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A Probabilistic Evaluation of the Safety of Babcock & Wilcox Nuclear Reactor Power Plants With Emphasis on Historically Observed Operational Events

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Prepared for
U.S. Nuclear Regulatory Commission

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A Probabilistic Evaluation of the Safety of Babcock & Wilcox Nuclear Reactor Power Plants With Emphasis on Historically Observed Operational Events

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ABSTRACT

This report summarizes a study performed by Brookhaven National Laboratory for the Office of Nuclear Reactor Regulation, Division of Engineering & System Technology (A/D for Systems), U.S. Nuclear Regulatory Commission. This study was requested by the NRC to assist their staff in assessing the risk significance of features of the Babcock & Wilcox (B&W) reactor plant design in the light of recent operational events. This study focuses on a critical review of submissions from the B&W Owners Group (BWOG) and as an independent assessment of the risk significance of "Category C" events at each operating B&W reactor. Category C events are those in which system conditions reach limits which require significant safety system and timely operator response to mitigate. A precursor study for each of the major B&W historical Category C events also was carried out. In addition, selected PRAs for B&W reactor plants and plants with other pressurized water reactor (PWR) designs were reviewed to appraise their handling of Category C events, thereby establishing a comparison between the risk profiles of B&W reactor plants and those of other PWR designs. The effectiveness of BWOG recommendations set forth in Appendix J of the BWOG SPIP (Safety and Performance Improvement Program) report (BAW-1919) also was evaluated.

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EXECUTIVE SUMMARY

This study was performed by the Risk Evaluation Group, Department of Nuclear Energy, Brookhaven National Laboratory (BNL) for the Office of Nuclear Reactor Regulation, Division of Engineering and System Technology (A/D for Systems), U.S. Nuclear Regulatory Commission. The objective of this study was to investigate the vulnerability of nuclear power plants with Babcock & Wilcox (B&W) reactors to severe transient events.

Despite implementation of improvements at B&W nuclear power plants following the accident at Three Mile Island Unit 2, a number of complex transient events have recurred at these plants during the past several years. Motivated by concern, the U.S. Nuclear Regulatory Commission (NRC) in early 1986, started a program of reexamining the basic design requirements of B&W reactors. One of the steps in the program called for a probabilistic risk assessment (PRA) of the B&W plants. Therefore, the B&W Owners Group (BWOG) submitted two reports (Phase I[1] and Phase II[2]) covering their risk assessment work. The BWOG work mainly evaluated the importance of "Category C" events, a class of the most serious transients historically observed at B&W plants.

BNL carried out a critical review of the BWOG reports as well as independently assessing the risk significance of Category C events at each of the operating B&W plants. Other specific tasks involved in the BNL study included: (i) a thorough review of existing PRAs for plants with B&W and Westinghouse reactors to appraise their handling of Category C type events, (ii) establishment of a comparison between the risk profiles of B&W plants and those of plants with other pressurized water reactor (PWR) designs, (iii) a precursor study based upon each of the B&W historical Category C events, and (iv) evaluation of the effectiveness of the BWOG recommendations in Appendix J of the BWOG SPIP (Safety and Performance Improvement Program) report (BAW-1919).

The main conclusions from this study are summarized in the following sections, according to the subject and the task involved.

E.1 Insights from Existing PRAs

A careful survey of the existing PRAs on Crystal River 3[3] and Oconee 3[4] revealed that, all of the initiating events observed in the historically occurring Category C events were taken into account in the selection of the initiating event. In general, Category C events were modelled and analyzed correctly. Neither of these PRAs, however, explicitly treats the problem of Pressurized Thermal Shock (PTS). For Oconee 3, the contribution to the total core damage frequency (CDF) from the accident sequences closely related to the Category C events was about 12.2%, of which 12% was due to undercooling transients. For Crystal River 3, the undercooling sequences contributed only about 0.5% to the total CDF, when the special case of station blackout was excluded. The contribution from overcooling sequences was also small (about 1%).

The existing PRAs for other PWR plants (Seabrook,[5] Millstone 3,[6] and Zion 1[7]) also were reviewed to evaluate their handling of events similar to B&W Category C events. Generally speaking, most of the initiating events related to Category C events were considered in these PRAs, although overcooling plant response was not explicitly treated in the Millstone 3 PRA. Also, the possibility of an overcooling transient leading to PTS was only investigated in the Seabrook PRA.

Accident sequences involving overcooling transients leading to core damage has no recognizable contribution to the total core damage frequency at these Westinghouse PWR plants. On the other hand, the risk significance of undercooling sequences varied from plant to plant. For Seabrook and Zion 1, undercooling sequences contributed roughly 5% and 2%, respectively, to the total frequency of core damage. For Millstone 3, undercooling sequences were significant contributors (about 37%) to the total frequency of core damage.

The results of these PRAs suggest that Category C events generally are somewhat less significant from the viewpoint of core-damage risk for other PWR plants as compared to B&W reactor plants.

E.2 Review of BWOG Phase I and Phase II Submittals

The Phase I and the Phase II submittals made to the NRC by the B&W Owners Group assessing the importance of Category C events and the risk associated with the B&W reactor design were carefully reviewed and the following conclusions regarding their technical accuracy and completeness were reached:

1. The frequencies of the transient initiating event relatively well established.

2. The event trees developed in the BWOG report are oversimplified. As a consequence, they are not fully representative of the historically observed Category C events. There is no clearly identifiable sequence on the event trees which adequately represents the observed scenario for each of the Category C events. Furthermore, in these event trees: (a) no explicit distinctions are made between the immediate success and temporary failure of the system, (b)operator's actions for recovery or control (such as emergency feedwater recovery) are not adequately displayed, and (c) the difference between minor overcooling and severe overcooling is not fully considered.

3. The branch-point probabilities used in quantifying the event trees were treated as if they were independent of each other and of the initiating events. As a result, the sequence frequencies may have been underestimated. Also, many of the plant-specific branch-point probabilities were obtained arbitrarily.

4. The BWOG report did not analyze cognitive human errors, nor the relationships between the error probabilities and the plant conditions.

5. Pressurized Thermal Shock (PTS) was not included in the BWOG analyses.

6. The frequencies of the Category C events were not directly addressed.

E.3 BNL's Independent Assessment of the Risk Significance of Category C Events at B&W Reactor Plants

Precursor Study

A precursor study was made on 12 B&W historical Category C events. The events that occurred at Davis Besse (September 9, 1977 and June 9, 1985) and Rancho Seco (March 20, 1978) were far more serious in terms of coming close to core melt, compared to the remainder of the historical Category C events. The exigency of these serious events almost exclusively stems from lack of cooling (i.e., undercooling). Our results suggest that, although Category C events involving over-cooling (such as those initiated by excessive feedwater or by a stuck-open secondary valve) occur rather frequently, for the most part they are relatively insignificant from the viewpoint of core damage.

E.4 Overall Conclusions

The major conclusions drawn from the results of this study are:

1. As a class, B&W plants do not have a core damage risk that is measurably greater than other PWR plants, although all the core damage scenarios that are similar to the historic Category C events are more likely to occur at B&W plants.

2. Overcooling events, which contribute over 99% of the predicted Category C frequency, are minor contributors to core damage at all plants.

3. The most significant Category C contributor to core damage at all B&W plants is undercooling, which is the least likely event to occur, but the most likely to lead to core damage. The dominant cause of core damage in these events is failure to re-establish feedwater and failure to establish feed-and-bleed.

4. Loss of ICS function is the most significant initiating event contributor to core damage from Category C events for all B&W plants.

5. Pressurized thermal shock (PTS) is more likely to occur at B&W plants than at other PWRs. However, PTS still is not a dominant contributor to core damage at the B&W plants.

6. With respect to the estimated overall CDF for each of the six B&W reactor plants, Category C events constituted a measurable contribution to four of these plants (Arkansas Nuclear One, Crystal River, Davis Besse, and Rancho Seco). However, these contributions do not dominate plant risk or significantly modify the plants' risk profile. The Category C contribution to overall CDF at the other two plants (Oconee and Three Mile Island) is minor.

7. Excessive feedwater was not found to be a significant contributor to core damage, either as an initiating event or as a subsequent failure.

8. Operator performance variations at B&W plants could have some impact on Category C risk significance.

E.5 References

1. Putney, B. et al., Risk Assessment Review of B&W Plant Transient History, Phase I, Science Applications International Corp., October 1986.

2. Averett, M. et al., Risk Assessment Review of B&W Plant Transient History - Phase II, B&W Owners Group, November 1986.

3. Science Applications International Corporation, "Probabilistic Risk Assessment for Crystal River Unit 3," February 1986 (Draft).

4. Sugnet, W. et al., "Oconee PRA - A Probabilistic Risk Assessment of Oconee Unit 3," NSAC-60, Nuclear Safety Analysis Center, EPRI, Palo Alto, California and Duke Power Co., Charlotte, North Carolina, June 1984.

5. Garrick, B.J. et al., "Seabrook Station Probabilistic Safety Assessment," PLG-0300, Pickard, Lowe and Garrick, Inc., December 1983.

6. Northeast Utilities Service Company, "Millstone Unit 3 Probabilistic Safety Study," August 1983.

7. Commonwealth Edison Company, "Zion Probabilistic Safety Study," 1981, and Revision 1, September 1982.

1. INTRODUCTION

Early in 1986, the NRC began to reexamine the design of Babcock and Wilcox Nuclear Power Plants because they were concerned about the occurrence of several operational transients. This concern was documented in a program plan,[1-1] which gave as its basis the following:

> "Since the TMI-2 accident, there has been a growing realization of the sensitivity of Babcock and Wilcox (B&W) plants to operational transients. By letter dated January 24, 1986, the acting EDO informed the Chairman of the Babcock and Wilcox Owners Group (B&WOG) that a number of recent events at B&W designed reactors lead us to conclude that there is a need to reexamine the basic design requirements for B&W reactors. The letter stated that the staff will reassess the overall safety of the B&W plants and determine whether the present set of requirements for these plants are appropriate for the long term and lead to a level of safety that is comparable to other pressurized water reactors."

The program plan developed included a probabilistically based assessment of the B&W plants, that is the subject of this report.

1.1 The Risk-Based Submittals of the B&W Owners Group (BWOG)

The BWOG risk assessment concentrated on the importance of the most serious class of transients historically observed at B&W plants. These transients are classified in the BWOG Safety and Performance Improvement Program (SPIP) Report[1-2] as Category C events. The definition of Category C events, adopted from the SPIP report, is that these events involve one or more of the following occurrences:

Parameter	Occurrence
Reactivity	Recriticality.
RCS Pressure	Pressure excursion causes either SI actuation or PORV/safety valve opening.
RCS Temperature	Exceeds PTS limit or results in loss of subcooled margin.
RCS Inventory	Pressurizer level goes offscale low with loss of subcooled margin or offscale high with PORV/safety valve lifting.
OTSG Pressure	Pressure excursion exceeds ASME code limit or results in OTSG isolation.
OTSG Inventory	Loss of all feedwater to both OTSGs, or a fill of either or both OTSGs to over 95% on the operating range.

The BWOG risk-based submittals consisted of a Phase I report and a Phase II report. The goals of the Phase I effort, as stated in the report,[1-3] are:

I. Assess the importance of B&W historical Category C events to core-melt risk. Category C transients are those wherein system conditions reach limits which require significant safety system and operator response to mitigate. There have been 13 such events and they are described herein.

II. Compare initiating event frequencies obtained from the transient history of all B&W units to the frequencies used in the Probabilistic Risk Assessments (PRAs).

III. Evaluate the dominant accident sequences, systems, and initiators from the PRAs. Compare these to the Category C events.

The goal of the Phase II effort, as stated in the report,[1-4] is:

IV. Generalize the results of the above analyses to all of the B&W units.

1.2 The Purpose of This Report

The NRC staff next requested Brookhaven National Laboratory (BNL) to perform an independent review and evaluation of the BWOG effort. Specifically, the tasks performed by BNL consisted of:

Review the existing PRAs on Crystal River 3 and Oconee 3 and assess their handling of Category C events (Section 2.1).

Review PRAs of other PWRs and assess their handling of events similar to B&W Category C events (Section 2.2).

Compare the risk profiles of B&W plants and other PWRs (Section 2.3).

Evaluate the technical accuracy and adequacy of the BWOG submittals in assessing the importance of Category C events (Section 3).

Develop an independent risk model to address the accuracy and adequacy of the concerns identified in the evaluation and quantify that model to identify the contribution of Category C events to core damage (Section 4.1).

Perform a precursor study to assess how close to core damage (in a probabilistic sense) each of the historical Category C events came (Section 4.2).

Combine the above information and determine the significance of Category C events to core damage risk at each of the operating B&W plants (Section 5).

Evaluate the effectiveness of the backfits planned by the BWOG for the SPIP in reducing the contribution of Category C events to core damage risk (Section 6).

References

1-1. Letter from D. M. Crutchfield/USNRC to H. R. Denton/USNRC on the subject of "Program Plan for B&W Design Reexamination," February 7, 1986.

1-2. B&W Owners Group, Safety and Performance Improvement Program, Revision 01, BAW-1919, The Babcock and Wilcox Company, August 1986.

1-3. Putney, B. et al., Risk Assessment Review of B&W Plant Transient History, Phase I, Science Applications International Corp., October 1986.

1-4. Averett, M. et al., Risk Assessment Review of B&W Plant Transient History - Phase II, B&W Owners Group, November 1986.

2. INSIGHTS FROM EXISTING PRAs

As the first step, we reviewed the results of a series of B&W and Westinghouse PRAs. By looking at how these PRAs evaluated Category C events and at the core damage obtained, it is possible to envision the potential differences in risk profile between B&W plants and Westinghouse plants, and to assess the extent to which they may be due to design differences which manifest themselves in the historically Category C events.

2.1 Insights from B&W PRAs

The insights from two B&W PRAs are discussed in this section. Section 2.1.1 deals with the Oconee-3 PRA,[2-1] and Section 2.1.2 with the Crystal River 3 PRA.[2-2]

2.1.1 Oconee-3

2.1.1.1 Summary of Results

Table 2.1 summarizes the results of the Oconee PRA, classified according to initiating event contribution to core melt frequency (CMF). Total loss of service water contributes 26% of the CMF. Other major contributors are large LOCAs (18%), which contribute about 1.5 times as much as small LOCAs (11%), and ATWS (11%): this is followed by steam-line/feedline break (9%), which will be discussed in more detail later in conjunction with Category C events.

Table 2.2 summarizes the results according to sequence classes. The dominant sequence classes are LOCAs (32%) and transient-induced LOCAs (32%) with early cooling failures. Various transient initiating events leading to early cooling failure also form a dominant sequence class (20%).

The generic data on initiating events from 35 PWRs (EPRI NP-801, 1982)[2-3] were used by the OPRA to generate a prior event-frequency distribution for each transient category. The operating experience of all three Oconee units then were used to generate plant-specific data on initiating events by means of the Bayesian updating process.

Most of the component failure data and maintenance unavailabilities used in the OPRA were found to be reasonable. The method used to develop the data base for the OPRA is the standard method used for plants with operating experience, i.e., Bayesian analysis to combine generic information obtained from industry experience with plant-specific data.

As part of the OPRA, a human reliability analysis was performed to assess the human contributions to core melt frequency. Four categories of human errors were studied: unavailability error (U), inadvertent actions (I), operator inhibits (OI), and operator fails to (OF). Of these, the first two are typically modelled at the subsystem or component level in the system fault trees, whereas the last two are generally modelled at the system level, i.e., at the top of the system fault tree. The failure to recover a system's operation or to find an alternative system, referred to as a "recovery error" in the OPRA, was normally modelled at the accident sequence level to ensure that any factors specific to the affected sequences are adequately considered. Dependences of human errors also are taken into consideration, although the dependence of human error on plant instrumentation was not specifically analyzed. Due to lack of reliable techniques and meaningful data, however, the human error probabilities quantified in the OPRA involve great uncertainty and subjectivity.

2.1.1.2 Handling of the Historically Observed B&W Category C Events

This section summarizes the findings obtained from the careful review of the Oconee PRA (OPRA) for assessing whether its core-melt risk profile adequately reflects the potential risk contribution of Category C events. Based on a survey of the characteristics of the transients, the B&W Owners Group report[2-4] concluded that the Category C events can be properly represented by transient sequences originated from four initiators, namely: (1) T_1 - Reactor/Turbine Trip, (2) T_2 - Loss of Main Feedwater, (3) T_3 - Excessive Feedwater, and (4) T_4 - Loss of Power to ICS. In this review, therefore, the sections of the OPRA dealing with the accident sequences induced by these four initiators were closely scrutinized to determine whether the Category C events are adequately analyzed. In addition, all the dominant cutsets listed in the OPRA were examined individually to decide whether they bear a close relationship with the Category C events.

a. Initiating Events

In the Oconee PRA, the transient sequences bearing the characteristics of Category C events were investigated within the framework of the event trees and fault trees developed to analyze transient initiating events and small-break LOCAs. In addition to the four initiating events analyzed in the B&W Owners Group Report, other initiators which have the potential of developing into the transient scenarios of the Category C event type, such as steamline break, partial loss of main feedwater, or feedwater-line break, were included in the analyses.

The generalized event tree for transient initiating events is applicable to 14 distinct initiators. Some of these sequences were eventually transferred to a small-break LOCA event tree, because they constitute transient-induced small LOCAs. The list of transient initiating events was obtained through refinement and grouping of a master list of 44 initiators, selected from a systematic appraisal of several sources. The loss of power to bus KI was selected as one of the transient initiating events because bus KI supplies most of the power to the ICS, which is closely related to the main feedwater system. Historical events at various B&W plants showed that such a failure constitutes an important initiator for the Category C events, because the ICS controls FW-control valves, startup FW-control valves, FW-pump speed, turbine control valves, and turbine-bypass valves. Besides the loss of ICS power, failures of other support systems, such as loss of instrument air or loss of service water, were also treated as initiators of the event tree for transient initiating events. Although these systems assume the important role of supporting the operations of EFW pumps, HPI pumps, FW control valves and turbine-bypass valves, transient sequences originating from these initiators are analyzed separately from the Category C events.

Next, a brief description is given regarding the response of the plant to each of the Category C transient initiating events, upon the scenarios that can lead to overcooling or undercooling of the RCS. The initiating event frequencies used in the OPRA also are shown.

1. Reactor/Turbine Trip (4.9/ry)

This represents several events that can lead to a reactor or turbine trip, but they were judged to have no significant direct impacts on the need for, or the availability of, plant systems. Following the trip, the plant usually stabilizes at hot-shutdown conditions, with the steam generators removing decay heat by operation of MFW and the TBS (turbine bypass system). Station power is provided by the Oconee substation via the startup transformer, which supplies power from the offsite grid. Core cooling, however, can be interrupted after a turbine trip when there are failures in the systems that provide RCS heat removal. For instance, the MFW runback creates the possibility for demand-type failures of the feedwater systems. Failure to transfer electric power to the startup transformer can cause dependence on emergency power, with the consequential loss of the RCPs (reactor coolant pumps) and an increased probability for the loss of all feedwater. A temporary interruption of RCS heat removal also can cause the pressurizer safety relief valves to open, creating the potential for a small-break LOCA if any of the valves fail to reseat.

2. Total Loss of Main Feedwater (0.64/ry)

According to the OPRA, there are several control-system and hardware malfunctions that can result in a loss of MFW to both steam generators. Once MFW is lost, both the reactor and the turbine will trip, and there will be an automatic demand for emergency feedwater (EFW) to provide backup cooling flow to the steam generators. If the EFW responds as designed, the subsequent plant response is similar to that of the turbine-trip event. If EFW is lost for about four minutes, the S.G. can dry out, causing RCS pressure to increase to the PORV or safety and relief valve (SRV) setpoints. The relief valves then will cycle to remove decay heat until feedwater flow to the steam generators is recovered. However, substantial makeup flow to the RCS must be provided to compensate for the RCS inventory loss through the relief valves. If both MFW and EFW are lost, core heat can be removed by manually opening the PORV and starting HPI flow to the core (feed and bleed). Core damage, however, can occur if the HPI cooling cannot be achieved.

3. Excessive Feedwater (9.2E-2/ry)

Control failures in the ICS, or MFW hardware malfunctions, can induce an excessive feedwater transient leading to overcooling of the RCS, shrinkage of the primary system and depressurization. A reactor trip is likely to occur in response to the resultant low RCS pressure, or temperature, or high flux. The overcooling ultimately causes an automatic or manual actuation of the HPI system, which could then repressurize the RCS to the pressurizer relief valve setpoint. Unless timely action is taken by the operator to control or terminate the HPI flow, a small-break LOCA may result if the relief valves stick open. To prevent overfilling of the steam generators, with the possibility of water being carried over into the turbine or spilling over into the steamlines, the MFW pumps are interlocked to trip on high water level in either steam generator. Therefore the excessive FW event may quickly turn from an overcooling transient to a loss-of-MFW (undercooling sequence).

4. Loss of Power to Bus 3KI (2.0E-2/ry)

This event involves loss of power to Bus 3KI, which supplies vital ac power for most of the instrumentation and control functions within the ICS. It results in tripping of the MFW pumps and loss of the capability to use the TBS. Furthermore, a number of control signals normally required to maintain the plant in a stable condition become unavailable to the operators. Historical events at B&W plants demonstrated that the failure ultimately leads to RCS overcooling, due to loss of control of steam generator cooling.

In addition to the transient initiators discussed above, the OPRA also treats partial loss of MFW, steamline-break, and feedwater/condensate line break as initiators for transient sequences leading to overcooling or undercooling of the RCS. They are summarized below.

5. Partial Loss of Main Feedwater (0.69/ry)

This event is an inappropriate reduction in flow to one or both steam generators without MFW becoming totally unavailable. These events usually bring about an RCS transient, that results in a reactor trip on high pressure. Unless MFW fails altogether after the trip, the subsequent transient response is analogous to the turbine-trip event. If MFW is completely lost, the transient sequence is classified as a total loss of MFW initiating event.

6. Steam-Line Break (3.0E-3/ry)

A steam-line break can also cause overcooling of the RCS due to depressurization and swelling of the secondary-side fluid and subsequent malfunction of the feedwater system. Depending upon the size of the break, the degree of overcooling can become severe enough to depressurize the RCS to saturated conditions. To prevent cavitation of the pump, the operators are instructed to trip the RCPs if the HPI system is automatically actuated on a low RCS pressure signal. The ensuing natural circulation, however, could be interrupted if sufficient voiding is formed in the RCS hot legs. As the RCS starts to heat up due to failure of removal of core heat, the resulting swelling of the coolant, together with the HPI flow, tend to collapse the RCS voids, thereby reestablishing communication between the core and the steam generators. If the HPI system is actuated and allowed to remain uncontrolled until the pressurizer becomes full of water, a small-break LOCA may occur if the safety relief valves fail to reseat after discharging liquid water. Because the faulty steam generator must be manually isolated, there is an increased possibility of losing RCS heat removal by the steam generators. All of these factors add to the potential for damaging the core starting from a steam-line break.

7. Feedwater/Condensate Line Break (9.3E-4/ry)

The OPRA studied two classes of breaks in the large feedwater/condensate line: (1) breaks that could happen inside the containment building and downstream from the isolation check valves, and (2) breaks that could occur elsewhere in the feedwater or condensate system. Initially, the RCS response to the first class would resemble that for a steamline break. After the steam generators dry out, however, the overriding effect would be a loss of secondary cooling, namely the undercooling of the RCS. For the second class of breaks, the RCS response would be similar to that for a loss of MFW, since the primary impact would be to interrupt the delivery of feedwater to the steam generators. Unless prompt manual action is taken to isolate the break or secure makeup to the hotwell, this initiating event may lead to loss of both feedwater systems, due to the interconnections between the suction sources for main and emergency feedwater.

b. Subsequent Plant Response

The function-oriented top events in the transient event tree and the small-break LOCA event tree are supported by a fault tree logic that relates failure of achieving a particular safety function to its causes, such as specific system or component failure, human errors, or the effect of initiating event. The structures of these trees, developed to support the event tree top events, were found to be logically sound and reasonably complete. The relevant system fault trees, such as those for Power Conversion System (PCS), Emergency Feedwater System (EFS), High Pressure Injection System (HPIS), and Integrated Control System (ICS) also were found to adequately represent the failure modes for each system. Therefore, Category C events, in general, are modelled correctly in the OPRA. However, we note that the potential for challenging the integrity of reactor vessels due to pressurized thermal shock (PTS), was not explicitly treated in the OPRA event tree nor in the fault tree analyses.

To evaluate the frequencies of core-melt sequences in a practical and efficient manner, the OPRA grouped the event tree sequences into six general categories or bins, each representing a type of core-melt sequence.

For each core-damage fault trees (CDFTs) are constructed by combining all the sequences, including all initiating events. The SETS code[2-5] was used to quantify the accident sequence for each bin, yielding the minimal cutsets at the bin-level. The quantitative results of the OPRA analyses are presented in Appendix D of the OPRA (Volume 4). The frequencies of all the dominant core-melt sequences, grouped according to the six bins, are presented including those of the important minimal cutsets. Based on individual scrutiny of all the dominant minimal cutsets, eight event tree sequences were determined to be closely related to the Category C events (Table 2.3). The approximate frequencies of these sequences add up to 6.6E-6/ry, roughly 12% of the total core melt frequency due to internal events. They are dominated by sequences belonging to core-melt bin III (i.e., TBU sequences) which involve failure to actuate HPI cooling following loss of all feedwater. Contributions to the total CMF from event tree sequences involving overcooling transients were relatively insignificant (roughly 6.0E-8/ry).

2.1.1.3 Description of Dominant Accident Sequences

A brief description of each dominant accident sequence follows. For convenience, the descriptions of the basic events (and their unavailabilities used in the OPRA calculations), which will be referred to in the following discussions are summarized in Table 2.4.

1. Event Tree Sequence $T_{10}QU_S$ (1.4E-8/ry)

The OPRA fault tree quantification revealed only one dominant cutset for this event tree sequence $T_{10}QU_S$,

T10*RCSRVLC* $\sum$ HPI (1.4E-8)

$\sum$ HPI denotes the unavailability of the HPI system, which was estimated to be 1.5E-4. This cutset implies that following a large break in a feedwater line, which shuts off the supply of both MFW and EFW, a pressurizer SRV fails to reclose after liquid relief and the HPI system also fails. Since the RCS inventory lost through the stuck-open SRV cannot be made up, core melt ensues.

2. Event Tree Sequence $T_{10}QY_SX_S$ (1.4E-8/ry)

This sequence also is initiated by a large break in the feedwater line resulting in a stuck-open SRV after discharging liquid. HPI is initially successful. However, since the break occurs inside the containment, the RBSS (reactor building spray system), which shares the suction source (BWST) with HPI pumps, is actuated. HPR (high pressure recirculation) fails to be established successfully upon depletion of the BWST inventory within two hours following the LOCA. The following cutset was dominant:

T10*CPT10I*RCSRVLC*YRBSH*(LPSUMPMF*RESUMPMF + XHPR2H) (1.4E-8)

3. Event Tree Sequence $T_{10}QX_S$ (2.8E-8/ry)

This sequence is initiated by a large break in the feedwater line resulting in failures of both MFW and EFW systems due to their common water supply. Main feedwater is not restored within ten minutes, and the pressurizer SRVs stick open after liquid relief. Fan coolers are operable and BWST inventory subsists for 12 hours. However, the HPR and LPR fail due to the operator's failure to initiate them. The only dominant cutset identified by the OPRA is:

T10*RCSRVLC*XHPLPR12H (2.8E-8)

4. Event Tree Sequence TQX_S (5.8E-8/ry)

Among the dominant event tree sequences in Table 3, this is the only sequence for which the origin of ultimate core damage can be traced back to overcooling of RCS caused by various transients including steam line break, secondary side depressurization of the steam generator, or loss of power provided by bus KI to the ICS. The following three minimal cutsets were dominant by the OPRA fault tree analyses.

T*RC4MVOCM*RCSRVLC*XHPLPR12H*MSRVC (2.1E-8)
T*RC4MVOCM*RCSRVLC*XHPLPR12H*IM41 (1.4E-8)
T*RC4MVOCM*RCSRVLC*XHPLPR12H*ICRDRTVO (1.4E-8)
Sum of other cutsets = 9.3E-9

These sequences originated from reactor turbine trip (due to all the transient initiating events, T = 7.0/ry), followed by additional failures such as MSRVC (two or more SG secondary-side main-steam relief valves fail to close) or IM41 (all four turbine bypass-valves fail to reclose due to common control faults) that have the potential of causing overcooling of RCS. Overcooling brings about primary system depressurization that actuate HPI, which, in turn, causes repressurization of the primary system. All the three cutsets contain basic events, RC4MVOCM, RCSRVLC, and XHPLPR12H. The occurrence of event RC4MVOCM (which is given a probability of 3.3E-2) implies that the PORV block valve is closed when PORV is demanded to open. This causes pressurizer SRVs to be challenged, and subsequently, the SRV fails to reclose after relieving liquid, creating a small-break LOCA. The fan coolers, meanwhile, successfully operate to maintain reactor building pressure below the setpoint of building spray, thus, allowing the BWST inventory to hold out for 12 hours. At that time, however, HPR and LPR fail due to operator's failure to initiate (event XHPLPR12H). Based on recent information (footnote on page D-17 of OPRA, Appendix D, Volume 4), however, the PORV block valve is closed 75-80% of the time to prevent PORV leakage. To reflect this, the probability of RC4MVOCM should be changed to 0.8. Although not explicitly shown, the three cutsets contain a basic event, QHPIH (operator fails to throttle HPI), which was conservatively assigned a value of 1.0 in the OPRA calculations. For the overcooling transients, operators generally have about 10 to 15 minutes, within which to throttle HPI so that a proper value for QHPIH would be 0.05. In addition, if a more up-to-date probability (8.0E-3) for the event MSRVC is used, the frequency of this sequence type would become roughly 1.1E-7 rather than the 5.8E-8 shown above, a difference which is not statistically significant in relation to the uncertainties in the analysis and does not alter the perception of the importance of this type of event to overall plant safety.

5. Event Tree Sequence T_2BU (1.2E-6/ry)

The OPRA identified eight dominant cutsets for this sequence:

T2*UTHPIH*EFUSTF*REFDW2	(7.7E-7)
T2*UTHPIH*EFM1A*EFM1B*REFDW1	(6.2E-8)
T2*UTHPIH*EFM17*EMF56*REFDW23	(3.5E-8)
T2*UTHPIH*EFTDPPR*EMF56*REFDW2	(2.3E-8)
T2*UTHPIH*EFM17*SWEFCCH*REFDW23	(1.4E-8)
T2*UTHPIH*EFTDPPR*SWEFCCH*REFDW2	(9.2E-9)
T2*UTHPIH*EFM17*SWEFPPAMF*SWEFPPBMF*REFDW23	(8.7E-9)
T2*UTHPIH*SW137MVF*EFM56*REFDW2	(6.2E-9)

Sum of other cutsets = 2.6E-7

Each of the cutsets is comprised of an initiating event, T2 (loss of MFW) and basic events involving failures of EFW, HPI cooling and feedwater recovery within 30 minutes of loss of all FW. It is noteworthy that all of the eight cutsets contain the basic event, UTHPIH (with probability of 0.01), indicating that failure of HPI cooling is largely attributable to the operator's failure to initiate the feed-and-bleed cooling following loss of all feedwater. The probability for operator's failure to recover feedwater in 30 minutes was assigned a value of 0.5 or 0.3, depending upon whether the number of recovery sources available is one or two. If the cutset also contains event EFM17, it was given a value of 0.2 based on a weighted average of REFDW2 and REFDW3 (with three recovery sources available). Failure of the EFW system is mostly due to local faults causing failures of turbine-driven EFW pumps, EFW discharge paths to the steam generators, or the suction flow path from UST (upper surge tank) to motor-driven EFW pumps.

6. Event Tree Sequence T'BU (4.2E-7/ry)

The initiator of these sequences is turbine/reactor trip due to any cause other than those directly causing a loss of main feedwater. Following the trip, MFW flow is lost and EFW fails to function properly. The operators fail to recover feedwater or initiate HPI cooling. Ten cutsets were determined to be dominant based on the OPRA fault tree quantifications:

T'UTHPIH*EFUSTF*MFM12*REFDW2	(6.2E-8)
T'UTHPIH*EFUSTF*IM2B*REFDW2	(5.9E-8)
T'UTHPIH*EFUSTF*IM2A*REFDW2	(5.9E-8)
T'UTHPIH*EFUSTF*MFSNGLH*MFM14*REFDW2	(2.5E-8)
T'UTHPIH*EFUSTF*MFM17*REFDW2	(2.1E-8)
T'UTHPIH*EFUSTF*IM15*REFDW2	(9.6E-9)
T'UTHPIH*EFUSTF*IM16*REFDW2	(9.6E-9)
T'UTHPIH*EFUSTF*MFSSH2*MFESU44H2*REFDW2	(8.6E-9)
T'UTHPIH*EFUSTF*IM1B*REFDW2	(6.8E-9)
T'UTHPIH*EFUSTF*IM1A*REFDW2	(6.8E-9)

Sum of other cutsets = 1.6E-7

The basic events UTHPIH, EFUSTF, and REFDW2 are common to all the cutsets signifying that the dominant cause for the failure of HPI cooling is the operator's failure to initiate it following loss of all feedwater. Also, that the failure of emergency feedwater systems is dominated by insufficient water in the upper surge tanks (USTs), which is the primary source of suction for the EFW pumps. Operating procedures require that the level in these tanks be at least five feet (30,000 gallons) whenever the EFW system is called upon to operate. The failure of operator to recover feedwater in 30 minutes corresponds to the case where two sources are available for the recovery. Loss of MFW following the trip, meanwhile, can be mainly ascribed to the occurrence of events, such as local faults in condensate system, common faults causing failure of CBPs (condensate booster pumps), or excessive feedwater flow through MFW control valves.

7. Event Tree Sequence T_{11}BU (6.0E-8/ry)

This sequence is characterized by the loss of ICS (integrated control system) resulting in tripping of the MFW pumps and, thus, loss of MFW. Emergency feedwater also fails and operators fail to recover feedwater or to initiate HPI cooling. The OPRA fault tree quantifications uncovered only one dominant cutset:

T11*UTHPIH*EFUSTF*REFDW1 (4.0E-8)
Sum of other cutsets = 2.0E-8

The failure modes of the HPI cooling and emergency feedwater system are identical to those discussed under 6. The failure probability for recovering the feedwater is larger (REFDW1 = 0.5), since only one recovery source was assumed to be available.

8. Event Tree Sequence $T_{10}BU$ (4.8E-6/ry)

The initiator for this sequence is a large break in the feedwater - or condensate-line resulting in loss of main and emergency feedwater inventory. Furthermore, the operator fails to recover feedwater or initiate HPI cooling. Only one cutset was preeminent among all the cutsets belonging to this sequence type:

T10*UTHPIH*REFDW1 (4.7E-6)
Sum of other cutsets = 7.6E-8

Despite the small initiator frequency (T_{10} = 9.3E-4/ry), the frequency obtained for this cutset is relatively large because it contains only two basic events (UTHPIH = 0.01, REFDW1 = 0.5). Since the initiator causes loss of both MFW and EFW, failures to restore the feedwater and to initiate HPI cooling in a timely manner lead to core melt.

2.1.2 Crystal River-3

2.1.2.1 Summary of Results

The results of the Crystal River-3 PRA are summarized on Tables 2.1 and 2.2. Table 2.1 presents the results by initiating event contribution to core damage frequency, and shows that the major contributor to core damage is a plant-specific support system initiator, loss of service water (2.3E-5, 40%). Other major contributors are small LOCAs (1.2E-5, 21%) and loss of offsite power (6.4E-6, 16%). No other initiating event contributes more than 8% to frequency of core damage.

Table 2.2 gives the results as sequence classes. The dominant sequence class is transients with failure of long-term cooling (3.1E-5, 53%), indicating that the traditional long-term cooling systems, along with auxiliary feedwater, are the plants primary vulnerability. However, this is not the case. This class appears primarily because of a plant-specific operating assumption. The PRA assumes that, given a transient where the RCS is nominally intact (no LOCA) and auxiliary feedwater is operating, some RCS makeup is still required in the long-term. In other PRAs, this type of sequence is usually assumed to end in success as long as AFWS can be operated over the long term. For Crystal River-3, however, makeup from at least one pump is assumed to be required within about 12 hours. This assumption is based on an analysis of the RCS pumps used at CR-3, which would have a normal post-trip leakage rate of about 10-15 gpm per pump. This rate is not big enough to be qualified as a LOCA, but when combined with the small RCS volume above the core indicative of B&W plants, it is enough to uncover the core in 10-12 hours if makeup is not provided. The major contributor to this sequence class is from loss of service water initiators, since service water provides one of the sources of cooling to the makeup pumps. The other sequence classes which are major contributors are LOCAs with loss of long-term cooling (1.8E-5, 30%) and station blackout (9.4E-6, 16%). The LOCA contribution indicates a vulnerability in the recirculation systems, and the contributors here are failures in the low pressure (removal of decay heat) parts of the system. Failures in the DHR system itself and in its cooling systems (the closed cooling system for decay heat removal and the raw water system) are the dominant contributors. Operator error also contributes to this sequence class (4% out of the 30%). The contribution from station blackout indicates a vulnerability in the onsite power system, along with loss of offsite power and failure to recover.

Overall, cognitive operator error only contributes 4% to total frequency of core melt. However, this is due primarily to a poor analysis of human reliability which is not consistent with the state-of-the-art, even though the PRA was recently completed. The PRA mentions all the recent developments in cognitive reliability analysis but does not use them. The screening numbers used are quite nonconservative and insufficiently documented. The use of these low screening values for human error eliminates most of the contribution and thus detailed analyses are generally not performed. Dependencies between cognitive actions are not evaluated. More sophisticated analyses performed in other recent PRAs have yielded values for cognitive error that are orders of magnitude higher than the CR-3 values for similar cognitive errors over the same or longer times. The contributions of cognitive human error to the frequency of core damage at CR-3 are probably seriously underestimated, thus masking one of the more obvious differences between B&W and other PWR designs, i.e., the shorter times available for action by the operator.

2.1.2.2 Handling of the Historically Observed B&W Category C Events

a. Initiating Events

All the initiating events observed in the historically occurring Category C events were considered in the selection process in the PRA. For the overcooling events, excessive feedwater and feed/steam line breaks were explicitly included, together with several contributing events, such as stuck open TBVs/ADVs and ICS failures. Failures of ICS, EFIC, or NNI equipment were considered as plant-specific compound initiating events, and were subsequently rejected based on engineering judgement of the design of these systems. All of these overcooling initiating events, and others which have not been historically observed as part of Category C events, were combined into the spurious HPI (transient initiator T6) model. The initiating events for Category C undercooling were also explicitly considered in the analysis.

b. Subsequent Plant Response

The plant's response to undercooling such as that observed in some historical B&W Category C events (initial loss of all feedwater) is modelled explicitly on the event trees. With the exception of station blackout (which is a special case), these sequences did not contribute significantly to core damage frequency (about 2.3E-7, less than 1%).

The plant's response to overcooling is treated as part of the event supporting logic. For example, the RCS integrity model shows that integrity can be lost (causing a LOCA) as a result of overcooling events (excessive feedwater, feed/steam line break, and failure of secondary valves) followed by automatic initiation of HPI and an unisolated primary safety or relief valve that is stuck open. The contribution of these events to frequency of core damage is about 1%. The analysis does not consider pressurized thermal shock (PTS) or core damage due to vessel rupture induced by PTS.

2.1.2.3 Description of Dominant Accident Sequences

TU (3.0E-5) - This sequence is a transient followed by successful secondary cooling but an inability to provide sufficient makeup flow within about 12 hours after the trip. The reason for this leading to core melt was discussed. Four initiating events contribute to this sequence, the dominant one being loss of service water, followed by failures in backup cooling to the makeup pumps and/or makeup pumps being in maintenance (2.3E-5). The remainder is contributed by reactor trip or loss of emergency ac bus A or B, along with maintenance and/or mechanical unavailability of makeup.

