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Precursors to Potential Severe Core Damage Accidents: 1982-83 A Status Report

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Abstract

This study is a continuation of earlier work that evaluated 1969–1981 and 1984–1994 events affecting commercial light-water reactors. One-hundred nine operational events that affected 51 reactors during 1982 and 1983 and that are considered to be precursors to potential severe core damage are described. All these events had conditional probabilities of subsequent severe core damage greater than or equal to 1.0×10^{-6}. These events were identified by first computer screening the 1982-83 licensee event reports from commercial light-water reactors to select events that could be precursors to core damage. Candidates underwent engineering evaluation that identified, analyzed, and documented the precursors. This report discusses the general rationale for the study, the selection and documentation of events as precursors, and the estimation of conditional probabilities of subsequent severe core damage for the events.

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Foreword

This report provides the results of the review and evaluation of 1982 and 1983 operational experience data by the Nuclear Regulatory Commission's ongoing Accident Sequence Precursor (ASP) Program. The ASP Program provides a safety significance perspective of nuclear plant operational experience. The program uses probabilistic risk assessment (PRA) techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include initiators, degradations of plant conditions, and safety equipment failures that could increase the probability of postulated accident sequences.

The primary objective of the ASP Program is to systematically evaluate U.S. nuclear plant operating experience to identify, document, and rank those operating events which were the most significant in terms of the potential for inadequate core cooling and core damage. In addition, the program has the following secondary objectives: (1) to categorize the precursor events for plant specific and generic implications, (2) to provide a measure which can be used to trend nuclear plant core damage risk, and (3) to provide a partial check on PRA-predicted dominant core damage scenarios.

The review and analysis of 1982 and 1983 licensee event reports (LERs) started in October 1994. The analyses documented in this report were performed primarily for historical purposes to obtain the two years of precursor data for the NRC's ASP Program which had previously been missing. In this effort, more than 2100 LERs were reviewed for potential precursors, and 435 LERs were selected for further detailed analysis. Once the draft report had been completed, it was sent to the respective licensees, and a number of them provided comments. In addition, each of the analyses in the report received an independent review by an NRC contractor. All of the comments received were evaluated for reasonableness and pertinence to the ASP analysis, and the conditional core damage probability calculations were revised where appropriate. This report documents the 109 precursors that were identified through this process - 54 for 1982 and 55 for 1983.

The most important precursor events for 1982-83 include: failures of the automatic reactor trip capability at both PWR units at a plant, a steam generator tube rupture at a PWR with subsequent failure of a pressurizer power-operated relief valve to close, problems with residual heat removal and/or residual heat removal service water systems at two BWRs, loss of offsite power-related events with a degraded emergency power system at two BWRs, a reactor trip with an inoperable auxiliary feedwater pump and an inoperable pressurizer power-operated relief valve at a PWR, and a loss of offsite power event with an inoperable turbine-driven auxiliary feedwater pump at a PWR.

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1. Introduction

The Accident Sequence Precursor (ASP) Program involves the review of licensee event reports (LERs) for operational events that have occurred at light-water reactors (LWRs). The program identifies and categorizes precursors to potential severe core damage accident sequences. The results of this work have been documented in NUREG/CR-2497, *Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report,*[1] in NUREG/CR-3591, *Precursors to Potential Severe Core Damage Accidents: 1980–1981, A Status Report,*[2] and in multiple volumes of NUREG/CR-4674,[3-13] which provides the results of the ASP Program for the years 1984 through 1994. Owing to changes in responsibility for the ASP Program within the U. S. Nuclear Regulatory Commission (NRC) during the 1984-85 time frame and because of demands for timely completion of the ASP analyses for 1984 and 1985, analysis of the 1982 and 1983 events was postponed. ASP analysis of the 1982-83 events was initiated in 1994 and the present report details the review and evaluation of operational events for those years, thereby completing ASP analysis and documentation for all events from 1969 to the present.

The requirements for reporting operating experience in LERs in the 1982-83 period are described in Regulatory Guide 1.16 (U.S. Nuclear Regulatory Commission, Regulatory Guide 1.16, Rev. 4, *Reporting of Operating Information, Appendix A: Technical Specifications,* August 1975).[14] These reporting requirements are different than current requirements (described in NUREG-1022, *Licensee Event Report System, Description of System and Guidelines for Reporting*[15-17]). The most important difference relative to the ASP analyses was that the pre-1984 LERs were not required to link plant trip information to reportable events. Plant trip information is important from an ASP perspective because one of the categories of events traditionally analyzed in the program includes plant trips with degraded safety systems. In order to be able to analyze these events without explicit information regarding the relationship between a trip and potentially unavailable equipment, assumptions had to be made about the relationship in some cases. Precursor events of this nature were termed "windowed" events and the specific assumptions associated with these events and their influence on ASP modeling are discussed in more detail in Chapter 2. It should be noted that because such assumptions were not required until 1984, the overall results of the 1982-83 ASP analyses may not be directly comparable with those from more recent years.

1.1 Background

The ASP Program owes its genesis to the Risk Assessment Review Group,[18] which concluded that "unidentified event sequences significant to risk might contribute... a small increment...[to the overall risk]." The report continues, "It is important, in our view, that potentially significant [accident] sequences, and precursors, as they occur, be subjected to the kind of analysis contained in WASH-1400."[19] Evaluations done for the 1969–1981 period were the first efforts at this type of analysis.

This study focuses on accident sequences in which, if additional failures had occurred, inadequate core cooling would have resulted and, as a consequence, could have caused severe core damage. Events considered to be potential precursors are analyzed, and a conditional probability for subsequent core damage is calculated. This is done by mapping failures observed during the event onto ASP accident sequence models. Those events with conditional probabilities of subsequent severe core damage $\geq 1.0 \times 10^{-6}$ are identified and documented as precursors. Detailed documentation is provided for all precursors with conditional probabilities $\geq 1.0 \times 10^{-5}$. However, the relatively large total number of precursors identified for the 1982-83 time frame meant that precursors with conditional probabilities between 1.0×10^{-5} and 1.0×10^{-6} could only be summarized.

1.2 Current Process

The current process for identifying, analyzing, and documenting precursors is described in detail in Chapter 2 and Appendix A. Each documented precursor analysis received an independent review by an NRC contractor (Oak Ridge National Laboratory, ORNL) and revisions were made as necessary to address comments generated by that review. A draft of the report was also provided to the affected licensees.

In addition to the events selected as accident sequence precursors, those involving (1) loss of containment function, (2) unusual failure modes or initiators, and (3) events that are impractical to analyze were identified. These events are also documented in this report.

The primary source of the 1982-83 operational event information is the NRC's sequence coding and search system (SCSS) data base. The data base contained approximately 4,400 LERs for 1982 and 4,800 for 1983. The ASP computer search algorithm used with SCSS selected 2,064 of these for engineering review as potential precursors. In addition, relevant volumes of NUREG-0090 (*Report to Congress on Abnormal Occurrences*)[20] and issues of *Nuclear Safety* were reviewed to identify other potentially interesting events. As a result of the engineering review process, 414 LERs were determined to be candidate precursors. An additional 21 LERs were examined because of potential relationships with events selected as candidate precursors. From these 435 LERs, 115 potential precursor events were identified (some involving multiple LERs). Of these 115 events, six were rejected after detailed reviews, leaving a total of 109 precursor events. Fifty-three of the events were analyzed and documented in detail (conditional core damage probabilities $\geq 1.0 \times 10^{-5}$) and 56 were analyzed and summarized (conditional core damage probabilities between 1.0×10^{-5} and 1.0×10^{-6}). The results of these analyses are shown in Tables 3.4–3.9. Forty-six events were determined to be impractical to analyze and 34 were documented as "interesting," containment-related, or shutdown-related events, but were not analyzed. Some of these events also involved multiple LERs.

Chapter 2 describes the selection and analysis process used for the review of 1982-83 events. Chapter 3 provides a tabulation of the precursor events, a summary of the more important precursors, and insights on the results. The remainder of this report is divided into seven appendices: Appendix A describes the process used to model events and the ASP models. Appendix B contains detailed documentation of the at-power precursors with conditional probabilities $\geq 1.0 \times 10^{-5}$. Appendix C contains summaries of the at-power precursors with conditional probabilities between 1.0×10^{-5} and 1.0×10^{-6}. Appendix D contains interesting shutdown-related events. Appendix E contains potentially significant events considered impractical to analyze. Appendix F presents the containment-related events, and Appendix G describes the "interesting" events that were not considered precursors to failure of core cooling.

2. Selection Criteria and Quantification

2.1 Accident Sequence Precursor Selection Criteria

The Accident Sequence Precursor Program identifies and documents potentially important operational events that have involved portions of core damage sequences, and quantifies the core damage probability associated with those sequences. Identification of precursors requires the review of operational events for instances in which plant functions that provide protection against core damage have been challenged or compromised. Based on previous experience with reactor plant operational events, it is known that most operational events can be directly or indirectly associated with four initiators: trip [which includes loss of main feedwater (LOFW) within its sequences], loss of offsite power (LOOP), small-break, loss-of-coolant accidents (LOCA), and steam generator tube ruptures (SGTR) (pressurized water reactors, PWRs, only). These four initiators are primarily associated with loss of core cooling. ASP Program staff members examine licensee event reports and other event documentation to determine the impact that operational events have on potential core damage sequences.

2.1.1 Precursors

This section describes the steps used to identify events for quantification. Figure 2.1 illustrates this process.

A computerized search of the SCSS data base at the Nuclear Operations Analysis Center (NOAC) of the Oak Ridge National Laboratory was conducted to identify LERs that met minimum selection criteria for precursors. This computerized search identified LERs potentially involving failures in plant systems that provide protective functions and those potentially involving core damage-related initiating events. Based on a review of the 1984–1987 precursor evaluations and all LERs for 1990, this computerized search successfully identified almost all precursors; the resulting subset of events is approximately one-third to one-half of the total LERs. It should be noted, however, that the computerized search scheme was not tested on the LER data base for the years prior to 1984. Moreover, since the LER reporting requirements for 1982-83 were different than for 1984 and later, the possibility exists that some 1982-83 precursor events were not included in the selected subset. Events described in NUREG-0900[20] and in issues of *Nuclear Safety* that potentially affected core damage sequences were also selected for review.

The events selected by the search of the SCSS data base underwent at least two independent reviews by different staff members to determine if the reported event should be examined in greater detail. This was a bounding review, meant to capture events that in any way appeared to deserve detailed review and to eliminate events that were clearly unimportant. This process involved eliminating events that satisfied predefined criteria for rejection and accepting all others as either potentially significant and requiring analysis, or potentially significant but impractical to analyze. All events identified as impractical to analyze at any point in the study are documented in Appendix E. Events were also eliminated from further review if they had little impact on core damage sequences or provided little new information on the risk impacts of plant operation—for example, short-term single failures in redundant systems, uncomplicated reactor trips, and LOFW events.

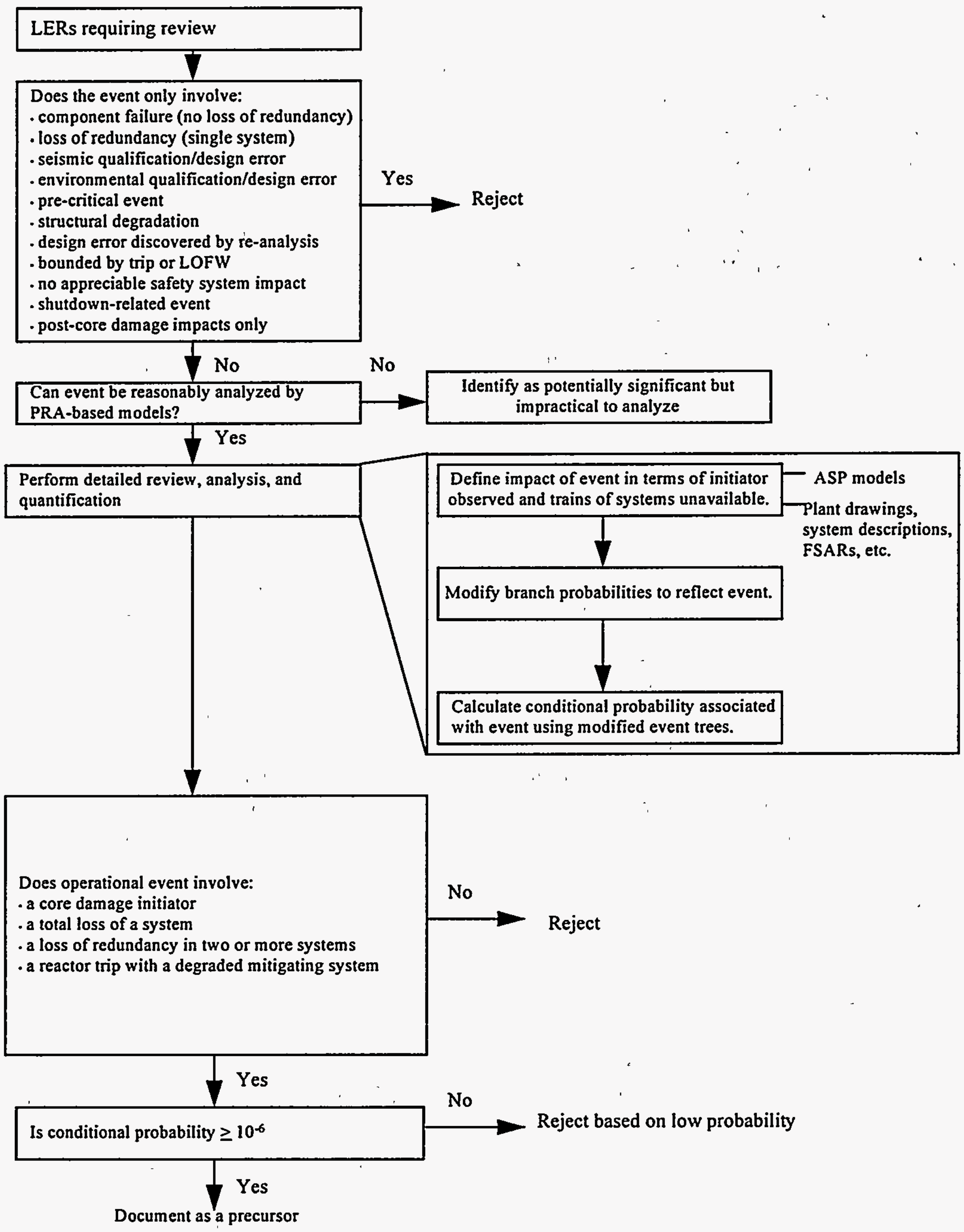

Figure 2.1 ASP Analysis Process

Selection Criteria and Quantification

LERs were eliminated from further consideration as precursors if they involved, at most, only one of the following:

- a component failure with no loss of redundancy,
- a short-term loss of redundancy in only one system,
- a seismic design or qualification error,
- an environmental design or qualification error,
- a structural degradation,
- an event that occurred prior to initial criticality,
- a design error discovered by reanalysis,
- an event bounded by a reactor trip or LOFW,
- an event with no appreciable impact on safety systems, or
- an event involving only post-core damage impacts.

Events identified for further consideration typically included the following:

- unexpected core damage initiators (LOOP, SGTR, and small-break LOCA);
- all events in which a reactor trip was demanded and a safety-related component failed;
- all support system failures, including failures in cooling water systems, instrument air, instrumentation and control, and electric power systems;
- any event in which two or more failures occurred;
- any event or operating condition that was not predicted or that proceeded differently from the plant design basis; and
- any event that, based on the reviewers' experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Events determined to be potentially significant as a result of this initial review were then subjected to a thorough, detailed analysis. This extensive analysis was intended to identify those events considered to be precursors to potential severe core damage accidents, either because of an initiating event, or because of failures that could have affected the course of postulated off-normal events or accidents. These detailed reviews were not limited to the LERs; they also used final safety analysis reports (FSARs) and their amendments, individual plant examinations (IPEs), and other information related to the event of interest.

The detailed review of each event considered the immediate impact of an initiating event or the potential impact of equipment failures or operator errors on the readiness of systems in the plant to mitigate off-normal and accident conditions. In the review of each selected event, three general scenarios (involving both the actual event and postulated additional failures) were considered.

1. If the event or failure was immediately detectable and occurred while the plant was at power, then the event was evaluated according to the likelihood that it and the ensuing plant response could lead to severe core damage.

2. If the event or failure had no immediate effect on plant operation (i.e., if no initiating event occurred), then the review considered whether the plant would require the failed items for mitigation of potential severe core damage sequences should a postulated initiating event occur during the failure period.

3. If the event or failure occurred while the plant was not at power, then the event was first assessed to determine whether it affected at-power or hot shutdown operation. If the event could occur only at cold shutdown or refueling shutdown, or the conditions clearly did not affect at-power operation, then its impact on continued decay heat removal during shutdown was assessed; otherwise it was analyzed as if the plant were at power. (Although no cold shutdown events were analyzed in the present study, some potentially significant shutdown-related events are described in Appendix D.)

For each actual occurrence or postulated initiating event associated with an operational event reported in an LER or multiple LERs, the sequence of operation of various mitigating systems required to prevent core damage was considered. Events were selected and documented as precursors to potential severe core damage accidents (accident sequence precursors) if the conditional probability of subsequent core damage was at least 1.0×10^{-6} (see Section 2.2). Events of low significance were thus excluded, allowing attention to be focused on the more important events. This approach is consistent with that used to define 1988–1993 precursors, but differs from that of earlier ASP reports, which addressed all events meeting the precursor selection criteria regardless of conditional core damage probability. As noted above, 109 operational events with conditional probabilities of subsequent severe core damage $\geq 1.0 \times 10^{-6}$ were identified as accident sequence precursors.

2.1.2 Potentially Significant Shutdown-Related Events

No cold shutdown events were analyzed in this study because the lack of information concerning plant status at the time of the event [e.g., systems unavailable, decay heat loads, reactor cooling system (RCS) heatup rates, etc.] prevented the development of models for such events. However, cold shutdown events such as a prolonged loss of residual heat removal (RHR) cooling during conditions of high decay heat can be risk significant. Sixteen shutdown-related events that may have potential risk significance are described in Appendix D.

2.1.3 Potentially Significant Events Considered Impractical to Analyze

In some cases, it was not practical to analyze the events because there was inadequate information or it was not possible to reasonably model the event within a probabilistic risk assessment (PRA) framework, considering the level of detail typically available in PRA models and the resources available to the ASP Program.

Forty-six events (some involving more than a single LER) identified as potentially significant were considered impractical to analyze for 1982-83. It is thought that such events are capable of affecting core damage sequences. However, the events usually involved component degradations in which the extent of the degradation could not be determined or the impact of the degradation on plant response could not be ascertained.

While a bounding analysis could have been conducted for many events classified as impractical to analyze, an assumption that the affected component or function was unavailable over a one-year period (as would be done using a bounding analysis) would result in the conclusion that a very significant condition existed. This conclusion would not be supported by the specifics of the event as reported in the LER(s) or by the limited engineering evaluation performed in the ASP Program. Thus, simple descriptions of such events are provided in Appendix E.

Selection Criteria and Quantification

2.1.4 Containment-Related Events

In addition to accident sequence precursors, events involving loss of containment functions, such as containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control, which are identified in the reviews of 1982-83 LERs, are documented in Appendix F. It should be noted that the SCSS search algorithm does not specifically search for containment-related events. If these events were identified for other reasons during the search, they were then examined and documented.

2.1.5 "Interesting" Events

Other events that provided insight into unusual failure modes with the potential for compromising continued core cooling but that were determined not to be precursors were also identified. These are documented as "interesting" events in Appendix G.

2.2 Precursor Quantification

Quantification of the significance of an accident sequence precursor involves determination of a conditional probability of subsequent severe core damage, given the failures observed during an operational event. This is estimated by mapping failures observed during the event onto the ASP models, which depict potential paths to severe core damage, and calculating a conditional probability of core damage through the use of event trees and system models modified to reflect the event. The effect of a precursor on event tree branches is assessed by reviewing the specifics of the operational event against system design information. Quantification results in a revised probability of core damage failure, given the operational event. The conditional probability estimated for each precursor is useful in ranking because it provides an estimate of the measure of protection against core damage that remains once the observed failures have occurred. Details of the event modeling process and calculated results can be found in Appendix A.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the light-water reactor population for the 1982–1986 time period, even though they are applied to sequences that are plant specific. Because of this, the conditional probabilities determined for each precursor cannot be rigorously associated with the probability of severe core damage resulting from the actual event at the specific reactor plant at which it occurred. Appendix A documents the accident sequence models used in the 1982-83 precursor analyses, and provides examples of the probability values used in the calculations.

The evaluation of precursors in this report considered equipment and recovery procedures believed to have been available at the various plants in the 1982-83 time frame. This includes features addressed in the current (1994) ASP models that were not considered in the analysis of 1984–1991 events, and only partially in the analysis of 1992-93 events. These features include the potential use of the residual heat removal system for long-term decay heat removal following a small-break LOCA in PWRs, the potential use of the reactor core isolation cooling system to supply makeup following a small-break LOCA in boiling water reactors (BWRs), and core damage sequences associated with failure to trip the reactor (this condition was previously designated anticipated transient without scram ("ATWS,") and not developed). In addition, the potential long-term recovery of the power conversion system for BWR decay heat removal has been addressed in the models.

Because of these differences in modeling, and the need to assume in the analysis of 1982-83 events that equipment reported as failed near the time of a reactor trip could have affected post-trip response (beginning in 1984 equipment response following a reactor trip had to be reported), the evaluations for these years may not be directly comparable with the results for other years.

Another difference between earlier and the most recent (1994) precursor analyses involves the documentation of the significance of precursors involving unavailable equipment without initiating events. These events are termed unavailabilities in this report, but are also referred to as "conditions." The 1994 analyses distinguish a precursor conditional core damage probability (CCDP), which addresses the risk impact of the failed equipment as well as all other nominally functioning equipment during the unavailability period, and an importance measure defined as the difference between the CCDP and the nominal core damage probability (CDP) over the same time period. This importance measure, which estimates the increase in core damage probability because of the failures, was referred to as the CCDP in pre-1994 reports, and was used to rank unavailabilities.

For most unavailabilities that meet the ASP selection criteria, the observed failures significantly affect the core damage model. In these cases, there is little difference between the CCDP and the importance measure. For some events, however, nominal plant response dominates the risk. In these cases, the CCDP can be considerably higher than the importance measure. For 1994 unavailabilities, the CCDP, CDP, and importance are all provided to better characterize the significance of an event. This is facilitated by the computer code used to evaluate 1994 events (the GEM module in SAPHIRE), which reports these three values.

The analyses of 1982-83 events, however, were performed with the event evaluation code (EVENTEVL) used in the assessment of 1984–1993 precursors. Because this code only reports the importance measure (increase in core damage probability) for unavailabilities, that value is reported on the calculation sheets. The CDP was calculated separately, and was added to importance to estimate the CCDP for the event. An example of the difference between a conditional probability calculation and an importance calculation is provided in Appendix A.

2.3 Review of Precursor Documentation

With completion of the initial analyses of the precursors and reviews by team members, a draft of this report containing the analyses was transmitted to an NRC contractor, Oak Ridge National Laboratories, for an independent review. The review was intended to (1) provide an independent quality check of the analyses, (2) ensure consistency with the ASP analysis guidelines and with other ASP analyses for the same event type, and (3) verify the adequacy of the modeling approach and appropriateness of the assumptions used in the analyses. In addition, the draft report was sent to the pertinent nuclear plant licensees. Comments received from the licensees were considered during resolution of comments received from ORNL.

2.4 Precursor Documentation Format

The 1982-83 precursors are documented in Appendices B and C. The at-power events with conditional core damage probabilities $\geq 1.0 \times 10^{-5}$ are contained in Appendix B and those with CCDPs between 1.0×10^{-5} and 1.0×10^{-6} are summarized in Appendix C. Appendix B provides a description of the event with additional information relevant to its assessment, the ASP modeling assumptions and approach used in the analysis, the analysis results, and a figure portraying the dominant core damage sequence postulated for each event. The conditional core damage probability calculations are documented; the documentation includes probability summaries for end states, the conditional probabilities for the more important sequences, and the branch probabilities used. Copies of the LERs are not provided with this report.

2.5 Potential Sources of Error

As with any analytic procedure, the availability of information and the modeling assumptions used can bias results. In this section, several of these potential sources of error are addressed.

1. *Evaluation of only a subset of 1982-83 LERs.* For 1969–1981 and 1984–1987, all LERs reported during the year were evaluated for precursors. For 1988–1994 and for the present ASP study of 1982-83 events, only a subset of the LERs were evaluated after a computerized search of the SCSS data base. While this subset is thought to include most serious operational events, it is possible that some events that would normally be selected as precursors were missed because they were not selected by the screening process. *Reports to Congress on Abnormal Occurrences*[20] (NUREG-0900 series) and operating experience articles in *Nuclear Safety* were also reviewed for events that may have been missed by the SCSS computerized screening.

2. *Inherent biases in the selection process.* Although the criteria for identification of an operational event as a precursor are fairly well defined, the selection of an LER for initial review can be somewhat judgmental. Events selected in the study were more serious than most, so the majority of the LERs selected for detailed review would probably have been selected by other reviewers with experience in LWR systems and their operation. However, some differences would be expected to exist; thus, the selected set of precursors should not be considered unique.

3. *Lack of appropriate event information.* The accuracy and completeness of the LERs and other event-related documentation in reflecting pertinent operational information for the 1982-83 events are questionable in some cases. Requirements associated with LER reporting at the time, plus the approach to event reporting practiced at particular plants, could have resulted in variation in the extent of events reported and report details among plants. In addition, usually only details of the sequence (or partial sequences for failures discovered during testing) that actually occurred are provided; details concerning potential alternative sequences of interest in this study must often be inferred.

   Finally, the lack of a requirement at the time to link plant trip information to reportable events meant that certain assumptions had to be made in analyzing certain kinds of 1982-83 events. Specifically, through use of the "Grey Books" (*Licensed Operating Reactors Status Report,* NUREG-0020)[21] it was possible to determine that system unavailabilities reported in LERs could have overlapped with plant trips. This was accomplished by assuming that the component could have been out of service for half of the test/surveillance period associated with that component. However, with the link between trips and events not being described in the LERs, it was often impossible to determine whether the component was actually unavailable during the trip or whether it was demanded during the trip.

Nevertheless, in order to avoid missing any important precursors for the time period, any reported component unavailability which overlapped a plant trip within half of the component's test/surveillance period, and which was believed not to have been demanded during the trip, was assumed to be unavailable concurrent with the trip. (If the component had been demanded and failed, the failure would have been reported; if it had been demanded and worked successfully, then the failure would have occurred after the trip.) Since such assumptions may be conservative, these events are distinguished from the other precursors listed in Tables 3.4–3.9. As noted previously, these events are termed "windowed" events to indicate that they were analyzed because the potential time window for their unavailability was assumed to have overlapped a plant trip.

4. *Accuracy of the ASP models and probability data.* The event trees used in the analysis are specific to plant class and reflect differences among plants in the eight plant classes that have been defined. The system models are structured to reflect the plant-specific systems, at least to the train level. While major differences between plants are represented in this way, the plant models utilized in the analysis may not adequately reflect all important differences. Modeling improvements that address these problems are being pursued in the ASP Program.

   Because of the sparseness of system failure events, data from many plants must be combined to estimate the failure probability of a multitrain system or the frequency of low- and moderate-frequency events (such as LOOPs and small-break LOCAs). Because of this, the response modeled for each event will tend toward an average response for the plant class. If systems at the plant at which the event occurred are better or worse than average (which is difficult to ascertain without extensive operating experience), the actual conditional probability for an event could be higher or lower than that calculated in the analysis.

   Known plant-specific equipment and procedures that can provide additional protection against core damage beyond the plant-class features included in the ASP event tree models were addressed for some plants in the 1982-83 precursor analysis. This information was not uniformly available; much of it was based on FSAR and IPE documentation available at the time this report was prepared. As a result, consideration of additional features may not be consistent in precursor analyses of events at different plants. However, multiple events that occurred at an individual plant or at similar units at the same site have been consistently analyzed.

5. *Difficulty in determining the potential for recovery of failed equipment.* Assignment of recovery credit for an event can have a significant impact on the assessment of the event. The approach used to assign recovery credit is described in detail in Appendix A. The actual likelihood of failing to recover from an event at a particular plant during 1982-83 is difficult to assess and may vary substantially from the values currently used in the ASP analyses. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, and others, concerning the likelihood of recovering from specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.

Selection Criteria and Quantification

6. *Assumption of a one-month test interval.* The core damage probability for precursors involving unavailabilities is calculated on the basis of the exposure time associated with the event. For failures discovered during testing, the time period is related to the test interval. A test interval of one month was assumed unless another interval was specified in the LER. See Ref. 1 for a more comprehensive discussion of test interval assumptions.

3. Results

This chapter summarizes the results of the review and evaluation of 1982-83 operational events. The primary objective of the ASP Program is the identification of operational events with conditional core damage probabilities of $\geq 1.0 \times 10^{-6}$ that satisfy at least one of the four following criteria: (1) an initiating event requiring safety system response, (2) the failure of a system required to mitigate the consequences of a core damage initiator, (3) degradation of more than one system required for mitigation, or (4) a trip or loss of feedwater with a degraded mitigating system. One-hundred nine such events identified for 1982-83 are documented in Appendices B and C.

A direct comparison of the results with those of earlier and more recent years is difficult without a substantial effort to reconcile differences in the analyses. As described in Section 2.2, changes in models used to evaluate events, and the assumptions required to link unavailable equipment with proximate trips in the 1982-83 period, prevent a detailed comparison with precursors identified for other years. Because of this, only limited and primarily qualitative results are provided here.

3.1 Tabulation of Precursor Events

The distribution of precursor probabilities for 1982-83 is shown in Table 3.1 in Section 3.3 and the precursor incidence rates for 1982-83 and other years are shown in Table 3.2 in Section 3.3. Table 3.3 in Section 3.4.2 presents the fraction of precursors occurring at BWRs in 1982-83 and in other years. The 1982-83 accident sequence precursor events are listed in Tables 3.4–3.9 at the end of this chapter. The following information is included in each table

- Name of the plant where the event occurred
- Docket/LER number associated with the event
- A brief description of the event
- Plant type
- Date(s) of the event
- Conditional probability of potential core damage associated with the event for initiating event and "windowed" initiating event-related precursors. For the "unavailability (Unavail)"-related precursors, the p(cd) value represents the increase in core damage probability over the duration of the event.
- Initiator associated with the event or "unavailability (Unavail)" if no initiator was involved

The 1982-83 precursor events are separated into three categories and then sorted by plant and by CCDP. The three categories include:

- unavailabilities that could have affected the course of postulated off-normal events or accidents,

- initiating events, including those in which a component or system failed to operate as expected,

- reactor trips that were assumed to overlap with a failed or unavailable system because the trip occurred in a time window during which the identified component could be assumed to be unavailable because of its test/surveillance period. As discussed in Section 2.5, owing to the lack of information in the 1982-83 LERs on the link between plant trips and failures/unavailabilities, the present study assumed that reported component unavailabilities that overlapped a plant trip within half of the component's test/surveillance period affected the post-trip response. Since such assumptions may be pessimistic for some unavailabilities, these events are distinguished from the other precursors listed in Tables 3.4–3.9. These events are termed "windowed" events to indicate that they were analyzed because the potential time window for their unavailability was assumed to have overlapped a plant trip.

The tables are grouped as follows:

- Table 3.4 - Precursors involving unavailabilities sorted by plant,
- Table 3.5 - Precursors involving initiating events sorted by plant,
- Table 3.6 - Precursors involving "windowed" initiating events sorted by plant,
- Table 3.7 - Precursors involving unavailabilities sorted by conditional core damage probability,
- Table 3.8 - Precursors involving initiating events sorted by conditional core damage probability,
- Table 3.9 - Precursors involving "windowed" initiating events sorted by conditional core damage probability.

3.1.1 Potentially Significant Shutdown-Related Events

Sixteen shutdown-related events that may have potential risk significance are described in Appendix D. As noted above, no cold shutdown events were analyzed in this study because of the lack of information concerning the status of key plant systems at the time of the event.

3.1.2 Potentially Significant Events Considered Impractical to Analyze

Forty-six potentially significant events were considered impractical to analyze for 1982-83. Typically, this event category includes events that are impractical to analyze because there is inadequate information or it is not possible to reasonably model the event within a probabilistic risk assessment framework, considering the level of detail typically available in PRA models. These potentially significant events are documented in Appendix E.

3.1.3 Containment-Related Events

Three containment-related events were found for 1982-83. This event category includes losses of containment functions, such as containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control. Descriptions of these events are located in Appendix F.

3.1.4 "Interesting" Events

Fifteen "interesting" events were found for 1982-83. This event category includes events that were not selected as precursors but that provided insight into unusual failure modes with the potential for compromising continued core cooling. Descriptions of these events are located in Appendix G of this report.

3.2 Important Precursors

There were 15 precursors in the 1982-83 period with conditional core damage probabilities $\geq 1.0 \times 10^{-4}$. Events with such probabilities have traditionally been considered important in the ASP Program. In alphabetical order by plant name, these events include the following (note that two of the events are discussed together; see Section 3.2.10).

3.2.1 Brunswick 2 (LER 324/82-005)

On January 16, 1982, Brunswick 2 experienced a scram caused by low condenser vacuum. Later, when operators attempted to align the suppression pool cooling, they discovered that both residual heat removal service water (RHRSW) loops were inoperable. Low suction header pressure lockout signals prevented pumps in both loops from starting.

An inspection of the suction header pressure switches found that their sensing lines were plugged with sediment that prevented the switches from sensing the actual header pressure, which was within acceptable limits. The suction header pressure switch for the A loop was also damaged. In addition, the power supply of the B loop suction header pressure switch was switched off, apparently having been left that way after prior maintenance work.

The operators recovered condenser vacuum and established decay heat removal using the secondary side. Both pressure switches were returned to service, the B service water loop was returned to service after approximately 4 hours and the A service water loop was made operable within about 8 hours (the service water system could also have been aligned if necessary to supply the RHR heat exchangers at Brunswick). The conditional core damage probability estimated for this scram with both trains of RHRSW initially unavailable is 2.3×10^{-4}.

3.2.2 Ginna (LER 244/82-003 and -005)

On January 25, 1982, at 0925, while the plant was operating at 100% power, a steam generator tube ruptured. The air ejector radiation monitor alarm indicated that the rupture was most likely in steam generator B. The continuing pressure drop resulted in an automatic reactor trip and actuation of safety injection.

Initially, operators cooled down the reactor by sending steam from both generators to the main condenser. The B steam generator was isolated at 0940, and natural circulation cooling in loop B was terminated. The water level in the B steam generator continued to rise in spite of the termination of feedwater flow because water was flowing through the ruptured tube. At 0955, the narrow-range water level indicator on the B steam generator went off scale high and the B main steam line started to fill.

At 1007, operators attempted to equalize the pressure differential between the reactor coolant system (RCS) and the secondary side of the B steam generator to stop flow through the ruptured tube. A pressurizer power-operated relief valve (PORV) was cycled three times before it stuck in an open position. The operator then closed the block valve to prevent further RCS water loss. There were steam bubbles in the reactor vessel upper head and in the top of the B steam generator tubes as well. The growth of the bubbles and increased safety injection flow rapidly filled the pressurizer. Loop A natural circulation was not affected by the steam bubbles.

One of the B steam generator safety valves cycled three times as a result of the overpressurization caused by continued flow through the ruptured tube. At 1038, safety injection was terminated to prevent further discharge of water through the safety valve. At 1040, the condensate system was secured to prevent further radioactive contamination of the condensate storage tanks and demineralizers. The operators used the steam generator PORV to relieve steam from the A steam generator.

At 1107, one safety injection pump was started in anticipation of an RCS pressure drop caused by the restart of the A reactor coolant pump (RCP). The RCP flow cooled and collapsed any remaining steam bubbles in the reactor upper vessel head and the B steam generator. This addition of flow led to another cycle of the B steam generator safety valve. Safety injection was stopped, but the valve continued to leak water at approximately 100 gpm. The pump was intermittently operated until 1235. At 1152, the pressurizer level returned on scale and a steam bubble was re-established.

At 1227, the RCS and B steam generator pressures equalized. RCS pressure was maintained at 25 psi below steam generator B pressure. At 1840, B steam generator water level returned on scale. Feed and bleed was then used to cool steam generator B.

At 0700, on January 26, 1982, the residual heat removal system was placed in operation, and the plant was declared to be in cold shutdown. The estimated conditional core damage probability for this event is 3.8×10^{-4}.

3.2.3 Hatch 2 (LER 366/82-084, -085)

On August 13, 1982, RHRSW pumps A and C were declared inoperable when they failed to meet their minimum flow requirements during testing procedures. The cavitrol trim (an anticavitation device) on the downstream side of the flow control valve in the outlet of the A loop RHRSW heat exchanger had both broken and bent tubes, which caused the restricted flow.

Also, on August 13, 1982, the D RHRSW pump failed to meet its minimum discharge pressure and flow requirement during testing. This was caused by the failure of the B pump discharge check valve to close. (It should be noted that failure of the check valve would not lead to failed flow from the B pump unless the D pump also failed.) The estimated increase in core damage probability due to the failed valves over the expected duration of this event (half of the surveillance interval, 360 hours) is 2.4×10^{-4}.

3.2.4 Hatch 2 (LER 366/83-042, -055 Rev. 1, -056)

On July 13, 1983 and on July 21, 1983, the reactor core isolation cooling (RCIC) pump failed to deliver the minimum required flow of 400 gpm because its electric governor remote (EGR) actuator was out of adjustment, and was declared unavailable. On July 14, 1983, the plant was starting up from a refueling outage and was at approximately 7% power when the unit started losing condenser vacuum. The turbine was tripped, and control room personnel scrammed individual rods with the scram switches at the scram timing panel. The objective was to reduce power quickly so that the mechanical vacuum pump could be placed in service before the decreasing vacuum reached the reactor feed pump low vacuum trip point. A reactor feed pump low vacuum trip results in a loss of feedwater flow to the vessel. Since RCIC was unavailable, operators were trying to avoid losing feedwater flow. The rod worth minimizer was bypassed and at one point the "emergency rod in" control was used to achieve the greatest possible insertion rate. After several rods had been inserted, one rod was found in an "out of sequence" position, and the reactor was manually scrammed.

Results

The next day, while the A loop of RHR was being put in the shutdown cooling mode to achieve a cold shutdown condition, the A loop heat exchanger outlet valve failed to open because the valve motor was faulted. The conditional core damage probability estimated for the scram with RCIC and RHR loop A unavailable is 1.5×10^{-4}. (It should be noted that the unavailability of RCIC had little effect on the probability of core damage for this event.)

3.2.5 Hatch 2 (LER 366/83-084)

On August 16, 1983, Hatch 2 experienced a reactor scram on low water level as a result of a spike in a reactor feed pump turbine control signal. The next day, as the unit was going from hot to cold shutdown, the RHR A loop heat exchanger outlet valve (2E11-F003A) failed to open because of a burned-out motor. When plant personnel attempted to open the valve, its position indication was lost and the personnel received a "valve overload" alarm. An investigation revealed that the valve's motor suffered an electrical fault when personnel tried to open the valve. The conditional core damage probability estimated for the scram with unavailable RHR train A is 1.4×10^{-4}.

3.2.6 Indian Point 2 (LERs 247/82-019 and -020)

During hot shutdown following a reactor trip caused by feed system perturbations on May 17, 1982, No. 23 motor-driven auxiliary feedwater (AFW) pump failed while in operation. The failure was caused by a damaged thrust bearing, which in turn was caused by the equalizing line check valve hanging up and negating positioning control of the balancing drum. Two days later, while in hot shutdown, No. 22 turbine-driven AFW pump failed while in operation. Erratic speed control by the governor on its steam turbine drive caused the AFW pump 22 to trip. The governor bearing was smoking. At the time the turbine driven pump was determined to be inoperable, No. 23 AFW motor-driven pump was still out of service. Since only motor-driven AFW pump 21 was operable, the plant started cooldown. The estimated conditional core damage probability for the trip with two inoperable AFW pumps is 1.2×10^{-4}.

3.2.7 LaSalle 1 (LER No. 373/82-093)

At approximately 0300 on August 21, 1982, a controlled shutdown was initiated because the condensate inventory was not sufficient for normal plant operation. Later, numerous condensate system alarms were received. Because of concern about condensate pump cavitation and the adequacy of the control rod drive (CRD) condensate supply, the unit was manually scrammed at 0536.

At an unspecified time on the same date, the RCIC system was inspected, and it was discovered that the RCIC turbine was leaking oil from its sight glass and that the oil level could not be maintained in the turbine. Accordingly, the RCIC was declared inoperable.

Initially after the scram, reactor makeup was supplied by the CRD system but high CRD suction and discharge filter differential pressures developed, and the CRD pump was tripped at 0745. Loss of CRD purge flow to the recirculation pump seals meant that the seals were cooled only by the reactor building closed cooling water (RBCCW) system. The seal temperature on recirculation pump 1B rose to 150°F and stabilized; however, the seal temperature on recirculation pump 1A continued to rise. By 0828, the temperature on the 1A recirculation pump reached 175°F, and the pump was tripped. Subsequently, the seal temperature rose to 235°F. At that time, around 0910, a drywell entry was made and the RBCCW flow to the seal was found to be low. The operations foreman increased flow from below 13 gpm to about 25 gpm over a period of about 1 minute and the seal temperature dropped abruptly to about 100°F. The resulting thermal stress completely fractured both the No. 1 and 2 (backup) seals, and water and steam began blowing out directly to the drywell around the seal assembly. The flow rate increased over time, eventually reaching about 27 gpm, based on one indication which was averaged over a two-hour period.

Around 1000, operators attempted to close the recirculation pump suction and discharge valves, but were unable to fully close the suction valve. The temperature of recirculation pump 1A seal continued to rise, exceeding 300°F. Around 1225, an operator entered the drywell again and manually closed the recirculation pump suction valve, stopping the leak. The conditional core damage probability estimated for this event is 1.1×10^{-4}.

3.2.8 LaSalle 1 (LER 373/83-117, -147)

On September 20, 1983, while LaSalle 1 was in cold shutdown, operators attempted to open the B residual heat removal heat exchanger outlet valve but were unable to do so. An inspection showed that the valve was experiencing hydraulic locking when water became trapped in the bonnet cavity. Since the bonnet cavity did not have any means to vent off water trapped inside it, the valve could become locked in the closed position. After the event, the motor operator for the valve was inspected. The motor windings were found to be burned and the motor was replaced.

On November 12, 1983, a few days after a scram, operators attempted to open the B RHR heat exchanger outlet valve but again were unable to do so. Again, hydraulic locking caused the valve failure. The valve motor operator was again found to be burned and the motor was replaced. The conditional core damage probability estimated for the scram with unavailable RHR train B is 1.4×10^{-4}.

3.2.9 Quad Cities 1 (LER 254/82-007, 009)

During normal operation on April 15, 1982, the outboard bearing on RHRSW pump D was found to be failed during a surveillance test. Investigation revealed that the pump bearing failed because there was excessive leakage of water from the adjacent packing into the bearing oil. Two weeks later, on April 30, RHRSW pump C was taken out of service for maintenance on the pump seal packing. Water that leaked from adjacent seal packing was found in the bearing oil reservoir. The licensee stated that while there was insufficient water to damage the bearing because of a loss of lubrication, continued operation could have possibly damaged the bearing. The pump seals were repacked and the oil in the bearing oil reservoir was replaced.

Three plant trips occurred around the time the bearing faults in the pumps were discovered (April 17, 19, and 30). The plant trip on April 17 involved a presumed loss of feedwater due to low condenser vacuum that was caused by a failure of a condensate demineralizer valve. The estimated conditional core damage probability for the scram, loss of feedwater, and degraded RHRSW is 1.7×10^{-4}.

3.2.10 Quad Cities 1 (LERs 254/82-012, -013, and -018)

During normal operation on June 22, 1982, the Unit 2 reactor experienced a trip as a result of feedwater pump trip and low water level. This was caused by the loss of bus 22 while the reserve auxiliary transformer 22 was being removed from service for maintenance. An equipment operator mistakenly pulled out the fuses for a

4-kV bus instead of the transformer fuses. This error disconnected power to the 2B reactor feedwater pump, which caused a low water level and initiated a trip. The Unit 2 main generator subsequently tripped, and all normal ac power to Unit 2 was lost. Upon the loss of offsite power (LOOP), both the Unit 2 and swing emergency diesel generators (EDGs) loaded their respective emergency buses. The swing diesel generator tripped when the A RHRSW pump was started, approximately 22 minutes after the fuses were pulled, and the EDG lockout relay actuated. To restart the EDG, the relay had to be manually reset by the equipment operator but this reset was delayed because the equipment operator had been sent to the switchyard to expedite the restoration of offsite power.

One day prior to the Unit 2 LOOP, the Unit 1 EDG was removed from service because the diesel generator cooling water pump was not providing water to the EDG during an HPCI flow rate test. Investigation revealed that the pump was air bound because air entered the suction line while RHRSW A was being drained for system modifications. The rotating element of the pump was replaced and the pump was returned to service in the late afternoon of June 22. When the Unit 2 LOOP occurred, Unit 1 operated without any EDGs available.

The estimated increase in conditional core damage probability for a postulated LOOP at Unit 1 while both EDGs were unavailable is 1.4×10^{-4}. The estimated conditional core damage probability for the Unit 2 plant-centered LOOP with one EDG inoperable is 1.1×10^{-4}.

3.2.11 Robinson 2 (LERs 261/83-004, -005, -007 and -016)

On April 19, 1983, following a reactor trip caused by failure of the turbine oil electrohydraulic oil pumps, A and B motor-driven AFW pumps started automatically. Within five minutes of the auto-start, pump B tripped. While the AFW pump was being tested to determine the cause of the trip, the pump casing was vented and a significant amount of vapor was released. It was determined that the buildup of vapor inside the casing caused the pump to cavitate, leading to low discharge pressure and tripping of the pump. An examination of the pumps the next day revealed high pump temperatures, indicating that there was a slight back-leakage of hot water through the discharge gate valves into the pump casings, which resulted in steam binding of the B AFW pump. A similar back-leakage through the discharge valves resulted in the binding of the turbine-driven AFW pump on July 21, 1983 (LER 261/83-016).

Ten days after the trip, during testing of the PORVs, valve RC-455C failed to meet the required cycle time, and on a subsequent attempt to cycle it, the valve failed to fully open. Inspections revealed that the PORV failure was caused by galling of the valve plug to the cage. The estimated conditional core damage probability for the trip, inoperable AFW pump, and failed PORV (which affects feed-and-bleed cooling) is 9.2×10^{-4}.

3.2.12 Salem 1 (LERs 272/83-011, -012, and -013)

On February 22, 1983, during routine startup of Salem 1 at 20% power, a manual trip was initiated following rapidly decreasing steam generator levels. Both reactor trip breakers failed to open three seconds earlier upon receipt of a valid low-low steam generator reactor trip signal. The low-low steam generator level was caused by the loss of a bus during the transfer to the auxiliary power transformer. An automatic safety injection occurred and No. 11 reactor coolant pump tripped. Loss of pressurizer spray increased the pressurizer pressure to the PORV setpoint and two PORVs actuated. The PORVs mitigated the transient and there was no damage to the reactor coolant system. The plant was then placed in safe shutdown. Investigations focused on the manual trip and the other related event, and the failure of the trip breakers was missed.

On February 25, 1983, during startup at 12% power, a low-low steam generator level signal was generated by the reactor trip system. Both reactor trip breakers failed to open and remained closed until operators manually tripped the plant 25 seconds later. During reviews of this event and the February 22 event, it was determined that the breakers had also failed on February 22. Investigation of the reactor trip system revealed that the breakers had failed to open automatically because the latch mechanism in the undervoltage trip attachment became mechanically bound. Since the manual trip operated the shunt trip device as well as the undervoltage trip attachment, the manual trip succeeded. Following the manual trip, the plant was placed in safe shutdown condition. Because the initial failure to trip was not discovered until after the second failure three days later, the impact of both transients was addressed in a single analysis. The combined conditional core damage probability estimate for the two failures to scram is 4.6×10^{-3}.

3.2.13 Salem 1 (LER 272/83-033 and -034)

On August 11, 1983, with Salem 1 operating at 99% power and Salem 2 operating at 100% power, both units tripped because debris was clogging the circulating water system (CWS) intake screens. The clogged intake screens led to a trip of the CWS and a decrease in condenser vacuum, which required rapid load reductions at both units. The combined load reduction and vacuum decrease tripped Unit 1 steam generator feedwater pump 11, which in turn led to a decrease in feedwater flow and steam generator levels. This then induced a Unit 1 trip. A few minutes later, all vital buses at Unit 1 experienced an undervoltage that was associated with the transfer of the group buses to the station power transformers. The undervoltage condition led to a LOOP. A low-low steam generator level signal occurred as well, and AFW turbine-driven pump 13 failed to start because the position indicator of the trip valve failed to show that the value had been left in a tripped position after a test. During the events at Unit 1, Unit 2 experienced the CWS trip, the turbine generator was successfully unloaded, and Unit 2 tripped. Unit 2 experienced no LOOP and was placed in a stable shutdown condition. The estimated conditional core damage probability for this event is 1.2×10^{-4}.

3.2.14 Salem 2 (LER 311/83-001 and 311/82-072)

On January 6, 1983, while Salem 2 was operating at 46% power, the reactor tripped due to low level in the number 21 steam generator. Following the trip, the operator noticed that trip breaker A had failed to open on the trip signal, but trip breaker B had opened and de-energized the rod drive mechanisms, resulting in a shutdown. It was later determined that trip breaker A undervoltage relay had malfunctioned due to dirt or corrosion which interfered with proper relay operation. A similar breaker failure occurred on August 20, 1982 during a surveillance test. The conditional core damage probability estimated for the reactor trip on January 6, 1983 is 4.4×10^{-4}.

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3.3 Distribution of Precursors as a Function of Conditional Core Damage Probability

One-hundred nine precursors [p(core damage) ≥ 1.0 x 10^{-6}] affecting 51 units were identified for the years 1982-83. The distribution of precursors as a function of conditional probability is shown for both years in Table 3.1.

Table 3.1 Distribution of Precursors as a Function of Conditional Core Damage Probability

Year	$10^{-3} \le p(cd) < 1$	$10^{-4} \le p(cd) < 10^{-3}$	$10^{-5} \le p(cd) < 10^{-4}$	$10^{-6} \le p(cd) < 10^{-5}$	Total number of precursors
1982	0	8	14	32	54
1983	1	6	24	24	55

As described previously, differences in the ASP models and analysis methods preclude a detailed comparison of the number of events identified in different calendar year periods. However, since differences between the models used to assess 1982-91 events are not expected to result in substantially different probabilities for many events, it is possible to compare the number of events that occurred in different time periods to develop a qualitative understanding of the changes in precursor incidence rates.

The frequency of precursors per critical reactor year in 1982-83, 1984-85, 1986-87, 1988-89 and 1990-91 is shown in Table 3.8.[1] Values for 1984-91 were developed from data in a recently published study.[22] The study also addressed the number of precursors identified in 1992-94. However, since changes in the analysis approach are believed to have had a significant effect on the number of precursors that were identified for those years, the 1992-94 period was not addressed herein.

Table 3.2 Precursor Incidence Rates

Range	1982-83	1984-85	1986-87	1988-89	1990-91
$P(cd) \ge 10^{-2}$	0	0.009	0	0	0
$10^{-3} \le p(cd) < 10^{-2}$	0.010	0.018	0.015	0	0.006
$10^{-4} \le p(cd) < 10^{-3}$	0.145	0.210	0.104	0.092	0.122
$10^{-5} \le p(cd) < 10^{-4}$	0.388	0.193	0.126	0.164	0.115
$10^{-6} \le p(cd) < 10^{-5}$	0.572	0.210	0.141	0.151	0.103
Total	1.113	0.639	0.387	0.408	0.346

[1]The observation periods are not equivalent; the number of critical reactor years increases from ~98 in 1982-83 to ~164 in 1990-91.

A comparison of frequencies in Table 3.2 for the five two-year periods indicates a reduction in the incidence rates for all precursors taken together and for precursors with CCDPs in the 10^{-6} - 10^{-5} and 10^{-5} - 10^{-4} ranges. This is consistent with the conclusions of Rasmuson and O'Reilly[22] for similarly significant precursors. The data in Table 3.2 also appears to be consistent with the conclusion in Rasmuson and O'Reilly[22] concerning the incidence rate for the most significant precursors; those with CCDPs $\geq 10^{-3}$. No change in the incidence rate is apparent for these events. With regard to precursors with CCDPs in the 10^{-4} - 10^{-3} range, it is not clear that a decreasing incidence rate exists (Rasmuson and O'Reilly[22] concluded that a decreasing trend existed, but this may have been the result of a large number of 1984 precursors in this range).

The reason for the apparent decrease in the incidence of less significant precursors, and the lack of a corresponding decrease for more significant precursors is unknown. Further analysis of the changes in precursor incidence rates over time appears to be warranted.

3.4 Insights

3.4.1 Sequence Contributions

Precursors with conditional probabilities $\geq 1.0 \times 10^{-4}$ that were identified for 1982-83 were reviewed to determine the sequences contributing most to the CCDP. These sequences include the observed plant state plus additional postulated failures required for core damage. They can generally be categorized as:

BWRs - based on nine events

- failure of RHR following a transient (67% of events $\geq 1.0 \times 10^{-4}$),
- station blackout (LOOP with emergency power failure) (22% of events $\geq 1.0 \times 10^{-4}$), and
- failure of high-pressure injection systems and failure to depressurize to allow the use of low-pressure systems following a transient (11% of events $\geq 1.0 \times 10^{-4}$),

PWRs - based on six events

- failure to trip and failure of emergency boration following a transient (33% of events $\geq 1.0 \times 10^{-4}$),
- failure of secondary-side cooling and failure of feed and bleed cooling following a transient (33% of events $\geq 1.0 \times 10^{-4}$),
- station blackout (LOOP with emergency power failure) (17% of events $\geq 1.0 \times 10^{-4}$), and
- failure of HPI following a steam generator tube rupture (17% of events $\geq 1.0 \times 10^{-4}$).

With the exception of PWR sequences associated with a failure to trip (the only PWR failure to trip occurred in 1982-83), the types of sequences contributing most to the CCDP are consistent with those observed in other years (the percent contributions vary based on the precursors observed in the yearly periods).

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3.4.2 Trends and Patterns

A review of the precursors identified for 1982-83 and a comparison with events identified in other time periods indicate the following trends and patterns.

1. Events at BWRs comprised a greater proportion of the precursors than the relative population of BWRs would indicate. In 1982-83, nine of the 15 precursors with p(cd) $\geq 1.0 \times 10^{-4}$ occurred at BWRs. This percentage, 0.60, is much higher than the percentage of BWR reactor years in 1982-83 (0.35). If all precursors are considered, the contribution for BWRs drops to 0.44, which is still above the percentage of reactor years. This greater proportion was also observed in 1984-85, as shown in Table 3.3. For the 1986-89 period, however, BWRs contributed a disproportionately smaller number of events. The reason for the disproportional contributions, and in particular the change in relative contribution between 1984-85 and 1986-87, is unknown (it does not correspond to a change in ASP event screening or models). As for the changes observed in precursor incidence rates, further analysis appears to be warranted.

Table 3.3. Fraction of Precursors at BWRs

	1982-83	1984-85	1986-87	1988-89	1990-91
Fraction of BWRs	0.35	0.35	0.36	0.34	0.34
Fraction of BWR Precursors	0.44	0.49	0.24	0.26	0.34
Fraction of BWR Precursors with CCDP $\geq 10^{-4}$	0.60	0.67	0.25	0	0.32

2. Two unusual types of events were observed in 1982-83: PWR failures to scram and BWR residual heat removal failures.

a. The only PWR failures to scram that have been observed occurred in 1983 at Salem 1 (the single observed BWR failure to scram occurred in 1980). Two failures occurred three days apart; the first failure was missed until the second was investigated. The failures were caused by mechanical binding in the breaker undervoltage trip mechanisms. Interestingly, trips with failure of one of the reactor trip breakers to open due to undervoltage relay problems occurred 1½ and 6 months earlier at Salem 2. One other failure of a single trip breaker to open during a reactor trip in a PWR has been identified by the ASP Program. The failure occurred at Shearon Harris on June 3, 1991.

b. A higher number of events involving BWR RHR failures occurred in 1982-83 than in other time periods. Six precursors of this type with p(cd) $\geq 1.0 \times 10^{-4}$ were identified in 1982-83, while one event was identified in 1980-81 and one in 1984-85. Multiple events occurred at three plants. This multiplicity of RHR-related events at individual plants may imply that many of these events were specific to individual plants and were not representative of performance of the industry as a whole. In general, few RHR-related events have been identified in the ASP Program. Three of the 1982-83 events occurred at Hatch 2 and involved valve failures. The three other RHR-related events occurred at Brunswick 2, LaSalle 1, and Quad Cities 1. The 1980-81 and 1984-85 events occurred at Brunswick 1 and LaSalle 1.

3. Event types that constituted some of the most important 1969-81 precursors were observed to a lesser extent in 1982-83 and in the following years. Nineteen of the 24 1969-81 precursors with p(cd) $\geq 1.0 \times 10^{-3}$ can be characterized as transients driven by electrical and instrumentation interactions, transients with AFW or HPCI/RCIC unavailabilities, and small-break LOCAs. Precursors identified in 1982-83 and 1984-87 were reviewed against these categories to determine changes in the number of these types of events that were observed in these time periods as opposed to the 1969-81 period. Since the number of reactor years in the 1969-81 and 1982-87 periods are approximately equal, differences in the number of events in each category can serve as an indication of differences in precursor characteristics in the two periods. This review indicated that two types of significant 1969-81 events (transients driven by electrical and instrumentation interactions (primarily in PWRs) and transients with AFW system unavailabilities (PWRs only)), occurred much less frequently in 1982-87 than in the preceding years. The reduction in the number of these events may be the result of efforts in the early 1980s to understand (and correct if undesirable) the impact of control system failures on plant operation. and the extensive scrutiny AFW system performance received following the TMI-2 accident. As with the observations concerning changes in precursor incidence and in the proportion of BWR precursors, further review to understand the causes of the observed changes appears desirable. The results of this review are discussed below.

a. *Transients with multiple failures caused by electrical and instrumentation interactions*. In these events, which occurred primarily in PWRs, an initial failure, such as the loss of a vital bus, unexpectedly impacted multiple components and often resulted in plant response that was not anticipated by the operators. Eight events with p(cd) $\geq 1.0 \times 10^{-3}$ were identified in 1969-81, including the Rancho Seco non-nuclear instrumentation bus failure (March 20, 1978) and the installation of dummy instrument signals at Zion 2 which resulted in the draining of the pressurizer (July 12, 1977). No events of this type with p(cd) $\geq 1.0 \times 10^{-4}$ occurred in 1982-83 or 1984-85. One such event, involving emergency power and high-pressure recirculation unavailability caused by water intrusion in instrument air lines, occurred in 1986-87 at Fort Calhoun (July 6, 1987).

b. *Ttransients with AFW system or combined HPCI/RCIC inoperability*. The AFW system in PWRs and the HPCI/RCIC systems in BWRs are the primary means of decay heat removal following a loss of normal feedwater. Six events involving a transient with AFW system inoperability with p(cd) $\geq 1.0 \times 10^{-3}$ were observed in 1969-81. Included in this set is the Three Mile Island 2 accident and two events involving clogged AFW pump suction strainers. No similar events occurred in 1982-83. For the combined period 1982-87, only one event with p(cd) $\geq 1.0 \times 10^{-3}$ was associated with an AFW system failure (Davis-Besse, June 9, 1985). AFW-related events with p(cd) $\geq 1.0 \times 10^{-4}$ were also observed in 1982-83 and 1984-87 (including an LOFW with AFW failure at San Onofre 1 with p(cd) just below $\geq 1.0 \times 10^{-3}$), but to a lesser extent than in 1969-81. Two BWR transients with HPCI/RCIC unavailability with p(cd) $\geq 1.0 \times 10^{-3}$ were observed in 1969-81. While no transients with p(cd) $\geq 1.0 \times 10^{-4}$ and with HPCI and RCIC unavailable were observed in 1982-83, an equal number of events were observed in the overall 1982-87 period. One occurred at Hatch 1 in conjunction with a small-break LOCA (May 15, 1985) and one occurred at Brunswick 2 following an LOFW (January 5, 1987). It should be noted that while the number of PWR transients with unavailable AFW systems decreased substantially after 1969-81, there was no decrease in the number of BWR transients with HPCI/RCIC unavailabilities.

c. *Small-break LOCA-related events*. These events typically involve a loss of reactor coolant through a relief valve that fails to reseat after it opens to relieve excessive RCS pressure or a failed reactor coolant pump seal (PWRs only). The lost inventory must be replaced using a high pressure injection system (HPI in PWRs and

HPCI/RCIC in BWRs). In addition to the TMI accident, which involved the loss of reactor coolant through a stuck-open PORV and operator termination of HPI, two additional LOCA-related events with p(cd) $\geq 1.0 \times 10^{-3}$ were observed in 1969-81: a stuck-open PORV at Davis-Besse (September 24, 1977) and a stuck-open safety valve with RCIC inoperable and RHR degraded at Brunswick (April 29, 1975). No LOCAs occurred in 1982-83 (the flow rate associated with the BWR recirculation pump seal failure at LaSalle 1 (LER 373/82-093) was too small for the event to be considered a LOCA). For 1984-87, two LOCA-related events with p(cd) $\geq 1.0 \times 10^{-4}$ were observed: an open relief valve (caused by water dripping from a ventilation duct onto control room instrumentation) with both HPCI and RCIC unavailable at Hatch 1 (May 15, 1985) and a LOCA associated with a letdown drain line rupture at Catawba 1 (June 13, 1986). One SGTR (a LOCA with inventory loss to the secondary side) with p(cd) $\geq 1.0 \times 10^{-4}$ was observed in the 1969-81 period (Prairie Island, October 2, 1979) compared with two in 1982-87 (at Ginna, on January 25, 1982, and North Anna on July 15, 1987). The incidence of high-probability small-break LOCA-related events is similar in the two time periods, although the causes of the events are somewhat different. For 1982-87, SGTRs contribute to a greater extent than in 1969-81, and stuck-open relief valves to a lesser extent.

Table 3.4 Precursors Involving Unavailabilities Sorted by Plant

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)[a]	Event Type
Brunswick 1	325/82-041	Both RHRSW Loops Simultaneously Inoperable	BWR	3/25/82	4.7×10^{-5}	Unavail.
Calvert Cliffs 1	317/83-046 317/83-049	One EDG and One Turbine-Driven AFW Pump Inoperable	PWR	8/16/83	5.8×10^{-6}	Unavail.
Cook 2	316/82-011	ESW Header and ECCS Train A Inoperable	PWR	1/28/82	1.5×10^{-6}	Unavail.
Cook 2	316/82-113	Long-Term Unavailability of SI Train B	PWR	12/29/82	1.4×10^{-6}	Unavail.
Crystal River 3	302/82-007	Unavailability of Two EDGs	PWR	1/23/82	2.8×10^{-5}	Unavail.
Duane Arnold	331/83-017 331/83-018	HPCI and RHRSW Train B Inoperable	BWR	5/23/83	1.7×10^{-5}	Unavail.
Fort Calhoun	285/82-009	Three of Four CCW Heat Exchangers Inoperable	PWR	4/11/82	5.7×10^{-6}	Unavail.
Hatch 1	321/82-070	HPCI and RCIC Simultaneously Unavailable	BWR	8/5/82	3.4×10^{-6}	Unavail.
Hatch 1	321/82-088	HPCI and RCIC Unavailable	BWR	9/24/82	2.5×10^{-6}	Unavail.
Hatch 2	366/82-084 366/82-085	RHRSW Pumps A, C, and D Failed	BWR	8/13/82	2.4×10^{-4}	Unavail.

Table 3.4 Precursors Involving Unavailabilities Sorted by Plant (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)[a]	Event Type
Hatch 2	366/82-095	RHRSW Loops A and B Unavailable	BWR	8/17/82	7.7×10^{-6}	Unavial
North Anna 2	339/82-009	PORVs Inoperable due to Low Nitrogen Pressure	PWR	3/8/82	4.3×10^{-6}	Unavail.
Peach Bottom 3	278/83-009	Two EDGs Inoperable	BWR	9/8/83	3.5×10^{-5}	Unavail.
Pilgrim	293/82-043 293/82-042	RCIC and HPCI Suction Valves Inoperable	BWR	9/30/82	5.8×10^{-6}	Unavail.
Prairie Island 1	282/82-015	Two EDGs Simultaneously Inoperable for 1.5 Hours	PWR	8/27/82	2.3×10^{-6}	Unavail.
Quad Cities 1	254/82-012 254/82-013 254/82-018	Postulated LOOP with Two EDGs Inoperable (Unit 1)	BWR	6/22/82	1.4×10^{-4}	Unavail.
Quad Cities 2	265/82-017 265/82-018	HPCI and One EDG Inoperable	BWR	10/1/82	3.6×10^{-6}	Unavail.
Sequoyah 1	327/82-048 327/82-050	One Motor-Driven AFW Pump and One EDG Inoperable	PWR	3/29/82	2.6×10^{-5}	Unavail.
Sequoyah 1	327/83-063	One of Two PORVs Failed to Open during Attempt to Reseat and Later Leaked after Maintenance	PWR	4/21/83	1.9×10^{-5}	Unavail.
Sequoyah 1	327/83-183 327/83-186	One EDG and Turbine-Driven AFW Pump Inoperable	PWR	11/17/83	3.1×10^{-5}	Unavail.
Summer	395/83-019	Both RHR Trains and One HPI Train Inoperable	PWR	3/17/83	1.0×10^{-6}	Unavail.
Turkey Point 3	250/83-007	Three AFW Pumps Unavailable	PWR	4/19/83	5.5×10^{-5}	Unavail.
Zion 1	295/82-025	Postulated Grid/Weather-Related LOOP with Two EDGs Inoperable for 24 Hours	PWR	8/11/82	3.8×10^{-6}	Unavail.

Table 3.4 Precursors Involving Unavailabilities Sorted by Plant (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)[a]	Event Type
Zion 1	295/82-033	Postulated Grid/Weather-Related LOOP with Two EDGs Inoperable for 24 Hours	PWR	10/15/82	1.4×10^{-6}	Unavail.
Zion 2	304/82-009	Unavailability of Both Motor-Driven AFW Pumps	PWR	4/9/82	3.4×10^{-5}	Unavail.
Zion 2	304/83-007	Postulated Grid/Weather-Related LOOP with 2 EDGs Unavailable	PWR	1/31/83	4.8×10^{-5}	Unavail.

[a] For unavailabilities, the p(cd) represents the increase in core damage probability over the duration of the event.

Table 3.5 Precursors Involving Initiating Events Sorted by Plant

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
ANO 1	313/83-015	Trip with One HPI Injection Valve Failed	PWR	6/16/83	2.9×10^{-6}	Trip
ANO 1	313/83-014	Trip with AFW Pump P-75 Inoperable	PWR	6/9/83	4.7×10^{-6}	Trip
Browns Ferry 1	259/83-006 259/83-007	Scram, MSRV and Its Vacuum Breaker Fail Open	BWR	2/5/83	4.4×10^{-5}	LOCA
Browns Ferry 2	260/83-074	Trip with HPCI Inoperable	BWR	11/10/83	3.2×10^{-5}	Trip
Brunswick 1	325/82-025	Scram with RCIC Inoperable	BWR	2/18/82	1.3×10^{-5}	Trip
Brunswick 1	325/82-054	Scram with RCIC Inoperable	BWR	6/7/82	1.4×10^{-5}	Trip
Brunswick 2	324/82-123	Scram with Emergency Bus E-3 De-energized	BWR	10/10/82	1.2×10^{-5}	Trip
Brunswick 2	324/82-005	Scram with Both RHRSW Loops Inoperable	BWR	2/16/82	2.3×10^{-4}	Trip
Cook 2	316/82-072	Control Room Instrument Distribution Bus IV Fails, Trip	PWR	8/24/82	1.3×10^{-6}	Trip
Cook 2	316/83-052	Control Room Instrument Distribution Bus IV Fails, Trip	PWR	6/23/83	1.0×10^{-6}	Trip
Davis-Besse 1	346/83-038 346/83-040	Trip with AFW Pump Inoperable and Failure of Two SFRCS Channels	PWR	7/25/83	8.2×10^{-5}	Trip
Ginna	244/82-003 244/82-005	Steam Generator Tube Rupture with One PORV Failed Open	PWR	1/25/82	3.8×10^{-4}	SGTR
Hatch 1	321/82-011 321/82-012	Trip with RCIC Inoperable	BWR	2/12/82	3.3×10^{-6}	Trip
Hatch 1	321/83-090 321/83-093	Manual Scram with HPCI and RCIC Unavailable	BWR	8/25/83	1.3×10^{-5}	Trip

Results

Table 3.5 Precursors Involving Initiating Events Sorted by Plant (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Hatch 2	366/82-081	Scram, Isolation, RCIC Failure, SRV Tailpipe Vacuum Relief Failed	BWR	8/25/82	$1.4x10^{-5}$	LOCA
Hatch 2	366/83-042 366/83-055 366/83-056	Reactor Trip with RCIC and RHR Loop A Unavailable	BWR	7/14/83	$1.5x10^{-4}$	Trip
Hatch 2	366/83-069	Scram with HPCI Unavailable	BWR	7/22/83	$6.2x10^{-6}$	Trip
Hatch 2	366/83-084	Scram with RHR Loop A Unavailable	BWR	8/17/83	$1.4x10^{-4}$	Trip
LaSalle 1	373/82-093	Scram and Multiple Failures	BWR	8/21/82	$1.1x10^{-4}$	Trip
LaSalle 1	373/83-057	Scram, LOFW with RCIC Inoperable	BWR	6/1/83	$2.1x10^{-5}$	Trip
Maine Yankee	309/83-002	Trip with MFW Inoperable and One Isolated Steam Generator	PWR	1/25/83	$8.6x10^{-5}$	Trip
McGuire 1	369/82-052	Loss of Vital I & C Bus and Trip	PWR	6/13/82	$3.1x10^{-6}$	Trip
Oconee 3	287/83-011	Trip with Loss of Main Feedwater and One AFW Pump Inoperable	PWR	10/13/83	$3.2x10^{-5}$	Trip
Peach Bottom 3	278/83-002 278/83-003	Trip with HPCI and ESF Bus 23 Inoperable	BWR	1/26/83	$3.4x10^{-5}$	Trip
Pilgrim	293/82-023 293/82-024	Trip with HPCI Failed: Controller Failure Resulting from Spray	BWR	8/13/82	$2.9x10^{-5}$	Trip
Pilgrim	293/83-007	LOOP during Shutdown	BWR	2/13/83	$9.7x10^{-5}$	LOOP
Quad Cities 1 (Impact on Unit 2)	254/82-012 254/82-013 254/82-018	Plant-Centered LOOP with One EDG Inoperable (Unit 1 Event Affected Unit 2)	BWR	6/22/82	$1.1x10^{-4}$	LOOP

Table 3.5 Precursors Involving Initiating Events Sorted by Plant (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Robinson 2	261/83-004 261/83-005 261/83-007 261/83-016	Trip with One AFW Pump Inoperable and One PORV Inoperable	PWR	4/19/83	$9.2x10^{-4}$	Trip
Salem 1	272/83-011 272/83-012 272/83-013	Trip with Automatic Reactor Trip Capability Failed	PWR	2/22/83	$4.6x10^{-3}$	Trip
Salem 1	272/83-033 272/83-034	LOOP with Turbine-Driven AFW Pump Inoperable	PWR	8/11/83	$1.2x10^{-4}$	LOOP
Salem 2	311/83-001 311/82-072	Trip with One Automatic Trip Breaker Failing to Open	PWR	1/6/83	$4.4x10^{-4}$	Trip
Salem 2	311/83-041	Trip with Number 2A Vital Bus De-energized	PWR	8/1/83	$1.2x10^{-6}$	Trip
San Onofre 3	362/83-099	Trip with Turbine-Driven AFW Pump Inoperable	PWR	10/31/83	$1.5x10^{-5}$	Trip
Sequoyah 1	327/83-100	Trip with AFW Pumps Unavailable	PWR	7/11/83	$5.7x10^{-6}$	Trip
St. Lucie 1	335/82-040	Trip with Loss of Grid Synchronization due to Shorted Generator Relay	PWR	9/2/82	$3.1x10^{-5}$	LOOP
St. Lucie 1	335/82-062	Trip with Inadvertent Safety Injection and Loss of Vital Power Supplies	PWR	11/26/82	$5.6x10^{-6}$	Trip
Susquehanna 1	387/83-051	Trip with RCIC System Unavailable Owing to Governor Valve Problem	BWR	3/22/83	$1.2x10^{-5}$	Trip
Susquehanna 1	387/83-120	Trip with RCIC System Unavailable Owing to Governor Valve Problem	BWR	8/28/83	$1.2x10^{-5}$	Trip
Trojan	344/83-002	Trip with MFW and Two AFW Pumps Unavailable	PWR	1/22/83	$9.7x10^{-5}$	Trip

Results

Table 3.5 Precursors Involving Initiating Events Sorted by Plant (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Trojan	344/83-012	AFW Pump Trip Following Reactor Trip	PWR	8/20/83	$3.0x10^{-5}$	Trip

Table 3.6 Precursors Involving Windowed Initiating Events Sorted by Plant

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
ANO 2	368/83-007 368/83-011 368/83-012	Trip with One Train of EFW Inoperable	PWR	2/14/83	4.1×10^{-5}	Trip
Beaver Valley 1	334/82-024	Trip with two CCW Pumps Inoperable	PWR	7/18/82	3.5×10^{-6}	Trip
Beaver Valley 1	334/83-008	Trip with Turbine-Driven AFW Pump Inoperable	PWR	2/18/83	5.9×10^{-6}	Trip
Brunswick 2	324/82-029	Scram with RHRSW System Degradations	BWR	2/3/82	3.4×10^{-5}	Trip
Brunswick 1	325/82-069	Scram with RCIC Inoperable	BWR	7/15/82	3.3×10^{-6}	Trip
Calvert Cliffs 1	317/82-054	Trip with One Turbine-Driven AFW Pump Inoperable	PWR	8/31/82	2.9×10^{-6}	Trip
Calvert Cliffs 1	317/83-076	Trip with the Motor-Driven AFW Pump Inoperable	PWR	12/30/83	7.7×10^{-6}	Trip
Calvert Cliffs 2	318/83-061	Trip with One LPSI Pump Inoperable	PWR	11/7/83	2.5×10^{-6}	Trip
Cooper	298/83-014	Trip with HPCI Unavailable	BWR	9/15/83	6.2×10^{-6}	Trip
Crystal River 3	302/82-041 302/82-051 302/83-037	Trip with One RHR Train Inoperable	PWR	6/8/82	4.8×10^{-6}	Trip
Crystal River 3	302/83-056 302/83-057	Trip with Turbine-Driven AFW Pump Inoperable	PWR	11/22/83	9.5×10^{-6}	Trip
Dresden 2	237/83-045 237/83-046 237/83-052	Core Spray A, LPCI A, and SDC A Inoperable; Scram	BWR	6/8/83	3.3×10^{-6}	Trip
Farley 2	364/82-022	Trip with One HPI Pump Inoperable	PWR	5/19/82	1.6×10^{-6}	Trip
Fitzpatrick	333/82-009	Trip with HPCI System Inoperable	BWR	2/10/82	4.8×10^{-6}	Trip
Hatch 1	321/83-122	Trip with HPCI Inoperable	BWR	12/28/83	6.5×10^{-6}	TRIP

Results

Table 3.6 Precursors Involving Windowed Initiating Events Sorted by Plant

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Indian Point 2	247/82-019 247/82-020	Trip with Two AFW Pumps Inoperable	PWR	5/17/82	$1.2x10^{-4}$	Trip
Indian Point 2	247/83-005	Trip with Turbine-Driven AFW Pump Inoperable	PWR	3/8/83	$3.9x10^{-6}$	Trip
LaSalle 1	373/82-107 373/82-099	Scram with RCIC and CRD Inoperable	BWR	8/12/82	$5.7x10^{-6}$	Trip
LaSalle 1	373/83-117 373/83-147	B RHR Heat Exchanger Outlet Valve Failure	BWR	9/30/83	$1.4x10^{-4}$	Trip
North Anna 1	338/82-021	Trip with One AFW Pump Inoperable	PWR	4/16/82	$1.8x10^{-6}$	Trip
North Anna 2	339/82-061	Trip with One LPI Pump Inoperable	PWR	9/5/82	$1.1x10^{-6}$	Trip
Palisades	255/82-002	Trip with AFW Auto-Initiation Inoperable	PWR	1/6/82	$5.0x10^{-6}$	Trip
Peach Bottom 2	277/83-028	Trip with Two HPSW Pumps Inoperable	BWR	12/23/83	$7.7x10^{-6}$	Trip
Peach Bottom 3	278/82-004	Trip with One LPCS and RHR Pump Inoperable	BWR	4/10/82	$3.3x10^{-6}$	Trip
Pilgrim	293/83-039	Trip with HPCI Inoperable	BWR	7/2/83	$5.2x10^{-6}$	Trip
Pilgrim	293/83-052	Trip with HPCI Inoperable	BWR	9/23/83	$5.2x10^{-6}$	Trip
Quad Cities 1	254/82-007 254/82-009	Trip with RHRSW Train B Inoperable	BWR	4/15/82	$1.7x10^{-4}$	Trip
Quad Cities 2	265/82-010	Trip with HPCI Inoperable	BWR	6/24/82	$4.7x10^{-6}$	Trip
Salem 1	272/82-041	Trip with Two Charging Pumps Inoperable	PWR	6/26/82	$1.1x10^{-6}$	Trip
Salem 1	272/82-056 272/82-053	Trip with One AFW Pump and One EDG Inoperable	PWR	7/31/82	$9.8x10^{-6}$	Trip
Salem 1	272/82-069	Trip with One Charging Pump Inoperable	PWR	8/31/82	$1.1x10^{-6}$	Trip
Salem 2	311/82-126	Trip with Two Charging Pumps Inoperable	PWR	10/18/82	$1.1x10^{-6}$	Trip

Table 3.6 Precursors Involving Windowed Initiating Events Sorted by Plant (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
San Onofre 2	361/83-063	Trip with Motor-Driven AFW Pump Inoperable	PWR	6/21/83	1.1×10^{-5}	Trip
Sequoyah 1	327/83-077	Trip with One Motor-Driven AFW Pump Unavailable	PWR	5/31/83	9.8×10^{-6}	Trip
St. Lucie 2	389/83-037 389/83-039	Trip with EDG Failure and Turbine-Driven AFW Pump Unavailable	PWR	7/28/83	3.7×10^{-6}	Trip
Summer	395/83-045	Trip with TDAFW Pump Inoperable Due to Incorrectly Set Speed Control	PWR	5/31/83	4.6×10^{-6}	Trip
Surry 2	281/83-005	Trip with AFW Pump Inoperable	PWR	2/11/83	3.8×10^{-6}	Trip
Surry 2	281/83-055	Trip with AFW Pump Inoperable	PWR	11/18/83	3.5×10^{-5}	Trip
Susquehanna 1	387/82-061	Trip with ESW Pumps B and D failed	BWR	12/22/82	4.3×10^{-5}	Trip
Susquehanna 1	387/83-103	Trip with RCIC System Unavailable Owing to Governor Valve Problem	BWR	7/7/83	1.4×10^{-5}	Trip
Susquehanna 1	387/83-106	Trip with HPCI Pump Failed	BWR	8/2/83	6.2×10^{-6}	Trip
Turkey Point 3	250/82-008	Trip with High Head Safety Injection Pump Failure	PWR	6/9/82	3.3×10^{-6}	Trip
Vermont Yankee	271/82-019	Trip with HPCI Inoperable	BWR	8/19/82	6.1×10^{-6}	Trip

Results

Table 3.7 Precursors Involving Unavailabilities Sorted by CCDP

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)[a]	Event Type
Hatch 2	366/82-084 366/82-085	RHRSW Pumps A, C, and D Failed	BWR	8/13/82	$2.4x10^{-4}$	Unavail.
Quad Cities 1[b]	254/82-012 254/82-013 254/82-018	Postulated LOOP with Two EDGs Inoperable (Unit 1)	BWR	6/22/82	$1.4x10^{-4}$	Unavail.
Turkey Point 3	250/83-007	Three AFW Pumps Unavailable	PWR	4/19/83	$5.5x10^{-5}$	Unavail.
Zion 2	304/83-007	Postulated Grid/Weather-Related LOOP with 2 EDGs Unavailable	PWR	1/31/83	$4.8x10^{-5}$	Unavail.
Brunswick 1	325/82-041	Both RHRSW Loops Simultaneously Inoperable	BWR	3/25/82	$4.7x10^{-5}$	Unavail.
Peach Bottom 3	278/83-009	Two EDGs Inoperable	BWR	9/8/83	$3.5x10^{-5}$	Unavail.
Zion 2	304/82-009	Unavailability of Both Motor-Driven AFW Pumps	PWR	4/9/82	$3.4x10^{-5}$	Unavail.
Sequoyah 1	327/83-183 327/83-186	One EDG and Turbine-Driven AFW Pump Inoperable	PWR	11/17/83	$3.1x10^{-5}$	Unavail.
Crystal River 3	302/82-007	Unavailability of Two EDGs	PWR	1/23/82	$2.8x10^{-5}$	Unavail.
Sequoyah 1	327/82-048 327/82-050	One Motor-Driven AFW Pump and One EDG Inoperable	PWR	3/29/82	$2.6x10^{-5}$	Unavail.
Sequoyah 1	327/83-063	Failure of Pressure-Operated Relief Valve	PWR	4/21/83	$1.9x10^{-5}$	Unavail.
Duane Arnold	331/83-017 331/83-018	HPCI and RHRSW Train B Inoperable	BWR	5/23/83	$1.7x10^{-5}$	Unavail.
Hatch 2	366/82-095	RHRSW loops A and B Unavailable	BWR	8/17/82	$7.7x10^{-6}$	Unavail.
Pilgrim	293/82-043 293/82-042	RCIC and HPCI Suction Valves Inoperable	BWR	9/30/82	$5.8x10^{-6}$	Unavail.

