THERMAL-HYDRAULIC ANALYSIS TO SUPPORT UPDATE OF SPECIFIC SUCCESS CRITERIA IN THE STANDARDIZED PLANT ANALYSIS RISK MODELS

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Executive Summary

Specific thermal-hydraulic success criteria from the suite of Standardized Plant Analysis Risk (SPAR) models have been identified as having apparent inconsistencies when compared to counterpart licensee probabilistic risk assessments (PRAs), other relevant SPAR models, or relevant engineering studies. The identified success criteria are for pressurized water reactors (PWRs) and boiling water reactors (BWRs) and involve a number of different initiating events / scenarios. To augment the technical basis for supporting or modifying these success criteria, MELCOR analyses have been performed.

First, a basis is provided for using a core damage surrogate of 2200 F (1204 C), based on MELCOR analyses for representative sequences and a consideration of historical core damage surrogates. Following this, the major plant characteristics for the two plants used for this analysis (Surry and Peach Bottom) are described, along with a description of the MELCOR models used to represent these plants. Next, the results of numerous MELCOR calculations are presented and these results are translated in to specific candidate SPAR model upgrades for Surry and Peach Bottom, as follows:

- Surry:
    - 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 LOCAs without operator action.
    - Revision of the success criteria for feed & bleed (loss of all feedwater) to require fewer power operated relief valves.
    - Updated timings for steam generator tube rupture events with minimal operator action.
    - Updated timings for alternating current (AC) power recovery for station blackout sequences.
    - Revision of success criteria for medium and large break LOCAs to modify the systems needed.
- Peach Bottom:
    - 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 heat-up and AC power recovery for station blackout sequences.

Finally, plans for future work in this effort are presented, including possibilities such as leveraging another State-of-the-Art Reactor Consequence Analysis (SOARCA) project model (Sequoyah), leveraging industry Modular Accident Analysis Program (MAAP) analysis, and/or developing additional new MELCOR models.

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Acronyms

A/C	Alternating current
ADS	Automatic depressurization system
AFW	Auxiliary feedwater
BAF	Bottom of active fuel
BWR	Boiling water reactor
CFR	*Code of Federal Regulations*
COR	MELCOR Core Package
CRD	Control rod drive injection
CS	Containment spray (PWR) or core spray (BWR)
CST	Condensate storage tank
CVH	Control volume hydrodynamics (MELCOR package)
CVTR	Carolinas-Virginia Tube Reactor
DC	Direct current or downcomer
DE	Double-ended
DEP	Depressurization
ECA	Emergency contingency action
ECCS	Emergency core cooling system
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ET	Event tree
FSAR	Final safety analysis report
HCTL	Heat capacity temperature limit
HDR	Heissdampfreaktor
HHSI	High-head safety injection
HPCI	High pressure core injection
HPSW	High-pressure service water
ISR	Inside spray recirculation
LHSI	Low-head safety injection
LBLOCA	Large break loss-of-coolant accident
LOCA	Loss-of-coolant accident
LPCI	Low-pressure core injection
LPI	Low-pressure injection
MAAP	Modular Accident Analysis Program
MCAP	MELCOR Code Assessment Program
MCP	Main coolant pump
MD	Motor-driven
MELCOR	Not an acronym
MFW	Main feedwater
MLOCA	Medium loss-of-coolant accident
MOU	Memorandum of understanding
MSPI	Mitigating System Performance Index
NPSH	Net positive suction head
NUPEC	Nuclear Power Engineering Coorporation
OSR	Outside spray recirculation
PCS	Power conversion system
PCT	Peak clad temperature
PORV	Power (or pilot) operated relief valve
PRA	Probabilistic risk assessment
PRT	Pressurizer relief tank
PRZ	Pressurizer
PWR	Pressurized water reactor
RCIC	Reactor core isolation cooling
RCP	Reactor coolant pump
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RHR	Residual heat removal
RPS	Reactor protection system
RPV	Reactor pressure vessel
RWST	Refueling water storage tank
SBLOCA	Small break loss-of-coolant accident
SC	Success criteria
SDP	Significance determination process
SG	Steam generator
SGTR	Steam generator tube rupture
SI	Safety injection
SOARCA	State-of-the Art Reactor Consequence Analysis
SORV	Stuck-open relief valve
SPAR	Standardized Plant Analysis Risk
SRV	Safety relief valve
TAF	Top of active fuel
TD	Turbine-driven
TMI	Three Mile Island
TRACE	TRAC/RELAP Advanced Computation Engine
VSLOCA	Very small loss-of-coolant accident

Introduction & Background

The success criteria in the NRC's Standardized Plant Analysis Risk (SPAR) models are largely based on the success criteria used in the associated licensee 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, situations exist where plants that should behave similarly from an accident sequence standpoint have different success criteria for specific scenarios. This is an issue that has been recognized for some time, but until recently, the infrastructure was not in place 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 changes to the corresponding SPAR models. The sequences analyzed are not necessarily the most probable sequences, due to 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, these assumptions are characterized in the results description.

This report summarizes the work that has been performed, as well as plans for additional work going forward. The following sections cover:

- 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 numerous MELCOR calculations,
- Expected application of the MELCOR results to the Surry and Peach Bottom SPAR models, and
- Planned future work.

Core Damage Definition

In order to perform success criteria supporting analysis, it is necessary to define what is meant by core damage (i.e., sequence success versus failure). The American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) PRA standard (ASME/ANS-RA-Sa-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 (Section 2-2.3, Supporting Requirement SC-A2).

For this project, a number of MELCOR calculations were run to identify a realistically conservative core damage surrogate. The MELCOR model used for this portion of the effort is not described thoroughly for two reasons: (1) all results are relative, meaning that a change in the model would generally not be expected to affect the delta between the surrogate core damage definition and the onset of rapid cladding oxidation, and (2) the model is based on the general purpose models used in the State-of-the Art Reactor Consequence Analysis (SOARCA) project, and will be documented thoroughly as part of that project.

[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*).

MELCOR version 1.8.6 was used to assess several possible surrogate definitions for a variety of PWR and BWR accident sequences. For the PWR (Surry), the following sequences were analyzed:

- Station blackout with a 182 gpm (0.689 m^3/min) per pump seal LOCA[2]
- Station blackout with a 500 gpm (1.89 m^3/min) per pump seal LOCA
- Hot leg LOCA for 2 inch (5.1 cm), 4 inch (10.2 cm), and 10 inch (25.4 cm) equivalent diameter break sizes

For the BWR (Peach Bottom), the analyzed sequences were:

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

Since 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 of ~1580 C to 1600 C). This is the point at which the reaction becomes more energetic and significant oxidation of the cladding is more likely. In reality, even this point is likely to be prior to the time at which a significant fission product release from the fuel will have occurred.

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, Supporting Requirement SC-A2) were considered. These include 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., 2200 F [1204 C]) is reached, relative to the time that the zirconium/water transition temperature range (1580 C – 1600 C) is reached. In all cases, except one, the proposed surrogate is reached prior to the oxidation transition temperature (noted as "Time Rapid Core Damage" in the figure). A peak cladding temperature of 2200 F (1204 C) achieves the following characteristics:

- Always precedes oxidation transition;
- It is not overly conservative;
- It is equally applicable for both PWRs and BWRs;
- The timing between 2200 F (1204 C) and oxidation transition is relatively similar between the different sequences analyzed; and
- It is consistent with the criteria contained in 10 CFR 50.46.

For the latter point, the conservatism (i.e., safety margin) in 10 CFR 50.46 is due to uncertainty in LBLOCA thermal-hydraulic analysis. For PRA usage, the margin is for a different reason (i.e., the desire to have a specific criterion that can be used for all sequences). For the reasons cited above, 2200 F (1204 C) peak cladding temperature is the surrogate that will be used to define core damage in the later SPAR model success criteria MELCOR analyses.

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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.