SX (1.2E-5) - This sequence is a small LOCA followed by successful high-pressure injection but failure of high-pressure recirculation. It is dominated by various combinations of failures in the removal system for decay heat and its cooling systems, and the equipment providing flow from the containment sump.

TBL1U (9.4E-6) - This sequence is a transient followed by loss of steam generator cooling and feed-and-bleed cooling and failure of the reactor coolant pump seals, leading to an early core melt. This sequence is completely dominated by loss of offsite power, followed by loss of all onsite power (station blackout).

AX (3.0E-6) - This sequence is a large LOCA followed by failure of low pressure recirculation. It is dominated by the same contributors as sequence SX.

RXZ (2.2E-6) - This sequence is a rupture in the steam generator tube followed by successful isolation of the steam generator cooling through the intact steam generator, successful depressurization and then failure of the decay heat removal system and failure to maintain injection. It is dominated by valve failures in the DHR system, in combination with an operator error in not refilling the BWST (which provides a suction source for injection).

TQX (4.4E-7) - This sequence is a transient-induced small LOCA, followed by failure of high-pressure recirculation. This sequence is dominated by the spurious HPI initiator (which, as previously alluded to, is due to overcooling-induced low-pressurizer pressure) followed by the primary safety or relief valves failing open, due to failure to terminate HPI flow (causing the LOCA), and failures of the high-pressure recirculation similar to those for sequence SX.

TBL1L2X (2.3E-7) - This sequence is a transient followed by failure of steam generator cooling, success of feed-and-bleed cooling, and failure of long-term cooling (recirculation). It is dominated by a loss of feedwater transient, mechanical failure of emergency feedwater and failure to recover (many mechanical failures are not recoverable in the short term).

2.2 Insights from Westinghouse PWR PRAs

A discussion of insights from three Westinghouse PRAs is presented in this section. Section 2.2.1 deals with the Seabrook PRA,[2-6] Section 2.2.2 deals with the Millstone-3 PRA,[2-7] and Section 2.2.3 deals with the Zion PRA.[2-8]

2.2.1 Seabrook

2.2.1.1 Summary of Results

The results of the Seabrook PRA are summarized on Tables 2.1 and 2.2. Table 2.1 presents the results by initiating event contribution to core damage frequency. The table shows that the major contributors to core damage are loss of offsite power (6.9E-5, 44%), general plant transients (3.8E-5, 24%), and small LOCAs (2.0E-5, 13%). No other initiating event contributes more than 5% to frequency of core damage. It is notable that plant-specific support system initiators are minor contributors to core damage.

Table 2.2 presents the results in terms of sequence classes. The major contributor is station blackout (6.9E-5, 44%), indicating that the major vulnerability of the plant's safety design is the onsite electric power system (including its support systems), and a moderately high loss of offsite power frequency and failure to recover curve. The other major contributors involve failure of long-term cooling. LOCAs, transients, and transient-induced LOCAs with the failure of long-term cooling together account for 41% of total frequency of core melt. Thus, it appears that the short term cooling function provided by the combined action of auxiliary feedwater, high-pressure injection, and/or feed-and-bleed cooling is generally reliable, whereas the long-term cooling function, which involves high-pressure recirculation (for LOCAs and feed-and-bleed) and long-term stabilization of AFWS heat removal (for transients) is not as reliable. It should be noted that a large portion of the contribution from transient with loss of long-term heat removal is due to operator error (14% out of 15%). This PRA is unusual in that it considered the need for the operator to take manual action in the long-term during transients when AFWS was available. Most other PRAs consider the sequence to be over if AFWS is successful and LOCA conditions do not exist. Overall, operator's error contributes 18% to the total frequency of core damage, and the treatment of operator error in the PRA is reasonable and consistent with the state-of-the-art at the time the PRA was performed.

2.2.1.2 Handling of the Type of Undercooling and Overcooling Transients Similar to the Historically Observed B&W Category C Events

a. Initiating Events

All of the initiating events which appeared in the Category C events are considered in the PRA. Excessive demand for feedwater and secondary steam (steamline break, open atmospheric dump valves) are explicitly identified as the main steam isolation valves are considered as part of the subsequent plant response; however, the failure probability is modified, based on the effect of any given initiating event, so that these are implicitly evaluated as compound initiating events. Total loss of feedwater also is handled as a compound initiator through the subsequent plant response. Initiators which can fail main feedwater are clearly identified and evaluated and their effects on the availability of auxiliary feedwater are propagated into determination of AFWS failure probability for that initiator. The explicitly modelled initiators included failures of instrumentation buses, power buses, and other support systems, and these compound initiators are adequately modelled. The failures of the instrument bus are the Westinghouse equivalent of the ICS and NNI failures seen in the B&W Category C events.

b. Subsequent Plant Response

The plant response to undercooling, analogous to that observed in some historical B&W Category C events (initial loss of all feedwater), is modelled explicitly and in detail on the event trees. The sequences modelled are very similar to B&W sequences, with the exception that available response times are generally longer. Other than station blackout (a special case) these sequences did not contribute significantly to frequency of core damage (about 5% of the total). Most of these sequences involved failures in support systems (ac, dc) which affected the availability of emergency feedwater and/or feed-and-bleed cooling.

The plant response to overcooling also is modelled explicitly on the event trees. Failure of secondary isolation when required (failure of turbine bypass valves or MSIVs to isolate or properly control the steam flow) is modelled on the trees as leading to serious overcooling and a subsequent RCS depressurization to below the HPI initiation setpoint. Excessive feedwater flow, while included in the PRA, is not considered to lead to serious overcooling in and of itself. This is reasonable for Westinghouse plants since the steam generator tubes are always completely covered: the massive increase in heat transfer observed in B&W plants when excessive feedwater flow covers previously uncovered tube sections would not be expected in a Westinghouse plant. The event trees explicitly model plant response during the overcooling in the following manner. If HPI fails to start, core damage is assumed; this probably is a very conservative assumption. If HPI successfully starts, repressurization can occur, but credit is given to the operator to prevent this by taking actions to control feedwater and HPI flow. If the operator's action fails, PTS is assumed to occur so there is a potential for rupture of the vessel and subsequent core damage. The case of repressurization causing a PORV to lift and stick open is not considered; however, the time allowed for the operator to prevent PTS is only 30 minutes, which is probably less than repressurization to the PORV setpoint would take. Thus, the analysis is conservative in assuming PTS (rather than just PORV lift) occurs in this time. Further, the analysis assumes a conditional probability of vessel rupture/core damage for these scenarios of 0.01, a very conservative (high) number for this type of event see (NUREG/CR-4183).[2-9] Even with all of these conservatisms, sequences involving overcooling leading to core melt had negligible contribution to the frequency of core damage.

2.2.2 Millstone Point-3

2.2.2.1 Summary of Results

The results of the Millstone Point-3 PRA are summarized in Tables 2.1 and 2.2. Table 2.1 shows that the dominant contributor is the plant-specific support system initiator, loss of a vital ac or dc bus (1.0E-5, 22%). This contribution is split between loss of an instrument ac bus (17%) and loss of a vital dc bus (5%). Other major contributors include loss of offsite power (6.6E-6, 15%), general transients (6.3E-6, 14%), medium LOCA (5.5E-6, 12%), and loss of PCS transients (4.8E-6, 11%). No other initiator contributes more than 8% to total frequency of core damage. The interesting feature of these results is that no single initiator clearly stands out as a significant dominant contributor.

Table 2.2 presents the results by event sequence class. The dominant contributor is transients with loss of early cooling (1.6E-5, 37%). All of the dominant sequences in this class involve failure of auxiliary feedwater combined with operator failure to recognize the need to establish feed-and-bleed, thus clearly identifying the dominant vulnerability at the plant. The next two dominant contributors are LOCAs with loss of long-term cooling (8.7E-6, 19%), and transients with loss of long term cooling (7.6E-6, 17%). The dominant contributors to these sequence classes are similar, with the LOCAs being dominated by failures in high or low pressure recirculation, and the transients being dominated by failures in high pressure recirculation in sequences where AFWS fails and feed-and-bleed cooling is successful. Thus, the recirculation system is the second major vulnerability at the plant. Finally, there is an ac power vulnerability which results in a significant contribution from station blackout (4.7E-6, 11%). As stated, cognitive human error was a key contributor to the most dominant sequence class, and also was a contributor to some other sequence classes; in total human error contributed to 47% of the total frequency of core damage. The analysis of cognitive human error is detailed, and while the quantification of human error probabilities is not up to present day standards (as for Seabrook), it is up to the standards for the time the PRA was performed, and adequately reflects the contribution of human errors to core damage.

2.2.2.2 Handling of the Type of Undercooling and Overcooling Transients Similar to the Historically Observed B&W Category C Events

a. Initiating Events

All of the initiating events which appeared in the Category C events are considered in the PRA. Excessive supply of feedwater and excessive demand for secondary steam (steamline break, open atmospheric dump valves) are explicitly identified as initiating events. Total loss of feedwater is handled implicitly as a compound initiator through the subsequent plant response. Initiators which can fail main feedwater are clearly identified and evaluated, and any effects on the availability of auxiliary feedwater are propagated into the determination of the AFWS failure probability for that initiator. The explicitly modelled initiators included failures of instrumentation buses, power buses, and other support systems, and these compound initiators are adequately modelled. Failures of the instrument bus are the Westinghouse equivalent of the ICS and NNI failures seen in the B&W Category C events.

b. Subsequent Plant Response

The plant's response to similar to that observed in some historical B&W Category C events (initial loss of all feedwater) is modelled explicitly and in detail on the event trees. The sequences modelled are very similar to B&W sequences, except that available response times are generally longer. These sequences constitute a significant contribution to core damage (1.6E-5, 37%, excluding station blackout, which is a special case). Most of these sequences involved failures in support systems (ac, dc) which affected the availability of emergency feedwater and/or feed-and-bleed cooling, but also there was a significant contribution from nonsupport system transients.

The plant's response to overcooling was not treated either explicitly or implicitly in the analysis. Although the initiating events which lead to overcooling were included in the analysis, the modelling of subsequent plant response was identical to that of other loss of PCS transients.

2.2.3 Zion Unit-1

2.2.3.1 Summary of Results

The risk potential from core damage of the Zion Unit-1 plant was first investigated in the Zion Probabilistic Safety Study (ZPSS). The results, however, were repeatedly reexamined and revised during subsequent studies including the Review and Evaluation of the ZPSS (NUREG/3300, commonly referred to as the Zion Review),[2-10] the Zion IDCOR baseline analysis[2-11] and the Zion ASEP rebaseline analysis.[2-12] The list of dominant contributors shown in Table 2.1 for Zion-1 was compiled mainly from the risk profile developed in the Zion ASEP rebaseline analysis (NUREG/CR-4550, Vol. 6, July 1986). This risk profile, published in the Appendix C of NUREG-1150 report[2-13] (App. A-1, Vol. 2, February 1987), was constructed basically by modifying and updating the dominant accident sequences in the Zion Review. In Table 2.1, the frequencies of some accident sequences, such as steam generator tube rupture (SGTR), steamline break, or ATWS, which did not appear in the Zion ASEP rebaseline core damage profile due to their insignificance, were evaluated on the results of event tree analyses in the original ZPSS.

Table 2.1 shows that a salient feature of the Zion-1 risk profile for core damage is that the frequency of core damage is dominated by a comparatively small number of accident sequences. The most dominant sequence is the failure of Component Cooling Water System (CCWS), leading to failure of all charging and safety injection (SI) pumps, and the development of RCP (reactor coolant pump) seal LOCAs. The dominant cause of the failure of CCWS is a rupture of pipe. This sequence alone is almost 74% of the total frequency due to internal events. The next three dominant sequences are small, large and medium LOCAs, each followed by failure of recirculation cooling, or, in the case of large LOCAs, also failure of the low-pressure injection. These three LOCA sequences account for about 17% of the total frequency of core damage (due to internal events) for Zion-1.

There are two accident sequences involving loss of offsite power, followed by loss of both AFWS (auxiliary feedwater system) and feed-and-bleed capability, with ac power restored during the periods of one to four hours and four to eight hours, respectively. The failure of AFWS and feed-and-bleed is essentially dominated by failures of various vital ac power buses. Several of the remaining sequences shown in the Zion ASEP rebaseline profile involve loss of offsite power, followed by loss of either CCWS or SWS, with successful restoration of ac power at various times.

2.2.3.2 Handling of the Type of Undercooling and Overcooling Transients Similar to the Historically Observed B&W Category C Events

a. Initiating Events

Virtually all of the initiating events which appeared in the Category C events are considered in the ZPSS. An event tree is specifically constructed to analyze loss of feedwater (FW) flow transients, including loss/reduction of FW flow in one steam generator and loss of FW flow in all steam generators.

An event tree also is developed to analyze the transient sequences involving rupture of the steamline and of the feedline inside the containment as well as inadvertent opening of steam relief valve or safety valves. Rupture of the steamline outside the containment and the steam dump valves failing open are treated by a separate event tree designated for loss of steam outside containment. Excessive (or increase in) FW flow in one or more steam generators is included in the category of initiating events of turbine trip (general).

b. Subsequent Plant Response

Table 2.1 shows that none of the Zion dominant sequences pertains directly to Category C events. Transient sequences bearing the characteristics of Category C events, such as steamline break (overcooling) or loss of MFW (undercooling), were analyzed in the original ZPSS by specifically constructing event trees. The frequencies included in Table 2.1 for these sequences were evaluated, based on the results of the ZPSS event-tree analyses. These sequences, however, were not identified as dominant contributors to core damage frequency either by the Zion Review or by the Zion ASEP rebaseline analysis; a few sequences originating from loss of MFW or reactor trip/turbine trip appeared in the IDCOR baseline profile of core damage, which was obtained by partially updating the ZPSS. The contribution of these sequences to the total core damage frequency (due to internal events), was less than 3%. The core damage profile of Zion-1 is unique in that it is dominated by a single initiating event (loss of component cooling water), which contributes about 74% of the total core-damage frequency.

2.3 Comparison of Risk Profile of B&W Plants vs. Westinghouse Plants

In comparing the risk profiles of B&W plants and the plants of other vendor designs, it is essential to bear in mind that the most distinctive difference between them is the use of once-through steam generators (OTSGs) by B&W plants, as opposed to U-tube steam generators by other vendors, as the major heat sink for removal of core heat. Unlike the U-tube steam generator, the once-through steam generator (OTSG), which produces superheated steam, generally has a comparatively small secondary-side water inventory, so that its heat capability for removal is rather sensitive to variations in the feedwater supply. Under extremely adverse conditions, malfunction of the feedwater system can cause the secondary-side of the OTSG to either dry out or become water solid in as little as four to five minutes. Since the primary-side thermal hydraulic condition directly reflects the extent of heat removal by the steam generators, excessive or insufficient feedwater supply can readily result in overcooling or undercooling of the RCS in the B&W reactors. In addition, the primary pressurizer volume in B&W plants is smaller than that of other PWRs, thus the speed and severity of primary response to changes in feedwater supply is greater in the B&W plants. Any remedial action which the operator must take to prevent core damage, regardless of whether it involves primary or secondary system operations (such as the initiation of feed-and-bleed cooling or EFW recovery) must be performed successfully within a relatively short time from the inception of the transient. For a given accident sequence, the length of time available for the operator to make correct diagnosis and take proper corrective action can be predicted from thermal hydraulic calculations. For example, calculations carried out for Oconee for total loss of feedwater transient with no FW recovery, indicated that pressurizer would become water solid in approximately ten minutes, and that core uncovery would begin about 30 minutes after the onset of the transient if the RCPs (reactor coolant pumps) are not tripped. The short time within which the operator must respond correctly and take proper action to avert the possible core damage has a great impact on predicted operator's performance.

For PWRs of other vendor designs, which employ U-tube steam generators, the operator generally has much longer times available to take necessary corrective actions, since the secondary-side inventory of the U-tube steam generator is considerably larger. Furthermore, the capability of heat removal by U-tube steam generators is less susceptible to change due to excessive supply of FW, because the entire U-tubes (heat-transfer areas) are normally submerged in the two-phase water mixture. Hence, increasing the secondary-side water level by supplying excessive feedwater will not readily induce overcooling of the RCS. On the other hand, the heat-transfer characteristic of OTSGs is more prone to change due to excessive supply of FW feedwater, because roughly the lower 60% section of the straight tubes (heat transfer areas) is normally in contact with two-phase water mixture. The remaining upper section of the straight tubes is usually exposed to super-heated steam. Thus, raising the secondary-side water level is tantamount to increasing the effective heat-transfer area, which can significantly promote the heat-transfer capability of the OTSGs.

To compare the risk profiles of B&W plants with other vendor designs with respect to the significance of Category C events, a careful survey of the core-damage profiles for Seabrook, Millstone 3, and Zion 1 was conducted, using the results in Tables 2.1 and 2.2. Depending upon the authors of the PRAs, there are variations in the detail and the means by which the Category C type events are treated. Generally speaking, most of the initiating events which appeared in the Category C events are considered in these PRAs, such as excessive demand for secondary steam (steamline break, open atmospheric dump valves or MSIV) or loss of feedwater. For Seabrook and Millstone 3, failure of instrumentation buses (Westinghouse equivalent of the ICS and NNI failure observed in the B&W Category C events) also is explicitly modelled as one of the initiators. The possibility of an overcooling transient leading to PTS (pressurized thermal shock), was expressly investigated only in the Seabrook PRA.

The undercooling sequences modelled in these PRAs are similar to those for B&W plants, except the response times available to the operator are generally longer. For Seabrook, excessive feedwater flow, while included in the PRA, is not considered to lead to serious overcooling for the reason suggested earlier. For Millstone 3, plant response to overcooling was not explicitly treated. Although the initiating events which lead to overcooling were included in the analysis, the modelling of subsequent plant response followed that of other loss of PCS transients.

The results of these PRAs indicate that sequences involving overcooling leading to core damage generally have no recognizable contribution to the total frequency of core damage at these Westinghouse PWR plants. The risk significance of undercooling sequences, on the other hand, varies from plant to plant. For Seabrook and Zion 1, undercooling sequences do not contribute significantly to core-damage frequency (about 5% and 2% respectively of the individual total CD frequency). By contrast, undercooling sequences were significant contributors to core-damage frequency (about 37%) at Millstone 3. Most of these sequences involve failure in support systems (ac, dc), which affect the availability of emergency feedwater and/or feed-and-bleed cooling.

It can be concluded that Category C events are somewhat less significant from the viewpoint of core damage risk for Westinghouse plants compared to B&W plants.

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Table 2.1
Comparison of Core-Damage Profiles
by Initiating Events

	Oconee-3	Crystal River-3	Zion-1	Seabrook	Millstone Point-3
Large LOCA	9.7E-6 (18%)	3.0E-6 (5%)	6.3E-6 (4%)	1.4E-6 (1%)	2.4E-6 (5%)
Medium LOCA	---	---	4.9E-6 (3%)	9.9E-7 (1%)	5.5E-6 (12%)
Small LOCA	6.1E-6 (11%)	1.2E-5 (21%)	1.6E-5 (10%)	2.0E-5 (13%)	1.6E-6 (3%)
Steam Generator Tube Rupture	2.7E-6 (5%)	2.2E-6 (4%)	2.2E-7 (neg)	1.7E-6 (1%)	1.7E-6 (4%)
Steamline/Feedline Break	4.9E-6 (9%)	---	1.0E-8* (neg)	8.4E-6 (5%)	3.5E-6 (8%)
Loss of PCS Transient	1.6E-6 (3%)	2.3E-7 (1%)	5.6E-7* (neg)	7.6E-6 (5%)	4.8E-6 (11%)
Spurious Safety Injection	2.5E-7 (1%)	4.4E-7 (1%)	3.1E-6* (2%)	1.1E-6 (1%)	5.3E-8 (neg)
Loss of Offsite Power	2.3E-6 (4%)	9.4E-6 (16%)	3.9E-6 (2%)	6.9E-5 (44%)	6.6E-6 (15%)
Total Loss of Service Water	1.4E-5 (26%)	2.3E-5 (40%)	---	2.5E-6 (2%)	---
Loss of a Service Water Train	---	---	---	---	7.1E-7 (2%)
Loss of an Emergency AC Bus	---	2.5E-6 (4%)	---	---	---
Loss of a Vital AC/DC Bus (Including ICS/NNI)	7.6E-7 (1%)	---	5.0E-8 (neg)	2.3E-6 (1%)	1.0E-5 (22%)
Loss of Component Cooling Water	---	---	1.2E-4 (74%)	1.4E-6 (1%)	---
Loss of Instrument Air	3.2E-6 (6%)	---	---	---	---
Interfacing System LOCA	1.4E-7 (neg)	---	1.0E-7 (neg)	1.8E-6 (1%)	1.9E-6 (4%)
Reactor/Turbine Trip	1.8E-6 (3%)	---	8.2E-7* (1%)	---	---
Vessel Rupture	---	---	---	2.7E-7 (neg)	---
Incore Instrument Tube Rupture	---	---	---	---	1.6E-7 (neg)
General of Other Transients	1.0E-6 (2%)	4.6E-6 (8%)	Neg.	3.8E-5 (24%)	6.3E-6 (14%)
ATWS	6.0E-6 (11%)	---	6.7E-6 (4%)	---	---
Total	5.4E-5	5.8E-5	1.6E-4	1.6E-4	4.5E-5

*Frequencies estimated based on the results of the ZPSS.

Table 2.2
Comparison of Core-Damage Profiles
by Sequence Classes

	Oconee-3	Crystal River-3	Zion-1	Seabrook	Millstone Point-3
LOCA (Including SGTR), Early Cooling Failure	1.7E-5 (32%)	Negl.	1.4E-6 (1%)	3.1E-6 (2%)	2.8E-6 (6%)
LOCA (Including SGTR), Late Cooling Failure	2.0E-6 (4%)	1.8E-5 (30%)	2.6E-5 (16%)	1.9E-5 (13%)	8.7E-6 (19%)
Transients (Including SLB), Early Cooling Failure	1.1E-5 (20%)	Negl.	1.4E-6 (1%)	6.5E-6 (4%)	1.6E-5 (37%)
Transients (Including SLB), Late Cooling Failure	1.9E-7 (neg)	3.1E-5 (53%)	Negl.	2.3E-5 (15%)	7.6E-6 (17%)
Transient Induced LOCA, Early Cooling Failure	1.7E-5 (32%)	Negl.	1.2E-4 (74%)	Negl.	Negl.
Transient Induced LOCA, Late Cooling Failure	2.9E-7 (1%)	4.4E-7 (1%)	3.1E-6 (2%)	2.0E-5 (13%)	Negl.
Station Blackout	1.4E-7 (neg)	9.4E-6 (16%)	3.0E-6 (2%)	6.9E-5 (44%)	4.7E-6 (11%)
Interfacing Systems LOCA	1.4E-7 (neg)	Negl.	1.0E-7 (neg)	1.8E-6 (1%)	1.9E-6 (4%)
ATWS	6.0E-6 (11%)	Negl.	6.7E-6 (4%)	1.3E-5 (8%)	2.8E-6 (6%)
Total	5.4E-5	5.8E-5	1.6E-4	1.6E-4	4.5E-5

Table 2.3
Dominant Event Tree Sequences Closely
Related to Category C Events (OPRA)

Bin	Sequence Type	Event Tree Sequence	Frequency (per reactor year)
I	F	$T_{10}QU_s$	1.4E-8
I	H	$T_{10}QY_sX_s$	1.4E-8
II	D	$T_{10}QX_s$	2.8E-8
II	E	TQX_s	5.8E-8
III	A	T_2BU	1.2E-6
III	C	$T'BU$	4.2E-7
III	D	$T_{11}BU$	6.0E-8
III	E	$T_{10}BU$	4.8E-6
Sum			6.6E-6

T: Summation of all transient frequencies.
T': Summation of all transient frequencies excluding loss of feedwater as an initiator.
T_2: Loss of main feedwater.
T_{10}: Feedline break.
T_{11}: Loss of ICS power bus KI.
B: Failure of RCS heat removal via the steam generators.
Q: Loss of RCS integrity.
U or U_s: Failure of core-heat removal by HPI cooling.
Y_s: Failure to maintain RCS makeup supply.
X_s: Failure to maintain long-term heat removal.

Table 2.4
Basic Events Appearing in the Dominant
Cutsets Related to Category C Events (OPRA)

Event	Mean Unavailability	Description
CPT10I	0.1	Feedwater-line break inside containment.
EFM1A	4.4E-3	Failure of EFW discharge path to SG3A due to local faults.
EFM1B	4.4E-3	Failure of EFW discharge path to SG3B due to local faults.
EFM17	5.6E-2	Local faults cause failure of turbine-driven EFW pump.
EFM56	4.9E-4	Suction flow from UST to motor-driven EFW pumps unavailable due to local faults.
EFTDPPR	2.4E-2	Turbine-driven EFW pump fails to run during the event.
EFUSTF	4.0E-4	Insufficient level in USTs (upper surge tanks).
ICRDRTVO	2.0E-3	Relay contact from CRD system indicating reactor trip fails to open.
IM1A	1.0E-3	Excessive feedwater flow through MFW control valve FDW-32.
IM1B	1.0E-3	Excessive feedwater flow through MFW control valve FDW-41.
IM2A	8.6E-3	AOV FDW-35 fails to close sufficiently due to local faults downstream of high-level limiter.
IM2B	8.6E-3	AOV FDW-44 fails to close sufficiently due to local faults downstream of high-level limiter.
IM15	1.4E-3	MFW pump A speed fails high due to local control faults.
IM16	1.4E-3	MFW pump B speed fails high due to local control faults.
IM41	2.0E-3	All four turbine-bypass valves fail to reclose after reactor trip due to common control faults.
LPSUMPMF	6.0E-4	Sump fails as suction source to pumps for recirculation mode; actual critical effect is flooding of HPI pump room via high activity waste tank.
MFM12	9.0E-3	Local faults in condensate system between hotwell and "F" heater bank.
MFM14	3.7E-3	Failure of hotwell pump train A due to local faults.
MFM17	3.0E-3	Common faults causing failure of CBPs (condensate booster pumps).
MFESU44H2	5.0E-3	Operator opens AOV FDW-44 to much via H/A stations.
MFSNGLH	1.0	Operators go to single train of MFW/CBP/HWP operation after reactor trip.
MFSSH2	0.25	Operator assumes manual control of AOV FDW-44 due to (minor) control system malfunctions.

Table 2.4 (Continued)

Event	Mean Unavailability	Description
MSRVC	3.0E-3	Two or more main steam relief vales (MSRVs) fail to reclose after opening.
RC4MVOCM	3.3E-2	RC-4 block valve was closed prior to demand.
RCSRVLC	0.1	Either of two spring-loaded pressurizer spray valves fails to close after liquid relief.
REFDW1	0.5	Failure of the operator to recover FW in 30 minutes; one source available for recovery.
REFDW2	0.3	Failure of the operator to recover FW in 30 minutes; two sources available for recovery.
REFDW23	0.2	Failure of the operator to recover FW in 30 minutes; in case where EFM17 is in the cutset.
RESUMPMF	0.1	Failure of operator to find and isolate leakage from sump via LWD99 and 103 before HPI pump motors flooded.
SWEFCCH	2.0E-4	MVs LPSW-513 and -518 are inadvertently left overthrottled.
SWEFPPAMF	1.1E-2	Cooling flow through EFW pump A fails.
SWEFPPBMF	1.1E-2	Cooling flow through EFW pump B fails.
SW137MVF	6.6E-3	MOV LPSW-137 fails to open.
UTHPIH	1.0E-2	Operators fail to initiate HPI cooling following loss of all feedwater.
XHPLPR12H	3.0E-4	Operators fail to attempt high- or low-pressure recirculation within 12 hours following a small-break LOCA.
XHPR2H	3.0E-3	Operators fail to attempt high-pressure recirculation within two hours after small-break LOCA.
YRBSH	0.5	Operators fail to terminate RB spray operation provided the RB cooling system is operating.

3. REVIEW OF BWOG PHASE I AND II SUBMITTALS

This section presents the results of the BNL review of the submittals made to the NRC by the B&W Owners Group on the risk profile of B&W plants and their relationship to historically observed operational events. The review has three parts. The first part (Section 3.1) discusses the BWOG submittals without comments, and represents a brief summary of the BWOG position. The second part (Section 3.2) is BNL's detailed technical critique of those submittals. The third part (Section 3.3) contains BNL's overall assessment of the submittals and the significance of the deficiencies.

3.1 Summary of the BWOG Submittals

The B&W Owners Group (BWOG) sponsored a two-phase program to assess the risk associated with the B&W reactor design.[3-1, 3-2]

A major question addressed by the BWOG is whether the so-called "Category C" events are risk-significant. The definition of "Category C" is taken from BAW-1919[3-3] and follows below.

Transients are classified in BAW-1919 (Chapter VI) according to a scheme which describes plant's response. As noted in BAW-1919, this is not a method of judging the safety significance of any transient. To quote: "Category "C" transients are those transients where system conditions reach limits which require safety system and operator response to mitigate. This transient is considered abnormal." More specifically, Category C events involve one or more of the following (adapted from BAW-1919, Chapter VI):

Parameter	Event
Reactivity	Recriticality.
RCS Pressure	Pressure excursion causes either SI actuation opening of the PORV or safety valve.
RCS Temperature	Exceeds PTS limit or results in loss of subcooled margin.
RCS Inventory	Pressurizer level goes offscale low with loss of subcooled margin, or offscale high with lifting of the PORV or safety valve.
OTSG Pressure	Pressure excursion exceeds ASME code limit or results in OTSG isolation.
OTSG Inventory	Loss of all feedwater to both OTSGs, or a fill of either or both to over 95% on operate range.

The scope of the Phase I effort, as stated in the report,[3-1] was:

I. Assess the importance of B&W historical Category C events to core-melt risk. Category C transients are those wherein system conditions reach limits which require significant safety system and operator response to mitigate.

II. Compare initiating event frequencies obtained from the transient history of all B&W units to the frequencies used in the Probabilistic Risk Assessments (PRAs).

III. Evaluate the dominant accident sequences, systems, and initiators, from the PRAs. Compare these to the Category C events.

The scope of the Phase II effort, as stated in the report,[3-2]

IV. Generalize the results of the above analyses to all of the B&W units.

The objectives to be met by this body of work were as follows. First, to determine if the number of Category C events experienced at the B&W plants was indicative of higher risk than previously predicted, and second, to establish if certain changes/upgrades of design may be of significant value in decreasing the risk of plant transients.

3.1.1 BWOG Risk Analysis Approach (Phases I and II)

This section discusses the approach used by the BWOG to meet the goals for assessing the risk significance of B&W plant transient history (Category C events in particular). The methods were developed and applied in two phases. The first dealt primarily with developing the approach and applying it to two plants with existing PRAs (Oconee and Crystal River). The second phase expanded the first to all operating B&W plants.

(a) Phase I Approach

In Phase I, the 13 historical Category C events were reviewed to determine what type of events they represented. Two general types of events had occurred, undercooling (loss of secondary heat removal or LOCA without injection) and overcooling (excessive secondary heat removal). Undercooling, if not corrected, will lead directly to core melt. Overcooling can lead to a LOCA if the HPI system is actuated and not controlled before repressurization and lifting of the primary safety/relief valves. All of the Category C events fit into one of these general types; however, they may be caused by different events. The purpose of this program is to identify what caused the Category C events, what subsequent plant responses occurred, and how these events might have progressed to core damage. This defines the "event envelope" to be considered to determine whether these events are significant in terms of core damage, both absolutely and relative to other contributors to core damage. This requirement led to the creation of four classes of Category C events. The following definitions are taken from the BWOG report.

<u>Excessive Feedwater</u> - This class includes all events where overcooling occurred through the steam generators due to excessive emergency feedwater, resulting in shrinkage of the primary system and depressurization. HPI was manually or automatically initiated (for all but Crystal River-3 (CR-3), 10/19/85). In one case, reheating and swelling of the primary system resulted in the opening of a PORV. The PORV subsequently reclosed and the transient was terminated. All of these events had the potential for opening the PORV and SRVs if the HPI was not throttled or reheat controlled. A small LOCA could have then resulted if the relief valves had failed to reclose. A final failure or operator error that terminated HPI during the LOCA would have to take place in order for core damage to occur.

<u>Loss of ICS Power</u> - These transients result in many spurious actuations and losses of control signals required to maintain the plant in a stable condition. The failure ultimately resulted in overcooling situations due to loss of control of steam generator cooling. Automatic HPI injection

signals were generated, but HPI flow was controlled to prevent the PORVs and SRVs from being forced open. The overcooling transients were ultimately terminated without inducing a LOCA condition. These events, with additional failures, could have resulted in stuck open safety relief valves or PORVs.

Blowdown of Secondary System - This class of events induces overcooling through depressurization of the secondary system. As in the previous categories, overcooling can cause a shrinkage of the primary system and manual or automatic actuation of the HPI system. In the cases discussed here, only a single relief valve has stuck open. The Oconee and CR-3 PRAs model more severe overcooling events, involving multiple valves and rapid overcooling to the HPI automatic initiation setpoints. Once HPI is actuated, primary system reheating could force open the PORV or SRVs with the inventory added to the system. If this were to happen, a LOCA would occur if the relief valves fail to reclose.

Loss of Main Feedwater - This last class of events can lead to core damage through two scenarios. If emergency feedwater is lost for a short time (approximately four minutes), the steam generator can dry out, causing the primary system to heat up and expand, relieving steam through the PORV. If feedwater is not recovered prior to 12 minutes, the SRV will open, relieving subcooled liquid. If feedwater is not recovered in approximately 40 minutes, core damage can occur if HPI cooling cannot be utilized. A second path to core damage can result if the primary relief valves fail to reclose and HPI is not actuated and maintained. All experienced Category C events fall into the former scenario; i.e., feedwater was lost and recovered prior to the requirement for HPI cooling. Of these events, only one resulted in core damage due to failure of the PORV to reclose and subsequent operator error in recognizing the LOCA (causing the PORV block valve to remain open) and terminating HPI flow.

It was decided that a risk model which treated these four classes of events and the potential core-damage responses discussed therein would adequately assess the significance of Category C events to plant risk. Part of that process included examining the Oconee and Crystal River PRAs[3-4,3-5] to determine to what extent these events were already included in the PRAs.

The analysis constructed event tree models for two initiating events: reactor/turbine trip and loss of main feedwater. These trees were specifically designed to consider the event types and scenarios discussed above. The trees were quantified by developing branch point probabilities from the system models in the PRAs. For some models, the fault trees were modified to meet the specific needs of the study.

The final sequence results from this analysis were compared with the results of the PRAs to see whether any new insights were gained which would (1) change the concept of what dominated plant risk, or (2) indicate whether these Category C events were significant contributors to core damage compared with the other contributors identified by the PRAs.

(b) Phase II Approach

In Phase II the analysis was extended to include the six operating B&W plants. First, it was necessary to identify plant-specific differences and develop plant-specific models from the Phase I event trees and system unavailabilities (top event failure probabilities). This task was accomplished partly by using information from other plant risk and reliability studies in addition to the Oconee and Crystal River studies used in Phase I, including the Arkansas Nuclear One - Unit 1 IREP study,[3-6] a draft PRA for Three Mile Island - Unit 1,[3-7] and the Davis-Besse EFW reliability study.[3-8] If plant-specific unavailabilities could not be obtained, the top events were quantified by using the studies mentioned above, and identifying which study analyzed a system similar to the one for which no data were available.

A major task in this phase entailed identifying the plant-specific differences which were thought to be relevant to the analysis. The report identified five systems/functions of interest:

- Main feedwater;
- Emergency feedwater;
- High-pressure injection, particularly as it is configured in providing core cooling following loss of all feedwater;
- Pressure relief for the reactor coolant system; and
- Pressure relief and heat rejection for the steam generators.

A set of event trees then was constructed, using the Phase I event trees as a guide. The analysts decided that four initiating events needed to be evaluated; reactor/turbine trip, loss of main feedwater, excessive main feedwater, and loss of power to the ICS. The plant-specific differences were taken into account and the failure probabilities for top events were assigned using the studies mentioned above. In some cases a failure probability was assigned for systems at certain plants based on the analysis of similarly configured systems at other plants. In addition, the frequencies of plant-specific initiating events were developed from B&W Document 51-1164148-00.[3-9] These values for top-event failure probability and initiating event frequency were used to quantify the four event trees for each plant on a plant-specific basis. Since the lack of detailed analysis of systems at certain plants put a greater level of uncertainty into the analysis, sensitivity studies were applied to the potentially important events which were believed to be the least well supported.

The results of the Phase I and Phase II reports are discussed in the following section.

3.1.2 BWOG Results

(a) Phase I Report

The BWOG Phase I report reviewed the PRA models for Oconee and CR-3 in the light of operations experience, and concluded that the initiating events and Category C sequences which occurred at the B&W plants are adequately portrayed by these PRAs. The dominant core-damage sequences appearing in the PRAs are mainly failures that incapacitate both the feedwater and HPI systems, or disable the HPI system for about 12 hours, sufficiently long to prevent core cooling due to loss of the primary system inventory through normal RCP (reactor coolant pump) seal leakage. In other words, these PRA results suggest that core-damage sequences are significantly more likely to develop from events that compromise primary system integrity or cooling while concurrently affecting the operability of the HPI system, rather than from events that do not impair operation of HPI. Thus, support systems such as component cooling water or electric power were very crucial from the viewpoint of core-damage risk. The Phase I report points out that, in order for core damage to occur, one of the following two scenarios must take place:

1. Loss of all feedwater for more than 30 minutes and failure of HPI cooling (feed-and-bleed).

2. A SRV LOCA (from either overcooling or undercooling) caused by a transient and failure of HPI system to maintain RCS integrity.

The importance of the HPI system was strongly emphasized. Failure of the HPI system at TMI-2 was ascribed to operator's error, which is said to be unlikely to recur at Oconee or CR-3 due to the operator's training and to the implementation of new operating procedures. The Phase I report also concluded that the frequencies of initiating event used in the Oconee and CR-3 PRAs are at worst representative and at best conservative.

To illustrate the systems and sequences involved in Category C events, two summary event trees were constructed, one for reactor/turbine trip, and the other for loss of MFW to highlight the scenarios that involve overcooling. Since the event tree developed for reactor/turbine trip to Category C events is generally applicable, the structures of these two event trees are completely identical, with the exception of the initiating event. As will be discussed shortly, simplified versions of these trees were used in the Phase II report to analyze all of the Category C events, including those initiated by excessive feedwater and loss of ICS bus. The development of these event trees in the Phase I report, thus paved the way for their analyses, performed in the Phase II report. These event trees, were quantified using the frequencies or initiating events listed in the Oconee and CR-3 PRAs, and the branch-point probabilities evaluated based on these PRA models. Descriptions are given in Appendix A of the Phase I report on the quantification of branch-point probabilities. No attempt was made to estimate the core-damage sequence frequencies of Category C events initiated by excessive feedwater or loss of an ICS bus, presumably because of their small frequencies of occurrence. The following estimates of core-damage frequency were made by quantifying the event trees:

	Oconee	CR-3
Reactor/Turbine Trip	1.9E-7	3.0E-7
Loss of MFW	4.6E-7*	8.6E-7

*The number shown in the Phase I report is 2.9E-7, which is in error because some of the CR-3 input data (including the initiating event frequency) were mistakenly used in quantifying the Oconee event tree.

In the Oconee and CR-3 PRAs, the total core-melt frequency due to internal events were estimated to be 5.4×10^{-5} (per reactor year) and 5.8×10^{-5} (per reactor year), respectively, for Oconee and CR-3. Since the results shown above represent only a small fraction of these total frequencies, the Phase I report concluded that Category C sequences are not significant to core-damage risk at Oconee and CR-3.