Table 3.7 Precursors Involving Unavailabilities Sorted by CCDP (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)[a]	Event Type
Calvert Cliffs 1	317/83-046 317/83-049	One EDG and One Turbine-Driven AFW Pump Inoperable	PWR	8/16/83	$5.8x10^{-6}$	Unavail.
Fort Calhoun	285/82-009	Three of Four CCW Heat Exchangers Inoperable	PWR	4/11/82	$5.7x10^{-6}$	Unavail.
North Anna 2	339/82-009	PORVs Inoperable due to Low Nitrogen Pressure	PWR	3/8/82	$4.3x10^{-6}$	Unavail.
Zion 1	295/82-025	Postulated Grid/Weather-Related LOOP with Two EDGs Inoperable for 24 Hours	PWR	8/11/82	$3.8x10^{-6}$	Unavail.
Quad Cities 2	265/82-017 265/82-018	HPCI and One EDG Inoperable	BWR	10/1/82	$3.6x10^{-6}$	Unavail.
Hatch 1	321/82-070	HPCI and RCIC Simultaneously Unavailable	BWR	8/5/82	$3.4x10^{-6}$	Unavail.
Hatch 1	321/82-088	HPCI and RCIC Unavailable	BWR	9/24/82	$2.5x10^{-6}$	Unavail.
Prairie Island 1	282/82-015	Two EDGs Simultaneously Inoperable for 1.5 Hours	PWR	8/27/82	$2.3x10^{-6}$	Unavail.
Cook 2	316/82-011	ESW Header and ECCS Train A Inoperable	PWR	1/28/82	$1.5x10^{-6}$	Unavail.
Cook 2	316/82-113	Long-Term Unavailability of SI Train B	PWR	12/29/82	$1.4x10^{-6}$	Unavail.
Zion 1	295/82-033	Postulated Grid/Weather-Related LOOP with Two EDGs Inoperable for 24 Hours	PWR	10/15/82	$1.4x10^{-6}$	Unavail.
Summer	395/83-019	Both RHR Trains and One HPI Train Inoperable	PWR	3/17/83	$1.0x10^{-6}$	Unavail.

[a] For unavailabilities, the p(cd) represents the increase in core damage probability over the duration of the event.
[b] The impact of this event on Quad Cities 2 is treated as a separate precursor, but is listed under the same LER number (see Table 3.5).

Table 3.8 Precursors Involving Initiating Events Sorted by CCDP

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Salem 1	272/83-011 272/83-012 272/83-013	Trip with Automatic Reactor Trip Capability Failed and Possible Positive MTC	PWR	2/22/83	$4.6x10^{-3}$	Trip
Robinson 2	261/83-004 261/83-005 261/83-007 261/83-016	Trip with One AFW Pump Inoperable and One PORV Inoperable	PWR	4/19/83	$9.2x10^{-4}$	Trip
Salem 2	311/83-001 311/82-072	Trip with One Automatic Trip Breaker Failing to Open	PWR	1/6/83	$4.4x10^{-4}$	Trip
Ginna	244/82-003 244/82-005	Steam Generator Tube Rupture with One PORV Failed Open	PWR	1/25/82	$3.8x10^{-4}$	SGTR
Brunswick 2	324/82-005	Scram with Both RHRSW Loops Inoperable	BWR	2/16/82	$2.3x10^{-4}$	Trip
Hatch 2	366/83-042 366/83-055 366/83-056	Reactor Trip with RCIC and RHR Loop A Unavailable	BWR	7/14/83	$1.5x10^{-4}$	Trip
Hatch 2	366/83-084	Scram with RHR Loop A Unavailable	BWR	8/17/83	$1.4x10^{-4}$	Trip
Salem 1	272/83-033 272/83-034	LOOP with Turbine-Driven AFW Pump Inoperable	PWR	8/11/83	$1.2x10^{-4}$	LOOP
Quad Cities 1 (impact on Unit 2)	254/82-012 254/82-013 254/82-018	Plant-Centered LOOP with One EDG Inoperable (Unit 1 Event Affected Unit 2)	BWR	6/22/82	$1.1x10^{-4}$	LOOP
LaSalle 1	373/82-093	Scram and Multiple Failures	BWR	8/21/82	$1.1x10^{-4}$	Trip
Pilgrim	293/83-007	LOOP during Shutdown	BWR	2/13/83	$9.7x10^{-5}$	LOOP
Trojan	344/83-002	Trip with MFW and Two AFW Pumps Unavailable	PWR	1/22/83	$9.7x10^{-5}$	Trip
Maine Yankee	309/83-002	Trip with MFW Inoperable and One Isolated Steam Generator	PWR	1/25/83	$8.6x10^{-5}$	Trip
Davis-Besse 1	346/83-038 346/83-040	Trip with AFW Pump Inoperable and Failure of Two SFRCS Channels	PWR	7/25/83	$8.2x10^{-5}$	Trip

Table 3.8 Precursors Involving Initiating Events Sorted by CCDP (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Browns Ferry 1	259/83-006 259/83-007	Scram, MSRV and Its Vacuum Breaker Fail Open	BWR	2/5/83	4.4×10^{-5}	LOCA
Peach Bottom 3	278/83-002 278/83-003	Trip with HPCI and ESF Bus 23 Inoperable	BWR	1/26/83	3.4×10^{-5}	Trip
Browns Ferry 2	260/83-074	Trip with HPCI Inoperable	BWR	11/10/83	3.2×10^{-5}	Trip
Oconee 3	287/83-011	Trip with Loss of Main Feedwater and One AFW Pump Inoperable	PWR	10/13/83	3.2×10^{-5}	Trip
St. Lucie 1	335/82-040	Trip with Loss of Grid Synchronization due to Shorted Generator Relay	PWR	9/2/82	3.1×10^{-5}	LOOP
Trojan	344/83-012	AFW Pump Trip Following Reactor Trip	PWR	8/20/83	3.0×10^{-5}	Trip
Pilgrim	293/82-023 293/82-024	Trip with HPCI Failed: Controller Failure Resulting from Spray	BWR	8/13/82	2.9×10^{-5}	Trip
LaSalle 1	373/83-057	Scram, LOFW with RCIC Inoperable	BWR	6/1/83	2.1×10^{-5}	Trip
San Onofre 3	362/83-099	Trip with Turbine-Driven AFW Pump Inoperable	PWR	10/31/83	1.5×10^{-5}	Trip
Hatch 2	366/82-081	Scram, Isolation, RCIC Failure, SRV Tailpipe Vacuum Relief Failed	BWR	8/25/82	1.4×10^{-5}	LOCA
Brunswick 1	325/82-054	Scram with RCIC Inoperable	BWR	6/7/82	1.4×10^{-5}	Trip
Brunswick 1	325/82-025	Scram with RCIC Inoperable	BWR	2/18/82	1.3×10^{-5}	Trip
Hatch 1	321/83-090 321/83-093	Manual Scram with HPCI and RCIC Unavailable	BWR	8/25/83	1.3×10^{-5}	Trip
Brunswick 2	324/82-123	Scram with Emergency Bus E-3 De-energized	BWR	10/10/82	1.2×10^{-5}	Trip
Susquehanna 1	387/83-051	Trip with RCIC System Unavailable Owing to Governor Valve Problem	BWR	3/22/83	1.2×10^{-5}	Trip

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Table 3.8 Precursors Involving Initiating Events Sorted by CCDP (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Susquehanna 1	387/83-120	Trip with RCIC System Unavailable Owing to Governor Valve Problem	BWR	8/28/83	1.2×10^{-5}	Trip
Hatch 2	366/83-069	Scram with HPCI Unavailable	BWR	7/22/83	6.2×10^{-6}	Trip
Sequoyah 1	327/83-100	Trip with AFW Pumps Unavailable	PWR	7/11/83	5.7×10^{-6}	Trip
St. Lucie 1	335/82-062	Trip with Inadvertent Safety Injection and Loss of Vital Power Supplies	PWR	11/26/82	5.6×10^{-6}	Trip
ANO 1	313/83-014	Trip with AFW Pump P-75 Inoperable	PWR	6/9/83	4.7×10^{-6}	Trip
Hatch 1	321/82-011 321/82-012	Trip with RCIC Inoperable	BWR	2/12/82	3.3×10^{-6}	Trip
McGuire 1	369/82-052	Loss of Vital I & C Bus and Trip	PWR	6/13/82	3.1×10^{-6}	Trip
ANO 1	313/83-015	Trip with One HPI Injection Valve Failed	PWR	6/16/83	2.9×10^{-6}	Trip
Cook 2	316/82-072	Control Room Instrument Distribution Bus IV Fails, Trip	PWR	8/24/82	1.3×10^{-6}	Trip
Salem 2	311/83-041	Trip with Number 2A Vital Bus De-energized	PWR	8/1/83	1.2×10^{-6}	Trip
Cook 2	316/83-052	Control Room Instrument Distribution Bus IV Fails, Trip	PWR	6/23/83	1.0×10^{-6}	Trip

Table 3.9 Precursors Involving Windowed Initiating Events Sorted by CCDP

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Quad Cities 1	254/82-007 254/82-009	Trip with RHRSW train B Inoperable	BWR	4/15/82	1.7×10^{-4}	Trip
LaSalle 1	373/83-117 373/83-147	B RHR Heat Exchanger Outlet Valve Failure	BWR	9/30/83	1.4×10^{-4}	Trip
Indian Point 2	247/82-019 247/82-020	Trip with Two AFW Pumps Inoperable	PWR	5/17/82	1.2×10^{-4}	Trip
Susquehanna 1	387/82-061	Trip with ESW Pumps B and D Failed	BWR	12/22/82	4.3×10^{-5}	Trip
ANO 2	368/83-007 368/83-011 368/83-012	Trip with One Train of EFW Inoperable	PWR	2/14/83	4.1×10^{-5}	Trip
Surry 2	281/83-055	Trip with AFW Pump Inoperable	PWR	11/18/83	3.5×10^{-5}	Trip
Brunswick 2	324/82-029	Scram with RHRSW System Degradations	BWR	2/3/82	3.4×10^{-5}	Trip
Susquehanna 1	387/83-103	Trip with RCIC System Unavailable Owing to Governor Valve Problem	BWR	7/7/83	1.4×10^{-5}	Trip
San Onofre 2	361/83-063	Trip with Motor-Driven AFW Pump Inoperable	PWR	6/21/83	1.1×10^{-5}	Trip
Salem 1	272/82-056 272/82-053	Trip with One AFW Pump and One EDG Inoperable	PWR	7/31/82	9.8×10^{-6}	Trip
Sequoyah 1	327/83-077	Trip with One Motor-Driven AFW Pump Unavailable	PWR	5/31/83	9.8×10^{-6}	Trip
Crystal River 3	302/83-056 302/83-057	Trip with Turbine-Driven AFW Pump Inoperable	PWR	11/22/83	9.5×10^{-6}	Trip
Calvert Cliffs 1	317/83-076	Trip with the Motor-Driven AFW Pump Inoperable	PWR	12/30/83	7.7×10^{-6}	Trip
Peach Bottom 2	277/83-028	Trip with Two HPSW Pumps Inoperable	BWR	12/23/83	7.7×10^{-6}	Trip
Hatch 1	321/83-122	Trip with HPCI Inoperable	BWR	12/28/83	6.5×10^{-6}	Trip
Susquehanna 1	387/83-106	Trip with HPCI Pump Failed	BWR	8/2/83	6.2×10^{-6}	Trip

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Table 3.9 Precursors Involving Windowed Initiating Events Sorted by CCDP (Cont.)

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	Event Type
Cooper	298/83-014	Trip with HPCI Unavailable	BWR	9/15/83	$6.2x10^{-6}$	Trip
Vermont Yankee	271/82-019	Trip with HPCI Inoperable	BWR	8/19/82	$6.1x10^{-6}$	Trip
Beaver Valley 1	334/83-008	Trip with Turbine-Driven AFW Pump Inoperable	PWR	2/18/83	$5.9x10^{-6}$	Trip
LaSalle 1	373/82-107 373/82-099	Scram with RCIC and CRD Inoperable	BWR	8/12/82	$5.7x10^{-6}$	Trip
Pilgrim	293/83-039	Trip with HPCI Inoperable	BWR	7/2/83	$5.2x10^{-6}$	Trip
Pilgrim	293/83-052	Trip with HPCI Inoperable	BWR	9/23/83	$5.2x10^{-6}$	Trip
Palisades	255/82-002	Trip with AFW Auto-Initiation Inoperable	PWR	1/6/82	$5.0x10^{-6}$	Trip
Fitzpatrick	333/82-009	Trip with HPCI System Inoperable	BWR	2/10/82	$4.8x10^{-6}$	Trip
Crystal River 3	302/82-041 302/82-051 302/83-037	Trip with One RHR Train Inoperable	PWR	6/8/82	$4.8x10^{-6}$	Trip
Quad Cities 2	265/82-010	Trip with HPCI Inoperable	BWR	6/24/82	$4.7x10^{-6}$	Trip
Summer	395/83-045	Trip with TDAFW Pump Inoperable due to Incorrectly Set Speed Control	PWR	5/31/83	$4.6x10^{-6}$	Trip
Indian Point 2	247/83-005	Trip with Turbine-Driven AFW Pump Inoperable	PWR	3/8/83	$3.9x10^{-6}$	Trip
Surry 2	281/83-005	Trip with AFW Pump Inoperable	PWR	2/11/83	$3.8x10^{-6}$	Trip
St. Lucie 2	389/83-037 389/83-039	Trip with EDG Failure and Turbine-Driven AFW Pump Unavailable	PWR	7/28/83	$3.7x10^{-6}$	Trip
Beaver Valley 1	334/82-024	Trip with Two CCW Pumps Inoperable	PWR	7/18/82	$3.5x10^{-6}$	Trip
Turkey Point 3	250/82-008	Trip with High Head Safety Injection Pump Failure	PWR	6/9/82	$3.3x10^{-6}$	Trip

4 Glossary

Accident. An unexpected event (frequently caused by equipment failure or some misoperation as the result of human error) that has undesirable consequences.

Accident sequence precursor. A historically observed element or condition in a postulated sequence of events leading to some undesirable consequence. For purposes of the ASP study, the undesirable consequence is usually severe core damage. The identification of an operational event as an accident sequence precursor does not of itself imply that a significant potential for severe core damage existed. It does mean that at least one of a series of protective features designed to prevent core damage was compromised. The likelihood of severe core damage, given that an accident sequence precursor occurred, depends on the effectiveness of the remaining protective features and, in the case of precursors that do not include initiating events, the probability of such an initiator.

Availability. The characteristic of an item expressed by the probability that it will be operational on demand or at a randomly selected future instant in time.

Common cause failures. Multiple failures attributable to a common cause.

Common mode failures. Multiple, concurrent, and dependent failures of identical equipment that fails in the same mode.

Components. Items from which equipment trains and/or systems are assembled (e.g., pumps, pipes, valves, and vessels).

Conditional probability. The probability of an outcome given that certain conditions exist.

Core damage. See *severe core damage.*

Core-melt accident. An event in a nuclear power plant in which core materials melt.

Coupled failure. A common cause or common mode failure of more than one piece of equipment. See *common cause failures* and *common mode failures.*

Degraded system. A system with failed components that still meets minimum operability standards.

Demand. A test or an operating condition that requires the availability of a component or a system. In this study, a demand includes actuations required during testing and because of initiating events. One demand is assumed to consist of the actuation of all redundant components in a system, even if these were actuated sequentially (as is typical in testing multiple-train systems).

Demand failure. A failure following a demand. A demand failure may be caused by a failure to actuate when required or a failure to run following actuation.

Dependent failure. A failure in which the likelihood of failure is influenced by the failure of other items. Common-cause failures and common-mode failures are two types of dependent failures.

Dominant sequence. The sequence in a set of sequences that has the highest probability of leading to a common end state.

Emergency core cooling system. Systems that provide for removal of heat from a reactor following either a loss of normal heat removal capability or a LOCA.

Engineered safety features. Equipment and/or systems (other than reactor trip or those used only for normal operation) designed to prevent, limit, or mitigate the release of radioactive material.

Event. An abnormal occurrence that is typically in violation of a plant's Technical Specifications.

Event sequence. A particular path on an event tree.

Event tree. A logic model that represents existing dependencies and combinations of actions required to achieve defined end states following an initiating event.

Failure. The inability to perform a required function. In this study, a failure was considered to have occurred if some component or system performed at a level below its required minimum performance level without human intervention. The likelihood of recovery was accounted for through the use of recovery factors. See *recovery factor*.

Failure probability. The long-term frequency of occurrence of failures of a component, system, or combination of systems to operate at a specified performance level when required. In this study, failure includes both failure to start and failure to operate once started.

Failure rate. The expected number of failures of a given type, per item, in a given time interval (e.g., capacitor short-circuit failures per million capacitor hours).

Front-line system. A system that directly provides a mitigative function included on the event trees used to model sequences to an undesired end state, in contrast to a support system, which is required for operability of other systems.

Immediately detectable. A failure is considered to be immediately detectable if it results in a plant response that is apparent at the time of the failure.

Independent. Two or more entities are said to be independent if they do not exhibit a common failure mode for a particular type of event.

Initial criticality. The date on which a plant goes critical for the first time in first-cycle operation.

Initiating event. An event that starts a transient response in the operating plant systems. In the ASP study, the concern is only with those initiating events that could lead to severe core damage.

Licensee event reports. Those reports submitted to the NRC by utilities who operate nuclear plants as described in 10 CFR 50.73. LERs describe abnormal operating occurrences at plants where, generally, the Technical Specifications have been violated.

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Multiple failure events. Events in which more than one failure occurs. These may involve independent or dependent failures.

Operational event. An event that occurs in a plant and generally constitutes a reportable occurrence under 10 CFR 50.73 as an LER.

Postulated event. An event that may happen at some time in the course of plant life.

Potential severe core damage. A plant operating condition in which, following an initiating event, one or more protective functions fail to meet minimum operability requirements over a period sufficiently long that core damage could occur. This condition has been called in other studies "core melt," "core damage," and "severe core damage," even though actual core damage may not result unless further degradation of mitigation functions occurs.

Precursor. See *accident sequence precursor.*

Reactor years. The accumulated total number of years of reactor operation. For the ASP study, operating time starts when a reactor goes critical, ends when it is permanently shut down, and includes all intervening outages and plant shutdowns.

Recovery factor (recovery class). A measure of the likelihood of not recovering a failure. Failures were assigned to a particular recovery class based on an assessment of the likelihood that recovery would not be affected, given event specifics. Considered in the likelihood of recovery was whether such recovery would be required in a moderate- to high-stress situation following a postulated initiating event.

Redundant equipment or system. A system or some equipment that duplicates the essential function of another system or other equipment to the extent that either may perform the required function regardless of the state of operation or failure of the other.

Reliability. The characteristic of an item expressed by the probability that it will perform a required function under stated conditions for a stated period of time.

Risk. A measure of the frequency and severity of undesired effects.

Sensitivity analysis. An analysis that determines the variation of a given function caused by changes in one or more parameters about a selected reference value.

Severe core damage. The result of an event in which inadequate core cooling was provided, resulting in damage to the reactor core. See *potential severe core damage.*

Technical Specifications. A set of safety-related limits on process variables, control system settings, safety system settings, and the performance levels of equipment that are included as conditions of an operating license.

Unavailability. The probability that an item or system will not be operational at a future instant in time. Unavailability may be a result of the item being tested or may occur as a result of malfunctions. Unavailability is the complement of availability.

Unit. A nuclear steam supply, its associated turbine generator, auxiliaries, and ESFs.

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5 References

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* Available for purchase from National Technical Information Service, Springfield, Virginia, 22161.

Results

Appendix A:
ASP MODELS

A.0 ASP Models

This appendix describes the methods and models used to estimate the significance of 1982-83 precursors. The modeling approach is similar to that used to evaluate 1984-91 operational events. Simplified train-based models are used, in conjunction with a simplified recovery model, to estimate system failure probabilities specific to an operational event. These probabilities are then used in event tree models that describe core damage sequences relevant to the event. The event trees have been expanded beyond those used in the analysis of 1984-91 events to address features of the ASP models used to assess 1994 operational events (Ref. 1) known to have existed in the 1982-83 time period.

A.1 Precursor Significance Estimation

The ASP Program performs retrospective analyses of operating experience. These analyses require that certain methodological assumptions be made in order to estimate the risk significance of an event. If one assumes, following an operational event in which core cooling was successful, that components observed to be failed were "failed" with a probability of 1.0, and components that functioned successfully were "successful" with a probability of 1.0, then one can conclude that the risk of core damage was zero, and that the only potential sequence was the combination of events that occurred. In order to avoid such trivial results, the status of certain components must be considered latent. In the ASP Program, this latency is associated with components that operated successfully—these components are considered to have been capable of failing during the operational event.

Quantification of precursor significance involves the determination of a conditional probability of subsequent core damage given the failures and other undesirable conditions (such as an initiating event or an unexpected relief valve challenge) observed during an operational event. The effect of a precursor on systems addressed in the core damage models is assessed by reviewing the specifics of the operational event against plant design and operating information, and translating the results of the review into a revised model for the plant that reflects the observed failures. The precursor's significance is estimated by calculating a conditional probability of core damage given the observed failures. The conditional probability calculated in this way is useful in ranking because it provides an estimate of the measure of protection against core damage remaining once the observed failures have occurred.

A.1.1 Types of Events Analyzed

Two different types of events are addressed in precursor quantitative analysis. In the first, an initiating event such as a loss of offsite power (LOOP) or a small-break loss of coolant accident (LOCA) occurs as a part of the precursor. The probability of core damage for this type of event is calculated based on the required plant response to the particular initiating event and other failures that may have occurred at the same time. This type of event includes the "windowed" 1982-83 events discussed in Section 2.2 of the main report.

The second type of event involves a failure condition that existed over a period of time during which an initiating event could have, but did not, occur. The probability of core damage is calculated based on the required plant response to a set of postulated initiating events, considering the failures that were observed. Unlike an initiating event assessment, where a particular initiating event is assumed to occur with a probability of 1.0, each initiating event is assumed to occur with a probability based on the initiating event frequency and the failure duration.

A.1.2 Modification of System Failure Probabilities to Reflect Observed Failures

The ASP models used to evaluate 1982-83 operational events describe sequences to core damage in terms of combinations of success and failure of mitigating systems following an initiating event. Each system model represents those combinations of train or component failures that will result in system failure. Failures observed during an operational event must be represented in terms of changes to one or more of the potential failures included in the system models.

If a failed component is included in one of the trains in the system model, the failure is reflected by setting the probability for the affected train to 1.0. Redundant train failure probabilities are conditional, which allows potential common cause failures to be addressed. If the observed failure could have occurred in other similar components at the same time, then the system failure probability is increased to represent this. If the failure could not simultaneously occur in other components (for example, if a component was removed from service for preventive maintenance), then the system failure probability is also revised, but only to reflect the "removal" of the unavailable component from the model.

If a failed component is not specifically included as an event in a model, then the failure is addressed by setting elements affected by the failure to failed. For example, support systems are not completely developed in the 1982-83 ASP models. A breaker failure that results in the loss of power to a group of components would be represented by setting the elements associated with each component in the group to failed.

Occasionally, a precursor occurs that cannot be modeled by modifying probabilities in existing system models. In such a case, the model is revised as necessary to address the event, typically by adding events to the system model or by addressing an unusual initiating event through the use of an additional event tree.

A.1.3 Recovery from Observed Failures

The models used to evaluate 1982-83 events address the potential for recovery of an entire system if the system fails. This is the same approach that was used in the analysis of most precursors through 1991.[1] In this approach, the potential for recovery is addressed by assigning a recovery action to each system failure and initiating event. Four classes were used to describe the different types of short-term recovery that could be involved:

[1] Later precursor analyses utilize time-reliability correlations to estimate the probability of failing to recover a failed system when recovery is dominated by operator action.

Recovery Class	Likelihood of Non-recovery[2]	Recovery Characteristic
R1	1.00	The failure did not appear to be recoverable in the required period, either from the control room or at the failed equipment.
R2	0.55	The failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from the control room did not appear possible.
R3	0.10	The failure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial operator burden.
R4	0.01	The failure appeared recoverable in the required period from the control room and was considered routine and procedurally based.

The assignment of an event to a recovery class is based on engineering judgment, which considers the specifics of each operational event and the likelihood of not recovering from the observed failure in a moderate to high-stress situation following an initiating event.

Substantial time is usually available to recover a failed residual heat removal (RHR) or BWR power conversion system (PCS). For these systems, the nonrecovery probabilities listed above are overly pessimistic. Data in Refs. 2 and 3 were used to estimate the following nonrecovery probabilities for these systems:

System	p(nonrecovery)
BWR RHR system	0.016 (0.054 if failures involve service water)
BWR PCS	0.52 (0.017 for MSIV closure)
PWR RHR system	0.057

It must be noted that the actual likelihood of failing to recover from an event at a particular plant is difficult to assess and may vary substantially from the values listed. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, etc., concerning the likelihood of recovering from specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.

A.1-4 Conditional Probability Associated with Each Precursor

As described earlier in this appendix, the calculation process for each precursor involves a determination of initiators that must be modeled, plus any modifications to system probabilities necessitated by failures observed

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[2]These nonrecovery probabilities are consistent with values specified in M.B. Sattison *et al.*, "Methods Improvements Incorporated into the SAPHIRE ASP Models," *in Proceedings of the U.S. Nuclear Regulatory Commission Twenty-Second Water Reactor Safety Information Meeting*, NUREG/CP-0140, Vol. 1, April 1995.

in an operational event. Once the probabilities that reflect the conditions of the precursor are established, the sequences leading to core damage are calculated to estimate the conditional probability for the precursor. This calculation process is summarized in Table A.1.

Several simplified examples that illustrate the basics of the precursor calculation process follow. The examples are not intended to describe a detailed precursor analysis, but instead to provide a basic understanding of the process.

The hypothetical core damage model for these examples, shown in Fig. A.1, consists of initiator I and four systems that provide protection against core damage: systems A, B, C, and D. In Fig. A.1, the up branch represents success and the down branch represents failure for each of the systems. Three sequences result in core damage if completed: sequence 3 [I /A ("/" represents system success) C D], sequence 6 (I A /B C D) and sequence 7 (I A B). In a conventional PRA approach, the frequency of core damage would be calculated using the frequency of the initiating event I, $\lambda(I)$, and the failure probabilities for A, B, C, and D [p(A), p(B), p(C), and p(D)]. Assuming $\lambda(I) = 0.1\ yr^{-1}$ and $p(A|I) = 0.003$, $p(B|IA) = 0.01$, $p(C|I) = 0.05$, and $p(D|IC) = 0.1$,[3] the frequency of core damage is determined by calculating the frequency of each of the three core damage sequences and adding the frequencies:

$$
\begin{aligned}
&0.1\ yr^{-1} \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)}\\
&+ 0.1\ yr^{-1} \times 0.003 \times (1 - 0.01) \times 0.05 \times 0.1 \text{ (sequence 6)}\\
&+ 0.1\ yr^{-1} \times 0.003 \times 0.01 \text{ (sequence 7)}\\
&= 4.99 \times 10^{-4}\ yr^{-1} \text{ (sequence 3)} + 1.49 \times 10^{-6}\ yr^{-1} \text{ (sequence 6)}\\
&+ 3.00 \times 10^{-6}\ yr^{-1} \text{ (sequence 7)} = 5.03 \times 10^{-4}\ yr^{-1}.
\end{aligned}
$$

In a nominal PRA, sequence 3 would be the dominant core damage sequence.

The ASP program calculates a conditional probability of core damage, given an initiating event or component failures. This probability is different than the frequency calculated above and cannot be directly compared with it.

Example 1. Initiating Event Assessment. Assume that a precursor involving initiating event I occurs. In response to I, systems A, B, and C start and operate correctly and system D is not demanded. In a precursor initiating event assessment, the probability of I is set to 1.0. Although systems A, B, and C were successful, nominal failure probabilities are assumed. Since system D was not demanded, a nominal failure probability is assumed for it as well. The conditional probability of core damage associated with precursor I is calculated by summing the conditional probabilities for the three sequences:

$$
\begin{aligned}
&1.0 \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)}\\
&+ 1.0 \times 0.003 \times (1 - 0.010) \times 0.05 \times 0.1 \text{ (sequence 6)}\\
&+ 1.0 \times 0.003 \times 0.01 \text{ (sequence 7)}\\
&= 5.03 \times 10^{-3}.
\end{aligned}
$$

[3] The notation p(B|IA) means the probability that B fails, given I occurred and A failed.

If, instead, B had failed when demanded, its probability would have been set to 1.0. The conditional core damage probability for precursor IB would be calculated as

$$1.0 \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} + 1.0 \times 0.003 \times 1.0 \text{ (sequence 7)} = 7.99 \times 10^{-3}.$$

Since B failed, sequence 6 cannot occur.

<u>Example 2. Condition Assessment</u>. Assume that during a monthly test, system B is found to be failed, and that the failure could have occurred at any time during the month. The best estimate for the duration of the failure is half of the test period, or 360 hours. To estimate the probability of initiating event I during the 360-hour period, the yearly frequency of I must be converted to an hourly rate. If I can only occur at power, and the plant is at power for 70% of a year, then the frequency for I is estimated to be $0.1\ yr^{-1}/(8760\ h/yr \times 0.7) = 1.63 \times 10^{-5}\ h^{-1}$.

If, as in example 1, B is always demanded following I, the probability of I in the 360-hour period is the probability that at least one I occurs (since the failure of B will then be discovered), or

$$1 - e^{-\lambda(I) \times \text{failure duration}} = 1 - e^{-1.63E\text{-}5 \times 360} = 5.85 \times 10^{-3}.$$

Using this value for the probability of I, and setting p(B) = 1.0, the conditional probability of core damage for precursor B is calculated by again summing the conditional probabilities for the core damage sequences in Fig. A.1:

$$5.85 \times 10^{-3} \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} + 5.85 \times 10^{-3} \times 0.003 \times 1.0 \text{ (sequence 7)}$$

$$= 4.67 \times 10^{-5}.$$

As before, since B failed, sequence 6 cannot occur. The conditional probability is the probability of core damage in the 360-hour period, given the failure of B. Note that the dominant core damage sequence is sequence 3, with a conditional probability of 2.92×10^{-5}. This sequence is unrelated to the failure of B. The potential failure of systems C and D over the 360-hour period still drive the core damage risk.

To understand the significance of the failure of system B, another calculation, an importance measure, is required. The importance measure that is used is equivalent to risk achievement worth on an interval scale (see Ref. 4). In this calculation, the increase in core damage probability over the 360-hour period due to the failure of B is estimated: p(cd | B) - p(cd). For this example the value is $4.67 \times 10^{-5} - 2.94 \times 10^{-5} = 1.73 \times 10^{-5}$, where the second term on the left side of the equation is calculated using the previously developed probability of I in the 360-hour period and nominal failure probabilities for A, B, C, and D.

For most conditions identified as precursors in the ASP Program, the importance and the conditional core damage probability are numerically close, and either can be used as a significance measure for the precursor. However, for some events—typically those in which the components that are failed are not the primary mitigating plant features—the conditional core damage probability can be significantly higher than the importance. In such cases, it is important to note that the potential failures of other components, unrelated to the precursor, are still dominating the plant risk.

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The importance measure for unavailabilities (condition assessments) like this example event were previously referred to as a "conditional core damage probability" in annual precursor reports before 1994, instead of as the increase in core damage probability over the duration of the unavailability. Because the computer code used to analyze 1982-83 events is the same as was used for 1984-93 evaluations, the results for 1982-83 conditions are also presented in the computer output in terms of "conditional probability," when in actuality the result is an importance.

A.2 Overview of 1982-83 ASP Models

Models used to rank 1982-83 precursors as to significance consist of system-based plant-class event trees and simplified plant-specific system models. These models describe mitigation sequences for the following initiating events: a nonspecific reactor trip [which includes loss of feedwater (LOFW) within the model], LOOP, small-break LOCA, and steam generator tube rupture [SGTR, pressurized water reactors (PWRs) only].

Plant classes were defined based on the use of similar systems in providing protective functions in response to transients, LOOPs, and small-break LOCAs. System designs and specific nomenclature may differ among plants included in a particular class but functionally they are similar in response. Plants in which certain mitigating systems do not exist, but which are largely analogous in their initiator response, are grouped into the appropriate plant class. ASP plant categorization is described in the following section.

The event trees consider two end states: success (OK), in which core cooling exists, and core damage (CD), in which adequate core cooling is believed not to exist. In the ASP models, core damage is assumed to occur following core uncovery. It is acknowledged that cladding and fuel damage will occur at later times, depending on the criteria used to define "damage," and that time may be available to recover core cooling once core uncovery occurs but before the onset of core damage. However, this potential recovery is not addressed in the models. Each event tree describes combinations of system failures that will prevent core cooling, and makeup if required, in both the short and long term. Primary systems designed to provide these functions and alternative systems also capable of performing these functions are addressed. The event trees are described in Section A.4.

The models used to evaluate 1982-83 events consider both additional systems that can provide core protection and initiating events not included in the plant-class models used in the assessment of 1984-91 events, and only partially included in the assessment of 1992-93 events. Response to a failure to trip the reactor is now addressed, as is an SGTR in PWRs. In PWRs, the potential use of the residual heat removal system following a small-break LOCA (to avoid sump recirculation) is addressed, as is the potential recovery of secondary-side cooling in the long term following the initiation of feed and bleed. In boiling water reactors (BWRs), the potential use of reactor core isolation cooling (RCIC) and the control rod drive (CRD) system for makeup if a single relief valve sticks open is addressed, as is the potential long-term recovery of the power conversion system (PCS) for decay heat removal in BWRs. These models better reflect the capabilities of plant systems in preventing core damage.

A.3 Plant Categorization

It was recognized early in the ASP Program that plant designs were sufficiently different that multiple models would be required to correctly describe the impact of an operational event in different plants. In 1985, substantial effort was expended to develop a categorization scheme for all U.S. LWRs that would permit grouping of plants with similar responses to a transient or accident at the system or functional level, and to subsequently develop eight sets of plant-class-specific event tree models. Much of the categorization and early event sequence work was done at the University of Maryland (Refs. 5 and 6). The ASP Program has generally employed these categorizations; however, some modifications have been required to more closely reflect the specific needs of the precursor evaluations.

In developing the plant categorizations, each reactor plant was examined to determine the systems used to perform the following plant functions required in response to initiating events to prevent core damage: reactor subcriticality, reactor coolant system (RCS) integrity, reactor coolant inventory, short-term core heat removal, and long-term core heat removal.

Functions solely related to containment integrity (containment overpressure protection and containment heat removal) and postaccident removal of reactivity are not included in the present ASP models (which only concern core damage sequences) and are not addressed in the categorization scheme.

For each plant, the systems utilized to perform each function were identified. Plants were grouped based on the use of nominally identical systems to perform each function; that is, systems of the same type and function without accounting for the differences in the design of those systems.

Three BWR plant classes were defined. BWR Class A consists of the older plants, which are characterized by isolation condensers (ICs) and feedwater coolant injection (FWCI) systems that employ the main feedwater (MFW) pumps. BWR Class B consists of plants that have ICs and a separate high-pressure coolant injection (HPCI) system instead of FWCI. BWR Class C includes the modern plants that have neither ICs nor FWCI. However, they have an RCIC system that Classes A and B lack. The Class C plants could be separated into two subgroups: those plants with turbine-driven HPCI systems and those with motor-driven high-pressure core spray (HPCS) systems. This difference is addressed instead in the models of the different plant systems.

PWRs are separated into five classes. One class represents most Babcock & Wilcox Company plants (Class D). These plants have the capability of performing feed and bleed without the need to open the power-operated relief valve (PORV). Combustion Engineering plants are separated into two classes: those that provide feed-and-bleed capability (Class G) and those that provide for secondary-side depressurization and the use of the condensate system as an alternative core cooling method, and for which no feed and bleed is available (Class H).[4]

The remaining two classes address Westinghouse plants. Class A is associated with plants that require the use of spray systems for core heat removal following a LOCA, and Class B is associated with plants that can utilize low to high-pressure recirculation for core heat removal.

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[4]Maine Yankee was designed by Combustion Engineering, but has a response to initiating events more akin to Westinghouse plants, so it was grouped with other Class B plants. Davis-Besse was also placed in Class B because its high-pressure injection (HPI) system design requires the operator to open the PORV for feed and bleed, as in most Westinghouse plants. The requirement to open PORVs for feed and bleed is an important difference between models for Westinghouse and Babcock & Wilcox plants. Plant response differences resulting from the use of different steam generator (SG) designs are not addressed in the models.

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Table A.2 lists the class associated with each plant.

A.4 Event Tree Models

The plant class event trees describe core damage sequences for four initiating events: nonspecific reactor trip, LOOP, small-break LOCA, and SGTR (in PWRs only). A separate event tree describes anticipated transient without scram (ATWS) sequences. Failure-to-trip sequences on the transient event tree are transferred to this tree. The event trees constructed are system based and include an event tree applicable to each plant class defined. For operational events that cannot be described using existing models, unique models are developed to describe sequences to core damage.

This section (1) describes the potential plant response to the initiating events listed above; (2) identifies the combinations of systems required for the successful mitigation of each initiator; and (3) briefly describes the criteria for success of each system-based function. The sequences are considered first for PWRs and then separately for BWRs. The event trees for Class B and D PWRs apply to the greatest number of operating PWRs and are therefore discussed first, followed by those for Classes G, H, and then A. For the BWR event trees, the plant Class C models are described first, because these are applicable to the majority of the BWRs, followed by discussions for the A and B Classes of BWRs, respectively.

The event trees are constructed with branch success as the upper branch and failure as the lower branch. Relief valve opening and RCP seal LOCA are indicated by up branches in the 1982-83 models. Each sequence path is read from left to right, beginning with the initiator and followed by subsequent systems required to preclude or mitigate core damage. Each sequence represents a series of branch successes and failures required to reach the sequence end state (OK or CD). The sequence as depicted on the event tree represents the logical combination of successes and failures required to reach the end state; it does not necessarily represent the actual sequence in which systems and functions would respond to an initiating event. However, short-term plant response is generally presented earlier in the sequence than long-term plant response.

The event trees can be found following the discussion sections and are grouped according to plant classes, beginning with the PWR classes and followed by the BWR classes. The trees are presented in the order shown in the following list. The abbreviations used in the event tree models are defined in the event tree branch descriptions in this section.

The trees are presented in the following order:

Fig. No.	Event tree
A.2	PWR Class A nonspecific reactor trip
A.3	PWR Class A loss of offsite power
A.4	PWR Class A small-break loss-of-coolant accident
A.5	PWR Class A steam generator tube rupture
A.6	PWR Class A anticipated transient without scram
A.7	PWR Classes B and D nonspecific reactor trip
A.8	PWR Classes B and D loss of offsite power
A.9	PWR Classes B and D small-break loss-of-coolant accident
A.10	PWR Classes B and D steam generator tube rupture

A.4.1 PWR Event Tree Models

The PWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following four initiating events: reactor trip, LOOP, small-break LOCA, and SGTR. The systems modeled in the event trees are those associated with the generic functions required in response to an initiating event. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality	Reactor trip and boration (following ATWS)
Reactor coolant system integrity	Addressed in small-break LOCA, SGTR and ATWS models plus trip and LOOP sequences involving failure of primary relief valves to close and RCP seal LOCA
Reactor coolant inventory	High-pressure injection (assumed required only following a LOCA)
Short-term core heat removal	Auxiliary feedwater
	Main feedwater

	Feed and bleed (high-pressure injection and PORV, PWR Classes A, B, D, and G)
	Secondary-side depressurization and use of condensate system (PWR Class H)
Long-term core heat removal	Auxiliary feedwater
	Main feedwater
	RCS cooldown and the use of the residual heat removal (RHR) system (following a LOCA with successful high-pressure injection).
	High-pressure recirculation (PWR Classes B and D) (also required to support RCS inventory for all classes)
	Secondary-side depressurization and use of condensate system (PWR Class H)
	Containment spray recirculation (PWR Classes A and G)

A.4.2 PWR Nonspecific Reactor Trip

The PWR nonspecific reactor trip event tree constructed for plant Classes B and D is shown in Fig. A.7. The event-tree branch descriptions follow (event tree branch designations are shown in brackets).

1. Initiating event (transient) [TRANS]. The initiating event for the tree is a transient or upset event that requires or is followed by a rapid shutdown of the plant. LOOP, small-break LOCA, and SGTR initiators are modeled in separate event trees. Medium and large-break LOCA and steam-line break (SLB) initiators are not addressed in the models described here.

2. Reactor trip [RT]. To achieve reactor subcriticality and thus halt the fission process, the reactor protection system (RPS) is required to insert control rods into the core. If the automatically initiated RPS fails, a reactor trip may be initiated manually. Failure to trip results in an ATWS response, described later.

3. Auxiliary feedwater [AFW]. AFW flow to the SGs must be provided following trip to remove the decay heat still being generated in the reactor core. Successful AFW operation requires flow from one or more AFW pumps to one or more SGs over a period of time ranging from 12 to 24 hours (typically, one pump to one SG is adequate).

4. Main feedwater [MFW]. In lieu of AFW, MFW can be utilized to remove the post-shutdown decay heat. Depending on the individual plant design, either main or AFW may be used as the primary source of secondary-side heat removal.

5. PORV challenged [PORV CHALL]. For sequences in which both reactor trip and steam generator feedwater flow (MFW or AFW) have been successful, the pressurizer PORV may or may not lift,

depending on the peak pressurizer pressure following the transient. (In most transients, these valves do not lift.) The upper branch indicates that the valve or valves were challenged and opened. Because of the multiplicity of relief and safety valves, it is assumed that a sufficient number will open if the demand from a pressure transient exists.

The lower branch indicates that the pressurizer pressure was not sufficiently high to challenge a relief valve. For the sequences in which AFW fails following a reactor trip, PORVs are assumed to be challenged for overpressure protection.

6. PORV reseats [PORV RESEAT]. Success for this branch requires the closure of any open relief valve once pressurizer pressure has decreased below the relief valve set point. If a PORV sticks open, Class B and D PWR plants are equipped with an isolation valve that allows manual termination of the blowdown. Failure of a primary-side relief valve to close results in a transient-induced LOCA that is modeled as part of this event tree.

7. High-pressure injection [HPI]. In the case of a transient-induced LOCA, HPI is required to provide RCS makeup to keep the core covered. Success for this branch requires introduction of sufficient borated water to keep the core covered, considering core decay heat. (Typically, one HPI pump or multiple centrifugal charging pumps is sufficient for this purpose.)

8. Feed and bleed [FEED & BLEED]. If normal methods of achieving decay heat removal via the SGs (MFW and AFW) are unavailable, core cooling can be accomplished in most plants by establishing a feed-and-bleed operation. This operation provides (1) heat removal via discharge of reactor coolant to the containment through the PORVs and (2) RCS makeup via injection of borated water from the HPI system. Except at Class D plants, successful feed and bleed requires the operator to open the PORVs manually. At Class D plants, the HPI discharge pressure is high enough to lift the primary-side safety valves, and feed and bleed can be accomplished without the operator manually opening a PORV. HPI success for feed and bleed is dependent on plant design but requires the introduction of sufficient amounts of borated water into the RCS to remove decay heat and provide sufficient reactor coolant makeup to prevent core damage. PORV success for feed and bleed is assumed to require all PORVs at the plant to be opened.

9. Recovery of secondary-side cooling [RECOV SEC SIDE COOLING]. Secondary-side cooling may be recovered following failure of AFW and MFW and successful initiation of feed and bleed but prior to depletion of the refueling water storage tank (RWST), eliminating the need to use containment sump recirculation for continued core cooling. Successful long-term recovery of secondary-side cooling (since the steam generators are dry, flow from one motor-driven AFW or MFW pump is required) and termination of feed and bleed cooling result in core cooling success.

10. RCS cooldown to RHR initiation pressure [RCS COOLDOWN]. Following initiation of HPI for RCS makeup following a transient-induced LOCA, substantial time (typically ~6 hours) is available before the RWST is depleted and sump recirculation is required. An RCS cooldown to the RHR initiation pressure [using the turbine bypass valves (TBVs) and main condenser, or the atmospheric dump valves (ADVs), in conjunction with AFW or MFW], and initiation of RHR will provide core cooling without the need for sump recirculation. This approach has been used in the mitigation of all historic PWR small-break LOCAs. Because RCS pressure is significantly reduced once RHR is initiated, HPI can

provide the limited makeup for a substantial period of time. Success for this branch requires an RCS cooldown to the RHR initiation pressure in time to allow initiation of RHR prior to RWST depletion.

11. Residual heat removal [RHR]. If the RCS can be cooled down and depressurized to the RHR initiation pressure, then the RHR system can be used for core cooling. Success for this branch requires the operation of one train of the RHR system. Many Class B and D PWR plants employ a common RHR pump suction line to supply RCS flow to both RHR trains. Multiple valves in this line must open for RHR success.

12. High-pressure recirculation [HPR]. Following a transient-induced LOCA or failure of secondary-side cooling and initiation of feed and bleed, continued core cooling and makeup are required. This requirement is satisfied by using HPI in the recirculation mode once the RWST is depleted, unless the plant can be placed on the RHR system beforehand. In this mode the HPI pumps recirculate reactor coolant collected in the containment sump and pass it through heat exchangers for heat removal. When MFW or AFW is available, heat removal is assumed to be required only to prevent HPI pump damage; if AFW or MFW is not available, HPR is required to remove decay heat as well. Typically, at Class B and D plants, the low-pressure injection (LPI) pumps are utilized in the HPR mode, taking suction from the containment sump, passing the pumped water through heat exchangers, and providing net positive suction head to the HPI pumps.

The event tree applicable to a PWR Class G nonspecific reactor trip is shown in Fig. A.12. Many of the event tree branches and the sequences leading to successful transient mitigation and core damage are similar to those following a nonspecific reactor trip transient for Class B plants (those branches are not discussed further). At Class G plants, however, the HPR system performs both the high- and low-pressure recirculation (LPR) function, taking suction directly from the containment sump without the aid of the low-pressure pumps. Decay heat is removed during recirculation by the containment spray recirculation (CSR) system.

1. Initiating event (transient) [TRANS]. The initiating event is a nonspecific reactor trip, similar to that described for PWR Classes B and D.

2. Reactor trip [RT].

3. Auxiliary feedwater or main feedwater [AFW or MFW].

4. PORV challenged/reseats [PORV CHALL/PORV RESEAT].

5. High-pressure injection [HPI].

6. Feed and bleed [FEED & BLEED].

7. Recovery of secondary-side cooling [RECOV SEC SIDE COOLING].

8. RCS cooldown to RHR initiation pressure [RCS COOLDOWN].

9. Residual heat removal [RHR].

10. Containment spray recirculation [CSR]. When secondary-side cooling and RHR are unavailable to remove decay heat, the CSR system operates to remove decay heat from the reactor coolant being recirculated. This is different than PWR Classes B and D, where the decay heat removal function can be performed by HPR.

11. High-pressure recirculation [HPR]. In the event of a transient-induced LOCA or feed and bleed, continued HPI via sump recirculation is needed to provide makeup once the refueling water tank (RWT) is depleted, unless the plant can be placed on the RHR system beforehand. In Class G plants, initiation of HPR realigns the HPI pumps to the containment sump. The use of LPI pumps for boosting suction pressure is not required.

The event tree for a PWR Class H nonspecific reactor trip is shown in Fig A.17. This class of plants is different than other PWR classes in that PORVs are not included in the plant design and feed and bleed cannot be used to remove decay heat in the event of MFW and AFW unavailability. If MFW or AFW cannot be recovered, the atmospheric dump valves can be used to depressurize the SGs to below the shutoff head of the condensate pumps, and these can be used, if available, for RCS cooling. The following is a description of event tree branches for Class H PWR that are different than those described for previous PWR classes.

1. Initiating event (transient) [TRANS]. The initiating event is a nonspecific reactor trip, similar to that described for the previous PWR classes.

2. Reactor trip [RT].

3. Auxiliary feedwater or main feedwater [AFW or MFW].

4. Safety relief valve (SRV) challenged [SRV CHALL]. The upper branch indicates that at least one safety valve has lifted as a result of the transient. In most transients in which reactor trip has been successful and MFW or AFW is available, these valves do not lift. In the case where both MFW and AFW are unavailable, at least one SRV is assumed to lift. The lower branch indicates that the pressurizer pressure was not sufficiently high to cause a relief valve to open.

6. SRV reseat [SRV RESEAT]. Success for this branch requires the closure of any open safety valve once pressurizer pressure has been reduced below the safety valve set point. Because only safety valves are used on this plant class, no block valves exist that can be closed to terminate flow from a stuck-open relief valve.

7. High-pressure injection [HPI].

8. Condensate pumps [COND]. If MFW and AFW are unavailable, the ADVs [or TBVs if the main steam isolation valves (MSIVs) are open] may be used on Class H plants to depressurize the SGs to the point that the condensate pumps can be used for SG cooling. Flow from one condensate pump to one SG is assumed adequate. In the event of the unavailability of MFW and AFW, failure to depressurize one SG to the operating pressure of the condensate system or unavailability of the condensate pumps is assumed to result in core damage.

9. RCS cooldown to RHR initiation pressure [RCS COOLDOWN].

10. Residual heat removal [RHR].

11. High-pressure recirculation [HPR]. The requirement for continued core cooling during mitigation of a transient-induced LOCA and following depletion of the RWT, if RHR has not been initiated, can be satisfied by using HPI in the recirculation mode. At Class H plants, initiation of HPR realigns the HPI pumps with the containment sump. The use of LPI pumps for boosting suction pressure is not required.

The event tree applicable to a PWR Class A nonspecific reactor trip is shown in Fig. A.2. Many of the event-tree branches and the sequences leading to successful transient mitigation and severe core damage are similar to those following a nonspecific reactor trip transient for Class B and G plants.

Like the Class G plants, Class A plants have a CSR system that provides decay heat removal during HPR. Use of CSR for decay heat removal was assumed to be required if AFW and MFW were unavailable. LPI pumps are required to provide suction to the HPI pumps during recirculation. The event tree branches and sequences are discussed further below.

1. Initiating event (transient) [TRANS]. The initiating event is a nonspecific reactor trip, similar to that described for the other PWR plant classes.

2. Reactor trip [RT].

3. Auxiliary feedwater or main feedwater [AFW or MFW].

4. PORV challenged/reseats [PORV CHALL/PORV RESEAT].

7. High-pressure injection [HPI].

8. Feed and bleed [FEED & BLEED].

9. Recovery of secondary-side cooling [RECOV SEC SIDE COOLING].

10. RCS cooldown to RHR initiation pressure [RCS COOLDOWN].

11. Residual heat removal [RHR].

12. Containment spray recirculation [CSR].

13. High-pressure recirculation [HPR]. The LPI pumps provide suction to the high-pressure pumps in the recirculation mode.

<u>Anticipated transient without scram</u>

The event trees constructed define potential plant response following an ATWS. Following a failure to scram, significant AFW flow is required for short-term core cooling, and injection of soluble boric acid is required to shut down the fission reaction. In addition, the primary relief valves, in conjunction with a negative moderator temperature coefficient, must limit RCS pressure to prevent the failure of RCS components. Failure to limit RCS pressure, provide adequate AFW to remove core heat, or inject soluble boric acid is assumed to result in core damage following a failure to trip.

Similar event trees are used for all PWR classes. These are shown in Figs. A.6, A.11, A.16, and A.21, respectively, for classes A, B and D, G, and H. Descriptions of event tree branches that are unique to the ATWS event trees follow. Branches on the ATWS tree that are also included on the transient event tree for the class are not described further.

1. Initiating event (ATWS) [ATWS]. The initiating event for this tree is a transient with failure to scram the reactor through either automatic or manual actuation of the RPS. This initiating event is an effective transfer from the transient event tree for sequences involving failure to scram.

2. Primary pressure limited [PRIMARY PRESSURE LIMITED]. ATWS analyses assume RCS components will fail unpredictably above ~3200 psi. If this occurs, core damage is assumed to result. Success for this branch requires RCS pressure to be limited to no greater than ~3200 psi. Primary pressure is limited by an adequately negative moderator temperature coefficient and by the operation of the primary SRVs and PORVs.

3. Auxiliary feedwater for ATWS [AFW (ATWS)]. AFW and the secondary side relief valves are required to remove core heat. Typically, twice the normal AFW flow is required until the fission process is terminated by the addition of boric acid.

4. Emergency boration [EMERGENCY BORATION (HPI+BORON)]. Injection of concentrated boric acid via the HPI or charging system is required to terminate the fission process. Emergency boration is manually initiated.

5. SRV and PORV reseat following ATWS pressure relief [PORV/SRV RESEAT (ATWS)]. All primary safety valves and the PORVs are (1) assumed to lift as a result of the high RCS pressure that accompanies an ATWS and (2) to discharge water. As a result of the passage of water through the valves, the valve failure-to-close probabilities are considerably higher than in the normal situation when only steam is relieved. Success for this branch requires the closure of all open safety valves and PORVs (if a PORV fails to close, its block valve can be closed by the operators).

   If a relief valve fails to close (down branch), a transient-induced LOCA results. Systems required to mitigate the LOCA are similar to those on the transient event tree. HPI is assumed to be successful since emergency boration is successful.

6. RCS cooldown to RHR initiation pressure [RCS COOLDOWN].

7. Residual heat removal [RHR].

8. High-pressure recirculation [HPR].

A.4.3 PWR Loss of Offsite Power

The event trees constructed define representative plant responses to a LOOP. A LOOP (without turbine runback on plants with this feature) will result in reactor trip due to (1) unavailability of power to the CRD mechanisms and (2) a loss of MFW because of the unavailability of power to components in the condensate and condenser cooling systems.

The PWR LOOP tree constructed for Class B and D plants is shown in Fig. A.8. Descriptions of the event tree branches follow.

1. Initiating event (LOOP) [LOOP]. The initiating event for the tree is a grid or switchyard disturbance to the extent that the generator must be separated from the grid and all offsite power sources are unavailable to plant equipment. The capability of a runback of the unit generator from full power to supply house loads exists at some plants but is not considered in the event tree. Only LOOPs that challenge the emergency power (EP) system and result in plant trip are addressed in the ASP Program.

2. Reactor trip given LOOP [RT (LOOP)]. Unavailability of power to the CRD mechanisms is expected to result in a reactor trip and rapid shutdown of the plant. If the reactor trip does not occur following a LOOP, the transient is considered to proceed to core damage (this may be conservative).

3. Emergency power [EP]. Given a LOOP and a reactor trip, electric power would be lost to all loads not backed by battery power. When power is lost, diesel generators (DGs) are automatically started to provide power to the plant safety-related loads. Emergency power success requires the starting and loading of a sufficient number of DGs to support safety-related loads in systems required to mitigate the LOOP and maintain the plant in a safe shutdown condition.

4. Auxiliary feedwater [AFW]. The AFW system functions to remove decay heat via the SG secondary side. Success requirements for this branch are equivalent to those following a nonspecific reactor trip and unavailability of MFW (both MFW and condensate pumps would be unavailable following a LOOP). Because specific AFW systems may contain different combinations of turbine-driven and motor-driven AFW pumps, the capability of the system to meet its success requirements will depend on the state of the EP system and the number of turbine-driven AFW pumps that are available.

5. PORV challenged [PORV CHALL]. The upper and lower states for this branch are similar to those following a nonspecific reactor trip. While a PORV may or may not lift, depending on the peak pressure following a particular event, the ASP models assume lift occurs following a LOOP with EP system failure.

6. PORV reseats [PORV RESEAT]. The success requirements for this branch are similar to those following a nonspecific reactor trip. However, for a situation in which emergency power is failed and the PORV fails to reseat, power is unavailable for block valve closure.

7. Seal LOCA [RCP SEAL LOCA]. In the event of a loss of EP following LOOP, both service water (SW) and component cooling water (CCW) are unavailable. This results in unavailability of RCP seal cooling and seal injection (since the charging pumps are also without power and cooling water). Unavailability of seal cooling and injection may result in seal failure after a period of time, depending

on the seal design.

The upper event tree branch represents the situation in which seal failure occurs prior to restoration of ac power. The lower branch represents the situation in which a seal LOCA does not occur.

8. Electric power recovered (long term) [OFFSITE POWER RECOV (LONG)]. Recovery of offsite power in the long term following failure of EP can prevent or allow mitigation of an RCP seal LOCA. If EP is successful, recovery of offsite power can still allow recovery of condenser cooling and facilitate placing the plant on the RHR system, thereby preventing the use of sump recirculation following a transient-induced LOCA.

   For sequences involving EP failure in which a seal LOCA has occurred, the success of long-term electric power recovery requires the restoration of ac power (either through recovery of offsite power or recovery of a DG) prior to core uncovery. For sequences involving EP failure in which a seal LOCA does not occur, the success of electric power recovery requires the recovery of ac power prior to battery depletion, typically 2 to 4 hours.

   If EP is successful, recovery of offsite power within 2 hours is assumed to allow sufficient time to recover the condenser, cool down the plant, and initiate RHR before depleting the RWST following a transient-induced LOCA, eliminating the need for sump recirculation. Recovery at 6 hours is assumed to allow recovery of secondary-side cooling in the event of an initial AFW failure.

9. High-pressure injection [HPI], feed and bleed [FEED & BLEED], residual heat removal [RHR] and high-pressure recirculation [HPR]. The success requirements for these branches are similar to those following a nonspecific reactor trip. Because the systems use motor-driven pumps, the capability of each system to meet its success requirements depends on the status of the DGs.

10. Recovery of secondary-side cooling [RECOV SEC SIDE COOLING] and RCS cooldown to RHR initiation pressure [RCS COOLDOWN]. Success requirements for these branches are similar to those following a nonspecific reactor trip. Prior recovery of offsite power is necessary to power secondary-side balance-of-plant loads.

The event tree constructed for the PWR Class G LOOP is shown in Fig. A.13. Most of the event tree branches and the sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, at Class G plants, decay heat removal during recirculation is provided by the CSR system, not the HPR system. The event tree branches and sequences different than those for PWR B LOOP are discussed below.

1. Initiating event (LOOP) [LOOP]. The initiating event is a LOOP similar to that described for Class B and D PWR plants. The following branches have functions and success requirements similar to those following a LOOP at PWRs associated with all of the plant classes defined.

2. Reactor trip given LOOP [RT (LOOP)].

3. Emergency power [EP].

4. Auxiliary feedwater [AFW].

5. PORV challenged/reseats [PORV CHALL/PORV RESEAT].

6. Seal LOCA [RCP SEAL LOCA].

7. Electric power recovered (long term) [OFFSITE POWER RECOV (LONG)].

8. High-pressure injection, feed and bleed, residual heat removal, and high-pressure recirculation [HPI, FEED & BLEED, RHR, HPR].

9. Recovery of secondary-side cooling and RCS cooldown to RHR initiation pressure [RECOV SEC SIDE COOLING and RCS COOLDOWN].

10. Containment spray recirculation [CSR]. The success requirements for this branch are similar to those following a nonspecific reactor trip. The CSR system is assumed to be required to provide decay heat removal for sequences in which secondary-side cooling is unavailable.

The event tree constructed for a PWR Class H LOOP is shown in A.18. Many of the event tree branches and sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, Class H plants do not have feed-and-bleed capability and rely instead on secondary-side depressurization and the condensate system as an alternative decay heat removal method. The condensate system is assumed to be unavailable following a LOOP, which limits the diversity of decay heat removal on this plant class following this initiator. The event branches and sequences are discussed further below.

1. Initiating event (LOOP) [LOOP]. The initiating event is a LOOP similar to that described for Class B and D PWR plants. The following branches have functions and success requirements similar to those following a LOOP at PWRs associated with all of the plant classes defined.

2. Reactor trip given LOOP [RT (LOOP)].

3. Emergency power [EP].

4. Auxiliary feedwater [AFW].

5. SRV challenged [SRV CHALL]. The function of this branch is similar to that described under the PWR Class H transient.

6. SRV reseat [SRV RESEAT]. Success requirements for this branch are similar to those described under the PWR Class H transient.

7. Seal LOCA [RCP SEAL LOCA].

8. Electric power recovered (long-term) [OFFSITE POWER RECOV (LONG)].

9. High-pressure injection, residual heat removal, and high-pressure recirculation [HPI, RHR, HPR].

10. RCS cooldown to RHR initiation pressure [RCS COOLDOWN].

The event tree constructed for the Class A LOOP is shown in Fig. A.3. All of the event tree branches and the sequences leading to successful mitigation and core damage are analogous to those following a LOOP at Class B plants with the addition of the CSR branch [CSR], which is required for decay heat removal during high-pressure recirculation if the plant cannot be cooled down and placed on the RHR system beforehand. Additional information on the use of the CSR system is provided in the discussion of the PWR Class A nonspecific reactor trip event tree.

A.4.4 PWR Small-Break Loss-of-Coolant Accident

Event trees were constructed to define the responses of PWRs to a small-break LOCA. The LOCA chosen for consideration is one that would require a reactor trip and continued HPI for core protection. Because of the limited amount of borated water available, the mitigation sequence also includes the requirement to recirculate borated water from the containment sump, unless the plant can be successfully cooled down and placed on the RHR system prior to RWST depletion.

The LOCA event tree constructed for PWR plant Classes B and D is shown in Fig. A.9. The event tree branches and the sequences leading to core damage follow.

1. Initiating event (small-break LOCA) [LOCA]. The initiating event for the tree is a small-break LOCA that requires reactor trip and continued HPI for core protection.

2. Reactor trip [RT]. Reactor trip success is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition. Failure to trip was considered to lead to core damage in the ASP models (this may be conservative).

3. Auxiliary feedwater or main feedwater [AFW or MFW]. Use of AFW or MFW was assumed to be necessary for some small breaks to reduce RCS pressure to the point where HPI is effective.

4. High-pressure injection [HPI]. Adequate injection of borated water from the HPI system is required to keep the core covered, thereby preventing excessive core temperatures and consequent core damage.

5. Feed and bleed [FEED & BLEED]. In the event AFW and MFW are unavailable following a small-break LOCA, core cooling can be provided using the feed-and-bleed mode. Depending on the size of the small break, opening the PORVs may not be required for success (opening a PORV is not required for success for Class D).

6. Recovery of secondary-side cooling [RECOV SEC SIDE COOLING]. Secondary-side cooling may be recovered following failure of AFW and MFW and successful initiation of feed and bleed, but prior to RWST depletion. Successful recovery of secondary-side cooling and reactor cooldown to the RHR initiation pressure will allow RHR to be used for core cooling. This plus reduced HPI to make up for flow from the break will avoid the need for sump recirculation.

7. RCS cooldown to RHR initiation pressure [RCS COOLDOWN]. Following initiation of HPI, substantial time (typically ~6 hours) is available before the RWST is depleted and sump recirculation

is required. An RCS cooldown to the RHR initiation pressure (using the TBVs and main condenser, or the ADVs, in conjunction with AFW or MFW) and initiation of RHR will provide core cooling without the need for sump recirculation. This approach has been used in the mitigation of all historic PWR small-break LOCAs. Because RCS pressure is significantly reduced once on RHR, HPI can provide the limited makeup for a substantial period of time. Success for this branch requires an RCS cooldown to the RHR initiation pressure in time to allow initiation of RHR prior to RWST depletion.

8. Residual heat removal [RHR]. If the RCS can be cooled down and depressurized to the RHR initiation pressure, then the RHR system can be used for core cooling. Success for this branch requires the operation of one train of the RHR system. Many PWR B and D Class plants employ a common RHR pump suction line to supply RCS flow to both RHR trains. Multiple valves in this line must open for RHR success.

9. High-pressure recirculation [HPR]. The requirement for continued core cooling following a LOCA is satisfied by using HPI in the recirculation mode once the RWST is depleted, unless the plant can be placed on the RHR system beforehand. In this mode the HPI pumps recirculate reactor coolant collected in the containment sump and pass it through heat exchangers for heat removal. When MFW or AFW is available, heat removal is assumed to be required only to prevent HPI pump damage; if AFW or MFW is not available, HPR is required to remove decay heat as well. Typically, at Class B and D plants, the LPI pumps are utilized in the HPR mode, taking suction from the containment sump, passing the pumped water through heat exchangers, and providing net positive suction head to the HPI pumps.

The event tree constructed for a small-break LOCA at Class G plants is shown in A.14. The LOCA event tree for Class G plants is similar to that for Class B and D plants except that long-term cooling is provided by the CSR system rather than by the HPR system. The event-tree branches and sequences are discussed further below.

1. Initiating event (small-break LOCA) [LOCA]. The initiating event is a LOCA similar to that described for Class B and D PWR plants. The following branches have functions and success requirements similar to those following a small-break LOCA at PWRs associated with all of the plant classes defined.

2. Reactor trip [RT].

3. Auxiliary feedwater and main feedwater [AFW and MFW].

4. High-pressure injection and feed and bleed [HPI and FEED & BLEED].

5. Recovery of secondary-side cooling [RECOV SEC SIDE COOLING].

6. RCS cooldown to RHR initiation pressure and RHR [RCS COOLDOWN and RHR].

7. Containment spray recirculation [CSR]. In the event that normal secondary-side cooling (AFW and MFW) is unavailable following a small-break LOCA, cooling via the CSR system during HPR is required to mitigate the transient. If AFW or MFW is available, CSR is assumed not to be required.

8. High-pressure recirculation [HPR].

The event tree constructed for a small-break LOCA at Class H PWR plants is shown in Fig. A.19. The event tree has been developed assuming that SG depressurization and condensate pumps can provide adequate RCS pressure reduction in the event of an unavailability of AFW and MFW to permit HPI and HPR to function in these plants. The event tree branches and sequences are similar to those following a transient-induced LOCA.

1. Initiating event (small-break LOCA) [LOCA]. The initiating event is similar to that described above for PWR Classes B, D, and G. The following branches have functions and success requirements similar to those discussed previously for this class.

2. Reactor trip [RT].

3. Auxiliary feedwater, main feedwater, and condensate [AFW, MFW, and COND].

4. High-pressure injection [HPI].

5. RCS cooldown to RHR initiation pressure [RCS COOLDOWN].

6. Residual heat removal [RHR].

7. High-pressure recirculation [HPR].

The event tree constructed for a small LOCA at Class A plants is shown in Fig. A.4. The LOCA event tree for Class A plants is similar to that for Classes B and D except that the CSR system is required in conjunction with HPR in sequences where secondary cooling is not provided.

As with the PWR transient and LOOP sequences, differences between plant classes are driven by the use of CSR on Class A and G plants and by the use of condensate pumps in lieu of feed and bleed on Class H PWRs.

A.4.5 PWR Steam Generator Tube Rupture

The event trees constructed define potential plant response following an SGTR. In the event of an SGTR, the nominal plant response is to provide RCS inventory makeup using the HPI system; detect and then isolate the ruptured SG by closing appropriate AFW, MFW, and MSIVs; and depressurize the RCS to below the SG relief valve reseat pressure using the intact SGs. This allows the relief valves to reseat and terminates flow from the RCS into the failed SG. If the break cannot be isolated, the RCS must be cooled down further and the RHR system must be placed in operation before RWST inventory is depleted. Failure to perform these functions is assumed to result in core damage.

The SGTR event trees constructed for PWR plant Classes B and D, G, and A are shown in Figs. A.10, A.15, and A.5, respectively. Descriptions of the branches that are unique to an SGTR response follow. Branches on the SGTR event tree that are also included on other event trees are not described further.

1. Initiating event (SGTR) [SGTR]. The initiating event is the failure of one SG tube, with resulting RCS flow from the primary to the secondary side of the SG. Simultaneous rupture of multiple tubes is not

addressed.

2. Reactor trip [RT]. Failure to trip the reactor following an SGTR is assumed to result in core damage (this may be conservative).

3. Auxiliary feedwater [SGTR]. AFW flow to the intact (unaffected) SGs must be provided to remove decay heat and cool the RCS to reduce its pressure to below the SG relief valve reseat point. Success for this branch requires flow from one or more AFW pumps to at least one intact SG.

4. Main feedwater [MFW]. The MFW system can be used for heat removal if AFW is unavailable. Most MFW systems isolate on safety injection, and subsequent operability is dependent on the type of pump driver; turbine-driven MFW pumps require steam from the non-affected SGs once the faulted SG is isolated.

5. High-pressure injection [HPI].

6. Ruptured SG isolated and RCS cooldown [RUPTURED SG ISOLATED AND RCS COOLDOWN]. Success requires the use of the ADVs or TBVs to reduce RCS pressure below the SG relief valve reseat pressure and isolation of the ruptured SG by closing open valves associated with feed, blowdown, and steam flow. This terminates flow from the tube rupture.

7. RCS cooldown below RHR pressure [RCS COOLDOWN BELOW RHR PRESSURE]. If the ruptured SG cannot be isolated, RCS cooldown is continued using the TBVs until RHR can be initiated. On plants with large ADV capacity, RCS cooldown may be accomplished without TBVs. Once on the RHR system, the SGs (which are no longer required for decay heat removal) can be isolated if necessary.

8. Residual heat removal [RHR].

The SGTR event tree constructed for Class H PWRs is shown in Fig. A.20. With the exception of one branch that addresses the potential use of the condensate system if both AFW and MFW fail, all branches are similar to those on the previous event trees.

1. Initiating event (SGTR) [SGTR].

2. Reactor trip [RT].

3. Auxiliary feedwater [AFW].

4. Main feedwater [MFW].

5. Condensate [COND]. In the event that both AFW and MFW are unavailable, the ADVs (or TBVs if the MSIVs are open) can be used on Class H PWR plants to depressurize the intact SGs to the point that the condensate pumps can be used for SG cooling. Flow from one condensate pump to one SG is assumed to be adequate.

6. High-pressure injection [HPI].

7. Ruptured SG isolated and RCS cooldown below SG relief valve setpoint [RUPTURED SG ISOLATED AND RCS COOLDOWN].

8. RCS cooldown below RHR pressure [RCS COOLDOWN BELOW RHR PRESSURE].

9. Residual heat removal [RHR].

A.4.6 BWR Event Tree Models

The BWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following three initiating events: trip, LOOP, and small-break LOCA. The systems modeled in the event trees are those associated with the generic functions required in response to an initiating event. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality	Reactor scram and standby liquid control (following failure to trip)
Reactor coolant system integrity	Addressed in small-break LOCA models and in trip and LOOP sequences involving failure of primary relief valves to reseat
Reactor coolant inventory	High-pressure injection systems [HPCI or HPCS, RCIC, CRD, FWCI]
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), condensate, low-pressure core spray (LPCS), residual heat removal service water (RHRSW) or equivalent]
Short-term core heat removal	Power conversion system (PCS)
	High-pressure injection systems [HPCI, RCIC, CRD, FWCI (BWR Class A)]
	Isolation condenser (IC) (BWR Classes A and B)
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), LPCS, condensate]
	Note: Short-term core heat removal to the suppression pool (all cases where power conversion system is faulted) requires use of the RHR system for heat removal in the long term.

Long-term core heat removal	Power conversion system
	Isolation condenser (BWR Class A)
	Residual heat removal (BWR Class C)
	Shutdown cooling (SDC) (BWR Classes A and B)
	Containment cooling (CC) (BWR Class A)
	Low-pressure coolant injection [CC mode (BWR Class B)]

A.4.7 BWR Nonspecific Reactor Trip

The nonspecific reactor trip event tree constructed for Class C BWR plants is shown in Fig. A.30. The event tree branches and the sequences leading to potential severe core damage follow [event tree branch designations are shown in brackets]. The Class C plants are discussed first because all but a few of the BWRs fit into the Class C category.

1. Initiating event (transient) [TRANS]. The initiating event is a transient or upset event that results in a rapid shutdown of the plant. Transients that are initiated by a LOOP or a small-break LOCA are modeled in separate event trees. Transients initiated by a large-break LOCA or large SLB are not addressed in the event trees described here; trees applicable to such initiators are developed separately if required.

2. Reactor shutdown [Rx SHUTDOWN]. To achieve reactor subcriticality and thus halt the fission process, the RPS commands insertion of the control rods into the core. Successful scram requires rapid insertion of control rods with no more than two adjacent control rods failing to insert. Failure to scram results in sequences associated with ATWS and are described later in this section.

3. Power conversion system [PCS]. Upon successful reactor scram, continued operation of the PCS would allow continued heat removal via the main condenser. This is considered successful mitigation of the transient. Continued operation of the PCS requires the MSIVs to remain open and operation of the condenser, turbine bypass system (TBS), condensate pumps, condensate booster pumps, and feedwater pumps.

4. SRVs close [SRVs CLOSE]. SRVs are assumed to lift following scram. Success for this branch requires the reseating of all but one open relief valve once the reactor pressure vessel (RPV) pressure decreases below the relief valve set point. If an SRV sticks open, a transient-induced LOCA is initiated. The response of Class C BWR plants to a single stuck-open relief valve is similar to the response when no SRV valve sticks open, and is represented by the upper branch on the event tree. The failure of two valves to close is represented by the middle branch; plant response is similar to a medium-break LOCA. The lowest branch represents the failure of more than two SRVs to close. This response is similar to a large-break LOCA.

5. Feedwater [FW]. Given unavailability of the PCS, continued delivery of feedwater to the RPV will keep the core from becoming uncovered. This, in combination with successful long-term decay heat

removal, will mitigate the transient, preventing core damage. For plants with turbine-driven feed pumps, PCS failure with subsequent feedwater success cannot involve MSIV closure or loss of condenser vacuum, because this would disable the feed pumps.

6. High-pressure coolant injection (or high-pressure core spray) [HPCI or HPCS]. The primary function of the HPCI or HPCS system is to provide makeup following small-break LOCAs while the reactor is at high pressure (not depressurized). The system is also used for decay heat removal following transients involving a loss of feedwater. Some later Class C plants are equipped with HPCS systems, but the majority are equipped with HPCI systems. HPCI or HPCS can provide the required makeup and short-term decay heat removal when the condenser and feedwater system are unavailable.

7. Reactor core isolation cooling [RCIC]. The RCIC system is designed to provide high-pressure coolant makeup for transients that result in LOFW. Both RCIC and HPCI (or HPCS) initiate when the reactor coolant inventory drops to the low-low level set point, taking suction from the condensate storage tank (CST) or the suppression pool. To prevent tripping of HPCI and RCIC pumps on high water level, HPCI is normally secured after HPCI/RCIC initiation when pressure and water level are restored. RCIC must then be operated until the RHR system can be placed in service. The RCIC system is also capable of providing successful makeup following a single stuck-open SRV.

8. Depressurization via manual actuation of the SRVs or the automatic depressurization system [SRVs/ADS]. In the event that the high-pressure systems have failed to provide adequate flow, the RPV can be depressurized to allow use of the low-pressure, high-capacity injection systems. The ADS will automatically initiate on high drywell pressure and low-low reactor water level, the availability of one train of the LPCI or LPCS systems, and following a time delay (which can be reset by the operator). The SRVs can also be opened by the operators to speed the depressurization process or if ADS fails to automatically actuate.

9. CRD injection [CRD PUMPS (INJ)]. In transient-induced sequences where heat removal and minimal core makeup are required (i.e., no more than one SRV sticks open), the CRD pumps can deliver coolant to the RPV.

10. Condensate system [COND]. Low-pressure injection can be provided by the condensate system if it is available following a loss of feedwater. Condensate is initially drawn from the condenser hotwell.

11. Low-pressure core spray [LPCS]. Low-pressure injection can be provided by the LPCS system if required. The LPCS system performs the same functions as the LPCI system (described below) except that the coolant, which is drawn from the suppression pool or the CST, is sprayed over the core.

12. Low-pressure coolant injection [LPCI]. The LPCI system can provide short-term heat removal and cooling water makeup if the reactor has been depressurized to the operating range of the low-head RHR pumps. At Class C plants, LPCI is a mode of the RHR system; thus, the RHR pumps operate during LPCI. LPCI takes suction from the suppression pool or the CST and discharges into the recirculation loops or directly into the reactor vessel. If LPCI is successful in delivering sufficient flow to the reactor, successful long-term heat removal is still required to mitigate core damage.

13. RHR service water or other injection source [RHRSW (INJ)]. This is a backup measure for providing

water to the reactor to reflood the core and maintain core cooling if other injection sources are unavailable. Typically, the SW pumps are aligned to the shell side of the RHR heat exchangers for delivery of water to one of the recirculation loops.

14. Residual heat removal [RHR]. Three modes of RHR are represented by this branch. In the shutdown cooling mode, coolant is circulated from the reactor by the RHR pumps through the RHR heat exchangers and back to the reactor vessel. In the suppression pool cooling mode, the RHR pumps and heat exchangers are aligned to take water from the suppression pool, cool it using the RHR heat exchangers, and return it to the suppression pool. In the containment spray mode, water from the suppression pool is first cooled using the RHR heat exchangers before being sprayed into the containment and returning to the suppression pool. Long-term core cooling success requires that heat transfer to the environment start within ~12 - 24 hours of the transient. RHR success following successful reactor scram and high- or low-pressure injection of water to the RPV will prevent core damage.

The event tree constructed for a BWR Class A nonspecific reactor trip is shown in A.22. The event tree is similar to that constructed for BWR Class C plants with the following exceptions: Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. The ICs can provide long-term core cooling provided no loss of inventory exists. Class A plants do not have LPCI systems, although they are equipped with LPCS; suppression pool cooling is provided by a system independent of the SDC system. The event tree branches different from those for Class C are discussed further below.

1. Initiating event (transient) [TRANS]. The initiating event is a nonspecific reactor trip similar to that described for Class C BWR plants. The following branches have functions and success requirements similar to those following a transient at BWRs associated with Class C.

2. Reactor shutdown [Rx SHUTDOWN].

3. Power conversion system [PCS].

4. SRVs close [SRVs CLOSE]. The three branches represent conditions in which all open SRVs close, one valve fails to close, and more than one valve fails to close. Following a transient with closure of all SRVs (upper branch), the IC can provide core cooling, as can MFW. If one SRV sticks open, MFW is required for RPV makeup and short-term core cooling, unless the RPV is depressurized so that low-pressure systems can be used. If more than one SRV sticks open, then the low-pressure systems can be utilized without the need for automatic or manual depressurization.

5. Isolation condenser and IC makeup [IC]. If PCS is not available and significant inventory has not been lost via the SRVs, then the IC system can provide decay heat removal and mitigate the transient. The IC system is essentially a passive system that condenses steam produced by the core, rejecting the heat to cooling water and returning the condensate to the reactor. Makeup is provided to the IC secondary-side as needed. The system does not provide makeup to the reactor vessel.

6. Feedwater [FW/FWCI]. MFW or feedwater coolant injection (FWCI) can provide short-term transient mitigation. MFW is required for makeup in transient-induced LOCA sequences and for heat removal in sequences when the IC system would have mitigated the transient but was not available. FWCI is

initiated automatically on low reactor level and uses the normal feedwater trains to deliver water to the reactor vessel. When feedwater is successful, long-term decay heat removal is required for complete transient mitigation. (PCS unavailability is assumed prior to MFW demand.)

7. Depressurization via SRV or ADS [SRVs/ADS].

8. CRD injection [CRD PUMPS (INJ)].

9. Condensate system [COND].

10. Low-pressure core spray [LPCS].

11. Fire water injection [FIREWATER OR OTHER (INJ)]. Fire water or other raw water systems can provide a capability similar to that provided by the RHRSW connection in Class C BWRs. As a backup source, if all normal core cooling is unavailable, fire water can be aligned with the LPCS injection line to provide water to the reactor vessel.

12. Shutdown cooling [SDC]. Like the shutdown cooling mode of the RHR system at Class C plants, the SDC system is a closed-loop system that performs the long-term decay heat removal function by circulating primary coolant from the reactor through the system's heat exchangers and back to the reactor vessel. Success requires the operation of at least one SDC loop.

13. Containment cooling [CC]. If the SDC system fails to provide long-term decay heat removal, the CC system can remove decay heat. The system utilizes dedicated pumps, drawing suction from the suppression pool, passing it through heat exchangers where heat is rejected to the service water system and then either returning it directly to the suppression pool or spraying it into the dry well.

The event tree constructed for a BWR Class B plant nonspecific reactor trip is shown in Fig. A.26. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same except that Class B plants are equipped with HPCI systems instead of FWCI systems, and they are equipped with an LPCI system that represents an additional capability for providing low-pressure injection. Also, at Class B BWRs, the CC system considered in the event tree utilizes the LPCI pumps rather than having its own dedicated pumps.