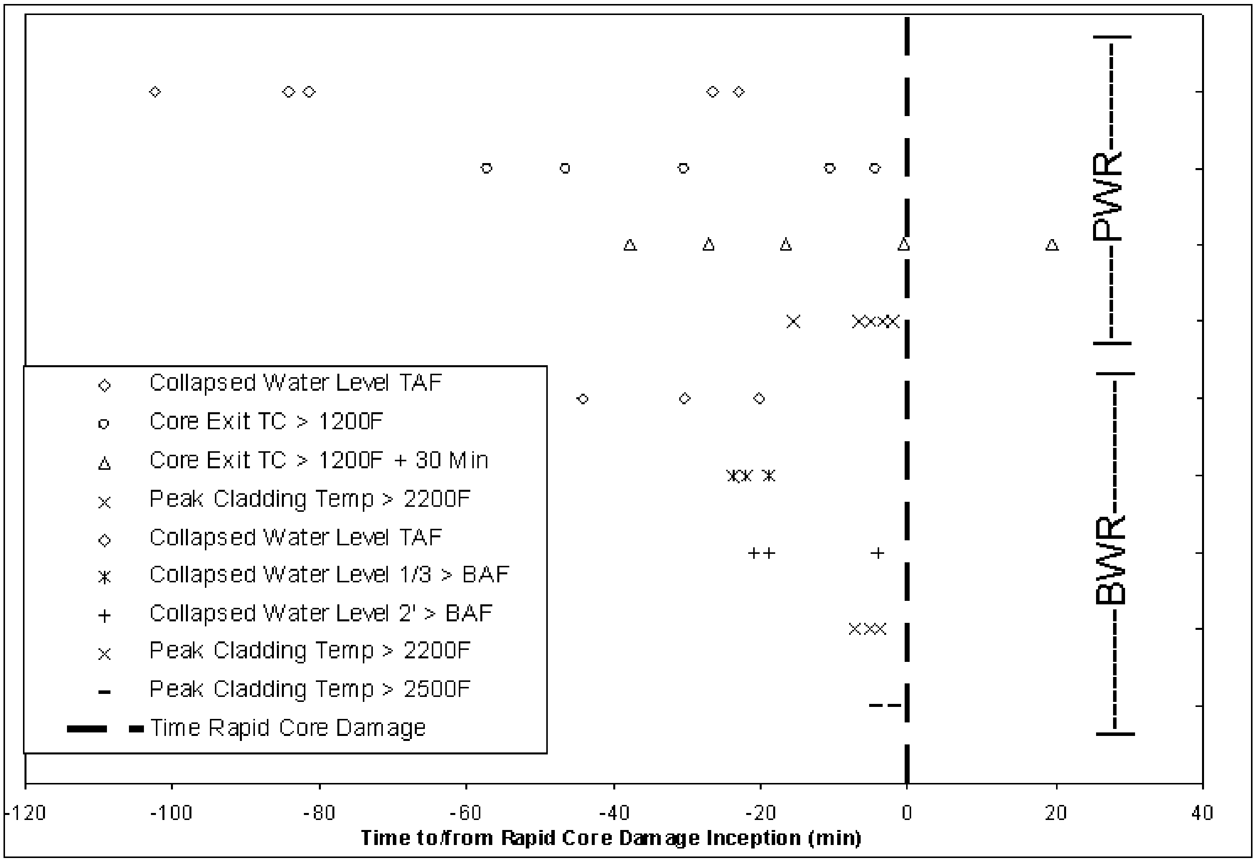

Figure 1: Summary of Core Damage Surrogate Calculations
1200 F = 649 C; 2200 F = 1204 C; 2500 F = 1371 C

Major Plant Characteristics

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

Surry

To the level of detail needed for this work, Surry Units 1 and 2 were considered to be identical. Each unit is a 3-loop Westinghouse with a sub-atmospheric containment. Each has 3 high-head safety injection pumps, and 2 low-head safety injection pumps. The latter are also required for high-pressure recirculation (in order to provide sufficient NPSH to the high head pumps when using the containment sump as a water source). The minimum technical specification RWST volume is 387,100 gallons (1,465 m^3). The water source for ECCS automatically transfers from the RWST to the containment sump when RWST water level drops below 13.5%. This transfer operation takes 2.5 minutes due to the time for the sump isolation valves to fully open.

The containment spray system in injection mode relies on 2 pumps rated @ 3200 gpm (12.1 m^3/min) per pump (which includes ~300 gpm [1.14 m^3/min] per pump of bleed-off flow[3]) and draws from the RWST. Containment spray automatically actuates at 25 psia (0.17 MPa)

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

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 utilizes 4 pumps (2 in containment and 2 outside of containment) which are each 3,500 gpm (13.2 m^3/min), and take suction from the containment sump.

Peach Bottom

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 RCIC system has a capacity of 600 gpm (2.3 m^3/min) at 150 to 1150 psig (1.0 to 7.9 MPa). The HPCI system capacity is 5,000 gpm (18.9 m^3/min). The CST is the preferred source until low level in the CST (< 5 ft [1.5 m]) 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 (3.4 and 10.3 bars), respectively. RCIC and HPCI systems will automatically isolate with a low steam line pressure of 75 psig (5.1 bars). RCIC and HPCI pump bearings are rated for 210 F (99 C). The high-capacity 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 suppression pool has a technical specific maximum temperature limit of 95 F (35 C), and a volume of 127,300 ft^3 (3,605 m^3).

MELCOR Model

Plant Representation

The Surry and Peach Bottom models used for this analysis are based on the models utilized in the SOARCA study[4]. Efforts have been taken to ensure that the models appropriately reflect the as-built / as-operated plant, including discussions with plant operation staff, plant 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 thus 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 that study. In another case (RCP seal leakage) the models used here differ from those used in the final SOARCA analysis. The modeling of RCP seal leakage is described in the Surry station blackout analysis section. Below is a brief overview of the Surry and Peach Bottom models, followed by some discussion of MELCOR's validation base.

Basic features of the Surry model, especially in cases where they differ from the SOARCA model, are outlined at the beginning of Appendix A. Included are reactor trip signals modeled, the ECCS injection set-points, 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 set-points.

Figure 2 shows a plan view of the MELCOR model for the Surry 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

[4] Documentation for the SOARCA study is expected to be externally available in early 2010.

the temperature ranges of interest here (i.e., those preceding core damage). The main coolant pumps are tripped on power failure or voiding in the loop. The core is nodalized into 10 axial levels and 5 radial rings. Each axial level is comprised of 2 thermal response (i.e., COR) nodes and 1 hydrodynamic (i.e., CVH) volume. Safety systems are modeled using injection points, and the relevant portions of the RPS and control systems are modeled using MELCOR control functions. For the secondary side, both TD-AFW and 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 fancoolers are also modeled.

