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

Draft Report for Comment

Office of Nuclear Regulatory Research

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COMMENTS ON DRAFT REPORT

Any party interested may submit comments on this report for consideration by the NRC staff. Comments may be accompanied by additional relevant information or supporting data. Please specify the report number draft NUREG-1953 in your comments and send them by December 15, 2010.

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ABSTRACT

Some specific thermal-hydraulic success criteria from the suite of Standardized Plant Analysis Risk (SPAR) models have apparent inconsistencies when compared to counterpart licensee probabilistic risk assessments, other relevant SPAR models, or relevant engineering studies. These inconsistencies are a natural outcome of the SPAR development process and the constraints that are placed upon it. Even so, the NRC staff wants to strengthen the technical basis for the SPAR models by performing targeted additional engineering analysis. The identified success criteria are for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) and involve a number of different initiating events and scenarios. This report describes MELCOR analyses performed to augment the technical basis for supporting or modifying these success criteria. The analyses are intended to be confirmatory in nature; they are not intended to be licensing-quality analyses.

First, this report provides a basis for using a core damage surrogate of 2,200 degrees Fahrenheit (1,204 degrees Celsius) peak cladding temperature, based on MELCOR analyses for representative sequences and a consideration of historical core damage surrogates. Following this are descriptions of the major plant characteristics for the two plants used for this analysis (Surry Power Station and Peach Bottom Atomic Power Station) and of the MELCOR models used to represent these plants. Finally, the report presents the results of many MELCOR calculations and translates these results into specific candidate SPAR model upgrades for Surry and Peach Bottom.

Potential SPAR model upgrades for Surry include the following:

- better quantification of the timing of core damage relative to refueling water storage tank depletion for small-break loss-of-coolant accidents (LOCAs)
- confirmation of the success criteria for small-break LOCAs without operator action
- revision of the success criteria for feed and bleed (loss of all feedwater) to require fewer pressurizer power-operated relief valves
- updated timings for steam generator tube rupture events with minimal operator action
- updated timings for alternating current power recovery for station blackout sequences
- revision of success criteria for medium and large-break LOCAs to modify the systems needed

Potential SPAR model upgrades for Peach Bottom include the following:

- additional credit for the reactor core isolation cooling system during an inadvertently opened relief valve event, and potential additional credit for the control rod drive injection system
- updated timings for suppression pool heatup and alternating current power recovery for station blackout sequences

FOREWORD

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

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

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

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ABBREVIATIONS AND ACRONYMS

ac	alternating current
ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AFW	auxiliary feedwater
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BAF	bottom of active fuel
BWR	boiling-water reactor
C	Celsius
CDF	core damage frequency
CFR	*Code of Federal Regulations*
cm	centimeter
COR	MELCOR core package
CRD	control rod drive injection
CST	condensate storage tank
CVH	control volume hydrodynamics (MELCOR package)
dc	direct current
ECA	emergency contingency action
ECCS	emergency core cooling system
EOP	emergency operating procedure
F	Fahrenheit
ft	feet
gpm	gallons per minute
HCTL	heat capacity temperature limit
HHSI	high-head safety injection
HPCI	high-pressure core injection
hr	hour
IORV	inadvertently open relief valve
K	Kelvin
LBLOCA	large-break loss-of-coolant accident
LHSI	low-head safety injection
LOCA	loss-of-coolant accident
LOMFW	loss of main feedwater
LOOP	loss of offsite power
LPCI	low-pressure core injection
LPCS	low-pressure core spray
m^3	cubic meters
m^3/min	cubic meters per minute
MCP	main coolant pump
MD-AFW	motor-driven auxiliary feedwater
MELCOR	Not an acronym
MFW	main feedwater
min	minute
MBLOCA	medium break loss-of-coolant accident
MPa	megapascal
MSIV	main steam isolation valve
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
PCT	peak cladding temperature
PORV	power- (or pilot-) operated relief valve
PRA	probabilistic risk assessment
PRT	pressurizer relief tank
psia	pounds per square inch absolute
psig	pounds per square inch gage
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPV	reactor pressure vessel
RWST	refueling water storage tank
SBLOCA	small-break loss-of-coolant accident
SC	success criteria
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SOARCA	State-of-the-Art Reactor Consequence Analyses
SPAR	standardized plant analysis risk
SRV	safety relief valve
TAF	top of active fuel
TD-AFW	turbine-driven auxiliary feedwater
TRACE	REAC/RELAP-V Advanced Computational Engine
WOG	Westinghouse Owners Group

1. INTRODUCTION AND BACKGROUND

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

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

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

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

The analyses performed are intended to be confirmatory in nature; they are not intended to be licensing-quality analysis.

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

2. DEFINITION OF CORE DAMAGE

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

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

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

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

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

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

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

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

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

- station blackout

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[2] Note that the seal leakage assumptions used here are different than those used in the SOARCA project. Also note that the leakage rate provided here is the leakage rate at full system pressure. As the system depressurizes, the leak rate will decrease.

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

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

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

- It always precedes oxidation transition.

- It is not overly conservative.

- It is equally applicable for both PWRs and BWRs.

- The timing between 1,204 degrees C (2,200 degrees F) and oxidation transition is relatively similar among the different sequences analyzed.

- It is consistent with the criteria contained in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." [10 CFR, 2007]

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

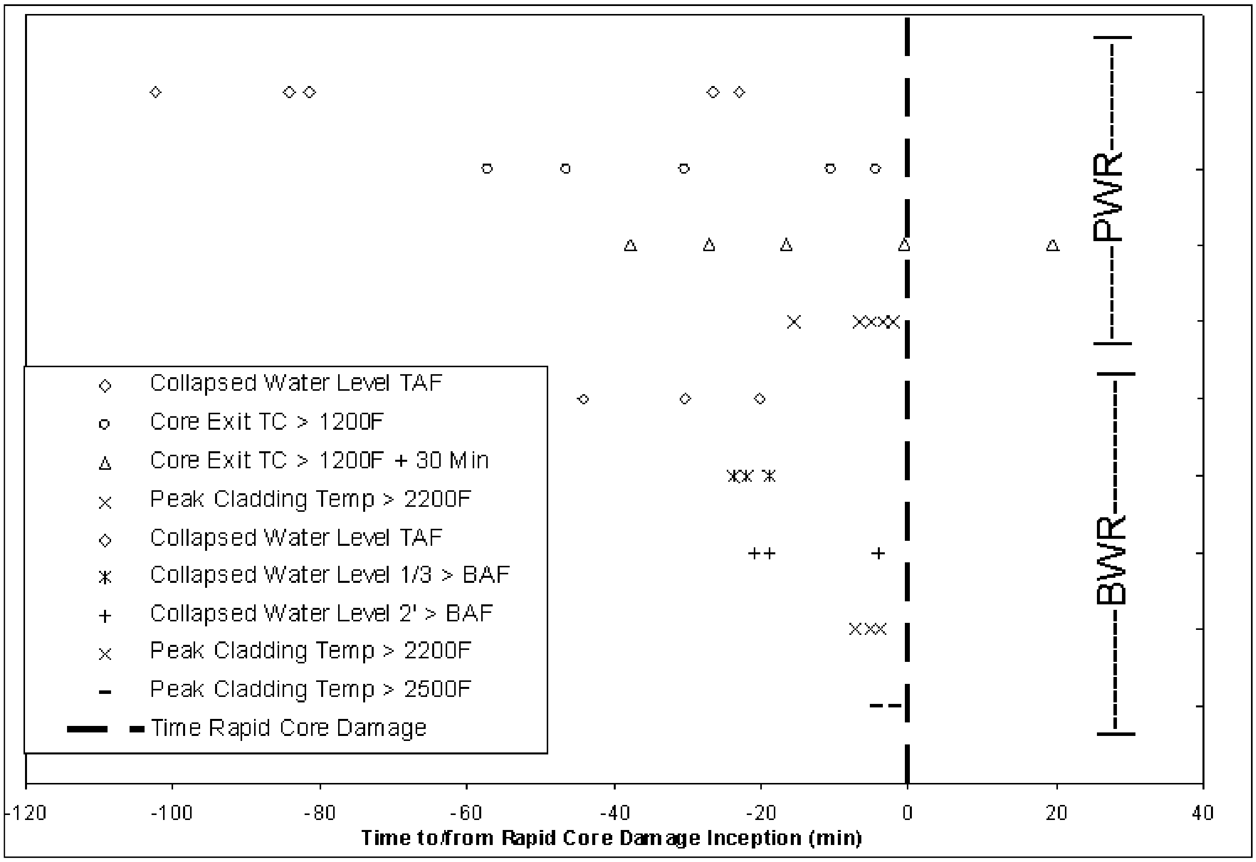

Figure 1 Summary of Core Damage Surrogate Calculations
(1,200 °F = 649 °C; 2,200 °F = 1,204 °C; 2,500 °F = 1,371 °C)

