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Evaluation of Operational Safety at Babcock and Wilcox Plants Volume 2 — Thermal-Hydraulic Results

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F O R M A L R E P O R T

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EVALUATION OF OPERATIONAL SAFETY AT BABCOCK AND WILCOX PLANTS VOLUME 2—THERMAL-HYDRAULIC RESULTS

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ABSTRACT

The Nuclear Regulatory Commission has initiated a research program to develop a methodology to assess the operational performance of Babcock and Wilcox plants and to apply this methodology on a trial basis. The methodology developed for analyzing Babcock and Wilcox plants integrated methods used in both thermal-hydraulics and human factors and compared results with information used in the assessment of risk. The integrated methodology involved an evaluation of a selected plant for each pressurized water reactor vendor during a limited number of transients. A plant was selected to represent each vendor, and three transients were identified for analysis. The plants were Oconee Unit 1 for Babcock and Wilcox, H. B. Robinson Unit 2 for Westinghouse, and Calvert Cliffs Unit 1 for Combustion Engineering. The three transients were a complete loss of all feedwater, a small-break loss-of-coolant accident, and a steam-generator overfill with auxiliary feedwater. Included in the integrated methodology was an assessment of the thermal-hydraulic behavior, including event timing, of the plants during the three transients. Thermal-hydraulic results are presented in this volume (Volume 2) of the report. An overview of all results, with emphasis on operator and hardware performance, is presented in Volume 1.

SUMMARY

The sensitivity of Babcock and Wilcox (B&W) nuclear power plants to operational transients has been a concern since the core damage accident at Three Mile Island Unit 2. The Nuclear Regulatory Commission (NRC) has initiated a research program to develop a methodology to assess the operational performance of B&W plants and to apply this methodology on a trial basis. The methodology was developed and applied at the Idaho National Engineering Laboratory.

The methodology developed for analyzing B&W plants integrated methods used in thermal-hydraulics, systems analysis, and human factors and compared the results with information used in the assessment of risk. The objectives of the analysis have included the following:

- Evaluating the effectiveness of plant systems and operator actions in lessening the severity of transients in B&W plants
- Comparing the differences in operational performance capabilities of selected B&W, Westinghouse, and Combustion Engineering plants
- Evaluating the significance of the findings relative to existing probabilistic and risk assessments (PRAs).

The integrated methodology involved evaluating a plant selected from each pressurized water reactor (PWR) vendor during three transients. The representative plants selected for analysis were Oconee Unit 1 for B&W, H. B. Robinson Unit 2 for Westinghouse, and Calvert Cliffs Unit 1 for Combustion Engineering. The three transients were a complete loss of all feedwater (LOFW), a small-break loss-of-coolant accident (SBLOCA), and a steam generator overfill with auxiliary feedwater (AFW).

The integrated methodology was then applied to assess the thermal-hydraulic behavior of the plants during the three transients and provide information on phenomenological behavior and event timing. The event timing results were combined with information obtained from plant emergency operating procedures and data obtained from task analysis at each of the plants to provide information on operator burden and performance. This volume (Volume 2) of NUREG/CR-4966 reports the evaluation of thermal-hydraulic response. Volume 1

describes the methodology, the evaluation of operator performance, and the evaluation of the significance of the findings relative to the PRAs. Important results and conclusions of the thermal-hydraulic analyses are summarized below.

The thermal-hydraulic responses of the Oconee, H. B. Robinson, and Calvert Cliffs plants were generally similar for the three transients analyzed, though there were differences in the timing of events. These differences relate primarily to design differences between the once-through and U-tube steam generators and their arrangement in the primary system. For the LOFW transient, significant thermal-hydraulic events generally occurred first at Oconee, then at Calvert Cliffs, and finally at H. B. Robinson. Core uncover and heatup began at 57 min after the initiation of LOFW at Oconee, at 101 min at Calvert Cliffs, and at 126 min at H. B. Robinson. Operator actions were required at each of the three plants to prevent core heatup in the total LOFW transient. Primary system feed and bleed was calculated to prevent core heatup if initiated within 31 min following LOFW at Calvert Cliffs, 49 min at Oconee, and 104 min at H. B. Robinson. Because of the high shutoff head of Oconee's high-pressure injection pumps, feed and bleed remains effective, even if only one of the three high-pressure injection pumps is functioning and the pressurizer power-operated relief valve cannot be opened. In this case, the single high-pressure injection pump must be started within 40 min of the initiation of LOFW. Although feed and bleed effectively cooled the core at all three plants, the Oconee systems can successfully respond over a wider range of conditions or failures.

Oconee, H. B. Robinson, and Calvert Cliffs remained in a safe condition during a SBLOCA initiated by a 0.01 ft² break with degraded high-pressure injection. No critical operator actions were identified as necessary to maintain the safety of the plants. Differences were calculated between plants relative to the depletion of reactor coolant and the availability of the steam generators as heat sinks. Less reactor coolant was lost at Oconee because the reactor vessel vent valves caused an earlier transition from single-phase liquid to a high-quality two-phase mixture at the break relative to the other two plants. The design of the once-through steam generator and hot leg at Oconee reduced the availability of the steam generators as

heat sinks compared to the other two plants after two-phase natural circulation stopped.

A steam generator overfill transient has the potential to be more severe at Oconee than at H. B. Robinson or Calvert Cliffs. The overfill transients investigated were initiated by a trip of the main feedwater pumps followed by an overfill with auxiliary feedwater (AFW). Because of their relatively small secondary volume, the steam generators filled more rapidly at Oconee than at the other two plants. The steam generators were full of liquid at 21 min after the main feed pumps tripped at Oconee, at 45 min at H. B. Robinson, and at 106 min at Calvert Cliffs. The steam generator overfill caused a more rapid cooldown at Oconee than at the other two plants. The only critical operator action identified for this transient was to terminate AFW before liquid entered the steam lines. Liquid in the steam lines could possibly cause a steam line failure owing to water hammer or exces-

sive hydrostatic loading, which would result in a more severe overcooling transient.

Results of the three plants studied were extended to other plants designed by the same vendor. For each of the three transients, the plants selected for analysis adequately represented the majority of plants for that vendor. The results of the Oconee calculations are expected to provide reasonable estimates of the thermal-hydraulic responses and event timings for all lowered-loop B&W plants. The Davis-Besse plant, with its low-head high-pressure injection pumps and raised loops, will respond somewhat differently during transients where these two design features are important. Davis-Besse does not have as great a flexibility for feed and bleed. Results from the H. B. Robinson and Calvert Cliffs calculations can generally be used to provide estimates of event timing for other Westinghouse and Combustion Engineering plants, respectively.

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EVALUATION OF OPERATIONAL SAFETY AT BABCOCK AND WILCOX PLANTS—VOLUME 2 THERMAL-HYDRAULIC RESULTS

1. INTRODUCTION

The sensitivity of Babcock and Wilcox (B&W) nuclear power plants to operational transients has been a concern since the core damage accident at Three Mile Island Unit 2. Recent events at some of the B&W plants have reinforced these concerns and has led the Nuclear Regulatory Commission (NRC) to "conclude that there is a need to reexamine the basic design requirements for B&W reactors."¹ Although the B&W Owners Group is taking the lead in this reexamination, the NRC initiated a research program to develop a methodology to assess the operational performance of these plants and to apply the methodology on a trial basis. Thermal-hydraulic results from this NRC-sponsored research program are reported herein.

A methodology has been developed to evaluate certain aspects of the operational safety of nuclear power plants through an integrated assessment of the plant phenomenological behavior and the performance of the plant operators. The methodology was applied to provide NRC with information that will aid in the assessment of the operational performance capabilities and limitations of B&W plants and provide a basis for judging the results from programs sponsored by the B&W Owners Group. Specifically, the objectives of this application are as follows:

1. Evaluate the effectiveness of plant systems and/or operator actions in lessening the severity of transients in B&W plants as defined by
 - a. The timing and magnitude of thermal-hydraulic behavior
 - b. The degree of reliance on safety systems and components and the depth of safety system defenses
 - c. The degree of dependence on operator performance (performance demands versus performance capability)
2. Compare the differences in operational performance capabilities and limitations of selected Combustion Engineering,

Westinghouse and B&W plants based on the results of the assessment outlined in the first objective.

3. Evaluate the significance of the findings in terms of the risk currently identified in existing probabilistic risk assessments (PRAs) for the selected plants.

To accomplish these objectives, three plants and three transients were selected for study. The three selected plants were Oconee Unit 1 representing B&W, H. B. Robinson Unit 2 representing Westinghouse, and Calvert Cliffs Unit 1 representing Combustion Engineering. The three selected transients were a complete loss of feedwater (LOFW), a small-break loss-of-coolant accident (SBLOCA), and a steam generator overfill with auxiliary feedwater. The integrated methodology was then applied to assess the thermal-hydraulic behavior of the plants during the three transients and provide information on phenomenological behavior and event timing. The event timing results were combined with information obtained from plant emergency operating procedures and data obtained from task analysis at each of the plants to provide information on operator burden and performance. This volume (Volume 2) of NUREG/4966 provides the evaluation of thermal-hydraulic responses. Volume 1 describes the methodology, the evaluation of operator performance, and the evaluation of the significance of the findings relative to the PRAs.

The remainder of this volume summarizes the conduct of and results from the thermal-hydraulic portion of the assessment of plant operational behavior for B&W plants. Section 2 describes how the three PWR vendor plants and the three transients were selected for analysis. Section 3 describes the characteristics of the plants. Section 4 presents the results of the thermal-hydraulic analysis. Section 5 presents the conclusions of this study. Section 6 presents references.