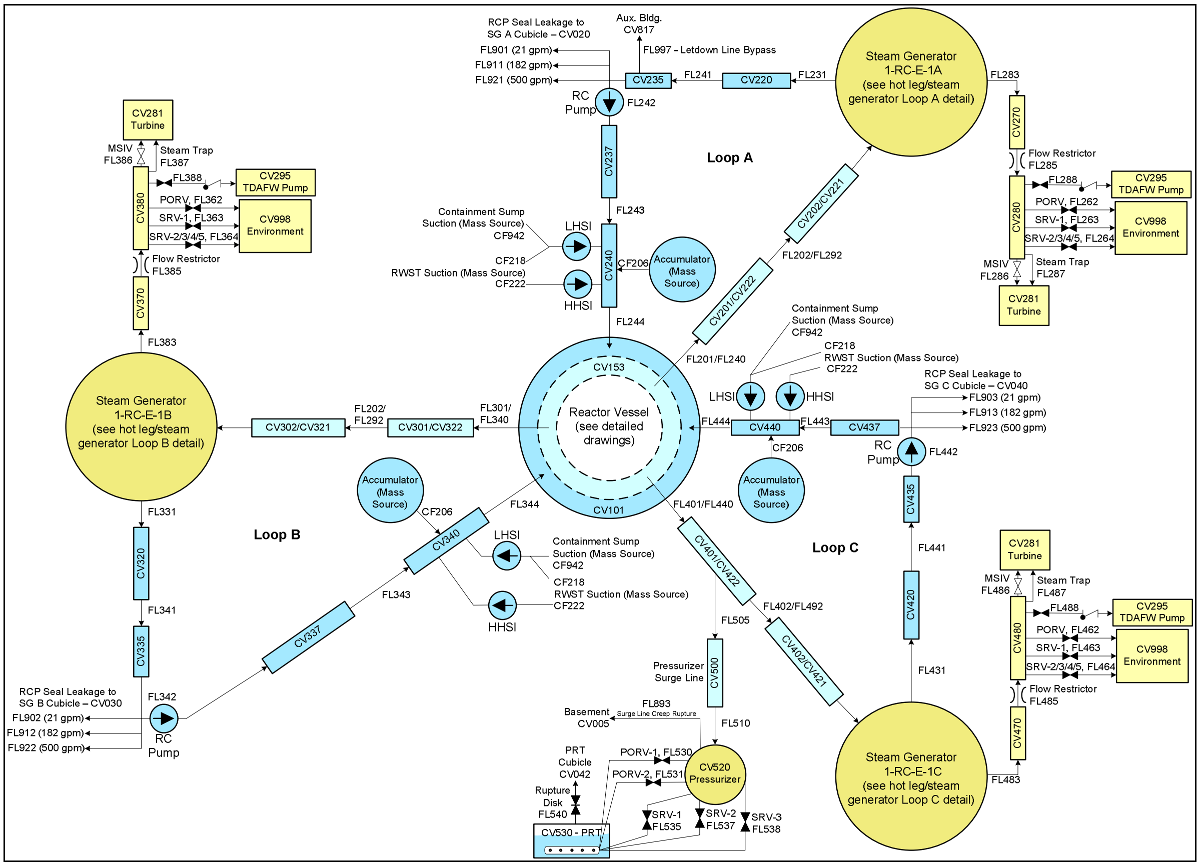

Figure 2: Plan View of the Surry MELCOR RCS Model

Figure 3 shows a schematic of the Peach Bottom MELCOR model including the reactor pressure vessel, wetwell and the safety systems. The drywell (not shown) has 4 control volumes representing the pedestal region, lower/upper drywell and the upper head. The vessel (excluding the core region) is represented by seven control volumes with connections to various safety systems including CRD, RCIC, HPCI, LPCS, and 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 SRVs (including dedicated ADS valves) are modeled with flow paths on two steam lines (a single steam line A, and combined steam line 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 5 radial rings. Like the Surry model, the core decay power is based on a number of ORIGEN calculations for each radial ring. Since no changes were made from the SOARCA model, Appendix B does not include the same introductory information as Appendix A.

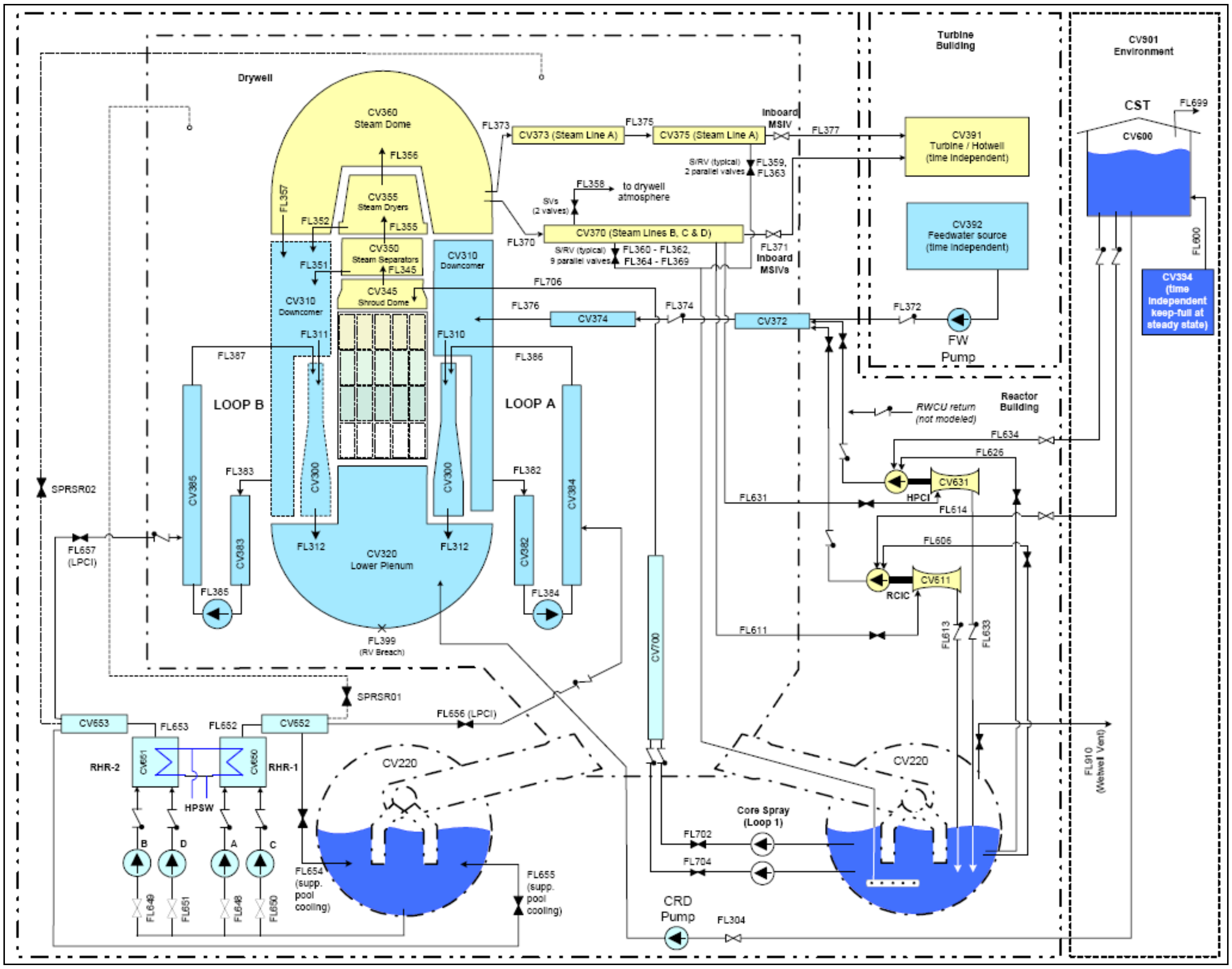

Figure 3: Schematic of the Peach Bottom RCS Nodalization

In order to model failure of PORVs or SRVs, one of three approaches is used as designated in the boundary condition descriptions for the various cases. Either (a) the relief valve can not stick open, (b) the relief valve sticks open on the first lift, or (c) the relief valve sticks open after n lifts, where n is a user-prescribed number. For Surry, the values used in the February 2009 interim documentation have been updated to be more realistic. The table below provides a synopsis of the basis for these numbers, including the specific value for each type of valve.

Table 1: Comparison of Values for Surry Stuck-Open Valves

	Surry IPE *(used for the February 2009 interim analysis)*		**Circa 2006 Surry PRA, SOARCA** *(used for all calculations in this document)*	
	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 for interim calculations (such as those presented in the February 2009 version of this document) was 622 lifts, which was based on a probability of failure per demand of 0.0037 at a cumulative probability of 0.9. In the current revision of the analysis, this number has been updated to 187 lifts, which reflects the same per demand failure probability at a cumulative failure probability of 0.5.

<u>MELCOR Validation</u>

The MELCOR code is designed to run best-estimate accident simulations. 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 (FSAR) Chapter 6, Chapter 15, and Chapter 19 audit calculations, and was the tool used for the post 9/11 security assessments and the SOARCA project. It has also previously been used to assess Significance Determination Process (SDP) issues. As such, it is an ideal tool for use in this project.

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

- Quench experiment 11, simulating a small break LOCA with late vessel depressurization to investigate response of overheated rods under flooding conditions
- The Three Mile Island Unit 2 accident (SBLOCA)
- 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 blow-down experiments characterizing single and two phase blow-down, liquid carryover, and water level swell
- General Electric Mark-III tests with steam blow-down into the suppression pool investigating vent clearing and heat transfer models
- Containment thermal-hydraulic phenomena studied in various experimental facilities including NUPEC (mixing and stratification), HDR (blow-down into containment), and CVTR (steam condensation in the presence of non-condensables)
- Small scale experiments to test condensation models including Wisconsin flat plate experiments and Dehbi tests

MELCOR Results

The detailed results for Surry and Peach Bottom are provided in Appendices A and B, respectively. The following subsections summarize these results in a standard format. First, a brief description of the scenario is provided. Next, key assumptions as well as operator actions are listed. Then, results are provided in tabular form. Finally, the timing to key events is provided, again in tabular form.