3. RELATIONSHIP TO THE ASME/ANS PRA STANDARD

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

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

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

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

4. MAJOR PLANT CHARACTERISTICS

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

4.1 Surry Power Station

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

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

4.2 Peach Bottom Atomic Power Station

As with Surry, to the level of detail needed for this analysis, Peach Bottom Units 2 and 3 were considered to be identical. Both are General Electric BWR/4s with Mark-I containment. Peach Bottom's reactor core isolation cooling (RCIC) system has a capacity of 600 gpm (2.3 m^3/min) at 150 to 1,150 pounds per square inch gage (psig) (1.0 to 7.9 MPa). The high-pressure core injection (HPCI) system capacity is 5,000 gpm (18.9 m^3/min). The condensate storage tank (CST) is the preferred source until low level in the CST (less than 5 feet (1.5 meters)) causes an automatic switchover to the suppression pool. The RCIC and HPCI turbines will automatically trip with a high turbine exhaust pressure of 50 psig and 150 psig (.34 and 1.03 MPa), respectively. RCIC and HPCI systems will automatically isolate with a low steamline pressure of 75 psig (.51 MPa). RCIC and HPCI pump bearings are rated for 210 degrees F (99 degrees C). The high-capacity low-pressure core injection (LPCI) system has a shutoff head of 295 psig (2.0 MPa). The volume of the CST is 200,000 gallons (756 m^3). The

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3 Note that the relationship between RWST volume and percent inventory is not intuitive, because zero percent corresponds to about 14,000 gallons (53 m^3), 13.5 percent corresponds to 66,000 gallons (250 m^3), about 97 percent corresponds to the technical specification limit, and 100 percent corresponds to 399,000 gallons (1,510 m^3).

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

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

suppression pool has a technical specific maximum temperature limit of 95 degrees F (35 degrees C), and a volume of 127,300 cubic feet (3,605 m^3).

5. MELCOR MODEL

5.1 Plant Representation

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

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

Figure 2 shows a plan view of the MELCOR model for the Surry reactor coolant system (RCS). All three RCS loops are modeled individually. The detailed nodalization of the RCS loop piping as well as the reactor core and vessel upper plenum allows modeling of the in-vessel and hot-leg counter-current natural circulation during core heatup. This feature has been shown to be relevant even within the temperature ranges of interest here (i.e., those preceding core damage). The main coolant pumps (MCPs) are tripped on power failure or voiding (related to pump vibration) in the loop[6]. The core is nodalized into 10 axial levels and five radial rings. Each axial level is comprised of two thermal response nodes (the MELCOR core package (COR)) and one hydrodynamic volume (the MELCOR control volume hydrodynamics package (CVH)). Safety systems are modeled using injection points, and the relevant portions of the reactor protection system and control systems are modeled using MELCOR control functions. For the secondary side, both turbine-driven auxiliary feedwater (TD-AFW) and motor-driven AFW (MD-AFW) are modeled (including provisions for water level control). The core decay power is based on a number of ORIGEN calculations for each radial ring. The containment is divided into nine control volumes representing the major compartments. Containment sprays and fan coolers are also modeled.

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

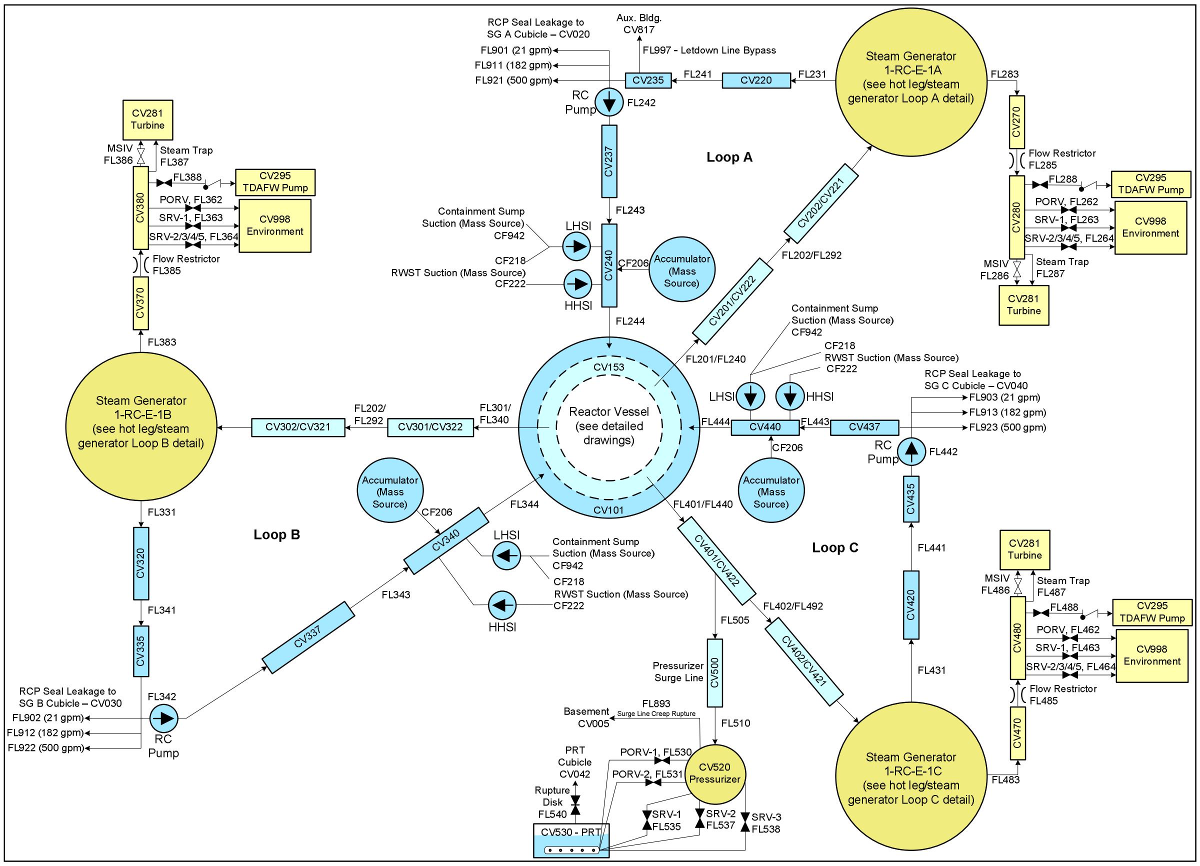

Figure 2 Plan View of the Surry MELCOR RCS Model

Figure 3 shows a schematic of the Peach Bottom MELCOR model, including the reactor pressure vessel (RPV), wetwell, and safety systems. The drywell (not shown) has four control volumes representing the pedestal, lower drywell, upper drywell, and upper head regions. The vessel (excluding the core region) is represented by seven control volumes with connections to various safety systems, including control rod drive injection (CRD), RCIC, HPCI, low-pressure core spray (LPCS), and residual heat removal (RHR) (vessel injection and containment cooling modes). The models for HPCI and RCIC include separate control volumes for the turbine exhausting into the suppression pool. All safety relief valves (SRVs), including dedicated automatic depressurization system (ADS) valves, are modeled with flow paths on two steamlines (a single steamline A, and a combined steamline for B, C, and D). The core nodalization is similar to the Surry model, with 10 axial levels (with a 2:1 COR:CVH ratio) and five radial rings. Like the Surry model, the core decay power is based on a number of ORIGEN calculations for each radial ring. Because no changes were made to the SOARCA model, Appendix B of this report does not include the same introductory plant model information for Peach Bottom as Appendix A does for Surry.