A.4.8 Anticipated Transient without Scram

The event trees constructed define potential plant response following an ATWS. Following a failure to automatically and manually scram or insert rods, the fission process is terminated by tripping the recirculation pumps and injecting soluble boron into the RPV. Availability of the PCS at this point terminates the transient. If PCS is unavailable, the operators further control power by lowering the RPV level to the top of the active fuel (TAF) and use HPCI or HPCS for makeup. Failing this, RPV pressure is lowered to allow the low-pressure systems to provide makeup.

Similar event trees are used for each BWR class (differences exist in the systems used for makeup, consistent with the systems available at each plant class). The event trees are shown in Figs. A.25, A.29, and A.33, respectively for classes A, B, and C. Descriptions of the event tree branches that are unique to ATWS follow.

Branches on the ATWS trees that are also included on the transient event trees are not discussed further.

1. Initiating event [ATWS]. The initiating event is an effective transfer from the transient event tree for sequences involving failure to scram.

2. Recirculation pump trip [RECIRC PUMP TRIP]. Success for this branch requires the automatic or manual trip of the recirculation pumps to reduce power.

3. Standby liquid control [SLCS]. The operators manually start the standby liquid control system to borate the RPV. This system must be initiated immediately following a failure to scram since it takes some time to be effective.

4. Power conversion system [PCS].

5. ADS inhibited and level controlled [ADS Inhibited and Level Controlled]. Failing to shut down the reactor manually or by alternative means, the operators must attempt to control power using the RPV level. The major actions are as follows. First, inhibit ADS. This both protects the containment (by avoiding a major transfer of hot RPV water to the suppression pool) and prevents the automatic actuation of LPCS and LPCI. Second, terminate injection. This excludes the standby liquid control system (SLCS) and CRD flow. The RPV level is deliberately lowered to the TAF, reducing reactivity and power. Third, restore injection. Were water level to fall below the TAF, there would be no assurance that core damage would be prevented. Hence, the level is reinstated.

6. High-pressure coolant injection [HPCI].

7. Manual reactor depressurization [Manual Depress for Core Cooling]. If the high-pressure systems are unavailable, the operators lower the RPV pressure to allow the use of the low-pressure systems for RPV makeup. This must be done carefully to avoid flushing boron from the core region.

8. Condensate, LPCS, LPCI (if available) [COND, LPCS, LPCI].

9. Residual heat removal or shutdown cooling and containment cooling [RHR or SDC and CC].

A.4.9 BWR Loss of Offsite Power

The event trees constructed define responses of BWRs to a LOOP in terms of sequences representing success and failure of plant systems. Only LOOPs that challenge the EP system and result in scram are addressed in the ASP program.

The event tree constructed for a LOOP at Class C BWR plants is shown in Fig. A.31. The event tree branches associated with sequences leading to core damage are described below (branches that are identical to those for a Class C transient are not further described).

1. Initiating event (LOOP) [LOOP]. The initiating event for a LOOP corresponds to any situation in which power from both the auxiliary and startup transformers is lost and scram occurs. This situation could result from grid disturbances or onsite faults.

2. Reactor shutdown [Rx SHUTDOWN]. Given a load rejection, a scram signal is generated. Successful scram is the same as for the transient trees: a rapid insertion of control rods with no more than two adjacent control rods failing to insert. The scram can be automatically or manually initiated. Failure to scram following a LOOP is assumed to result in core damage (this may be conservative).

3. Emergency power [EP]. Emergency power is provided by DGs at almost all plants. The DGs receive an initiation signal when an undervoltage condition is detected. Emergency power success requires starting and loading a sufficient number of DGs to support safety-related loads in systems required to mitigate the transient and maintain the plant in a safe shutdown condition.

4. LOOP recovery (long-term) [EP REC (LONG)]. Success for this branch requires recovery of offsite power or diesel-backed ac power before the station batteries are depleted, typically 2 to 4 hours.

5. SRVs close [SRVs CLOSE].

6. HPCI (or HPCS) and RCIC [HPCI or HPCS and RCIC]. Success requirements for these branches are identical to those following a transient at Class C BWRs. Either RCIC or HPCI (or HPCS) can provide the makeup and short-term core cooling required following most transients, including failure of the EP system. HPCI and RCIC only require dc power and sufficient steam to operate the pump turbines. HPCS systems utilize a motor-driven pump but are diesel backed and utilize dedicated SW cooling.

7. Depressurization via SRV or the ADS [SRVs/ADS].

8. CRD injection [CRD PUMPS (INJ)]. Given the availability of emergency power to the CRD pumps, success requirements for this branch following a LOOP are identical to those following a transient. Manual restart of the CRD pumps is required following the LOOP.

9. LPCS, LPCI, and RHR service water injection [LPCS, LPCI, and RHRSW (INJ)]. Given the availability of emergency power, success requirements for these branches following a LOOP are identical to those following a transient.

10. Residual heat removal [RHR]. Given the availability of emergency power, the success requirements for this branch are similar to those following a nonspecific reactor trip transient at Class C BWRs. Success for any one of the three modes associated with RHR can provide the long-term decay heat removal required for transient mitigation. If emergency power fails, it must be recovered to power long-term decay heat removal equipment. However, long-term decay heat removal is not required until ~12 - 24 hours after the LOOP (well beyond the time at which emergency power must be recovered to avoid battery depletion).

The event tree constructed for a LOOP at Class A BWR plants is shown in Fig. A.23. The event tree is similar to that constructed for Class C BWR plants with the major exception that Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. However, given a LOOP, FWCI would normally be unavailable, because it is not backed by emergency power. Also, additional long-term core cooling is not required with IC success, as long as no transient-induced LOCA exists. In EP failure sequences, the IC system is the only system that can provide core cooling because FWCI would be without power. The event tree branches that are different from those for a BWR Class A transient and a BWR Class C LOOP

(LOOP-related branches only) are further discussed below.

1. Initiating event (LOOP) [LOOP]. The initiating event is a LOOP similar to that described for Class C BWRs.

2. Reactor shutdown [Rx SHUTDOWN].

3. Emergency power [EP].

4. LOOP recovery (long term) [EP REC (LONG)].

5. SRVs close [SRVs CLOSE].

6. Isolation condenser and IC makeup [IC].

7. Feedwater [FW/FWCI]. The feedwater system can provide short-term core cooling and makeup for transient mitigation. However, MFW success requires normal power supplies in most plants. If EP can be supplied to the MFW pumps (from a gas turbine, for example), then MFW can provide short-term core cooling and makeup.

8. Depressurization via SRV or ADS [SRVs/ADS].

9. CRD injection [CRD PUMPS (INJ)]. Given the availability of emergency power to the CRD pumps, success requirements for this branch following a LOOP are identical to those following a transient. Manual restart of the CRD pumps is required following the LOOP.

10. LPCS and fire water injection [LPCS and FIREWATER OR OTHER (INJ)]. Success requirements for these branches are similar to those following a nonspecific reactor trip at Class A BWRs. With interim high-pressure cooling unavailable, either LPCS or as a last resort fire water or another water source can be used to provide low-pressure water for core makeup. LPCS pumps and valves require EP to operate. Plants typically have one engine-driven fire pump that can run during a LOOP without emergency power.

11. SDC and containment cooling [SDC and CC]. Given the availability of EP or recovery of offsite power, success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs.

The event tree constructed for a BWR plant Class B LOOP is shown in Fig. A.27. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that Class B plants are equipped with HPCI systems instead of FWCI systems and an LPCI system, which represents an additional capability for providing low-pressure injection. At Class B BWRs, the CC system utilizes the LPCI pumps rather than having its own dedicated pumps. In EP failure sequences, either the IC or HPCI system can provide the required core cooling for short-term transient mitigation. However, if an SRV sticks open (transient-induced LOCA), then the IC cannot provide the makeup needed, and HPCI is required. The IC can also provide long-term cooling, but when only HPCI is operable, recovery of emergency power is necessary to power SDC-related loads.

A.4.10 BWR Loss-of-Coolant Accident

The event trees constructed define the response of BWRs to a LOCA in terms of sequences representing success and failure of plant systems. The LOCA chosen for consideration is a small-break LOCA that would require a reactor scram and continued operation of high-pressure systems. A large-break LOCA would require operation of the high-volume/low-pressure systems and is not addressed in the models.

The LOCA event tree constructed for BWR Class C plants is shown in Fig. A.32. The event-tree branches associated with core damage sequences follow (only branches that are different from Class C transient sequences are described).

1. Initiating event (small LOCA) [Small LOCA]. Any breach in the RCS on the reactor side of the MSIVs that results in coolant loss in excess of the capacity of one CRD pump and a reactor scram is considered to be a LOCA. A small-break LOCA is considered to be one in which losses are not great enough to reduce the system pressure to the operating range of the low-pressure systems.

2. Reactor shutdown [Rx SHUTDOWN].

3. PCS, MFW, HPCI or HPCS, and RCIC [PCS, MFW, HPCI or HPCS and RCIC].

4. Depressurization via SRV or ADS [SRVs/ADS].

5. Control rod drive injection [CRD PUMPS (INJ)].

5. Condensate, LPCS, LPCI, and RHR service water [COND, LPCS, LPCI, and RHRSW (INJ)].

6. Residual heat removal [RHR].

The small-break LOCA event tree constructed for BWR Class A plants is shown in Fig. A.24. The event tree branches associated with sequences leading to core damage follow (only branches that are different from BWR Class A transient branches are described).

1. Initiating event (small-break LOCA) [Small LOCA]. The initiating event is a small-break LOCA similar to that described for BWR Class C plants.

2. Reactor shutdown [Rx SHUTDOWN].

3. Power Conversion System [PCS].

4. Feedwater [FW/FWCI].

5. Depressurization via SRV or ADS [SRVs/ADS].

6. CRD injection [CRD PUMPS (INJ)].

7. Condensate, low pressure core spray, and fire water injection [COND, LPCS, and FIREWATER OR OTHER (INJ)].

8. Shutdown cooling and containment cooling [SDC and CC].

The small-break LOCA event tree constructed for BWR Class B plants is shown in Fig. A.28. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that Class B plants are equipped with HPCI systems instead of FWCI systems and have a LPCI system, which provides an additional capability for low-pressure injection. At Class B BWRs, the CC system uses the LPCI pumps rather than having its own dedicated pumps.

A.5 Branch Models and Probability Estimates

Branches included in the event tree models address potential initiating events, system and function failures, relief valve challenge, and electric power recovery. Frequencies and probabilities associated with the event tree branches were developed from data when available, or from simplified models. These models consider the probability that major components, such pumps and motor-operated valves required to operate for system success, would fail on demand.

The system models are train based, and address system configurations where success requires from one-of-one to four-of-four trains. In addition, a separate serial element (to consider, for example, the two series suction valves in many RHR systems), operator action to start a manually actuated system, and operator action to recover an initially faulted system can be addressed. The failure probabilities used in the models are conditional, which allows the impact of a component discovered to be failed to be distinguished from the impact of a component that is unavailable because of preventive maintenance. The structure of the models is described in Ref. 7.

The system models typically employ an independent failure probability of 0.01 for a supercomponent that includes a motor-driven pump or motor-operated valve, plus associated manual valves, check valves, circuit breakers, controllers, and maintenance unavailabilities. This value is consistent with values developed for similar component groupings during the 1982-83 time frame (see, for example, Ref. 8). The conditional probability of failure of the second and third like component is assumed to be 0.1 and 0.3. These values are similar to generic multiple Greek letter β and γ values (see, for example, Ref. 9).

The probabilities of failing to actuate manually actuated systems used in the 1982-83 evaluations are shown in Table A.3. These values typically assume a failure probability of 0.001 for unburdened action and 0.01 for burdened action. While these values may appear conservative for the mid-1990s, they are considered reasonable for the 1982-83 period.

The sources of initiating event frequencies and branch failure probabilities are listed in Table A.4. System, nonrecovery, and operator failure probabilities used in the analysis of individual events are listed at the end of the calculation summaries provided for each precursor in Appendix C.

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*Available for purchase from National Technical Information Service, Springfield, Virginia, 22161.

Table A.1. Rules for Precursor Calculation

1. *Event sequences requiring calculation.* If an initiating event occurs as part of a precursor (i.e., the precursor consists of an initiating event plus possible additional failures), then use the accident sequence model associated with the initiator; otherwise, use all accident sequence models affected by the observed unavailability.

2. *Initiating event probability.* If an initiating event occurs as part of the precursor, then the initiating event probability used in the calculation is 1.0. If an initiating event does not occur as part of the precursor, then the probability used for the initiating event is developed assuming a constant hazard rate. Event durations (the period of time during which the failure existed) are based on information included in the event report, if provided. If the event is discovered during testing, then half of the test period (15 days for a 30-day test interval) is typically assumed, unless a specific failure duration is identified.

3. *Branch (system) failure probability estimation.* For event tree branches for which no failed or degraded condition is observed, a probability equal to the nominal branch failure probability is assigned. For event tree branches associated with a failed system, a probability equal to the numeric value associated with the recovery class is assigned. For event tree branches that include a degraded system, the estimated failure probability is modified to represent the loss of redundancy.

4. *Nonrecovery probability.* If an initiating event or a total system failure occurred as a part of the precursor, the basic event representing the probability of not recovering from the failure is revised to reflect the potential for recovery of the specific failures observed during the event. For condition assessments, the probability of nonrecovery is estimated under the assumption that an initiating event has occurred.

5. *Failures in Support Systems.* If the support system is not included in the ASP models, the impact of the failure is addressed by setting the affected components to be failed. The modeling of a support system failure recognizes that as long as the failure remains unrecovered, all affected components are unavailable; however, if the support system failure is recovered, all affected components are also recovered. This can be modeled through multiple calculations that address the impact of failure and success of the failed component. Calculated core damage probabilities for each case are normalized based on the likelihood of not recovering from the support system failure. (Except for emergency power, support systems are not addressed in the 1982-83 ASP models.)

Table A.2 ASP Reactor Plant Classes

Plant name	Plant class	Plant name	Plant class
ANO - Unit 1	PWR Class D	Nine Mile Point 1	BWR Class A
ANO - Unit 2	PWR Class G	North Anna 1	PWR Class A
Beaver Valley 1	PWR Class A	North Anna 2	PWR Class A
Big Rock Point	BWR Class A	Oconee 1	PWR Class D
Browns Ferry 1	BWR Class C	Oconee 2	PWR Class D
Browns Ferry 2	BWR Class C	Oconee 3	PWR Class D
Browns Ferry 3	BWR Class C	Oyster Creek	BWR Class A
Braidwood 1	PWR Class B	Palisades	PWR Class G
Braidwood 2	PWR Class B	Peach Bottom 2	BWR Class C
Brunswick 1	BWR Class C	Peach Bottom 3	BWR Class C
Brunswick 2	BWR Class C	Pilgrim 1	BWR Class C
Calvert Cliffs 1	PWR Class G	Point Beach 1	PWR Class B
Calvert Cliffs 2	PWR Class G	Point Beach 2	PWR Class B
Cook 1	PWR Class B	Prairie Island 1	PWR Class B
Cook 2	PWR Class B	Prairie Island 2	PWR Class B
Cooper Station	BWR Class C	Quad Cities 1	BWR Class C
Crystal River 3	PWR Class D	Quad Cities 2	BWR Class C
Davis-Besse	PWR Class B	Rancho Seco	PWR Class D
Dresden 2	BWR Class B	Robinson 2	PWR Class B
Dresden 3	BWR Class B	Salem 1	PWR Class B
Duane Arnold	BWR Class C	Salem 2	PWR Class B
Farley 1	PWR Class B	San Onofre 1	Unique
Farley 2	PWR Class B	San Onofre 2	PWR Class H
Fitzpatrick	BWR Class C	San Onofre 3	PWR Class H
Fort Calhoun	PWR Class G	Sequoyah 1	PWR Class B
Ginna	PWR Class B	Sequoyah 2	PWR Class B
Grand Gulf 1	BWR Class C	St. Lucie 1	PWR Class G
Haddam Neck	PWR Class B	St. Lucie 2	PWR Class G
Hatch 1	BWR Class C	Summer 1	PWR Class B
Hatch 2	BWR Class C	Surry 1	PWR Class A
Indian Point 2	PWR Class B	Surry 2	PWR Class A
Indian Point 3	PWR Class B	Susquehanna 1	BWR Class C
Kewaunee	PWR Class B	Three Mile Island 1	PWR Class D
LaCrosse	Unique	Trojan	PWR Class B
LaSalle 1	BWR Class C	Turkey Point 3	PWR Class B
Maine Yankee	PWR Class B	Turkey Point 4	PWR Class B
McGuire 1	PWR Class B	Vermont Yankee	BWR Class C
McGuire 2	PWR Class B	Yankee Rowe	PWR Class B
Millstone 1	BWR Class A	Zion 1	PWR Class B
Millstone 2	PWR Class G	Zion 2	PWR Class B
Monticello	BWR Class C		

Table A.3 Operator Action Failure Probabilities

Operator action	Failure probability
BWRs	
Control rod drive water use	0.01
Initiation of RHR service water, fire water	0.01
Shutdown cooling	0.00001
Standby liquid control initiation	0.01
PWRs	
Condensate/MFW recovery	0.001
Containment spray recirculation	0.001
Emergency core cooling recirculation	0.001
Failure to block stuck-open PORVs	0.001
SG depressurization	0.01
Use feed and bleed to cool core	0.01

Table A.4 Branch Failure Data Sources

Initiator or branch	Source
Reactor trip frequency	D. Mackowiak, C. Gentillon, and K. Smith, *Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments*, NUREG/CR-3862, 1985.
LOOP frequency	P. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, U. S. Nuclear Regulatory Commission, NUREG-1032, June 1988 and J. Minarick, *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, Oak Ridge National Laboratory, Oak Ridge, TN, 1989.
Small-break LOCA frequency	J. Minarick *et al.*, *Precursors to Potential Severe Core Damage Accidents: 1986, A Status Report*, NUREG/CR-4674, U. S. Nuclear Regulatory Commission, Vol. 5, Table 4.2.
SGTR frequency	D. Ericson *et al.*, *Analysis of Core Damage Frequency: Internal Events Methodology*, NUREG/CR-4550, Vol. 1, Rev. 1, U. S. Nuclear Regulatory Commission, 1990.
PORV challenge rates	R. Bertucio and J. Julius, *Analysis of Core Damage Frequency: Surry, Unit 1, Internal Events*, NUREG/CR-4550, Vol. 3, Rev. 1, U. S. Nuclear Regulatory Commission, 1990.
LOOP nonrecovery probability	P. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, U. S. Nuclear Regulatory Commission, June 1988 and J. Minarick, *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, Oak Ridge National Laboratory, Oak Ridge, TN, 1989.
RCP seal LOCA probability	T. Wheeler *et al.*, *Analysis of Core Damage Frequency from Internal Events: Expert Judgement Elicitation*, NUREG/CR-4550, Vol. 2, U. S. Nuclear Regulatory Commission, 1989.
System failure probabilities	Simplified system models

