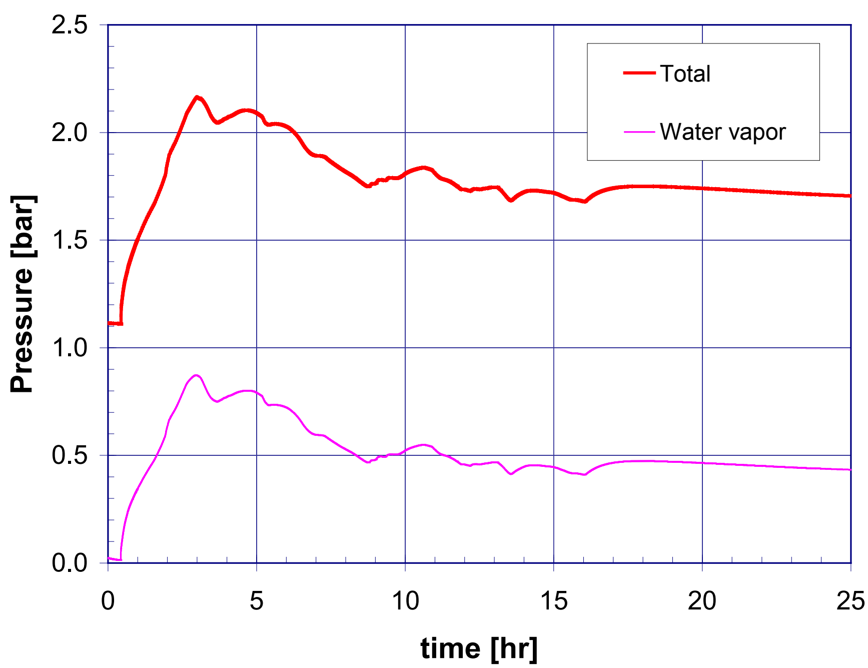

Figure A.1.2.7 Containment Pressure (Case A.1.2)

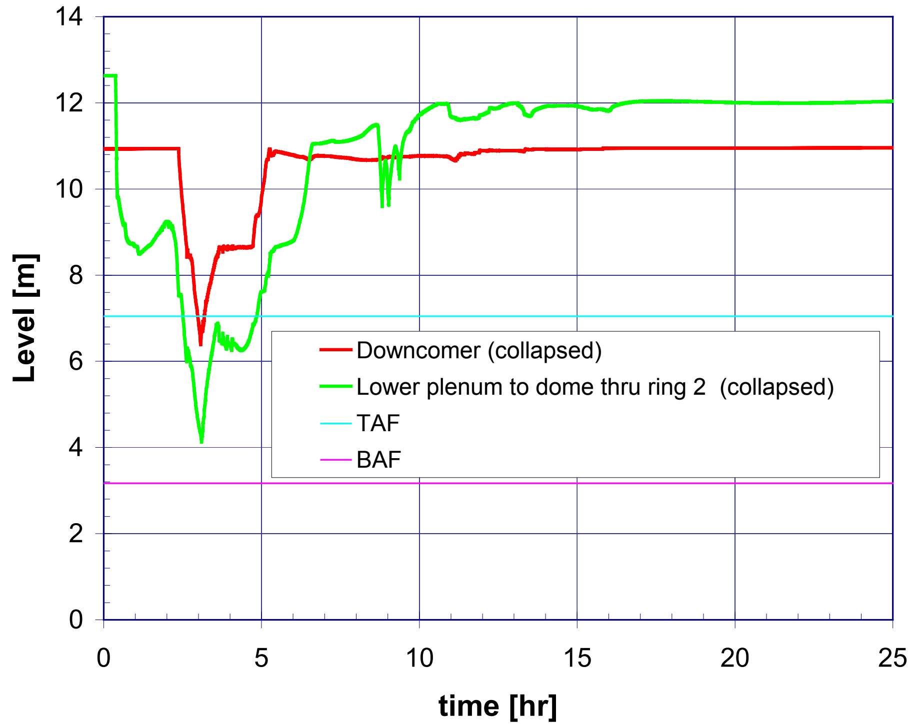

Figure A.1.2.8 Water Level in the Vessel (Case A.1.2)

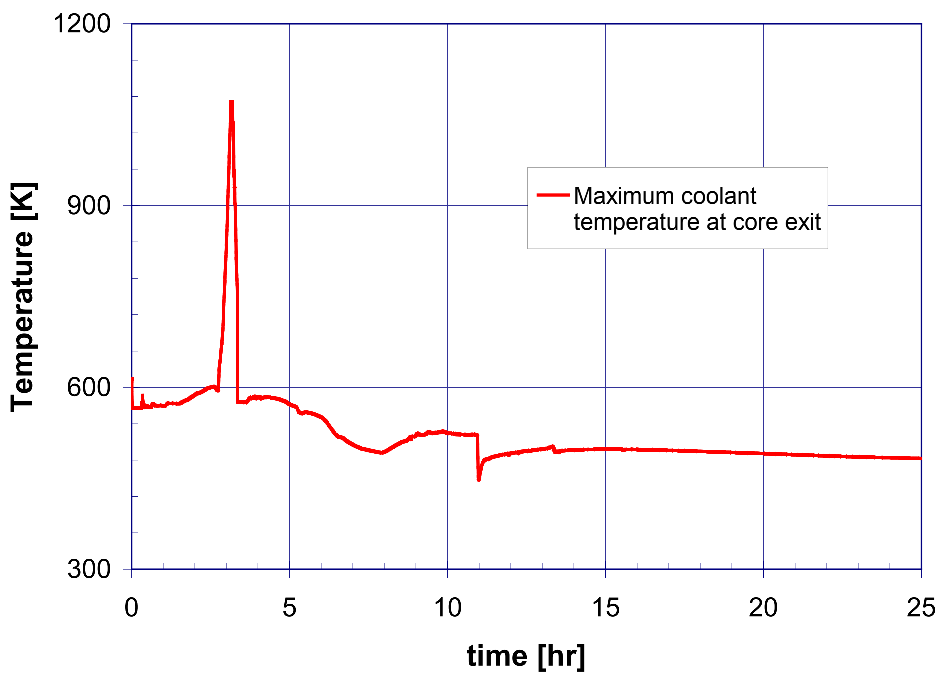

Figure A.1.2.9 Core Exit Temperature (Case A.1.2)

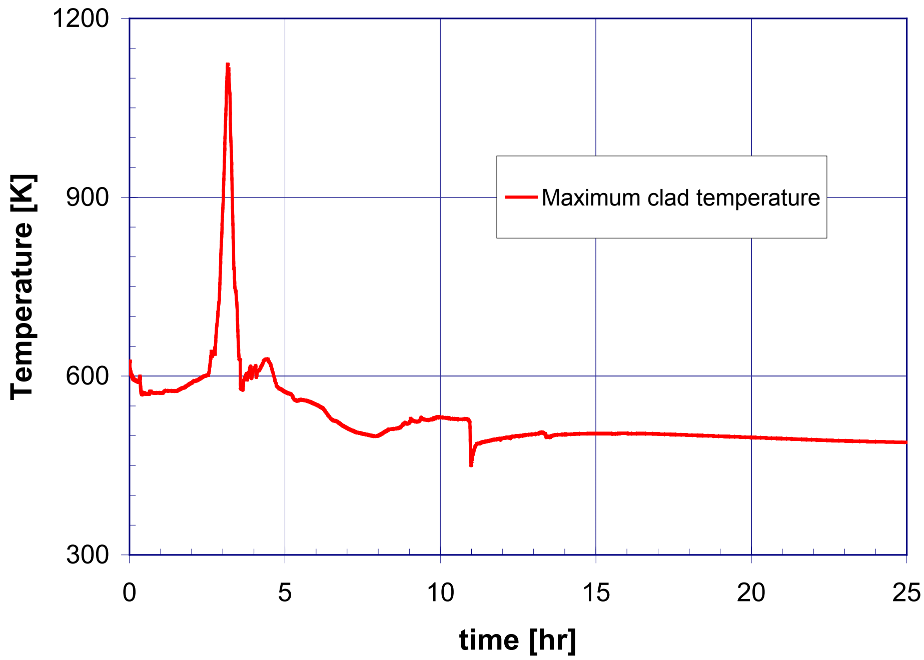

Figure A.1.2.10 Peak Cladding Temperature (Case A.1.2)

Scram results in a closure of the turbine throttle valve and the opening of the condenser steam dump valves. The dump valves control attempts to keep T_{avg} constant at 564.8 K. This action of the dump valves keeps the main steam line pressures below the opening pressures of the SG PORVs and SRVs (the PORVs open eventually).

At 18 minutes, the operators trip all reactor coolant pumps in preparation for initiating the B&F. At 20 minutes, the operators fully open one of the two pressurizer PORVs and manually trigger the S signal that initiates the ECCS. As in the base case, an early automatic S signal does not occur. The RCS pressure reaches a minimum of 12.7 MPa at 45 seconds, just slightly above the low-pressure S signal setpoint of 12.63 MPa. Figures A.1.2.2 show the RCS pressure.

The manually initiated S signal activates the ECCS at 20 minutes. Credit is given to one of two SI pumps and one of two RHR pumps. Both charging pumps are assumed to be unavailable. Actual injection by the SI pump begins at 20.7 minutes. During the period of the ECCS injection mode, the RCS pressure is too high for the RHR pump to inject.

All MSIVs close at 1.48 hours because of high containment pressure. This closure seals off the dump valves. The steam lines pressure rises until the SG-PORVs open at 1.72 hours. The PORVs remain closed after 3.38 hours. No SG relief valves open during the calculation.

Dryout of the steam generators occurs during 1.1 to 1.8 hours depending on the loop; see Figure A.1.2.1b. Consequently, the RCS pressures rise from about 1.5 to 2.6 hours, and the SI pump is deadheaded from 2.06 to 3.04 hours (Figure A.1.2.2b).

At 7.74 hours, the RWST is drawn down to the level (so-called Lo-2) where the switch to the ECCS recirculation mode occurs. The model assumes that Lo-2 is attained when the volume delivered by the ECCS pumps and containment sprays is 187,736 gallons, which was calculated to represent the RWST initially at 89 percent full. In this calculation, drawdown of the RWST results from the one operating SI pump only: the other pumps and sprays are not available. In the recirculation mode, one RHR pump takes suction from the containment sump. After passing through the RHR heat exchanger, the pump's output then flows into all credited SI and charging pumps. These pumps are configured in parallel, for final injection into the cold and/or hot legs. In the present scenario, the RHR heat exchanger provides no cooling because of the assumed unavailability of the service water, and the RHR pump's outflow is into the one available SI pump and then in two cold legs. Figure A.1.2.3 shows the injections resulting from the various pumps and modes. Cold legs A and D each receive water injected at the plotted rates. (A footnote in Section 2.1 describes later correction of this treatment, but it is of little consequence for this scenario since all loops are intact.). Here, by convention, during the recirculation mode, the injection is associated with the pump(s) downstream of the RHR pump. If the primary pressure falls low enough to allow it, the RHR pump would be aligned to inject directly to the RCS instead of feeding the other pumps. The plot would then associate the flow with the RHR pump.

During the time from 2.06 to 3.04 hours, the SI pump is deadheaded by high RCS pressure. At all other times later than 20.7 minutes, the ECCS injection occurs continuously. No water-level-dependent control of the injection is invoked, with the result that the pressurizer is overflowing at most times after 1 hour. At times later than about 5 hours, the flow through the PORV is almost pure liquid. The temperature of this liquid is never less than 205 °C (478 K), while the saturation temperature at the containment pressure is never more than 123 °C (396 K). Therefore, this effluent quickly becomes steam in the containment, and the fan cooler is thus effective in removing heat from the plant.

From the time that the recirculation mode is initiated, the sump is filled to a depth of about 8 m above its bottom. During this mode, the temperature of the water increases from 85° C (at 7.74 hours) to 96° C (369 K) (at 25 hours). Therefore, adequate suction for the recirculation mode exists. Figure A.1.2.4 shows the temperature of the sump water and the saturation temperature. In the model's single-node treatment of the containment, control volume CV800 represents the sump, dome and all other parts of the containment.

Besides the two ECCS pumps, the scenario also credits one of the four containment fan coolers. Containment sprays and accumulators are not credited. The fan cooler operates at all times, including during the pre-transient steady state. However, the S signal causes it to change to the so-called accident mode (Figure A.1.2.5a). From about 5 to 25 hours, the fan cooler is removing heat at around 30 to 20 MW. As Figure A.1.2.5b shows, this cooling rate is comparable to the decay heat at times later than about 5 hours. The energy extraction is primarily due to steam condensation (Figure A.1.2.6 shows the condensation rate). The containment pressure (Figure A.1.2.7) is around 0.2 MPa during this time range, with the water vapor partial pressure around 0.05 MPa.

Figure A.1.1.8 shows the water level in the vessel. During the time from 2.5 to 4.9 hours, as a consequence of the period when the SI pump is deadheaded, the core uncovers partially. Figure A.1.1.9 shows the temperature of the coolant at the core exit (maximized over the radius). The highest clad temperature (maximized over the core) occurs during this time (1,123 K at 3.15 hours), as Figure A.1.1.10 shows.

The calculation is ended at 27.8 hours with the core kept covered by injection in the recirculation mode from one SI pump, while the fan cooler removes heat at a rate very close to the decay heat rate. Gap releases and gross core damage in the form of the relocation of any structures are avoided. However, 4.7 kg of hydrogen are generated as a result of oxidation reactions in the core.

A.2.2.3 Results of Case A.1.3 (No Charging Pump Injection)

Case A.1.3 considers the effect of the injection by one charging pump. In the base case, this injection takes place during the time from scram until the time when the VCT depletes. In the base case, the VCT depletes at 10.9 minutes. That this injection need be accounted for starting only at scram assumes that normal-operation charging is canceled by letdown prior to scram. The letdown is not modeled but is assumed to end at the scram. In the sensitivity case, no injection by a charging pump occurs at any time (i.e., unavailability of charging is related to failure of the charging pump or train and not a failure associated with realignment to ECCS injection).

As Figure A.1.3.1 shows, the minimum RCS pressure that occurs soon after scram is sufficiently deepened (relative to the base case) by the lack of charging pump injection that the ECCS actuation pressure is attained at 64 seconds. In the base case, the ECCS is not activated until 20 minutes, by the manual S signal. However, despite being activated, the available ECCS pumps cannot inject until after the manual opening of PORV-1, and the only consequence of the early ECCS actuation is switching the fan cooler to accident mode.

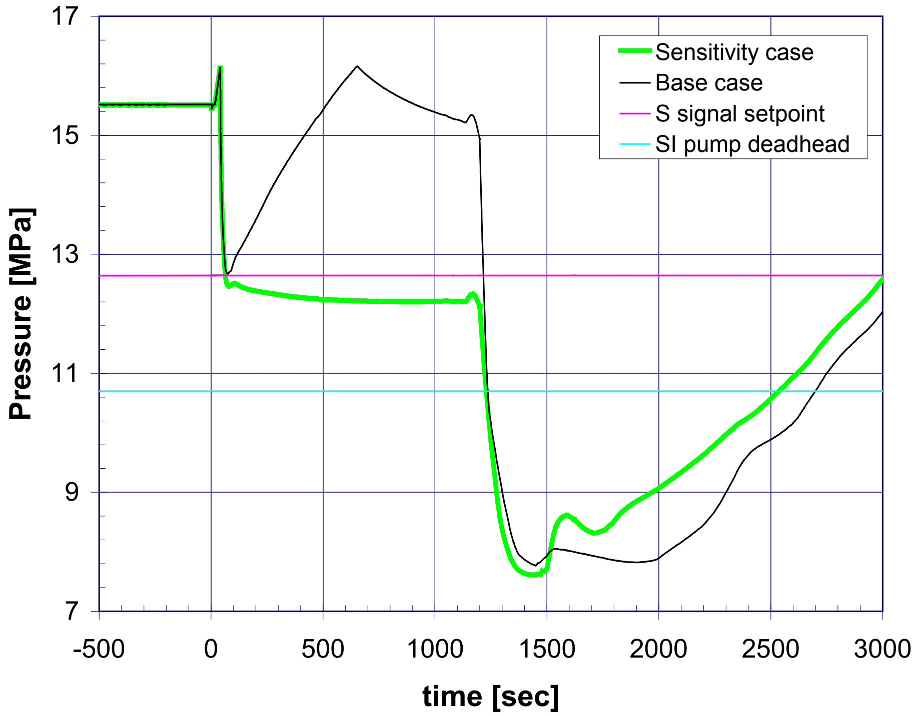

Figure A.1.3.1 Pressure in the Pressurizer (Case A.1.3)

Relative to the base case, at early times the RCS pressure is generally lower in the sensitivity case. Later, however, there is little effect on the pressure. As a result, the history of the SI pump deadhead, core uncovery, and core damage is similar in both the base and the sensitivity cases. The coolant temperature at the core exit first exceeds 922 K at 1.62 hours; the maximum cladding temperature first exceeds 1,478 K at 1.83 hours; and the first failure of the core support plate occurs at 2.23 hours. These timings compare with 1.68, 1.91, and 2.27 hours, respectively, in the base case.

A.2.2.4 Results of Case A.1.4 (Depressurized Steam Generators)

Case A.1.4 assumes that at 10 minutes, operators depressurize all four steam generators (e.g., in anticipation of SG injection via an alternate low pressure source – though such injection is not credited here). This is accomplished by manually closing all MSIVs, then manually fully opening all SG PORVs. These actions are modeled as occurring instantly at 10 minutes.

Figure A.1.4.1 compares the base and sensitivity cases with respect to the pressure in the pressurizer. Initially (i.e., after the operator action takes place at 10 minutes), the depressurization of the secondary side has a notable effect on the primary-side pressure. After about half an hour, however, the base-case versus the sensitivity-case primary pressures are similar, leading to a similar history of core uncovery and core damage. The coolant temperature at the core exit first exceeds 922 K at 1.60 hours. The maximum cladding temperature first exceeds 1,478 K at 1.81 hours. The first failure of the core support plate occurs at 2.27 hours. These timings compare with 1.68, 1.91, and 2.27 hours, respectively, in the base case.

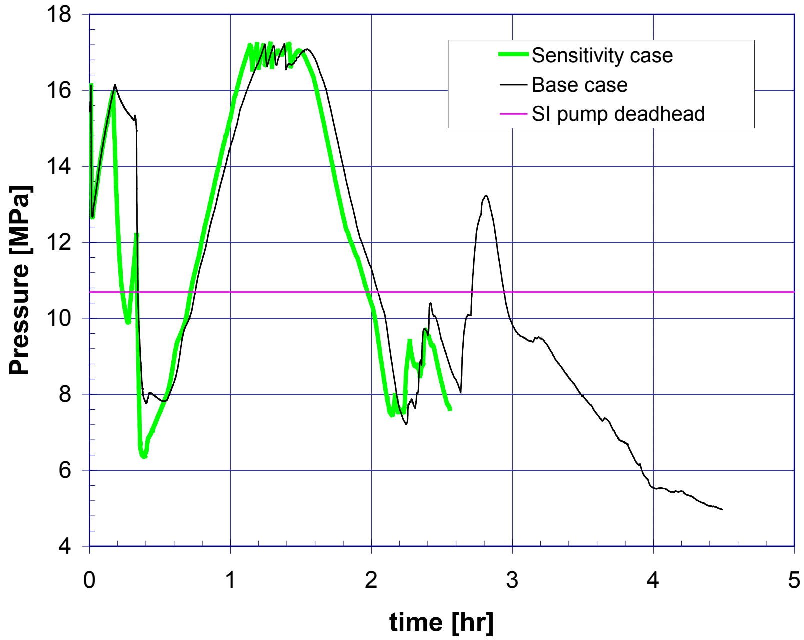

Figure A.1.4.1 Pressure in the Pressurizer (Case A.1.4)

In an undocumented variant, the four steam generators are depressurized at scram. The core damage timings are similar: the coolant temperature at the core exit first exceeds 922 K at 1.76 hours, and the maximum cladding temperature first exceeds 1478 K at 1.98 hours.

A.2.2.5 Results of Case A.1.5 (One Steam Generator Safety Valve Is Stuck Open)

In this case, the assumed unavailability of the condenser steam dump valves and steam generator PORVs causes a cycling of the steam generator safety valves. Additionally, this sensitivity case assumes that one of the safety valves becomes stuck in the open position after 25 open-and-close cycles. The affected valve is FL376, the safety valve of lowest opening pressure in the pressurizer-loop steam generator, which opens at 8.205 MPa. This event occurs at 24.3 minutes.

It may be stated that the postulated sticking event is relatively unlikely. Reference [A.1] quotes NUREG/CR–7037 to present 263 as the number of valve lifts that yields a 50 percent cumulative probability of becoming stuck in the open position, estimated to apply to the steam generator relief valves at the Surry plant. This statistic implies a per-close sticking probability of 0.002632, which in turn implies a cumulative probability of 6.4 percent for becoming stuck in the open position in 25 or less demands to close. Two considerations led to imposing a comparatively early time for this valve to stick open. First, it is desired that the event should have interesting consequences, so it should precede the period of the SI pump deadhead. Second, the steam generator relief valves naturally cycle less frequently as the sequence progresses. For example, as a variant to the present calculation, were FL376 allowed to cycle indefinitely, it would have cycled 38 times at 43 minutes and 40 times at 67 minutes.

The count of pressurizer-loop steam generator safety valve cycles attains 25 cycles at 24.3 minutes, so in this sensitivity case the valve remains open starting at this time. All MSIVs are still open, so the stuck valve acts to depressurize all steam generators. At 30.5 minutes, however, the high containment pressure trip point is reached, and all MSIVs close. The pressurizer-loop SG continues to depressurize until it reaches atmospheric pressure at about 35 minutes. Pressures in the other steam generators recover, but the SRVs do not cycle again before the problem is ended. Figure A.1.5.1 depicts these events and shows the pressures in the four steam generator domes.

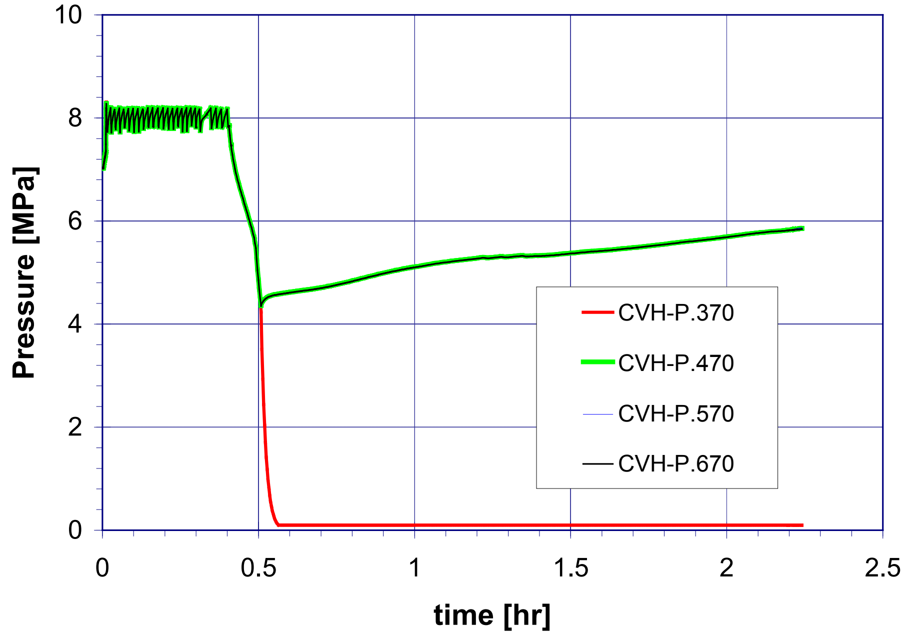

Figure A.1.5.1 Pressure in the Steam Generator Domes (Case A.1.5)

Figure A.1.5.2 compares the RCS pressure of the sensitivity case with that of the base case. These pressures are similar, and they lead to a similar history of core uncovery and core damage. The coolant temperature at the core exit first exceeds 922 K at 1.66 hours. The maximum cladding temperature first exceeds 1,478 K at 1.88 hours. And the first failure of the core support plate occurs at 2.23 hours. These timings compare with 1.68, 1.91, and 2.27 hours, respectively, in the base case.

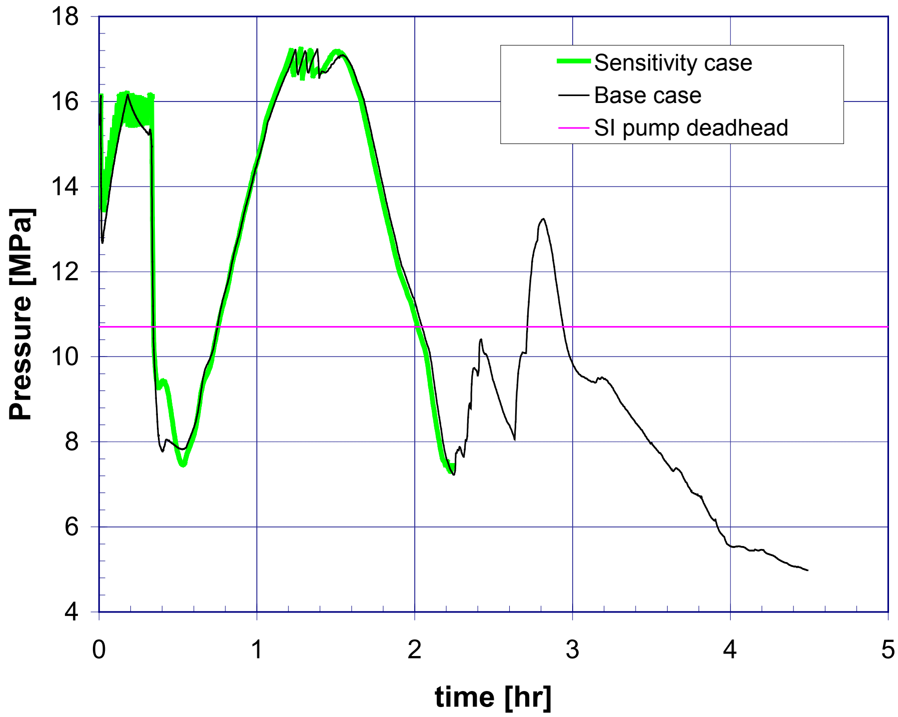

Figure A.1.5.2 Pressure in the Pressurizer (Case A.1.5)

A.2.2.6 Results of Case A.1.6 (Accumulators Credited)

This case credits the four accumulators, which are assumed to be unavailable in the base case.

Before 5.1 hours, the sensitivity case results are identical to those of the base case, because the RCS pressure is above the accumulator injection pressure (4.515 MPa) before this time. Therefore, the cases are identical with respect to core damage, which occurs at around 1.8 hours depending on the surrogate measure.

A.2.2.7 Results of Case A.1.7 (All Fan Coolers Are Credited)

This case credits all four fan coolers, while the base case credits only one.

The greater containment cooling has little effect on in-vessel processes, especially during the ECCS injection mode, when core damage occurs in the base case. As a result, the history of core uncovery and core damage is similar in the two cases. In the sensitivity case, the coolant temperature at the core exit first exceeds 922 K at 1.68 hours; the maximum cladding temperature first exceeds 1,478K at 1.91 hours; and the first failure of the core support plate occurs at 2.31 hours. These timings compare with 1.68, 1.91, and 2.27 hours, respectively, in the base case.

A.2.2.8 Results of Case A.1.8 (MELCOR Version 2.1)

Case A.1.8 is the same as the base case, except that the calculations use MELCOR version 2.1 (Release 3649) instead of version 1.8.6.

Figure A.1.8.1 compares the results with respect to the pressure in the pressurizer. Through the times at which the various core damage surrogate measures indicate, the differences are very small. Later, there are some timing differences in the pressures and in other phenomena. In the sensitivity case, the coolant temperature at the core exit first exceeds 922 K at 1.68 hours; the maximum cladding temperature first exceeds 1,478 K at 1.91 hours; and the first failure of the core support plate occurs at 2.07 hours. These timings compare with 1.68, 1.91, and 2.27 hours, respectively, in the base case.

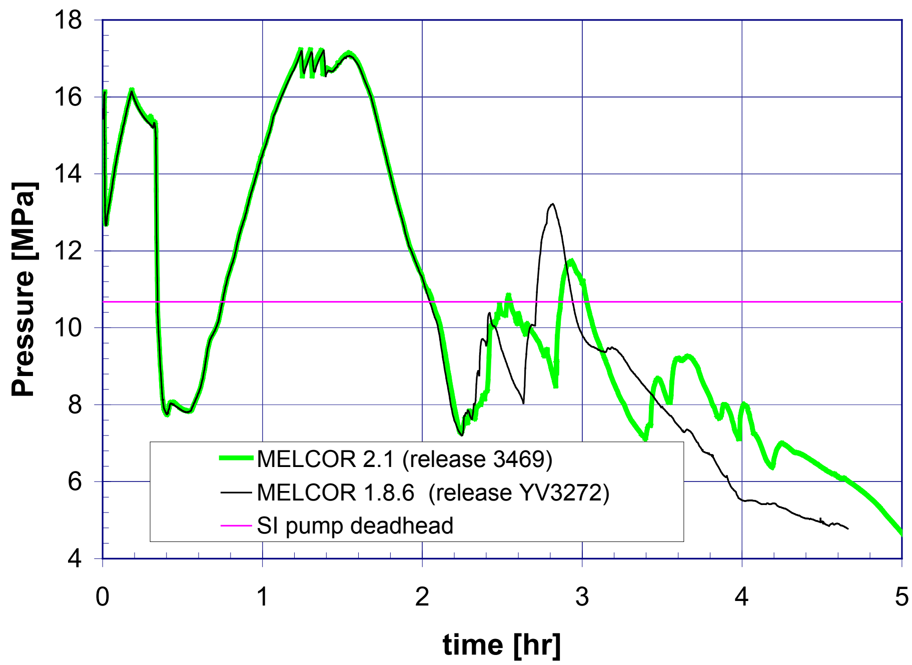

Figure A.1.8.1 Pressure in the Pressurizer (Case A.1.8)

It is noted that the Byron MELCOR model is under continuous development, and has reached Revision 8 as this report is being finalized. (Except for the results reported in this paragraph, all results that apply to the Loss of DC Bus 111 scenario have been obtained using a working Revision 5 of the model[23]). Figure A.1.8.2 compares the pressure in the pressurizer, as predicted by Revision 8 of the model and MELCOR 2.1, with the base-case calculation. (The thin black curves of Figures A.1.8.1 and A.1.8.2 are identical.) Core damage occurs at 2.16 hours in the Revision 8, version 2.1 calculation, according to the surrogate that associates core damage with peak cladding temperature in excess of 1478 K. In the base case (model Revision 5, MELCOR version 1.8.6), the core damage time by the same surrogate is 1.91 hours.

Among the many modeling changes that may contribute to the different results, two changes have been identified with their specific effects. In Revision 8, the condenser steam dump valves attempt to control the average RCS legs temperature, T_{avg}, to a target value of 570 K. The Revision 5 value for the target is 564.8 K. As the base-case write-up noted in connection with Figure A.1.1.3a, the low RCS pressure setpoint for an early automatic initiation of the S signal was narrowly missed in that calculation at about one minute after initiation of the transient. In Revision 8, the pressure setpoint would actually be attained were the target temperature taken as 564.8 K. To avoid the S signal, the target temperature is set higher in Revision 8. The desired effect in the first minute (i.e., suppression of the signal) is thereby achieved, but an ongoing consequence is that, according to Revision 8, the steam generators provide less cooling at all times. The reduced cooling accounts for the higher primary pressure that Figure A.1.8.2 shows in the time range from 0.1 to 0.3 hours.

A second modeling change for Revision 8 is the use of a larger value for the opening height of the flow path that connects the boiler with the separator in each steam generator. The opening height defines the range of elevations over which gas versus liquid can enter the flow path. In the present case, it may be considered an adjustable parameter since its value cannot be derived from the steam generator design.

[23] See footnote 1 in Section 2 of the main report.

The larger value results in better numerical stability, especially during the pre-transient part of the calculation. The larger value also affects the steam generator level drop during the interval after feedwater loss and before scram, such that the scram (due to low water level) occurs earlier. The time of scram is 33 seconds in the case of the Revision 8 calculation, versus 40 seconds in the base case. As case A.1.2 showed, an earlier scram time can have long-term mitigating effects. Here, the time of core damage has been slightly postponed.

Generally, it is considered that the differences between the predictions of Revisions 5 and 8 of the model are not drastic, and not surprising given the many intervening input model changes.

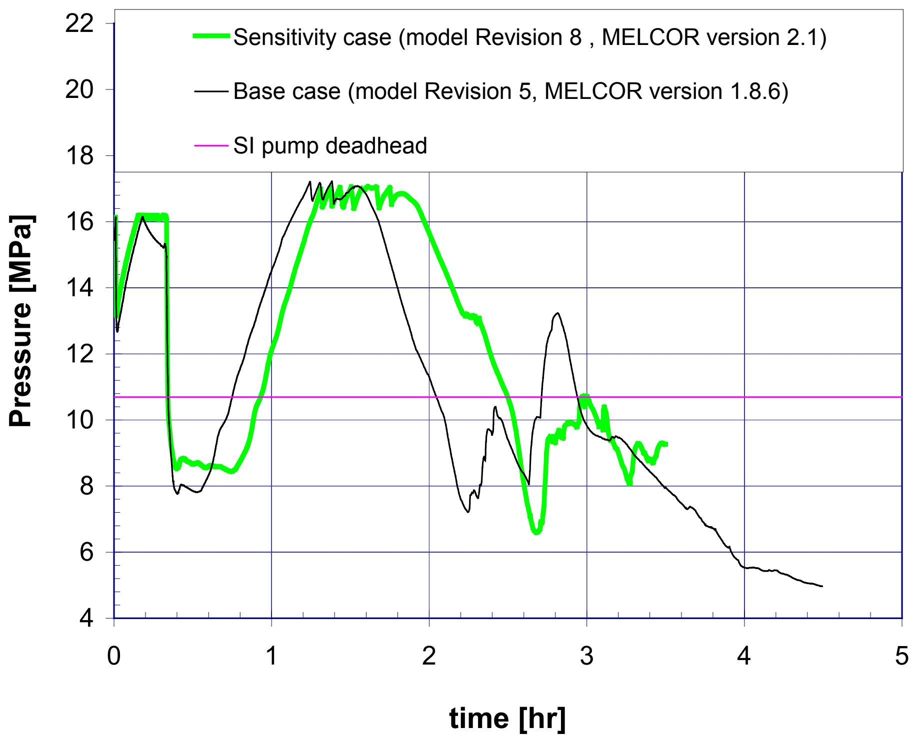

Figure A.1.8.2 Pressure in the Pressurizer (Case A.1.8 Variant)

A.2.2.9 Results of Case A.1.9 (Reduced Decay Heat)

This case is the same as the base case until the reactor trip. After the reactor trip, the decay heat is 80 percent of the base-case decay heat. Full power is assumed before the reactor trip. This assumption is overly conservative, but convenient since reducing the fission power would require re-working the steady-state model. In this case, this assumption proves to be acceptable since the prediction is that core damage is avoided.

Figure A.1.9.1 compares the primary pressures for both the base and sensitivity cases. In the sensitivity case, the period of SI pump deadheading is shorter and occurs later. The core is partially uncovered from 2.12 to 3.45 hours. The hottest cladding temperature peaks at 2.71 hours at 964 K. Thus, MELCOR predicts that gap releases and gross core damage are avoided.

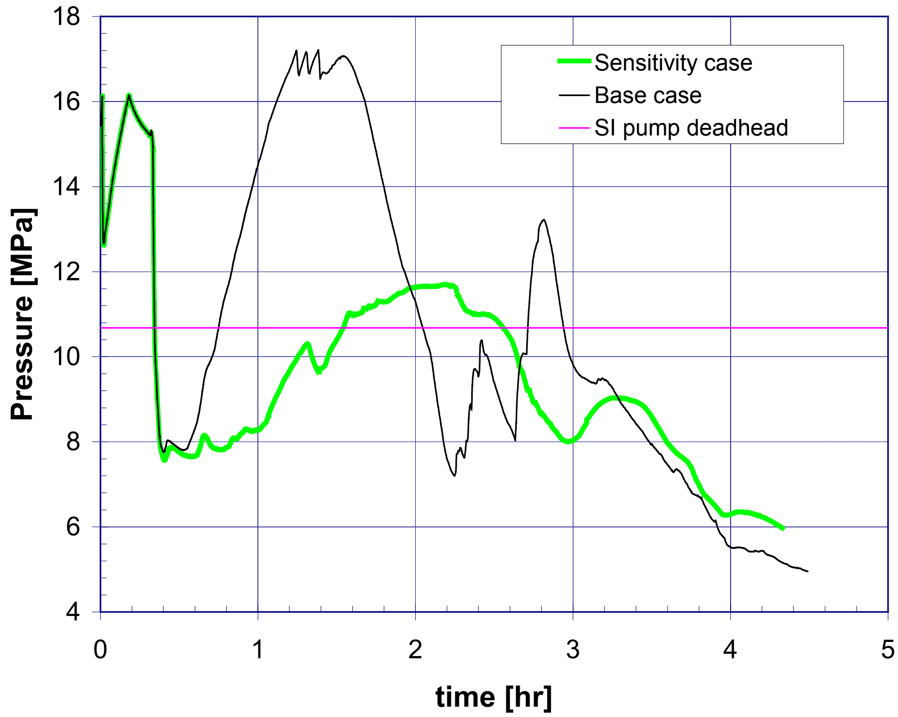

Figure A.1.9.1 Pressure in the Pressurizer (Case A.1.9)

A.2.3 Summary: Loss of DC Bus 111 with Bleed and Feed Scenario

The base-case and sensitivity-case calculations, in which the reactor trip occurs at 40 seconds or earlier, show that the RCS pressure drop provided by opening one pressurizer PORV at 20 minutes is sufficient to allow one SI pump to begin to inject soon thereafter (at about 21 minutes). Later, however, in the base case (Case A.1.1) and in all of the sensitivity cases, pressures high enough to deadhead the pump eventually return to the RCS. The return of higher pressures is due to the loss of steam generator cooling after the steam generators dry out. In the base case, the period of the SI pump deadhead is 0.75 to 2.04 hours. Due to absence of injection, the core partly uncovers and gross core damage results (e.g., relocations of fuel and support structures occur). In the base case, the temperature of the hottest cladding first exceeds 1,478 K (2,200 °F) at 1.91 hours.

Gross core damage (e.g., material relocations) is avoided in two sensitivity cases. In the first (Case A.1.2), an earlier reactor trip at 11 seconds is prescribed. The SI pump deadhead and partial core uncovery still occur and core heatup still takes place; however, the maximum cladding temperature remains below 1,123 K. MELCOR therefore predicts that no gap releases will occur. In the second case (A.1.9), the decay heat is uniformly reduced to 80 percent of its base-case value. Again, the SI pump deadhead, partial core uncovery, and core heatup still occur, but since the maximum cladding temperature reaches only 964 K, no gap releases or gross core damage occurs.

The other sensitivity cases investigate the following factors:

- the role of injection by a charging pump before the ECCS signal (Case A.1.3 discounts such injections)
- the influence of manual steam generator depressurization (Case A.1.4 imposes depressurization on all four steam generators at 10 minutes)

- the effect of a stuck-open steam generator relief valve (Case A.1.5)
- the effect of accumulator injections (Case A.1.6 credits all accumulators, which are discounted in the base case)
- the effect of additional fan coolers (Case A.1.7 credits all four fan coolers, whereas the base case credits only one)
- uncertainties due to different MELCOR code versions (Case A.1.8 repeats the base case by using MELCOR version 2.1 instead of version 1.8.6)

These factors introduce little variation in the times to core damage. See Section 4.1.1 of the main report for additional details.

A.3 Byron Scenarios and Results: Loss of Coolant During Shutdown

A.3.1 Break Outside the Containment during Shutdown

The Byron "Shutdown" MELCOR model is described briefly in Section 2.1 of the main report. The model applies to a phase of the shutdown operations occurring before refueling but while the RCS is still closed. The loop stop valves (which serve the same function as nozzle dams at other plants) are open, the RCS is full of water, and 24 hours have elapsed since reactor subcriticality. The main differences between the MELCOR shutdown model and the MELCOR full-power model are summarized in the following bullets:

- The decay heat is calculated at a shifted time t' = t + 24 hours, where t is the simulation time. During the pre-transient part of the calculation (i.e., during -5000 s < t < 0 s), the decay heat is constant at 21.29 MW, which is the decay heat at 24 hours.
- Initial water and steam temperatures throughout the RCS (primary and secondary sides) are 466.5 K (380 °F). Initial pressures are the corresponding saturation pressure (1.35 MPa).
- Decay heat removal is provided by the closed-loop operation of one RHR pump. The pump takes suction from the pressurizer-loop (loop D) hot leg. The outflow passes through an RHR heat exchanger and is then injected into the cold legs of loops A and D. The heat exchanger is modeled as described in the ERI-maintained calculation notes. In particular, the shell-side water temperature is 314 K (105 °F), and the heat exchanger has an efficiency of 0.609. Water at 466.5 K is cooled to 373.4 K by this heat exchanger. The pumping rate is controlled by feedback logic to keep the coolant temperature at the core exit constant.
- There is a provision for a break outside the containment in the RHR lines that feed the cold legs of loops B and C. The assigned break flow area is such that the initial discharge rate is about 1,000 gpm (0.063 m^3/s). These lines are not in use for injection before the start of the accident. The break of the common line is modeled with two flow paths, one in each cold leg.

Note that injections are to the loops D and A cold legs (modeled by control volumes CV348 and CV448), while the breaks are from the loops B and C cold legs (modeled by control volumes CV548 and CV648). In general, the model's control volumes that are designated CV*ixy* with *i* = 3, 4, 5, and 6 correspond respectively to loops D, A, B, and C in plant nomenclature (see footnote in Section 2.1 for more information on the loop numbering). For this reason, the breaks do not result in an immediate impairment of the discharge of the RHR-cooled water into the RCS.

The base-case break, which connects two cold legs to the environment via a broken RHR injection line, may not be realistic because of certain check valves inside the containment that are neglected by the model. Sensitivity Case A.2.10 addresses this concern by considering a break from a hot leg.

In this report, shutdown modeling is considered a first foray into this subject and consequently, there is a preference for the shutdown model not to depart too much from the main full-power model. For this reason, a phase of the shutdown operations is considered in which, for example, the head is still on the vessel. However, MELCOR is fully capable of modeling all other phases of shutdown operations that possibly have greater risk significance. Such phases may include those where the head is removed and the core region is connected to the spent fuel pool by a continuous path through water.

Actual shutdown operations would likely have water-solid conditions in the RCS at this time, with the pressure well above the saturation pressure. The readily available modeling/operational data applicable to the water-solid state, however, were not complete. Moreover, numerical difficulties related to water incompressibility were also anticipated. As an expedient to meet these difficulties, it was decided to model the plant in the state that it has attained just prior to when the operators place the PORVs in cold overpressure protection. Saturated conditions hold at this time, but it occurs shortly before the time when the RCS becomes water-solid. The adopted model is set up with saturated conditions and a steam bubble volume of 1 m^3 in the pressurizer. (However, see sensitivity Case A.2.11.) The RCS temperature is assumed to be at 466.5 K (380 °F), rather than 449.8 K (350 °F), which actually holds at the start of the period of water-solid conditions. The selected temperature is consistent with Step 21 of a Byron shutdown-related document referred to as 1BOA-SD-2. As noted, the assumption about the initial pressure (i.e., saturated instead of super-pressurized) is made partly for computational convenience, but it is not considered consequentially significant because the transient includes a break that may be assumed to quickly lower the pressure to near saturation (i.e., had the RCS been initially prescribed as super-pressurized). In fact, sensitivity case A.2.11 demonstrates that this assumption is correct.

Because the actual disposition of the steam generators at this point in the shutdown sequence is not known, the assumption that the temperature on the primary and secondary sides of the RCS is the same is made to minimize the thermal interaction between the two sides.

The transient initiator is the opening of a break outside the containment at time t = 0. Main FW and all forms of AFW are assumed to be unavailable. One out of four containment fan coolers is available. Table A.5 summarizes the availability of various systems.

To investigate the variability of the MELCOR predictions, thirteen sensitivity cases are defined in addition to the base case. The highlighted entries in Table A.6 show how each sensitivity case differs from the base case.

Case A.2.1 is the base case. No operator actions are credited, thus insuring that core damage will eventually occur as a result of the lack of decay heat removal following a loss of suction by the RHR pump at the pressurizer-loop hot leg.

Case A.2.2 assumes that operators act at 0.5 hours to initiate ECCS injections (with the pumps operating according to their normal-operation configurations) with one charging and one SI pump. The pumps take suction from the RWST, which is credited to be refilled as required.

Case A.2.3 is similar to Case A.2.2, but with the actions occurring at 1.0 hours.

Case A.2.4 is similar to the base case, except that it assumes a more stringent treatment of the RHR pump suction loss.

Case A.2.5 combines Cases A.2.2 and A.2.4.

Case A.2.6 considers an attempt to enhance steam generator cooling, relative to what arises naturally in the base case. Starting at 0.5 hours, the condenser steam dump valve is held partially open to implement a gradual secondary-side depressurization. AFW is available.

Cases A.2.7a through A.2.7d are similar to the base case, except that the break flow areas differ from that of the base case by various constant factors.

Case A.2.8 is similar to the base case except the decay heat differs from that of the base case by a constant factor of 0.75.

In Case A.2.9, the break is inside the containment and allows the effluent to condense and return to the containment sump. Otherwise, it is similar to Case A.2.2, except that a successful switchover to the recirculation mode (ECCS pumps take suction from the sump) is credited when the RWST level reaches Lo-2.

Case A.2.10 considers a break from one hot leg of the same flow area as the net area of the base case's two cold legs breaks.

Case A.2.11 redefines the pre-transient initial conditions so that a water-solid, sub-cooled primary RCS is prescribed.

The base-case calculation for the Byron shutdown scenario is described in detail in Section A.3.2.1. Sections A.3.2.2 through A.3.2.11 describe the sensitivity calculations, typically in less detail. A summary discussion pertaining to this scenario appears in Section A.3.3 (which also appears in the main report).

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Table A.5 Shutdown: System, Phenomenological, and Operator Action Assumptions

Primary Side	Considered to represent the situation just before operators take the RCS water-solid and place the PORVs in cold over-pressure protection mode: • Initial temperature 466.5 K; saturation pressure 1.35 MPa • Initial decay heat 21.29 MW • RCPs off • Full inventory ○ Steam bubble at the top of the pressurizer volume ~1 m^3
Secondary Side	• Initial temperature 466.5 K; saturation pressure 1.35 MPa[1] • FW not available[1] • PORVs available • Condenser dump valve closed; turbine throttle valve closed • MSIVs open
ECCS	Not available in the base case. (Sensitivity cases consider the recovery of one charging pump and one SI pump operating according to their normal-operation ECCS configurations.)
Operator actions	None for the base case. (Sensitivity cases consider alignment of one charging pump and one SI pump according to their normal-operation ECCS configurations, and the opening of the condenser dump valve.)
Other	One fan cooler operates. Sprays are not available.

[1] Lack of information on the initial temperature of the RCS secondary sides during the considered time frame leads to the assumption that their initial temperature and pressure are the same as those of the primary side. In this case, if the secondary sides remain closed, water levels stay high and there is no demand for feedwater. However, sensitivity case A.2.6 considers available auxiliary feedwater in the case that the boil-off of secondary-side water inventory is promoted by a partial opening of the condenser steam dump valve.

Table A.6 Sensitivity Cases for the Byron Shutdown Scenario

Case	Time of Operator Actions (hours)	Treatment of Loss of RHR Pump Suction on Low HL-A Level	Auxiliary FW and Condenser Dump Valve	Break	Decay Heat	Pre-Transient Thermodynamic Conditions in the Primary RCS
A.2.1 (Base)	Never	As in base case(1)	Not available	Outside containment 1,000 gpm (0.063 m^3/s)	Initially, 21.29 MW corresponding to 24 hours after scram	466.5 K; 1.35 MPa (saturation); ~1 m^3 steam in the pressurizer
A.2.2	0.5(2)	As in base case	Not available	As in base case	As in base case	As in base case
A.2.3	1.0(2)	As in base case	Not available	As in base case	As in base case	As in base case
A.2.4	Never	Pump trips: rate falls to zero irreversibly	Not available	As in base case	As in base case	As in base case
A.2.5	0.5(2)	Pump trips: rate falls to zero irreversibly	Not available	As in base case	As in base case	As in base case
A.2.6	0.5(3)	As in base case	Available(3)	As in base case	As in base case	As in base case
A.2.7a	Never	As in base case	Not available	Break flow area 75% of base case	As in base case	As in base case
A.2.7b	Never	As in base case	Not available	Break flow area 10% of base case	As in base case	As in base case
A.2.7c	Never	As in base case	Not available	Break flow area 5% of base case	As in base case	As in base case
A.2.7d	Never	As in base case	Not available	Break flow area 500% of base case	As in base case	As in base case
A.2.8	Never	As in base case	Not available	As in base case	Decay heat uniformly reduced to 75% of base case	As in base case
A.2.9	0.92(4)	Pump trips: rate falls to zero irreversibly	Not available	Inside containment 1,000 gpm (0.063 m^3/s)	As in base case	As in base case
A.2.10	Never	As in base case	Not available	From a hot leg: same flow area	As in base case	As in base case
A.2.11	Never	As in base case	Not available	As in base case	As in base case	466.5 K; 2.54 MPa; water-solid

1 The pump rate becomes zero only while there is less than 1 L of water in the lower part of the pressurizer-loop hot leg.
2 Operators initiate normal-operation ECCS modes with one charging pump and one SI pump. Suction is from the RWST, which is refilled as necessary.
3 The condenser steam dump valve is held partially open starting from 0.5 hours.
4 Operators initiate normal-operations ECCS modes with one charging pump and one SI pump. Suction is from the RWST until Lo-2, then from one RHR pump (which itself draws from the containment sump); one RHR heat exchanger is credited.

A.3.2 Results

The following remarks about the pre-transient part of the calculations apply to most of the base and sensitivity cases (i.e., cases A.2.1 through A.2.7, A.2.9, and A.2.10). These cases have in common the pre-transient calculation described in this paragraph.) As noted in the definition of the scenario, isothermal initial conditions (466.5 K) are applied to both the primary and secondary sides of the RCS. The purpose of assuming equal initial temperatures on both sides is to minimize the role of the steam generators, since their disposition at this point in the shutdown sequence is not known. Because the RCPs are assumed to be turned off, temperature gradients develop comparatively slowly during the pre-transient steady state. During this time, the interactions of the primary and secondary sides are sometimes sizable. For example, the net rate of heat transfer to the steam generators attains a peak of about 10 MW about 300 seconds into the calculation. Therefore, a comparatively long steady-state calculation is made extending from about 5,000 seconds to 0 seconds. During this time, the decay heat and the RHR pumping rate are clamped to 21.29 MW and 51.26 kg/s. The pumping rate provides 21.29 MW of heat removal for water cooled from 466.5 K to 373.4 K. Steam generator water levels stay high during the steady state, so there is no demand for FW. The condenser steam dump valve is allowed to operate during the steady state in order to reduce small secondary-side pressure rises. The turbine throttle valve is always closed during all shutdown calculations. Small flows through the dump valve (1 kg/s) occur during the steady state. These flows are zero at all times later than -400 seconds (except in sensitivity case A.2.6, where the dump valve is used during the transient). At t = 0 seconds, the pressures are 1.30 MPa in the pressurizer; 1.41 MPa in the vessel dome; and 1.32 MPa in the steam lines common header. A coolant flow of about 100 kg/s occurs in each loop from natural circulation, and heat is being transferred to the steam generators at a net rate of about 0.49 MW. The radially-averaged liquid temperature is 466.1 K at the core exit; 464.3 K in the pressurizer; and the steam in the common header is 466.1 K. There is a steam bubble volume of 1.2 m^3 at the top of the pressurizer. Figures A.2a and A.2b show some results from the steady-state calculation.