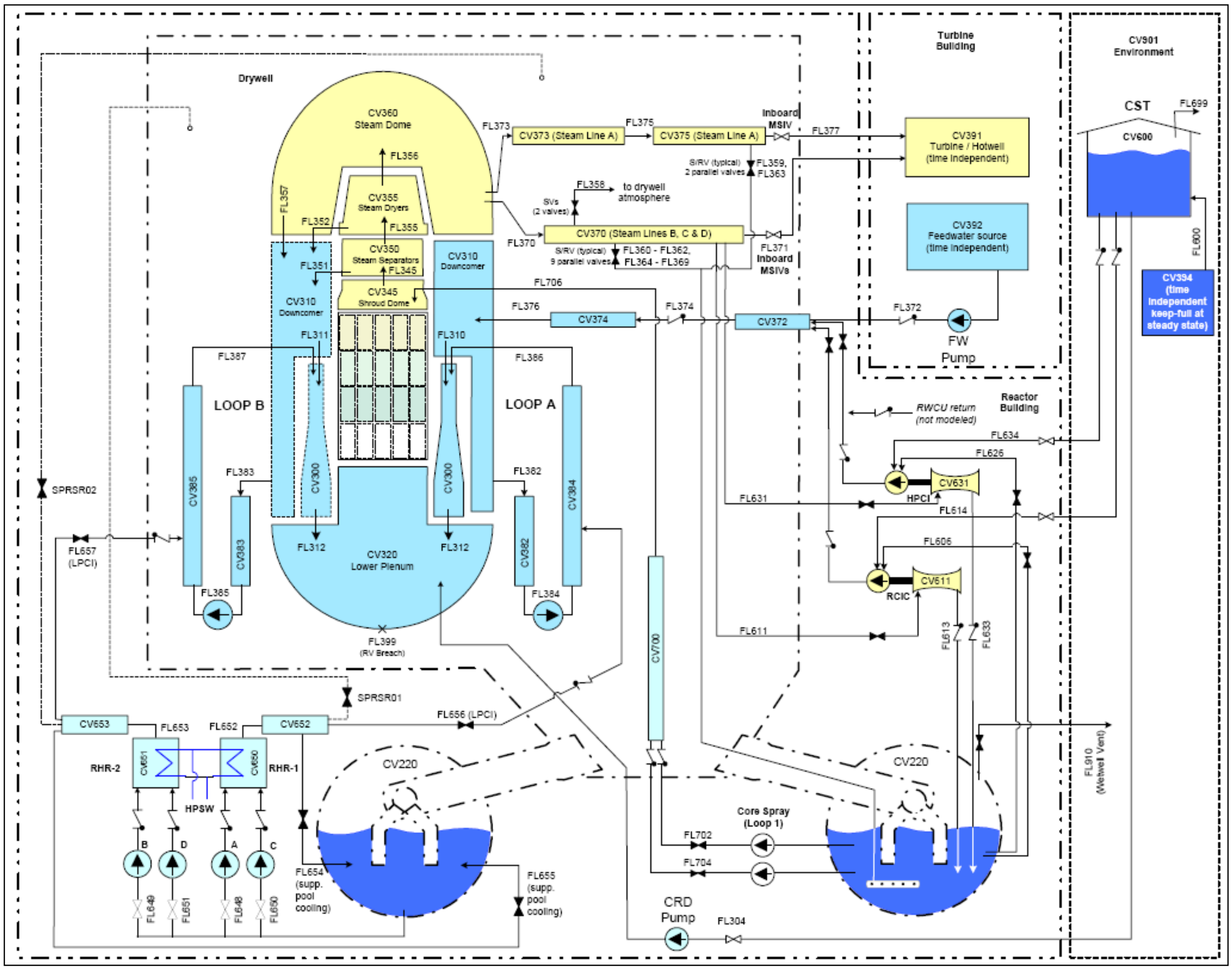

Figure 3 Schematic of the Peach Bottom RCS Nodalization

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

For the purposes of this analysis, a simplified treatment of valve cycling and failure is adopted for this intermediate situation. Table 2 provides a synopsis of the basis for the values used, including the specific value used for each type of valve. The values in Table 2 are tabulated using the following formula:

$$Cumulative\ Probability\ of\ Failure = 1 - (1 - P_D)^n$$

where P_D equals the probability of failure per demand and n equals the number of lifts. A median value (cumulative probability equal to 0.5) was used in this report. Two key limitations to this approach are (i) its use of a constant failure probability per demand and (ii) its assumption that the failure probability is the same regardless of whether the valve is passing steam, water, or a two-phase mixture.

Table 2 Comparison of Values for Surry Stuck-Open Valves

	Surry Individual Plant Examination		Circa 2006 Surry PRA, SOARCA (used in the present analyses)	
Valve	**Probability of Sticking Open per Demand**	**# of Lifts for Cumulative Probability of Sticking open = 0.5**	**Probability of Sticking Open per Demand**	**# of Lifts for Cumulative Probability of Sticking Open = 0.5**
Pressurizer PORV	0.0123	56	0.0028	247
Main Steamline PORV	0.0123	56	0.0058	119
Pressurizer SRV	0.0123	56	0.0027	256
Main Steamline SRV	0.0123	56	0.0027	256

For Peach Bottom, the value used is 187 lifts, which corresponds to a cumulative failure probability of 0.5 for a probability of failure per demand of 0.0037[7].

5.2 MELCOR Validation

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

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

- Quench experiment 11, simulating an SBLOCA with late vessel depressurization to investigate response of overheated rods under flooding conditions
- The Three Mile Island Unit 2 accident
- Loss-of-fluid test (LOFT) LP-FP-2, simulating LBLOCA
- Russian Academy of Sciences MEI experiments involving a spectrum of LOCA sizes to study critical flow and vessel response
- NEPTUN experiments to test pool boiling models and void fraction treatment
- General Electric level swell and vessel blowdown experiments characterizing single- and two-phase blowdown, liquid carryover, and water level swell

[7] Note that this value may be different that the final value used in the SOARCA project.

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

- Containment thermal-hydraulic phenomena studied in various experimental facilities, including Nuclear Power Engineering Corporation for mixing and stratification, Heissdampfreaktor for blowdown into containment, and Carolinas-Virginia Tube Reactor for steam condensation in the presence of noncondensables

- Small-scale experiments to test condensation models, including Wisconsin flat plate experiments and Dehbi tests

6. MELCOR RESULTS

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

A number of different scenarios are analyzed. The following scenarios are analyzed for Surry:

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

The following scenarios are analyzed for Peach Bottom:

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

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

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

This series of cases investigates the timing to RWST depletion (and thus switchover to recirculation) for small-break LOCAs (SBLOCAs) in which operators take very few actions. In reality, the operators would enter procedure E-0, “Reactor Trip or Safety Injection,” transition to E-1, “Loss of Reactor or Secondary Cooling,” and later to ES-1.2, “Post LOCA Cooldown and Depressurization.”