A number of different scenarios are analyzed. For Surry, the analyzed scenarios are

- Small LOCAs to investigate the dependency on sump recirculation
- Feed & bleed (during loss of all feedwater) to investigate the number of PORVs and HHSI pumps needed
- Steam generator tube rupture events to provide updated sequence timings
- Station blackout sequences to provide update sequence timings, and
- Medium and large break LOCAs to look at the systems needed for successful inventory control

For Peach Bottom, the analyzed scenarios are:

- Inadvertently opened relief valve cases to investigate the effects of RCIC, HPCI, and CRD injection
- Station blackout sequences to investigate the time for A/C recovery, the time for suppression pool heatup, and the times associated with the loss of RCIC and HPCI

In many cases the analyzed sequence progressions make assumptions about unavailability of systems and/or operator actions that are not taken. These assumptions often stem from the particular sequence in the event tree that is being informed, which may not be the most probable sequence. In other cases, these characteristics are not included due to resource constraints. In all cases, these assumptions are articulated in the following sections.

<u>SBLOCA Dependency on Sump Recirculation (Surry)</u>

This series of cases investigates the timing to RWST depletion (and thus switchover to recirculation) for small break LOCAs in which operators take minimal action. The varied parameters are break size (0.5 inch, 1 inch, 2 inch (1.3 cm, 2.5 cm, 5.1 cm)) and containment spray function (available or not available). In addition, sensitivity cases are performed to look at the effects of securing HHSI pumps and secondary side cooldown. These sensitivity cases demonstrate the impact of HHSI and secondary-side cooldown on RCS pressure and RHR entry timing. Due to project resource considerations, they take a simplified scoping approach, and do not necessarily represent the actual plant operating procedures. For this reason, they should be used with caution.

Results are provided in the tables below. For 2 inch (5.1 cm) breaks, the system depressurizes causing the primary side pressure to be below the maximum 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, since this causes the 0.5 inch (1.3 cm) break to become a 1.9 inch (4.8 cm) break[5]. Note that operator action to reduce injection and thus limit PORV cycling was not modeled. For 0.5 inch (1.3 cm) cases where the PORV does not stick

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

open, the system does not depressurize. Finally, for the 1 inch (2.54 cm) cases, the break is not large enough to cause depressurization (due to 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:

- For the 0.5" (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 to prevent 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) after containment pressure drops below 12 psia (0.083 MPa)
- RCPs trip at 10% voiding (same as assumption used in SOARCA)
- Manual RCS cooldown and depressurization are not modeled, except in a simplified manner for the sensitivity cases

Table 2: Surry SBLOCA Sump Recirculation Results

<table>
<tr><th>Case</th><th>Size (in)</th><th>HHSI Pumps</th><th>PORV Treatment</th><th>Sprays</th><th>Secondary Side Cooldown</th><th>Core Uncover (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 2 HHSI pumps are secured at 15 minutes
[3] It is assumed that 2 HHSI pumps are secured at 15 minutes, and the 3rd pump is secured at 30 min followed by secondary side cool down at 100 F/hr (55.6 C/hr)
[4] These cases reach RHR entry conditions (both temperature and pressure) prior to heatup
1 inch = 2.54 cm; 2 inch = 5.1 cm; 0.5 inch = 1.3 cm

Table 3: 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	-	-
Spray Injection	-	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 > 2200F)	11.9	9.93	10.0	-	-	-

2200 F = 1204 C

Table 4: 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	-	-	-
Spray Injection	-	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 > 2200F)	-	-	9.61	-	25.1	21.4

2200 F = 1204 C

Feed & Bleed PORV Success Criteria (Surry)

The initiating event of interest for this calculation is loss of main feedwater. 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. The injection source is HHSI (initially from RWST) and the bleed path is the PORV(s). Containment sprays actuate periodically due to the increase in containment pressure from primary coolant being expelled through the PORV into the PRT, and subsequent release of primary coolant into containment after the PRT rupture disk fails. The analysis performed here demonstrates that 1 PORV provides a sufficient bleed path to maintain quasi-steady conditions on the primary side. 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. For this case, RWST level is depleted after ~ 9.5 hours, with much of the depletion being caused by the actuation of the high-capacity containment spray system (see plots in Appendix A). The case does eventually proceed to core damage, but only because HHSI recirculation (which would actuate upon RWST depletion) is not modeled.

Key assumptions and operator actions:

- For this simulation, reactor trip is assumed to happen immediately upon loss of main feedwater
- Operators secure containment sprays (and reset to allow subsequent actuation) after containment pressure drops below 12 psia (0.083 MPa)
- HHSI recirculation is not modeled
- The PORV opens automatically due to high pressure set-point (i.e., no manual operator action)
- Manual RCS depressurization and cooldown is not modeled
- RCPs trip at 10% voiding

Table 5: Surry Feed & Bleed PORV Success Criteria Results

Case	Auto SI		HHSI Pumps			PRZ PORVs (automatically open)			Core Uncovery (hr)	Core Damage (hr)
	Yes	No	1	2	3	0	1	2		
1	X		X				X		10.9[1]	13.5[1]

[1] Core damage is an artifact of the assumed unavailability of HHSI recirculation

Table 6: Surry Feed & Bleed PORV Success Criteria Key Timings

Event	Time (hr)
Reactor Trip, MFW, MD-AFW, TD AFW Unavailable	0
SG dryout	1.11
PRT rupture disk open	1.56
SI signal (containment pressure > 1.22 bar)	1.96
MCP trip (10% void)	2.05
Spray started in injection mode (containment pressure > 1.72 bar)	3.84
RWST Depletion (<13.5%)	9.43
Core uncovery	10.90
Core damage (max temp > 2200F)	13.53

1.22 bar = 0.122 MPa; 2200 F = 1204 C

<u>SGTR Event Tree Timing (Surry)</u>

These calculations look at the time available to take corrective actions for events involving spontaneous (as opposed to accident-induced / consequential) tube rupture events. For reference, the effective leak size of a 1-tube rupture is ~ 1 inch (2.5 cm). Past operating experience for steam generator tube rupture events suggests that, in some cases, the time between the initiating event and initiation of RHR can be significant. For example, this timing ranges from 3.25 hours to 21.5 hours, for the events covered in NUREG/CR-6365, “Steam Generator Tube Failures” (April 1996). Even so, the results provided below show that there is substantial time available for corrective actions due to the availability of secondary side heat removal. At 24 hours, the fuel temperatures for all three cases are stable at < 550 F (288 C).

Key assumptions and operator actions:

- Operators secure either 1 or 2 HHSI pumps at 15 minutes (depending on the case) and manually control auxiliary feedwater to maintain SG level (standard practice)
- HHSI recirculation is not modeled
- RCPs trip at 10% voiding

- Manual isolation of the faulted SG is not modeled
- Manual actions to model long-term heat removal (EOP ECA 3.1/3.2) are not modeled

Table 7: Surry SGTR Results

Case	No. Tubes	HHSI Pumps	TD-AFW	MD-AFW	Core Uncover	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 8: Surry SGTR Key Timings

Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)
Reactor Trip	0.048	0.012	0.012
HHSI Initiates (3 pumps)	0.051	0.013	0.013
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

[1] Recall that since the RCS leak location is the ruptured steam generator tube(s), a substantial amount of water is expelled from the system via the SG relief valves (rather than in to containment).

PWR Station Blackout (Surry)

A number of simulations were run for station blackout sequences to investigate the effects of RCP seal failures, PORV operation, and TD-AFW availability on the time available to recover A/C power and re-establish core cooling. Along with the variations in system conditions, some other factors that affect the time to core damage are the time to depletion of the emergency CST tank for cases with TD-AFW available, and the system pressure / occurrence of natural circulation (Case 4). These cases all assume infinite DC power, which is an intentional modeling artifact to investigate timing. In reality, battery depletion would cause a loss of DC power prior to core damage for Cases 4 and 6, and perhaps for Case 8.