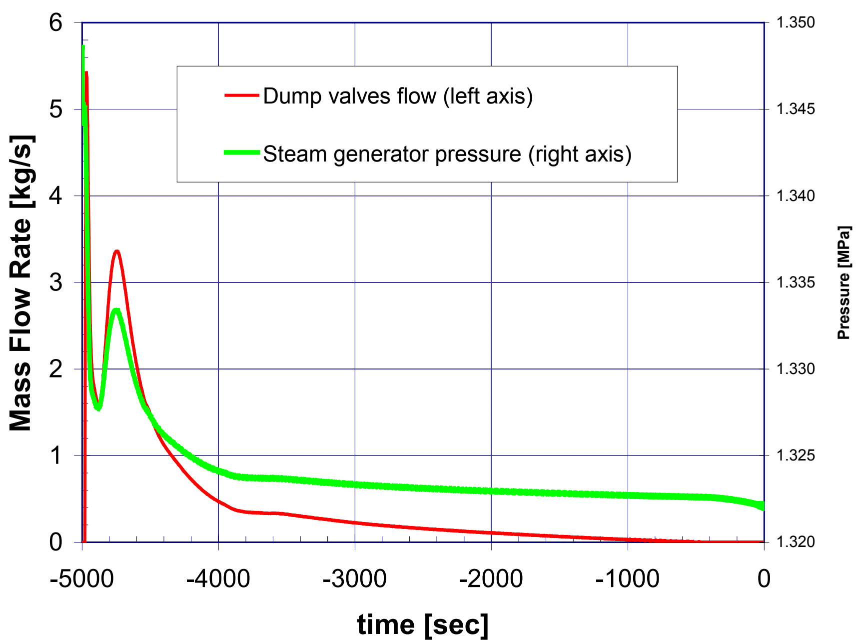

Figure A.2a Dump Valve Flow and Steam Generator Pressure during A.2.x Steady State

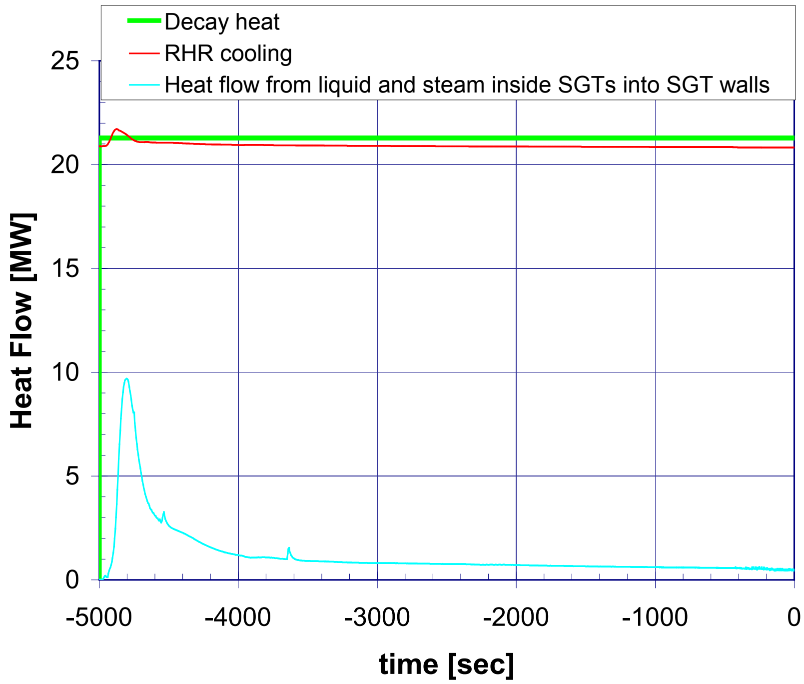

Figure A.2b RHR and Steam Generator Cooling during A.2.x Steady State

A.3.2.1 Results of Case A.2.1 (Base Case)

The scenario progression is outlined in Table A.7 and is illustrated in Figures A.2.1.1 to A.2.1.12. The break at *t* = 0 seconds (after a pre-transient steady-state period lasting 5,000 seconds) initiates the accident. The break occurs outside the containment, in the line that runs from RHR pump B to cold legs B and C. Note that this pump is not in use for cooling. The RHR pump that is in use is RHR pump A, which injects to the cold legs of Loops D and A. These locations are modeled by control volumes CV348 and CV448. There are therefore two MELCOR break flow paths, one to directly connect each B and C cold leg to the environment. These locations are modeled by control volumes CV548 and CV648. The break flow areas (9.4 cm^2 for each flow path) are assigned such that the early combined liquid flow is about 1,000 gpm (0.063 m^3/s).

The base case assumes that no operator actions occur, thus making eventual core damage inevitable. MFW and AFW are assumed to be unavailable; the condenser steam dump valves are assumed to remain shut. The steam generator PORVs are available, but the pressure remains too low for them to open during the calculation. The secondary sides of the steam generators, therefore, are effectively closed during the scenario.

Figures A.2.1.1a and A.2.1.1b show the break flows. Initially, the combined break flow from the cold legs of Loops B and C is about 39 kg/second, or 710 gpm (0.045 m^3/s). However, the RCS pressure (see Figure A.2.1.2) increases and produces a generally upward trend in the break flow rate during the first hour. The reason is that the breaks reduce the flow through the core of the injected water during this time. The resulting core heatup increases the pressure. The break flow exceeds 1,000 gpm (0.063 m^3/s) at about 11 minutes. The rising RCS pressure ends only at about 2.8 hours, as a result of decreasing steam production as more and more of the core uncovers.

Table A.7 Timing of Key Events Predicted for the MELCOR Model Shutdown Base-Case Calculation

Event Description	**Time (hours)[1]**
Break in RHR-B lines outside the containment	0
Reduced RHR-A pump rate because of low water level in hot leg D	~0.7
Core partially uncovered	0.75
Pressurizer empties (i.e., contains less than 10 kg of liquid)	2.64
Zero RHR pump rate because of dry hot leg D	~2.7
Coolant temperature at core exit reaches 922 K (1,200 °F)	2.93
First gap release[2]	3.16
Peak cladding temperature reaches 1,478K (2,200 °F)	3.34
Calculation end	3.90

1 The tabulated time is the problem time. Decay heat is calculated at the problem time plus 24 hours and is initially 21.29 MW.

2 In MELCOR, gap releases occur by rings when the axially-maximized clad temperature reaches 1,173 K in the given ring.

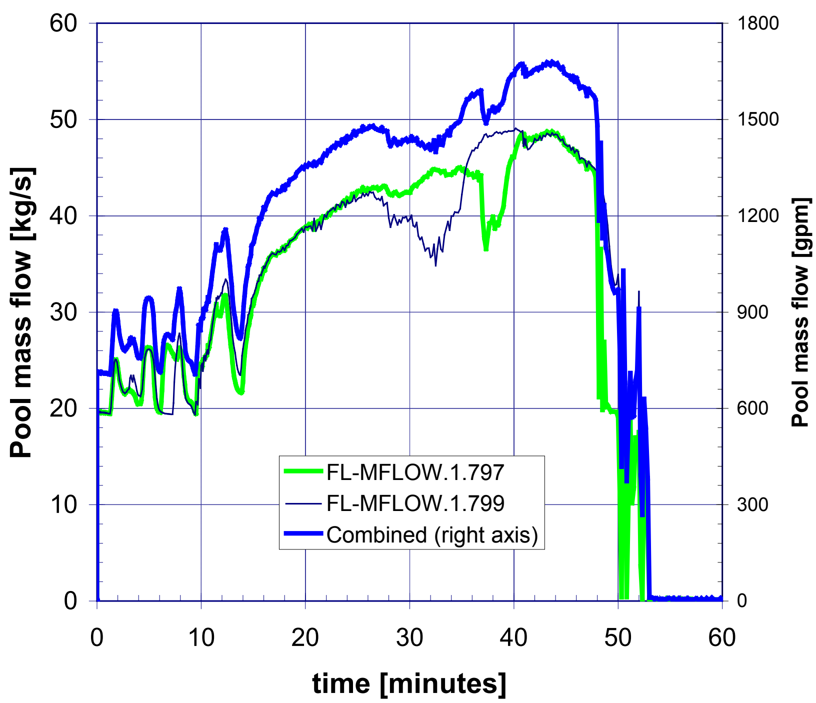

Figure A.2.1.1a Break Flows (Case A.2.1 short-term)

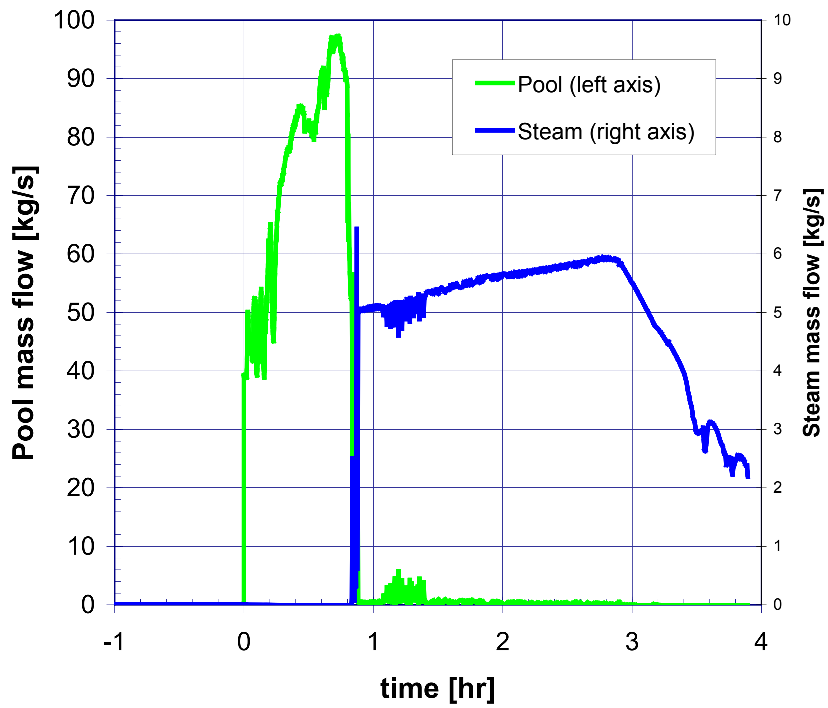

Figure A.2.1.1b Break Flows (Case A.2.1 longer-term)

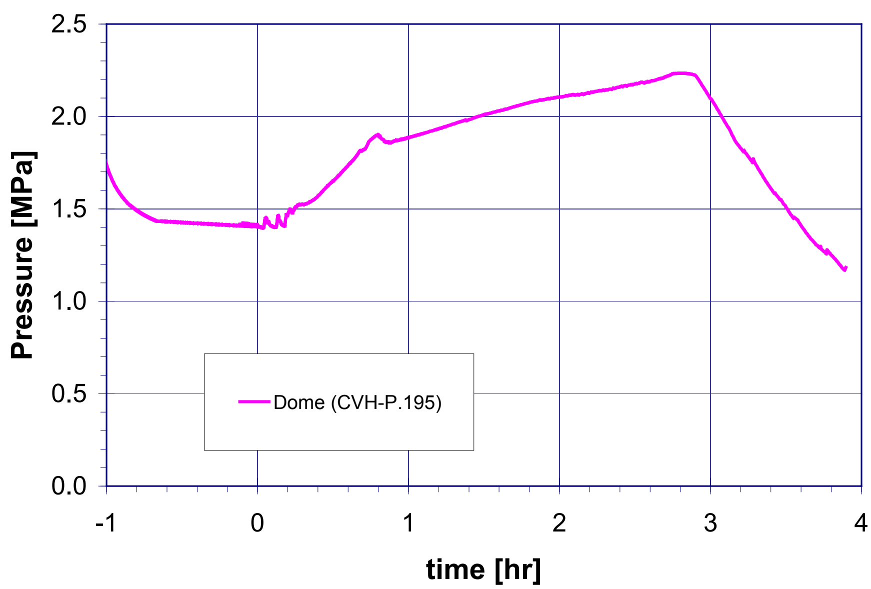

Figure A.2.1.2 RCS Pressure (Case A.2.1)

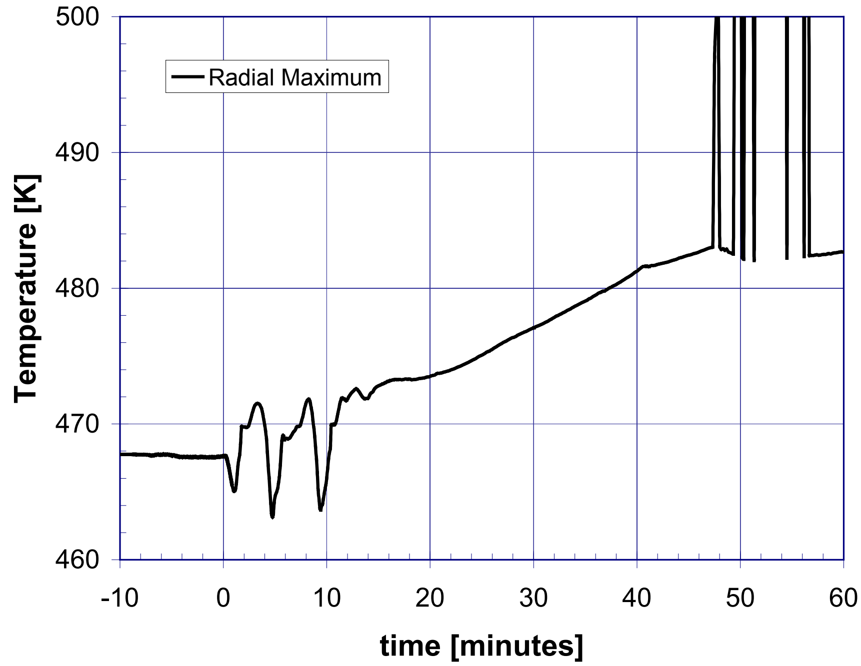

Figure A.2.1.3a Core Exit Temperature (Case A.2.1 short-term)

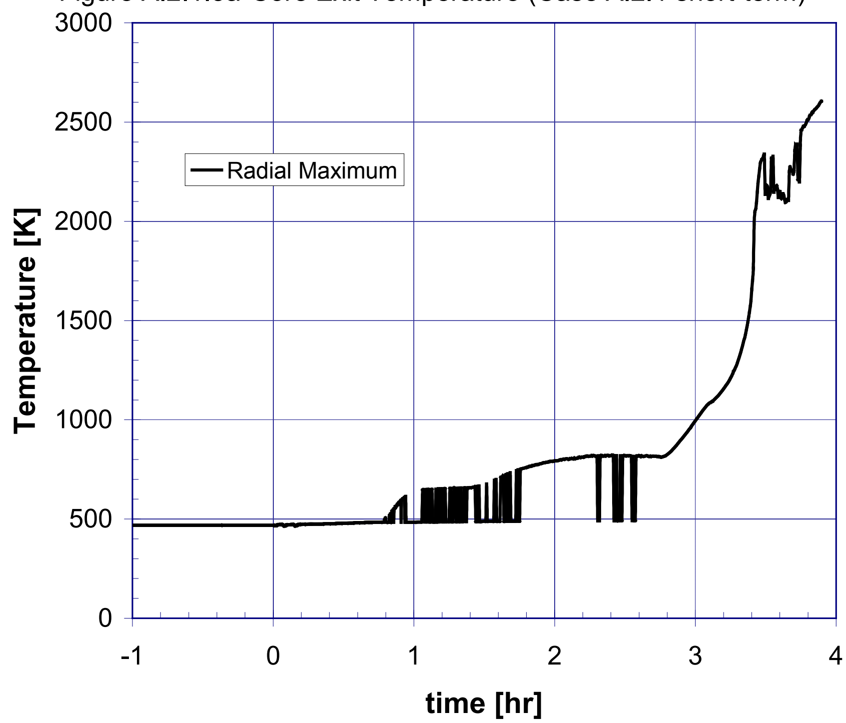

Figure A.2.1.3b Core Exit Temperature (Case A.2.1 longer-term)

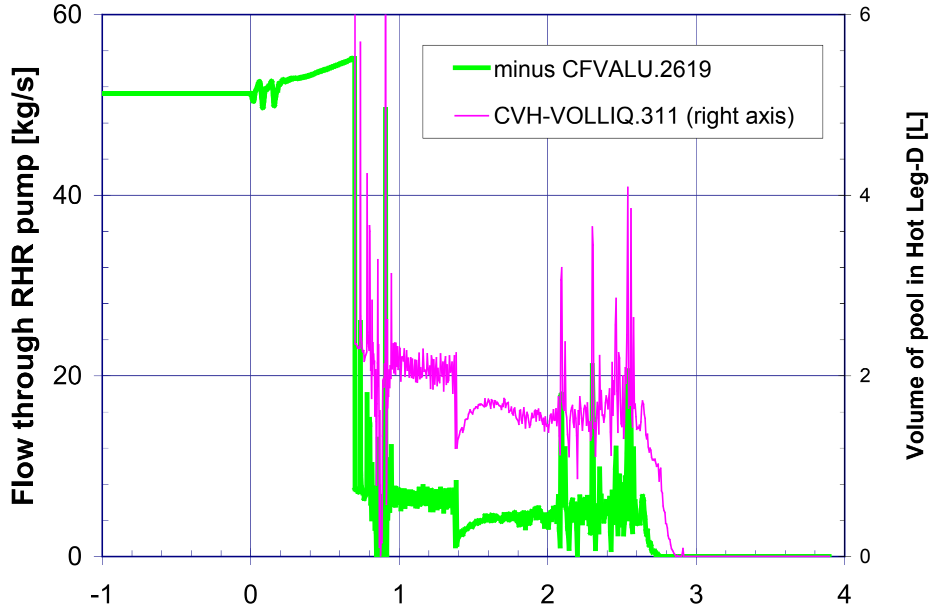

Figure A.2.1.4 RHR Pump Rate and Liquid Volume in Hot Leg D (Case A.2.1)

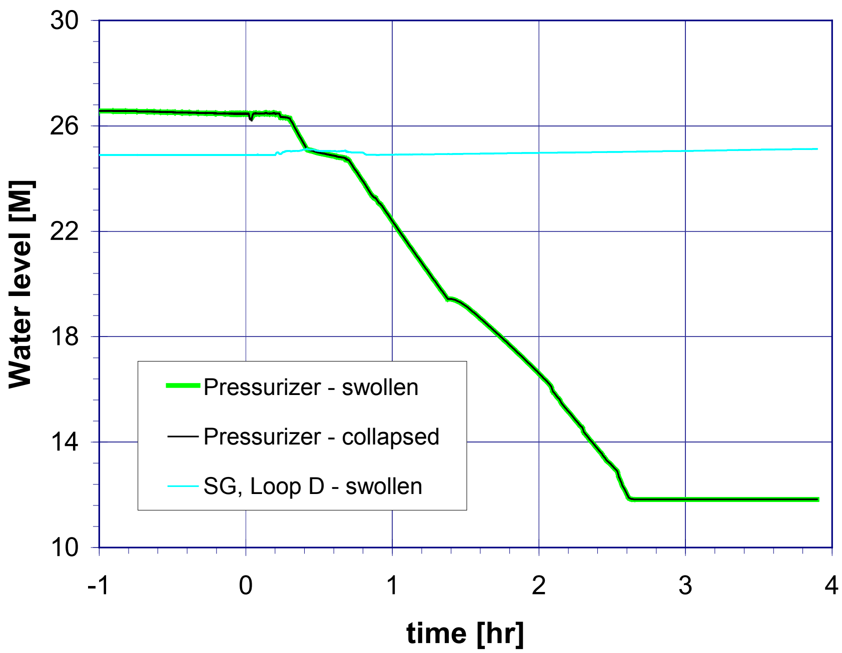

Figure A.2.1.5 Water Levels in the Pressurizer and Steam Generator D (Case A.2.1)

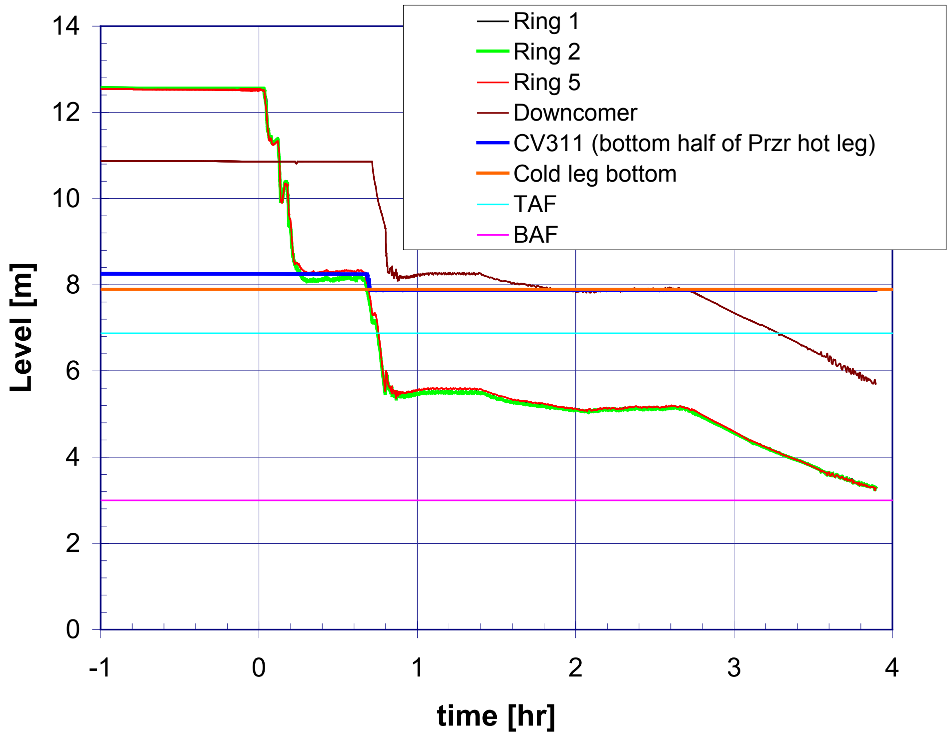

Figure A.2.1.6a Water Levels in the Vessel (Case A.2.1)

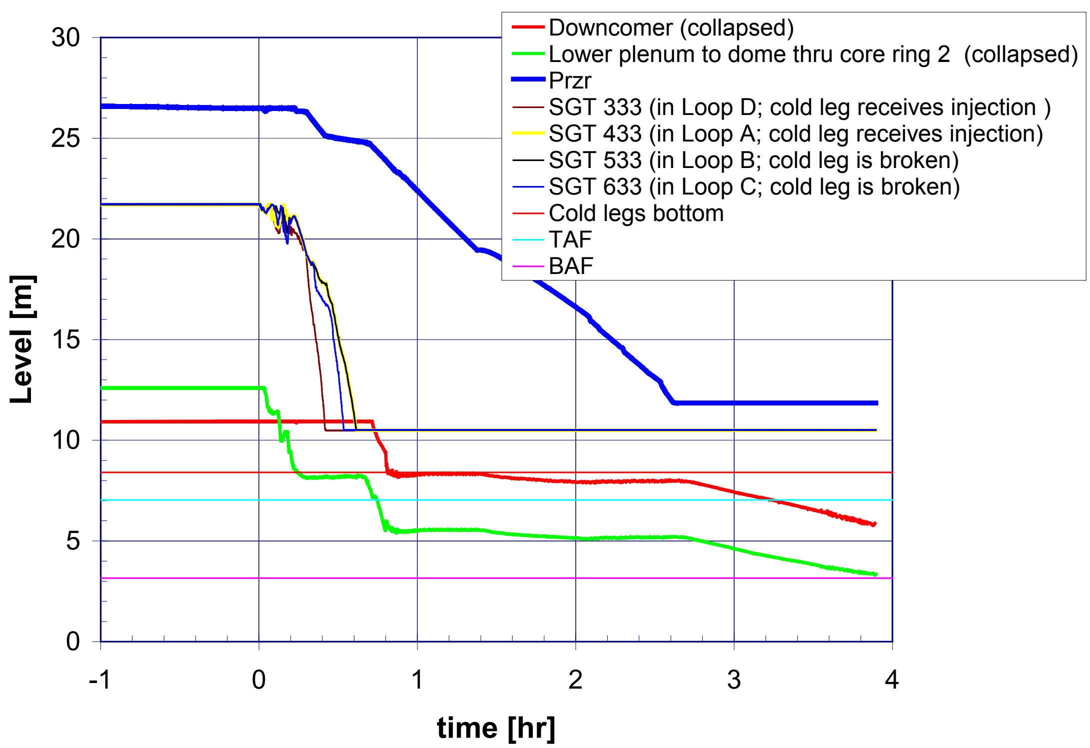

Figure A.2.1.6b Water Levels in the RCS (Case A.2.1)

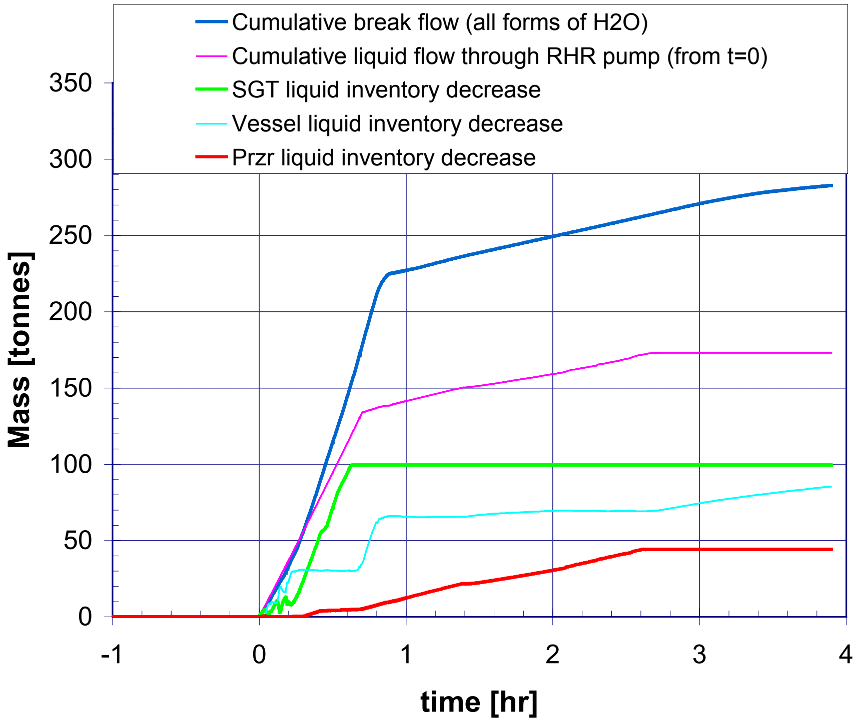

Figure A.2.1.7 Cumulative Flows and Inventory Decreases (Case A.2.1)

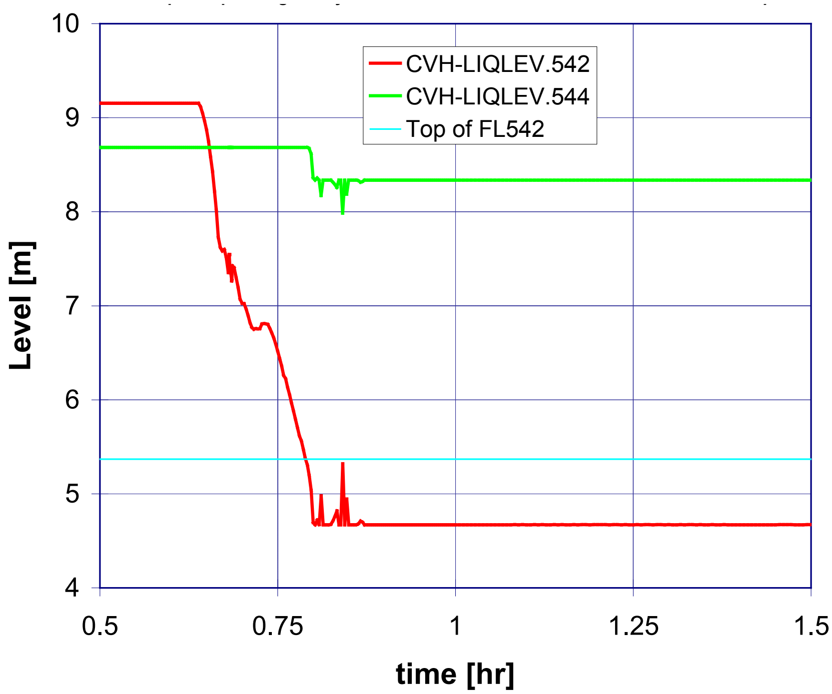

Figure A.2.1.8 Clearing of the Loop Seal in Loop B (Case A.2.1)

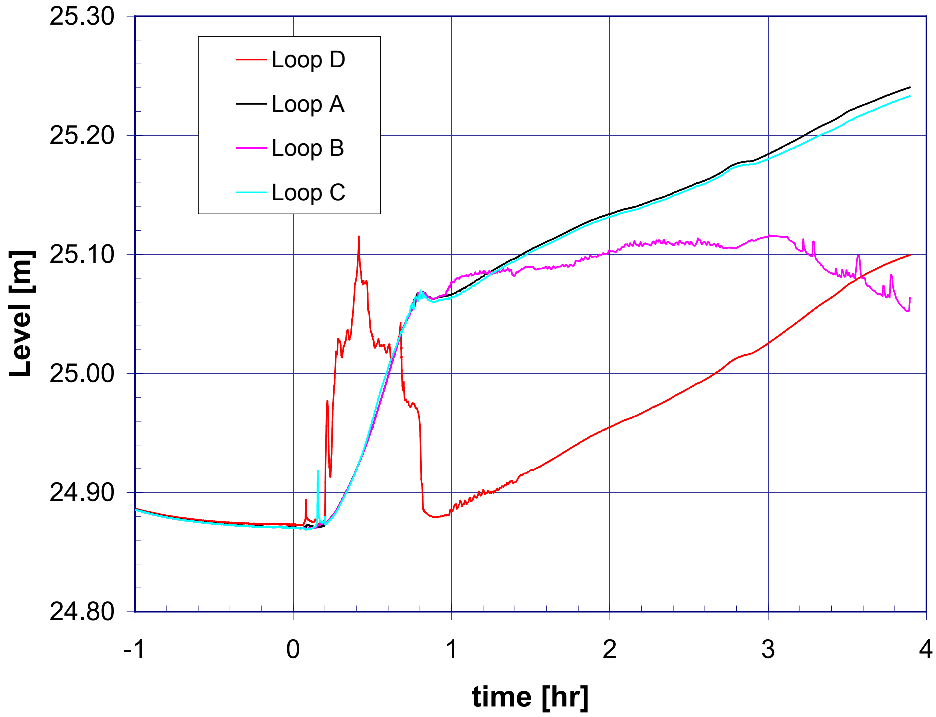

Figure A.2.1.9 Water Levels in the Steam Generators (Case A.2.1)

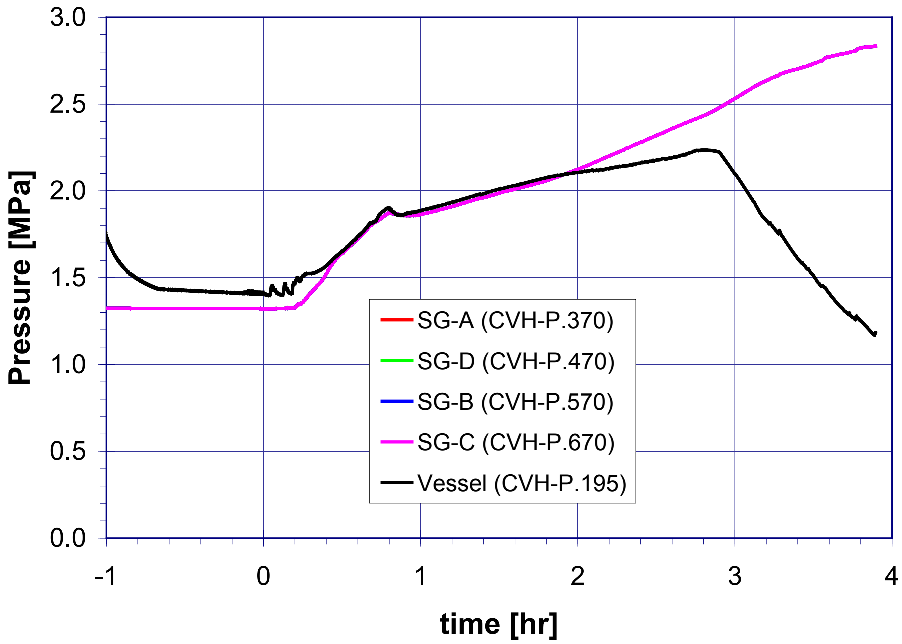

Figure A.2.1.10 Pressures in the Steam Generators and Vessel Dome (Case A.2.1)

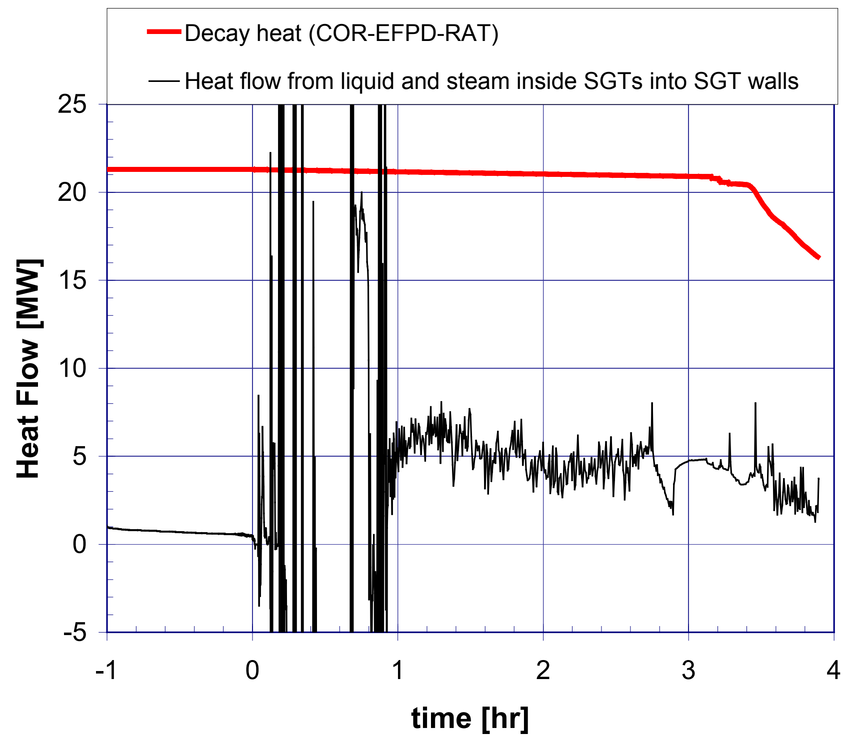

Figure A.2.1.11a Decay Heat and Heat Transferred to Steam Generators (Case A.2.1)

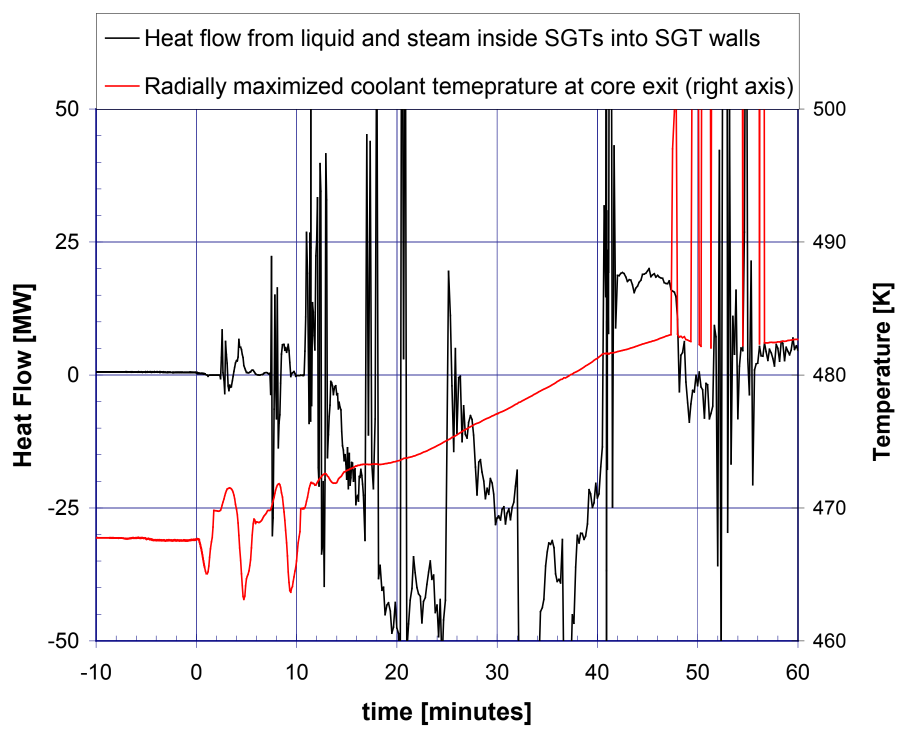

Figure A.2.1.11b Core Exit Temperature and Heat Transferred to Steam Generators (Case A.2.1)

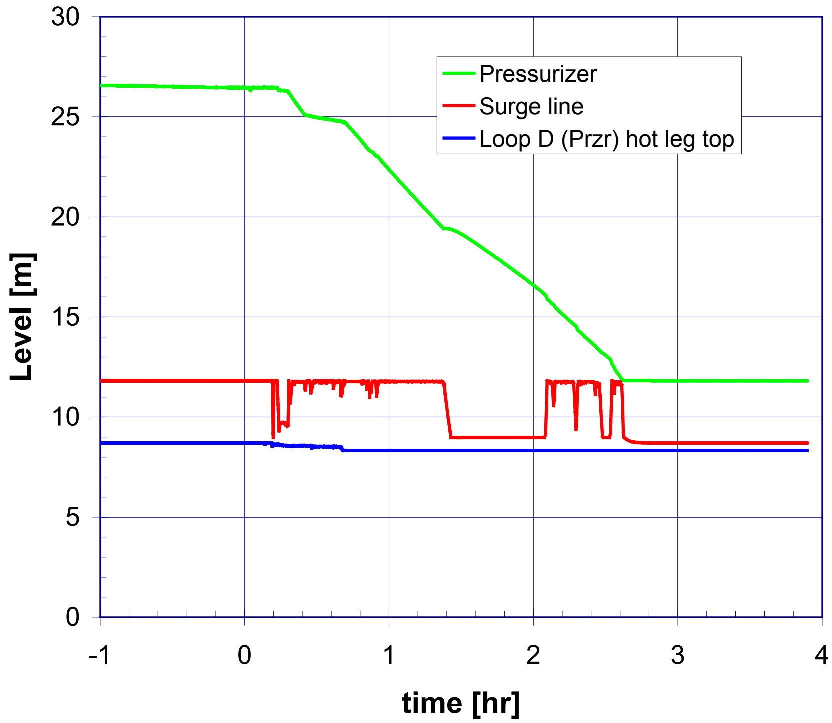

Figure A.2.1.12 Pressurizer, Surge Line, and Hot Leg Top Water Levels (Case A.2.1)

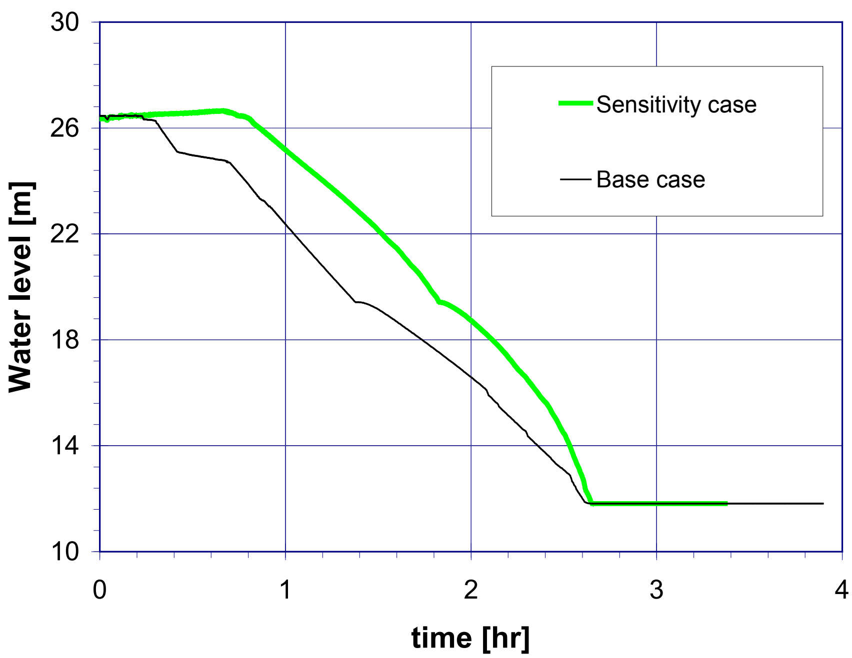

Figure A.2.1.13 Pressurizer Water Levels: Base Case and Case of Active SPARC Model in the Surge Line to Pressurizer Flow Path (Case A.2.1)

Figure A.2.1.3 shows that the coolant temperatures at the core exit also increase. Figure A.2.1.4 shows that the RHR pump flow rate increases according to its automatic control, which attempts to keep the core exit temperature constant. The increase in the pumping rate depends on the assumed details of the pump control logic (but see a sub-variant of Case A.2.4).

Figure A.2.1.3a shows oscillations in the core-exit coolant temperature during the first ten minutes of the transient. Related oscillations also occur in the break flow rate, RCS pressure, and the RHR pumping rate. These oscillations are caused by heat exchange with the steam generators and are discussed in more detail below. In Figures A.2.1.3a and A.2.1.3b, large temperatures that appear intermittently after 48 minutes indicate times when a core-exit control volume has momentarily dried out. At such times, these figures display a vapor temperature. The lower liquid temperature is displayed at times when liquid exists.

The RHR pump takes suction from control volume CV311, the up-stream bottom part of the pressurizer hot leg that has a liquid volume of 0.75 m^3 when full. CV311 remains full until about 0.7 hours. It is assumed that the RHR pump may pump at any rate up to its zero-head maximum (5,000 gpm (0.315 m^3/s), or 275 kg/second for a water density of 872.4 kg/m^3), as long as the leg contains any significant amount of water. (See the write-up for Case A.2.4 below for more discussion.) The actual rate during such times reflects throttling implemented by the control and is typically 50 to 60 kg/second (Figure A.2.1.4). Once CV311 becomes essentially empty, the RHR pumping rate is effectively the rate at which (for whatever reason) water re-appears in CV311. On the other hand, sensitivity Case A.2.4 considers that the RHR pump will trip irreversibly the first time that CV311 becomes essentially empty. In Figure A.2.1.4, the RHR pump operates at a reduced rate from about 0.7 to 2.7 hours, during which time CV311 typically contains from 1 to 2 L of water. This water is continuously replenished by a flow from the pressurizer and is pumped out as fast as it flows in. As Figure A.2.1.5 shows, the pressurizer empties at about 2.6 hours. Only then does the RHR pump stop altogether. This treatment of RHR suction loss is reconsidered in sensitivity Case A.2.4, which shows that the times to core damage are a little earlier if the RHR pump trips irreversibly at 0.7 hours. In that case, the peak cladding temperature exceeds 1,478 K about 8 minutes earlier than in the base case; the difference is greater for earlier-indicating surrogate measures.

Figure A.2.1.6a shows the water levels in the vessel. Generally, for analyses of the Byron plant scenarios, the core water level is defined by summing the heights of the collapsed water columns in the stack of control volumes that runs from the lower plenum to the dome, by way of the second computational ring of core control volumes. This figure shows that in the present case, there is not a significant difference among water level definitions that use other rings, except that the outer rings define a slightly higher elevation for the plateau that extends from about 0.3 to about 0.7 hours. This plateau is high enough to keep the core completely covered. The initial fall of the core water level to this plateau is mainly a pressure effect. The core water level is depressed by the steam building up in the dome until the level reaches the top of the hot legs, thus allowing the steam to escape into the RCS loops. The downcomer remains full until about 0.7 hours. In addition to the vessel water levels, Figure A.2.1.6b co-plots the water levels in the steam generator tubes (SGTs) and the pressurizer water level. This figure shows that the downcomer remains full until shortly after the SGTs are empty (at about 0.65 hours). Once the downcomer is no longer full, the core water level falls rapidly. The active fuel uncovers for the first time at 0.75 hours, and the top-third of the active fuel uncovers for the first time at 0.79 hours. Soon, however, drainage from the pressurizer substantially stops further uncovery, and another plateau with one-third to one-half of the core uncovered extends from about 0.8 to 2.7 hours. This plateau ends shortly after the pressurizer empties (2.64 hours). According to the measure that associates core damage with a maximum core exit coolant temperature in excess of 922 K, core damage occurs at 2.93 hours. The first gap release occurs at 3.16 hours. The core damage criterion of peak cladding temperature in excess of 1,478 K occurs at 3.34 hours. After 3.65 hours, the gap inventory of the entire core has been released. The calculation ended at 3.90 hours because of numerical problems associated with the melting of the heat structure that represents the upper core support plate.

Figure A.2.1.7 co-plots the cumulative break flow; cumulative RHR pump flow; and decreases in liquid inventory inside the steam generator tubes, vessel liquid inventory, and pressurizer liquid inventory. Consideration of these curves (and their slopes) leads to the following approximate account of the break and the regions that feed it. Until about 15 minutes after the start of the transient, the water lost to the break comes from the core and the SGTs. From then until about 0.65 hours, the SGTs by themselves are the predominant source. The final source (i.e., after about 2.7 hours) is identifiable as the vessel, while pressurizer draining is the main source during approximately 0.9 to 2.6 hours. The flow through the RHR pump approximately equates to the break flow, except during the range of times from about 0.3 to 0.9 hours (when the break flow is larger) and after about 2.7 hours (when there is no RHR pump flow).

Recall that in Figure A.2.1.1b, the break flow transitions from all liquid to all steam during the time range from about 0.8 to about 0.9 hours. This transition is a result of the clearing of the loop seal in Loop B (which is one of the broken loops). Figure A.2.1.8 illustrates the clearing of the seal; flow path FL 542 represents the loop seal. MELCOR allows gas to enter a flow path once the water level in the upstream control volume (in this case, CV 542) is low enough to uncover the top of the flow path. This partial uncovery occurs at 0.79 hours and is attributable to hydrostatic pressure. Although the flow path remains covered on the downstream side, steam is able to traverse the water column standing in CV544 and thus reach the break. No other loop seal clears. The clearing of only one loop seal is typical: once the steam flows freely into a loop that has just cleared, any clearing tendency will be reduced in the other loops.

The calculation was carried out with the assumption that the initial secondary-side temperatures and pressures are the same as those on the primary side. In Figure A.2.1.5, the secondary-side water level in a typical loop (Loop D) remains relatively constant compared with the water level in the pressurizer. This result indicates that the assumed lack of FW is not important. Figure A.2.1.9 shows the water levels in all of the steam generators on a fine scale. Figure A.2.1.10 shows the SG pressures. The pressures of the four SGs coincide because they remain connected by the common header as long as the MSIVs are open; scenario definition holds them open indefinitely in anticipation of sensitivity cases that make use of the condenser steam dump valves.

Figures A.2.1.11a and A.2.1.11b show the heat flow from the fluid inside the steam generator tubes into the tube walls. Although the problem’s initial conditions were chosen with the intent to minimize the interaction between the primary and secondary sides of the RCS, the cooling provided by the steam generators is nevertheless significant, amounting to around 5 MW in the times later than about 1 hour (Figure A.2.1.11a). In the first hour, the heat flows are erratic, large in magnitude, and sometimes negative (i.e., from the secondary side to the primary side). The heat of condensation (either given up to the tube walls when vapor condenses on the tubes’ inner surfaces, or drawn from the tubes’ surfaces when liquid films there evaporate) results in the large magnitude. These erratic heat flows cause the oscillations of the core exit temperature mentioned above in connection with Figure A.2.1.3a, as Figure A.2.2.11b shows.

The slow emptying of the pressurizer is striking; it is not complete until 2.64 hours. Figure A.2.1.12 shows the water levels in the pressurizer, surge line, and hot leg top. There are times when the pressurizer water is suspended above an empty surge line. At 1.5 hours for example, both the surge line and the hot leg top are empty; the pressure in the pressurizer is 1.9481 MPa; and the pressure in the hot leg top is 2.0114 MPa. In MELCOR, the pressure of a control volume may always be considered to apply at the top of any water column it contains. If the control volume is dry at that time, the height dependence of the pressure is small because of the typically low density of gas. At 1.5 hours, the pressurizer water level and liquid density imply a pressure of 2.0111 MPa at the bottom of the pressurizer. In other words, the conditions are nearly those of a hydrostatic equilibrium: the pressure difference acting to drive a flow is small and even slightly negative at this time, according to this estimate. Note also that as Figure A.2.1.7 shows, the mass rate of pressurizer draining is similar to the RHR pumping rate and the break flow. Regarding the break flow during the relevant times (e.g., 1.5 to 2.6 hours; see Figure A.2.1.6b); it has the low mass rate appropriate to a steam flow (see Figure A.2.1.1b). These behaviors come about as the water that drains from the pressurizer is immediately pumped to the intact cold legs and moves from there to the downcomer. During the relevant times, the downcomer water level is too low for the water to pass on to the broken cold legs. It instead enters the core, boils, passes the cleared loop, and leaves the RCS as steam through the break. The rate of pressurizer draining (typically about 4.7 kg/second) is actually less than the rate at which water can be vaporized by the available decay heat (about 10.7 kg/second). The actual rate is affected by the modeling of the two-phase flows in the pressurizer/surge line/hot leg regions that lead to the (near) hydrostatic pressure balance referred to above. Two of these aspects are investigated in the variant calculations described in the following two paragraphs. This pressure balance prevents the pressurizer water from reaching the break as liquid via larger, gravity-driven flows.

The pressurizer draining is affected significantly by a MELCOR phenomenological modeling parameter, referred to as the momentum exchange length, which defines the length over which momentum is exchanged between gas and liquid when these two phases simultaneously pass through a flow path. This parameter is intended to model counter-current flow limitation. Whereas the Byron model uses 0.1 m for this parameter for the RCS flow paths, two sub-variants of Case A.2.1 (undocumented apart from remarks in this paragraph) use 0.01 m and 1.0 m. The modified values are applied to the flow paths that connect the surge line to the hot leg and to the pressurizer. The former case gives 1.9 hours for the time when the pressurizer empties, while the latter case gives 3.2 hours. Although this variation is sizable, pressurizer draining can be considered slow in all the cases.

MELCOR has a set of models referred to collectively as the SPARC model, which describes interactions between liquid and gas bubbles. Inactive by default, the SPARC model may be activated for individual flow paths, though activating the model increases the computational time. In the base case, the model is active for the flow path that connects the hot leg to the surge line, but the model is inactive for the flow path that connects the surge line to the pressurizer. For another sub-variant of Case A.2.1 that is undocumented apart from remarks in this paragraph, the SPARC model is activated for the surge line to pressurizer flow path. Figure A.2.1.13 compares the pressurizer draining with and without an activated SPARC model for this flow path. Activation of the model delays the onset of pressurizer draining but later results in faster draining, so that the time for the pressurizer to empty is similar in both calculations. According to the measure that associates core damage with the spatially-maximized cladding temperature exceeding 1,478 K, the time to core damage is 3.44 hours with the SPARC model activated for the surge line to pressurizer flow path, versus 3.34 hours in the base case.

A.3.2.2 Results of Case A.2.2 (Operator Actions at 0.5 Hours)

This scenario progression is outlined in Table A.8 and illustrated in Figures A.2.2.1 to A.2.2.4. In this case, one charging pump and one SI pump become available at 0.5 hours. The pumps are assumed to operate according to their normal-operations ECCS modes. In particular, because the pressure in the vessel is low, the pumps inject at their maximum rates. They take their suction from the RWST, which is assumed to be refilled as necessary. As a result, the switchover to the recirculation mode never occurs. The pumps inject into the cold legs of Loops A and D (i.e., the intact legs).

Table A.8 Timing of Key Events Predicted for the MELCOR Model Shutdown Sensitivity-Case 2 Calculation

Event Description	**Time (hours)**
Break in RHR-B lines outside the containment	0
Initiation of injection by one charging and one SI pump	0.50
Reduced RHR-A pump rate because of the low hot leg water level	~1.4
Core partially uncovered	1.59
Pressurizer empties (i.e., contains less than 10 kg of liquid)	1.93
Latest time to refill the RWST	2.96
Calculation ends	6.00
Coolant temperature at core exit reaches 922 K (1,200 °F)	Do not occur
First gap release	
Peak clad temperature reaches 1,478 K (2,200 °F)	

Figure A.2.2.1 shows the flow rates of the break and the combined injections related to the charging and SI pumps. The pumps inject at a combined constant rate of 79.8 kg/second at all times after 0.5 hours. The separate contributions of the SI and charging pumps are 43.6 and 36.2 kg/second, respectively. As the analogous base-case figures (Figures A.2.1.1a/b) show, the peak break flow rate that occurs in the absence of the injections is about 97 kg/second and occurs at about 0.7 hours. The ability of the injected water to cool the core is limited, because the water need not pass through the core. The water can reach the breaks by way of the downcomer. For these reasons, the injections are insufficient to maintain the water level in the core before the start of core uncovery. After about 1.8 hours, however, the core water level is maintained with roughly the top quarter of the core uncovered. This finding is based on collapsed water levels in the second innermost computational ring. Figure A.2.2.2 shows the water levels in the vessel.