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

The varied parameters are break size (0.5 inch, 1 inch, and 2 inch (1.3 cm, 2.5 cm, and 5.1 cm)), the assumption on relief valve sticking, and containment spray function (available or not available). In all twelve cases investigated, the break location is the horizontal section of the cold leg. In addition, sensitivity cases are performed to look at the effects of securing HHSI pumps (Cases 2a and 6a) and also performing secondary-side cooldown (Cases 2b and 6b). These sensitivity cases demonstrate the impact of HHSI and secondary-side cooldown on RCS pressure and RHR entry timing. Because of project resource considerations, the modeling uses a simplified scoping approach and does not necessarily represent the actual plant operating procedures. For this reason, the results should be used with caution. Results are provided in Table 3, Table 4, and Table 5. In addition to the key timing tables below, plots for various results of interest are provided in Appendix A, Section A.2.

For the 2-inch (5.1-cm) breaks investigated, the reactor coolant system depressurizes due to the break. The loss of high-head injection following RWST depletion (high-head recirculation was not modeled) further reduces the primary side pressure to less than the maximum pressure for LHSI recirculation, and thus HHSI recirculation is not necessary. The same is true for 0.5-inch (1.3-cm) breaks when the PORV is assumed to stick open after 247 lifts, because this causes the 0.5-inch (1.3-cm) break to become a 1.9-inch (4.8-cm) break.[9] Note that operator action to reduce injection (in response to PORV cycling) and thus limit pressurizer PORV cycling was not modeled. Also note that some cases do include throttlng HHSI for the purpose of scoping operator actions to depressurize and cooldown. For 0.5-inch (1.3-cm) cases in which the PORV does not stick open, the system does not depressurize. Finally, for the 1-inch (2.54-cm) cases, the break is not large enough to cause depressurization (because of HHSI injection) and the PORV does not open. As a result, the system pressure is still high at the time of RWST depletion. Loss of HHSI at RWST depletion causes depressurization, but not enough to allow for LHSI recirculation.

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

- For the 0.5-inch (1.3-cm) breaks, the PORV sticks open after 247 cycles unless (a) it does not lift that many times [Case 6b] or (b) noted otherwise [Cases 7 and 8].
- Operators do not throttle injection for the purpose of preventing valve chattering, which is relevant for 0.5-inch (1.3-cm) breaks.
- Operators do not take action to refill the RWST.
- Operators secure containment sprays (and reset to allow subsequent actuation) per EOPs after containment pressure drops below 12 psia (0.083 MPa).
- RCPs trip at 10 percent voiding.
- Operator actions for manual cooldown and depressurization are not modeled, except in a simplified manner for sensitivity cases 2b and 6b.
- MD/TD-AFW is available.

[9] The equivalent diameter of the PORV is 1.39 inches (3.53 cm).

Table 3 Surry SBLOCA Sump Recirculation Results

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

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

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

Event	Case 1 (hr)	Case 2 (hr)	Case 2a (hr)	Case 2b (hr)	Case 3 (hr)	Case 4 (hr)
Reactor trip	0.03	0.03	0.03	0.03	0.01	0.01
HSSI injection	0.03	0.03	0.03	0.03	0.01	0.01
LHSI injection	-	-	-	2.02	-	-
First actuation of contain. sprays	-	2.65	3.29	-	-	1.76
RWST depletion (< 13.5%)	5.83	4.30	5.80	-	3.12	2.63
Spray recirculation	-	4.30	5.80	-	-	2.63
LHSI recirculation	-	-	-	-	3.38	2.86
Accumulator starts to inject	6.38	4.92	5.83	0.82	0.23	0.23
RCP trip (10% void)	7.38	5.76	6.73	1.41	-	-
Core uncovery	9.23	7.32	7.9	-	-	-
Core damage (max temp > 2,200 °F)[1]	11.9	9.93	10.0	-	-	-

1 2,200 °F = 1204 °C.

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

Event	Case 5 (hr)	Case 6 (hr)	Case 6a (hr)	Case 6b (hr)	Case 7 (hr)	Case 8 (hr)
Reactor trip	0.01	0.01	0.01	0.01	0.01	0.01
HSSI injection	0.01	0.01	0.01	0.01	0.01	0.01
LHSI injection	-	-	-	3.49	-	-
PORV stuck open	0.83	0.83	4.65	-	-	-
First actuation of contain. sprays	-	2.20	5.30	-	-	3.23
RWST depletion (<13.5%)	4.14	3.43	7.45	-	8.17	5.52
Spray recirculation	-	3.43	7.45	-	-	5.53
LHSI recirculation	4.72	3.97	-	-	26.6	-
Accumulator starts to inject	4.15	3.44	7.18	1.10	8.28	5.65
RCP trip (10% void)	-	4.68	5.00	13.8	11.7	10.3
Core uncovery	-	-	8.77	-	17.8	14.4
Core damage (max temp > 2,200 °F)[1]	-	-	9.61	-	25.1	21.4

[1] 2,200 °F = 1204 °C.

6.2 Feed-and-Bleed PORV Success Criteria (Surry)

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

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

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

The analysis performed here demonstrates that one PORV provides a sufficient bleed path to maintain quasi-steady conditions on the primary side.[10] Further, it is not necessary for the operators to manually open the PORV, as the HHSI at Surry will cause the valve to automatically open due to high pressure. Even in the absence of operator action, the capacity of one HHSI pump is sufficient to remove decay heat for either the nominal (Surry) or elevated (e.g., Virgil C. Summer Nuclear Station) power levels. Nevertheless, it is important to note that other differences between Surry and the higher power-level three-loop plants (most notably steam generator (SG) type) have not been addressed.

In the absence of further operator action, these cases do eventually proceed to core damage in these analyses because HHSI recirculation (which would actuate upon RWST depletion) is not modeled. However, at least 8 hours is available prior to RWST depletion, and an additional 3.5 to 4 hours is available until core damage occurs. This timing information can be used to inform related sequences that include human failure events associated with refilling the RWST or aligning the HHSI water source to the containment sump. In addition to the results and key timings in Table 6 and Table 7 below, plots for various results of interest are provided in Appendix A, Section A.3.

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

- Operators secure containment sprays (and reset to allow subsequent actuation) per the EOPs after containment pressure drops below 12 psia (0.083 MPa).
- HHSI recirculation is not modeled; thus the time to core damage is driven by RWST depletion (the timing of which is affected by the assumption that operators do not take early action to start HHSI).
- The PORV is aligned for automatic operation and opens when the RCS pressure increases above the high pressure setpoint (i.e., no manual operator action).
- Manual RCS depressurization and cooldown is not modeled.
- RCPs trip at 10 percent voiding; in actuality, Function Restoration Procedure FR-H.1 would have the operators stop all RCPs.

10 Note that for Cases 2 and 3, SRV 1 briefly lifts because of the actuation of HHSI (PORV 2 was disabled for the calculation). This brief actuation is judged to be inconsequential to the overall progression of the event.

Table 6 Surry Feed-and-Bleed PORV Success Criteria Results

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

1 Nominal equals 2,546 MWt (Surry) and 113.9% equals 2,900 MWt (Beaver Valley, Harris, and Summer); 2,900 MWt is the highest present power level of the three-loop Westinghouse plants.
2 Low SG level is < 19% of narrow-range span, while low-low SG level is < 16% of narrow-range span, based on Technical Specification 2.3-3 (January 2008).
3 Core uncovery and damage late in the simulation are artifacts of the assumed unavailability of HHSI recirculation.

Table 7 Surry Feed-and-Bleed PORV Success Criteria Key Timings

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

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

6.3 Steam Generator Tube Rupture Event Tree Timing (Surry)

These calculations assess the time available to take corrective actions for events involving spontaneous (as opposed to accident-induced or consequential) tube rupture events. In addition to the results and key timings in Table 8 and Table 9 below, plots for various results of interest are provided in Appendix A, Section A.4. For reference, the effective leak size of a one-tube rupture is about 1 inch (2.5 cm) effective diameter. Past operating experience for SGTR events suggests that, in some cases, the time between the initiating event and initiation of RHR can be significant (e.g., this timing ranges from 3.25 hours to 21.5 hours for the events covered in a study conducted in the mid-1990s)[11]. Here, very few operator actions are assumed. In reality, the operators would be expected to enter E-0, “Reactor Trip or Safety Injection,” transition to E-3, “Steam Generator Tube Rupture,” and later to one of three post-SGTR procedures (based on plant conditions).