2. PLANT AND TRANSIENT SELECTION

It was necessary to select a plant for each PWR reactor vendor and evaluate a limited number of transients. Section 2.1 describes how the three plants chosen to represent each of the PWR vendors were selected. Section 2.2 describes how the three transients used in the study were selected. Volume 1 of this report further details the selection process.

2.1 Plant Selection

One plant was selected from each of the three major vendors of PWRs to allow vendor-to-vendor comparisons of thermal-hydraulic response, event timing, and the time available for necessary operator actions. The plants were selected based on a weighting of five criteria. These criteria were (1) the availability of thermal-hydraulic code input models, (2) the availability of plant information to support human factors and thermal-hydraulic analysis needs, (3) the availability of a suitable PRA, (4) the typicality of the plant relative to other plants designed by the same vendor, and (5) the similarity of the plant's thermal power, size, and primary system configuration to the selected plants of the other reactor vendors. Based on the established criteria, the following plants were selected for study: Oconee Unit 1 for B&W, H. B. Robinson Unit 2 for Westinghouse, and Calvert Cliffs Unit 1 for Combustion Engineering.

A major goal of the analyses was to use consistent methods to enhance the capability to make good comparisons between the thermal-hydraulic results for the three plants. The consistency of the results was obtained by using the same computer code for all the thermal-hydraulic calculations and using standardized thermal-hydraulic modeling techniques. The RELAP5/MOD2 computer code,² Cycle 36.04, was used to perform all thermal-hydraulic calculations described in this study. A brief description of the RELAP5/MOD2 computer code appears in Appendix A.

The standardized models of Oconee and H. B. Robinson were assembled with relatively minor changes from previously existing models^{3,4} developed for the analysis of pressurized thermal

shock. A RELAP5/MOD2 model of Calvert Cliffs Unit 1 was developed based on a TRAC-PF1⁵ model⁶ generated by Los Alamos National Laboratory.

The RELAP5 models used nodalizations that were as consistent as possible considering geometric differences between plants. For example, the nodalizations of the cold legs and pressurizer were virtually identical in all three models. Minor differences appear in the nodalization of the reactor vessel because of geometric differences between plants. Larger differences appear in the nodalizations of the hot legs and steam generators because of the significant design differences in these components. The boundary conditions used in the calculations were also selected to be as consistent as possible. These boundary conditions included auxiliary feedwater temperature, high-pressure injection temperature, normalized core decay power, and assumed operator actions and equipment failures.

2.2 Transient Selection

Based on the study objectives, as well as on time and resource constraints, three transients were judged sufficient to provide significant results from the outlined methodology. Selection of the transients was based on several criteria. Each transient selected should (a) severely challenge the operators and/or safety systems of the plants, (b) highlight differences in the response of the operators and/or safety systems for each of the three plants selected, (c) be expected to occur at a reasonably high frequency, based on plant operating history, (d) be estimated to make a reasonable contribution to core melt frequency for one or more plants, and (e) have the potential to be successfully analyzed within the time and resource constraints, using available human factors methods and thermal-hydraulic models. Based on these criteria, the following transients were selected for study: a complete loss of feedwater, a small-break loss-of-coolant accident with degraded high-pressure injection, and a steam generator overfill with auxiliary feedwater.

3. PLANT DESCRIPTIONS

This section briefly describes the three nuclear steam supply systems of the three plants selected.

3.1 Oconee

Oconee Unit 1 is a PWR configured in a two-by-four design, with two hot legs, four cold legs, and two once-through steam generators, as shown in Figures 3-1 and 3-2. The reactor has a rated core power of 2568 MWt. Oconee is a lowered-loop plant, i.e., the steam generator outlets are below the elevation of the top of the core. Table 3-1 compares selected parameters for Oconee Unit 1 with the plants designed by the other PWR vendors. The major systems of Oconee Unit 1 that have unique features are discussed below. Further details are found in References 7 and 8.

The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 177 fuel bundles. Communication between the vessel upper plenum and downcomer is possible through eight reactor vessel vent valves located above the hot legs. The reactor vessel vent valves start opening when the differential pressure between the upper plenum and the downcomer exceeds 0.125 psid and are fully open at 0.25 psid.⁹

The hot legs connect the vessel with the once-through steam generators. As the hot legs rise to connect with the top of the steam generators, they form a “candy cane” shape. High-point vent valves are placed at the top of the candy cane to allow venting of vapor, which might inhibit natural circulation. Primary fluid makes a single pass down through the steam generators. Feedwater enters the downcomer on the secondary side of the steam generator in an annular region formed by a cylindrical baffle between the outer shell and the tube bundle. After falling through the downcomer, the feedwater flows upward, parallel to the tube bundle, and vaporizes to steam. A gap in the baffle region allows steam to be drawn from the tube bundle into the downcomer to heat the feedwater. The design allows for superheating of the exit steam. Emergency feedwater (EFW) is supplied directly to the upper portions of the tube bundle. EFW is supplied by two motor-driven pumps, one dedicated to each generator and by a turbine-driven pump that feeds both steam generators.

The Oconee emergency core cooling system (ECCS) includes high-pressure injection (HPI),

core flood tanks, and low-pressure injection (LPI) systems. The HPI system contains three centrifugal pumps and piping such that the flow is injected into all four cold legs. The HPI shutoff head is about 2900 psia at Oconee, well above the primary system safety relief valve setpoint.

The integrated control system (ICS) controls the generation of power. The ICS generates a power demand signal and controls the reactor, feedwater, and steam generator pressure to obtain the demanded power.

3.2 H. B. Robinson

H. B. Robinson Unit 2 is a three-loop PWR employing three vertical U-tube steam generators, one in each loop, as illustrated in Figures 3-3 and 3-4. Rated core power is 2300 MWt. Systems that have unique features are described below. Table 3-1 compares selected plant parameters for H. B. Robinson Unit 2 with values for Oconee Unit 1 and Calvert Cliffs Unit 1. Further details are found in References 10 and 11.

The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 157 fuel bundles. The hot legs connect the vessel with the vertical U-tube steam generators. The vertical U-tube steam generators are double-pass heat exchangers. Reactor coolant enters the bottom of the steam generator, passes up, then down, through the U-tubes, and exits at the bottom of the steam generator. Feedwater enters the downcomer on the secondary side of the steam generator in an annular region formed by a baffle between the outer shell and the tube bundle. After falling through the downcomer, the feedwater flows upward and is heated as it passes the tube bundle. A portion of the flow past the tube bundle is vaporized to saturated steam, which then flows through the separators, driers, and into the steam lines. The remainder of the flow past the tube bundle is saturated liquid, which is separated from the steam in the separators and driers and then flows back into the downcomer, where it heats the feedwater. Auxiliary feedwater (AFW) is injected into the downcomer of each steam generator. There are three AFW pumps, two motor-driven and one steam-driven. The AFW pumps discharge to common headers, allowing each pump to feed all the steam generators.

7-11025

Figure 3-1. Oconee reactor coolant system elevation diagram.

The ECCS contains three HPI pumps, three accumulators, and two LPI pumps. The HPI pumps discharge into a common header, which is connected to each cold leg of the reactor coolant system. The HPI pumps have a shutoff head of 1470 psia. H. B. Robinson also has three high-head low-capacity charging pumps.

3.3 Calvert Cliffs

Calvert Cliffs Unit 1 is a two-by-four loop PWR with two hot legs, four cold legs, and two vertical U-tube steam generators, as illustrated in Figures 3-5 and 3-6. The reactor has a rated core power of 2700 MWt. Systems with unique design features are described below. Table 3-1 compares selected plant parameters for Calvert Cliffs Unit 1 with values for Oconee Unit 1 and H. B. Robinson Unit 2. Further details are found in References 12 and 13.

The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 217 fuel bundles. The hot legs connect the vessel with the vertical U-tube steam generators. The vertical U-tube steam generators are double-pass heat exchangers. The internal components of these steam generators are arranged similarly to those described for the H. B. Robinson plant. AFW is injected into the downcomer of each steam generator. There are three AFW pumps, one motor driven and two steam driven. However, one steam-driven pump is a spare and is not normally used. The AFW pumps discharge to common headers, allowing each pump to feed both steam generators.

The ECCS contains three HPI pumps, four safety injection tanks, and two LPI pumps. The HPI pumps discharge into a common header, which is connected to the four cold legs. The HPI pumps have a shutoff head of 1275 psia. Calvert Cliffs also has three high-head low-capacity charging pumps.

7-11022

Figure 3-2. Oconee reactor coolant system loop arrangement.

Table 3-1. Comparison of reactor parameters at Oconee, H. B. Robinson, and Calvert Cliffs

Parameter	Oconee	H. B. Robinson	Calvert Cliffs
Core power (MWt)	2,568	2,300	2,700
Number of core fuel bundles	177	157	217
Steam generator design	Once-through	U-tube	U-tube
Number of hot legs	2	3	2
Number of cold legs	4	3	4
RCS fluid volume (ft ³)	12,096	9,111	11,101
Pressurizer volume (ft ³)	1,500	1,300	1,500
Effective PORV area ^a (ft ²)	0.00652	0.02126	0.01512
Maximum makeup (gpm)	250	231	132
Separate makeup and HPI pumps?	No	Yes	Yes
HPI shutoff head (psia)	2,900	1,470	1,275
Accumulator pressure ^b (psia)	600	615	200
LPI shutoff head (psia)	215	175	180
Total AFW flow ^c (gpm)	1,800	1,200	800
Steam generator volume ^d (ft ³)	6,824	14,187	16,010
Steam generator mass ^d (lbm)	79,706	279,000	273,000

a. The total PORV area used in the RELAP5/MOD2 thermal-hydraulic models.

b. Minimum pressure of the cover gas in the accumulator. Referred to as the core flood tank at Oconee and the safety injection tank at Calvert Cliffs.

c. Flow corresponds to a steam generator pressure of 1000 psia.

d. Represents the total coolant in the secondary sides of all steam generators.