(b) Phase II Report

Phase II of the risk-assessment review generalizes the results of the Phase I analyses to all B&W plants. An evaluation of each of the plants was performed within the context of the event trees developed in the Phase I report, although the event trees used in the Phase II analyses are considerably simpler (particularly those related to excessive FW and loss of ICS bus initiating events). Quantification of the branch-point probabilities, described in Appendix A of that report, relies primarily on information obtained from plant-specific risk and reliability analyses. Where such information was unavailable, the quantification was based on comparative studies of similar systems at other plants. The report warns that conclusions drawn from results for those plants, for which no specific analysis was performed, must be dealt with carefully. Sensitivity studies were carried out to assess the impact of these uncertainties, by varying the unavailabilities of HPI cooling or EFW, and the results are presented in Table 3.1. By using plant-specific initiating event frequencies, the event trees corresponding to each initiator were quantified separately for each plant. The report concluded:

1. Category C events are not likely to be significant contributors to the frequency of core damage for Oconee, CR-3, ANO-1, or Rancho Seco, particularly when compared to other potential contributors, such as station blackout, loss of service water or external events. For Davis-Besse and TMI-1, however, Category C events (particularly loss of MFW) may have some impact on the overall risk of core damage.

2. The large number of diverse pump trains available for core cooling (MFW, EFW, and HPI) significantly enhance the reliability of core cooling if there is no common-cause vulnerability.

3. The potential of inducing a stuck-open-SRV LOCA through actuation of HPI was an insignificant contributor.

4. The TMI-1 results were mainly driven by the relatively high unavailabilities estimated for feed-and-bleed cooling and for the EFW system. Similarly, the Davis-Besse results were ascribed to the high probability of failure assigned to feed-and-bleed cooling.

The following shortcomings and limitations on Phase II analyses were pointed out:

1. No detailed assessment of operator's action to recover lost safety functions was performed, and few such actions were included.

2. The analyses addressed only sequences similar to Category C events that already have occurred at B&W plants. Therefore, the results do not reflect the contributions of other important initiators, such as the loss of offsite power, loss of service water, or external events.

3. Some common cause failures, such as that of support systems (service water or electric power) were not included in the analyses.

For Oconee and CR-3 some discrepancies can be found between the frequency estimates for core damage shown in Table 3.1 (for reactor/turbine trip and loss of MFW) and those of the Phase I analyses. These differences can be attributed to the use of more recent data in the Phase II analyses on both the frequencies of plant-specific initiating events and some of the branch-point probabilities.

The results shown in Table 3.1 suggest that, for Oconee, CR-3, ANO-1, and Rancho Seco, the contributions by the Category C events amount to no more than several percent of the total core melt frequency due to internal events. Although there is no information on the total frequency of core melt for Davis-Besse or TMI-1 due to incomplete PRAs, the BWOG report concluded that the Category C events could be significant contributors to the total core-melt frequency at these plants.

3.1.3 Submittals' Conclusions Regarding Importance of Category C Events

From the results of the CR-3 and Oconee PRAs, the Phase I report asserts that core-damage events are significantly more likely to occur from events that compromise RCS integrity or core heat removal, and simultaneously affect the operability of the HPI system, than from events that do not impair operation of HPI. This is consistent with the findings made in the importance analysis performed for CR-3, which are summarized in Figure 4-1 of the Phase I report.[3-1] The importance analysis indicates that the systems of high importance are the support systems (service water and ac power), followed by the HPI and LPI systems. The power conversion system (PCS), including MFW and ICS, and EFW systems were of much lower importance. This distinction is chiefly due to the fact that the HPI system, with its relatively high reliability, is capable of mitigating loss of all MFW and EFW and that the service water and ac power support both HPI and the feedwater systems. The Category C events, which are still multiple failures away from core damage, were thus concluded to be not significant to core-melt risk at CR-3 and Oconee. The Phase II report drew the identical conclusion for most of the B&W plants, based on the generalized analyses performed. The only exceptions are TMI and Davis-Besse, for which the Category C events had some impact on overall risk of core damage. These different results, however, are attributed mainly to differences in the analyses as well as the unavailability estimate for failure of feed-and-bleed cooling.

3.2 BNL Assessment of BWOG Work

This section discusses our review of the BWOG Phases I and II risk assessments. The section is divided into two major parts; the first discusses the review of the BWOG event tree analysis, and the second discusses the BNL assessment of the BWOG input data and event-tree quantification. As stated, the Phase I work was a development of the methodology, and the Phase II work was a customizing of the Phase I work for the six individual B&W plants.

3.2.1 Review of Event Tree Analysis

This section discusses the event tree analysis performed in the B&W Owners Group Phase I and II risk assessments (referred to jointly as the Owners Group report), identifying areas where the analysis may be deficient. It does not assess the significance of these potential deficiencies, which is determined by a comparison of the Owners Group results with the final results of our risk study (see Sections 4 and 5). Some of the potential deficiencies identified did not prove to be deficiencies, from the standpoint of contributions to total frequency of core damage.

The Phase I and Phase II event trees are reviewed as a single unit, even though they appear to be different. In actuality, the two sets of trees are virtually identical in functional representation, but a few event names were changed and there was some simplification. The major difference between the two sets is that the Phase II trees were individually tailored to each plant and each initiator; in general, the Phase II trees are a tailored subset of the Phase I trees. There are two exceptions to this. The first is that Phase II eliminated consideration of the case where all of the primary safety/relief valves (PORVs and SRVs) fail to open. This simplifying assumption is reasonable, and is commonly made in more recent PRAs, based on the fact that rupture of the vessel due to inability to provide adequate primary steam relief was never identified as a contributor to core melt in any of the numerous PRAs for which this sequence was considered. The second exception is that Phase II did not consider minor overcooling events, only those events which lead to overcooling significant enough to result in a safety-injection signal. In Phase I, consideration was given to overcooling which involved a secondary steam leak large enough to result in overcooling (plant parameters leaving the normal post-trip window) but small enough that a safety-injection signal would not occur and the operator would normally start high pressure makeup from the RWST manually. This point is discussed further below.

To measure the validity of the event trees, we attempted to identify on those trees an event sequence which represented each of the Category C events which had occurred. This should always be possible, since these events actually occurred and therefore were credible. This is true, despite the assertion in the Owners Group report that plant modifications had rendered certain of these events no longer credible. What the report actually meant was that the root cause which caused those events no longer existed in the plant in its previous form, so that the particular Category C event could no longer occur as a result of that root cause. However, the particular combination of system successes and failures which were observed in the Category C events are not unique to that root cause, and the same combination could occur due to other root causes. Thus, the events are always "credible," just not necessarily as "likely" as they once were (due to the modifications to the plant). Therefore, the event trees should contain sequences which represent each of the Category C events which has occurred.

We found that the event trees did not contain sequences representing each of the Category C events. In attempting to identify why this was the case, we isolates the potential deficiencies previously mentioned. In addition, we also identified other areas which could have been investigated more fully. All of these areas are discussed below.

(a) Levels of Overcooling

An overcooling event can occur at two levels. A minor overcooling occurs when the plant leaves its normal post-trip parameters but does not reach a safety injection setpoint. Substantial overcooling occurs when the overcooling is so rapid that nothing can be done to prevent a safety injection signal from occurring. In the Phase I report, this difference was addressed only in the case of overcooling due to excessive secondary steam flow. Excessive flow represented by the failure (open) of a single secondary steam valve (SRV, TBV, ADV, etc.) was considered minor overcooling. Failure of a second valve (or equivalent) was substantial overcooling. In all cases, overcooling due to overfeed was considered to be substantial overcooling. In Phase II, this distinction was removed and only substantial overcooling was considered.

The level of overcooling affects the plant's response. If a minor overcooling occurs, the operator has time to start high pressure makeup manually, in a controlled way (using one pump rather than all pumps). If a substantial overcooling occurs, the pressure drop is too rapid to limit in this manner, and a safety-injection signal will result, causing all the pumps to start and a substantial amount of flow to be injected into the RCS. The effects of these two different levels are interesting. With regard to repressurization to the PORV/SRV setpoints, the minor overcooling case (with its smaller pressure drop) will repressurize faster, even though only one high-pressure pump is operating. Thus, for minor overcooling there is less time for the operator to take action to control the repressurization and prevent the valve(s) from opening. On the other hand, with regard to driving the system solid and causing a serious PRS (see next section), the substantial overcooling case will inject much more coolant and this will result in PTS occurring more rapidly. Thus, there is less time for the operator to take action to control HPI flow and prevent PTS during substantial overcooling. The differences in timing will affect the probabilities of human error, and thus it seems premature to us to fail to fully consider the difference between minor and substantial overcooling in the Owners Group report.

The Phase I work did not fully consider minor versus substantial overcooling except for that caused by a steam leak. In reviewing the Category C events, it was noted that there were two levels of overcooling induced by overfeed. In cases where automatic feedwater control failed, minor overcooling resulted if the operator took timely action to terminate the overfeed by establishing manual control over feedwater flow. Substantial overcooling only occurred when manual control was not established. The Owners Group event trees only contained a single event for feedwater control, failure of which resulted in substantial overcooling. They should have considered two events, one for failure of automatic control and one for subsequent failure of manual control, resulting in minor and substantial overcooling, respectively. Further, consideration should have been applied to the main feedwater system, not just emergency feedwater.

(b) Consideration of Pressurized Thermal Shock (PTS)

The consequences of the occurrence of a serious PTS following an overcooling event are not considered in the Owners Group report. By serious PTS, we mean those PTS events which have a high probability of causing rupture of the vessel and core damage. To determine if such events existed and what they might be, we reviewed the results of NUREG/CR-3770,[3-10] which evaluated PTS at B&W plants using Oconee as a model. This report showed that, while several sequences could lead to what are generally referred as PTS conditions, only those which include overcooling (low RCS temperature) and uncontrolled HPI flow resulting in driving the primary system solid (high RCS pressure) were associated with high conditional probabilities of core damage. While the Owners Group report did identify sequences with these characteristics, the only consideration in the analysis was that the primary safety valves would pass water and this would increase the probability that they would stick open.

We believe that the probability that PTS would result in core damage should have been considered for cases where HPI flow (repressurization) is not controlled by the operator, and it drives the system solid. As discussed in the previous section, this is not necessarily the same as failing to control repressurization in time to prevent primary relief valves from lifting. In the case of minor overcooling, the probability that PTS occurs would be related to the amount of time between the initial valve lift and the time at which the single running HPI pump adds sufficient inventory to result in a solid system. For substantial overcooling, the available analyses suggest that the difference between these two times is negligible because of the large initial depressurization and the high rate of HPI flow. Therefore, it is reasonable to assume that the PTS will result in all cases where repressurization is not controlled prior to relief valve lift. We felt that, at a minimum, the analysis should have carefully considered scenarios involving very low RCS temperature with very high RCS pressure.

(c) Consideration of EFW Recovery

Some of the Category C events began as undercooling events in which emergency feedwater was recovered. Subsequent overcooling may or may not have occurred. However, the Owners Group report does not distinguish between events where feedwater initially works properly, or where it fails and is restored. The distinction is an important one since in the case of EFW recovery the primary pressure rise causes the primary relief valve(s) to open whereas when EFW works initially this valve opening will not occur. Thus, the former case provides an opportunity for a LOCA, which the latter does not. In addition, the time spent in attempting to recover EFW affects the time available to establish feed-and-bleed cooling, which will have an effect on the probability of human error. This also was not modelled in the Owners Group report (the handling of human reliability is discussed in more detail later). Thus, we feel that EFW recovery should have been modelled directly on the event trees.

(d) Analysis of Cognitive Human Errors

Cognitive human errors are failures of the operator (the term operator refers to the entire control-room team) to acknowledge control-room indications, properly interpret their meaning in terms of plant conditions, or properly decide what response is necessary. The Owners Group report does not analyze these errors, or the relationships between them and the plant conditions. It is extremely important to do this since such errors were shown to be major contributors to risk of core damage in many PRAs, especially the more recent ones which use the more advanced analyses tools developed in recent years.

A key feature of these errors is that they are time dependent, and in general the less time available, the more likelihood failure. To properly assess the relationship between B&W risk versus the risk from other PWRs these errors must be analyzed in detail, since one of the key features of distinction between the plant types is the difference in time available (less for B&W) for the operator to perform certain essential actions.

Another important reason for evaluating these errors is the dependencies they have with each other and with the specific plant conditions. For example, in cases of undercooling due to loss of all feedwater, the operator has the choice of two actions. These are recovery of EFW, which should be attempted first, and then attempts should be made to establish feed-and-bleed cooling if EFW recovery is unsuccessful. These decisions are part of the same thought-path, and are highly dependent. If the operator fails to recognize the need to recover EFW, the probability of failing to recognize the need for feed-and-bleed is less than if the need for EFW was recognized but could not recover in time. In the latter case, it was recognized that undercooling exists and the operator is proceeding in an approved manner, where as in the former case the situation was missed entirely. Another example is the operator recognizing the need to control repressurization following of an overcooling event. In this case, the operator is in the process of responding to the overcooling itself, trying to locate the cause and terminate it, when it is necessary to diagnose that the continued addition of high pressure makeup will soon result in primary relief valves opening, or PTS. This dependency is referred to as making two diagnoses "closely in time," and the ability to make the second diagnosis is affected by the continuing response to the first. Thus, the probability of failure is greater than if the second diagnosis was required during an otherwise stable plant condition. Finally, there is the dependence of the operator on plant's instrumentation. In conditions when instrumentation failures occur, like the failure of an ICS bus, the operator's potential for confusion is increased as is the probability of failure. This should be included in the assessment of cognitive human errors. The state-of-the-art model for analysis of human reliability (HRA) allow estimates of human error probability (HEP) which account for all of the dependencies discussed above, and therefore, the required cognitive decisions should have been included in the event trees, along with detailed assessments of these HEPs.

<u>Summary of Conclusions on Event Tree Analysis</u>

In general, we believe that the event-tree models used in the Owners Group report should have been much more detailed and that additional thermal/hydraulic analysis is necessary to address certain issues and to properly establish the time-frames in which certain conditions occur. This latter information is required to assess the response times available to the operator, which is essential to estimating the HEPs. The four specific areas of greater concern are:

1. The trees do not explicitly distinguish between immediate system success and temporary system failure.

2. Operator actions for recovery or control are not adequately displayed or evaluated.

3. Category C events are not satisfactorily addressed within the simplified tree structure.

4. Pressurized thermal shock is not addressed.

We feel that the event tree analysis in the Owners Group report is overly simplified to address the significance of Category C events, the risk contribution of similar type events at B&W plants, and the relative risk of B&W plants versus other PWRs.

3.2.2 Review of Data and Modelling

To assess the significance of B&W historical category C events to core-damage risk, two event trees are developed in the B&W Phase I report. These trees elucidate the safety-function failures and sequences involved in the transients started by reactor/turbine trip (T_1) and loss of MFW (T_2) events, respectively. Emphasis is laid upon illustrating the scenarios that involve overcooling (the first three classes of category C events) and the LOCAs induced by overcooling. For the latter, attention is confined to failure of the pressurizer PORV or SRVs to reclose following primary pressure relief necessitated by RCS repressurization due to HPI actuation and reheating of the primary system. The end states of the various sequences represented by the ramifications of the event trees are classified into four categories: core damage (CD), non-core damage (OK), minor overcooling (OC1), and severe overcooling (OC2). The branch-point split fractions of the top events in event trees are evaluated as probabilities of function failures, based on models from the Oconee PRA (OPRA) and CR-3 PRA. The frequency of each core-damage sequence is quantified by multiplying the relevant branch-point split fractions on the event tree with the corresponding initiator frequency. Only those sequences classified as CD (core damage) are quantified to obtain the core damage frequencies for T_1 and T_2 events. The event trees do not specifically consider the core damage which may result from PTS (pressurized thermal shock) induced by severe overcooling and the subsequent RCS repressurization due to automatic HPI actuation.

In the B&W Phase II report, which essentially generalizes the results of the Phase I analyses to all of the B&W units, the core damage frequencies for T_1 and T_2 events are quantified, based on a slightly simplified version of the event trees. Additionally, two more simplified event trees are used to compute the core-damage frequencies associated with other category C transients initiated by T_3 (Excessive Feedwater) and T_4 (Loss of ICS Bus) events. The approaches taken in quantifying the top events and sequence frequencies of core damage for Phase I and Phase II are practically identical. This section outlines the BNL reviews on the initiator frequencies and the branch-point split fractions of the event trees used in quantifying the results shown in the Phase I and the Phase II reports, by making proper reference to those presented in the Oconee PRA (OPRA) and the Crystal River 3 (CR-3) PRA.

1. Review of the Initiating Event Frequencies

The data base used in obtaining the plant-specific frequencies of initiating event for the Phase I and the Phase II tasks is completely identical; it is contained in B&W Document 51-1164148-00.[3-9] The data base encompasses approximately 42.2 reactor years of operation. The initiating events for the event tree quantifications include reactor trip/turbine trip (T_1), loss of main feedwater (T_2), excessive feedwater (T_3), and loss of ICS power (T_4). The number of occurrences of these events at each of the B&W plants during the period surveyed are divided by the corresponding reactor operating years to obtain the initiating event frequencies (tabulated in Table 4-1 of the Phase II report). Since the data base for TMI-1 is too small to yield meaningful results, its values are taken from the TMI PRA. For the same reason, average values are assigned to all of the plants for T_3 events. It is also pointed out that, as a result of certain design changes, several of the historical events which make up the T_4 data base will no longer cause a loss of ICS bus.

In Table 3-3 of the Phase I report, both the generic and plant-specific frequencies of initiating events for various transients are listed for Oconee and CR-3. For convenience, the frequencies of initiating events from the Phase I and Phase II reports for Oconee and CR-3 are compared in Table 3.2.

The table shows some discrepancies in the plant-specific frequencies listed in the two reports. A closer examination revealed that the plant-specific frequencies tabulated in Table 3-3 of the Phase I report in reality correspond to those presented in the Oconee PRA (OPRA) and the CR-3 PRA, respectively. They also are the frequencies employed in quantifying the T_1 and T_2 event trees developed in the Phase I report. The initiating-event frequencies given in the Oconee PRA were evaluated by applying a two-stage Bayesian update to the plant-population data (for Oconee 1, 2, and 3) recorded from their effective service date (1973 for Unit 1 and 1974 for Units 2 and 3) through to the end of March 1980. The plant population data used in the Bayesian analysis did not include more recent data from 1981-1985. Also, in quantifying the loss of feedwater (T_2) event tree for Oconee illustrated in Figure 4-5 of the Phase I report, the CR-3 initiating event frequency (2.1/ry) was mistakenly used instead of the correct Oconee value (0.64/ry). A more detailed discussion on this matter will be given later. For the Phase II task, the event trees for the four different category C event initiators were quantified using the individual plant-specific frequencies listed in Table 4-1 of that report.

2. Review of Event Tree Quantifications

Prior to quantifying the sequence frequency of core damage associated with each of the transient scenarios depicted by the event trees, it is essential to estimate the branch-point split fractions for the top events on the event tree. To accomplish this, the relevant fault trees from the Oconee PRA or the CR-3 PRA were modified so that they properly reflect the supporting logics of the top events, as well as the specific requirements for the analysis. Such modifications were needed because the event trees developed in the B&W reports are structurally different from those presented in the Oconee PRA or the CR-3 PRA. As mentioned previously, the former was created specifically to illustrate the transient scenarios of the category C events, such as overcooling. Consequently, their top events are closely connected with failures of safety function which will lead to overcooling, undercooling or other related core-damage states. By contrast, the event tree for transient-initiating events in the Oconee PRA has a more generalized structure, which is applicable to more than 14 different transients including the four category C event initiators. Its top events, thus, denote failures of safety function in a much broader sense, such as loss of RCS integrity (in general), or failure of RCS heat removal via the steam generators. One salient feature of the Oconee PRA methodology is that it adopts relatively simple event trees, the top events of which are logically augmented by large and elaborate fault trees. Since the core-damage frequencies in the Oconee PRA are quantified by solving the intricate fault trees based on bin-level Boolean expressions, it is virtually impossible to find directly from the Oconee PRA the branch-point split fractions suitable to quantify the B&W event trees. Proper selection as well as certain modification, therefore, were necessary to use the fault trees in the Oconee PRA or in the CR-3 PRA. These were essentially done in the B&W reports, and the resulting fault trees were solved to find the minimal cutsets for the top events. The branch-point split fraction was obtained by summing up the probabilities of all the minimal cutsets produced by solving the relevant fault trees. The list of dominant minimal cutsets and their probabilities are summarized in the Appendix A of the B&W Phase I report, for each of the event tree top events. We did not make independent SETS code calculations to check the correctness of the branch-point probabilities shown in the B&W report. However, we carefully assessed their adequacy and checked the probabilities assigned to each of the basic events, by reference to those presented in the Oconee PRA. Some of our findings are briefly discussed below.

(i) In evaluating the branch-point split fraction of the event trees for event S1 (see p.A-8 of Phase I report), a probability of 5.5E-2 was assigned to the basic event, MSRVC1 (any of the Main Steam Safety Relief Valve fails to reclose), for both CR-3 and Oconee. According to the Oconee PRA (OPRA), however, the probability of this basic event is 4.0E-2. Also, the probability for IM41 is taken to be 3.6E-3, while the correct value is 2.0E-3 (see p.A-9). With these corrections, the branch-point split fraction for S1 became 5.9E-2 (as compared to 7.1E-2 shown). These changes also will alter the conditional probability for S2 evaluated on p.A-9, although they have negligible impact on the computed frequency of core damage.

(ii) In reviewing the event tree quantification illustrated in Figure 4-5 for Oconee, it was revealed that, instead of using the correct Oconee values, the CR-3 values were mistakenly assigned to the initiating event frequency, as well as to the branch-point probabilities for the top events, E1, S1, S2, U1, P1, and Q1. The correct Oconee values were used for the remainder of the events. We requantified the event tree, employing the correct Oconee values listed in Table 4-1 of the report, and the frequency of core damage for the T_2 events was recalculated as 4.6E-7 (as compared to 2.9E-7 shown on p.4-20 of the report).

(iii) For Oconee, the probability of the basic event, RC4MVOCM (RC-4 block valve was closed prior to demand) was taken to be 3.3E-2, based on the OPRA. As remarked in the OPRA, however, the PORV block valve is actually closed about 75-80% of the time to circumvent PORV leakage. The probability of RC4MVOCM, therefore, should be updated to 0.8, in accordance with their suggestion. Such updating was made in the Phase II report, but not in the Phase I report. This change resulted in an increase of the frequency of core damage for T_1 events from 1.9E-7 to 7.3E-7. A similar increase ensued in the frequency for T_2 events, from 4.6E-7 to 7.3E-7. For both cases, this change also significantly increased the frequencies of the sequences whose end states are classified as OC1 (minor overcooling) and OC2 (severe overcooling). This signifies that, if these overcooling sequences were further pursued for core-damage scenarios, there could be an appreciable increase in the frequency of core damage associated with these sequences.

(iv) The probability of failure of Emergency Feedwater System (event E1) evaluated for Oconee, 6.7E-5, is comparable to those (1.7E-5 for Unit 1, 6.2E-5 for Unit 2, and 1.5E-5 for Unit 3) obtained in a recent analysis[3-11] based on updated information on component failure for Oconee EFW systems.

To sum up, the branch-point probabilities presented in the Phase I report were based on a sound approach. However, since no specific description is given on the fault trees used to generate the minimal cutsets, it is impossible to ascertain whether system dependency (e.g., through support system) is properly taken into account.

For the Phase II task, the branch-point probabilities used to quantify the event trees were estimated primarily from existing reliability analyses and PRAs, such as the Oconee PRA or the CR-3 PRA. If data based on actual operating experience were available, they were chosen to compute the unavailabilities of the event-tree top events, in preference to those obtained from fault-tree quantifications. For those plants which had not been assess for risk, the branch-point probabilities were deduced by comparing the system configurations with those of plants for which there were reliability analyses. To scrutinize the uncertainties introduced by lack of detailed analysis, sensitivity studies were carried out by varying the branch-point probabilities of some of the potentially important failures of safety function denoted by the event tree's top events. The probability values varied, and the results of the sensitivity studies are summarized in Table 4-5 and Figure 4-25 of the Phase II report, respectively.

A concise summary of the event tree branch-point probabilities used in the Phase II task is given in Table 3.3, with emphasis placed upon identifying the sources of the probability data. It can be observed that the probabilities for Event M (loss of MFW following reactor trip) were evaluated from actual data from plant operation, which are available for most plants except TMI-1. For TMI-1, the value reported in TMI-1 PRA was adopted. The sources of the branch-point probabilities for the remainder of the events are briefly reviewed below for each of the plants.

(i) Oconee: With the exception of Event E2, the branch-point probabilities were obtained either from the Oconee PRA or by quantifying the relevant fault trees presented in the Oconee PRA. For Event E2, which is not analyzed in the Oconee PRA, the value was taken from the Oconee PTS, (pressurized thermal shock) report, DPC-RS-1001.

(ii) CR-3: All of the branch-point probabilities were obtained either from the CR-3 PRA or by quantifying the fault trees developed in the CR-3 PRA.

(iii) ANO-1: The probability for the event, E1, was estimated by solving the fault trees developed in ANO-1 IREP study. The value assigned to U1 also is partly based on the results shown in the IREP study. CR-3 values were used for events E2, S, and Q1 (sensitivity study only), while Oconee value was used for Event X.

(iv) Davis-Besse: The failure probability of U1 was independently estimated, based on the success criteria of opening the PORV and successful operation of both charging pumps. The unavailability of EFW is based on the results published in the Davis-Besse EFW Reliability Study. CR-3 values were used for events E2, S, and X.

(v) Rancho-Seco: CR-3 values are used for all of the events, except X, for which Oconee value was used.

(vi) TMI-1: The branch-point probabilities were almost exclusively based on TMI-1 PRA results. For event X, the failure probabilities of both DHR and the high-pressure recirculation system were evaluated from information in the TMI-1 PRA. The conditional failure probability of both DHR and high-pressure recirculation, given the success of HPI, also was calculated using the values from the TMI-1 PRA.

In general, event trees must be quantified with due regard for the interdependence of branch-point probabilities, and for the correlation between initiating event type and branch-point probabilities. There are widely different methods for handling this, much of the effort in a typical PRA is spent in sorting out these relationships. The Owners Group report treats branch point probabilities as if they were largely independent. The report acknowledges this deficiency, but declares that it is not significant to the results. The reasons are not made clear. We can clearly state that the quantification method used by the Owners Group does not in any way account for the effects of failures of support system (other than loss of ICS, which is treated as an initiating event) in causing dependent failures of front-line (mitigating) safety systems. Therefore, individual sequence frequencies may be underestimated as a result.

3.3 Summary and Conclusions

This section summarizes the results of the BNL review of the BWOG submittals and offers our conclusions about them.

3.3.1 Submittals' Achievement of Own Goal

As described in the introductory remarks of the Phase I report, the principal tasks of the Phase I study are (1) assessment of the importance of B&W historical "Category C" events to core-melt risk, (2) comparison of initiating event frequencies obtained from the transient history of all B&W units with those used in the PRAs, and (3) evaluation of the dominant accident sequences, systems, and initiators from the PRAs; comparison of those with category C events.

Of these tasks, the comparison of the initiating event frequencies obtained from all B&W historical Category C events with those used in the PRAs (the goal of Task 2) and the evaluation of dominant accident sequences, systems and initiators from the PRAs (the goal of the first half of Task 3) were satisfactorily attained. The latter goal was achieved by scrutinizing the dominant core-melt sequences from the Oconee PRA and the CR-3 PRA according to their initiating events, and identifying some of the plant's equipment and operational features crucial to core-damage frequency. There is no discussion, however, of the core-melt sequences directly connected to overcooling transients given in the Oconee PRA, possibly because of their relative insignificance.

The assessment of the importance of B&W historical Category C events to core-melt risk (the goal of Task 1) forms the core of all the tasks of the Phase I study. As discussed in Section 3.2.1, however, the event trees developed in the Phase I report, though logically sound, are incomplete. Some of the key factors, particularly those related to operator's action in handling the EFW system or HPI, are not treated in sufficient detail. Also, neither the Phase I study nor the Oconee PRA explicitly treat the core-melt sequences which might result from PTS (pressurized thermal shock) originating from severe overcooling. Further, most of the event-tree top events in the Phase I reports are implicitly embedded in the portion of the Oconee PRA fault trees dealing with overcooling scenarios. Since the event tree's branch-point probabilities computed in the Phase I report were based not only on the fault trees but also on the basic events data developed in the Oconee PRA, it is not surprising that the Phase I study did not uncover anything significantly different from what was already found in the Oconee PRA. The conclusion drawn in the Phase I report, that the category C sequences are not significant contributors to core-melt risk at CR-3 and Oconee, is somewhat premature. Whether the goal of Task 1 was met by the Phase I study, therefore, is questionable. For the Phase II study, which was to generalize the results of Phase I study to all of the B&W units, a similar comment can be made, since the risk analyses performed in the Phase II report were based on further simplified event trees.

3.3.2 Key Points Identified in the Review

1. The frequencies of transient initiating events are relatively well established.

2. The frequencies of Category C events themselves are not directly addressed in the submittal. In one area, it nearly did so: plant states OC1 and OC2 are defined on the event trees. However, other intermediate plant states also are worth defining and quantifying.

3. Event tree branch-points are quantified as if they were independent of each other and of the initiating event. The submittal's authors are aware of what is wrong with this, and argue that it does not matter in this application.

4. According to the submittal, to damage the core, it is necessary to have a relief valve or seal LOCA without makeup or lose all heat removal. Other overcooling scenarios, such as PTS, are absent from the model. This is of central importance to the conclusions.

5. The simplified event-tree structures employed in the submittal can be argued to be formally adequate for discussing core-damage scenarios, but their simplicity obscures questions (e.g., of the relationships between human errors) which could affect the quantification of core-damage frequency and certainly affect discussion of Category C events.

6. Many of the plant-specific branch-point split fractions are obtained in an arbitrary way. The submittal is candid about this, but it seriously undermines the plant comparisons. The submittal is candid about this, too. There are also a number of plain errors in the submittal.

3.3.3 Significance of Issues Raised in This Review

This review identified areas in the Owners Group report which may have been improperly modelled. Many of these will be evaluated quantitatively in Section 4; however, it is not possible to address every area, nor to address them independently. Therefore, this section presents a qualitative discussion of whether each of the items is likely to have a significant effect on the results. This assessment of each area discussed in the review is summarized below.

1) A more rigorous treatment of system interdependencies could have a significant effect, but (as suggested in the submittal) may not in this case. One key dependency with the ICS is handled directly as an initiator, and appears to be properly handled. PRAs have shown other support systems to be significant contributors to core damage; but, for the types of events being examined for this evaluation, they are not likely to have much effect. One exception may be loss of offsite power and the emergency power dependency, where further investigation may show some significance.

2) The identification of minor overcooling versus substantial overcooling would serve primarily an informational purpose. Although it effects plant response, especially in the area of PTS, it is unlikely that specifically including the minor overcooling events (which are less severe) would significantly change the overall frequency of core damage as a result of overcooling.

3) Proper consideration of recovery of emergency feedwater (EFW) has the potential to significantly alter the results. For example, in the Davis-Besse event of 6/9/85, EFW was recovered long into the event. In a probabilistic sense, the plant was much closer to core damage just before EFW recovery (when recovery was still uncertain) than it was just after EFW recovery. However, in the Owners Group report, this event was treated as if EFW had succeeded initially; thus, the treatment masks the fact that the plant was "within" EFW recovery failure and feed-and-bleed failure of core damage. Had the operator failed to recover EFW in the next few minutes, there would have been a relatively short time in which to decide to start bleed-and-feed

cooling. This type of sequence is potentially very important, and was not properly treated.

4) Consideration of PTS could significantly effect the results. This scenario is one of the more obvious potential end-states for serious overcooling events. Failure to consider it could mask the differences between B&W and other PWRs with regard to the frequency and severity of overcooling.

5) A more detailed assessment of cognitive human errors could have a significant effect on the results. The complex actions required in some cases, combined with instrumentation failure due to loss of ICS and/or related busses, make response to certain Category "C" events quite difficult and prone to failure. In some of the observed Category "C" events, multiple errors were made by the operator and additional potential errors, which could have led to core melt, were certainly credible. These were not adequately treated in the Owners Group report.

6) Many errors were found in the numerical treatments given in the submittals. These errors detract from the reports, but they are secondary to the more fundamental issues raised here.

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Table 3.1
Estimates of Core Damage Frequency Attributable to Category C Events by B&WOG

Results Core Damage Frequency Per Reactor Year	Oconee	CR-3	ANO-1	Davis-Besse	Rancho Seco	TMI-1
Reactor/ Turbine Trip	3.7×10^{-7}	3.3×10^{-7}	7.6×10^{-7}	1.4×10^{-6}	3.7×10^{-8}	2.3×10^{-6}
Loss of Main Feedwater	5.7×10^{-7}	9.0×10^{-7}	3.0×10^{-6}	8.6×10^{-6}	2.4×10^{-7}	7.4×10^{-6}
Excessive Feedwater	2.6×10^{-7}	9.2×10^{-8}	3.2×10^{-7}	1.0×10^{-6}	2.6×10^{-8}	3.9×10^{-6}
Loss of ICS Bus	5.5×10^{-7}	1.6×10^{-6}	2.7×10^{-6}	8.6×10^{-6}	1.4×10^{-7}	1.8×10^{-6}
Totals	1.8×10^{-6}	2.9×10^{-6}	6.8×10^{-6}	2.0×10^{-5}	5.8×10^{-7}	1.5×10^{-5}

Table 3.2
Comparison of Initiating Event Frequencies for Oconee and CR-3

Initiating Event	Table 3-3 of Phase I Report Frequency (/ry) Generic	Table 3-3 of Phase I Report Frequency (/ry) Plant-Specific	Table 4-1 of Phase II Report Frequency (/ry) Plant-Specific
(a) Oconee			
T_1	2.9	4.9	2.9
T_2	1.3	0.64	0.72
T_3	0.12	0.092	0.12
T_4	0.024	0.02	0.67
(b) CR-3			
T_1	2.9	5.1	3.7
T_2	2.1	2.1	2.3
T_3	0.12	0.13	0.12
T_4	-	-	1.3

Table 3.3
Summary of Event Tree Branch-Point Probabilities
Used in the Phase II Report and Their Sources

Event	Oconee	CR-3	ANO-1	Davis-Besse	Rancho-Seco	TMI-1**
M	3.8E-2 Plant Operation Data	0.14 Plant Operation Data	0.27 Plant Operation Data	3.8E-2 Plant Operation Data	0.1 Plant Operation Data	2.4E-2 TMI-1 PRA
E1	6.7E-5 Calculation using Oconee PRA Fault Trees	2.2E-4 Calculation using CR-3 Fault Trees	4.9E-4 Calculation using ANO-1 IREP Fault Trees	8.5E-5 3-Train System in IMPELL Report*	2.2E-4 CR-3 Value Used	9.4E-4 TMI-1 PRA
E2	8.4E-4 Oconee PTS Report DPC-RS-1001	4.0E-2 CR-3 PRA	4.0E-2 CR-3 Value Used	4.0E-2 CR-3 Value Used	4.0E-2 CR-3 Value Used	4.1E-5 TMI-1 PRA
S	7.1E-2 Oconee PRA	9.5E-2 CR-3 PRA	1.0 for Loss of ICS 9.5E-2 Otherwise CR-3 Value Used	9.5E-2 CR-3 Value Used	9.5E-2 CR-3 Value Used	4.6E-2 TMI-1 PRA
U1	1.0E-2 Oconee PRA	1.2E-4 CR-3 PRA	1.4E-3 IREP Study & Indep. Calculation	0.1 Independent Estimation	1.2E-4 CR-3 Value Used	3.4E-2 TMI-1 PRA
P	0.8 Fraction of Time Block Valve is Closed	6.1E-2 CR-3 PRA	1.0 Fraction of Time Block Valve is Closed	PORV Cannot be Challenged	6.1E-2 CR-3 Value Used	7.5E-2 TMI-1 PRA
Q1	2.6E-4 Oconee PRA	5.4E-5 CR-3 PRA	5.4E-5 Sensitivity Study CR-3 Value Used	PORV Cannot be Challenged	5.4E-5 CR-3 Value Used	1.3E-4 TMI-1 PRA
X	3.6E-4 Calculation using Oconee PRA Fault Trees	1.4E-3 Calculation using CR-3 PRA Fault Trees	3.6E-4 Oconee Value Used	1.4E-3 CR-3 Value Used	3.6E-4 Oconee Value Used	4.3E-4 Calculation using TMI-1 PRA Values

Table 3.3 (Continued)

NOTES:

*0210401333-Davis-Besse Nuclear Power Station Reliability Analysis based on NUREG-0611, IMPELL, November 1, 1982.
**TMI-1 PRA data is from an unpublished draft.

M: Loss of MFW following reactor trip.
E1: Failure of EFW after loss of MFW.
E2: Excessive EFW after loss of MFW.
S: Excessive secondary steam relief.
U1: Failure of feed-and-bleed cooling.
P: Failure of pressure relief via PORV.
Q1: PORV fails to close after pressure relief.
X: Failure of long-term heat removal.

4. REVISED MODEL FOR B&W PLANT TRANSIENTS

To account for the review comments made in Section 3, BNL constructed a revised model to evaluate the significance of Category C events. The revised model was used in two ways. First, a sequence analysis was made, designed to assess the contribution of Category C events to core damage in operating B&W plants (that is, accounting for changes in plant design and excluding the observed frequency of Category C events) (Section 4.1). Second, a precursor analysis using the observed Category C events was performed, to determine how close (in a probabilistic sense) these observed events came to core damage.

4.1 Event Tree Quantification and Sequence Assessment

The accident sequence analysis consisted of constructing and quantifying an event tree representing the B&W plant's response to the types of events seen in the historical Category C events. The process followed, and results obtained in the analysis is described in this section.

4.1.1 Event Tree Development

Early in the study we developed a generic transient event tree which would be all-inclusive of possible plant responses to the transients of interest. Thus, a reasonable level of completeness was assured. For more restrictive cases where certain responses were not possible, the tree needed only to be reduced rather than a new tree constructed (that is, all the trees in the analysis were subsets of the master tree). The development of the event tree was aided by an event sequence diagram (ESD). The ESD laid out the overall logic of the plant's response, and the tree was developed to represent that logic. The most important guideline followed in constructing the ESD was that the logic had to accurately reflect all of the actual Category C events (one of the deficiencies identified in the Owners' Group report). Other areas of potential deficiency in the Owners' Group report discussed in Section 3 were also treated in the ESD construction (Figure 4.1). The ESD structure can be generalized as follows:

- each box represents a required system response, which may or may not include a requirement for operator action,
- the arrows indicate the logical path being followed for each response,
- successful response exits an event box to the right, failed response exits an event box downward (in a few cases upward), and
- the end states are OK (no overcooling), OC1 (minor overcooling), OC2 (severe overcooling), core vulnerable, and core damage.

As the figure shows, the ESD treats two levels of overcooling, minor and severe. Minor overcooling is a level which results in a slow drop in temperature and pressure, giving the operator adequate time to take action to recover RCS pressure before HPI initiation. Severe overcooling is a rapid overcooling, which results in automatic HPI initiation within minutes. The ESD shows two ways to get a minor overcooling. The first is an overfeed of either main or emergency feedwater due to loss of automatic flow control, which is controlled by the operator before it progresses to the point of automatic HPI initiation. The second is the case where there is a small secondary steam leak, resulting in a slow blowdown of a steam generator (e.g., a single secondary valve stuck open). Severe overcooling can occur in three ways. First, an overfeed of either main or emergency feedwater where the operator does not take manual control of feedwater flow (completely uncontrolled feedwater flow). The second is a large secondary steam leak resulting in a rapid blowdown of the steam generator (e.g., two or more secondary steam valves stuck open. The third is a minor overcooling event which the operator ignores for a long time, allowing it to develop into a severe overcooling. In this case, the operator failed to respond to the pressure drop induced by the continued minor overcooling by either initiating sufficient makeup flow (from one HPI pump) or by locating and terminating the cause of the overcooling.