The RCP seal leakage rates and timing are taken from the WOG 2000 seal leakage model (WCAP-15603, “WOG2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs,” Rev. 1, May 2002) for “new” high-temperature seals, as modified by the staff’s associated Safety Evaluation Report[6], which is the seal leakage model used in the current Surry SPAR model. This is the same model that is invoked in a later PRA guidance topical report, WCAP-16141, “RCP Seal Leakage PRA Model Implementation Guidelines for Westinghouse PWRs,” August 2003. The SER 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 4 possible leakage rates for use in PRAs, with the onset of increased leakage occurring at 13 minutes in all cases. The leakage rates and their conditional probabilities are reproduced in Table 9, along with some associated timings from the Westinghouse Emergency Response Guidelines as reproduced in the Surry SPAR v3.46 model documentation (July 2008). For the current work, cases are run for three of these leakage sizes (21 gpm [0.079 m^3/min], 182 gpm [0.689 m^3/min] and 500 gpm [1.89 m^3/min]).

[6] Memorandum from Michael Johnson and Jared Wemiel to Herbert Berkow, *Safety Evaluation Report for the Westinghouse Owners Group (WOG) Topical Report WCAP-15603, Revision 1, “WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs,”* April 8, 2003. [ML030980126]

Table 9: RCP Seal Leakage Details

Seq. #	Leak Rate at > 13 minutes (gpm)	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
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 calculations performed here are in good agreement with the results shown in Table 9. For analogous cases (i.e., those with TD-AFW available and no secondary-side depressurization):

- time to core uncovery is ~ 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 ~ 4 hours for the intermediate leakage rate of 182 gpm/RCP (0.689 m^3/min/RCP), as compared to 3 hours in the Westinghouse calculations
- time to core uncovery is ~ 13 hours for the normal leakage rate of 21 gpm/RCP (0.079 m^3/min/RCP) , which is identical to the Westinghouse calculation result

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

WCAP-16396-NP (January 2005) provides arguments for why the NRC's safety evaluation of the WOG 2000 model, and the WOG 2000 model itself, result in conservative estimation of RCP seal leak rates. These conservatisms are associated with both the leak rates assumed, as well as the timing of seal failure (which is reported to vary from 8 minutes to 40 minutes, as compared to the 13 minutes used in the WOG 2000 model). This topical report quantitatively assesses the effects of these conservatisms on accident progression timings (specifically the time for loss of pressurizer level and core uncovery). The topical report concludes that the conservatisms can have a substantial effect on 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." These conclusions lead 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 won't affect the overall conclusions drawn from the models. Even so, the potential conservatisms could affect intermediate PRA results, such as the human failure probability associated with a particular action.

Regarding 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 clad temperature of 2200 F [1204 C]), and
- recovery of HHSI at 1.64 hours, (i.e., ½ hour prior to core damage).

As shown in Figure 5, for the former sensitivity case recovery of injection at this point was not sufficient to avert fuel melting. For the latter case, 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.

Key assumptions and operator actions:

- Operators manually control auxiliary feedwater to maintain SG level (standard practice)
- Infinite DC power for control of TD-AFW
- SRV sticks open on the 1st lift for some cases (as specified below)
- For cases with RCP seal failure, failure is assumed to occur at 13 minutes[7]
- Manual actions for rapid secondary-side depressurization are not modeled

Table 10: Surry Station Blackout Results

<table>
<tr><th>Case</th><th>Seal leakage rate[1] after failure (gpm per pump)</th><th>Seal failure (min)</th><th>SRV stuck-open</th><th>TD-AFW</th><th>Core Uncover (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td rowspan="2">500</td><td rowspan="2">13</td><td rowspan="4">N/A[2]</td><td>No</td><td>1.4</td><td>2.1</td></tr>
<tr><td>2</td><td>Yes</td><td>1.6</td><td>2.3</td></tr>
<tr><td>3</td><td rowspan="4">21</td><td rowspan="4">-</td><td>No</td><td>2.3</td><td>3.4</td></tr>
<tr><td>4</td><td>Yes</td><td>13.3</td><td>16.3</td></tr>
<tr><td>5</td><td rowspan="2">1st lift off</td><td>No</td><td>2.1</td><td>2.6</td></tr>
<tr><td>6</td><td>Yes</td><td>13.0</td><td>13.8</td></tr>
<tr><td>7</td><td rowspan="2">182</td><td rowspan="2">13</td><td rowspan="2">N/A[2]</td><td>No</td><td>2.0</td><td>3.1</td></tr>
<tr><td>8</td><td>Yes</td><td>3.9</td><td>4.8</td></tr>
</table>

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

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

[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

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

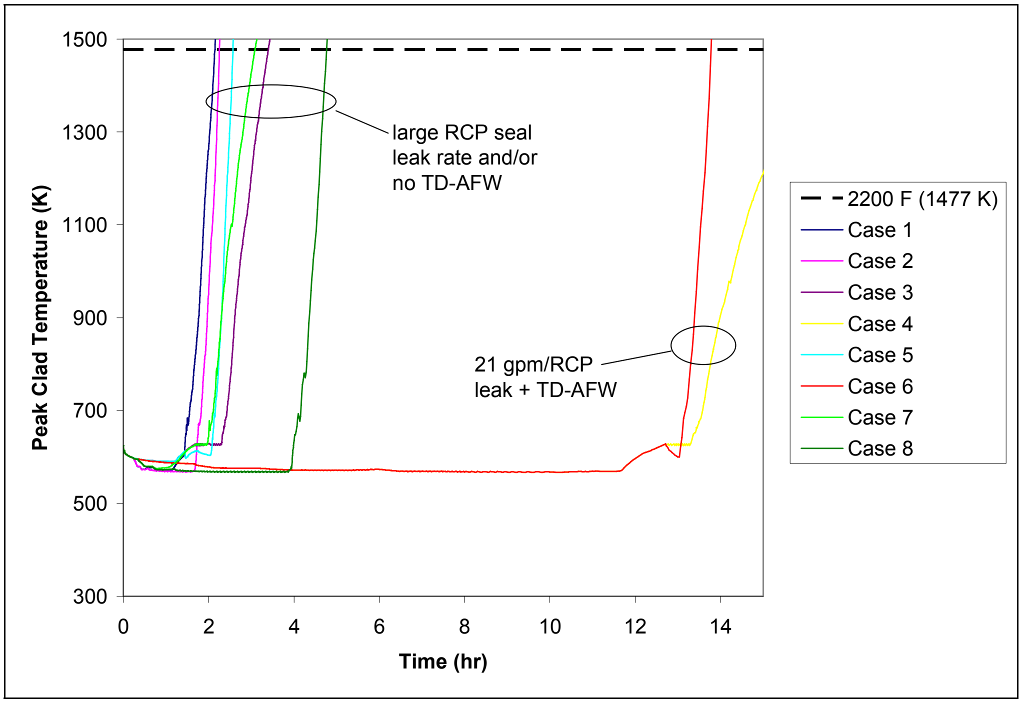

Figure 4: PCT Signatures for All Surry SBO Cases

Table 11: Surry Station Blackout Key Timings (Cases 1-2)

Event	Case 1 (hr)	Case 2 (hr)
Reactor Trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0
Seal leakage (21 gpm/pump)	0	0
Seal failure (500 gpm/pump)	0.22	0.22
Primary side SG tubes water level starts to decrease	0.52	0.52
Primary side SG tubes dry	0.96	0.98
SG dryout	1.16	-
Core uncovery	1.40	1.63
Gap release	1.92	2.15
Core damage (max temp > 2200F)	2.14	2.25

500 gpm = 1.89 m^3/min; 21 gpm = 0.076 m^3/min; 2200 F = 1204 C

Table 12: Surry Station Blackout Key Timings (Cases 3-6)

Event	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 > 2200F)	3.40	16.33	2.57	13.80

21 gpm = 0.076 m^3/min; 2200 F = 1204 C

Table 13: Surry Station Blackout Key Timings (Cases 7-8)

Event	Case 7 (hr)	Case 8 (hr)
Reactor Trip, RCP trip, MFW/TD-AFW/MD-AFW	0	0
Seal leakage (21 gpm/pump)	0	0
Seal failure (182 gpm/pump)	0.22	0.22
Primary side SG tubes water level starts to decrease	1.04	1.01
Primary side SG tubes dry	1.52	2.22
SG dryout	1.22	-
Core uncovery	1.98	3.88
Gap release	2.63	4.00
Core damage (max temp > 2200F)	3.09	4.77