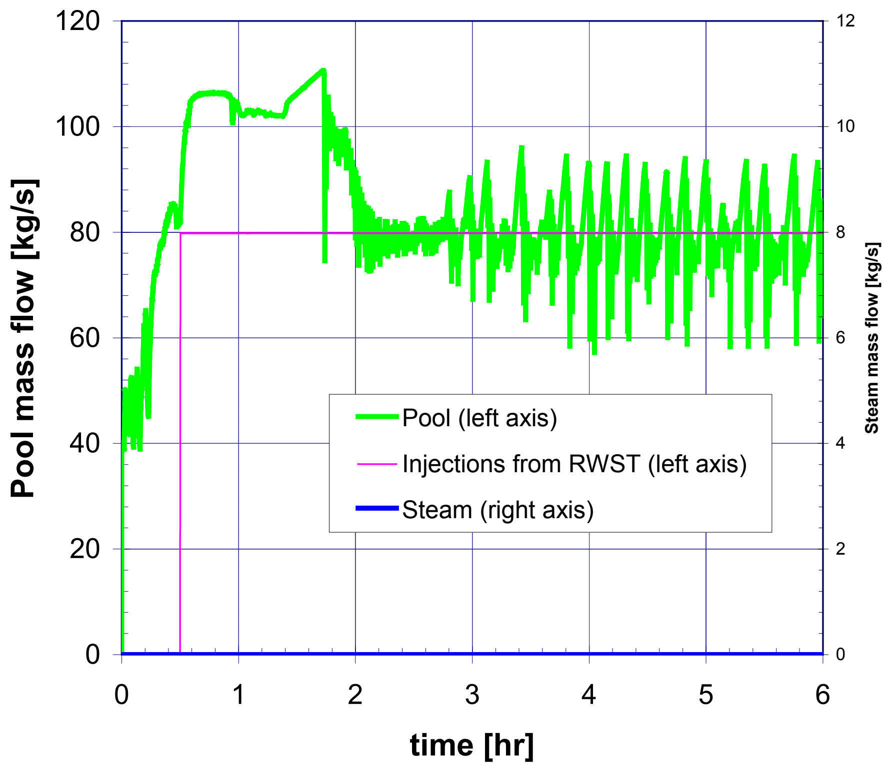

Figure A.2.2.1 Break and ECCS Injections Flow Rates (Case A.2.2)

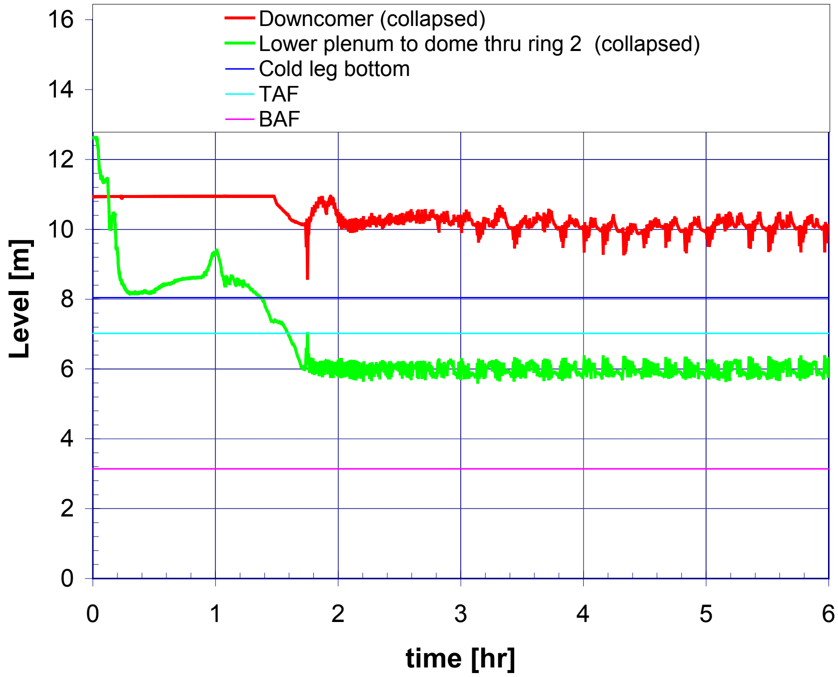

Figure A.2.2.2 Water Level in the Reactor Pressure Vessel (Case A.2.2)

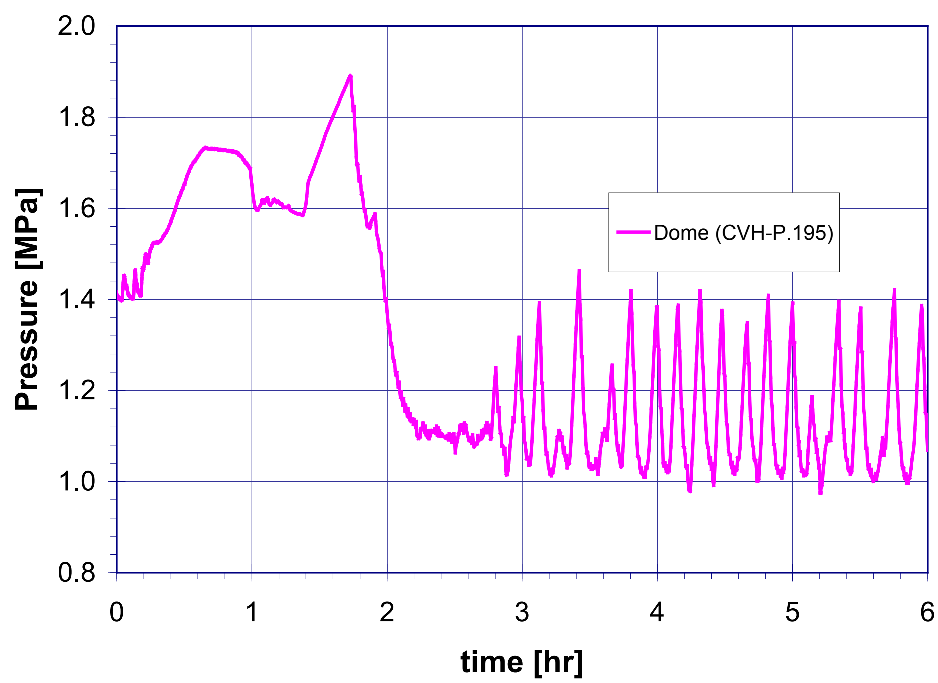

Figure A.2.2.3 RCS Pressure (Case A.2.2)

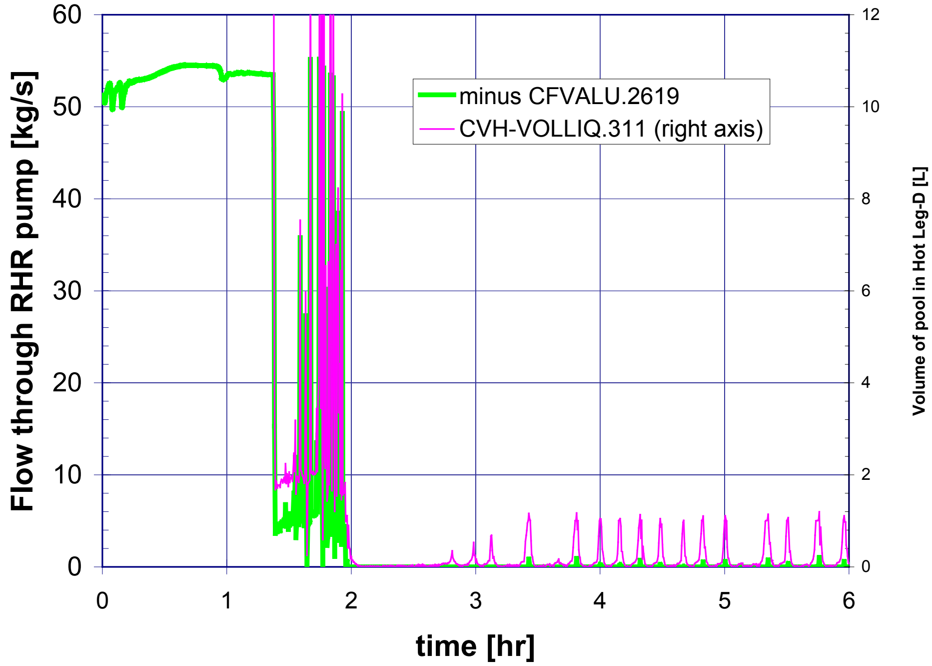

Figure A.2.2.4a RHR Pump Rate and Liquid Volume in Hot Leg D (Case A.2.2)

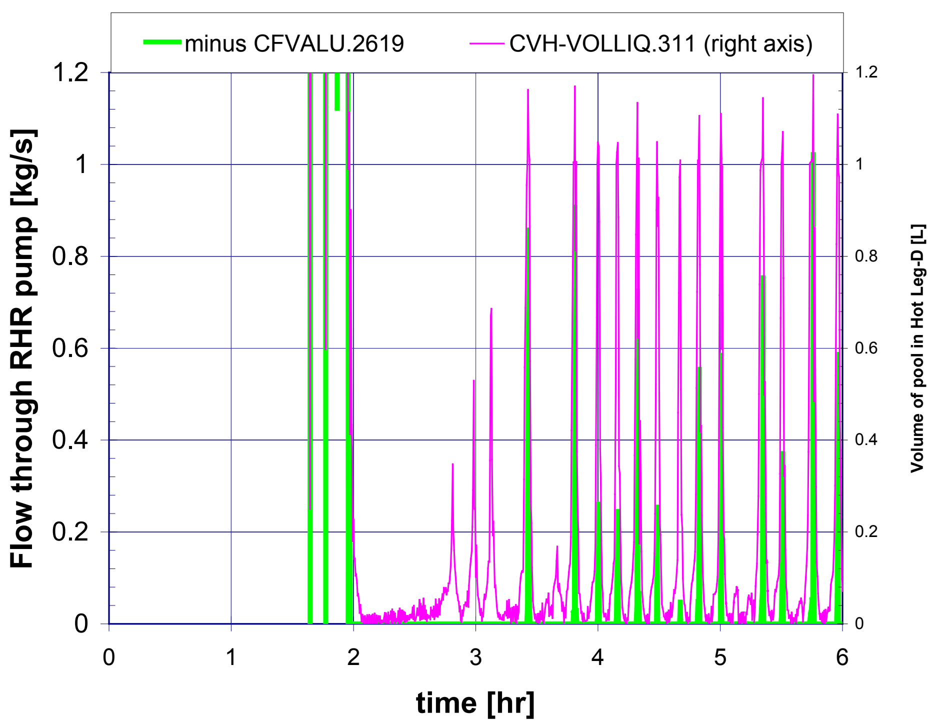

Figure A.2.2.4b RHR Pump Rate and Liquid Volume in Hot Leg D – Narrower Range (Case A.2.2)

Figure A.2.2.3 shows the pressure in the vessel. Periodic peaks are seen starting at about 2.8 hours. At the same time, there are peaks and dips, respectively, in the core and downcomer water levels (Figure

A.2.2.2). These features are presumed to be results of spontaneous oscillations of the water level and water flow between the core and cooler regions that periodically quench the steaming, thereby bringing about the pressure oscillations that sustain the oscillations of flow and water level. This behavior is also reflected in the activity of the RHR pump (Figures A.2.2.4a/b). The RHR pump is considered to have full suction until about 1.4 hours. From then until about 2.0 hours, when the pressurizer empties, the hot leg bottom (i.e., control volume CV311, with a liquid inventory of 0.75 m^3 when full) sporadically contains from 1 to 11 L of water, and the RHR pump operates at reduced rates according to the assumed modeling of suction loss. The rate is set to zero only for a liquid inventory of less than 1 L. At nearly all times after 2.0 hours, the hot leg bottom is essentially dry (less than 1 L of liquid). At the times of the pressure peaks, however, the liquid volume briefly exceeds 1 L and the RHR pump injects at a much reduced rate. The duration of these injections is about one minute, and the peak injection rates are about 1 kg/second. The ad hoc assumptions about RHR suction loss are reconsidered in sensitivity Case A.2.5, where substantially similar results are obtained if the RHR pump trips irreversibly at 1.38 hours.

At 2.96 hours, the cumulative injection by the charging and SI pumps reaches 187,736 gallons. This is the assumed volume delivered from the RWST after being drawn to Lo-2 from an initial 89 percent full status. The assumption is that by 2.96 hours the RWST will either be refilled, replenished at a rate to balance the depletion or an alternative injection source has been established.

The only surrogate measures that ever indicate core damage are those for a partial core uncovery. The first indication of water level at TAF occurs at 1.59 hours, and the first indication of an uncovery of the top third of the core occurs at 2.14 hours. The maximum cladding temperature never exceeds 560 K. The calculation ends at 6 hours.

A.3.2.3 Results of Case A.2.3 (Operator Actions at 1.0 Hours)

This case is the same as Case A.2.2, except that the injections from one charging pump and one SI pump begin at 1.0 hours instead of 0.5 hours.

Figure A.2.3.1 shows the water levels in the vessel (core and downcomer) for the present sensitivity case. Figure A.2.3.2 compares the water level in the core in the present sensitivity case versus the water level in Case A.2.2. The time of the start of partial core uncovery in Case A.2.3 is 0.75 hours, which is the same as the base case. In comparison, the earlier time of operator actions credited in Case A.2.2 postpones this time until 1.59 hours. The maximum amount of core uncovery, however, is only slightly greater in the case of the later operator actions.

The only surrogate measures that ever indicate core damage are those related to partial core uncovery. The first indication of the water level at TAF occurs at 0.75 hours, and the first indication of uncovery of the top third of the core occurs at 0.79 hours. The maximum cladding temperature attains a peak of 705 K at 1.01 hours. The calculation ends at 6 hours.

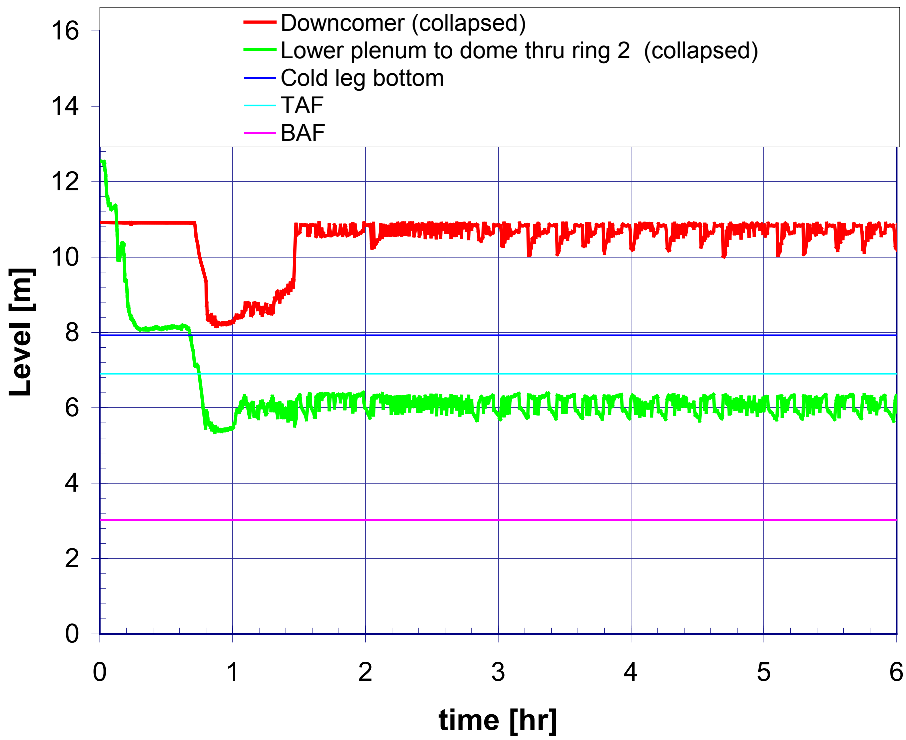

Figure A.2.3.1 Water Levels in the Vessel (Case A.2.3)

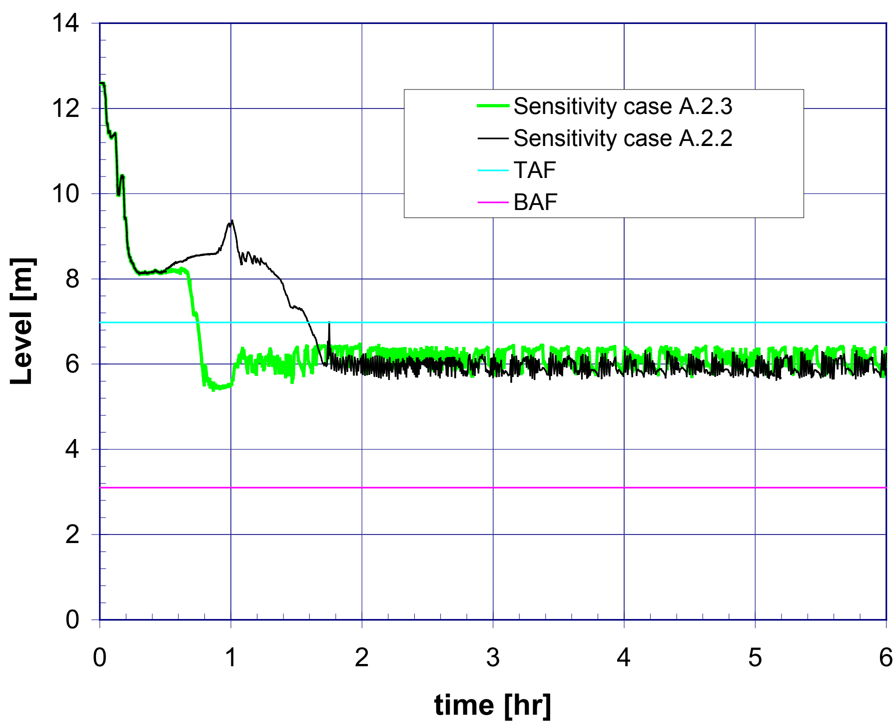

Figure A.2.3.2 Water Levels in the Core (Case A.2.3)

A.3.2.4 Results of Case A.2.4 (RHR Pump Trips Irreversibly on Suction Loss)

This case reconsiders the assumptions about the behavior of the RHR pump as it loses suction at the pressurizer-loop hot leg. In all other cases (except Cases A.2.5 and A.2.9), it is assumed that the RHR pump may pump at any rate up to its zero-head maximum any time that the leg contains any significant amount of water. The actual rate is set by the feedback control that attempts to keep the core exit coolant temperature constant. Since MELCOR sources are used to model the pump, no water removal from the hot leg may occur if no pool is there, so the assumption is that the pump rate becomes zero when the leg bottom (modeled by control volume CV311) contains less than 1 L of water. For a numerically smoother implementation of this control, a factor ranging linearly from 0.0 at 1 L of hot leg liquid inventory to 1.0 at 11 L is applied to the rates of draw and injection. This factor allows the RHR pumping rate to become equal to the rate at which (for whatever reason) water reappears in CV311. In the base case, this logic leads to a period of a low RHR pump rate that lasts from about 0.7 to 2.7 hours, during which time the hot leg liquid inventory, sustained by pressurizer draining, fluctuates at around 1.5 L. The present sensitivity case considers the consequences of this period of possibly unrealistic RHR pump activity.

For a reconsideration of the modeling of RHR pump suction loss, first note that, according to Section 6 of the Byron/Braidwood FSAR (pp. 6.5-25), an RHR pump has a net positive suction head (NPSH) requirement of about 3.6 m when operating at 3,000 gpm (0.189 m^3/s). Less NPSH would be required at the smaller pump rates that arise in the Shutdown configuration. Also according to the FSAR, the elevation of an RHR pump has a center line about 8.8 m below the water level in the sump, when the sump is full. Considering the additional head provided by the elevation of the hot leg above the water level of the full sump (estimated at about 4.2 m), it seems clear that the NPSH requirement will be met any time that there is liquid in the hot leg, on the assumption that the suction point is at the leg bottom. (As discussed later in this section, the effects will likely be small if the suction actually is taken from a point above the bottom). In fact, after dryout of the hot leg, the RHR pump will continue to have suction until it pumps out the water in the part of its suction line above the lowest elevation that provides the required NPSH. Since the corresponding volume cannot be estimated without detailed knowledge about the line, it is defensible to suppose that the pump loses suction when the hot leg empties. It is expected that this time should differ negligibly from 0.70 hours, which is the time when the liquid inventory in control volume CV311 first fell below 11 L in the base case. (Later in the base case, the pump rate was throttled according to the ad hoc treatment described above, so that the hot leg did not rigorously empty until about 2.7 hours.) In the context of this discussion and for a more stringent treatment of the RHR pump operation as the pump loses suction, this sensitivity case therefore assumes that the pump trips irreversibly at 0.70 hours. Otherwise, the sensitivity case is the same as the base case.

Figure A.2.4.1 compares the sensitivity and base cases with respect to their pressurizer water levels. The differences indicate that the action of the pump in the base case helps to draw down the pressurizer, so that it empties about ten minutes earlier than it does in the sensitivity case. Figure A.2.4.2 compares the vessel water levels and shows minor differences in favor of deeper coverage in the base case. Somewhat larger differences appear in Figure A.2.4.3 for the (radially maximized) temperature of the coolant at the core exit. In particular, during the time from about 0.7 to 2.7 hours, the sensitivity case does not show the sporadic drying out of the outer core exit regions, which occurs in the base case as a result of the intermittent low flow-rate pumping of the RHR pump. (Figure A.2.4.3 shows the temperature of the vapor when no liquid is present.) The time at which the core exit temperature reaches 922 K is 2.52 hours in the sensitivity case versus 2.93 hours in the base case. The time at which the peak cladding temperature reaches 1,478 K is 3.21 hours in the sensitivity case versus 3.34 hours in the base case.

It can be concluded that the base-case treatment of the RHR pump suction loss, which allows pumping at reduced rates while there is very little liquid in the hot leg, leads to 5 to 30 minute delays of core damage depending on the surrogate measure and relative to results obtained with bounding assumptions about pump operation as suction is lost (i.e., irreversible trip).

Case A.2.4 assumes an irreversible pump trip at what is still a rather small volume, 11 L, in CV311. However, because the hot leg is emptying fast at this time, little difference would arise given any other choice for a trip on the water level of the (partially full) hot leg. For example, the time when the water level is at the hot leg mid-plane—CV311 is full but CV310 is empty—is just 1.7 minutes earlier than the time adopted for the present calculation.

In a sub-variant of this sensitivity case (not documented apart from these remarks), during the beginning of the transient the RHR pumping rate and the temperature of the water returning from the RHR heat exchanger are clamped to 51.26 kg/second and 373.3 K, respectively. These are the values that hold at the end of the pre-transient part of the calculation. Otherwise, the sub-variant case is similar to sensitivity Case A.2.4. Figures A.2.4.4a/b compare the (radially-maximized) temperatures of the coolant at the core exit that result in the sub-variant sensitivity and base cases. The fine scales of Figure A.2.4.4a show that the oscillations in this temperature that occur during the first ten minutes of the transient are not a result of the RHR pumping rate control algorithm. In fact, they are a result of heat exchange with the steam generators, which is discussed in the base-case write-up. Figure A.2.4.4b is very similar to Figure A.2.4.3, which indicates that the particular assumptions of the RHR pumping rate control algorithm are not important in determining the times of core damage. The time at which the peak cladding temperature reaches 1,478 K is 3.20 hours in the sub-variant case versus 3.21 hours in the sensitivity case.

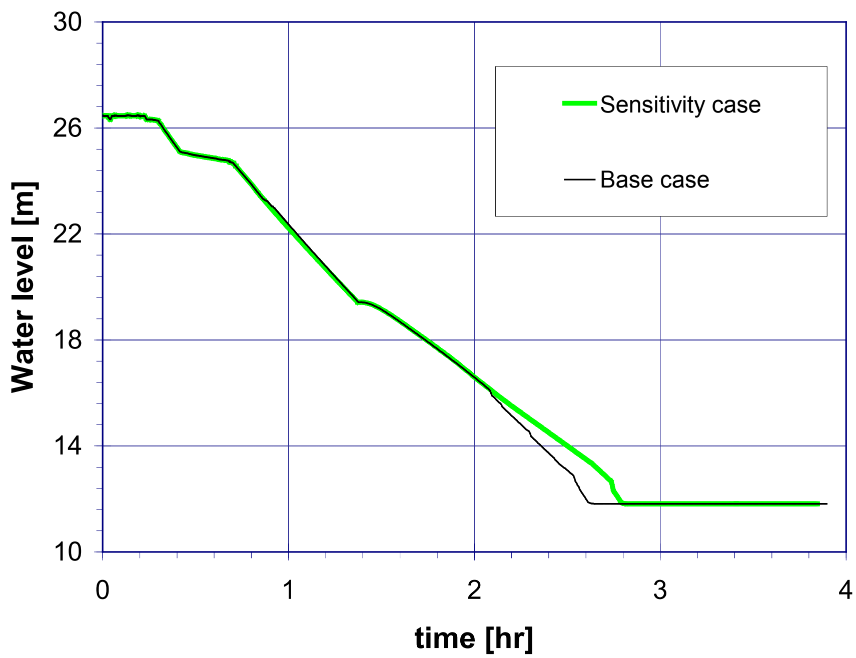

Figure A.2.4.1 Water Levels in the Pressurizer (Case A.2.4)

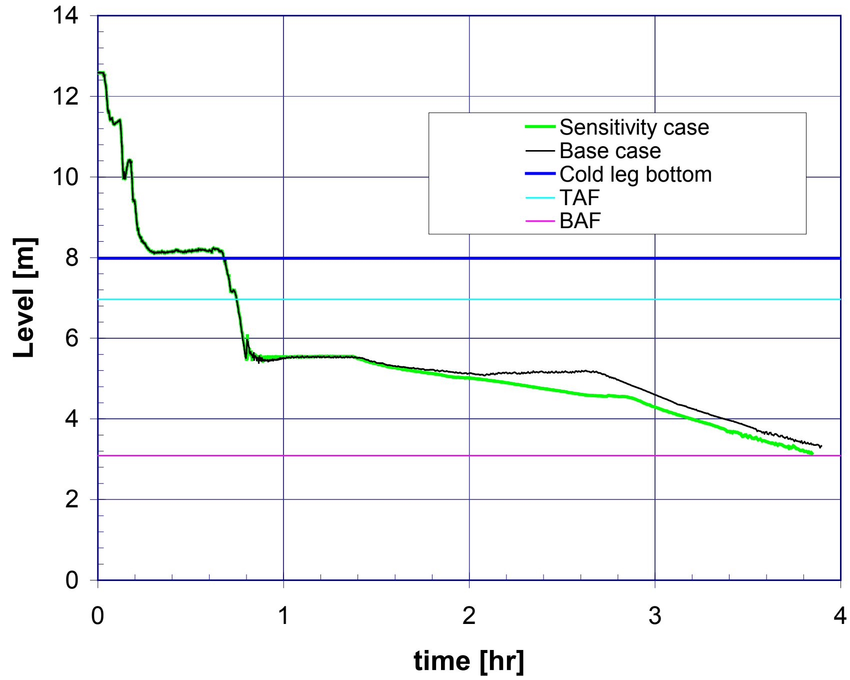

Figure A.2.4.2 Water Levels in the Core (Case A.2.4)

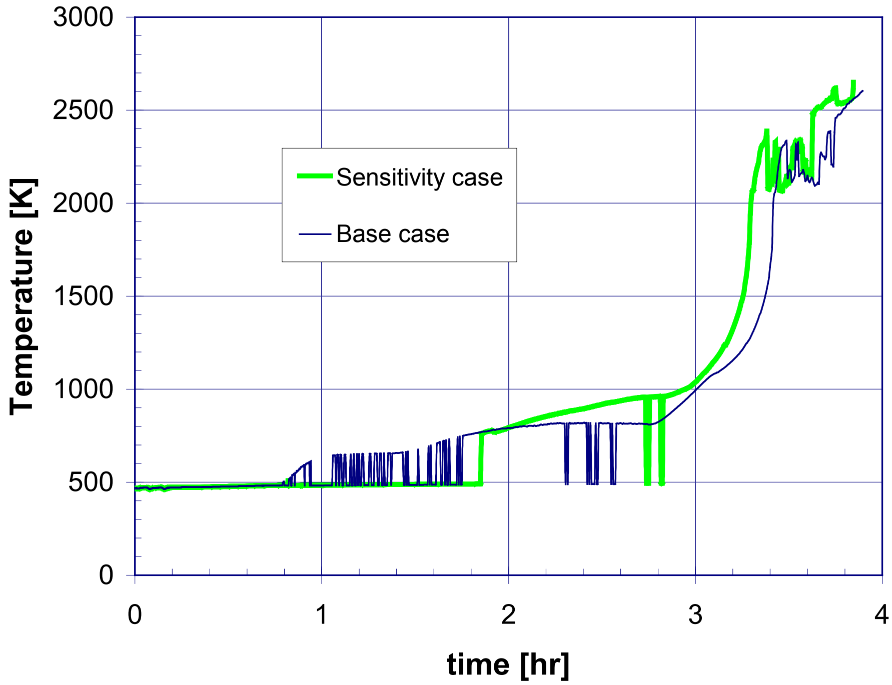

Figure A.2.4.3 Temperatures of Coolant at the Core Exit (Case A.2.4)

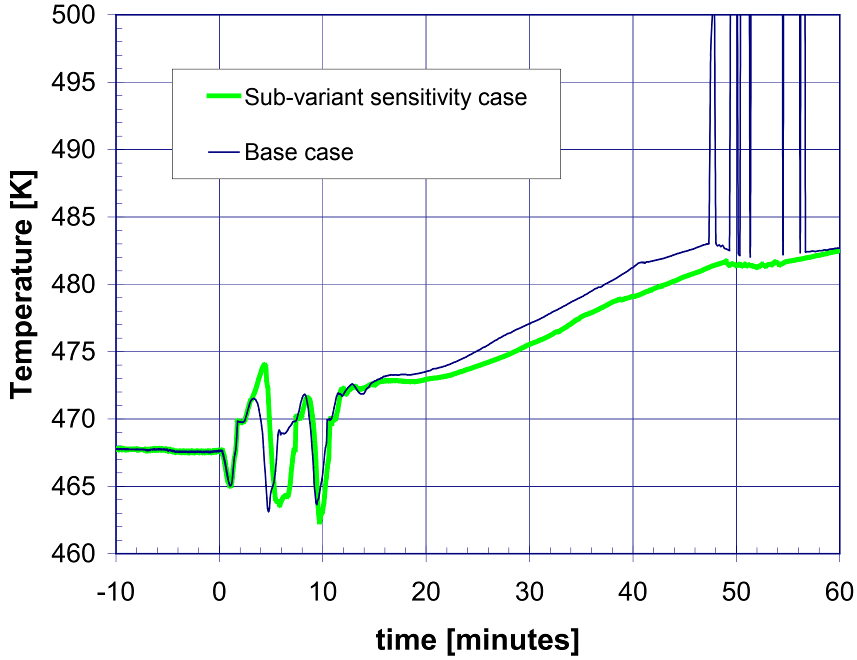

Figure A.2.4.4a Temperatures of Coolant at the Core Exit (Case A.2.4 short-term)

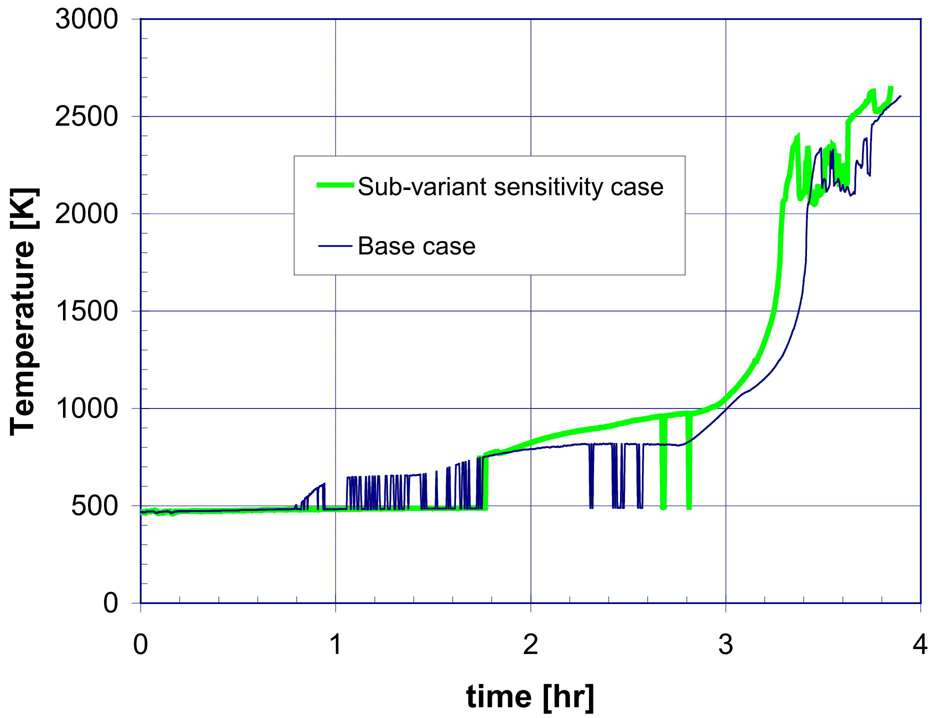

Figure A.2.4.4b Temperatures of Coolant at the Core Exit (Case A.2.4 longer-term)

A.3.2.5 Results of Case A.2.5 (Operator Actions at 0.5 Hours; RHR Pump Trips Irreversibly on Suction Loss)

Case A.2.5 combines Cases A.2.2 and A.2.4. That is, as in Case A.2.2, injections by one charging pump and one SI pump begin at 0.5 hours. As in Case A.2.4, instead of continuing to pump at a reduced rate during times when the water level in the hot leg is low, the RHR pump trips irreversibly. The time of 1.38 hours for the pump trip in the present calculation is identified as the time when the liquid inventory in control volume CV311 first falls below 11 L in Case A.2.2. (Later in Case A.2.2, the pump rate is throttled so that it remains appreciable until about 2.0 hours.)

Figure A.2.5.1 compares the (radially maximized) coolant temperatures at the core exit as predicted for sensitivity cases A.2.5 and A.2.2. It can be concluded that the base-case treatment of the RHR pump loss of suction, which allows pumping at reduced rates while there is very little liquid in the hot leg, is not important to the prediction of core damage avoidance in Case A.2.2.

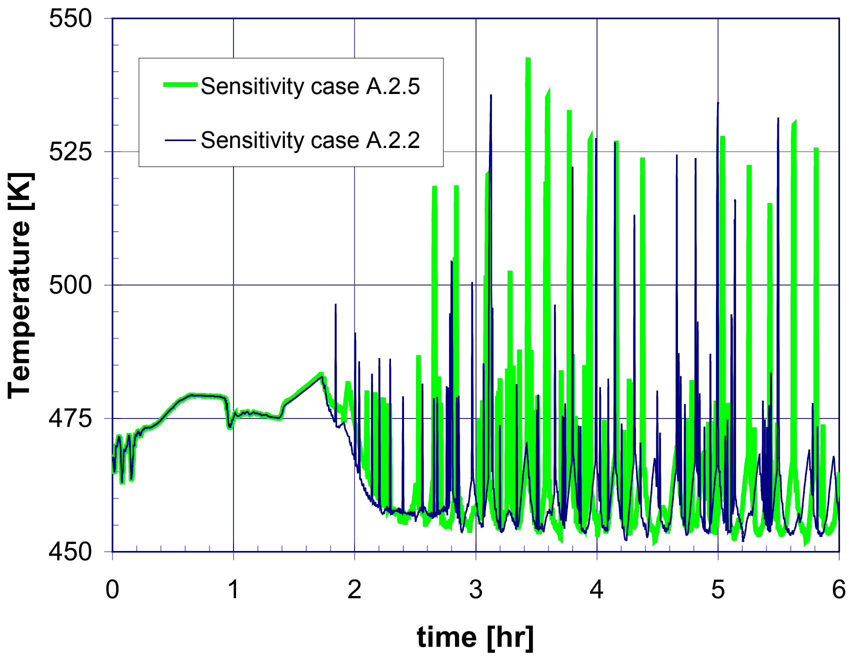

Figure A.2.5.1 Core Exit Coolant Temperature (Case A.2.5)

A.3.2.6 Results of Case A.2.6 (Enhanced Steam Generator Cooling)

Case A.2.6 is the same as the base case, except for the assumption that operators hold the condenser steam dump valve partially open starting at 0.5 hours as an attempt to enhance the cooling provided by the steam generators.

As noted previously, the general definition of the shutdown scenario is premised on the assumption of initial isothermal conditions on both sides of the RCS. This assumption was considered adequate for minimizing the role of the secondary side given the insufficient availability of plant-specific operational information, including the actual role of the secondary side. The present sensitivity case starts from the same initial state, but it now considers operator actions intended to enhance the secondary-side cooling. Although the sensitivity case is therefore likely to model an unrealistic situation, it is included for modeling insights. Note also that Step 22c of 1BOA-SD-2 (a Byron document related to shutdown) seems to indicate that if any RHR pumps are running the operators will not dump steam, which is contrary to the operator action assumed here.

Base-case Figures A.2.1.11a/b show that despite the approximately isothermal conditions that hold on both sides of the RCS at the start of the transient, the steam generators provided significant cooling up to and after the onset of core damage in the base case. In the base case, the secondary sides remained closed during the transient. That is to say, the turbine throttle and condenser steam dump valves remained closed: pressures remained below the actuation setpoints of the PORVs and SRVs, and no demand for FW ever occurred since water levels stayed high. The present sensitivity case investigates enhancement of this cooling by depressurizing the secondary sides and providing FW. The sensitivity case assumes that the dump valve remains closed until 0.5 hours, when it is then held open to a fixed fraction (4.5 percent) of its fully opened position for the rest of the calculation. (When opened fully, the dump valve's flow area, 0.109 m^2, is such as to provide a steam flow consistent with full-power plant data – steam flow 1345.2 kg/second at main steam line pressure 8.274 MPa-abs.) The AFW is also assumed available in this sensitivity case. Injections will maintain the water level as demanded, until the condensate storage tank is depleted. Calculations for a supplemental scenario, discussed at the end of this section, motivate the choice of the particular dump valve opening fraction.

Figure A.2.6.1 compares the steam generator cooling in the sensitivity and base cases. As in the base case, the cooling is erratic, and sometimes negative, during the first hour, making the comparison unclear. In the time from about 0.9 to about 2.2 hours, in the sensitivity case the steam generators provide 2 to 3 times as much cooling as they do in the base case. Even so enhanced, however, the cooling is generally less than the decay heat. After about 2.3 hours, there is no longer much difference.

Figure A.2.6.2a shows the effect on primary pressure. Figure A.2.6.2b compares the secondary pressures. Figure A.2.6.2c co-plots the primary and secondary pressures in the sensitivity case only, while Figure A.2.6.2d does the same for the base case. Figures A.2.6.2e and A.2.6.2f are temperature analogs to Figures A.2.6.2a and A.2.6.2b. They compare the base and sensitivity cases with respect to the liquid temperature inside a typical steam generator tube (Figure A.2.6.2e) versus the liquid temperature outside the steam generator tube (Figure A.2.6.2f). In the sensitivity case, both the primary-side and the secondary-side liquid temperatures decrease as a result of the operator action (the primary-side temperature meaning in this case the temperature of the liquid inside the steam generator tube). Figure A.2.6.2g shows that the temperature difference between the primary- and secondary-side liquid is affected very little by the operator action (i.e., during the time from about 0.5 to 1.5 hours).

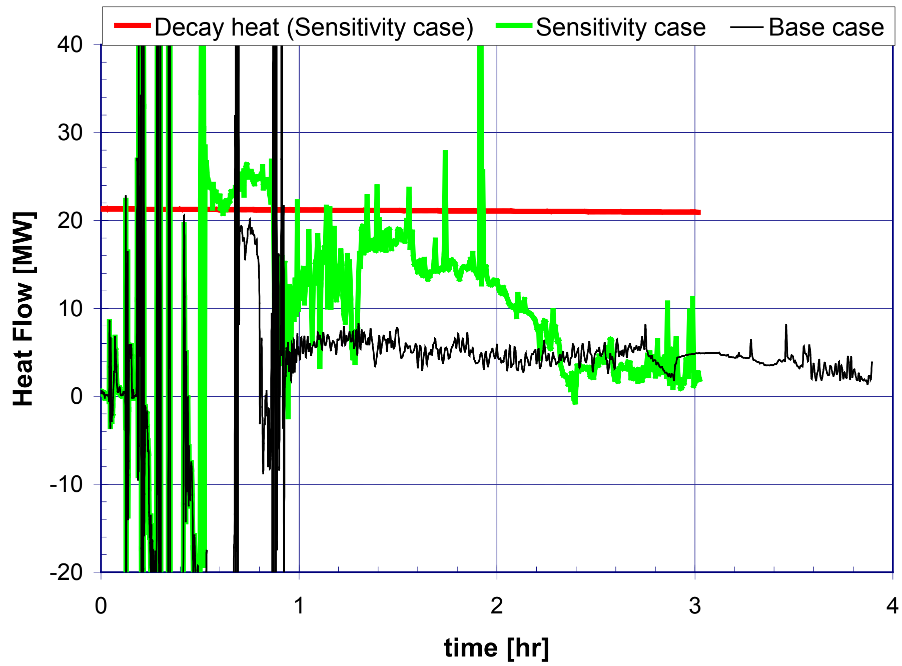

Figure A.2.6.1 Decay Heat and Heat Transferred to Steam Generators (Case A.2.6)

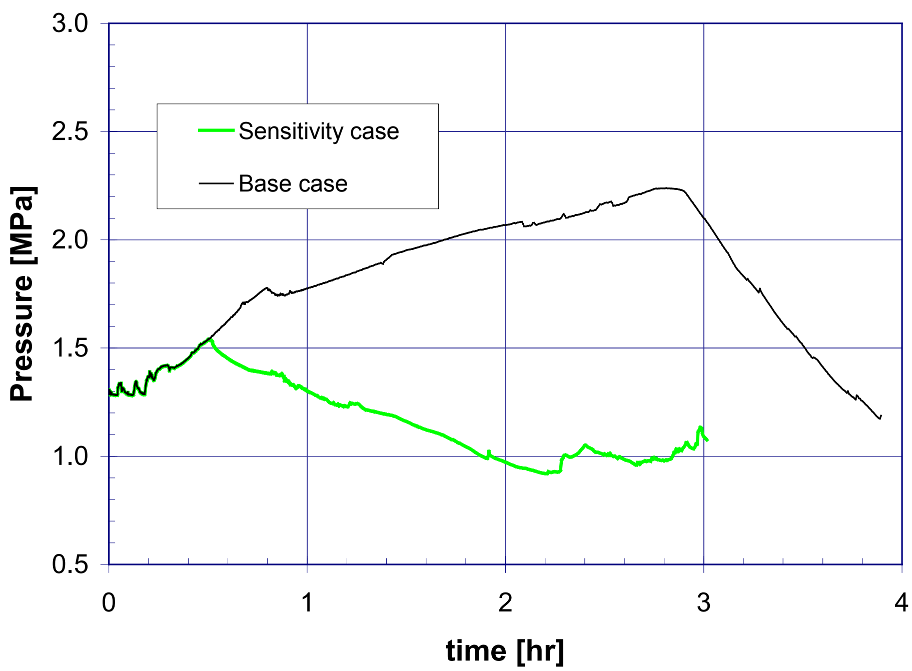

Figure A.2.6.2a Pressures at the Pressurizer (Case A.2.6)

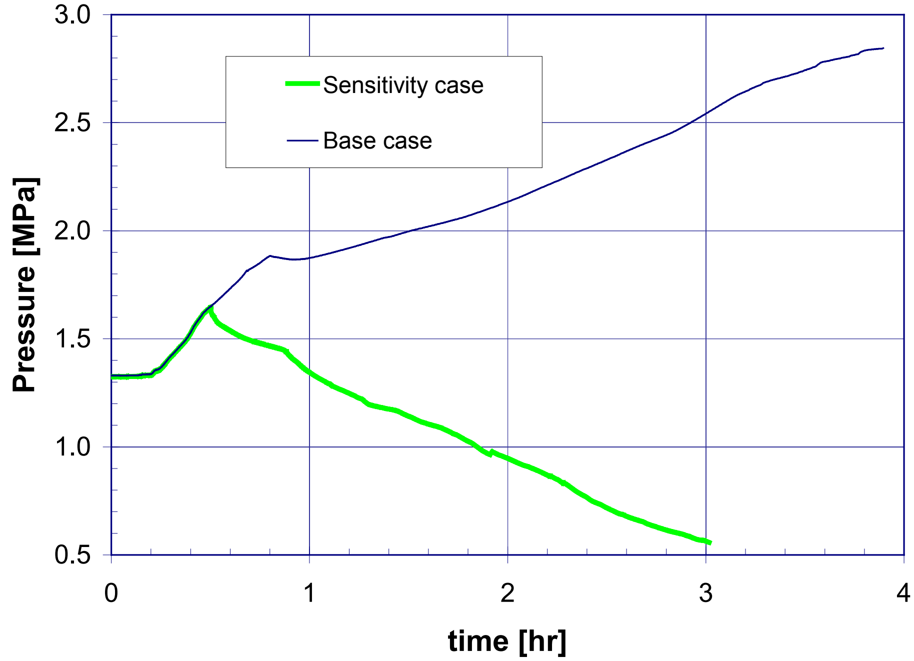

Figure A.2.6.2b Pressures at the Steam Lines Common Header (Case A.2.6)

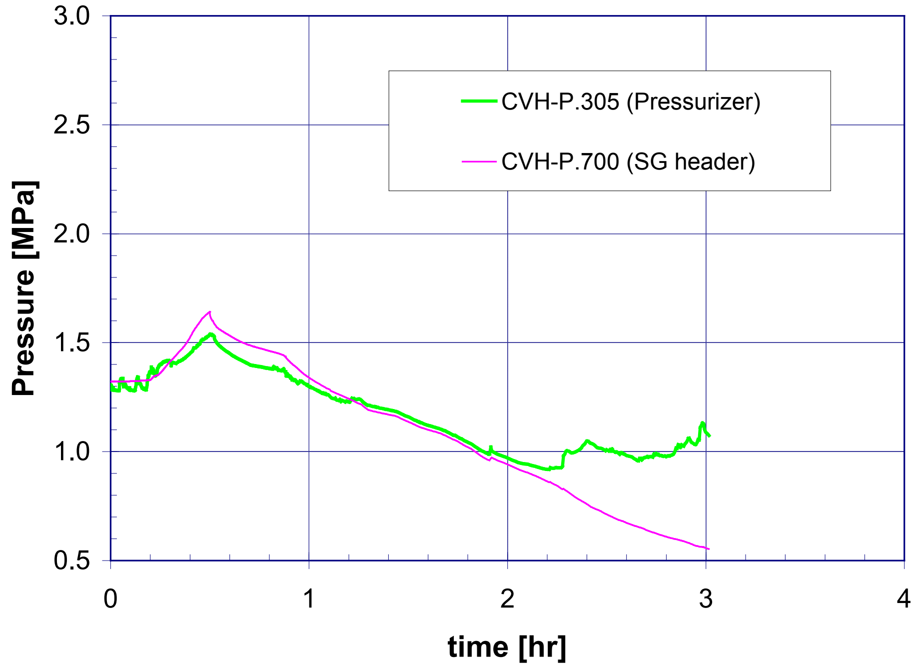

Figure A.2.6.2c Primary and Secondary Pressures (Sensitivity Case for A.2.6)

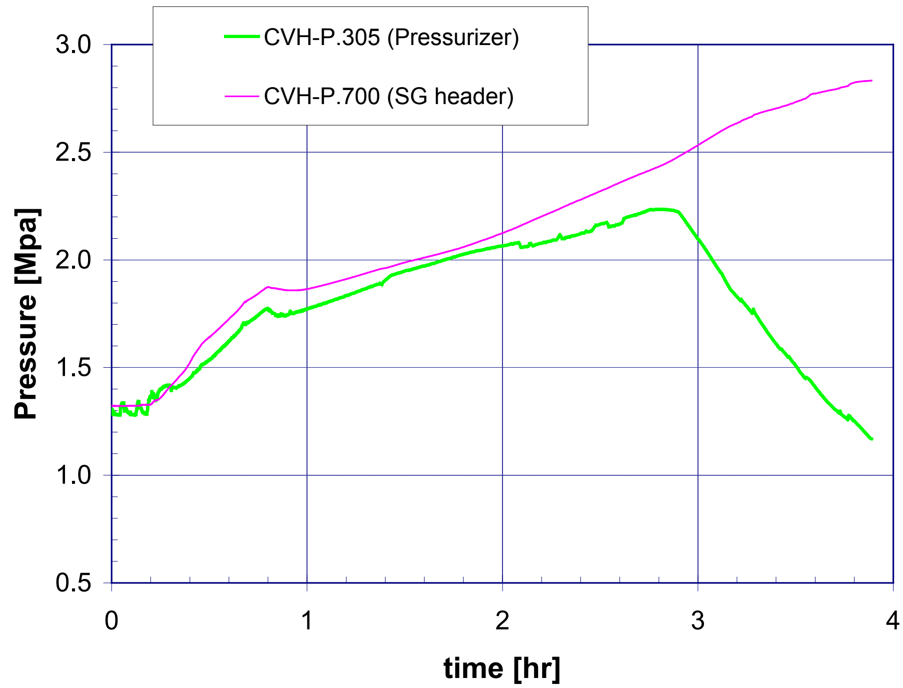

Figure A.2.6.2d Primary and Secondary Pressures (Base Case for A.2.6)

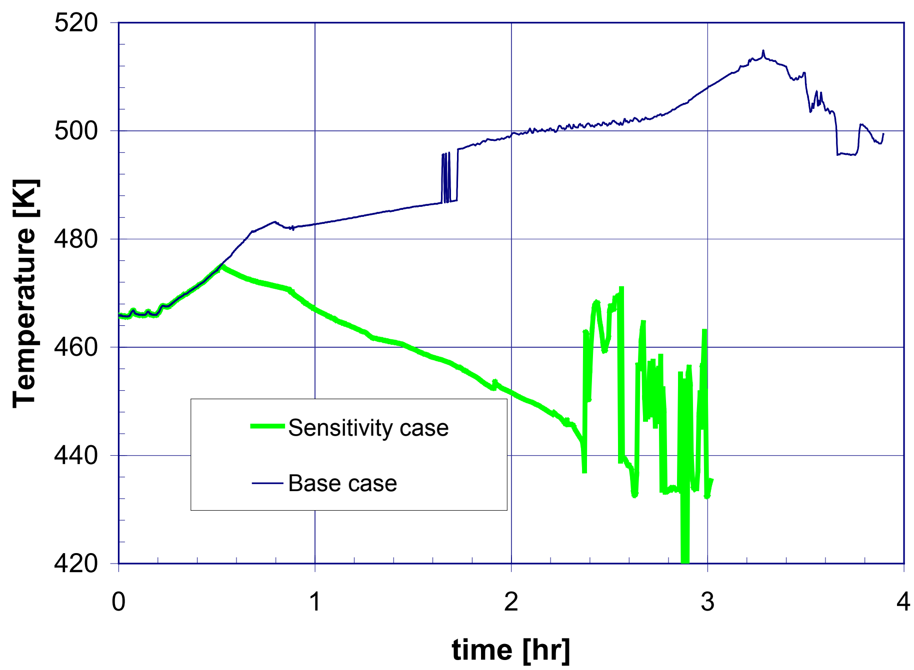

Figure A.2.6.2e Primary-Side Liquid Temperature - Loop A SGT; CV430 (Case A.2.6)

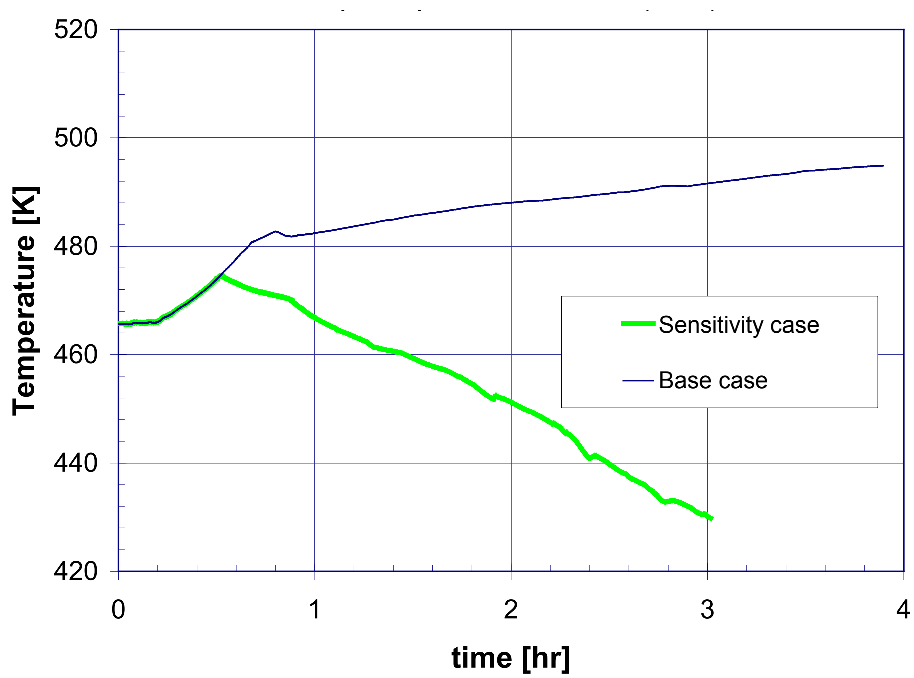

Figure A.2.6.2f Secondary-Side Liquid Temperature - Loop A Boiler; CV460 (Case A.2.6)

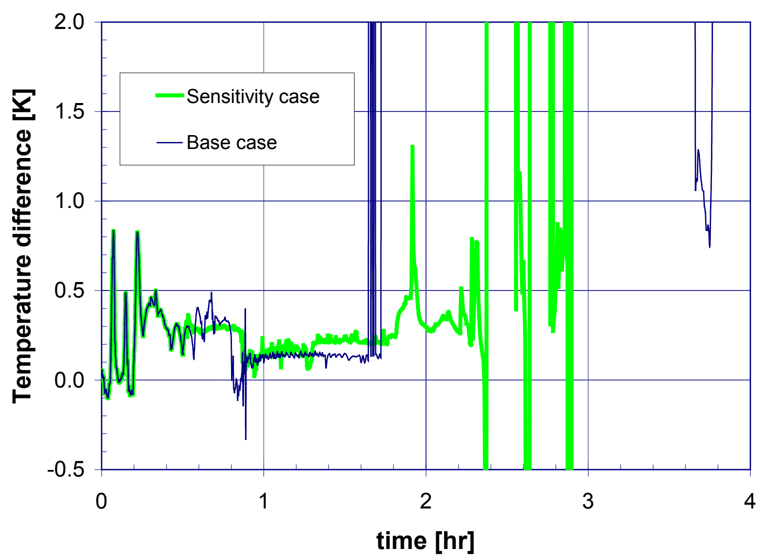

Figure A.2.6.2g Differences Between Primary- and Secondary-Side Liquid Temperatures (Case A.2.6)

Whereas in the base case the steam generator water levels remain roughly constant during the transient (see Figure A.2.1.9 in the base-case write-up), in the sensitivity case falling water levels result in demands for AFW starting at 1.45 hours. The temperature of the injected water is 322 K (i.e., by assumption, the value appropriate to full-power operations). At first, these injections of cooler feedwater help enhance the cooling, but finally (i.e., after about 2.3 hours) they make little difference.