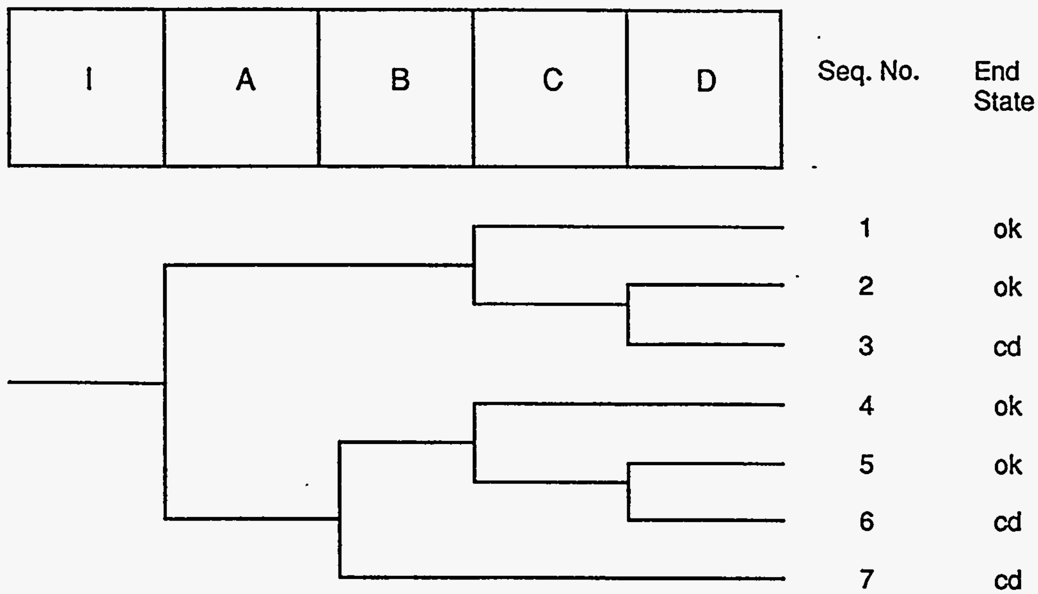

Fig. A.1 Hypothetical core damage model

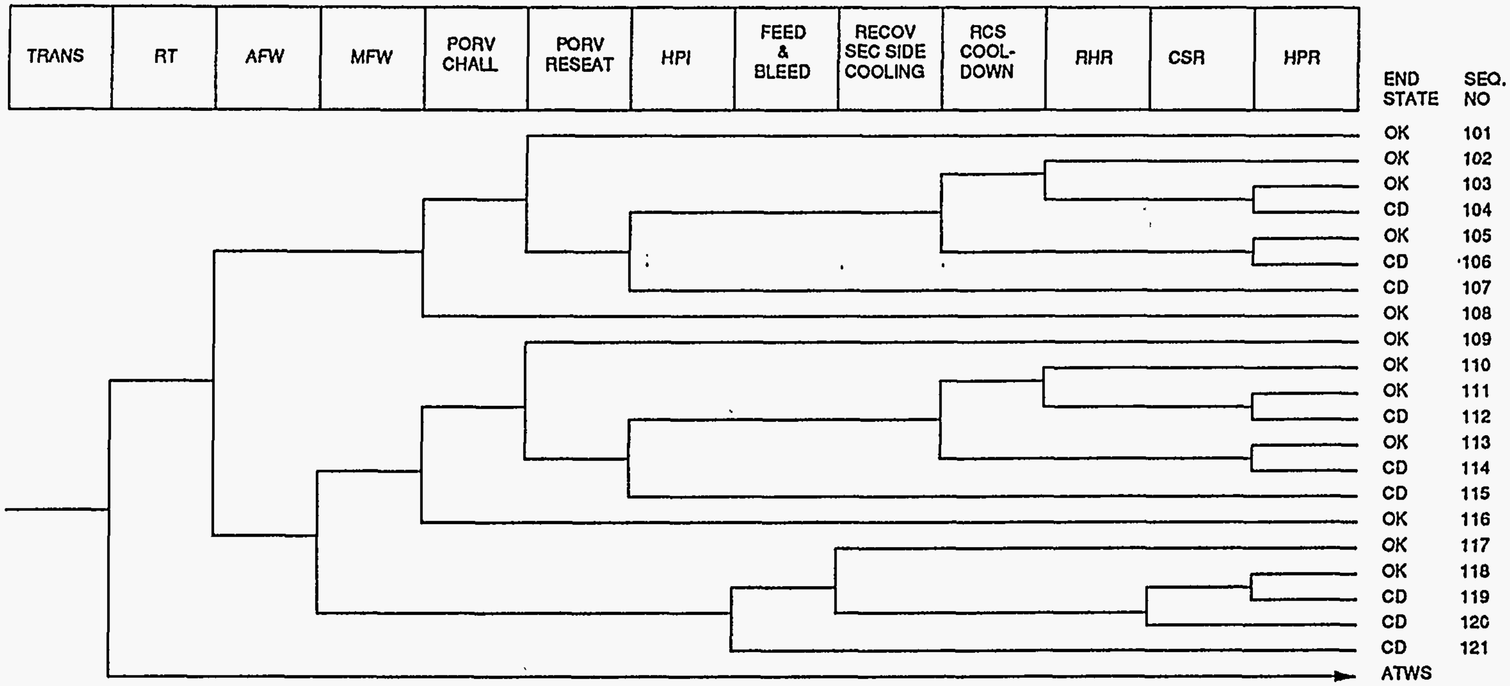

Fig. A.2 PWR Class A nonspecific reactor trip

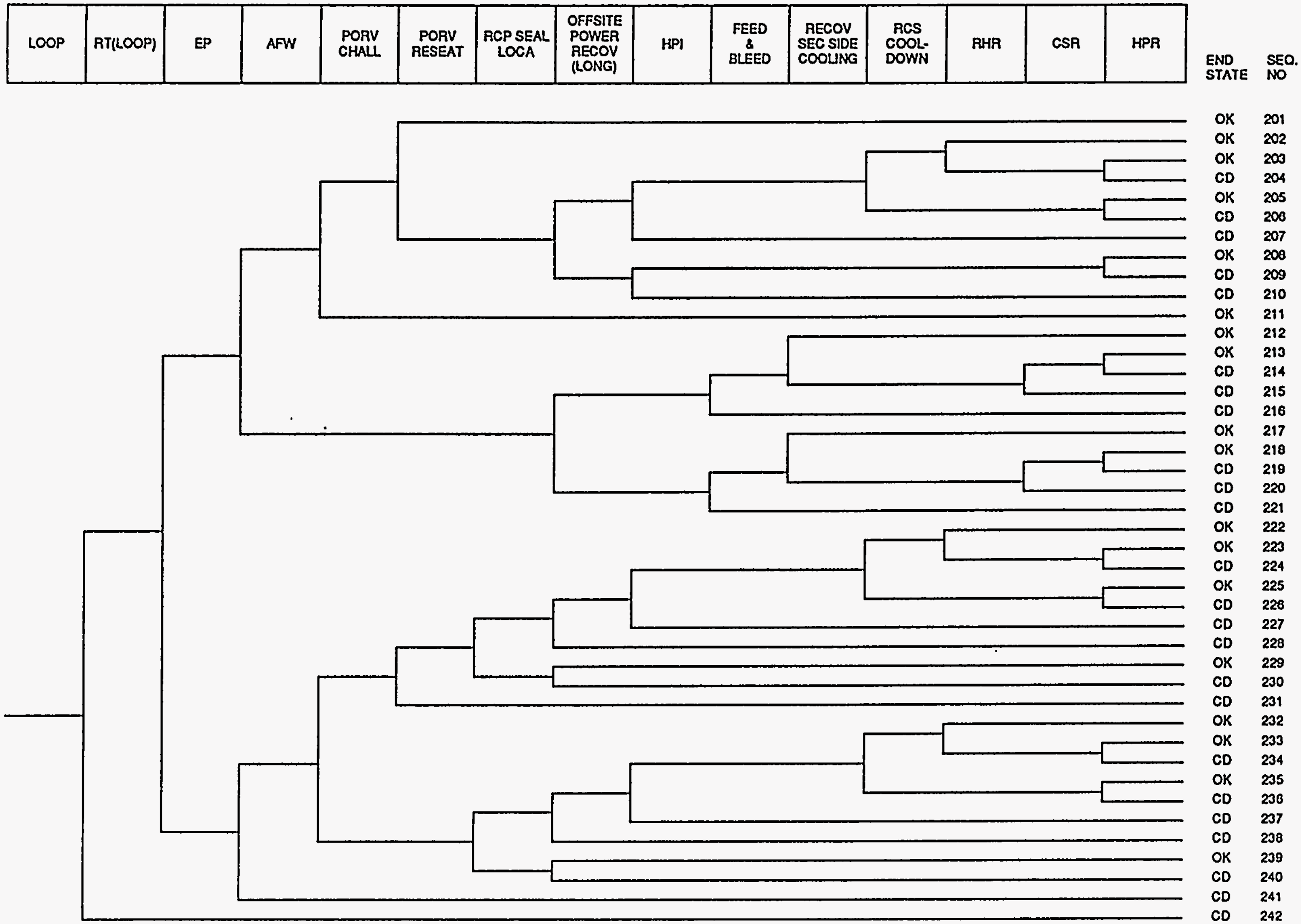

Fig. A.3 PWR Class A loss of offsite power

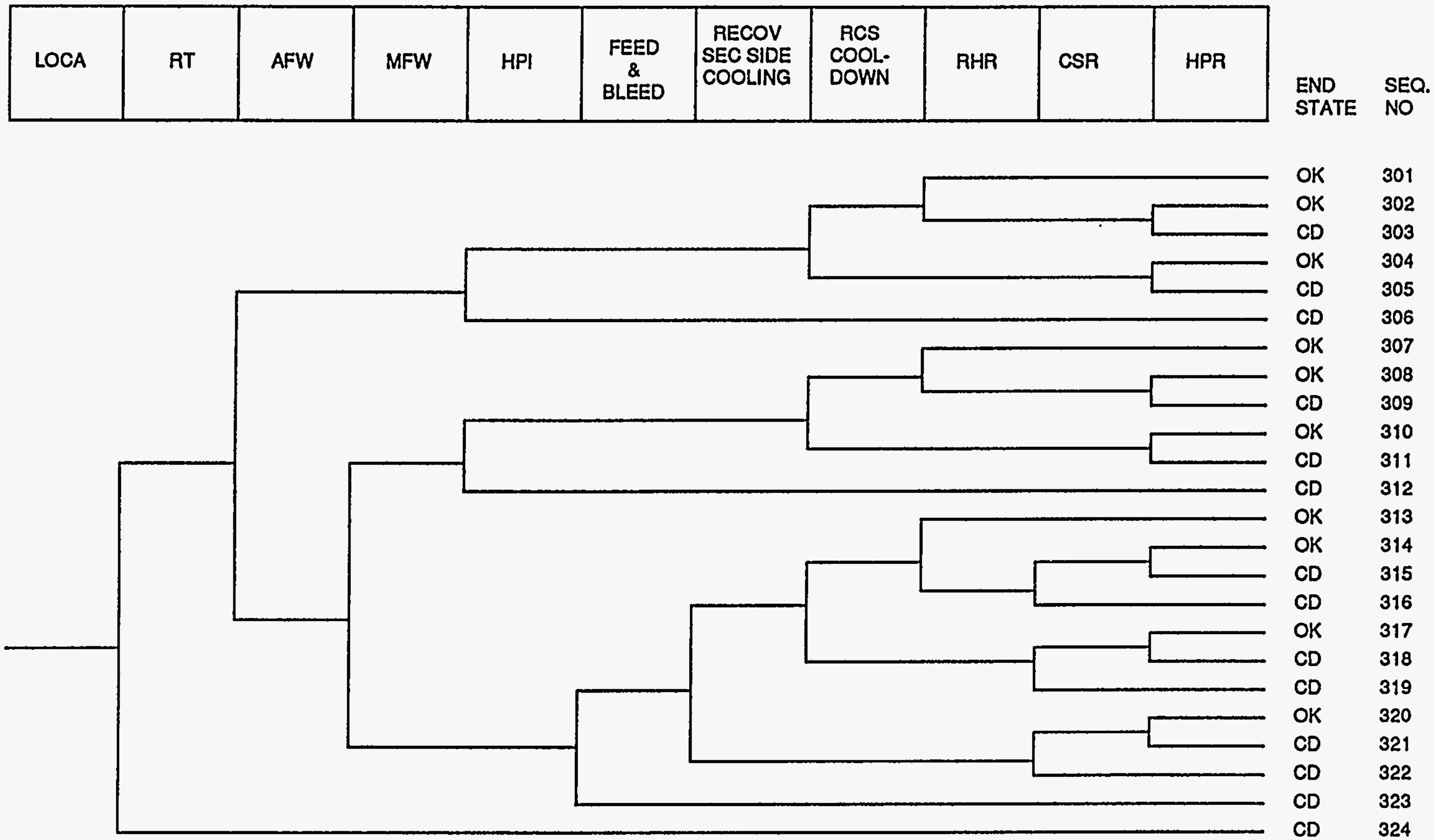

Fig. A.4 PWR Class A small-break loss-of-coolant accident

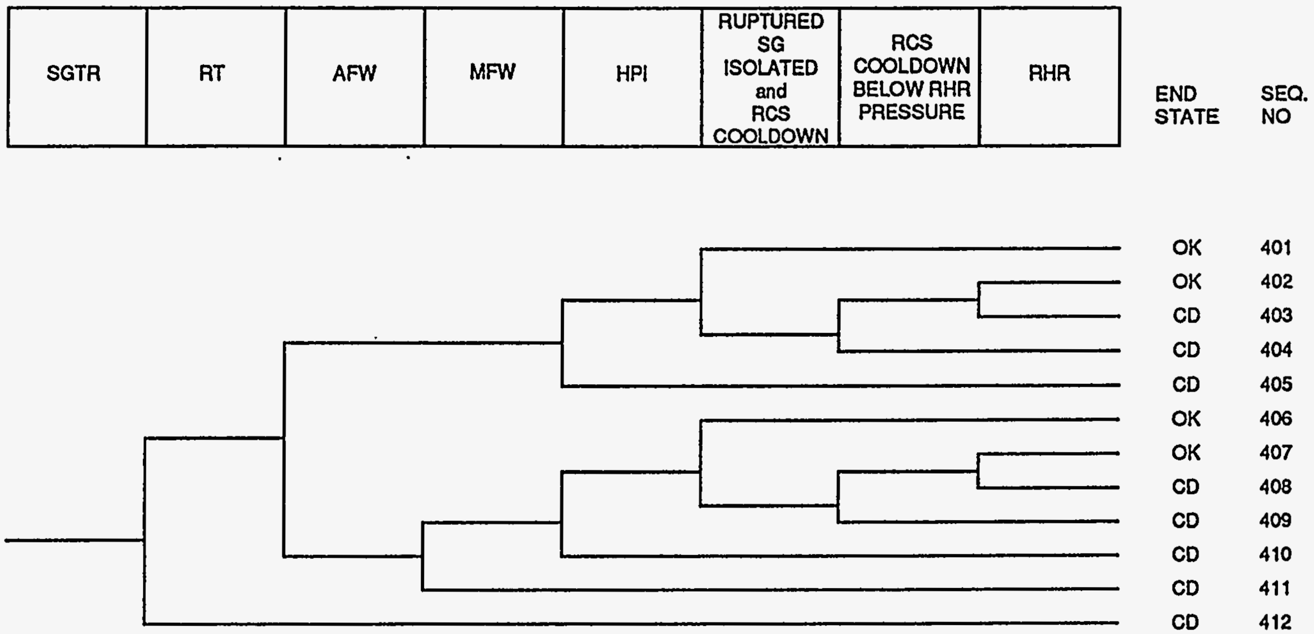

Fig. A.5 PWR Class A steam generator tube rupture

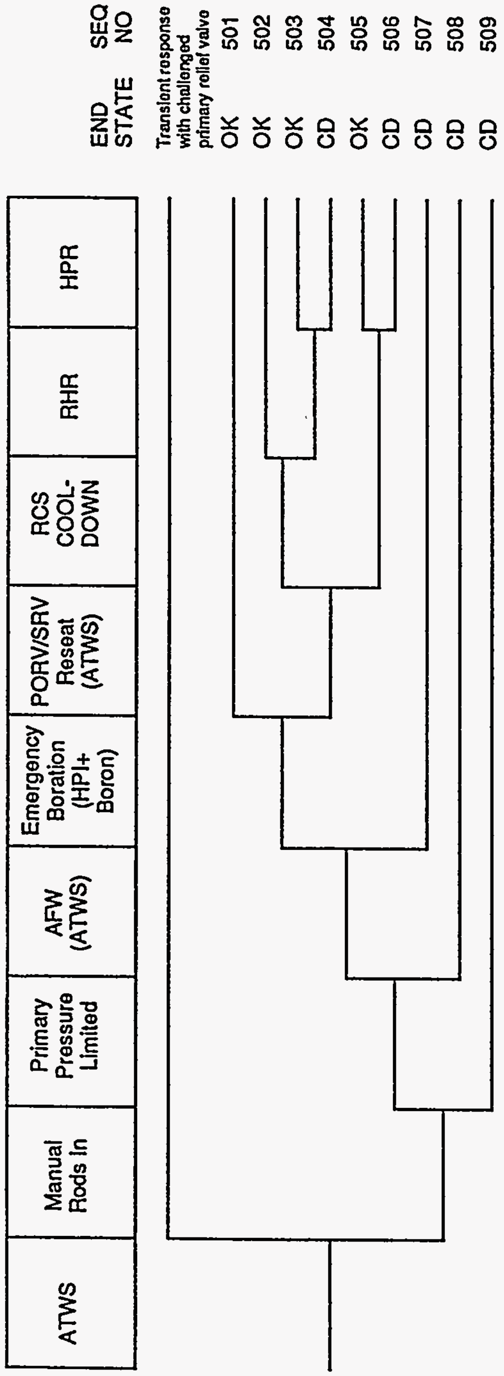

Fig. A.6 PWR Class A anticipated transient without scram

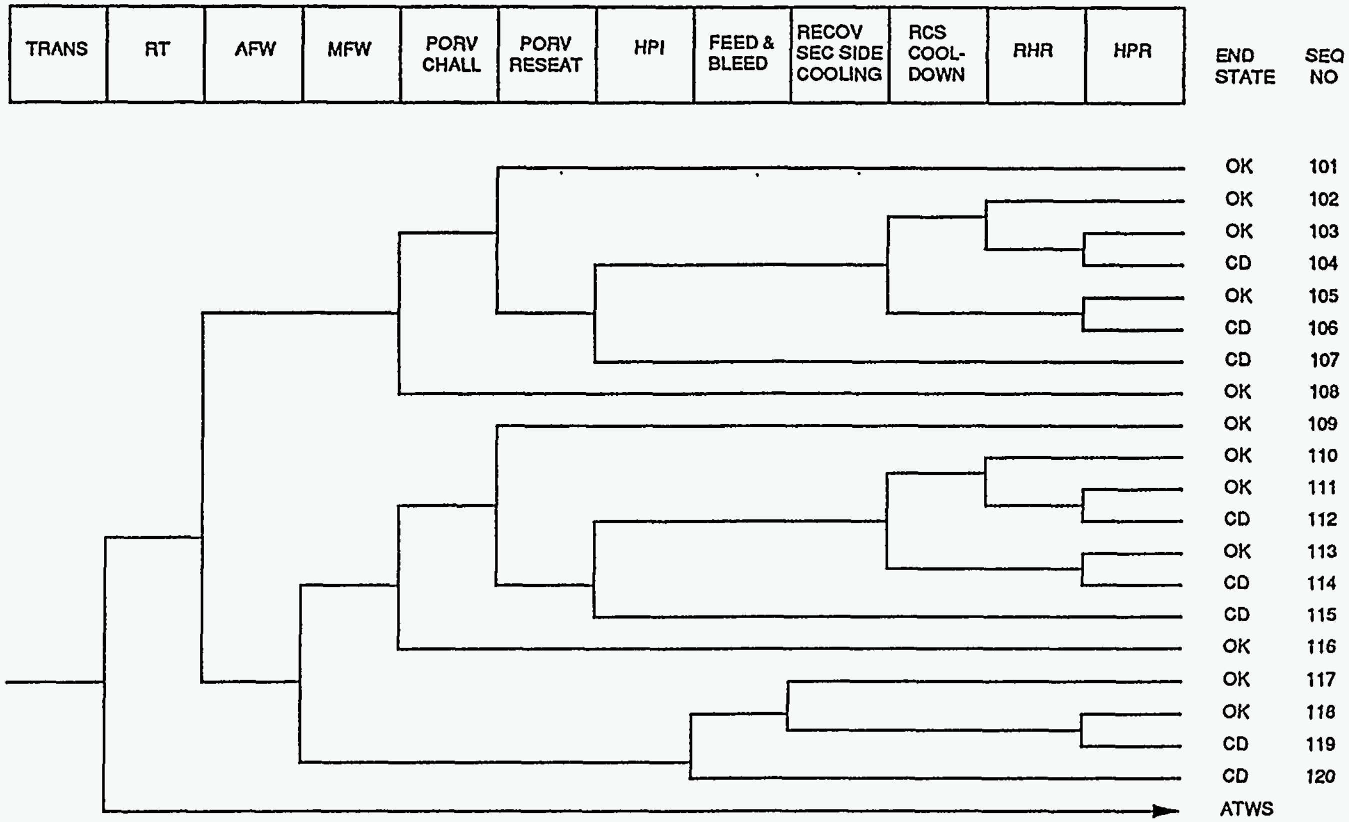

Fig. A.7 PWR Classes B and D nonspecific reactor trip

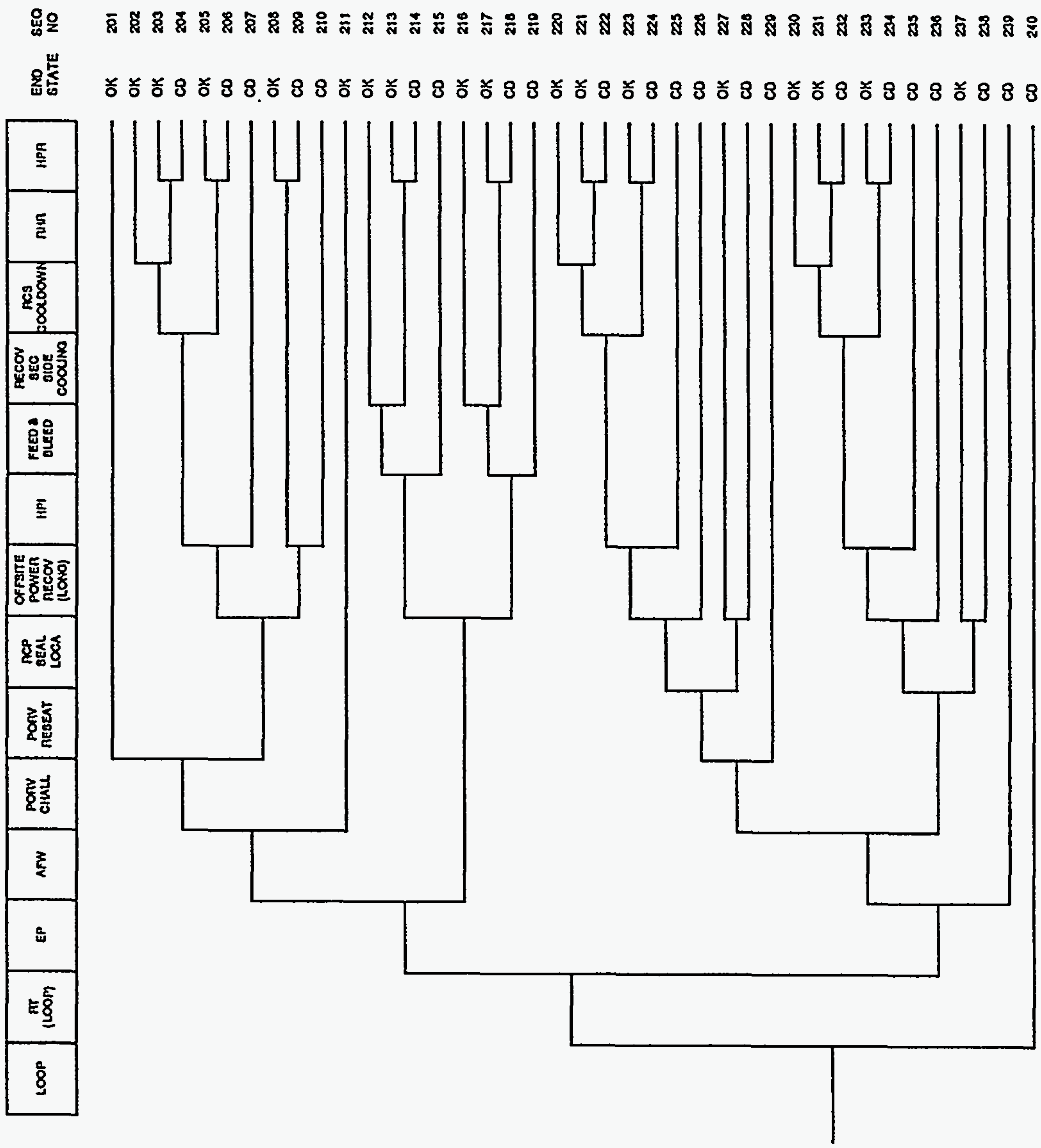

Fig. A.8 PWR Classes B and D loss of offsite power

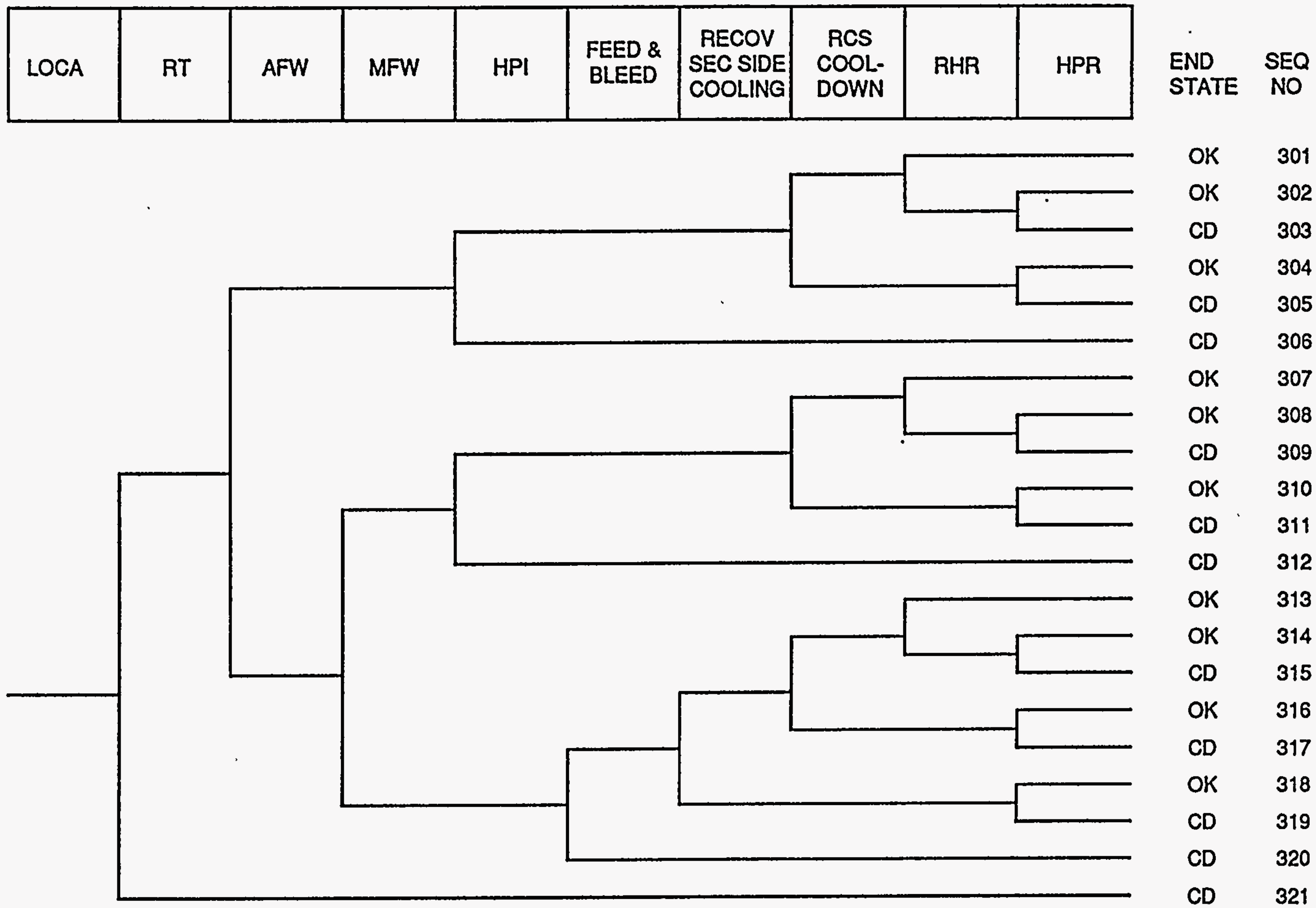

Fig. A.9 PWR Classes B and D small-break loss-of-coolant accident

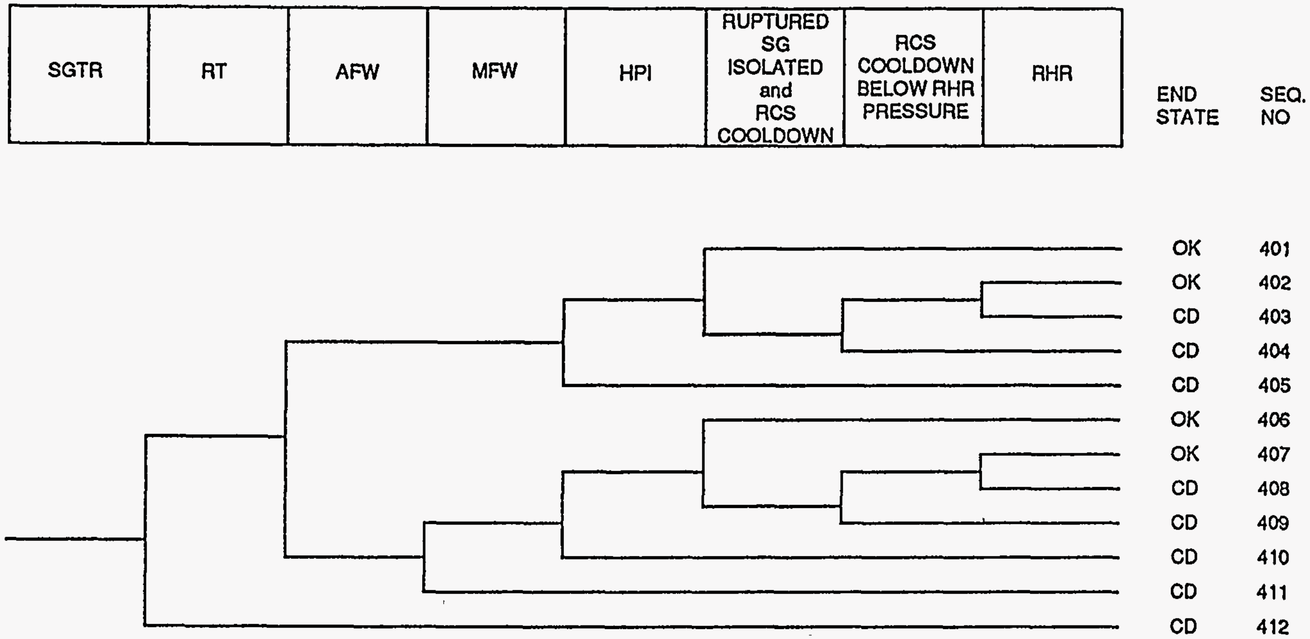

Fig. A.10 PWR Classes B and D steam generator tube rupture

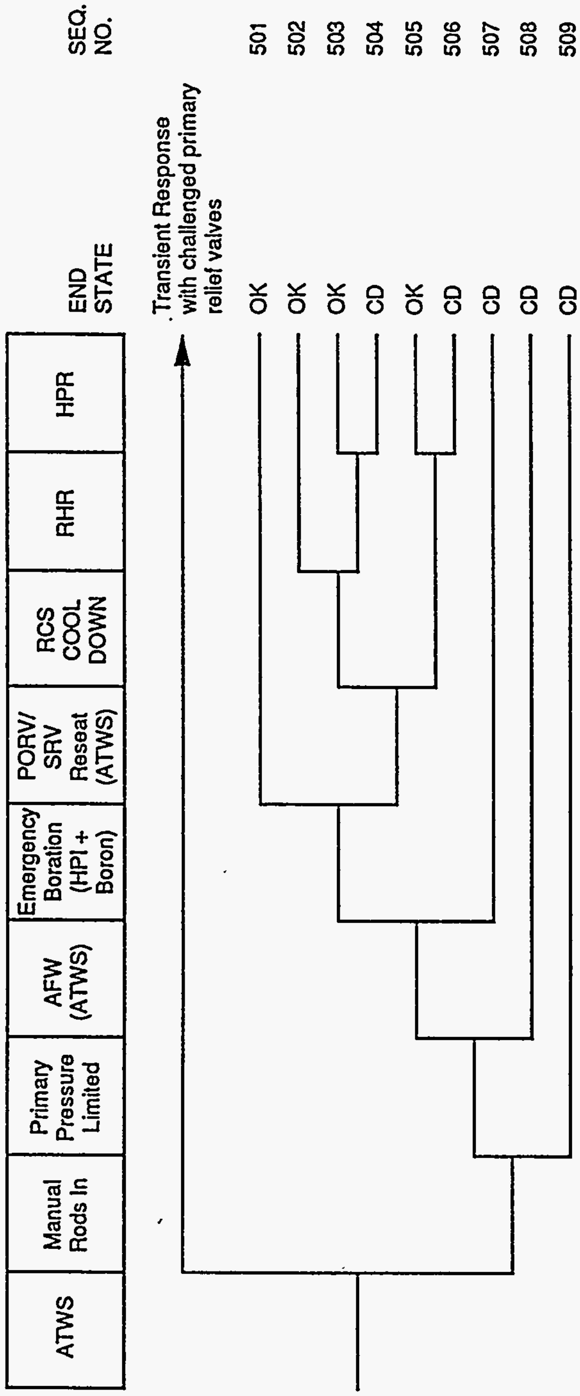

Fig. A.11 PWR Classes B and D anticipated transient without scram

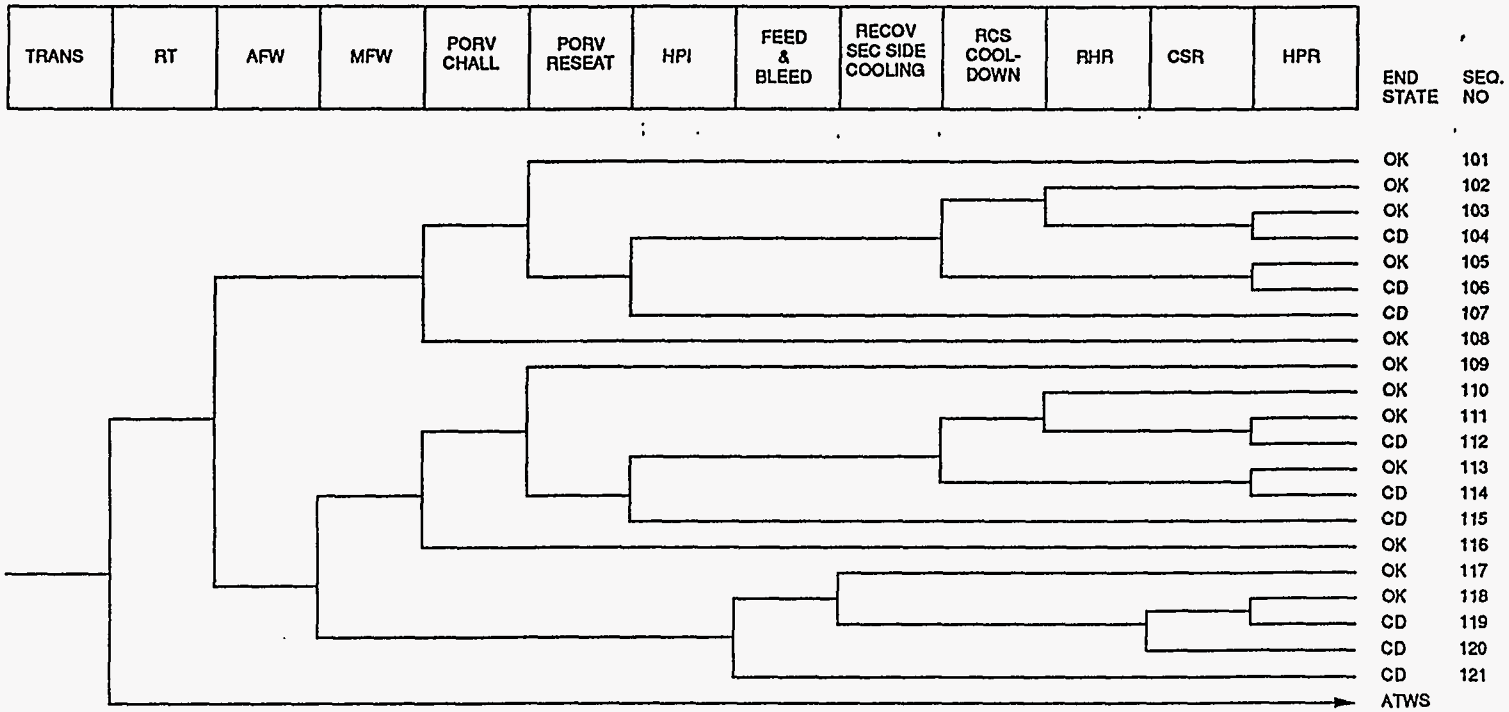

Fig. A.12 PWR Class G nonspecific reactor trip

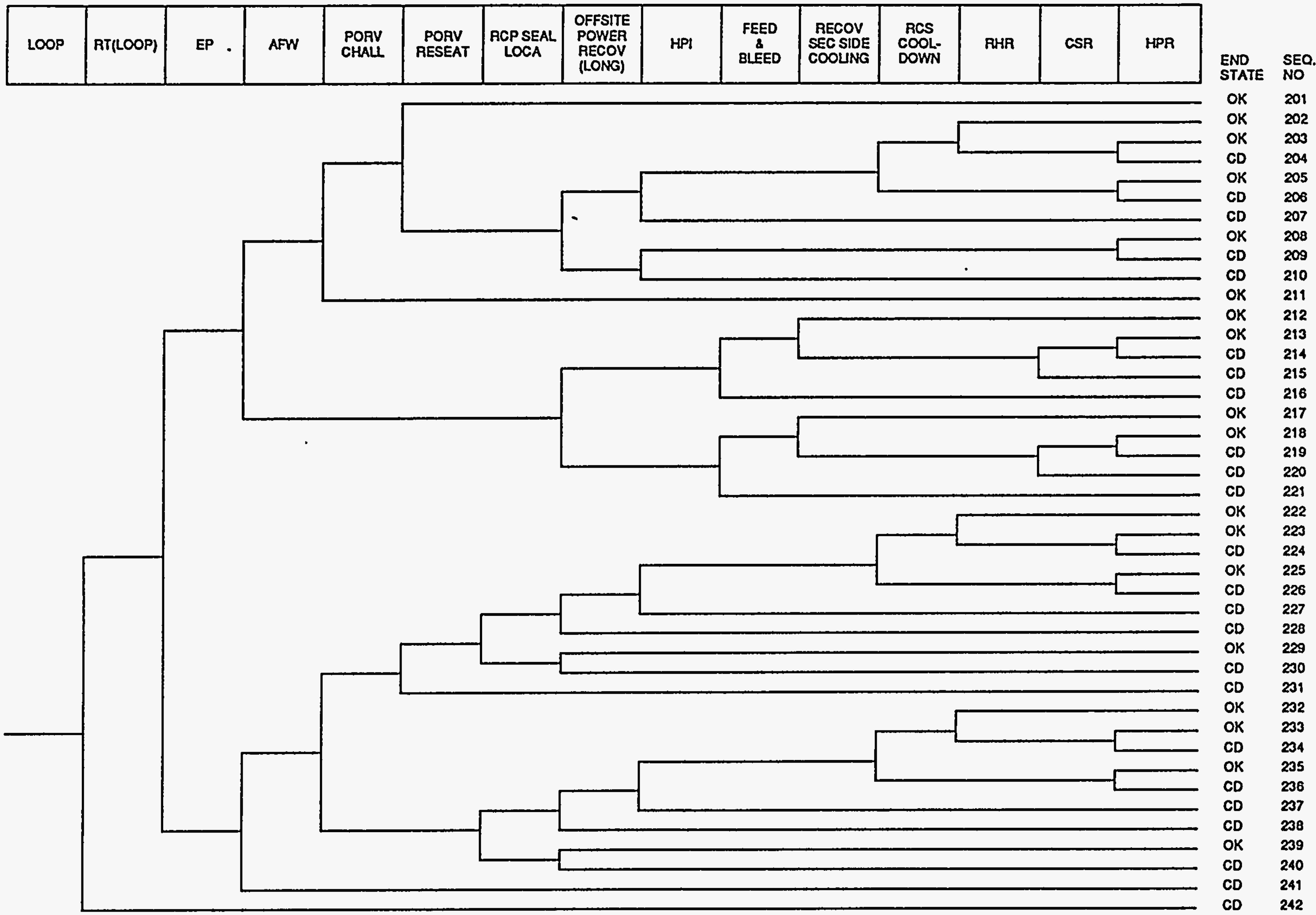

Fig. A.13 PWR Class G loss of offsite power

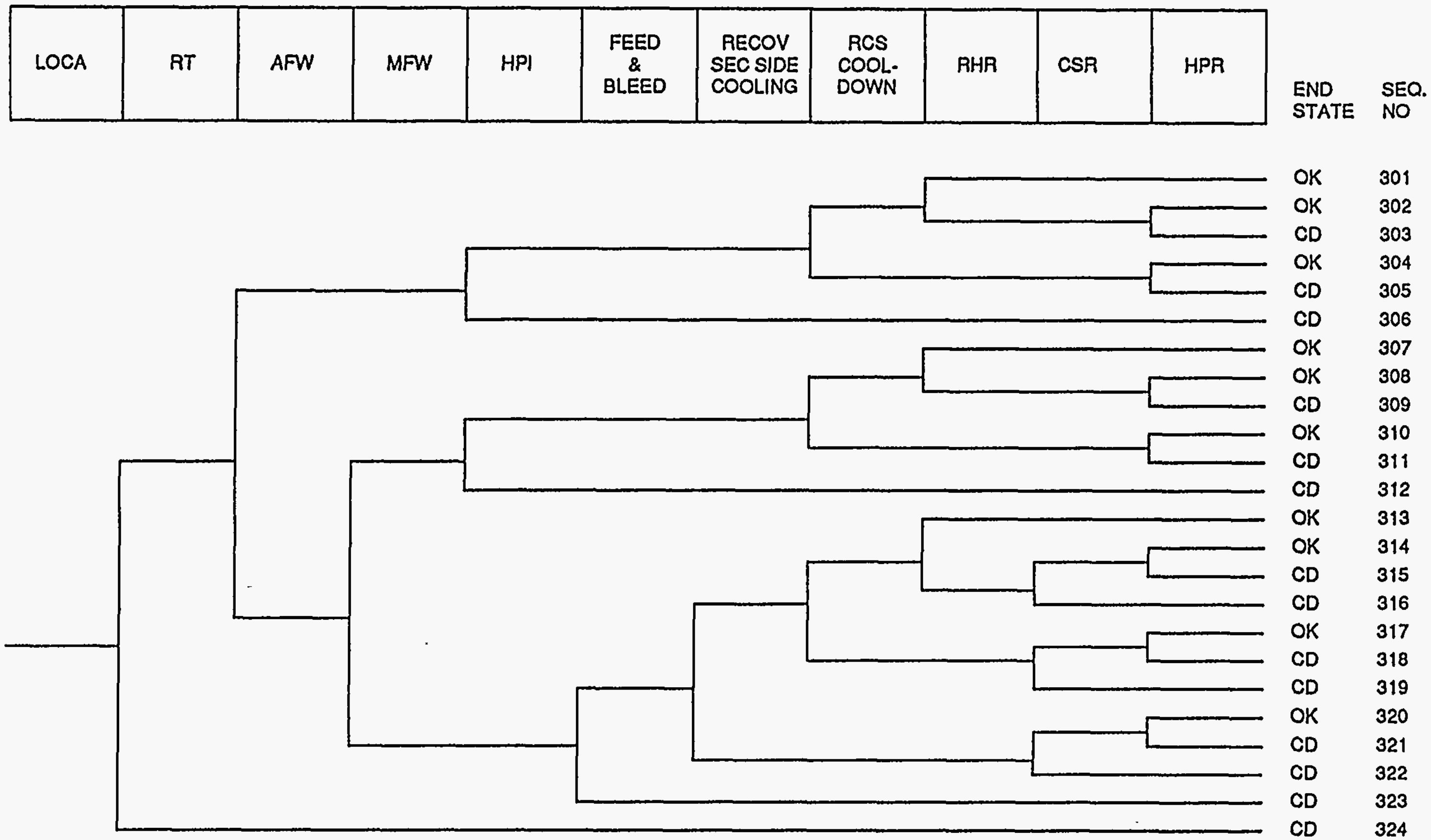

Fig. A.14 PWR Class G small-break loss-of-coolant accident

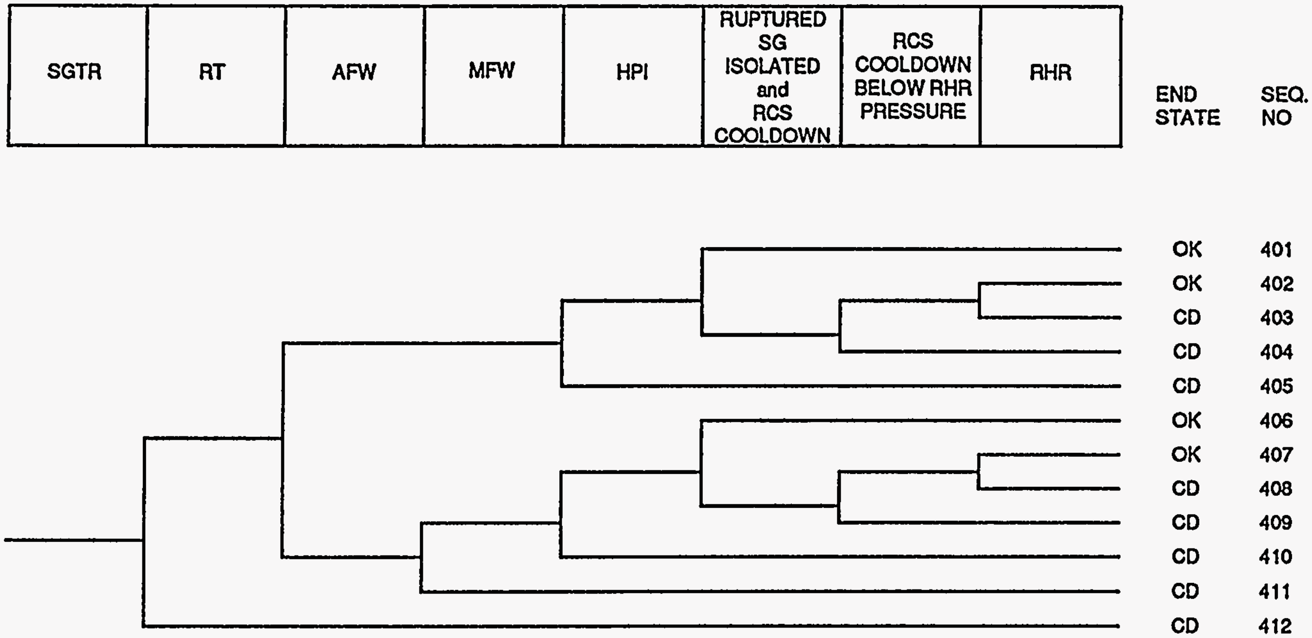

Fig. A.15 PWR Class G steam generator tube rupture

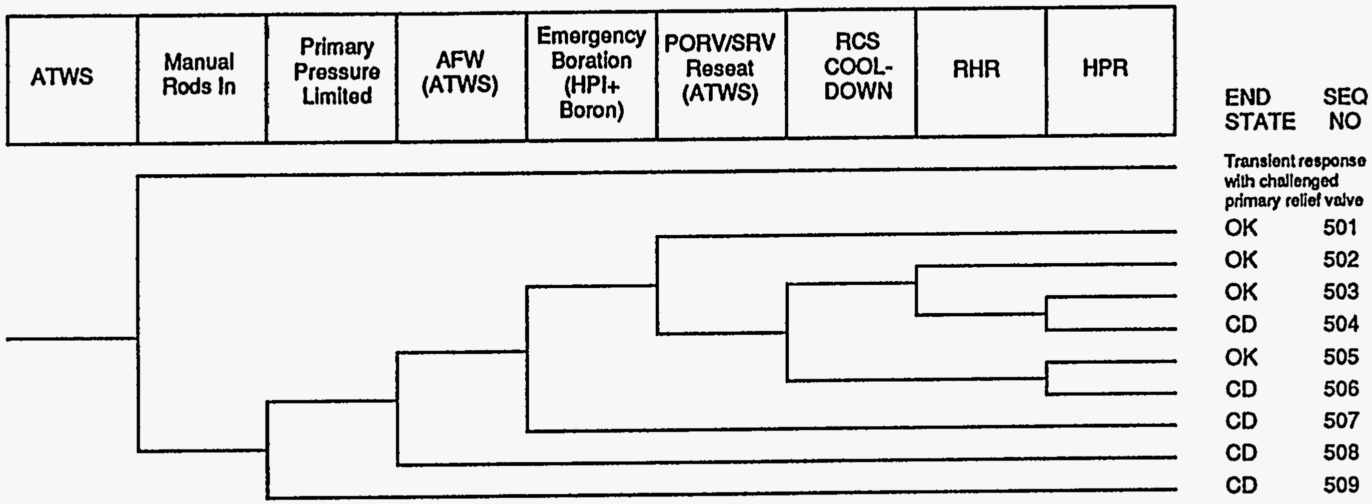

Fig. A.15 PWR Class G anticipated transient without scram

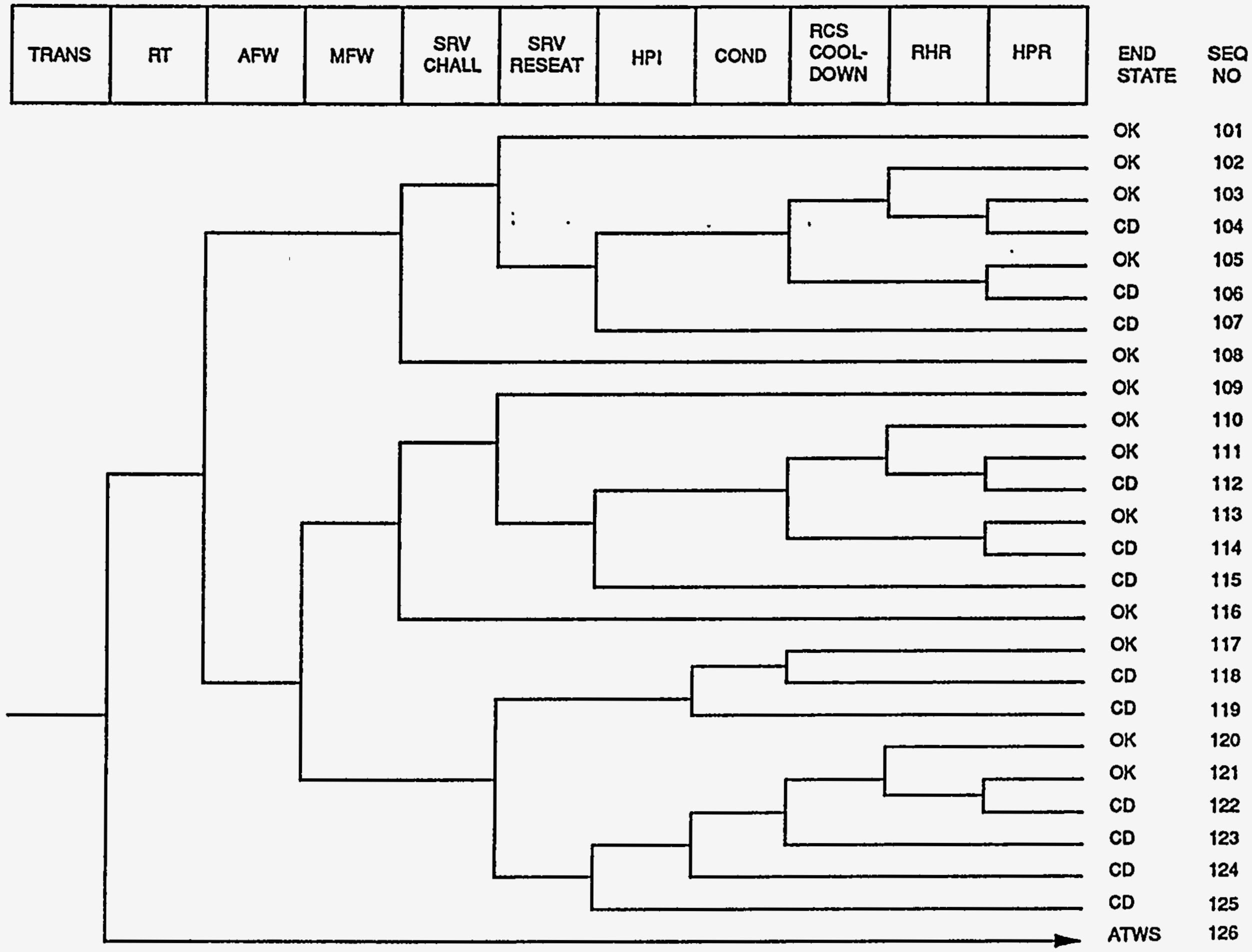

Fig. A.17 PWR Class H nonspecific reactor trip

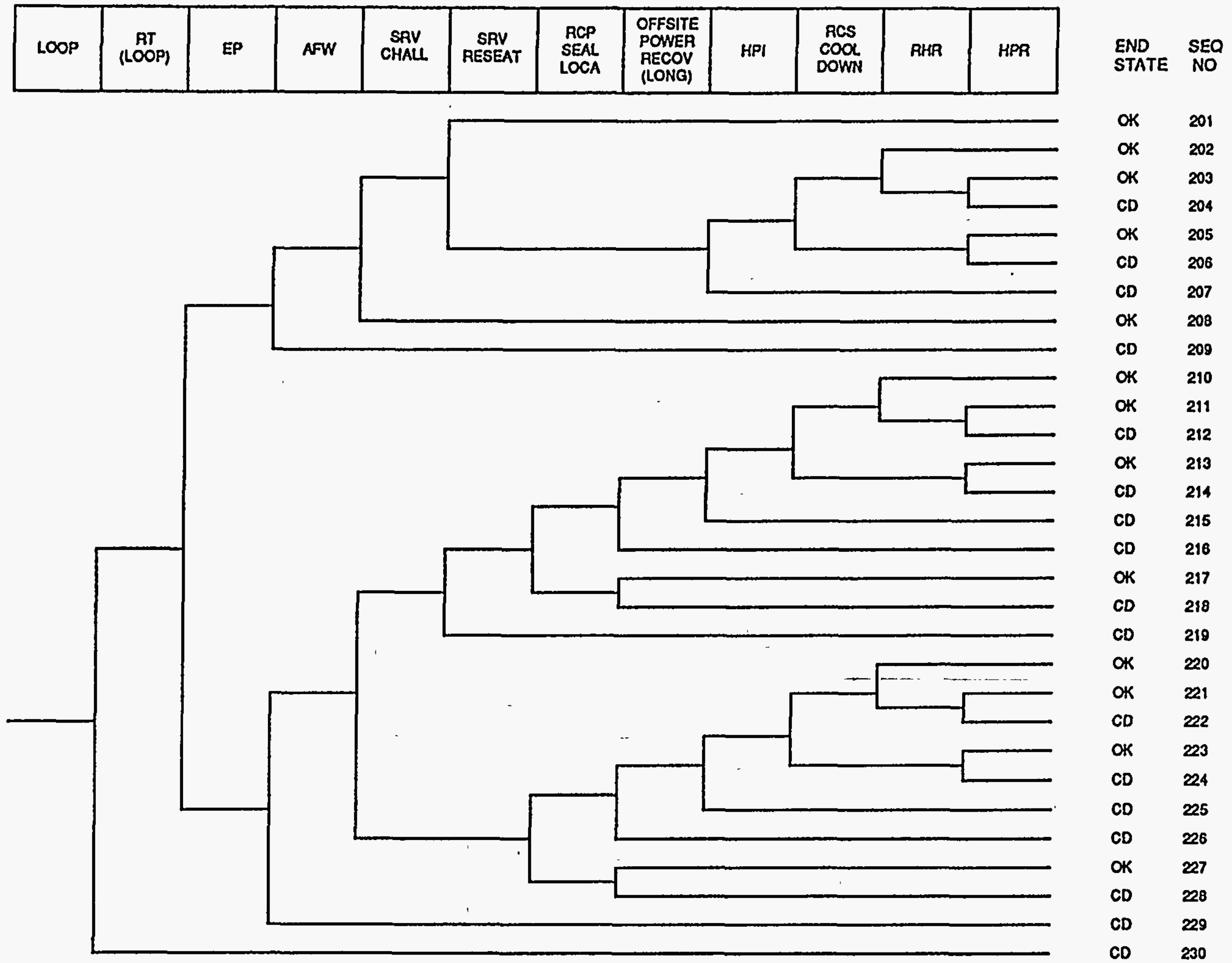

Fig. A.18 PWR Class H loss of offsite power

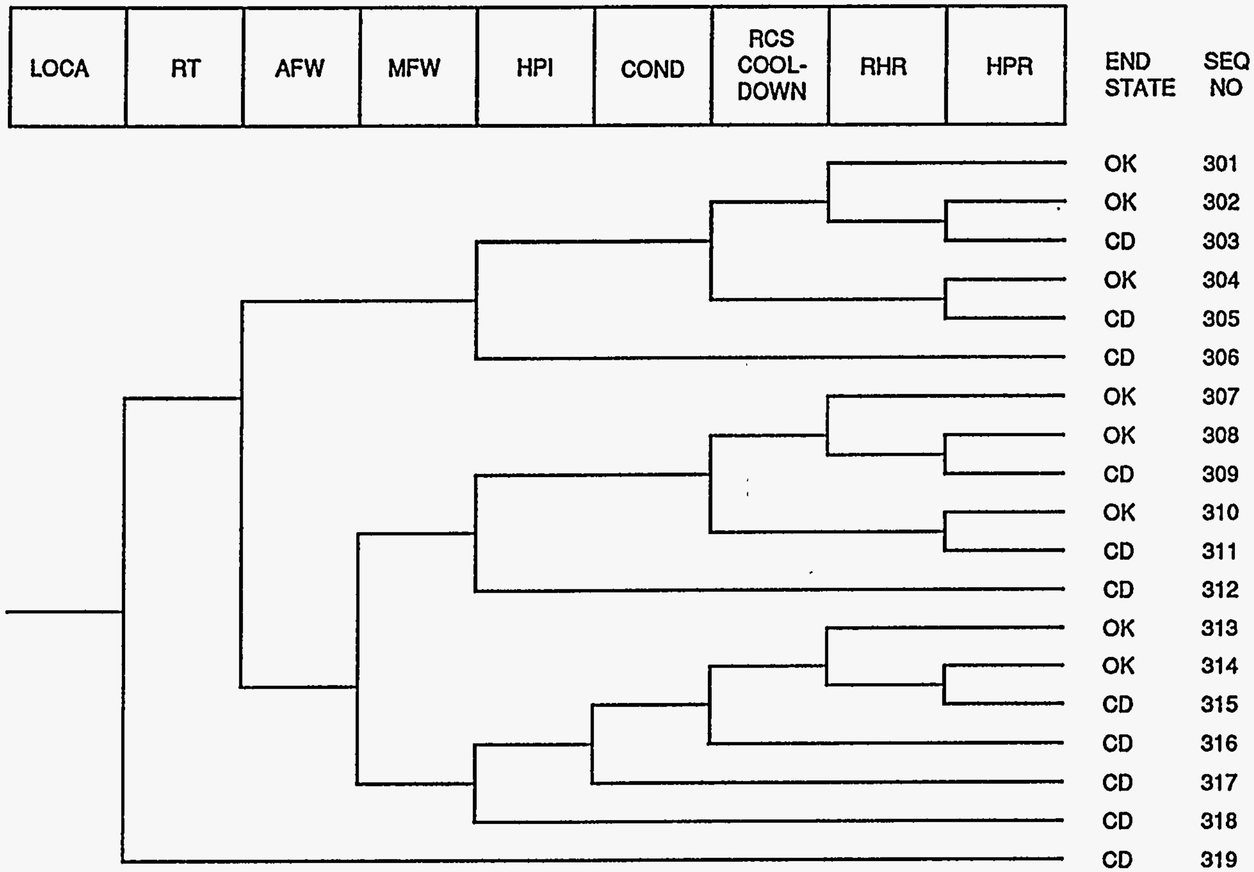

Fig. A.19 PWR Class H small-break loss-of-coolant accident

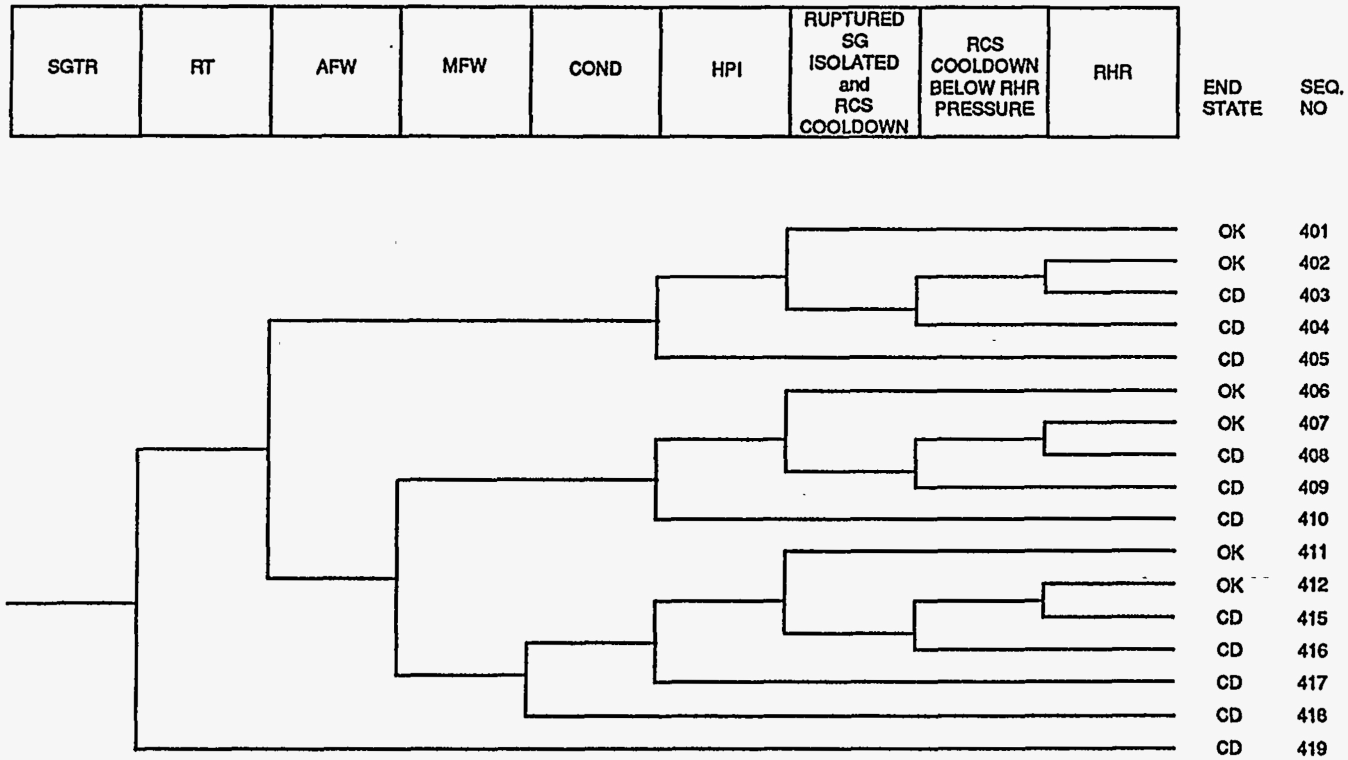

Fig. A.20 PWR Class H steam generator tube rupture

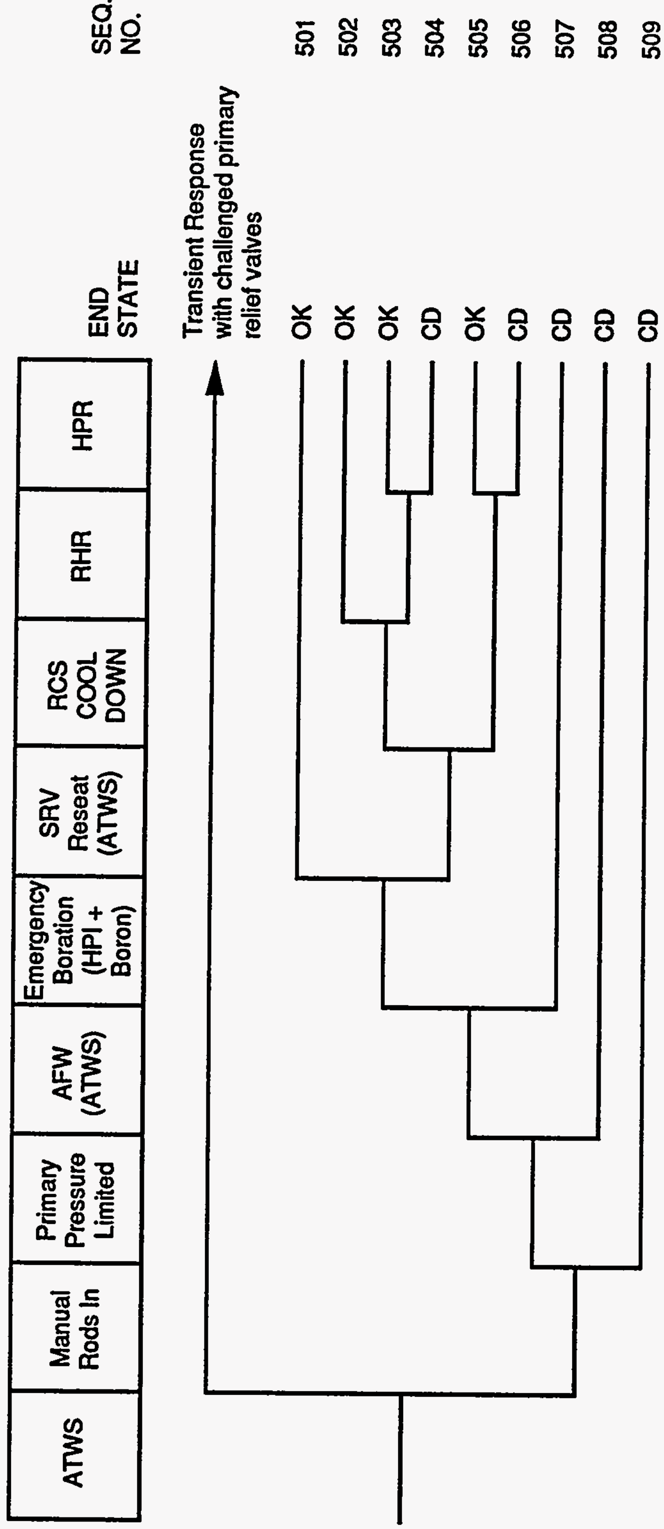

Fig. A.21 PWR Class H anticipated transient without scram

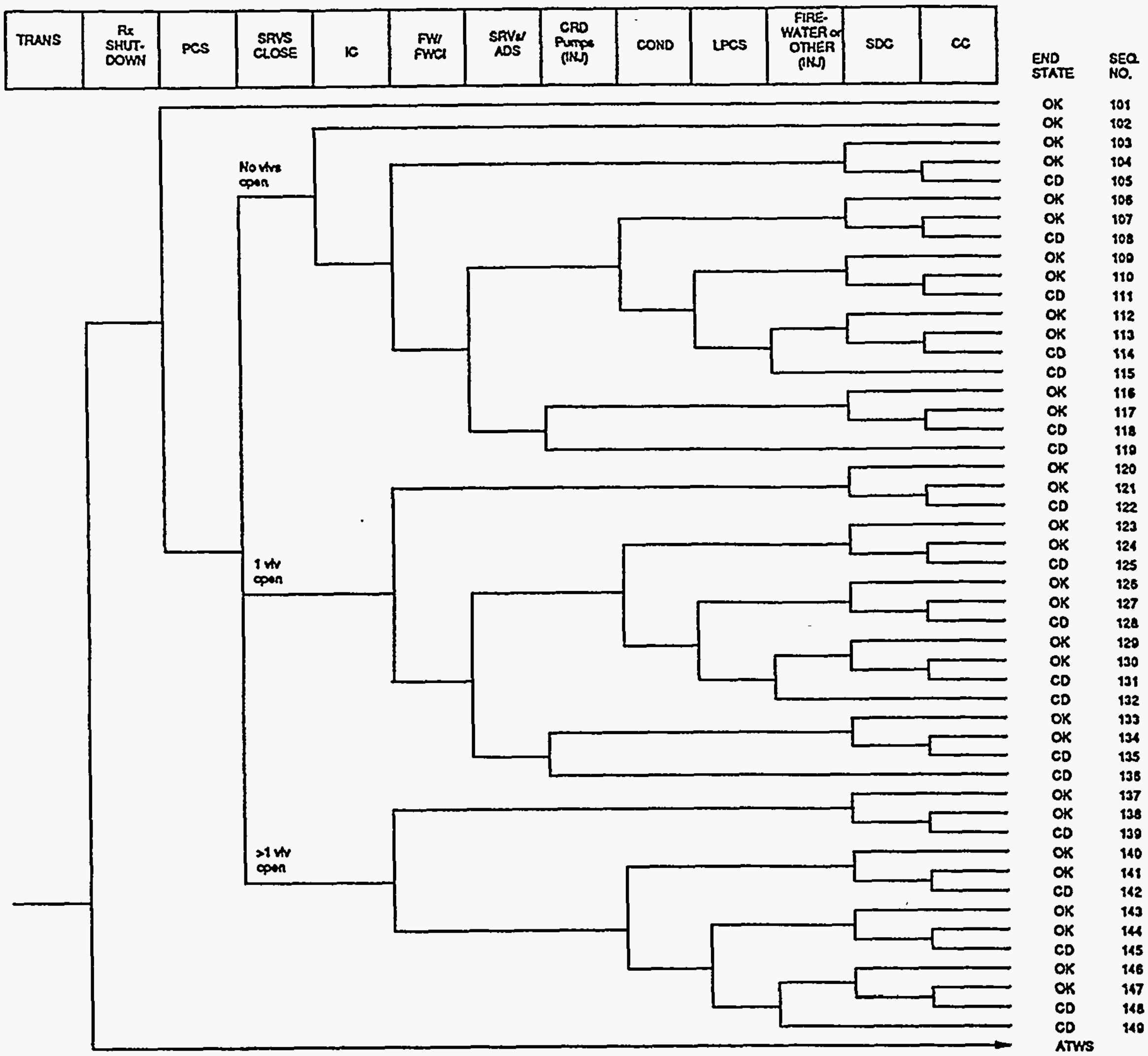

Fig. A.22 BWR Class A nonspecific reactor trip

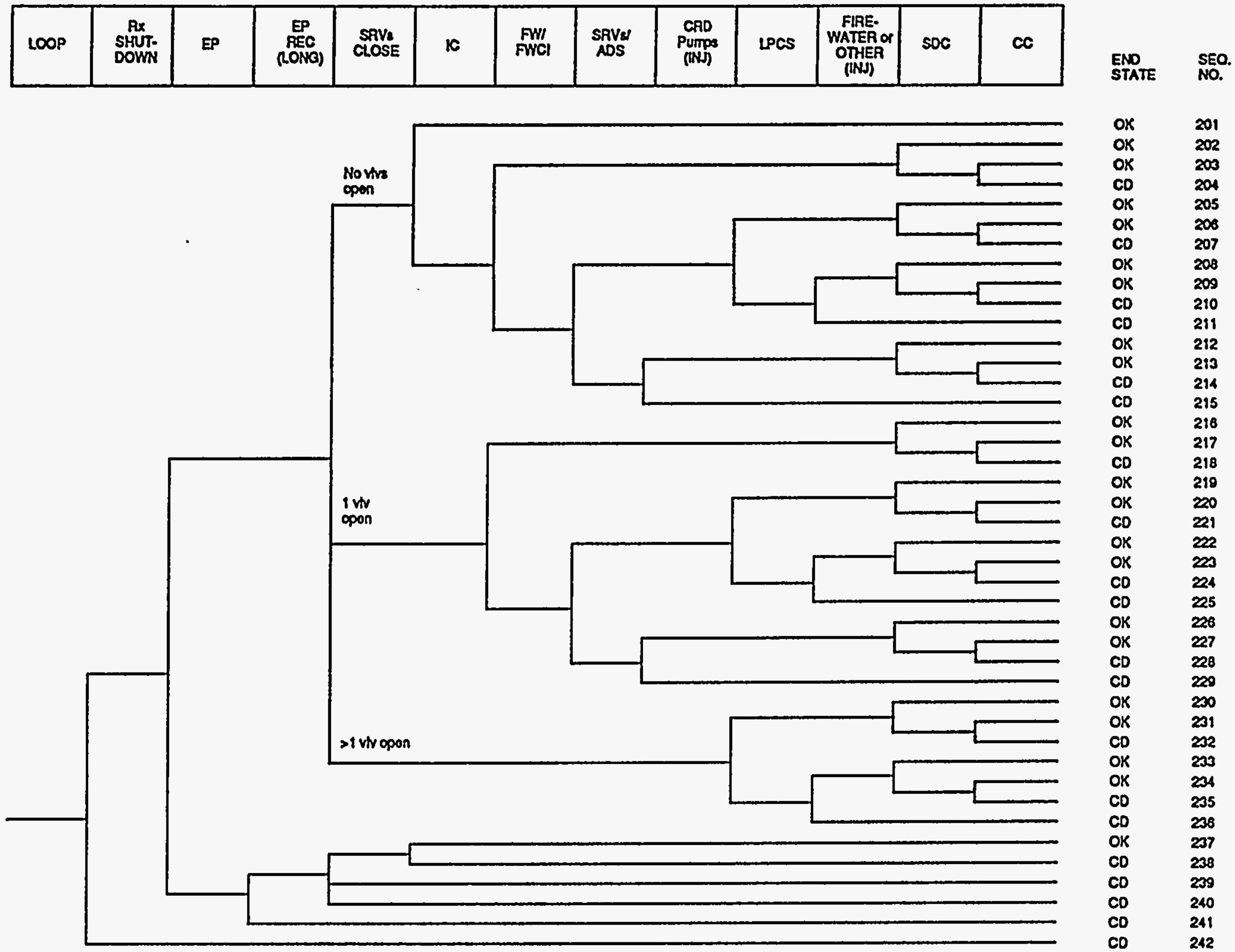

Fig. A.23 BWR Class A loss of offsite power

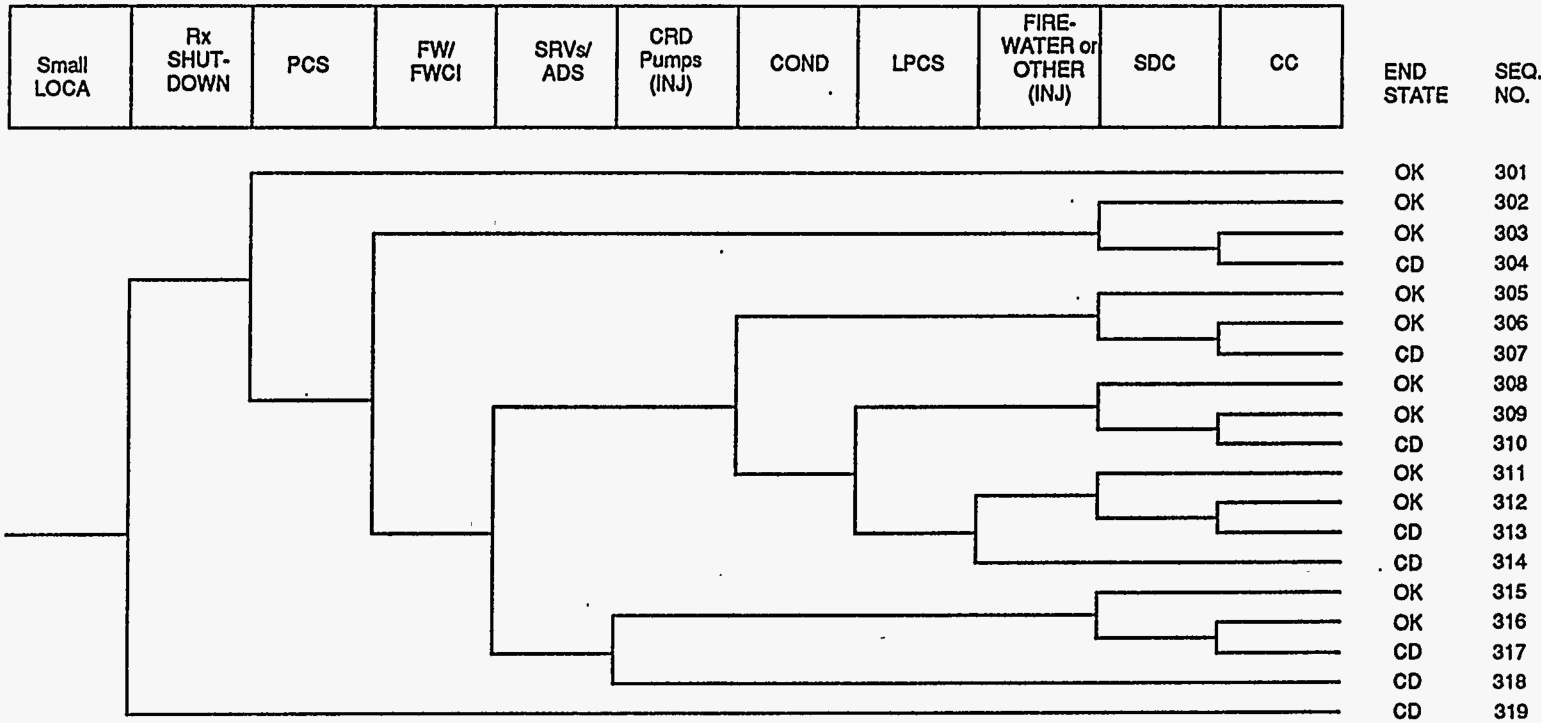

Fig. A.24 BWR Class A small-break loss-of-coolant accident

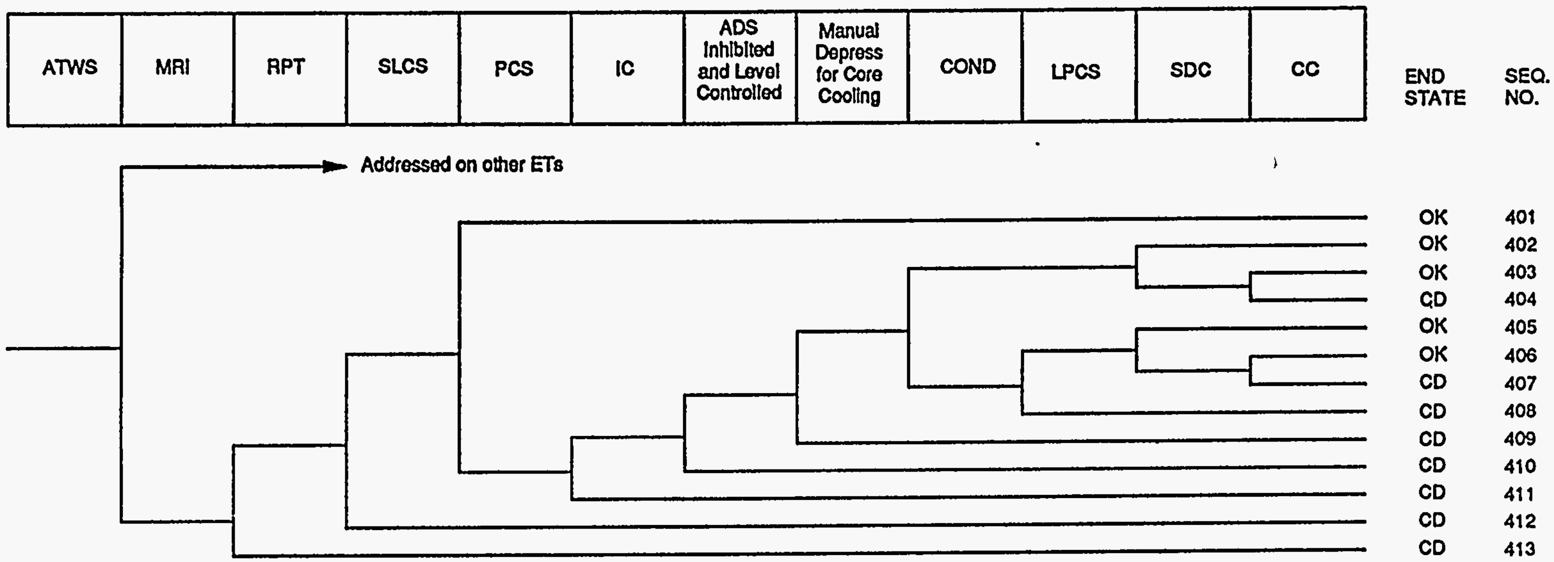

Fig. A.25 BWR Class A anticipated transient without scram

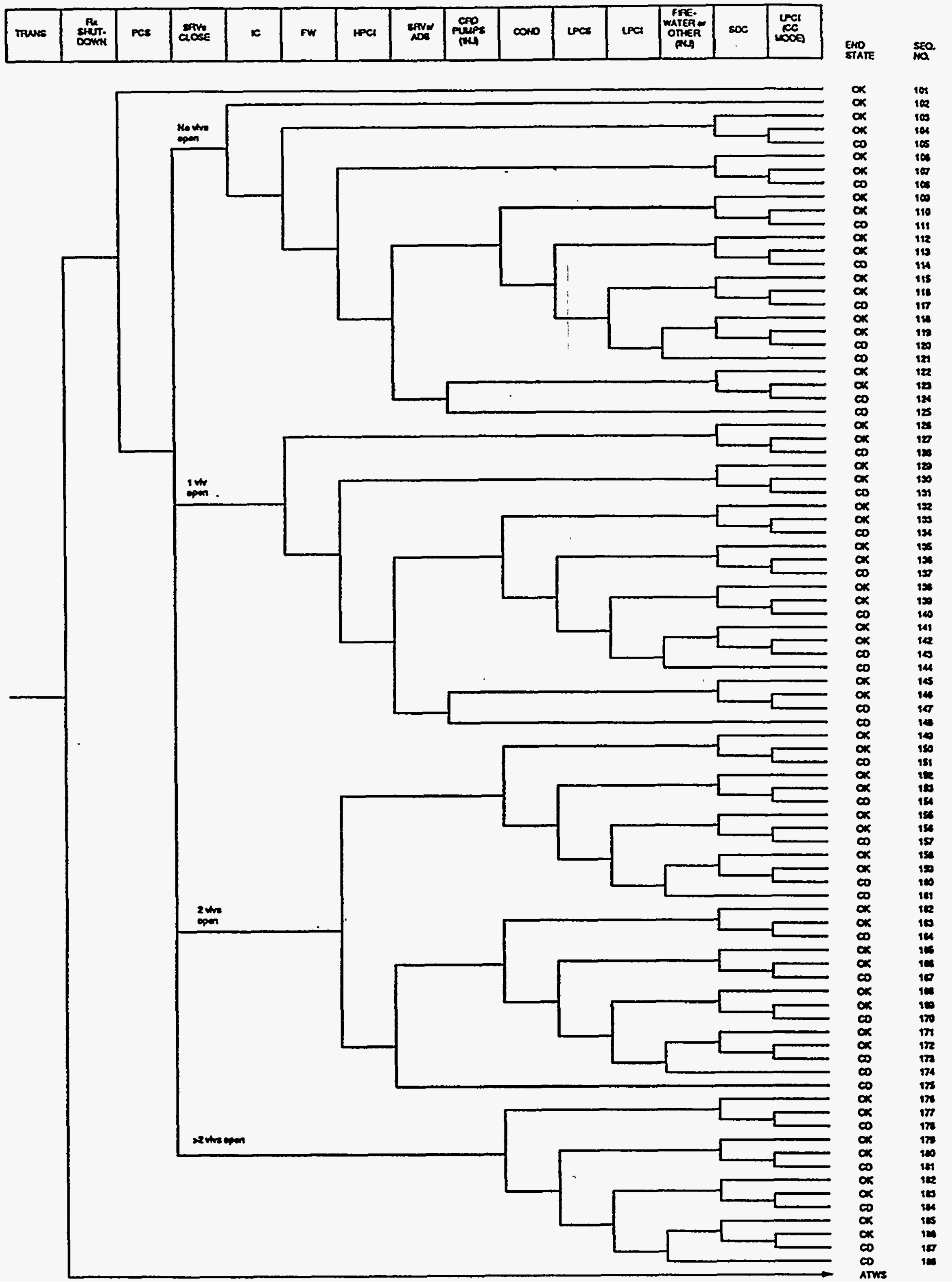

Fig. A.26 BWR Class B nonspecific reactor trip

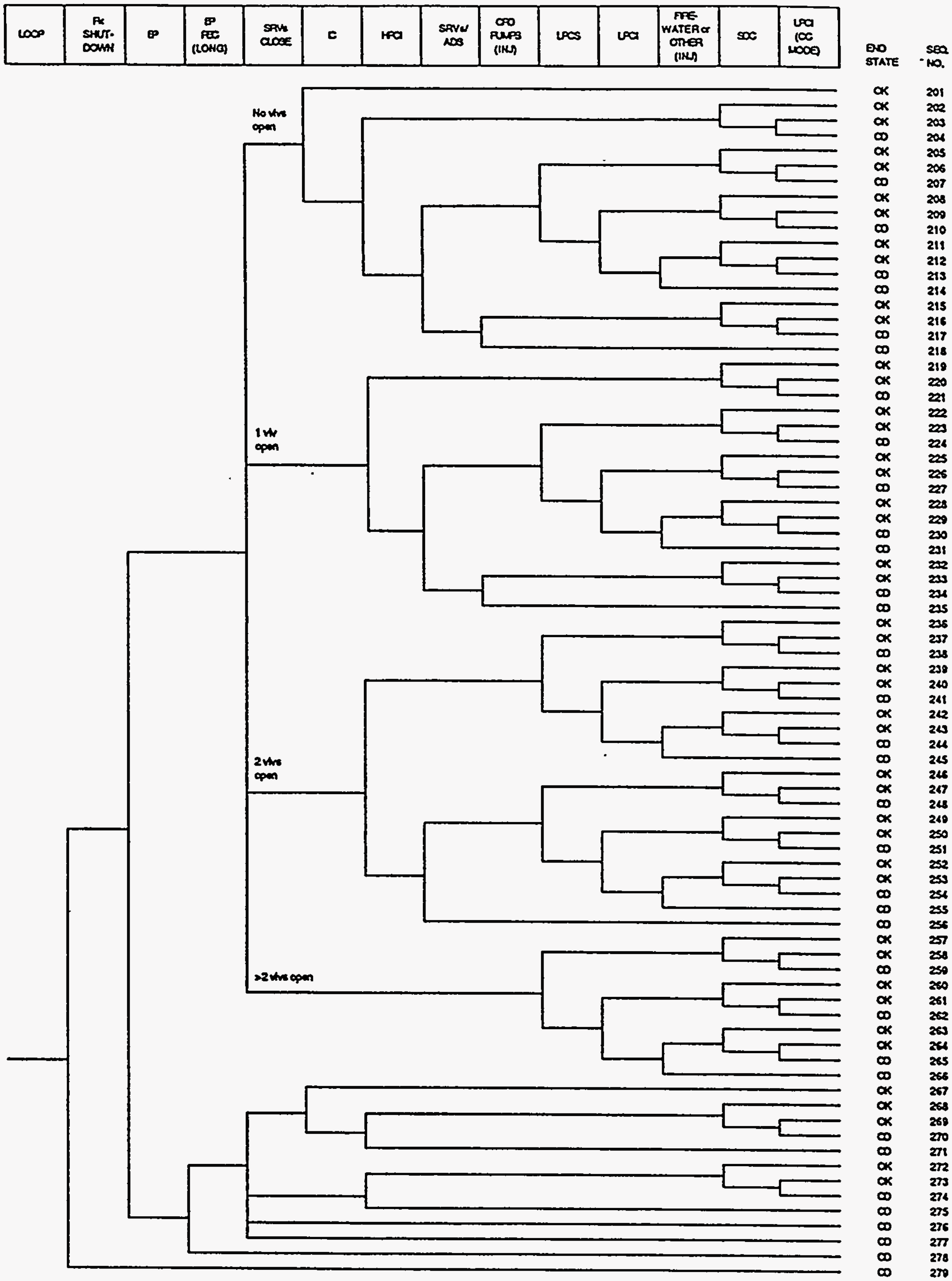

Fig. A.27 BWR Class B loss of offsite power

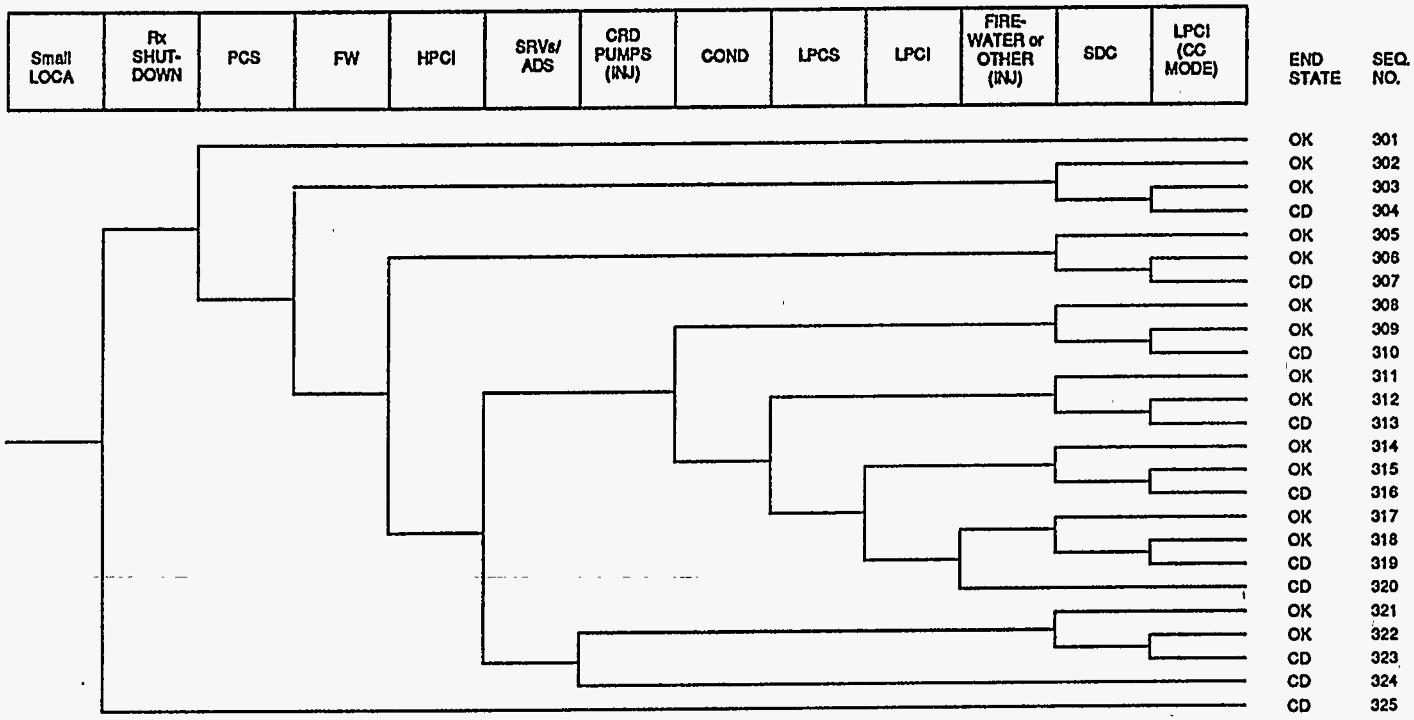

Fig. A.28 BWR Class B small-break loss-of-coolant accident

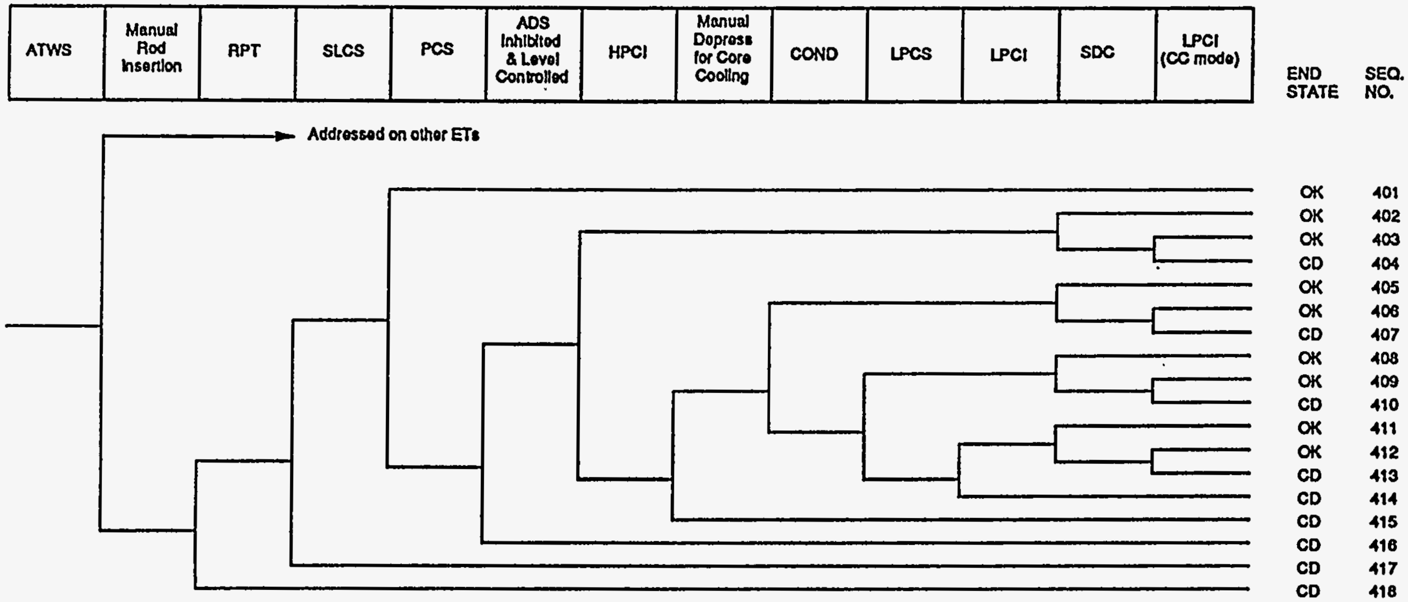

Fig. A.29 BWR Class B anticipated transient without scram

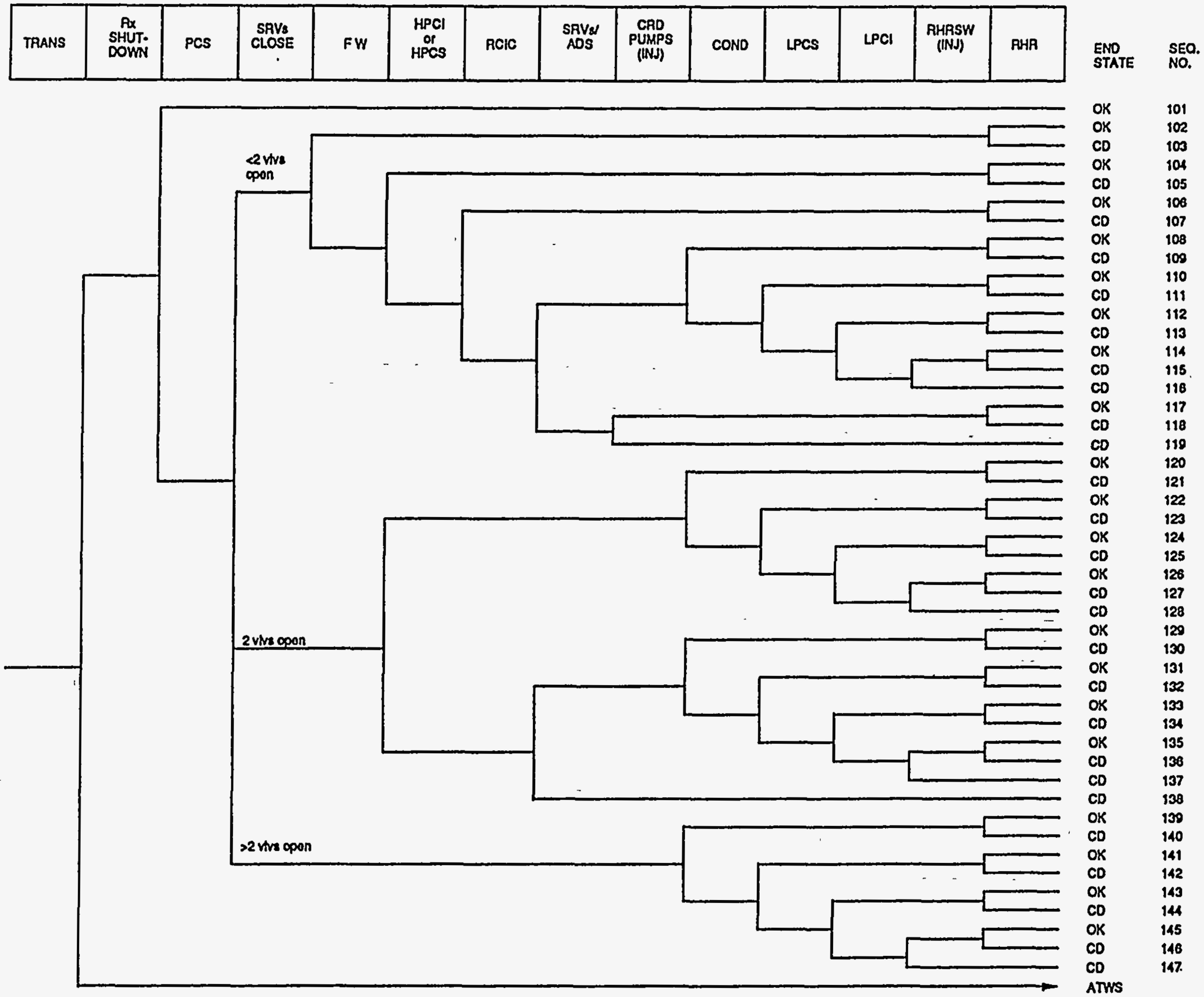

Fig. A.30 BWR Class C nonspecific reactor trip

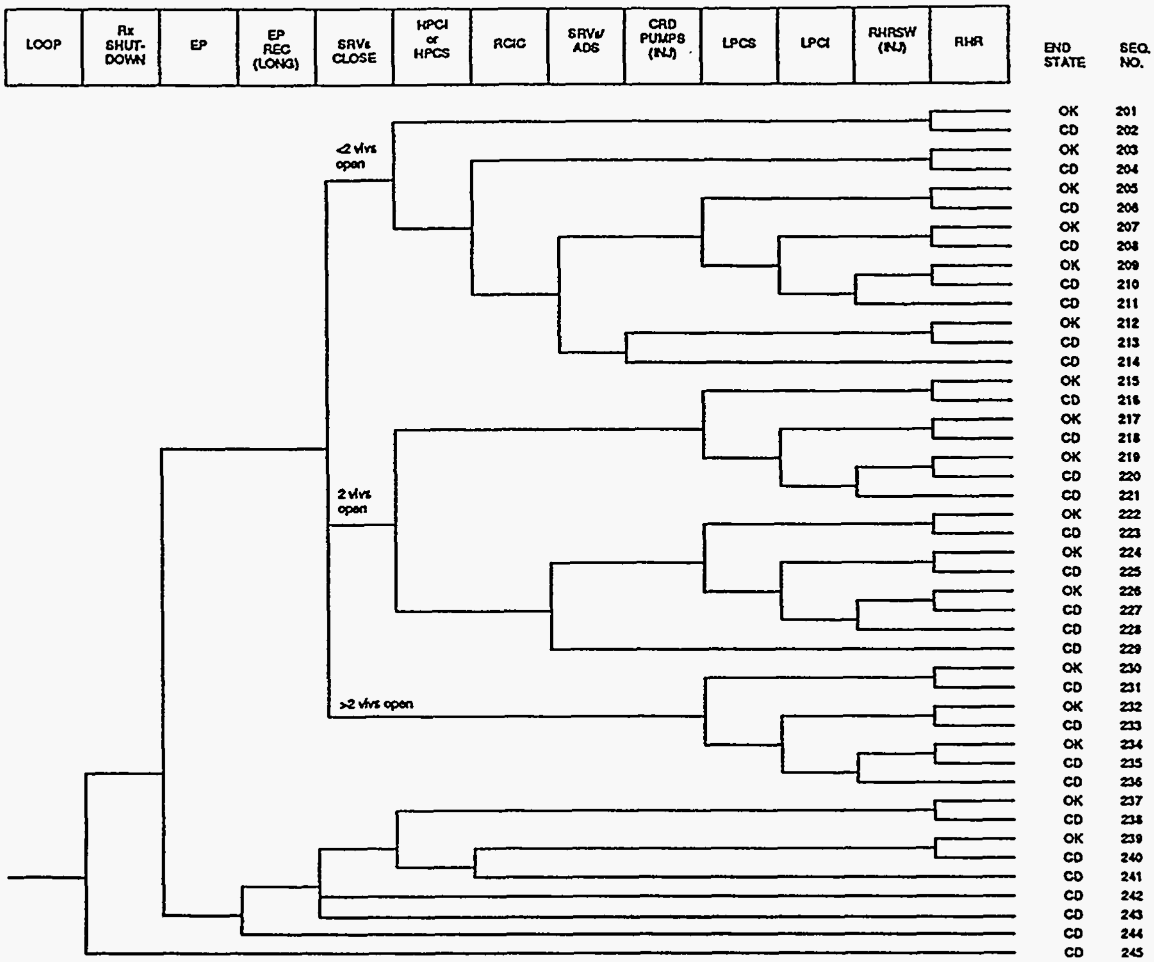

Fig. A.31 BWR Class C loss of offsite power

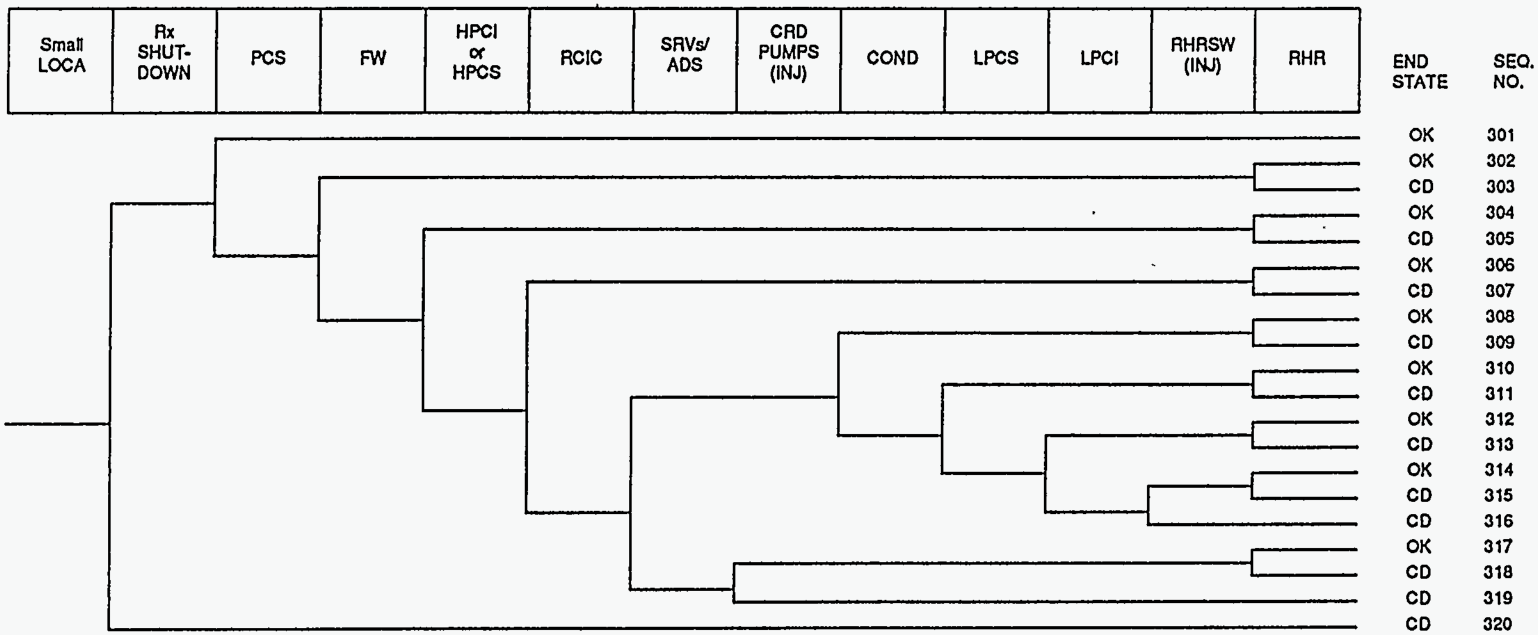

Fig. A.32 BWR Class C small-break loss-of-coolant accident

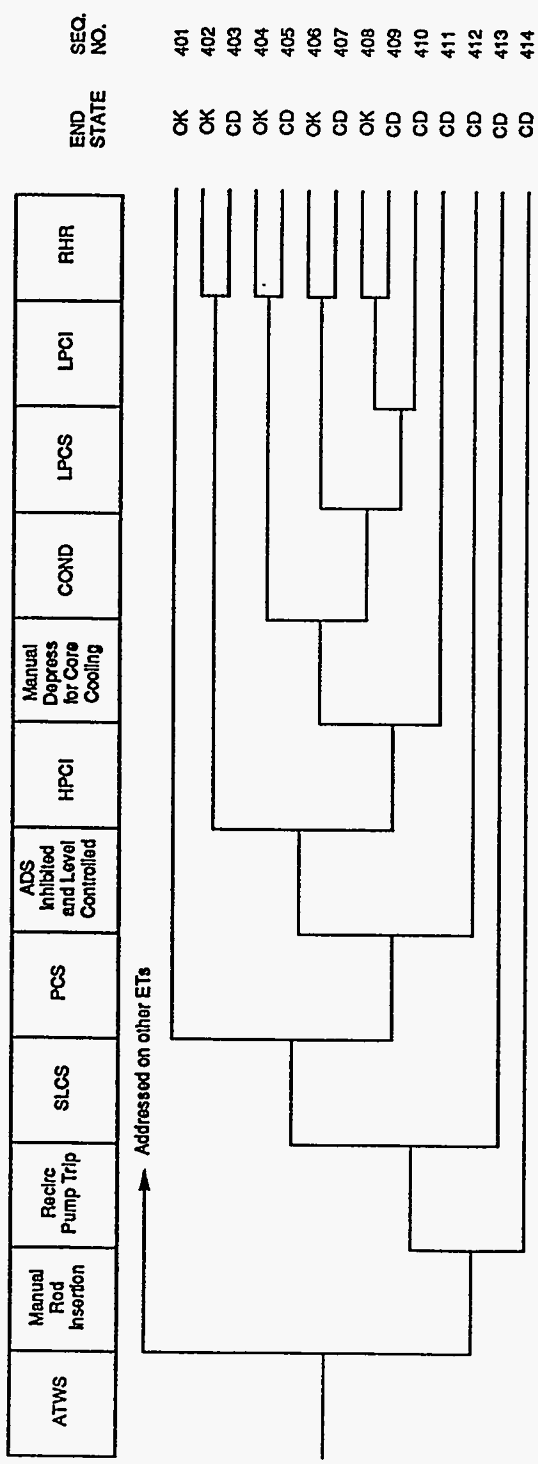

Fig. A.33 BWR Class C anticipated transient without scram

Appendix B:
Precursors with Conditional Core Damage Probabilities $\geq 1.0 \times 10^{-5}$

B.0 Precursors with Conditional Core Damage Probabilities (CCDPs) ≥ 1.0 x 10^{-5}

B.0.1 Accident Sequence Precursor Program Event Analyses for 1982-83

This appendix documents 1982 and 1983 operational events selected as accident sequence precursors with CCDPs ≥ 1.0 x 10^{-5}.

Licensee Event Reports (LERs) describing operational events at commercial nuclear power plants were reviewed for potential precursors if

- the LER was identified as requiring review based on a computerized search of the sequence Coding and Search System data base maintained at Oak Ridge National Laboratory, or
- the LER was identified as requiring review by the NRC Office for Analysis and Evaluation of Operational Data, or
- the LER was discussed in NUREG-0900 (*Report to Congress on Abnormal Occurrences*) or in issues of *Nuclear Safety* and appeared to be a potential precursor.

B.0.2 Precursors Identified

Fifty-three precursors with CCDPs ≥ 1.0 x 10^{-5} were identified from the 1982-1983 LERs reviewed. Events in this group were identified as precursors if they met one of the following precursor selection criteria and the conditional core damage probability (CCDP) estimated for the event was at least 1.0 x 10^{-5}:

- the event involved the total failure of a system required to mitigate effects of a core damage initiator.
- the event involved the degradation of two or more systems required to mitigate effects of a core damage initiator,
- the event involved a core damage initiator such as a loss of offsite power or small-break loss-of-coolant accident, or
- the event involved a reactor trip or loss of feedwater with a degraded safety system.

The 53 precursors identified are listed in Table B.1. Note that only 52 separate events are listed because in one instance two precursors are discussed together. In this case, an event associated with one unit at Quad Cities had an impact on the second unit (see page B.5-1).

Table B.1 List of ASP Events with CCDPs ≥ 1.0 x 10^{-5} [a]

Event Identifier	Plant	Description	Page
244/82-003 244/82-005	Ginna	Steam Generator Tube Rupture with One PORV Failed Open	B.1-1

Precursors

Event Identifier	Plant	Description	Page
247/82-019 247/82-020	Indian Point 2	Trip with Two AFW Pumps Inoperable	B.2-1
250/83-007	Turkey Point 3	Three AFW Pumps Unavailable	B.3-1
254/82-007 254/82-009	Quad Cities 1	Trip with RHRSW Train B Inoperable	B.4-1
254/82-012 254/82-013 254/82-018	Quad Cities 1 & 2 (Two separate precursors covered)	Postulated LOOP with Two EDGs Inoperable (Unit 1) and a Plant-Centered LOOP with One EDG Inoperable (Unit 2)	B.5-1
259/83-006 259/83-007	Browns Ferry 1	Scram, MSRV and its Vacuum Breaker Fail Open	B.6-1
260/83-074	Browns Ferry 2	Trip with HPCI Inoperable	B.7-1
261/83-004 261/83-005 261/83-007 261/83-016	Robinson 2	Trip with One AFW Pump Inoperable and One PORV Inoperable	B.8-1
272/83-011 272/83-012 272/83-013	Salem 1	Trip with Automatic Reactor Trip Capability Failed and Possible Positive MTC	B.9-1
272/83-033 272/83-034	Salem 1	LOOP with Turbine-Driven AFW Pump Inoperable	B.10-1
278/83-002 278/83-003	Peach Bottom 3	Trip with HPCI and ESF Bus 23 Inoperable	B.11-1
278/83-009	Peach Bottom 3	Two EDGs Inoperable	B.12-1
281/83-055	Surry 2	Trip with AFW Pump Inoperable	B.13-1
287/83-011	Oconee 3	Trip with Loss of Main Feedwater and One AFW Pump Inoperable	B.14-1
293/82-023 293/82-024	Pilgrim	Trip with HPCI Failed: Controller Failure Resulting from Spray	B.15-1
293/83-007	Pilgrim	LOOP During Shutdown	B.16-1
302/82-007	Crystal River 3	Unavailability of Two EDGs	B.17-1
304/82-009	Zion 2	Unavailability of Both Motor-Driven AFW Pumps	B.18-1
304/83-007	Zion 2	Postulated Grid/Weather-Related LOOP with 2 EDGs Unavailable	B.19-1

Event Identifier	Plant	Description	Page
309/83-002	Maine Yankee	Trip with MFW Inoperable and One Isolated Steam Generator	B.20-1
311/83-001 311/82-072	Salem 2	Trip with One Automatic Trip Breaker Failing to Open	B.21-1
316/82-113	Cook 2	Long-Term Unavailability of SI Train B	B.22-1
321/83-090 321/83-093	Hatch 1	Manual Scram with HPCI and RCIC Unavailable	B.23-1
324/82-005	Brunswick 2	Scram with Both RHRSW Loops Inoperable	B.24-1
324/82-029	Brunswick 2	Scram with RHRSW System Degradations	B.25-1
324/82-123	Brunswick 2	Scram with Emergency Bus E-3 Deenergized	B.26-1
325/82-025	Brunswick 1	Scram with RCIC Inoperable	B.27-1
325/82-041	Brunswick 1	Both RHRSW Loops Simultaneously Inoperable	B.28-1
325/82-054	Brunswick 1	Scram with RCIC Inoperable	B.29-1
327/82-048 327/82-050	Sequoyah 1	One Motor-Driven AFW Pump and One EDG Inoperable	B.30-1
327/83-063	Sequoyah 1	One of Two PORVs Failed to Open During Attempt to Reseat and Later Leaked after Maintenance	B.31-1
327/83-183 327/83-186	Sequoyah 1	One EDG and Turbine-Driven AFW Pump Inoperable	B.32-1
331/83-017 331/83-018	Duane Arnold	HPCI and RHRSW Train B Inoperable	B.33-1
335/82-040	St. Lucie 1	Trip with Loss of Grid Synchronization Due to Shorted Generator Relay	B.34-1
339/82-009	North Anna 2	PORVs Inoperable Due to Low Nitrogen Pressure	B.35-1
344/83-002	Trojan	Trip with MFW and Two AFW Pumps Unavailable	B.36-1
344/83-012	Trojan	AFW Pump Trip Following Reactor Trip	B.37-1

Precursors

Event Identifier	Plant	Description	Page
346/83-038 346/83-040	Davis-Besse 1	Trip with AFW Pump Inoperable and Failure of Two SFRCS Channels	B.38-1
361/83-063	San Onofre 2	Trip with Motor-Driven AFW Pump Inoperable	B.39-1
362/83-099	San Onofre 3	Trip with Turbine-Driven AFW Pump Inoperable	B.40-1
366/82-081	Hatch 2	Scram, Isolation, RCIC Failure, SRV Tailpipe Vacuum Relief Failed	B.41-1
366/82-084 366/82-085	Hatch 2	RHRSW Pumps A, C, and D Failed	B.42-1
366/83-042 366/83-055 366/83-056	Hatch 2	Reactor Trip with RCIC and RHR Loop A Unavailable	B.43-1
366/83-084	Hatch 2	Scram with RHR Loop A Unavailable	B.44-1
368/83-007 368/83-011 368/83-012	ANO 2	Trip with One Train of EFW Inoperable	B.45-1
373/82-093 373/82-094	LaSalle 1	Scram and Multiple Failures	B.46-1
373/83-057	LaSalle 1	Scram, LOFW with RCIC Inoperable	B.47-1
373/83-117 373/83-147	LaSalle 1	B RHR Heat Exchanger Outlet Valve Failure	B.48-1
387/82-061	Susquehanna 1	Trip with ESW Pumps B and D Failed	B.49-1
387/83-051	Susquehanna 1	Trip with RCIC System Unavailable Owing to Governor Valve Problem	B.50-1
387/83-103	Susquehanna 1	Trip with RCIC System Unavailable Owing to Governor Valve Problem	B.51-1
387/83-120	Susquehanna 1	Trip with RCIC System Unavailable Owing to Governor Valve Problem	B.52-1

[a]Acronyms used in table are defined as follows: low-pressure coolant injection (LPCI), power-operated relief valve (PORV), auxiliary feedwater (AFW), residual heat removal service water (RHRSW), loss of offsite power (LOOP), emergency diesel generator (EDG), main steam relief valve (MSRV), high-pressure safety injection (HPCI), moderator temperature coefficient (MTC), engineered safety feature (ESF), main feedwater (MFW), safety relief valve (SRV), emergency feedwater (EFW), loss of feedwater (LOFW), reactor core isolation cooling (RCIC), steam and feedwater rupture control system logic channels (SFRCS), emergency service water (ESW)

B.0.3 Event Documentation

This appendix provides documentation for 53 precursor events (two precursors are covered under event B.5) with CCDPs $\geq 1.0 \times 10^{-5}$. Analysis documentation and precursor calculation sheets for each precursor are attached. The precursors are in docket/LER number order. For each precursor in this appendix, an event analysis sheet is included. This provides a description of the operational event, event-related plant design information, the assumptions and approach used to model the event, and analysis results. For additional information on interpreting the computer output from the analyses, see *Event Evaluation Computer Code: Description and User Manual*, ORNL/NRC/LTR-87/2/R1, October 1992.

Precursors

B.1 LER No. 244/82-003 and -005

Event Description: Steam Generator Tube Rupture with One PORV Failed Open

Date of Event: January 25, 1982

Plant: Ginna

B.1.1 Summary

On January 25, 1982, while operating at 100% power, the Ginna B steam generator experienced a tube rupture. The resulting plant transient included significant primary system depressurization, actuation of the safety injection system and minor releases of radioactive materials from the plant. During the transient, a pressurizer power-operated relief valve (PORV) failed to close after being used to reduce primary and secondary pressure below the steam generator safety valve setting. The estimated conditional core damage probability for this event is 3.8×10^{-4}.

B.1.2 Event Description

On January 25, 1982, at 0925, while the plant was operating at 100% power, a tube rupture occurred in steam generator B. Multiple control room alarms alerted the operators to a reactor coolant system (RCS) rapid depressurization. The air ejector radiation monitor alarm indicated that the rupture was likely in steam generator B. The continuing pressure drop resulted in an automatic reactor trip and an automatic safety injection actuation. All three high-pressure injection (HPI) pumps started. The safety injection actuation resulted in an automatic containment isolation and trip of the operating charging pumps. All safety systems functioned properly. Both reactor coolant pumps were manually stopped, and natural circulation cooling in the RCS was verified. The pressurizer emptied, and the RCS depressurization reached a minimum of 1200 psig. A small steam bubble formed during natural circulation in the upper head, but was collapsed when safety injection flow refilled the RCS.

Initially, operators cooled down the reactor by steaming both steam generators to the main condenser. The B steam generator was isolated at 0940, and natural circulation cooling in loop B was terminated. The B steam generator water level continued to rise in spite of the termination of feedwater flow to the steam generator, due to flow through the ruptured tube. At 0955, the narrow-range water level indicator on B steam generator went off scale high, and subsequently the B main steam line started to fill.

At 0957, the safety injection actuation circuitry was reset, thus resetting the containment isolation system. Instrument air, and thus control of the air-operated valves inside containment, was restored.

At 1007, operators attempted to equalize the pressure differential between the RCS and the secondary side of the B steam generator to stop flow through the tube rupture. A pressurizer PORV was cycled three times before it stuck open. The operator attempted to close the valve, but the valve would not close. The operator then closed the block valve to prevent further RCS water loss. Steam bubbles in the reactor vessel upper head and in the top of the B steam generator tubes occurred as well. The growth of the bubbles and increased safety injection flow resulted in the rapid filling of the pressurizer. Loop A natural circulation was not affected by the steam bubbles.

One of the B steam generator safety valves cycled three times as a result of the overpressurization caused by continued flow through the ruptured tube (apparently caused by the inappropriate isolation of the B steam generator atmospheric dump valves). At 1038, safety injection was terminated to prevent further water discharge through the safety valve. At 1040, the condensate system was secured to prevent further radioactive contamination of the condensate storage tanks and demineralizers. The operators used the steam generator (SG) PORV to relieve steam from the A steam generator.

At 1042, the pressurizer heaters were reenergized (after having tripped at 0928 on low pressurizer level) to re-establish a steam bubble in the pressurizer. At 1052, the rupture disk on the pressurizer relief tank burst due to the addition of water from the letdown line relief valve, the pressurizer PORV, and the relief valve for the reactor coolant pump (RCP) seal return line.

At 1107, one safety injection pump was started in anticipation of an RCS pressure drop due to the restart of the A RCP. At 1119, the B steam generator safety valve lifted and closed. At this time, the B steam line had flooded sufficiently to cause water, rather than steam, to be released. At 1121, the A RCP was started. The RCP flow cooled and collapsed any remaining steam bubbles in the reactor upper vessel head and the B steam generator. This addition of flow led to another cycle of the B steam generator safety valve. Safety injection was stopped, but the valve continued to leak water at approximately 100 gpm.

At 1152, the pressurizer level returned on scale, and a steam bubble was re-established. At 1202, normal letdown from the RCS to the chemical and volume control system was re-established. Due to the B steam generator safety valve leak, the RCS continued to leak through the tube rupture in steam generator B. Operators re-started one safety injection pump at 1212 in response to the continued decrease in pressurizer level. The pump was intermittently operated until 1235. The safety relief valve on steam generator B stopped leaking at approximately 1225.

At 1227, the RCS and B steam generator pressures equalized. RCS pressure was maintained at 25 psia below steam generator B pressure. At 1840, B steam generator water level returned on scale. Secondary side feed and bleed was then used to cool steam generator B.

At 0700, on January 26, 1982, the residual heat removal system was placed in operation, and the plant was declared to be in cold shutdown.

B.1.3 Additional Event-Related Information

The ruptured B steam generator tube was located at row 42, column 55 on the hot-leg side of the steam generator. The rupture was approximately 4 inches long and 0.7 inch wide at its center. The rupture was fish-mouth shaped, and pointed outward along the tube column. The tube appeared ballooned at the rupture location, and had a wall thickness of less than 5% of the nominal thickness. Markings on the exterior of the tube had the appearance of fretting wear. Damage to sixteen additional tubes that had been plugged in steam generator B was identified. Foreign objects and tube fragments were found in the steam generator. An examination of steam generator A revealed the existence of some small foreign objects as well. The most probable cause of damage to steam generator B was a piece of metal that was left inside during a 1975 repair when a large ring was removed to increase the efficiency of the recirculation flow. The ring was cut into pieces to be removed, but one piece was left inside.

Additional information on this event is included in the *Report to Congress on Abnormal Occurrences, January - March 1982,* NUREG-0090, Vol. 5, No. 1.

B.1.4 Modeling Assumptions

This event was modeled as a steam generator tube rupture initiating event. Since a second pressurizer PORV was available and the leaking valve could have been used for depressurization during this event by opening its block valve, the model was not revised to reflect the stuck-open pressurizer PORV. The potential for the steam generator relief valve that leaked during the event to have stuck open was addressed by revising the failure probability for SG.ISO.AND.RCS.COOLDOWN to 0.1, a typical failure probability for a relief valve, once it has passed water.

B.1.5 Analysis Results

The estimated conditional core damage probability for this event is 3.8×10^{-4}. The dominant sequence, highlighted on the event tree in Figure B.1.1, involved the successful operation of auxiliary feedwater and the failure of high-pressure injection.

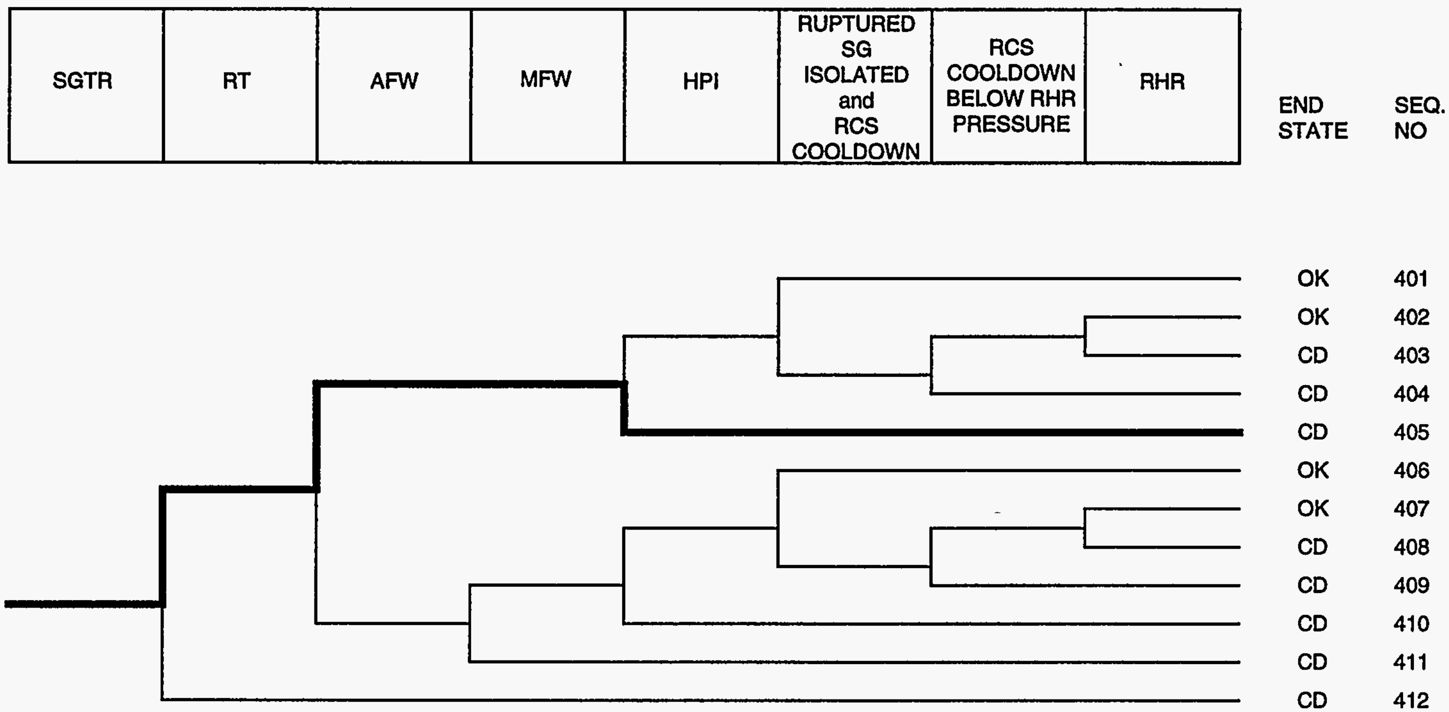

Figure B.1.1 Dominant core damage sequence for LER 244/82-003 and -005

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 244/82-003 and -005
Event Description: Steam Generator Tube Rupture
Event Date: January 25. 1982
Plant: Ginna

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

SGTR	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
SGTR	3.8E-04
Total	3.8E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
405	sgtr -rt -afw hpi	CD	2.7E-04	8.9E-01
404	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN rcs.cool.below.rhr	CD	6.0E-05	1.0E-01
412	sgtr rt	CD	2.8E-05	1.0E-01
403	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN -rcs.cool.below.rhr rhr	CD	2.2E-05	5.7E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
403	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN -rcs.cool.below.rhr rhr	CD	2.2E-05	5.7E-03
404	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN rcs.cool.below.rhr	CD	6.0E-05	1.0E-01
405	sgtr -rt -afw hpi	CD	2.7E-04	8.9E-01
412	sgtr rt	CD	2.8E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\pwrb8283.cmp
BRANCH MODEL: c:\asp\1982-83\ginna.82
PROBABILITY FILE: c:\asp\1982-83\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail

trans	2.6E-04	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	1.6E-05	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	5.7E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.3E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.1E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	9.9E-02	1.0E+00	
offsite.pwr.rec/seal.loca	6.0E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	8.2E-03	1.0E+00	
SG.ISO.AND.RCS.COOLDOWN	1.0E-02 > 1.0E-01	1.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.0E-02 > 1.0E-01		
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.2 LER No. 247/82-019 and -020

Event Description: Transient with Two AFW Pumps Inoperable

Date of Event: May 17, 1982

Plant: Indian Point 2

B.2.1 Summary

During hot shutdown following a trip on May 17, 1982, No. 23 auxiliary feedwater (AFW) motor-driven pump failed while in operation, due to a damaged thrust bearing. Two days later, also during hot shutdown, No. 22 turbine-driven AFW pump failed while in operation, due to a governor failure. When the turbine-driven pump failed, AFW pump No. 23 was out of service, due to repairs on the damaged thrust bearing. Only one AFW pump was available, so the plant commenced cooldown. The estimated conditional core damage probability of this event is 1.2×10^{-4}.

B.2.2 Event Description

During hot shutdown following a reactor trip on May 17, 1982, No. 23 AFW pump failed while in operation. The pump failure was due to a damaged thrust bearing. The bearing damage was caused by the oil pressure equalizing line check valve hanging up, thus negating positioning control of the balancing drum. The pump was taken out of service to replace the bearing and check valve internals. Two days later, during hot shutdown, No. 22 turbine-driven AFW pump failed while in operation. AFW pump 22 tripped due to erratic speed control by the governor on its steam turbine drive. The governor bearing was observed smoking. The governor and governor valve were rebuilt. At the time the turbine-driven pump was determined inoperable, No. 23 AFW motor-driven pump was still out of service. Since only motor-driven AFW pump 21 was operable, the plant commenced cooldown.

The reactor trip which occurred on May 17, 1982 was initiated by feedwater system perturbations (NUREG-0020).

B.2.3 Additional Event-Related Information

Indian Point 2 has three AFW pumps. Two (21 and 23) are motor-driven pumps and one (22) is a turbine-driven pump. Each motor-driven pump feeds two steam generators. Bearing design on the 21 AFW pump is identical to the 23 AFW pump which failed.

The single turbine-driven pump can feed all four steam generators. In the event of a loss of offsite power and a loss of emergency power, the turbine-driven pump can supply all steam generators. Because Indian Point was concerned about a potential common-cause failure of the 21 pump, the bearing oil pressure equalizing line check valve internals was removed.

B.2.4 Modeling Assumptions

Since the AFW pump failures occurred following the trip, the event was modeled as a transient with degraded AFW. The turbine-driven pump failure, although observed two days after the trip, was assumed to be present at the time of the trip (i.e., it was assumed that the turbine-driven pump had not been demanded at the time of the trip).

For sequences involving AFW, given a successful reactor trip, one of three AFW pumps provides success. Based on the nature of the AFW pump 23 and pump 22 failures, the conditional failure probability of the remaining AFW pump was assumed to be 0.1. For sequences involving AFW given a trip failure anticipated transient without scram (ATWS), two of three pumps are assumed necessary for success. Since two of the three pumps were inoperable, AFW was assumed failed for ATWS sequences. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common-cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event.

Since the reactor trip was initiated due to feedwater system perturbations, but there is no information indicating that main feedwater (MFW) was inoperable, MFW was assumed operable at the time of the event.

B.2.5 Analysis Results

The estimated conditional core damage probability for this event is 1.2×10^{-4}. The dominant core damage sequence, shown on the event tree in Figure B.2.1, involves a successful reactor trip, failure of AFW, failure of MFW, and failure of feed and bleed.

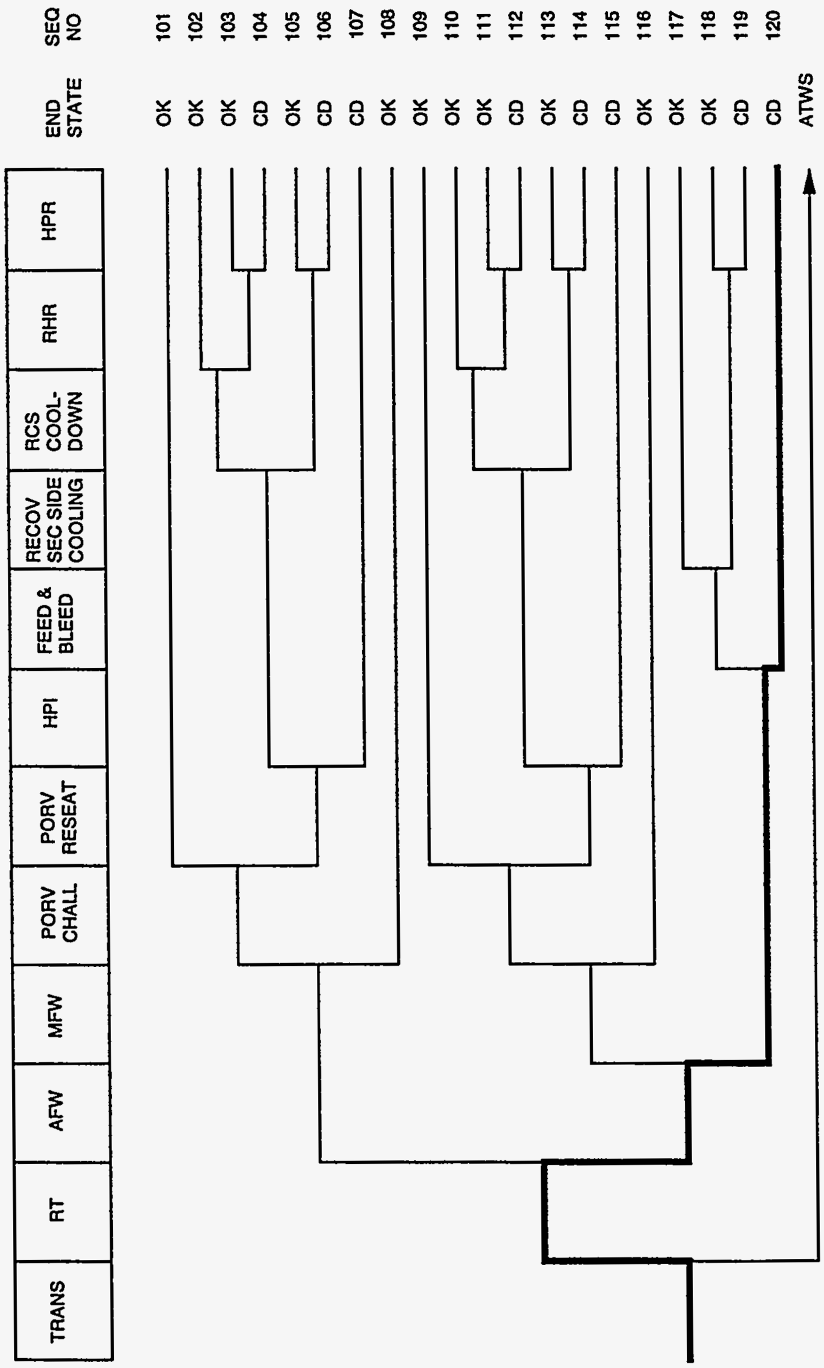

Figure B.2.1 Dominant core damage sequence for LER 247/82-019 and -020

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 247/82-019 and -020
Event Description: Transient with two AFW pumps inoperable
Event Date: May 17. 1982
Plant: Indian Point 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.2E-04
Total	1.2E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW mfw feed.bleed	CD	9.4E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01
119	trans -rt AFW mfw -feed.bleed recov.sec.cool hpr	CD	3.0E-06	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW mfw -feed.bleed recov.sec.cool hpr	CD	3.0E-06	1.5E-01
120	trans -rt AFW mfw feed.bleed	CD	9.4E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\ipoint2.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail

trans	1.3E-03	1.0E+00	
loop	3.1E-05	1.7E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 1.0E-01	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
AFW/ATWS	4.3E-03 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > Failed		
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	5.4E-04	8.9E-01	
seal.loca	2.1E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	1.9E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	8.0E-02	1.0E+00	
offsite.pwr.rec/seal.loca	6.0E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	5.6E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.3 LER No. 250/83-007

Event Description: Three Auxiliary Feedwater Pumps Unavailable

Date of Event: April 14, 1983 through April 19, 1983

Plant: Turkey Point 3

B.3.1 Summary

Manual valves in the steam supply lines to the B and C auxiliary feedwater (AFW) pump turbines were found to be closed on April 19, 1983. Since AFW pump A was out of service at the time, all of the AFW pumps were unavailable. The increase in core damage probability, or importance, over the duration of the event is 5.5×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 1.5×10^{-6}, resulting in an estimated conditional core damage probability (CCDP) of 5.6×10^{-5}.

B.3.2 Event Description

On April 19, 1983 with the unit at full power, manual valves 3-084A and 3-086B on the steam supply lines to the B and C auxiliary feedwater pump turbines, respectively, were found to be closed. This rendered the B and C pumps inoperable. Since AFW pump A was out of service at the time, all of the AFW pumps were unavailable. The cause of the event was determined to be human error in tagging the valves and lack of independent verification of the tag locations and valve positions. The two manual valves were immediately locked open and AFW pumps B and C were returned to service within an hour.

B.3.3 Additional Event-Related Information

Because of modifications to the A AFW pump and the common redundant steam supply piping, manual valves 001B, 002B, 001C, and 002C in the AFW pump turbine steam supply lines were closed on March 26, 1983. Valves 001B and 002B are in series in the steam supply line from steam generator 3A to the AFW pump B turbine. Valves 001C and 002C supply the AFW pump C turbine. On April 11, 1983, valves 001B and 001C were opened to perform a hydro test, and they were supposed to be reclosed on April 14. On April 19, valve 001B was found closed, but with its clearance tag removed. The tag was found on valve 3-084A, which was also closed. Valve 001C was found open and its tag was located on valve 3-086B, which was closed. Thus, steam supplies were isolated to all AFW pump turbines.

The plant has a standby steam generator feedwater (SSGFW) system consisting of two 100% capacity motor-driven pumps. This system is shared with Turkey Point 4. Although the SSGFW system is not safety related, it is powered from multiple onsite and offsite sources.

B.3.4 Modeling Assumptions

It is assumed that valves 3-084A and 3-086B were mistakenly closed on April 14, 1983. With valves 001B, 002B, 001C, and 002C closed continuously since April 14, and Unit 4 in refueling, there was no steam available for AFW pumps B and C for at least five days or 120 hours. AFW pump A was out of service during this period so all three AFW pumps were unavailable. Therefore, all three trains of AFW were modeled as unavailable for five days. The errors which caused the steam supply valves to be incorrectly closed would have to have been restored locally. This restoration would be complicated by the mislocated tags. To reflect this situation the AFW nonrecovery probability was increased to 0.55.

The failure probability for the SSGFW system was estimated as described in the analysis of LER 251/92-007 in *Precursors To Potential Severe Core Damage Accidents: 1992 A Status Report*, NUREG/CR-4674, ORNL/NOAC-232, Vol. 18. This system requires one of the two pumps to operate and realignment of one valve to be successful. An operator failure rate of 0.01 was assumed. Since the SSGFW system is placed into service prior to attempting feed and bleed, the operator failure rate for initiating feed-and-bleed was increased to 0.2, consistent with the Turkey Point probabilistic risk assessment (PRA). The SSGFW system failure probability, 0.011, was calculated as:

$$\begin{aligned}&(\text{PMPA} \times \text{PMPB}) + \text{VLV1} + \text{OPR}\\&= (0.01 \times 0.01) + 0.0004 + 0.01\\&= 0.011\end{aligned}$$

This value was incorporated into the model by modifying the nonrecovery probability of the main feedwater (MFW). Transient, loss-of-offsite power (LOOP), loss-of-coolant accident (LOCA), and steam generator tube rupture (SGTR) were used as potential initiators in the unavailability analysis.

B.3.5 Analysis Results

The increase in core damage probability over the duration of the event is 5.5×10^{-5}. The base-case CDP (not shown in calculation) is 1.5×10^{-6}, resulting in an estimated CCDP of 5.6×10^{-5}. The contributions of the postulated LOCA and SGTR initiators are negligible compared to those due to a transient or LOOP. The dominant core damage sequence, shown in Figure B.3.1, involves a transient, successful reactor trip, failure of AFW, failure of main feedwater, and failure of feed and bleed.