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Figure 3-3. H. B. Robinson reactor coolant system elevation diagram.

7-11024

Figure 3-4. H. B. Robinson reactor coolant system flow schematic.

7-11026

Figure 3-5. Calvert Cliffs reactor coolant system elevation diagram.

7-11020

Figure 3-6. Calvert Cliffs reactor coolant system loop arrangement.

4. THERMAL-HYDRAULIC ANALYSIS

This section presents and compares the results of the thermal-hydraulic analyses of the three selected PWRs for the three chosen transients. Baseline thermal-hydraulic calculations were performed for each plant for each of the three selected transients. All transients were initiated from 100% rated core power. The baseline transients included conservative assumptions concerning hardware failure and minimal operator actions and should not be considered a best-estimate response of the plant to the initiating event. Sensitivity calculations were performed to investigate alternate forms of core cooling in the LOFW transients.

Section 4.1 describes the differences in plant response between Oconee, H. B. Robinson, and Calvert Cliffs during a LOFW transient. Sections 4.2 and 4.3 compare the thermal-hydraulic responses of the three plants during the SBLOCA and steam generator overfill transients, respectively. Detailed discussions of the individual thermal-hydraulic models and plant responses appear in Appendixes B, C, and D for Oconee, H. B. Robinson, and Calvert Cliffs, respectively. Section 4.4 discusses the applicability of the thermal-hydraulic results to other PWRs.

4.1 Loss of Feedwater

LOFW transients at Oconee, H. B. Robinson, and Calvert Cliffs were simulated using the RELAP5/MOD2 computer code. Each plant was assumed to be operating at 100% rated core power when the transient was initiated by the trip of all main feed pumps (MFPs). Baseline and sensitivity calculations were performed for each plant. In the baseline calculations, normal operation of all automatic plant systems was assumed except that auxiliary feedwater (AFW) failed, resulting in a total loss of feedwater. The only operator action simulated in the baseline transient was to trip the reactor coolant pumps (RCPs). This operator action was simulated because continued RCP-operation would have had a major impact on the hydraulics of the reactor coolant system (RCS) once voiding began, particularly affecting the time of core dryout and heatup.

The calculated sequence of events for the baseline LOFW transients are compared in Table 4-1. Each LOFW transient was initiated at 40 s when the MFPs tripped. The table indicates that signifi-

cant events generally occurred first at Oconee, then at Calvert Cliffs, finally at H. B. Robinson. For example, steam generator dryout at Oconee occurred 950 s earlier than at Calvert Cliffs and 3450 s earlier than at H. B. Robinson. The RCS reached saturation temperature and began voiding 2100 s earlier at Oconee than at Calvert Cliffs and 3900 s earlier than at H. B. Robinson. Core dryout and heatup began 2650 s earlier at Oconee than at Calvert Cliffs and 4150 s earlier than at H. B. Robinson.

One significant exception in the timing of events relates to RCP trip. The RCPs were tripped at Calvert Cliffs before they were tripped at the other two plants. At Oconee and H. B. Robinson, the RCPs were tripped 30 s after the loss of minimum subcooling (20°F at Oconee and 25°F at H. B. Robinson) as directed by the emergency operating procedures. At Calvert Cliffs, however, the operator is directed to trip the RCPs to reduce the heat load on the steam generators shortly after he enters the LOFW emergency operating procedure. Thus, the Calvert Cliffs RCPs were tripped based on the estimated time it would take the operators to reach this procedural step. If this procedural step had been ignored in the calculation, the RCPs would never have been tripped, invalidating the desired comparisons between plants. (Calvert Cliffs procedures do direct the operator to trip the RCPs based on low hot leg subcooling coincident with low RCS pressure, but this combination did not occur during the transient.) If the Calvert Cliffs RCPs had been tripped based solely on subcooling similar to the other two calculations, it is estimated that steam generator dryout would occur at 1060 s, the RCPs would be tripped at 2300 s, the RCS would reach saturation temperature at 3100 s, and core dryout would begin at 5100 s.

The different responses of the three plants during a LOFW transient are illustrated in Figure 4-1, which shows total fluid mass, including steam, on the secondary side of the steam generators. Owing to design differences between the once-through and U-tube steam generators, the initial liquid inventory at Oconee was only about 30% of the initial inventory at the other two plants. Consequently, the steam generators were depleted of liquid and dried out much earlier at Oconee than at Calvert Cliffs or H. B. Robinson. Although the initial liquid inventory was nearly identical at Calvert Cliffs and H. B. Robinson, the steam generators dried

Table 4-1. Sequence of events for the LOFW transients

Time (s)			Event
Oconee	Calvert Cliffs	H.B. Robinson	
40	40	40	MFPs tripped
40	70	40	Reactor tripped; turbine tripped
43	70	41	Turbine bypass valves opened
300	1250	3750	Steam generator dryout
305	836	637	Pressurizer heaters off
536	1272	4750	PORV opened for first time
975	2900	4750	Pressurizer liquid full
1515	460	5260	RCPs tripped
1900	4000	5800	RCS at saturation temperature
2180	4400	6300	Natural circulation stopped
3450	6100	7600	Core heatup started

Figure 4-1. A comparison of total steam generator fluid masses in the LOFW transient.

out much earlier at Calvert Cliffs because of differences in the timing of reactor trip. Since an MFP trip does not cause reactor trip at Calvert Cliffs, the steam generator liquid inventory decreased rapidly until a low steam generator level signal caused a reactor trip. Since an MFP trip initiates a reactor trip at H. B. Robinson, the steam generator liquid inventory was conserved. MFP trip also caused a reactor trip at Oconee.

The steam generator liquid inventory at Calvert Cliffs was also consumed more rapidly after reactor trip because of its lower no-load temperature, 532°F versus about 550°F at the other two plants. The no-load temperature at each plant was determined through the action of the control systems, which stabilized steam generator pressures following reactor trip, as illustrated in Figure 4-2. The response of the Calvert Cliffs control systems to the RCP trip at 460 s briefly perturbed nearly all of the primary and secondary thermal-hydraulic parameters. The steam generators depressurized at both Calvert Cliffs and H. B. Robinson after dryout, as the plant control systems opened steam line valves in a futile attempt to control RCS fluid temperatures. The depressurization of the steam generators at Calvert Cliffs and H. B. Robinson caused the fluid mass (recall Figure 4-1) to decrease to near zero, owing to the small density of steam at atmospheric pressure. The steam generator fluid mass, after dryout, was somewhat higher at Oconee because of its higher steam pressure and correspondingly higher steam density.

The thermal-hydraulic response of the RCS during a LOFW transient is illustrated in Figure 4-3, which shows hot leg fluid temperatures. The hot leg temperatures responded similarly at all three plants except for timing differences. The hot leg temperatures decreased following reactor trip, stabilized in response to plant control systems, increased after the steam generators dried out, and then stabilized after reaching the saturation temperature corresponding to the setpoint of the pressurizer power-operated relief valves (PORVs). The subcooling in the hot legs, shown in Figure 4-4, disappeared at 1900 s, 4000 s, and 5800 s at Oconee, Calvert Cliffs, and H. B. Robinson, respectively. The loss of subcooling coincided with the RCS reaching saturation temperature and allowed bulk boiling to begin in the core.

The thermal-hydraulic response of the RCS during a LOFW transient is further illustrated by Figures 4-5 and 4-6, which compare RCS pressures and pressurizer levels, respectively. The pressures and levels responded similarly, except for the timing

differences between plants. The pressures and levels initially decreased in response to the normal cooldown of the RCS following reactor trip. At Oconee, the thermal expansion of the RCS fluid after the dryout of the steam generators was primarily responsible for the subsequent increase in RCS pressure and pressurizer levels. However, at H. B. Robinson and Calvert Cliffs, the pressurizer pressure and level control systems returned the pressure and level to normal posttrip values.

After steam generator dryout at H. B. Robinson and Calvert Cliffs, the heatup and thermal expansion of the RCS fluid caused the pressure and level to increase. Eventually at each plant, the PORVs opened as necessary to relieve the pressure, and the pressurizer filled with liquid. Once hot leg subcooling was lost, the PORVs could not relieve the volume expansion caused by boiling in the core as easily as when the RCS was subcooled. At Oconee, the pressure increased and the pressurizer code safety relief valves (SRVs) opened several times. At Calvert Cliffs, the pressure increased to within 10 psi of the SRV opening setpoint, but the pressure never exceeded the setpoint. At H. B. Robinson, the pressure did not approach the setpoint of the SRVs, but the PORVs were continuously open for longer periods of time than before the loss of subcooling. The loss of mass through the PORVs caused voiding to occur in the RCS, evidenced by the decrease in pressurizer levels that occurred shortly after the loss of subcooling. The voiding of the RCS is also illustrated in Figure 4-7, which shows void fractions in the upper plenum. The mixture level dropped out of the upper plenum and into the core when the void fraction reached unity at 3450 s, 6100 s, and 7600 s at Oconee, Calvert Cliffs, and H. B. Robinson, respectively. The fuel rods at the top of the core began heating up shortly thereafter. The baseline LOFW calculations were terminated soon after core heatup began.

The calculations indicate that operator action would be required to prevent core uncover in a total LOFW transient at all three plants. The thermal-hydraulic responses of the three plants were similar except for the timing of events. The significant events of steam generator dryout, RCS saturation, and core uncover occurred first at Oconee, then at Calvert Cliffs, finally at H. B. Robinson. The differences between plants were primarily caused by the small initial liquid inventory in the Oconee steam generators and the delayed reactor trip at Calvert Cliffs.