Despite the lower primary pressure, Figure A.2.6.3 shows that the break flow is greater in the sensitivity case. The reason appears to be that lower RCS pressure results in faster pressurizer draining

(Figure A.2.6.4), supporting (during the draining) a larger RHR pumping rate (Figure A.2.6.5). The greater break flow results when most of the water drawn by the RHR pump from the hot leg re-enters the RCS at the Loop A and D cold legs and then leaves through the breaks in the cold legs of Loops B and C, after traversing the downcomer without passing through the core.

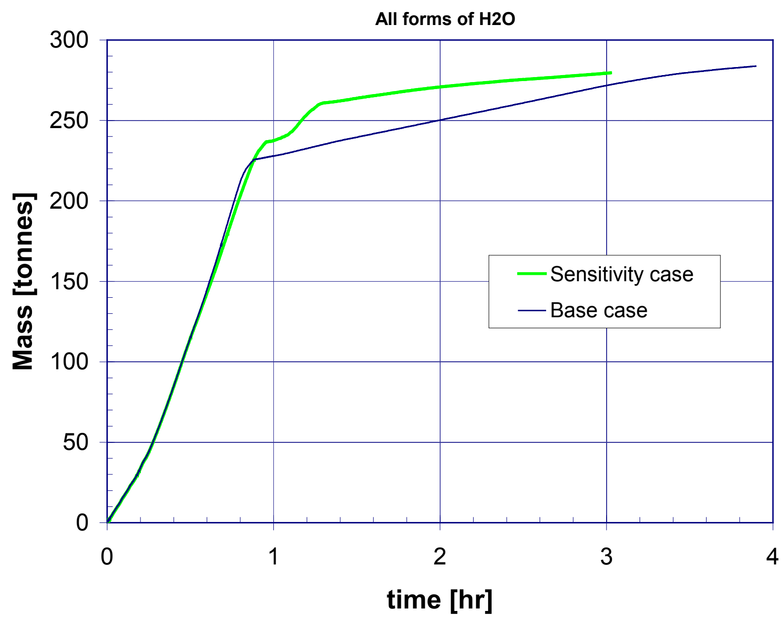

Figure A.2.6.3   Cumulative Break Flow (Case A.2.6)

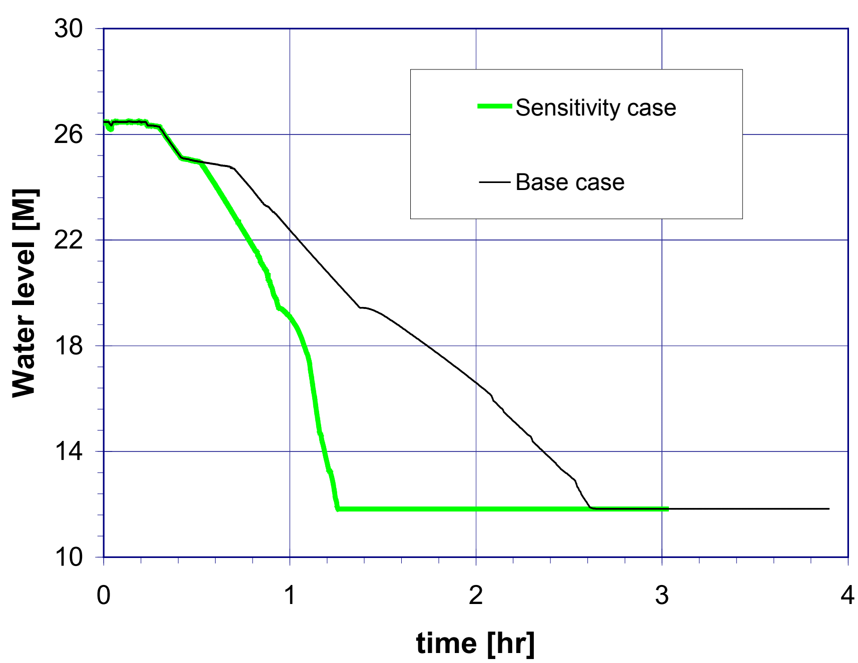

Figure A.2.6.4   Swollen Water Levels in the Pressurizer (Case A.2.6)

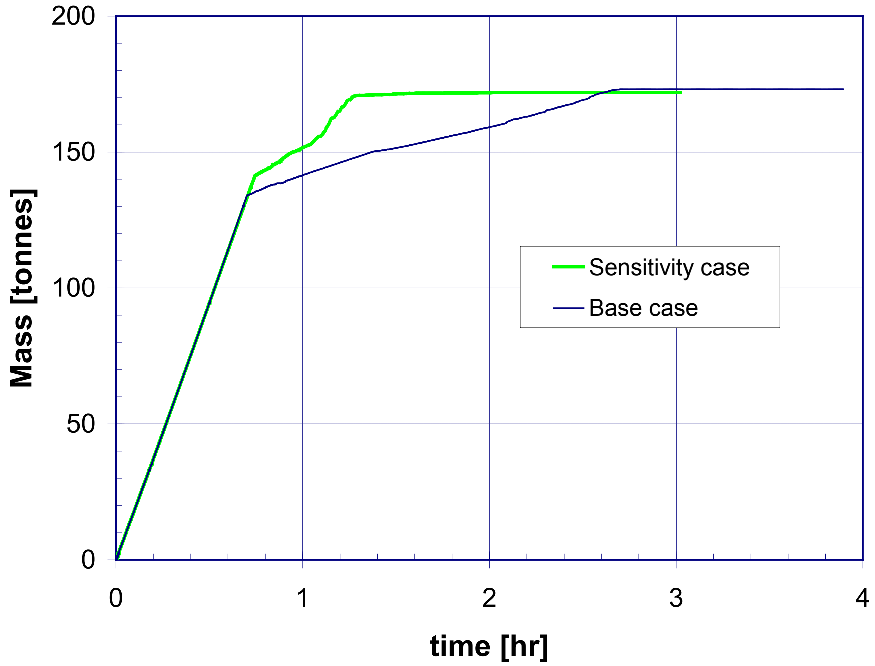

Figure A.2.6.5 Cumulative Flow through the RHR Pump (Case A.2.6)

Figure A.2.6.6 shows water levels in the core. Because of the faster coolant inventory loss in the sensitivity case, the core more quickly attains the amounts of uncovery that enable core damage to be indicated according to the surrogate measures based on temperature and other phenomena. But both cases reach a plateau with approximately the top-third of the core uncovered in about the same time. This plateau persists until the pressurizer empties.

The sensitivity case shows that the assumed operator action does initially enhance the steam generator cooling. Eventually, however, the cooling becomes similar to that of the base case, i.e., less than half the decay heat, and core damage is not avoided. Unexpectedly, effects of the action on the break flow actually hasten the time to core damage, relative to the base case.

The time at which the core exit temperature reaches 922 K is 1.83 hours in the sensitivity case versus 2.93 hours in the base case. The time at which the peak cladding temperature reaches 1,478 K is 2.22 hours in the sensitivity case versus 3.34 hours in the base case. The calculation is stopped at 3.02 hours due to numerical problems associated with the melting of the heat structure that represents the upper core support plate.

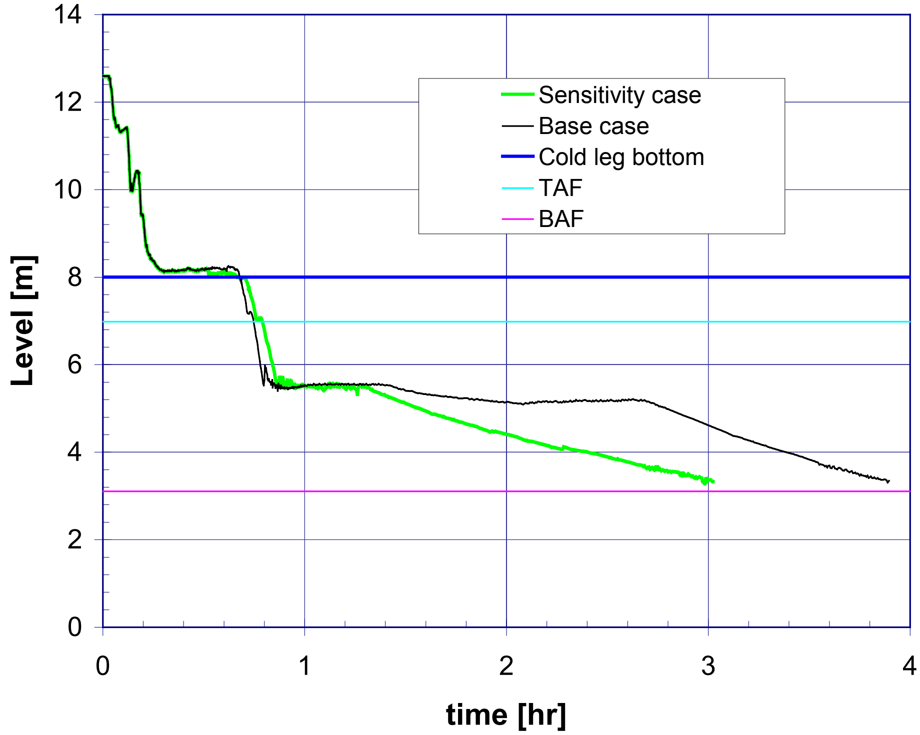

Figure A.2.6.6 Water Levels in the Core (Case A.2.6)

The remainder of this section provides some motivation for the set-up of Case A.2.6. Case A.2.6 assumed a fixed opening fraction of 4.5 percent for the condenser steam dump valve. Note that the dump valve has a feedback control mode that automatically adjusts the opening fraction in an attempt to keep the secondary-side pressure constant. This automatic control is included in the full-power model. However, preliminary calculations for case A.2.6 showed that considerable modifications of the model's feedback control logic (such as revisions of the gain factor between error signal and valve opening, and integration times) would be required in order for the calculation to yield constant pressure in the case of a break scenario at low power. Such updates seem unjustified, since there are no plant data to substantiate them. Moreover, it is not clear that the automatic dump valve control mode would be available during shutdown operations. Therefore, it was decided to assume for the sensitivity calculation a fixed dump valve opening that provides a gradual though uncontrolled secondary-side depressurization.

To determine the appropriate fraction, a scenario resembling a station blackout was simulated. There is no break, but the RHR pump is assumed to stop at the start of the transient. First, the situation of a closed dump valve was considered. Figure A.2.6.7a shows the primary- and secondary-side pressures that result. The secondary-side pressure rise is large, but not enough to open the PORVs or SRVs. Steam generator water levels stay high. Although a steady natural circulation flow around the primary loops occurs, the steam generators carry off only about 60 percent of the decay heat. The primary-side pressure quickly rises until the SRVs begin to cycle.

Figure A.2.6.7b shows similar plots pertaining to a calculation in which the steam dump valve is opened to 4.5 percent at 345 seconds, approximately when the secondary-side pressure first begins to rise in the absence of such an action. With the 4.5 percent fraction, the secondary-side pressure remains fairly constant. In this case, the steam generator cooling amounts to about 95 percent of the decay heat until about 1.3 hours. The temperature at the core exit remains fairly constant after the opening of the dump valve and is only about 10 K higher than its pre-transient value. Because the primary system is water-solid, the primary pressure rise is again large, from 1.35 to about 5 MPa. But in this case, it is not sufficient to open the SRVs. Steam generator water levels fall, and starting from about 1.3 hours AFW injections occur. The injection of the cooler water (322 K) results in cooling in excess of the decay heat, causing the primary pressure to decrease.

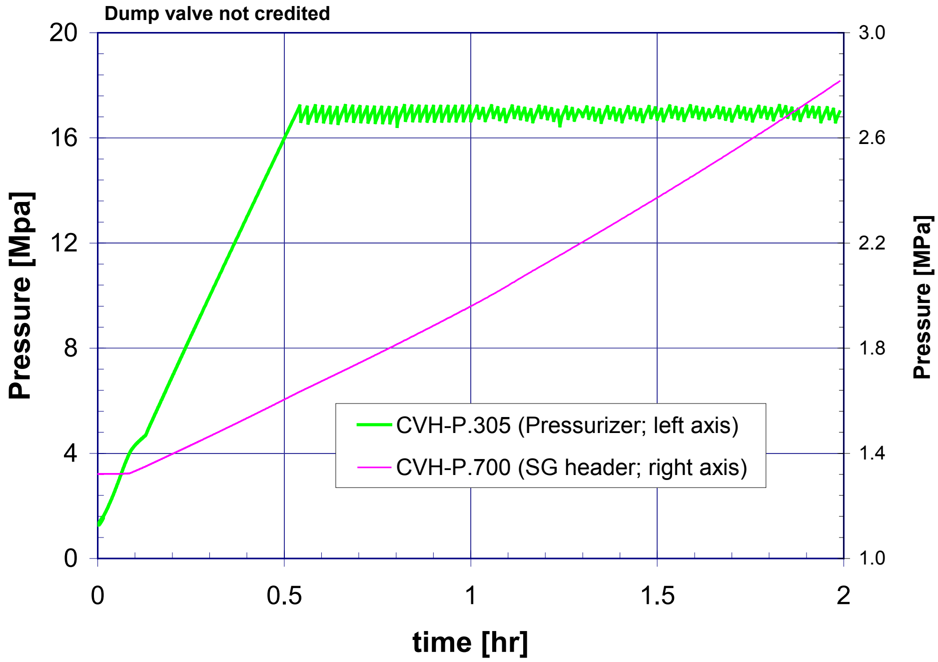

Figure A.2.6.7a Pressures (Supplemental Calculation with RHR Pump Trip and No Break)

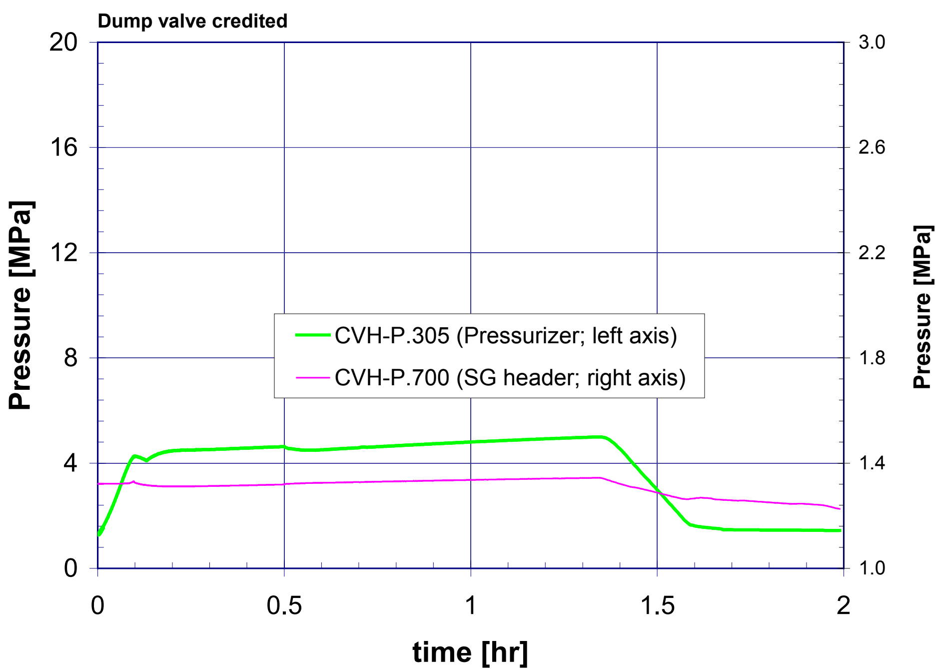

Figure A.2.6.7b Pressures (Supplemental Calculation Variant with RHR Pump Trip and No Break)

For Case A.2.6, the assumed time for opening the dump valve, 0.5 hours, is reasonable for an operator action. This time also is approximately consistent with the time when RHR cooling is lost in the base case (i.e., about 0.7 hours).

A.3.2.7 Results of Cases A.2.7a, A.2.7b, A.2.7c, and A.2.7d (Other Break Flow Areas)

These four sensitivity cases are the same as the base case, except that the flow areas of the break differ from the base-case value.

In Case A.2.7a, the flow area is 75 percent of the value that applies to the base case. The resulting accident sequence is similar to that of the base case, except the events are delayed by 0.1 to 0.5 hours, depending on the time. Figure A.2.7a.1 compares the core water levels in the base and sensitivity cases. In the sensitivity case, the core starts to uncover at 1.01 hours; the time at which the core exit temperature reaches 922 K is 3.27 hours; and the time at which the peak cladding temperature reaches 1,478 K is 3.71 hours. In the base case, these same events occur at 0.75, 2.93, and 3.34 hours. The calculation is stopped at 4.42 hours due to numerical problems associated with the melting of the heat structure that represents the upper core support plate.

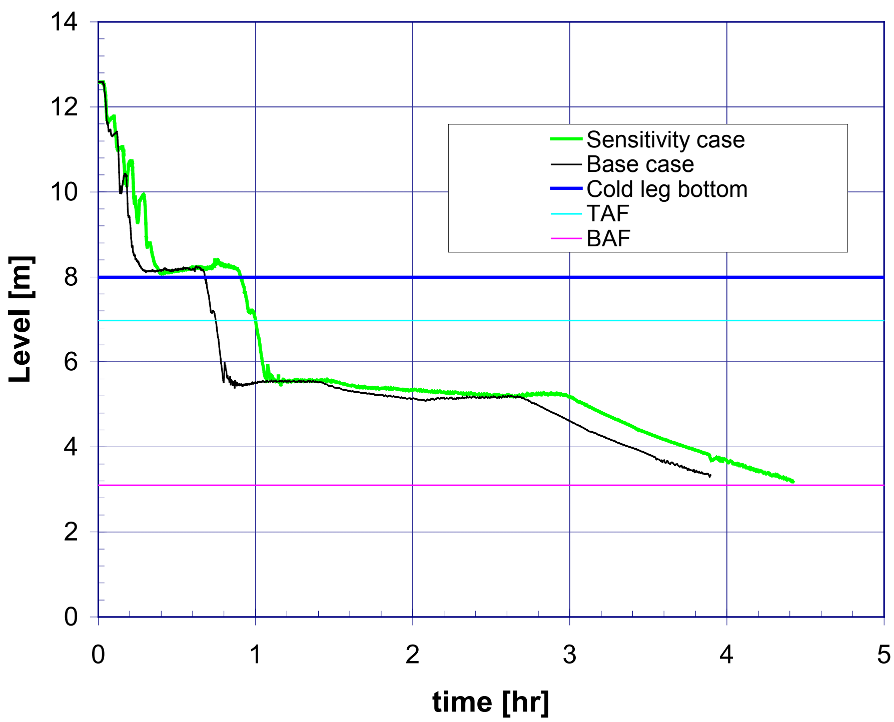

Figure A.2.7a.1 Water Levels in the Core (Case A.2.7a)

In Case A.2.7b, the flow area is 10 percent of the value that applies to the base case, for a nominal initial break flow rate of 100 gpm (0.0063 m^3/s). Table A.9 provides timings of some of the key events. Figure A.2.7b.1 shows the break flows. During the first 2 hours, the break flow rate is close to 100 gpm (0.0063 m^3/s), but it is as much as 185 gpm (0.0117 m^3/s) later. Water levels are shown in Figure A.2.7b.2. The RHR pump (Figure A.2.7b.3) has full suction until about 7.3 hours, when the pressurizer empties, but pumping at reduced rates continues until about 9.3 hours. Uncovery of the core starts at 7.87 hours in the second computational ring. The time at which the peak cladding temperature reaches 1,478 K is 11.38 hours. The calculation is stopped at 13.9 hours.

Table A.9 Timing of Key Events Predicted for the MELCOR Model Shutdown Sensitivity-Case A.2.7b Calculation

Event Description	Time (hours)
Break in RHR-B lines outside the containment[(1)]	0
Reduced RHR-A pump rate due to low water level in hot leg D	~7.3
Core partially uncovered	7.87
Pressurizer empties (i.e., contains less than 10 kg of liquid)	7.26
Zero RHR pump rate due to dry hot leg D	9.32
Coolant temperature at core exit reaches 922 K (1,200 °F)	9.62
First gap release	10.17
Peak clad temperature reaches 1,478 K (2,200 °F)	11.38
Calculation ends	13.89

[1] Nominal break flow, 100 gpm (0.0063 m^3/s) (vs. 1,000 gpm (0.063 m^3/s) in the base case).

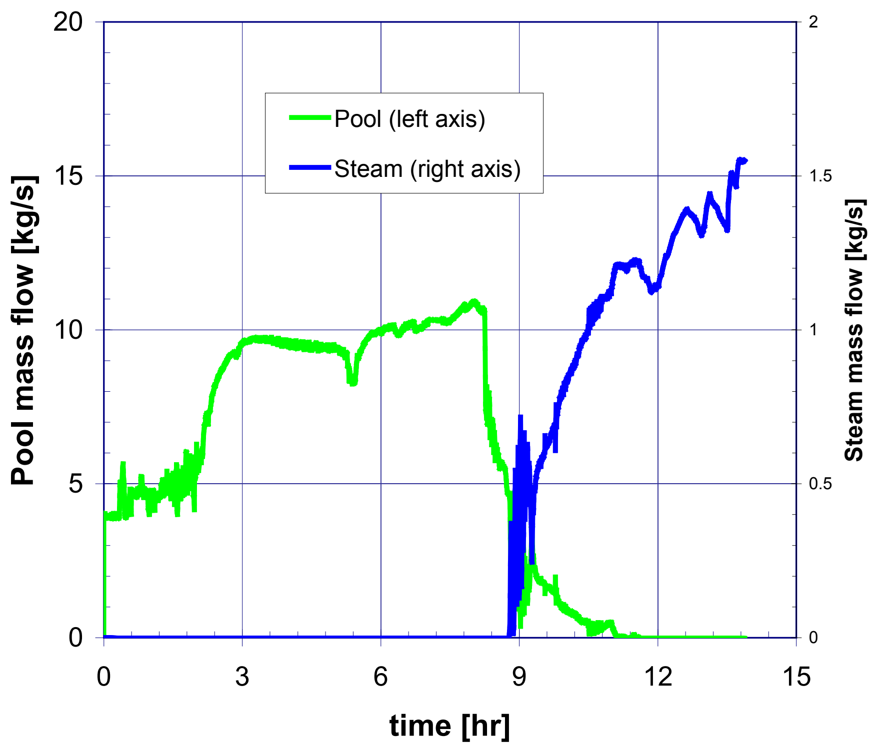

Figure A.2.7b.1 Break Flows (Case A.2.7b)

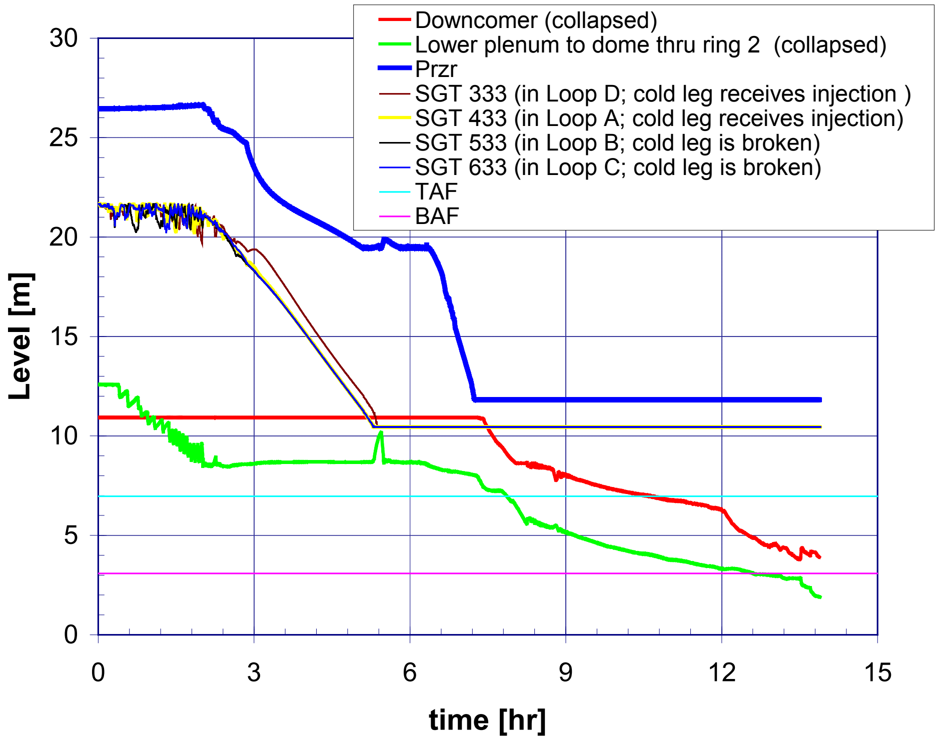

Figure A.2.7b.2 Water Levels in the RCS (Case A.2.7b)

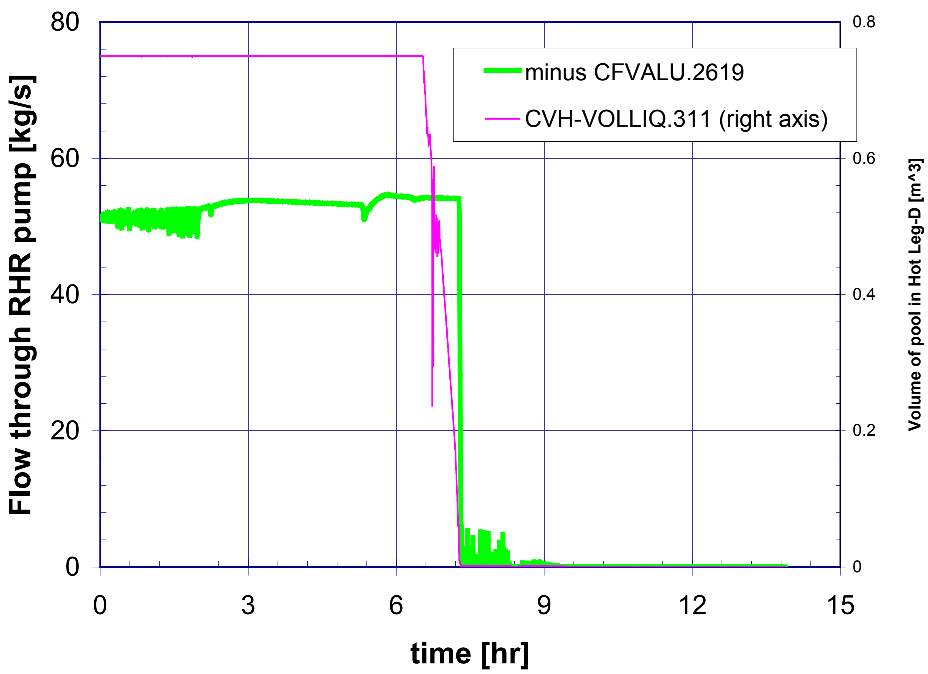

Figure A.2.7b.3 RHR Pump Rate and Liquid Volume in Hot Leg D (Case A.2.7b)

In Case A.2.7c, the flow area is 5 percent of the value that applies to the base case for a nominal initial break flow rate of 50 gpm (0.0032 m^3/s). During the first 4 hours, the break flow rate increases from about 35 to 60 gpm (0.0022 to 0.0038 m^3/s), but the peak value (occurring at about 17 hours) is about 97 gpm (0.0061 m^3/s). The pressurizer empties at about 14.6 hours. The RHR pump has full suction until about 15.5 hours, but pumping at reduced rates continues until about 18.9 hours. Uncovery of the core starts at 16.81 hours in the second computational ring. The time at which the peak cladding temperature reaches 1,478 K is 23.92 hours. The calculation is stopped at 24.8 hours.

In Case A.2.7d, the flow area is 500 percent of the value that applies to the base case for a nominal initial break flow rate of 5,000 gpm (0.315 m^3/s). Table A.10 provides timings of some key events. Figure A.2.7d.1 shows the break flows. From 5 to 10 minutes, the break flow rate is about 5,250 gpm (0.331 m^3/s). Figure A.2.7d.2 shows the water levels. The RHR pump (Figure A.2.7d.3) operates at a reduced rate from about 11 to 15 minutes. It then has a second period of full suction before finally stopping altogether at 21.5 minutes. The pressurizer empties at 20.2 minutes. Uncovery of the core starts at 11.5 minutes in the second computational ring. The time at which the peak cladding temperature reaches 1,478 K is 39.8 minutes. The calculation is stopped at 1.5 hours.

Table A.10 Timing of Key Events Predicted for the MELCOR Model Shutdown Sensitivity-Case A.2.7d Calculation

Event Description	**Time (minutes)**
Break in RHR-B lines outside the containment(1)	0
Reduced RHR-A pump rate due to low water level in hot leg D	~11 – 15
Core partially uncovered	11.5
Pressurizer empties (i.e., contains less than 10 kg of liquid)	20.2
Zero RHR pump rate due to dry hot leg D	21.5
Coolant temperature at core exit reaches 922 K (1,200 °F)	35.1
First gap release	35.2
Peak clad temperature reaches 1,478 K (2,200 °F)	39.8
Calculation ends	91

1 Nominal break flow, 5,000 gpm (0.315 m^3/s) (vs. 1,000 gpm (0.063 m^3/s) in the base case).

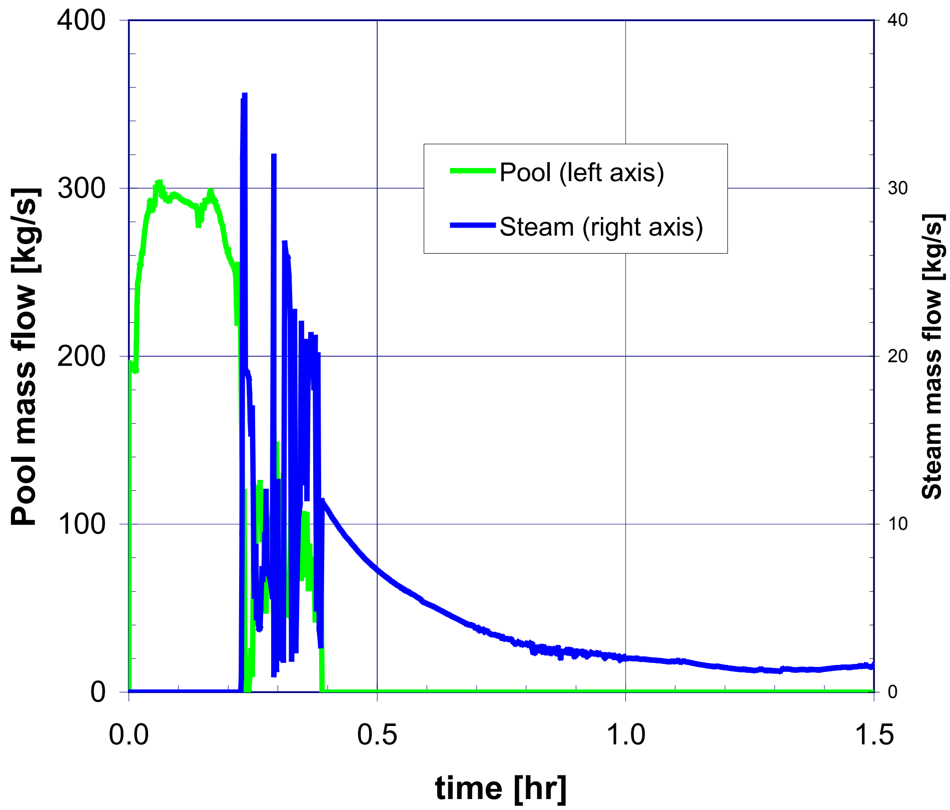

Figure A.2.7d.1 Break Flows (Case A.2.7d)

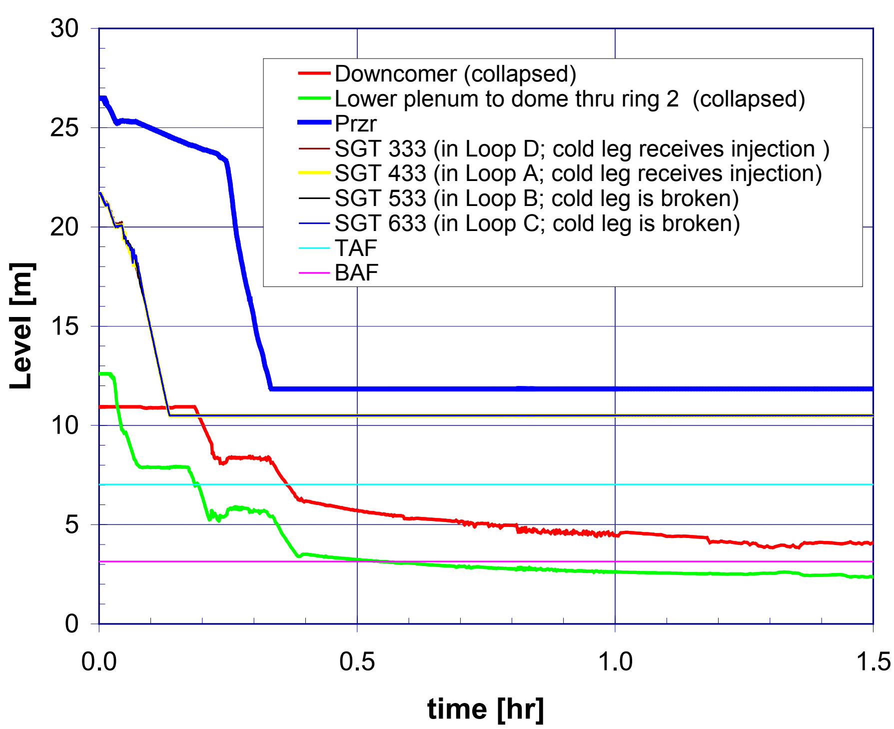

Figure A.2.7d.2 Water Levels in the RCS (Case A.2.7d)

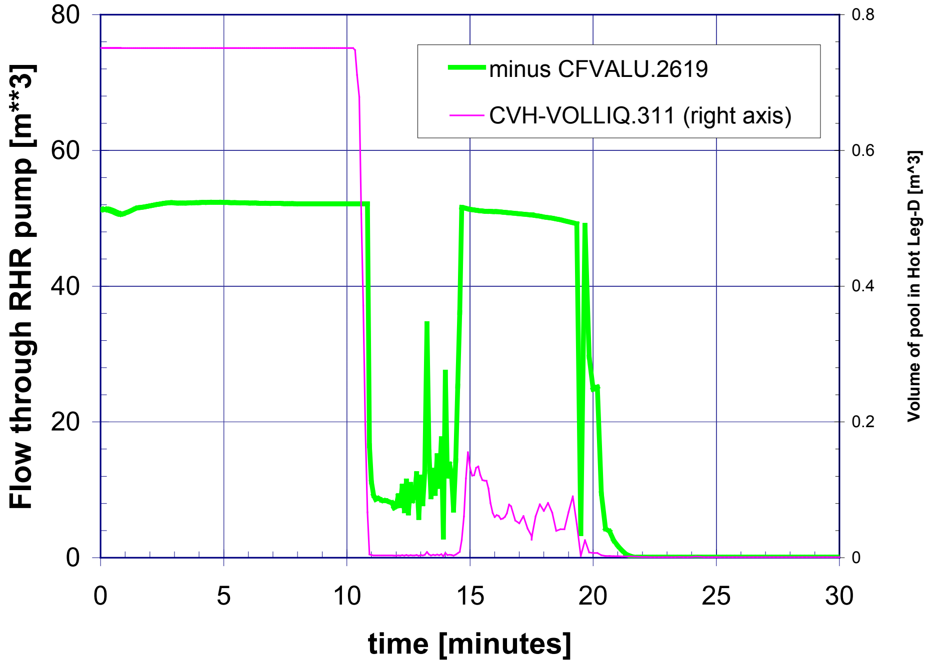

Figure A.2.7d.3 RHR Pump Rate and Liquid Volume in Hot Leg D (Case A.2.7d)

A.3.2.8 Results of Case A.2.8 (Reduced Decay Heat)

Case A.2.8 is the same as the base case, except that the decay heat is uniformly reduced to 75 percent of its value in the base case. The initial decay heat is therefore 15.97 MW. This case receives a special pre-transient calculation with a clamped, appropriately reduced rate of RHR pumping relative to the base case. Note that the initial decay heat is quite similar to the decay heat at 72 hours, which is estimated as 15.85 MW. Case A.2.8 therefore serves as an approximation to a modified scenario in which the transient begins at 72 hours. However, the present treatment with a uniform reduction of the decay heat by a constant factor is not exactly equivalent to evaluating the unmodified decay heat curve at any later times, and the assumed initial conditions (e.g., RPV head still in place) would likely not actually be applicable at 72 hours.

The resulting accident sequence is similar to that of the base case, except for small delays of the various events. Figure A.2.8.1 compares the core water levels. In the sensitivity case, the core starts to uncover at 0.80 hours; the time at which the core exit temperature reaches 922 K is 2.98 hours; and the time at which the peak cladding temperature reaches 1,478 K is 3.46 hours. In the base case, these same events occur at 0.75, 2.93, and 3.34 hours, respectively. The calculation was stopped at 4.37 hours due to numerical problems associated with the melting of the heat structure that represents the upper core support plate.

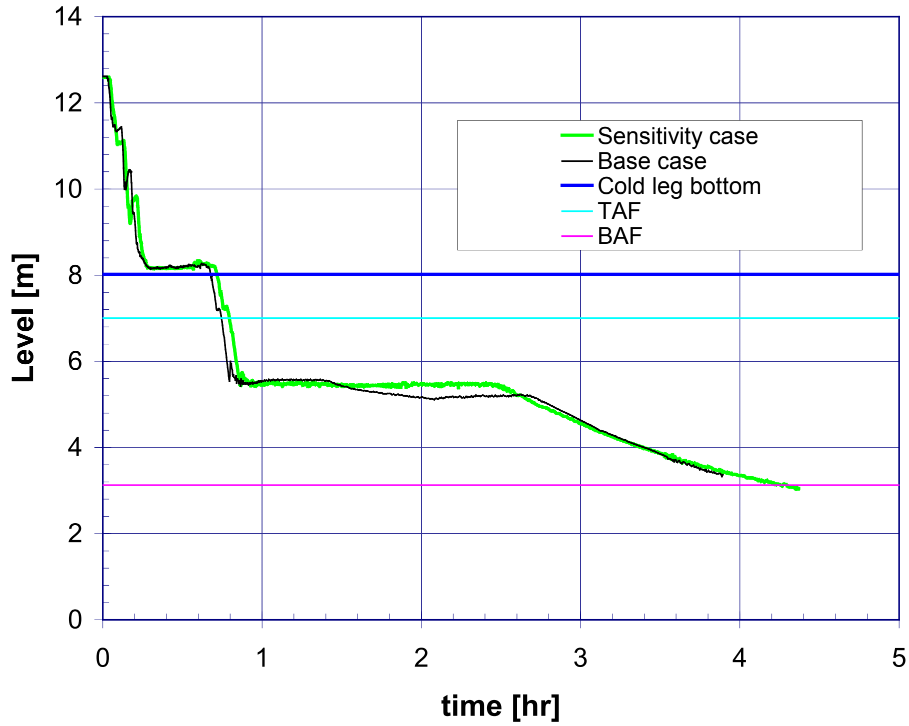

Figure A.2.8.1 Water Levels in the Core (Case A.2.8)

A.3.2.9 Results of Case A.2.9 (Break inside the Containment with the ECCS Recirculation Mode)

This case considers that the break of the line running from RHR pump B to the cold legs of loops B and C occurs inside the containment instead of outside. As in Case A.2.4, the unaffected RHR pump A is supposed to trip irreversibly on the low hot leg water level, which should occur at 0.71 hours in the present case. As in Cases A.2.2 and A.2.3, the ECCS injections from the RWST with one charging and one SI pump are credited. They begin at 0.92 hours. As modeled, the injections result from the ECCS actuation on high containment pressure; in actuality it may be due to operator action if the aforementioned actuation signal is disabled in this mode of plant operation. In contrast with the other cases, this case assumes that the ECCS is successfully switched over to the recirculation mode, where the containment sump becomes the water source, when the RWST water level is drawn down to Lo-2. This event occurs at 3.37 hours. In the assumed configuration, RHR pump A draws from the sump during the recirculation mode. Its discharge passes one RHR heat exchanger, then is fed into one charging and one SI pump, which finally inject into cold legs A and D (i.e., the two intact cold legs).

Many results are similar to those of Case A.2.2. Figure A.2.9.1 shows the water levels around the RCS. Figure A.2.9.2 shows the ECCS injections. By about 1 hour, a stable state is achieved and the core water level is steady, with no more than about the top third of the fuel uncovered. The top of the active fuel uncovers for the first time at 0.76 hours, and the top third uncovers for the first time at 0.81 hours. No other surrogate measures ever indicate core damage. The calculation is stopped at 7.4 hours.

In this case, because the break is inside the containment, containment pressurization sufficient to reach the ECCS actuation setpoint occurs. All Shutdown calculations credit one fan cooler, but only in this case does it play a significant role. ECCS actuation switches the cooler from normal mode to ECCS mode, and further containment pressurization is prevented. Figure A.2.9.3 shows the containment pressure. Figure A.2.9.4 shows the rate of heat extraction provided by the fan cooler, in comparison with the decay heat.

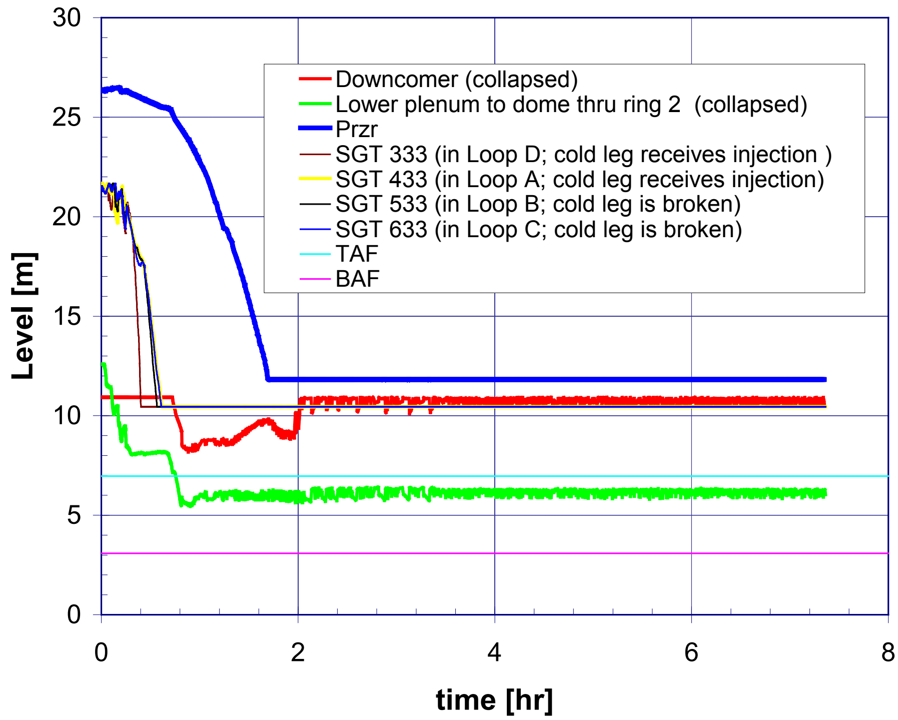

Figure A.2.9.1 Water Levels in the RCS (Case A.2.9)

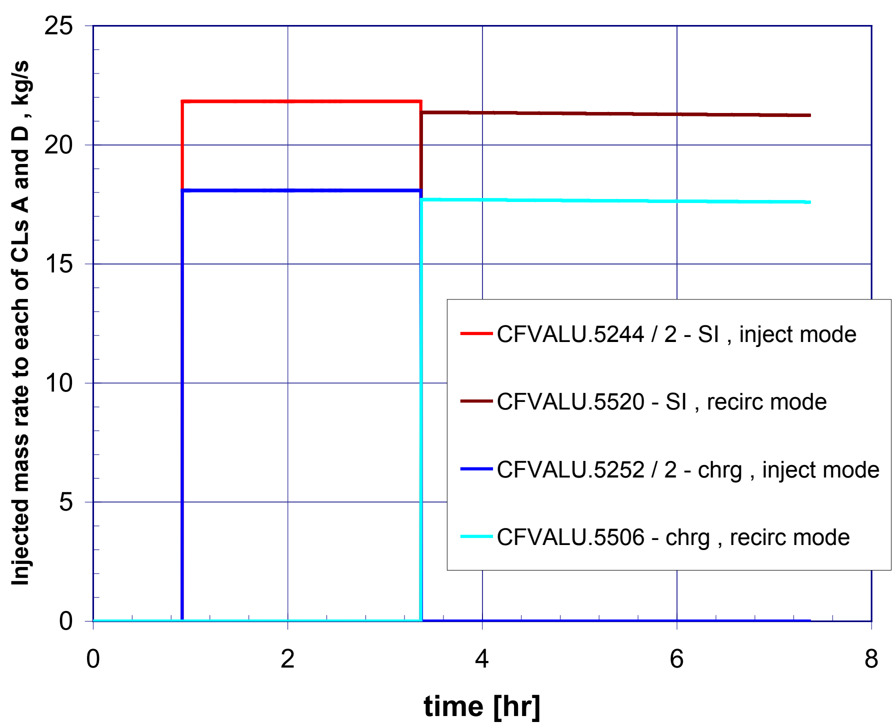

Figure A.2.9.2 ECCS Injections Flow Rates (Case A.2.9)

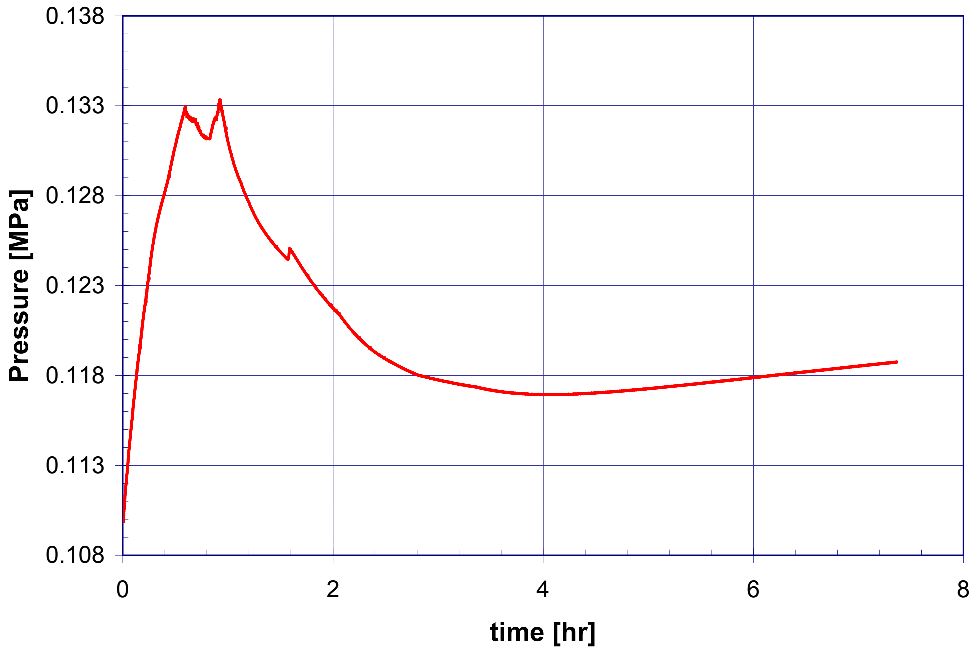

Figure A.2.9.3 Containment Pressure (Case A.2.9)

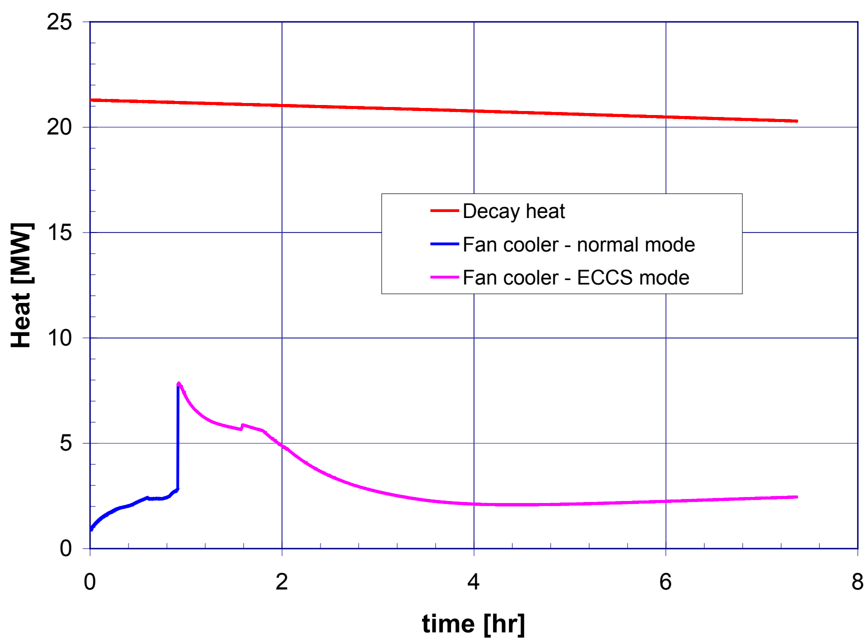

Figure A.2.9.4 Decay Heat and Fan Cooler Heat Removal (Case A.2.9)

A.3.2.10 Results of Case A.2.10 (Break from a Hot Leg)

This case considers that the break occurs in the suction line of the unused RHR pump (instead of that pump's injection line, as the base case supposes). The break therefore connects the loop C hot leg (modeled by control volume CV 611) with the environment, instead of connecting the cold legs of loops B and C with the environment, as occurs in the base case.

Figure A.2.10.1 shows the break flow. The flow area of the break flow path, 18.8 cm^2, is assumed to be twice the area of the two base-case cold leg break flow paths, so that any different behavior of the sensitivity-case break flow is not merely a matter of different flow areas. Recall that the area was prescribed in order to obtain a net break flow of around 1,000 gpm (0.063 m^3/s) early in the transient in the base case. Figure A.2.10.2 compares the sensitivity and base cases with respect to the primary pressure. Whereas in the base case the RHR cooling is impaired from the start of the transient, in the sensitivity case the efficacy of RHR cooling is barely or not at all reduced until the actual loss of suction at the (intact) hot leg. In the base case, even though the suction was not lost until later, starting from t=0 some of the returning cooled water spilled through the breaks after traversing the downcomer without passing the core. In the base case, the impaired cooling during the first hour caused the pressure to rise and therefore increased the break flow. This impairment and pressure rise do not occur in the sensitivity case, and the break flow stays around 650 gpm (0.041 m^3/s) during the first hour.