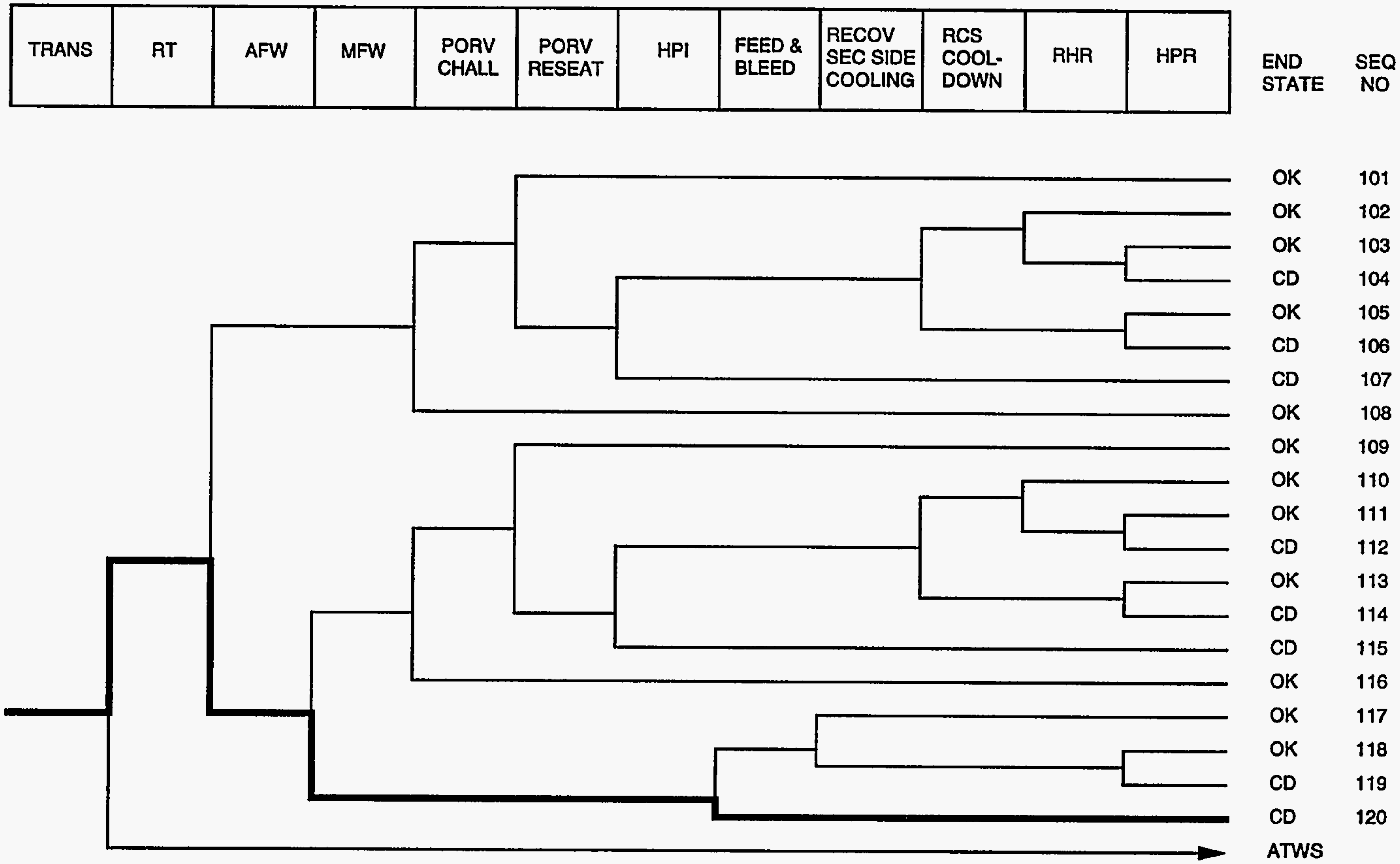

Figure B.3.1 Dominant core damage sequence for LER 250/83-007

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 250/83-007
Event Description: Three AFW pumps unavailable due to tagging error
Event Date: 4/14/83 - 4/19/83
Plant: Turkey Point 3

UNAVAILABILITY, DURATION= 120

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.1E-01
LOOP	1.4E-03
LOCA	1.6E-04
SGTR	2.0E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.0E-05
LOOP	2.5E-05
LOCA	3.9E-08
SGTR	2.2E-07
Total	5.5E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW mfw feed.bleed	CD	2.7E-05	6.0E-03
215	loop -rt(loop) -ep AFW -offsite.pwr.rec/-ep.and.afw feed.bleed /loop	CD	2.1E-05	9.3E-02
508	trans rt -prim.press.limited AFW/ATWS	CD	3.0E-06	1.0E-01
239	loop -rt(loop) ep AFW/EP	CD	1.9E-06	8.3E-02
219	loop -rt(loop) -ep AFW offsite.pwr.rec/-ep.and.afw feed.bleed /loop	CD	1.6E-06	9.3E-02
214	loop -rt(loop) -ep AFW -offsite.pwr.rec/-ep.and.afw -feed.bleed /loop recov.sec.cool hpr	CD	6.7E-07	9.3E-02
226	loop -rt(loop) ep -AFW/EP porv.chall/sbo -porv.reseat/ep seal .loca offsite.pwr.rec/seal.loca	CD	(3.0E-07)	6.8E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW mfw feed.bleed	CD	2.7E-05	6.0E-03
508	trans rt -prim.press.limited AFW/ATWS	CD	3.0E-06	1.0E-01
214	loop -rt(loop) -ep AFW -offsite.pwr.rec/-ep.and.afw -feed.bleed /loop recov.sec.cool hpr	CD	6.7E-07	9.3E-02
215	loop -rt(loop) -ep AFW -offsite.pwr.rec/-ep.and.afw feed.bleed /loop	CD	2.1E-05	9.3E-02
219	loop -rt(loop) -ep AFW offsite.pwr.rec/-ep.and.afw feed.bleed	CD	1.6E-06	9.3E-02

```
     /loop
226  loop -rt(loop) ep -AFW/EP  porv.chall/sbo -porv.reseat/ep  seal  CD      ( 3.0E-07 )  6.8E-02
     .loca  offsite.pwr.rec/seal.loca
239  loop -rt(loop) ep  AFW/EP                                   CD        1.9E-06    8.3E-02
** non-recovery credit for edited case
```

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\asp\models\tpoint82.cmp
BRANCH MODEL: c:\asp\models\tpoint3.82
PROBABILITY FILE: c:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	9.5E-04	1.0E+00	
loop	6.7E-05	1.7E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	1.5E-03 > 1.0E+00	4.5E-01 > 5.5E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
AFW/ATWS	1.2E-02 > 1.0E+00	1.0E+00	
Branch Model: 2.OF.3			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
AFW/EP	1.5E-03 > 1.0E+00	4.5E-01 > 5.5E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
mfw	1.9E-01	1.1E-02	
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	7.5E-04	8.9E-01	
feed.bleed	2.1E-02	1.0E+00	2.0E-01
feed.bleed/loop	2.1E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.6E-01	1.0E+00	

offsite.pwr.rec/-ep.and.-afw	2.4E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	7.1E-02	1.0E+00	
offsite.pwr.rec/seal.loca	6.2E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.6E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.4 LER No. 254/82-007, -009

Event Description: Transient with RHRSW Train B Inoperable

Date of Event: April 15, 1982

Plant: Quad Cities 1

B.4.1 Summary

During normal operation on April 15, 1982, residual heat removal service water (RHRSW) pump D outboard bearing was found to be failed due to excessive leakage of water from the adjacent packing to the oil in the bearing. On April 30, RHRSW pump C was taken out of service for maintenance on the pump seal packing. Water which leaked from adjacent seal packing was found in the bearing oil reservoir. Three plant trips had occurred around the time of the faults in the pumps (April 17, 19, and 30). The conditional core damage probability estimated for this event is 1.7×10^{-4}.

B.4.2 Event Description

During normal operation on April 15, 1982, RHRSW pump D outboard bearing was found to be failed during a surveillance test. Investigation revealed that the pump bearing failed due to excessive leakage of water from the adjacent packing to the oil in the bearing. The bearing and packing was replaced and the pump was returned to service on April 22. A few days later, on April 30, RHRSW pump C was taken out of service for maintenance on the pump seal packing. Water which leaked from adjacent seal packing was found in the bearing oil reservoir. The licensee stated that while there was insufficient water to cause bearing damage due to a loss of lubrication, continued operation could have possibly resulted in bearing damage. The pump was declared inoperable. The pump seals were repacked and the oil in the bearing oil reservoir was replaced. The pump was returned to service later that day.

Three plant trips occurred around the time of the discovery of the bearing faults in the pumps (April 17, 19, and 30). The plant trip on April 17 involved a reactor scram due to low condenser vacuum due to a condensate demineralizer valve failure. The plant trip on April 19 involved a reactor scram due to high main steam line flow. The plant trip on April 30 (Licensed Operating Reactors, Status Summary Report, NUREG-0020, published monthly, hereafter referred to as NUREG-0020) involved a trip on low reactor water level due to a B reactor feedpump discharge valve closure.

B.4.3 Additional Event-Related Information

The residual heat removal service water system provides cooling water to the residual heat removal (RHR) system heat exchangers. RHR is a two-train system (A and B) which provides three functions: suppression pool cooling, containment spray, and shutdown cooling. Each train has two RHR pumps and one heat exchanger. Suppression pool cooling is used to remove heat from the suppression pool whenever the water temperature exceeds 95°F. Containment spray is used in the event of a nuclear system break within the primary containment to prevent excessive containment pressure and temperature by condensing steam and cooling noncondensable gases. Shutdown cooling can be used during normal shutdown and cooldown to remove decay heat, once the reactor coolant temperature is low enough that the steam supply pressure is not sufficient to maintain turbine shaft gland seals or vacuum in the main condensers. RHR requires the use of one pump and one functioning heat exchanger (and thus one train of RHRSW) for suppression pool cooling, containment spray, and shutdown cooling. RHRSW is a two-train system (A and B). Each RHRSW train has two pumps and one heat exchanger. Pumps A and B supply heat exchanger A for RHR train A. Pumps C and D supply heat exchanger B for RHR train B. RHRSW also has a crosstie which enables the RHRSW pumps to provide coolant to the RHR system for use as an alternative injection system. Two RHRSW pumps supplying flow to one heat exchanger are sufficient for all RHR modes. One RHRSW pump is sufficient to provide the alternative injection source for RHR.

B.4.4 Modeling Assumptions

RHRSW (and thus RHR) were assumed to be degraded at the time of the trip on April 17, 1982. The event was modeled as a transient with PCS initially unavailable due to main steam isolation valve (MSIV) closure. The demineralizer valve failure was assumed to be recoverable on the same time scale as the MSIVs. Assuming that the water was present in the lube oil for both pumps C and D at the time of the transient, two of the four RHRSW pumps were assumed to fail during their mission time, and potential failure of the other two pumps from similar causes was assumed. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event. Since the ASP model assumes that common cause failure of the RHR pumps dominates the failure of RHR and does not directly account for the failure of RHRSW pumps leading to RHR failure, the RHR failure probability was modified to reflect the degraded state of RHRSW in this event. The conditional train probabilities for RHRSW pumps shown in Table 1 were combined and added to the probability of RHR failure as follows

$$P(RHRSW) = P(A|DC)*P(B|ADC)$$

$$P(RHR)_{NEW} = P(RHR)_{OLD} + P(RHRSW)$$

$$P(RHR)_{NEW} = P(RHR)_{OLD} + 0.15.$$

Table 1. RHRSW Pump Train Failure to Start and Run Conditional Failure Probabilities

Train	Conditional Failure Probability
P(1)	0.01
P(2|1)	0.1
P(3|12)	0.3
P(4|123)	0.5

The suppression pool cooling mode of RHR would also be affected in the same manner. Thus, P(RHRSW) was added to the branch probability for RHR(SPCOOL) in the same manner as described above. The same modifications were made to RHR/-LPCI and RHR(SPCOOL)/-LPCI. Since there would still be ample time to recover RHR given LPCI success, the nonrecovery probability for RHR/-LPCI was set to the same nominal nonrecovery probability as that for RHR.

The nonrecovery probability for RHR was revised to 0.054 to reflect the RHRSW failure (see Appendix A). For sequences involving potential RHR or power conversion system (PCS) recovery, the nonrecovery estimate was revised to 0.054 x 0.017 (PCS nonrecovery given MSIV closure), or 9.2E-4.

A sensitivity study was performed assuming that the water leak into the bearing oil reservoir for pump C was not sufficient to cause pump C to fail. RHR, RHR(SPCOOL), RHR/-LPCI, and RHR(SPCOOL)/-LPCI were modified to reflect only one failed RHRSW pump ($p = 0.015$).

B.4.5 Analysis Results

The estimated conditional core damage probability is 1.7×10^{-4}. The dominant sequence involves a successful reactor shutdown, failure of the power conversion system, successful feedwater recovery, and failure of RHR, and is highlighted in the event tree in Figure B.4.1. The estimated conditional core damage probability for the sensitivity study (with RHRSW pump C operable) is 2.8×10^{-5}. The dominant sequence remains the same.

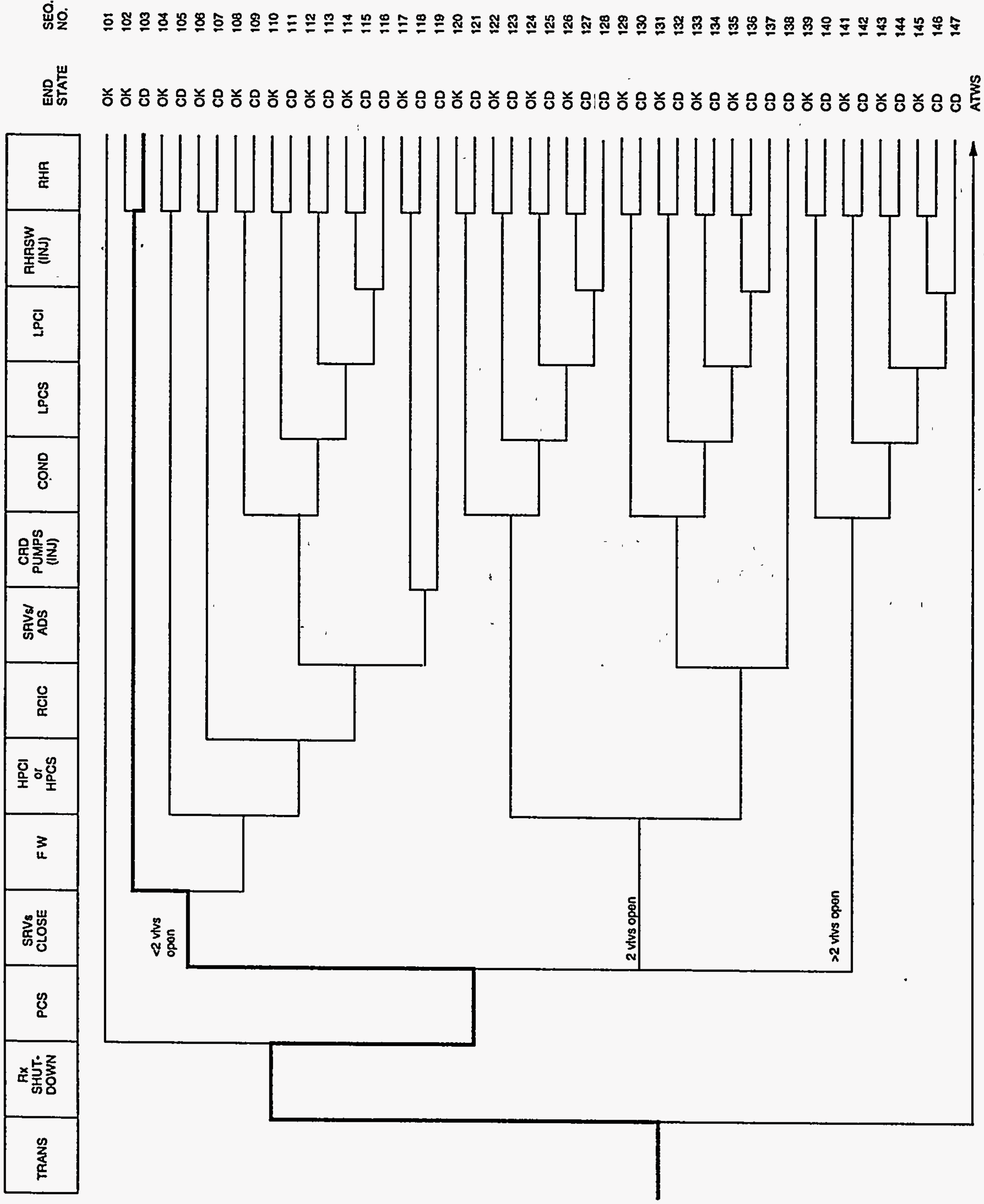

Figure B.4.1 Dominant core damage sequence for LER 254/82-007, -009

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 254/82-007
Event Description: Transient with RHRSW train B inoperable
Event Date: April 15, 1982
Plant: Quad Cities 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.7E-04
Total	1.7E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	9.8E-05	6.1E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	4.9E-05	3.1E-04
121	trans -rx.shutdown PCS srv.ftc.2 -hpci -cond RHR	CD	6.8E-06	3.5E-02
403	trans rx.shutdown -rpt -slcs PCS -ads.inhibit -hpci RHR(SPCOO L)	CD	5.0E-06	9.9E-02
123	trans -rx.shutdown PCS srv.ftc.2 -hpci cond -lpcs RHR	CD	3.5E-06	1.8E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	9.8E-05	6.1E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	4.9E-05	3.1E-04
121	trans -rx.shutdown PCS srv.ftc.2 -hpci -cond RHR	CD	6.8E-06	3.5E-02
123	trans -rx.shutdown PCS srv.ftc.2 -hpci cond -lpcs RHR	CD	3.5E-06	1.8E-02
403	trans rx.shutdown -rpt -slcs PCS -ads.inhibit -hpci RHR(SPCOO L)	CD	5.0E-06	9.9E-02

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\quadcit1.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	2.9E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.5E-01 **	1.6E-02 > 5.4E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-01 **	8.3E-03 > 9.2E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.5E-01 **	1.0E+00 > 5.4E-02	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.5E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.5E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	

ep.rec	4.9E-02	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.5 LER No. 254/82-012, -013, and -018

Event Description: Postulated LOOP with 2 EDGs Inoperable (Unit 1) and Plant-centered LOOP with one EDG Inoperable (Unit 2)

Date of Event: June 22, 1982

Plant: Quad Cities 1 and 2

B.5.1 Summary

During normal operation on June 22, 1982, Unit 2 reactor experienced a loss of offsite power (LOOP) and a trip while the reserve auxiliary transformer 22 was being removed from service for maintenance. Both the Unit 2 and swing emergency diesel generators (EDGs) loaded to their respective emergency buses. The swing diesel generator tripped when the A residual heat removal service water (RHRSW) pump was started. One day prior to the Unit 2 loss of offsite power, the Unit 1 EDG was removed from service due to the failure of the diesel generator cooling water pump to provide flow to the EDG during a high-pressure coolant injection (HPCI) flow rate surveillance test. Thus, when the Unit 2 LOOP occurred, Unit 1 began operating without any EDGs available. The estimated increase in core damage probability, or importance, over the duration of the postulated LOOP at Unit 1 with both EDGs inoperable is 1.4×10^{-4}. The base-case core damage probability (CDP) over the duration of the event is 2.2×10^{-6}, resulting in an estimated conditional core damage probability (CCDP) of 1.4×10^{-4}. The conditional core damage probability estimate for the plant-centered LOOP with one EDG inoperable at Unit 2 is 1.1×10^{-4}.

B.5.2 Event Description

During normal operation on June 22, 1982, at 0526 hours, Unit 2 reactor experienced a trip due to a reactor feedwater pump trip and subsequent low water level due to a loss of bus 22 while the reserve auxiliary transformer 22 was being removed from service for maintenance. An equipment operator mistakenly pulled out the fuses for a 4-kV bus instead of pulling the transformer fuses. The error disconnected power to the 2B reactor feedwater pump, which caused a low water level and initiated a trip. The Unit 2 main generator subsequently tripped and all normal ac power to Unit 2 was lost. Following the LOOP, both the Unit 2 and swing emergency diesel generators (EDGs) loaded to their respective emergency buses. Approximately 22 minutes later, the swing diesel generator tripped when the A residual heat removal service water (RHRSW) pump was started. Because of an error in the design of the EDG protective relaying, the EDG underexcitation relay was unblocked and thus tripped when the RHRSW pump was initiated. Actuation of the underexcitation relay tripped the EDG lock-out relay as well. To restart the EDG, the relay had to be manually reset by the equipment operator. The resetting of the lock-out relay was delayed since the equipment operator had been sent to the switchyard to expedite the restoration of offsite power. Subsequent to the event, the under-excitation relays were temporarily removed on all three diesel generators until a permanent design change could be completed.

One day prior to the Unit 2 LOOP, the Unit 1 EDG was removed from service due to the failure of the diesel generator cooling water pump to provide flow to the EDG during a HPCI flow rate surveillance test.

Investigation revealed that the pump was air bound due to air which entered the suction line while RHRSW A was being drained to install system modifications. The rotating element of the pump was replaced and the pump was returned to service in the late afternoon of June 22nd. When the Unit 2 LOOP occurred, Unit 1 began operating without any EDGs available.

LER 254/82-018 indicates that on June 26th, the Unit 1 EDG cooling water pump was again removed from service to reduce the vibration of the pump due to misalignment of the motor and pump. The motor and pump were re-aligned, and the pump was returned to service. This event occurred after the LOOP and was not modeled.

B.5.3 Additional Event-Related Information

Quad Cities Units 1 and 2 each have one EDG (EDG 1 and EDG 2) dedicated to that unit. They share a common swing EDG (EDG 1/2). EDGs 1 and 2 supply emergency power buses 14-1 and 24-1, respectively, which power core spray pumps 1B and 2B, RHR pumps 1C, 1D, 2C, and 2D, and RHRSW pumps 1C, 1D, 2C, and 2D. The swing EDG can be aligned to power either bus 13-1 or 23-1. Bus 13-1 supplies core spray pump 1A, RHR pumps 1A and 1B, and RHRSW pumps 1A and 1B. Bus 23-1 supplies core spray pump 2A, RHR pumps 2A and 2B, and RHRSW pumps 2A and 2B. The emergency power buses are automatically fed from the EDGs on a loss of offsite power. Unit 1 bus 14-1 and Unit 2 bus 24-1 can be cross-tied by closing two normally open breakers. This provides recovery if normal power is available on the other unit (plant-centered LOOP).

Two 250-V dc and two 125-V dc batteries are shared between both units. Each battery is sized to power its respective loads for 4 hours. Unit 1 batteries are charged from bus 14-1 through bus 19, and Unit 2 batteries are charged from bus 24-1 through bus 29. An alternative charger can be powered from bus 13-1 and 23-1, and can charge either unit's battery. The 480-V ac buses power the battery chargers on each unit, and can also be cross-tied.

B.5.4 Modeling Assumptions

This event was modeled as two separate events. The first analysis considers a postulated LOOP with two EDGs inoperable for Unit 1 and assumes that both of the EDGs were inoperable for up to half the surveillance period on the EDGs, 15 days. One train of emergency power (EP) was set to failed to reflect the failure of EDG 1/2, and the other train was set to unavailable to reflect EDG 1's unavailability due to maintenance. Since power can be recovered from Unit 2 following a plant-centered LOOP at Unit 1 (only Unit 2 would be without offsite power in this case), only dual-unit LOOPs (grid-and weather-related) were considered in this analysis. The frequency of LOOP and the probability of not recovering offsite power was revised to reflect this.

The second analysis considers the plant-centered LOOP which occurred at Unit 2 and the inoperability of the swing EDG. The LOOP frequency and the probabilities of failing to recover offsite power in the short term and before battery depletion were modified for a plant-centered LOOP using the models described in *Revised LOOP Frequency and PWR Seal LOCA Models,* ORNL/NRC/LTR-89/11, August 1989. One train of emergency power was set to failed to reflect the failure of EDG 1/2, and all associated equipment powered by the swing EDG was set to unavailable. The probability of failing to recover offsite power prior to battery depletion was revised to reflect the potential for recovery from Unit 1. The probability of failing to recover power from Unit 1 was assumed to be 0.1 (see Appendix A).

In this event, Unit 1 remained operating during the Unit 2 LOOP. Had Unit 1 tripped and experienced a LOOP during the Unit 2 LOOP, Unit 1 would have experienced a station blackout.

B.5.5 Analysis Results

The estimated increase in core damage probability over the duration of the postulated LOOP at Unit 1 is 1.4×10^{-4}. The base-case CDP (not shown in calculation) is 2.2×10^{-6}, resulting in an estimated CCDP of 1.4×10^{-4}. The dominant sequence, highlighted on the event tree in Figure B.5.1, involved a successful reactor shutdown, failure of emergency power, and failure to recover offsite power prior to battery depletion. The estimated conditional core damage probability for the Unit 2 plant-centered LOOP with one EDG inoperable is 1.1×10^{-4}. The dominant sequence involves a successful reactor shutdown, successful emergency power, successful HPCI, and failure of RHR.

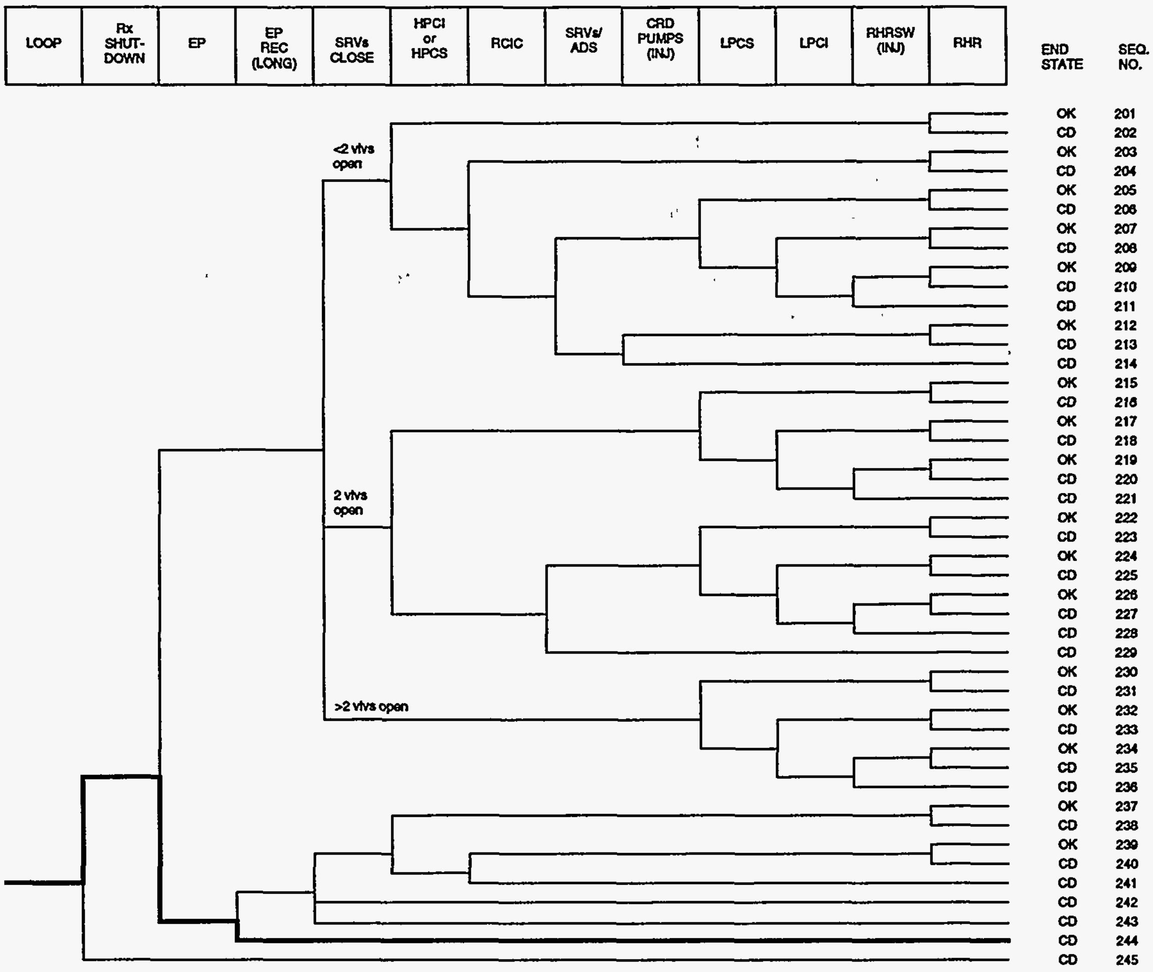

Figure B.5.1 Dominant core damage sequence for LER 254/82-012, -013, and -018

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 254/82-012, -013 and -018
Event Description: Postulated LOOP with two EDGs inoperable (Unit 1)
Event Date: June 22, 1982
Plant: Quad Cities 1

UNAVAILABILITY, DURATION= 360

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	6.7E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	1.4E-04
Total	1.4E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

Sequence	End State	Prob	N Rec**
244 LOOP -rx.shutdown EP EP.REC	CD	1.4E-04	6.6E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

Sequence	End State	Prob	N Rec**
244 LOOP -rx.shutdown EP EP.REC	CD	1.4E-04	6.6E-01

** non-recovery credit for edited case

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\quadcit1.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
LOOP	1.6E-05 > 2.8E-06	5.3E-01 > 6.6E-01	
Branch Model: INITOR			

Initiator Freq:	1.6E-05 > 2.8E-06		
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.5E-04	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
EP	2.9E-03 > 1.0E+00	8.7E-01 > 1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02 > Unavailable		
EP.REC	4.9E-02 > 2.1E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.9E-02 > 2.1E-01		
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 254/82-012, -013 and -018
Event Description: Plant-centered LOOP with one EDG inoperable (Unit 2)
Event Date: June 22, 1982
Plant: Quad Cities 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	5.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	1.1E-04
Total	1.1E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
242	LOOP -rx.shutdown EP -EP.REC srv.ftc.2	CD	3.2E-05	4.3E-01
241	LOOP -rx.shutdown EP -EP.REC srv.ftc.<2 hpci rcic	CD	2.1E-05	2.1E-01
245	LOOP rx.shutdown	CD	1.8E-05	5.0E-02
244	LOOP -rx.shutdown EP EP.REC	CD	1.6E-05	4.3E-02
202	LOOP -rx.shutdown -EP srv.ftc.<2 -hpci RHR	CD	1.2E-05	7.9E-03
243	LOOP -rx.shutdown EP -EP.REC srv.ftc.>2	CD	5.5E-06	4.3E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
202	LOOP -rx.shutdown -EP srv.ftc.<2 -hpci RHR	CD	1.2E-05	7.9E-03
241	LOOP -rx.shutdown EP -EP.REC srv.ftc.<2 hpci rcic	CD	2.1E-05	2.1E-01
242	LOOP -rx.shutdown EP -EP.REC srv.ftc.2	CD	3.2E-05	4.3E-01
243	LOOP -rx.shutdown EP -EP.REC srv.ftc.>2	CD	5.5E-06	4.3E-01
244	LOOP -rx.shutdown EP EP.REC	CD	1.6E-05	4.3E-02
245	LOOP rx.shutdown	CD	1.8E-05	5.0E-02

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\quadcit2.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	6.9E-04	1.0E+00	
LOOP	1.6E-05 > 1.4E-05	5.3E-01 > 5.0E-01	
Branch Model: INITOR			
Initiator Freq:	1.6E-05 > 1.4E-05		
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
LPCS	2.0E-03 > 2.0E-02	1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
LPCI	1.1E-03 > 2.0E-03	1.0E+00	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
Train 4 Cond Prob:	5.0E-01 > Unavailable		
Serial Component Prob:	1.0E-03		
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.0E-03	1.6E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
Train 4 Cond Prob:	5.0E-01 > Unavailable		
RHR.AND.PCS.NREC	1.5E-04 > 1.0E-03	8.3E-03	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
Train 4 Cond Prob:	5.0E-01 > Unavailable		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
EP	2.9E-03 > 5.7E-02	8.7E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02		
EP.REC	4.9E-02 > 6.4E-03	1.0E+00 > 1.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.9E-02 > 6.4E-03		
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02

ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.6 LER No. 259/83-006 and -007

Event Description: Scram, MSRV and its Vacuum Breaker Fail Open

Date of Event: February 5, 1983

Plant: Browns Ferry 1

B.6.1 Summary

Unit 1 was operating at approximately 100% power when a reactor scram occurred. A main steam relief valve (MSRV) was opened manually to control reactor pressure but when operators attempted to close it, they were unable to do so. The MSRV tailpipe vacuum relief valve failed open during the event, venting steam into the drywell instead of directing it to the suppression pool. The conditional core damage probability estimated for the event is 4.4×10^{-5}.

B.6.2 Event Description

On February 5, 1983, Unit 2 was operating at 100% power, when the operator began a surveillance test of the main turbine overspeed trip system. At that time something, possibly a turbine trip, occurred which caused a reactor scram. MSRV 1-1-22 was manually opened to control reactor pressure, but operators were unable to close it.

At the same time, the vacuum relief valve for the MSRV 1-1-22 tailpipe was stuck open. The main steam relief valves at Browns Ferry sit on stub headers attached to the main steam lines. Each MSRV's exhaust is routed via a tailpipe to a quencher submerged in the suppression pool. After MSRV operation, steam in the tailpipe will condense, drawing a vacuum. This tends to draw a slug of water up from the suppression pool into the tailpipe. Subsequent operation of the associated MSRV could propel this water slug into the quencher, causing damage. To prevent this, each tailpipe is equipped with two vacuum breakers which open after MSRV operation to limit negative pressure in the tailpipe.

At the time of the event, the disk in a vacuum breaker for MSRV 1-1-22 was partially separated from its hinge arm and jammed in the open position, which allowed steam to vent continuously to the drywell. The peak drywell pressure attained and the specific leak rate were not noted in the licensee event report for this event, but it was noted that leakage into the drywell exceeded a Technical Specification limit of 5 gpm. Therefore, it may be assumed that steam leakage into the drywell was in excess of 200 cubic feet per minute, possibly substantially in excess. It is unclear why the drywell pressure did not exceed the 2.45 psig loss-of-coolant accident (LOCA) setpoint during this event.

The failure of the MSRV was initially attributed to its pilot valve so, after the unit was shut down, the pilot valve on 1-1-22 was replaced. The vacuum relief valve for the MSRV was found stuck open and was replaced also. Four other vacuum relief valves were also found to be damaged and were replaced.

Three days after the event described, the unit was restarted. The intention was to test the repaired MSRV when reactor pressure reached 250 psig. However, when reactor pressure reached 178 psig, MSRV 1-1-22 opened spontaneously and could not be closed until the reactor was shut down again.

In addition, the movement indicating arm attached to the tailpipe vacuum breaker for MSRV 1-1-22 was apparently bent out of its normal alignment, causing the vacuum breaker to again stick open. Once again, an unspecified quantity of steam leakage greater than 5 gpm entered the drywell until the unit was shut down.

This time, MSRV 1-1-22 was replaced. More rigorous inspection of the MSRV found that the pilot valve inlet tube mounting bracket had broken, and a piece of debris had lodged under the seat of the main valve.

The position indicating arms were removed from all vacuum relief valves to prevent further valve failures.

B.6.3 Additional Event-Related Information

A safety evaluation report provided with licensee event report 259/83-007 indicates that this event, involving a stuck-open SRV relieving directly to the drywell instead of through quenching headers to the suppression pool, may be represented as a loss-of-coolant accident (LOCA) with a break area of approximately 0.15 square feet.

B.6.4 Modeling Assumptions

This event was modeled as a LOCA, and it was assumed that the initiating event was nonrecoverable. A break area of 0.15 square feet is defined as an intermediate break in Browns Ferry analyses. To represent the medium-break LOCA, the low flow-rate makeup systems, control rod drive (CRD) and reactor core isolation cooling (RCIC), were set to failed, since they were assumed to be inadequate to make up losses.

The second event occurred three days after shutdown, with a reduced decay heat load and at temperature and pressure conditions only slightly above those which would permit alignment of the residual heat removal (RHR) system. Presumably, given a stuck-open relief valve, reactor pressure was sufficiently reduced to allow resumption of shutdown cooling within a short time. The second event was therefore considered to be little different from a routine trip during startup, and the event was not analyzed.

B.6.5 Analysis Results

The conditional core damage probability estimated for this event is 4.4×10^{-5}. The dominant core damage sequences, highlighted on the event tree in Figure B.6.1, involve a failure to trip following the LOCA [this anticipated transient without scram (ATWS) sequence is not developed in the model], and the observed LOCA, failure of the power conversion system (PCS), the feedwater system (FW), high-pressure coolant injection (HPCI), and the automatic depressurization (ADS). Excluding the potentially conservative ATWS sequence, a conditional core damage probability of 8.6×10^{-6} is estimated.

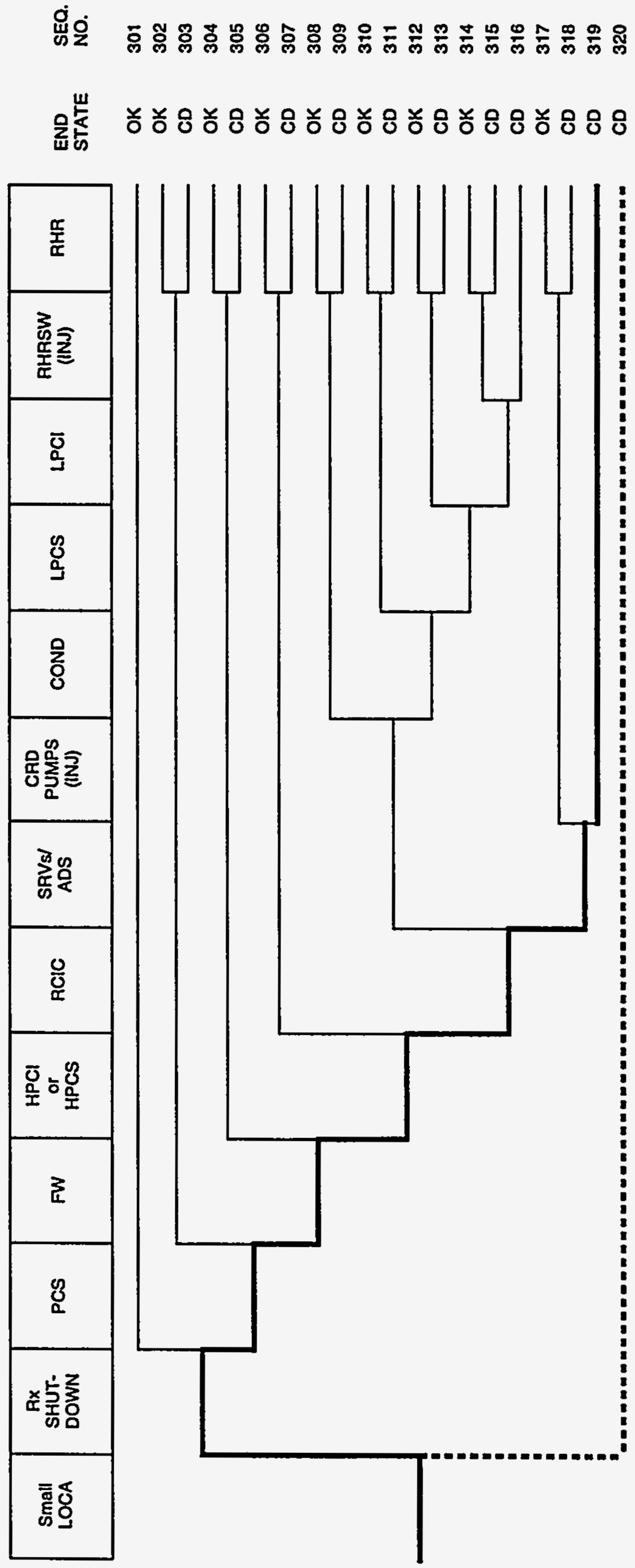

Figure B.6.1 Dominant core damage sequence for LER 259/83-006 and -007

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 259/83-006 and -007
Event Description: Scram, MSRV and its vacuum breaker fail open
Event Date: February 5, 1983
Plant: Browns Ferry 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOCA	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOCA	4.4E-05
Total	4.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
320	LOCA rx.shutdown	CD	3.5E-05	1.0E-01
319	LOCA -rx.shutdown pcs mfw hpci RCIC srv.ads CRD(INJ)	CD	6.8E-06	1.7E-01
303	LOCA -rx.shutdown pcs -mfw rhr	CD	1.8E-06	1.4E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
303	LOCA -rx.shutdown pcs -mfw rhr	CD	1.8E-06	1.4E-02
319	LOCA -rx.shutdown pcs mfw hpci RCIC srv.ads CRD(INJ)	CD	6.8E-06	1.7E-01
320	LOCA rx.shutdown	CD	3.5E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\brown1.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail

trans	1.7E-03	1.0E+00	
loop	1.6E-05	2.4E-01	
LOCA	3.3E-06 > 3.3E-06	6.7E-01 > 1.0E+00	
Branch Model: INITOR			
Initiator Freq:	3.3E-06		
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	4.6E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
CRD(INJ)	1.0E-02 > 1.0E+00	1.0E+00	1.0E-02
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.5E-04	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	7.5E-03	8.7E-01	
ep.rec	1.4E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.7 LER No. 260/83-074

Event Description: Trip with HPCI Inoperable

Date of Event: November 10, 1983

Plant: Browns Ferry 2

B.7.1 Summary

Unit 2 was operating at approximately 98% power when a reactor scram occurred. Reactor vessel level dropped sufficiently to provide an auto-initiation signal to the high pressure coolant injection (HPCI) system. HPCI started and immediately isolated when a turbine exhaust rupture diaphram ruptured, rendering HPCI inoperable. The conditional core damage probability estimated for the event is 3.2×10^{-5}.

B.7.2 Event Description

On November 10, 1983, while operating at essentially full power, Unit 2 experienced a scram. Reactor vessel level dropped sufficiently to result in HPCI auto-initiation; however, HPCI immediately isolated when its turbine exhaust rupture diaphragm ruptured.

The cause of the failure was not determined with certainty. An exhaust diaphragm rupture which occurred during testing five days earlier had been attributed to inadequate draining of condensate from HPCI steam lines. Apparently, the November 5 rupture disk failure may have been caused by the impact of a slug of water which accelerated in the steam exhaust line after the turbine started. While the disk rupture patterns were found to be similar in both events, the November 10 failure was tentatively attributed to control system problems. Testing conducted later, in February of 1984, suggested that improper HPCI control system behavior could lead to exhaust line pressure fluctuations, perhaps great enough to cause failure of the rupture disk. Adjustments were made to the control system to minimize these fluctuations.

B.7.3 Additional Event-Related Information

High-pressure makeup sources at Browns Ferry include the turbine-driven main feedwater pumps, HPCI, the reactor core isolation cooling system (RCIC) and the control rod drive (CRD) pumps. For events involving isolation of the reactor vessel, only HPCI can provide high flow-rate (5,000 gpm) makeup to the reactor.

B.7.4 Modeling Assumptions

This event was modeled as a scram with HPCI assumed unavailable and not recoverable. Because the HPCI auto-initiation reported indicates that reactor vessel level had dropped to -51.5 inches below instrument zero, it can be assumed that the main steam isolation valves (MSIVs) isolated, causing an initial loss of main feedwater and power conversion systems. The nonrecovery probability for the power conversion system (PCS) was revised to 0.017 to reflect initial assumed closure of main steam isolation valves.

B.7.5 Analysis Results

The conditional core damage probability estimated for this event is 3.2×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.7.1, involves the observed trip, unavailability of the power conversion system, failure of two safety relief valves (SRVs) to close, unavailability of HPCI, and failure of the automatic depressurization system (ADS).

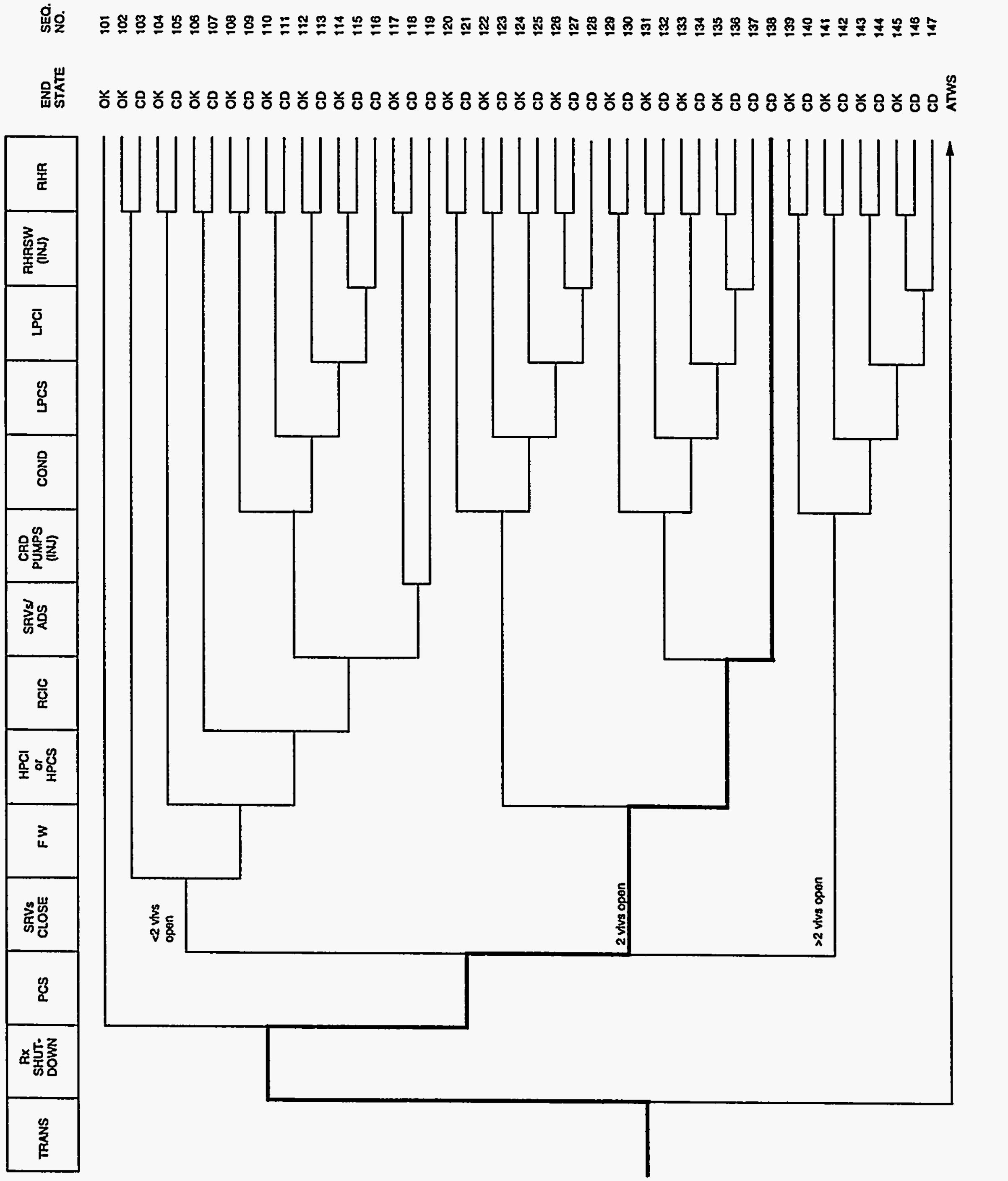

Figure B.7.1 Dominant core damage sequence for LER 260/83-074

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 260/83-074
Event Description: Scram with HPCI inoperable
Event Date: November 10, 1983
Plant: Browns Ferry 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.2E-05
Total	3.2E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.6E-05	7.0E-01
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
119	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI rcic srv.ads c rd(inj)	CD	3.6E-06	1.7E-01
107	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI -rcic RHR.AND.PC S.NREC	CD	3.3E-06	9.1E-05
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
107	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI -rcic RHR.AND.PC S.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI rcic srv.ads c rd(inj)	CD	3.6E-06	1.7E-01
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.6E-05	7.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\brown2.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	2.0E-03	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
HPCI	2.9E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-02 > 1.0E+00		
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	7.5E-03	8.7E-01	
ep.rec	1.4E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.8 LER No. 261/83-004, -007 and -016

Event Description: Transient with One AFW Pump and One PORV Inoperable

Date of Event: April 19, 1983

Plant: Robinson 2

B.8.1 Summary

On April 19, 1983, following a reactor trip, A and B motor-driven auxiliary feedwater (AFW) pumps started automatically. The B AFW pump tripped due to low discharge pressure caused by pump cavitation resulting from buildup of vapor in the pump's casing. On April 29, 1983, a pressurizer power operated relief valve (PORV) failed to meet required cycle time and also on a third attempt to cycle, the valve failed to fully open. The estimated conditional core damage probability for this event is 9.2×10^{-4}.

B.8.2 Event Description

On April 19, 1983, following a reactor trip caused by failure of the turbine electro-hydraulic oil pumps, A and B motor-driven AFW pumps started automatically. Within five minutes of the auto-start, pump B tripped. Visual inspection of the B pump breaker revealed no damage, and the breaker did not appear to have tripped on overcurrent. The periodic AFW component test was performed. The test requires that the pump casing be vented prior to running the pump. When the casing was vented, a significant amount of vapor was released from the pump casing. Thus, it was determined that the pump tripped due to low discharge pressure caused by pump cavitation resulting from vapor buildup inside the pump casing. Following the test on pump B, the same test was performed on pump A. Pump A casing did not release any vapor when vented. A later examination of the pumps on April 20, 1983 revealed high pump temperatures. The temperature indicated that a slight backleakage of hot water through the discharge gate valves into the pump casings existed. Both pumps were again vented, but no vapor was released.

A similar occurrence of backleakage through the discharge valves resulted in the binding of the turbine-driven AFW pump on July 21, 1983 (LER 261/83-016). The plant was operating at 79% power when the turbine-driven AFW pump was declared inoperable due to steam binding. The plant was shut down when the limiting condition for operation time limit expired, seven days later. The pump discharge valves were repaired and a leakage evaluation was performed with satisfactory results.

On April 29, 1983, during testing of the PORVs, valve RC-455C failed to meet the required cycle time, and on a subsequent attempt to cycle the valve, the valve failed to fully open. Inspections revealed that the cause of the PORV failure was galling of the valve plug to the cage. The valve was rebuilt and returned to service approximately thirteen days later. A stem and valve plug manufactured from materials designed to reduce the chance of galling and a stem guide bushing intended to improve the valve plug's ability to seat were installed.

B.8.3 Additional Event-Related Information

The AFW system at Robinson 2 is a three-train system consisting of two motor-driven pumps and a turbine-driven pump. Either motor-driven pump or the turbine-driven pump is capable of supplying secondary side cooling to any of three steam generators.

In addition to providing overpressure protection, the PORVs are used in conjunction with the safety injection system to provide bleed and feed cooling should the AFW and main feedwater (MFW) systems fail.

B.8.4 Modeling Assumptions

AFW pump B was declared inoperable following a reactor trip. Assuming the PORV was faulted at the time of the trip as well (which is likely since these valves are usually cycled only when shutdown), this event was modeled as a transient with one train of AFW set to failed and the feed-and-bleed (FEED.BLEED) branch probability set to 1.0.

The mechanism which failed AFW pump B could have occurred in the other pumps as well. LER 261/83-016 reported a similar problem with the turbine-driven pump three months later. To reflect the potential impact of this failure mode in all three pumps, the serial component probability (which represents common cause effects among the three different design pumps in the AFW model) was revised to 3.0E-2 [p(pump A fails from steam binding given pump B failed from steam binding)*p(pump C fails given pump B and pump A failed)], 0.1*0.3, using typical ASP Program conditional failure probabilities). Because potential common cause effects were addressed using the serial component failure probability, the AFW train failure probabilities were revised to reflect the unavailability of pump B (independent faults).

The failure probability for AFW following ATWS was also revised to reflect the potential common cause failure of all three pumps. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event.

The serial component of the feed-and-bleed branch probability represents the failure of the PORVs. In the models, both PORVs are assumed to be needed for proper accident mitigation using feed and bleed. Thus, to represent the effect of one PORV inoperable, the serial component of FEED.BLEED was set to 1.0. Because PORV RC-455C partially opened, the PORVs were assumed to be available to support pressure relief.

B.8.5 Analysis Results

The estimated conditional core damage probability for this event is 9.2×10^{-4}. The dominant sequence, highlighted on the event tree in Figure B.8.1, involved the observed trip, failure of AFW, failure of main feedwater, and failure of feed and bleed.

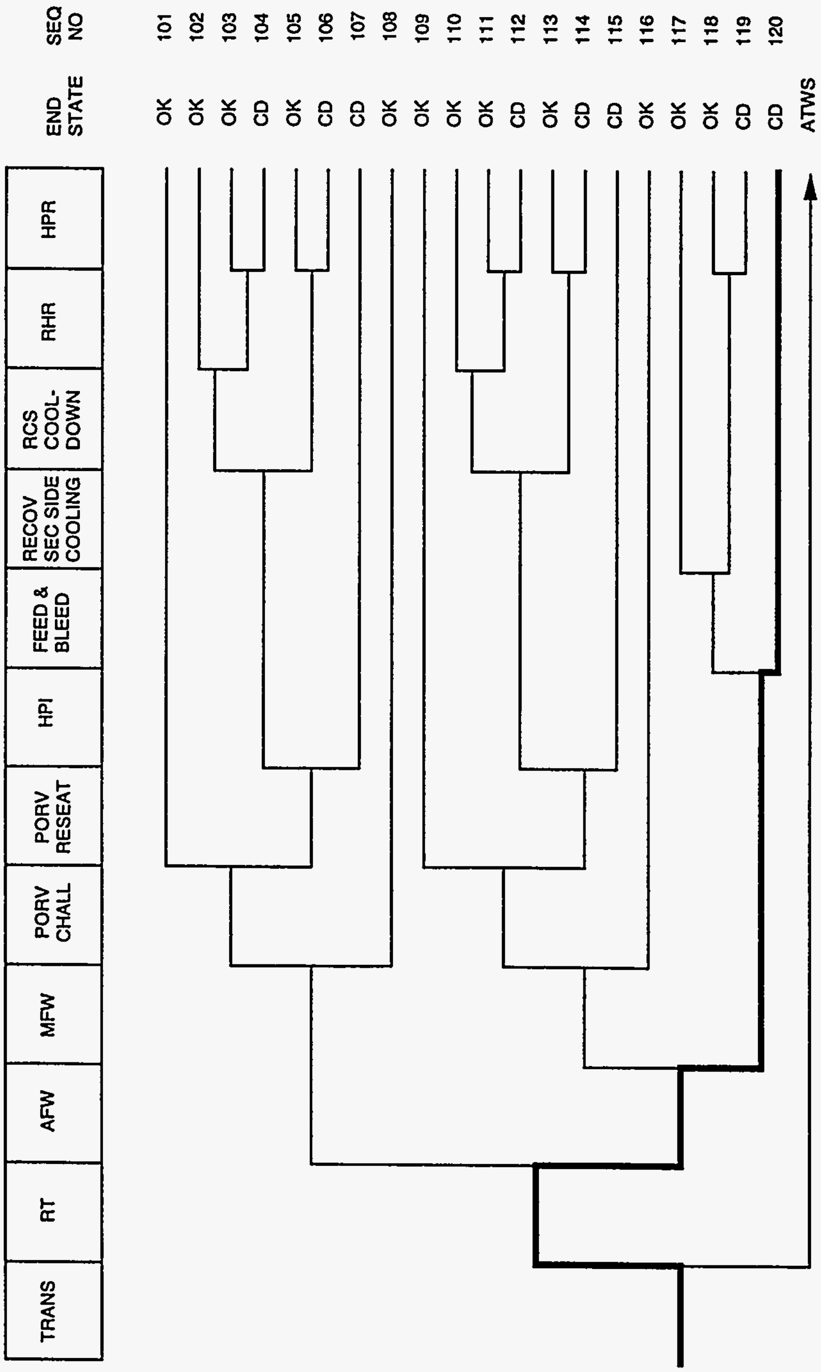

Figure B.8.1 Dominant core damage sequence for LER 261/83-004, -007, and -016

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 261/83-004. -005. -007. and -016
Event Description: Transient with one AFW pump and one PORV inoperable
Event Date: April 19. 1983
Plant: Robinson

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	9.2E-04
Total	9.2E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW mfw FEED.BLEED	CD	9.1E-04	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW mfw FEED.BLEED	CD	9.1E-04	1.5E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\robinson.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.0E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	

rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 3.1E-02	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04 > 3.0E-02		
AFW/ATWS	4.3E-03 > 1.9E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > 1.9E-01		
afw/ep	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.5E-03	8.9E-01	
FEED.BLEED	2.0E-02 > 1.0E+00	1.0E+00	1.0E-02
Branch Model: 1.OF.3+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Serial Component Prob:	2.0E-02 > 1.0E+00		
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	3.1E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.9 LER No. 272/83-011 and -012

Event Description: Transient with Automatic Reactor Trip Capability Failed

Date of Event: February 22, 1983

Plant: Salem 1

B.9.1 Summary

On February 22, 1983, during routine startup of Salem 1 at 20% power, both reactor trip breakers failed to open automatically on receipt of a low-low steam generator level reactor trip signal. A manual trip was initiated approximately 3 seconds after the automatic trip breaker failed to open, and was successful. Investigation focused on the reasons for the manual trip and the fact that both reactor trip breakers had failed was not revealed. A similar event occurred on February 25th, at 12% power. Both reactor trip breakers failed to open automatically on receipt of a valid low-low steam generator level reactor trip signal. A manual trip was initiated 25 seconds later, and was successful. Following the trip, the plant was placed in a stable shutdown condition. Investigation revealed that mechanical binding of the latch mechanism in the breaker undervoltage trip attachment failed both breakers in both events. These attachments were replaced with new devices and tested extensively. The combined conditional core damage probability estimate for these events is 4.6×10^{-3}.

B.9.2 Event Description

On February 22, 1983, during routine startup of Salem 1 at 20% power, a manual trip was initiated due to rapidly decreasing steam generator levels. Both reactor trip breakers failed to open 3 seconds earlier upon receipt of a valid low-low steam generator reactor trip signal. The event occurred due to a transient which was initiated by the loss of No. 1F 4-kV Group bus during the transfer to the auxiliary power transformer. An automatic safety injection occurred and No. 11 reactor coolant pump tripped for no apparent reason. Loss of pressurizer spray increased the pressurizer pressure to the power-operated relief valve (PORV) setpoint and two PORVs actuated. The PORVs mitigated the transient and no damage to the reactor coolant system occurred. The plant was then placed in safe shutdown. Investigations focused on the manual trip and the other related event, and the failure of the trip breakers was missed.

On February 25, 1983, during startup at 12% power, a low-low steam generator level signal was generated by the reactor trip system. Both reactor trip breakers failed to open and remained closed until operators manually tripped the plant 25 seconds later. During reviews of this event and the February 22nd event, it was determined that the breakers had also failed during the February 22nd event. Investigation of the reactor trip system revealed that the breakers had failed to open automatically due to mechanical binding of the latch mechanism in the undervoltage trip attachment. Since the manual trip operated the shunt trip device as well as the undervoltage trip attachment, the manual trip succeeded. Following the manual trip, the plant was placed in a safe shutdown condition.

B.9.3 Additional Event-Related Information

The Salem 1 reactor protection system (RPS) uses two independent channels and trains which consist of sensors, transmitters, relays and trip breakers to detect and protect against unsafe plant conditions. When an unsafe plant condition occurs, the RPS signals the trip breakers to open and de-energize the rod drive mechanisms, resulting in a shutdown, and also transfers the information to the safeguards equipment cabinet which in turn determines the type of accident and loads and starts the safety systems needed to mitigate the effect of the initiating event. The reactor trip breakers are ac circuit breakers positioned in series. When either trip breaker is tripped open, holding power to the control rods is lost and the rods drop into the core. Two mechanisms could open the trip breaker at the time of the event. The first mechanism is the undervoltage trip coil which, upon de-energization, would trip open the breaker. When the RPS signals the trip breakers, de-energization of the undervoltage trip coils occurs and the breakers open. The second mechanism for tripping open the breakers is through energizing the shunt trip coil. The shunt trip coils, once energized, will open the breakers. A manual scram would energize the shunt trip coils and open the breakers. Following the Salem failure-to-trip, the RPS was reconfigured to automatically actuate the shunt trip coils as well.

B.9.4 Modeling Assumptions

Because the initial failure to trip was not discovered until after the second failure, the impact of both transients was addressed in a single analysis. Both events were modeled as transients with a portion of the reactor trip system failed. The reactor trip system is modeled as a double-train system with a nonrecovery probability. Both trains represent the automatic portion of the reactor trip system. Each train is assumed to be dominated by the failure of a reactor trip breaker. One of two trip breakers must open in order for the reactor to automatically scram. The nonrecovery probability is the likelihood that operators will not manually scram the reactor. In this event, both trip breakers failed to operate correctly, thus both reactor trip system trains were set to failed. The operators successfully manually scrammed the reactor, so the nonrecovery probability was left at its default value.

The February 25, 1983, transient occurred at relatively low power. This low power level may have required reduced AFW flow and fewer relief valves for primary pressure protection than assumed in the ATWS model (see appendix A) used in this analysis. However, since development of such specialized success criteria is beyond the scope of this effort, both transients were assessed using the same model.

B.9.5 Analysis Results

The conditional core damage probability estimate for each transient is 2.3×10^{-3}, resulting in an overall estimate for the combined event of 4.6×10^{-3}. The dominant sequences are all postulated ATWS sequences, with the highest contributor involving failure to trip, a successful AFW given ATWS, and failure of the emergency boration system. The dominant sequence is highlighted on the event tree in Figure B.9.1.

LER 272/83-013 reported the occurrence of an all-rods-out positive moderator temperature coefficient (MTC) prior to the transient on February 22nd. Since the value of the MTC at the time of the trip breaker failures is not known, the high MTC was not addressed in the analysis. According to NUREG/CR 4550, Vol. 3, Rev.1, Part 1, *Analysis of Core Damage Frequency: Surry, Unit 1 Internal Events,* an MTC of -7pcm/F is the critical value above which, if the plant is at high power, RCS pressure cannot be maintained below 3200 psi; transients initiated at low power have no restrictions on MTC, since the PORVs would be able to maintain the pressure below 3200 psi, and high power is assumed to be above 25% power. In this event, power was never greater than 20%. However, had the transient been initiated at a power above 25%, when the MTC was greater than -7 pcm/F, mitigation may have been affected since RCS pressure may have exceeded 3200 psi.

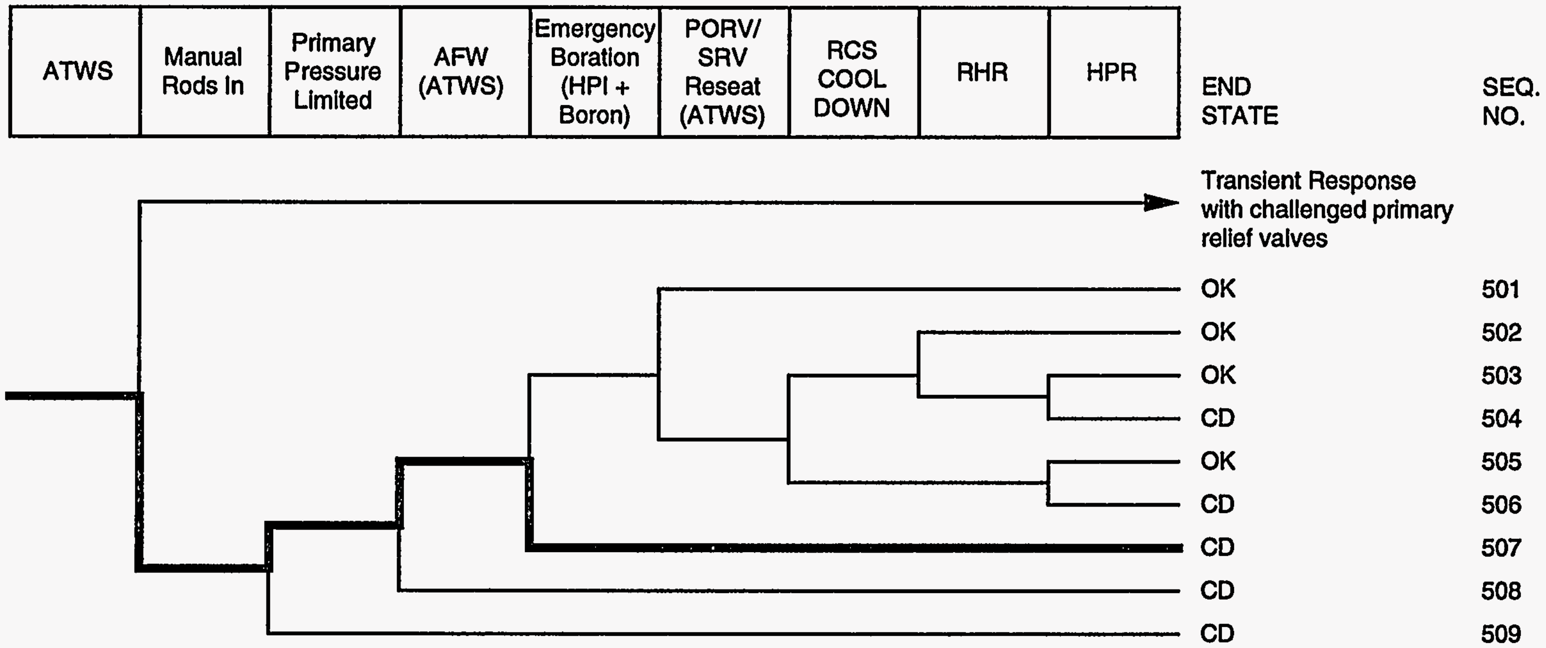

Figure B.9.1 Dominant core damage sequence for LER 272/83-011 and -012

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 272/83-011 and -012
Event Description: Transient with automatic trip breakers failed
Event Date: February 22. 1983
Plant: Salem 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	2.3E-03
Total	2.3E-03

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
507	trans RT -prim.press.limited -afw/atws emrg.boration	CD	9.9E-04	1.0E-01
509	trans RT prim.press.limited	CD	8.8E-04	1.0E-01
508	trans RT -prim.press.limited afw/atws	CD	4.3E-04	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
507	trans RT -prim.press.limited -afw/atws emrg.boration	CD	9.9E-04	1.0E-01
508	trans RT -prim.press.limited afw/atws	CD	4.3E-04	1.0E-01
509	trans RT prim.press.limited	CD	8.8E-04	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\saleml.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.2E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
RT	2.8E-04 > 1.0E+00	1.0E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	1.5E-03 > Failed		
Train 2 Cond Prob:	1.9E-01 > Failed		
rt(loop)	0.0E+00	1.0E+00	
afw	3.8E-04	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-05	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	5.4E-04	8.9E-01	
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.10 LER No. 272/83-033 and -034

Event Description: Loss of Offsite Power with Auxiliary Feedwater Pump 13 Failed

Date of Event: August 11, 1983

Plant: Salem 1

B.10.1 Summary

On August 11, 1983, with Salem 1 operating at 99% power and Salem 2 operating at 100% power, both units tripped due to debris which clogged the circulation water system (CWS) intake screens. A decrease in condenser vacuum led to the trip of feedwater pump 11 and an undervoltage condition which resulted in a loss of offsite power (LOOP) at Unit 1. Following the LOOP at Unit 1, auxiliary feedwater (AFW) pump 13 (a turbine-driven pump) failed to start. A LOOP did not occur at Unit 2, and Unit 2 was brought to a stable shutdown condition. The conditional core damage probability estimated for the event at Unit 1 is 1.2×10^{-4}.

B.10.2 Event Description

On August 11, 1983, with Salem 1 operating at 99% power and Salem 2 operating at 100% power, both units tripped due to debris which clogged the CWS intake screens. The clogged intake screens led to a trip of the CWS and a decrease in condenser vacuum which required rapid load reductions at both units. The combined load reduction and vacuum decrease resulted in a Unit 1 steam generator feedwater pump 11 trip, which in turn led to a decrease in feedwater flow and steam generator levels, which then induced a Unit 1 trip. A few minutes later, an undervoltage condition occurred on all vital buses at Unit 1 associated with the transfer of the group buses to the station power transformers. The undervoltage condition led to a LOOP. At the same time, a low-low steam generator level signal occurred at Unit 1, but Unit 1 AFW turbine-driven pump 13 failed to start due to the trip valve which was left in the tripped position following a test due to a malfunction of the valve position indicator. During these events at Unit 1, Unit 2 experienced the CWS trip, the turbine generator was successfully unloaded, and Unit 2 tripped. Unit 2 experienced no LOOP and was placed in a stable shutdown condition.

B.10.3 Additional Event-Related Information

Salem 1 AFW system consists of two motor-driven pumps and one turbine-driven pump. The motor-driven pumps are actuated on the receipt of a safety injection signal, a low-low steam generator level signal or the trip of both steam generator feedwater pumps. The turbine-driven pump is actuated upon the loss of offsite power, the receipt of a low-low steam generator level signal for two of four steam generators, or an undervoltage in group buses using one-of-two-taken-twice logic.

B.10.4 Modeling Assumptions

This event was modeled as a plant-centered LOOP at Unit 1 with a degraded AFW system. Offsite power non-recovery probabilities and the probability of seal LOCA were modified as shown in Table 1 to reflect those values associated with a plant-centered LOOP (see ORNL/NRC/LTR 89/11, *Revised LOOP Recovery and PWR Seal LOCA Models,* August 1989).

Table 1. Revised LOOP Probabilities

Event	Default Probability	Revised Probability
LOOP short-term nonrecovery	0.53	0.5
Seal LOCA probability	0.27	0.23
Offsite power nonrecovery prior to battery depletion given no seal LOCA	7.0E-2	4.3E-2
Offsite power nonrecovery given seal LOCA	0.57	0.48
Offsite power nonrecovery within 2 hours (OFFSITE.PWR.REC/-EP.AND.-AFW)	0.22	0.14
Offsite power nonrecovery within 6 hours (OFFSITE.PWR.REC/-EP.AND.AFW)	6.7E-2	9.9E-4

AFW pump 13 failed to start due to a trip valve which was left in its tripped position during a test. AFW pump 13 is the turbine-driven pump and is modeled as train 3 in the ASP model. In the ASP model, this train was set to failed for the AFW (AFW) and AFW given failure of emergency power (AFW/EP) branches. These are the two branches associated with the LOOP sequences.

B.10.5 Analysis Results

The estimated conditional core damage probability for the LOOP at Unit 1 in this analysis is 1.2 x 10^{-4}. The dominant core damage sequence involves the observed LOOP and successful reactor trip, failure of emergency power, and failure of AFW given the failure of emergency power and is highlighted on the event tree shown in Figure B.10.1.

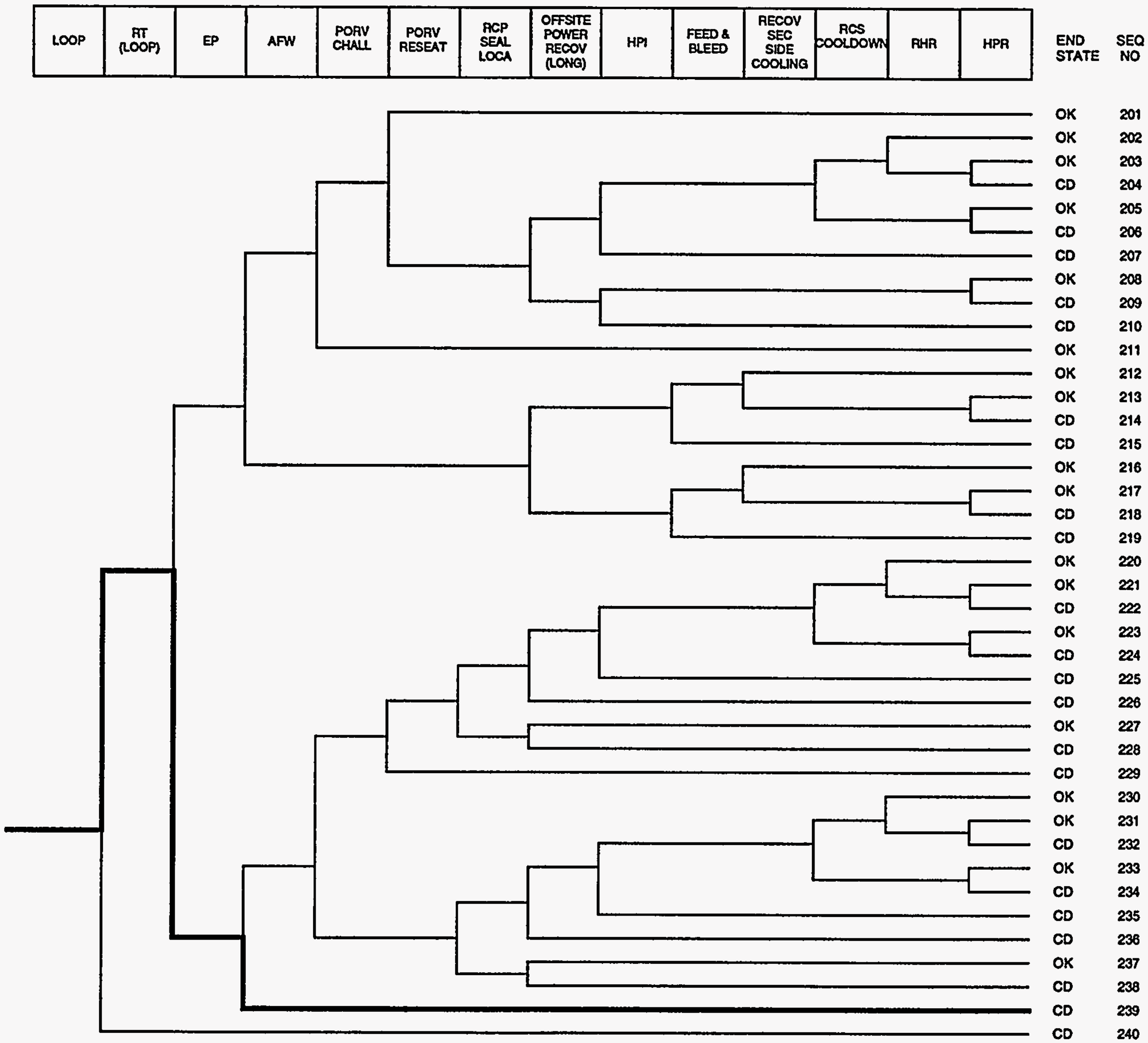

Figure B.10.1 Dominant core damage sequence for LER 272/83-033 and -034

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 272/83-033 and -034
Event Description: Plant centered LOOP with turbine-driven AFW pump inop.
Event Date: August 11, 1983
Plant: Salem 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	5.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	1.2E-04
Total	1.2E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
239	LOOP -rt(loop) ep AFW/EP	CD	8.2E-05	1.5E-01
226	LOOP -rt(loop) ep -AFW/EP porv.chall/sbo -porv.reseat/ep SEAL .LOCA OFFSITE.PWR.REC/SEAL.LOCA	CD	1.7E-05	2.9E-01
215	LOOP -rt(loop) -ep AFW -OFFSITE.PWR.REC/-EP.AND.AFW feed.bleed	CD	1.5E-05	2.2E-01
228	LOOP -rt(loop) ep -AFW/EP porv.chall/sbo -porv.reseat/ep -SEAL .LOCA OFFSITE.PWR.REC/-SEAL.LOCA	CD	5.2E-06	2.9E-01
229	LOOP -rt(loop) ep -AFW/EP porv.chall/sbo porv.reseat/ep	CD	3.2E-06	2.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
215	LOOP -rt(loop) -ep AFW -OFFSITE.PWR.REC/-EP.AND.AFW feed.bleed	CD	1.5E-05	2.2E-01
226	LOOP -rt(loop) ep -AFW/EP porv.chall/sbo -porv.reseat/ep SEAL .LOCA OFFSITE.PWR.REC/SEAL.LOCA	CD	1.7E-05	2.9E-01
228	LOOP -rt(loop) ep -AFW/EP porv.chall/sbo -porv.reseat/ep -SEAL .LOCA OFFSITE.PWR.REC/-SEAL.LOCA	CD	5.2E-06	2.9E-01
229	LOOP -rt(loop) ep -AFW/EP porv.chall/sbo porv.reseat/ep	CD	3.2E-06	2.9E-01
239	LOOP -rt(loop) ep AFW/EP	CD	8.2E-05	1.5E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\salem1.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.2E-03	1.0E+00	
LOOP	1.6E-05 > 1.4E-05	5.3E-01 > 5.0E-01	
Branch Model: INITOR			
Initiator Freq:	1.6E-05 > 1.4E-05		
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 2.3E-03	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
afw/atws	4.3E-03	1.0E+00	
AFW/EP	5.0E-02 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.0E-02 > Failed		
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-05	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	5.4E-04	8.9E-01	
SEAL.LOCA	2.7E-01 > 2.3E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.7E-01 > 2.3E-01		
OFFSITE.PWR.REC/-EP.AND.-AFW	2.2E-01 > 1.4E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.2E-01 > 1.4E-01		
OFFSITE.PWR.REC/-EP.AND.AFW	6.7E-02 > 9.9E-04	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.7E-02 > 9.9E-04		
OFFSITE.PWR.REC/SEAL.LOCA	5.7E-01 > 4.8E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.7E-01 > 4.8E-01		
OFFSITE.PWR.REC/-SEAL.LOCA	7.0E-02 > 4.3E-02	1.0E+00	
Branch Model: 1.OF.1			

Train 1 Cond Prob:	7.0E-02 > 4.3E-02		
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.11 LER No. 278/83-002 and -003

Event Description: Transient with HPCI and ESF Bus 23 Inoperable

Date of Event: January 26, 1983

Plant: Peach Bottom 3

B.11.1 Summary

While Peach Bottom Unit 3 was operating at 860 MWe on January 26, 1983, the isolation valve on the high pressure coolant injection (HPCI) system turbine exhaust vacuum breaker failed to fully close during a local leak rate test. Surveillance testing of backup systems was initiated. During the surveillance test, the reactor core isolation cooling system (RCIC) turbine throttle valve motor breaker tripped during the reset capability test for the turbine remote throttle valve. Shutdown was initiated, and RCIC was returned to service within 30 minutes. HPCI was repaired and restored to service late on January 27. On January 27, an HEA relay on the E23 emergency safeguard bus tripped the normal supply and locked out the backup and diesel supplies. The E234 emergency auxiliary load center de-energized and tripped the offgas air ejector, which resulted in reducing the condenser vacuum. Operators manually scrammed the reactor approximately 20 minutes later in anticipation of a low vacuum scram. The conditional core damage probability estimated for the event is 3.4×10^{-5}.

B.11.2 Event Description

While Peach Bottom Unit 3 was operating at 860 MWe on January 26, 1983, the isolation valve for the HPCI system turbine exhaust vacuum breaker failed to fully close during a local leak rate test. A manual valve in the same line was closed, HPCI was declared inoperable, and surveillance testing of redundant backup systems was initiated. The HPCI vacuum breaker isolation valve limit torque operator prevented the valve from fully closing. Examination of the valve internals revealed that the gear train lubricating grease had solidified. The limitorque operator was cleaned and new lubricant was added, and HPCI was restored to service late on January 27. On the 26th, while testing redundant backup systems and when the RCIC surveillance test was performed, the turbine throttle valve motor breaker tripped during the remote throttle valve reset capability test. Shutdown was initiated. Within 30 minutes, the linkage on the RCIC turbine throttle valve motor breaker thermal reset switch was adjusted and the surveillance test was successful on the valve and RCIC. On January 27, an HEA relay on the E23 emergency safeguard bus tripped the normal supply and locked out the backup and diesel supplies. The E234 emergency auxiliary load center de-energized and tripped the offgas air ejector, which resulted in reducing the condenser vacuum. Operators manually scrammed the reactor approximately 20 minutes later in anticipation of a low vacuum scram. It was later determined that the HEA relay trip may have been caused by a defective activating device. All devices were to be checked during the refueling outage.

B.11.3 Additional Event-Related Information

The motor-operated HPCI vacuum breaker isolation valve is normally open to allow the vacuum breakers to break vacuum between the HPCI turbine exhaust line and the suppression pool air space. The valve closes automatically if primary coolant pressure is less than 100 psig and drywell pressure is greater than 2 psig. The closure of the valve isolates the suppression pool air space from the HPCI turbine exhaust line during periods when HPCI is not required, and seals the HPCI exhaust line check valve with suppression pool water after HPCI operation is no longer required. With the vacuum breaker line isolated, initial operation of HPCI would not be affected; however, cycling of the HPCI could lead to failure.

The emergency safeguard bus 23 powers the high pressure service water (HPSW) pump B, residual heat removal (RHR) system pump B, and emergency service water (ESW) pump A. The HPSW system is a four-train system which supplies cooling water to the RHR heat exchangers. One of four pumps supplying one of four RHR heat exchangers is sufficient to properly cool the RHR system. Pumps B or D can be crosstied to the RHR system for another source of injection. The RHR system is a four-train system with four pumps and four heat exchangers. It operates in four modes: low pressure coolant injection (LPCI) mode, suppression pool cooling (SPCOOL) mode, containment spray mode, and shutdown cooling (SDC) mode. LPCI provides coolant makeup to the reactor vessel from the suppression pool. Suppression pool cooling is used to remove heat from the suppression pool whenever the water temperature exceeds 95°F. Containment spray is used in the event of a nuclear system break within the primary containment to prevent excessive containment pressure and temperature by condensing steam and cooling noncondensable gases. Shutdown cooling can be used during normal shutdown and cooldown to remove decay heat once the reactor coolant temperature is low enough that the steam supply pressure is not sufficient to maintain turbine shaft gland seals or vacuum in the main condenser. Successful operation of RHR requires the use of at least one pump and one heat exchanger. The ESW system provides cooling to the pumps and rooms of the emergency core cooling systems (ECCS) and cooling to the emergency diesel generator (EDG) jacket coolers in the event that normal service water is lost. The ESW system has two main pump trains, A and B, which provide cooling water to the various systems. One of the two pump trains is sufficient to supply cooling to all ECCS and EDG jacket coolers.

B.11.4 Modeling Assumptions

Since the HPCI valve was not fixed until late on January 27, HPCI was assumed to be inoperable at the time of the trip on January 27. One train of RHR and LPCI was unavailable due to the loss of ESF bus 23. It was assumed that since the bus failure was due to the HEA relay for bus 23 only, the other ESF buses would not be affected by the failure of bus 23. Since ESW and HPSW had redundant pumps which were not inoperable due to the loss of bus 23 at the time of the trip and normal service water was operable, HPSW and service water cooling were assumed to be functioning. Since the loss of bus 23 made HPSW pump B unavailable, HPSW injection had only one pump (pump D) available for injection. RCIC was inoperable for less than thirty minutes during the time the HPCI vacuum breaker line was isolated. Since RCIC was inoperable for such a short time over the two day period of the event, it was assumed operable at the time of the trip. The loss of condenser vacuum would result in the closure of the main steam isolation valves (MSIVs), thus making the power conversion (PCS) and feedwater (FW) systems unavailable. Thus, the event was modeled as a transient with HPCI failed, one train of RHR and LPCI unavailable, PCS and FW unavailable, and the HPSW injection [(RHRSW(INJ))] probability modified to reflect the unavailability of one of the pumps.

B.11.5 Analysis Results

The estimated conditional core damage probability for this event is 3.4×10^{-5}. The dominant sequence involved the observed transient with failure of PCS and RHR and is highlighted on the event tree in Figure B.11.1. A slightly less probable sequence involves failure of two safety relief valves (SRVs) to close following lift, failure of HPCI, and failure of the automatic depressurization system (ADS).