Sensitivity calculations were performed to investigate alternate forms of core cooling in a total LOFW transient. One alternate form of core cooling

Figure 4-2. A comparison of steam generator pressures in the LOFW transient.

Figure 4-3. A comparison of hot leg fluid temperatures in the LOFW transient.

Figure 4-4. A comparison of hot leg subcoolings in the LOFW transient.

Figure 4-5. A comparison of RCS pressures in the LOFW transient.

Figure 4-6. A comparison of normalized pressurizer levels in the LOFW transient.

Figure 4-7. A comparison of upper plenum void fractions in the LOFW transient.

investigated was feed and bleed. Feed and bleed was initiated by activating the pumps necessary to supply water to the RCS and opening primary-side valves to depressurize the RCS. The purpose of the sensitivity calculations was to determine the latest time feed and bleed could be initiated at each plant to prevent core heatup. The times calculated are not the absolute latest time at which feed and bleed could be initiated since core heatup, rather than core damage, was chosen as the success criterion. Feed and bleed was successfully initiated at 1880 s at Calvert Cliffs, 3000 s at Oconee, and 6300 s at H. B. Robinson.

The effect of feed and bleed on the RCS mass balance is summarized in Figures 4-8 and 4-9, which present the flow into and out of the RCS. The flow into the RCS represents all sources of high-pressure water that would normally be used during feed and bleed at each plant. Specifically, the flow represents the output of all three high-pressure injection (HPI) pumps at Oconee, three HPI pumps and three charging pumps at H. B. Robinson, and two HPI pumps and three charging pumps at Calvert Cliffs. The flow out of the RCS represents the flow through both pressurizer PORVs at H. B. Robinson and Calvert Cliffs and the flow through the pressurizer PORV and both hot leg high-point vent valves (HPVVs) at Oconee. The HPI pumps develop more head at Oconee than at the other two plants. The Oconee HPI pumps can deliver flow at pressures above 2525 psia, the opening setpoint of the pressurizer SRVs. The HPI shutoff head is 1470 psia at H. B. Robinson and 1275 psia at Calvert Cliffs. More liquid entered the RCS at the start of feed and bleed at Oconee because of its high-head HPI pumps. The flow into the RCS at H. B. Robinson and Calvert Cliffs was initially supplied only by the relatively low-capacity charging pumps. At Oconee, the RCS began to refill immediately after feed and bleed started, because the HPI flow exceeded the flow out of the RCS. The refill at Oconee was aided by the relatively low mass flow out of the RCS that was a consequence of the RCS being sufficiently voided to allow steam flow out the PORV and HPVVs at the start of feed and bleed. Thus, feed and bleed can be used to prevent core heatup at Oconee virtually up to the time of core uncover. However, at the other two plants, feed and bleed must be initiated well before the beginning of core uncover, about 1300 s and 4200 s before core uncover at H. B. Robinson and Calvert Cliffs, respectively. At these two plants, the flow through the PORVs exceeded the charging flow, resulting in continued voiding of the RCS

after the initiation of feed and bleed. The RCS did not begin to refill until after HPI flow was initiated. The flow into the RCS exceeded the flow out after 5800 s at Calvert Cliffs and 7600 s at H. B. Robinson. Also note that Figure 4-9 provides a relative indication of the effective PORV area at the three plants. Oconee has the smallest PORV area whereas H. B. Robinson has the largest.

Figure 4-10 shows calculated RCS pressure for the three plants during feed and bleed. At Oconee and H. B. Robinson, the RCS was two-phase when feed and bleed was initiated, although the RCS was subcooled at Calvert Cliffs. At all three plants, the RCS depressurized at or shortly after the start of feed and bleed. Once begun, the depressurization continued at Oconee and H. B. Robinson. Feed and bleed caused the Calvert Cliffs RCS to depressurize rapidly until saturation pressure was reached and bulk boiling began in the core. The PORVs were then not able to relieve the fluid expansion caused by boiling and RCS pressure increased. The RCS depressurized slowly once the RCS had voided sufficiently to allow steam to exit through the PORVs. The pressure increase and slow depressurization calculated at Calvert Cliffs indicated less capability for feed and bleed cooling than at the other two plants. The major difference in feed and bleed capability between Calvert Cliffs and Oconee was related to the high-head HPI pumps at Oconee versus the relatively low-head pumps at Calvert Cliffs. The differences between the capability to feed and bleed at Calvert Cliffs and H. B. Robinson were related to several factors, each of which made it more difficult to feed and bleed at Calvert Cliffs. These factors included 40% less charging flow, 30% less relief capacity through the PORVs, and 20% more core power at Calvert Cliffs. Thus, the results indicate that feed and bleed must be started earlier at Calvert Cliffs than at the other two plants.

Feed and bleed can be used to successfully cool the core in a total LOFW transient if initiated early enough for the three plants studied. Feed and bleed was calculated to prevent core uncover if initiated at 31 min following MFP trip at Calvert Cliffs, 49 min at Oconee, and 104 min at H. B. Robinson. Logic indicates that feed and bleed would also be successful, with more margin for safety, if initiated earlier. Hand calculations indicate that if the start of feed and bleed had been delayed by an additional 5 min, a minor core heatup would have occurred at each plant.

Other sensitivity studies were performed to evaluate additional methods of core cooling in total LOFW

Figure 4-8. A comparison of flows into the RCS during feed and bleed.

Figure 4-9. A comparison of flows out of the RCS during feed and bleed.

Figure 4-10. A comparison of RCS pressures during feed and bleed.

transients. At Oconee, the ability to feed and bleed with degraded HPI and degraded depressurization capability was demonstrated. A sensitivity calculation showed that with only one HPI pump running, feed and bleed could still successfully prevent core uncover if initiated at 3000 s. In this sensitivity calculation, the RCS began refilling near 4600 s. Another sensitivity calculation shows that one HPI pump delivering flow at the setpoint pressure of the pressurizer SRVs is sufficient to prevent core uncover if initiated by 2400 s.

Sensitivity calculations were also performed to investigate the use of the condensate pumps at H. B. Robinson and the condensate booster pumps at Calvert Cliffs to feed the steam generators. These sensitivity calculations simulated the operator actions directed by the emergency operating procedures to establish core cooling with these pumps. The most important procedural step was to depressurize the steam generators below the shutoff head of the condensate pumps. The purpose of the sensitivity calculations was to determine the latest time by which the operators must act to establish condensate flow in order to prevent dryout of the steam generators. The sensitivity calculations indicate that the condensate pumps could be successfully used to feed the steam

generators prior to dryout at both plants. The depressurization of the steam generators began at 190 s at Calvert Cliffs and at 2800 s at H. B. Robinson. However, if the Calvert Cliffs calculation had been performed in strict adherence to the letter of the emergency operating procedures, the depressurization would have been abandoned as unsuccessful because the minimum steam generator level dropped below the value where the procedures direct the operators to isolate the steam generators and initiate feed and bleed. However, the calculated results were sufficiently close to the boundary between successful and unsuccessful depressurization as defined by the procedures that the uncertainty in the calculation could alter the outcome of the operation. The major source of uncertainty in the calculation was the time between the trip of the MFPs and the reactor trip on low steam generator level. The estimated uncertainty in the time of reactor trip corresponded to about 120 s in the time available to initiate the depressurization of the steam generators. In order to prevent confusion to the operators, it appears worthwhile to review the Calvert Cliffs procedures and supporting calculations relative to the depressurization operation to maximize the probability of success according to the criteria appearing in the procedures.

4.2 Small-Break Loss-of-Coolant Accident

Each plant was assumed to be operating at 100% rated core power prior to the start of the SBLOCA transient. The transient was initiated by opening a 0.01 ft² break in the cold leg between the HPI nozzle and the reactor vessel. The break was located on the side of the cold leg, at the elevation of the cold leg centerline. Normal operation of all automatic plant systems was represented except that only one HPI pump was assumed to be available. Operator action to trip the RCPs based on emergency operating procedures was simulated. The RCPs were tripped 30 s after the minimum hot leg subcooling decreased to 25°F at H. B. Robinson and 20°F at Oconee. At Calvert Cliffs, two RCPs were tripped 30 s after pressurizer pressure decreased to 1725 psia. The other two RCPs were tripped 30 s after the RCS pressure and subcooling decreased to 1300 psia and 30°F, respectively. The RCPs were tripped because continued RCP operation would have had a major impact on the hydraulics of the RCS once voiding began. Operator action to throttle AFW once the liquid levels in the steam generators returned to normal posttrip levels was also simulated at H. B. Robinson. The throttling of AFW prevented the SBLOCA results from being obscured by a steam generator overfill. A similar action would have been simulated at Calvert Cliffs had the calculation not been terminated before the normal posttrip levels were reached. In the Oconee calculation, the integrated control system (ICS) automatically throttled feedwater after normal posttrip levels were reached.

Table 4-2 compares the calculated sequence of events for the SBLOCA transient in the three PWRs. Each SBLOCA was initiated at 40 s, when the break opened. The table indicates that the significant events occurring early in the transient happened at about the same time in all three calculations. For example, reactor trip occurred between 76 and 82 s at all three plants. The safety injection signal was generated between 104 and 125 s. Forced circulation was lost in the RCS between 177 and 214 s owing to RCP trip. The similar response of all three plants early in the transient is also illustrated in Figure 4-11, which shows pressurizer liquid levels. The levels decreased after the break opened at 40 s. The levels decreased more rapidly after reactor trip owing to the normal posttrip cooldown of the RCS. Each pressurizer emptied of liquid at about the same time. Although the

pressurizer was empty at Calvert Cliffs, the level did not indicate zero because it was based on differential pressure and the static head of steam appears as a liquid level.