In all cases where HPI is initiated (either manually of automatically) the operator must manually control HPI flow and limit RCS repressurization. Failure to do so can result in lifting primary safety/relief valve(s) and subsequently serious PTS (defined as driving the RCS solid during an overcooling event). Where a single HPI pump was manually started, failure to control HPI in the short term results in lifting a primary PORV or S/RV, and failure in the longer term results in driving the system solid (serious PTS). Where HPI started automatically, repressurization and overfill happen so closely together that no distinction is made, and PTS will always result if HPI is not controlled in the short term. Whenever PTS occurs, the ESD considers whether rupture of the vessel occurs. Whenever a primary valve opens, the ESD considers whether the valve sticks open, causing a LOCA.

In addition to overcooling events, the ESD also treats undercooling events, which occur when both main and emergency feedwater initially fail. The undercooling can be terminated by either restoration of emergency feedwater or by initiating feed-and-bleed cooling. When EFW is recovered, both the potential for a LOCA (due to stuck open primary valves) or for subsequent overcooling (uncontrolled EFW flow after recovery) are considered. The overcooling case is handled in the same manner as the overcooling situations. All this discussion is shown on the ESD.

The ESD was used to construct an event tree from which the top events were developed directly. The definitions of the top events are:

INIT. EVENT — Initiating Event - One of the four initiating events evaluated in the study, (1) turbine/reactor trip, (2) loss of main feedwater, (3) excessive feedwater, and (4) loss of ICS. These initiating events were the same as those in the Owners' Group report.

MFW — Main Feedwater System Continues to Operate - This event represents the continued operation of the main feedwater system in supplying cooling water to the secondary side of the steam generators. By implication, this also includes continued operation of the entire power conversion system (main steam, turbine bypass, condenser, and condensate systems in addition to main feedwater). Success is defined as supplying enough flow to remove decay heat.

EFW — Automatic Actuation of Emergency Feedwater System is Successful - This event represents the automatic actuation and operation of the emergency feedwater system to supply coolant to the secondary side of the steam generators when the MFW system is no longer operating. Success is defined as at least one EFW pump starting and supplying enough cooling water to the steam generators from the condensate storage tank to remove decay heat.

OA-EFW-R Operator Recognizes the Need to Recover Emergency Feedwater - This event represents the recognition by the operator that there is no feedwater flow to the steam generators and that it is necessary to attempt to recover flow from the EFW system. It applies to cases where both MFW and EFW, as defined above, have failed. Success is defined as the operator recognizing within 15 minutes of the loss of MFW and EFW (25 minutes for loss of offsite power) that there is a need to recover emergency feedwater.

EFW-R Emergency Feedwater Recovery is Successful - This event represents the actions and equipment required to implement the decision discussed in OA-EFW-R, when the operator has made that decision successfully. Success is defined as the operator performing the actions required to establish EFW flow to the steam generators from the condensate storage tank using at least one EFW pump within 20 minutes of the loss of MFW and EFW (30 minutes for loss of offsite power). Success includes that the equipment required has failed in a way which is recoverable.

FW-AC Automatic Control of Feedwater Flow (Main or Emergency, as appropriate) is Successful) - This event represents the automatic control of the feedwater flow in cases where feedwater is available, from whichever feedwater system is operating, to limit flow to only that which is required to remove decay heat. Success of the feedwater events discussed above already was defined as supply of at least enough flow to remove decay heat, thus at this point in the tree we are concerned only with preventing overcooling by preventing excess flow. Success is defined as the automatic flow control system limiting feedwater flow, such that decay heat is matched and an overcooling does not occur.

OA-FW-MC Operator Recognizes the Need to Take Manual Control of Feedwater Flow (Main or Emergency, as appropriate) - This event represents the operator recognizing that a rapid overcooling is taking place as a result of excessive feedwater flow and that there is a need to take manual control of feedwater to limit the extent of that overcooling. This applies in cases where FW-AC has failed. Success is defined as the operator recognizing the need to take control of feedwater flow within five minutes of the start of the overcooling.

FW-MC

Manual Control of Feedwater Flow is Successful - This event represents the performance of the action required to implement the decision arrived at in OA-FW-MC. Success is defined as the operator taking control of feedwater flow within five minutes of the start of the rapid overcooling and limiting the extent of the overcooling such that the RCS pressure does not quickly reach the initiation pressure for automatic high-pressure injection thus preventing a severe overcooling. (NOTE: This establishes the definitions of minor and severe overcooling used throughout the analysis. Minor overcooling (OC1) is an overcooling event where the RCS pressure remains above the HPI initiation setpoint. Severe overcooling (OC2) is an overcooling event where the RCS pressure drops below the HPI initiation setpoint.)

SECON. INT.

Secondary is Intact (i.e., no excess steam leakage) - This event represents the proper operation of the secondary system to prevent excess heat removal (overcooling) due to excessive steam glow in cases where overcooling due to excessive feedwater flow has not already occurred. Success is defined as automatic control of secondary steam flow, such that steam flow does not exceed that necessary to remove decay heat.

LEAK SIZE

Size of Secondary Steam Leak (upward branch is "small," downward branch is "large") - This event represents the size of the secondary steam leak in the cases where secondary integrity has failed. A small leak will result in a minor overcooling and is generally considered to result from a steam leak equivalent to a single stuck upon steam dump valve or turbine bypass valve. A large leak will result in a severe overcooling and is considered to result from a steam leak equivalent to a greater number of stuck-open valves. As previously defined, a severe overcooling will result in a rapid RCS pressure drop to the HPI initiation setpoint (within five minutes) whereas a minor overcooling will result in a much slower pressure drop, allowing automatic HPI initiation to be prevented. There is no real "success" definition for this event, however, the upward (or success) branch for this event is defined as a small steam leak (minor overcooling).

PS/RV-C-UC

Primary Safety/Relief Valves Close After Opening Due to Undercooling - This event represents the reclosing of the RCS pressurizer safety/relief valves for the case where they have opened due to an undercooling condition for any length of time (both MFW and EFW fail, regardless of whether EFW is eventually recovered). This will prevent the occurrence of a small LOCA. Success is defined as the reclosing of any and all primary code safety and power operated relief valves (PORV) which opened as a result of the undercooling condition. This would include any action taken by the operator to close the PORV block valve to isolate a stuck open PORV.

OA-HPI-M

Operator Recognizes the Need to Manually Initiate High Pressure Injection Flow (for Total Loss of Feedwater Case This Event Represents Need to Establish Feed-and-Bleed Cooling) - As indicated in the title, this event has two definitions. In those cases where a minor overcooling has occurred (FW-AC fails and FW-MC succeeds or Secon. Int. fails and Leak Size is small), this represents the recognition by the operator that RCS pressure is dropping (due to the overcooling) and that a HPI pump should be started to stop the pressure drop: this prevents the pressure from reaching the HPI initiation setpoint, thus preventing the minor overcooling from becoming a severe overcooling. Success is defined as the operator recognizing this need and making this decision within 60 minutes of the start of a minor overcooling. It should be noted that this option is not allowed in the severe overcooling case since it will not prevent the overcooling from causing the RCS pressure to drop below the HPI initiation setpoint.

For the case where an undercooling has occurred (MFW and EFW fail and either OA-EFW-R or EFW-R fails) this event represents the recognition by the operator that no secondary cooling is available and that a once-through (i.e., feed-and-bleed) cooling should be initiated by opening the PORV and initiating high pressure injection. Success is defined as the operator recognizing this need and making this decision within 40 minutes of the loss of secondary cooling (50 minutes for loss of offsite power). It should be noted that this is 20 minutes past the time by which the operator has to recover EFW.

HPI-M

Manual Initiation of High Pressure Injection is Successful (for Total Loss of Feedwater Case This Event Represents Feed-and-Bleed Cooling) - This event represents the operator action and equipment success required to implement the decisions made for event OA-HPI-M. For the minor overcooling case, success is defined as initiation of a single HPI pump to inject coolant into the RCS, and realignment of that pump to take suction from the borated water storage tank (BWST) within 60 minutes of the start of the overcooling. For the undercooling case, success is defined as initiation of HPI, with operation of at least one pump train injecting coolant into the RCS from the BWST, and locking open the pressurizer PORV within 40 minutes of the loss of secondary cooling (50 minutes for loss of offsite power).

OA-OCT

Operator Recognizes the Need to Terminate the Overcooling Event - This event represents the recognition of the operator that the RCS pressure is dropping due to a minor overcooling event and the need to take action to terminate the minor overcooling in order to prevent it from progressing to a severe overcooling. This is an alternative (or backup) to the initiation of an HPI pump to prevent a minor overcooling from becoming a severe overcooling. It is considered as a backup, because the operators symptomatic response should cause then to think about HPI initiation before to attempting to isolate the cause of the overcooling. Success is defined as the operator recognizing this need and making this decision within 60 minutes of the start of the overcooling. It should be noted that, like the event OA-HPI-M, no credit is given for this action for the severe overcooling case (because insufficient time is available for action).

OCT — Overcooling is Successfully Terminated - This event represents the operator action and equipment successes required to implement the decision associated with OA-OCT. Success depends on the conditions which are causing the overcooling, whether it is a minor steam generator overfeed due to imprecise manual control or a small secondary side steam leak, or some combination of both. The root cause of the overcooling is also important, since this affects the recoverability. In general terms, success is defined as terminating the overcooling within minutes of its start.

HPI-A — Automatic Initiation of High Pressure Injection is Successful - This event represents the automatic initiation of HPI, when the RCS pressure drops below the HPI initiation setpoint. This can result from two basic causes, severe overcooling or small LOCA. Severe overcooling is caused in a number of ways, as follows: (1) failure of MFW-AC and either OA-FW-MC or FW-MC, (2) failure of Secon. Int. and Leak Size large, or (3) any minor overcooling followed by failure of both manual HPI (OA-HPI-M or HPI-M) and overcooling termination (OA-OCT or OCT). Small LOCA is caused by a stuck-open safety/relief valve (failure of PS/RV-C-UC) following an undercooling event (failure of MFW and EFW). For all cases, success is defined as the automatic actuation of at least on HPI pump train taking suction from the BWST and injection coolant into the RCS.

OA-HPI-MC — Operator Recognizes the Need to Take Manual Control of High Pressure Injection Flow Rate - This event represents the operator recognizing that pressurizer level is rising and RCS pressure is increasing due to the effect of high pressure injection flow, and that there is a need to reduce (or terminate) this flow to prevent the RCS from repressurizing to the point where the primary safety/relief valves would open. The definition of success depends on whether a minor overcooling exists (small initial depressurization, one HPI pump running in manual mode, i.e., success of OA-HPI-M and HPI-M) or a severe overcooling exists (large initial depressurization, all HPI pumps running in automatic mode, i.e., success of HPI-A). For the minor overcooling, success is defined as the operator recognizing the conditions and making the correct decision within five minutes of the manual initiation of HPI. For the severe overcooling case, success is defined as the operator recognizing the conditions and making the correct decision within ten minutes of the automatic initiation of HPI. In both cases, success will terminate repressurization before it reaches the point where the safety/relief valves would open.

HPI-MC — Manual Control of High Pressure Injection Flow Rate is Successful - This event represents the operator actions and system successes required to implement the decision associated with OA-HPI-MC. Additional actions may be required of the operator other than just controlling HPI flow, such as varying secondary heat removal rate. This is sequence dependent, but the general definition of success is that the necessary actions are taken to prevent repressurization and primary safety/relief valve opening (including the success of equipment) within five minutes (for minor overcooling) or ten minutes (for severe overcooling).

PTS — Pressurized Thermal Shock Conditions Do Not Occur (i.e., upward branch means no PTS, downward branch means PTS conditions are reached) - This event represents operator recognition of increasing pressurizer level after the opening of the primary safety/relief valves, the realization of the need to terminate HPI flow prior to the occurrence of a severe PTS, and the actions and equipment successes required to implement this decision. The only PTS events with which we are concerned are those where the pressurizer becomes completely filled with water; thus, success of this event simply requires that the action be completed before this would occur. This event is applied only to the minor overcooling case, where the pressurizer is filled slowly because only one HPI pump is running. In this case, the actions must be completed within 15 minutes of the opening of the safety/relief valves (20 minutes following manual initiation of HPI). The nature of the severe overcooling event is such that severe PTS occurs a very short time after the safety/relief valves open, so no credit is given for this action for that case (a detailed discussion is included elsewhere in the report).

PTS-CD — Occurrence of Pressurized Thermal Shock Does Not Result in Core Damage by Vessel Rupture (i.e., upward branch means no core damage, downward branch means vessel rupture and core damage) - This event represents the survival of the reactor vessel following the occurrence of a serious PTS (failure of PTS for the minor overcooling case, failure of OA-HPI-MC or HPI-MC for the severe overcooling case). Success is defined as no rupture of the RCS caused by PTS. Failure of this event is assumed to result in core damage.

PS/RV-C-RP	Primary Safety/Relief Valves Fail to Close After Opening Due to Repressurization - This event represents the failure of the primary code safety valves or power operated relief valve (PORV) to reclose after they have opened due to RCS repressurization following an overcooling (failure to prevent repressurization to the valve setpoint, i.e., failure of OA-HPI-MC or HPI-MC). Success is defined in the same manner as for PS/RV-C-UC, except for the cause of the valve opening. Two cases are quantified: (1) no PTS (success at event PTS), which assumes that only steam is passed through the valves, and (2) PTS without vessel rupture (failure at event PTS but success at event PTS-CD, which assumes that water is passed through the valves.
LTC	Long Term Cooling Following Feed-and-Bleed or LOCA is Successfully Established - This event represents the decision, action, and equipment required to establish long-term cooling whenever either a LOCA has occurred or feed-and-bleed has been implemented. For the LOCA case, success is defined as the successful initiation and operation of at least one train of high-pressure recirculation, taking suction from the containment sump and returning the water to the core, along with successful initiation and operation of at least one train of containment cooling. For the feed-and-bleed case, success is defined as either the successful initiation and operation of one train of residual heat removal (if the PORV can be reclosed or isolated) or else the use of the same method as used for the LOCA case.

The event tree itself is not presented in this report since it is quite large (461 sequences), but it accurately reflects the logic of the ESD. In our opinion, it is not necessary that such a detailed event tree be developed whenever a PRA is done for a B&W plant. Many of the paths shown on the ESD and reflected in the event tree will be of very low frequency, and therefore not significant contributors to overall frequency of core damage. Our detailed study will identify which paths to core damage are significant and which are not, and will thus help to settle the issue of which Category C events are likely to contribute to core damage, and should be considered in future PRAs and which should not. This will be discussed later in this report.

To summarize, by constructing an event tree with a relatively large number of top events, a deeper insight could be gained into the possible impact of certain events, such as those related to cognitive human errors, EFW recovery, or PTS, on both the accident sequence frequencies and on the total frequency of core damage. More specifically, the event tree developed in the

BNL study was structured to include the following features, which were not in the event trees used in the BWOG Phase II analyses.

1. Minor overcoolings which do not cause automatic actuation of HPI are included in the analysis. Furthermore, a clear distinction is drawn between minor overcooling and severe overcooling.

2. The causes of minor or severe overcoolings, such as feedwater runaway (loss of automatic FW control, loss of manual FW control, or both) or secondary blowdown (one stuck-open steam valve versus two or more stuck-open steam valves) are clearly distinguished.

3. Recovery of EFW is modelled as an event-tree top event.

4. The possible impacts of PTS (pressurized thermal shock) are explicitly analyzed.

5. Cognitive human errors, such as those related to EFW recovery, manual control of FW, manual control of HPI flow, initiation of feed-and-bleed cooling, or termination of overcooling, are explicitly treated through the event-tree top events.

6. The event tree is detailed enough to be used in the precursor study of all the major historical Category C events that have occurred at B&W plants.

Furthermore, the relatively detailed structure of the event tree enables a prediction to be made of the frequency of occurrence of Category C events and also to identify and distinguish the risk-significance of various Category C events according to their initiators, transient types and the ultimate causes leading to core damage. With simple event trees, it would be difficult to make such predictions or draw such distinctions.

4.1.2 Data Base

(a) Initiating Event Frequencies

The initiating events considered for the event-tree quantifications are (1) reactor/turbine trip, (2) loss of main feedwater, (3) excessive feedwater, and (4) loss of ICS power. The initiating-event frequencies, which are plant-specific, were taken directly from B&W Owners Group report, Phase II[4-1] as noted in Table 4.1.

(b) Branch-Point Split Fractions

The event tree containing 21 top events was quantified by employing the data on branch-point split fractions summarized in Table 4.2. About one-half of the 21 top events also appear in the event trees developed in the B&W Owners Group analyses. The branch-point split fractions for those top events having this commonality were adopted directly from the Owners Group reports (Phase I and Phase II), if they were judged to be reasonable. In part, this approach was taken to facilitate identification of the characteristic difference between the event tree developed in this study and those presented in the Owners Group reports. The top events belonging to this category include numbers 1, 2, 5, 8, 9, 10, 12, 17, 20, and 21. For those top events closely connected with operator's actions (i.e., 3, 6, 11, 13, 16, and 18), the essential details of the methods used in estimating the branch-point probabilities were discussed elsewhere in this report (Section 4.1.3 on Human Reliability Analysis).

The event tree is structured to explicitly analyze the core-damage sequences associated with pressurized thermal shock (PTS), a phenomenon brought about by stresses caused by the combination of severe overcooling and high pressures. The failure probabilities for the reactor vessel (i.e., top event 19) required to quantify the PTS sequences were computed, following a recommendation made by the NRC staff. They are based on Figure 5.8 of Reference 4-3, a plot of the conditional probability of vessel failure (given the existence of a flaw) versus final coolant temperature. If the final (minimum) coolant temperature that can be attained during an overcooling transient can be predicted, a conditional probability of vessel failure can be obtained from this plot, which is then multiplied by 0.03, the probability for the preexistence of a flaw. The structure of the event tree suggests that different vessel-failure probabilities are required for quantifying the PTS sequences, depending upon whether the overcooling is induced by feedwater runaway, secondary-side blowdown, or the combinations of both. More specifically, the PTS core-damage sequences resulting from overcooling and the subsequent repressurization can be grouped into the following several types (shown in Table 4.3), according to the controllability of feedwater and the intactness of the secondary-side system.

For each type of the overcooling transients shown in Table 4.3, the final (minimum) coolant temperature attainable during the transient was estimated from the results of various thermal hydraulic calculations[4-3] exclusively performed for Oconee 1 to assess the risk associated with PTS. To compute the probability of vessel failure applicable to each of the overcooling types, the cumulative probability of minimum coolant temperature (achievable during such a transient) was assumed to have a S-shaped distribution, as sketched in Figure 4-2. To characterize the range of overcooling, the probability curve was conveniently subdivided into four temperature bins, T_1 through T_4. The fractional probability that the minimum coolant temperature might fall into a particular temperature bin then is assigned to each of the four temperature bins (see Table 4.3), based on the results of thermal hydraulic analyses and other considerations, such as the time available for the operator to recover the feedwater or to isolate the secondary system plagued with stuck-open valves. For instance, it was estimated that 80% of all the overcooling transients of type 4 would end up with minimum coolant temperature in the range represented by bin T_1, and 20% in the temperature range denoted by bin T_2 (Table 4.3).

The calculated probabilities of vessel failure are summarized in Table 4.4 for all the overcooling types.

To arrive at the failure probabilities shown in the second column of Table 4.4, the conditional failure probabilities corresponding to the mean temperatures of the four temperature bins are first obtained separately, using Figure 5.8 of Reference 4-3. A weighted average of these vessel-failure probabilities then are calculated, based on the fractional distribution of the temperature bins shown in Table 4.3. Due to insufficient detail in the event-tree structure, only one input data (for top event 19) is permissible to represent overcooling types 4, 5, and 6. Similarly, only one input value can be assigned to event 19 to encompass overcooling types 7 and 8. For these two cases, therefore, the relevant failure probabilities were combined and reduced to a single value, by applying the weighted-averaging process, taking into account the possibilities for concurrent occurrence of feedwater runaway and secondary-side blowdown. For transients initiated by loss of ICS power, the human error probabilities (i.e., Events 3, 6, 11, 13, and 16) are multiplied by a factor of five. In addition, the probabilities of top events 8 and 9 were taken to be 0.3 and 0.67, respectively, for those plants which do not have MSIVs. The vessel-failure probability (Event 19) also was adjusted to conform with this assumption.

For the remaining top events (i.e., events 4, 7, 14, and 15), the probability (0.2) assigned to event 4 (EFW-R) was developed by examining feedwater recovery analyses from other PRAs. The primary source was the Oconee PRA,[4-4] which developed a series of nonrecovery factors for different situations as follows:

FW within 30 minutes (one source available)	0.5
FW within 30 minutes (two sources available)	0.3
FW within 30 minutes (three sources available)	0.1
FW within 30 minutes (including Safe Shutdown Facility)	0.1
FW within 30 minutes (special failure mode case)	0.3

In addition, the Millstone 3 PRA[4-5] determined a nonrecovery factor for the AFW system (all trains) of 0.13. The Zion PRA[4-6] only gave credit for recovery of the turbine-driven train of AFW, a nonrecovery factor of 0.5. Based on these studies, and our estimate of the recovery actions available to the operators in most cases (including the potential to recover any emergency feedwater train, limited restoration of main feedwater, and depressurization of the steam generators so that the condensate system can be used directly to supply feedwater flow) the 0.2 nonrecovery factor was selected as a reasonable estimate. A sensitivity study was performed on this value (see Section 4.1.5 (a)). The values specified for events 7 and 14, are, at best, rough estimates based on limited data and certain engineering judgement. The unavailability data assigned to event 15 for Oconee and CR-3 were deduced from Oconee PRA fault trees and Owners Group report (Phase I), respectively.

4.1.3 Human Reliability Analysis

This section documents the assumptions and models used in the quantification of human error probabilities (HEPs). The human reliability analysis for this study focuses on cognitive errors, i.e., errors in judgement and decision making, as opposed to manipulative errors, i.e., errors in performing the actions decided upon. These cognitive errors generally were shown to be more significant in terms of risk (in past PRAs) than manipulative errors, so it was decided early in the project that our limited time and resources would be best applied to cognitive errors.

The human errors to be evaluated were identified in the process which developed the event sequence diagram (ESD) and event tree (ET) presented previously. The process used generally followed that described in the SHARP report (EPRI-NP-3583),[4-7] but in a free-form manner. A brief description of that process would is:

The historical Category C events and the event trees developed by the Owner's Group were reviewed and an initial ESD was developed. This ESD was intended to depict the key elements of plant's response to the initiating events of concern. The response was reviewed in a step-by-step manner to identify which key elements required a decision on operator's part in order to be successful. A draft event tree was constructed, which included as top events these operator decisions.

An initial evaluation of the event tree was performed. We determined, using conservative screening estimates, what top events on the event tree (both operator action and system unavailability) were potentially significant in terms of sequence frequency and model accuracy. In addition, refinements in initial assumptions were made to be certain the modelled response was reasonable and as simple as possible. Then a final ESD and ET were constructed.

The remaining cognitive actions of the operator were evaluated in detail. The human cognitive reliability (HCR) model[4-8] was selected to be used for each error, unless the required input information could not be reasonably estimated; then, the time-dependent HEP curve (screening) from NUREG/CR-1278[4-9] was used. Each action was quantified using the appropriate model.

(a) The Human Cognitive Reliability (HCR) Model

The HCR model was developed for EPRI and is fully documented in a draft report by NUS Corporation (NUS-4531).[4-8] Summary information on the approach can be found in various conference papers.[4-10,4-11,4-12] This section will not describe the basis for the model in detail, but will simply present the model and its application for this study.

The equation for determining the cognitive human error probability of an operating crew is:

$$P(t) = \exp - \left[\frac{t/T_{1/2} - C_{\gamma i}}{C_{\eta i}} \right]^{\beta_i}$$

P(t) is the probability that the operators will fail to make the proper decision in the available time "t". This time "t" is not the time from when the sequence begins (plant trip) to when the action must be completed, but is the time from when the symptoms (or compelling signal) of an abnormal condition occurs to the time by which a decision must be made to allow sufficient time for response. Thus, "t" is, in actuality, the total time available from the initiating event minus the time it takes for the symptoms of the abnormal condition to show up minus the expected amount of time for the operating crew to perform the manipulations to implement the cognitive decision.

The parameters $C_{\gamma i}$, $C_{\eta i}$, and β_i are correlation coefficients assigned according to the type of mental processing (cognitive behavior) associated with the action. The three types of behavior are considered; skill, rule, and knowledge. Skill-based behavior occurs when the operator is well trained, is motivated to perform the task, and has experience in performing the task with no ambiguity. Rule-based behavior occurs when the operator has a clearly understood set of rules to follow in responding to a well-understood transient or situation. Knowledge-based behavior occurs when the previous definitions do not apply, or the operator must understand the condition of the plant, interpret some instrument readings, or make a difficult diagnosis. A logic aid for selecting the type of behavior for any action is provided in the HCR report, and is reproduced as Figure 4.3. The parameters for each behavior type is given in Table 4.5.

$T_{1/2}$ is the median response time taken by a crew to make the cognitive decision. That is, it is the time by which one-half of the crews will have taken the appropriate action and one-half will not have. If a sufficient amount of data exists, this value can be determined directly. On the other hand, if there are insufficient data then it is necessary to use simulator data, task analysis, and expert judgement to estimate a median response time under "normal" conditions. This value then is adjusted by the use of performance-shaping factors to account for the difference between normal (or nominal) conditions and the actual conditions which are expected during a real event. Table 4.6 give the three performance-shaping factors and their related coefficient values. $T_{1/2}$ is determined through the following formula:

$$T_{1/2} = T_{1/2}\text{, nominal} \times (1 + K1 + K2 + K3)$$

The criteria for selecting a particular level for each performance-shaping factor are given in Table 4.7.

This model was used to quantify the cognitive human errors used in this study. A summary is presented in the next section.

(b) Cognitive HEP Quantification for Specific Actions

The specific cognitive actions of the operator considered in this study were defined in the section on event sequence diagram and event tree development. The parameters used for the quantification of these actions, along with the final HEPs, are given in Table 4.8.

Each of the actions considered is listed in the first column of the table. The behavior type (second column) was selected based on which definition (presented in the previous section), best fit the action. The next three columns deal with determining the time available for the operator to make the required decision. The column labeled "Total Time Available" gives the total amount of time from the first indication of the specific abnormal condition to the time at which an adverse result will occur if no action is taken. These times were developed in two ways. For all actions other than OA-HPI-M undercooling case and OA-EFW-R, these time were determined through a review of analyses performed to evaluate overcooling events. The sources of information included NUREG/CR-3770,[4-3] NUREG/CR-3706,[4-13] and preliminary results of analyses performed by EG&G Idaho for a NRC project to evaluate B&W plant transients. For the two actions not handled in this manner, the times were taken directly from the Oconee PRA, plus additional information provided by the BWOG. The column labelled "Action Time" is the average time required to perform the action once the cognitive decision to proceed has been made. This time must be subtracted from the total time available, to get the time available to make the decision (the next column). In most cases the action time is indicated as "negligible." These actions can all be accomplished from the control room and, once the decision is made to take them, the time required is quite short in comparison to the length and uncertainty of the time available to initiate the action. Thus, for all intents and purposes it can generally be assumed that the full available time is available for making the decision. Two operator actions have significant action times associated with them; five minutes for OA-EFW-R and ten minutes for OA-OCT. These actions can sometimes be accomplished from the control room, but often involve action out in the plant. Further, even when the response can occur from the control room, these particular actions can involve restorative actions away from the main control panels. For this reason, these action times were selected (based on expert judgement of the complexity of the action and the potential distance from the control room) for the amount of time required to perform the necessary manipulations.

When the behavior type and the available decision time are known for each action, the remaining piece of information required for the HCR model is the "median response time." This is the subject of the next five columns. The quantity required is the adjusted median response time, that is, adjusted for performance-shaping factors as discussed previously. However, as noted, consideration of the performance-shaping factors is only required if the median response time is estimated from simulator data, task analysis, or expert judgement. If data from actual events is available, this can be used to directly determine the median response time. This data was available for most of the actions evaluated. If actual data was used, then the entry in the column labelled "Median Response Time" is "N/A." In these cases, the performance-shaping factors are provided for informational purposes only, and the value in the column labelled "Adjusted Median Response Time" is estimated from data. This data is shown, for each event evaluated, in Tables 4.9 and 4.10. (Note: In all cases the amount and accuracy of the data was not sufficient to distinguish time intervals closer than one-half minute.) For most of the actions (Table 4.9), the data on the Category C events was taken from the time sequence descriptions provided in the B&W Owner's Group TAP reports.[4-14] For the undercooling (bleed-and-feed) case of OA-HPI-M and for recovery of emergency feedwater (OA-EFW-R) (Table 4.10), the median response times were taken from a series of simulator trials for loss of secondary cooling performed at the Oconee plant simulator and supervised/documented by EG&G-Idaho. These decisions require the same diagnosis because both involve responding to a complete loss of secondary cooling; however, the bleed-and-feed decision involves the additional diagnosis that EFW is unrecoverable. This additional diagnosis drives the high (seven minute) median response time for this event. The control room operator would try to recover EFW from the control room and, failing that, may dispatch an auxiliary operator to the EFW room to attempt local recovery or perform secondary recovery actions. The Oconee simulator trials clearly demonstrated that different operating crews interpreted quite differently the amount of effort required in attempting recovery of secondary cooling before implementing bleed-and-feed. The control room operator would not decide to implement bleed-and-feed unless the auxiliary operator reported back that secondary cooling would be difficult (or impossible) to recover in a short time. This value then was adjusted for the performance-shaping factor under stress (grave) which would occur during an actual event. The selection of performance-shaping factors for each event was based on expert judgement of which definition (as previously presented) best fit the action being considered. During simulator trials the quality of the operator/plant interface and the operator's experience levels are the same as during an actual event, therefore, it is only necessary to modify the observed median response time for the stress factor (which is always assumed to be optimum for simulator trials).

The last column gives the output of the HCR model for each action, except that a lower bound cutoff of 1E-3 was applied (below which it was felt no model can be confidently taken for short-term operator response). This value was the minimum HEP permitted for any action. The values given in that column are the baseline HEPs used for our analysis.

There is one exception to the process presented above, that being the event PTS. For this event, the data were so limited that it was not possible to estimate the median response time, either from actual occurrences or expert judgement. For this event, the quantification model used was the nominal time-reliability correlation from NUREG/CR-1278.[4-9] This model only considers the amount of time available to make the decision. For the 20 minutes available for this action, a HEP of 1E-2 is obtained from that correlation.

There are two other considerations in addition to the baseline values discussed above. First is the effect of failures in key support systems which could complicate a diagnosis and decision, particularly a loss of instrumentation. Second is the consideration of dependencies between certain decision processes. For both of these cases, our analysis is modelled after that in the Connecticut Yankee PSS (performed by Northeast Utilities).[4-16] For accounting for the effect of support system unavailabilities, a series of multipliers is used. Under particular conditions, the baseline HEPs are multiplied by a set factor to account for the increased difficulty in diagnosing the event. If there is a substantial loss of safety-related instrumentation or equipment, a maximum HEP value (0.5) is applied. The particular conditions of concern and their associated modification factors are given in Table 4.11. For handling the effect of dependencies between cognitive errors, a simple equation is used. If an operator's probability of correctly deciding to perform action "B" is dependent on whether it was correctly decided to attempt action "A," the probability of failing to correctly make decision "B" given that the operator failed at "A" is:

$$P(B|A) = P(B)/P(A)$$

where P(A) and P(B) are the independent baseline HEPs for the cognitive actions in question. Where the calculated value of $P(B|A)$ would exceed 0.5 using this equation, 0.5 is used as the conditional HEP. There were only two instances where we felt that conditional HEPs were required. First was where the operator had failed to recognize the need to restore EFW when all feedwater was lost. It was felt that the probability that the operator would subsequently recognize the need to establish bleed-and-feed cooling was dependent on the initial failure, since they are both responses to a loss of all primary heat removal. The second case was when the operator had failed to recognize the need to manually initiate HPI during a minor overcooling. It was felt that the probability that the need to terminate the overcooling in the short term would subsequently be recognized was dependent on the initial failure, since they are both responses to the overcooling-induced drop in RCS pressure and pressurizer level. The results of the evaluation also are shown on Table 4.11.

4.1.4 Results of Event Tree Quantifications

Of the 461 transient sequences embedded in the event tree, 186 sequences are considered to have an end-state of core damage. Depending on the ultimate cause leading to core damage, these sequences can be divided into three groups: (1) small LOCA and failure of long-term cooling (104 sequences), originated from both overcooling and undercooling with failure to actuate HPI. This also includes two sequences involving complete loss of FW, with successful initiation of feed-and-bleed cooling, but failure of long-term cooling, (2) failure of feed and bleed (4 sequences), and (3) vessel failure caused by PTS (78 sequences).

The transient sequences delineated by the event tree were quantified for all of the major B&W plants, i.e., Oconee, Crystal River 3, Davis-Besse, ANO-1, Rancho Seco, and TMI-1. The quantified results are presented in Table 4.12, along with those shown in the Owners Group report (Phase II, page 4-5), which were obtained by using their own event trees. Since the Owners Group reports did not explicitly analyze the PTS sequences, the equivalent of their results obtained in this study are tabulated in the column denoted "Subtotal." The frequencies of core damage shown in the last column (denoted "Total") represent the sum of all the contributions to the core-damage frequency, including that due to PTS.

The BNL results from Table 4.12 are also presented graphically in two ways on Figures 4.4 through 4.7. In Figures 4.4 and 4.5, the relative contribution of each of the three core-damage sequence groups to total Category C core damage frequency is presented for each of the six plants. The figures show that no cooling (failure of feed-and-bleed) is the dominant contributor, contributing over 50% for four of the six plants (over 90% for two of those four). For the two plants which have under a 50% contribution from this sequence group, the contribution is still a significant percentage (over 35%). The next most dominant contributor is LOCA (with failure of long-term cooling), which contributes between 30% and 53% of the Category C core-damage frequency for four of the six plants, but is a rather negligible contributor to the other two. Finally, the last group, PTS exceeds a 10% contribution for only one of the six plants.

Figures 4.6 and 4.7 gives the relative contribution of each of the four initiating events considered to the total Category C core-damage frequency for each of the six plants. Loss of ICS is the dominant contributing over 50% for five of the six plants and 30% for the sixth (for which it is still the highest contributor). The next most dominant initiating event is loss of main feedwater, which contributes about 25% (or more) for five out of the six plants. The other initiating events contribute relatively less overall, with reactor/turbine trip contributing less than 10% to four of the six plants (20% and 26% for the other two) and excessive feedwater contributing 5%, or less, to five of the six plants (20% for the sixth).

Comparing the core-damage frequencies (attributable to Category C events) obtained in this study with those in the B&WOG report, the results compare favorably with each other for Oconee and Davis-Besse, but the BNL results for Crystal River 3, ANO-1, and Rancho Seco are considerably higher (by a factor of about 9, 4.5, and 26, respectively). On the other hand, the BNL results for TMI-1 are lower by a factor of roughly 8. To clarify the reasons behind these differences, a close scrutiny was given to the branch-point probabilities (shown in Table 4.2) and to the initiating event frequencies (shown in Table 4.1). The following insights were gained. Note that the numerical values for the initiating event frequencies (Table 4.1) and the branch-point probabilities of events 1, 2, 5, 8, 9, 10, 12, 17, 20, and 21 (Table 4.2) used by BNL were identical to those used in the B&WOG report. The branch-point probabilities for the remaining 11 events required in the BNL event-tree quantifications, however, were estimated by BNL (see Section 4.1.2 and 4.1.3).

1. Oconee - For Oconee, which is not equipped with MSIVs, it was assumed that there is a probability of 0.2 that loss of ICS could induce a secondary-side blowdown, leading to overcooling. This assumption may contribute to the somewhat higher BNL results for the case of loss of ICS power. The otherwise good agreement between of the two studies can be ascribed, in part, to the relatively reasonable data on unavailability assigned to the feed-and-bleed cooling in the B&WOG analysis. The more elaborate treatment of cognitive human errors in the BNL analysis did not yield appreciably different results. Also, the low unavailability of EFW system (6.7E-5) played a crucial role in maintaining comparatively low frequencies for Category C event core-damage for Oconee as a whole.

2. Crystal River 3 - The frequencies of core damage (due to Category C events) computed by BNL are substantially higher, primarily because of our more exhaustive treatment of cognitive human errors. The unavailability of feed-and-bleed cooling (1.2E-4) used in the B&WOG analysis is believed to be overly optimistic, because of the small probability of human error (1.0E-4) assigned. As compared to Oconee, the core-damage frequencies computed for CR-3 are higher, because not only the unavailabilities of MFW (0.14 vs. 3.8E-2), EFW (2.2E-4 vs. 6.7E-5) and FW-AC (0.04 vs. 8.4E-4) are higher for CR-3, but also the initiating event frequencies in general are higher also.

3. Davis-Besse - The unavailability of the EFW system for Davis-Besse used in the BNL analysis was taken from the B&WOG report (Phase II), which assigns a value of 8.5E-5, based on a three-train system. The failure probability of HPI cooling was estimated to be 0.1 in the B&WOG report, by prescribing the need for operating both charging pumps and opening the PORV (since Davis-Besse does not have a high head HPI system). This estimate may still be optimistic, in view of the fact that the BNL result (1.2E-5), which takes into account EFW recovery (with a nonrecovery factor of 0.2), is only about 40% lower than that obtained by the B&WOG study (2.0E-5). The frequencies of Category C event core damage are virtually predominated by the failure of feed-and-bleed cooling, since a stuck-open SRV following an overcooling is improbable at Davis-Besse due to lack of a high head HPI system. The frequencies of core damage due to SRV LOCA shown in Table 4.12 are mainly those ascribable to undercooling, which leads to RCS pressurization and challenging of the SRVs.

4. ANO-1 - Although BNL results allow EFW recovery, they are still about a factor of 4.5 higher than those estimated by the B&WOG study. The discrepancy is chiefly imputable to the underestimate of the unavailability of feed-and-bleed cooling in the B&WOG analysis, particularly that due to human cognitive errors. It is interesting to briefly compare the input data (branch-point probabilistics) for ANO-1 and CR-3 used in the event tree quantifications. Table 4.2 shows that the unavailabilities of both MFW (0.27 vs. 0.14) and EFW (4.9E-4 vs. 2.2E-4) for ANO-1 are higher roughly by a factor of 2. The unavailabilities of HPI-M (1.4E-3 vs. 1.2E-4) and HPI-A (2.1E-4 vs. 2.4E-5) for ANO-1 are also higher by about a factor of 10. On the other hand, the failure probability of LTC (long-term cooling) for ANO-1 is approximately a factor of 4 lower (3.6E-4 vs. 1.4E-3). The combined frequency of Category C initiating events are also about 25% lower for ANO-1. All these factors are reflected in the results for ANO-1, which indicate that the core-damage frequency due to Category C events is dominated (about 90%) by failure of feed-and-bleed cooling.

5. Rancho Seco - Besides Oconee, Rancho Seco is another plant which does not have MSIVs. Consequently, the probability of a secondary-side blowdown induced by loss of ICS power was assumed to be 0.2, just as for Oconee. It is noteworthy that the branch-point probabilities for Rancho Seco bear a close resemblance to those for CR-3 (see Table 4.2). Differences can be found only in MFW (0.1 vs. 0.14 for CR-3) and LTC (3.6E-4 vs. 1.4E-3 for CR-3). In the B&WOG report, the core-damage frequency imputable to Category C events was estimated to be a factor of five smaller for Rancho Seco (5.8E-7) compared with that for CR-3 (2.9E-6). The BNL results for these two plants, however, are much closer (1.5E-5 for Rancho Seco versus 2.5E-5 for CR-3), mainly because of the detailed treatment of cognitive human errors and the imposition of the aforementioned assumption regarding the secondary-side blowdown upon loss of ICS power.