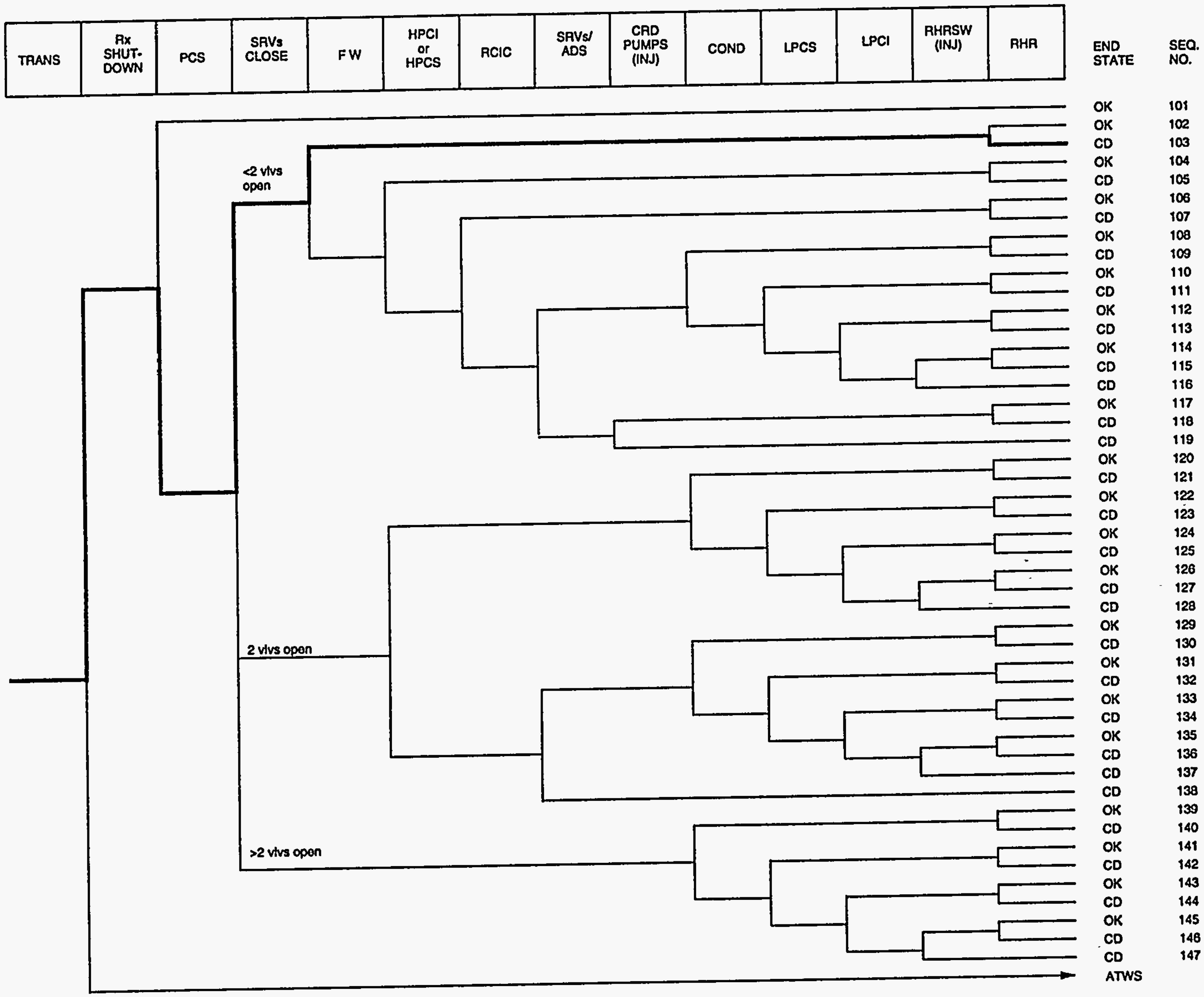

Figure B.11.1 Dominant core damage sequence for LER 278/83-002 and -003

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 278/83-002
Event Description: Transient with HPCI and ESF bus 23 inop
Event Date: January 26. 1983
Plant: Peach Bottom 3

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.4E-05
Total	3.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	1.2E-05	1.8E-02
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.1E-05	4.9E-01
107	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI -rcic RHR.AND.PC S.NREC	CD	4.2E-06	6.6E-03
119	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI rcic srv.ads c rd(inj)	CD	2.5E-06	1.2E-01
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -HPCI RHR.AND.PCS.NREC	CD	1.9E-06	2.9E-03
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	1.2E-05	1.8E-02
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -HPCI RHR.AND.PCS.NREC	CD	1.9E-06	2.9E-03
107	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI -rcic RHR.AND.PC S.NREC	CD	4.2E-06	6.6E-03
119	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI rcic srv.ads c rd(inj)	CD	2.5E-06	1.2E-01
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.1E-05	4.9E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\peach3.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	4.8E-04	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
HPCI	2.9E-02 > 1.0E+00	7.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-02 > 1.0E+00		
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
LPCI	1.1E-03 > 1.3E-03	1.0E+00	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
Serial Component Prob:	1.0E-03		
RHRSW(INJ)	2.0E-02 > 3.0E-02 **	1.0E+00	1.0E-02
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	2.0E-02		
RHR	1.5E-04 > 3.0E-04	1.6E-02 > 5.4E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
RHR.AND.PCS.NREC	1.5E-04 > 3.0E-04	8.3E-03 > 2.8E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		

rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 2.3E-03	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
Serial Component Prob:	2.0E-03		
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	7.5E-03	8.7E-01	
ep.rec	6.1E-02	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.12 LER No. 278/83-009

Event Description:	Two EDGs inoperable
Date of Event:	September 8, 1983
Plant:	Peach Bottom 3

B.12.1 Summary

During low-power operation on September 8, 1983, with one emergency diesel generator (EDG) out for maintenance, surveillance tests were being performed on the other EDGs. During the surveillance test, the breaker between EDG E-1 and bus E-13 failed to close. The estimated increase in core damage probability, or importance, over the duration of this event is 3.5×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 5.6×10^{-7}, resulting in an estimated conditional core damage probability (CCDP) of 3.5×10^{-5}.

B.12.2 Event Description

During low-power operation on September 8, 1983, with one emergency diesel generator out for maintenance, surveillance tests were being performed on the other EDGs. During the surveillance test, the breaker between EDG E-1 and bus E-13 failed to close. The breaker failure in conjunction with the inoperable EDG which was out for maintenance required the unit to be in cold shutdown within 24 hours. The breaker was removed, and immediate testing failed to reveal the problem. The breaker was successfully tested for operability several times and was returned to service within 30 minutes.

B.12.3 Additional Event-Related Information

Peach Bottom Units 2 and 3 receive offsite power from two separate sources. If both offsite sources are lost, auxiliary power is supplied to both Unit 2 and Unit 3 from four onsite EDGs which are shared between the units. Each EDG automatically starts, but requires battery power to do so. Each EDG starts automatically on total loss of offsite power, low reactor water level, or high drywell pressure coincident with low reactor pressure. Each diesel generator can be manually started and loaded locally.

B.12.4 Modeling Assumptions

Since it is unknown how long the first EDG was out for maintenance and how long the breaker was degraded before it was discovered, it was assumed that both EDGs were inoperable for the maximum allowable period allowed by the Technical Specifications (one EDG can be inoperable for up to seven days before requiring the unit to shut down). Thus, the event was modeled as the unavailability of two EDGs [(two trains of emergency power (EP)] during a postulated loss of offsite power (LOOP) for a duration of seven days. The third train of EP was set to unavailable to reflect the EDG which was out for maintenance. The first train of EP was set to failed to reflect the failure of the EDG breaker to close (it was assumed that this failure could have also occurred on the remaining trains of emergency power). The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event.

B.12.5 Analysis Results

The estimated increase in core damage probability over the duration of this event is 3.5×10^{-5}. The base-case CDP (not shown in calculation) is 5.6×10^{-7}, resulting in an estimated CCDP of 3.5×10^{-5}. The dominant sequence involved a LOOP with successful reactor shutdown, failure of emergency power (station blackout), and failure to recover offsite power prior to battery depletion, and is highlighted on the event tree in Figure B.12.1.

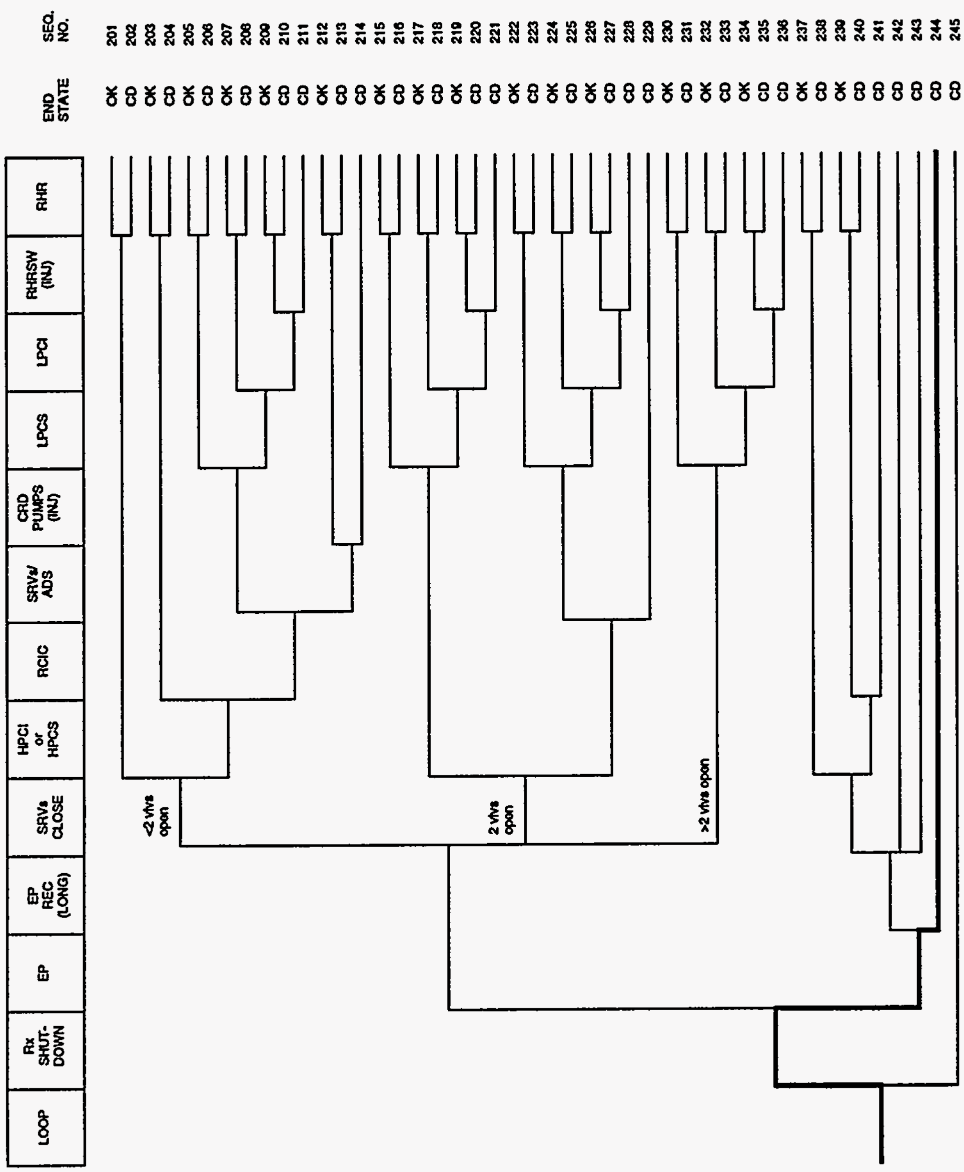

Figure B.12.1 Dominant core damage sequence for LER 278/83-009

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 278/83-009
Event Description: Two EDGs inoperable
Event Date: September 8. 1983
Plant: Peach Bottom 3

UNAVAILABILITY. DURATION= 168

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	8.1E-02
LOOP	6.4E-04
LOCA	3.7E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	0.0E+00
LOOP	3.5E-05
LOCA	0.0E+00
Total	3.5E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

Sequence	End State	Prob	N Rec**
244 loop -rx.shutdown EP ep.rec	CD	3.4E-05	2.1E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

Sequence	End State	Prob	N Rec**
244 loop -rx.shutdown EP ep.rec	CD	3.4E-05	2.1E-01

** non-recovery credit for edited case

Note: For unavailabilities. conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\peach3.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	4.8E-04	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	4.6E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.5E-04	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
EP	7.5E-03 > 1.0E+00	8.7E-01	
Branch Model: 2.OF.3			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02		
Train 3 Cond Prob:	1.9E-01 > Unavailable		
ep.rec	6.1E-02	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.13 LER No. 281/83-055

Event Description: Trip with AFW Pump Inoperable

Date of Event: November 18, 1983

Plant: Surry 2

B.13.1 Summary

Auxiliary feedwater (AFW) pump B was found failed due to steam binding on November 18, 1983. On November 20, 1982, it was found failed due to a failed lube oil cooler. Surry 2 experienced a trip on November 16. The conditional core damage probability estimated for this event is 3.5×10^{-5}.

B.13.2 Event Description

Surry unit 2 was operating at full power on November 18, 1983, when the B motor-driven AFW pump failed to provide flow when started. An investigation determined that a leaking check valve was allowing backflow into the pump, which became steam bound. A similar problem was experienced by the pump on December 6, 1983. The turbine-driven AFW pump at Surry experienced a steam-binding problem on November 20; however, the relevant licensee event report indicates that the pump had been operable previously. In addition, AFW pump B was found to have a failed lube oil cooler during maintenance efforts on November 20. There was a reactor trip reported on November 16, 1983.

B.13.3 Additional Event-Related Information

None.

B.13.4 Modeling Assumptions

As the problems with motor-driven auxiliary feedwater pump (MDAFWP) B reported on November 18 and 20 were believed to have been latent during the trip on November 16, this event was modeled as a trip with that AFW pump inoperable. It was assumed that failure of the other AFW pumps from the same cause was possible. Although the specific failure discovered was not reported to be present in redundant systems, the potential for common cause failure was believed to exist. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event. This was implemented in the model by setting the serial component failure probability equal to the conditional probability that the remaining pumps would fail, given failure of pump B (0.1 x 0.3). Since failure of either remaining pump would have rendered AFW inoperable for ATWS mitigation, the failure probability of AFW during ATWS was calculated as 0.1 + 0.1 = 0.2.

B.13.5 Analysis Results

The conditional core damage probability estimated for this event is 3.5×10^{-5}. The dominant sequence for this event, highlighted on the event tree in Figure B.13.1, involves a transient with reactor trip success, failure of main and auxiliary feedwater, and failure of feed-and-bleed cooling.

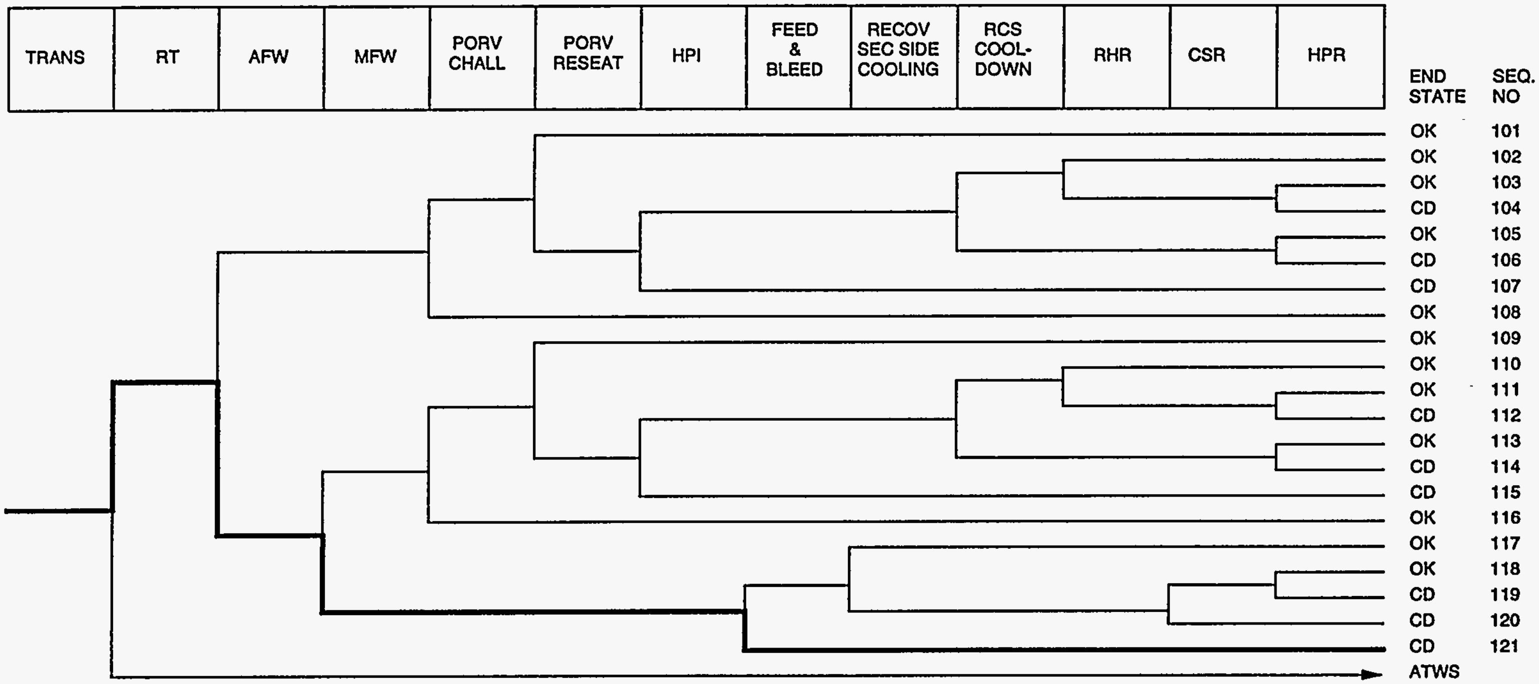

Figure B.13.1 Dominant core damage sequence for LER 281/83-055

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 281/83-055
Event Description: Trip with AFW pump inop
Event Date: November 18, 1983
Plant: Surry 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.5E-05
Total	3.5E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
121	trans -rt AFW mfw feed.bleed	CD	2.7E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	5.6E-06	1.0E-01
119	trans -rt AFW mfw -feed.bleed recov.sec.cool -csr hpr	CD	8.8E-07	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW mfw -feed.bleed recov.sec.cool -csr hpr	CD	8.8E-07	1.5E-01
121	trans -rt AFW mfw feed.bleed	CD	2.7E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	5.6E-06	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\pwra8283.cmp
BRANCH MODEL: d:\asp\models\surry2.82
PROBABILITY FILE: d:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.9E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	

sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 3.1E-02	4.5E-01	
Branch Model: 1.OF.3+ser			
Event Identifier: 281/83-055			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > 1.0E+00		
Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04 > 3.0E-02		
AFW/ATWS	4.3E-03 > 2.0E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > 2.0E-01		
afw/ep	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.5E-03	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
csr	7.5E-04	1.0E+00	
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.14 LER No. 287/83-011

Event Description: Loss of Feedwater Transient with One EFW Pump Inoperable

Date of Event: October 13, 1983

Plant: Oconee 3

B.14.1 Summary

On October 13, 1983, following a loss of main feedwater, the 3B motor-driven emergency feedwater (EFW) pump started. Approximately 30 minutes later, it was discovered that there was no low pressure service water (LPSW) flow through the 3B EFW pump due to a failed solenoid valve. The pump was declared inoperable. The estimated conditional core damage probability for this event is 3.2×10^{-5}.

B.14.2 Event Description

On October 13, 1983, following a loss of main feedwater (MFW) caused by a loss of instrument air pressure, the 3B motor-driven EFW pump emergency started. Approximately 45 minutes later, it was discovered that no LPSW was flowing through the 3B EFW pump. The pump was shut down and declared inoperable. Investigation revealed that the solenoid that operates 3B EFW motor cooler outlet valve 3SV-203 had failed. The outlet valve did not open when the pump was started. The plunger of valve 3SV-203 was found to be stuck. The valve was replaced and the 3B EFW pump was declared operable the next day. To help identify low LPSW flow to the EFW pumps, the computer points for flow were to be changed to provide an alarm if flow is low while the pumps are running.

On October 18, 1983, an additional problem was found related to the 3B EFW pump. While operating at 100% power, a performance test on 3B EFW pump was performed. Feedwater valve 3FDW-382 was closed per procedure. Upon completion of the test, the valve was reopened by placing the valve position switch to its normally open position in the control room. The control panel indicated the valve was open when it was only partially open. Visual inspection of the valve during the next shift revealed that the valve was only 10% open. Attempts to close the valve from the control room failed, and the valve and the pump were declared inoperable. The direct cause was unknown, but the valve was successfully cycled from the control room approximately 40 minutes later. This event was reported in LER 287/83-012.

B.14.3 Additional Event-Related Information

Oconee 3 has a three-train EFW system. Two trains have motor-driven EFW pumps (3A and 3B). Each motor-driven pump is aligned to one steam generator. The third train has a turbine-driven pump which is aligned to both steam generators. In the event that main feedwater is lost, the EFW pumps provide secondary side cooling to the steam generators. One pump delivering flow to one steam generator provides sufficient secondary side cooling in the event that main feedwater is lost and the reactor has tripped. LPSW provides motor cooling to both motor-driven EFW pumps and turbine jacket cooling and bearing oil cooling for the turbine-driven EFW pump.

B.14.4 Modeling Assumptions

The event was modeled as a transient with loss of MFW and one motor-driven EFW pump inoperable because of the loss of LPSW cooling. The valve failure reported in LER 287/83-012 was assumed not to further impact the EFW system during the loss of feedwater. MFW was set to failed to reflect the loss of feedwater. Because of the loss of instrument air pressure, MFW was assumed to be nonrecoverable. In revising the EFW system failure probability, it was assumed that the type of valve fault observed on pump 3B could also have affected the motor-driven pump 3A. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event.

B.14.5 Analysis Results

The estimated conditional core damage probability for this event is 3.2×10^{-5}. The dominant sequence, highlighted on the event tree in Figure B.14.1, involved a plant trip with the failure of EFW, the failure of MFW and the failure of feed and bleed.

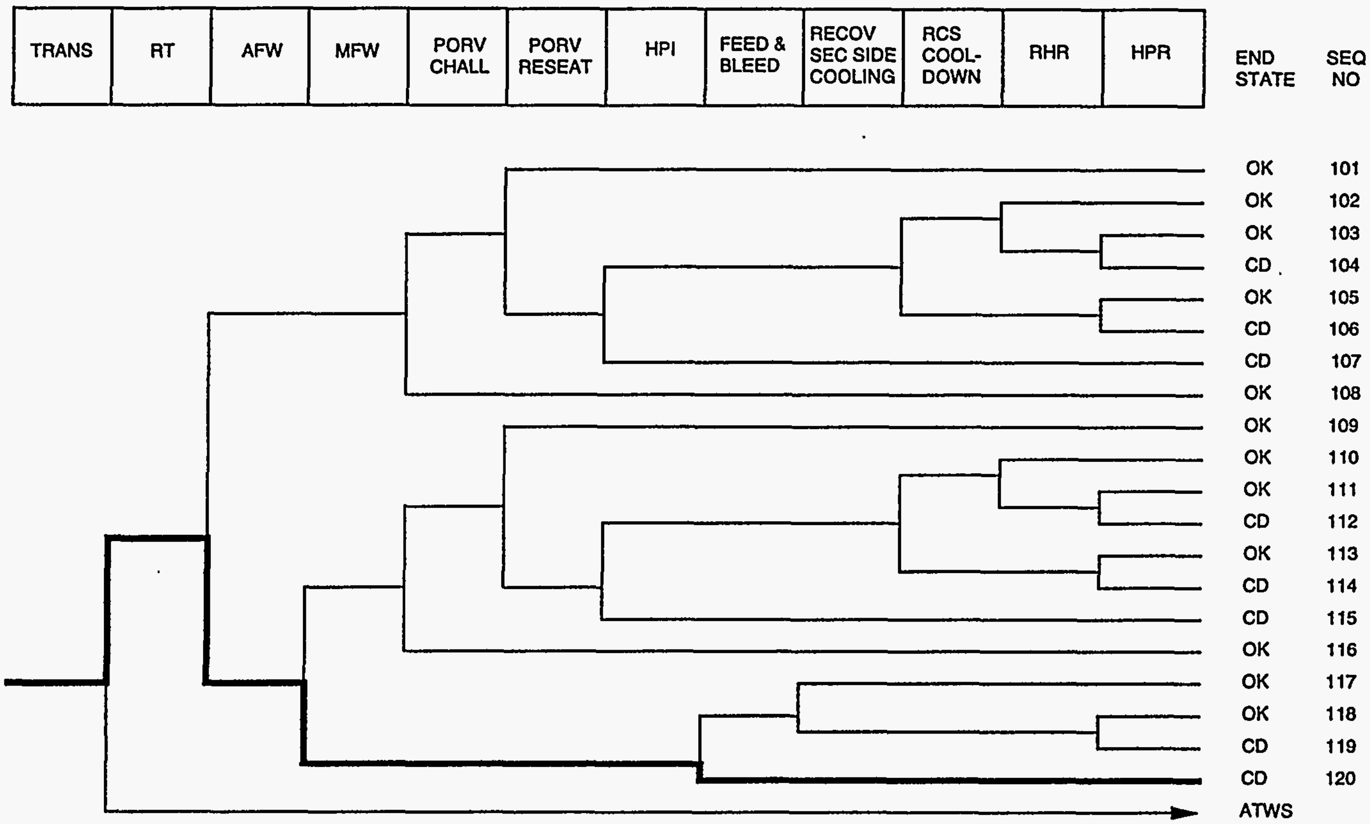

Figure B.14.1 Dominant core damage sequence for LER 287/83-011.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 287/83-011
Event Description: LOFW transient with one EFW pump inop
Event Date: October 13. 1983
Plant: Oconee 3

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.2E-05
Total	3.2E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW MFW feed.bleed	CD	2.4E-05	4.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	4.2E-06	1.0E-01
119	trans -rt AFW MFW -feed.bleed recov.sec.cool hpr	CD	2.3E-06	4.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW MFW -feed.bleed recov.sec.cool hpr	CD	2.3E-06	4.5E-01
120	trans -rt AFW MFW feed.bleed	CD	2.4E-05	4.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	4.2E-06	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\pwrb8283.cmp
BRANCH MODEL: d:\asp\models\oconee3.82
PROBABILITY FILE: d:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	2.4E-04	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	2.4E-06	5.4E-01	

sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 5.3E-03	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04		
AFW/ATWS	4.3E-03 > 1.5E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > 1.5E-01		
afw/ep	5.0E-02	3.4E-01	
MFW	2.0E-01 > 1.0E+00	3.4E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-01 > 1.0E+00		
porv.chall	8.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	1.0E-02	1.1E-02	
porv.reseat/ep	1.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	3.0E-04	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	5.7E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	6.0E-04	8.9E-01	
seal.loca	0.0E+00	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	1.6E-01	1.0E+00	
offsite.pwr.rec/seal.loca	0.0E+00	1.0E+00	
offsite.pwr.rec/-seal.loca	4.5E-01	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.15 LER No. 293/82-024 and -023

Event Description: Scram and HPCI Failure

Date of Event: August 13, 1982

Plant: Pilgrim

B.15.1 Summary

During recovery from a scram on August 13, 1982 (LER 293/82-023), the high-pressure coolant injection (HPCI) system tripped after 5 minutes owing to high reactor water level. After restarting the HPCI pump, attempts to bring it past idle speed were unsuccessful. The estimated conditional core damage probability for the event is 2.9×10^{-5}.

B.15.2 Event Description

On August 13, 1982, a scram occurred when a removable hand rail fell against main steam hi-flow instrumentation and generated a containment isolation signal (LER 293/82-023). During recovery from the scram, the HPCI system tripped after 5 minutes, owing to high reactor water level. After restarting the HPCI pump, attempts to bring it past idle speed were unsuccessful. Eleven manual safety relief valve (SRV) actuations were required to control pressure. Investigation of the HPCI system revealed that the HPCI gland seal condenser gasket had failed, causing wetting of the HPCI control circuitry. The control circuits were dried and calibrated, and gasket repair was accomplished.

B.15.3 Additional Event-Related Information

The HPCI system consists of a single turbine-driven pump that can provide primary coolant makeup at a rate of 4250 gpm. The HPCI pump is provided with two suction sources. The primary source is the condensate storage tank (CST), with the suppression pool providing the secondary source. These are interlocked to ensure that only one source is aligned at a time. The system is designed to swap from the CST to the suppression pool on low CST or high suppression pool level.

B.15.4 Modeling Assumptions

This event was modeled as a transient initiator with the power conversion system (PCS) unavailable and HPCI failed due to control circuit wetting and not recoverable. The PCS system was assumed unavailable because a containment isolation signal was generated when the hand rail fell against the main steam hi-flow instrumentation; this signal is expected to have closed the main isolation valves (MSIVs).

The main feedwater system is motor driven at Pilgrim, and was assumed to be available following closure of the MSIVs.

The nonrecovery probability for sequences involving residual heat removal (RHR) or PCS recovery was revised to reflect the MSIV isolation (see Appendix A).

B.15.5 Analysis Results

The estimated conditional core damage probability for the event is 2.9×10^{-5}. The dominant sequence, highlighted on the event tree in Figure B.15.1, involves a transient initiator followed by successful reactor shutdown, failure of the power conversion system, failure of two SRVs to close, HPCI unavailability and automatic depressurization system (ADS) failure.

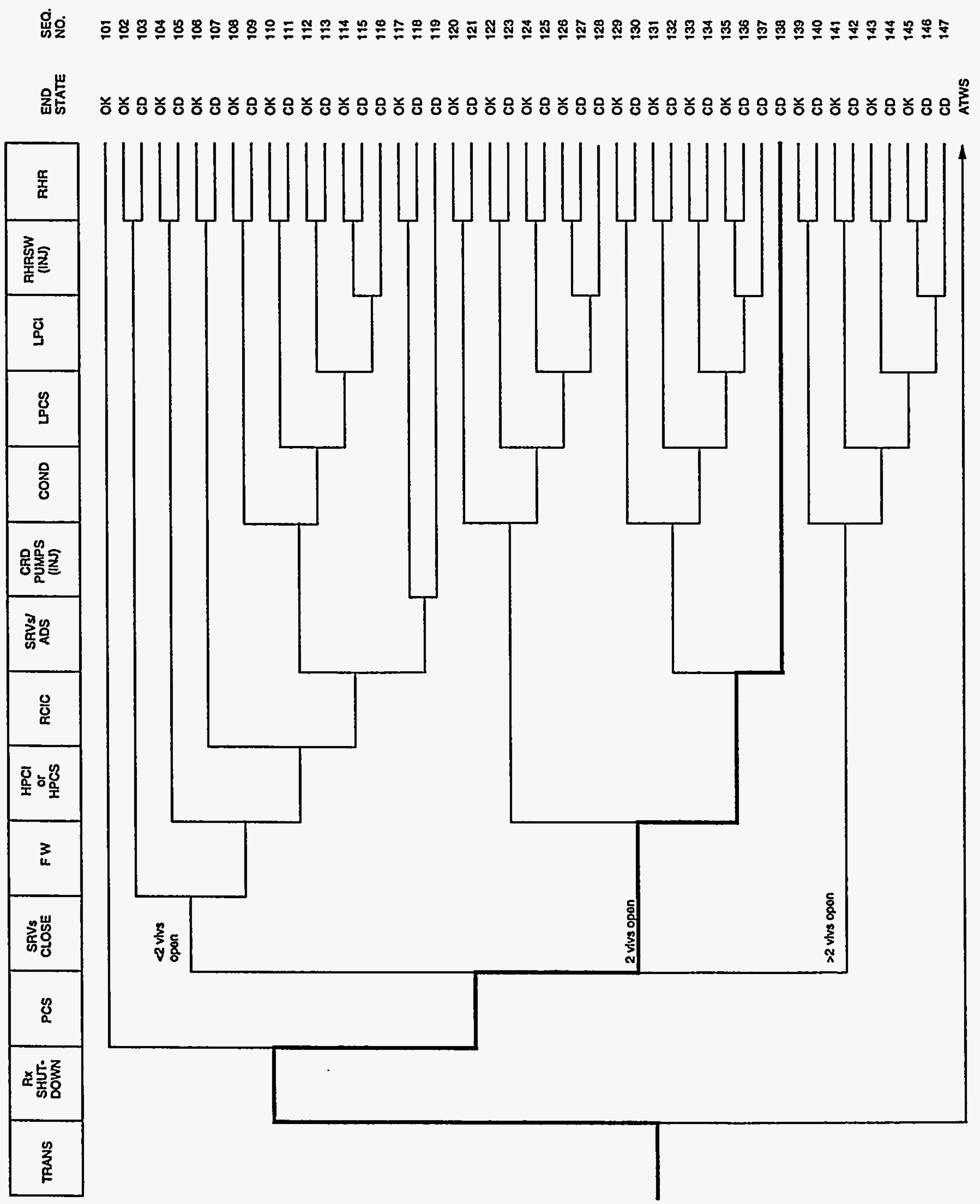

Figure B.15.1 Dominant core damage sequence for LER 293/82-024 and -023

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 293/82-024
Event Description: Scram and HPCI failure
Event Date: August 13. 1982
Plant: Pilgrim

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	2.9E-05
Total	2.9E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.6E-05	7.0E-01
103	trans -rx.shutdown PCS srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	9.0E-06	2.5E-04
119	trans -rx.shutdown PCS srv.ftc.<2 mfw HPCI rcic srv.ads c rd(inj)	CD	1.0E-06	1.7E-01
107	trans -rx.shutdown PCS srv.ftc.<2 mfw HPCI -rcic RHR.AND.PC S.NREC	CD	9.5E-07	9.1E-05
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	9.0E-06	2.5E-04
107	trans -rx.shutdown PCS srv.ftc.<2 mfw HPCI -rcic RHR.AND.PC S.NREC	CD	9.5E-07	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 mfw HPCI rcic srv.ads c rd(inj)	CD	1.0E-06	1.7E-01
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.6E-05	7.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\pilgrim.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.2E-03	1.0E+00	
loop	2.0E-05	4.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
HPCI	2.9E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-02 > 1.0E+00		
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	3.1E-02	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.16 LER No. 293/83-007

Event Description: LOOP During Shutdown

Date of Event: February 13, 1983

Plant: Pilgrim

B.16.1 Summary

On February 13, 1983, a loss of offsite power (LOOP) occurred following a load rejection and scram caused by salt buildup on insulators in the switchyard. The LOOP occurred during the process of washing down portions of the switchyard to remove the salt deposits that had accumulated during a heavy ocean storm. The conditional core damage probability estimated for the event is 9.7×10^{-5}.

B.16.2 Event Description

On February 13, 1983, during a shutdown condition resulting from a load reject, a LOOP occurred. The load reject occurred when a heavy ocean storm caused a salt buildup on switchyard insulators, creating arcing to ground and the subsequent opening of breakers. During the process of washing down the isolated portion of the switchyard, melting ice and salt deposits on the remaining inservice portion of the switchyard created a separate ground that caused the inservice breakers to open, resulting in a LOOP. The emergency diesel generators (EDGs) started and other safety-related equipment functioned as designed. A secondary offsite power source was available as backup to the EDGs. After completion of the washdown, power was restored to the startup transformer and preparations for startup commenced.

B.16.3 Additional Event-Related Information

Pilgrim has two safety-related 4160 V ac buses. Both of these buses can be powered from the unit auxiliary transformer (UAT) or the startup transformer (SUT). Upon loss of the UAT following a reactor trip, the safety-related buses are transferred to the SUT. If the SUT is lost, the EDGs are started to power safety-related loads. If an EDG fails, the 23-kV secondary offsite source automatically powers the bus.

B.16.4 Modeling Assumptions

This event was modeled as a severe weather-induced loss of offsite power with all equipment available to respond to the event. The probabilities of failing to recover offsite power in the short term and before battery depletion were modified using the models described in *Revised LOOP Frequency and PWR Seal LOCA Models,* ORNL/CRC/LTR-89/11, August 1989.

The 23-kV line is unusual because it is used following the failure of the EDGs to start. The Pilgrim IPE indicates that 18 failures of the 345-kV lines occurred between September 13, 1975, and February 21, 1989. Of these 18 LOOPs, 7 were caused by severe weather. In three of these severe-weather-induced LOOPs, the 23-kV line was also lost. Therefore, the conditional probability that the 23-kV line is lost, given that the 235-kV lines are lost due to a severe-weather-induced LOOP, was set to 0.43 (3/7). Because the 23-kV line would close in automatically following the failure of the EDGs, the EDG nonrecovery value was modified to include the probability that the 23-kV line would be unavailable. Breaker failures and control system failures were assumed to be not significant, given the high unavailability of the line under these conditions.

The probabilities of failing to recover offsite power in the short term and before battery depletion were set to 0.9 and 5.5E-2, respectively.

B.16.5 Analysis Results

The estimated conditional core damage probability for the severe weather-induced LOOP is 9.7×10^{-5}. The dominant sequence, highlighted on the event tree in Figure B.16.1, involves a LOOP initiating event, successful reactor shutdown, failure of the emergency power system, and failure to restore offsite power before battery depletion.

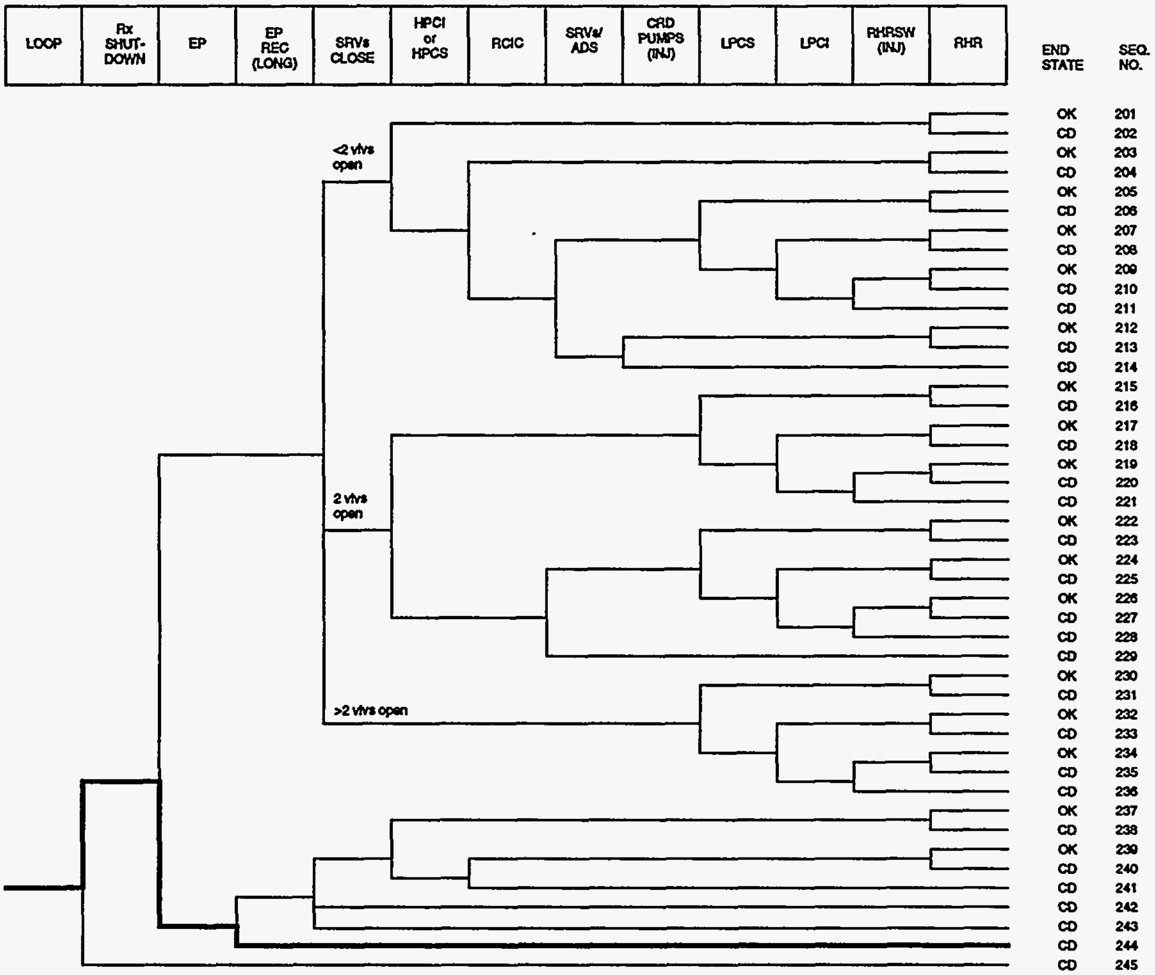

Figure B.16.1 Dominant core damage sequence for LER 293/83-007

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 293/83-007
Event Description: Loop during shutdown
Event Date: February 13, 1983
Plant: Pilgrim

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	9.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	9.7E-05
Total	9.7E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
244	LOOP -rx.shutdown EP EP.REC	CD	5.2E-05	3.3E-01
245	LOOP rx.shutdown	CD	3.1E-05	9.0E-02
202	LOOP -rx.shutdown -EP srv.ftc.<2 -hpci rhr	CD	1.1E-05	1.4E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
202	LOOP -rx.shutdown -EP srv.ftc.<2 -hpci rhr	CD	1.1E-05	1.4E-02
244	LOOP -rx.shutdown EP EP.REC	CD	5.2E-05	3.3E-01
245	LOOP rx.shutdown	CD	3.1E-05	9.0E-02

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\pilgrim.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail

trans	1.2E-03	1.0E+00	
LOOP	2.0E-05 > 2.0E-05	4.3E-01 > 9.0E-01	
Branch Model: INITOR			
Initiator Freq:	2.0E-05		
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.5E-04	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
EP	2.9E-03 > 2.9E-03	8.7E-01 > 3.7E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02		
EP.REC	3.1E-02 > 5.5E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	3.1E-02 > 5.5E-02		
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.17 LER No. 302/82-007

Event Description: Two EDGs Inoperable

Date of Event: January 23, 1982

Plant: Crystal River 3

B.17.1 Summary

On January 23, 1982, during normal operation, the starting air pressure for emergency diesel generator (EDG) B was too low for automatic start and the air pressure low alarm did not alarm. On January 25, EDG A was fast started but did not excite and maintain voltage. Thus, both EDGs were inoperable. The estimated increase in core damage probability, or importance, over the duration of this event is 2.8×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 3.4×10^{-7}, resulting in an estimated conditional core damge probability (CCDP) of 2.8×10^{-5}.

B.17.2 Event Description

At 0700 on January 23, 1982, during normal operation, the starting air pressure for EDG B was found to be too low for automatic start and the air pressure low alarm did not alarm. The air tank blowdown valve EGV-16 was not completely shut. The valve was shut and EDG B was fast started and restored to operation at 0705. Maintenance was initiated on the alarm circuit. Investigation revealed that a bad circuit card had led to the alarm malfunction. The circuit card was replaced and the alarm circuit tested satisfactorily. At 0714 on January 25, EDG A was fast started but did not excite and maintain voltage. Maintenance was initiated on the excitation and voltage control circuit but no problem could be found. EDG A was restored at 1307 the same day.

B.17.3 Additional Event-Related Information

Crystal River 3 has two EDGs which provide power to two engineered safeguards (ESF) buses in the event of a loss of normal power supply to the ESF buses. Both EDGs automatically start on either a low ESF bus voltage or an ESF actuation signal.

B.17.4 Modeling Assumptions

This event is modeled as an unavailability of both EDGs for a period of 24 hours given a postulated LOOP. The cause for the failure of EDG A was never determined, so EDG A was assumed to be failed for a period of 15 days (half the surveillance period) prior to the discovery of the failure. EDG B was assumed to be failed for a period of 24 hours (approximately two shifts) since the diesel generator rooms are frequently checked, the leaking air would make a noticeable noise, and the air compressor would run continuously. Thus, both trains were assumed inoperable for approximately 24 hours. Both trains of emergency power (EP) were set to failed and the recovery probability was modified to reflect the ability of the operators to recover the EDGs locally (p=0.55, see Appendix A of this report).

B.17.5 Analysis Results

The estimated increase in core damage probability over the duration of this event is 2.8×10^{-5}. The base-case CDP (not shown in calculation) is 3.4×10^{-7}, resulting in an estimated CCDP of 2.8×10^{5}. The dominant sequence involves a successful reactor shutdown following a postulated LOOP, failure of emergency power (station blackout) and failure to recover offsite power prior to battery depletion and is highlighted in the event tree in Figure B.17.1.

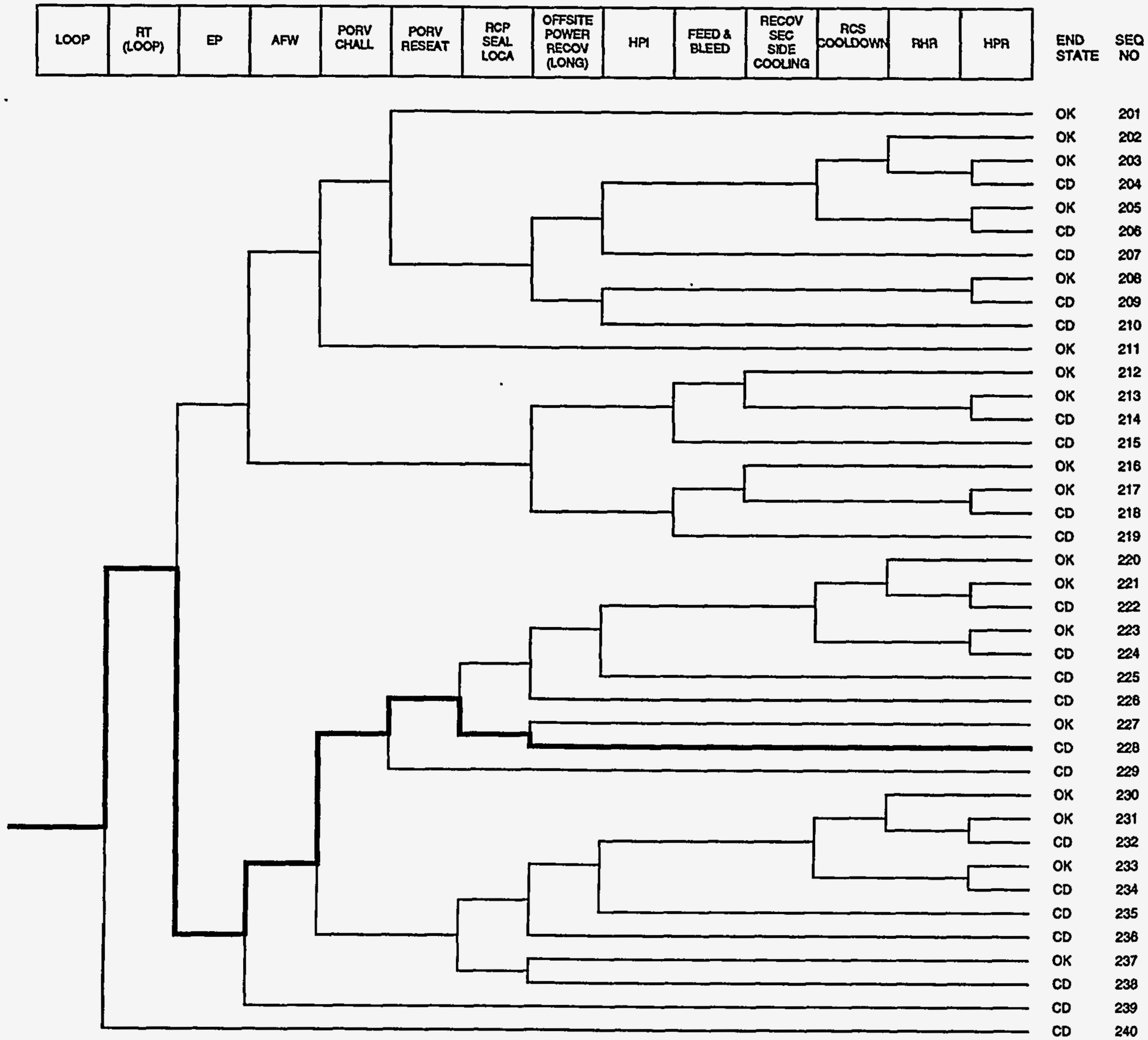

Figure B.17.1 Dominant core damage sequence for LER 302/82-007

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 302/82-007
Event Description: Two EDGs inoperable
Event Date: January 23, 1982
Plant: Crystal River 3

UNAVAILABILITY, DURATION= 24

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	1.4E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	2.8E-05
Total	2.8E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
228	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep -seal .loca offsite.pwr.rec/-seal.loca	CD	2.2E-05	1.8E-01
226	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep seal .loca offsite.pwr.rec/seal.loca	CD	3.4E-06	1.8E-01
239	loop -rt(loop) EP afw/ep	CD	1.3E-06	6.2E-02
229	loop -rt(loop) EP -afw/ep porv.chall/sbo porv.reseat/ep	CD	7.6E-07	1.8E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
226	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep seal .loca offsite.pwr.rec/seal.loca	CD	3.4E-06	1.8E-01
228	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep -seal .loca offsite.pwr.rec/-seal.loca	CD	2.2E-05	1.8E-01
229	loop -rt(loop) EP -afw/ep porv.chall/sbo porv.reseat/ep	CD	7.6E-07	1.8E-01
239	loop -rt(loop) EP afw/ep	CD	1.3E-06	6.2E-02

** non-recovery credit for edited case

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp

BRANCH MODEL: c:\aspcode\models\criver3.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.8E-03	1.0E+00	
loop	1.8E-05	3.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	1.3E-03	4.5E-01	
afw/atws	7.0E-02	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
porv.chall	8.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	1.0E-02	1.1E-02	
porv.reseat/ep	1.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	3.0E-04	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
EP	2.9E-03 > 1.0E+00	8.9E-01 > 5.5E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02 > Failed		
seal.loca	6.0E-02	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	4.3E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	2.1E-01	1.0E+00	
offsite.pwr.rec/seal.loca	7.6E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	3.1E-01	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.18 LER No. 304/82-009

Event Description: Unavailability of Two Motor-Driven AFW Pumps

Date of Event: April 9, 1982

Plant: Zion 2

B.18.1 Summary

On April 9, 1982, while monthly periodic testing of the auxiliary feedwater (AFW) pumps was being performed, service water valves 2SVSW131 and 2SVSW130 failed to open to supply cooling to motor-driven AFW pumps 2B and 2C oil coolers upon the start of the pumps. Accumulation of silt from the service water system caused the valves to stick closed. The event was modeled as an unavailability of both AFW motor-driven pumps. The estimated increase in core damage probability, or importance, over the duration of this event is 3.4×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 1.9×10^{-6}, resulting in an estimated conditional core damage probability (CCDP) of 3.6×10^{-5}.

B.18.2 Event Description

On April 9, 1982, during monthly periodic testing of the AFW pumps, service water valves 2SVSW131 and 2SVSW130 failed to open to supply cooling to motor-driven AFW pumps 2B and 2C oil coolers upon the start of the pumps. After valve 2SVSW130 for pump 2C was tapped, the valve opened and service water flowed to the 2C pump oil cooler. The AFW turbine-driven pump was operable during the time of this event. Investigation revealed that years of accumulation of silt from the service water system caused the valves to stick closed. The valves were cleaned and opened properly when tested again.

B.18.3 Additional Event-Related Information

The AFW system of Zion 2 has three pumps. Two pumps are motor driven and one is turbine driven. Each motor-driven pump can supply cooling to two of four steam generators. The single turbine-driven pump is capable of providing cooling to all four steam generators and can provide cooling in the event of a loss of offsite power as well as a loss of emergency power.

B.18.4 Modeling Assumptions

This event was modeled as an unavailability of both motor-driven AFW pumps. Since the length of time in which both service water valves were stuck closed is unknown, the duration of the event was taken to be half the surveillance period of the AFW pumps (15 days or 360 hours). All four initiating events were examined, each with its default value as the initiating event frequency.

B.18.5 Analysis Results

The estimated increase in core damage probability over the duration of this event is 3.4×10^{-5}. The base-case CDP (not shown in calculation) is 1.9×10^{-6}, resulting in an estimated CCDP of 3.6×10^{-5}. The dominant sequence involved a postulated transient with a successful reactor trip, failure of AFW, failure of main feedwater (MFW), and failure of feed and bleed and is shown on the event tree in Figure B.18.1.

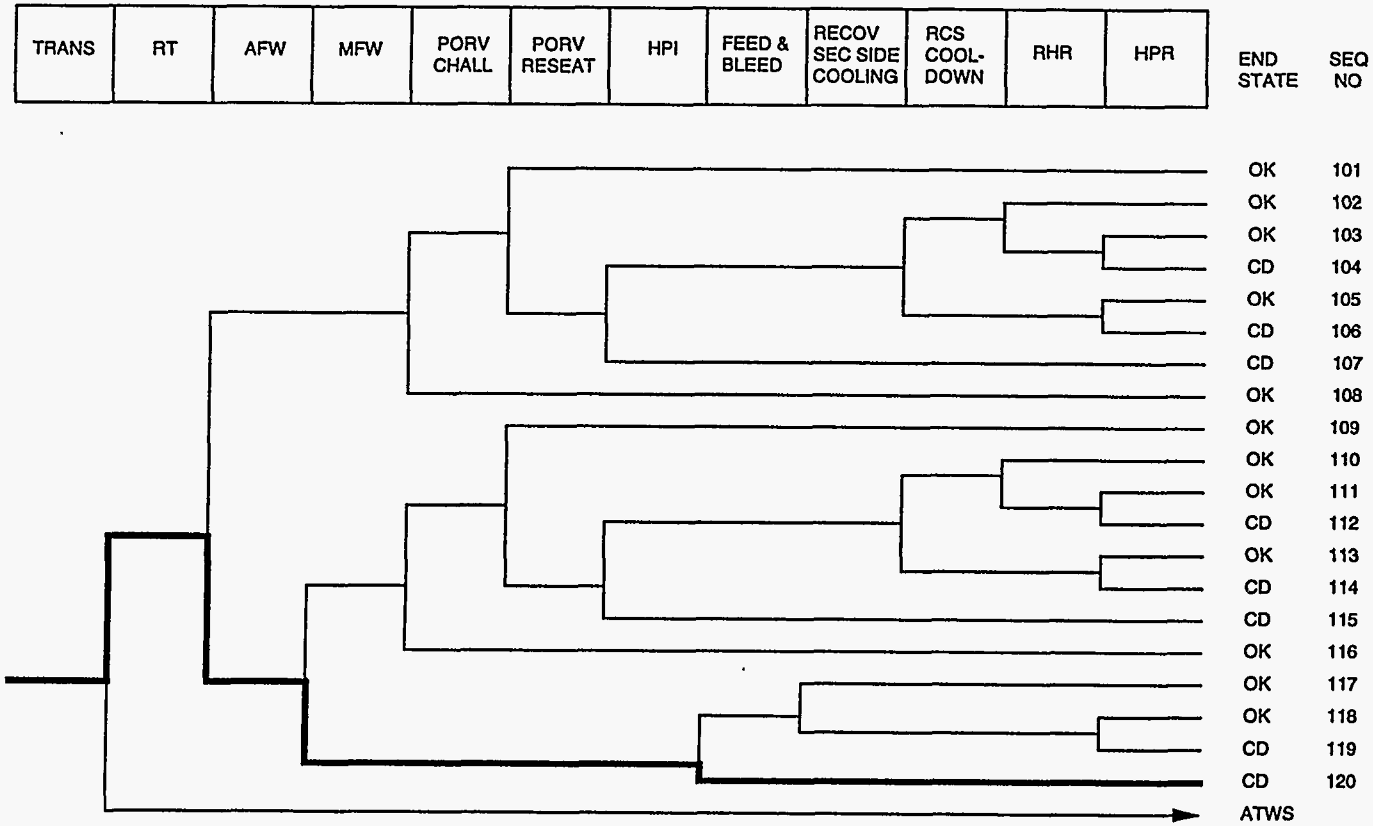

Figure B.18.1 Dominant core damage sequence for 304/82-009

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 304/82-009
Event Description: Unavailability of two motor-driven AFW pumps
Event Date: April 19, 1982
Plant: Zion 2

UNAVAILABILITY, DURATION= 360

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	5.4E-01
LOOP	3.1E-03
LOCA	4.7E-04
SGTR	5.9E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.1E-05
LOOP	2.1E-06
LOCA	2.2E-08
SGTR	9.1E-07
Total	3.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW mfw feed.bleed	CD	1.9E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	1.2E-05	1.0E-01
215	loop -rt(loop) -ep AFW -offsite.pwr.rec/-ep.and.afw feed.bleed	CD	1.9E-06	2.4E-01
411	sgtr -rt AFW mfw	CD	9.1E-07	1.5E-01
119	trans -rt AFW mfw -feed.bleed recov.sec.cool hpr	CD	6.2E-07	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW mfw -feed.bleed recov.sec.cool hpr	CD	6.2E-07	1.5E-01
120	trans -rt AFW mfw feed.bleed	CD	1.9E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	1.2E-05	1.0E-01
215	loop -rt(loop) -ep AFW -offsite.pwr.rec/-ep.and.afw feed.bleed	CD	1.9E-06	2.4E-01
411	sgtr -rt AFW mfw	CD	9.1E-07	1.5E-01

** non-recovery credit for edited case

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\zion2.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 5.0E-02	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04		
AFW/ATWS	4.3E-03 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > Failed		
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-05	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	5.4E-04	8.9E-01	
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	3.1E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.19 LER No. 304/83-007

Event Description: Postulated Grid/Weather-Related LOOP with 2 EDGs Inoperable

Date of Event: January 31, 1983

Plant: Zion 2

B.19.1 Summary

On January 31, 1983, during normal operations, the 0 diesel generator failed to accept a load greater than 50% during an operability check for taking residual heat removal (RHR) 2A pump out for servicing. The 2B diesel generator was also declared inoperable due to an oil leak. Rampdown of the unit was started. The estimated increase in core damage probability, or importance, over the duration of this event is 4.8×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 1.9×10^{-6}, resulting in an estimated conditional core damage probability (CCDP) of 5.0×10^{-5}.

B.19.2 Event Description

On January 31, during normal operations, the 0 diesel generator failed to accept a load greater than 50% during an operability check for taking RHR 2A pump out for servicing. The 2B diesel generator was also declared inoperable due to an oil leak. These failures resulted in two Technical Specification violations (3.15.2.C and 3.15.2.H). Investigation of the 0 diesel generator revealed that the 0 diesel generator turbo charger ET-18EK9V had seized, thus reducing capacity to 50%. The turbo charger was replaced along with the 2B diesel generator oil filter gasket.

B.19.3 Additional Event-Related Information

Zion 2 has three emergency diesel generators, each rated at 4,000 kW and cooled by service water. Two diesel generators are specifically dedicated to Zion 2. Diesel generator 2A feeds 4-kV bus 248, and diesel generator 2B feeds 4-kV bus 249. One diesel generator (diesel generator 0) is connected to both Zion 1 bus 147 and Zion 2 bus 247. The buses are electrically interlocked to prevent the diesel from supplying both buses at the same time. All diesel generators have a 50,000-gallon storage tank which is sufficient for seven days of operation as well as a 600-gallon day tank equipped with automatic level controls. The diesels start automatically upon receipt of an automatic or manually initiated safety injection signal, loss of power to any two of four 4-kV non-engineered safety feature (ESF) buses, and an undervoltage on the 4-kV bus served by the diesel generator. Diesel generator 2B bus 249 supplies power to auxiliary feedwater pump 1C, RHR pump A, and charging pump A. Diesel generator 0 bus 247 supplies power to safety injection pump A and charging pump B. In addition to the diesel generators, power from the Unit 1 station auxiliary transformer (SAT) can be manually aligned to supply power to Unit 2.

B.19.4 Modeling Assumptions

Since auxiliary power can be supplied from Unit 1, plant-centered losses of offsite power (LOOPs) would not be of particular importance in this event. LOOPs which affected both units (i.e., when Unit 1 could not provide auxiliary power to Unit 2), such as grid-related and weather-related LOOPs, would be of importance given both dedicated emergency diesel generators (EDGs) were inoperable. Thus, this event was modeled as a postulated grid-related/weather related LOOP with two EDGs inoperable. The LOOP frequency, the offsite power recovery probabilities, and the probability of seal loss-of-coolant accident (LOCA) were modified as shown in Table B.23.1 to reflect those values associated with grid-related and weather-related LOOPs (see ORNL/NRC/LTR 89/11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989). The first train of emergency power was set to failed to reflect the failed EDG since it was assumed that the fault discovered in EDG 0 could also have occurred in the other EDGs. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as a part of the postulated event. The second train of emergency power was set to failed to reflect the assumption that the oil leak in EDG 2B was severe enough to prevent the EDG from operating if needed. The corresponding system trains which rely on these diesels for power given the loss of offsite power were also modified to reflect their unavailability. Since the length of time in which both faults were present is unknown, a duration of half the surveillance period on the diesels was chosen (15 days or 360 hours).

Table B.19.1 Revised LOOP Probabilities

Event	Default Probability	Revised Probability
LOOP frequency	1.6E-5	2.8E-6
LOOP short-term nonrecovery	0.53	0.66
Seal LOCA probability	0.27	0.42
Offsite power recovery prior to battery depletion given no seal LOCA	0.031	0.14
Offsite power recovery prior to battery depletion given seal LOCA	0.57	0.77
Offsite power recovery within 2 hours (OFFSITE.PWR.REC/-EP.AND.-AFW)	0.22	0.52
Offsite power recovery within 6 hours (OFFSITE.PWR.REC/-EP.AND.AFW)	0.067	0.32

B.19.5 Analysis Results

The estimated increase in core damage probability over the duration of this event is 4.8×10^{-5}. The base-case CDP (not shown in calculation) is 1.3×10^{-6}, resulting in an estimated CCDP of 5.0×10^{-5}. The dominant sequence highlighted on the event tree in Figure B.19.1 involved a postulated LOOP with emergency power failure (station blackout), an RCP seal LOCA, and failure to recover offsite power before core uncovery.

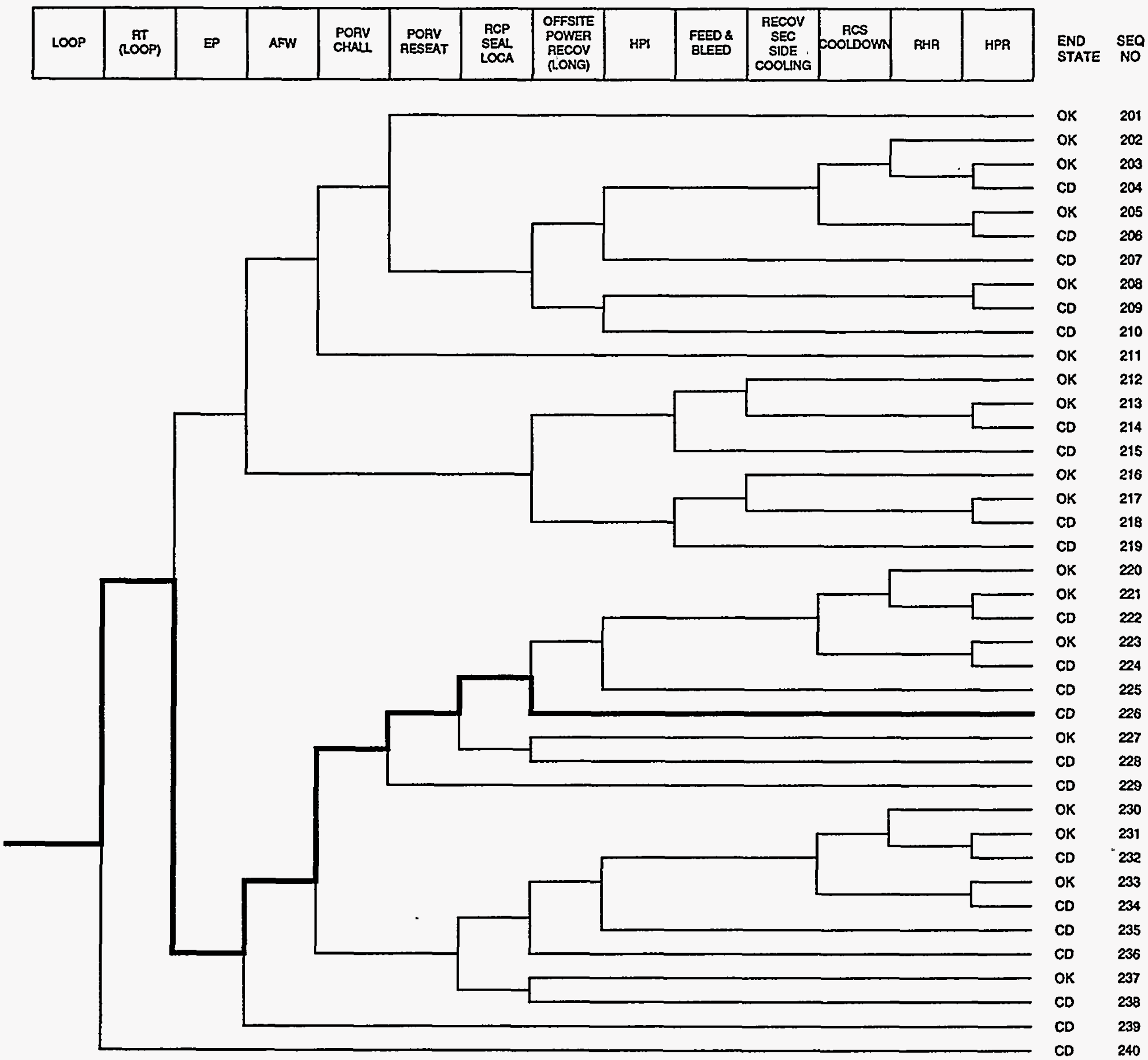

Figure B.19.1 Dominant core damage sequence for LER 304/83-007

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 304/83-007
Event Description: Postulated grid/weather related LOOP with 2 EDGs inop.
Event Date: January 31, 1983
Plant: Zion 2

UNAVAILABILITY, DURATION= 360

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	6.7E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	4.8E-05
Total	4.8E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
226	LOOP -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep SEAL .LOCA OFFSITE.PWR.REC/SEAL.LOCA	CD	3.5E-05	5.8E-01
228	LOOP -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep -SEAL .LOCA OFFSITE.PWR.REC/-SEAL.LOCA	CD	8.8E-06	5.8E-01
229	LOOP -rt(loop) EP -afw/ep porv.chall/sbo porv.reseat/ep	CD	2.2E-06	5.8E-01
239	LOOP -rt(loop) EP afw/ep	CD	1.9E-06	2.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
226	LOOP -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep SEAL .LOCA OFFSITE.PWR.REC/SEAL.LOCA	CD	3.5E-05	5.8E-01
228	LOOP -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep -SEAL .LOCA OFFSITE.PWR.REC/-SEAL.LOCA	CD	8.8E-06	5.8E-01
229	LOOP -rt(loop) EP -afw/ep porv.chall/sbo porv.reseat/ep	CD	2.2E-06	5.8E-01
239	LOOP -rt(loop) EP afw/ep	CD	1.9E-06	2.0E-01

** non-recovery credit for edited case

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\zion2.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
LOOP	1.6E-05 > 2.8E-06	5.3E-01 > 6.6E-01	
Branch Model: INITOR			
Initiator Freq:	1.6E-05 > 2.8E-06		
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 1.3E-03	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04		
AFW/ATWS	4.3E-03 > 7.0E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > 7.0E-02		
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
HPI	1.0E-05 > 1.0E-02	8.9E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	1.0E-02 > Unavailable		
FEED.BLEED	2.0E-02 > 3.0E-02	1.0E+00	1.0E-02
Branch Model: 1.OF.3+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	1.0E-02 > Unavailable		
Serial Component Prob:	2.0E-02		
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
RHR	2.2E-02 > 4.0E-02	7.0E-02	1.0E-03
Branch Model: 1.OF.2+ser+opr			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Serial Component Prob:	2.0E-02		

RHR.AND.HPR	1.0E-03 > 1.0E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HPR	4.0E-03 > 4.0E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	4.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
EP	5.4E-04 > 1.9E-01	8.9E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02 > Failed		
Train 3 Cond Prob:	1.9E-01		
SEAL.LOCA	2.7E-01 > 4.2E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.7E-01 > 4.2E-01		
OFFSITE.PWR.REC/-EP.AND.-AFW	2.2E-01 > 5.2E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.2E-01 > 5.2E-01		
OFFSITE.PWR.REC/-EP.AND.AFW	6.7E-02 > 3.2E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.7E-02 > 3.2E-01		
OFFSITE.PWR.REC/SEAL.LOCA	5.7E-01 > 7.7E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.7E-01 > 7.7E-01		
OFFSITE.PWR.REC/-SEAL.LOCA	3.1E-02 > 1.4E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	3.1E-02 > 1.4E-01		
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.20 LER No. 309/83-002

Event Description: Transient with MFW Inoperable and One Isolated Steam Generator

Date of Event: January 25, 1983

Plant: Maine Yankee

B.20.1 Summary

On January 25, 1983, Maine Yankee tripped from a full load while isolating an electrical ground. Main feedwater (MFW) was unavailable after the trip and auxiliary feedwater (AFW) auto-started and provided cooling to the steam generators. Approximately 15 minutes after the trip, indications were received that a main feedwater line break had occurred. The estimated conditional core damage probability for this event is 8.6 x 10^{-5}.

B.20.2 Event Description

On January 25, 1983, Maine Yankee tripped from full load while an electrical ground was being isolated. Main feedwater was not available after the trip due to the trip of the turbine-driven pump and maintenance on both motor-driven pumps [NUREG-0090-Vol. 6-No.11]. Approximately 15 minutes later, a loud noise was heard in the plant machine shop, and a containment fire detector alarmed. Containment humidity also began to rise. The containment was entered for inspection and a feedline leak was discovered near the number 2 steam generator inlet nozzle. Station cooldown was initiated to permit close access for further inspection and repairs. Further investigation revealed that the leak likely occurred due to water hammer, which resulted in the failure of an existing crack in the feed pipe. The feedline leak was at most 100 gpm [NUREG-0090-Vol.6-No.11] and all AFW pumps were functional at the time of the incident.

B.20.3 Additional Event-Related Information

The Maine Yankee MFW system consists of one turbine-driven pump and two motor-driven pumps. AFW system consists of two motor-driven pumps and one turbine-driven pump. Any one of the three AFW pumps can supply sufficient water to remove decay heat from the steam generators.

B.20.4 Modeling Assumptions

This event was modeled as a transient with MFW failed. The MFW branch probability was set to failed, and the nonrecovery probability was set to 1.0 to reflect the likelihood that operators would not have been able to recover MFW within the allowable time during the transient.

The ~100 gpm leak rate experienced from the feedwater line to SG2 was relatively large compared to the output of a single AFW pump. Therefore, AFW success was assumed to require operation of two of three AFW pumps. The AFW branch failure probability was revised to reflect this.

B.20.5 Analysis Results

The estimated conditional core damage probability for this event is 8.6×10^{-5}. The dominant sequence, highlighted on the event tree in Figure B.20.1, involves a successful reactor trip, the failure of AFW, the failure of MFW, and the failure of feed and bleed.

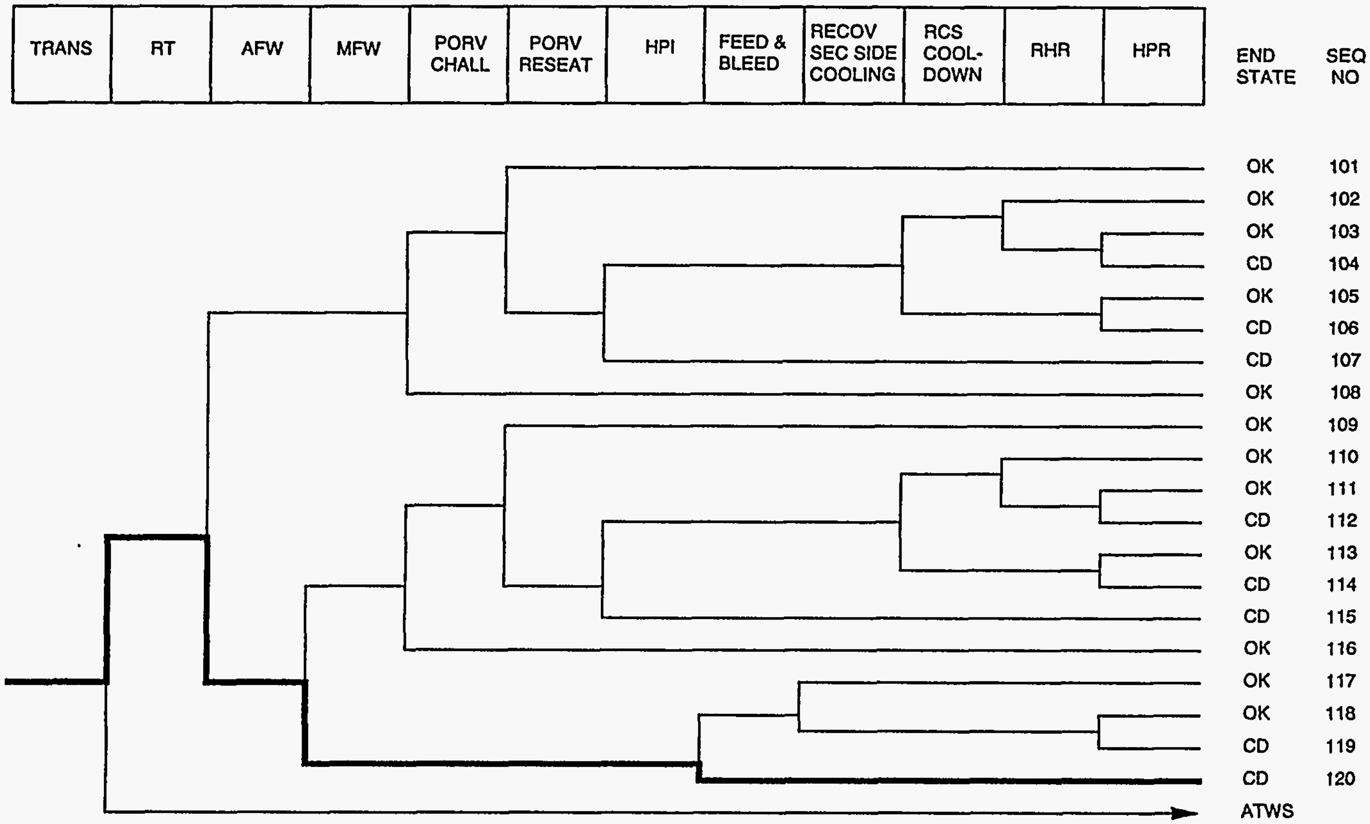

Figure B.20.1 Dominant core damage sequence for LER 309/83-002

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 309/83-002
Event Description: Transient with MFW inoperable and one SG isolated
Event Date: January 25. 1983
Plant: Maine Yankee

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	8.6E-05
Total	8.6E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt afw MFW feed.bleed	CD	8.2E-05	4.5E-01
119	trans -rt afw MFW -feed.bleed recov.sec.cool hpr	CD	2.7E-06	4.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt afw MFW -feed.bleed recov.sec.cool hpr	CD	2.7E-06	4.5E-01
120	trans -rt afw MFW feed.bleed	CD	8.2E-05	4.5E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\myank82.cmp
BRANCH MODEL: c:\aspcode\models\myankee.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	6.5E-04	1.0E+00	
loop	2.0E-05	5.8E-01	
loca	2.4E-06	5.4E-01	

sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	6.1E-03***	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
MFW	1.9E-01 > 1.0E+00	3.4E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.9E-01 > Failed		
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	3.1E-02	7.0E-02	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	5.5E-02	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	3.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	1.1E-01	1.0E+00	
offsite.pwr.rec/seal.loca	6.5E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	2.3E-01	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced
*** branch probability reflects the requirement for 2 of 3 AFW pumps for success.

B.21 LER No. 311/83-001 and 311/82-072

Event Description: Transient with One Automatic Trip Breaker Failing to Open

Date of Event: January 6, 1983

Plant: Salem 2

B.21.1 Summary

On January 6, 1983, while Salem 2 was operating at 46% power, the reactor tripped due to low level in the number 21 steam generator. Following the trip, the operator noticed that trip breaker A had failed to open on the trip signal, but trip breaker B had opened and de-energized the rod drive mechanisms, resulting in a shutdown. It was later determined that trip breaker A undervoltage relay had malfunctioned due to dirt or corrosion which interfered with proper relay operation. A similar breaker failure occurred on August 20, 1982 during a surveillance test. The conditional core damage probability estimated for the reactor trip on January 6, 1983 is 4.4×10^{-4}.

B.21.2 Event Description

On January 6, 1983, while Salem 2 was operating at 46% power, the reactor tripped due to low level in the number 21 steam generator. Following the trip, the operator noticed that trip breaker A had failed to open on the trip signal, but trip breaker B had opened and de-energized the rod drive mechanisms, resulting in a shutdown. It was later determined that the trip breaker A undervoltage relay had malfunctioned due to dirt or corrosion which interfered with proper relay operation. This dirt or corrosion resulted from the infrequent operation of the breaker, which led to insufficient self-cleaning of the relay. The debris accumulated and caused a mechanical binding of the undervoltage relay.

On August 20, 1982, during a surveillance test with the plant at 82% power, reactor trip breaker B failed to trip as required. Trip breaker A was operable. Investigation revealed that the cause of the B trip breaker was binding of the undervoltage coil. The coil was replaced and trip breaker B was reinstalled and satisfactorily tested.

B.21.3 Additional Event-Related Information

The Salem 2 reactor protection system (RPS) uses independent channels and trains which consist of sensors, transmitters, relays and trip breakers to detect and protect against unsafe plant conditions. When an unsafe plant condition occurs, the RPS signals the trip breakers to open and de-energize the rod drive mechanisms, resulting in a reactor shutdown. The reactor trip breakers are ac circuit breakers positioned in series. When either trip breaker is tripped open, holding power to the control rods is lost and the rods drop into the core. At the time of this event, one mechanism, de-energization of the undervoltage coils, could open the trip breaker. A second mechanism for tripping open the breakers was installed after the February 1983 Salem anticipated transient without scram (ATWS). This mechanism energizes the shunt trip coil. The shunt trip coils, once energized, will open the breakers.

B.21.4 Modeling Assumptions

The August 20, 1982 surveillance test failure was considered incidental to the event of interest. The January 6, 1983 reactor trip was modeled as a trip with the reactor trip system degraded. One train of the reactor trip (RT) system was set to failed. Manual scram capability was not affected by the failure of the reactor trip breakers so the RT nonrecovery probability (which models the manual scram capability of the RT system) was not modified. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure of the other train of the reactor trip system was included in the analysis.

B.21.5 Analysis Results

The conditional core damage probability for this event is 4.4×10^{-4}. The dominant sequence is a postulated ATWS sequence involving a failure to trip, success of auxiliary feedwater (AFW) given ATWS, and failure of emergency boration, and is highlighted on the event tree shown in Figure B.21.1.

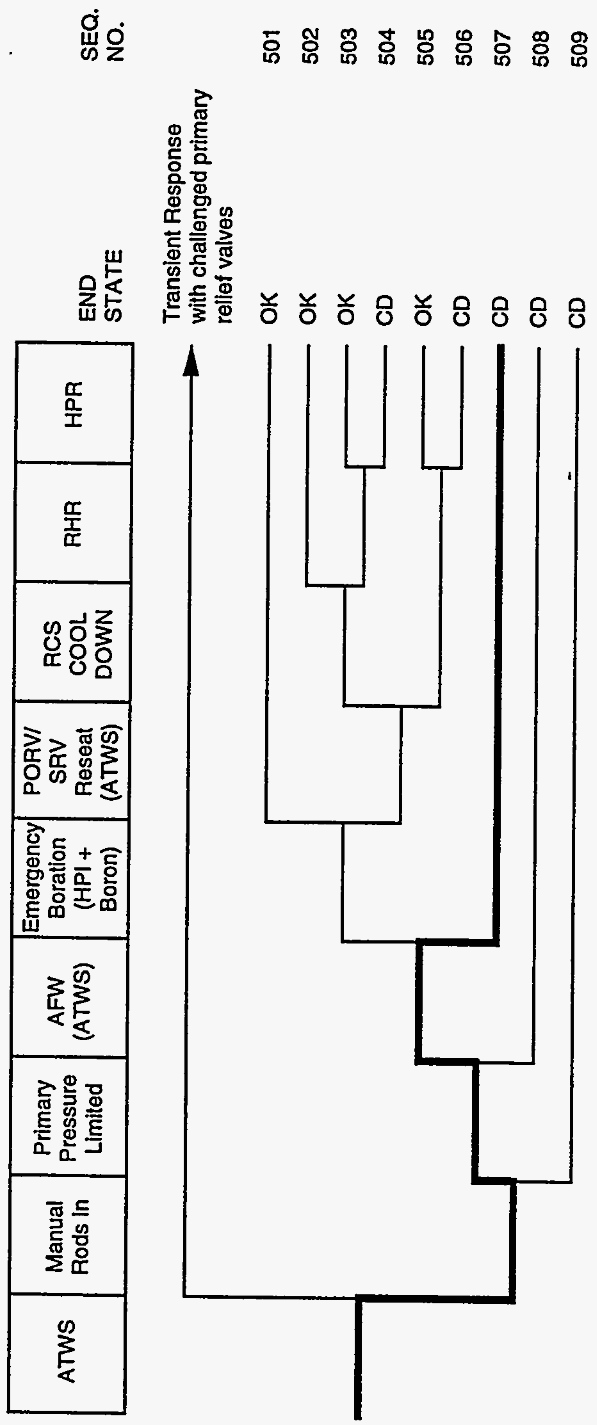

Figure B.21.1 Dominant core damage sequence for LER 311/83-001 and 311/82-072

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 311/83-001 and 311/82-072
Event Description: Transient with one automatic trip breaker failing to open
Event Date: January 6. 1983
Plant: Salem 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	4.4E-04
Total	4.4E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

Sequence	End State	Prob	N Rec**
507 trans RT -prim.press.limited -afw/atws emrg.boration	CD	1.9E-04	1.0E-01
509 trans RT prim.press.limited	CD	1.7E-04	1.0E-01
508 trans RT -prim.press.limited afw/atws	CD	8.1E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

Sequence	End State	Prob	N Rec**
507 trans RT -prim.press.limited -afw/atws emrg.boration	CD	1.9E-04	1.0E-01
508 trans RT -prim.press.limited afw/atws	CD	8.1E-05	1.0E-01
509 trans RT prim.press.limited	CD	1.7E-04	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\salem2.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	2.1E-03	1.0E+00	

loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
RT	2.8E-04 > 1.9E-01	1.0E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	1.5E-03 > Failed		
Train 2 Cond Prob:	1.9E-01		
rt(loop)	0.0E+00	1.0E+00	
afw	3.8E-04	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-05	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	5.4E-04	8.9E-01	
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.22 LER No. 316/82-113

Event Description: Long-Term Unavailability of SI Train B

Date of Event: December 29, 1982

Plant: Cook 2

B.22.1 Summary

An emergency core cooling system (ECCS) flow balance test determined that safety injection (SI) train B could not meet test flow requirements. Subsequent investigation determined that the B train check valve was obstructed by a second check valve disk, which was lodged in the outlet of the valve. This condition was believed to have existed for a prolonged period. The increase in core damage probability (CDP), or importance, over the duration of the event is 1.4×10^{-6}. The base-case CDP over the duration of the event is 9.0×10^{-5}, resulting in an estimated conditional core damage probability (CCDP) of 9.1×10^{-5}.