The RCS pressures also responded similarly in the early portion of the transient, as shown in Figure 4-12. The pressure responses of H. B. Robinson and Calvert Cliffs were similar throughout the transient. The Oconee pressure increased later in the calculation, diverging from the others. The divergence in pressures was caused by differences in steam generator heat transfer, which will be discussed later.

The RCS mass balances during the SBLOCA are shown in Figures 4-13 and 4-14, which present the flow into and out of the RCS. The total flow into the RCS was similar at all three plants. The total flow at Oconee, which was the result of a single HPI pump, was slightly less than at the other two plants. The total flow at the other two plants was the result of a single HPI pump combined with charging pumps. The flow from the Oconee HPI pump generally exceeded the HPI flow at the other two plants, but Oconee does not have separate charging pumps, and the total flow was slightly less at Oconee. The flow out the break (Figure 4-14) was similar at each plant until 1700 s. The similarity of the break flows was a consequence of the identical break size, the similar flow of water into the RCS, and the similar RCS fluid volumes of the three PWRs. These similarities were responsible for the close agreement in the calculated responses early in the transient.

One significant difference observed between the plants relates to the time the flow out the break transitioned from single-phase liquid to a high-quality, two-phase mixture. The transition in the fluid state at the break was responsible for a sharp decrease in the break flow. With the reduction in break flow, the flow into the RCS matched or exceeded the flow out the break. Prior to the reduction in break flow, the flow out the break exceeded the flow into the RCS, causing a gradual voiding of the RCS at each plant. The transition in fluid state at the break occurred between 1750 and 2600 s earlier at Oconee than at the other two plants. The different transition time was caused by the reactor vessel vent valves present at B&W plants. Steam generated in the Oconee core flowed into the upper plenum, through the vent valves, into the downcomer, through the broken cold leg, and out the break. At H. B. Robinson and Calvert Cliffs, steam generated in the core did not reach the break until a RCP loop seal cleared. After the loop seal cleared, steam flowed from the core, through a hot leg, steam generator, and cold leg to reach the break. Because of the steam path

Table 4-2. Sequence of events for the SBLOCA transients

Time (s)			Event
Oconee	Calvert Cliffs	H.B. Robinson	
40	40	40	Break opened
76	79	82	Reactor tripped; turbine tripped
114	120	114	Pressurizer emptied
125	104	110	SI signal
214	136/177	205	RCPs tripped
	215	130	150 Upper head began voiding
1700	2600	1770	Natural circulation stopped
1700	4320	3453	Break transitioned to two-phase
2300	4320	3453	RCS began refilling

Figure 4-11. A comparison of normalized pressurizer liquid levels in the SBLOCA transient.

Figure 4-12. A comparison of RCS pressures in the SBLOCA transient.

Figure 4-13. A comparison of the flow into the RCS in the SBLOCA transient.

Figure 4-14. A comparison of break flows in the SBLOCA transient.

through the vent valves, the reduction in break flow occurred earlier, causing the voiding of the RCS to stop earlier at Oconee than at the other two plants. The longer period of voiding at H. B. Robinson and Calvert Cliffs resulted in more depletion of RCS inventory compared to Oconee. However, the core remained covered with liquid during all three SBLOCA calculations, as illustrated by Figure 4-15. The figure shows calculated void fractions in the upper plenum volume immediately above the core. Each reactor core in each plant remained cool throughout the transient.

All three plants remained in a safe condition during the SBLOCA transient initiated by a 0.01 ft² break with degraded HPI. No critical operator action necessary to maintain the safety of the plants was identified. However, differences between plants were observed in the ability of the operators to recover following a SBLOCA. Figures 4-16 through 4-18 compare RCS and steam generator pressures at the three plants. At H. B. Robinson and Calvert Cliffs, the RCS and steam generator pressures are closely coupled after the initial depressurization of the RCS. The coupling of the pressures indicate close thermal contact between the RCS and steam generators. The steam generators are available as heat sinks throughout

the transient. Thus, the operators can depressurize the RCS, a necessary step to reach residual heat removal entry conditions, by depressurizing the steam generators. However, at Oconee the RCS and steam generator pressures diverged during the transient owing to slow pressurization of the RCS. The RCS pressurized because the core decay heat exceeded the cooling resulting from the steam generators, break, and HPI. The steam generators were effectively lost as heat sinks when natural circulation stopped at 1700 s. Thereafter, the break and HPI provided decay heat removal. Unlike H. B. Robinson and Calvert Cliffs, the Oconee RCS could not be depressurized using the steam generators. The operator could depressurize the RCS by momentarily operating the RCPs or opening the pressurizer PORV.

The difference between plants in the effectiveness of the steam generators as heat sinks is the result of design differences between the once-through and U-tube steam generators. Once natural circulation is lost in a once-through steam generator by voiding at the top of the hot legs, the steam generators cannot condense steam unless the liquid level on the primary side of the steam generator drops below the level on the secondary side. (The level on the primary side remained above the

Figure 4-15. A comparison of upper plenum void fractions in the SBLOCA transient.

Figure 4-16. A comparison of RCS and steam generator pressures in the SBLOCA transient at H. B. Robinson.

Figure 4-17. A comparison of RCS and steam generator pressures in the SBLOCA transient at Calvert Cliffs.

Figure 4-18. A comparison of RCS and steam generator pressures in the SBLOCA transient at Oconee.

top of the tubes during the Oconee SBLOCA calculation.) At H. B. Robinson and Calvert Cliffs, the steam generators remained available as heat sinks after the loss of two-phase natural circulation owing to reflux cooling. In reflux cooling, steam produced in the core flows through the hot legs and into the steam generators, where some is condensed. The condensate then drains down the U-tubes through the hot leg and back into the upper plenum. Because the steam generators remained available as heat sinks, H. B. Robinson and Calvert Cliffs have an additional RCS depressurization mechanism for certain break sizes that is not available at Oconee.

The SBLOCA calculations indicate that the steam generator safety relief valves are more likely to be challenged following reactor trip at Oconee than at H. B. Robinson or Calvert Cliffs. The additional challenges to the steam generator safety relief valves increase the probability that one will not fully reseal after opening. Operating experience indicates that failure of safety relief valves to fully reseal is common. The leak associated with a safety relief valve that does not reseal can further complicate a transient.

4.3 Steam Generator Overfill

Each plant was assumed to be operating at 100% rated core power prior to the start of the steam generator overfill transient. The transient was initiated by a trip of the main feed pumps (MFPs). AFW was initiated automatically and continued to run, causing an overfill of the steam generators. At Oconee, the AFW is normally throttled automatically, but this control function was assumed to fail for this transient. At H. B. Robinson and Calvert Cliffs, AFW is normally throttled manually, but this manual action was not simulated. No manual actions were simulated in any of the calculations. Normal operation of all automatic plant systems was assumed except for failure of the automatic throttling of AFW at Oconee.

Table 4-3 compares the calculated sequence of events for the steam generator overfill transients. Each transient was initiated at 40 s when the MFPs tripped. The MFP trip immediately caused reactor and turbine trips at Oconee and H. B. Robinson. At Calvert Cliffs, the reactor trip was delayed until 70 s, when the reactor and turbine tripped on low steam generator level. AFW was initiated within a few seconds of reactor trip at H. B. Robinson and Calvert Cliffs. The initiation of AFW flow was delayed at Oconee until steam generator levels

dropped below the low-level limit. Figure 4-19 presents a comparison of the total AFW flow delivered to the steam generators in the three plants. The total flow represents the output of two motor-driven and one turbine-driven AFW pump at Oconee and H. B. Robinson, and one motor-driven and one turbine-driven pump at Calvert Cliffs. The flow at H. B. Robinson and Calvert Cliffs corresponds to the rate automatically delivered by the AFW flow control valves. Because of the failure assumed at Oconee, AFW was delivered at its maximum rate. The largest AFW flow was delivered at Oconee; the smallest flow was delivered at Calvert Cliffs.

The effect of the AFW overfill on the total fluid mass on the secondary side of the steam generators is shown in Figure 4-20. The mass decreased significantly following reactor trip at Oconee and Calvert Cliffs, while remaining relatively constant at H. B. Robinson. At H. B. Robinson, AFW was initiated shortly after reactor trip, preventing a significant decrease in steam generator mass. At Oconee, AFW flow to the steam generators was delayed until the startup liquid levels reached their low-level limit about 140 s after reactor trip. (In a LOFW transient without an AFW failure, AFW would normally maintain the steam generators at their low-level limit.) At Calvert Cliffs, the sharp decrease in mass was caused by a 30-s delay between the time of MFP and reactor trips. The Oconee steam generators filled more rapidly than those at the other two plants. The steam generators were liquid full at 1308 s at Oconee and at 2750 s at H. B. Robinson. The Calvert Cliffs steam generators had not filled when the calculation was stopped at 4640 s, but a hand calculation indicates that the generators would fill near 6400 s. The steam generators filled with liquid earlier at Oconee because of their smaller volume and the larger AFW flow rate. The time required to fill the Oconee steam generators would have been larger, closer to the times obtained for the other plants, if the Oconee AFW flow had been assumed to fail at an intermediate rate rather than the maximum rate as assumed in the calculation.

The response of the steam generators is further illustrated in Figure 4-21, which shows steam generator pressures. The AFW overfill depressurized the Oconee steam generator, while the pressures remained relatively stable at the other two plants. The relatively severe depressurization at Oconee was caused by the higher AFW flow rate, the smaller steam generators, and the injection of AFW into the steam space near the top of the steam

Table 4-3. Sequence of events for the steam generator overfill transients

Time (s)			Event
Oconee	Calvert Cliffs	H.B. Robinson	
40	40	40	MFPs tripped
40	70	40	Reactor tripped; turbine tripped
44	77	42	Pressurizer heaters on
172/192	72	41/57	AFW initiated
200	3934	1287	AFW not throttled
464	N/A	1440	Pressurizer heaters tripped
1308	6400 ^a	2750	Steam generators liquid full

a. Determined by hand calculation, transient terminated at 4640 s before steam generator was full.