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

Even with few operator actions assumed, the results provided below show that there is substantial time for corrective actions because of available secondary-side heat removal. At 24 hours, the fuel temperatures for all three cases are stable at less than 550 degrees F (288 degrees C), although additional actions would be eventually required (e.g., refilling the CST). For these analyses, the faulted steam generator relief valves were not allowed to stick open, despite cycling a large number of times (e.g., > 15,000). If the valve stuck-open, core damage would result earlier (the inability of the faulted steam generator to hold pressure would result in continued significant flow through the break after primary and secondary side pressure equalize following RWST depletion). An additional calculation will be performed to determine the effect of this assumption, and this calculation will be included in the final version of this report. Preliminary results suggest that tens of hours are still available.

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

- Main steamline isolation valves close on reactor trip.

- Operators secure either one or two HHSI pumps at 15 minutes (depending on the case) and manually control auxiliary feedwater to maintain SG level (standard practice).

- The faulted steam generator PORV does not stick open, regardless of the number of lifts, and regardless of whether it passes water. The other two steam generators' PORVs do not stick open (based on the 119 cycle criteria; see Table 2) until after 24 hours.

- HHSI recirculation is not modeled.

- RCPs trip at 10 percent voiding.

- Manual isolation of the faulted SG is not assumed (i.e., operators fail to perform this action).

- Manual actions to model long-term heat removal (EOP Emergency Contingency Action (ECA) 3.1/3.2) are not modeled.

Table 8 Surry SGTR Results

Case	**No. Tubes**	**HHSI Pumps**	TD-AFW	MD-AFW	**Core Uncovery**	**Core Damage**
1	1	3/2	Yes		No[1]	No[1]
2	5				No[1]	No[1]
3	1	3/1			No[1]	No[1]

[1] Based on a 24-hour mission time.

Table 9 Surry SGTR Key Timings

Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)
Reactor trip	0.048	0.012	0.048
HHSI initiates (3 pumps)	0.051	0.013	0.051
1 of 3 HHSI pumps secured	0.25	0.25	N/A
2 of 3 HHSI pumps secured	N/A	N/A	0.25
RWST depletion (< 13.5%)[1]	10.68	5.58	14.06
MCP trip (10% void)	17.81	11.71	20.20
Core damage	> 24 hours		

[1] Recall that, because the RCS leak location is the ruptured SG tubes, a substantial amount of water is expelled from the system via the SG relief valves (rather than into containment).

6.4 PWR Station Blackout (Surry)

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

In the emergency operating procedures, the operators would first enter E-0, "Reactor Trip or Safety Injection," which would direct them to ECA-0.0, "Loss of All AC Power." If ac power is recovered, the operators will transition to ECA-0.1, "Loss of All AC Power Recovery without SI Required" and/or ECA-0.2, "Loss of All AC Power Recovery with SI Required." If ac power is not recovered and the core-exit thermocouples rise past 1,200 degrees F (649 degrees C), the operators will transition to SACRG-1, "Severe Accident Control Room Guideline Initial Response."

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

The RCP seal leakage rates and timing are taken from the seal leakage model used in the current Surry SPAR model: the Westinghouse Owners Group (WOG) 2000 seal leakage model for "new" high-temperature seals, WCAP-15603, "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," issued May 2003 [West., 2003], as modified by the NRC staff's associated April 2003 safety evaluation report [NRC, 2003b].[12] The safety

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

evaluation report for WCAP-15603 makes a few modifications to the WCAP-15603 model, including the disallowance of credit for the third RCP seal. The resulting model has outcomes associated with four possible leakage rates for use in PRAs, with the onset of increased leakage occurring at 13 minutes in all cases. Table 10 reproduces the leakage rates and their conditional probabilities, along with some associated timings from the Westinghouse Emergency Response Guidelines as reproduced in the Surry SPAR v3.52 model documentation of July 2008. The current analysis ran cases for three of these leakage sizes (21 gpm per pump [0.079 m^3/min], 182 gpm per pump [0.689 m^3/min] and 500 gpm per pump (1.89 m^3/min)).[13]

Table 10 Reactor Coolant Pump Seal Leakage Details

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

1 Assumes availability of TD-AFW
2 21 gpm = 0.079 m^3/min; 76 gpm = 0.29 m^3/min; 182 gpm = 0.689 m^3/min; 480 gpm = 1.82 m^3/min.

The results of the present analysis are in good agreement with those from the Westinghouse Emergency Response Guidelines (Table 10). For analogous cases (i.e., those with TD-AFW available and no secondary-side depressurization) the following conditions apply:

- Time to core uncovery is about 1.5 hours for the largest leakage rate of 500 gpm/RCP (1.89 m^3/min/RCP), as compared to 2 hours in the Westinghouse calculations.

- Time to core uncovery is about 4 hours for the intermediate leakage rate of 182 gpm/RCP (0.68 m^3/min/RCP), as compared to 3 hours in the Westinghouse calculations.

- Time to core uncovery is about 13 hours for the normal leakage rate of 21 gpm/RCP (0.079 m^3/min/RCP), which is identical to the Westinghouse calculations.

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

Topical report WCAP-16396-NP, "WOG 2000 Reactor Coolant Pump Seal Performance for Appendix R Solutions," issued January 2005 [West., 2005] provides discussion for why the NRC's safety evaluation of the WOG 2000 model—and the WOG 2000 model itself—result in conservative estimates of RCP seal leak rates. These conservatisms are associated with both the leak rates assumed and the timing of seal failure (which is reported to vary from 8 minutes to 40 minutes, as compared with the 13 minutes used in the WOG 2000 model). This topical report quantitatively assesses the effects of these conservatisms on accident progression timings (specifically, the time for loss of pressurizer level and core uncovery). The topical report

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

concludes that the conservatisms can substantially affect the assessment of coping strategies, but that the conservatisms are "unlikely to affect any conclusions drawn from PRA models for internal events from at-power conditions." [West., 2005] These conclusions led to the decision not to request NRC review of a less conservative model. If applied here, these conclusions suggest that the timings to core damage calculated here are conservative, but that these conservatisms will not affect the overall conclusions drawn from the models. Even so, the potential conservatisms could affect intermediate PRA results, such as the human error probability associated with a particular action.

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

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

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

As shown in Figure 5, the sensitivity case where HHSI was recovered at 2.14 hours occurred too late to avert fuel melting. For the case where HHSI was recovered at 1.64 hours, recovery of injection was sufficient to avert fuel melting. A best-estimate time could be developed by running calculations using an intermediate time (e.g., 15 minutes) for this case, as well as running similar sensitivities for other cases. In addition to the results and key timings in Table 11, Table 12, Table 13, Table 14 and Figure 4 below, plots for various results of interest are provided in Appendix A, Section A.5.

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

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

- There is infinite dc power for control of TD-AFW for Cases 4 and 6 (i.e., mimics successful blind feeding).

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

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

- For cases with RCP seal failure, failure is assumed to occur at 13 minutes.[14]

- Manual operator actions for rapid secondary-side depressurization are not modeled.

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[14] Note that this differs from the seal failure model used in the SOARCA project.