182 gpm = 0.689 m^3/min; 21 gpm = 0.076 m^3/min; 2200 F = 1204 C

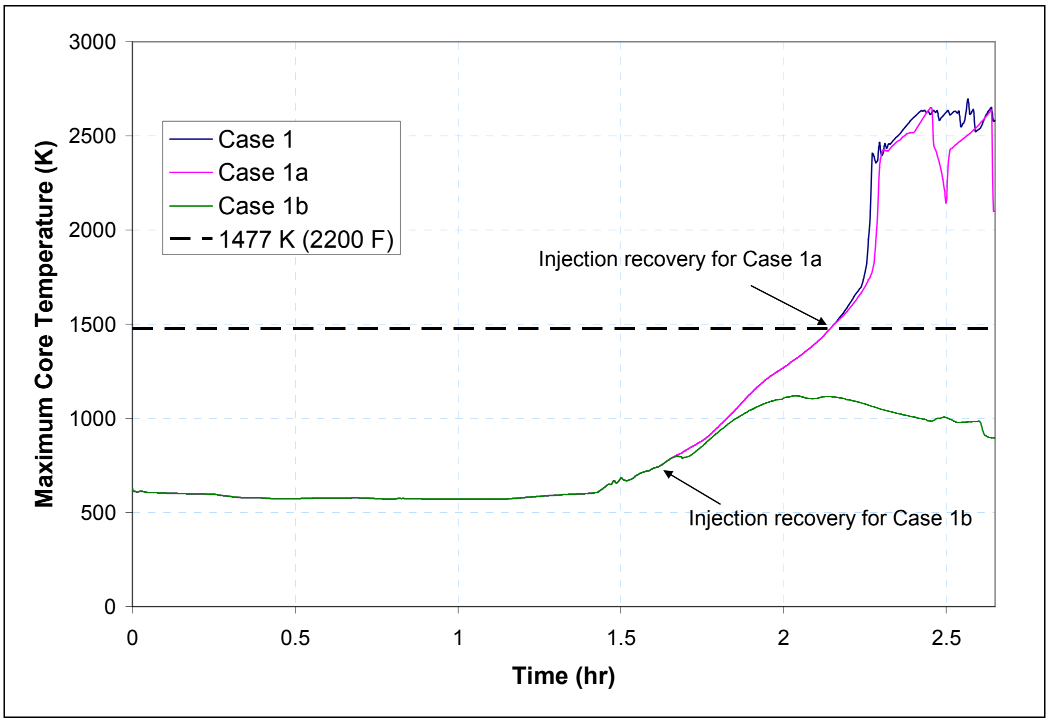

Figure 5: Surry Injection Recovery Sensitivity Cases

Accumulator Injection (Surry)

The final set of Surry sequences investigates the need for accumulators (as well as HHSI and LHSI) for a spectrum of LOCA break sizes. Break sizes from 2 inches (5.1 cm) all the way up to a double-ended break were analyzed, as shown in the table below. By convention, the breakdown in the LOCA spectrum for most Westinghouse PWRs is ½ inch to 2 inches (SBLOCA), 2 inches to 6 inches (MLOCA) and 6 inches and greater (LBLOCA). The break location for the current analyses is always the horizontal section of the cold leg in the pressurizer loop.

As will be shown below, some of these accidents progress very quickly, with core uncovery taking place within the first minute (for large LOCAs). Given this situation, it is of interest to look at the degree of margin between the peak clad temperature for cases that are deemed successful, relative to the core damage definition being used. Table 14 presents these figures of merit, demonstrating that the highest MLOCA PCT (for a success case) is 483 F (268 C) from the core damage definition used here, and the highest LBLOCA PCT (for a success case) is 706 F (392 C) from the core damage definition. This demonstrates that there is significant margin in these cases, which helps in counteracting the additional model uncertainty that might be expected for these more quickly-evolving accidents.

Table 14: PCT Ranges for Accumulator Success Cases

Range of Break Size	Range of PCT for Success Cases	Range of Margin: 2200F – PCT (1204C – PCT)
MLOCA (2-inch to 6-inch)	606F – 1717F (319C – 936C)	483F – 1594F (268C – 885C)
LBLOCA (6-inch to double-ended)	606F – 1494F (319C – 812C)	706F – 1594F (392C – 885C)

The results in Table 15 are distilled here to identify the minimal equipment needed to avoid core damage. For medium LOCAs:

- For 6 inch (15.2 cm) breaks, Cases 2, 6, and 8 demonstrate that any 2 of the following 3 would be adequate (1 HHSI, 1-of-2 accumulators, and/or 1 LHSI)
- For 4 inch (10.2 cm) breaks, Case 13 demonstrates that 1-of-2 accumulators and 1 LHSI is not adequate, leaving 2 remaining success paths: 1 HHSI and 1-of-2 accumulators <u>or</u> 1 HHSI and 1 LHSI
- For 2 inch (5.1 cm) breaks, Case 9 (1 HHSI) would be expected to be successful with either 1-of-2 accumulators (which would start to inject at ~ 50 minutes, which is over 4 hours prior to heatup) or 1 LHSI (which would allow for HHSI recirculation)
- Cases 16 and 17 demonstrate that injection is sufficient for removing decay heat in the absence of AFW

The resulting minimal equipment success criteria for medium LOCAs is 1 HHSI and 1-of-2 accumulators <u>or</u> 1 HHSI and 1 LHSI.

For large LOCAs:

- For 6 inch (15.2 cm) breaks, Cases 2, 6, and 8 demonstrate that any 2 of the following 3 would be adequate (1 HHSI, 1-of-2 accumulators, and/or 1 LHSI)
- For 8 inch (20.3 cm) breaks, Cases 3 and 18 confirm the above; note that the 6-inch case with 1-of-2 accumulators and 1 LHSI (Case 6) is viewed as a limiting case in terms of large LOCAs, because larger breaks will depressurize faster
- For 10 inch (25.4 cm) breaks, Cases 4 and 19 (in conjunction with Case 6) confirm the above
- For the double-ended break, Case 10 demonstrates that only LHSI is necessary

The resulting minimal equipment success criteria for large LOCAs is 1 HHSI and 1-of-2 accumulators <u>or</u> 1 HHSI and 1 LHSI <u>or</u> 1-of-2 accumulators and 1 LHSI.

Key assumptions and operator actions:

- Break is in the horizontal section of the cold leg, in the pressurizer loop
- RCPs trip at 10% voiding
- HHSI recirculation is not modeled

Table 15: Surry LOCA (Accumulator Injection) Results

Case	Break Size (in)	# HHSI Pumps	# Accum.	# LHSI Pumps	AFW?	Core Uncovery (hr)	Core Recovery (hr)	Core Damage (hr)
9	2	1	0	0	Yes	5.45	No	6.1[1]
15		0	2	1		0.41	No	0.73
1	4	1	0	1		0.09	Yes	No
11		1	0	0		0.09	Yes	No
12		0	0	1		0.10	Yes	0.27
13		0	1	1		0.10	Yes	0.27
14		0	2	1		0.10	Yes	No
2	6	1	0	1		0.04	Yes	No
5		0	0	1		0.04	Yes	0.16
6		0	1	1		0.04	Yes	No
7		1	0	0		0.07	Yes	0.28
8		1	1	0		0.08	Yes	No
16		1	0	1	No	0.04	Yes	No
17		1	1	0		0.06	Yes	No
3	8	1	0	1	Yes	0.02	Yes	No
18		1	1	0		0.01	Yes	No
4	10	1	0	1		0.01	Yes	No
19		1	1	0		0.01	Yes	No
10	DE	0	0	1		0.02	Yes	No

[1] Note that core damage results from the inability to go to HHSI recirculation, due to the unavailability of LHSI
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 16: Surry LOCA (Accumulator Injection) Key Timings (2” Breaks)

Event	Case 9 (hr)	Case 15 (hr)
Reactor Trip	0.01	0.003
HHSI Injection	0.01	-
RCP Trip (10% void)	0.28	0.07
Spray Injection	1.14	-
Core uncovery (water < TAF)	5.45	0.41
LHSI Injection	-	-
Maximum clad temperature timing (maximum temperature)	6.1 (1478K[1])	0.73 (1477K[1])
Core covered	-	-