Figure 4-19. A comparison of total AFW flows in the overfill transient.

Figure 4-20. A comparison of total steam generator fluid masses in the steam generator overfill transient.

Figure 4-21. A comparison of steam generator pressures in the steam generator overfill transient.

generators, which allowed more condensation than in the other two plants (where AFW was injected below or just above the liquid level).

The effect of the overfill on RCS fluid temperatures is shown in Figure 4-22. The hot leg fluid temperatures shown in the figure were typical of all RCS fluid temperatures, since the RCPs were not tripped in this transient. The hot leg temperatures decreased and then stabilized normally following reactor trip. However, the continued injection of cold AFW eventually overcooled the RCS. The overcooling began at 200 s at Oconee, 1230 s at H. B. Robinson, and 3640 s at Calvert Cliffs. The time that the overcooling began at each plant corresponds roughly to the time that the liquid levels in the steam generators increased above the values at which they would normally be throttled. The largest cooldown rate occurred at Oconee; the smallest rate occurred at Calvert Cliffs. The rate of cooldown of the average RCS temperature from the time of reactor trip to the time the steam generators filled with liquid was 256°F/hr at Oconee, 93°F/hr at H. B. Robinson, and 41°F/hr at Calvert Cliffs.

The effect of the overfill on pressurizer liquid levels is shown in Figure 4-23. The pressurizer levels all decreased following reactor trip and the subsequent cooldown of the RCS. At Oconee, the pressurizer level decreased continuously because the makeup flow was unable to compensate for the shrinkage of the RCS fluid caused by the cooldown. At H. B. Robinson, the pressurizer level was controlled near its normal posttrip value by the plant control systems. At Calvert Cliffs, the plant control systems acted to increase the pressurizer level to near its normal posttrip value. The H. B. Robinson and Calvert Cliffs pressurizer levels were thus stabilized following reactor trip but then decreased again following the start of the overcooling of the RCS. The pressurizer levels were then again returned to their normal values by the response of the pressurizer level control systems. The RCS pressures, shown in Figure 4-24, had trends similar to the pressurizer liquid levels. The RCS pressures decreased following reactor trip and then increased due to the action of the pressurizer pressure and level control systems. However, at Oconee the pressurizer heaters were tripped at 464 s because of low pressurizer level. The RCS pressure then decreased along with the RCS temperature and pressurizer level. The RCS pressure did not decrease far enough to generate a safety injection signal in any of the calculations.

The only critical operator action identified for this transient was to terminate AFW before liquid

entered the steam lines. Liquid in the steam lines could possibly cause a steam line failure due to water hammer or excessive hydrostatic loading. A steam line failure under these conditions would result in a more severe overcooling transient.

4.4 Applicability of Results to Other Plants

The purpose of this section is to evaluate the applicability of the results of the three transients at Oconee, H. B. Robinson, and Calvert Cliffs to other plants based on a brief review of their typicality to other plants designed by the same vendor. This evaluation of applicability is intended to provide an overview of how the calculated results might extrapolate to other plants. The extrapolations are not intended to provide definitive event timings for other plants. A rigorous extension of these results to other plants would require a comprehensive program, well beyond the scope of the current effort.

The extrapolation of the results contained in Sections 4.1 through 4.3 to other plants is organized based on reactor vendor. Section 4.4.1 describes the applicability of the calculated results for Oconee Unit 1 to other B&W plants. Sections 4.4.2 and 4.4.3 describe the applicability of the results for H. B. Robinson Unit 2 and Calvert Cliffs Unit 1 to other Westinghouse and CE plants, respectively.

4.4.1 Babcock & Wilcox. A brief review was performed to determine how well Oconee Unit 1 represented the other operating B&W plants. This review was based on a variety of sources of information^{14,15,16} that may not always reflect the most recent modifications to the plants. Hardware differences identified by this review are summarized in Table 4-4. The table illustrates that Oconee Unit 1 is generally representative of all operating, lowered-loop B&W plants. The RCS fluid volume is essentially the same at all B&W plants and the power varied by only 8% or less. The most significant difference is that the three Oconee units do not have a system to detect a steam line break, whereas all the other plants do. The design of the steam line break detection system varies significantly between plants. The table also illustrates that Oconee Unit 1 has significant differences from the raised-loop plant, Davis-Besse Unit 1. These differences include the HPI and makeup systems, number of reactor vessel vent valves, size of the pressurizer PORV, and steam line break detection system. An evaluation of the effect of the differences between plants on the

Figure 4-22. A comparison of hot leg temperatures in the steam generator overfill transient.

Figure 4-23. A comparison of RCS normalized pressurizer levels in the steam generator overfill transient.

Figure 4-24. A comparison of RCS pressures in the steam generator overfill transient.

Table 4-4. A comparison of B&W plant parameters

Parameter	Oconee ^a	ANO-1 ^b	CR-3 ^c	DB-1 ^d	RS ^e	TMI-1 ^f
Core power (MWt)	2568	2568	2452	2772	2772	2535
RCS fluid volume (ft ³)	11500	11500	11500	11500	11500	11500
Loop configuration	Lowered	Lowered	Lowered	Raised	Lowered	Lowered
HPI shutoff head (psi)	3000	3000	2800	1733	3000	2800
Separate makeup pumps?	No	No	No	Yes	No	No
Number of reactor vessel vent valves	8	8	8	4	8	8
PORV bore (in.)	1.094	1.094	1.156	1.500	1.156	1.094
Steam line break detection system?	No	Yes	Yes	Yes	Yes ^g	Yes

a. Represents all three units at Oconee.

b. Arkansas Nuclear One Unit 1.

c. Crystal River Unit 3.

d. Davis-Besse Unit 1.

e. Rancho Seco Unit 1.

f. Three Mile Island Unit 1.

g. Does not have main steam line isolation valves.

applicability of the results to both lowered-loop and raised-loop designs follows.

The results shown for Oconee Unit 1 in Sections 4.1 through 4.3 are thought to be generally applicable to all the other lowered-loop plants. The differences between Oconee and the other lowered-loop plants in core power and PORV size should result in only relatively minor differences in the timing of events. Other differences between plants caused by differences in the design of the ICS and the balance of plant are insignificant because of the transients selected in this study. In transients in which the ICS or the balance of plant plays an important role, the differences between plants in these components could become more significant. The major difference between the Oconee units and the other lowered-loop plants is that Oconee does not

have a steam line break detection system. This system should not play an important role in the three transients analyzed. The steam line break detection system acts to automatically isolate the steam generators when they are depressurized to about 600 psia. Since the lowest calculated pressure was about 700 psia (occurring in the steam generator overfill transient), this system would not have played an important role in any of the three transients for any of the lowered-loop plants. However, it is conceivable that this system might be activated in the steam generator overfill transient for certain plants because of the possible variation of the results (owing to code uncertainty or different assumptions regarding initiating event or equipment failures). If the steam line break detection system were actuated in both steam generators, Δ FW would be

automatically terminated in some plants, preventing the continued overfilling of the steam generators. The AFW flow rate would also be limited at TMI Unit 1 owing to a cavitating venturi in the AFW line, which would reduce the cooldown rate.

The results of the Oconee feed and bleed sensitivity calculations are also qualitatively applicable to the other lowered-loop plants. Because of the similarity of core powers, steam generators, and RCS fluid volumes, the time to steam generator dryout and RCS heatup rates after dryout should be very similar. Since all the lowered-loop plants have high-head HPI, it should be possible to establish feed and bleed as demonstrated in the sensitivity calculations. However, the time available to establish feed and bleed, particularly in the case with degraded HPI and the PORV unavailable, could be sensitive to relatively minor differences in the head and flow characteristics of the HPI pumps. Since these pump characteristics were not evaluated during this study, the time available to establish feed and bleed in the degraded HPI case should be considered a rough estimate of the time available in other plants. Similarly, the time required to fill the steam generators in the overfill transient depends on the AFW flow rate. Although the AFW flow capacities at the other plants are qualitatively similar to the Oconee value, the head and flow characteristics of these AFW pumps were not rigorously reviewed.

The calculated results for Oconee are not as applicable for Davis-Besse, the raised-loop plant. In the LOFW transient, the steam generators would dry out and the RCS would reach saturation temperature slightly earlier because of the higher power at Davis-Besse. However, the time between the loss of RCS subcooling and the start of core heatup would tend to be delayed because of the raised-loop configuration, in which more of the RCS liquid can drain back into the vessel, thus providing additional cooling time. The capacity to feed and bleed at Davis-Besse is not as great as at Oconee because Davis-Besse does not have high-head HPI pumps. Feed and bleed must be established earlier at Davis-Besse than at Oconee to prevent core heatup. Although the latest time at which the operators can successfully establish feed and bleed at Davis-Besse is not a result of this study, previous calculations indicate that feed and bleed can prevent core heatup if initiated within 20 min of the MFP trip.¹⁷ Because of the low head of the HPI pumps and the relatively low capacity of the high-head makeup pumps at Davis-Besse, feed and bleed probably would not prevent core heatup if the

PORV could not be opened, a different result than obtained for Oconee. In the SBLOCA transient, additional cold water would be added to the RCS at Davis-Besse because of the contribution of the makeup pumps. It is not known if the additional flow into the RCS would be sufficient to prevent the voiding at the top of the hot legs, the subsequent loss of natural circulation, and the loss of the steam generators as heat sinks as calculated for Oconee. The reduced number of reactor vessel vent valves at Davis-Besse is thought to be insignificant for all three transients. In the overfill transient, the Oconee results appear to be as applicable to Davis-Besse as to any of the other B&W plants.