Table 11 Surry Station Blackout Results

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

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

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

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

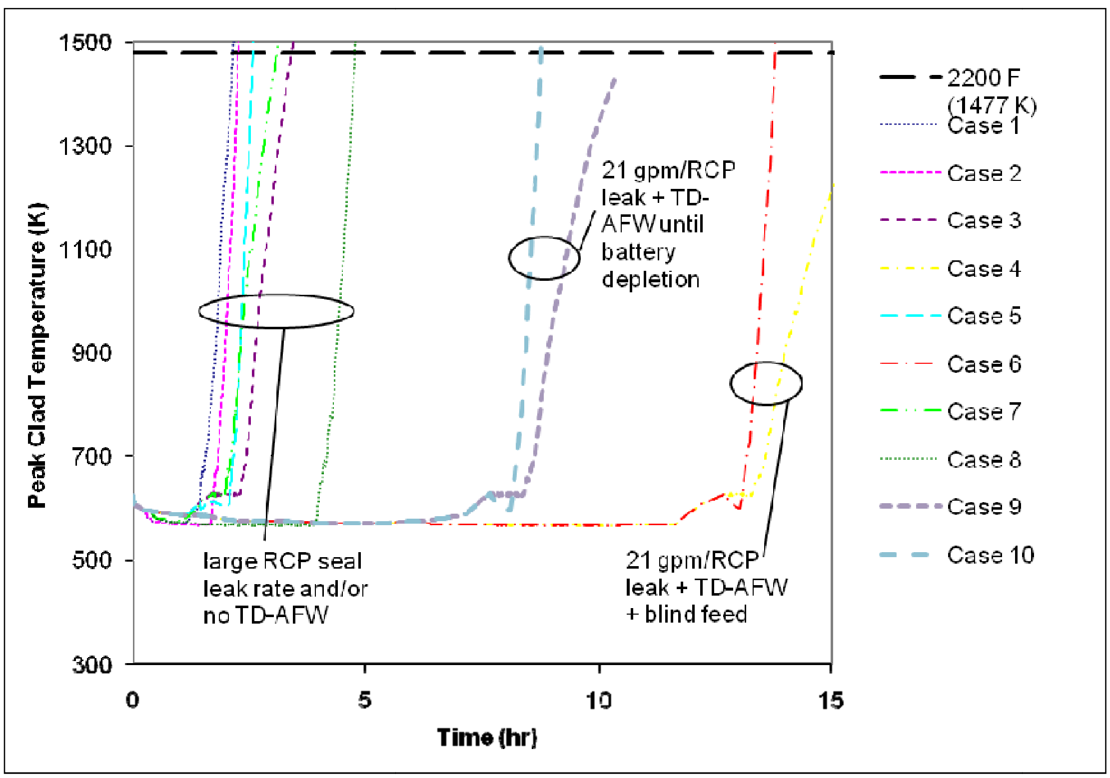

Figure 4 PCT Signatures for all Surry Station Blackout Cases

Table 12 Surry Station Blackout Key Timings (Cases 1–2)

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

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

Table 13 Surry Station Blackout Key Timings (Cases 3–6)

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

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

Table 14 Surry Station Blackout Key Timings (Cases 7–10)

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

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

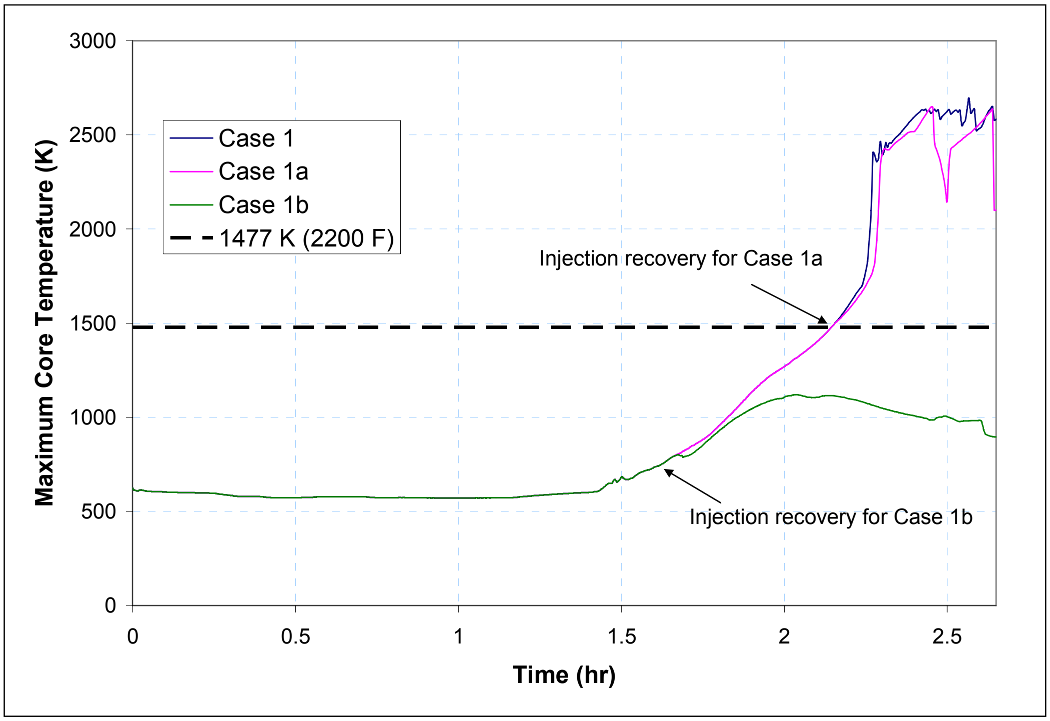

Figure 5 Surry Injection Recovery Sensitivity Cases

6.5 PWR Medium and Large Break LOCA Initial Response (Surry)

The final set of Surry sequences investigates combinations of accumulators, HHSI, and LHSI for a spectrum of LOCA break sizes for the early phase of the accident (e.g., the first few hours). Break sizes from 2 inches (5.1 cm) to a double-ended break were analyzed, as shown in Table 16. Although some calculations are simulated into the long-term cooling phase, the calculations are only intended to inform success criteria for the early injection phase of the accident.

By convention, the breakdown in the LOCA spectrum for most Westinghouse PWRs is 0.5 inch (1.3 cm) to 2 inches (5.1 cm) (SBLOCA), 2 inches (5.1 cm) to 6 inches (15.2 cm) (medium break LOCA (MBLOCA)) and 6 inches (15.2 cm) and greater (LBLOCA). The break location for the current analyses is always the horizontal section of the cold leg in the pressurizer loop. Very few operator actions are modeled. In reality, the operators would enter E-0, “Reactor Trip or Safety Injection,” and transition to E-1, “Loss of Reactor or Secondary Coolant.” Depending on the course of the accident, they would then transition to one of several ES-1.x series supplemental emergency procedures.

As will be shown below, some of these accidents progress very quickly, with core uncovery taking place within the first minute (for large-break LOCAs). Since quickly-evolving accidents can be more challenging to simulate from a thermal-hydraulic standpoint, it is of interest to look at the degree of margin between the PCT (for cases that are deemed successful) and the core damage definition being used. Table 15 presents these figures, demonstrating that the highest MBLOCA PCT (peak cladding temperature) (for a success case) is 483 degrees F (268 degrees C) from the core damage definition used here, and the highest LBLOCA PCT (for a success case) is 706 degrees F (392 degrees C) from the core damage definition. This demonstrates that there is significant margin in these cases, which helps to counteract the additional model uncertainty that might be expected for these quickly evolving accidents. In addition to the key timings in Table 17 through Table 23 below, plots for various results of interest are provided in Appendix A, Section A.6.