6. TMI-1 - According to B&WOG, although TMI-1 is equipped with MSIVs, they are slow closing valves which require local operations. Thus, TMI-1 was treated as though it does not have MSIVs, and loss of ICS was assumed to cause secondary-blowdown, leading to RCS overcooling with a probability of 0.2. The unavailability of EFW (1.36E-4) used in the BNL event-tree quantification also was an updated value provided by the B&WOG. Table 4.2 shows that the failure probability of FW-AC (4.1E-5) at TMI-1 is the lowest among all the B&W plants. These relatively low unavailabilities and the generally lower initiating event frequencies, gave a BNL estimate of core-damage frequency attributable to Category C events of 1.9×10^{-6}, which is about a factor of eight smaller than that in the B&WOG report. As will be discussed later (Section 4.1.5, Sensitivity Analyses), the use of the updated (and smaller) unavailability data for EFW and the consideration of EFW recovery accounted for the discrepancy between the BNL and the B&WOG results.

An examination of the individual sequence frequencies revealed that the core-damage frequencies shown in Table 4.12 for each of the B&W plants are dominated by a limited number of transient sequences. To illustrate this point, a few dominant core-damage sequences for each of the plants are listed in Table 4.13, along with their frequencies and initiating events. A terse description of the transient scenarios for these dominant sequences is presented separately in Table 4.14. For CR-3, Davis-Besse, ANO-1, and Rancho Seco, the core-damage frequencies for Category C are dominated by SEQ461 and

SEQ457, which involve complete loss of FW, followed by the operator's failure to recognize the need to either recover FW or initiate feed-and-bleed cooling. It is noteworthy that, in the majority of cases, these dominant core-damage sequences are initiated by either loss of ICS or loss of MFW. For Oconee and TMI-1, SEQ211, which involves severe overcooling due to secondary-side blowdown and the eventual development of a stick-open SRV LOCA and long-term cooling failure, is one of the leading contributors to the Category C events core-damage frequency. The same sequence also appears in the list of dominant core-damage sequences (due to Category C events) for Rancho Seco.

The structure of the present event tree permits rough estimates to be made of the frequencies of the various types of Category C events for each initiating event. There are five types of Category C events, as discussed and represented on the event sequence diagram. They are (1) minor overcooling due to excessive feedwater flow, (2) severe overcooling due to excessive feedwater flow, (3) minor overcooling due to secondary blowdown, (4) severe overcooling due to secondary blowdown, and (5) undercooling. Distribution of Category C event frequencies according to their types are shown in Tables 4.15, 4.17, 4.19, 4.21, 4.23, and 4.25 for each of the plants, respectively. These results are also presented graphically for each plant on the left hand side of Figures 4.8, 4.9, and 4.10. With the exception of Oconee, roughly 60% of the total Category C events frequencies can be attributed to overcooling due to secondary blowdown, and the remainder to overcooling due to excessive feedwater. For Oconee, about 80% is due to secondary-side blowdown. Contribution to the frequency of Category C event by undercooling sequences is extremely small (in most cases, less than 0.02%). A large fraction of the overcooling sequences can be classified as minor overcooling. For Rancho Seco, for example, minor overcooling accounts for about 71% of the total Category C event frequency, as opposed to 29% for severe overcooling.

These results can be contrasted with the risk potential of Category C events, by looking at the contributions to the total Category C core-damage frequency induced by each of the five types of Category C events. These results are summarized in Tables 4.16, 4.18, 4.20, 4.22, 4.24, and 4.26 for each plant, respectively. These results also are presented graphically for each plant on the right hand side of Figures 4.8, 4.9, and 4.10. These figures give a ready comparison between the contribution of each Category C event type to the total Category C frequency, and the contribution of each event type to the total core-damage frequency due to Category C events. In contrast with the trend found in the distribution of Category C event frequencies, which indicates a negligible contribution by undercooling sequences, a substantial portion of the total frequency of core-damage can be ascribed to undercooling sequences (35% for Oconee, 46% for TMI-1, and more than 55% for the remainder). For Davis-Besse and ANO-1 undercooling sequences contribute nearly 98% and 91% respectively to the total frequency of core damage due to Category C events. On the other hand, the risk potential of minor overcooling sequences are less significant compared to that of severe overcooling sequences. For Rancho Seco, for example, the severe overcooling sequences contribute about 29% to the total frequency of core damage, compared with 8.5% by minor overcooling sequences, although the latter has much higher frequency of occurrence. We can conclude that despite their large contributions to the frequency of occurrence of Category C events, the minor overcooling sequences are relatively insignificant to risk compared to severe overcooling sequences. Further, since undercooling events dominate the Category C core-damage frequency while contributing little to the Category C event frequency, it can be stated that just looking at Category C event frequency (as the events are presently defined) is not a reasonable predictor of core-damage risk. That is, the events which dominate Category C event frequency are not the types of events which are most likely to lead to core damage.

4.1.5 Sensitivity Analyses

(a) Sensitivity Studies on EFW Failure Probability and EFW Recovery Probability

In quantifying the BNL event trees, it was revealed that the Category C event core damage frequency is rather sensitive to the unavailability of EFW used in the computation. To illustrate this finding, the results of sensitivity studies are shown in Table 4.27 for Davis-Besse and TMI-1. For the former, raising the unavailability of EFW from 8.5E-5 to 1.6E-3 alone caused the core-damage frequency to increase from 1.2E-5 to 2.3E-4, almost a proportional increase. The EFW unavailability of 1.6E-3 was an estimate[4-16] based on the old Davis-Besse two-train system, equipped with steam turbine-driven pumps. It was also the value used for the precursor study to be discussed later (Section 4.2). A similar trend can be found for TMI-1. By simply changing the unavailability of EFW from 1.36E-4 to 9.4E-4, the value used in the B&WOG report, the core-damage frequency increases from 1.9×10^{-6} to 7.0×10^{-6}. For TMI-1, therefore, the discrepancy between our results and those of the B&WOG report is partly caused by the use of the updated EFW unavailability data in the BNL analyses. These results also suggest that improving the availability of EFW system is an effective means of lowering the frequency of core damage frequency associated with Category C events.

As pointed out previously, the BNL analyses explicitly consider EFW recovery as one of the event-tree top events. The results of sensitivity studies on the EFW recovery probability are shown in Table 4.28 for the same two plants. For Davis-Besse, a sensitivity study in which the nonrecovery factor of EFW was raised from 0.2 to 0.5, increased the core damage frequency from 1.2E-5 to 2.5E-5. For TMI-1, raising the EFW nonrecovery factor from 0.2 to 0.5 also increased the core damage frequency from 1.9×10^{-6} to 2.6×10^{-6}. The consideration of EFW recovery in the BNL analyses thus partially accounts for the discrepancies between the BNL results and those of the B&WOG report for these two plants. However, the use of the higher nonrecovery factor does not change the results sufficiently to affect any conclusions regarding the risk profile of B&W plants with respect to the Category C events evaluated or of the relative importance of each type of Category C event.

(b) HRA Sensitivity Study

Most HEPs were assessed using data from the historic Category "C" events. The data was a blend of responses from various B&W plants, dominated by Crystal River, Rancho Seco, and Davis Besse. However, two key responses which related to total loss of feedwater scenarios did not have sufficient data on observed responses in real-life situations. These actions were OA-EFW-R and OA-HPI-M (bleed-and-feed case), as discussed in Section 4.1.5. For these actions, simulator trial data from the Oconee plant simulator using four Oconee crews was used. Because Oconee is considered to be a well-operated plant with well-designed control rooms and well-trained crews, there was some concern that this data might be optimistic for other B&W plants. In particular, concern was expressed that two key performance-shaping factors could be different for other plants (the quality of the operator/plant interface and the level of operator training/experience). To determine the significance of these factors, a sensitivity analysis was performed on Crystal River and Davis Besse (because of the similarity in results between Crystal River and the other three B&W plants, the magnitude of the sensitivity effect is expected to be similar to that observed for Crystal River).

Three sensitivity cases were evaluated using the HCR model:

Case 1: Operator's experience level not as high as Oconee. (An operator's experience level of average rather than good for PSF K_1 from Tables 4.6 and 4.7).
Case 2: Operator/plant interface not as good as Oconee. (A plant interface quality level of fair rather than good for PSF K_2 from Tables 4.6 and 4.7).
Case 3: Combination of Cases 1 and 2.

By adjusting the median response times obtained from the Oconee simulator data to account for these PSF modifications, as specified in the HCR model in Section 4.1.3, new HEPs were obtained for the event tree model. The new baseline values obtained for Crystal River and Davis Besse were:

Case 1: OA-EFW-R 3E-2, 3E-2; OA-HPI-M 2E-2, 0.1.
Case 2: OA-EFW-R 5E-2, 5E-2; OA-HPI-M 3E-2, 0.1.
Case 3: OA-EFW-R 1E-1, 1E-1; OA-HPI-M 5E-2, 0.2.

For purposes of comparison, the original values were 1E-2, 1E-2, 9E-3, and 6E-2.

The results of this study are shown in Table 4.29. Comparing these results with the baseline results on Table 4.10 shows that the only contributor to core damage affected is feed-and-bleed (total loss of all core cooling). For Crystal River, Case 1 yields an increase in the feed-and-bleed contributor of a factor of three, and an increase in the total frequency of core damage of a factor of two. Case 2 yields factors of five and three, respectively. Case 3 yields factors of six and four. For Davis-Besse, the increase factors are: Case 1: two and two, Case 2: two and two, and Case 3: three and three. Thus, the effect of changed performance-shaping factors could have a significant effect on the results, depending on the particular plant and combination of factors. However, it is not significant enough to markedly change our conclusions concerning the effect of Category C events on the risk profile of B&W plants, nor which Category C events are most important to risk.

4.2 Precursor Study

The primary objective of the precursor study performed by BNL was to estimate approximately of how close the historical Category C events were to the state of core damage. The relatively detailed structure of the event tree developed in this study enables close representation of the actual transient scenarios for most of the major historical Category C events at B&W plants (see B&WOG Phase I report for detail) during the past several years. For the precursor study, the path of a particular event tree sequence, which closely delineates the transient scenario of a given historical Category C event, is first traced, as illustrated by the solid line in Figure 4.11. At each plateau of the sequence path, the conditional probability of going to core damage is calculated. At each plateau, there is a corresponding event tree branch point, which separates into an upper branch (denoting the top-event success path) and a lower branch (top-event failure path). The ramifications of the event tree encompassed by these two branches, whose end states are classified as "core damage," then are quantified for their sequence frequencies, and summed up. The conditional probability of interest can be obtained by dividing the sum by the probability of reaching that particular plateau, which is simply the sequence frequency computed starting from the initiating event frequency up to the branch point before to reaching the plateau. It is evident that the initiating event frequency does not enter into the conditional probability because it is cancelled out in the course of the division.

The precursor study briefly described above was performed for 12 B&W historical Category C events, and the results are presented in Figures 4.11 through 4.21. There is a separate number shown inside a bracket underneath the conditional probability at each plateau of the event tree sequence path. These numbers were obtained in a manner similar to that described above, except that, when the core-damage frequencies were summed up for calculating the conditional probability at each plateau, only the core-damage sequences encompassed by the lower branch (top-event failure path) of the corresponding branch-point were included in the summation. These numbers are useful in determining the relative importance of the top events for avoiding core damage. Specifically, the highest number in parentheses occurs at the top event, whose failure is most likely to lead to core damage, thus indicating which failure that did not actually occur during the sequence dominates the conditional core damage probability. Also, the core damage frequencies used in computing the conditional probabilities include contributions from PTS sequences.

The conditional probabilities computed in the present precursor study signify, at various stages of the transients, how close the actual events were to core damage. Referring to the Davis-Besse event (6/9/85) (Figure 4.20), for example, following the successive losses of MFW and EFW, there was almost a 4% probability that core damage would have occurred. The conditional probability for core damage was slightly decreased when the operator recognized the need to recover the EFW, and was greatly reduced once the EFW was successfully restored. Since the only conceivable way of having a core damage thereafter was to have a stuck-open SRV with long-term cooling failure, the conditional probability of core damage further went down when SRVs closed successfully after opening (event 10) and the operator realized the necessity to manually control HPI flow (event 16) to prevent the pressurizer from becoming water solid (the probability of core damage became zero, since Davis-Besse does not have a high head HPI system).

Table 4.30 summarizes the largest conditional probability of core damage attained during each of the historical Category C events. The events that occurred at Davis-Besse (9/9/77 and 6/9/85), and Rancho Seco (3/20/78) appear to be far more serious in terms of coming close to core damage compared to the remainder of the historical Category C events. It is interesting to note that the exigency of these serious events almost exclusively stems from lack of cooling (i.e., undercooling). The results of precursor study shown in Table 4.30 indicate that although Category C events involving overcooling (such as those initiated by excessive FW or by stuck open secondary valve) occur rather frequently, they are relatively insignificant to core damage.

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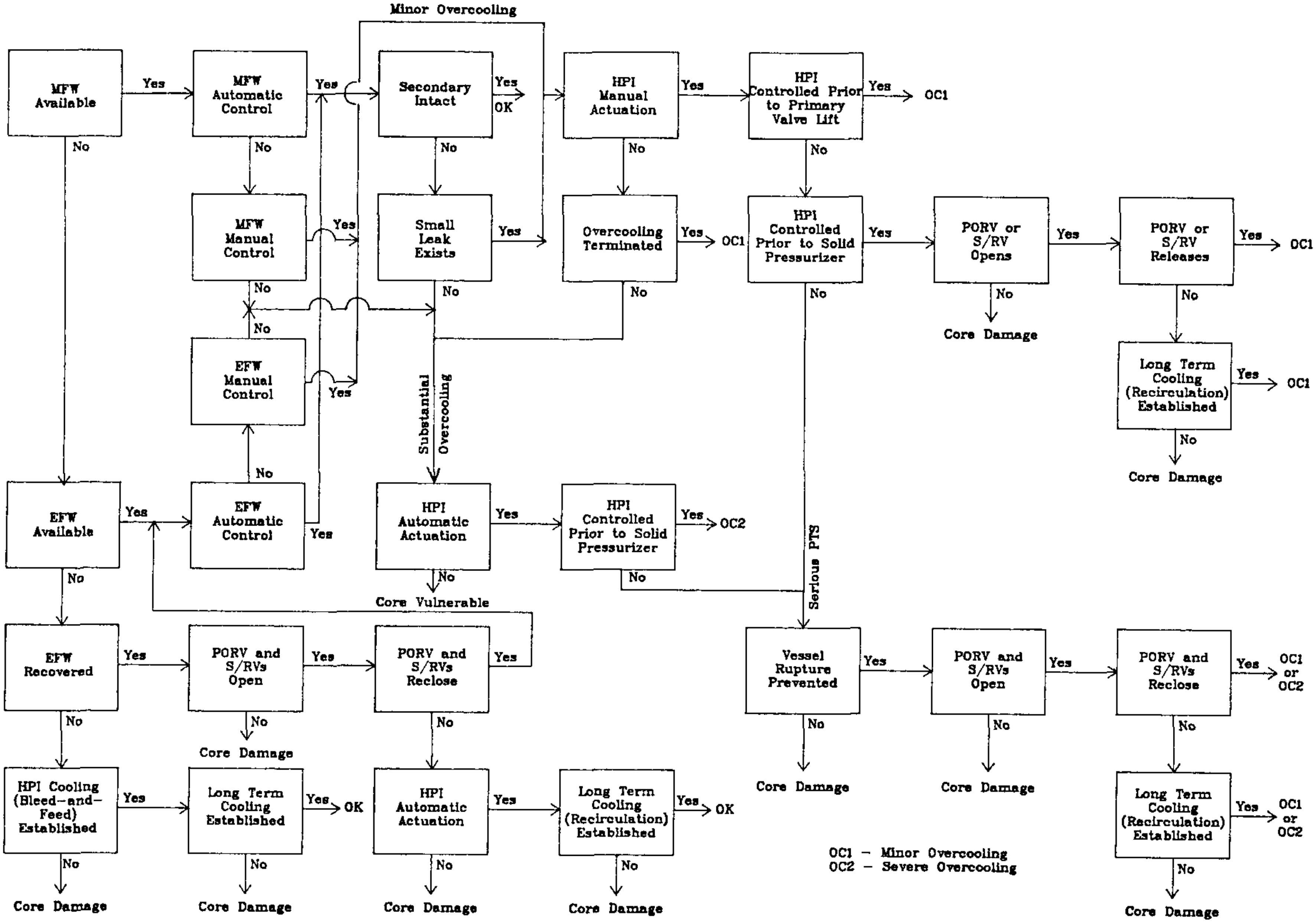


Figure 4.1. ESD (Event Sequence Diagram).

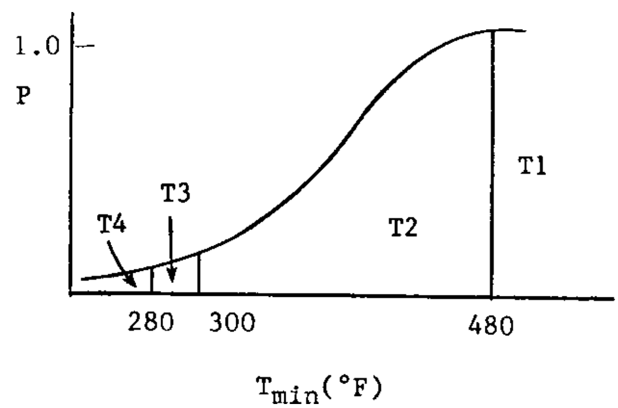


Figure 4.2. Distribution of minimum coolant temperature for overcooling transients.

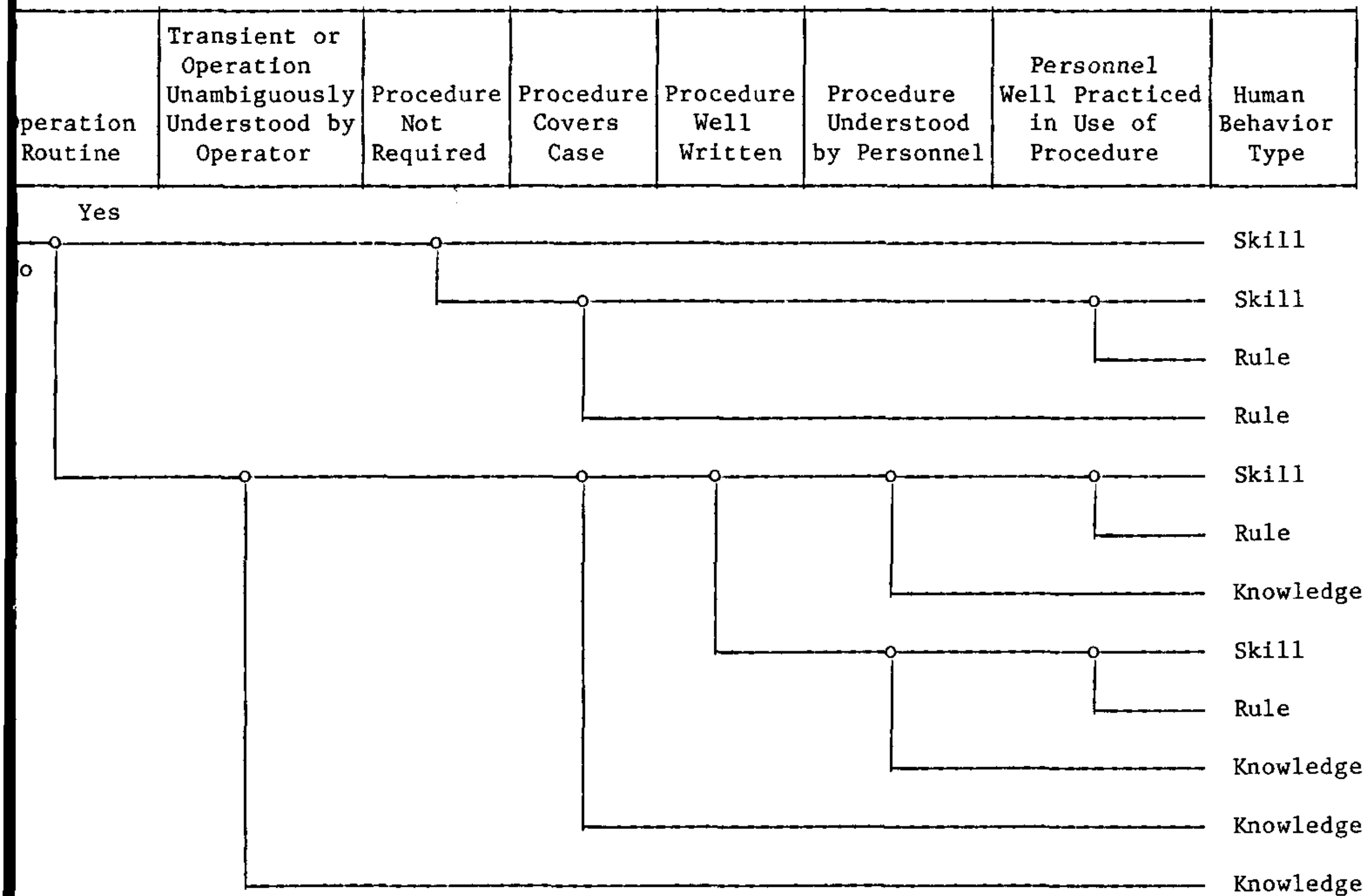


Figure 4.3. Logic tree to aid in selection of expected behavior type.

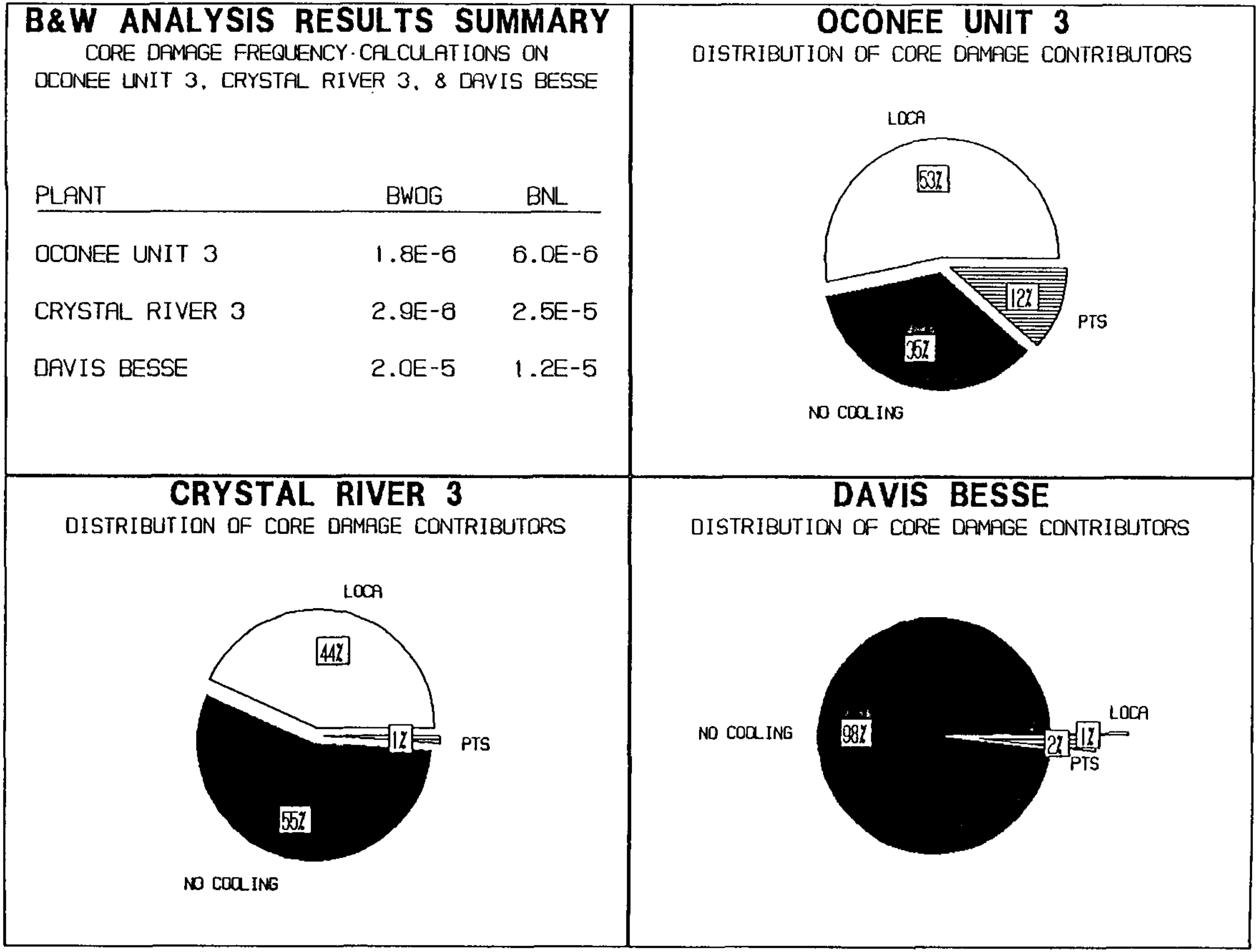
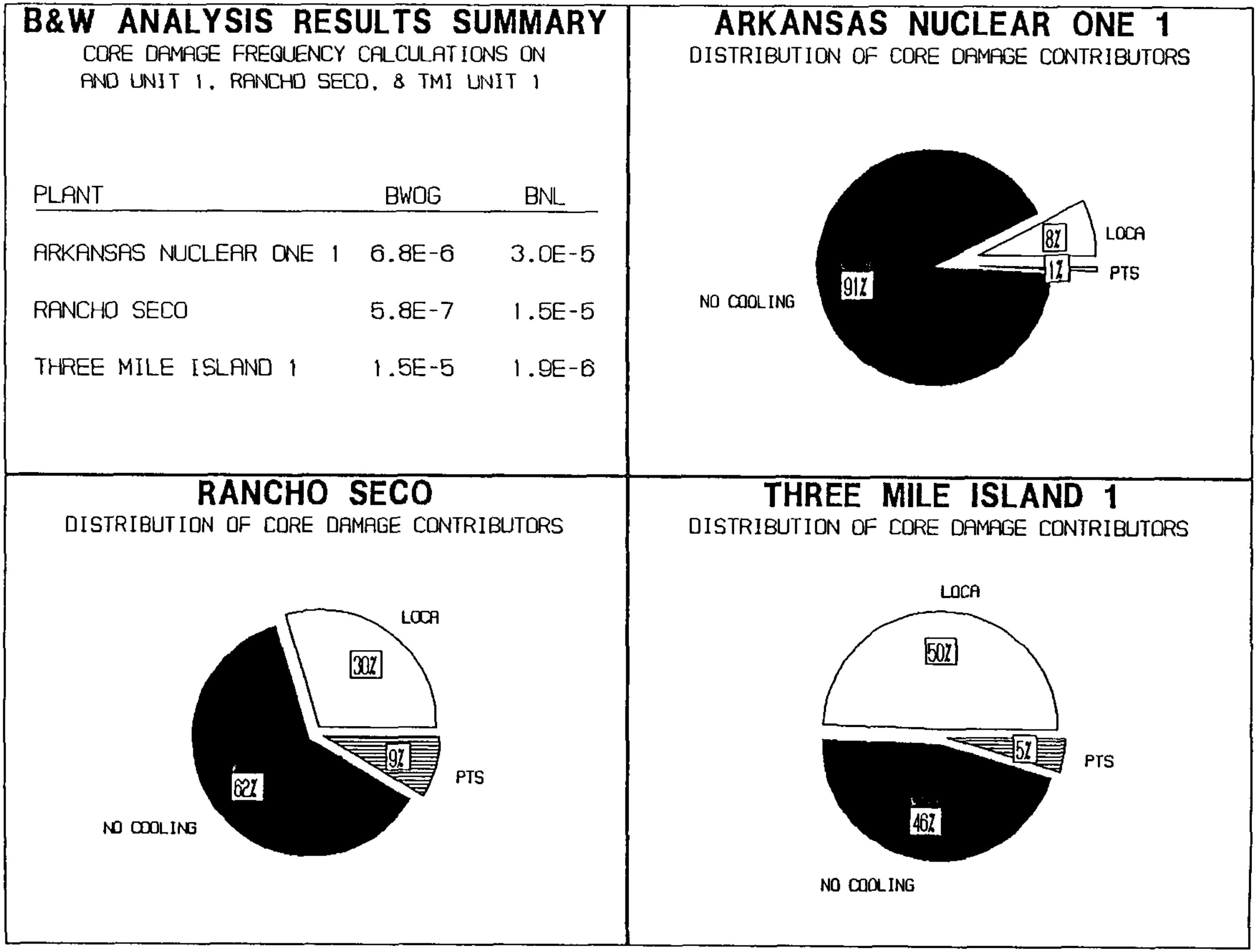


Figure 4.4 Distribution of core damage contributors, Oconee-3, CR-3, and Davis Besse.



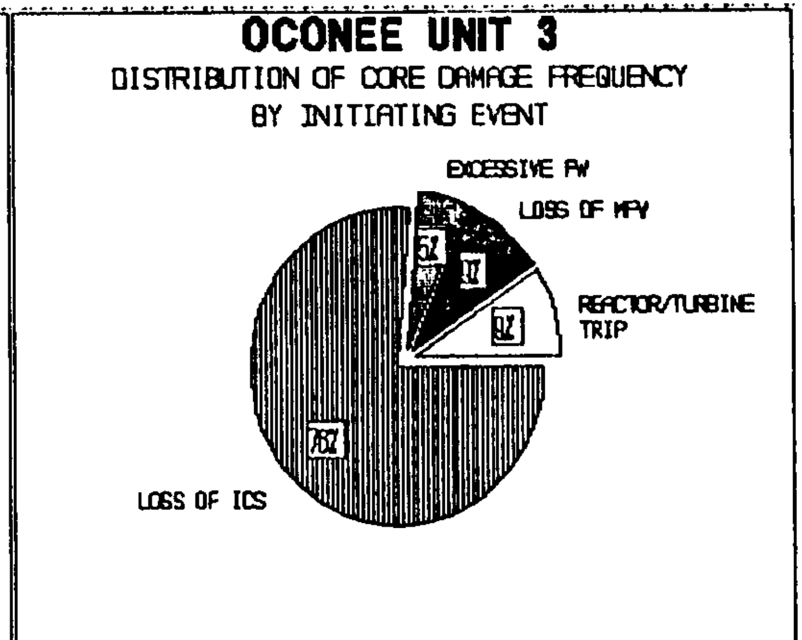
PLANT	BWOG	BNL
ARKANSAS NUCLEAR ONE 1	6.8E-6	3.0E-5
RANCHO SECO	5.8E-7	1.5E-5
THREE MILE ISLAND 1	1.5E-5	1.9E-6

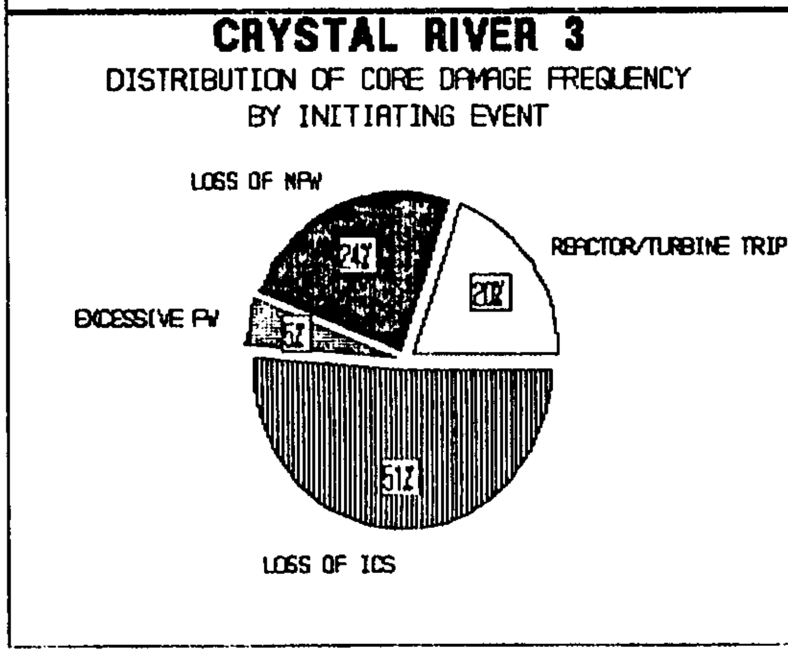
Figure 4.5 Distribution of core damage contributors, ANO-1, Rancho Seco, and TMI-1.

B&W ANALYSIS RESULTS SUMMARY

CORE DAMAGE FREQUENCY CALCULATIONS ON
OCONEE UNIT 3, CRYSTAL RIVER 3, & DAVIS BESSE

PLANT	BWOG	BNL
OCONEE UNIT 3	1.8E-6	6.0E-6
CRYSTAL RIVER 3	2.9E-6	2.5E-5
DAVIS BESSE	2.0E-5	1.2E-5





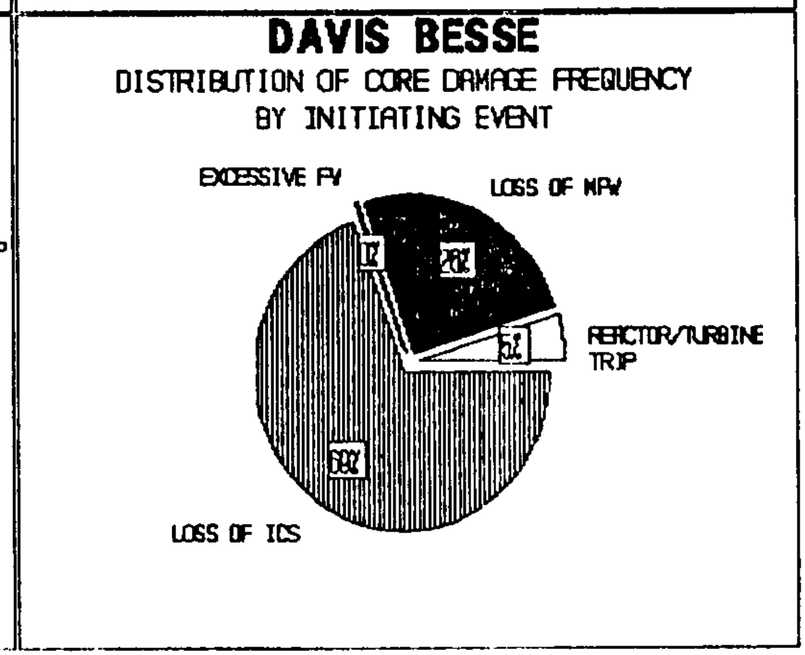


Figure 4.6 Distribution of core damage frequency by initiating event, Oconee-3, CR-3, and Davis-Besse.

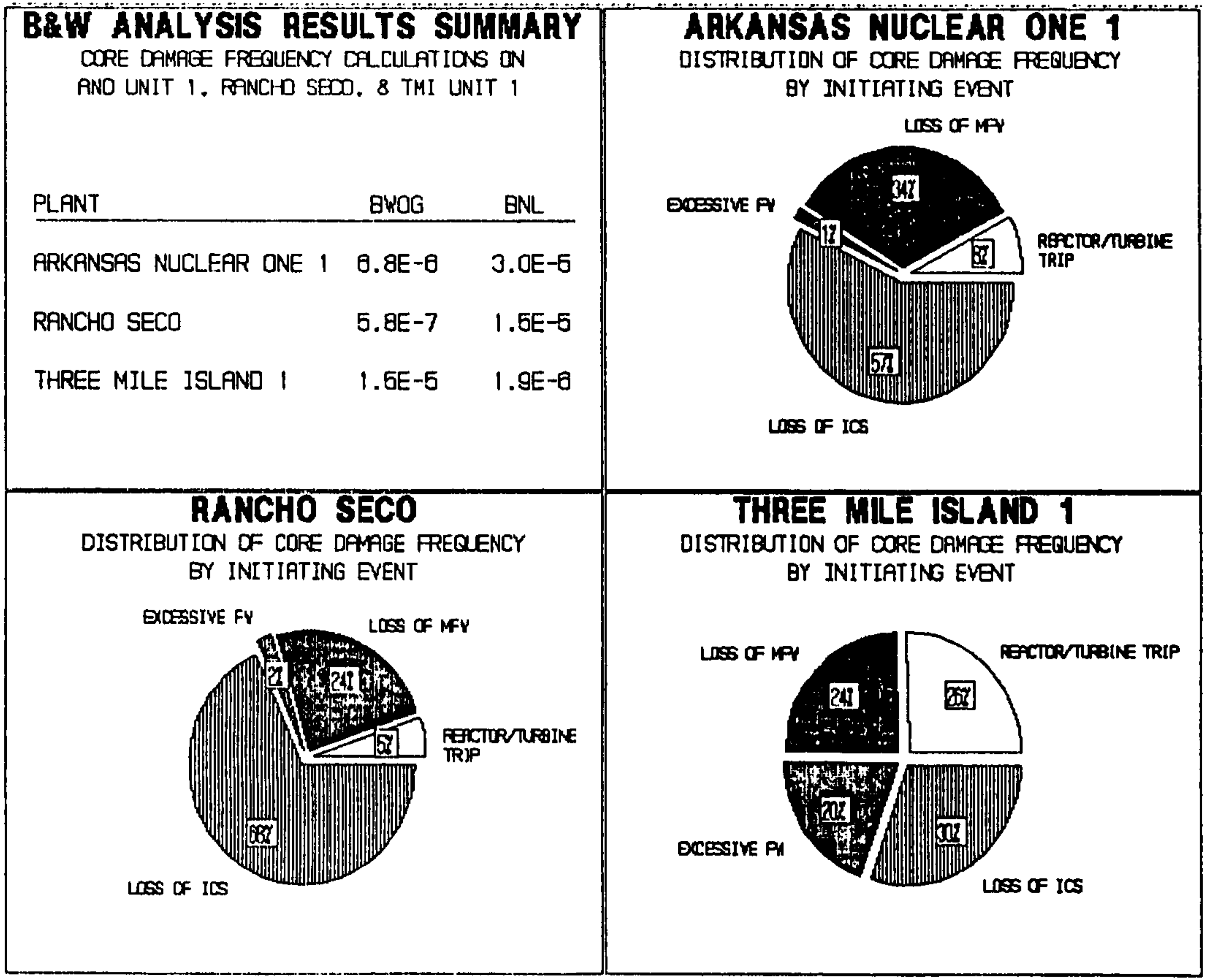


Figure 4.7 Distribution of core damage frequency by initiating event, ANO-1, Rancho Seco, and TMI-1.