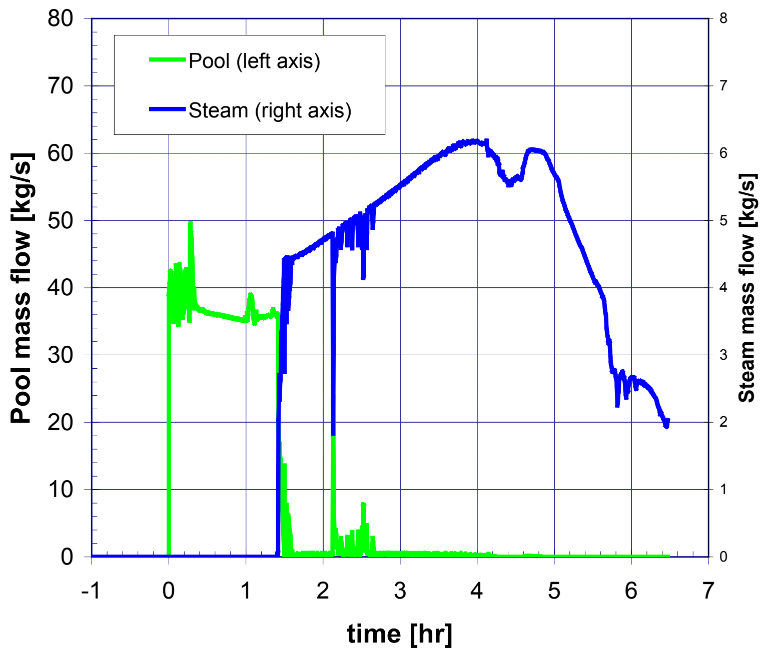

Figure A.2.10.1 Break Flows (Case A.2.10)

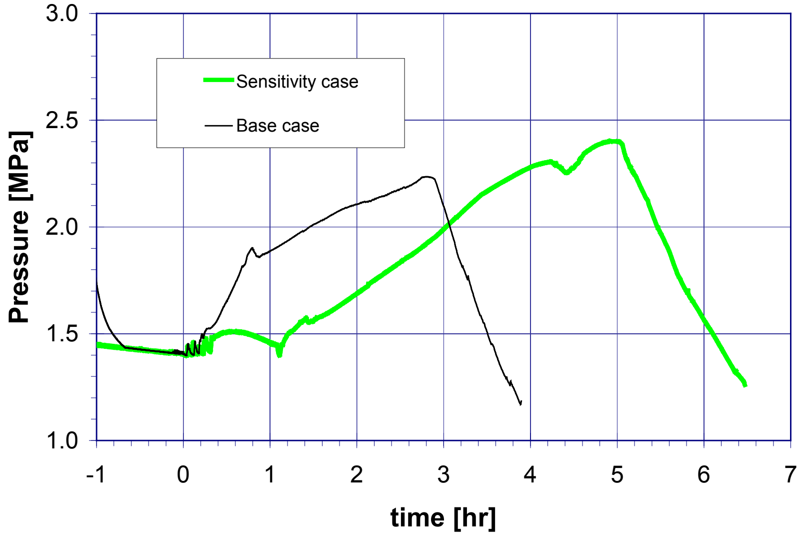

Figure A.2.10.2 Vessel Dome Pressures (Case A.2.10)

Figure A.2.10.3 shows the water levels at various locations. The water levels in the loop D hot leg (which the figure shows, and from which the RHR pump takes suction), and the loop C hot leg (with the break, and not shown in the figure) are similar at all times. A void first occurs there around 1.4 hours. After this time, the break flow is predominately steam (Figure A.2.10.1). Also after 1.4 hours, the RHR pump operates only at a reduced rate, which is equal on average to the rate at which water reappears in hot leg D due to pressurizer draining (Figure A.2.10.4). The slopes of the curves in Figure A.2.10.4 indicate that from about 1.4 hours until 2.65 hours (when the pressurizer empties), water leaves the pressurizer, is pumped to the vessel, boils in the core, and finally leaves the RCS through the break as steam, all at approximately equal mass rates. Draining the pressurizer is very similar to the base-case behavior. In the base case, the pressurizer empties at 2.64 hours.

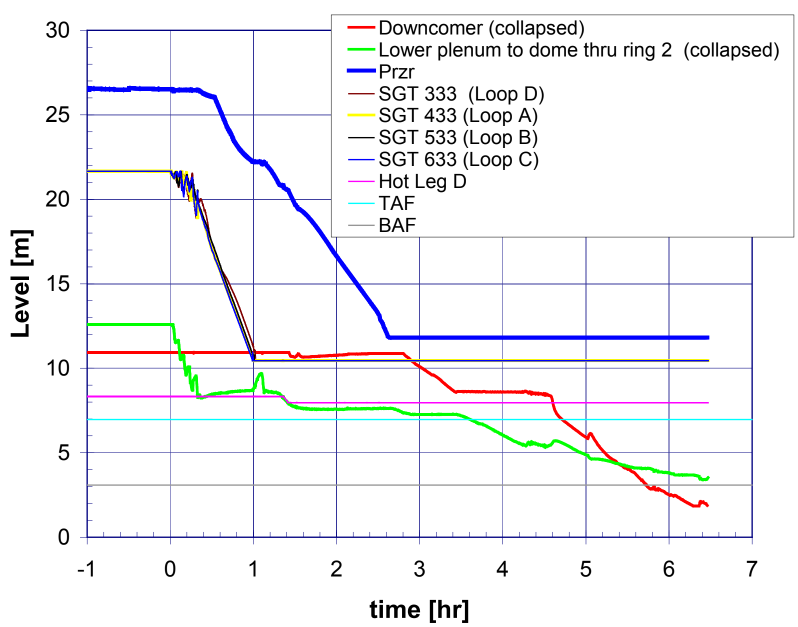

Figure A.2.10.3 Water Levels in the RCS (Case A.2.10)

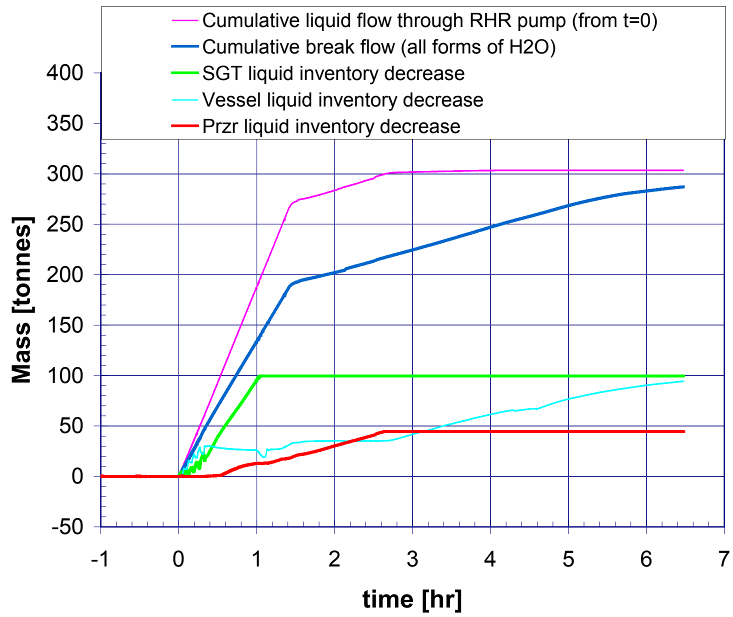

Figure A.2.10.4 Cumulative flows and Inventory Decreases (Case A.2.10)

Uncovery of the core starts at 3.62 hours; significant heatup of the core begins at about 5 hours. The time to core damage, according to the measure that associates core damage with the spatially-maximized cladding temperature becoming greater than 1,478 K, is 5.59 hours in the case of the hot leg break versus 3.34 hours in the base case.

A.3.2.11 Results of Case A.2.11 (Initially Sub-cooled Primary RCS)

This case is the same as the base case, except for the initial primary-side pressure. Actual shutdown operations would likely have water-solid conditions in the RCS, with pressure well above the saturation pressure. Because of anticipated possible numerical difficulties related to water incompressibility, the base-case calculation was set up with saturated conditions instead of sub-cooled conditions. In this sensitivity case, a solid and sub-cooled state during the pre-transient period is generated by adding above the pressurizer a fictitious water pool under a cover of nitrogen gas with an initial pressure of 2.57 MPa. As in the base case, the pre-transient calculation begins at -5,000 seconds. During most of the pre-transient, the nitrogen buffers the RCS pressure against large fluctuations that would otherwise occur, until the thermal interactions between the primary and secondary sides of the RCS become steady. Figures A.2a and A.2b describe these interactions in the base case. The fictitious pool is valved out of the problem at -50 second. At the start of the transient (i.e., t = 0 s when the break opens), the pressure at the top of the completely full pressurizer is 2.54 MPa (354 psig). In the base case, the initial pressurizer pressure is 1.29 MPa, the saturation value. The base-case primary-side temperature of 466.5 K (380 °F) is assumed to also apply to this sensitivity case. The selected temperature is consistent with Step 21 of 1BOA-SD-2 (a Byron document related to shutdown), whereas typical primary-side conditions at this point in a shutdown operation would be 360 psig (2.48 MPa-g) and 350 °F (450 K). The initial conditions on the secondary side of the RCS are the same for this sensitivity calculation as they are in the base case. In particular, the temperature is also 466.5 K and the pressure is the saturation value.

Figures A.2.11.1 and A.2.11.2 compare the primary pressure and the break flows in the sensitivity case versus the base case. Once the transient is initiated by the opening of the break, the pressure quickly falls to saturation in the sensitivity case. This results in similar primary pressures and similar break flows during most of the transient in the two cases. All results are therefore similar. The time to core damage according to the measure that associates core damage with the spatially-maximized cladding temperature becoming greater than 1,478 K is 3.47 hours in case the RCS is initially water-solid and sub-cooled, versus 3.34 hours in the base case.

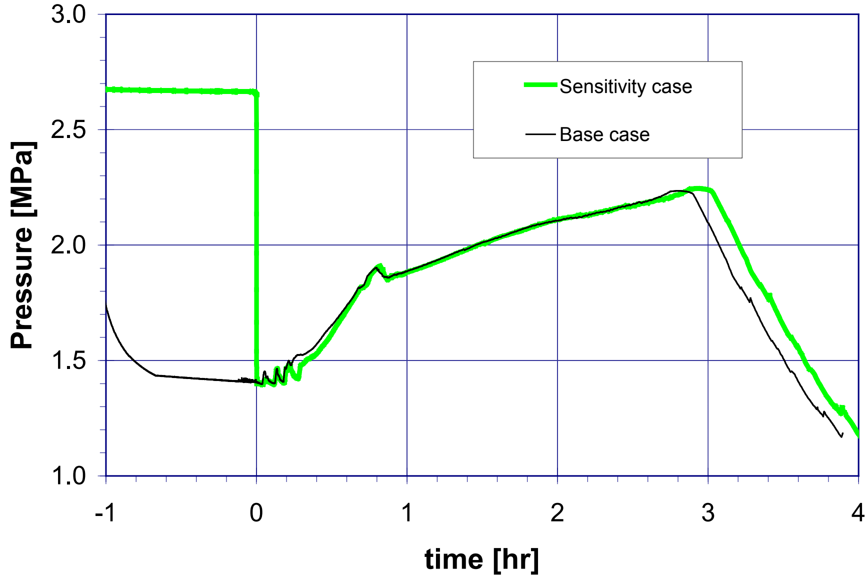

Figure A.2.11.1 Vessel Dome Pressures (Case A.2.11)

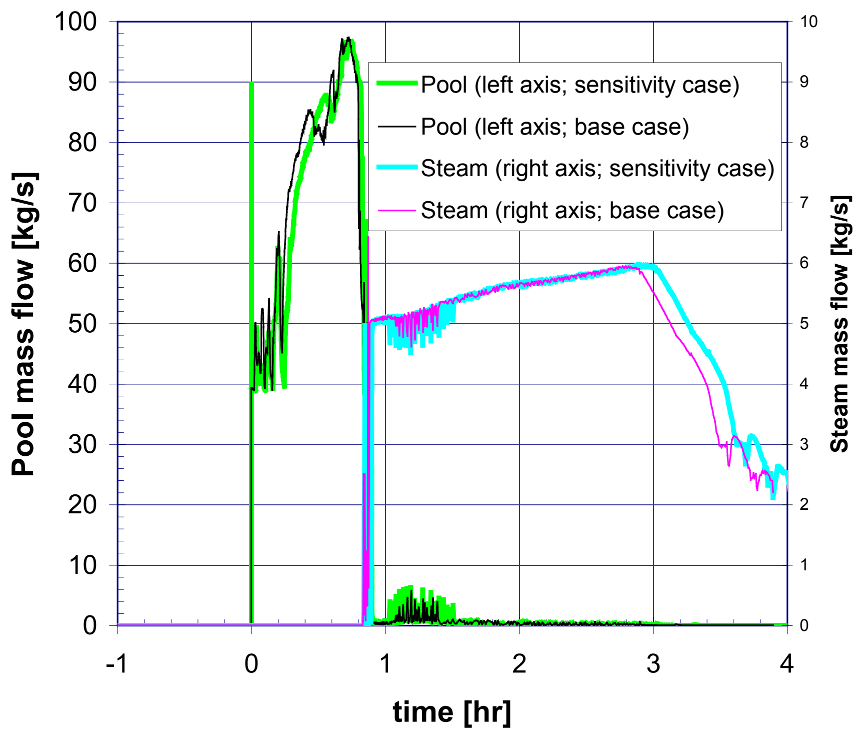

Figure A.2.11.2 Break Flows (Case A.2.11)

A.3.3 <u>Summary: Shutdown Scenario</u>

This scenario considers a break outside of containment in piping that connects an RHR pump to two cold legs. Initial temperatures around the RCS are 466.5 K. The RHR pump unaffected by the break is in use in a closed-loop configuration to remove the decay heat, which is 21.29 MW initially and corresponds to

24 hours since scram. The break flow area produces an initial break flow of about 1,000 gpm (0.063 m^3/s).

No operator actions are credited in the base case, thus leading to core damage when the RHR pump loses suction as inventory is lost and the hot leg empties. Core damage occurs at 3.34 hours according to the measure that associates core damage with the temperature of the hottest cladding exceeding 1,478 K (2,200 °F).

Variants of the base case that also lead to core damage consider the following:

- A modified treatment of the behavior of the RHR pump on very low hot leg water levels (Case A.2.4 predicts core damage 0.13 hours earlier with the assumption that the pump trips irreversibly)

- An attempt to bring about more steam generator cooling (Case A.2.6 initiates a slow secondary-side depressurization starting at 0.5 hours and predicts core damage 1.12 hours earlier)

- Break flows of different sizes (Case A.2.7a reduces the break flow area to 75 percent of the base-case value and predicts core damage 0.37 hours later. Case A.3.7b reduces the break flow area to 10 percent of the base-case value and predicts core damage at 11.38 hours. Case A.3.7c reduces the break flow area to 5 percent of the base-case value and predicts core damage at 23.92 hours. Case A.3.7d increases the break flow area to 500 percent of the base-case value and predicts core damage at 39.8 minutes.)

- Lower decay heat (Case A.2.8 reduces the decay heat to 75 percent of the base-case value and predicts core damage 0.12 hours later)

- Different break location (Case A.2.10 considers a hot leg break and predicts core damage 2.25 hours later)

- Initially sub-cooled RCS (Case A.2.11 starts the transient from water-solid and super-pressurized primary-side conditions, and predicts core damage 0.13 hours later).

In the above bullets, the times to core damage refer to the measure that associates core damage with the temperature of the hottest cladding exceeding 1,478 K (2,200 °F).

Gross core damage is avoided in Case A.2.2, wherein, at 0.5 hours, operators initiate ECCS injections with one charging pump and one SI pump taking suction from an RWST that is assumed to be refilled as necessary. Core uncovery is not prevented. The time that the top third of the core is first uncovered is 2.14 hours. However, gap releases and the relocation of fuel or structural material are not predicted to occur.

Variations of Case A.2.2 that also avoid gross core damage consider the following:

- A later time for the same actions (Case A.2.3 credits the actions to occur at 1.0 hour)

- A modified treatment of the behavior of the RHR pump on very low hot leg water levels (Case A.2.5 applies to Case A.2.2 the same treatment considered in Case A.2.4),

- Available switchover to the ECCS sump recirculation mode (Case A.2.9 reinterprets the break as occurring inside the containment, so water accumulates in the sump).

A.4 Reference

A.1 H. Esmaili, D. Helton, D. Marksberry, R. Sherry, P. Appignani, D. Dube, M. Tobin, R. Buell, T. Koonce, and J. Schroeder, "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models – Surry and Peach Bottom," U. S. Nuclear Regulatory Commission, NUREG–1953 (September 2011).

APPENDIX B: DETAILS OF THE SURRY SGTR ANALYSIS

B.1 Introduction

B.1.1 Background

Section 2.2 of the main report describes the Surry plant and the MELCOR model. A more complete description of the model is provided by the SNL Surry model documentation. This appendix includes details of the models and calculations for the Surry nuclear power plant.

B.1.2 Objectives and Scope

The objectives of this Appendix are to document the calculations specific to the Surry plant that encompass detailed results for the basic scenario and its sensitivity cases. These simulations are limited in scope to considerations important for determining the relevant probabilistic risk assessment (PRA) level-1 system success criteria and sequence timing.

This Appendix analyzes an SGTR sequence sometimes herein referred to as SGTR-9. The emphasis is on the variability of the results that may arise from variations of the model, variations of the plant's initial state, or variations of assumed operator actions that still fall within the SGTR-9 definition. To this end, the discussion defines a base case and 20 sensitivity cases. Of these 21 cases, SNL defined and computed 20 cases. ERI defined the final case and carried out the MELCOR computation using a suitably modified version of the basic model. The results of interest are mostly confined to the time of core damage. Various criteria (referred to as core damage surrogates) are used to define these times.

B.1.3 Outline

Section B.2 focuses on results that describe one basic scenario consisting of the SGTR-9. The scenario and its sensitivity cases are defined in Section B.2.1. Detailed results specific to the base and sensitivity cases appear in subsections of Section B.2.2. Summary results pertaining to all cases appear in Section B.2.3 and also in Section 3.2 of the main report.

All detailed results appear in the subsections of Section B.2.2; each subsection pertains to a particular base or sensitivity case. To simplify the Figure numbering, a four-character code is used within these subsections according to the numbering scheme explained in the previous Appendix. After "B," which designates Surry, the next character numbers the scenario (for Surry, there is only one scenario). The character after that numbers the sensitivity case; the base case is counted as the first case. The last character in the string is the count among the Figures that apply to a given sensitivity case in a given scenario. For example, the first figure in the second sensitivity case that pertains to the SGTR scenario is Figure B.1.2.1. Figures not pertaining to a specific scenario and sensitivity case are not numerous, and are designated by a two-character code. The first character denotes Surry and the second character reflects a count over this Appendix. A similar two-character code is used for all the Tables.

B.2 Accident Scenario and Results

B.2.1 Steam Generator Tube Rupture Sequence SGTR-9

Figure B.1 is an event tree that indicates 24 possible sequences that may result from an SGTR accident. SGTR-9 ends in core damage following a failure to initiate a primary- and secondary-side cooldown; to refill the RWST; and to provide long-term heat removal.

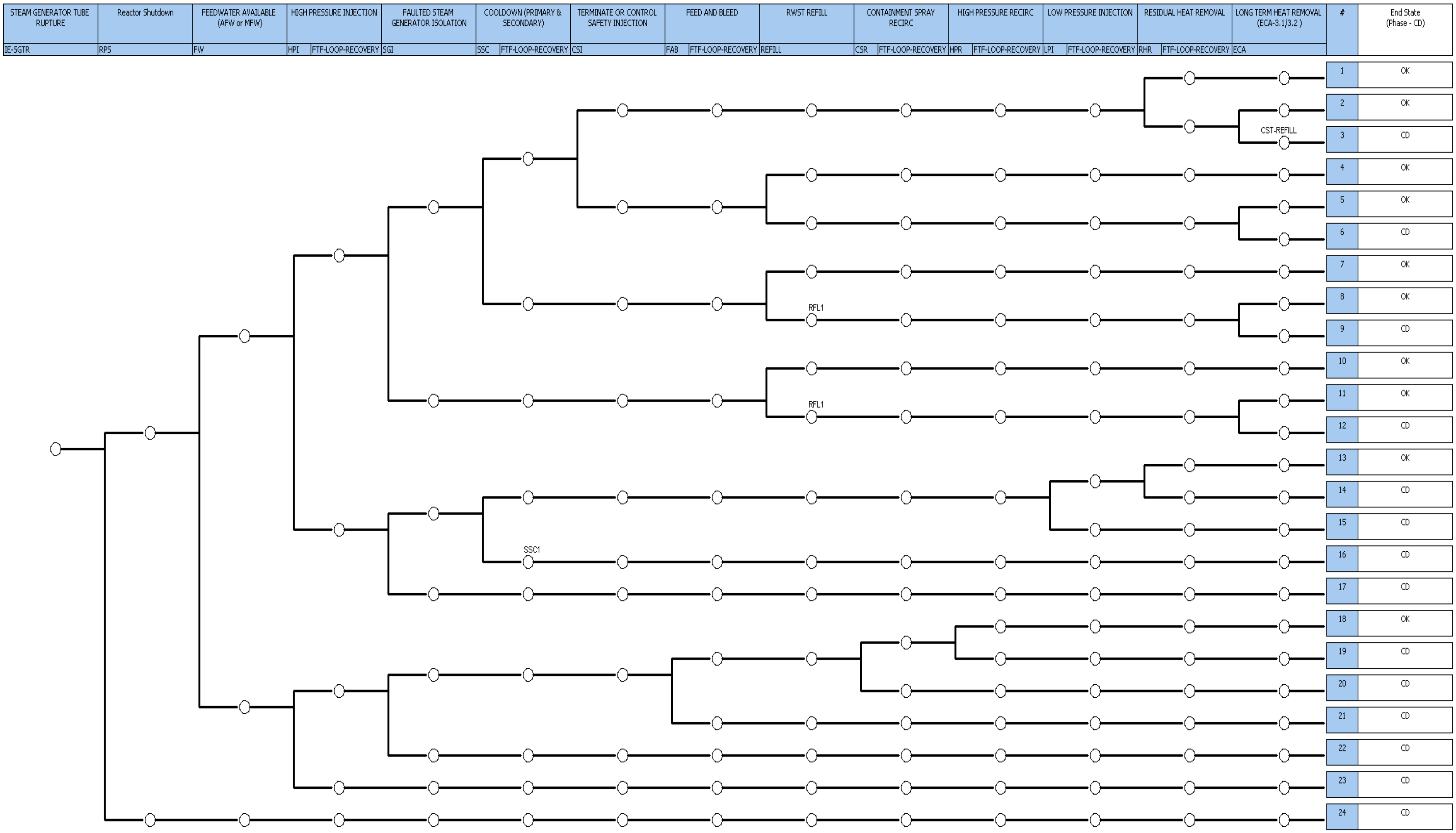

Figure B.1 Surry SPAR (v8.17) SGTR Event Tree

Table B.1 gives the timings at which certain operator actions may be assumed to occur.

In the MELCOR model, the events in Figure B.1 and Table B.1 are taken into account by a setup that includes:

- An initiating steam generator tube rupture at time $t = 0$
- Scram, MSIV closure, MFW trips, and RCP trips at finite model-predicted times[24]
- Available HHSI, LHSI, and containment sprays with the RWST as the source
- Available turbine-driven and motor-driven AFW with CST as the source
- Operator actions E-3 (from Table B.1) occurring at a hard-wired time (17.88 minutes)
- No provision for steam generator cooldown (e.g., by controlled depressurization)
- Unavailable recirculation mode for all systems
- No refilling of the RWST or CST

The base-case calculation also assumes no termination or control of HHSI by operators, no B&F, and no RHR, although Table B.1 identifies interim SOARCA-recommended timings for these actions. Some of these actions are considered by the sensitivity cases, however. Note that the time of the operator actions, 17.88 minutes, was hard-wired into the deck by the model developers. The rationale for this precise time is unknown, but it does not differ importantly from the SOARCA-recommended time of 15 minutes.

To investigate the variability of the predictions, 20 sensitivity cases are defined in addition to the base case. The highlighted entries in Table B.2 show how each sensitivity case differs from the base case.

- Case B.1 is the base case.

- Cases B.2 and B.3 consider the effective number of tubes that rupture.

- Case B.4 considers the effect of a slightly higher initial water inventory in the RWST.

- Cases B.5 and B.6 consider the effect of reducing the number of HHSI pumps.

- Cases B.7, B.8, and B.9 adjust the conditions assumed to lead to failure in the open position of a secondary-side PORV. Cases B.10, B.11, and B.15 apply the adjusted conditions together with the larger effective number of ruptured tubes considered in Case B.2.

- Cases B.12, B.13, and B.14 adjust the conditions assumed to lead to failure in the open position of a pressurizer PORV.

- Case B.16 considers the effects of operators taking control of the HHSI system, so that level is controlled in the core (see event CSI in Figure 2.1). Maintaining level lower in the RCS would be outside the EOPs, and is included for illustrative purposes.

- Case B.17 adds to Case B.16 the opening of one pressurizer PORV to prepare for B&F operations (see event B&F in Figure 2.1). The criterion to start B&F would not have been obtained, and the calculation is included for illustrative purposes, though it could be viewed as an error of commission.

- Case B.18 adds to Case B.16 the opening of two pressurizer PORVs in preparation for B&F operations (see event B&F in Figure 2.1 and caution above).

[24] It was pointed out during review that Step 1 of E-3 directs the operators to trip the RCPs. Since the prescribed sequence credits operator actions downstream of this procedural step, this is a specific failure assumed by the analysis. Note that the SPAR sequence does not consider RCP trip, and generally, early RCP trip would be expected to make this particular scenario more benign (the RCPs act to push more primary side fluid out of the break, which there is more than adequate heat removal for the additional energy they impart in the system).

- Cases B.19 and B.20 add the availability of the RHR, which is subject to permitting temperature and pressure conditions, to Cases B.17 and B.18, respectively (see event RHR in Figure 2.1 and caution above).

- Case B.21 adjusts the kinetics of the zirconium steam-oxidation reaction. The Prater and Courtright correlation for the rate coefficient is used instead of the Urbanic-Heidrick correlation, which is used in all other cases.

Table B.1 List of Operator Actions for the SGTR Sequence

Operator Actions Description[†]	**Time (hh:mm)**
Spontaneous SGTR	00:00
Automatic reactor SCRAM	00:03
E-0 – Step 1. Verify Reactor Trip – Status OK E-0 – Step 2. Verify Turbine Trip – Status OK E-0 – Step 3. Initiate E-0 Attachment 1, “System Alignment Verification” E-0 – Step 4. Check SI Initiated – Status True (Low Pressurizer Pressure) E-0 – Step 5. Verify AC Emergency Buses Energized – Status OK E-0 –Step 6 RCS Average Temp – Trending to 547 ºF, Turbine Bypass Valve (TBV) closed E-0 – Step 7 Pressurizer PORVs and Spray Valves – Closed E-0 – Step 8 RCP Trip – At least one charging pump is running (omitted here) E-0 – Step 9 Steam Generators Faulted – Pressure is increasing E-0 – Step 10 Steam Generator Tubes Ruptured – Status: High radiation Steam Generator NR increasing quickly Go to E-3	00:03 to 00:15
E-3 – Step 1 Check RCP Trip – Status OK; HHSI is running (omitted here) E-3 – Step 2 Identify Ruptured Steam Generator – Status True E-3 – Step 3 Isolate Ruptured Steam Generator – Status: Steam Generator PORV setpoint to 1,035 psig PORV operating and cannot be closed Steam Generator blowdown valves are closed Steam supply from ruptured Steam Generator to TDAFW is isolated Main Steam Turbine Valve (MSTV) from ruptured Steam Generator is closed E-3 – Step 4 Check Ruptured Steam Generator Level – NR level > 18% AFW isolated to ruptured SG E-3 – Step 5 Ruptured Steam Generator Pressure > 350 psig – Status True Go to ECA-3.1	00:15
ECA-3.1 – Operators fail to implement in the base case	
Assumed timing of the CSI[‡] (throttling the high-pressure injection (HPI), if applicable)	00:15
Assumed timing of the pressurizer (opening the PORVs, if applicable)	00:60
Assumed timing of B&F, if applicable	00:60
Assumed timing of the RHR, if applicable	00:15 after entry conditions

[†] From interactions with Surry operators as part of the SOARCA project.
[‡] CSI on procedure E-3, Step 23.

Table B.2 Sensitivity Cases for Surry SGTR

Case	# Tubes	RWST Inventory(1)	HHSI(2)	Secondary PORV Failure(3)	Pressurizer PORV Failure(3)	CSI(4)	FAB(5) (# of PRZ PORVs)	RHR(6)	Alternate Oxidation Model(7)
B.1	1	TS	3→2	CDF = 0.5	CDF = 0.5	No	0	No	No
B.2	5	TS	3→2	CDF = 0.5	CDF = 0.5	No	0	No	No
B.3	0.5	TS	3→2	CDF = 0.5	CDF = 0.5	No	0	No	No
B.4	1	Full	3→2	CDF = 0.5	CDF = 0.5	No	0	No	No
B.5	1	TS	1→1	CDF = 0.5	CDF = 0.5	No	0	No	No
B.6	1	TS	2→2	CDF = 0.5	CDF = 0.5	No	0	No	No
B.7	1	TS	3→2	CDF = 0.15	CDF = 0.5	No	0	No	No
B.8	1	TS	3→2	CDF = 0.85	CDF = 0.5	No	0	No	No
B.9	1	TS	3→2	Water or CDF = 0.5	CDF = 0.5	No	0	No	No
B.10	5	TS	3→2	CDF = 0.15	CDF = 0.5	No	0	No	No
B.11	5	TS	3→2	CDF = 0.85	CDF = 0.5	No	0	No	No
B.12	1	TS	3→2	CDF = 0.5	CDF = 0.15	No	0	No	No
B.13	1	TS	3→2	CDF = 0.5	CDF = 0.85	No	0	No	No
B.14	1	TS	3→2	CDF = 0.5	Water or CDF = 0.5	No	0	No	No
B.15	5	TS	3→2	Water or CDF = 0.5	CDF = 0.5	No	0	No	No
B.16[8]	1	TS	3→2	CDF = 0.5	CDF = 0.5	Yes	0	No	No
B.17[8]	1	TS	3→2	CDF = 0.5	CDF = 0.5	Yes	1	No	No
B.18[8]	1	TS	3→2	CDF = 0.5	CDF = 0.5	Yes	2	No	No
B.19[8]	1	TS	3→2	CDF = 0.5	CDF = 0.5	Yes	1	Yes	No
B.20[8]	1	TS	3→2	CDF = 0.5	CDF = 0.5	Yes	2	Yes	No
B.21	1	TS	3→2	CDF = 0.5	CDF = 0.5	No	0	No	Yes: P & C

1 TS: Technical Specification level; Full: maximum level
2 3→2: 3/3 HHSI pumps start and 1 is isolated at 17.9 minutes per procedure; 1→1: 1/3 HHSI pumps starts and it is not isolated at 17.9 minutes; 2→2: 2/3 HHSI pumps start and none are isolated at 17.9 minutes
3 CDF: Cumulative Distribution Function (e.g., 0.5 implies a 50% chance of failure for the given number of valve cycles); Water: valve failure due to two-phase flow
4 CSI: operators take control of safety injection
5 B&F – operators open 1 or 2 pressurizer PORVs
6 RHR: residual heat removal, available 15 minutes after entry conditions are persistently achieved (PRCS < 450 psi, TRCS < 350 ºF, hot leg water level above center line)
7 P & C: Prater and Courtright correlation for rate coefficients used instead of the default correlation
8 Cases for controlling HHSI above TAF (i.e., below the normal pressurizer level) and B&F are counter to EOPs for the prescribed conditions. They are included for illustrative purposes, though they could be viewed as representing errors of commission.

Section B.2.2.1 describes in detail the base-case calculation. Sections B.2.2.2 through B.2.2.21 describe the sensitivity calculations, typically in less detail. A summary discussion pertaining to this scenario appears in Section B.2.3, which is also in the main report.

B.2.2 Results

B.2.2.1 Results of Case B.1 (Base Case)

The sequence of successful versus failure events that defines scenario SGTR-9 is given in Figure B.1. The operator actions during the accident sequence are outlined in Table B.1. The scenario progression is outlined in Table B.3 and is illustrated in Figures B.1.1.1 to B.1.1.8.

The SGTR (initiating at t = 0 seconds after a pre-transient steady-state period lasting 100 seconds) is modeled by two flow paths, each leading to the boiler of the steam generator of loop A. One originates in the steam generator inlet plenum, the other in the steam generator outlet plenum. (Note that loop A is not the loop that contains the pressurizer.) Each has as a flow area the full cross-sectional area (3.045 cm^2) of one steam generator tube to model a double-ended break of one tube. Figures B.1.1.1 and B.1.1.2 show the primary pressure and the break flows, respectively.

The MFW stops at 2.9 minutes due to the scram. The RCPs continue to pump in all of the loops, by assumption that the operators fail to trip the RCPs as directed by procedure.

High-pressure injection (HHSI) into all three cold legs begins at 3.1 minutes. The water source is the RWST. Containment sprays and low-pressure injection (LHSI), also with the RWST as the source, are credited to be available; however, the sprays actuation setpoint (containment pressure above 0.172 MPa) is never attained, nor is the RCS pressure ever less than the LHSI deadhead pressure (1.18 MPa).

Motor-driven AFW starts at 3.5 minutes; turbine-driven AFW starts at 4.0 minutes. Steam generators A (with the steam generator tube break), B, and C (in the pressurizer loop) all receive motor-driven and turbine-driven AFW.

At the hard-wired time of 17.88 minutes, various operator actions are modeled. Both types of AFW are turned off to all steam generators. The pressure-dependent, per-loop HHSI rate is reduced to two-thirds of its true value, while injection at this reduced rate continues into all three legs. This treatment approximately models operator turn-off of one of the three HHSI pumps. The PORV in steam generator A becomes stuck open at 37.4 minutes from excessive cycling.

Motor-driven AFW starts up again at 50.3 and 51.0 minutes in steam generator C and steam generator B, respectively. The injection rate is controlled to maintain the level in the steam generator downcomers to 23.376 m. The water level in steam generator A is above target at this time, thus ensuring a zero injection rate. Figure B.1.1.4 shows the steam generator downcomer levels. The injected water does not boil and does not leave the steam generators, but it builds up in the boilers and domes while the water levels in the downcomers are kept near the target by the control. Finally, at 8.40 hours, the water levels in the downcomers exceed the target, and the AFW stops. The AFW in these loops never resumes in the scenario.

Table B.3 Timing of Key Events Predicted for the MELCOR Model SGTR Base-Case Calculation (Case B.1)

Event Description	**Time**
Spontaneous SGTR (double-ended rupture on one tube)	0
SCRAM, MFW trip, turbine control valves close	2.9 minutes
Steam dump valves open and modulate	Not Simulated†
Steam dump valves close (RCS temperature < 547 ºF)	Not Simulated†
HHSI initiated into all loops; source is the RWST	3.1 minutes
Motor-driven AFW starts in all loops	3.5 minutes
Turbine-driven AFW starts in all loops	4 minutes
Credited operator actions to isolate faulted steam generator: ‡ Secure AFW delivery Isolate blowdown valve Isolate Main Steam Turbine Valve (MSTV) Set Steam Generator PORV to 1,035 psig Isolate steam flow to TDAFW Check that PORV is closed below 1,035 psig	15 minutes
Operators turn off all motor-driven and turbine-driven AFW	17.9 minutes
Rate of HHSI into all loops reduced to 2/3 of full capacity (to model operator turn-off of one HHSI pump)	17.9 minutes
Faulted steam generator (i.e., loop A) PORV begins to cycle	30.1 minutes
Faulted steam generator, flooded	33.2 minutes
Faulted steam generator, PORV becomes stuck fully open	37.4 minutes
Motor-driven AFW resumes in loop C	50.3 minutes
Motor-driven AFW resumes in loop B	51 minutes
End of HHSI (RWST attains the switch-over level but the switch to recirculation mode fails)	8.35 hours
Accumulator injections begin	8.63 hours
Motor-driven AFW stops in loops B and C due to high levels	8.40 hours
RCPs trip in all loops (void fractions in RCP flow paths exceed 0.1)	12.32 hours
Motor-driven AFW resumes in loop A	14.27 hours
All AFW becomes unavailable due to exhausted CST (96,000 gallons drawn by all types of AFW); motor-driven AFW stops in loop A	27.83 hours
SG of loop A drys out (SGs of loops B and C remain full)	~31 hours
Swollen level below the TAF (occurs first in ring 5)	34.67 hours
Turn-on of counter-current natural circulation model (all loops)	35 hours
More than 0.1 kg hydrogen generated by oxidation in core	36.87 hours
PORV of SG in loop C opens and cycles for rest of calculation	37.13 hours
PORV of SG in loop B opens momentarily	37.77 hours
Calculation ends (PCT exceeds 3,000 ºF)	38.15 hours

† The automatic operation of steam dump valves is not represented in the Surry MELCOR model.

‡ Timing sequence from Appendix B, Section 5.3 of the SOARCA documentation (i.e., NUREG–1935; see Reference [B.1]). Only the first action is modeled, and as modeled it occurs at 17.9 minutes.

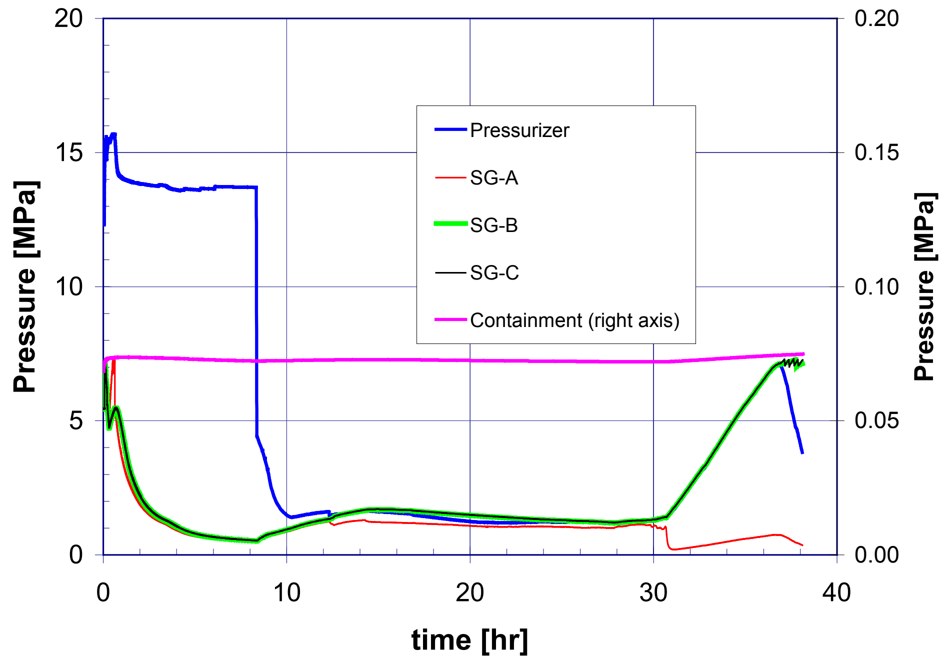

Figure B.1.1.1 Pressures (Case B.1)

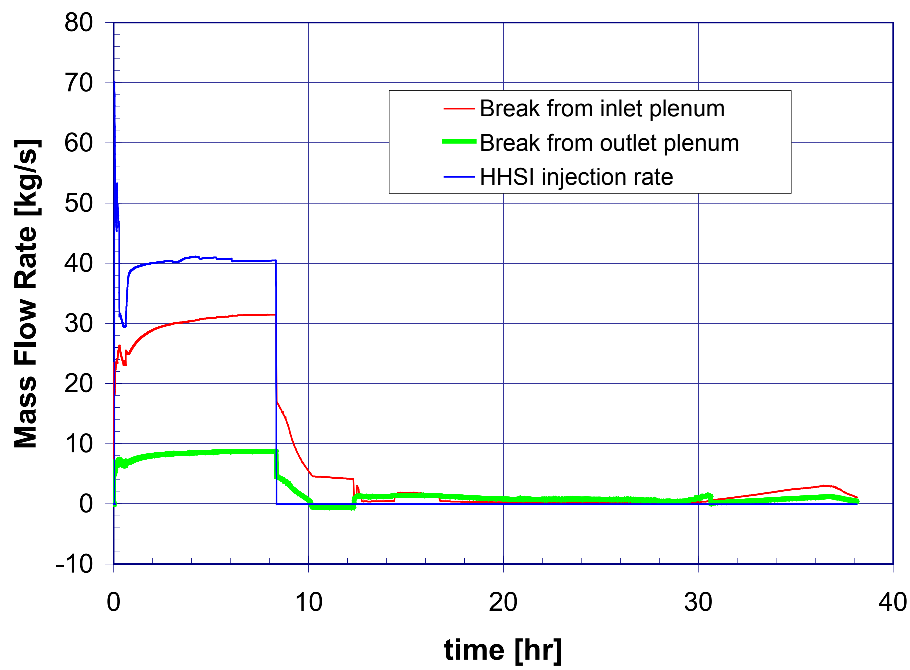

Figure B.1.1.2 SGTR Break Flows and HHSI Rate (Case B.1)

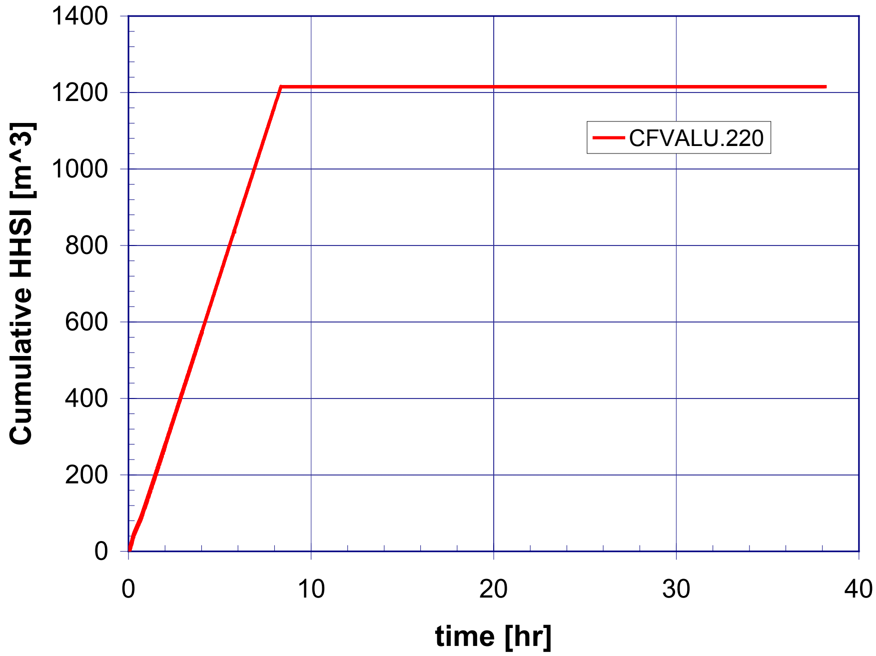

Figure B.1.1.3 Cumulative HHSI (Case B.1)

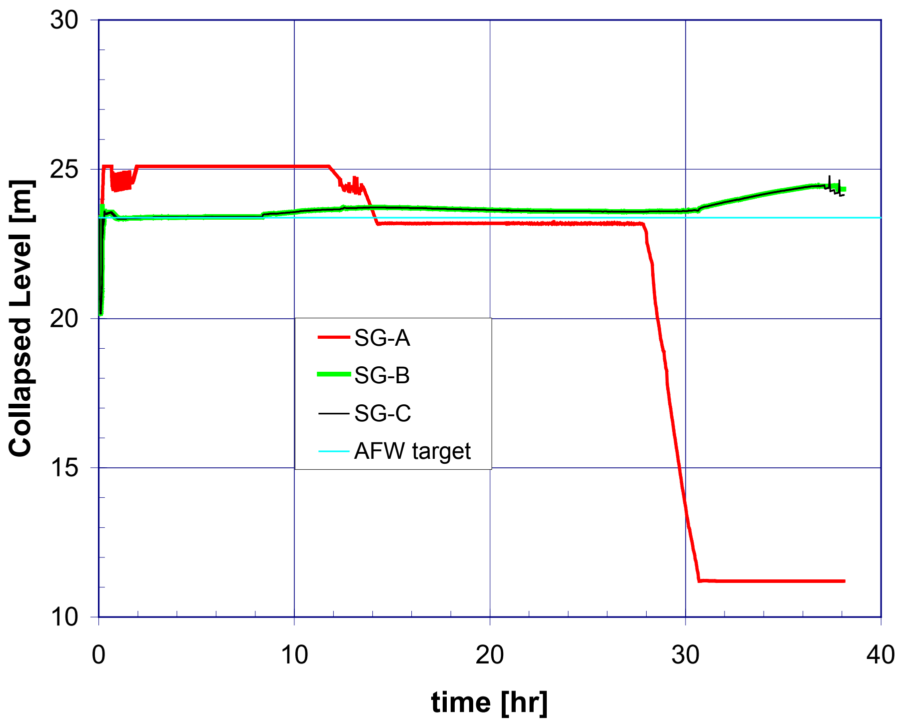

Figure B.1.1.4 Steam Generator Downcomer Water Levels (Case B.1)

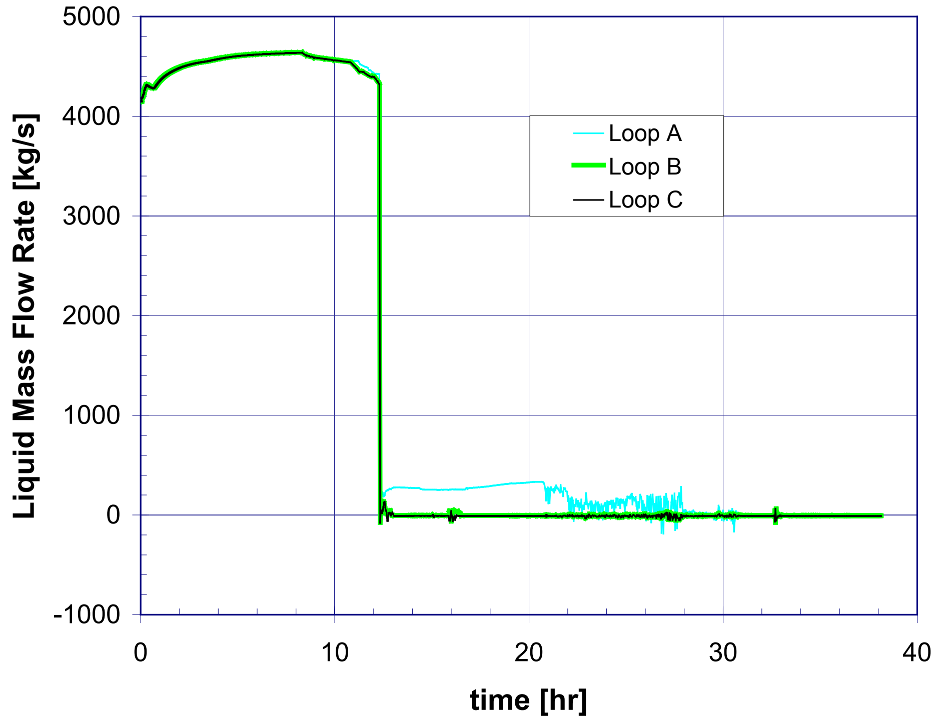

Figure B.1.1.5 Coolant Flow through the Cold Legs (Case B.1)

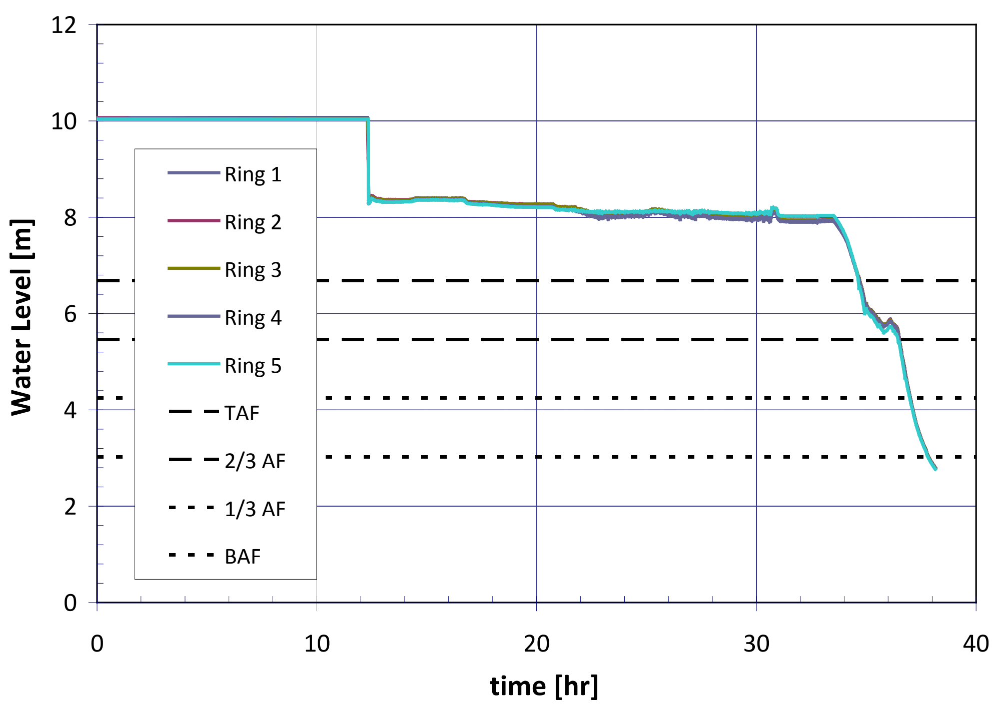

Figure B.1.1.6a Water Level in the Core (Case B.1)

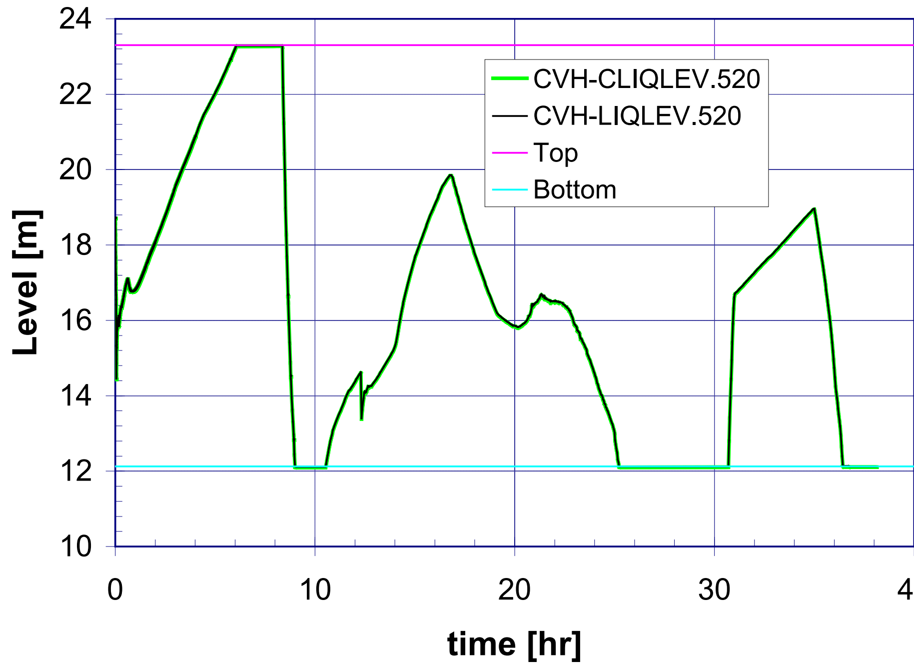

Figure B.1.1.6b Water Level in the Pressurizer (Case B.1)

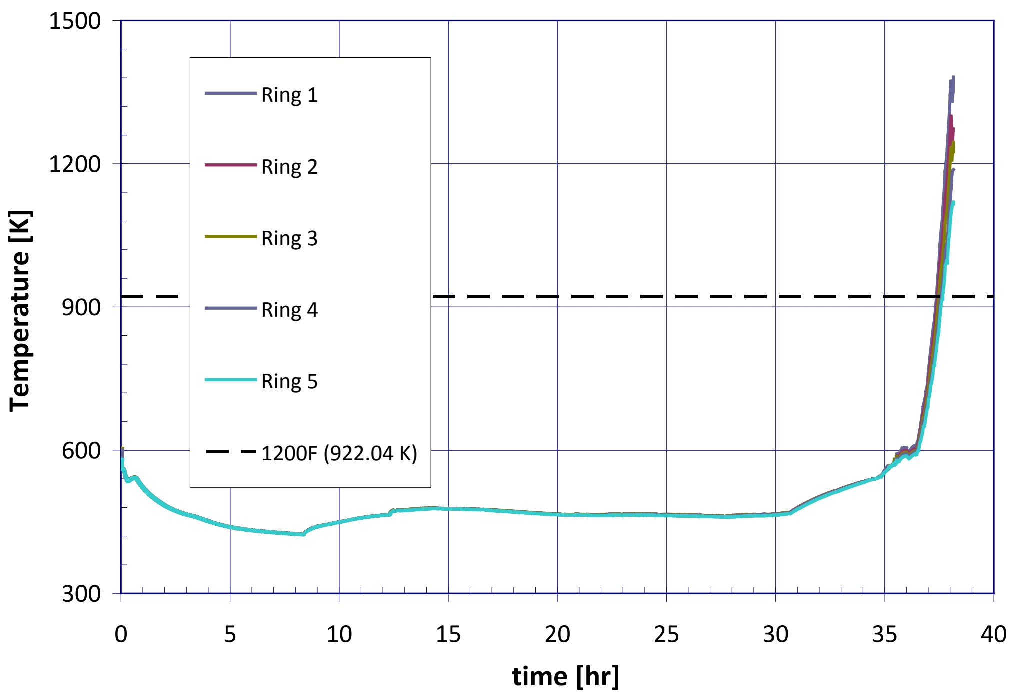

Figure B.1.1.7 Core Exit Temperature (Case B.1)

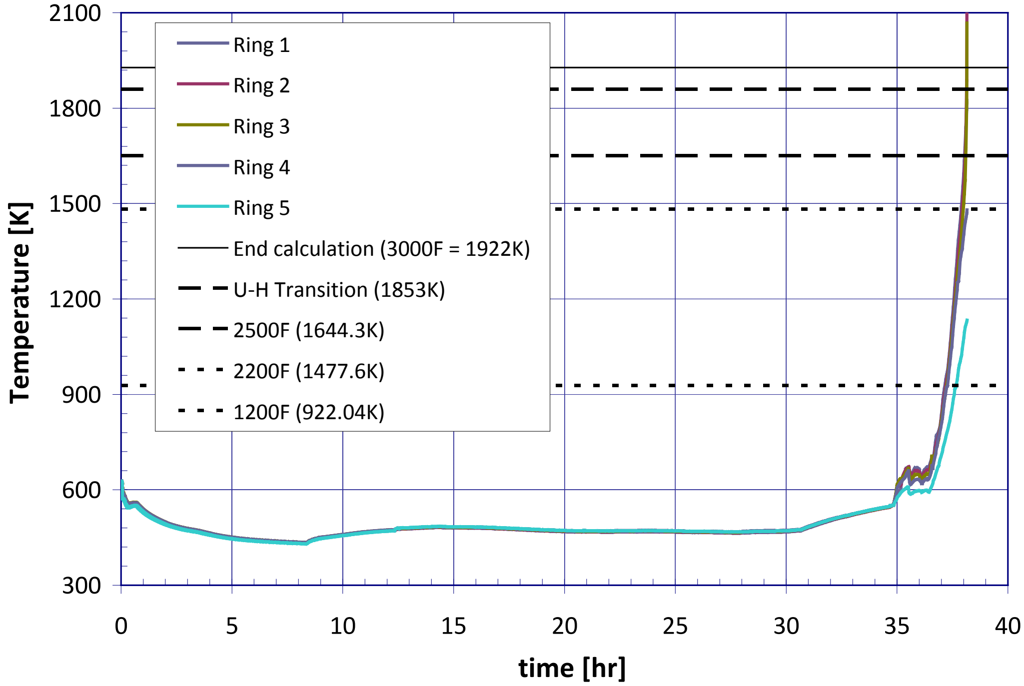

Figure B.1.1.8 Peak Cladding Temperature (Case B.1)

Figure B.1.1.3 shows the cumulative volume of water injected by the HHSI system. The injection mass rate appears in Figure B.1.1.2, co-plotted with the break mass flow rates. At 8.35 hours, HHSI has injected all available inventory of water in the RWST (1,215 m^3), so the switchover to the recirculation mode is required for the further use of HHSI (and, in general, LHSI and containment sprays – but these systems, though available, do not attain their setpoints and/or permissive conditions in this base-case calculation). According to the scenario definition, the switchover fails. No more injection to the primary system occurs in the calculation, except accumulator injections. Accumulators begin injecting for the first time at 8.63 hours. Accumulator injections stop at about 10 hours, resume at about 19 hours, and finally stop at about 21 hours.