Table 15 PCT Ranges for Accumulator Success Cases

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

The results in Table 16 are distilled here to identify the minimal equipment needed to avoid core damage *during the injection phase*. For MBLOCAs, the minimal equipment is the following:

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

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

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

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

For large-break LOCAs, the minimal equipment is the following:

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

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

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

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

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

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

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

Table 16 Surry MBLOCA and LBLOCA Results

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

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

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

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

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

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

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

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

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

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

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

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

Event	Case 14 (hr)	Case 22 (hr)	Case 25 (hr)	Case 28 (hr)
Reactor trip	0.003	0.003	0.003	0.003
HHSI injection	-	0.004	0.004	0.004
RCP trip (10% void)	0.04	0.04	0.04	0.03
First actuation of containment sprays	0.07	0.08	0.08	0.07
Core uncovery (water < TAF)	0.10	0.09	0.09	0.09
LHSI injection	0.73	-	0.30	-
Maximum cladding temperature timing (max. temperature)	0.73 (1183K)	0.21 (807K)	0.32 (1054K)	0.26 (721K)
Core covered	0.79	0.39	0.39	0.41

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

Event	Case 2 (hr)	Case 5 (hr)	Case 6 (hr)	Case 7 (hr)
Reactor trip	0.002	0.002	0.002	0.002
HHSI injection	0.002	-	-	0.002
RCP trip (10% void)	0.02	0.02	0.02	0.02
First actuation of containment sprays	0.03	0.03	0.03	0.03
Core uncovery (water < TAF)	0.04	0.04	0.04	0.07
LHSI injection	0.13	0.14	0.18	-
Maximum cladding temperature timing (maximum temperature)	0.15 (774K)	0.16 (1477K[1])	0.16 (990K)	0.28 (1477K[1])
Core covered	0.19	N/A	0.20	N/A

[1] Actual peak temperature would be higher; this value corresponds to the surrogate used in this project for core damage (2200F [1204 C])

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

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

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

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

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

Event	Case 4 (hr)	Case 19 (hr)	Case 24 (hr)	Case 10 (hr)
Reactor trip	0.001	0.001	0.001	0.001
HHSI injection	0.001	0.001	-	-
RCP trip (10% void)	0.008	0.008	0.006	0.001
First actuation of containment sprays	0.008	0.008	0.008	0.005
Core uncovery (water < TAF)	0.01	0.008	0.02	0.022
LHSI injection	0.04	-	0.05	0.005
Maximum cladding temperature timing (maximum temperature)	0.08 (850 K)	0.30 (835 K)	0.04 (640 K)	0.036 (1043 K)
Core covered	0.12	0.87	0.06	0.053

6.6 Inadvertent Open Relief Valve Success Criteria (Peach Bottom)

The first scenario of interest for Peach Bottom deals with an inadvertent/stuck open relief valve. For this simulation, the reactor is tripped and a safety relief valve (SRV1) opens at time zero, unless noted otherwise. LPCI is available for all cases. The availability of RCIC, HPCI, and CRD injection is varied to assess their effects.

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

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

The calculations summarized in Table 24 and Table 25 demonstrate that any of the injection options considered will prevent heatup before depressurization to LPCI entry. In the case of HPCI, the injection capacity is such that depressurization to LPCI entry doesn't occur for 9 hours. For cases with only CRD injection, CRD prevents significant heatup even when the second CRD pump is not started until 20 minutes after the initiating event. For cases with no high-pressure injection, the system still depressurizes to LPCI entry conditions before core damage would occur, with a maximum cladding temperature of 939 degrees C (1,722 degrees F). Finally, a sensitivity case was run to look at the effect of the assumption that the reactor trips at t = 0, as opposed to tripping on one of the automatic trip signals. This sensitivity case was run for the more limiting of the CRD cases and demonstrated that the reactor tripped shortly after the start of the transient (8 seconds), leading to a PCT that is 110 K higher, but still more than 500 K below the onset of core damage. In addition to the key timing tables below, plots for various results of interest are provided in Appendix B, Section B.1.

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

- Operator actions to reclose the SRV, start residual heat removal in suppression pool cooling mode, and perform an emergency depressurization are either not initiated or are unsuccessful.
- Reactor trip and one SRV are stuck open at time zero (except for Case 4a).
- RCIC is run in inventory control mode.

- Post-scram CRD flow ranges from 110 gpm (0.416 m^3/min) at high pressure (1020 psia, 7.0 Mpa) to 180 gpm (0.681 m^3/min) at low pressure (14.7 psia, 0.1 Mpa) for one pump, or 210 gpm (0.795 m^3/min) to 300 gpm (1.14 m^3/min) for two pumps.

- RCIC and HPCI isolate on low steamline pressure of 75 psig (0.52 Mpa).

Table 24 Peach Bottom Inadvertent Open SRV Results

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

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

Table 25 Peach Bottom Inadvertent Open SRV Key Timings (Cases 1–5)

Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)	Case 4 (hr)	Case 4a (hr)	Case 5 (hr)
SRV 1 open	0	0	0	0	0	0
Reactor trip	0	0	0	0	< 0.01[1]	0
Downcomer level first reaches L2	0.07	0.07	0.07	0.07	0.03	0.07
RCIC/HPCI first started (CST injection mode)	0.08	0.08	-	-	-	-
2nd CRD pump started	-	-	0.17	0.33	0.33	-
Downcomer level reaches L1	0.37	8.93	0.32	0.32	0.24	0.26
Downcomer level below TAF	0.37	8.93	0.35	0.33	0.25	0.28
LPCI first started	0.51	8.93	0.59	0.58	0.53	0.57
RCIC/HPCI pump isolation: low steamline pressure < 0.52 Mpa (75 psig)	0.82	5.59	-	-	-	-
Maximum cladding temperature timing (max temperature)	No heatup	No heatup	0.78 (786 K)	0.76 (830 K)	0.67 (941 K)	0.75 (1,212 K)

[4] Reactor trips at 8 seconds on low RPV level.

6.7 BWR Station Blackout (Peach Bottom)

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

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

For cases with both HPCI and RCIC unavailable, core damage occurs at 0.8 or 1.2 hours, depending on the assumption about SRVs sticking open. Recovery of injection at the time of core damage was demonstrated to quickly arrest heatup. For cases in which dc is lost after 2 hours, core damage occurs at 4 to 5 hours. For cases in which the SRV sticks open after 187 lifts or HCTL depressurization is performed, core damage ranges from 7 to 11 hours. (Note that the operators would initiate HCTL depressurization to protect containment even without a low-pressure injection source.) For cases in which the SRV does not stick open and HCTL depressurization is not performed, RCIC or HPCI (depending on which is assumed available) fails after approximately 12 hours because of loss of NPSH, and core damage occurs after 19 hours. Considering all cases, the time lag from uncovery of the top of active fuel (TAF) to the time of core damage ranges from 0.5 to 1.8 hours. In addition to the results and key timings in Table 26 to Table 29 below, plots for various results of interest are provided in Appendix B, Section B.2.

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

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

Table 26 Peach Bottom Station Blackout Results

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

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

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

Table 27 Peach Bottom Station Blackout Key Timings (Cases 1, 1a, and 2)

Event	Case 1 (hr)	Case 1a (hr)	Case 2 (hr)
Reactor trip, MSIV closure	0	0	0
Downcomer level reaches L2	0.16	0.16	0.16
Downcomer level reaches L1	0.50	0.50	0.27
Downcomer level below TAF	0.50	0.50	0.27
Gap release: 900 °C (1,652 °F)	1.02	1.02	0.69
Core damage: max temp > 1204 °C (2,200 °F)	1.17	1.17	0.79
HPCI, RCIC, CRD Injection start	-	1.17	-
ADS actuated	-	1.24	-
Downcomer level recovers above TAF	-	1.27	-
SRV sticks open due to high # of cycles	1.75	-	-

Table 28 Peach Bottom Station Blackout Key Timings (Cases 3–6)

Event	Case 3 (hr)	Case 4 (hr)	Case 5 (hr)	Case 6 (hr)
Reactor trip, MSIV closure	0	0	0	0
Downcomer level first reaches L2	0.16	0.16	0.16	0.16
RCIC started (CST injection mode)	0.17	0.17	0.17	0.17
RCIC fails due to loss of dc	-	-	2.00	-
HCTL limit reached	2.46 (no action taken)	2.46	2.46 (no action taken)	2.46 (no action taken)
SRV sticks open due to high # of cycles	-	-	-	2.47
RCIC NPSH limit exceeded	11.57	-	-	-
RCIC pump isolation: low steam line pressure < 0.52 MPa (75 psig)	-	3.90	-	3.92
RCIC injection ends due to CST level < 5 ft (1.5 m)	14.43	-	-	-
Downcomer level reaches L1	17.68	5.61	3.25	5.62
Downcomer level below TAF	17.68	5.61	3.25	5.62
Gap release: 900 °C (1,652 °F)	19.06	6.99	4.04	7.00
Core damage max temp > 1,204 °C (2,200 °F)	19.42	7.17	4.25	7.18
Exhaust pressure exceeded: 0.35 MPa (50 psig)	20.14	-	-	-