4.4.2 Westinghouse. Results of the H. B. Robinson plant analysis have been reviewed to determine their applicability to Westinghouse plants in general. Westinghouse, unlike the other vendors, has supplied nuclear reactor systems in two-, three-, and four-loop configurations. Westinghouse plant design characteristics were reviewed in those areas deemed important for the three analyzed transients. Design parameters were taken from Reference 18 for older plants and Final Safety Analysis Reports^{19,20,21} for plants that started operation after 1979. This information may not always reflect the most recent modifications to the plants. Selected plant parameters are shown in Table 4-5. The H. B. Robinson analysis results are then extended to other Westinghouse plants based on these and other selected design differences between plants. Note that results may not be directly extrapolated owing to possible system differences not identified in this review.

Westinghouse has supplied nuclear steam supply systems (NSSS) of the two-, three-, and four-loop configuration. The NSSS core powers range from 600 MWt for older plants to 3425 MWt for the new plants currently coming on-line. The increase in core power has also increased the number of coolant loops. The lower core power NSSS are two-loop plants with the new NSSS having four loops. All Westinghouse plants use vertical U-tube steam generators. However, several variations in steam generator design have been employed through the years. The applicability of the H. B. Robinson results to the other plants will be discussed in more detail relative to each transient below.

The steam generator secondary mass, RCS volume-to-power ratio, PORV capacity, and HPI characteristics are the important system parameters during a LOFW transient. Steam generator secondary mass determines time to steam generator dryout and the start of primary system heatup. NUREG-0611 shows steam generator dryout times from

Table 4-5. A comparison of Westinghouse plant parameters

Plant	Rated Core Power (MWt)	RCS Fluid Volume (ft ³)	Volume-to- Power Ratio (ft ³ /MW)	Normalized PORV Capacity (lb/hr-MW)	HPI Shutoff Head (psi)
H. B. Robinson	2300	9167	3.98	95.5 ^a	1430
Trojan	3411	12540	3.67	61.6 ^a	2600
Salem 1	3338	12811	3.83	63.0 ^a	2760
Surry 1,2	2449	9200	3.75	86.0 ^a	2600
Farley	2652	9723	3.66	79.2 ^a	2600
Turkey Point 3,4	2200	9343	4.24	95.1 ^a	1517
North Anna 1	2775	9957	3.58	76.0 ^a	2550
Prairie Island 1,2	1650	6450	3.91	108.5 ^a	2168
Ginna	1300	6245	4.80	117.8 ^a	1485
Point Beach 1,2	1519	6450	4.24	117.9 ^a	1539
Zion 1,2	3250	12000	3.75	64.6 ^a	2600
Haddam Neck	1825	8800	4.82	115.1 ^a	2948
Indian Point 2	2758	12224	4.43	78.7 ^a	1517
Indian Point 3	3025	12032	3.97	78.7 ^a	1517
Beaver Valley	2652	9716	3.66	79.9 ^b	2536
D. C. Cook	3400	12612	3.71	61.8 ^b	2514
San Onofre 1	1347	6450	4.79	80.0 ^a	2600
Kewaunee	1650	6191	3.75	106.0 ^a	2167
Braidwood	3425	12257	3.58	61.3 ^a	1580
Millstone 3	3425	12240	3.57	61.3 ^a	1515
Wolf Creek ^d	3411	11393	3.34	61.6 ^a	1515
Yankee Rowe	600	2940	4.90	118.0 ^c	844

a. Represents the relief capacity for one PORV. This plant has two PORVs.

b. Represents the relief capacity for one PORV. This plant has three PORVs.

c. Represents the relief capacity for one PORV. This plant has one PORVs.

d. Wolf Creek is representative of Catawba 1&2, Commanche Peak 1&2, McGuire 1&2, and Watts Bar 1&2.

two-thirds to one and a half times that in H. B. Robinson, with the majority of the plants having larger steam generator dryout times.

The RCS volume-to-power ratio for H. B. Robinson is 3.98 ft³/MW. The RCS volume-to-power ratio for other Westinghouse reactors is within 15% of the H. B. Robinson value. The newer plants have the smallest volume-to-power ratios at around 3.56 ft³/MW. With similar volume-to-power ratios, primary system coolant heatup after steam generator dryout would be simi-

lar, though plants with the smaller ratios would heatup faster.

The PORV capacity and HPI injection rate are the other two critical items for LOFW transients. The PORV capacity and HPI flow rate should be viewed in relation to each other. A system with less relief capacity but higher shutoff head for the HPI may be as desirable as more relief capacity and lower shutoff head. Each Westinghouse plant has two or three PORVs, with the exception of Yankee Rowe. The PORV rated relief capacity to core

power ratio was reviewed for Westinghouse plants. H. B. Robinson has a ratio of 95.1 lbm/hr/MWt. Values for other Westinghouse plants range from 61.6 to 118 lbm/hr/MWt. However, H. B. Robinson has the lowest HPI shutoff head, with the exception of Yankee Rowe. Several of the Westinghouse designed plants have HPI systems capable of delivering flow at the PORV setpoint pressure.

In summary of the LOFW transient, H. B. Robinson provides plant response similar to the other Westinghouse plants. The timing of steam generator dryout may be slightly earlier in a few plants, but the majority of plants would dry out later. The heatup of the RCS should be very similar for all Westinghouse plants after steam generator dryout. The PORV and HPI capabilities for H. B. Robinson make the timing for initiation feed and bleed conservative compared to most of the other Westinghouse plants. This results from the lower shutoff head of the HPI pumps at H. B. Robinson.

The results of the H. B. Robinson SBLOCA calculation are thought to be generally applicable to the other Westinghouse plants. Similar phenomena are expected in all Westinghouse plants. However, the event timing is expected to be affected by the different sizes of the plants. For example, events would occur more rapidly at the older plants because of their smaller RCS fluid volumes. Event timing for other plants should be similar to that calculated for H. B. Robinson if the break was sized so that the ratio of break area to RCS fluid volume was identical to the ratio used in the H. B. Robinson SBLOCA calculation.

Since Westinghouse plants use vertical U-tube generators, plant response for the steam generator overfill transient should be similar to H. B. Robinson. Similar phenomena is expected in all the plants. However, the time required to fill the steam generators depends directly on the volume of the steam generators and the AFW flow rate, which were not closely reviewed in this study.

4.4.3 Combustion Engineering. A brief review was performed to determine how well Calvert Cliffs Unit 1 represented the other operating CE plants. This review was based on a variety of sources of information²²⁻²⁶ that may not always reflect the most recent modifications to the plants. In particular, the information for the newer CE plants, including Waterford Unit 3, San Onofre Units 2 and 3, and Palo Verde Units 1, 2, and 3, was limited. The results of the review are summarized in Table 4-6. Based on the brief review, we conclude

that Calvert Cliffs Unit 1 was a representative CE plant. The applicability of the Calvert Cliffs results to the other plants will be discussed in more detail relative to each transient below. However, no attempt was made to determine the applicability of the Calvert Cliffs results to the Maine Yankee plant because of significant design differences. Maine Yankee has three loops, each with its own steam generator, whereas the other CE plants have a two-by-four loop configuration with two steam generators.

The results of the calculation of the LOFW transient at Calvert Cliffs Unit 1 are generally thought to be representative of the other CE plants. Considering the spectrum of design variations in CE plants, Calvert Cliffs has intermediate values for rated core power, total RCS fluid volume, and volume-to-power ratios. According to Reference 22, the time required to dry out the steam generators in the older CE plants is within 12% of the value for Calvert Cliffs. Based on the RCS volume-to-power ratio, the time between steam generator dryout and loss of RCS subcooling and core uncover are expected to be within about 20% of the value for Calvert Cliffs. No differences are expected between Calvert Cliffs Unit 1 and Unit 2. The Calvert Cliffs results appear directly applicable to Millstone 2 and St. Lucie. Thermal-hydraulic events, such as when the RCS reaches saturation temperature, are expected to occur earlier at the newer CE plants, such as Arkansas Nuclear One Unit 2, Waterford Unit 3, and the San Onofre and Palo Verde units. Conversely, events are expected to occur later at the older CE plants, such as Fort Calhoun and Palisades, because of their lower volume-to-power ratio. If any of the CE plants have a direct reactor trip on MFP trip, steam generator dryout and core uncover would occur much later than at Calvert Cliffs, where reactor trip was caused by low steam generator level.

The newer CE plants do not have pressurizer PORVs. Consequently, these plants do not have the capability to cool the core using feed and bleed. The relatively low-head HPI pumps in these plants cannot deliver flow at the setpoint of the pressurizer SRVs. Thus, these plants cannot cool the core using the HPI system to pump flow out of the SRVs.