B.22.2 Event Description

An ECCS flow balance test of the safety injection system determined that SI train B could not provide the flow rate required of it. A radiographic exam was performed and on December 29, 1982, it was recognized that a second valve disk was stuck in SI train B check valve SI-152S. The check valve was repaired on December 30, and the system subsequently passed a flow balance test.

It was not known precisely when the check valve failed; however it was reported that check valve SI-152S was found to be "leaking excessively" in May 1981. In October, 1981, the valve was opened, the disk was found to be missing and it was replaced. No ECCS flow balance test was performed at that time.

It is possible that the check valve was failed before or at the time of the May testing. Reconstructed test data from 1979 indicated that SI train B flow may have been around 243 gpm at that time, with a minimum of 300 gpm required for the train to be considered operable. Certainly the valve was failed between October 1981, and discovery of the extra disk on December 29, 1982.

B.22.3 Additional Event-Related Information

None.

B.22.4 Modeling Assumptions

This event was modeled as an unavailability of one train of high-pressure injection (HPI) for one operating year (one year is the maximum unavailability period normally considered in ASP analyses).

Cook reported in the licensee event report for this event that flow from the degraded HPI pump in conjunction with the flow from a charging pump would provide most of the flow required from a single HPI pump. As the charging pumps are included in the ASP model as a distinct backup source of high-pressure injection (successful operation of both charging pumps is assumed to provide adequate high-pressure injection), this potential augmentation of a degraded HPI pump by a charging pump was not credited in the ASP analysis. Other events were reported during the year, which were affected by the SI train B unavailability. These analyses are reported separately.

B.22.5 Analysis Results

The increase in core damage probability (CDP), or importance, over the duration of the event is 1.4×10^{-6}. The dominant sequence, highlighted on the event tree in Figure B.22.1, involves a steam generator tube rupture, reactor trip and auxiliary feedwater (AFW) success, and failure of HPI. The base-case CDP over the duration of the event is 9.0×10^{-5} (not shown on calculation sheet), resulting in an estimated CCDP of 9.1×10^{-5}.

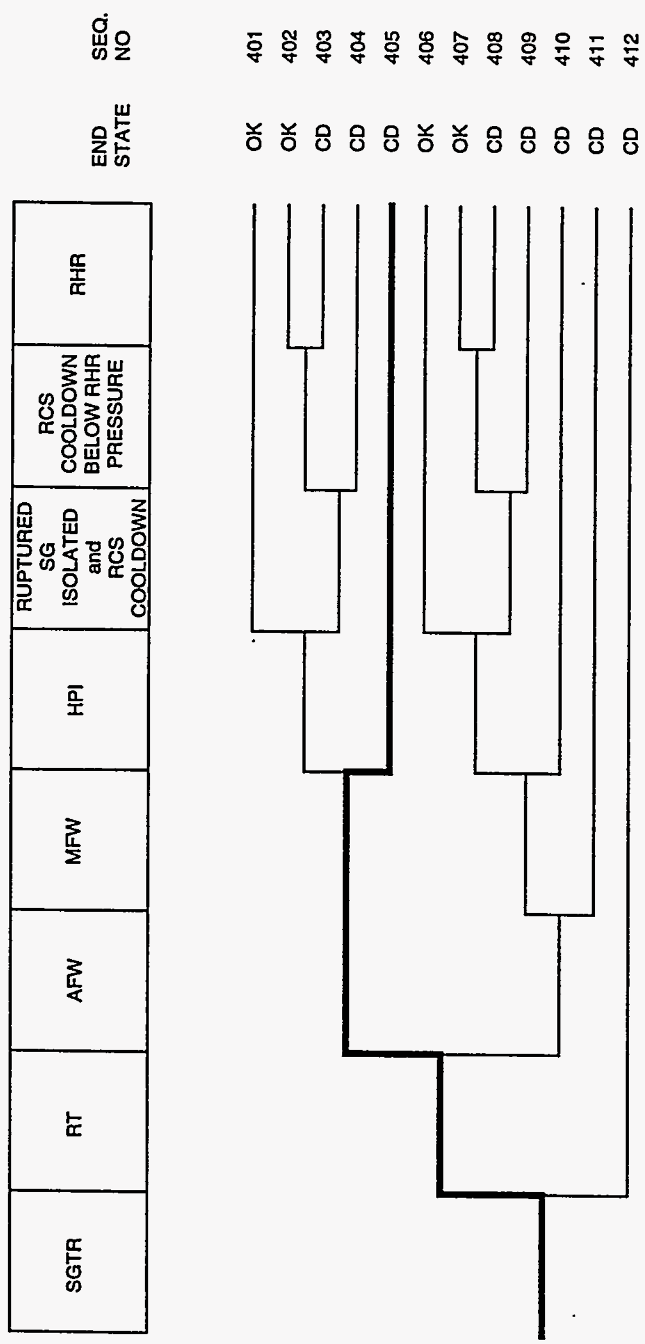

Figure B.22.1 Dominant core damage sequence for LER 316/82-113

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 316/82-113
Event Description: Long-term unavailability of SI train B
Event Date: December 29. 1982
Plant: Cook 2

UNAVAILABILITY. DURATION= 6132

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	4.0E+00
LOOP	2.3E-02
LOCA	7.9E-03
SGTR	1.0E-02

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	6.9E-10
LOOP	3.7E-10
LOCA	6.3E-07
SGTR	8.0E-07
Total	1.4E-06

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
405	sgtr -rt -afw HPI	CD	8.0E-07	8.9E-01
306	loca -rt -afw HPI	CD	6.3E-07	4.8E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
306	loca -rt -afw HPI	CD	6.3E-07	4.8E-01
405	sgtr -rt -afw HPI	CD	8.0E-07	8.9E-01

** non-recovery credit for edited case

Note: For unavailabilities. conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: d:\asp\models\pwrb8283.cmp
BRANCH MODEL: d:\asp\models\cook2.82
PROBABILITY FILE: d:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
Event Identifier: 316/82-113			
trans	6.5E-04	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	3.8E-04	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
HPI	1.0E-05 > 1.0E-04	8.9E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > 1.0E+00		
Train 3 Cond Prob:	1.0E-02		
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	5.7E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.5E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	1.6E-01	1.0E+00	
offsite.pwr.rec/seal.loca	6.9E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	5.2E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.23 LER No. 321/83-090, Rev. 1, -093

Event Description: Manual Scram with HPCI and RCIC Unavailable

Date of Event: August 25, 1983

Plant: Hatch 1

B.23.1 Summary

On August 24, 1983, the high-pressure coolant injection (HPCI) pump was manually tripped during surveillance testing following receipt of a low oil pressure alarm. Operators attempted to run HPCI a second time, and the same problems were encountered. On August 25, 1983, the reactor core isolation cooling system (RCIC) tripped on low bearing oil pressure, and a low flow alarm for the B RCIC area cooler was received in the control room. The unit was brought to cold shutdown because of the concurrent HPCI and RCIC unavailabilities. The conditional core damage probability estimated for the event is 1.3×10^{-5}.

B.23.2 Event Description

On August 24, 1983, a HPCI turbine bearing oil pressure low alarm was received in the control room when HPCI was started for its pump operability test. HPCI was manually tripped. The plant operators attempted to run HPCI a second time, and the same problems were encountered.

On August 25, 1983, RCIC tripped on low bearing oil pressure during the performance of the RCIC pump operability test. Also during the test, a low flow alarm for the B RCIC area cooler was received in the control room. Later on August 25, 1983, while washing down the cooling coils for the B RCIC area cooler, water drenched the RCIC turbine exhaust high-pressure switches, causing a turbine trip and an alarm in the control room. As a result of the combined HPCI and RCIC unavailabilities, the unit was brought to cold shutdown. The shutdown was initiated by a manual scram.

The cause of the HPCI turbine bearing oil pressure low alarm was attributed to the HPCI turbine thrust bearing oil supply valve being closed and providing insufficient oil supply to the bearing. Also, the HPCI turbine governor end radial bearing oil supply valve was found to be open too much (providing oil at 30 psig when it should have been 10 to 12 psig). The valves were repositioned, and the oil pressure on the HPCI turbine was adjusted.

A visual inspection of the HPCI turbine overspeed trip ball and tappet assembly revealed a broken ball, thus making the HPCI turbine overspeed trip nonfunctional. It is believed that this damage occurred when the HPCI turbine was started on August 24, 1983, and ran erratically. The HPCI turbine overspeed trip ball and tappet assembly was replaced. HPCI was tested and returned to operability on August 29, 1983.

The cause of the RCIC low bearing oil pressure alarm and trip was attributed to setpoint drift of the RCIC low bearing oil pressure switch. The cause of the RCIC B area cooler low flow alarm was a sensing element (pitot tube) partially stopped up with dust. The oil pressure switch was recalibrated and the pitot tube was cleaned and blown out. The water-drenched RCIC turbine exhaust switches were dried out and their calibration was checked.

B.23.3 Additional Event-Related Information

The HPCI and RCIC systems are the primary source of reactor pressure vessel (RPV) makeup in the event of a loss of feedwater.

B.23.4 Modeling Assumptions

This event is modeled as a reactor trip with both RCIC and HPCI unavailable. Neither HPCI nor RCIC was deemed repairable, and a nonrecovery probability of 1.0 was assigned to each of them.

B.23.5 Analysis Results

The conditional core damage probability estimated for this event is 1.3×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.23.1, involves the observed transient, failure of the power conversion system, failure of feedwater, HPCI failure, RCIC failure, failure of the safety relief valves (SRVs) to reduce pressure, and failure of the control rod drive system to provide RPV makeup.

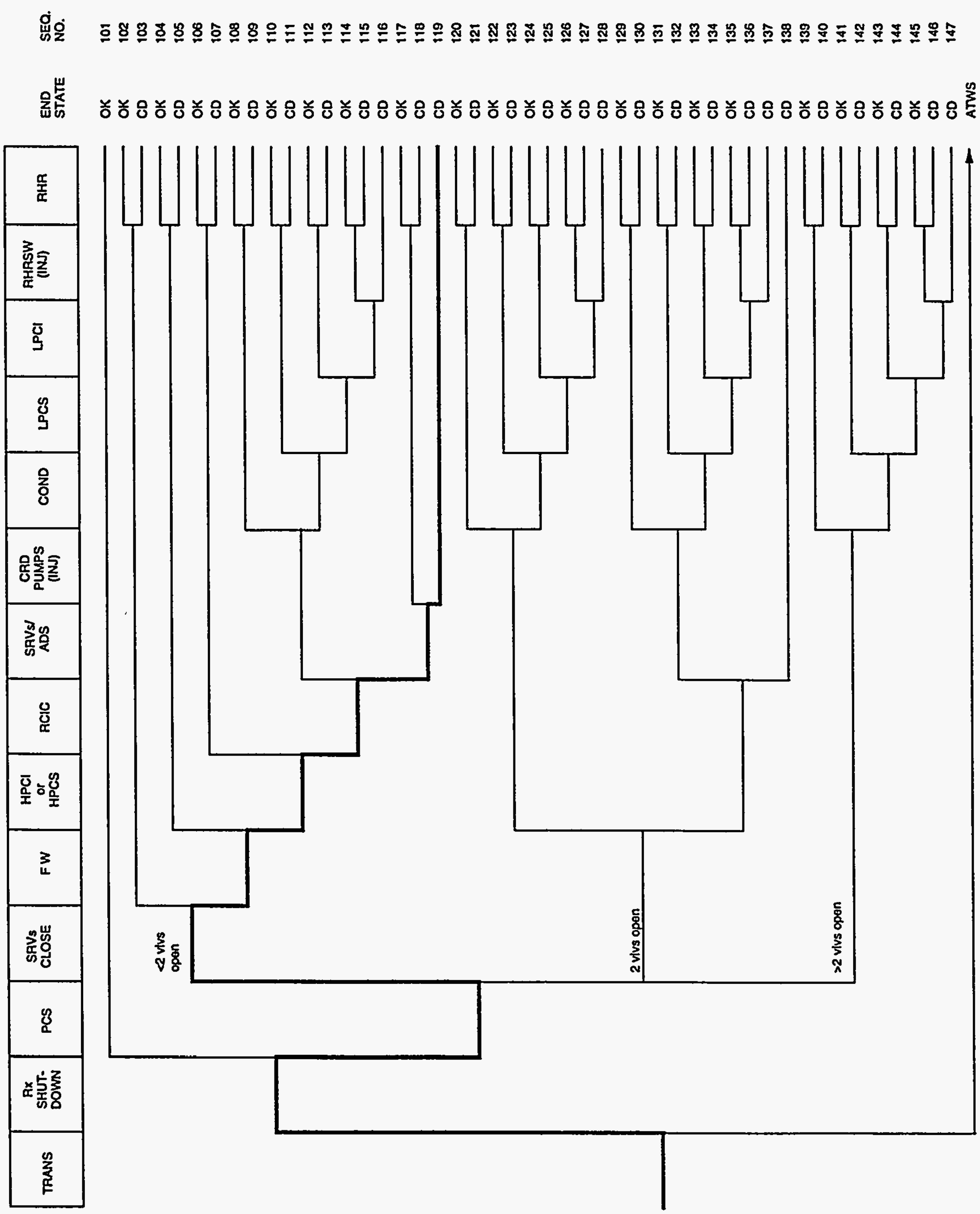

Figure B.23.1 Dominant core damage sequence for LER 321/83-090

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 321/83-090
Event Description: Manual scram with HPCI and RCIC unavailable
Event Date: August 25. 1983
Plant: Hatch 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.3E-05
Total	1.3E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rx.shutdown pcs srv.ftc.<2 mfw HPCI RCIC srv.ads c rd(inj)	CD	6.6E-06	2.4E-01
138	trans -rx.shutdown pcs srv.ftc.2 HPCI srv.ads	CD	2.8E-06	7.0E-01
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw rhr.and.pcs.nrec	CD	1.6E-06	7.3E-03
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
109	trans -rx.shutdown pcs srv.ftc.<2 mfw HPCI RCIC -srv.ads -c ond rhr	CD	2.1E-07	3.6E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw rhr.and.pcs.nrec	CD	1.6E-06	7.3E-03
109	trans -rx.shutdown pcs srv.ftc.<2 mfw HPCI RCIC -srv.ads -c ond rhr	CD	2.1E-07	3.6E-03
119	trans -rx.shutdown pcs srv.ftc.<2 mfw HPCI RCIC srv.ads c rd(inj)	CD	6.6E-06	2.4E-01
138	trans -rx.shutdown pcs srv.ftc.2 HPCI srv.ads	CD	2.8E-06	7.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

```
SEQUENCE MODEL:      d:\asp\models\bwrc8283.cmp
BRANCH MODEL:        d:\asp\models\hatch1.82
PROBABILITY FILE:    d:\asp\models\bwr8283.pro
```

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.6E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	4.6E-01	3.4E-01	
HPCI	2.9E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	2.9E-02 > 1.0E+00		
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.5E-04	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.24 LER No. 324/82-005

Event Description: Scram with both RHRSW Loops Inoperable

Date of Event: January 16, 1982

Plant: Brunswick 2

B.24.1 Summary

After a reactor scram, operators attempted to align suppression pool cooling but were unable to do so because both residual heat removal (RHR) system service water (SW) loops were found to be inoperable. The conditional core damage probability estimated for the event is 2.3×10^{-4}.

B.24.2 Event Description

On January 16, 1982, Brunswick 2 experienced a scram due to low condenser vacuum. After the scram, a group 1 isolation occurred and the main steam isolation valves (MSIVs) closed. Operators aligned the reactor core isolation cooling system (RCIC) to supply makeup water to the reactor. Later, when operators attempted to align suppression pool cooling, they discovered that both RHRSW loops were inoperable. Low suction header pressure lockout signals prevented start of pumps in both loops. Operators reset the group 1 isolation, reopened the MSIVs, re-established condenser vacuum, and realigned the main feedwater power conversion system (PCS) for makeup and decay heat removal.

An inspection of the suction header pressure switches found that their sensing lines were partially plugged with sediment, which may have prevented the switches from sensing the actual header pressure, which was within acceptable limits. The suction header pressure switch for the A loop was also found to be damaged. In addition, the power supply of the B loop suction header pressure switch was found to be switched off, apparently having been left that way after prior maintenance work. The pressure switch power feed breaker was reclosed, the RHRSW B loop interlock cleared, and the associated RHR train was started and aligned for suppression pool cooling. RHRSW train B was tested and declared operable approximately 4 hours after the scram. The A service water loop was made operable approximately 8 hours after the scram.

B.24.3 Additional Event-Related Information

"Operating Experience Feedback Report - Service Water System Failures and Degradations," NUREG-1275, V. 3, and "Brunswick Nuclear Power Station Unit 2 Loss of Residual Heat Removal Service Water on January 16, 1982," AEOD/Engineering Evaluation Report E236, USNRC, provide additional detail about this event.

B.24.4 Modeling Assumptions

This event was modeled as a scram with both trains of RHRSW initially unavailable. The RHRSW pumps at Brunswick maintain a positive pressure differential between the tube and shell side of the RHR heat exchangers, which prevents leakage of primary coolant into the service water (SW) system. Adequate decay heat removal can be provided in the event of RHRSW pump unavailability using the SW pumps to directly supply the RHR heat exchangers, if one valve (FO68A/B) in each train is locally opened. This action is addressed in the Brunswick IPE, and an operator failure probability of 0.01 was estimated. Because of the unavailability of the RHRSW trains, all modes of RHR, except low-pressure coolant injection (LPCI), were modeled as initially failed.

As this event involved a loss of condenser vacuum and MSIV isolation, the power conversion system was assumed to be failed and nonrecoverable in the short term (PCS was assumed to be recoverable in the long term).

To recover decay heat removal capability using RHR, operators needed to either recover the inoperable RHRSW pumps or align the service water system to supply the RHR heat exchangers. In the event, since the PCS had been recovered and was being used for decay heat removal, the operators focused on correcting the RHRSW suction header pressure switch problems and restoring RHRSW.

If the PCS had not been recovered, RHR could have been recovered by locally opening SW valves FO68A and B.

To address this action, the nonrecovery probability for RHR was revised to 0.01 to reflect the probability of the operators failing to open FO68A and B.

For sequences involving potential RHR and PCS recovery, the nonrecovery estimate was revised to 0.01 x 0.017 [probability of not aligning SW multiplied by PCS long-term nonrecovery given MSIV closure (see Appendix A)], or 1.7E-4.

B.24.5 Analysis Results

The conditional core damage probability estimated for this event is 2.3×10^{-4}. The dominant core damage sequence, highlighted on the event tree in Figure B.24.1, involves the observed scram, failure of the power conversion system, and RHR failure.

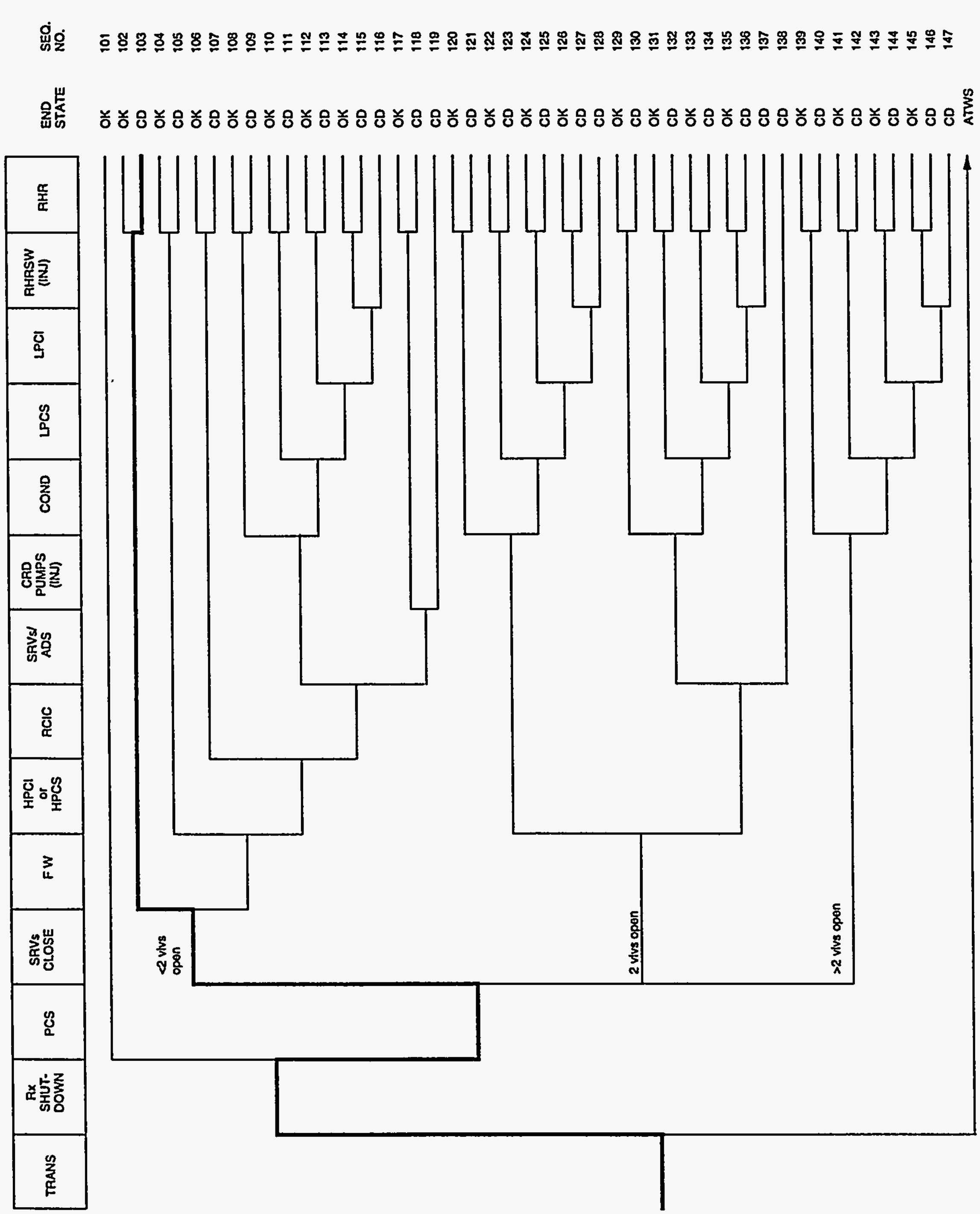

Figure B.24.1 Dominant core damage sequence for LER 324/82-005

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 324/82-005
Event Description: Scram with both RHRSW loops inoperable
Event Date: January 16, 1982
Plant: Brunswick 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	2.3E-04
Total	2.3E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	1.2E-04	1.1E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	6.0E-05	5.7E-05
403	trans rx.shutdown -rpt -slcs PCS -ads.inhibit -hpci RHR(SPCOOL)	CD	3.3E-05	9.9E-02
121	trans -rx.shutdown PCS srv.ftc.2 -hpci -cond RHR	CD	8.4E-06	6.6E-03
123	trans -rx.shutdown PCS srv.ftc.2 -hpci cond -lpcs RHR	CD	4.3E-06	3.4E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	1.2E-04	1.1E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	6.0E-05	5.7E-05
121	trans -rx.shutdown PCS srv.ftc.2 -hpci -cond RHR	CD	8.4E-06	6.6E-03
123	trans -rx.shutdown PCS srv.ftc.2 -hpci cond -lpcs RHR	CD	4.3E-06	3.4E-03
403	trans rx.shutdown -rpt -slcs PCS -ads.inhibit -hpci RHR(SPCOOL)	CD	3.3E-05	9.9E-02

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\bruns2.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.1E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
RHRSW(INJ)	2.0E-02 > 1.0E+00	1.0E+00	1.0E-02
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	2.0E-02 > 1.0E+00		
RHR	1.5E-04 > 1.0E+00 **	1.6E-02 > 1.0E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.0E+00 **	8.3E-03 > 1.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.0E+00 **	1.0E+00	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.0E+00 **	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.0E+00 **	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	

slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.25 LER No. 324/82-029

Event Description: Scram with RHRSW System Degradations

Date of Event: February 3, 1982

Plant: Brunswick 2

B.25.1 Summary

Brunswick Unit 2 was operating at approximately 73% power when an unsuccessful attempt was made to start "A" residual heat removal (RHR) service water (SW) pump. At the same time that A RHRSW pump was inoperable, the emergency power supply for RHRSW pump B was out of service for maintenance. A scram occurred the same day that the RHRSW system degradations were detected. The conditional core damage probability estimated for the event is 3.4×10^{-5}.

B.25.2 Event Description

On February 3, 1982, Brunswick 2 personnel were investigating a signal regarding A RHRSW pump when they unsuccessfully attempted to start the pump. They determined that an RHRSW loop I low-suction header pressure lockout signal had been generated by a failed pressure switch, rendering loop I inoperable. At the same time, the No. 4 emergency diesel generator (EDG) was out of service due to maintenance. As a result, RHRSW pump 1B did not have an emergency power supply, and RHRSW loop II was declared inoperable.

A high main steamline radiation indication resulted in a scram on the same date.

B.25.3 Additional Event-Related Information

None.

B.25.4 Modeling Assumptions

It was assumed that the RHRSW failure existed at the time of the scram, which occurred on the same day. This event was modeled as a transient with one loop of RHRSW inoperable. Failure of the RHRSW train II low suction pressure switch prevented automatic or manual start of the train II RHRSW pumps. The RHRSW pumps at Brunswick maintain a positive pressure differential between the tube and shell side of the RHR heat exchangers, which prevents primary coolant leakage into the service water (SW) system. Adequate decay heat removal can be provided using the SW pumps once one valve (FO68A/B) in each train is locally opened. An operator error probability of 0.01 was estimated for this action in the Brunswick individual plant examination (IPE). Because of the unavailability of RHRSW train II, one train of RHR was modeled as failed in all modes except low-pressure coolant injection (LPCI). Although the specific failure discovered was apparently not present in the other train at the same time, other common cause modes remained and could have affected system performance. A common cause failure probability of 0.1 was assumed for train I and, as this value dominates the component failure probabilities in the system, the probability of RHR failure was set to 0.1 in the model.

The licensee event report indicates that a high main steamline radiation signal was received during the event. It was assumed that this signal caused an automatic isolation of the main steam isolation valves, resulting in power conversion system (PCS) and main feedwater unavailabilities (PCS was assumed to be recoverable in the long term). The probability of RHR nonrecovery was revised to 0.01 to reflect the potential failure of the operators to open FO68A and B. For sequences involving potential RHR or PCS recovery, the nonrecovery estimate was revised to 0.01 x 0.17 [PCS long-term nonrecovery given main steam isolation valve (MSIV) closure (see Appendix A)], or 1.7E-4.

B.25.5 Analysis Results

The conditional core damage probability estimated for this event is 3.4×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.25.1, involves the observed scram, failure of the power conversion system, main feedwater success, and RHR failure.

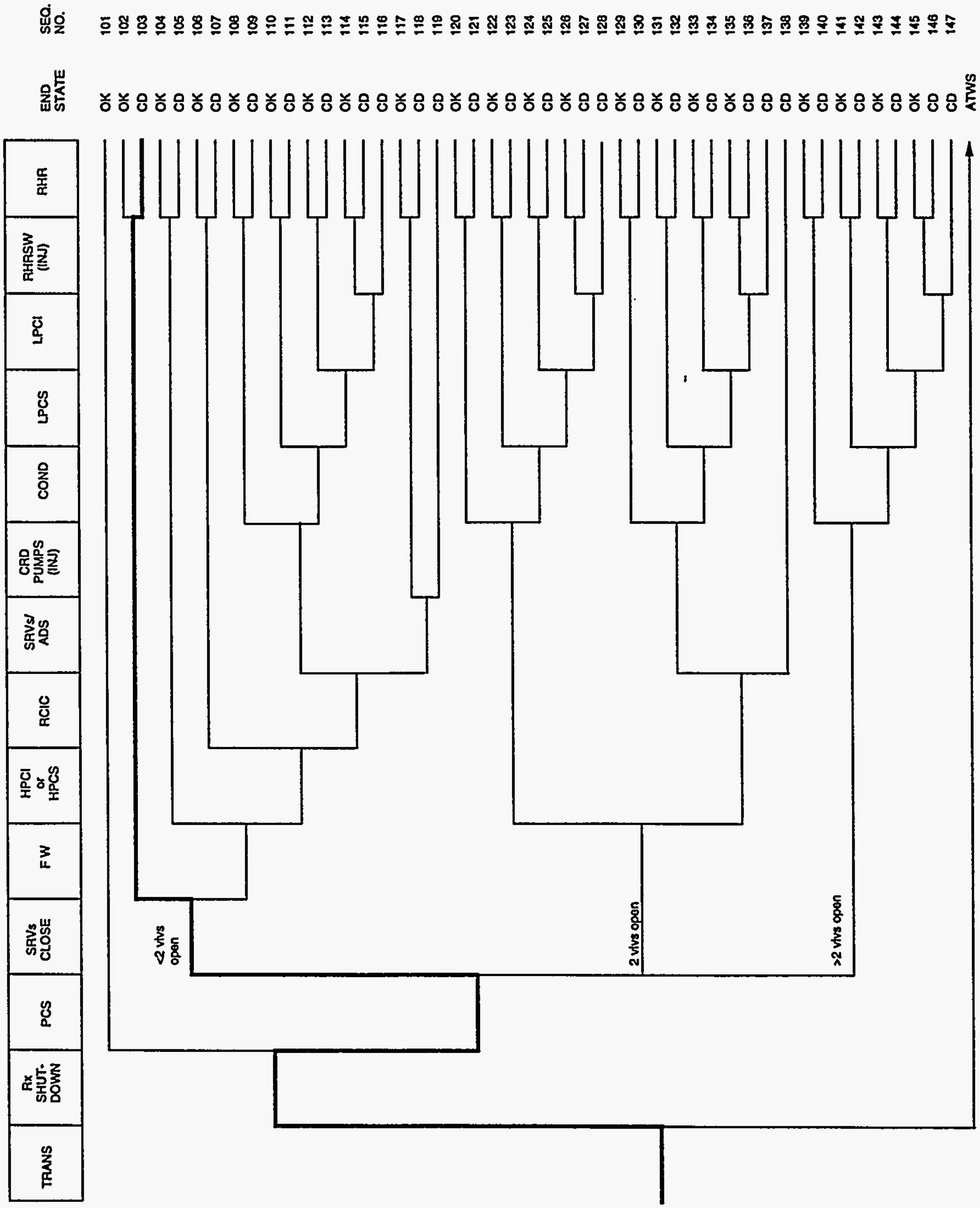

Figure B.25.1 Dominant core damage sequence for LER 324/82-029

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 324/82-029
Event Description: Scram and RHRSW degradations
Event Date: February 3, 1982
Plant: Brunswick 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.4E-05
Total	3.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	1.8E-05	1.1E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	9.0E-06	5.7E-05
403	trans rx.shutdown -rpt -slcs PCS -ads.inhibit -hpci RHR(SPCOOL)	CD	3.3E-06	9.9E-02
121	trans -rx.shutdown PCS srv.ftc.2 -hpci -cond RHR	CD	8.5E-07	6.6E-03
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	1.8E-05	1.1E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	9.0E-06	5.7E-05
121	trans -rx.shutdown PCS srv.ftc.2 -hpci -cond RHR	CD	8.5E-07	6.6E-03
403	trans rx.shutdown -rpt -slcs PCS -ads.inhibit -hpci RHR(SPCOOL)	CD	3.3E-06	9.9E-02
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\bruns2.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.1E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.0E-01 **	1.6E-02 > 1.0E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.0E-01 **	8.3E-03 > 1.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.0E-01 **	1.0E+00 > 1.0E-02	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.0E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		

Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.0E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.26 LER No. 324/82-123

Event Description: Scram with Emergency Bus E-3 De-energized

Date of Event: October 10, 1982

Plant: Brunswick 2

B.26.1 Summary

Brunswick Unit 1 was operating at approximately 17% power during an attempted controlled shutdown when operators experienced difficulty in transferring bus 2D from the unit auxiliary transformer (UAT) to the startup auxiliary transformer (SAT). When the UAT breaker was opened, the output breakers from both the SAT and the number 3 emergency diesel generator (EDG) failed to close, resulting in scram and loss of power to bus E-3. The conditional core damage probability estimated for the event is 1.2×10^{-5}.

B.26.2 Event Description

On October 10, 1982, operators were performing a controlled shutdown of Brunswick Unit 2 when they attempted to transfer electrical bus 2D from the UAT to the SAT. When the UAT breaker opened, the feeder breaker from the SAT failed to close in. Subsequent troubleshooting determined that the breaker charging spring motor shaft had broken, and the breaker was not charged. When the SAT feeder breaker failed to close in on the bus, EDG 3 should have started and powered the bus, but it failed to do so and buses 2D and E-3 were de-energized. An investigation of that failure determined that inappropriately calibrated relays were simultaneously providing open and close signals to the EDG output breaker. As a result, the EDG output breaker was prevented from closing.

B.26.3 Additional Event-Related Information

De-energization of bus E-3 caused a scram and main steam isolation valve (MSIV) isolation. In addition, the 2A core spray and residual heat removal (RHR) pumps were rendered inoperable by the loss of their normal and emergency power supplies.

B.26.4 Modeling Assumptions

This event was modeled as a scram and MSIV isolation with core spray pump 2A and RHR pump 2A inoperable. With MSIVs isolated, the power conversion and feedwater systems were initially unavailable. Because of the multiple failures required to render the 2A core spray and RHR pumps inoperable, the potential for common cause failure of other pumps was assumed to be small. The nonrecovery probability for the power conversion system (PCS) was revised to 0.017 to reflect the MSIV closure (see Appendix A). Combining this value with the estimated long-term RHR nonrecovery probability of 0.016 results in a combined nonrecovery probability for RHR and PCS of 2.7E-4.

B.26.5 Analysis Results

The conditional core damage probability estimated for this event is 1.2×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.26.1, involves the observed scram, failure of the power conversion system, main feedwater success, and failure of RHR.

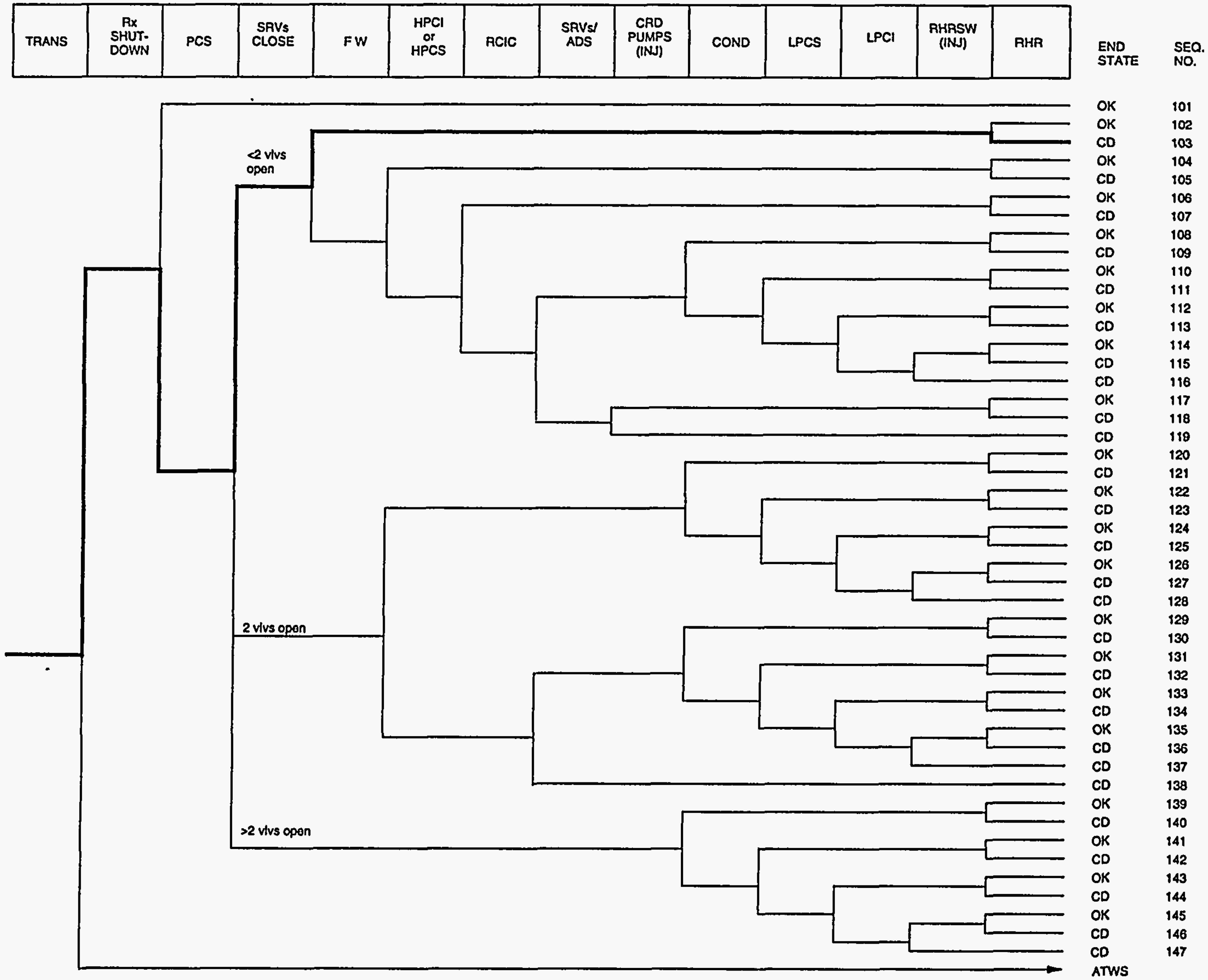

Figure B.26.1 Dominant core damage sequence for LER 324/82-123

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 324/82-123
Event Description: Scram with emergency bus E-3 deenergized
Event Date: October 10. 1982
Plant: Brunswick 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.2E-05
Total	1.2E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.4E-06	9.1E-05
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.4E-06	9.1E-05
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\bruns2.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.1E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
LPCS	2.0E-03 > 2.0E-02	1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > 1.0E+00		
LPCI	1.1E-03 > 1.3E-03	1.0E+00	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
Serial Component Prob:	1.0E-03		
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 3.0E-04	1.6E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
RHR.AND.PCS.NREC	1.5E-04 > 3.0E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05

rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 2.3E-03	1.0E+00	1.0E-03
Branch Model: 1.0F.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
Serial Component Prob:	2.0E-03		
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.27 LER No. 325/82-025

Event Description: Scram with RCIC Inoperable

Date of Event: February 18, 1982

Plant: Brunswick 1

B.27.1 Summary

After a reactor scram, the reactor core isolation cooling (RCIC) system and high-pressure coolant injection (HPCI) systems were signalled to start by a low reactor water level. A failure of the turbine speed controller caused RCIC to immediately trip, however. The conditional core damage probability estimated for this event is 1.3×10^{-5}.

B.27.2 Event Description

On February 18, 1982, Brunswick 1 experienced a scram, followed by a low reactor water level start signal for HPCI and RCIC. While HPCI apparently performed normally, RCIC started and immediately tripped on high exhaust pressure. This was attributed to an improperly calibrated turbine speed control system. RCIC was successfully restarted and operated manually to provide reactor vessel makeup.

B.27.3 Additional Event-Related Information

Following a similar event described in LER 325/82-069, the RCIC control system problems were eventually attributed to a control system design error. The RCIC electronic governor module was found not to have a reference signal common to the RCIC speed controller. Accordingly, variations between the two circuits in sensed ground potential caused unpredictable control system behavior. The flawed control system design apparently had existed since plant startup.

B.27.4 Modeling Assumptions

This event was modeled as a scram, with RCIC assumed unavailable, but recoverable. As reactor vessel level dropped to the auto-start setpoint for RCIC and HPCI, the reactor was assumed to have isolated. This isolation resulted in the unavailability of the power conversion and feedwater systems. The nonrecovery probability for the power conversion system (PCS) was revised to 0.017 to reflect the main steam isolation valve (MSIV) closure (see Appendix A). Combining this value with the estimated long-term RHR nonrecovery probability of 0.016 results in a combined nonrecovery probability for residual heat removal (RHR) and PCS of 2.7E-4.

B.27.5 Analysis Results

The conditional core damage probability estimated for this event is 1.3×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.27.1, involves the observed scram and failure of the power conversion system and RHR.

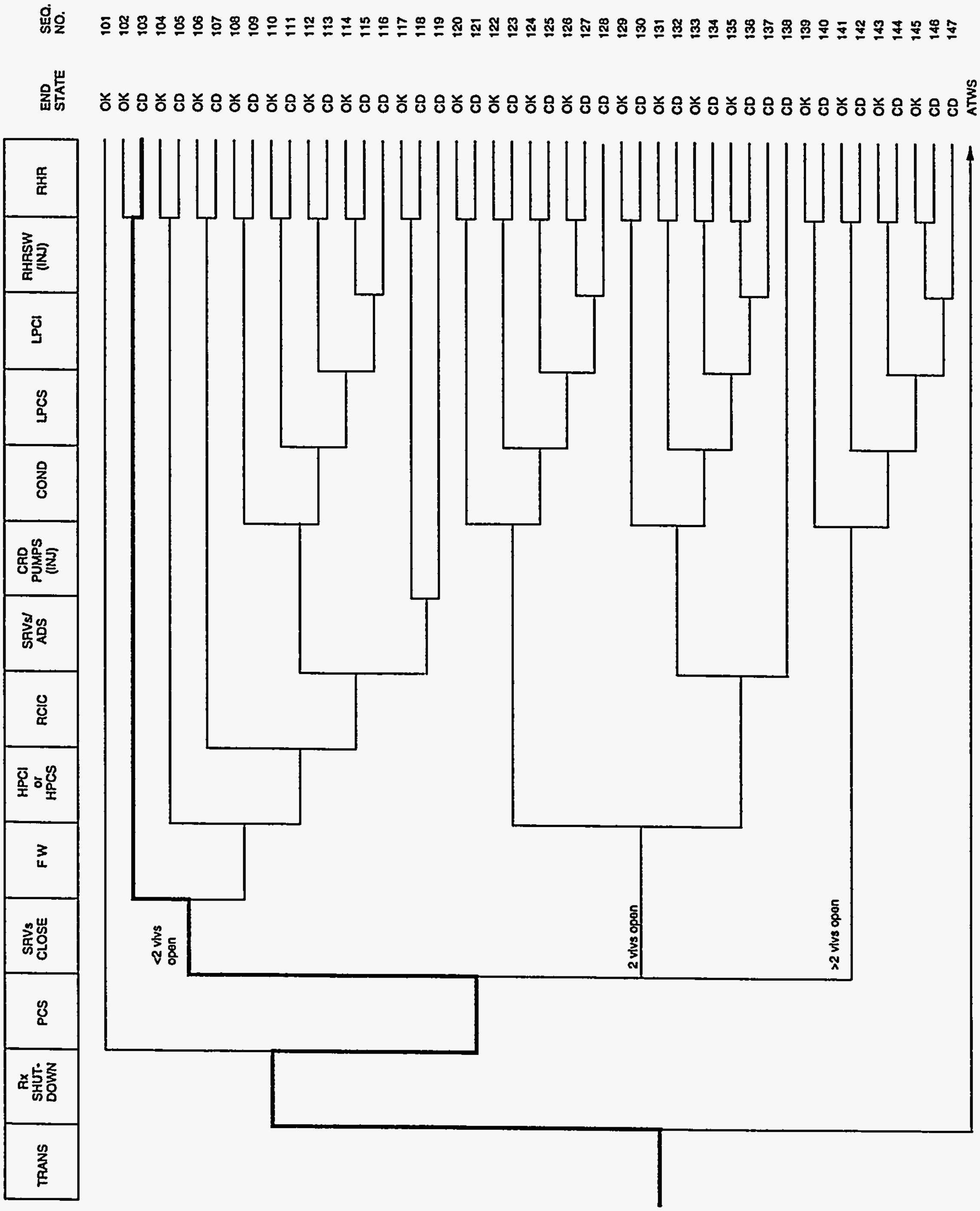

Figure B.27.1 Dominant core damage sequence for LER 325/82-025

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 325/82-025
Event Description: Scram with RCIC inoperable
Event Date: February 18. 1982
Plant: Brunswick 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.3E-05
Total	1.3E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.2E-06	1.2E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.2E-06	1.2E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp

BRANCH MODEL: c:\asp\1982-83\bruns1.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.0E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.28 LER No. 325/82-041

Event Description: Both RHRSW Loops Simultaneously Inoperable

Date of Event: March 25, 1982

Plant: Brunswick 1

B.28.1 Summary

Loop B of the residual heat removal (RHR) service water (SW) system was found to be inoperable during a period when loop A was tagged out for maintenance. The estimated increase in core damage probability, or importance, over the duration of the event is 4.7×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 2.9×10^{-7}, resulting in an estimated conditional core damage probability (CCDP) of 4.7×10^{-5}.

B.28.2 Event Description

During power operations, an auxiliary operator discovered that the motor cooler isolation valves to 1B and 1D pumps were open with the pumps not running. Investigating, he found that the breaker which supplied the motor cooler isolation valves had tripped, de-energizing the valves as well as the B loop RHRSW low-suction header pressure switches. In turn, this rendered the B loop of RHRSW inoperable. At the same time, loop A RHRSW was tagged out for maintenance.

B.28.3 Additional Event-Related Information

Loop B of RHRSW had been flushed approximately two days prior to the event, which entailed manipulation of the breaker in question.

B.28.4 Modeling Assumptions

It was assumed that RHRSW loop B was unavailable from the time of the system flush until discovery of the mispositioned breaker. Loop A was assumed to have been unavailable throughout this period as well. This event was modeled as a two-day unavailability of RHRSW and, accordingly, of RHR. The RHRSW pumps at Brunswick maintain a positive pressure differential between the tube and shell side of the RHR heat exchangers, which prevents primary coolant leakage into the service water (SW) system. Adequate decay heat removal can be provided using the SW pump if valves FO68A and B are locally opened. This action is addressed in the Brunswick individual plant examination (IPE), and an operator error probability of 0.01 was estimated. To address this section, the nonrecovery probability for RHR was revised to 0.01 to reflect the probability of the operators failing to open FO68A and B. For sequences involving potential RHR and PCS recovery, the nonrecovery estimate was revised to 0.01×0.52 (see Appendix A), or 5.2×10^{-3}.

B.28.5 Analysis Results

The estimated increase in core damage probability over the duration of the event is 4.7×10^{-5}. The base-case CDP (not shown in calculation) is 2.9×10^{-7}, resulting in an estimated CCDP of 4.7×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.28.1, involves a transient with failure of the power conversion system, main feedwater success, and RHR failure.

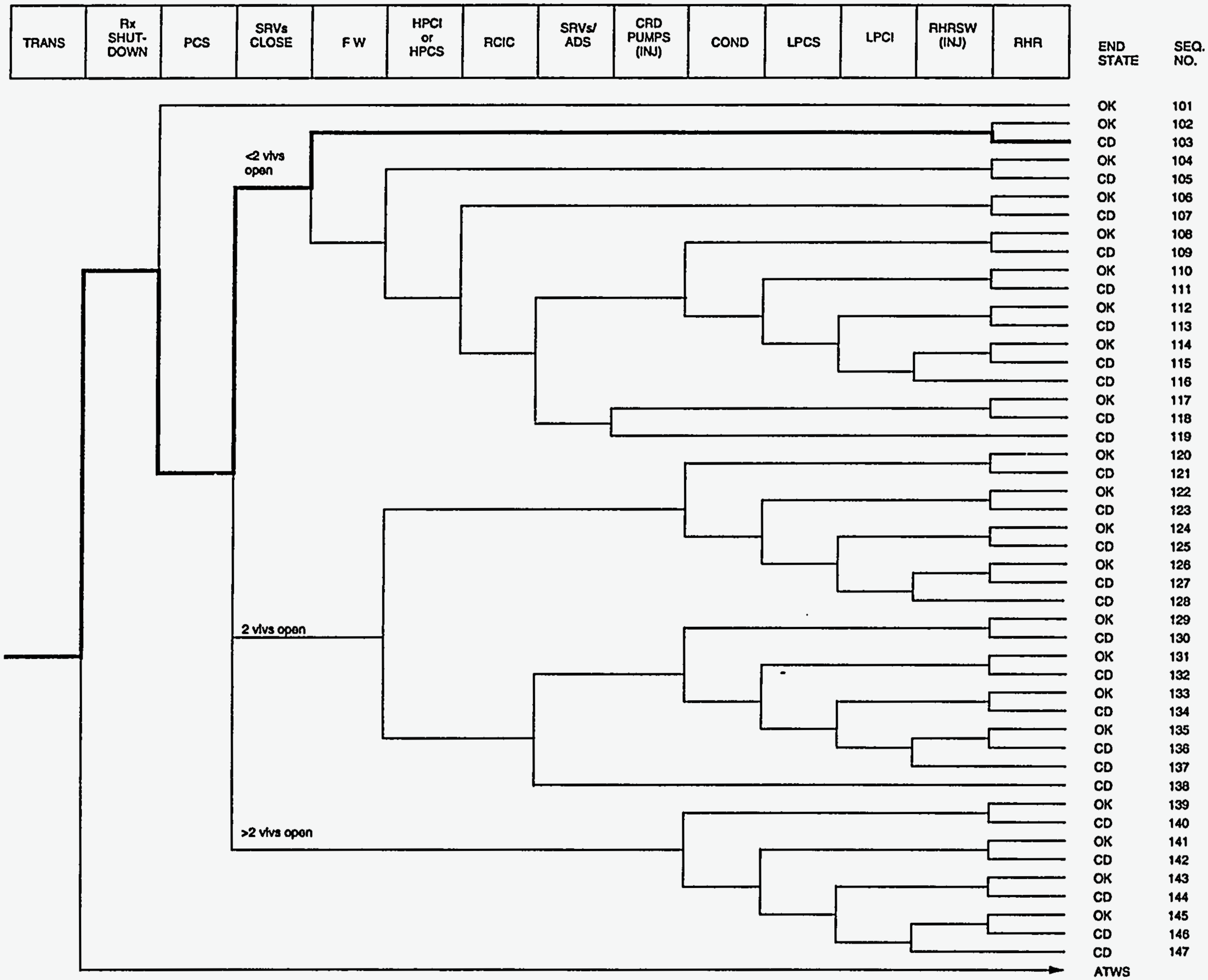

Figure B.28.1 Dominant core damage sequence for LER 325/82-041

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 325/82-041
Event Description: Both RHRSW loops simultaneously inoperable
Event Date: March 25. 1982
Plant: Brunswick 1

UNAVAILABILITY. DURATION= 48

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	5.0E-02
LOOP	2.8E-04
LOCA	1.1E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	4.4E-05
LOOP	2.8E-06
LOCA	1.8E-07
Total	4.7E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	3.7E-05	4.6E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-03
202	loop -rx.shutdown -ep srv.ftc.<2 -hpci RHR	CD	2.7E-06	3.6E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	3.7E-05	4.6E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-03
202	loop -rx.shutdown -ep srv.ftc.<2 -hpci RHR	CD	2.7E-06	3.6E-03

** non-recovery credit for edited case

Note: For unavailabilities. conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\bruns1.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.0E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	4.6E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
RHRSW(INJ)	2.0E-02 > 1.0E+00	1.0E+00	1.0E-02
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	2.0E-02 > 1.0E+00		
RHR	1.5E-04 > 1.0E+00	1.6E-02 > 1.0E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01 > 1.0E+00		
Train 3 Cond Prob:	3.0E-01 > 1.0E+00		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
RHR.AND.PCS.NREC	1.5E-04 > 1.0E+00	8.3E-03 > 5.2E-03	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01 > 1.0E+00		
Train 3 Cond Prob:	3.0E-01 > 1.0E+00		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
RHR/-LPCI	0.0E+00 > 1.0E+00	1.0E+00 > 1.0E-02	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00 > 1.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.0E+00	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01 > 1.0E+00		
Train 3 Cond Prob:	3.0E-01 > 1.0E+00		
Train 4 Cond Prob:	5.0E-01 > 1.0E+00		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.0E+00	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00 > 1.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02

man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.29 LER No. 325/82-054

Event Description: Scram with RCIC Inoperable

Date of Event: June 7, 1982

Plant: Brunswick 1

B.29.1 Summary

After a reactor scram, the reactor core isolation cooling (RCIC) system and high-pressure coolant injection (HPCI) systems were signalled to start by a low reactor water level. A failure of the turbine speed controller caused RCIC to immediately trip, however. The conditional core damage probability estimated for the event is 1.4×10^{-5}.

B.29.2 Event Description

On June 7, 1982, a blown main steam isolation valve (MSIV) fuse caused a scram, followed by a low reactor water level start signal for HPCI and RCIC. While HPCI apparently performed normally, RCIC started and immediately tripped due to a failed electronic governor module.

B.29.3 Additional Event-Related Information

Following a similar event described in LER 325/82-069, the RCIC control system problems were eventually attributed to a control system design error. The RCIC electronic governor module was found not to have a reference signal common to the RCIC speed controller. Accordingly, variations between the two circuits in sensed ground potential caused unpredictable control system behavior. The flawed control system design apparently had existed since plant startup.

B.29.4 Modeling Assumptions

The HPCI and RCIC low-level auto-start setpoint is assumed to be the same as the reactor isolation setpoint, so the power conversion and main feedwater systems were assumed to be unavailable. This event was modeled as a scram and isolation, with RCIC unavailable and not recoverable. The long-term nonrecovery probability for the power conversion system (PCS) was revised to 0.017 to reflect the apparent isolation (see Appendix A). Combining this value with the estimated long-term residual heat removal (RHR) nonrecovery probability of 0.016 results in a combined nonrecovery probability for RHR and PCS of 2.7×10^{-4}.

B.29.5 Analysis Results

The conditional core damage probability estimated for this event is 1.4×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.29.1, involves the observed scram, failure of the power conversion system, main feedwater recovery, and RHR failure. The RCIC failure does not affect this sequence.

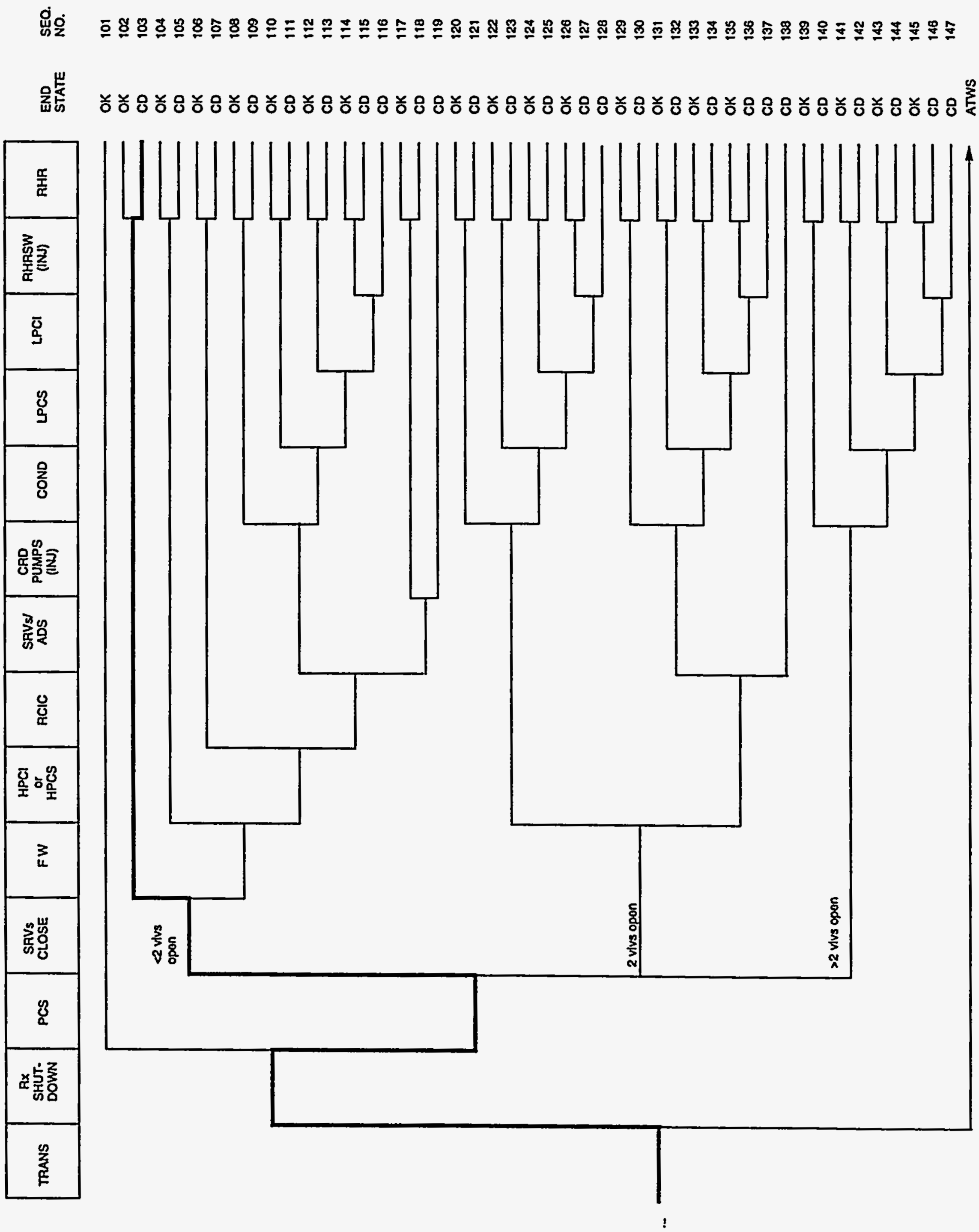

Figure B.29.1 Dominant core damage sequence for LER 325/82-054

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 325/82-054
Event Description: Scram with RCIC inoperable
Event Date: June 7. 1982
Plant: Brunswick 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.4E-05
Total	1.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\bruns1.82

PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.0E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.30 LER Nos. 327/82-048 and -050

Event Description: Unavailability of One Emergency Diesel Generator and One Motor-Driven Auxiliary Feedwater Pump

Date of Event: April 13, 1982

Plant: Sequoyah 1

B.30.1 Summary

On April 8, 1982, an automatic control valve in the auxiliary feedwater (AFW) system at Sequoyah Unit 1 failed to open on demand. The failure was caused by a faulty soldered connection to the electrohydraulic actuator of the valve. A similar event occurred on April 10, 1982. In this case, it was found that the servo valve oil passages were blocked by an accumulation of foreign matter in the filters. In both instances, the motor-driven pump in train B was rendered inoperable by the failures. Five days later (April 13, 1982), emergency diesel generator (EDG) 1A-A was declared inoperable when power fuses opened in the control circuitry. The estimated increase in core damage probability, or importance, over the duration of the event is 2.6×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 2.2×10^{-6}, resulting in an estimated conditional core damage probability (CCDP) of 2.8×10^{-5}.

B.30.2 Event Description

On April 8, 1982, Sequoyah Unit 1 was operating at 100% power when an automatic control valve (1-PCV-3-132) in the AFW system was declared inoperable due to failure to open on demand. A similar event occurred on April 10, 1982. The first event was due to a faulty soldered connection to the electrohydraulic actuator of the valve. The connector was repaired and the valve was returned to service on April 8, 1982. In the second event, it was found that the servo valve oil passages were blocked by an accumulation of foreign matter in the filters. The valve was replaced, and the control valve was returned to service on April 11, 1982.

On April 13, 1982, with Unit 1 still at 100% power, emergency diesel generator 1A-A was declared inoperable when power fuses opened in the control circuitry. The failure was due to a broken lead in the annunciator horn which had shorted to ground, causing the fuses to open due to excessive current. The horn was replaced, and the EDG was declared operable on April 13, 1982.

B.30.3 Additional Event-Related Information

The failure of valve 1-PCV-3-132 disables train B of the AFW system, since it is located in the discharge line of motor-driven pump (MDP) B. EDG 1A-A is one of two diesel generators that provide emergency power to Unit 1.

B.30.4 Modeling Assumptions

These events are modeled as a combined unavailability of one EDG and an AFW MDP. EDG 1A-A is assumed to have been inoperable for half of the 30-day surveillance period prior to April 13, 1982, i.e. starting on March 29, 1982. Similarly, MDP B is assumed to have been unavailable starting on March 24, 1982, 15 days prior to April 8, 1982. Using these assumptions, the period during which both systems were unavailable began on March 29, 1982. The end of the overlap period is April 8, 1982, when the valve which made MDP B inoperable was first returned to service and before EDG 1A-A was discovered to be inoperable. This gives an overlap period of 10 days or 240 hours, longer than the overlap associated with the second valve failure on April 10, 1982. To reflect the inoperability of EDG 1A-A, train 1 of the emergency power system was failed. EDG 1B-B was therefore subject to failure due to the same (common) cause. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event. In the AFW model, the unavailability of train B was represented by setting train 1 to failed. This recognizes the potential for a similar failure in the other train due to common cause. To represent the unavailability of power from EDG 1A-A, the second AFW train was made unavailable. In the high-pressure injection (HPI) system model, the train 2 safety injection (SI) pump was made unavailable due to the loss of EDG 1A-A. Train 1, the other SI pump, was thus susceptible only to random failures. Since train 3 of the HPI model represents the two charging pumps, which have a 2 of 2 success criterion, this train was made unavailable. Feed-and-bleed operations use the HPI system. Therefore the modifications made to the HPI model were also made to the FEED.BLEED model. In the HPR model, train 2, which represents the same SI pump used in the HPI system, was made unavailable due to the loss of EDG 1A-A, leaving train 1 subject to random failures. Finally, train 2 in the RHR and RHR.AND.HPR models and the serial component in the RHR model (representing the series RHR suction valves) were also set to unavailable due to the loss of the EDG. A loss-of-offsite power (LOOP) was used as the potential initiator for the unavailability analysis. The base-case CDP (not shown in calculation) is 2.2×10^{-6} and the CCDP is 2.8×10^{-5}.

B.30.5 Analysis Results

The increase in core damage probability over the duration of the event is 2.6×10^{-5}. The base-case CDP (not shown in calculation) is 2.2×10^{-6}, resulting in an estimated CCDP of 2.8×10^{-5}. The dominant core damage sequence, shown in Figure B.30.1, involves a postulated LOOP, failure of emergency power, an RCP seal LOCA, and failure to recover offsite power prior to core uncovery.

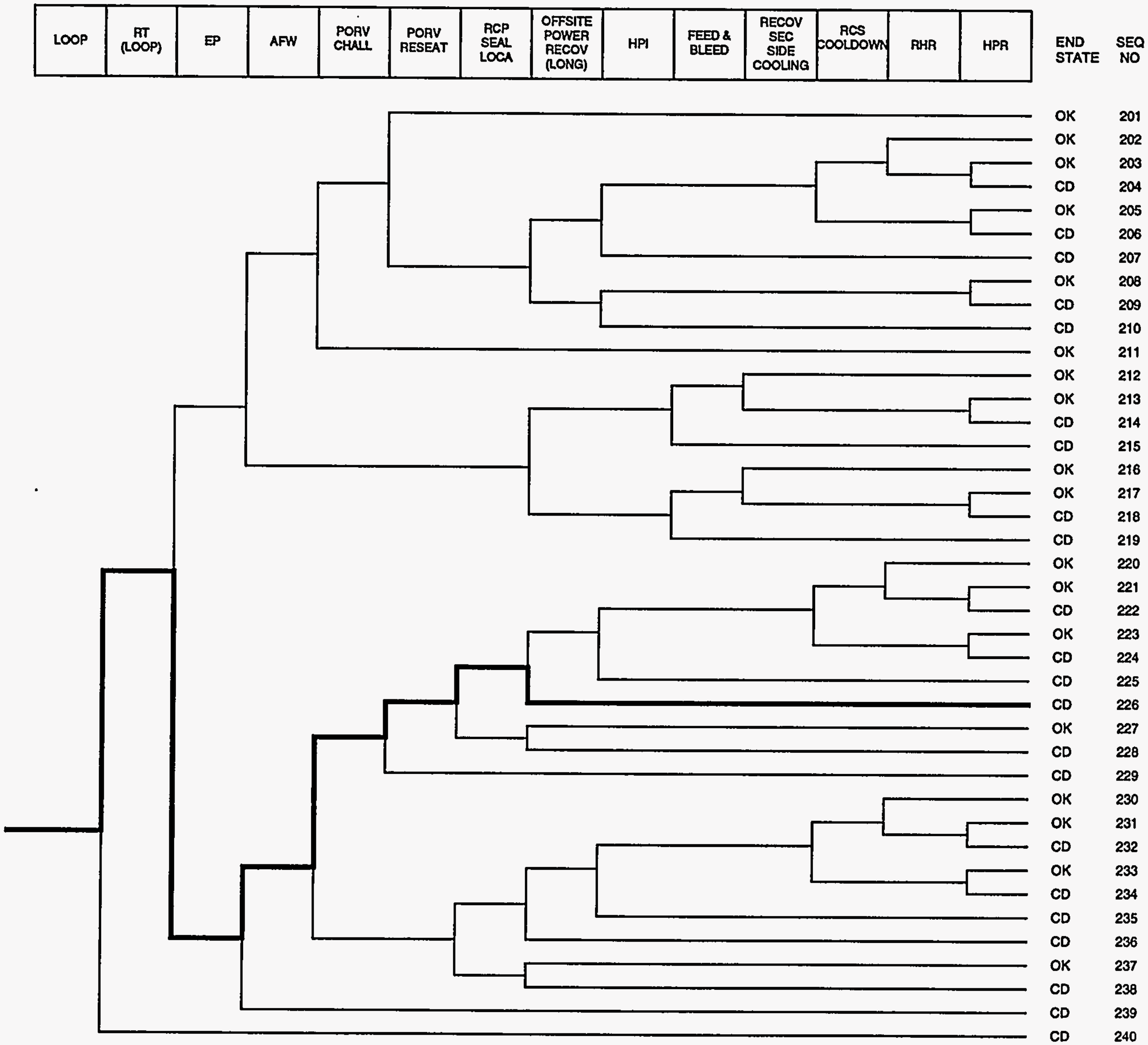

Figure B.30-1 Dominant core damage sequence for LER Nos. 327/82-048 and -050

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 327/82-048
Event Description: EDG unavailable. AFW MDP discharge valve fails
Event Date: April 13. 1982
Plant: Sequoyah 1

UNAVAILABILITY. DURATION= 240

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	2.1E-03

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	2.6E-05
Total	2.6E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
226	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep seal .loca offsite.pwr.rec/seal.loca	CD	1.5E-05	4.7E-01
228	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep -seal .loca offsite.pwr.rec/-seal.loca	CD	4.9E-06	4.7E-01
229	loop -rt(loop) EP -afw/ep porv.chall/sbo porv.reseat/ep	CD	2.0E-06	4.7E-01
239	loop -rt(loop) EP afw/ep	CD	1.7E-06	1.6E-01
215	loop -rt(loop) -EP AFW -offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	1.6E-06	2.4E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
215	loop -rt(loop) -EP AFW -offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	1.6E-06	2.4E-01
226	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep seal .loca offsite.pwr.rec/seal.loca	CD	1.5E-05	4.7E-01
228	loop -rt(loop) EP -afw/ep porv.chall/sbo -porv.reseat/ep -seal .loca offsite.pwr.rec/-seal.loca	CD	4.9E-06	4.7E-01
229	loop -rt(loop) EP -afw/ep porv.chall/sbo porv.reseat/ep	CD	2.0E-06	4.7E-01
239	loop -rt(loop) EP afw/ep	CD	1.7E-06	1.6E-01

** non-recovery credit for edited case

Note: For unavailabilities. conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\asp\1982-83\pwrb8283.cmp

BRANCH MODEL: c:\asp\1982-83\sequoy1.82
PROBABILITY FILE: c:\asp\1982-83\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.6E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 5.0E-02	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04		
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
PORV.RESEAT	2.0E-02 > 2.0E-02	1.1E-02 > 5.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-02		
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
HPI	1.0E-05 > 1.0E-02	8.9E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	1.0E-02 > Unavailable		
FEED.BLEED	2.0E-02 > 3.0E-02	1.0E+00	1.0E-02
Branch Model: 1.OF.3+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	1.0E-02 > Unavailable		
Serial Component Prob:	2.0E-02		
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
RHR	2.2E-02 > 1.0E+00	5.7E-02	1.0E-03
Branch Model: 1.OF.2+ser+opr			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Serial Component Prob:	2.0E-02 > 1.0E+00		
RHR.AND.HPR	1.0E-03 > 1.0E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HPR	4.0E-03 > 4.0E-02	1.0E+00	1.0E-03

Branch Model: 1.0F.2+opr			
Train 1 Cond Prob:	4.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
EP	2.9E-03 > 5.7E-02	8.9E-01	
Branch Model: 1.0F.2			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02		
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.31 LER No. 327/83-063

Event Description: Failure of Power Operated Relief Valve

Date of Event: April 21, 1983 through September 12, 1983

Plant: Sequoyah 1

B.31.1 Summary

On April 21, 1983, one of two power operated relief valves (PORVs) failed to open when operators attempted to reseat it, apparently because of a failed solenoid. The same PORV was found to be leaking through following valve maintenance on September 12, 1983. Failure of the PORV resulted in unavailability of feed and bleed. The increase in core damage probability (CDP) over the duration of the event, or importance, is 1.9×10^{-5}. The base-case CDP over the duration of the event is 6.0 $^{-5}$x 10 , resulting in an estimated conditiional core probability of 7.9×10^{-5}.

B.31.2 Event Description

On April 21, 1983, with the unit at power, one of two PORVs failed to open when operators attempted to reseat it. The associated block valve was closed and power removed. Following maintenance during an outage on September 12, 1983, with the unit in Mode 3, the same PORV was found to be leaking through. The block valve was again closed and power removed. The most probable cause for the first event was determined to be a failed solenoid coil. The second event was attributed to the valve not fully closing. No root cause for this condition was provided. To correct the problem, it was decided to replace the PORV during the next refueling outage.

B.31.3 Additional Event-Related Information

None.

B.31.4 Modeling Assumptions

The NUREG-1150 analysis of Sequoyah 1 (NUREG/CR-4550, Vol. 5, Rev.1, Part 1) indicates that both PORVs are required for successful feed and bleed. Since one PORV was presumably unavailable, the feed and bleed branch was modelled by setting the failure probability of the serial component, i.e. the PORVs, to 1. The probabilities for failure of the PORVs to reseat once challenged (branch models PORV.RESEAT and PORV.RESEAT/EP) were also revised to reflect this. The Sequoyah final safety analysis report (FSAR) specifies a maximum time between PORV tests of 18 months or between each refueling shutdown, whichever occurs sooner. The date of the last refueling outage prior to the April 23, 1983 valve failure was not provided. In the absence of this information, the time the PORV was estimated to be unavailable was assumed to be half the maximum time between required PORV tests, i.e., nine months or 6570 hours. Since this is greater than one year of reactor operation, the unavailability time was taken to be 6,132 hours (one reactor year, assuming the unit was critical 70% of the time). Transient (TRANS), loss-of-offsite power (LOOP), loss-of-coolant accident (LOCA), and steam generator tube rupture (SGTR) were used as potential initiators in the unavailability analysis.

B.31.5 Analysis Results

The increase in CDP, or importance, estimated for the event is 1.9×10^{-5}. Adding this value to the nominal CDP in the unavailability period, 6.0×10^{-5} (not shown on calculation sheet), results in an estimated conditional core damage probability of 7.9×10^{-5} The dominant sequence, highlighted on the event tree shown in Figure B.31.1, involves a postulated transient, failure of auxiliary feedwater and main feedwater, and failure of feed and bleed.

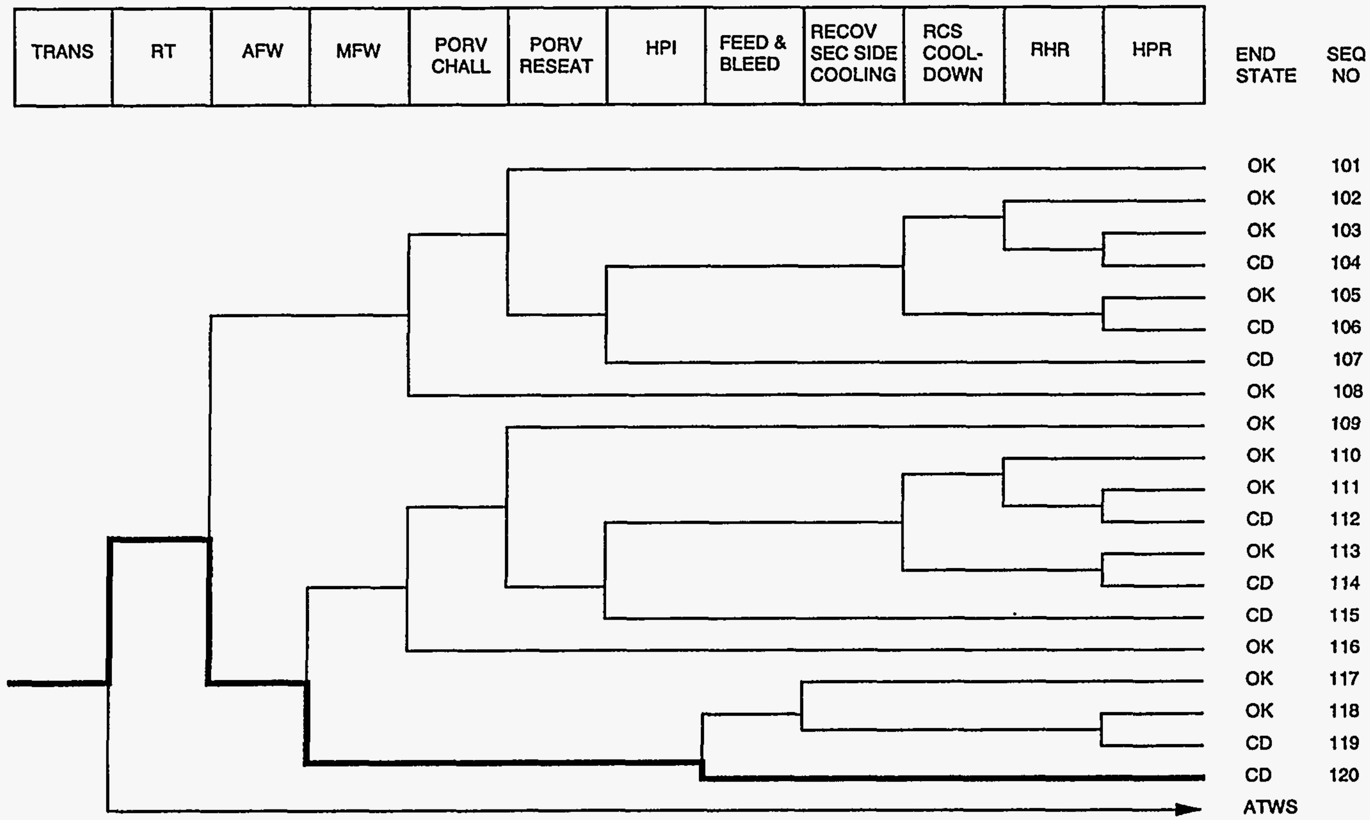

Figure B.31.1 Dominant core damage sequence for LER 327/83-063

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 327/83-063
Event Description: PORV fails to open during attempt to reseat
Event Date: 9/12/83
Plant: Sequoyah 1

UNAVAILABILITY, DURATION= 6132

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	9.6E+00
LOOP	5.3E-02
LOCA	7.9E-03
SGTR	1.0E-02

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.1E-05
LOOP	7.3E-06
LOCA	9.0E-08
SGTR	0.0E+00
Total	1.9E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt afw mfw FEED.BLEED	CD	1.1E-05	1.5E-01
215	loop -rt(loop) -ep afw -offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	7.8E-06	2.4E-01
219	loop -rt(loop) -ep afw offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	5.6E-07	2.4E-01
229	loop -rt(loop) ep -afw/ep porv.chall/sbo PORV.RESEAT/EP	CD	(1.3E-06)	4.7E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt afw mfw FEED.BLEED	CD	1.1E-05	1.5E-01
215	loop -rt(loop) -ep afw -offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	7.8E-06	2.4E-01
219	loop -rt(loop) -ep afw offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	5.6E-07	2.4E-01
229	loop -rt(loop) ep -afw/ep porv.chall/sbo PORV.RESEAT/EP	CD	(1.3E-06)	4.7E-01

** non-recovery credit for edited case

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\asp\1982-83\pwrb8283.cmp
BRANCH MODEL: c:\asp\1982-83\sequoy1.82
PROBABILITY FILE: c:\asp\1982-83\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.6E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	3.8E-04	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
PORV.RESEAT	2.0E-02 > 1.0E-02	1.1E-02	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-02 > 1.0E-02		
PORV.RESEAT/EP	2.0E-02 > 1.0E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-02 > 1.0E-02		
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-05	8.9E-01	
FEED.BLEED	2.0E-02 > 1.0E+00	1.0E+00	1.0E-02
Branch Model: 1.OF.3+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	1.0E-02		
Serial Component Prob:	2.0E-02 > 1.0E+00		
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	5.7E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.32 LER Nos. 327/83-183 and -186

Event Description: Unavailability of One Emergency Diesel Generator and the Turbine-Driven Auxiliary Feedwater Pump

Date of Event: December 2, 1983

Plant: Sequoyah 1

B.32.1 Summary

Emergency diesel generator (EDG) 1A-A tripped due to high crankcase pressure during surveillance testing. The diesel suffered extensive damage which was determined to be due to incorrect assembly. Ten days earlier, stroke testing of the turbine-driven auxiliary feedwater (TDAFW) pump revealed that the steam supply valve was inoperable because of a failed Limitorque operator-geared limit switch. The estimated increase in core damage probability, or importance, over the duration of the event is 3.1×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 1.6×10^{-6}, resulting in an estimated conditional core damage probability (CCDP) of 3.3×10^{-5}.

B.32.2 Event Description

During performance of surveillance testing of emergency diesel generator 1A-A on December 12, 1983, it tripped due to high crankcase pressure. Unit 1 was operating at 100% power at the time of the event. Investigation of the failure revealed significant damage to the diesel, which resulted from improper torquing of the wrist pin bolts on one cylinder. The EDG was repaired and returned to service on December 12, 1983. This event was reported under LER 327/83-186. On December 2, 1983, with Unit 1 at 0% power, stroke testing of the turbine-driven auxiliary feedwater pump revealed that the steam supply valve was inoperable. The cause was a failed limitorque operator-geared limit switch. The switch was replaced and the valve was verified operable on December 12, 1983. This event was reported under LER 327/83-183.

B.32.3 Additional Event-Related Information

LER 327/83-183 states that the steam supply valve could have been in a failed condition during power operation. EDG 1A-A is one of two diesel generators that provide emergency power to Unit 1.

B.32.4 Modelling Assumptions

These events are modeled as a combined unavailability of one EDG and the TDAFW pump. Since steam supply valve 1-FCV-1-18 was found to be failed on December 2, 1983, it was assumed that it was failed on November 17, 1983, 15 days after the last pump test one month earlier. On December 12, 1983, EDG 1A-A was found to be failed. Given the fact that the failure was due to incorrect assembly, it was assumed that it would have been unavailable since the last test one month earlier on November 11, 1983. Therefore, both systems would have been unavailable from November 17, 1983 until December 12, 1983 when the EDG was the first of the two systems to be returned to service. This unavailability period of 25 days (600 hours) was reduced to 168 hours to reflect the fact that the unit tripped on November 24, 1983. With EDG 1A-A inoperable, train 2 of the emergency power system model was failed to reflect the assumption that EDG 1B-B would not be likely to fail due to the same cause since, according to LER 327/83-186, a similar failure had not occurred in a factory-assembled unit since 1979. In the AFW model, train 3, which represents the TDP, was failed. The single train in the AFW/EP model represents the TDP, so it was also failed. Trains 1 and 2 represent the two motor-driven pumps (MDPs). Since one MDP would be unavailable due to the failure of EDG 1A-A, train 2 was set to unavailable, making train 1 only susceptible to independent (random) failures. Similarly, the high-pressure injection (HPI) and high-pressure recirculation (HPR) system models were modified by setting train 2 to unavailable due to the loss of EDG 1A-A. Since train 3 of the HPI model represents the two charging pumps, which have a 2 of 2 success criteria, the failure probability was also set to unavailable. Feed-and-bleed operations use the HPI system. Therefore, the modifications made to the HPI model were also made to the FEED.BLEED model. A loss of offsite power (LOOP) was used as the potential initiator for the unavailability analysis.