Table 29 Peach Bottom Station Blackout Key Timings (Cases 7–10)

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

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

Table 30 and Table 31 below map the MELCOR calculations presented in Section 5 with the most closely corresponding SPAR model sequences and provide the relative risk contribution of these sequences. Note that at the initiator heading level (e.g., LOMFW), the right-most column gives the relative contribution of all SPAR sequences from that initiator class (e.g., 9.97%), while the subsequent rows give the relative contributions from the subset of sequences studied in this report (e.g., LOMFW-16 = 9.32%). Regarding loss of offsite power / station blackout, the initiator class relative contribution is for all loss of offsite power events (e.g., switchyard-centered), whereas the analyses in this report focus on station blackout events. Finally, for the station blackout sequences, the nomenclature of having multiple sequence numbers reflects transfers amongst two or more event trees. For instance, “LOOP-17-45” indicates the sequence with end-state #17 from the LOOP event tree, which transfers to the SBO event tree and results in end-state #45 from that event tree. All relevant event trees are provided in Appendix C.

It is also of interest to look at the quantitative timings to core uncovery and ac power recovery used in the Surry SPAR model relative to those from the present analysis (as provided in Section 6.4). Table 32 provides this comparison. A key difference between the SPAR model and the present analyses arises for sequences with AFW available and a stuck-open relief valve. SPAR assumes that the relief valve sticks open early in the event, whereas in the present analyses the relief valves are not challenged (when AFW is available) until much later (e.g., 8 hours). This difference results in a very large delta in the time to core damage. A second key difference is the SPAR assumption that offsite power must be recovered before battery depletion (i.e., no opportunity for preventing core damage following battery depletion), as compared to the present analysis where the calculation is continued beyond battery depletion until the core damage surrogate is reached.

Table 30 Mapping of MELCOR Analyses to the Surry 1 & 2 SPAR (v3.52) Model

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

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

Table 31 Mapping of MELCOR Analyses to the Peach Bottom 2 SPAR (v3.50) Model

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

Table 32 Comparison of Surry Station Blackout Results to the SPAR Model

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

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

Table 33 and Table 34 below (1) summarize the scenarios that have been investigated, (2) recap the boundary and initial condition variations studied using MELCOR, (3) highlight the relevant parts of the existing Surry and Peach Bottom SPAR success criteria, and (4) propose changes to these models based on the MELCOR analysis. Where appropriate, insights are offered on how these results may be applied to SPAR models for other, similar plants. Application of the results to Surry and Peach Bottom, as well as extension of these results to other plants, is being rigorously evaluated; the basis for changes to the SPAR models will be documented in a separate report.

Table 33 Potential Success Criteria Updates Based on Surry Results

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

<table>
<tr><th>Initiator/Aspect of Interest</th><th>MELCOR Variations</th><th>Affected Portion of Existing SPAR Model</th><th>Proposed Changes</th></tr>
<tr><td>Station Blackout (Section 6.4)</td><td>• RCP seal leakage rate: 21, 182, 500 gpm/pump (0.076, 0.689, 1.89 m^3/min)
• SRV stuck-open: 1^{st} lift, never
• TD-AFW: available, unavailable, blind-feeding successful
• dc power: unavailable, depletes at 4 hr, infinite</td><td>Time to recover ac power (and re-establish AFW cooling and RCS makeup capability)</td><td>• Table 32 provides a comparison of the timings between SPAR and the MELCOR analyses. In many cases, the MELCOR results confirm the current modeling assumption. In some cases, the timings will be further investigated to potentially reduce conservatism.
• Sensitivity cases for this scenario suggest that recovery of ac power at 30 minutes or more prior to core damage provides adequate time to establish injection and stop fuel heatup.</td></tr>
<tr><td>Medium-Break LOCA (Section 6.5)[2]</td><td rowspan="2">• Break size: 2, 4, 6, 8, 10 inches, double-ended (5.1, 10.2, 15.2, 20.3, 25.4 cm)
• # of HHSI pumps: 0, 1
• # of LHSI pumps: 0, 1
• # of accumulators: 0, 1, 2
• AFW availability</td><td>Success criteria for the injection phase for the MBLOCA event tree: 1 HHSI train and (1 accumulator in each intact loop or 1 AFW train)</td><td>Based on the MELCOR analyses, the resulting minimal equipment success criteria for the injection phase for medium-break LOCAs is 1 HHSI train and (1 accumulator in either intact loop or 1 low-pressure injection train). Note that the former criteria would not be sufficient for the recirculation phase, because LHSI is necessary to accomplish HHSI recirculation. Also note the above criteria intentionally excludes AFW as it was found not to be necessary for the injection phase.</td></tr>
<tr><td>Large-Break LOCA (Section 6.5)[3]</td><td>Success criteria for inventory control during injection phase for the LBLOCA event tree: 1 Accumulator in each intact loop and 1 low-pressure injection train</td><td>Based on the MELCOR analyses, the resulting minimal equipment success criteria for the injection phase for large-break LOCAs is 1 low-pressure injection train and (1 accumulator in either intact loop or 1 HHSI train).</td></tr>
</table>

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

2 Historically 2-inch (5.1-cm) to 6-inch (15.2-cm) equivalent diameter [NRC, 1990] and [NRC, 1999] (Appendix J).

3 Historically greater than 6-inch (15.2-cm) equivalent diameter [NRC, 1990] and [NRC, 1999] (Appendix J).

Table 34 Potential Success Criteria Updates Based on Peach Bottom Results

Class	MELCOR Variations	Affected Portion of Existing SPAR Model	Proposed Changes
IORV (Section 6.6)	• Injection source: RCIC, HPCI, CRD, none • Timing of 2nd CRD pump initiation: 10 min 20 min	Effectiveness of injection source for core cooling until low-pressure pumps can provide makeup	• For RCIC, four plants (Cooper, Monticello, Perry, Vermont Yankee) may be modified to credit RCIC for this function. The calculation confirms the treatment in all other SPAR models. • For HPCI, the calculation confirms the treatment in all SPAR models. • For CRD, the MELCOR analysis support additional credit for this injection source. Subsequent evaluation will look at variability in CRD flows and concerns relative to CRD trip on run-out at lower pressures.
Station Blackout (Section 6.7)	• Injection: HPCI, RCIC, none • Operator actions: HCTL depress., none • SRV behavior: stuck open at t = 0, stuck open at 187 lifts, never sticks • Recovery time: 1.2 hours, never • DC power: none, 2 hours, infinite	Time to recover ac power (and reestablish core cooling)	• For complete loss of ac/dc, calculations suggest that credit for recovery of offsite power can be extended to 1 hour (currently credit for 30 minutes is given in the SPAR models). • For complete loss of ac/dc commensurate with a stuck-open SRV, calculations suggest that credit for recovery of offsite power can be extended to one half-hour (currently no credit is given in any of the SPAR models). • For cases with infinite dc and RCIC/HPCI loss because of NPSH, current SPAR models are in agreement with these results. • For cases with 2 hours of dc, calculations suggest that 2 hours can be credited for boiloff (currently no credit is given in any of the SPAR models for boiloff). • For the maximum time for injection without suppression pool cooling (HCTL depressurization cases), the SPAR models are in agreement with these calculations, with the exception of Grand Gulf and Nine Mile Point 2 for RCIC.

8. CONCLUSION

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

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