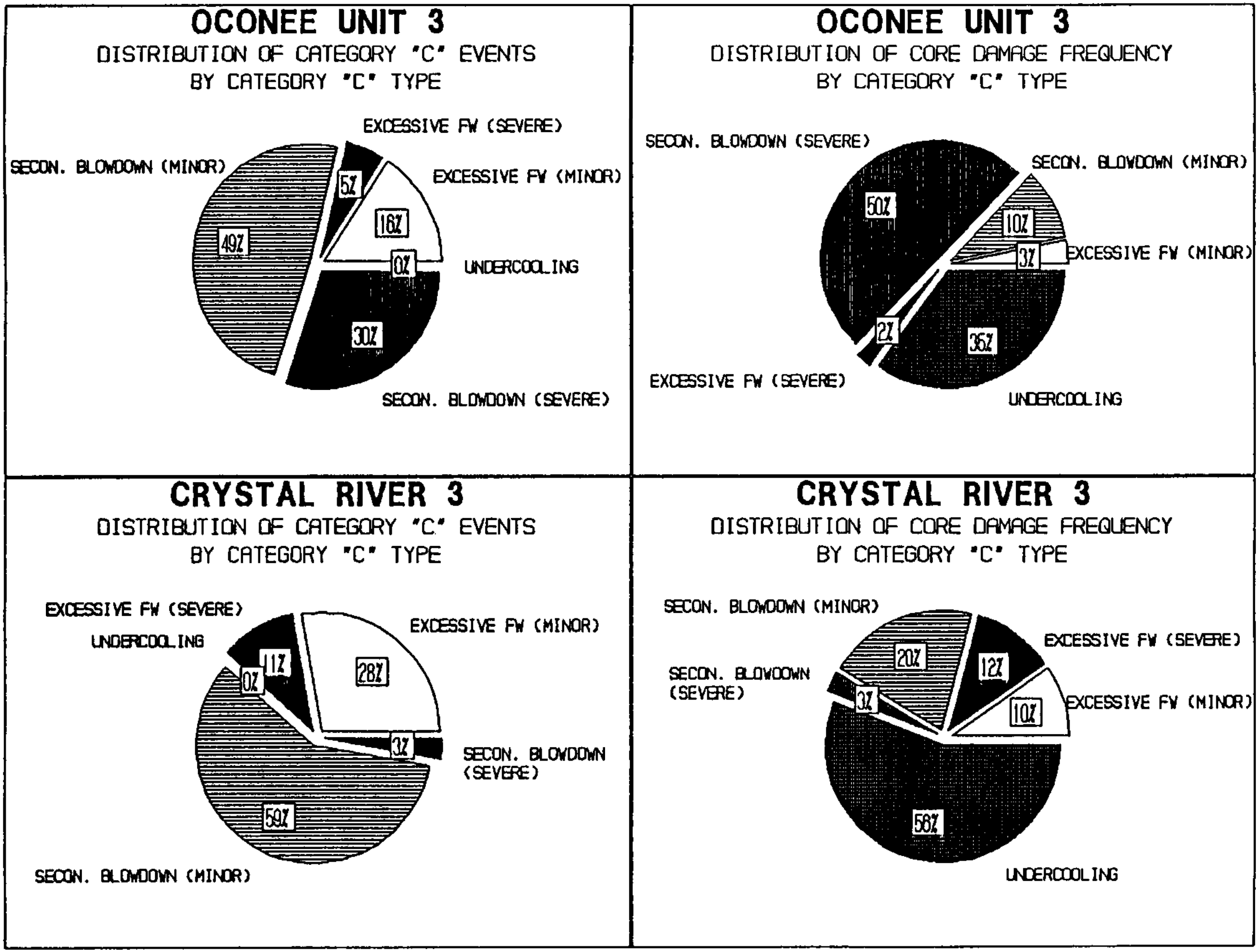


Figure 4.8 Distribution of Category C events and core damage frequency by Category C type, Oconee-3 and CR-3.

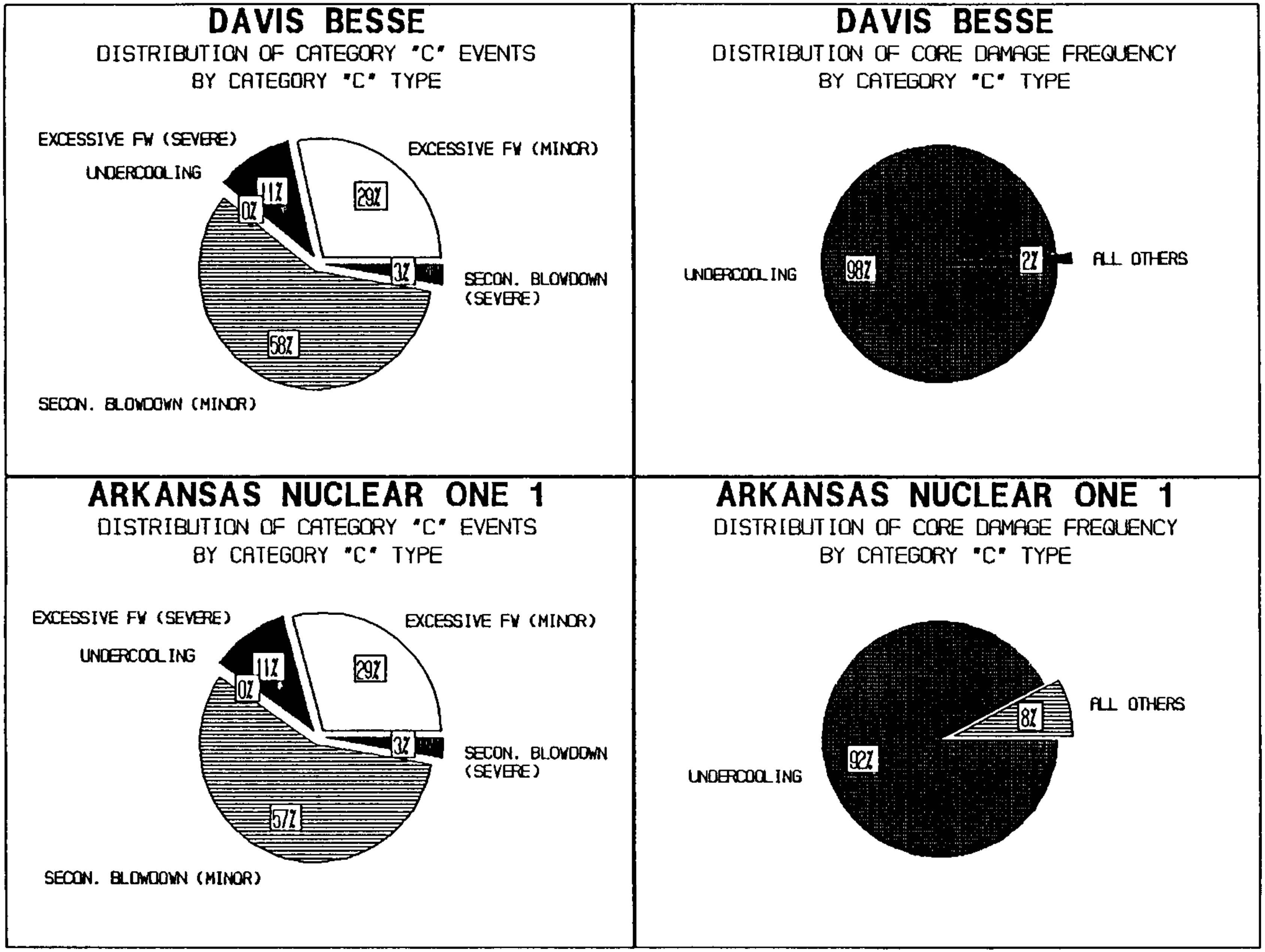


Figure 4.9 Distribution of Category C events and core damage frequency by Category C type, Davis-Besse and ANO-1.

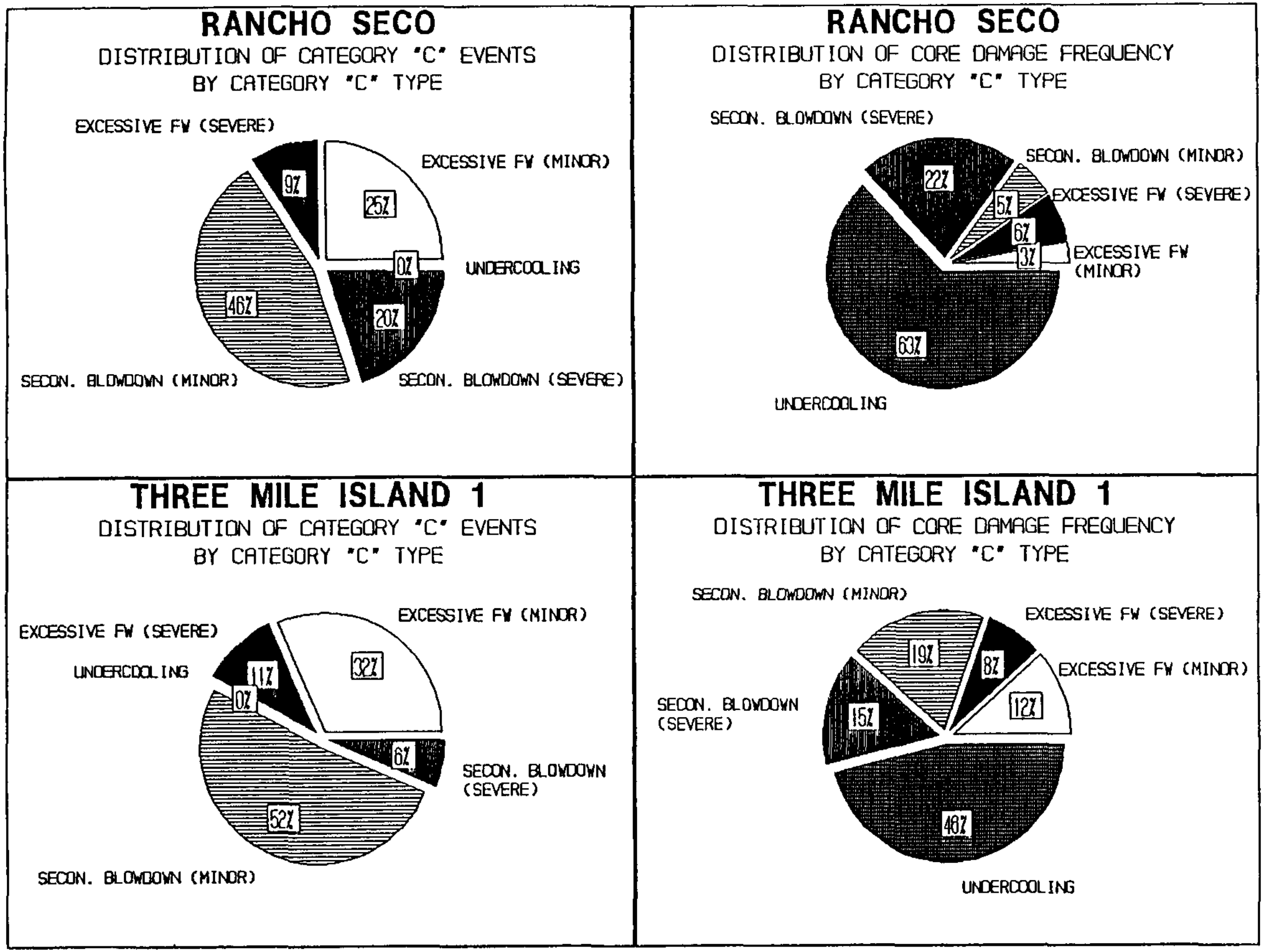


Figure 4.10 Distribution of Category C events and core damage frequency by Category C type, Rancho Seco and TMI-1.

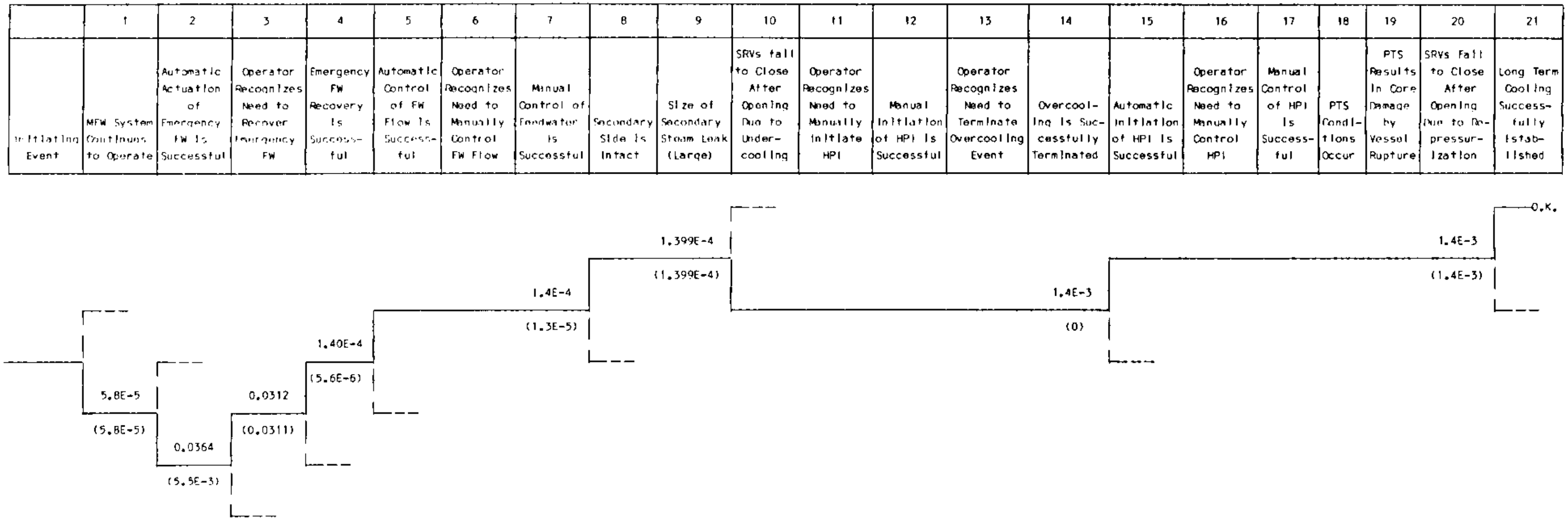


Figure 4.11. Davis-Besse 9/9/77.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery is Success-ful	Automatic Control of FW Flow is Success-ful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater is Successful	Secondary Side is Intact	Size of Secondary Steam Leak (Large)	SRVs fail to Close After Opening Due to Under-cooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcool-ing is Suc-cessfully Terminated	Automatic Initiation of HPI is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI is Success-ful	PTS Condi-tions Occur	PTS Results in Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Re-pressur-ization	Long Term Cooling Success-fully Estab-lished

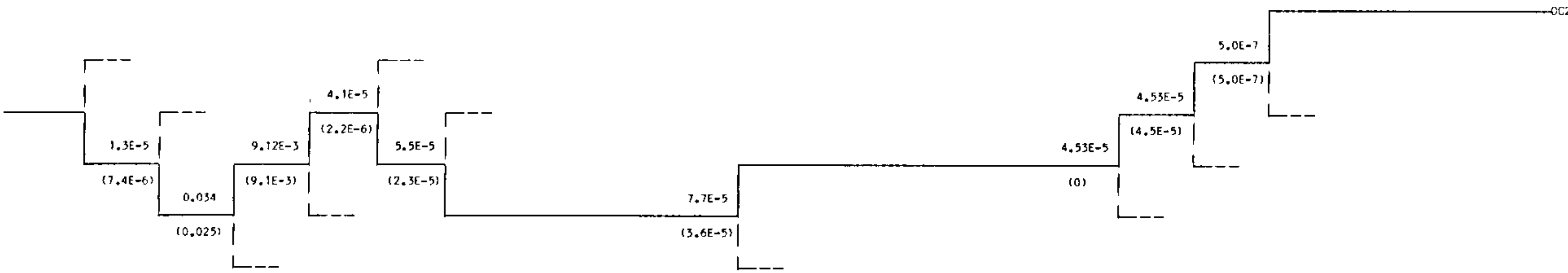


Figure 4.12. RS 3/20/78.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW Is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery Is Successful	Automatic Control of FW Flow Is Successful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater Is Successful	Secondary Side Is Intact	Size of Secondary Steam Leak (Large)	SRVs Fail to Close After Opening Due to Undercooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI Is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcooling Is Successfully Terminated	Automatic Initiation of HPI Is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI Is Successful	PTS Conditions Occur	PTS Results In Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Repressurization	Long Term Cooling Successfully Established

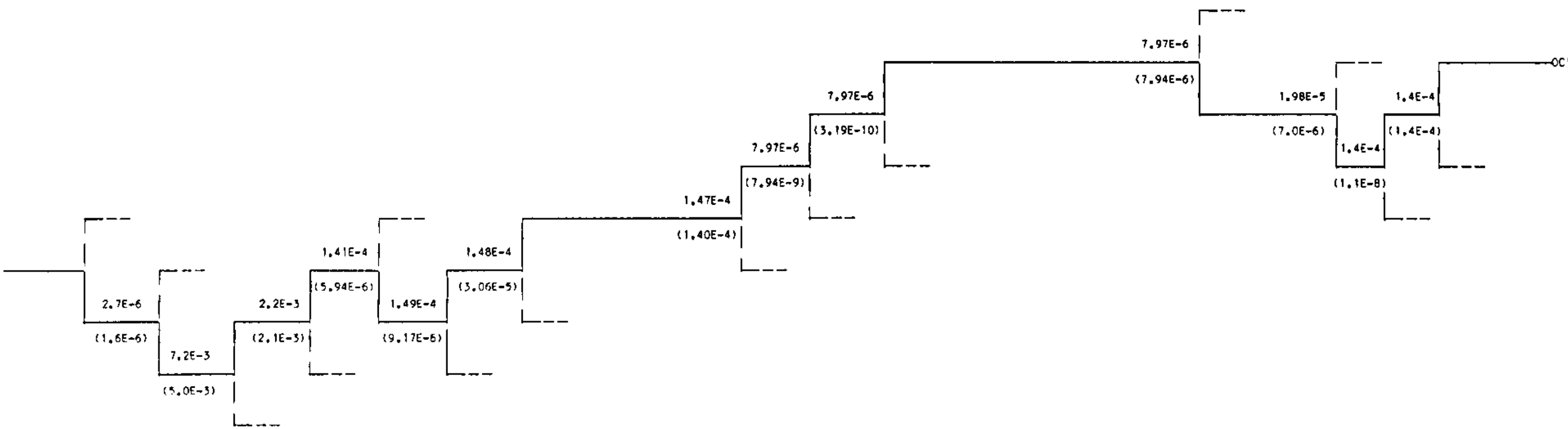


Figure 4.13. CR-3 2/26/80.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery is Success-ful	Automatic Control of FW Flow is Success-ful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater is Successful	Secondary Side is Intact	Size of Secondary Steam Leak (Large)	SRVs fail to Close After Opening Due to Under-cooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcool-ing is Suc-cessfully Terminated	Automatic Initiation of HPI is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI is Success-ful	PTS Condi-tions Occur	PTS Results in Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Re-pressur-ization	Long Term Cooling Success-fully Estab-lished

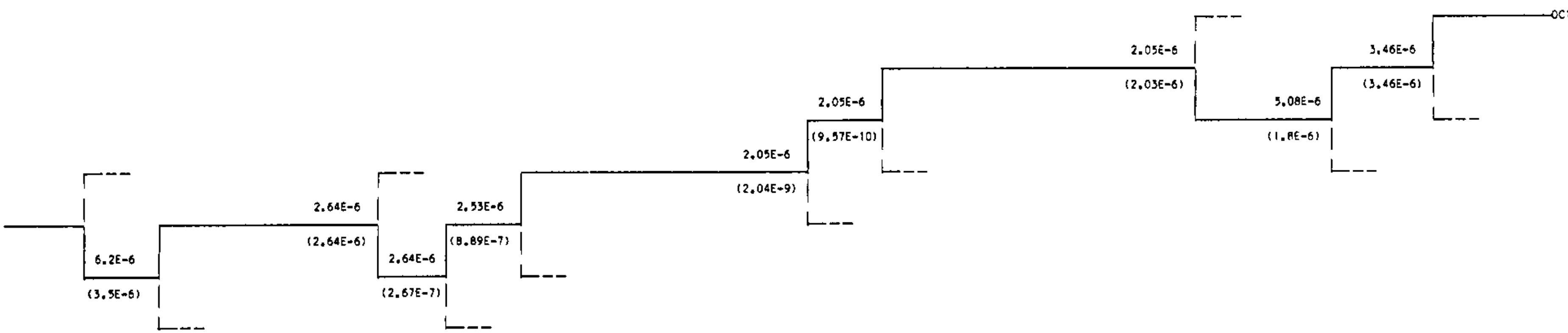


Figure 4.14. ANO 4/7/80.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW Is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery Is Successful	Automatic Control of FW Flow Is Successful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater Is Successful	Secondary Side Is Intact	Size of Secondary Steam Leak (Large)	SRVs fail to Close After Opening Due to Under-cooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI Is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcooling Is Successfully Terminated	Automatic Initiation of HPI Is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI Is Successful	PTS Conditions Occur	PTS Results in Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Re-pressurization	Long Term Cooling Successfully Established

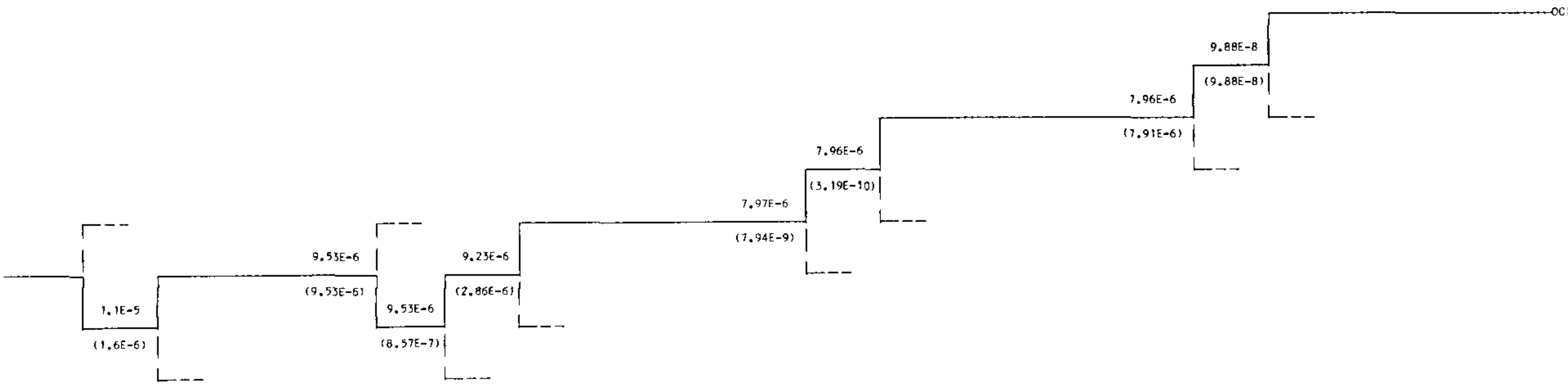


Figure 4.15. CR 10/9/85, CR 6/16/81.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW Is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery Is Successful	Automatic Control of FW Flow Is Successful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater Is Successful	Secondary Side Is Intact	Size of Secondary Steam Leak (Large)	SRVs fail to Close After Opening Due to Under-cooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI Is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcooling Is Successfully Terminated	Automatic Initiation of HPI Is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI Is Successful	PTS Conditions Occur	PTS Results In Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Re-pressurization	Long Term Cooling Successfully Established

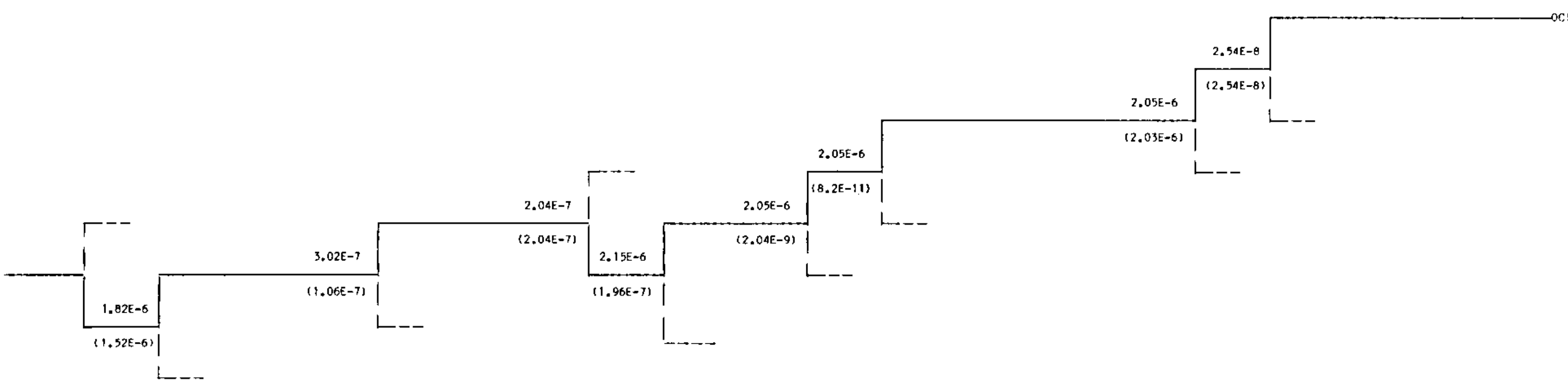


Figure 4.16. RS 10/2/85, RS 6/17/81.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW Is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery Is Successful	Automatic Control of FW Flow Is Successful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater Is Successful	Secondary Side Is Intact	Size of Secondary Steam Leak (Large)	SRVs fail to Close After Opening Due to Under-cooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI Is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcooling Is Successfully Terminated	Automatic Initiation of HPI Is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI Is Successful	PTS Conditions Occur	PTS Results in Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Re-pressurization	Long Term Cooling Successfully Established

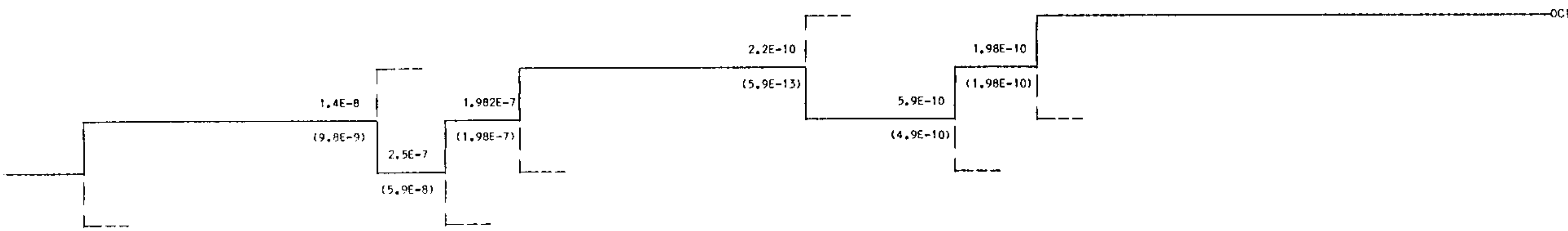


Figure 4.17. Davis-Besse 3/2/84.

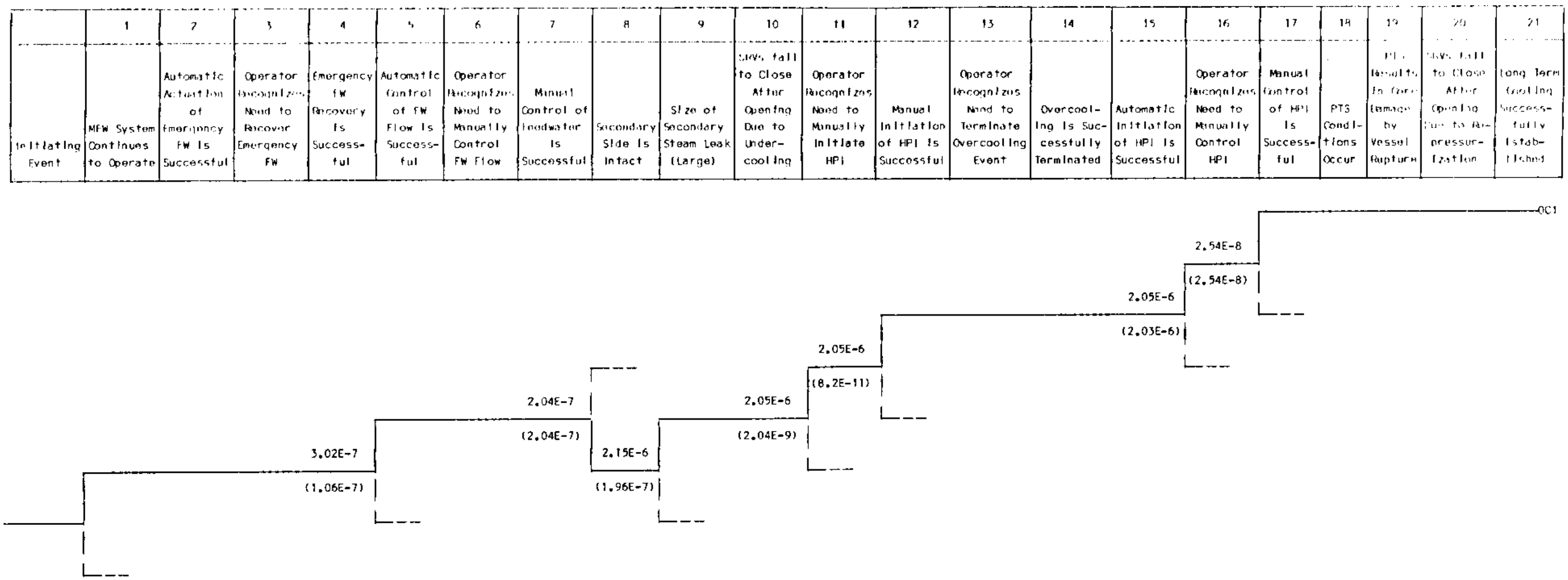


Figure 4.18. RS 3/19/84 (Phase I).

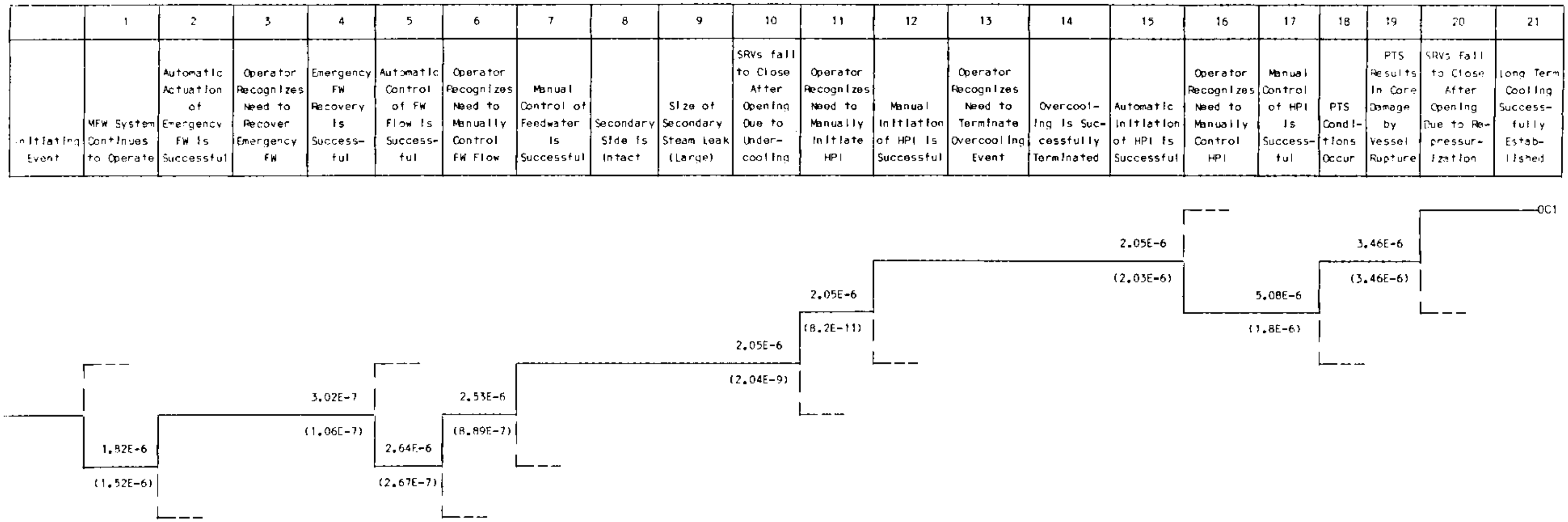


Figure 4.19. RS 3/19/84 (Phase II).

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW Is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery Is Successful	Automatic Control of FW Flow Is Successful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater Is Successful	Secondary Side Is Intact	Size of Secondary Steam Leak (Large)	SRVs Fail to Close After Opening Due to Undercooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI Is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcooling Is Successfully Terminated	Automatic Initiation of HPI Is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI Is Successful	PTS Conditions Occur	PTS Results In Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Repressurization	Long Term Cooling Successfully Established

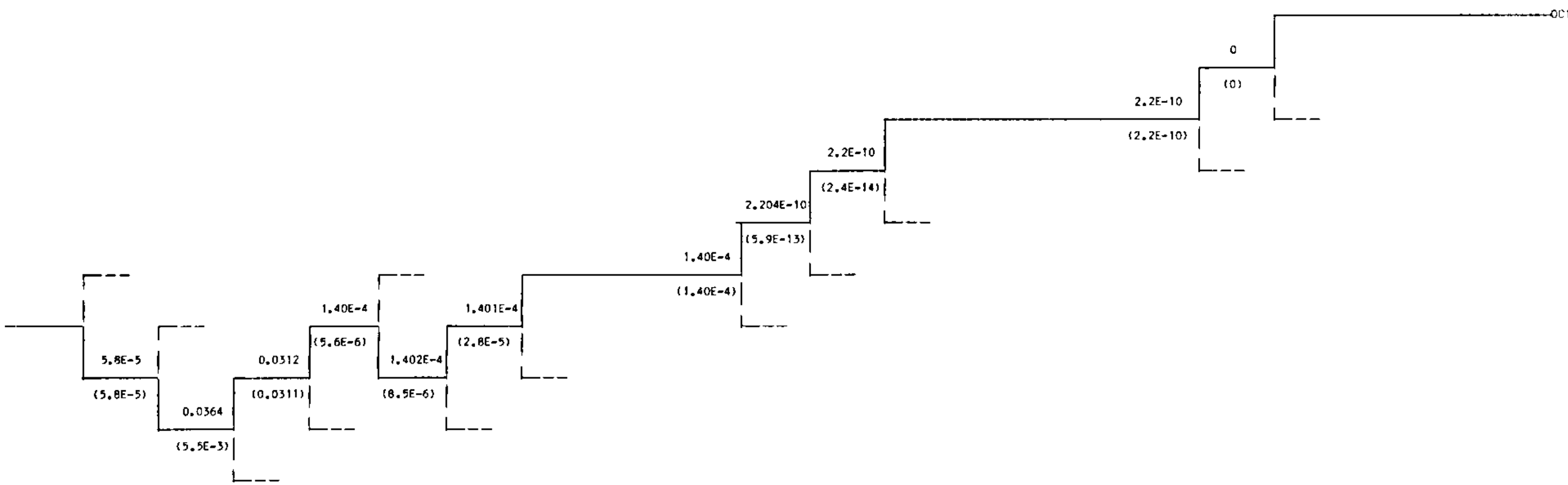


Figure 4.20. Davis-Besse 6/9/85.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
Initiating Event	MFW System Continues to Operate	Automatic Actuation of Emergency FW Is Successful	Operator Recognizes Need to Recover Emergency FW	Emergency FW Recovery Is Successful	Automatic Control of FW Flow Is Successful	Operator Recognizes Need to Manually Control FW Flow	Manual Control of Feedwater Is Successful	Secondary Side Is Intact	Size of Secondary Steam Leak (Large)	SRVs Fail to Close After Opening Due to Undercooling	Operator Recognizes Need to Manually Initiate HPI	Manual Initiation of HPI Is Successful	Operator Recognizes Need to Terminate Overcooling Event	Overcooling Is Successfully Terminated	Automatic Initiation of HPI Is Successful	Operator Recognizes Need to Manually Control HPI	Manual Control of HPI Is Successful	PTS Conditions Occur	PTS Results in Core Damage by Vessel Rupture	SRVs Fail to Close After Opening Due to Repressurization	Long Term Cooling Successfully Established

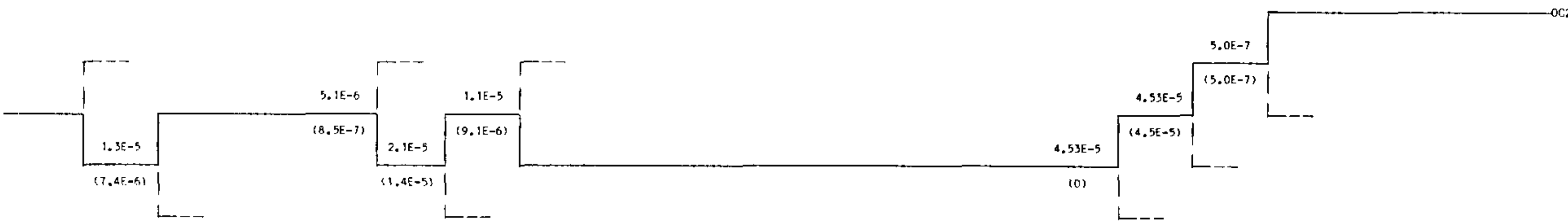


Figure 4.21. RS 12/26/85

Table 4.1
B&W Plant-Specific Initiating Event Frequencies
(Per Reactor/Year)*

Initiating Event	Oconee	CR-3	ANO-1	Davis-Besse	Rancho-Seco	TMI-1
Reactor/turbine trip	2.9	3.7	1.8	4.3	1.7	3.0
Loss of MFW	0.72	2.3	2.7	1.0	2.0	0.23
Excessive FW	0.12	0.12	0.12	0.12	0.12	0.12
Loss of ICS Power	0.67	1.3	1.0	1.0	0.83	0.054
Total	4.41	7.42	5.62	6.42	4.65	3.404

*The plant-specific frequencies were developed in Reference 1-4.

Table 4.2
Summary of Branch-Point Probabilities

Event No.	Event-tree Top Event	Oconee	CR-3	ANO-1	Davis-Besse	Rancho Seco	TMI-1
1	MFW	3.8E-2	0.14	0.27	3.8E-2	0.1	2.4E-2
2	EFW	6.7E-5	2.2E-4	4.9E-4	8.5E-5	2.2E-4	1.36E-4
3	OA-EFW-R	1.0E-2	1.0E-2	1.0E-2	1.0E-2	1.0E-2	1.0E-2
4	EFW-R	0.2	0.2	0.2	0.2	0.2	0.2
5	FW-AC	8.4E-4	0.04	0.04	0.04	0.04	4.1E-5
6	OA-FW-MC	0.06	0.06	0.06	0.06	0.06	0.06
7	FW-MC	0.2	0.2	0.2	0.2	0.2	0.2
8	Secon. Int.	7.1E-2	9.5E-2	9.5E-2	9.5E-2	9.5E-2	4.6E-2
9	Leak Size	0.15	4.5E-2	4.5E-2	4.5E-2	4.5E-2	4.5E-2
10	PS/RV-C-UC	0.1	0.1	0.1	0.1	0.1	0.1
11	OA-HPI-M[1]	1.0E-3	1.0E-3	1.0E-3	1.0E-3	1.0E-3	1.0E-3
12	HPI-M	1.0E-2	1.2E-4	1.4E-3	1.2E-4 0.1 for Feed & Bleed	1.2E-4	3.4E-2
13	OA-OCT[2]	1.0E-3	1.0E-3	1.0E-3	1.0E-3	1.0E-3	1.0E-3
14	OCT	0.2	0.2	0.2	0.2	0.2	0.2
15	HPI-A	1.4E-4	2.4E-5	2.1E-4	2.4E-5	2.4E-5	1.0E-4
16	OA-HPI-MC[3]	0.09	0.09	0.09	0.09	0.09	0.09
17	HPI-MC	5.0E-3	5.0E-3	5.0E-3	0.0	5.0E-3	5.0E-3
18	PTS	0.05	0.05	0.05	0.05	0.05	0.05
19	PTS-CD[4]	1.0E-10, 1.0E-5 1.1E-8, 1.1E-5	1.0E-10, 1.0E-5 1.1E-8, 1.1E-5	1.0E-10, 1.0E-5 1.1E-8, 1.1E-5	1.0E-10, 1.0E-5 1.1E-8, 1.1E-5	1.0E-10, 1.0E-5 1.1E-8, 1.1E-5	1.0E-10, 1.0E-5 1.1E-8, 1.1E-5
20	PS/RV-C-RP	0.1 (liq.) 9.6E-3 (steam)	0.1 (liq.) 9.6E-3 (steam)	0.1 (liq.) 9.6E-3 (steam)	0.0	0.1 (liq.) 9.6E-3 (steam)	0.1 (liq.) 9.6E-3 (steam)
21	LTC	3.6E-4	1.4E-3	3.6E-4	1.4E-3	3.6E-4	4.3E-4

[1] Conditional probability of Event 11: = 0.5, given failure of Event 3; = 0.009, given success of Event 3, but, failure of Event 4 (0.045 is used for loss of ICS). However, for Davis-Besse only, 0.06 and 0.3 (loss of ICS) are used instead of 0.009 and 0.045.
[2] Conditional probability of Event 13: = 0.5, given failure of Event 11.
[3] Conditional probability of Event 16: = 0.4, given success of Event 12.
[4] This probability varies depending on the cause and severity of overcooling (a detailed explanation is given in the main text).