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

Table 17: Surry LOCA (Accumulator Injection) Key Timings (4” Breaks)

Event	Case 1 (hr)	Case 11 (hr)	Case 12 (hr)	Case 13 (hr)	Case 14 (hr)
Reactor Trip	0.003	0.003	0.003	0.003	0.003
HHSI Injection	0.003	0.004	-	-	-
RCP Trip (10% void)	0.040	0.04	0.04	0.04	0.04
Spray Injection	0.075	0.08	0.07	0.07	0.07
Core uncovery (water < TAF)	0.086	0.09	0.10	0.10	0.10
LHSI Injection	0.294	-	0.33	0.45	0.73
Maximum clad temperature timing (max. temperature)	0.339 (982K)	0.53 (1209K)	0.27 (1477K[1])	0.27 (1477K[1])	0.73 (1183K)
Core covered	0.378	>0.83	>0.42	0.50	0.79

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

Table 18: Surry LOCA (Accumulator Injection) Key Timings (6” Breaks Group 1)

Event	Case 2 (hr)	Case 5 (hr)	Case 6 (hr)
Reactor Trip	0.002	0.002	0.002
HHSI Injection	0.002	-	-
RCP Trip (10% void)	0.019	0.02	0.02
Spray Injection	0.033	0.03	0.03
Core uncovery (water < TAF)	0.042	0.04	0.04
LHSI Injection	0.125	0.14	0.18
Maximum clad temperature timing (maximum temperature)	0.150 (774K)	0.16 (1478K[1])	0.16 (990K)
Core covered	0.194	0.23	0.20

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

Table 19: Surry LOCA (Accumulator Injection) Key Timings (6” Breaks Group 2)

Event	Case 7 (hr)	Case 8 (hr)	Case 16 (hr)	Case 17 (hr)
Reactor Trip	0.002	0.002	0.002	0.002
HHSI Injection	0.002	0.002	0.002	0.002
RCP Trip (10% void)	0.02	0.02	0.022	0.022
Spray Injection	0.03	0.03	0.032	0.032
Core uncovery (water < TAF)	0.07	0.08	0.042	0.061
LHSI Injection	-	-	0.128	-
Maximum clad temperature timing (maximum temperature)	0.28 (1478K[1])	0.04 (592K)	0.152 (775K)	0.04 (575K)
Core covered	0.68	0.10	0.194	0.120

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

Table 20: Surry LOCA (Accumulator Injection) Key Timings (≥ 8” Breaks)

Event	Case 3 (hr)	Case 18 (hr)	Case 4 (hr)	Case 19 (hr)	Case 10 (hr)
Reactor Trip	0.002	0.002	0.001	0.001	0.001
HHSI Injection	0.002	0.002	0.001	0.001	-
RCP Trip (10% void)	0.009	0.009	0.008	0.008	0.001
Spray Injection	0.014	0.013	0.008	0.008	0.005
Core uncovery (water < TAF)	0.017	0.014	0.011	0.008	0.022
LHSI Injection	0.069	-	0.044	-	0.005
Maximum clad temperature timing (maximum temperature)	0.097 (851K)	0.400 (1085K)	0.075 (850K)	0.297 (835K)	0.036 (1043K)
Core covered	0.142	0.908	0.117	0.867	0.053

SRV/RCIC 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 SRV1 is opened 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 effect. The calculations below demonstrate that any of the injections options considered will prevent heatup prior to 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 2nd 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 prior to core damage, with a maximum cladding temperature of 939 C (1722 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 110C higher, but still over 500C below the onset of core damage.

Key assumptions and operator actions:

- Reactor trip and 1 SRV stuck open at time zero (unless noted otherwise)
- 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 1 pump, or 210 gpm (0.795 m^3/min) to 300 gpm (1.14 m^3/min) for 2 pumps
- RCIC/HPCI isolate on low steam line pressure of 75 psig (0.52 MPa)

Table 21: Peach Bottom Stuck-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 Uncover (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>N/A</td><td>N/A</td></tr>
<tr><td>2</td><td rowspan="5">No</td><td>Yes</td><td>N/A</td><td>N/A</td></tr>
<tr><td>3</td><td rowspan="4">No</td><td>1 @ t=0 and 2 @ t=10 min</td><td>N/A</td><td>N/A</td></tr>
<tr><td>4</td><td rowspan="2">1 @ t=0 and 2 @ t=20 min</td><td>N/A</td><td>N/A</td></tr>
<tr><td>4a[1]</td><td>N/A</td><td>N/A</td></tr>
<tr><td>5</td><td>No</td><td>N/A</td><td>N/A</td></tr>
</table>

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

Table 22: Peach Bottom Stuck-Open SRV Key Timings (Case 1-5)

Event	Case 1 (hr)	Case 2 (hr)	Case 3 (hr)	Case 4 (hr)	Case 4a (hr)	Case 5 (hr)
SRV1 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 steam line pressure < 0.52 MPa (75 psig)	0.82	5.59	-	-	-	-
Maximum cladding temperature: 1204 C (2200 F)	No heatup	No heatup	0.78 (513 C)	0.76 (557 C)	0.67 (668C)	0.75 (939 C)

[1] Reactor trips at 8 seconds on low RPV level

BWR Station Blackout (Peach Bottom)

Numerous simulations were run for Peach Bottom to look at the timing associated with suppression pool heatup, specifically during a station blackout event. These calculations investigate variations in the availability of injection sources, the behavior of the SRVs, implementation of HCTL-based depressurization, and the time to battery depletion. A sensitivity case was also performed to look at the effect of recovery, similar to the Surry station blackout sensitivities described previously. 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 regarding SRVs. Recovery of injection at the time of core damage was demonstrated to quickly arrest heatup. For cases where DC is lost after 2 hours, core damage occurs at 4 to 5 hours. For cases where 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 even without a low-pressure injection source to protect containment.) For cases where the SRV does not stick open and where HCTL depressurization is not performed, RCIC or HPCI (depending on which is assumed available) fail after approximately 12 hours due to loss of NPSH, and core damage occurs after 19 hours. Considering all cases, the time lag from uncovery of the top of active fuel to the time of core damage ranges from 0.5 to 1.8 hours.

Key assumptions and operator actions:

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

Table 23: 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 Uncover (hr)</th><th>Core Damage (hr)</th></tr>
<tr><td>1</td><td rowspan="3">No</td><td rowspan="7">No</td><td>No</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>No</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>5.6</td><td>7.2</td></tr>
<tr><td>5</td><td>2 hrs 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>5.6</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>9.1</td><td>10.8</td></tr>
<tr><td>9</td><td>2 hrs 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>8.9</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 1204 C (2200 F) quickly arrests further heatup.

Table 24: 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 (1652 F)	1.02	1.02	0.69
Core damage: max temp > 1204 C (2200 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 25: 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 (1.5m)	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 (1652 F)	19.06	6.99	4.04	7.00
Core damage max temp > 1204 C (2200 F)	19.42	7.17	4.25	7.18
Exhaust pressure exceeded: 0.35 MPa (50 psig)	20.14	-	-	-

Table 26: 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 (1.5m)	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 (1652 F)	18.96	10.59	4.63	10.46
Core damage max temp > 1204 C (2200 F)	19.31	10.8	4.85	10.68
Exhaust pressure exceeded: 1.04 MPa (150 psig)	-	-	-	-

Application of MELCOR Results to Surry and Peach Bottom SPAR Models

Table 27 and Table 28 below (i) provide a summary of the scenarios that have been investigated, (ii) re-cap the boundary and initial condition variations studied using MELCOR, (iii) highlight the relevant parts of the existing Surry and Peach Bottom SPAR models, and (iv) provide proposed changes to these models based on the MELCOR analysis. Where appropriate, insights are offered as to how these results may be applied to SPAR models for other, similar plants. Extension of these results to these other plants is subject to change, based on additional interactions that are planned with internal stakeholders (e.g., the Office of Nuclear Reactor Regulation - NRR).