The results of the Calvert Cliffs feed and bleed calculation are generally applicable for the older CE plants, which do have PORVs, except for Maine Yankee. Calvert Cliffs has a slightly lower PORV flow capacity than the other plants when normalized to rated core power. The HPI shutoff

Table 4-6. A comparison of Combustion Engineering plant parameters

Plant	Rated Core Power (MWt)	RCS Fluid Volume (ft ³)	Volume-to- Power Ratio (ft ³ /MW)	Normalized PORV Capacity (lb/hr-MW)	HPI Shutoff Head (psi)
Calvert Cliffs 1,2	2700	11101	4.11	56.7 ^a	1257
Arkansas Nuclear One Unit 2	2815	9976	3.54	0.0	1517
Fort Calhoun	1565	7066	4.52	69.7 ^a	1387
Maine Yankee	2640	11026	4.18	57.0 ^a	2471
Millstone 2	2700	11015	4.08	59.8 ^a	1213
Palisades	2560	12400	4.84	60.5 ^a	1257
St. Lucie	2700	11101	4.11	59.8 ^a	1357
Waterford 3	3390	11800	3.48	0.0	1473
San Onofre 2,3	3390	11800	3.48	0.0	1495
Palo Verde 1,2,3	3800	14153	3.72	0.0	1863

a. Represents the relief capacity for one PORV. This plant has two PORVs.

head at Calvert Cliffs is as low or lower than all the CE plants with PORVs, except for Millstone Unit 2. The maximum makeup flow rate is similar in each plant. Thus, the Calvert Cliffs results conservatively bound the ability to feed and bleed in the older CE plants. However, the differences in flexibility to feed and bleed between the older CE plants are thought to be relatively minor, except for Fort Calhoun, which has significantly more capability to feed and bleed because of its lower power and higher relative relief capacity.

The results of the Calvert Cliffs SBLOCA calculation are thought to be generally applicable to the other CE plants. Similar phenomena are expected in all the plants. However, the event timing is expected to be affected by the different sizes of the

plants. For example, for the same break size, events would occur more rapidly at Fort Calhoun and Arkansas Nuclear One Unit 1 because of their smaller RCS fluid volume. Event timing for other plants should be similar to that calculated for Calvert Cliffs if the break was sized so that the ratio of break area to RCS fluid volume was identical to the ratio used in the Calvert Cliffs SBLOCA calculation.

The results of the Calvert Cliffs steam generator overfill calculation are also thought to be generally applicable to the other CE plants. Similar phenomena would be expected in all the plants. However, the time required to fill the steam generators depends directly on the AFW flow rate, which was not closely reviewed in this study.

5. CONCLUSIONS

The thermal-hydraulic responses of Oconee, H. B. Robinson, and Calvert Cliffs were generally similar for the three transients analyzed. The observed differences in plant responses were primarily related to design differences between the once-through and U-tube steam generators and their arrangement in the primary system. Conclusions drawn from the comparison of the LOFW, SBLOCA, and steam generator overfill transients follow.

Operator action is required to prevent core uncover and heatup in a total LOFW transient at Oconee, H. B. Robinson, and Calvert Cliffs. The thermal-hydraulic responses of all three plants to a total LOFW transient are similar except for the timing of events. Significant thermal-hydraulic events generally occurred first at Oconee, then at Calvert Cliffs, finally at H. B. Robinson. The steam generators dried out about 4 min after MFP trip at Oconee, about 20 min at Calvert Cliffs, and about 62 min at H. B. Robinson. The RCS reached saturation temperature and lost subcooling at 31 min after MFP trip at Oconee, at 66 min at Calvert Cliffs, and at 96 min at H. B. Robinson. Core uncover and heatup began at 57 min after MFP trip at Oconee, at 101 min at Calvert Cliffs, and at 126 min at H. B. Robinson. The differences in plant responses were primarily caused by the small initial liquid inventory in the Oconee steam generators and the delayed reactor trip at Calvert Cliffs.

Primary system feed and bleed can be used to successfully cool the core during a total LOFW, if initiated early enough, at all three plants. Feed and bleed was calculated to prevent core heatup if initiated at 31 min following MFP trip at Calvert Cliffs, 49 min at Oconee, and 104 min at H. B. Robinson. Hand calculations show that if the start of feed and bleed had been delayed by more than about 5 min from the times given above, core heatup would have begun at each plant.

Oconee has extensive capability to feed and bleed. Because of its high-head HPI pumps, feed and bleed can be used to prevent core heatup at nearly any time prior to the start of core uncover. Feed and bleed can also be successfully used at Oconee even with degraded HPI and without the PORV and HPVVs. A sensitivity calculation shows that one HPI pump delivering flow against the set-point pressure of the pressurizer safety relief valves is sufficient to prevent core uncover if initiated within 40 min of MFP trip. The capability to feed

and bleed at H. B. Robinson and Calvert Cliffs is not as extensive as at Oconee because of the lower head of the HPI pumps.

Condensate pumps can be used to prevent steam generator dryout in H. B. Robinson and Calvert Cliffs if the operators act to depressurize the steam generators during a LOFW transient. To prevent steam generator dryout, the operators must begin depressurizing the steam generators at about 2.5 min after MFP trip at Calvert Cliffs and at 46 min at H. B. Robinson. Based on these results, it appears worthwhile to review the Calvert Cliffs emergency operating procedures and supporting thermal-hydraulic calculations relative to the depressurization of the steam generators to maximize the probability of successfully using the condensate booster pumps according to the criteria appearing in the procedures. The condensate pumps can probably be used to provide core cooling at all three plants, even after steam generator dryout. However, the time by which condensate flow must be established to prevent core heatup was not determined analytically in this study.

Oconee, H. B. Robinson, and Calvert Cliffs remained in a safe condition during a SBLOCA initiated by a 0.01 ft² break with degraded HPI. No critical operator action necessary to maintain the safety of the plants was identified. The thermal-hydraulic responses of the three plants were similar early in the SBLOCA. One significant difference observed between the plants relates to the time that the break flow transitioned from single-phase liquid to a high-quality, two-phase mixture. Because of the reactor vessel vent valves present in B&W plants, the transition to two-phase flow at the break occurred between 29 and 43 min earlier at Oconee than at the other two plants. The earlier break flow transition resulted in less depletion of the RCS liquid inventory. Differences between plants were also observed relative to the ability of the operators to recover from the SBLOCA, owing to the lack of natural circulation that results from voiding in the hot leg piping at Oconee. At Oconee, the operator can depressurize the RCS, a step necessary to reach residual heat removal entry conditions, by opening the pressurizer PORV or momentarily operating the reactor coolant pumps. At H. B. Robinson and Calvert Cliffs, the operator has the additional option to depressurize the RCS by depressurizing the steam generators. Thus, H. B. Robinson and Calvert Cliffs have an additional RCS depressurization mechanism for certain break sizes that is not available at Oconee.

A steam generator overfill transient has the potential to be more severe at Oconee than at H. B. Robinson or Calvert Cliffs. The overfill transients investigated were initiated by a trip of the MFPs followed by an overfill with AFW. Because of their relatively small size, the steam generators filled more rapidly at Oconee than at the other two plants. The steam generator levels exceeded their normal posttrip values within 4 min of reactor trip at Oconee, within 20 min at H. B. Robinson, and within 65 min at Calvert Cliffs. The steam generators were full of liquid at 21 min after MFP trip at Oconee, at 45 min at H. B. Robinson, and at 106 min at Calvert Cliffs. The steam generator overfill caused a more rapid cooldown in Oconee than at the other two plants. The rate of cooldown of the average RCS temperature from the time of reactor trip to the time the steam generators filled with liquid was 256°F/hr at Oconee, 93°F/hr at H. B. Robinson, and 41°F/hr at Calvert Cliffs. The only critical operator action identified for this transient was to terminate AFW before liquid entered the steam lines. Liquid in the steam lines could possibly cause a steam line failure from water hammer or excessive hydrostatic loading. A steam line failure would result in a more severe overcooling transient.

Transients causing reactor trip challenge the steam generator safety relief valves more frequently at Oconee than at Calvert Cliffs or H. B. Robinson. The larger number of challenges at Oconee may result in more failures of the valves to fully reseal, thus further complicating the transient.

Results of the three plants studied have been extended to other plants designed by the same vendor. Effort was made to estimate the applicability of the results based on a brief review of the typicality of the three plants studied to other plants designed by the same vendor. A rigorous extension of these results to other plants was not undertaken and would require a comprehensive program, which is well beyond the scope of the current effort. The estimation of applicability was only intended to provide a quick overview of how the calculated results might extrapolate to other plants. In each case, the plant selected for analysis adequately represented the majority of plants for that vendor.

Seven of the eight B&W plants are lowered-loop with similar system volumes and rated core powers within 8% of the Oconee values. The results of the Oconee calculations give reasonable estimates of the thermal-hydraulic responses and event timings for the other lowered-loop plants. Davis-Besse, with its low-head HPI and raised loops, will respond somewhat differently during transients where these two design features are important. Davis-Besse does not have as great a flexibility for primary system feed and bleed because of the low-head HPI.

Results of the H. B. Robinson and Calvert Cliffs calculations can be used to provide estimates of event timing for Westinghouse and Combustion Engineering plants, respectively. Both Westinghouse and Combustion Engineering have, in general, supplied plants with similar configurations. This similarity allows extrapolation of the thermal-hydraulic results.

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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

The Nuclear Regulatory Commission has initiated a research program to develop a methodology to assess the operational performance of Babcock and Wilcox plants and to apply this methodology on a trial basis. The methodology developed for analyzing Babcock and Wilcox plants integrated methods used in both thermal-hydraulics and human factors and compared results with information used in the assessment of risk. The integrated methodology involved an evaluation of a selected plant for each pressurized water reactor vendor during a limited number of transients. A plant was selected to represent each vendor, and three transients were identified for analysis. The plants were Oconee Unit 1 for Babcock and Wilcox, H. B. Robinson Unit 2 for Westinghouse, and Calvert Cliffs Unit 1 for Combustion Engineering. The three transients were a complete loss of all feedwater, a small-break loss-of-coolant accident, and a steam-generator overflow with auxiliary feedwater. Included in the integrated methodology was an assessment of the thermal-hydraulic behavior, including event timing, of the plants during the three transients. Thermal-hydraulic results are presented in this volume (Volume 2) of the report. An overview of all results, with emphasis on operator and hardware performance, is presented in Volume 1.

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