Note: For loss of ICS, the human error probabilities (i.e., Events 3, 6, 11, 13, and 16) are multiplied by a factor of 5. Also, for Oconee, Rancho Seco, and TMI-1, Event 8 (Secon. Int.) and Event 9 (Leak Size) are taken to be 0.3 and 0.67, respectively. In addition, parts of the vessel failure probabilities (Event 19) are changed (1.1E-8 to 4.5E-6, and 1.1E-5 to 6.4E-5).

Table 4.3
Grouping of Overcooling Transients Into Types

Overcooling Type No.	FW Automatically Controlled	FW Manually Controlled	No. of Stuck-Open Secondary Valves	OC Classification	Fractional Distribution of Temp. Bins T_1	T_2	T_3	T_4
1	Yes	Not needed	0	No Core Damage	No PTS Involved			
2	Yes	Not needed	1	Minor OC	No PTS Possible			
3	Yes	Not needed	2 (or more)	Severe OC	0.2	0.3	0.3	0.2
4	No	Yes	0	Minor OC	0.8	0.2	0	0
5	No	Yes	1	Minor OC	0.5	0.4	0.1	0
6	No	Yes	2 (or more)	Severe OC	0.1	0.3	0.3	0.3
7	No	No	0	Severe OC	0.2	0.4	0.2	0.2
8	No	No	1 or 2 (or more)	Severe OC	0	0.3	0.3	0.4

Table 4.4
Computed Conditional Probabilities of Vessel Failure

Overcooling Type No.	Vessel Failure Probability	Actual Values Used in Event Tree Quantifications
1	Not Applicable	None
2	Less than 10^{-10}	10^{-10}
3	10^{-5}	10^{-5}
4	$3x10^{-10}$	$1.1x10^{-8}$ for initiators other than loss of ICS
5	$3.3x10^{-7}$	
6	$1.5x10^{-5}$	$4.5x10^{-6}$ for loss of ICS
7	$9.7x10^{-6}$	$1.1x10^{-5}$ for initiators other than loss of ICS
8	$1.9x10^{-5}$	$6.4x10^{-5}$ for loss of ICS

Table 4.5
Interim HCR Correlation Parameters

Cognitive Processing Type	βi	$C\gamma i$	$C_\eta i$*
Skill	1.2	0.7	0.407
Rule	0.9	0.6	0.601
Knowledge	0.8	0.5	0.791

*Decimals carried on $C_\eta i$ to ensure that $P(t) = 1$ at $t = 0$.

Table 4.6
HCR Model Performance-Shaping Factors and Related Coefficients

	Coefficients
Operator Experience (K_1)	
1. Expert, well trained	-0.22
2. Average knowledge training	0.00
3. Novice, minimum training	0.44
Stress Level (K_2)	
1. Situation of grave emergency	0.44
2. Situation of potential emergency	0.28
3. Active, no emergency	0.00
4. Low activity, low vigilance	0.28
Quality of Operator/Plant Interface (K_3)	
1. Excellent	-0.22
2. Good	0.00
3. Fair	0.44
4. Poor	0.78
5. Extremely poor	0.92

Table 4.7
Criteria Used to Assess Performance-Shaping Factor (PSF) Levels

Performance Shaping Factor	Level	Criteria
K_1 Experience	1	Licensed with more than five years experience.
	2	Licensed with more than six months experience.
	3	Licensed with less than six months experience.
K_2 Interface	1	Same as 2, but with Advanced Operator Aids to help in accident situations.
	2	Displays carefully integrated with SPDS to help operator.
	3	Displays human engineered, but require operator to integrate information.
	4	Displays available, but not human engineered.
	5	Displays needed to alert operator not directly visible to operators.
K_3 Stress	1	Problem with vigilance, unexpected transient with no precursors.
	2	Optimal situation, crew carrying out small load adjustments.
	3	Mild stress situation, part-way through accident with high work load or equivalent.
	4	High stress situation, emergency with operator feeling threatened.

Table 4.8

Estimation of Baseline Human Error Probabilities

Operator Action	Behavior Type	Total Time Available	Action Time	Available Decision Time	Median Response Time	Stress Level	Quality of Oper./Plant Interface	Operator Experience	Adjusted Median Response Time	Baseline HEP
OA-HPI-M										
Overcooling Case	Skill	60	Negl.	60	N/A	Nominal	Good	Average	2.5	<1E-3
Undercooling (BF Case)[1]	Rule	40 (norm.)[2]	Negl.	40	7	Grave Emer.	Good	Expert	10	9E-3[3]
		50 (LOSP)	Negl.	50	7	Grave Emer.	Good	Expert	10	2E-3
OA-FW-MC	Skill	5	Negl.	5	N/A	Pot. Emer.	Fair	Average	3	6E-2
OA-HPI-MC										
Severe O.C. Case	Rule	10	Negl.	10	N/A	Pot. Emer.	Fair	Average	4.5	9E-2
Minor O.C. Case	Rule	5	Negl.	5	N/A	Pot. Emer.	Fair	Average	4.5	4E-1
OA-OCT	Knowledge	60	10	50	N/A	Pot.Emer.	Poor	Average	2.5	<1E-3
OA-EFW-R[1]										
Normal Case	Rule	20	5	15	3	Grave Emer.	Good	Expert	4.3	1E-2
LOSP Case	Rule	30	5	25	3	Grave Emer.	Good	Expert	4.3	<1E-3
PTS	Rule	20	Negl.	20	Unk.	Grave Emer.	Fair	Average		1E-2[4]

[1]Oconee simulator data exclusively used for these actions.
[2]25 minutes for Davis-Besse.
[3]6E-2 for Davis-Besse.
[4]1278 nominal model used.

Table 4.9
Observed Crew Response Times and Median Response Time Selection
(Time in Minutes Rounded to Nearest Half Minute)

OA-HPI-M (Overcooling Case)							
0.5	0.5	1.0	2.0	3.0	3.0	3.5	4.5
Median = 2.5							
OA-FW-MC							
2.0	3.0	4.0					
Median = 3.0							
OA-HPI-MC							
0.5	3.0	4.0	5.0	6.0	>8		
Median = 4.5							
OA-OCT							
1.5	2.0	2.5	5	6			
Median = 2.5							

Table 4.10
Observed Crew Response Times From Oconee Simulator Trials
(Times in Minutes Rounded to Nearest Half Minute)

OA-EFW-R			
1.5	3.0	3.0	3.0
Median = 3.0			
OA-HPI-M (Undercooling Case)			
6.0	7.0	8.0	
Median =7.0			
(Note: Fourth trial did not require bleed-and-feed action)			

Table 4.11
HEP Modification Factors for Support System Conditions

One ICS/NNI/VDC/VAC Bus Unavailable	HEP x 2
Total Loss of ICS	HEP x 5
Total Loss of VDC or VAC	0.5
Loss of Offsite Power, One EAC Bus Unavailable	HEP x 2
Station Blackout	0.5
Maximum Allowable Value After Modification	0.5

Only baseline HEPs are modified, conditional HEPs are never modified.
For action taken after recovery of support systems revert to nominal value.

Conditional HEPs

OA-HPI-M/OA-EFW-R (Undercooling Case)	0.5
OA-OCT/OA-HPI-M (Overcooling Case)	0.5

Table 4.12
Summary of Core Damage Frequency Attributable to Category C Events
(Per Reactor/Year)

(A) Oconee

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	4.7×10^{-7}	6.5×10^{-8}	5.3×10^{-7}	3.7×10^{-7}	3.0×10^{-8}	5.6×10^{-7}
Loss of MFW	1.2×10^{-7}	4.2×10^{-7}	5.5×10^{-7}	5.7×10^{-7}	7.4×10^{-9}	5.5×10^{-7}
Excessive FW	2.9×10^{-7}	2.7×10^{-9}	2.9×10^{-7}	2.6×10^{-7}	3.1×10^{-8}	3.2×10^{-7}
Loss of ICS power	2.3×10^{-6}	1.6×10^{-6}	3.9×10^{-6}	5.5×10^{-7}	6.2×10^{-7}	4.6×10^{-6}
Total	3.2×10^{-6}	2.1×10^{-6}	5.3×10^{-6}	1.8×10^{-6}	6.9×10^{-7}	6.0×10^{-6}

(B) Crystal River-3

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	4.2×10^{-6}	7.8×10^{-7}	5.0×10^{-6}	3.3×10^{-7}	5.3×10^{-8}	5.0×10^{-6}
Loss of MFW	2.8×10^{-6}	3.4×10^{-6}	6.2×10^{-6}	9.0×10^{-7}	3.3×10^{-8}	6.2×10^{-6}
Excessive FW	1.1×10^{-6}	2.5×10^{-8}	1.1×10^{-6}	9.2×10^{-8}	3.1×10^{-8}	1.2×10^{-6}
Loss of ICS power	3.1×10^{-6}	9.6×10^{-6}	1.3×10^{-5}	1.6×10^{-6}	1.4×10^{-7}	1.3×10^{-5}
Total	1.1×10^{-5}	1.4×10^{-5}	2.5×10^{-5}	2.9×10^{-6}	2.6×10^{-7}	2.5×10^{-5}

(C) Davis-Besse

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	5.3×10^{-9}	5.0×10^{-7}	5.1×10^{-7}	1.4×10^{-6}	5.8×10^{-8}	5.6×10^{-7}
Loss of MFW	3.2×10^{-8}	3.1×10^{-6}	3.1×10^{-6}	8.6×10^{-6}	1.4×10^{-8}	3.1×10^{-6}
Excessive FW	1.5×10^{-10}	1.4×10^{-8}	1.4×10^{-8}	1.0×10^{-6}	3.0×10^{-8}	4.4×10^{-8}
Loss of ICS power	2.8×10^{-8}	8.3×10^{-6}	8.3×10^{-6}	8.6×10^{-6}	1.1×10^{-7}	8.4×10^{-6}
Total	6.5×10^{-8}	1.2×10^{-5}	1.2×10^{-5}	2.0×10^{-5}	2.1×10^{-7}	1.2×10^{-5}

able 4.12 (Continued)

(D) ANO-1

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	5.5×10^{-7}	1.7×10^{-6}	2.2×10^{-6}	7.6×10^{-7}	2.6×10^{-8}	2.3×10^{-6}
Loss of MFW	9.3×10^{-7}	9.3×10^{-6}	1.0×10^{-5}	3.0×10^{-6}	3.8×10^{-8}	1.0×10^{-5}
Excessive FW	2.9×10^{-7}	1.1×10^{-7}	4.0×10^{-7}	3.2×10^{-7}	3.1×10^{-8}	4.3×10^{-7}
Loss of ICS power	6.4×10^{-7}	1.7×10^{-5}	1.7×10^{-5}	2.7×10^{-6}	1.1×10^{-7}	1.7×10^{-5}
Total	2.4×10^{-6}	2.8×10^{-5}	3.0×10^{-5}	6.8×10^{-6}	2.1×10^{-7}	3.0×10^{-5}

(E) Rancho Seco

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	4.9×10^{-7}	2.5×10^{-7}	7.5×10^{-7}	3.7×10^{-8}	2.4×10^{-8}	7.7×10^{-7}
Loss of MFW	6.2×10^{-7}	3.0×10^{-6}	3.6×10^{-6}	2.4×10^{-7}	2.8×10^{-8}	3.6×10^{-6}
Excessive FW	2.9×10^{-7}	1.8×10^{-8}	3.0×10^{-7}	2.6×10^{-8}	3.1×10^{-8}	3.4×10^{-7}
Loss of ICS power	3.1×10^{-6}	6.1×10^{-6}	9.2×10^{-6}	2.8×10^{-7}	1.2×10^{-6}	1.0×10^{-5}
Total	4.5×10^{-6}	9.4×10^{-6}	1.4×10^{-5}	5.8×10^{-7}	1.3×10^{-6}	1.5×10^{-5}

(F) TMI-1

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	3.4×10^{-7}	1.3×10^{-7}	4.8×10^{-7}	2.3×10^{-6}	5.9×10^{-9}	4.8×10^{-7}
Loss of MFW	3.0×10^{-8}	4.3×10^{-7}	4.6×10^{-7}	7.4×10^{-6}	4.5×10^{-10}	4.6×10^{-7}
Excessive FW	3.4×10^{-7}	5.3×10^{-9}	3.4×10^{-7}	3.9×10^{-6}	3.1×10^{-8}	3.7×10^{-7}
Loss of ICS power	2.3×10^{-7}	3.0×10^{-7}	5.2×10^{-7}	1.8×10^{-6}	4.9×10^{-8}	5.7×10^{-7}
Total	9.4×10^{-7}	8.7×10^{-7}	1.8×10^{-6}	1.5×10^{-5}	8.6×10^{-8}	1.9×10^{-6}

Table 4.13
Dominant Core Damage Sequences Attributable to Category C Events

Plant	Sequence Rank	Dominant Sequences	Sequence Frequency (per reactor year)	Initiating Event
Oconee	1	SEQ211	$2.2x10^{-6}$	Loss of ICS
	2	SEQ461	$1.1x10^{-6}$	Loss of ICS
CR-3	1	SEQ461	$7.2x10^{-6}$	Loss of ICS
	2	SEQ461	$2.5x10^{-6}$	Loss of MFW
	3	SEQ457	$2.4x10^{-6}$	Loss of ICS
	4	SEQ12	$1.4x10^{-6}$	Reactor/Turbine Trip
Davis-Besse	1	SEQ457	$4.8x10^{-6}$	Loss of ICS
	2	SEQ461	$2.1x10^{-6}$	Loss of ICS
	3	SEQ456	$1.6x10^{-6}$	Loss of MFW
	4	SEQ456	$1.1x10^{-6}$	Loss of ICS
ANO-1	1	SEQ461	$1.2x10^{-5}$	Loss of ICS
	2	SEQ461	$6.6x10^{-6}$	Loss of MFW
	3	SEQ457	$4.2x10^{-6}$	Loss of ICS
	4	SEQ457	$2.4x10^{-6}$	Loss of MFW
Rancho Seco	1	SEQ461	$4.6x10^{-6}$	Loss of ICS
	2	SEQ211	$2.6x10^{-6}$	Loss of ICS
	3	SEQ461	$2.2x10^{-6}$	Loss of MFW
	4	SEQ457	$1.6x10^{-6}$	Loss of ICS
TMI-1	1	SEQ456	$2.1x10^{-7}$	Loss of MFW
	2	SEQ211	$2.1x10^{-7}$	Loss of ICS
	3	SEQ12	$1.9x10^{-7}$	Reactor/Turbine Trip
	4	SEQ461	$1.8x10^{-7}$	Loss of ICS

Table 4.14
Description of Dominant Core Damage Sequences

Core Damage Sequence No.	Sequence Formula and Description of Sequence Scenario
SEQ12	$\overline{E1}*\overline{E5}*E8*\overline{E9}*\overline{E11}*\overline{E12}*E16*\overline{E18}*E20*E21$ Following reactor and turbine trip, MFW continues and is automatically controlled. A secondary valve sticks open, causing minor overcooling. HPI is manually actuated, but operator fails to recognize the need to control HPI flow. A SRV sticks open and long-term cooling also fails.
SEQ211	$E1*\overline{E2}*\overline{E5}*E8*E9*\overline{E15}*E16*\overline{E19}*E20*E21$ Due to loss of ICS, MFW is lost, but EFW is actuated and automatically controlled. Two or more secondary valves stick open, causing severe overcooling and automatic actuation of HPI. However, operator fails to recognize the need to control HPI flow. A SRV sticks open causing a small LOCA. Long-term cooling also fails.
SEQ456	$E1*E2*\overline{E3}*E4*\overline{E11}*E12$ Following reactor/turbine trip, both MFW and EFW are lost. Operator recognizes the need to recover EFW, but cannot recover it. Operator also recognizes the need to manually initiate feed and bleed, but the initiation fails.
SEQ457	$E1*E2*\overline{E3}*E4*E11$ Similar to SEQ456, except that operator fails to recognize the need to initiate feed and bleed cooling.
SEQ461	$E1*E2*E3*E11$ Following reactor/turbine trip, both MFW and EFW are lost. Operator fails to recognize the need to both recover EFW and manually initiate feed and bleed cooling.

Note: E1 and $\overline{E1}$ denote respectively failure and success of top event E1.

Table 4.15

Oconee: Distribution of Category "C" Events by Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Category "C" Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	2.9	1.83E-3	6.0E-4	0.175	0.0309	7.4E-6	0.208
Loss of MFW	0.72	4.5E-4	1.5E-4	0.0434	7.7E-3	4.8E-5	0.052
Excessive FW	0.12	0.0902	0.0298	0	0	3.1E-7	0.12
Loss of ICS power	0.67	3.2E-4	2.5E-4	0.066	0.135	4.5E-5	0.201
Total	4.41	0.093	0.031	0.284	0.174	1.01E-4	0.58

Table 4.16

Oconee: Distribution of Core Damage Frequency by Category "C" Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Core Damage Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	2.9	3.7E-9	2.7E-9	3.6E-7	1.3E-7	6.6E-8	5.6E-7
Loss of MFW	0.72	9.3E-10	6.7E-10	8.8E-8	3.3E-8	4.3E-7	5.5E-7
Excessive FW	0.12	1.8E-7	1.3E-7	0	0	2.7E-9	3.2E-7
Loss of ICS power	0.67	6.9E-10	1.1E-8	1.4E-7	2.8E-6	1.6E-6	4.6E-6
Total	4.41	1.9E-7	1.4E-7	5.9E-7	3.0E-6	2.1E-6	6.0E-6

Table 4.17
Crystal River 3: Distribution of Category "C" Events by Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Category "C" Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	3.7	0.111	0.037	0.322	0.015	1.1E-4	0.49
Loss of MFW	2.3	0.069	0.023	0.2	0.009	5.1E-4	0.30
Excessive FW	0.12	0.09	0.03	0	0	3.7E-6	0.12
Loss of ICS power	1.3	0.029	0.023	0.11	5.3E-3	2.9E-4	0.17
Total	7.4	0.3	0.113	0.632	0.0293	9.1E-4	1.08

Table 4.18
Crystal River 3: Distribution of Core Damage Frequency by Category "C" Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Core Damage Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	3.7	8.9E-7	5.2E-7	2.6E-6	2.2E-7	8.2E-7	5.0E-6
Loss of MFW	2.3	5.5E-7	3.3E-7	1.6E-6	1.3E-7	3.6E-6	6.2E-6
Excessive FW	0.12	7.2E-7	4.2E-7	0	0	2.7E-8	1.2E-6
Loss of ICS power	1.3	2.4E-7	1.6E-6	9.2E-7	3.6E-7	9.7E-6	1.3E-5
Total	7.4	2.4E-6	2.9E-6	5.1E-6	7.1E-7	1.4E-5	2.5E-5

Table 4.19

Davis Besse: Distribution of Category "C" Events by Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Category "C" Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	4.3	0.129	4.27E-2	0.375	1.76E-2	1.4E-5	0.56
Loss of MFW	1.0	0.03	9.9E-3	0.087	4.1E-3	8.5E-5	0.13
Excessive FW	0.12	0.09	0.03	0	0	3.9E-7	0.12
Loss of ICS power	1.0	0.022	0.018	0.087	4.1E-3	8.5E-5	0.13
Total	6.42	0.271	0.101	0.549	2.6E-2	1.8E-4	0.94

Table 4.20

Davis Besse: Distribution of Core Damage Frequency by Category "C" Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Core Damage Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	4.3	2.9E-11	4.2E-8	7.5E-13	1.6E-8	5.1E-7	5.6E-7
Loss of MFW	1.0	6.6E-12	9.8E-9	1.7E-13	3.7E-9	3.1E-6	3.1E-6
Excessive FW	0.12	2.0E-11	2.9E-8	0	0	1.4E-8	4.4E-8
Loss of ICS power	1.0	5.2E-12	8.7E-8	1.9E-13	1.8E-8	8.3E-6	8.4E-6
Total	6.42	6.1E-11	1.7E-7	1.1E-12	3.8E-8	1.2E-5	1.2E-5

Table 4.21
ANO-1: Distribution of Category "C" Events by Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Category "C" Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	1.8	0.0541	0.0179	0.157	7.4E-3	2.4E-4	0.236
Loss of MFW	2.7	0.0812	0.0268	0.235	0.0111	1.3E-3	0.355
Excessive FW	0.12	0.0902	0.0298	0	0	1.6E-5	0.12
Loss of ICS power	1.0	0.0224	0.0176	0.0871	4.1E-3	4.9E-4	0.132
Total	5.62	0.248	0.0921	0.479	0.0226	2.05E-3	0.84

Table 4.22
ANO-1: Distribution of Core Damage Frequency by Category "C" Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Core Damage Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	1.8	1.1E-7	7.9E-8	3.2E-7	3.2E-8	1.7E-6	2.3E-6
Loss of MFW	2.7	1.7E-7	1.2E-7	4.8E-7	4.8E-8	9.5E-6	1.0E-5
Excessive FW	0.12	1.8E-7	1.3E-7	0	0	1.1E-7	4.3E-7
Loss of ICS power	1.0	4.7E-8	3.7E-7	1.8E-7	8.5E-8	1.66E-5	1.7E-5
Total	5.62	5.1E-7	7.0E-7	9.8E-7	1.7E-7	2.8E-5	3.0E-5

Table 4.23
Rancho Seco: Distribution of Category "C" Events by Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling Minor	Excessive FW Overcooling Severe	Secondary Blowdown Overcooling Minor	Secondary Blowdown Overcooling Severe	Undercooling	Total Category "C" Frequency
Reactor/turbine trip	1.7	0.0511	0.0169	0.148	7.0E-3	3.7E-5	0.223
Loss of MFW	2.0	0.0601	0.0198	0.174	8.2E-3	4.4E-4	0.263
Excessive FW	0.12	0.0902	0.0298	0	0	2.6E-6	0.12
Loss of ICS power	0.83	0.0186	0.0146	0.0789	0.16	1.8E-4	0.272
Total	4.65	0.22	0.081	0.4	0.175	6.6E-4	0.88

Table 4.24
Rancho Seco:Distribution of Core Damage Frequency by Category "C" Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling Minor	Excessive FW Overcooling Severe	Secondary Blowdown Overcooling Minor	Secondary Blowdown Overcooling Severe	Undercooling	Total Core Damage Frequency
Reactor/turbine trip	1.7	1.1E-7	7.5E-8	3.0E-7	3.0E-8	2.6E-7	7.7E-7
Loss of MFW	2.0	1.2E-7	8.8E-8	3.6E-7	3.6E-8	3.0E-6	3.6E-6
Excessive FW	0.12	1.85E-7	1.3E-7	0	0	1.8E-8	3.4E-7
Loss of ICS power	0.83	4.1E-8	6.6E-7	1.6E-7	3.3E-6	6.2E-6	1.0E-5
Total	4.65	4.6E-7	9.5E-7	8.2E-7	3.4E-6	9.5E-6	1.5E-5

Table 4.25
TMI-1: Distribution of Category "C" Events by Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Category "C" Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	3.0	9.2E-5	3.1E-5	0.132	6.2E-3	9.8E-6	0.138
Loss of MFW	0.23	7.1E-6	2.3E-6	0.0101	4.8E-4	3.1E-5	0.0106
Excessive FW	0.12	0.0902	0.0298	0	0	3.9E-7	0.12
Loss of ICS power	0.054	1.2E-6	9.7E-7	5.3E-3	0.0109	7.3E-6	0.0162
Total	3.404	0.0903	0.0298	0.147	0.0176	4.9E-5	0.29

Table 4.26
TMI-1: Distribution of Core Damage Frequency by Category "C" Type (Per Reactor/Year)

Initiating Event	I. E. Frequency	Excessive FW Overcooling		Secondary Blowdown Overcooling		Undercooling	Total Core Damage Frequency
		Minor	Severe	Minor	Severe		
Reactor/turbine trip	3.0	2.2E-10	1.6E-10	3.2E-7	3.1E-8	1.3E-7	4.8E-7
Loss of MFW	0.23	1.7E-11	1.2E-11	2.4E-8	2.4E-9	4.3E-7	4.6E-7
Excessive FW	0.12	2.2E-7	1.5E-7	0	0	5.4E-9	3.7E-7
Loss of ICS power	0.054	3.3E-12	4.7E-11	1.4E-8	2.6E-7	3.0E-7	5.7E-7
Total	3.404	2.2E-7	1.5E-7	3.6E-7	2.9E-7	8.7E-7	1.9E-6

Table 4.27
Results of Sensitivity Study on
The Unavailability of EFW for Davis-Besse and TMI-1

(A) Davis-Besse (Unavailability of EFW = 1.6E-3 instead of 8.5E-5)

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	9.9×10^{-8}	9.4×10^{-6}	9.5×10^{-6}	1.4×10^{-6}	5.8×10^{-8}	9.6×10^{-6}
Loss of MFW	6.1×10^{-7}	5.8×10^{-5}	5.8×10^{-5}	8.6×10^{-6}	1.4×10^{-8}	5.8×10^{-5}
Excessive FW	2.8×10^{-9}	2.6×10^{-7}	2.7×10^{-7}	1.0×10^{-6}	3.0×10^{-8}	2.9×10^{-7}
Loss of ICS power	5.3×10^{-7}	1.6×10^{-4}	1.6×10^{-4}	8.6×10^{-6}	1.1×10^{-7}	1.6×10^{-4}
Total	1.2×10^{-6}	2.3×10^{-4}	2.3×10^{-4}	2.0×10^{-5}	2.1×10^{-7}	2.3×10^{-4}

(B) TMI-1 (Unavailability of EFW = 9.4E-4 instead of 1.36E-4)

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	3.5×10^{-7}	9.2×10^{-7}	1.3×10^{-6}	2.3×10^{-6}	5.9×10^{-9}	1.3×10^{-6}
Loss of MFW	5.3×10^{-8}	2.9×10^{-6}	3.0×10^{-6}	7.4×10^{-6}	4.5×10^{-10}	3.0×10^{-6}
Excessive FW	3.4×10^{-7}	3.7×10^{-8}	3.7×10^{-7}	3.9×10^{-6}	3.1×10^{-8}	4.0×10^{-7}
Loss of ICS power	2.3×10^{-7}	2.1×10^{-6}	2.3×10^{-6}	1.8×10^{-6}	4.9×10^{-8}	2.3×10^{-6}
Total	9.7×10^{-7}	6.0×10^{-6}	7.0×10^{-6}	1.5×10^{-5}	8.6×10^{-8}	7.0×10^{-6}

Table 4.28
Results of Sensitivity Study on
The Emergency Feedwater Nonrecovery Factor for Davis-Besse

(A) Davis-Besse (EFW nonrecovery factor = 0.5 instead of 0.2)

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	1.0×10^{-8}	1.1×10^{-6}	1.1×10^{-6}	1.4×10^{-6}	5.8×10^{-8}	1.2×10^{-6}
Loss of MFW	6.2×10^{-8}	6.9×10^{-6}	7.0×10^{-6}	8.6×10^{-6}	1.4×10^{-8}	7.0×10^{-6}
Excessive FW	2.8×10^{-10}	3.2×10^{-8}	3.2×10^{-8}	1.0×10^{-6}	3.0×10^{-8}	6.1×10^{-8}
Loss of ICS power	4.8×10^{-8}	1.7×10^{-5}	1.7×10^{-5}	8.6×10^{-6}	1.1×10^{-7}	1.7×10^{-5}
Total	1.2×10^{-7}	2.5×10^{-5}	2.5×10^{-5}	2.0×10^{-5}	2.1×10^{-7}	2.5×10^{-5}

(B) TMI-1 (EFW nonrecovery factor = 0.5 instead of 0.2)

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	3.4×10^{-7}	2.6×10^{-7}	6.0×10^{-7}	2.3×10^{-6}	5.9×10^{-9}	6.1×10^{-7}
Loss of MFW	3.3×10^{-8}	8.2×10^{-7}	8.6×10^{-7}	7.4×10^{-6}	4.5×10^{-10}	8.6×10^{-7}
Excessive FW	3.4×10^{-7}	1.0×10^{-8}	3.5×10^{-7}	3.9×10^{-6}	3.1×10^{-8}	3.8×10^{-7}
Loss of ICS power	2.3×10^{-7}	4.6×10^{-7}	6.9×10^{-7}	1.8×10^{-6}	4.9×10^{-8}	7.4×10^{-7}
Total	9.4×10^{-7}	1.6×10^{-6}	2.5×10^{-6}	1.5×10^{-5}	8.6×10^{-8}	2.6×10^{-6}

Table 4.29
Results of Sensitivity Studies on Human Reliability Analysis
for CR-3 and Davis-Besse

(A) CR-3

(1) Case 1

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	4.2×10^{-6}	2.2×10^{-6}	6.3×10^{-6}	3.3×10^{-7}	5.3×10^{-8}	6.4×10^{-6}
Loss of MFW	2.8×10^{-6}	9.6×10^{-6}	1.2×10^{-5}	9.0×10^{-7}	3.3×10^{-8}	1.2×10^{-5}
Excessive FW	1.1×10^{-6}	7.0×10^{-8}	1.2×10^{-6}	9.2×10^{-8}	3.1×10^{-8}	1.2×10^{-6}
Loss of ICS power	3.1×10^{-6}	2.6×10^{-5}	2.9×10^{-5}	1.6×10^{-6}	1.4×10^{-7}	3.0×10^{-5}
Total	1.1×10^{-5}	3.8×10^{-5}	4.9×10^{-5}	2.9×10^{-6}	2.6×10^{-7}	5.0×10^{-5}

(2) Case 2

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	4.2×10^{-6}	3.5×10^{-6}	7.7×10^{-6}	3.3×10^{-7}	5.3×10^{-8}	7.7×10^{-6}
Loss of MFW	2.8×10^{-6}	1.6×10^{-5}	1.8×10^{-5}	9.0×10^{-7}	3.3×10^{-8}	1.8×10^{-5}
Excessive FW	1.1×10^{-6}	1.1×10^{-7}	1.2×10^{-6}	9.2×10^{-8}	3.1×10^{-8}	1.3×10^{-6}
Loss of ICS power	3.1×10^{-6}	4.2×10^{-5}	4.5×10^{-5}	1.6×10^{-6}	1.4×10^{-7}	4.5×10^{-5}
Total	1.1×10^{-5}	6.2×10^{-5}	7.2×10^{-5}	2.9×10^{-6}	2.6×10^{-7}	7.2×10^{-5}

(3) Case 3

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	4.2×10^{-6}	5.1×10^{-6}	9.2×10^{-6}	3.3×10^{-7}	5.3×10^{-8}	9.3×10^{-6}
Loss of MFW	2.8×10^{-6}	2.2×10^{-5}	2.5×10^{-5}	9.0×10^{-7}	3.3×10^{-8}	2.5×10^{-5}
Excessive FW	1.1×10^{-6}	1.6×10^{-7}	1.3×10^{-6}	9.2×10^{-8}	3.1×10^{-8}	1.3×10^{-6}
Loss of ICS power	3.1×10^{-6}	5.9×10^{-5}	6.2×10^{-5}	1.6×10^{-6}	1.4×10^{-7}	6.3×10^{-5}
Total	1.1×10^{-5}	8.6×10^{-5}	9.8×10^{-5}	2.9×10^{-6}	2.6×10^{-7}	9.9×10^{-5}

Table 4.29 (Continued)

(B) Davis-Besse

(1) Case 1

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	5.2×10^{-9}	7.4×10^{-7}	7.5×10^{-7}	1.4×10^{-6}	5.8×10^{-8}	8.0×10^{-7}
Loss of MFW	3.2×10^{-8}	4.5×10^{-6}	4.6×10^{-6}	8.6×10^{-6}	1.4×10^{-8}	4.6×10^{-6}
Excessive FW	1.5×10^{-10}	2.1×10^{-8}	2.1×10^{-8}	1.0×10^{-6}	3.0×10^{-8}	5.0×10^{-8}
Loss of ICS power	2.7×10^{-8}	1.5×10^{-5}	1.5×10^{-5}	8.6×10^{-6}	1.1×10^{-7}	1.5×10^{-5}
Total	6.4×10^{-8}	2.0×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	2.1×10^{-7}	2.0×10^{-5}

(2) Case 2

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	5.3×10^{-9}	8.8×10^{-7}	8.9×10^{-7}	1.4×10^{-6}	5.8×10^{-8}	9.5×10^{-7}
Loss of MFW	3.3×10^{-8}	5.4×10^{-6}	5.4×10^{-6}	8.6×10^{-6}	1.4×10^{-8}	5.5×10^{-6}
Excessive FW	1.5×10^{-10}	2.5×10^{-8}	2.5×10^{-8}	1.0×10^{-6}	3.0×10^{-8}	5.4×10^{-8}
Loss of ICS power	3.1×10^{-8}	1.9×10^{-5}	1.9×10^{-5}	8.6×10^{-6}	1.1×10^{-7}	1.9×10^{-5}
Total	6.9×10^{-8}	2.5×10^{-5}	2.5×10^{-5}	2.0×10^{-5}	2.1×10^{-7}	2.6×10^{-5}

(3) Case 3

	SRV LOCA, Long-term Cooling Failure	Feed and Bleed	Subtotal	B&W Owners Group Report	PTS	Total
Reactor/turbine trip	5.0×10^{-9}	1.3×10^{-6}	1.3×10^{-6}	1.4×10^{-6}	5.8×10^{-8}	1.3×10^{-6}
Loss of MFW	3.1×10^{-8}	7.7×10^{-6}	7.7×10^{-6}	8.6×10^{-6}	1.4×10^{-8}	7.7×10^{-6}
Excessive FW	1.4×10^{-10}	3.5×10^{-8}	3.5×10^{-8}	1.0×10^{-6}	3.0×10^{-8}	6.5×10^{-8}
Loss of ICS power	3.5×10^{-8}	2.2×10^{-5}	2.2×10^{-5}	8.6×10^{-6}	1.1×10^{-7}	2.3×10^{-5}
Total	7.1×10^{-8}	3.1×10^{-5}	3.1×10^{-5}	2.0×10^{-5}	2.1×10^{-7}	3.2×10^{-5}

Table 4.30
Summary of Precursor Study Results

Plant	Date	Initiating Event	Maximum Conditional Core Damage Probability Attained
Davis-Besse	9/9/77	Loss of MFW	0.036
Rancho-Seco	3/20/78	Loss of ICS	0.034
CR-3	2/26/80	Loss of MFW	7.2E-3
ANO-1	4/7/80	Excessive FW	6.2E-6
CR-3	6/16/81	Excessive FW	1.1E-5
Rancho-Seco	6/17/81	Stuck Open Steam Valve	2.2E-6
Davis-Besse	3/2/84	Stuck Open Steam Valve	2.5E-7
Rancho-Seco (Phase I)	3/19/84	Reactor/Turbine Trip	2.2E-6
Rancho-Seco (Phase II)	3/19/84	Stuck Open Steam Valve	5.1E-6
Davis-Besse	6/9/85	Loss of MFW	0.036
Ranch-Seco	10/2/85	Stuck Open Steam Valve	2.2E-6
CR-3	10/9/85	Excessive FW	1.1E-5
Rancho-Seco	12/26/85	Loss of ICS	4.5E-5

5. CONCLUSIONS ON THE RISK PROFILE OF B&W PLANTS

This section consolidates the results presented in Section 4 and develops conclusions regarding the role played by Category C events in the overall perception of the risk profile of B&W plants.

5.1 Summary of B&W Risk Profile

Whether Category C events are significant to risk depends, in part, on plant-specific design features and the unavailabilities of certain systems, such as the HPI system (for feed-and-bleed cooling) or the emergency feedwater (EFW) system. The results of BNL event-tree analyses and precursor study indicate that, in general, Category C events involving undercooling are less likely to happen but are considerably more important from a risk perspective than those involving overcooling, which have much higher likelihood of occurrence. One fundamental reason for this is explained in the following.

With the PTS (pressurized thermal shock) sequences determined to be of minor importance to core-melt risk, the structure of the BNL event tree suggests that one other major way in which core damage could result from overcooling sequences is the occurrence of a stuck-open SRV, followed by failure of HPI and long-term cooling. [The HPI must be operating in order for the RCS repressurization (following depressurization due to overcooling) and the stuck-open SRV to take place.] For the sequences involving overcooling transient scenarios to end in core damage, the event tree shows that multiple failures of a relatively large number of top events must occur. This tends to lower their frequencies, and, hence, their contribution to the frequency of core damage frequency attributable to Category C events. For undercooling sequences, on the other hand, failure of a relatively small number of crucial top events suffices to bring about core damage. For example, if restoration of feedwater is unattainable within about 30 minutes from the onset of its total loss, failure to initiate feed-and-bleed cooling can immediately threaten the integrity of the core. The sequence involving failure of the operator to initiate HPI (feed-and-bleed) cooling was identified as the foremost contributor to the core-melt frequency among all the Category C event sequences. Although a stuck-open SRV LOCA induced by undercooling, with failure to actuate HPI, can also result in damage to the core, its contribution to the frequency of core damage due to Category C events was relatively small.

Regardless of whether the ultimate cause leading to core damage is failure to initiate HPI (feed-and-bleed) cooling, or a stuck-open SRV LOCA with failure to maintain HPI and long-term cooling, the availability and successful operation of the HPI system was found to play a prominent role in reducing the risk of core damage associated with Category C events. This conclusion is in general agreement with that in the B&WOG report.

Because of the crucial importance of the HPI system to core-melt risk, the unavailabilities of the HPI system must be estimated as accurately as possible in evaluating the risk significance of Category C events. In the BNL analysis of the event tree special care was taken to adequately model key factors, such as cognitive human errors involved in initiating the HPI (feed-and-bleed) cooling. However, despite the elaborate treatment of human errors, the core-damage frequencies attributable to Category C events for

Oconee and TMI-1 were relatively small compared to those computed for other B&W plants (see Section 4, Table 4.12). For Oconee, this may partly be due to the sound human reliability analysis performed in the Oconee PRA, which provided the basis for evaluating the branch-point probabilities used in the B&WOG reports. For TMI-1, the unavailability of feed-and-bleed cooling (0.034) used in the B&WOG analysis is appropriate. For Oconee and TMI-1, another important contributing factor to their relatively low frequency of core damage from Category C events is the low unavailabilities of their EFW systems (6.7E-5 for Oconee, 1.36E-4 for TMI-1), which are features of the plant-specific design. A low EFW unavailability was found to play a key role in suppressing the contribution to the total frequency of core damage attributable to Category C events. The data on branch-point probability in Table 3 show that the unavailabilities of FW-AC for Oconee and TMI-1 (8.4E-4 for Oconee and 4.1E-5 for TMI-1) are noticeably smaller than that for all other plants (0.04). Moreover, the unavailabilities of MFW and LTC (long-term cooling) also are comparatively small. All of these factors contribute to the relatively small core damage frequencies of Category C events for these two plants. For Oconee, this frequency (6.0E-6) amounts to roughly 11% of the total frequency of core damage due to internal events (5.4E-5 based on the Oconee PRA). Although no reliable information is available for TMI-1 on the total frequency due to internal events, the BNL estimate of the Category C events core-damage frequency (1.9E-6) is believed to contribute only a small fraction of it. For Oconee and TMI-1, therefore, Category C events appear to be relatively insignificant to risk.

In contrast the BNL analyses performed for CR-3, ANO-1, Davis-Besse, and Rancho Seco, with similar treatment of cognitive human errors, yielded significantly larger frequencies of core damage (due to Category C events). The result for CR-3, for example, shows that Category C events may have an impact on overall risk of core damage, because their contribution to the total frequency of core melt due to internal events is about 30%. The PRAs, for ANO-1, Davis-Besse, and Rancho Seco lacked detailed information on the total frequency of core damage (due to internal events). However, for each of these plants, the BNL result could represent a significant fraction of its total core-damage frequency. Thus, it can be reasonably concluded that Category C events are relatively significant for risk at CR-3, ANO-1, Davis-Besse, and Rancho Seco.