The RCPs in all three loops stop at 12.32 hours when, nearly simultaneously, the void fraction passing the RCP flow paths exceeds 0.1. After the RCPs trip, the coolant continues to flow at reduced rates by natural circulation, as Figure B.1.1.5 shows. This figure shows the cold leg liquid flows. Immediately after the RCPs trip, the vessel dome (control volume CV160) and the regions immediately below it (control volumes CV1R4, where R=1, …, 5 stands for rings 1 through 5) abruptly empty out. (See Figure 2.2.1 in the main report for the vessel nodalization diagram.) Control volumes CV1R2 (which have the upper core support plate and TAF as the bottom) and CV7R8 (which have the upper core support plate and TAF as the top) stay nearly full. Being nearly full, they should be considered to deeply cover the core. Nevertheless, a control function, included in the SNL-prepared deck for evaluating the “water level below TAF” core damage surrogate, indicates this condition at this time (i.e., ~12.3 hours). This built-in control function has been defined by summing the heights of stacked water columns only up through CV7R8, whose tops are the TAF. This control function indicates at this time because of actually rather small gas volumes that develop in CV7R8 after RCP trip. The present analysis introduces a measure of water level that sums up column heights up through and including CV1R4 (i.e., almost to the top of the RPV). According to this measure, the (ring-dependent) water levels stay above the TAF until about 35 hours. Figure B.1.1.6a shows the so-defined levels. These levels stay fairly constant at about 8 m from 12.3 to 33 hours (TAF is at 6.72 m). Note that there are flow paths through the upper core support plate, so it is reasonable to credit water above the plate as covering the TAF. The level measure that sums water column heights up through CV1R4 is adopted in this analysis to redefine the “water level below TAF” core damage surrogate, since times of core damage obtained with this level measure are in much better agreement with those obtained with the other surrogates than are the times defined by the built-in “level below TAF” control function. The condition is considered to be satisfied when the lowest ring-dependent level is below the TAF, although in fact the levels defined by different rings are similar (Figure B.1.1.6a).

Figure B.1.1.6b shows the water level in the pressurizer. Even though it is completely full during the time from 6.1 to 8.4 hours, no liquid outflow through the pressurizer SRVs or PORVs occurs, because the pressure remains below the opening setpoints of all of these valves at all times during the calculation. The lowest opening pressure is 16.2 MPa for PORV-1.

After the RCPs trip, the natural-circulation RCS coolant flow rate in loop A (whose steam generator is the faulted one) is much greater than in the other loops. As a result, cooling is provided predominately by steam generator A. While steam generator B and steam generator C levels stay approximately constant and above the AFW target, the water level of steam generator A falls until the motor-driven AFW starts at 14.27 hours, in this loop only. Injection continues from this time and the water level of steam generator A is controlled to about 23 m until 27.83 hours, when the AFW source (i.e., the CST, credited to supply 96,000 gallons) is depleted. The water level of steam generator A then falls and dryout occurs at about 31.0 hours. The TAF uncovers, first in ring 5 at 34.67 hours. Rapid increase of the core exit temperature (CET) defined by the vapor temperatures in control volumes CV1R2 (see Figure B.1.1.7) starts at about 36.6 hours. Oxidation of Zircaloy in the core starts at about 36.9 hours. At 37.13 hours, the steam generator C PORV begins to cycle and the steam generator C water level begins to drop. This provides some cooling that lowers the vessel pressure but is insufficient to stop the rapid heatup. The “CET above 1,200 ºF (922 K)” core damage surrogate is met at 37.40 hours. A control function automatically stops the calculation when the peak cladding temperature (Figure B.1.1.8) has persisted above 3,000 ºF for 30 seconds, which occurs at 38.15 hours in this base-case calculation.

B.2.2.2 Results of Case B.2 (5-Tube Rupture)

Figures B.1.2.1 through B.1.2.5 show the results of calculations for this case. The larger break depressurizes the RCS and results in a lower RCS pressure relative to Case B.1 (the base case). The HHSI rate is therefore higher and causes an earlier depletion of the RWST at 4.40 hours. Additionally, a small amount of water is drawn and injected by the LHSI system. Containment sprays do not actuate. After the RCPs trip (loop C at 4.58 hours and loops A and B at 4.87 hours), the natural circulation of coolant is much greater in loop A than in the other loops. Therefore, steam generator A supplies cooling and receives motor-driven AFW until depletion of the CST at 18.27 hours. The core begins to uncover at 22 hours, oxidation of the core begins at about 22.6 hours, and the calculation is terminated at 23.11 hours because the PCT is in excess of 3,000 ºF.

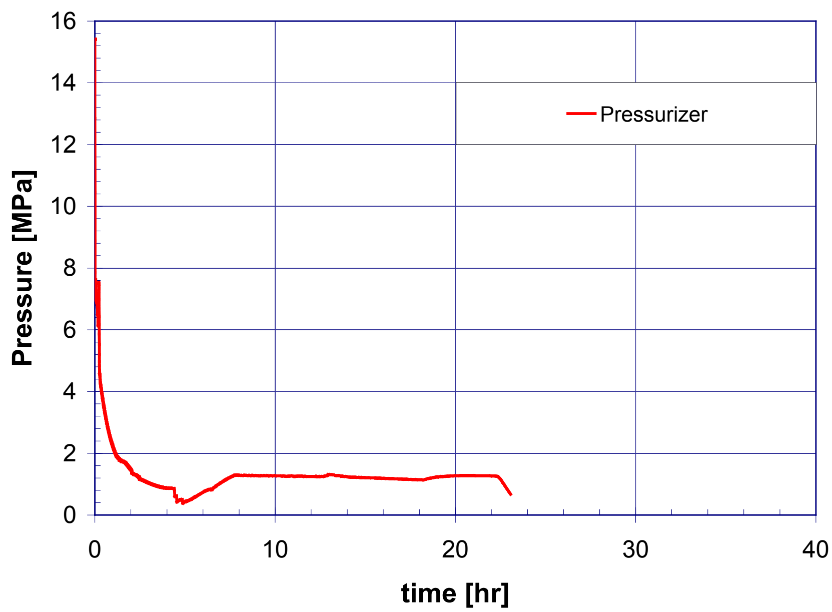

Figure B.1.2.1 Primary RCS Pressure (Case B.2)

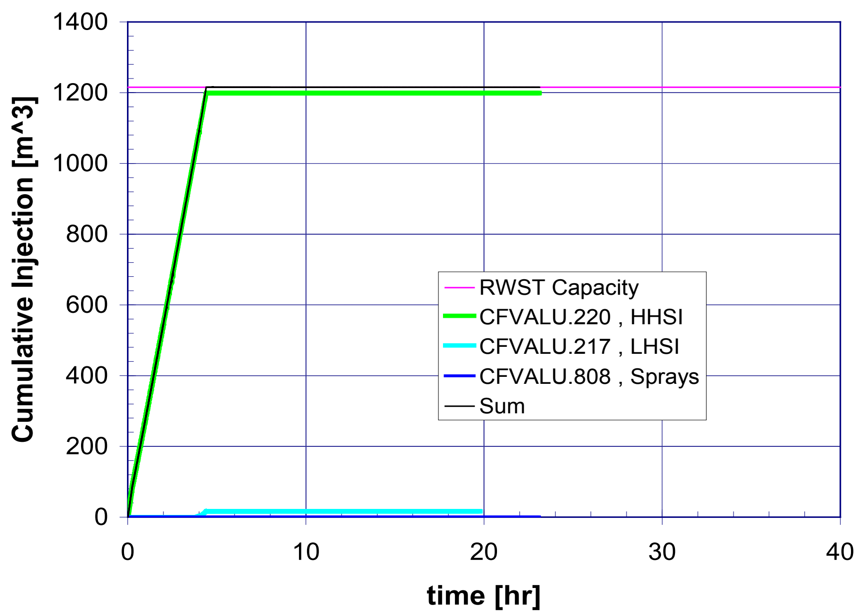

Figure B.1.2.2 Cumulative Safety Injections and Sprays (Case B.2)

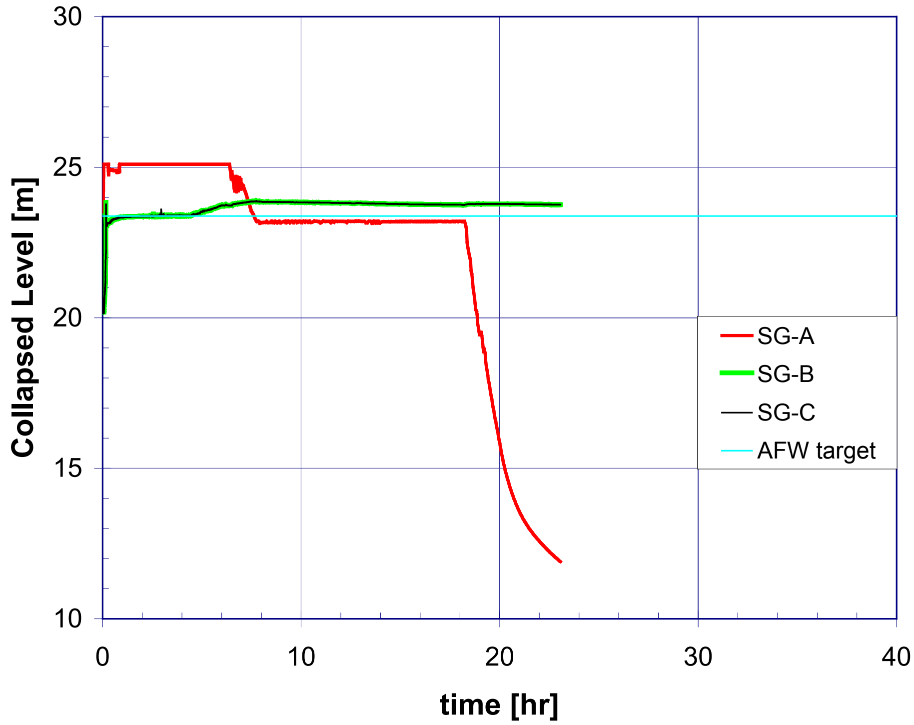

Figure B.1.2.3 Steam Generator Downcomer Water Levels (Case B.2)

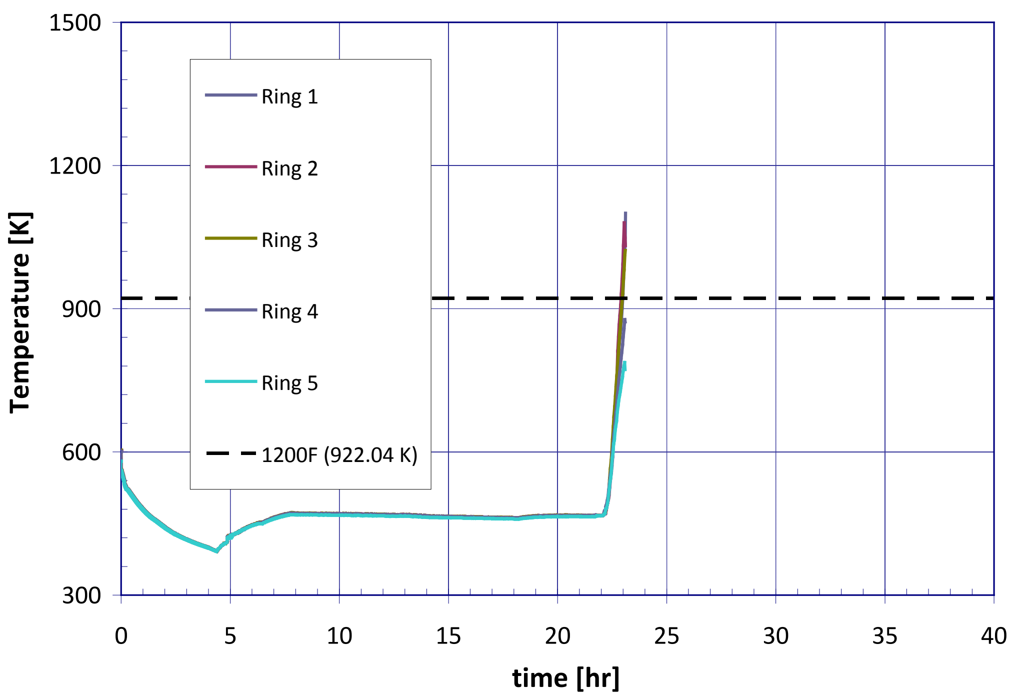

Figure B.1.2.4 Core Exit Temperature (Case B.2)

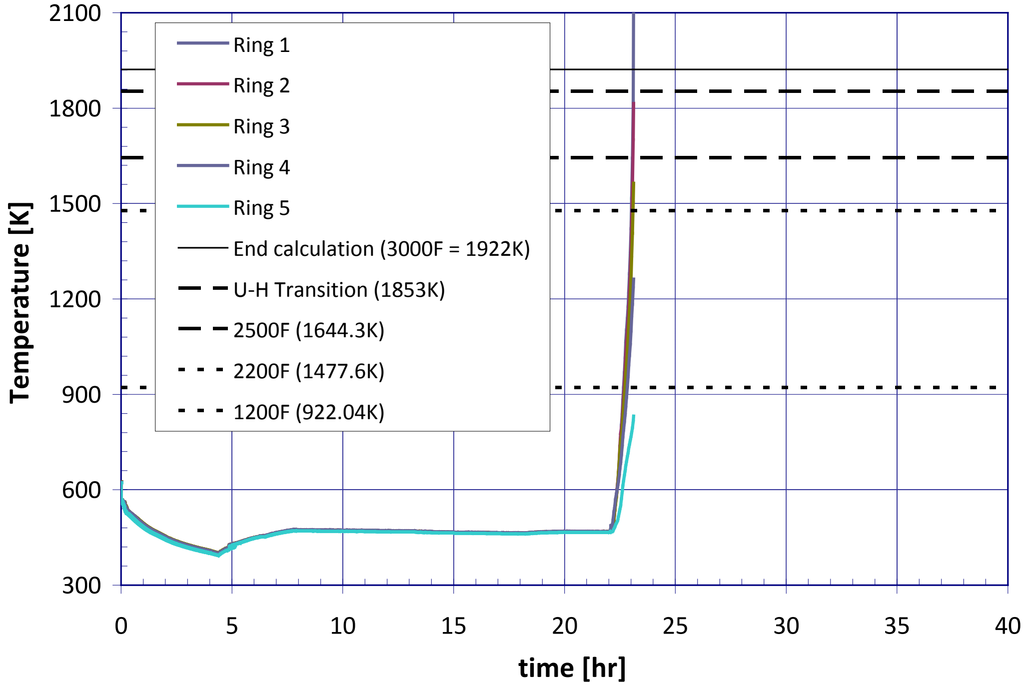

Figure B.1.2.5 Peak Cladding Temperature (Case B.2)

B.2.2.3 Results of Case B.3 (Half-Tube Rupture)

Figures B.1.3.1 through B.1.3.5 show the results of calculations for this case. The small break affords no depressurization of the RCS. Consequently, shortly after the operator action at 17.88 minutes (when the HHSI is reduced and AFW is stopped), pressurizer PORV-1 opens (19.5 minutes). PORV-1 then becomes stuck in the open position at 1.06 hours after cycling 495 times. Consequently, the containment pressurizes and attains the setpoint for spray actuation at 3.20 hours. The extra draw on the RWST, relative to the base case in which the sprays do not actuate, results in an earlier depletion of the RWST at 3.55 hours. The RCPs trip simultaneously at 4.24 hours. The break flows become negative after about 4.9 hours, due to the primary pressure falling below the pressure in steam generator A. These backflows into the primary system disrupt the natural circulation around loop A that had previously provided cooling. Also, there is no natural liquid circulation around the other loops. Therefore, the steam generators cannot provide any cooling. The calculation ends with the CST not exhausted but with no demand for AFW. The core begins to uncover at 6.65 hours, oxidation of the core begins at about 7.1 hours, and the calculation is terminated at 7.91 hours because of a PCT in excess of 1922 K (3,000 ºF).

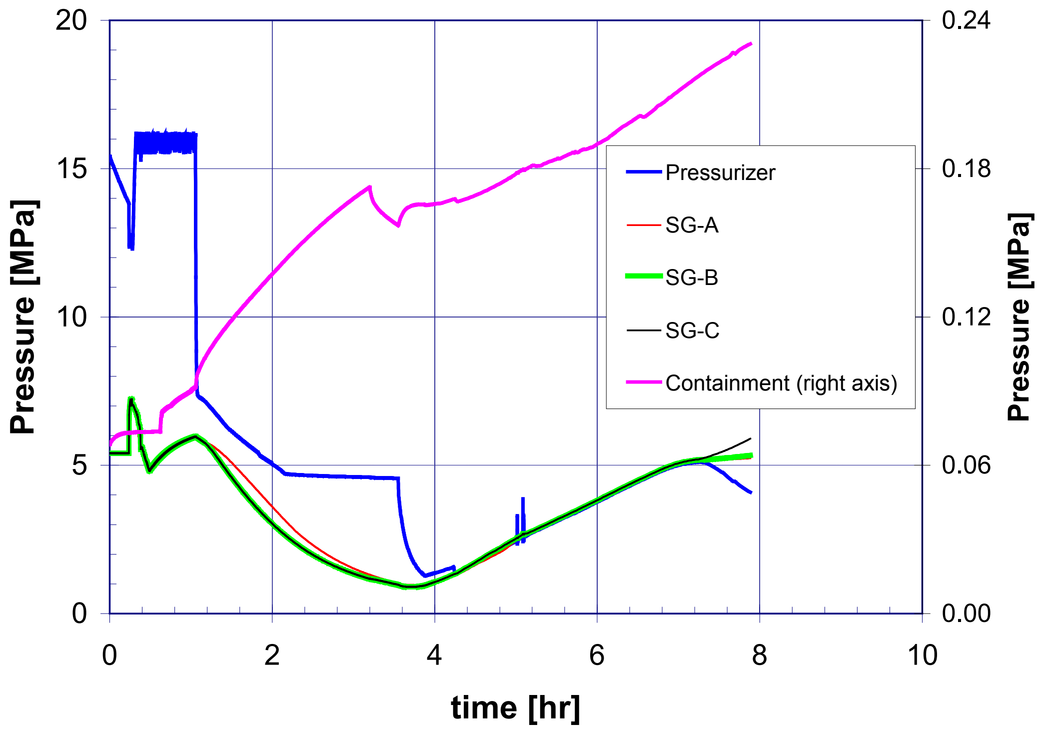

Figure B.1.3.1 Pressures (Case B.3)

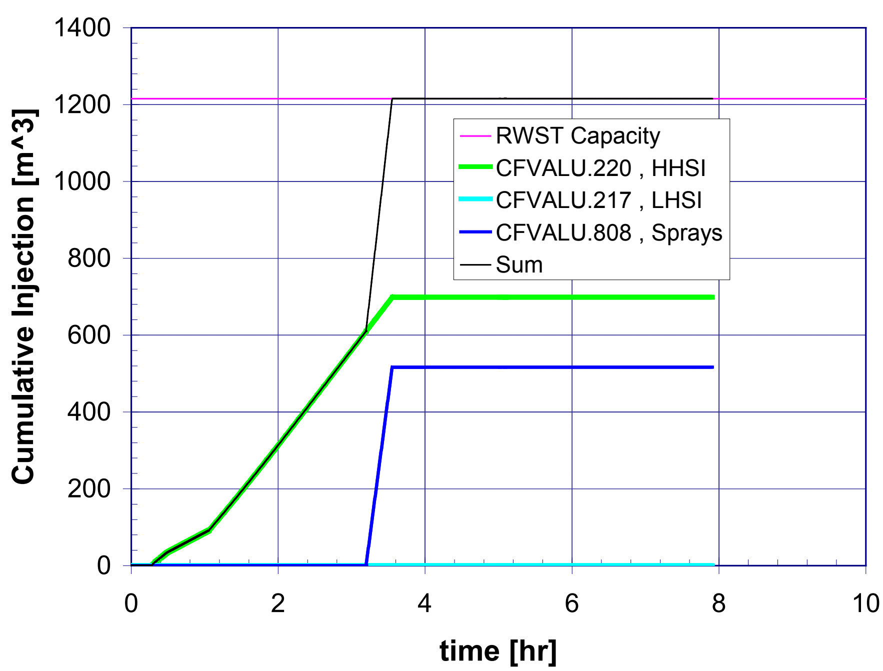

Figure B.1.3.2 Cumulative Safety Injections and Sprays (Case B.3)

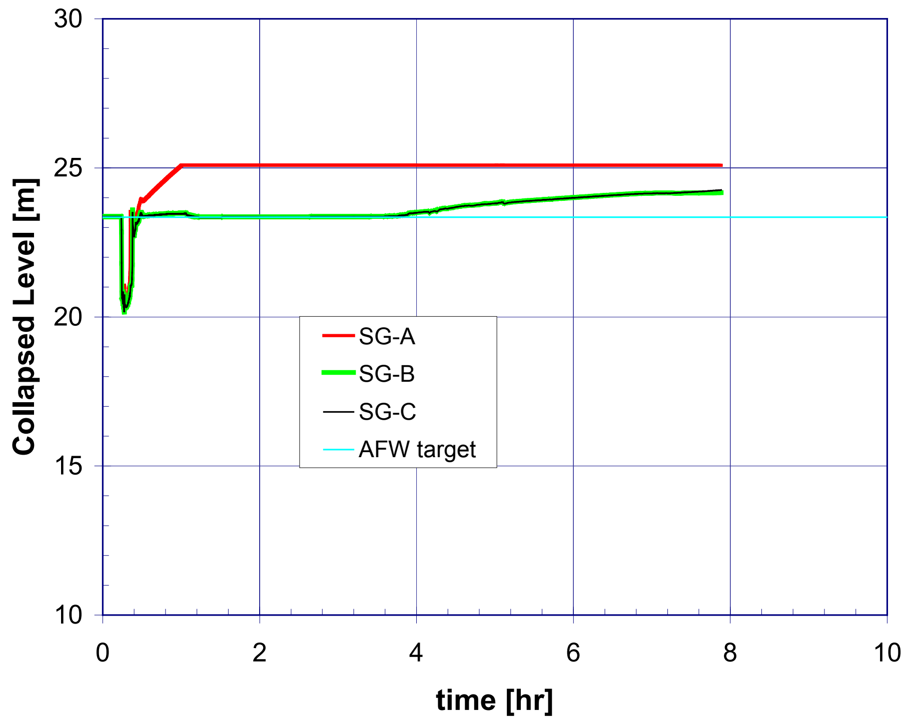

Figure B.1.3.3 Steam Generator Downcomer Water Levels (Case B.3)

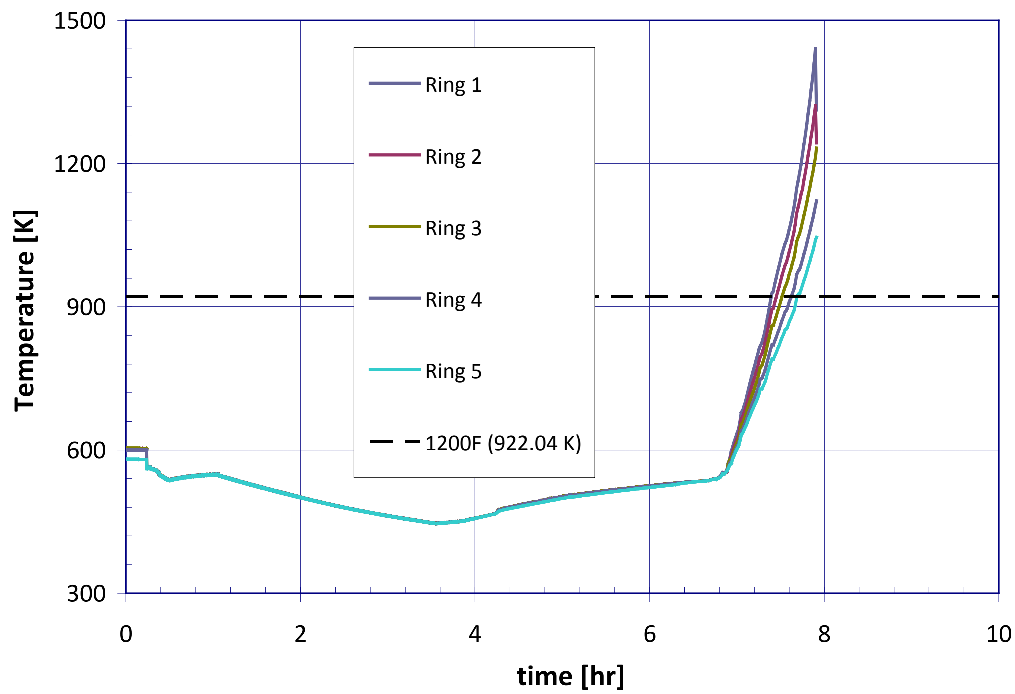

Figure B.1.3.4 Core Exit Temperature (Case B.3)

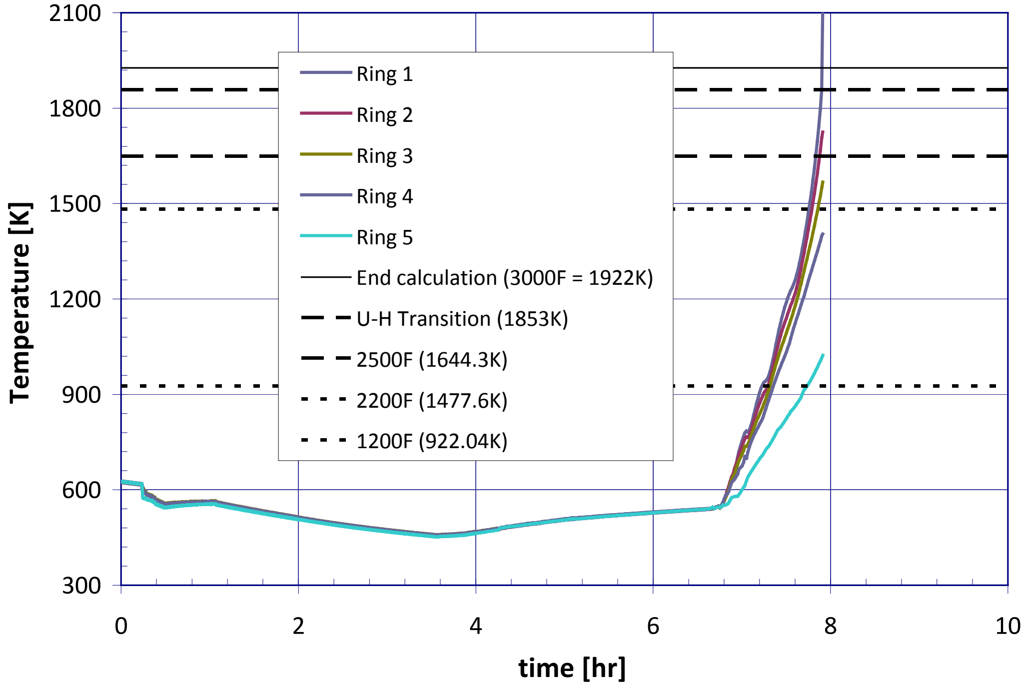

Figure B.1.3.5 Peak Cladding Temperature (Case B.3)

B.2.2.4 Results of Case B.4 (Additional RWST Inventory)

Figures B.1.4.1 and B.1.4.2 show the results of the calculations for this case, which uses the maximum value rather than the technical specification value for the initial water inventory of the RWST. This makes 1,260 m^3 of water available rather than the base-case value of 1,215 m^3. Progression of the accident is very similar to that of Case B.1. The core begins to uncover at 35.17 hours, oxidation of the core begins at about 37.4 hours, and the calculation is terminated at 38.68 hours because of PCT in excess of 1922 K (3,000 ºF).

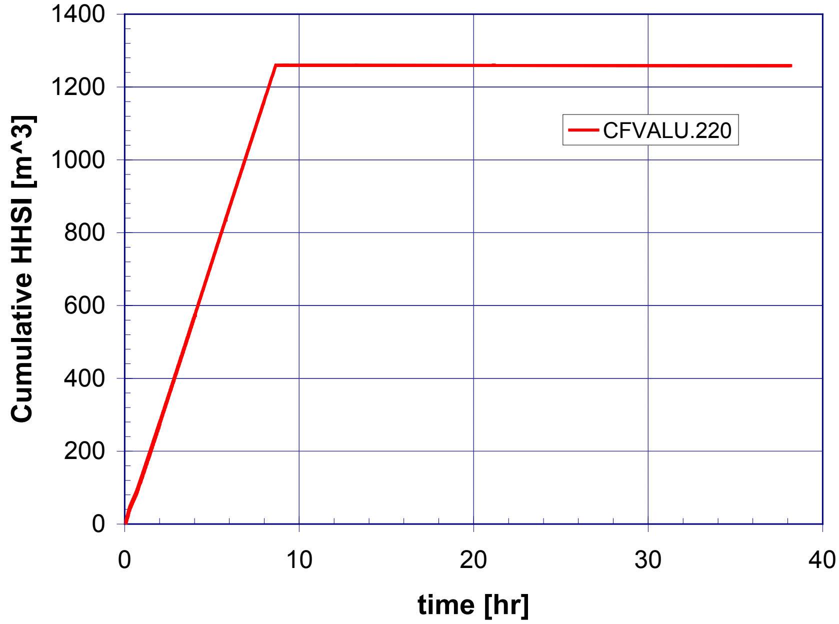

Figure B.1.4.1 Cumulative HHSI (Case B.4)

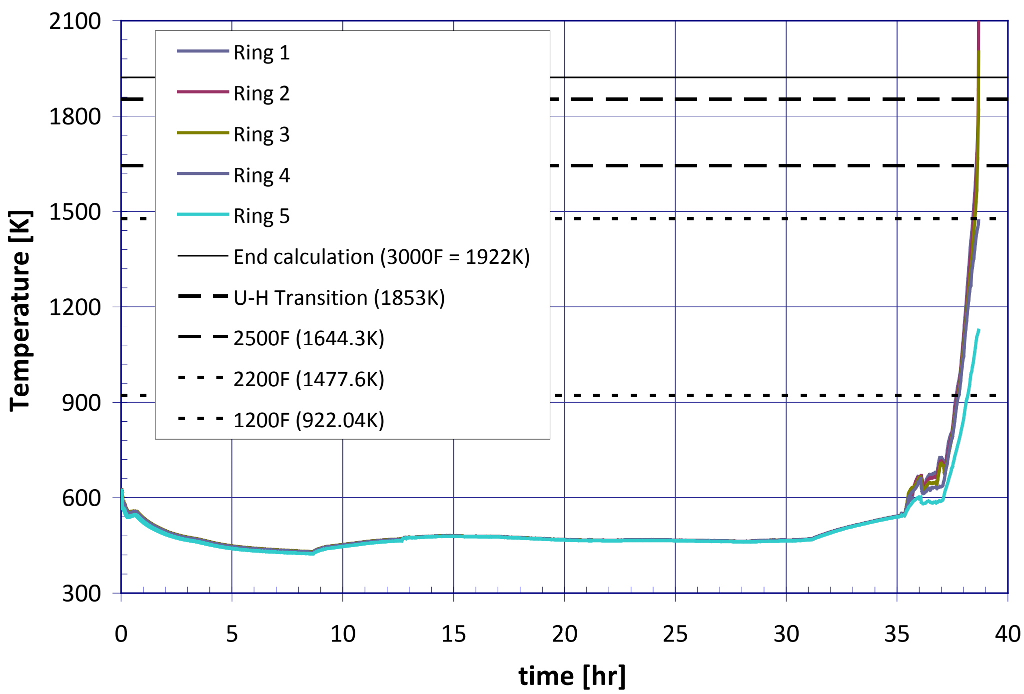

Figure B.1.4.2 Peak Cladding Temperature (Case B.4)

B.2.2.5 Results of Case B.5 (Reduced HHSI Availability—One Pump)

Figures B.1.5.1 through B.1.5.3 show the results of calculations for this case, which only used one HHSI pump. In Base Case B.1, three pumps are used until 17.88 minutes, then two are used. A comparison of Figure B.1.5.1 with Figure B.1.1.1 shows that during the time of the HHSI, the vessel pressure is significantly lower in the sensitivity case than in Case B.1. As a result, on a per-pump basis the injection rate is higher in the sensitivity case. In Case B.1, the net injection rate with two pumps is typically about 40 kg/second. In the sensitivity case, the net injection rate with one pump is typically about 30 kg/second. The reduced net injection rate of the sensitivity case lengthens the time to RWST depletion, but by much less than a factor of two that one would associate with using only half the number of pumps. Specifically relative to Case B.1, the time to deplete the RWST is lengthened by 3.1 hours from 8.35 hours in Case B.1 to 11.45 hours in Case B.5. The time to core damage is lengthened by about 4.9 hours. Qualitatively, the progression of the accident is very similar to that of Case B.1. The core begins to uncover at 39.30 hours, oxidation of the core begins at about 41.7 hours, and the calculation is terminated at 43.03 hours because of a PCT in excess of 1922 K (3,000 ºF).

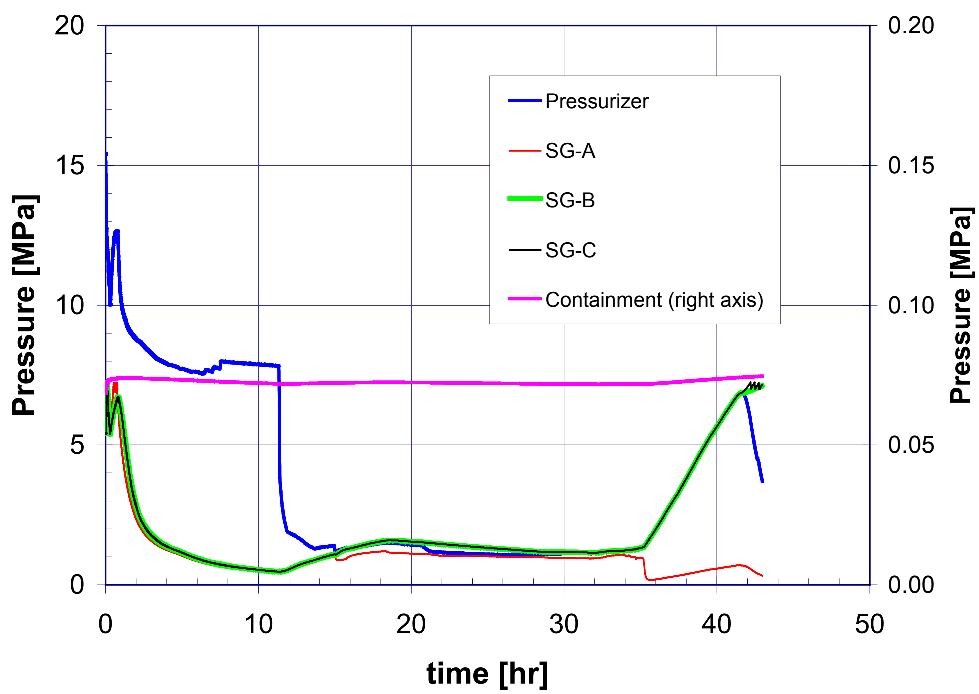

Figure B.1.5.1 Pressures (Case B.5)

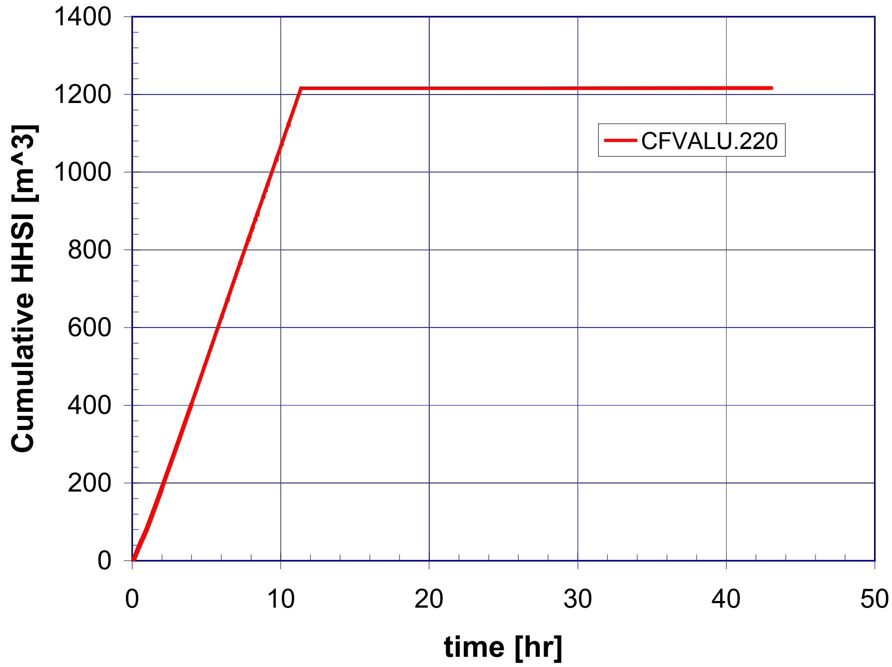

Figure B.1.5.2 Cumulative HHSI (Case B.5)

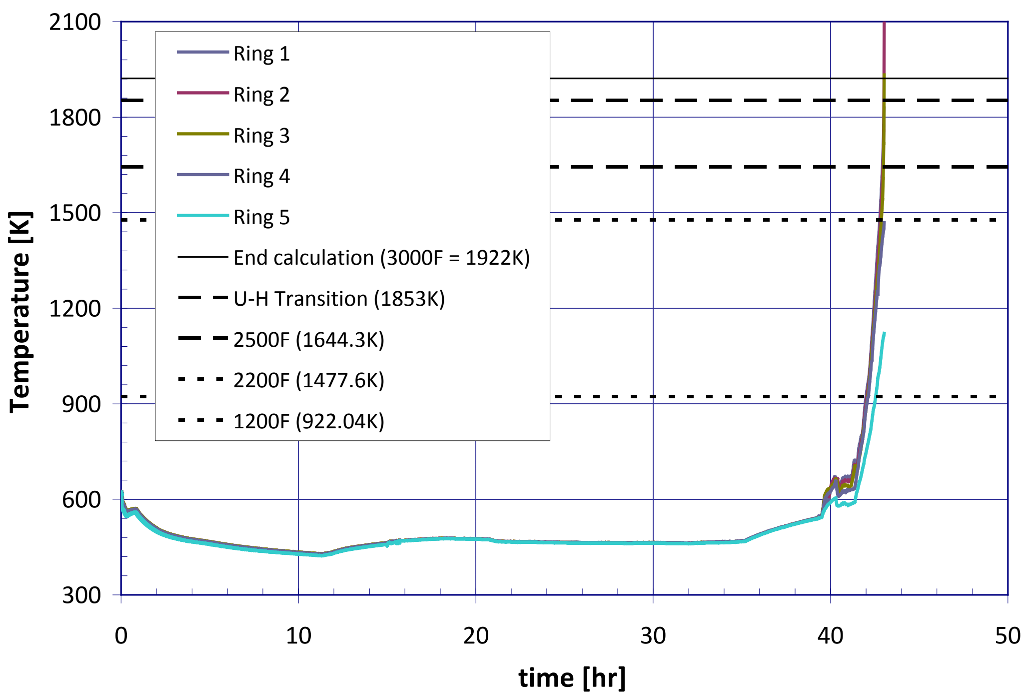

Figure B.1.5.3 Peak Cladding Temperature (Case B.5)

B.2.2.6 Results of Case B.6 (Reduced HHSI Availability—Two Pumps)

In this case, only two HHSI pumps are used. Case B.1 (the base case) used three until 17.88 minutes, then two. The progression of the accident is very similar to that of Case B.1. Core damage is postponed by about five minutes.

B.2.2.7 Results of Case B.7 (Earlier Failure in the Open Position of the Secondary PORV)

This case assumes that the loop A secondary-side PORV will become stuck in the open position after cycling the number of times for which the cumulative probability of valve failure is 0.15. (The probability for Case B.1 (the base case) is 0.50, with the result that the PORV of steam generator A will become stuck in the open position at 37.4 minutes.) As a result, the steam generator A PORV will become stuck open at 36.2 minutes. The progression of the accident differs negligibly from that of Case B.1.

B.2.2.8 Results of Case B.8 (Later Failure in the Open Position of Secondary PORV)

This case assumes that the loop A secondary-side PORV will become stuck open after cycling the number of times for which the cumulative probability of valve failure is 0.85. (In Case B.1 (the base case), the probability of valve failure is 0.50, with the result that the PORV of steam generator A will become stuck open at 37.4 minutes.) As a result, the steam generator A PORV will become stuck open at 39.4 minutes. The progression of the accident differs negligibly from that of Case B.1.

B.2.2.9 Results of Case B.9 (Failure in the Open Position of Secondary PORV by Water)

This case assumes that the loop A secondary-side PORV will become stuck open if the two-phase fluid passes through the valve. This condition is modeled as equivalent to significant liquid standing in the steam line. Additionally, as in Case B.1 (base case), the PORV will become stuck open after cycling the number of cycles that corresponds to a 0.50 cumulative probability of becoming stuck. In Case B.1, only becoming stuck after excessive cycling is considered, with the result that the PORV of steam generator A will become stuck open at 37.4 minutes. As a result, the steam generator A PORV will become stuck in the present case at 33.2 minutes, due to the two-phase flow. The progression of the accident differs negligibly from that of Case B.1.

B.2.2.10 Results of Case B.10 (5-Tube Rupture with Earlier Failure in the Open Position of Secondary PORV)

This case combines the conditions of Case B.2 (the rupture of five steam generator tubes) with those of Case B.7 (the steam generator A PORV becomes stuck open at the number of cycles corresponding to the 0.15 cumulative probability of sticking). As a result, the steam generator A PORV will be stuck open at 13.5 minutes compared with 13.8 minutes in Case B.2. The progression of the accident is very similar to that of Case B.2. Core damage occurs at about 18 minutes earlier than in Case B.2.

B.2.2.11 Results of Case B.11 (5-Tube Rupture with a Later Failure in the Open Position of Secondary PORV)

This case combines the conditions of Case B.2 (the rupture of five steam generator tubes) with those of Case B.8 (the steam generator A PORV becomes stuck open at the number of cycles corresponding to 0.85 cumulative probability). As a result, the steam generator A PORV becomes stuck in the open position at 14.6 minutes compared with 13.8 minutes in Case B.2. The progression of the accident is very similar to that of Case B.2. Core damage occurs about 11 minutes earlier than in Case B.2.

Note that relative to Case B.2, core damage occurs earlier in both Cases B.10 and B.11, although one case has an earlier stuck-open PORV and the other a later stuck-open PORV. However, ERI ran Case B.2 because SNL's calculation was not yet complete. It is possible that a small machine dependence caused the ERI-run Case B.2 to reach core damage relatively late, so the SNL-run Cases B.10 and B.11 do not bracket Case B.2.

B.2.2.12 Results of Case B.12 (Earlier Failure in the Open Position of One Pressurizer PORV)

Case B.12 was defined to consider the consequences of assuming that pressurizer PORV-1 will become stuck open after cycling the number of times for which the cumulative probability of valve failure is 0.15 (instead of 0.50 as Case B.1 assumes). Because PORV1 never opens in Case B.1, all results from this calculation are identical to those of Case B.1.

B.2.2.13 Results of Case B.13 (Later Failure in the Open Position of One Pressurizer PORV)

Case B.13 was defined to consider the consequences of assuming that pressurizer PORV-1 will become stuck open after cycling the number of times for which the cumulative probability of valve failure is 0.85 (instead of 0.50 as Case B.1 assumes). Because PORV-1 never opens in Case B.1, all results from this calculation are identical to those of Case B.1.

B.2.2.14 Results of Case B.14 (Failure in the Open Position of One Pressurizer PORV by Water)

Case B.14 was defined to consider the consequences of assuming that pressurizer PORV-1 will be stuck open when the pressurizer fills up with liquid and water passes the PORV. In fact, all results from this calculation are identical to those of Case B.1 (the base case) because, although the pressurizer fills completely during the time from 6.1 to 8.4 hours, the pressure always remains below the opening setpoints of all SRVs and PORVs.

B.2.2.15 Results of Case B.15 (5-Tube Rupture, with Failure in the Open Position of Secondary PORV by Water)

This case combines the conditions of Case B.2 (the rupture of five steam generator tubes) with those of Case B.9 (loop A secondary-side PORV becomes stuck open when the steam line begins to fill up with liquid). As a result, the steam generator A PORV becomes stuck open at 10.5 minutes compared with 13.8 minutes in Case B.2. The progression of the accident is very similar to that of Case B.2. Core damage occurs about 19 minutes earlier than in Case B.2.

B.2.2.16 Results of Case B.16 (HHSI Injection Rate Controlled To Maintain the Water Level in the Core)

Figures B.1.16.1 through B.1.16.4 show the results of calculations for this case, which assumes that operators take control of the HHSI rate to maintain the level in the core in a range of 7.4 to 8.1 m (i.e., above the TAF at 6.72 m). Such action would be outside the EOPs (maintaining level in the pressurizer would be expected), but the case is included for illustrative purposes.

The control is initiated at 32.3 minutes resulting from a high water level in steam generator A. At this time, the core level is above the target, so the HHSI is shut off. Trips of the three RCPs occur nearly simultaneously at 2.88 hours. HHSI resumes with a low average rate at 8.80 hours and a more significant average rate at about 11.6 hours. As in Case B.1 (the base case), after the RCPs trip the coolant flow by natural circulation is predominately in loop A. Motor-driven AFW is supplied to steam generator A (and to steam generator B and steam generator C in minute amounts) until the CST is exhausted at 9.77 hours. Dryout of steam generator A occurs at about 12.3 hours. Ongoing HHSI keeps the core covered and cool.

Relative to Case B.1, the level-dependent HHSI control substantially reduces the average HHSI rate, so the RWST is still not depleted as of 72 hours. By extrapolation, it will deplete at about 88 hours. Thus, core damage is avoided in this case, at least to times in excess of 72 hours.