Table 27: Surry (Units 1 & 2) Success Criteria Update

Feature of interest	Class	MELCOR Variations	Affected Portion of Existing SPAR Model	Proposed Changes
SBLOCA dependency on sump recirculation	Small LOCA	• Break size: 0.5, 1, 2 inches • # of containment spray pumps operating: 0, 2 • PORV treatment: sticks open at 247 lifts, does not stick open	SBLOCA event tree timing for sequences with failure to achieve recirculation	For sequences without modeling of controlled cooldown via operator action, it has not been demonstrated that all break sizes will depressurize to RHR conditions prior to 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, and this may suggest changes to timing issues for particular sequences.
Feed & bleed PORV success criteria	-	• # of PRZ PORVs available: 1	SC for Early Decay Heat Removal for Loss of Main FW event tree: 1 AFW train or **2 PORVs** and 1 Charging train	The following revised success criteria can be applied to Surry, and other plants that are similar[a]: 1 AFW train or [1 charging train and 1 PORV automatically opened in pressure-relief mode]
SGTR event tree timing	-	• # 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 even with minimal 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 re-visiting, pending an assessment of other factors (e.g., consideration of safe/stable state versus mission time).

Feature of interest	Class	MELCOR Variations	Affected Portion of Existing SPAR Model	Proposed Changes
Station blackout		• RCP seal leakage rate: 21, 182, 500 gpm/pump • SRV stuck-open model: 1st lift, never • TD-AFW availability: yes, no	Time to recover A/C power (and re-establish AFW cooling and/or RCS makeup capability)	• For cases w/ TD-AFW and large seal LOCAs (182 or 500 gpm), the calculations confirm the timings currently used in SPAR. • For cases w/ TD-AFW and no large seal LOCA, core damage occurs well after anticipated battery depletion. SPAR currently does not credit AC recovery after battery depletion. • For cases w/out TD-AFW and large seal LOCAs (182 or 500 gpm), the calculations provide the basis for extending core damage time from 1 to 2 (or more) hours, for similar plants[a]. • 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.
Accumulator injection	Medium LOCA[b]	• Break size: 2, 4, 6, 8, 10 inches, double-ended • # of HHSI pumps: 0, 1 • # of LHSI pumps: 0, 1 • # of accumulators: 0, 1, 2 • AFW availability[d]	Success criteria for Inventory Control during Injection Phase for the MLOCA event tree: 1 Charging train and (**2 Accumulators** or 1 AFW train)	For similar plants[a]: 1 HHSI and [1 LHSI or 1-of-2 accumulators on the non-broken cold legs]
	Large LOCA[c]		Success criteria for Inventory Control during Injection Phase for the LBLOCA event tree: **2 Accumulators** and 1 LPI train	Calculations suggest that for Surry, any 2 of the following 3 is sufficient: 1 HHSI and/or 1-of-2 accumulators and/or 1 LHSI. Applicability to other similar plants has not been evaluated yet.

[a] In this case, similar plants would be those with high-volume/high-head SI (CVCS) pumps - 150 gpm (0.568 m^3/min) @ 2500 psi (17.2 MPa), large volume steam generators (series 51 and F) and core thermal power ≤ 2900 Mwt; plants in this category are Beaver Valley 1&2, Farley, North Anna, Harris, Summer, and Surry

[b] Historically 2” to 6” equivalent diameter (from NUREG-1150 and NUREG/CR-5750, Appendix J)

[c] Historically greater than 6” equivalent diameter (from NUREG-1150 and NUREG/CR-5750, Appendix J)

[d] Conventionally, AFW is not needed for success for large LOCA; the break size is large enough to remove decay heat and the system fully depressurizes. This entry in the table refers to the confirmatory (in the sense of decay heat removal without AFW) calculations for a 6-inch (15.2 cm) break documented in Case 16 and Case 17 from Table 15.

500 gpm = 1.89 m^3/min; 182 gpm = 0.689 m^3/min; 21 gpm = 0.076 m^3/min; 0.5 inch = 1.3 cm; 1 inch = 2.5 cm; 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 28: Peach Bottom (Units 2 & 3) Success Criteria Update

Feature of interest	Class	MELCOR Variations	Affected Portion of Existing SPAR Model	Proposed Changes
SRV/RCIC Success Criteria	-	• Injection source: RCIC, HPCI, CRD, none • Timing of 2nd CRD pump initiation: 10 mins., 20 mins.	Effectiveness of injection source for core cooling until low-pressure pumps can provide makeup.	• For RCIC, 4 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, additional discussion is needed due to variability in CRD flows and concerns relative to CRD trip on run-out at low pressures.
Suppression Pool Heatup	Station Blackout	• 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 A/C Power (and re-establish 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 ½ hour (currently no credit is given in any of the SPAR models). • For cases with infinite DC and RCIC/HPCI loss due to 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). • Regarding 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.

Future Work

The work outlined above is very useful for informing success criteria for Surry and Peach Bottom, as well as other similar plants as noted in Table 27 and Table 28. Interactions are ongoing between RES and NRR to define the next phase of this work. These interactions may lead to a shift in focus in the work from the current tact (looking at a handful of success criteria for a couple of plants) to a different tact (looking at a couple of success criteria for a broader range of plants). Several opportunities for advancing this work with regard to either or both of the above approaches are outlined next.

A third plant may be analyzed within the SOARCA project, and some work has already been performed to this end, including the development of a preliminary MELCOR input deck. This plant is a 4-loop Westinghouse with an ice condenser containment. As such, it represents an additional benefit in that it covers a different RCS type (4-loop versus 3-loop) and a different containment type (ice condenser versus large, sub-atmospheric). This plant is generally representative of other ice condensers, having a high thermal power, low containment spray actuation set-point, and intermediate head SI. At a minimum, the work already performed as part of SOARCA (which includes upgrade of the plant model to MELCOR version 1.8.6, accounting for steam generator replacement, a new enhanced lower plenum model, incorporation of SOARCA best practices and enhanced containment modeling) would provide a good starting point for completion of this MELCOR deck.

As part of an ongoing MOU with EPRI, RES plans to engage in discussions with EPRI as to how the two entities can collaborate on this effort. Like NRC, EPRI has been focused on the use of phenomenological tools to improve the realism in success criteria determinations. In the case of EPRI, the relevant tool is the MAAP code. Over the past few years, EPRI has been developing an applications guidance document for MAAP4 which covers guidance to assure quality, uncertainty and sensitivity analysis guidance, BWR and PWR applications guidance, benchmarks and model assessments, and applicability considerations. This document is expected to be completed during the summer of 2009. Potential collaborations could lead to industry-led analysis of sequences that are of mutual interest, which would then be reviewed by NRC staff, compared to related analysis for consistency, and used (with engineering judgment) as the basis for success criteria updates. Alternatively, or in addition, MAAP4 input decks provided by industry would provide a good starting point for the development of additional MELCOR decks. Lastly, interactions with industry will be beneficial from the standpoint of identifying additional studies that have been done subsequent to MSPI, related to plant similarity. On a related note, RES tentatively plans to present the success criteria work done to date at the December 2009 MAAP Users' Group Meeting.

As part of a separate project, RES is investigating the possibility of executing a number of LOCA calculations for a variety of plant types using the TRACE code. These calculations would form the basis for re-assignment of LOCA size ranges, for the purpose of better alignment between PRA LOCA modeling and the phenomenological differences that prompt treatment of small, medium, and large LOCAs separately. This work is being done as part of a project to update the initiating event frequencies for LOCA PRA modeling in the SPAR models using the results of the LOCA expert elicitation (NUREG-1829). The results would be directly applicable to some of the success criteria of interest for the current project.

Finally, RES is considering a project in Fiscal Year 2010 to develop additional MELCOR decks. These decks would be designed to handle Level 1 PRA issues, meaning that they would focus on primary systems and relevant containment systems, but would not have the functionality associated with severe accident analysis (e.g., radionuclide transport, hydrogen combustion). The focus of this work will be highly influenced by the ongoing interactions with NRR regarding project focus. RES plans to present the success criteria work done to date at the September 2009 MELCOR Code Assessment Program (MCAP) meeting.

Conclusion

A realistically conservative core damage definition surrogate has been defined based on accident simulations. MELCOR analyses have been performed for two plants, looking at a range of initiating events and sequences. These results have been mapped to specific changes that are envisioned for the relevant SPAR models. In addition, SPAR models for similar plants that may also utilize these results have been identified. RES is continuing to work in this area, and plans to present the ongoing work at the September 2009 MCAP meeting and the December 2009 MAAP Users' Group meeting.