Since four of the six plants had no PRAs, it is desirable to identify the significance of Category C events to the risk profile of B&W plants in an absolute sense. That is, is the contribution to total frequency of core damage due to these events significant on its face, regardless of its relation to other contributors or to contributors at other types of PWRs? To assess this, it is necessary to compare the Category C contribution to some "significance scale." For this purpose, the ranking scale used in the Integrated Safety Assessment Program (ISAP) was adapted.

In ISAP, an absolute ranking scale was used to determine the significance of various safety/backfit issues. The issues were assigned priorities based on the assessment of their contribution of core damage frequency into four categories; High, Medium, Low, and Drop.[5-1] These four categories can be used directly as a measure of the significance of Category C events at B&W plants, as follows:

Significance Level	CD Frequency Range (per year)
High	CD $\geq$ 5E-5
Medium	5E-6 $\leq$ CD $<$ 5E-5
Low	5E-7 $\leq$ CD $<$ 5E-6
Insignificant	CD $<$ 5E-7

To determine the significance level of Category C events, it is necessary first to look at each type of Category C event and each plant separately, since straightforward conclusions about Category C events over all B&W plants are not obvious. Differences in significance between the plants are substantial.

The results were previously presented in a number of ways; by Category C type, by initiating event, and by contributor to core damage. These are defined as "event descriptors," and each will be individually ranked:

Category C Type:
- C1 - Excessive Feedwater (Minor)
- C2 - Excessive Feedwater (Severe)
- C3 - Secondary Blowdown (Minor)
- C4 - Secondary Blowdown (Severe)
- C5 - Undercooling

Initiating Event:
- T1 - Reactor/Turbine Trip
- T2 - Loss of Main Feedwater
- T3 - Excessive Feedwater
- T4 - Loss of ICS Function

Core Damage Contributor:
- D1 - Transient Induced LOCA
- D2 - No Cooling (Feed-and-Bleed Failure)
- D3 - Pressurized Thermal Shock

Each of these event descriptors was given a significance level for each plant as indicated by the results on core-damage frequency presented in the tables in the previous section. The results are shown in matrix form in Table 5.1. For example, C1 (Excessive Feedwater - Minor) was a low-significance contributor to CR3 (Crystal River 3) and ANO (Arkansas Nuclear One), and an insignificant contributor to the other four plants. Similarly, T4 (Loss of ICS Function) was a contributor of medium significance to CR3, DB (Davis-Besse), ANO, and RS (Rancho Seco) and a low significance contributor to the other two plants. No event descriptors had a high significance level for any plant.

To get an overall level of significance across the group of B&W plants, each event descriptor was given a score based on the following point allocation:

- 10 points for each plant ranked high.
- 5 points for each plant ranked medium.
- 1 point for each plant ranked low.
- 0 points for each plant ranked insignificant.

The maximum score is therefore 60 (all six plants ranked high for that descriptor) and the minimum score is 0. Looking at the matrix, C1 gets a significance score of 2 (two lows and four insignificants) and T4 gets a score of 22 (four mediums and 2 lows). The results, in Table 5.2, show that the most significant event descriptors are C5 (undercooling events), T4 (loss of ICS function initiators), and D2 (loss of cooling core damage sequences).

A similar ranking can be performed for each plant, summing the point totals for each plant across the 12 plant descriptors (Table 5.2). As the table shows, Category C events are the most significant at Crystal River 3, followed by Arkansas, Rancho Seco, and Davis-Besse, all of which have a Category C significance level in the medium range. Category C events are of minor significance at Oconee and Three Mile Island.

5.2 BNL Results vs. Previous B&W PRAs

In assessing the contribution to core damage from Category C events, we compared the results of this study with the comparable data from the PRAs for Oconee and Crystal River which have PRAs available.[5-2, 5-3] The results are shown on Table 5.3.

For Oconee, BNL has found that the PRA underestimated the contribution to core damage from 1) excessive feedwater, 2) loss of ICS, 3) transient-induced LOCA, and 4) PTS. The other Category C contributions are pretty close in the two studies. Of the four items noted, only the loss of ICS and transient-induced LOCAs are significant from the standpoint of effect on the overall results. Even so, the overall effect on the plant frequency of core damage is only about 10%, and the core-damage profile of the plant (the perception of what constitutes the plant vulnerabilities) is not altered.

For Crystal River, BNL found that the PRA significantly underestimated the contribution from 1) loss of main feedwater, 2) excessive feedwater, 3) loss of ICS, 4) transient-induced LOCA, 5) no cooling (loss of all cooling), and 6) PTS. All but the loss of PTS are significant from the standpoint of effect on the overall results. The overall effect on the plant frequency of core damage is about 40% (which is not very significant in a statistical sense) but the core-damage profile of the plant is altered by the identification of additional vulnerabilities which were previously thought to be of little or no concern.

5.3 Overall Conclusions

The major conclusions drawn from this study are:

1. As a class, B&W plants do not have core-damage risk which is measurably greater than other PWRs, although all the scenarios which are similar in nature to the historic Category C events are more likely to occur at B&W plants. This does not lead to the conclusion that B&W plants generically have higher risk of core damage. This result derives from the fact that there are other, more dominant, contributors to core damage which are not related to these Category C events nor to any other feature of the B&W-specific design, but which apply to all PWRs.

2. Overcooling events, which contribute over 99% of predicted Category C frequency, are minor contributors to core damage at all plants. Thus, Category C frequency is not a valid indicator of the risk of core damage.

3. The most significant contributor to core damage at all B&W plants is undercooling, which is the least likely Category C event to occur but the most likely to lead to core damage. The dominant cause of core damage in these events is failure to reestablish feedwater and failure to establish feed-and-bleed.

4. Among all the initiating events, loss of ICS function is the most significant contributor to core damage due to Category C events for all plants. This is due primarily to the confusion of the operator in dealing with events involving large-scale upset in instrumentation and also (for plants without MSIVs or with slow acting MSIVs) due to the increased probability of severe overcooling from large secondary blowdown.

5. Pressurized thermal shock (PTS) is more likely to occur at B&W plants than at other PWRs. However, PTS is still not a significant contributor to core damage at B&W plants.

6. With respect to the estimated overall CDF for each of six B&W reactor plants, Category C events comprise a measurable contribution to four of these plants (Arkansas Nuclear One, Crystal River, Davis Besse, and Rancho Seco). The Category C contribution to overall CDF at the other two plants (Oconee and Three Mile Island) is relatively insignificant. These two plants are less vulnerable because both have lower unavailability for their emergency feedwater systems and the automatic control of their feedwater systems. TMI also has lower frequencies of Category C initiators.

7. Excessive feedwater is not a significant contributor to core damage, either as an initiating event or as a subsequent failure.

References

5-1. Atefi et al., "Review of Risk Based Evaluation of Integrated Safety Assessment Program (ISAP) Issues for Millstone Unit 1," Science Applications International Corporation, December 1985.

5-2. Sugnet, W. et al., "Oconee PRA - A Probabilistic Risk Assessment of Oconee Unit 3," NSAC/60, Nuclear Safety Analysis Center, EPRI, Palo Alto, California and Duke Power Co., Charlotte, NC, June 1984.

5-3. Science Applications International Corporation, Probabilistic Risk Assessment for Crystal River Unit 3, February 1986 (Draft).

Table 5.1
Significance of Category C Events by Category C Type, Initiating Event, and Core Damage Contributor

Event Descriptor / Significance Level	C_1	C_2	C_3	C_4	C_5	T_1	T_2	T_3	T_4	D_1	D_2	D_3
(10) High $CD \geq 5E-5$												
(5) Medium $5E-6 \leq CD < 5E-5$			CR3		CR3 DB ANO RS	CR3	CR3 ANO		CR3 DB ANO RS	CR3	CR3 DB ANO RS	
(1) Low $5E-7 \leq CD < 5E-6$	CR3 ANO	CR3 ANO RS	OC3 ANO RS	OC3 CR3 RS	OC3 TMI	OC3 DB ANO RS	OC3 DB RS	CR3	OC3 TMI	OC3 ANO RS TMI	OC3 TMI	OC3
(0) Insignificant $CD < 5E-7$	OC3 DB RS TMI	OC3 DB TMI	DB TMI	DB ANO TMI		TMI	TMI	OC3 DB ANO RS TMI		DB		CR3 DB ANO RS TMI

Table 5.2
Category C Significance Scores for Each Category C Type, Initiating Event, and Core Damage Contributor

Descriptor	Score*
C_5 - Undercooling	22
T_4 - Loss of ICS	22
D_2 - No Cooling (Feed & Bleed)	22
T_2 - Loss of Main Feedwater	13
C_3 - Secondary Blowdown (Minor)	11
T_1 - Reactor/Turbine Trip	9
D_1 - Transient Induced LOCA	9
C_2 - Excessive FW (Severe)	3
C_4 - Secondary Blowdown (Severe)	3
C_1 - Excessive FW (Minor)	2
T_3 - Excessive FW	1
D_3 - Pressurized Thermal Shock	1

Category "C" Significance Scores for Each Plant

Plant Name	Score**
CR3 - Crystal River 3	39
ANO - Arkansas Nuclear One	24
RS - Rancho Seco	21
DB - Davis-Besse	17
OC3 - Oconee Unit 3	9
TMI - Three Miles Island	4

* 0-6 Low/Insignificant
7-30 Medium
30-60 High

** 0-12 Low/Insignificant
13-60 Medium
61-120 High

Table 5.3
Category "C" Event Core Damage Frequency -
BNL Study vs. PRA*

	Oconee		CR-3	
	BNL Study	PRA	BNL Study	PRA
Reactor/Turbine Trip	5.6E-7	1.8E-6	5.0E-6	4.6E-6
Loss of MFW	5.5E-7	1.6E-6	6.2E-6	2.3E-7
Excessive FW	3.2E-7	Negl.	1.2E-6	Negl.
Loss of ICS	4.6E-6	6.0E-8	1.3E-5	Negl.
Transient-Induced LOCA	3.2E-6	5.9E-7	1.1E-5	4.4E-7
No Cooling	2.1E-6	2.0E-6	1.4E-5	Negl.
PTS	6.9E-7	Not Avail.	2.6E-7	Not Avail.
Total CDF	5.8E-5**	5.4E-5	8E-5**	5.8E-5

*PRA results presented include, insofar as possible, only the CD contribution from events similar to those included in the BNL study.
**Estimated Total CDF from the applicable PRA modified by the differences noted in the BNL study.

6. CONCLUSIONS ON THE EFFECTIVENESS OF BWOG PLANNED BACKFITS

As part of the BWOG Safety and Performance Improvement Program (SPIP), the Owners Group recommended a number of plant modifications to reduce the frequency and severity of transients at B&W plants. These recommendations are documented in Appendix J of the SPIP report.[6-1] We reviewed these recommendations to identify those which would help to reduce the frequency of core damage from Category C events.

BNL also examined the potential risk-benefit achievable from selected recommendations, as discussed below. Since the details of any specific implementation of these BWOG recommendations are not known, no plant- specific benefit can be determined. Rather, the maximum risk impact was estimated, assuming that the recommendation was properly developed and implemented: this provides a perspective on the more effective areas for action and potential benefit.

Since the time of this evaluation, Appendix J has been updated to include additional recommendations approved by the BWOG and to identify recommendations that were rejected or superseded. These changes or the inclusion of additional recommendations as a result of recent updates to Appendix J were not factored into the evaluation.

6.1 Areas Covered by the BWOG Recommendations

The 154 BWOG recommendations first were reviewed and separated into twelve broad categories (Table 6.1 through 6.12). Each recommendation is described by its number and title (as assigned in the SPIP report). The categories used are:

Improvements Affecting ICS/NNI (Table 6.1)
Improvements Affecting Main Feedwater (Table 6.2)
Improvements Affecting Instrument Air (Table 6.3)
Improvements in Plant Operations (Table 6.4)
Improvements Affecting Main Steam (Table 6.5)
Improvements Affecting Plant Electrical Supply (Table 6.6)
Improvements Affecting Motor-Operated Valves (Table 6.7)
Improvements Affecting Plant Administration (Table 6.8)
Improvements Affecting the Main Turbine Systems (Table 6.9)
Improvements Affecting Main Steam/Feedwater Isolation (Table 6.10)
Improvements Affecting Emergency Feedwater (Table 6.11)
Improvements Affecting the Reactor Protection System (Table 6.12)

(More detailed technical information on any of these recommendations appears in Appendix J of the SPIP report.)

6.2 Potential of the BWOG Recommendations for Category C Core-Damage Reduction

The goal of this section is to identify the BWOG recommendations which will reduce the contribution of Category C events to the frequency of core damage for all the B&W plants evaluated (Section 5 discusses the present contribution and its significance). Based on the results presented in Section 4 and discussed further in Section 5, areas of improvement were identified which would most likely produce the necessary reduction. It is very important to note that only the effect of these improvements on the frequency of Category C core damage is evaluated in this section. Other benefits which may accrue to other parts of the risk profiles of these B&W plants were not within the scope of this study, nor could they be evaluated with the models or information compiled in this study.

The first area to be considered is failures in the integrated control system (ICS). These were shown to be the dominant contributor to core damage from Category C events in the B&W plants studied. Further, in looking at the causes of their dominance, it was clear that the desired reduction in core damage could not be achieved without improving the ICS area. To summarize the results, for the four plants where Category C events are significant contributors to frequency of core damage, the contribution from loss of ICS versus the total Category C contribution is:

Crystal River 3: 1.3E-5 out of 2.5E-5, or about 50%.
Davis Besse: 8.4E-6 out of 1.2E-5, or about 70%.
ANO-1: 1.7E-5 out of 3.0E-5, or about 55%.
Rancho Seco: 1.0E-5 out of 1.5E-5, or about 65%.

For the two plants where Category C events are not significant contributors to core-damage frequency, the results are:

Oconee: 4.6E-6 out of 6.0E-6, or about 75%.
Three Mile Island: 5.7E-7 out of 1.9E-6, or about 30%.

The reason for these results is two-fold: 1) ICS failure causes loss of main feedwater and may also induce other effects on the secondary systems which increase the severity of events and 2) ICS failure causes a substantial upset in the instruments which, even though backed up by safety-grade instrumentation, can result in a substantial amount of confusion for the operator, increasing the chance of error.

The BWOG recommendations list a large set of improvements in the ICS area (Table 6.1). Some of them, if properly implemented, would significantly reduce core damage from ICS upset events. In particular, implementation of numbers TR-001-ICS, TR-002-ICS, TR-004-ICS, TR-013-ICS, TR-038-ICS, TR-102-ICS, and TR-104-ICS would have the most significant effect on reducing the number of plant trips due to failures in the ICS (a significant reduction in frequency of initiating event). Recommendations TR-033-ICS and TR-036-ICS would reduce the severity of the effect of the loss on plant systems (such as main feedwater). Recommendations TR-154-ICS and, to a lesser extent, TR-012-ICS, TR-034-ADM, and TR-035-ADM could largely reduce the problem of the operator's confusion for the remaining occurrences of total system upset. In addition, TR-017-MFW (Table 6.2) would augment feedwater availability during loss of ICS, and TR-060-OPS (Table 6.4) would further enhance the operator's response during loss of ICS.

The combination of these effects would be such that the contribution from loss of ICS core damage presented above would become so small as to no longer be a dominant contributor to the Category C core-damage frequency. The maximum amount of the reduction would, for all intents and purposes, be equal to the total percentage contribution shown above for the present plant configurations. Therefore, that the new Category C frequencies would be:

Crystal River 3:	1.2E-5
Davis Besse:	3.6E-6
ANO-1:	1.3E-5
Rancho Seco:	5.0E-6
Oconee:	1.4E-6
Three Mile Island:	1.3E-6

Thus, in addition to Oconee and Three Mile Island, Davis Besse and Rancho Seco would have the contribution of Category C events to the frequency of core damage reduced to the level of not being significant by implementing these recommendations.

Following implementation of the ICS fixes, the next logical area to consider for possible improvement (based on our results) is in the reliability of feedwater and/or bleed-and-feed capability. We showed that failure of these functions was a dominant contributor to core damage from Category C events. For the two plants where Category C events would still be significant contributors to core-damage frequency, the contribution from total loss of feedwater and bleed-and-feed versus total Category C contribution, assuming the ICS fixes suggested are implemented in a manner to achieve minimum Category C frequency, is:

Crystal River 3:	4.2E-6 out of 1.2E-5, or about 35%.
ANO-1:	1.0E-5 out of 1.3E-5, or about 75%.

For the four plants where Category C events would now not be considered to be significant contributors to core damage frequency, the results are:

Davis Besse:	3.6E-6 out of 3.6E-6, or about 100%.
Rancho Seco:	3.3E-6 out of 5.0E-6, or about 65%.
Oconee:	4.9E-7 out of 1.4E-6, or about 35%.
Three Mile Island:	5.6E-7 out of 1.3E-6, or about 45%.

The reasons for this contribution to core damage are three-fold: 1) a high overall frequency of loss of main feedwater events, 2) relatively high probability of human error for failure to properly respond, and 3) a relatively low probability of recovering feedwater once it has failed. The contributions given above constitute the maximum possible reduction in core-damage frequency if it were possible to completely eliminate these total loss of cooling sequences.

The BWOG recommendations include several items which would enhance performance in the three areas mentioned above, most of which pertain to the main feedwater system (Table 6.2). The particular recommendations with the greatest potential to reduce the frequency of core damage are as follows (all are from Table 6.2 unless otherwise noted). For the high frequency of loss of main feedwater events, recommendations TR-014-MFW, TR-015-MFW, TR-066-MFW, TR-074-MFW, and TR-052/053/054-SFI (Table 6.10) should be particularly effective. For the low probability of feedwater recovery, recommendations TR-070-MFW and TR-071-MFW (for main feedwater recovery) and TR-055/056/057-ADM (Table 6.8, for emergency feedwater recovery) potentially are the most effective. Recommendations TR-018-MFW and TR-067-MFW would be effective for both the reduction of loss of main feedwater event frequency and the enhancement of main feedwater recoverability. In the area of operator response to total loss of feedwater, recommendations TR-064-OPS (Table 6.4) and TR-060-OPS (Table 6.4) would enhance the operator's response in recognizing the need to recover EFW and recognizing the need to initiate bleed-and-feed, respectively. In addition, the modification of Davis-Besse bleed-and-feed capability which was discussed with the BNL team, but does not appear on the list of BWOG recommendations, is deemed to be essential to enhance bleed-and-feed reliability at that plant.

The implementation of these recommendations potentially could result in a combined reduction of an order of magnitude in the frequency of core damage due to these total loss of cooling event sequences. After requantifying the model for the suggested fixes, the minimum frequency of Category C core damage for each plant is calculated to be:

Crystal River 3:	8.6E-6
Davis Besse:	5.0E-7
ANO-1:	3.0E-6
Rancho Seco:	1.8E-6
Oconee:	9.9E-7
Three Mile Island:	8.0E-7

Thus, Category C events would be reduced to insignificant contributors for all B&W plants (although, as previously presented, no reduction is required for Oconee and Three Mile Island and the ICS-related fixes would be sufficient for Davis Besse and Rancho Seco).

These conclusions reflect the maximum risk effectiveness of five selected BWOG recommendations. Details of implementation would determine how much of this benefit would be realized at a specific plant. For example, the design features of Davis Besse are such that the potential improvement cited above may not be fully achievable. At present, deficiencies in bleed-and-feed manifest themselves as a benefit in preventing transient-induced LOCAs. If bleed-and-feed capability is improved it is likely that some of this benefit will be lost to an increased frequency of core damage due to LOCA, although the extent cannot be quantified without more detailed information on the new Davis Besse bleed-and-feed capabilities. The final result would probably fall more in the 2E-6 range rather than 5E-7. However, these results indicate that many of the recommendations were well directed towards responding to the concerns about the operational and design features of B&W plants.

References

6-1. B&W Owners Group, Safety and Performance Improvement Program, Revision 01, BAW-1919, The Babcock and Wilcox Company, August 1986.

Table 6.1
Recommendations Made on the ICS/NNI System

Recommendation Number	Recommendation
TR-001-ICS	Replace RC flow signal input to ICS with equivalent signals based on RC pump status.
TR-002-ICS	Eliminate plant transients and trips due to a single failure of a T_{hot} and T_{cold} signal. Implement a modification to automatically detect invalid RC temperature input to ICS.
TR-003-ICS	Remove startup feedwater flow correction to MFW flow from the ICS.
TR-004-ICS	Implement a modification to automatically detect an invalid input to ICS of turbine header pressure.
TR-005-ICS	The auctioneering circuitry for the neutron flux signals should be removed from the RPS and relocated in the ICS.
TR-006-ICS	Delete FW temperature correction to FW demand function from ICS.
TR-007-ICS	For all plants with greater than 35°F superheat, remove the BTU limits as an active control function in the ICS. However, leave BTU limits alarm function active so that manual actions can be taken.
TR-008-ICS	To improve reactor runback capability, restore the high pressure reactor trip setpoint to 2355 psig and set the ULD setpoint for runback on loss of one MFWP to match the capacity of one MFWP.
TR-009-ICS	Improvement to ICS tune control circuits for reducing trip and transients due to loss of all MFW because of unacceptable tune control circuits.
TR-010-ICS	Incorporate ICS control circuit modification to reduce automatic-to-manual upsets.
TR-011-ICS	Determine if the grid frequency error circuit has been detuned to the extent that it is inoperable. This is to eliminate upset at full power due to ICS feature that is not needed.
TR-012-ICS	Review procedures, annunciators, indicators, alarms, etc., to determine if operator has necessary information to detect loss of all NNI power vs. loss of NNI-X power or NNI-Y power.

Table 6.1 (Continued)

Recommendation Number	Recommendation
TR-013-ICS	Install the necessary equipment to prevent loss of ± 24V power to the ICS or NNI due to the loss of a single power source.
TR-021-ICS	Identify the causes of and develop solutions to correct MFW pump control problems.
TR-032-ICS	Evaluate the restoration of ICS/NNI power and make appropriate changes to assure that the plant will remain in a safe state on restoration of power.
TR-033-ICS	Make appropriate changes to assure that plant will go to a known, safe state without any operator action required on loss of ICS/NNI power.
TR-034-ADM	Review training records to ensure that operators have had training on loss of ICS power.
TR-035-ADM	Familiarize operators with Rancho Seco event.
TR-036-ICS	Evaluate turbine bypass valve position on loss of ICS.
TR-037-ICS	Evaluate MFW pump speed control on loss of ICS power.
TR-038-ICS	Develop and implement a recommended preventive maintenance program for ICS/NNI.
TR-039-ICS	Wire the power supply monitors in the ICS/NNI cabinets directly to the output bus after the auctioneering diodes.
TR-102-ICS	Install redundant dc power supplies for NNI-Y (AP&L only).
TR-103-ICS	Fuse external power leaving the ICS/NNI cabinet assemblies (FPC only).
TR-104-ICS	Incorporate automatic selection of valid input signals for ICS/NNI.
TR-105-ICS	Each utility should perform a field verification of ICS/NNI drawings and update them accordingly.
TR-106-ICS	Unused hardware should be removed from the ICS/NNI cabinets to avoid exposing these systems to additional failure points.
TR-107-ICS	Improve ICS maintenance and tuning methods to correct post-trip MFW system control problems and develop a periodic surveillance/tuning program.

Table 6.1 (Continued)

Recommendation Number	Recommendation
TR-154-ICS	Provide the operator with unambiguous status of indicators and recorders in the main control room on loss of ICS/NNI power or signal.

Table 6.2
Recommendations Made on Main Feedwater Supply System

Recommendation Number	Recommendation
TR-014-MFW	Install a monitoring system in main FW pump trip circuitry to document the primary causes of MFWP trips.
TR-015-MFW	Determine if a low suction pressure trip is needed. Then decide what trip or response to low suction pressure should be implemented.
TR-016-MFW	Investigate response of oil system and pressure switches for evidence of abnormal pressure pulses and reliability of the pressure switches in the MFWP system.
TR-017-MFW	Evaluate the MFWP control systems and their interaction with the ICS system. Implement a program to identify improvements needed in both control systems.
TR-018-MFW	Provide training to operators and maintenance personnel and assure procedures are adequate for line-up, operation and maintenance of MFW system components.
TR-019-MFW	Assure there are sufficient annunciators and trip signals in the FW supply system.
TR-020-MFW	Ensure procedures have adequate instruction for switching MFW pump oil supply from auxiliary to shaft-drive and vice-verse.
TR-066-MFW	Check all MFW and condensate system protective circuits, interlocks and motors, etc., to ensure that a single electrical failure will not cause a loss of both feedwater trains.
TR-067-MFW	Evaluate the setpoints and functions of the automatic MFW pump trip features. Wherever possible, eliminate these trip functions altogether.
TR-068-MFW	Develop a post-maintenance testing program for MFW pump turbines and governor controls to check each feed pump unit for operability in manual and automatic modes.
TR-069-MFW	Eliminate automatic control of the MFW block valve except following a reactor trip, in which case, it should still close automatically.
TR-070-MFW	Provide capability to override a close signal to the MFW block valve to enable CRO to stop the block valve at any intermediate position during valve closure.

Table 6.2 (Continued)

Recommendation Number	Recommendation
TR-071-MFW	Install valve position indication for the startup and MFW regulating valves (and low load control valves at applicable plants).
TR-072-MFW	Eliminate the transfer from the startup to main FW flowmeter when the main FW block valve opens. Continuously use the main FW flow meter.
TR-073-MFW	Eliminate high MFW pump discharge pressure trips as a common occurrence.
TR-074-MFW	Schedule I&C calibration and inspection work so as to minimize the number of times the main FW pumps and turbines instrumentation and control equipment is disturbed during power operation.
TR-075-MFW	Modify control scheme for heater drain pump recirculation control valves to reduce or eliminate their occasional erratic shifting between maintaining minimum flow requirements and tank level (for ANO-1 only).
TR-076-MFW	Eliminate automatic trip of the "preferred" MFW pump after a reactor trip (for ANO-1 only).
TR-077-MFW	Review and upgrade preventive maintenance on auxiliary boilers.
TR-078-MFW	Add an indicator to the control room apron near the MFW pump controls for MFW pump discharge pressure.
TR-079-MFW	Put MFW regulating valves, main block valves, and startup control valves and all of the operators for these valves on a refueling frequency for an operational check.
TR-080-MFW	Assess the feasibility of adding instrumentation to permit CRO to determine the performance of MFWPT shaft driver oil pump without having to secure the running auxiliary oil pump.
TR-081-MFW	Move control room MFW flow indication from the back panel to the apron (for Rancho Seco only).
TR-082-MFW	Evaluate the need to add or enhance the functional capability to automatically bypass Powdex (or condensate demineralizer) units on high differential pressure.
TR-83-MFW	Add MFWPT lube oil purifiers.

Table 6.2 (Continued)

Recommendation Number	Recommendation
TR-84-MFW	Correct the problem with feed pump turbine shaft sealing to reduce water induction into turbine oil sump.
TR-085-MFW	Modify or repair as appropriate the main feed pump recirculation valve/controller for automatic control during startup and shutdown.
TR-086-MFW	Find a suitable resolution for the problem of the first stage FW heaters not properly draining.
TR-087-MFW	Determine need for adding capability for flushing the feed pump turbine governor control oil system.
TR-088-MFW	Eliminate automatic plant runback on low MFW pump discharge pressure or establish a setpoint that will offer some chance of a successful runback.
TR-089-MFW	Eliminate potential for physical damage by vibration, excessive loading due to improper installation or personnel abuse, etc., to condensate and MFW pneumatic valve operator air supply lines.
TR-090-MFW	Add valve position indicator in control room for the inlet control/isolation valves to Deaerator FW tank (for Davis-Besse only).
TR-091-MFW	Develop a long term solution to eliminate the need for an auxiliary operator to open a DFT drain line after reactor trips (for Davis-Besse only).
TR-092-MFW	Assess the cause of frequent feed booster pump low suction pressure alarms during plant startup and implement correct action to eliminate this (for Davis-Besse only).
TR-093-MFW	Assess alternatives and implement a modification to allow full power operation using only two hotwell pumps (for Oconee units only).
TR-094-MFW	Assess alternatives and implement a modification to reduce the effects of flashing of 4th stage FW heater drains to the DFT (for Davis-Besse only).
TR-095-MFW	Clean/flush the condensate pump motor coolers supplied by the turbine building cooling water system (for Davis-Besse only).
TR-098-MFW	The MFW system design should include operational, automatic overfill protection to prevent loss of heat sink or water inventory in the main steam lines.

Table 6.3
Recommendations Made on Instrument Air System

Recommendation Number	Recommendation
TR-120-IAS	For critical air-operated valves, check O-rings and other seals within pneumatic components.
TR-121-IAS	Make appropriate personnel aware of importance of instrument air system, prohibition of use for tools and need to immediately report air system damage.
TR-122-IAS	Instrument air system should be systematically inspected for leaks.
TR-123-IAS	For instrument air systems, provisions should be made to protect against failures possible with dessicant type driers.
TR-124-IAS	Identify and inspect instrument air system metal lines with high vibration, and when cracks are found, replace with reinforced flexible tubing.
TR-125-IAS	Operability testing of critical air-operated valves should be performed in the preventive maintenance program and, compare with design basis stroking times.
TR-126-IAS	Each utility should compare their instrument air system configuration with the functional target criteria.
TR-127-IAS	For instrument air system, each utility should review its preventive maintenance programs, identifying parameters for trending to help determine periodic maintenance.
TR-128-IAS	For instrument air system, each utility should review its training and loss-of-air response procedures and make appropriate changes as needed.
TR-129-IAS	Install and automatic bypass line around both the drier and the filters in the ANO-1 instrument air system.
TR-130-IAS	Expand procedure for the loss of instrument air to include the information described in Section 4 of the source document (for ANO-1 only).
TR-131-IAS	Investigate feasibility of routing instrument air compressor intakes to the exterior (for Oconee units only).
TR-132-IAS	Add an after drier to the instrument air line to reduce the dew point below - 20°F (for Oconee units only).

Table 6.3 (Continued)

Recommendation Number	Recommendation
TR-133-IAS	Add a filtration system downstream of the last drier in the Oconee units instrument air system.
TR-134-IAS	Install control room-operated isolation valves with manual bypass at the key line feeding each unit's auxiliary building instrument air system header (for Oconee units only).
TR-135-IAS	Install automatic isolation valves that could limit the extent to which the instrument air system is affected from air leaks (for Oconee units and CR-3).
TR-136-IAS	A dew point monitor should be installed downstream of the instrument air system driers (for Duke, FPC and TED only).
TR-137-IAS	Check accumulators in instrument air system for water buildup. Install drain valve on the bottom of accumulators to allow blowdown of water where necessary (for all operating plants except ANO-1).
TR-138-IAS	Install a check valve after each compressor aftercooler in the instrument air system (for DPC and FPC only).
TR-139-IAS	Install on/off status and remote start of instrument and air compressors (in the instrument air system) in the control rooms (for DPC and FPC only).
TR-140-IAS	Assign high maintenance priority to an out-of-service air compressor and maintain sufficient spare parts to repair a compressor within a week (for Oconee units only).
TR-141-IAS	Install an automatic bypass valve to bypass both the drier and filters upon low instrument air header pressure (for FPC, Supply System and TVA).
TR-142-IAS	The components of instrument air system should be designed to withstand maximum flow generated by all the compressors until manual bypass valves are used (for FPC only).
TR-143-IAS	Inspect accumulators and their check valves in the instrument air system (for FPC, GPUN, SMUD, and TED only).
TR-144-IAS	Develop or upgrade a loss of instrument air procedure. The operators and site personnel should be trained on the proper use of the procedure (for FPC, SMUD and TVA only).
TR-145-IAS	Install automatic isolation valves at several points within the instrument air lines at CR-3 plant.

Table 6.3 (Continued)

Recommendation Number	Recommendation
TR-146-IAS	Loss-of-air procedure for instrument air system should note importance of quickly bypassing driers and filters when excessive flow rates are experienced (for TMI-1 only).
TR-147-IAS	Normally closed positions are recommended for IA-V12 and SA-V12 at TMI-1.
TR-148-IAS	Install automatic isolation valves at specified points within the instrument air lines at Rancho-Seco.
TR-149-IAS	Instrument air system components should be designed to withstand maximum flow generated by all compressors for the period that it would take for manual bypass to be used (for TED only).
TR-150-IAS	The ESFAS signal to close specified motor control valves and isolate service and control air should be eliminated (for TVA only).
TR-151-IAS	Eliminate apparent inconsistencies in instrument-air valve designations on various drawings (for TVA only).
TR-152-IAS	Establish same run time for the various compressors in the instrument-air system (for TVA only).
TR-153-IAS	A plant specific air-system failure evaluation should be made to ensure that air-system failures will not affect the ability to control the plant during an air outage.

Table 6.4
Recommendations Made on Plant Operations

Recommendation Number	Recommendation
TR-026-OPS	Operability of SG shell thermocouples should be verified during every refueling outage and periodically during plant operation.
TR-051-OPS	Conduct post-maintenance and surveillance PORV testing which should include an in-service functional test.
TR-058-OPS	Use highest emergency classification level when making initial notification to NRC.
TR-059-OPS	Personnel who make emergency notifications should be trained to assure that they are familiar with the type of information which must be provided.
TR-060-OPS	Stress in operator training that emergency operating procedures are to be followed explicitly, even when such procedures are considered as drastic actions.
TR-061-OPS	Establish a means of systematically identifying high priority operator tasks requiring specific short-term training.
TR-062-OPS	Maintain a high SPDS (Safety Parameter Display System) availability by corrective and preventive maintenance.
TR-063-OPS	Ensure that P/T graphs are provided in the control room. Provide procedural guidance on making P/T plots when the SPDS is not available.
TR-064-OPS	Operator training to reset turbine-driven EFW pumps after overspeed trips should be part of formal training programs and should include hand-on training.
TR-065-OPS	Improve communication between the control room and certain plant areas at Rancho Seco.
TR-099-OPS	Include guidance on excessive MFW, throttling AFW and throttling HPI in plant procedures.

Table 6.5
Recommendations Made on Main Steam System

Recommendation Number	Recommendation
TR-023-MSS	Determine need to replace MSSV release nut cotter pins on all MSSVs.
TR-024-MSS	Determine causes to correct anomalous post-trip performance of MSSVs.
TR-048-MSS	Review turbine bypass and atmospheric dump valve preventive maintenance programs and revise as necessary.
TR-049-MSS	Review and revise steam trap preventive maintenance programs.
TR-050-MSS	Include in plant operating procedures provisions for opening steam trap bypass valves during startup of MSS and draining turbine bypass header valves prior to startup or cooldown.
TR-096-MSS	Evaluate design of turbine bypass and atmospheric dump systems.
TR-108-MSS	Investigate using maximum allowable set pressure for the lowest set MSSVs (for TMI-1 only).
TR-109-MSS	Ensure that relief valves that are not automatically isolated from main steam system post trip are in a preventive maintenance and surveillance test program.
TR-110-MSS	Davis-Besse should provide continuous EFW flow as a function of level.

Table 6.6
Recommendations Made on Plant Electrical System

Recommendation Number	Recommendation
TR-112-PES	Review switchyard maintenance procedures to assure there is no mechanism for loss of offsite power.
TR-113-PES	Review breaker control power distribution to determine effects of a loss of the battery bus.
TR-114-PES	Evaluate hardware to assure diesel generators cannot be synchronized to the grid out of phase.
TR-115-PES	Test diesel generators to assure they will carry loads under expected sequential loading conditions.
TR-116-PES	Review dc changing system and assure the charging voltage does not exceed the voltage rating of plant equipment.
TR-117-PES	Modify inverter overcurrent protection to ensure the breakers/fuses open on overcurrent before the inverters fail.
TR-118-PES	Evaluate loadings on ac and dc vital buses to assure adequate margin exists for normal fluctuations in voltage or frequency without trip of equipment.
TR-119-PES	Implement preventive maintenance procedures for electrical buses.

Table 6.7
Recommendations Made on Motor-Operated Valves

Recommendation Number	Recommendation
TR-041-MOV	Confirm by field inspection all design data required to size operators and valves for all safety related motor-operated valves.
TR-042-MOV	Obtain analytical methods used by the valve and operator vendors.
TR-043-MOV	For all safety-related valves, assure that torque switch bypass limit switch is set to open after valve is unseated.
TR-044-MOV	For all safety-related wedge seating valves, position open direction torque switches to the highest allowable setpoints.
TR-045-MOV	Ensure that maintenance procedures provide proper instructions for setting torque switches and bypass limit switches.
TR-046-MOV	All safety-related motor-operated valves should be challenged to open and close under differential pressures which simulate worst operational and accident conditions.
TR-047-MOV	Institute formal training programs on motor-operated valves.

Table 6.8
Recommendations Made on Plant Administration

Recommendation Number	Recommendation
TR-027-ADM	Ensure that calibration techniques for power range imbalance are in accordance with B&W site instructions.
TR-028-ADM	Include training on power/imbalance control during transient Xenon conditions in the operator training program.
TR-029-ADM	Ensure that TAP reports include specific information regarding events where human errors occur.
TR-040-ADM	Use the Transient Assessment Committee's Trip Investigation/ Root Cause Determination Program.
TR-055-ADM	Coordinate activities of plant operations, security and radcon (health physics) personnel to facilitate timely access to critical equipment.
TR-056-ADM	Move chain link fences as necessary to provide better access to critical components.
TR-057-ADM	Consider ways to improve access to critical components where problems have been identified with gaining access to critical components because of Appendix R fire barriers.

Table 6.9
Recommendations Made on Main Turbine System

Recommendation Number	Recommendation
TR-025-MTS	Review EHC system for loss of input power.
TR-030-MTS	Raise ART (anticipatory reactor trip) on turbine trip arming point from current rating of 20% power to a higher level.
TR-100-MTS	Review MSR draw tank level control system and drain line configuration for reliability improvements.
TR-101-MTS	Operator training should include emphasis on generator excitation, voltage control and operation and testing of overspeed protection controller and governor valve speed limiter.

Table 6.10
Recommendations Made on Main Steam/Feedwater Isolation System

Recommendation Number	Recommendation
TR-052-SFI	AP&L, GPUN, and SMUD need to filter their steam generator level signals in the Steam Feedwater Rupture Control System (SFRCS).
TR-053-SFI	Correct overheating problems that can lead to electric power supply malfunctions and also correct problems caused by degraded voltage power supplies (AP&L, GPUN, and SMUD).
TR-054-SFI	Redesign MSIV pneumatic hardware to assure this equipment is exercised during surveillance testing (AP&L only).

Table 6.11
Recommendations Made on Emergency Feedwater System

Recommendation Number	Recommendation
TR-022-EFW	Review EFIC system to ensure that adequate margin exists between the low SG level setpoint for normal control and the low SG level setpoint for EFW actuation.
TR-097-EFW	Evaluate the design of the EFW flow control valves.

Table 6.12
Recommendations Made on Reactor Protection System

Recommendation Number	Recommendation
TR-031-RPS	Increase setpoint for high pressure reactor trip from current valve of 2300 psig to 2355 psig.
TR-111-RPS	Review safety systems (RPS, ESFAS, EFW) surveillance procedures for checking which channel is available for testing prior to initiation of test.

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13. ABSTRACT (200 words or less)

This report summarizes a study performed by Brookhaven National Laboratory for the Office of Nuclear Reactor Regulation, Division of Engineering & System Technology (A/D for Systems), U.S. Nuclear Regulatory Commission. This study was requested by the NRC to assist their staff in assessing the risk significance of features of the Babcock & Wilcox (B&W) reactor plant design in the light of recent operational events. This study focuses on a critical review of submissions from the B&W Owners Group (BWOG) and as an independent assessment of the risk significance of "Category C" events at each operating B&W reactor. Category C events are those in which system conditions reach limits which require significant safety system and timely operator response to mitigate. A precursor study for each of the major B&W historical Category C events also was carried out. In addition, selected PRAs for B&W reactor plants and plants with other pressurized water reactor (PWR) designs were reviewed to appraise their handling of Category C events, thereby establishing a comparison between the risk profiles of B&W reactor plants and those of other PWR designs. The effectiveness of BWOG recommendations set forth in Appendix J of the BWOG SPIP (Safety and Performance Improvement Program) report (BAW-1919) also was evaluated.

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