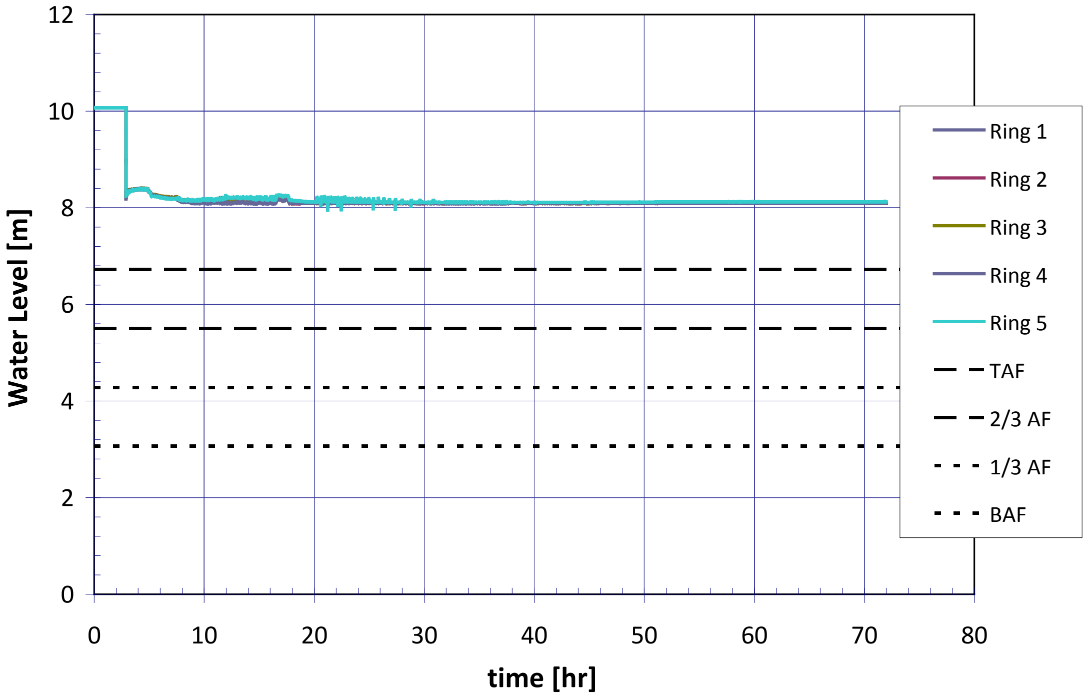

Figure B.1.16.1 Water Level in the Core (Case B.16)

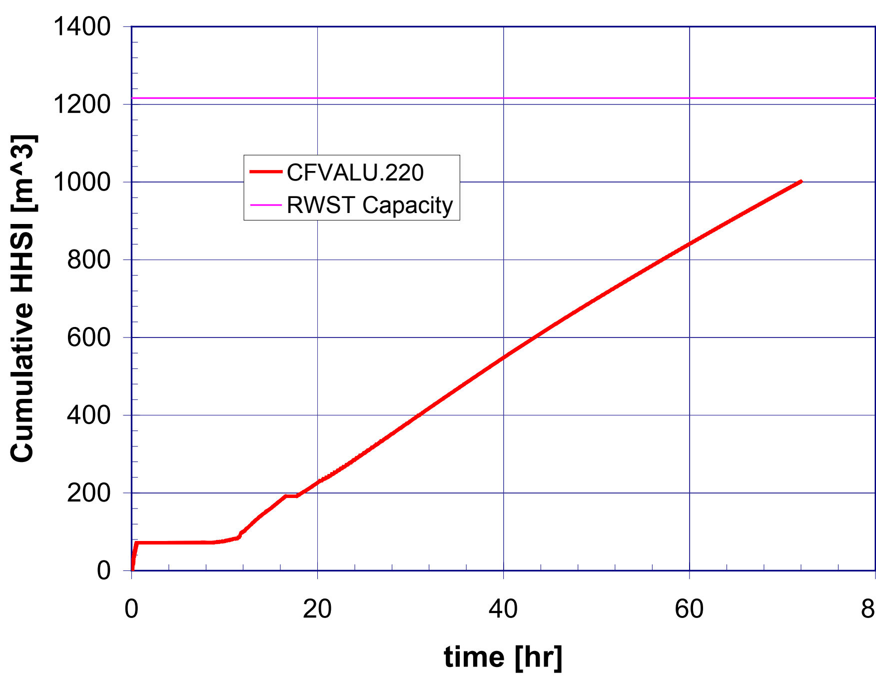

Figure B.1.16.2 Cumulative HHSI (Case B.16)

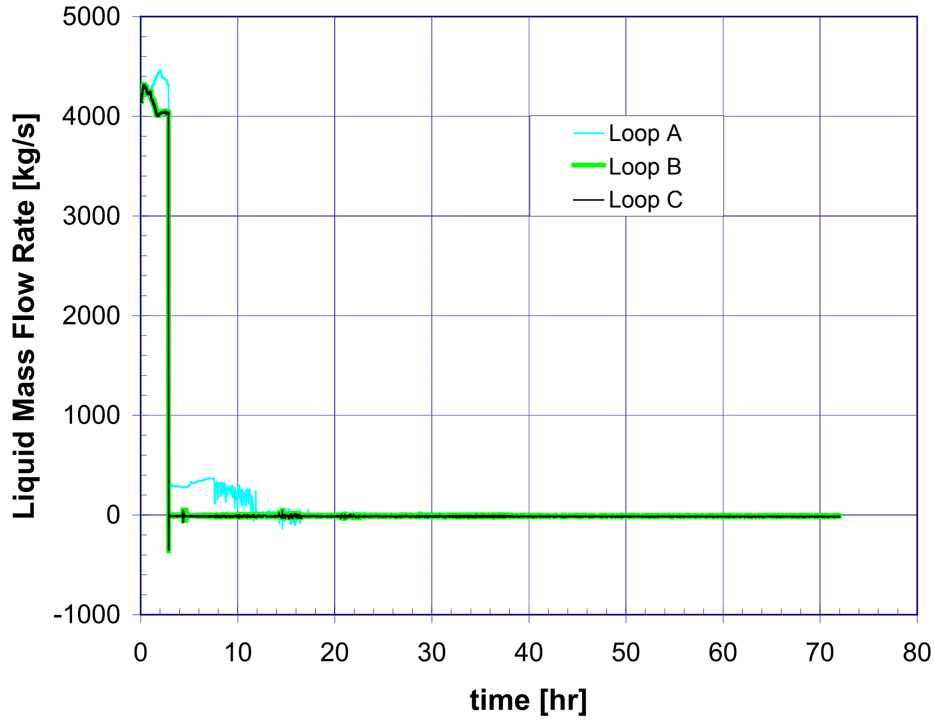

Figure B.1.16.3 Coolant Flow through the Cold Legs (Case B.16)

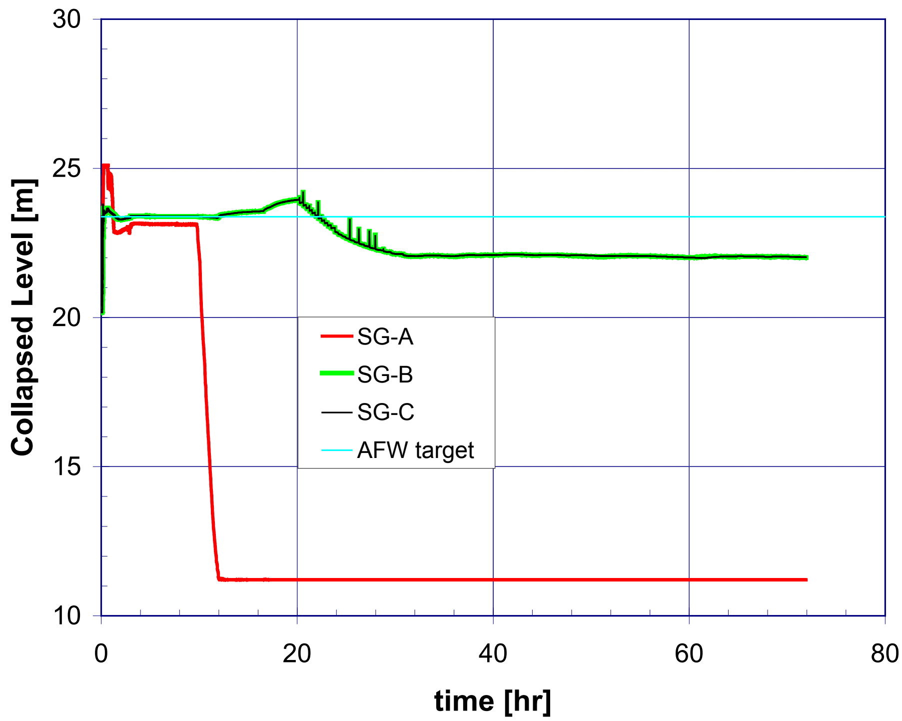

Figure B.1.16.4 Steam Generator Downcomer Water Levels (Case B.16)

B.2.2.17 Results of Case B.17 (Controlled HHSI with One Open Pressurizer PORV)

Figures B.1.17.1 through B.1.17.3 show the results of calculations for this case. This case adds the opening of pressurizer PORV-1 to the level-dependent control of the HHSI rate described in Section B.2.2.16. Actuation of PORV-1 takes place at the same time and condition (32.3 minutes due to high water level in steam generator A), when the injection control is initiated. Opening the PORV is a preparatory step for B&F (error of commission for the prescribed conditions given that the B&F criteria would not be expected), but as in all of the SGTR scenarios, a switchover to the ECCS recirculation modes is not available.

As a consequence of the open PORV, the containment pressurizes and attains the setpoint for spray actuation at 3.36 hours. The primary pressure, however, is never reduced to below the maximum primary pressure of 1.18 MPa that allows the LHSI. The extra draw on the RWST from the sprays relative to Case B.1 (the base case) in which the sprays do not actuate, results in an earlier depletion of the RWST at 3.99 hours.

The RCPs trip nearly simultaneously at 56.8 minutes. The break flows become negative after about 1.5 hours. After that, there is no primary natural liquid circulation around loop A (or any other loop). Therefore, the steam generators cannot provide any cooling. The calculation ends at 6.86 hours because of a peak cladding temperature above 3,000 ºF) with AFW still being supplied to the steam generators of loops A and C.

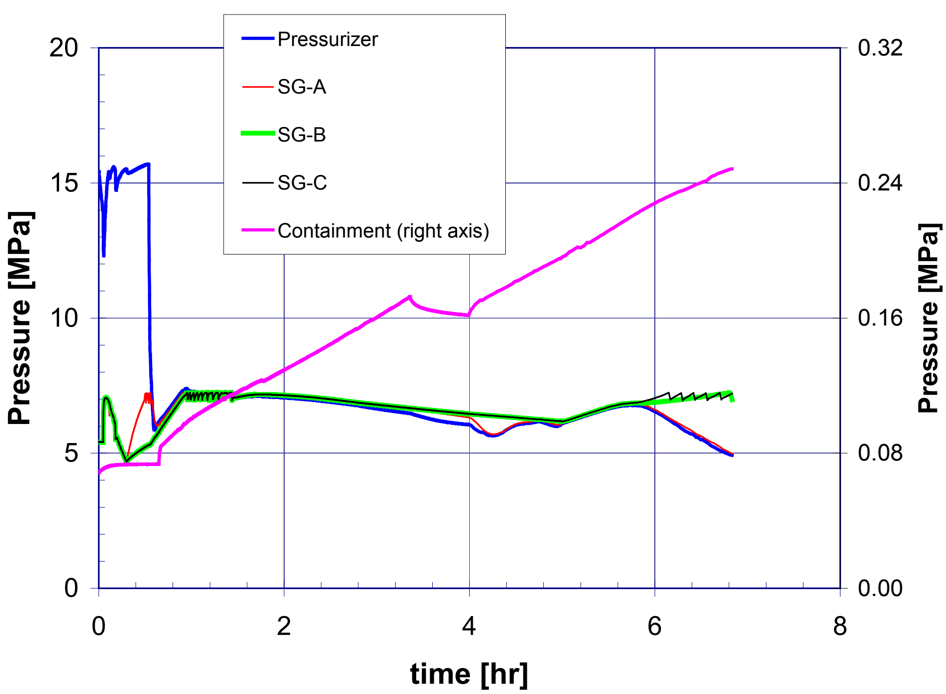

Figure B.1.17.1 Pressures (Case B.17)

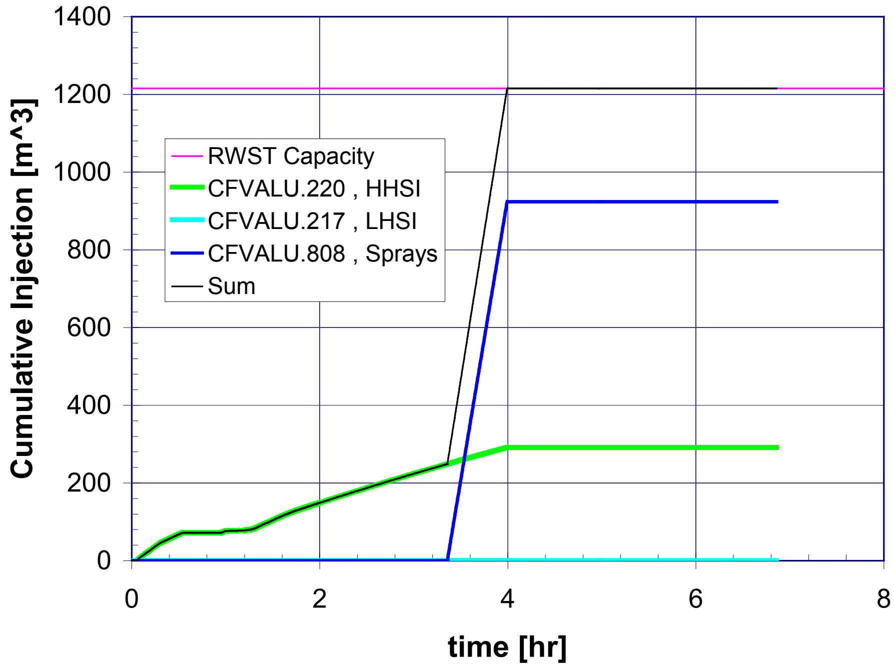

Figure B.1.17.2 Cumulative Safety Injections and Sprays (Case B.17)

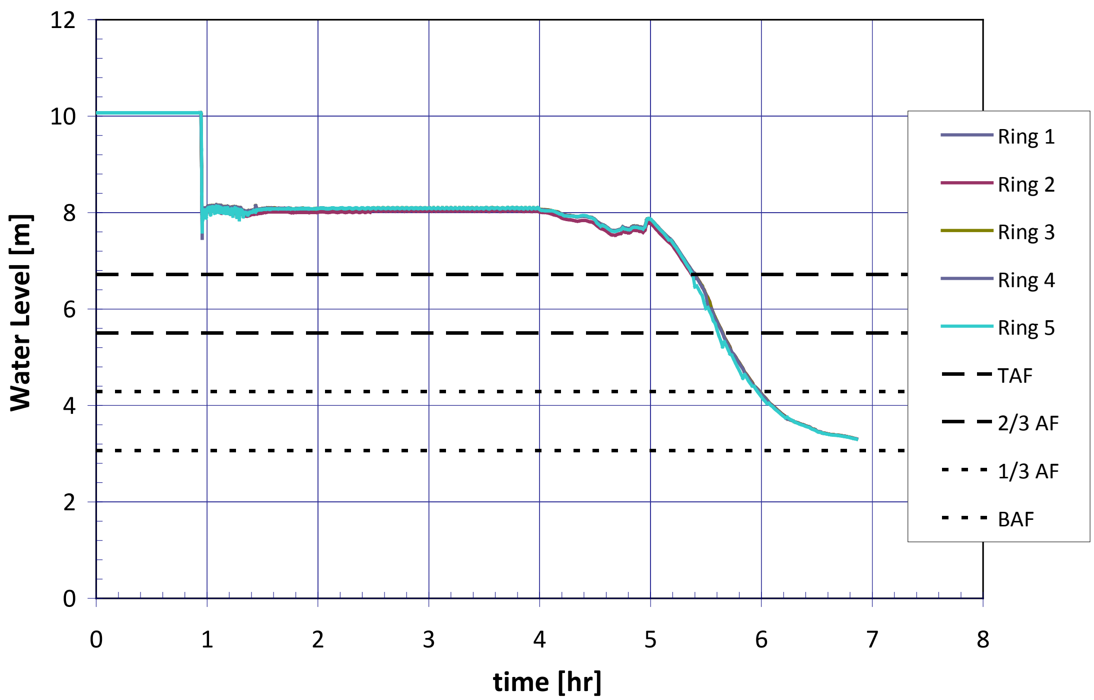

Figure B.1.17.3 Water Levels in the Core (Case B.17)

B.2.2.18 Results of Case B.18 (Controlled HHSI with Two Open Pressurizer PORVs)

Figures B.1.18.1 through B.1.18.3 show the results of calculations for this case. This case is similar to Case B.17 except operators open both PORV-1 and PORV-2. Again, the case is included for illustrative purposes. The accident progression is qualitatively similar to that of Case B.17. Spray actuation occurs at 2.12 hours; RWST depletion occurs at 2.77 hours; and the RCPs trip at 44.7 minutes. The calculation ends at 5.36 hours due to a peak cladding temperature above 3,000 ºF, with AFW still being supplied to steam generator A.

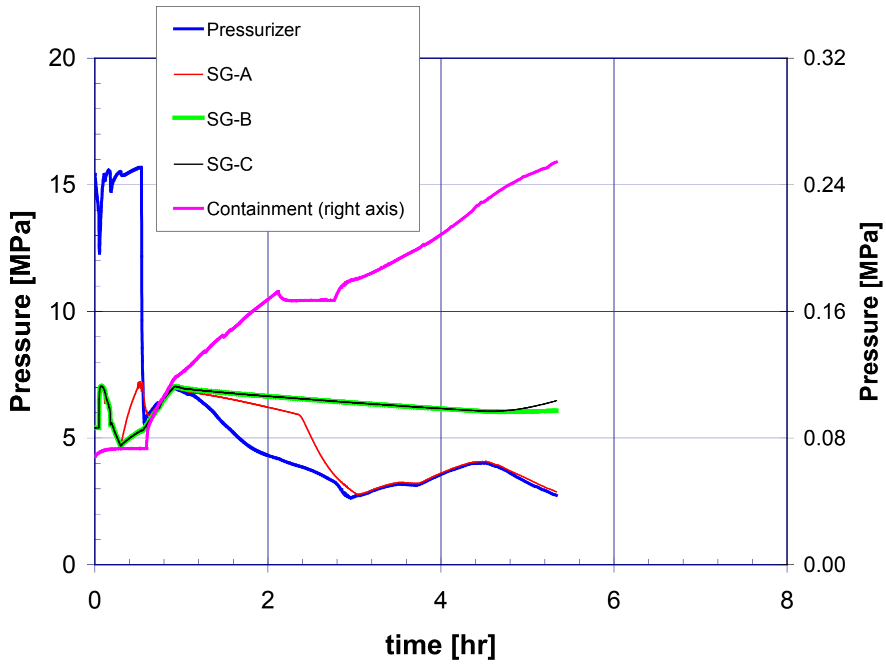

Figure B.1.18.1 Pressures (Case B.18)

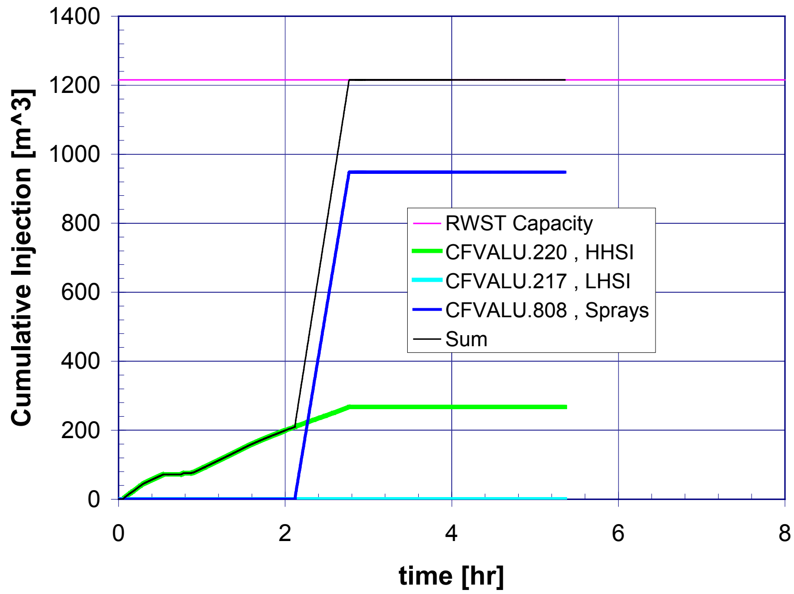

Figure B.1.18.2 Cumulative Safety Injections and Sprays (Case B.18)

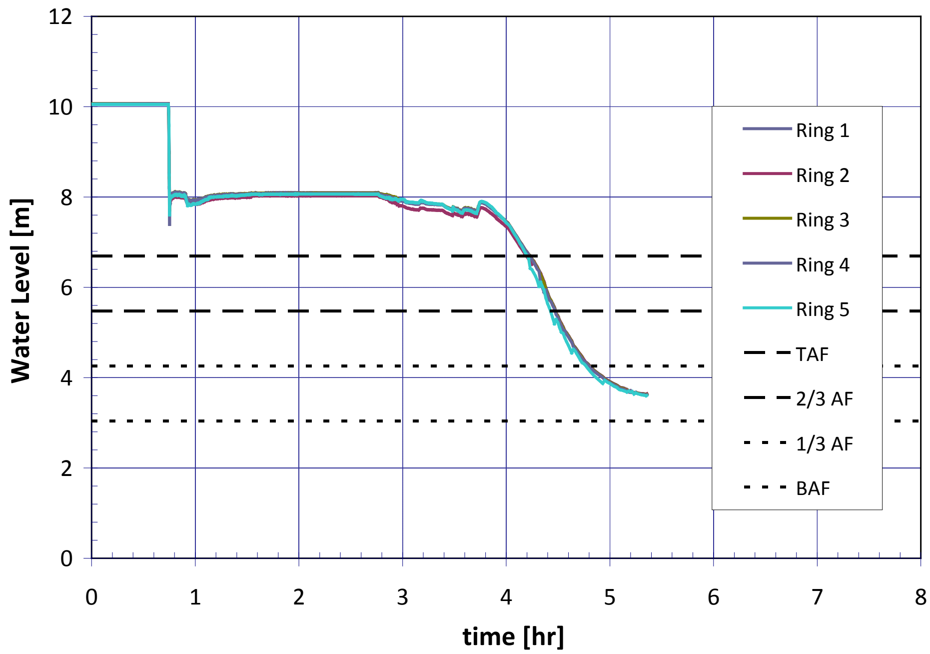

Figure B.1.18.3 Water Levels in the Core (Case B.18)

B.2.2.19 Results of Case B.19 (Controlled HHSI with One Open Pressurizer PORV, and an Available RHR)

The assumptions of this case are identical to those of Case B.17 except that the RHR was declared unavailable in all previous cases, but it is available here. This system draws water from a hot leg, passes it through a heat exchanger, and re-injects it into two cold legs. This system may start up only if the average temperature of the liquid in the vessel is below 450 K. Since this condition is not achieved in Case B.17, the results of Case B.19 are identical to those of Case B.17. Again, the case is included for illustrative purposes.

B.2.2.20 Results of Case B.20 (Controlled HHSI with Two Open Pressurizer PORVs and an Available RHR)

The assumptions for this case are identical to those of Case B.18, except that RHR is available here. This system draws water from a hot leg, passes it through a heat exchanger, and re-injects it into two cold legs. This system may start up only if the average temperature of liquid in the vessel is below 450 K; Case B.18 does not achieve this condition. The minimum vessel liquid reached in Case B.20 is 479 K. Case B.19 misses the permissive temperature even more widely. The results of Case B.20 are therefore identical to those of Case B.18. Again, the case is included for illustrative purposes.

B.2.2.21 Results of Case B.21 (Alternative Correlation for Oxidation Reaction Rate Coefficients)

This case differs from Case B.1 (the base case) only by the use of the Prater and Courtright correlation for the temperature-dependent rate coefficient that describes the kinetics of the reaction in which zirconium is oxidized by steam. All other calculations use the MELCOR default (i.e., the Urbanic and Heidrick correlation). The accident progression is very similar to that of Case B.1. The success criterion of hydrogen production attaining 1 percent of the amount that corresponds to steam oxidation of the whole-core zirconium inventory is met at 38.00 hours in Case B.21, versus 37.63 hours in Case B.1. The end of the calculation, due to PCT exceeding 3,000 °F (1,922 K), occurs at 38.22 hours, versus 38.15 hours in Case B.1.

B.2.3 Summary: SGTR-9 Scenario

In the base-case scenario (Case B.1), core damage occurs at about 38 hours (e.g., 37.90 hours according to the surrogate measure that identifies core damage with the peak cladding temperature (PCT) exceeding 2,200 °F (1,478 K).

Core damage is much earlier (5.1 to 7.8 hours by the PCT >2,200 °F [1,478 K] measure) in the sensitivity cases that lead to a stuck-open or held-open pressurizer PORV. A stuck-open PORV occurs in the half-tube break case (Case B.3), because the small break does not depressurize the vessel, thus leading to excessive PORV cycling. Held-open PORVs are assumed in several cases (Cases B.17 through B.20) as one of the conditions that would be established by operators as part of initiating B&F (an error of commission for the prescribed conditions). Whether the PORVs are stuck open or held open, the resulting steam flow to the containment actuates the sprays and consequently, depletes the RWST more quickly.

The sensitivity cases that result in actual reduction of the HHS injection rate (i.e., Cases B.5, B.6, and B.16) lead to longer times to core damage because depletion of the RWST is postponed. Note that Cases B.17 through B.20 are not included in this category. Although water-level-dependent control of HHSI is credited in these cases, the injection rate tends to be higher than in the base case because these cases also call for held-open PORV(s).

The one case in which core damage is avoided (i.e., Case B.16, as of 72 hours) assumes that no more HHSI occurs than is required to maintain core water level somewhat above the TAF (included for illustrative purposes).

The cases that consider a five-tube break (i.e., Cases B.2, B.10, B.11, and B.15) give intermediate times (around 23 hours) because of lower vessel pressures relative to the base case, thus leading to higher HHSI rates and earlier RWST depletion.

These results show that specific boundary condition assignments dominate the sources of variability for the analyzed scenario. These include the number of tubes ruptured, operator action to initiate B&F, and operator action to maintain core water level just above the TAF to conserve the RWST inventory. The importance of the latter two specific items is dominated by the scenario assumption that recirculation is unavailable. As such, the key conclusion is that specific boundary condition assignments drive the results, and which specific boundary condition assignments are important depends on the specifics of the scenario. This is in contrast to a situation where all boundary condition specifications contribute relatively equally to result variability.

The times to core damage range from 4.20 to 43.03 hours, depending on the sensitivity case and on the core damage surrogate measure. This range excludes one sensitivity case (Case B.16, with successful operator initiation of water-level-dependent control of HHSI), in which no core damage is indicated as of 72 hours.

With very few exceptions, the order in which the various surrogate measures indicate core damage is independent of the sensitivity case. In most cases, the final measure of PCT above 3,000 °F (1,922K) is indicated around two hours after one-third of the core is uncovered.

In summary, Table B.4 compares the 21 cases with respect to certain events in the accident progression. A comparison of the times of core damage according to various surrogate measures appears in Section 4.2 of the main report.

Table B.4 Comparison of Times of Events (in Hours) among SGTR Base and Sensitivity Calculations

Case	Flooding[1] of SG in loop A	RWST Depleted	RCP Trip (A, B, C)[2]	Restart of AFW to SG in loop A	CST Depleted	Dryout[3] of SG in loop A	Core Water Level at TAF	Calculat-ion End Time [4]
B.1 (Base)	0.52	8.35	12.32	14.27	27.83	31.0	34.67	38.15
B.2	0.17	4.40	4.87, 4.87, 4.58	7.75	18.27	{11.87}	22.00	23.11
B.3	Does not occur	3.56	4.23	Does not occur		{25.10}	6.65	7.91
B.4	0.52	8.67	12.63	4.35	28.33	31.4	35.17	38.68
B.5	0.67	11.37	14.97	18.33	32.12	35.4	39.30	43.03
B.6	0.56	8.40	12.33	14.30	27.90	30.9	34.73	38.22
B.7	0.52	8.37	12.27	14.30	27.87	30.8	34.67	38.16
B.8	0.52	8.37	12.3	14.30	27.87	30.9	34.67	38.18
B.9	0.52	8.33	12.27	14.27	27.83	30.8	34.63	38.13
B.10	0.17	4.42	4.97, 4.95, 4.95	7.43	18.00	{11.88}	21.70	22.82
B.11	0.17	4.42	4.95	7.48	18.10	{11.88}	21.80	22.92
B.12 – B.14	Same as Case B.1							
B.15	0.17	4.40	4.85, 4.53, 4.85	7.45	17.97	{11.88}	21.67	22.79
B.16[6]	0.52	~88[5]	2.87	1.26	9.77	12.2	> 72.00	72.00
B.17[6]	0.52	3.99	0.94	3.97	Does not occur	{23.27}	5.38	6.86
B.18[6]	0.52	2.77	0.74	2.34		{23.28}	4.20	5.36
B.19[6]	Same as case B.17							
B.20[6]	Same as case B.18							
B.21	0.52	8.37	12.30	14.27	27.83	30.8	34.67	38.22

1 Defined as the earliest time when liquid passes the steam generator outlet (FL283).
2 Only one time is tabulated if the three pumps trip nearly simultaneously.
3 In cases where dryout does not occur, the final downcomer water level is tabulated in braces. (Dryout corresponds to 11.20 m.)
4 Calculations end when peak cladding temperature exceeds 3,000 ºF, except for Case 16.
5 Extrapolation.
6 Cases for controlling HHSI above TAF (i.e., below the normal pressurizer level) and B&F are counter to EOPs for the prescribed conditions. They are included for illustrative purposes, though they could be viewed as representing errors of commission.

B.3 Reference

B.1 R. Chang, J. Schaperow, T. Ghosh, J. Barr, C. Tinkler, and M. Stutzke, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," U. S. Nuclear Regulatory Commission, NUREG–1935 (January 2012).

APPENDIX C: DETAILS OF PEACH BOTTOM SBO ANALYSIS

C.1 Introduction

C.1.1 Background

The Peach Bottom plant and its MELCOR model are described in Section 2.3 of the main report. The SNL Peach Bottom model documentation provides a more detailed description of the model. This Appendix highlights details of the calculations performed at ERI using the Peach Bottom model[25].

C.1.2 Objectives and Scope

The objectives of this Appendix are to document the calculations specific to the Peach Bottom plant that encompass detailed results for the basic scenario and its sensitivity cases. These simulations are limited in scope to considerations important for determining the relevant PRA level-1 system success criteria and sequence timing.

C.1.3 Outline

Section C.2 is devoted to results that describe one basic scenario consisting of an SBO with the RCIC operational until batteries are depleted. Section C.2 defines the scenario and its sensitivity cases. Detailed results specific to the base and sensitivity cases appear in subsections of Section C.2.2. Summary results pertaining to all cases are in Section C.2.3. These summary results are also in Section 3.3 of the main report.

All detailed results are in the subsections of Section C.2.2. Each subsection pertains to a particular base or sensitivity case. To simplify the Figure numbering, a four-character code is used within these subsections, according to the numbering scheme defined in the previous Appendices. After "C," which designates Peach Bottom, the next character numbers the scenario; in the case of Peach Bottom, there is only one scenario. The character after that numbers the sensitivity case; the base case is counted as the first case. The last character in the string is the count among the Figures that apply to a given sensitivity case of a given scenario. For example, the first figure in the second sensitivity case that pertains to the SBO with RCIC scenario is Figure C.1.2.1. Figures not pertaining to a specific scenario and sensitivity case are not numerous, and are designated by a two-character code. The first character denotes Peach Bottom, and the second character is a count over the Appendix. A similar two-character code is used for all the Tables.

C.2 Accident Scenario and Results

C.2.1 SBO with RCIC Operation until Battery Depletion

The transient initiator is a loss of all onsite power, except for batteries that are credited to support only the operation of the RCIC system. Table C.1 tabulates the assumptions that define the scenario. No system is credited to function except the RCIC, which is available until 8 hours, the assumed time that the batteries will be depleted. Due to the loss of all AC power, RHR pumps, low-pressure core sprays, and service water are not available; and FW, recirculation pumps, and the CRD hydraulic system fail at the

[25] As mentioned in Section 2, it was discovered after completion of the analyses that the value used for the high-vessel-level shutoff of RCIC in automatic control is lower than the actual value by 10 inches (0.25 m). This deviation has an effect similar to Case C.8 which looks at a different RCIC pump curve, in that it affects the duration and frequency of RCIC duties while dc power is available, and thus the vessel level at the assumed time of battery depletion. Case C.8 showed a modest change in accident timing. To evaluate the effect of this setpoint deviation, Case 1 has been rerun with the correct setpoint. Relative to the results of Case 1 as documented below (with the low setpoint), with corrected setpoint the end of the last RCIC duty cycle occurs 0.65 hours later, and core damage (identified with the PCT exceeding 1478K) occurs 0.97 hours later.

start of the transient. These last three systems do, however, operate during the pre-transient steady state, whose duration is 120 seconds. It is assumed that the Automatic Depressurization System (ADS) is unavailable because of the power loss or, alternatively, because operators act to inhibit ADS activation since the RCIC is to be used. The SRVs are assumed to cycle at their actuation setpoint indefinitely, assuming that the SRVs will not fail while cycling.

Table C.1 Assumptions Applicable to SBOs with the RCIC Operating until Battery Depletion

Reactor Coolant System	Relief valves are allowed to cycle indefinitely without failing. (A sensitivity case considers one stuck-open relief valve.)
ECCS	None available except RCIC.
Operator Actions	Operators are assumed to act to inhibit automatic depressurization. A sensitivity case considers operator control of RCIC.
Other	Batteries deplete at 8 hours, failing the RCIC. Sensitivity cases consider other battery depletion times. Another case considers that the RCIC may continue to run in an uncontrolled mode after battery depletion.

The 8-hour battery depletion time may be characterized as representing the beginning-of-battery-life value or the value that describes mid-battery-life with effective load shedding. Also note that 8 hours is the Peach Bottom SBO coping time given in NUREG–1776, though this is a coincidence and not part of the sequence definition given that the licensing-basis coping time involves the use of alternate alternating current after 1 hour, whereas here the sequence being modeled is one where the operators fail to set up the alternate alternating current source within the required time (or it is unavailable) [C.1]. Moreover, the 8-hour battery depletion time makes the calculation a hybrid between the following two SPAR sequences:

- LOOP-xx-39-30 with a core damage frequency of approximately 10^{-8} per year; 2-hour battery depletion time.

- LOOP-xx-39-9 representing successful operator actions to extend the RCIC operation beyond battery depletion. This is a non-minimal sequence in the base model because the action is assumed to fail. This sequence also includes failure branches for containment venting and late injection.

These remarks are based on the v8.19 (March 2011) version of the Peach Bottom 2 SPAR model.

The assumption that the SRVs do not become stuck-open is made because the relevant sequence (LOOPxx-39-59-6) is roughly two orders of magnitude lower in frequency than the one mentioned above. However, a sensitivity case considers the implication of failure of one SRV in open position following repeated cycling.

As described in Section 2.3 of the main report, the model includes automatic and manual modes for the RCIC. In the automatic mode and while there is power, the RCIC turns on and off between low and high vessel water levels with a 2.8 m deadband. While on, it pumps at the maximum rate. If the RCIC mode is specified as automatic, it stops pumping upon loss of power, which in this scenario occurs at 8 hours due to battery depletion. In the manual mode, while there is power, operators throttle the pump so that it operates continuously at a variable rate, thus maintaining a constant water level. If the RCIC mode is specified as manual, after battery depletion the RCIC turbine/pump operation continues at the speed it had when the batteries became depleted. The base case supposes that the RCIC mode is automatic. Hence, RCIC injection stops at 8 hours in the base case. A sensitivity case assumes that the RCIC mode is manual, which allows it to continue to run until it trips on one of the conditions described in Section 2.3 of the main report.

To investigate the variability of the predictions, ten sensitivity cases are defined in addition to the base case. The highlighted entries in Table C.2 show how each sensitivity case differs from the base case as follows:

- Case C.1 is the base case. The automatic mode is specified for the RCIC; hence, it fails at 8 hours because of depleted batteries.
- Case C.2 specifies the manual mode for the RCIC, allowing it to run on after 8 hours at a fixed rate until tripped by other conditions.
- Case C.3 repeats the base case, except that it assumes that the lowest-setpoint SRV (FL359) becomes stuck open after excessive cycling.
- Cases C.4 and C.5 repeat the assumptions of the base case, but with 2 and 5 hours as the battery depletion time, respectively.
- Case C.6 repeats the base case, but supposes that the event occurs while the reactor is operating at 90% power. The decay heat is scaled relative to the base case.
- Case C.7 supposes that the lowest-setpoint SRV (FL359) may not open. Otherwise, Case C.7 is similar the base case.
- Case C.8 reduces the maximum RCIC flow rate to 90% of its base-case value.
- Case C.9 decreases the flow losses and the momentum exchange length, in the flow paths that model the two SRVs with the lowest opening pressures.
- Case C.10 applies a factor of 0.8 to the built-in correlation for the heat transfer coefficients that apply to the nucleate boiling curve for pool boiling and the transition boiling curve for film boiling.
- Case C.11 sets the initial temperature of the CST to 283 K; other calculations use 300 K.

The base-case calculation for the SBO with the RCIC operation until battery depletion is described in detail in Section C.2.2.1. Sections C.2.2.2 through C.2.2.11 describe the sensitivity calculations, typically in less detail. Summary discussion pertaining to this scenario appears in Section C.2.3 (which also appears in the main report).

Table C.2 Sensitivity Cases for Peach Bottom SBO with RCIC Operation until the Battery Depletion

Case	RCIC Mode(1)	SRVs Are Stuck Open(2)	Battery Depletion Time, Hours	Factor for Decay Heat (3)	Number of Unavailable SRVs(4)	Factor for RCIC Flow Rate(5)	SRV Parameters(6)	Factor for Pool Boiling Model(7)	Initial CST Temperature
C.1 (Base)	Automatic	No	8	1.0	0	1.0	Base	1.0	300K
C.2	Manual	No	8	1.0	0	1.0	Base	1.0	300K
C.3	Automatic	Yes	8	1.0	0	1.0	Base	1.0	300K
C.4	Automatic	No	2	1.0	0	1.0	Base	1.0	300K
C.5	Automatic	No	5	1.0	0	1.0	Base	1.0	300K
C.6	Automatic	No	8	0.9	0	1.0	Base	1.0	300K
C.7	Automatic	No	8	1.0	1	1.0	Base	1.0	300K
C.8	Automatic	No	8	1.0	0	0.9	Base	1.0	300K
C.9	Automatic	No	8	1.0	0	1.0	See note (6)	1.0	300K
C.10	Automatic	No	8	1.0	0	1.0	Base	0.8	300K
C.11	Automatic	No	8	1.0	0	1.0	Base	1.0	283K

1 In automatic mode, RCIC stops when batteries deplete. In Case C.2, manual mode is specified, allowing RCIC to run on at a constant rate after battery depletion.
2 In Case C.3, the SRV of lowest opening pressure (FL359) is stuck open after cycling 187 times.
3 In Case C.6, a factor of 0.9 is applied to the base-case decay heat curve. This factor models occurrence of the transient while the reactor is at 90% of full power.
4 In Case C.7, the SRV of lowest opening pressure (FL359) is assumed to be unavailable to open.
5 In Case C.8, the base-case RCIC maximum flow rate, 37.7 kg/s, is reduced to 34.0 kg/s.
6 The parameters considered are K-loss and choked flow discharge coefficients, and the momentum exchange length. The base-case values are 5.0, 1.0, and 17.9 m, respectively. In case C.9, the respective values are 3.0, 1.5, and 0.5 m.
7 In Case C.10, a factor of 0.8 is applied to the built-in correlations for pool and film boiling heat transfer coefficients.

C.2.2 Results

C.2.2.1 Results of Case C.1 (Base Case)

The key events for this scenario are listed in Table C.3 and illustrated in Figures C.1.1.1 to C.1.1.4.

Table C.3 Timing of Key Events in the Base-Case Calculation

Event Description	**Time (hours)**
Loss of onsite power except for batteries; trip of reactor, feedwater, recirculation pumps, and CRD hydraulic system	0 s
Beginning of first (of four) RCIC duties	0.18 (10.5 minutes)
End of last (of four) RCIC duties	7.1
Battery depletion (RCIC becomes unavailable)	8.0
Start of core uncovery	9.6
Core-exit coolant exceeds 922 K (1,200 °F)	10.44
Maximum cladding temperature exceeds 1,478 K (2200 °F)	10.99
End of calculation	11.9

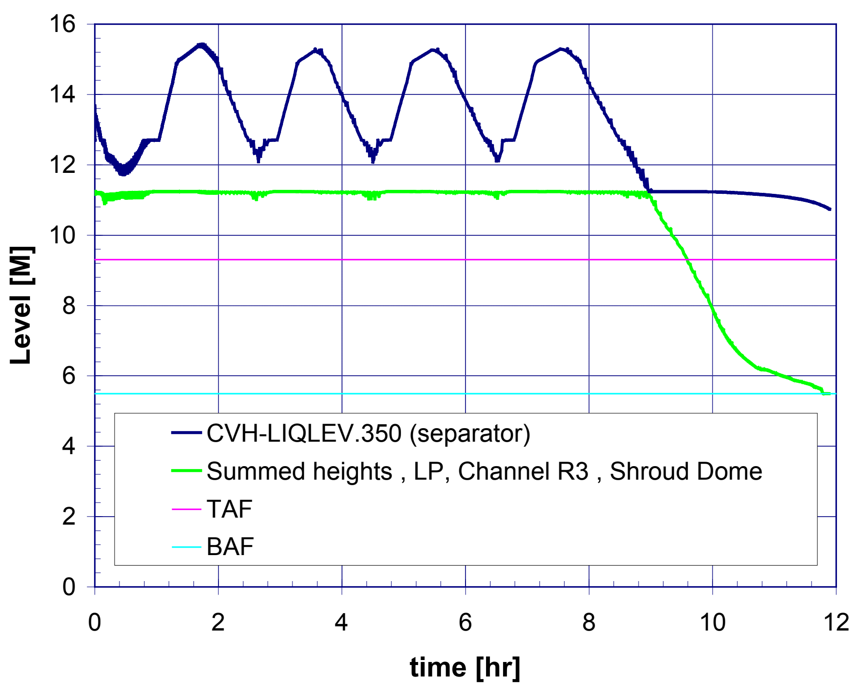

Figure C.1.1.1 Water Levels in the Vessel (Case C.1)

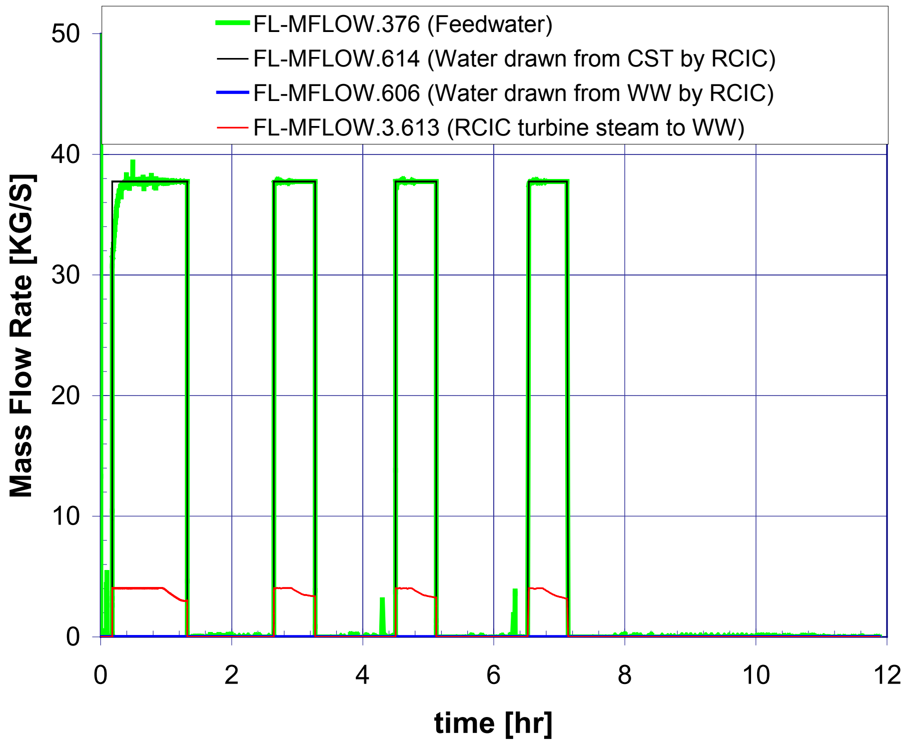

Figure C.1.1.2 Injection by RCIC (Case C.1)

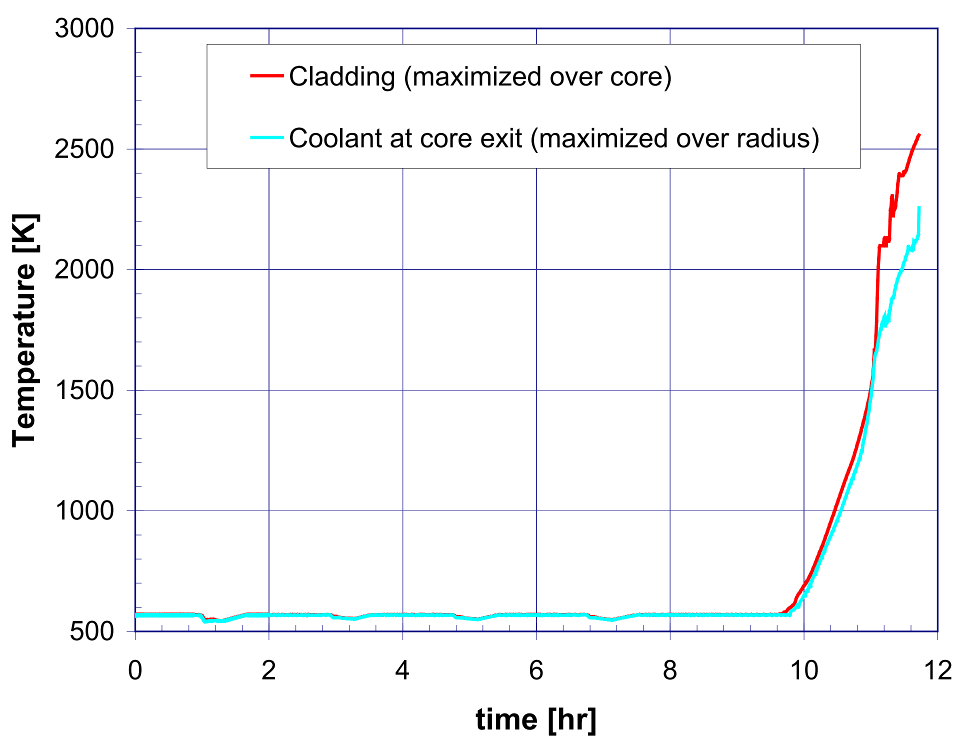

Figure C.1.1.3 Maximum Cladding and Core Exit Coolant Temperatures (Case C.1)

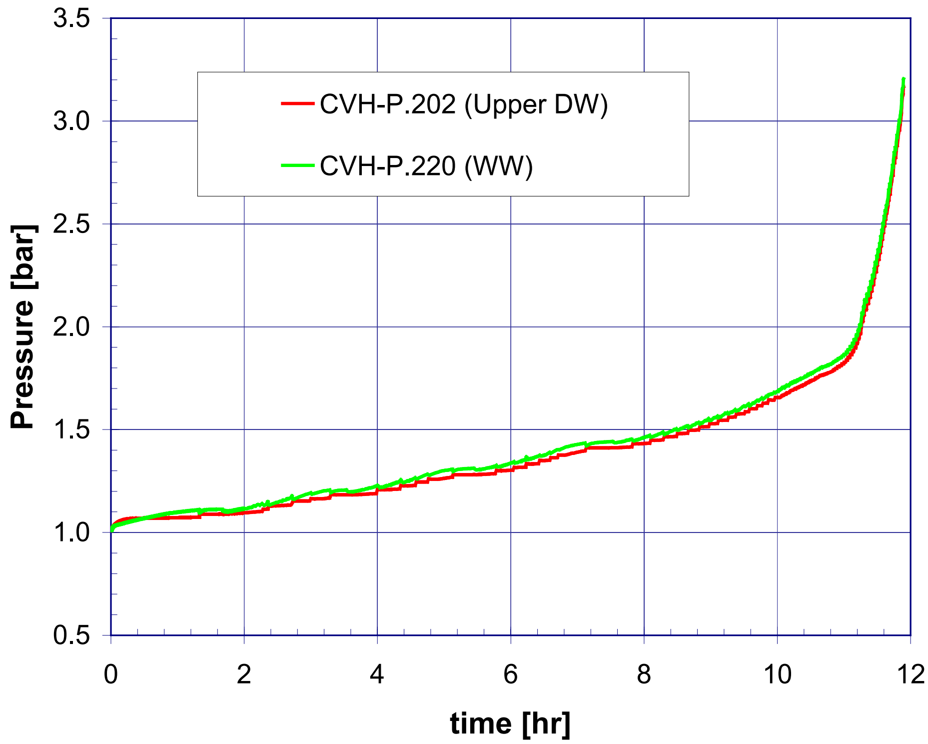

Figure C.1.1.4 Containment Pressures (Case C.1)

Loss of onsite power (except for batteries) at *t* = 0 seconds initiates the transient after a pre-transient steady-state period lasting 120 seconds. The batteries are credited to support only the RCIC system, and they deplete at 8 hours.

In the assumed automatic mode, RCIC turns on at low vessel water level and off at high vessel water level. Figure C.1.1.1 shows vessel water levels, and Figure C.1.1.2 shows the RCIC injection rate. Note that water pumped by the RCIC enters the vessel through MELCOR flow paths that model the FW piping. The figure legend labels the flow into the vessel as “Feedwater” (green curve) in reference to this modeled FW piping, although FW, as a system, is unavailable in this scenario. Two curves showing the RCIC-provided water entering the FW piping are also shown. One corresponds to a flow path coming from the CST; the other corresponds to a flow path coming from the wetwell. Only the first is finite in this calculation because depletion of the CST does not occur.

In Figure C.1.1.1, the water level in the core is defined by summing the heights of the columns of swollen water from the lower plenum to the shroud dome by way of the third of five computational core-region rings. The sum is not carried to higher levels because the top of the shroud dome control volume is higher than the bottom of the separator control volume. The separator is the region whose range of elevation contains the elevations of the RCIC control points, and its level can be considered to define the vessel water level until the other curve (shown in green in Figure C.1.1.1) begins to decrease, at about 9 hours. The RCIC control is based on the water level in the downcomer.

The RCIC turns on for the first time at 10.5 minutes (Figure C.1.1.2). Altogether, the RCIC is on for four duties occurring at roughly regular intervals during the time from 10.5 minutes until 7.1 hours. The average duration of a duty is 45 minutes. The last duty ends at 7.1 hours. By the time another duty would be called for, the batteries have depleted and the RCIC cannot restart.

Outflows through the SRVs begin at 30 seconds. These outflows continue steadily, except during the times that the RCIC injects. There are 11 flow paths used to model the SRVs, but only two of them—FL359 and FL360—ever open. Moreover, there is very little flow through FL360 after 30 minutes. At the end of the calculation (11.9 hours), FL359 has opened or closed 1426 times, and FL360 has opened or closed 48 times.

The core begins to uncover at about 9.6 hours. The temperature of the coolant at the core exit exceeds 922 K (1,200 °F) at 10.44 hours; the first gap release is at 10.68 hours; and the maximum cladding temperature exceeds 1,478 K (2,200 °F) at 10.99 hours. Figure C.1.1.3 shows the core exit temperature and the maximum cladding temperature. Failures of the core support plate begin at 11.25 hours. The calculation is stopped at 11.9 hours, since gross core damage is underway.

Figure C.1.1.4 shows the pressure in the containment. The maximum temperature of water in the wetwell is 105 °C (this occurs at the end of the calculation). The wetwell water is always subcooled by at least 13 K.

C.2.2.2 Results of Case C.2 (RCIC in Manual Mode)

This case assumes that the RCIC operates in manual mode, which allows it to run on after battery depletion until it is tripped by other conditions. The key events for this scenario are listed in Table C.4 and are illustrated in Figures C.1.2.1 to C.1.2.3.

Table C.4 Timing of Key Events – Case 2

Event Description	**Time (hours)**
Loss of onsite power (except for batteries); trip of reactor, feedwater, recirculation pumps, CRD hydraulic system	0
Beginning of RCIC injection (at maximum rate) due to low water level	0.18 (10.5 minutes)
End of RCIC injection due to high water level	1.36
Resumption of RCIC injection with variable throttling to maintain constant water level (RCIC manual mode)	1.89
Battery depletion (RCIC injection rate locked in)	8.0
RCIC trip due to flooding of the main steam line	13.87
Start of core uncovery	16.9
Core-exit coolant exceeds 922 K (1,200 °F)	17.92
Maximum cladding temperature exceeds 1,478 K (2,200 °F)	18.57
End of calculation	20.0

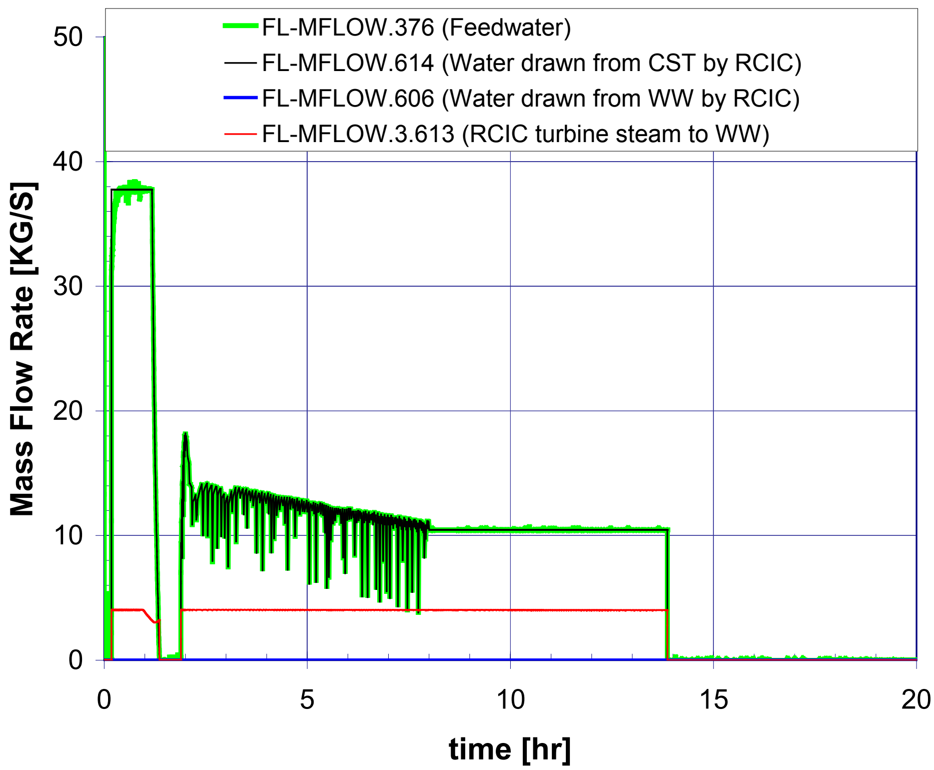

Figure C.1.2.1 Injection by RCIC (Case C.2)

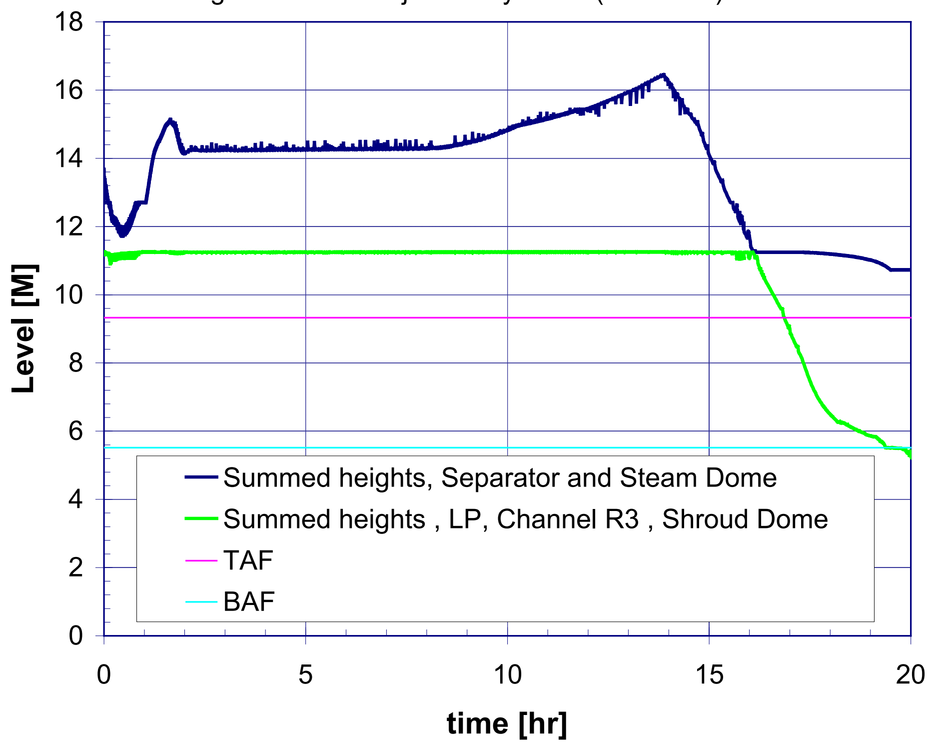

Figure C.1.2.2 Water Levels in the Vessel (Case C.2)

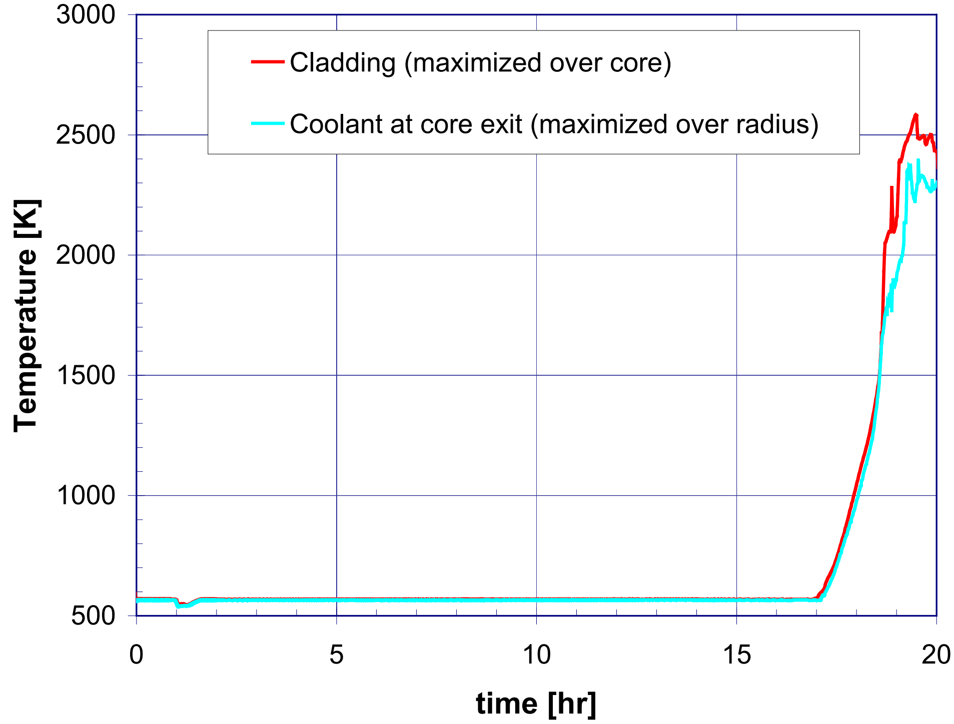

Figure C.1.2.3 Maximum Cladding and Core Exit Coolant Temperatures (Case C.2)

The RCIC manual mode versus the automatic mode, as represented in the model need some discussion. In automatic mode, the RCIC runs only at full speed but turns on and off between wide water level limits. In manual mode, on the other hand, before battery depletion, operators throttle the RCIC so that it runs continuously at a variable rate and maintains a constant level in the vessel. However, the logic specified by the model causes the first RCIC activity to be as if the mode were automatic, regardless of the mode specified by the user. That is, the first RCIC startup occurs on low level, and the RCIC runs at full speed until it shuts off on high level. The RCIC startup with throttling begins subsequently, as the falling level approaches the level that operators are assumed to maintain when they operate the RCIC manually.

Switching the user-specified mode from automatic to manual during the transient is not considered, for the following reason. The manual throttling is implemented in the model using a look-up table that sets the RCIC injection rate as a linearly decreasing function of water level. The assumption is that in the manual mode following a battery depletion, the RCIC maintains the rate it had at the instant the battery was depleted. This assumption makes the water level a controlling variable for the RCIC injection rate at all times after the depletion. If manual operation had been in effect only for a short time after the switch from a hypothetical period of automatic operation, the water level could be any value within the deadband. Assuming that the manual mode was in effect from the start of the transient ensures that at the time of battery depletion, the level will remain under control near the value that yields a quasi-steady state, wherein the RCIC injection rate equals the boil-off rate. After battery depletion, this RCIC injection rate will be locked in, but it will continue to equal the boil-off rate, at least approximately, for some time. This condition/assumption will apply only if the transient is not so short that the time of RCIC throttling is never reached.

Figure C.1.2.1 depicts the RCIC injection rate. The period of a throttled RCIC injection begins at 1.89 hours. After battery depletion at 8 hours, the RCIC holds the rate it had at 8 hours. Because this rate exceeds the boil-off rate as the decay heat decreases, the level in the vessel rises (see Figure C.1.2.2). The steam line floods at 13.87 hours, which trips the RCIC and makes it unavailable for the rest of the scenario. Other RCIC trip conditions, which are listed in Section 2.3 of the main report, include a high turbine exhaust pressure and a pump bearing failure. These conditions have not yet been attained at the time of the trip due to flooding. Note that the MELCOR model does not account for the effect of RCIC room heatup on pump bearing temperature.

The core begins to uncover at about 16.9 hours. The temperature of the coolant at the core exit exceeds 922 K (1,200 °F) at 17.92 hours. The first gap release is at 18.20 hours, and the maximum cladding temperature exceeds 1,478 K (2,200 °F) at 18.57 hours. Figure C.1.2.3 shows the core exit temperature and the maximum cladding temperature. Failures of the core support plate begin at 18.87 hours. The calculation is stopped at 20 hours because gross core damage is underway.

C.2.2.3 Results of Case C.3 (One Stuck-Open SRV)

This case repeats the base case (C.1), except that it assumes that the lowest setpoint SRV (FL359) is stuck in the open position after excessive cycling. The cause of this occurrence is the 50 percent assumed cumulative probability of becoming stuck and a per-demand-to-close probability of 0.0037. The cumulative sticking probability is assumed for this calculation. The sticking probability per demand to close is the deck default, whose value is discussed in the SNL Peach Bottom model documentation. The sticking assumptions are equivalent to assuming that FL359 becomes stuck after opening 187 times. This event occurs at 2.45 hours. Until then, the sensitivity case is identical to the base case.

Figure C.1.3.1 shows the vessel pressure, which falls after the SRV becomes stuck in the open position. At 3.88 hours, the pressure falls below 0.62 MPa and trips the RCIC. Although the pressure later recovers somewhat, the model assumes that the RCIC cannot be restarted.

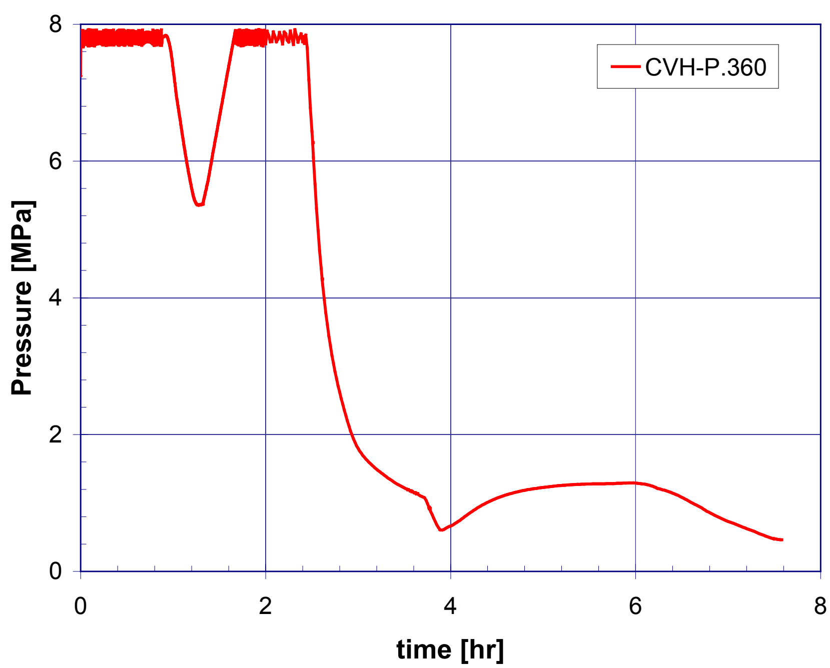

Figure C.1.3.1 Pressure in the Reactor Pressure Vessel (Case C.3)

Figures C.1.3.2 and C.1.3.3 show the water level in the vessel and the cladding and coolant temperatures, respectively. The core begins to uncover at about 5.9 hours. The temperature of the coolant at the core exit exceeds 922 K (1,200 °F) at 6.80 hours; the first gap release is at 6.96 hours; and the maximum cladding temperature exceeds 1,478 K (2,200 °F) at 7.15 hours. Failures of the core support plate begin at 7.49 hours. The calculation is stopped at 7.6 hours, because gross core damage is underway.

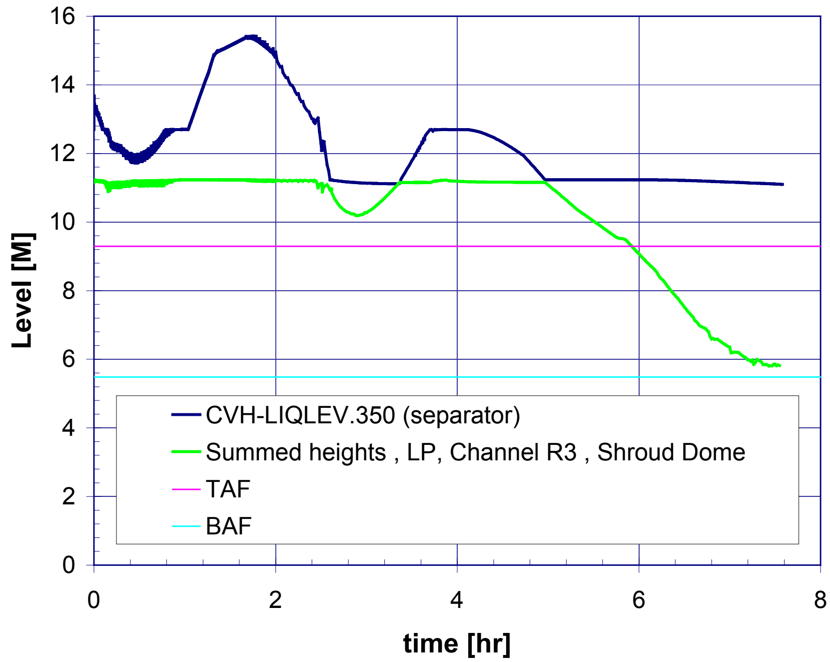

Figure C.1.3.2 Water Levels in the Reactor Pressure Vessel (Case C.3)

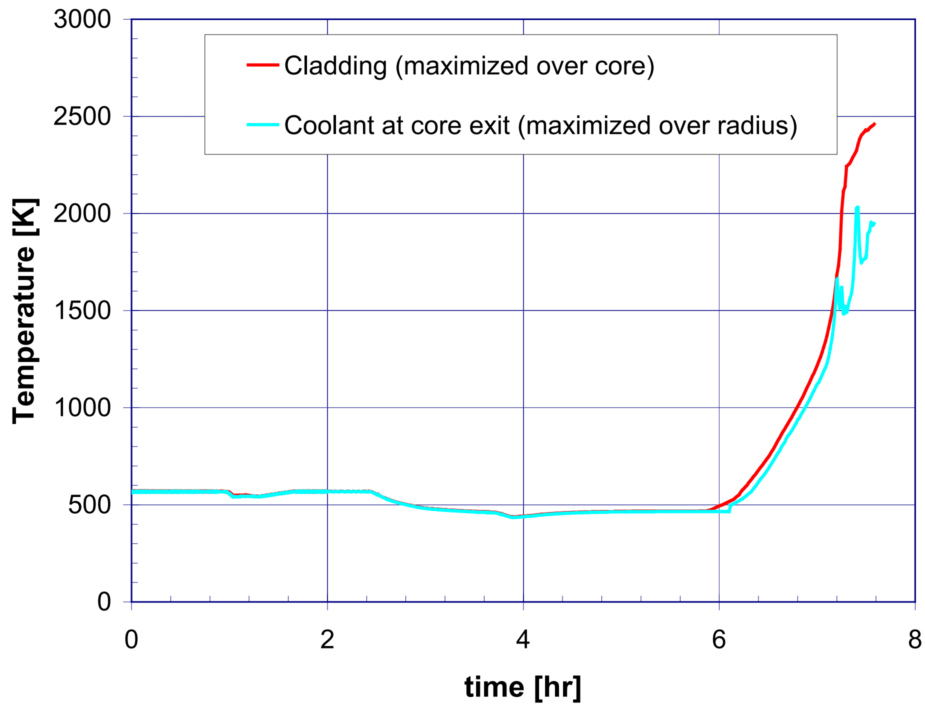

Figure C.1.3.3 Maximum Cladding and Core Exit Coolant Temperatures (Case C.3)

C.2.2.4 Results of Case C.4 (Batteries Deplete at 2 Hours)

This case repeats the base case (C.1), except that battery depletion is assumed to occur at 2 hours instead of at 8 hours.

Figures C.1.4.1 through C.1.4.3 show the RCIC injection rate, vessel water level, and clad/coolant temperatures. The first RCIC duty ends at 1.32 hours because of the normal shutoff on high level. The next duty would have started at around 2.5 hours, but since that time is after the battery has depleted, there is no RCIC activity at any time after 1.32 hours.

The core begins to uncover at about 3.3 hours. The temperature of the coolant at the core exit exceeds 922 K (1,200 °F) at 3.90 hours; the first gap release is at 4.05 hours; and the maximum cladding temperature exceeds 1,478 K (2,200 °F) at 4.27 hours. Failures of the core support plate begin at 4.43 hours. The calculation is stopped at 5 hours, because gross core damage is underway.

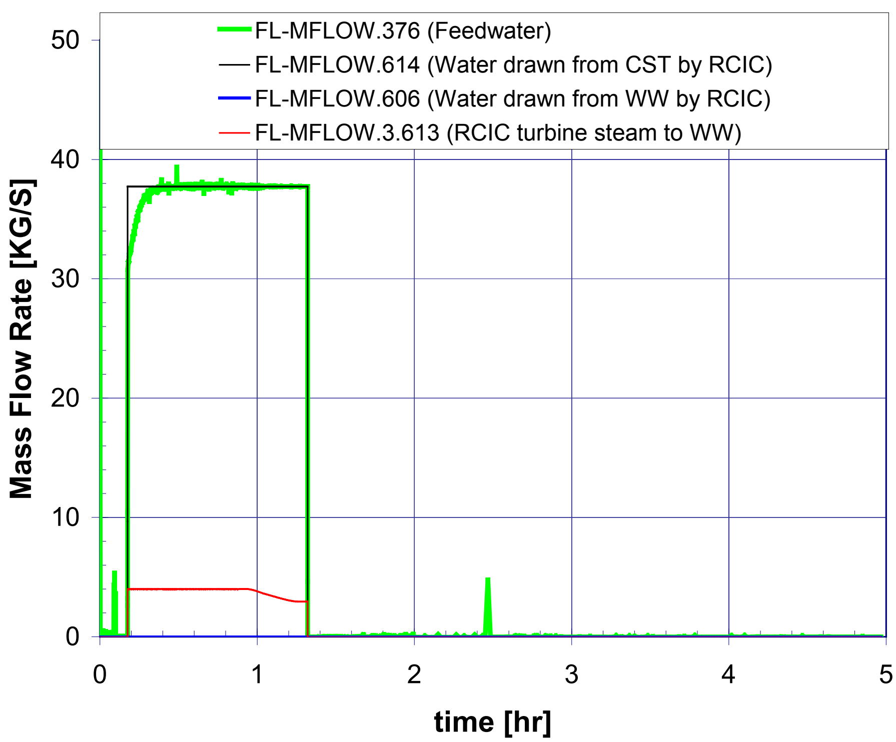

Figure C.1.4.1 Injection by RCIC (Case C.4)

Regarding the figure above, the brief jump in the signature at approximately 2.5 hours corresponds to the uncovering of the vessel-side flow junction in the MELCOR model, and should be viewed as a numerical (rather than physical) event.

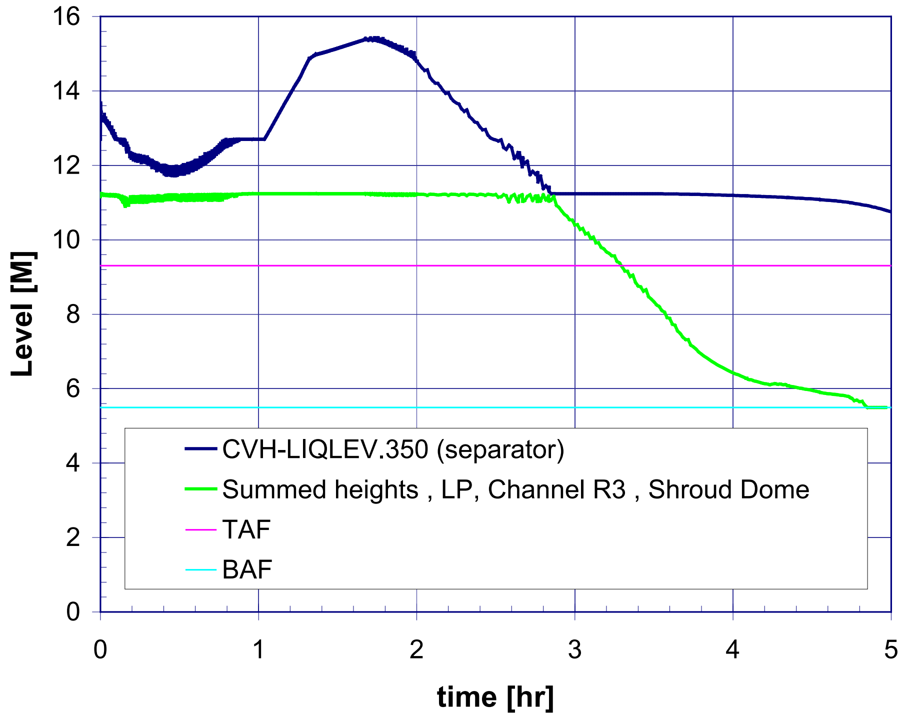

Figure C.1.4.2 Water Levels in the Vessel (Case C.4)

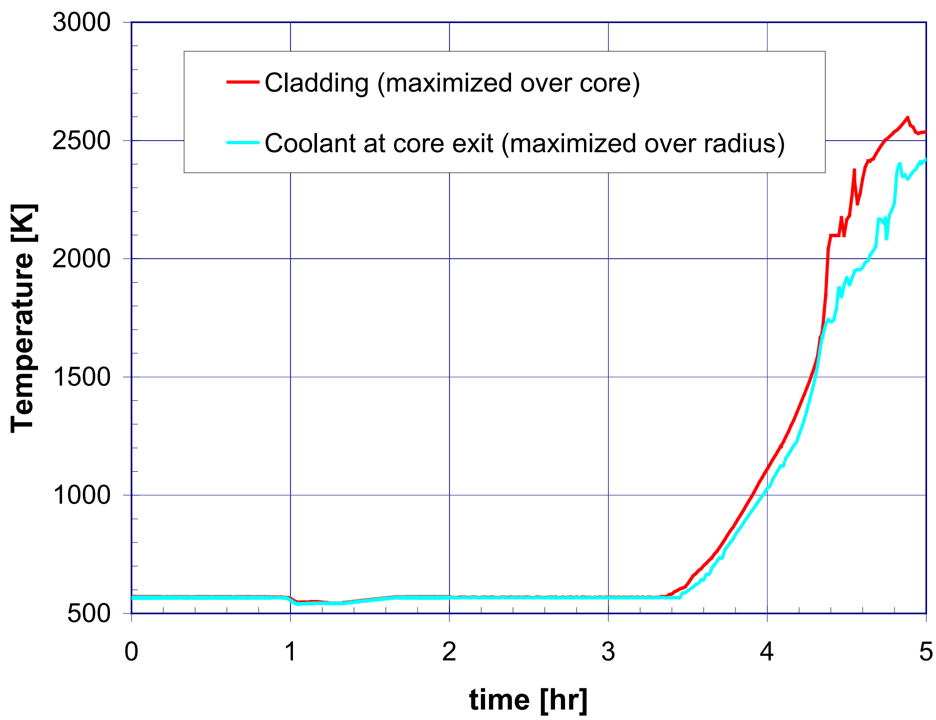

Figure C.1.4.3 Maximum Cladding and Core Exit Coolant Temperatures (Case C.4)

C.2.2.5 Results of Case C.5 (Batteries Deplete at 5 Hours)

This case repeats the base case (C.1), except that battery depletion is assumed to occur at 5 hours instead of at 8 hours.

Figures C.1.5.1 through C.1.5.3 show the RCIC injection rate, vessel water level, and clad/coolant temperatures. At 5 hours, the third RCIC duty is in progress (the RCIC is injecting at the maximum rate). The RCIC stops at this time and never starts again.

The core begins to uncover at about 6.8 hours. The temperature of the coolant at the core exit exceeds 922 K (1,200 °F) at 7.58 hours; the first gap release is at 7.78 hours; and the maximum cladding temperature exceeds 1,478 K (2,200 °F) at 8.03 hours. Failures of the core support plate begin at 8.22 hours. The calculation is stopped at 10 hours, because gross core damage is underway.

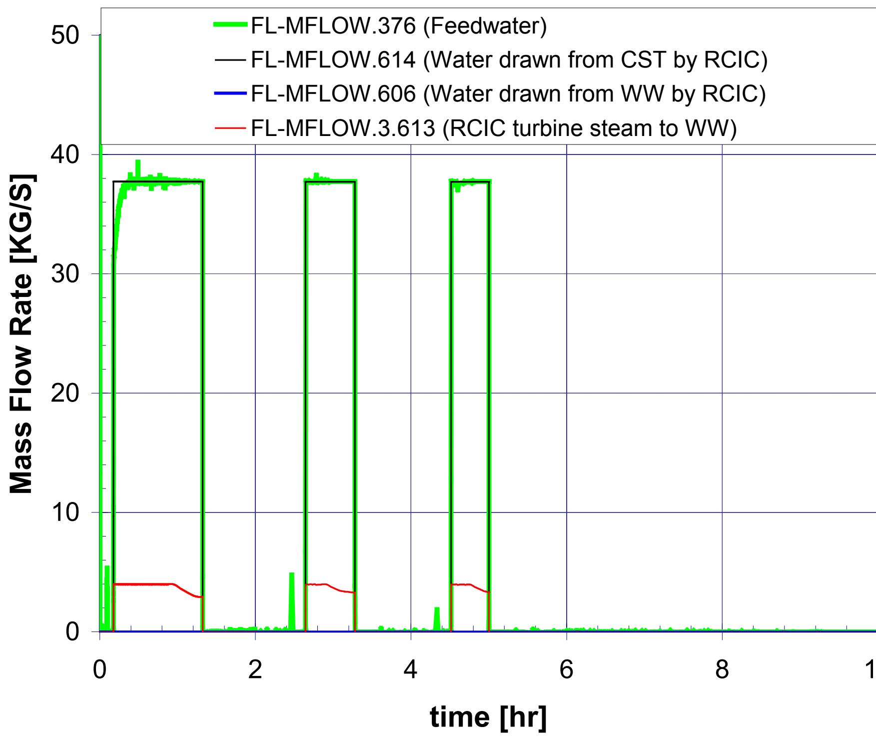

Figure C.1.5.1 Injection by RCIC (Case C.5)

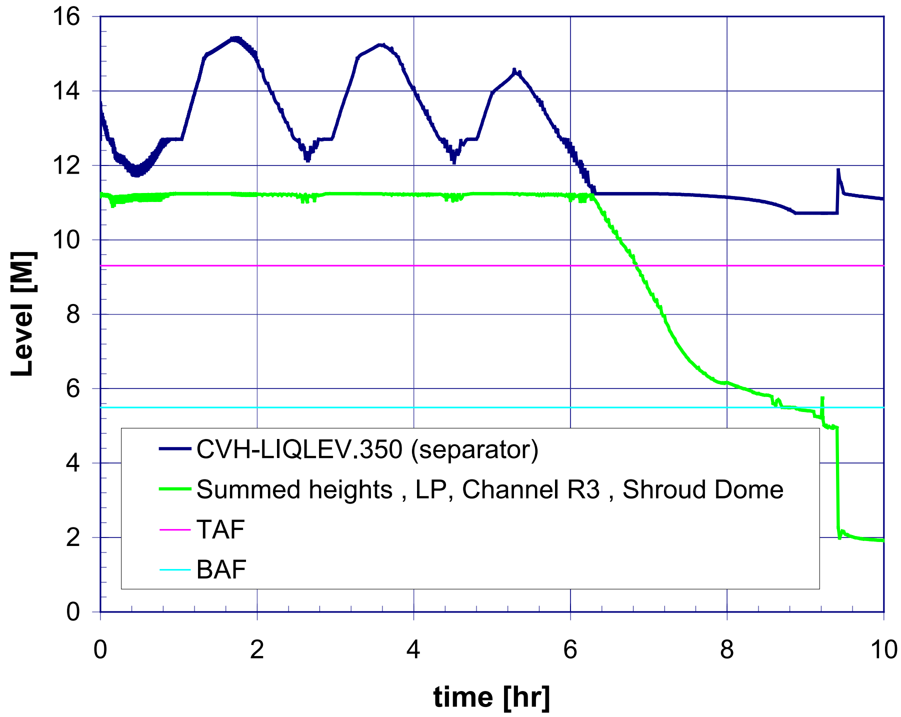

Figure C.1.5.2 Water Levels in the Vessel (Case C.5)

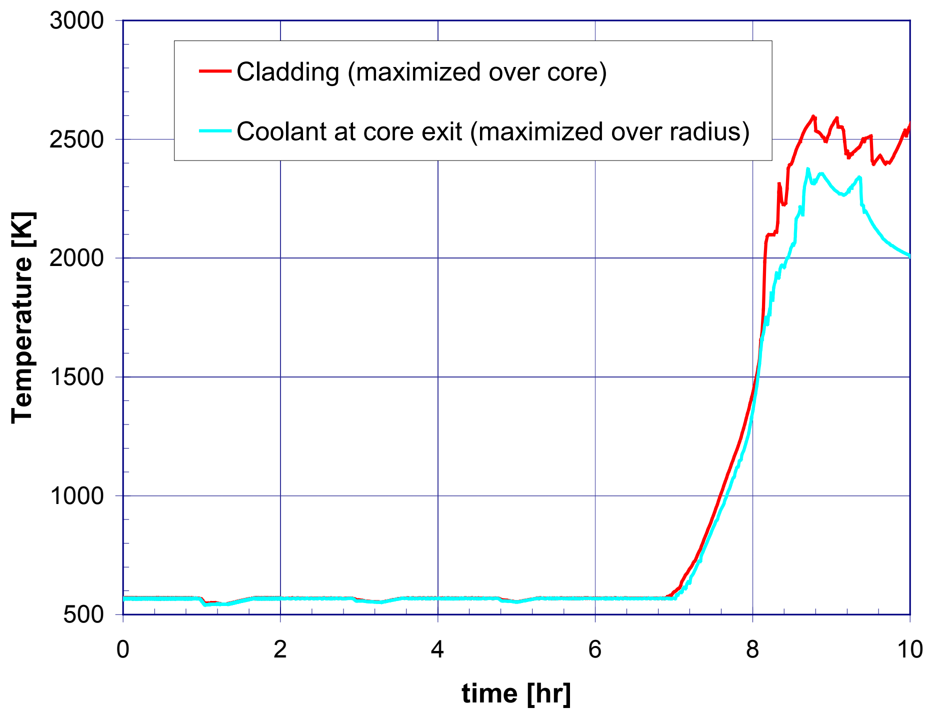

Figure C.1.5.3 Maximum Cladding and Core Exit Coolant Temperatures (Case C.5)

C.2.2.6 Results of Case C.6 (90 percent Decay Heat)

This case repeats the base case (C.1), except that the decay heat and likewise the fission heat before the scram are 90 percent of the base-case values.

Figure C.1.6.1 shows the RCIC injection rate. As in the base case, the RCIC finishes four complete duties and is idle on high vessel level at 8 hours, at which time the batteries deplete and preclude any further RCIC activity. Because of the lower boil-off rate, the first RCIC turn-on and the last turn-off occur at slightly later times than in the base case, and the average duration of a duty is shortened slightly from 45 minutes in the base case to 42 minutes in the sensitivity case. The onset of core damage is delayed for the same reason. Figures C.1.6.2 and C.1.6.3 show, respectively, the levels in the vessel and the cladding and coolant temperatures. The core begins to uncover at about 10.0 hours. The temperature of the coolant at the core exit exceeds 922 K (1,200 °F) at 10.95 hours; the first gap release is at 11.23 hours; and the maximum cladding temperature exceeds 1,478 K (2,200 °F) at 11.58 hours. The calculation is stopped at 11.9 hours.

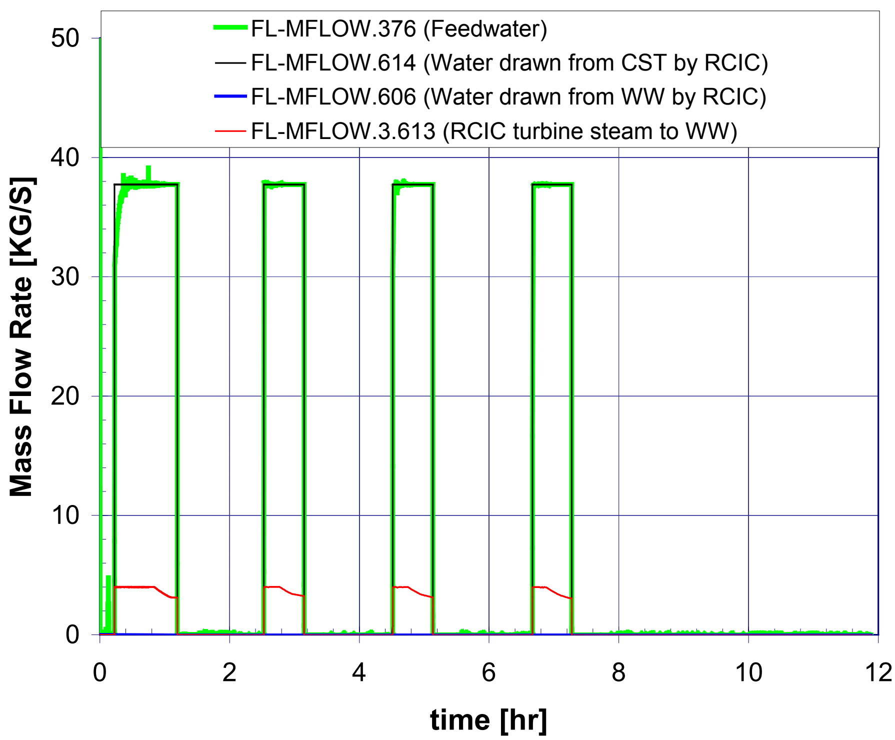

Figure C.1.6.1 Injection by RCIC (Case C.6)

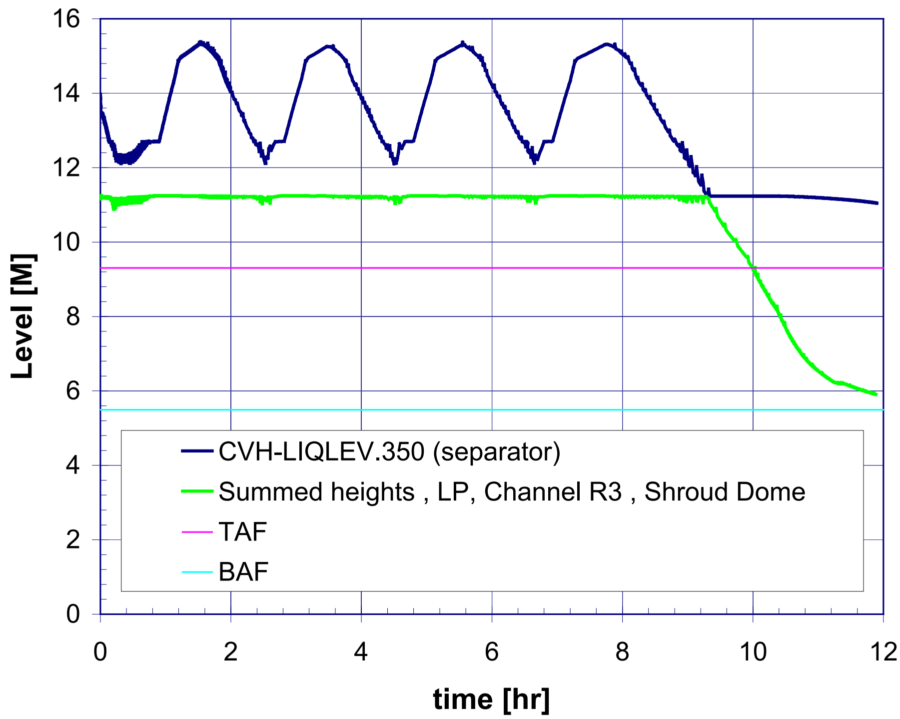

Figure C.1.6.2 Water Levels in the Vessel (Case C.6)

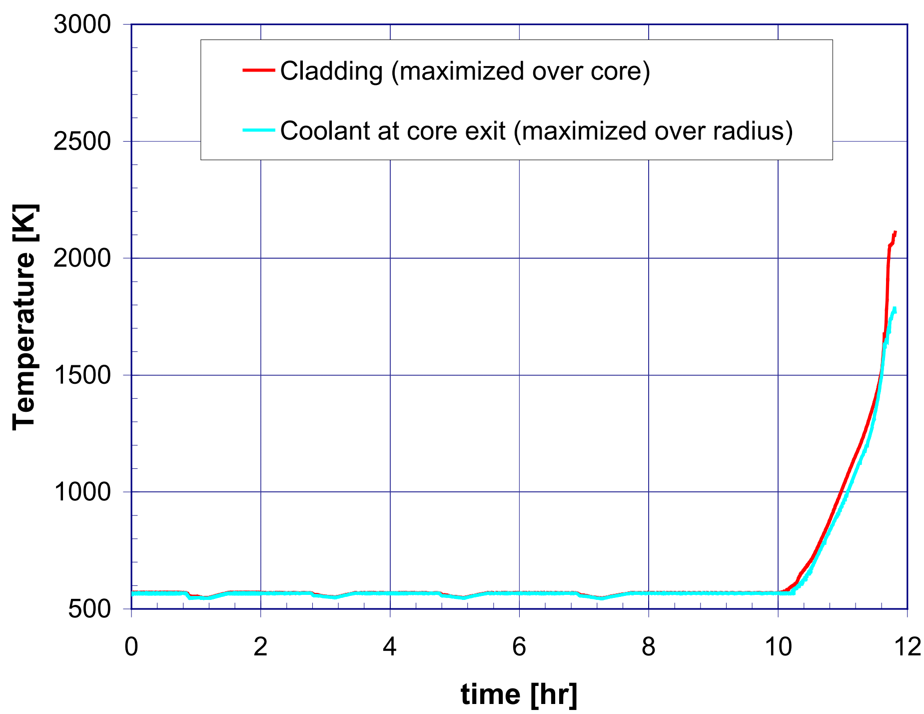

Figure C.1.6.3 Maximum Cladding and Core Exit Coolant Temperatures (Case C.6)

C.2.2.7 Results of Case C.7 (One Unavailable SRV)

This case repeats the base case (C.1), except that the SRV with the lowest opening pressure (FL359, opening at 7.815 MPa) is assumed to be stuck closed throughout the scenario.

Whereas in the base case the vessel is relieved predominately by FL359, but with a small contribution from FL360; in the sensitivity case, the vessel is relieved predominately by FL360, but with a small contribution from FL361. The other eight flow paths that model SRVs—flow paths FL362 through FL369—do not open in either calculation. The opening pressure of FL360 is just 1 psi higher than that of FL359, and that of FL361 is likewise just 1 psi higher than that of FL360. (A 1 psi offset is assumed in the MELCOR model. The actual design (FSAR) setpoints for these valves is the same value.) Therefore, the results of this sensitivity case are very similar to those of the base case. At the end of the calculation (11.5 hours), FL360 has opened or closed 1,292 times, and FL361 has opened or closed 28 times.

Event timings characteristic of core damage are similar to those of the base case. The core begins to uncover at about 9.6 hours. The temperature of the coolant at the core exit exceeds 922 K (1,200 °F) at 10.47 hours; the first gap release is at 10.70 hours; and the maximum cladding temperature exceeds 1,478 K (2,200 °F) at 11.01 hours. Failures of the core support plate begin at 11.28 hours. The calculation is stopped at 11.5 hours, because gross core damage is underway.

C.2.2.8 Results of Case C.8 (Reduced RCIC Flow Rate)

This case repeats the base case (C.1), except that the maximum RCIC flow rate is reduced to 90 percent of its base-case value.

As in the base case, the RCIC completes four duties. Because of the reduced flow rate, the duration of the duties is longer. The last duty ends at 7.72 hours instead of at 7.12 hours as in the base case. This leads to behavior in the sensitivity case very similar to that of the base case, but with a delay of about 0.6 hours. Figure C.1.8.1 compares the cladding and core exit temperatures predicted by the sensitivity calculation with the base-case results.

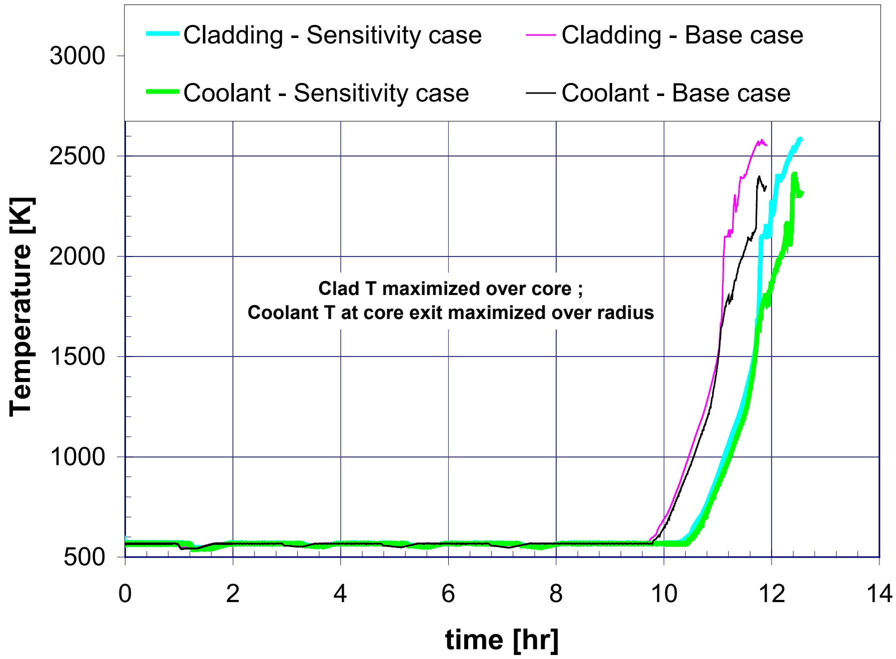

Figure C.1.8.1 Maximum Cladding and Core Exit Coolant Temperatures (Case C.8)

C.2.2.9 Results of Case C.9 (SRV Parameters)

This case considers uncertainty in some of the parameters that describe the SRVs. The two SRVs with the lowest opening pressures are considered, because in the base case (C.1) they are the only SRVs that ever open. They are modeled by FL359 and FL360. The base-case values for the K-loss coefficient, the choked flow discharge coefficient, and the momentum exchange length are 5.0, 1.0, and 17.9 m, respectively. The last parameter is left unassigned by user input in the base-case model and has therefore been assigned by MELCOR. The large value reflects the long flow length of the SRV flow paths. The sensitivity case assigns these three parameters (for FL359 and FL360 only) as 3.0, 1.5, and 0.5 m. The changes made to the K-loss and choked flow discharge coefficients both tend to increase the flow. The shorter momentum exchange length reduces the ability of SRV gas flows to entrain liquid, which probably has been overestimated in the base case[26].

As in the base case, only FL359 and FL360 ever open. Figure C.1.9.1 compares the cladding and core exit temperatures predicted by the sensitivity calculation with the base-case results. The differences are small, which is understandable since the considered differences only influence the duration that the SRVs are open and the flow rate during that time, but not the integrated flow (i.e., because their pressure setpoints control the opening and closing of the SRVs).

[26] MELCOR has a simple entrainment model that uses the momentum exchange length, in addition to other user-specified inputs (junction opening heights). The latter determines whether liquid/gas may be drawn into a flow path at any given time (based on liquid level), and thereby whether entrainment/flooding is a possibility. Meanwhile, the momentum exchange length value affects entrainment and flooding (when they may occur) by controlling the balance between the resulting force between the two flows (which tends to couple them and reduce slip) and buoyancy (which tends to separate pool from atmosphere). A larger value tends to promote coupling between pool and atmosphere, while a smaller one tends to enhance their separation.

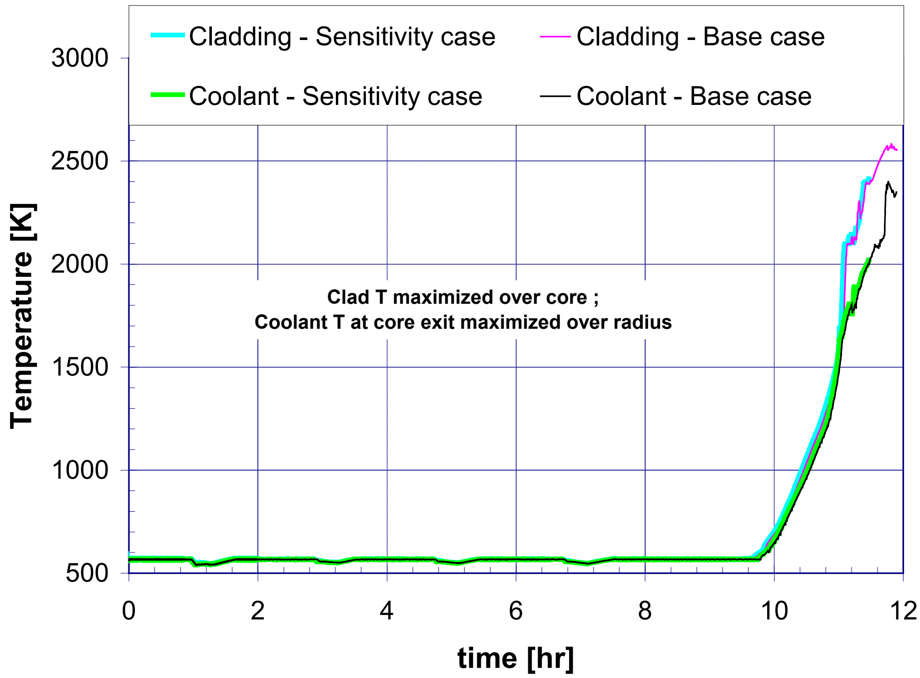

Figure C.1.9.1 Maximum Cladding and Core Exit Coolant Temperatures (Case C.9)

C.2.2.10 Results of Case C.10 (Heat Transfer Coefficients for Boiling)

This case repeats the base case (C.1), except that a factor of 0.8 is applied to the built-in correlation for the heat transfer coefficients that apply to the nucleate boiling curve for pool boiling (MELCOR sensitivity card 1241[1]) and the transition boiling curve for film boiling (MELCOR sensitivity card 1242[1]).

Figure C.1.10.1 compares the cladding and core exit temperatures predicted by the sensitivity calculation with the base-case results. The differences are very small.

C.2.2.11 Results of Case C.11 (Initial Temperature of Water in the CST)

This case repeats the base case (C.1), except that the initial temperature of the CST is set to 283 K (other calculations use 300 K).

Figure C.1.11.1 compares the cladding and core exit temperatures predicted by the sensitivity calculation with the base-case results. The sensitivity-case results are well described as a delayed version of the base-case results. The delay is about 0.4 hours.

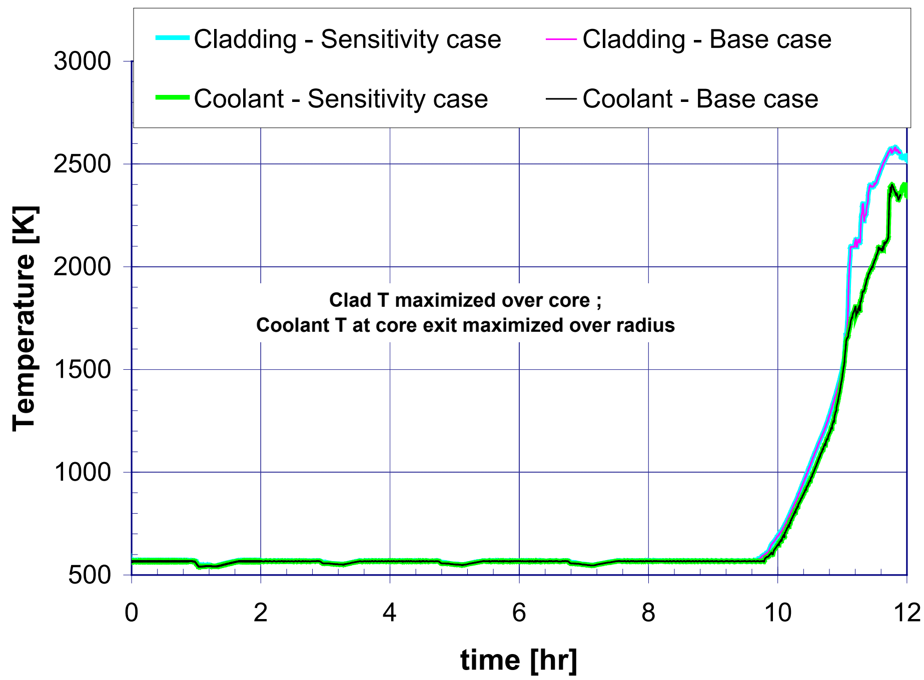

Figure C.1.10.1 Maximum Cladding and Core Exit Coolant Temperatures (Case C.10)

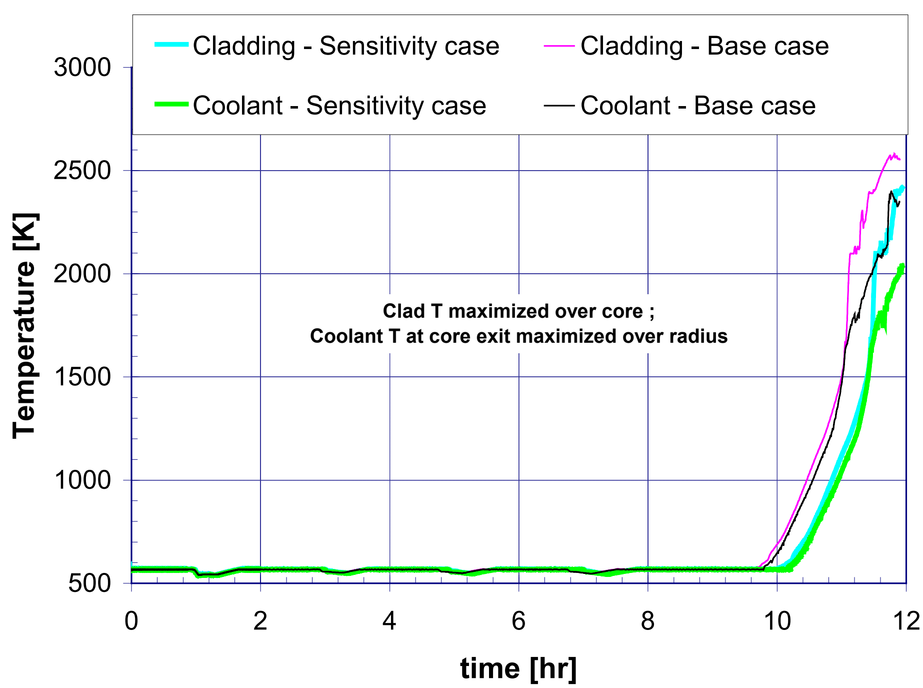

Figure C.1.11.1 Maximum Cladding and Core Exit Coolant Temperatures (Case C.11)

C.2.3 Summary: SBO with RCIC Scenario

In the base-case scenario (C.1), the unavailability of the RCIC after 8 hours (i.e., the assumed battery depletion time) results in core damage that starts at 10.99 hours, according to the criterion that the PCT exceeds 1,478 K (2200 °F). Two sensitivity cases (C.4 and C.5) are identical to the base case except that battery depletion is assumed to occur at 2 hours and at 5 hours. Core damage (according to the same criterion) occurs in these cases, at 4.27 and 8.03 hours, respectively.

One sensitivity case (C.2) considers the RCIC continuing to run after battery depletion in uncontrolled mode, until it is shut-off from the flooding in the steam lines at 13.87 hours. In this case, core damage occurs at 18.57 hours (according to the same criterion).

In the base case (C.1), the SRV with the lowest opening pressure opens and closes a total of 1,426 times by the end of the calculation (i.e., 11.9 hours). A sensitivity case (C.3) considers this valve to be stuck in the open position after opening 187 times. This event occurs at 2.45 hours. Core damage, according to the same criterion, occurs at 7.15 hours, after the RCIC became unavailable at 3.88 hours due to low vessel pressure caused by the stuck-open valve.

Another sensitivity case (C.7) assumes that the SRV with the lowest opening pressure fails in a closed position throughout the calculation. The results are similar to those of the base case, because the vessel is relieved by the other SRVs whose opening pressures are only slightly higher.

In a sensitivity case that uniformly reduces the decay heat to 90 percent of the base-case value (C.6), core damage according to the same criterion occurs at 11.58 hours.

Sensitivity cases that consider adjusting some parameters that affect flow through the SRVs (Case C.9) and the correlation for heat transfer coefficients during boiling (Case C.10) show results that are not significantly different from those of the base case.

See Section 4.3 of the main report for additional details.

C.3 Reference

C.1 W.S. Raughley, "Regulatory Effectiveness of the Station Blackout Rule," U. S. Nuclear Regulatory Commission, NUREG–1776 (August 2003).

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11. ABSTRACT (200 words or less)

This report augments the existing collection of contemporary Level 1 PRA success criteria analyses for the purpose of (i) maintaining and enhancing the Standardized Plant Analysis Risk (SPAR) models being developed by the NRC; (ii) supporting the NRC's risk analysts when addressing specific issues in the Accident Sequence Precursor (ASP) program and the Significance Determination Process (SDP); and (iii) informing other ongoing and planned initiatives. This report focuses on investigations of (a) the effect of modeling assumptions on key figures-of-merit such as the time of depletion of the refueling water storage tank; (b) the relative conservatism or non-conservatism associated with commonly used and potential core damage surrogates; (c) the time required to arrest fuel heatup if equipment is recovered following a core heatup; and (d) the comparison of MELCOR to the Modular Accident Analysis Program (MAAP) for characterizing the safety margins of selected PRA sequences through an uncertainty quantification process.

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