B.32.5 Analysis Results

The estimated increase in core damage probability over the duration of the event is 3.1×10^{-5}. The base-case CDP (not shown in calculation) is 1.6×10^{-6}, resulting in an estimated CCDP of 3.3×10^{-5}. The dominant core damage sequence, shown in Figure B.32.1, involves a postulated LOOP, failure of emergency power, and loss of AFW.

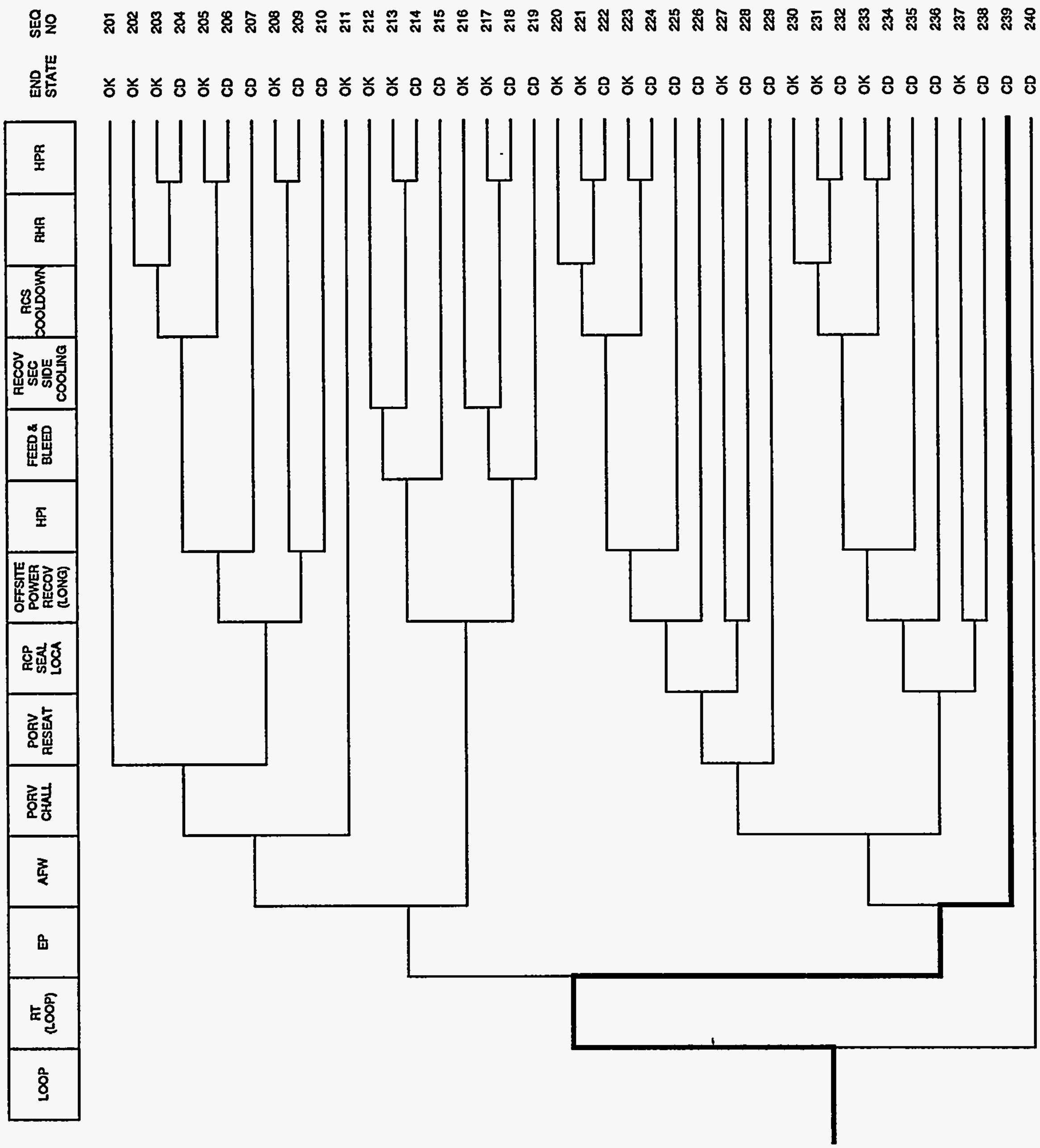

Figure B.32.1 Dominant core damage sequence for LER Nos. 327/83-183 and -186

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 327/83-183 and -186
Event Description: EDG trips during test. valve on TDAFW pump fails test
Event Date: December 2. 1983
Plant: Sequoyah 1

UNAVAILABILITY. DURATION= 168
NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	1.5E-03

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	3.1E-05
Total	3.1E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
239	loop -rt(loop) EP AFW/EP	CD	2.2E-05	1.6E-01
226	loop -rt(loop) EP -AFW/EP porv.chall/sbo -porv.reseat/ep seal .loca offsite.pwr.rec/seal.loca	CD	5.9E-06	3.1E-01
228	loop -rt(loop) EP -AFW/EP porv.chall/sbo -porv.reseat/ep -seal .loca offsite.pwr.rec/-seal.loca	CD	2.0E-06	3.1E-01
229	loop -rt(loop) EP -AFW/EP porv.chall/sbo porv.reseat/ep	CD	7.8E-07	3.1E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
226	loop -rt(loop) EP -AFW/EP porv.chall/sbo -porv.reseat/ep seal .loca offsite.pwr.rec/seal.loca	CD	5.9E-06	3.1E-01
228	loop -rt(loop) EP -AFW/EP porv.chall/sbo -porv.reseat/ep -seal .loca offsite.pwr.rec/-seal.loca	CD	2.0E-06	3.1E-01
229	loop -rt(loop) EP -AFW/EP porv.chall/sbo porv.reseat/ep	CD	7.8E-07	3.1E-01
239	loop -rt(loop) EP AFW/EP	CD	2.2E-05	1.6E-01

** non-recovery credit for edited case

Note: For unavailabilities. conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\asp\1982-83\pwrb8283.cmp
BRANCH MODEL: c:\asp\1982-83\sequoy1.82
PROBABILITY FILE: c:\asp\1982-83\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.6E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 2.0E-02	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
afw/atws	4.3E-03	1.0E+00	
AFW/EP	5.0E-02 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.0E-02 > Failed		
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
PORV.RESEAT	2.0E-02 > 2.0E-02	1.1E-02 > 5.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-02		
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
HPI	1.0E-05 > 1.0E-02	8.9E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	1.0E-02 > Unavailable		
FEED.BLEED	2.0E-02 > 3.0E-02	1.0E+00	1.0E-02
Branch Model: 1.OF.3+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	1.0E-02 > Unavailable		
Serial Component Prob:	2.0E-02		
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
RHR	2.2E-02 > 1.0E+00	5.7E-02	1.0E-03
Branch Model: 1.OF.2+ser+opr			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Serial Component Prob:	2.0E-02 > 1.0E+00		
RHR.AND.HPR	1.0E-03 > 1.0E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HPR	4.0E-03 > 4.0E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	4.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
EP	2.9E-03 > 5.0E-02	8.9E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02 > Failed		

seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.33 LER No. 331/83-017 and -018

Event Description: HPCI and RHRSW Loop B Inoperable

Date of Event: May 23, 1983

Plant: Duane Arnold

B.33.1 Summary

During normal operation on May 23, 1983, the pressure differential across the residual heat removal service water (RHRSW) system strainer 1S-90B increased. The strainer jammed with river water debris, causing the drive motor coupling shear pins to shear. On the same day, during routine surveillance testing, the high-pressure coolant injection (HPCI) pump did not meet Technical Specification output pressure flow requirements. Investigation revealed that the HPCI speed indication circuit for the turbine was out of calibration. The estimated increase in core damage probability, or importance, over the duration of the event is 1.7×10^{-5}. The base-case core damage probability (CDP) over the duration of the event is 1.0×10^{-7}, resulting in an estimated conditional core damage probability (CCDP) of 1.7×10^{-5}.

B.33.2 Event Description

During normal operation on May 23, 1983, the pressure differential across the residual heat removal service water system strainer 1S-90B increased. The strainer jammed with river water debris, causing the drive motor coupling shear pins to shear as per design. The B loop of RHRSW was declared inoperable, and the redundant loop was satisfactorily tested. The strainers were cleaned and the shear pins were replaced. On the same day, during routine surveillance testing, the HPCI pump did not meet Technical Specification output pressure flow requirements. Technical Specifications require an output pressure of 1,050 psig at 3,700 RPM and 3000 gpm. The test measured an output pressure of 780 psig. Investigation revealed that the HPCI speed indication circuit for the turbine was out of calibration. The turbine speed was actually lower than indicated. The speed circuit was recalibrated and the pump tested satisfactorily.

B.33.3 Additional Event-Related Information

The RHRSW system provides cooling water to the residual heat removal (RHR) system heat exchangers. The RHR system provides three functions: suppression pool cooling, containment spray, and shutdown cooling. Suppression pool cooling is used to remove heat from the suppression pool whenever the water temperature exceeds 95°F. Containment spray is used in the event of a nuclear system break within the primary containment to prevent excessive containment pressure and temperature by condensing steam and cooling noncondensable gases. Shutdown cooling can be used during normal shutdown and cooldown to remove decay heat, once the reactor coolant temperature is low enough that the steam supply pressure is not sufficient to maintain turbine shaft gland seals or vacuum in the main condenser. RHR requires the use of at least one heat exchanger (and thus RHRSW) for all three modes.

RHRSW is a two-loop system (A and B). Each loop has two pumps (1P22A and 1P22C, and 1P22B and 1P22D, respectively) and one heat exchanger (1E201A and 1E201B). Each pair of pump discharge lines connects to a common line which flows through a self-cleaning strainer and then to an RHR heat exchanger. If the strainers become clogged, flow to the RHR heat exchangers is degraded. One pump supplying one heat exchanger is sufficient to cool all RHR.

RHRSW also has a crosstie which enables the RHRSW pumps to provide coolant to the RHR system for use as an alternative injection system. Flow from the RHRSW common lines proceeds through the strainers to the crosstie line. The crosstie line contains two motor-operated valves in series, which must be opened for injection. One pump is sufficient to provide the alternative injection source for RHR.

B.33.4 Modeling Assumptions

HPCI was assumed to be inoperable for half its surveillance period, 360 hours. Since HPCI did not pass Technical Specification requirements due to a miscalibrated turbine speed indication circuit, the HPCI pump was assumed to be failed and nonrecoverable. The clogged strainers were assumed to lead to degraded RHR. Since the RHR system model is composed of four pump trains and two of the four trains flow through one heat exchanger which is cooled by RHRSW, two of the four trains were assumed to be failed. Since it is likely that the RHRSW loop A strainer could also get clogged with river debris, the first two trains of RHR and RHR (SPCOOL) were set to failed to reflect the potential for a common cause failure due to river debris in loop A. The probability of RHRSW injection failure was also revised to reflect the potential failure of both RHRSW loops. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event. It is unlikely that clogged strainers would go unnoticed for more than 24 hours, so this event was modeled as the unavailability of HPCI, two RHR pump trains, and degraded RHRSW injection for a period of 24 hours. The nonrecovery probability for RHR was revised to 0.054 to reflect the RHRSW failures (see Appendix A). For sequences involving potential RHR or power conversion system (PCS) recovery, the nonrecovery estimates were revised to 0.054 x 0.52 (PCS nonrecovery), or 0.028.

B.33.5 Analysis Results

The estimated increase in core damage probability over the duration of this event is 1.7×10^{-5}. The base-case CDP (not shown in calculation) is 1.0×10^{-7}, resulting in an estimated CCDP of 1.7×10^{-5}. The dominant sequence is a postulated transient with a successful reactor shutdown, failure of PCS, successful feedwater, and failure of RHR, and is shown in Figure B.33.1.

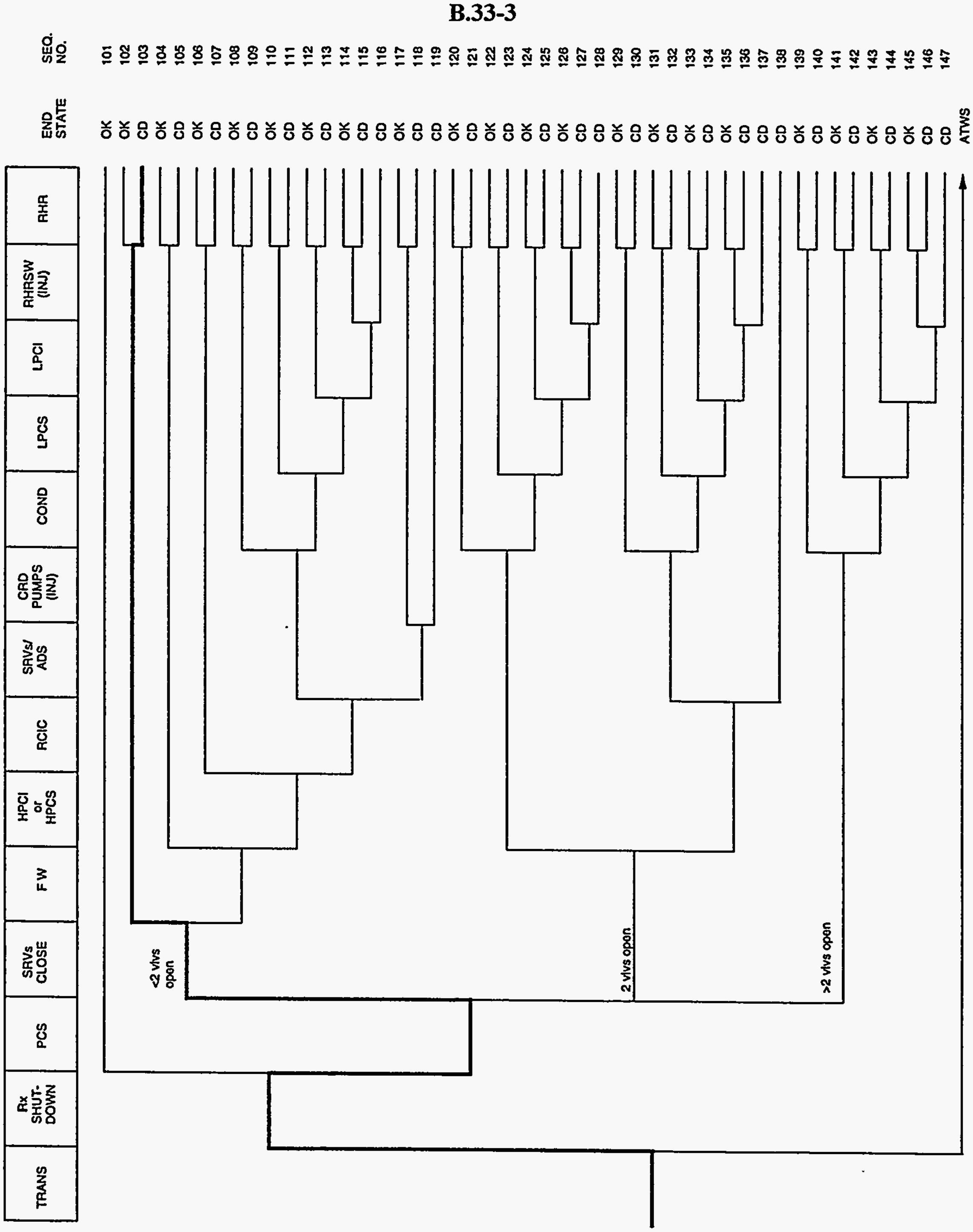

Figure B.33.1 Dominant core damage sequence for LER 331/83-017 and -018

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 331/83-017
Event Description: HPCI and RHRSW loop B inop
Event Date: May 23. 1983
Plant: Duane Arnold

UNAVAILABILITY. DURATION= 24

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	2.3E-02
LOOP	1.4E-04
LOCA	5.3E-05

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.6E-05
LOOP	1.2E-06
LOCA	7.3E-08
Total	1.7E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.4E-05	2.6E-02
107	trans -rx.shutdown pcs srv.ftc.<2 mfw HPCI -rcic RHR.AND.PC S.NREC	CD	1.5E-06	9.4E-03
204	loop -rx.shutdown -ep srv.ftc.<2 HPCI -rcic RHR	CD	1.1E-06	1.9E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.4E-05	2.6E-02
107	trans -rx.shutdown pcs srv.ftc.<2 mfw HPCI -rcic RHR.AND.PC S.NREC	CD	1.5E-06	9.4E-03
204	loop -rx.shutdown -ep srv.ftc.<2 HPCI -rcic RHR	CD	1.1E-06	1.9E-02

** non-recovery credit for edited case

Note: For unavailabilities. conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp

BRANCH MODEL: d:\asp\models\duarnold.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	9.5E-04	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
HPCI	2.9E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-02 > 1.0E+00		
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
RHRSW(INJ)	2.0E-02 > 0.012	1.0E+00	1.0E-02
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	2.0E-02 > 0.012		
RHR	1.5E-04 > 1.5E-01 **	1.6E-02 > 5.4E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-01 **	8.3E-03 > 2.8E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.5E-01 **	1.0E+00 > 5.4E-02	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.5E-01 **	1.0E+00 > 5.4E-02	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	6.6E-02	1.0E+00	

rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.34 LER No. 335/82-040

Event Description:	Reactor Trip and Loss of Grid Synchronization Due to Shorting of Generator Relay During Testing
Date of Event:	September 2, 1982
Plant:	St. Lucie 1

B.34.1 Summary

On September 2, 1982, personnel conducting a test of a generator trip relay short circuited it, which caused the generator breakers to open and a reactor/turbine trip. The spurious operation of the generator breakers allowed the unit to slip out of synchronization with the grid. Transfer of the vital buses to startup power did not occur and the emergency power system was actuated. The conditional core damage probability estimated for this event is 3.1×10^{-5}.

B.34.2 Event Description

During full power operation, a generator trip relay was briefly shorted while being tested. This caused the generator breakers to open and a synchronizing inhibit timer to start. By the time the reactor tripped due to a turbine overspeed trip, the timer had cycled, so transfer of the vital buses to startup power did not occur. The diesel generators started automatically and loaded properly. Offsite power and normal plant status were restored about 28 minutes after the short circuit occurred.

B.34.3 Additional Event-Related Information

A similar bus loss was reported in LER 335/79-028.

B.34.4 Modelling Assumptions

Since this event, in effect, isolated the plant from offsite power, it was modeled as a plant-centered loss of offsite power (LOOP). However, this is probably conservative since the event involved a failure to transfer only the vital buses. Changes to LOOP-related branch probabilities to reflect the plant-centered LOOP are shown in the following table:

Branch	Description	Probability
SEAL.LOCA	Probability that an RCP seal LOCA will occur.	4.0×10^{-2}
OFFSITE.PWR.REC/ -EP.AND-AFW	Probability of failing to recover offsite power within 2 hours given that EP and AFW are successful.	1.4×10^{-1}
OFFSITE.PWR.REC/ -EP.AND.AFW	Probability of failing to recover offsite power within 6 hours given that EP is successful but AFW fails.	9.9×10^{-4}
OFFSITE.PWR.REC/ SEAL.LOCA	Probability of failing to recover offsite power given the occurrence of an RCP seal LOCA.	4.8×10^{-1}
OFFSITE.PWR.REC/ -SEAL.LOCA	Probability of failing to recover offsite power given that there is no RCP seal LOCA.	2.2×10^{-5}

EP - emergency power
AFW - auxiliary feedwater
RCP - reactor coolant pump

B.34.5 Analysis Results

The conditional core damage probability (CCDP) estimated for this event is 3.1×10^{-5}. The dominant core damage sequence, shown in Figure. B.34.1, involves the effective LOOP, successful reactor trip, failure of emergency power (EP), success of AFW, power-operated relief valve (PORV) challenge, and failure of the PORVs to reseat.

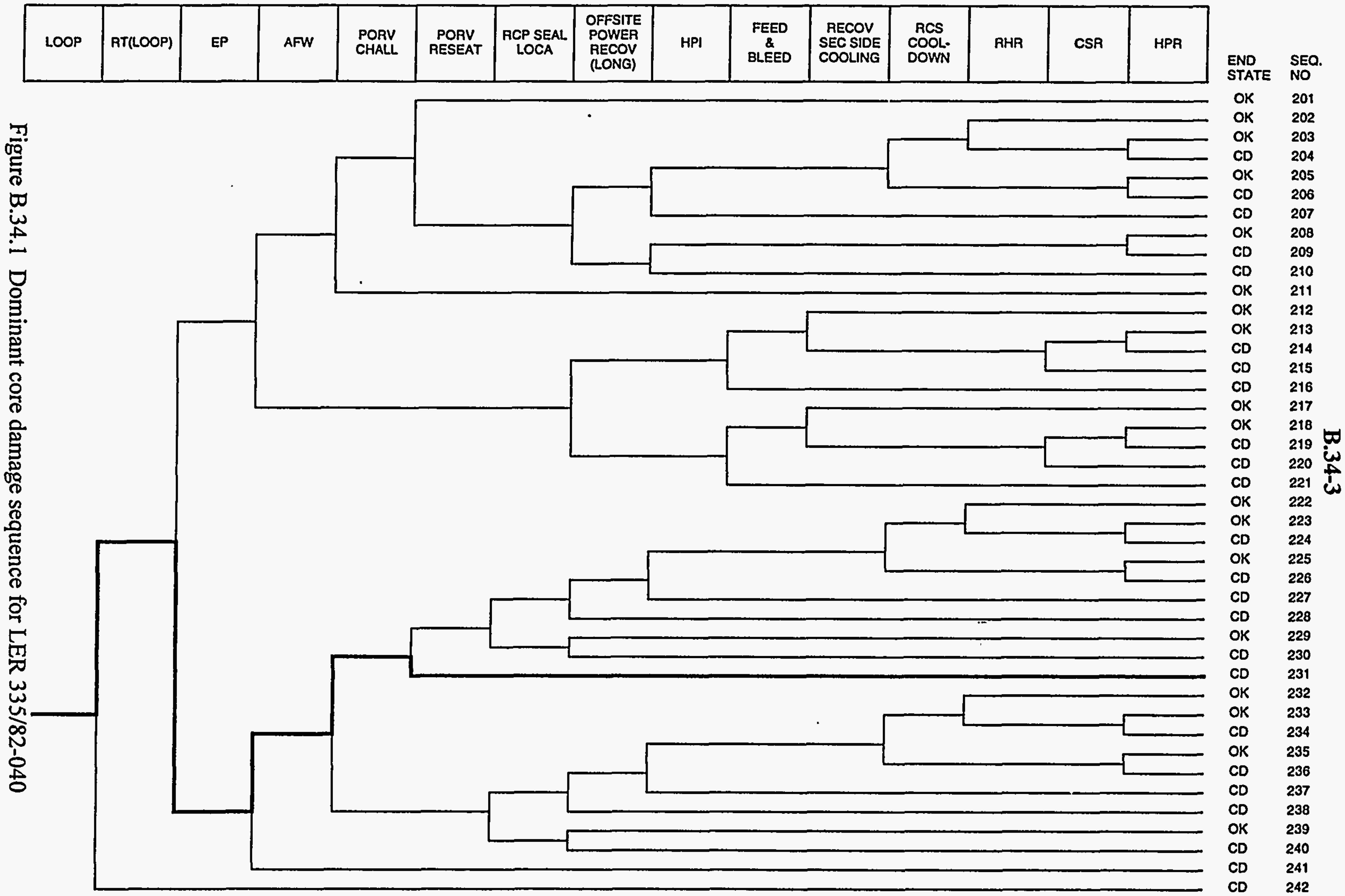

Figure B.34.1 Dominant core damage sequence for LER 335/82-040

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 335/82-040
Event Description: Reactor trip due to shorted generator relay
Event Date: 9/2/82
Plant: St. Lucie 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	2.1E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	3.1E-05
Total	3.1E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
231	loop -rt(loop) ep -afw/ep porv.chall/sbo porv.reseat/ep	CD	1.0E-05	1.8E-01
228	loop -rt(loop) ep -afw/ep porv.chall/sbo -porv.reseat/ep SEAL .LOCA OFFSITE.PWR.REC/SEAL.LOCA	CD	9.9E-06	1.8E-01
241	loop -rt(loop) ep afw/ep	CD	9.1E-06	6.4E-02
216	loop -rt(loop) -ep afw -OFFSITE.PWR.REC/-EP.AND.AFW feed.bleed	CD	1.1E-06	9.4E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
216	loop -rt(loop) -ep afw -OFFSITE.PWR.REC/-EP.AND.AFW feed.bleed	CD	1.1E-06	9.4E-02
228	loop -rt(loop) ep -afw/ep porv.chall/sbo -porv.reseat/ep SEAL .LOCA OFFSITE.PWR.REC/SEAL.LOCA	CD	9.9E-06	1.8E-01
231	loop -rt(loop) ep -afw/ep porv.chall/sbo porv.reseat/ep	CD	1.0E-05	1.8E-01
241	loop -rt(loop) ep afw/ep	CD	9.1E-06	6.4E-02

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\models\pwrg8283.cmp
BRANCH MODEL: c:\asp\models\sluciel.82
PROBABILITY FILE: c:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	7.2E-04	1.0E+00	
loop	6.7E-05	2.1E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	3.8E-04	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	8.0E-03	7.0E-02	1.0E-03
csr	4.0E-03	1.0E+00	1.0E-03
hpr	1.5E-04	1.0E+00	
ep	2.9E-03	8.9E-01	
SEAL.LOCA	4.8E-02 > 4.0E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.8E-02 > 4.0E-02		
OFFSITE.PWR.REC/-EP.AND.-AFW	2.5E-01 > 1.4E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.5E-01 > 1.4E-01		
OFFSITE.PWR.REC/-EP.AND.AFW	5.7E-02 > 9.9E-04	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.7E-02 > 9.9E-04		
OFFSITE.PWR.REC/SEAL.LOCA	6.0E-01 > 4.8E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-01 > 4.8E-01		
OFFSITE.PWR.REC/-SEAL.LOCA	1.1E-02 > 2.2E-05	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.1E-02 > 2.2E-05		
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.35 LER No. 339/82-009

Event Description: PORVs Inoperable due to Low Nitrogen Pressure

Date of Event: March 8, 1982

Plant: North Anna 2

B.35.1 Summary

On March 8, 1982, while cooling down for a refueling outage, the pressurizer PORVs were determined to be inoperable due to low nitrogen pressure. The increase in core damage probability (CDP), or importance, over the duration of the event is 4.3×10^{-6}. The base-case CDP over the duration of the event is 8.1×10^{-5}, resulting in an estimated conditional core damage probability (CCDP) of 8.5×10^{-5}.

B.35.2 Event Description

On March 8, 1982, while the unit was in Mode 4 and cooling down to begin a refueling outage, the pressurizer power-operated relief valves (PORVs) were declared inoperable due to low nitrogen supply tank pressure.

The nitrogen tanks were refilled and the PORVs were restored to operability after about 3 hours. The PORVs were again declared inoperable due to inadequate nitrogen supply later in the same day, and the nitrogen tank was replenished again. A similar event occurred on March 10, 1982.

B.35.3 Additional Event-Related Information

The licensee event report for this event indicates that the PORV nitrogen supply failures were caused by excessive use of nitrogen, excessive system leakage, and an inadequate makeup supply. The nitrogen supply is required to operate the PORVs when instrument air is unavailable.

B.52.4 Modeling Assumptions

It was assumed that the PORV nitrogen system failures were latent during the prior operating cycle. Since the duration of the failures was not known, they were assumed to have existed for one-half of the annual operating cycle or 3,066 hours. It was assumed that the PORV nitrogen supply would have been required for PORV operation during postulated loss of offsite power (LOOP) events and that the PORVs would therefore not have been available for feed-and-bleed operations during LOOPs.

B.52.5 Analysis Results

The increase in CDP, or importance, estimated for the event is 4.3×10^{-6}. Adding this value to the nominal CDP in the unavailability period, 8.1×10^{-5}, results in an estimated CCDP of 8.5×10^{-5}. The dominant sequence, highlighted on the event tree shown in Figure B.35.1, involves a postulated transient, failure of auxiliary feedwater and main feedwater, and failure of feed and bleed.

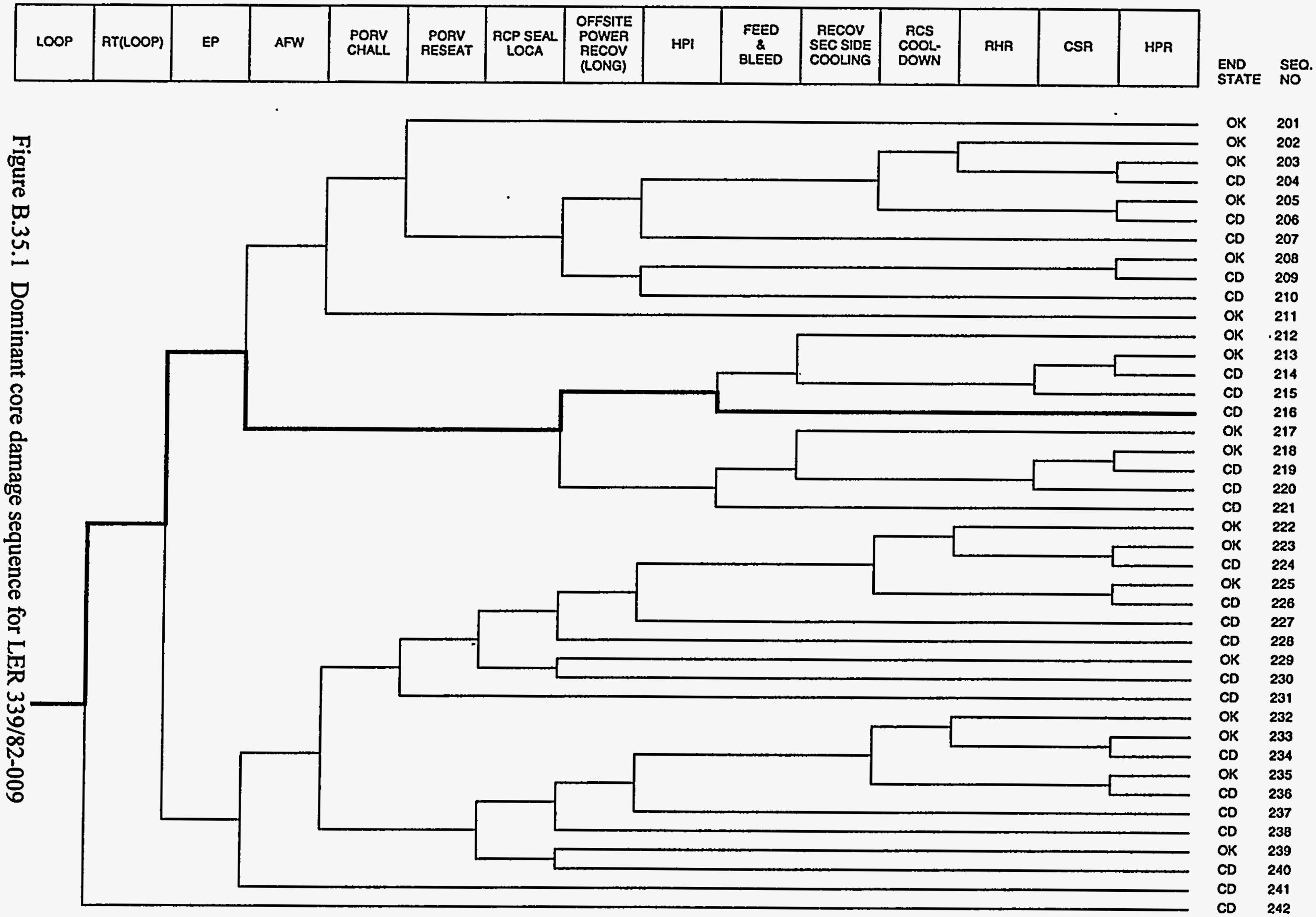

Figure B.35.1 Dominant core damage sequence for LER 339/82-009

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 339/82-009
Event Description: PORVs inoperable due to low nitrogen pressure
Event Date: March 8. 1982
Plant: North Anna 2

UNAVAILABILITY. DURATION= 3066

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	2.6E-02

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	4.3E-06
Total	4.3E-06

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

State	Sequence	End	Prob	N Rec**
216	loop -rt(loop) -ep afw -offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	4.0E-06	2.4E-01
221	loop -rt(loop) -ep afw offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	2.9E-07	2.4E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

State	Sequence	End	Prob	N Rec**
216	loop -rt(loop) -ep afw -offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	4.0E-06	2.4E-01
221	loop -rt(loop) -ep afw offsite.pwr.rec/-ep.and.afw FEED.BLEED	CD	2.9E-07	2.4E-01

** non-recovery credit for edited case

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: d:\asp\models\pwra8283.cmp
BRANCH MODEL: d:\asp\models\nanna2.82
PROBABILITY FILE: d:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	8.0E-04	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	

Event Identifier: 339/82-009

afw	3.8E-04	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.5E-03	8.9E-01	
FEED.BLEED	2.0E-02 > 1.0E+00 **	1.0E+00	1.0E-02
Branch Model: 1.OF.3+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Serial Component Prob:	2.0E-02		
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	5.7E-02	1.0E-03
csr	7.5E-04	1.0E+00	
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.2E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	6.7E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.7E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	7.0E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.36 LER No. 344/83-002

Event Description: Reactor Trip with Main Feedwater and Two Auxiliary Feedwater Pumps Unavailable

Date of Event: January 22, 1983

Plant: Trojan

B.36.1 Summary

On January 22, 1983 a high-high steam generator level caused the main feedwater (MFW) pump to trip. The auxiliary feedwater (AFW) pumps auto started, but were manually shut down by the operator. The levels in two steam generators then decreased sufficiently to cause a reactor trip. The operator attempted to restart the AFW pumps, but was unsuccessful. Action was taken to reset the AFW pumps locally while the motor-driven nonengineered safety feature (ESF) AFW pump was started, which reestablished feedwater to the steam generators. Feedwater flow to the steam generators was lost for seven minutes. The conditional core damage probability estimated for this event is 9.7×10^{-5}.

B.36.2 Event Description

On January 22, 1983 with the unit at 4% power in mode 2, a high-high steam generator level caused the MFW pump to trip. The AFW pumps auto started, but the operator, assuming MFW was still operating, manually shut down the AFW pumps. The levels in steam generators B and C then decreased to the low-low level setpoint, which caused a reactor trip. Realizing that the MFW pump had tripped, the operator attempted to restart the AFW pumps. The diesel-driven pump would not start and the steam-driven pump started but tripped on overspeed. An operator was sent to reset the AFW pumps locally while the motor-driven non-ESF AFW pump was started, which re-established feedwater to all four steam generators. Feedwater flow to the steam generators was lost for seven minutes.

B.36.3 Additional Event-Related Information

The diesel-driven AFW pump was secured before reaching full speed. This prevented restart from the control room. Both pumps were reset locally and restarted.

B.36.4 Modeling Assumptions

This event was modeled as a transient with MFW and two AFW pumps unavailable. This was reflected in the analysis by setting train 1 of main feedwater (MFW) and trains 1 and 2 of AFW (the steam- and diesel-driven pumps) to unavailable. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event.

B.36.5 Analysis Results

The estimate of the conditional core damage probability (CCDP) resulting from this event is 9.7×10^{-5}. The dominant core damage sequence, shown in Figure B.36.1, involves a transient, successful reactor trip, failure of the AFW system, the unavailability of the MFW system, and failure of feed and bleed.

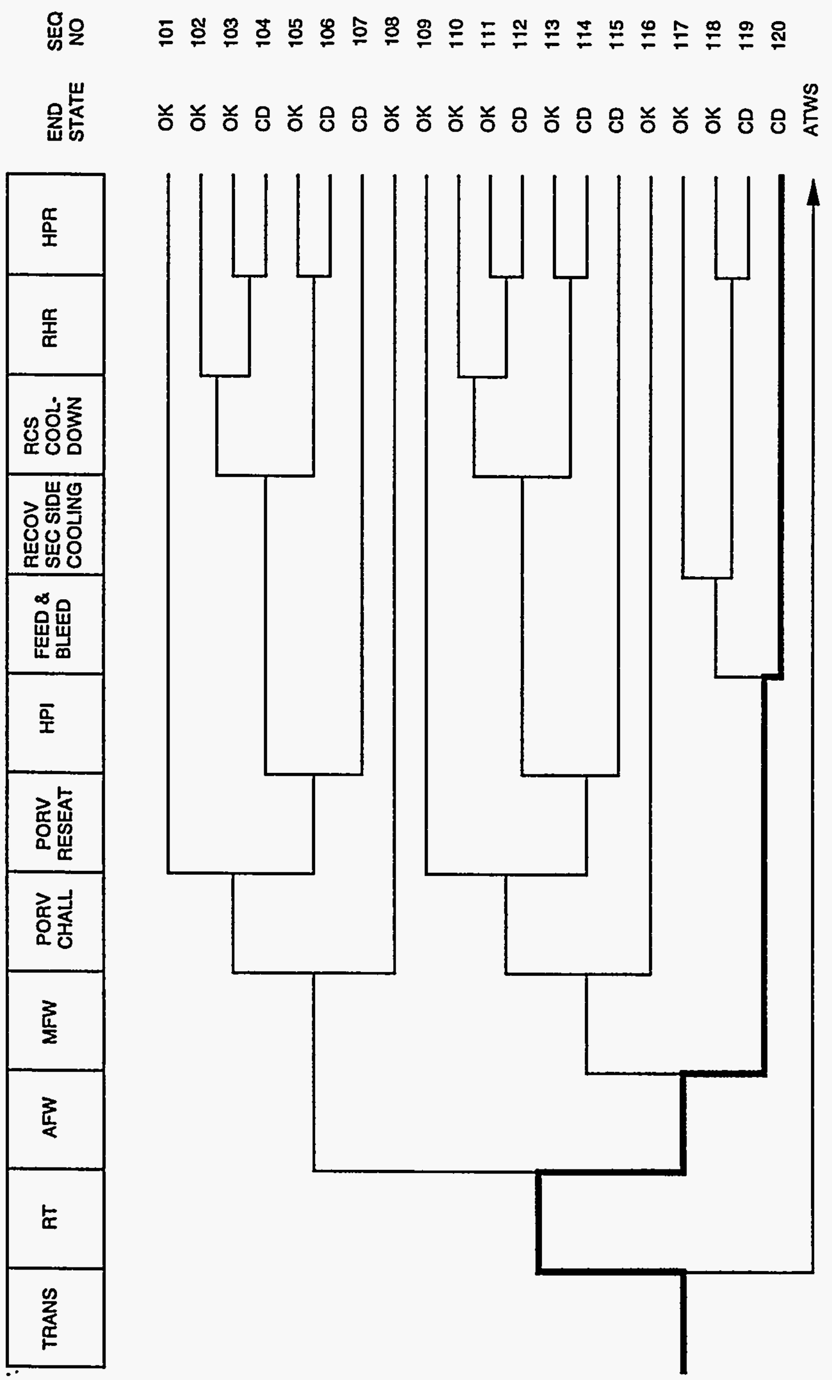

Figure B.36.1 Dominant core damage sequence for LER 344/83-002

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 344/83-002
Event Description: Reactor trip with MFW and two AFW pumps unavail.
Event Date: 1/22/83
Plant: Trojan

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	9.7E-05
Total	9.7E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW MFW feed.bleed	CD	9.3E-05	1.5E-01
119	trans -rt AFW MFW -feed.bleed recov.sec.cool hpr	CD	3.0E-06	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW MFW -feed.bleed recov.sec.cool hpr	CD	3.0E-06	1.5E-01
120	trans -rt AFW MFW feed.bleed	CD	9.3E-05	1.5E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\models\pwrb8283.cmp
BRANCH MODEL: c:\asp\models\trojan.82
PROBABILITY FILE: c:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.6E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	

AFW	3.3E-04 > 2.0E-02	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	5.0E-02 > Unavailable		
Train 3 Cond Prob:	2.0E-02		
Serial Component Prob:	2.8E-04		
AFW/ATWS	1.0E-01 > 2.0E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.0E-01 > 2.0E-02		
AFW/EP	2.8E-03 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.2+ser			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	5.0E-02 > Unavailable		
Serial Component Prob:	2.8E-04		
MFW	2.0E-01 > 1.0E+00	3.4E-01	1.0E-03
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	2.0E-01 > Unavailable		
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-05	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	1.0E+00	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.3E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.1E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	9.9E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.9E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	6.1E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.37 LER Number 344/83-012

Event Description: Auxiliary Feedwater Pump Tripped Following Reactor Trip

Date of Event: August 20, 1983

Plant: Trojan

B.37.1 Summary

After a reactor trip occurred on August 20, 1983, the diesel-driven auxiliary feedwater (AFW) pump auto started but tripped due to overspeed. Several attempts to restart the pump were unsuccessful. The other AFW pumps operated as required. The event was analyzed as an AFW pump failure in conjunction with the reactor trip. The conditional core damage probability (CCDP) estimated for this event is 3.0×10^{-5}.

B.37.2 Event Description

On August 20, 1983 a reactor trip occurred at 100% power due to a spurious main turbine high vibration signal. The diesel-driven auxiliary feedwater pump auto started but tripped due to overspeed. Several attempts to restart the pump in automatic mode failed. The steam-driven AFW pump and non-ESF motor-driven AFW pump supplied flow as required.

B.37.3 Additional Event-Related Information

The apparent cause of the diesel AFW pump overspeed was procedural deficiencies for restoration of the pump following annual maintenance combined with human error. Following the failure, the diesel engine controls were adjusted and the engine was tested successfully.

B.37.4 Modeling Assumptions

The event was modeled as an AFW pump failure in conjunction with a reactor trip. Train 1 of AFW, representing the diesel-driven pump, was failed. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event. Since success of AFW given an anticipated transient without scram (ATWS) requires that both of the auto-start AFW pumps operate, this branch (AFW/ATWS) was assumed to be failed.

B.37.5 Analysis Results

The conditional core damage probability estimated for this event is 3.0×10^{-5}. The dominant accident sequence, shown in Figure B.37.1, consists of the transient followed by a failure to trip the reactor, successful limiting of reactor coolant system pressure (<3200 psi), and failure of AFW for ATWS mitigation.

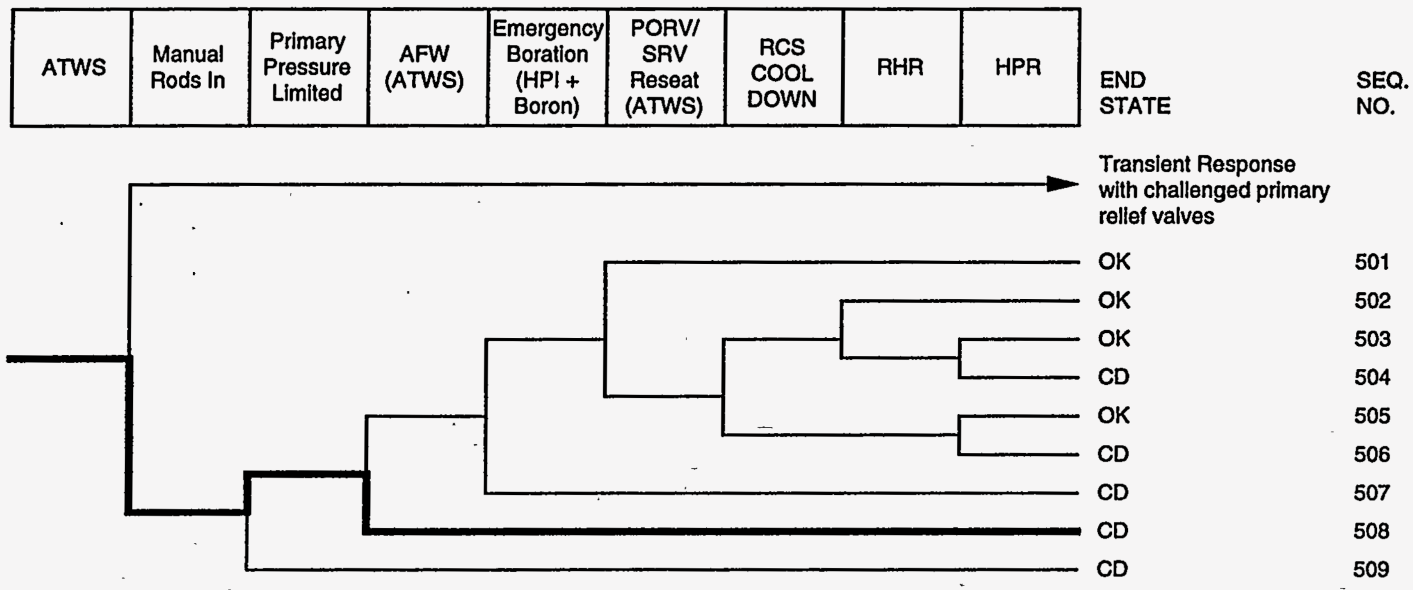

Figure B.37.1 Dominant core damage sequence for LER 344/83-012

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 344/83-012
Event Description: Diesel AFW pump trip after reactor trip
Event Date: 8/20/83
Plant: Trojan

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	3.0E-05
Total	3.0E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01
120	trans -rt AFW mfw feed.bleed	CD	1.2E-06	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW mfw feed.bleed	CD	1.2E-06	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\models\pwrb8283.cmp
BRANCH MODEL: c:\asp\models\trojan.82
PROBABILITY FILE: c:\asp\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.6E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.3E-04 > 1.3E-03	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	5.0E-02		
Train 3 Cond Prob:	2.0E-02		
Serial Component Prob:	2.8E-04		
AFW/ATWS	1.0E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.0E-01 > Failed		
AFW/EP	2.8E-03 > 5.0E-02	3.4E-01	
Branch Model: 1.OF.2+ser			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04		
mfw	2.0E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-05	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.3E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.1E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	9.9E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.9E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	6.1E-02	1.0E+00	

sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
* forced

B.38 LER Nos. 346/83-038 and -040

Event Description: Inoperability of One Auxiliary Feedwater Pump and Reactor Trip Due to Trip of Two Steam and Feedwater Rupture Control System Logic Channels

Date of Event: July 25, 1983

Plant: Davis-Besse 1

B.38.1 Summary

On July 25, 1983, the reactor was tripped by a spurious steam and feedwater rupture control system signal. After the reactor trip, auxiliary feedwater (AFW) pump 1-1 was discovered to be inoperable. The conditional core damage probability estimated for this event is 8.2×10^{-5}.

B.38.2 Event Description

On July 25, 1983, a trip alarm was received on steam and feedwater rupture control system (SFRCS) Logic Channel 3, causing a half trip of Actuation Channel 1. The failure was traced to a failed 48V dc/dc power supply that apparently caused the overvoltage trip device to actuate. While the power supply was being replaced, SFRCS Logic Channel 1 spuriously tripped. This caused a full trip of Actuation Channel 1, resulting in a reactor trip. The cause of this failure was believed to be a momentary failure of the 48-V power supply in Logic Channel 1. The root cause of both failures was overheating of the power supplies due to improper design of the SFRCS cabinets. Fans were installed in the cabinets to ensure adequate cooling.

Following the reactor trip, AFW pump 1-1 was declared inoperable due to its failure to respond to speed change signals from both automatic and manual control. The problem was found to be due to a slipping clutch between the speed changer motor and the governor; attempts to adjust the clutch were unsuccessful. During this event, AFW pump 1-2 and the startup motor driven feed pump were operable. The unit was shut down and a refueling outage was initiated.

B.38.3 Additional Event-Related Information

Failures similar to those documented in LER 346/83-038 were reported in LERs 346/82-051 and 346/83-019. Previous occurrences involving the inoperability of AFW pumps due to defective governors were reported in LERs 346/81-037 and 346/81-045.

B.38.4 Modeling Assumptions

These events are modeled as a reactor trip with one AFW pump inoperable. The redundant train of the AFW system was assumed to be vulnerable to a similar type of failure. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event.

The actuation of SFRCS was assumed to have isolated the main feedwater pumps. A nonrecovery probability of 0.1 was assumed for the main feedwater (MFW) pumps. In addition, the startup feed pump could have been locally aligned to provide water to the steam generators (SGs). A failure probability of 0.55 (see Appendix A) was employed here to recognize the local actions that would have been required for the alignment and the dependency between the recovery of MFW and the use of the startup feed pump.

B.38.5 Analysis Results

The conditional core damage probability estimated for this event is 8.2×10^{-5}. The dominant core damage sequence, shown in Figure B.38.1, involves the initiating transient, a successful reactor trip, failure of AFW, loss of MFW, and failure of feed and bleed.

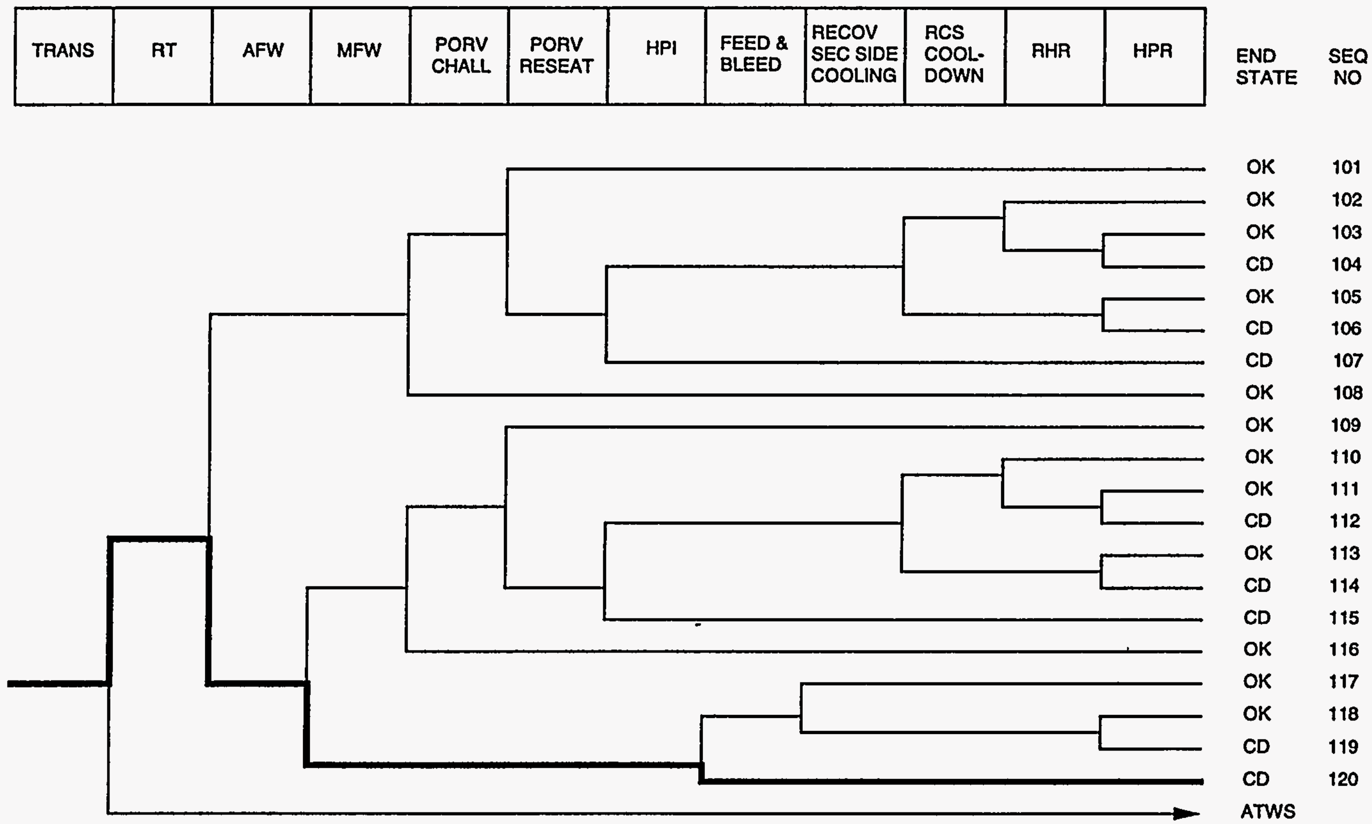

Figure B.38.1 Dominant core damage sequence for LER Nos. 346/83-038 and -040

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 346/83-038
Event Description: Reactor trip with one AFW pump inoperable
Event Date: July 25, 1983
Plant: Davis Besse

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	8.2E-05
Total	8.2E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
120	trans -rt AFW MFW feed.bleed	CD	5.2E-05	2.5E-02
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01
119	trans -rt AFW MFW -feed.bleed recov.sec.cool hpr	CD	2.4E-06	2.5E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW MFW -feed.bleed recov.sec.cool hpr	CD	2.4E-06	2.5E-02
120	trans -rt AFW MFW feed.bleed	CD	5.2E-05	2.5E-02
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\pwrb8283.cmp
BRANCH MODEL: c:\asp\1982-83\dbesse.82
PROBABILITY FILE: c:\asp\1982-83\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.2E-03	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	

AFW	5.0E-03 > 1.0E-01	4.5E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01		
AFW/ATWS	9.5E-02 > 1.0E+00	1.0E+00	
Branch Model: 2.OF.2			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01		
AFW/EP	5.0E-03 > 1.0E-01	4.5E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01		
MFW	2.0E-01 > 1.0E+00	3.4E-01 > 5.5E-02	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-01 > Unavailable		
porv.chall	8.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	1.0E-02	1.1E-02	
porv.reseat/ep	1.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	1.0E-03	8.9E-01	
feed.bleed	1.1E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	5.7E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	0.0E+00	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.7E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	1.6E-01	1.0E+00	
offsite.pwr.rec/seal.loca	0.0E+00	1.0E+00	
offsite.pwr.rec/-seal.loca	4.5E-01	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.39 LER No. 361/83-063

Event Description: Trip with Motor-Driven AFW Pump Inoperable

Date of Event: June 21, 1983

Plant: San Onofre 2

B.39.1 Summary

A few days after a trip on Unit 2, an endurance test was performed on one of the Unit 3 motor-driven auxiliary feedwater (AFW) pumps (3P-504). During the test, the outboard bearing on the motor failed, and the pump was declared inoperable. Upon examination, it was discovered that the oil grooves in the AFW pump bearings were inadequate. Presumably, the bearings were not being lubricated properly. Similar bearing defects were discovered in the other Unit 3 motor-driven AFW pump (3P-141), and one of the two motor-driven AFW pumps on Unit 2 (2P-504). A trip had occurred on Unit 2 a few days earlier, while the AFW pump was presumably inoperable. The conditional core damage probability estimated for the event at Unit 2 is 1.1×10^{-5}.

B.39.2 Event Description

While Unit 2 was shut down and Unit 3 was still precritical, a 48-hour endurance test was performed on the Unit 3 motor-driven AFW pump, 3P-504. During the run, the outboard bearing on the motor failed, rendering the pump inoperable. It was discovered later that the bearing oil grooves had been machined incorrectly and the bearing was apparently not being properly lubricated as a result. Similar problems were then identified with the other Unit 3 motor-driven AFW pump, 3P-141, and with the 2P-141 motor-driven AFW pump on Unit 2.

Low condenser vacuum caused a trip on Unit 2 on June 16, 1983.

B.39.3 Additional Event-Related Information

A supplemental report to the licensee event report for this event indicated that the bearing machining error would not by itself cause bearing failure. However the report did indicate that the machining error, in conjunction with other normally acceptable conditions, could cause bearing failure. The other conditions which might be required were not specified.

B.39.4 Modeling Assumptions

Since normally occurring conditions in conjunction with the bearing machining error could cause pump failure, it was assumed in this analysis that AFW pump 2P- 504 was inoperable in the long term. [The ASP models, like most PRA models, assume long-term (24 hours) operability for AFW success.] As the bearing condition apparently existed since the pump's manufacture, it was assumed that the AFW pump was inoperable at the time of the Unit 2 trip which occurred 5 days earlier. This event was modeled as an inoperability of train 1 of AFW in conjunction with the unit trip.

The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event.

The Unit 3 failures were not modeled, as Unit 3 was still in precritical status at the time.

B.39.5 Analysis Results

The conditional core damage probability estimated for this event is 1.1×10^{-5}. The dominant core damage sequence, shown in Figure B.39.1, involves reactor trip, failure of main and auxiliary feedwater, and failure to supply makeup from the condensate system.

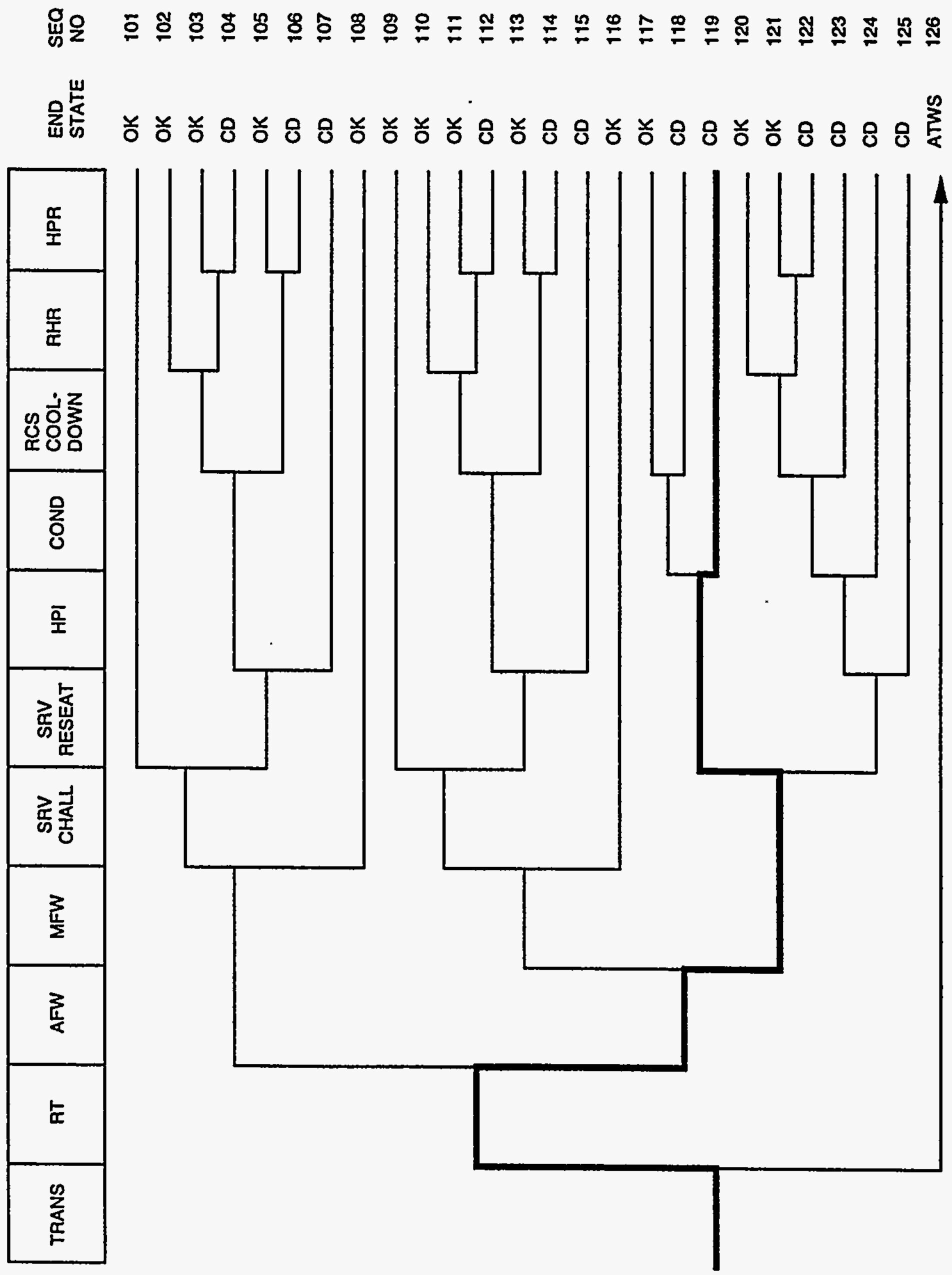

Figure B.39.1 Dominant core damage sequence for LER 361/83-063

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 361/83-063
Event Description: Trip with AFW pump inoperable
Event Date: June 21. 1983
Plant: San Onofre 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.1E-05
Total	1.1E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW mfw -srv.reseat cond	CD	5.5E-06	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	4.2E-06	1.0E-01
118	trans -rt AFW mfw -srv.reseat -cond rcs.cooldown	CD	3.1E-07	1.5E-01
509	trans rt prim.press.limited	CD	2.5E-07	1.0E-01
507	trans rt -prim.press.limited -AFW/ATWS emrg.boration	CD	2.4E-07	1.0E-01
107	trans -rt -AFW srv.chall srv.reseat hpi	CD	2.1E-07	8.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
107	trans -rt -AFW srv.chall srv.reseat hpi	CD	2.1E-07	8.9E-01
118	trans -rt AFW mfw -srv.reseat -cond rcs.cooldown	CD	3.1E-07	1.5E-01
119	trans -rt AFW mfw -srv.reseat cond	CD	5.5E-06	1.5E-01
507	trans rt -prim.press.limited -AFW/ATWS emrg.boration	CD	2.4E-07	1.0E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	4.2E-06	1.0E-01
509	trans rt prim.press.limited	CD	2.5E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\pwrh8283.cmp
BRANCH MODEL: c:\asp\1982-83\sanono2.82
PROBABILITY FILE: c:\asp\1982-83\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.7E-03	1.0E+00	
loop	2.0E-05	5.8E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	3.8E-04 > 5.3E-03	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04		
AFW/ATWS	4.3E-03 > 1.5E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > 1.5E-01		
afw/ep	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
cond	2.5E-02	1.0E+00	1.0E-02
srv.chall	4.0E-02	1.0E+00	
srv.chall/afw	1.0E+00	1.0E+00	
srv.chall/loop	1.0E-01	1.0E+00	
srv.chall/sbo	1.0E+00	1.0E+00	
srv.reseat	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
rhr	7.1E-03	5.7E-02	1.0E-03
hpr	2.0E-03	1.0E+00	
ep	2.9E-03	8.9E-01	
seal.loca	5.5E-02	1.0E+00	
offsite.pwr.rec/seal.loca	7.6E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	3.4E-01	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cooldown	1.0E-03	1.0E+00	1.0E-03
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	
emrg.boration	0.0E+00	1.0E+00	1.0E-02

* branch model file
** forced

B.40 LER No. 362/83-099

Event Description: Trip with Turbine-Driven AFW Pump Inoperable

Date of Event: October 31, 1983

Plant: San Onofre 3

B.40.1 Summary

San Onofre 3 was operating at 62% power when a loss of main feedwater caused a reactor trip. An emergency feedwater actuation signal (EFAS) was generated, but the turbine-driven emergency feedwater (EFW) pump failed to start. The conditional core damage probability estimated for the event is 1.5×10^{-5}.

B.40.2 Event Description

During Mode 1 operation at 62% power, San Onofre Unit 3 experienced a loss of feedwater and tripped after the feed pumps experienced problems with their suction supply. An EFAS signal was generated when the unit tripped, but the turbine-driven EFW pump failed to start. The pump was found to be tripped, was reset, and was successfully restarted. The reason for the EFW pump trip was unknown at the time of the licensee event report (LER), but investigation was ongoing.

B.40.3 Additional Event-Related Information

None.

B.40.4 Modeling Assumptions

This event was modeled as a loss of feedwater with the turbine-driven EFW pump inoperable.

B.40.5 Analysis Results

The conditional core damage probability estimated for this event is 1.5×10^{-5}. The dominant sequence for this event, highlighted on the event tree in Figure B.40.1, involves a failure of main and emergency feedwater, and failure to supply makeup from the condensate system.

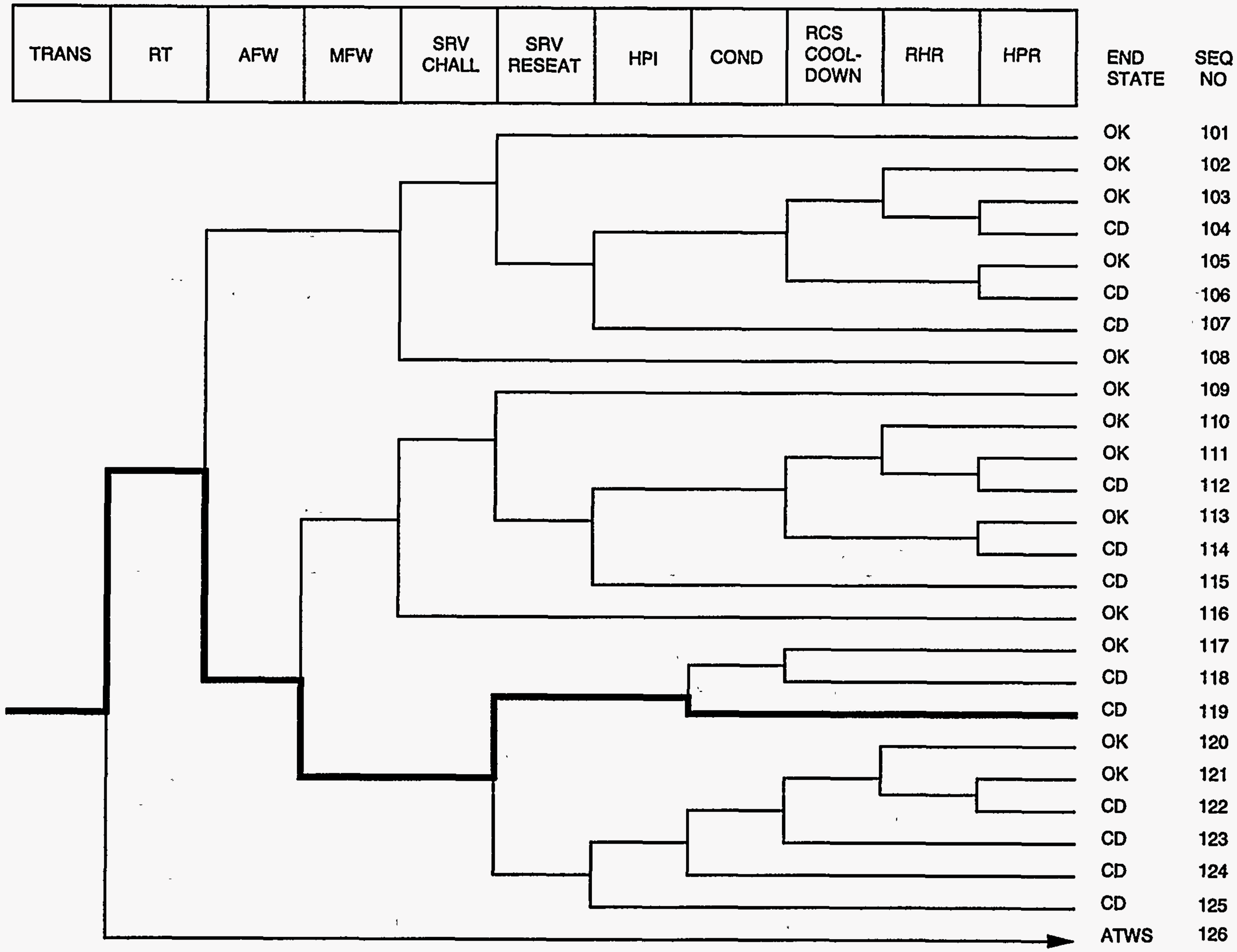

Figure B.40.1 Dominant core damage sequence for LER 362/83-099

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 362/83-099
Event Description: Trip with turbine-driven AFW pump inoperable
Event Date: October 31, 1983
Plant: San Onofre 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.5E-05
Total	1.5E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rt AFW MFW -srv.reseat cond	CD	1.2E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	1.1E-06	1.0E-01
118	trans -rt AFW MFW -srv.reseat -cond rcs.cooldown	CD	6.6E-07	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
118	trans -rt AFW MFW -srv.reseat -cond rcs.cooldown	CD	6.6E-07	1.5E-01
119	trans -rt AFW MFW -srv.reseat cond	CD	1.2E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	1.1E-06	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\pwrh8283.cmp
BRANCH MODEL: c:\asp\1982-83\sanono2.82
PROBABILITY FILE: c:\asp\1982-83\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.7E-03	1.0E+00	
loop	2.0E-05	5.8E-01	
loca	2.4E-06	5.4E-01	

sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	

Event Identifier: 362/83-099

AFW	3.8E-04 > 2.3E-03	4.5E-01	
Branch Model: 1.OF.3+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
AFW/ATWS	4.3E-03 > 4.0E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.3E-03 > 4.0E-02		
afw/ep	5.0E-02	3.4E-01	
MFW	2.0E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-01 > Failed		
cond	2.5E-02	1.0E+00	1.0E-02
srv.chall	4.0E-02	1.0E+00	
srv.chall/afw	1.0E+00	1.0E+00	
srv.chall/loop	1.0E-01	1.0E+00	
srv.chall/sbo	1.0E+00	1.0E+00	
srv.reseat	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
rhr	7.1E-03	5.7E-02	1.0E-03
hpr	2.0E-03	1.0E+00	
ep	2.9E-03	8.9E-01	
seal.loca	5.5E-02	1.0E+00	
offsite.pwr.rec/seal.loca	7.6E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	3.4E-01	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cooldown	1.0E-03	1.0E+00	1.0E-03
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	
emrg.boration	0.0E+00	1.0E+00	1.0E-02

* branch model file
** forced

B.41 LER No. 366/82-081

Event Description: Scram, Isolation, RCIC Failure, SRV Tailpipe Vacuum Relief Failed

Date of Event: August 25, 1982

Plant: Hatch 2

B.41.1 Summary

Hatch 2 was operating at power when a main steam isolation valve (MSIV) failed closed, causing a scram and isolation. Multiple problems and equipment failures, including failure of the reactor core isolation cooling (RCIC), leakage of reactor coolant into the drywell, and leakage of reactor coolant into the reactor building, complicated the scram recovery. The conditional core damage probability estimated for the event is 1.4×10^{-5}.

B.41.2 Event Description

During power operation at Hatch Unit 2, a valve disk in an MSIV in steamline C separated from its valve stem and closed suddenly. The sudden isolation of one of four main steam lines caused a reactor pressure increase and void collapse. The increased moderator density resulting from the void collapse caused a sudden power increase which in turn caused a scram. Increased flow in the three open main steamlines also caused a high-steam flow group 1 isolation, closing the rest of the MSIVs.

With the MSIVs closed, reactor pressure began increasing and two safety/relief valves (SRVs) opened. The high-pressure coolant injection (HPCI) system and RCIC system started, then tripped on reactor high level. Operators opened the MSIV bypasses in preparation for resetting the isolation and restarted RCIC to provide makeup. The RCIC system flow rate was inadequate to maintain reactor inventory and HPCI was restarted.

B.41.3 Additional Event Information

During a scram, reactor coolant is used to hydraulically drive pistons to which control rods are attached. Water from nitrogen-charged accumulators is routed beneath the pistons and water from the over-piston spaces is drained to a tank, the scram discharge volume (SDV). The SDV is normally kept vented and drained. A scram signals the vents and drains to close, to protect reactor coolant system integrity. Water from the over-piston spaces and control rod drive seal leakoff collects in the SDV until the scram is reset, at which time the SDV inventory is aligned to drain to the reactor building equipment drain sump (RBEDS).

The SDV drain valve failed to fully close during this event. This opened a direct path between the reactor coolant system and the RBEDS, outside of containment. Flow was limited by the control rod drive seals and the 2-inch SDV drain line, but reactor coolant drained to the sump continuously. Enough inventory transferred to the sump that steam began flowing from an uncapped drain connection in the RCIC space. The steam release in the RCIC space caused actuation of the fire protection system and the combined effects of the steam and fire protection spray caused RCIC instrumentation to fail, and the system shut down.

At the same time, an SRV tailpipe vacuum relief valve failed. SRVs relieve the suppression pool via "tailpipes." After SRV operation, steam in the tailpipes condenses, creating a partial vacuum. The vacuum can then draw water into the tailpipe from the suppression pool. Subsequent operation of the associated SRV will accelerate the water in the tailpipe, which can result in damage to the tailpipe or the attached quencher. To prevent this, each tailpipe is equipped with vacuum relief valves. During this event, one of the tailpipe relief valves failed open, allowing steam to vent directly to the drywell and causing a drywell high-pressure accident/scram signal. This scram signal could not be reset until drywell pressure was reduced. At the same time, it was difficult to reduce drywell pressure because the drywell chillers were isolated by the accident signal.

Operators aligned a reactor feedpump to provide makeup and the steam was dumped to the main condenser. The high drywell pressure accident/scram signal was jumpered out and a drywell chiller was restarted. Drywell pressure was reduced and the scram was reset, finally isolating the flow path from the reactor to the RBEDS.

B.41.4 Modeling Assumptions

This event was modeled as a transient and isolation with RCIC inoperable. The long-term nonrecovery probability for the power conversion system (PCS) was revised to 0.017 to reflect the initial MSIV isolation. Accordingly, for sequences involving potential residual heat removal (RHR) or PCS nonrecovery, the nonrecovery estimate was revised to 0.016 x 0.11, or 2.7E-4 (see Appendix A).

B.41.5 Analysis Results

The conditional core damage probability estimated for this event is 1.4×10^{-5}. The dominant core damage sequence, shown in Figure B.41.1, involves the observed transient, failure of PCS, recovery of main feedwater, and failure of RHR.

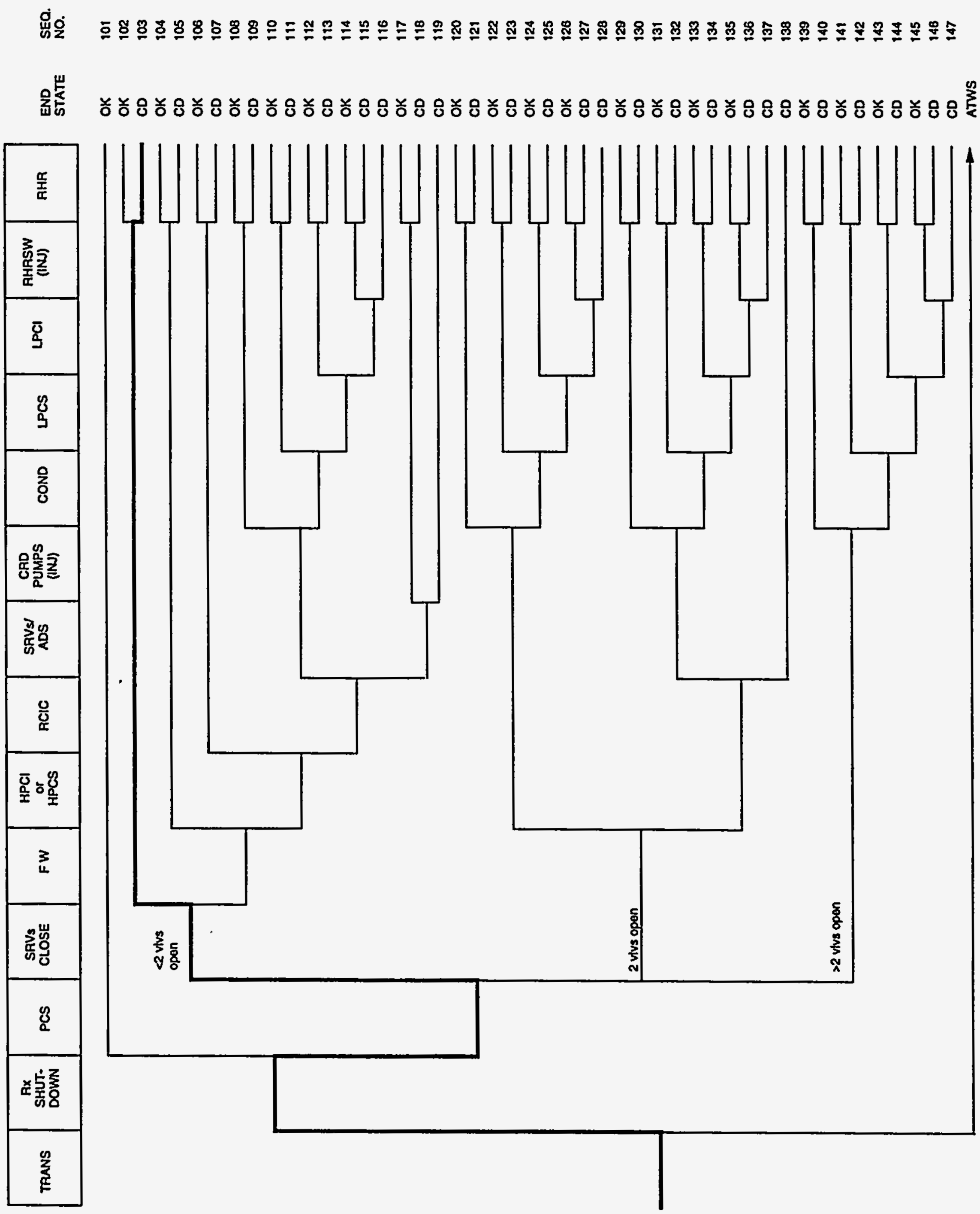

Figure B.41.1 Dominant core damage sequence for LER 366/82-081

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 366/82-081
Event Description: Scram. isolation. RCIC and SRV vacuum relief valve failure
Event Date: August 25. 1982
Plant: Hatch 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.4E-05
Total	1.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\hatch2.82

PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.42 LER No. 366/82-084, -085

Event Description: RHRSW Pumps A, C, and D Failed

Date of Event: August 13, 1982

Plant: Hatch 2

B.42.1 Summary

On August 13, 1982, residual heat removal service water (RHRSW) pumps A and C were declared inoperable when they failed to meet their minimum flow requirements. On the same day, the D RHRSW pump was declared inoperable when it failed to meet its minimum flow requirements. The increase in core damage probability over the duration of the event, or importance, is 2.4×10^{-4}. The base-case core damage probability (CDP) over the duration of the event is 2.7×10^{-6}, resulting in an estimated conditional core damage probability (CCDP) of 2.4×10^{-4}.

B.42.2 Event Description

On August 13, 1982, RHRSW pumps A and C were declared inoperable when they failed to meet their minimum flow requirements during testing procedures. The cavitrol trim (an anticavitation device) on the downstream side of the flow control valve in the outlet of the A loop RHRSW heat exchanger was found to have both broken and bent tubes, which caused the restricted flow.

On August 13, 1982, the D RHRSW pump failed to meet its minimum discharge pressure and flow requirement during testing. The cause of the failure was the failure of the B pump discharge check valve to close. The check valve was freed and the D pump was returned to service within 8 hours.

B.42.3 Additional Event-Related Information

The RHRSW system provides cooling water from the ultimate heat sink (the Altamaha River) to remove decay heat via the residual heat removal (RHR) heat exchangers. By means of a crosstie with the RHR system, the RHRSW system also can supply makeup to the reactor coolant system (RCS) when all emergency core cooling systems have failed.

B.42.4 Modeling Assumptions

RHRSW (and thus RHR) were assumed to be degraded at the time the event was reported on August 13, 1982. RHRSW train I (pumps A and C) was assumed to be inoperable, due to the failure of the cavitrol trim device, which was discovered by flow testing. In addition, testing on August 13, 1982 identified that the B pump discharge check valve was stuck in the open position. The train II discharge check valve was stuck in the open position. Train II (pumps B and D) was assumed to be operable, but with an increased failure probability. With the B pump discharge check valve stuck open, train II would be operable only if pump B was operable.

Failure of pump B was assumed to fail the entire train by allowing diversion of flow from pump D. It was not known how long the failures in the RHRSW system had existed, so it was assumed in the analysis that they had existed for half of a one-month surveillance interval, or 360 hours.

The analysis assumed that common cause failures would dominate the system failure probability, given the observed failure in RHRSW loop A. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event. Although the specific failure discovered during testing (in this case, the cavitrol trim device) was apparently not present in loop B at the time of the loop A failure, other common cause failure modes remained and could have affected system performance. Interestingly, the licensee noted that a similar problem had occurred with the loop B cavitrol trim device during the previous year.

The probability of RHR failing is given by the probability of RHRSW train II failing, which was estimated as follows:

$P_{\text{rhrsw loop 2}}$ = 0.1 (Probability that loop II flow control valve fails, given that loop I flow control valve has failed.)

+ .01 (Probability that RHRSW pump B fails to start and run.)

= 0.11

The nonrecovery probability for RHR was revised to 0.054 to reflect the RHRSW failures (see Appendix A). For sequences involving potential RHR or power conversion system (PCS) recovery, the nonrecovery estimate was revised to 0.054 x 0.052 (PCS nonrecovery), or 0.028.

B.42.5 Analysis Results

The estimated increase in core damage probability over the duration of the event is 2.4×10^{-4}. The base case CDP (not shown in calculation) is 2.1×10^{-6}, resulting in an estimated CCDP of 2.4×10^{-4}. The dominant sequence, highlighted in the event tree in Figure B.42.1, involves a postulated trip during the unavailability period, failure of the power conversion system, success of feedwater, and failure of RHR.

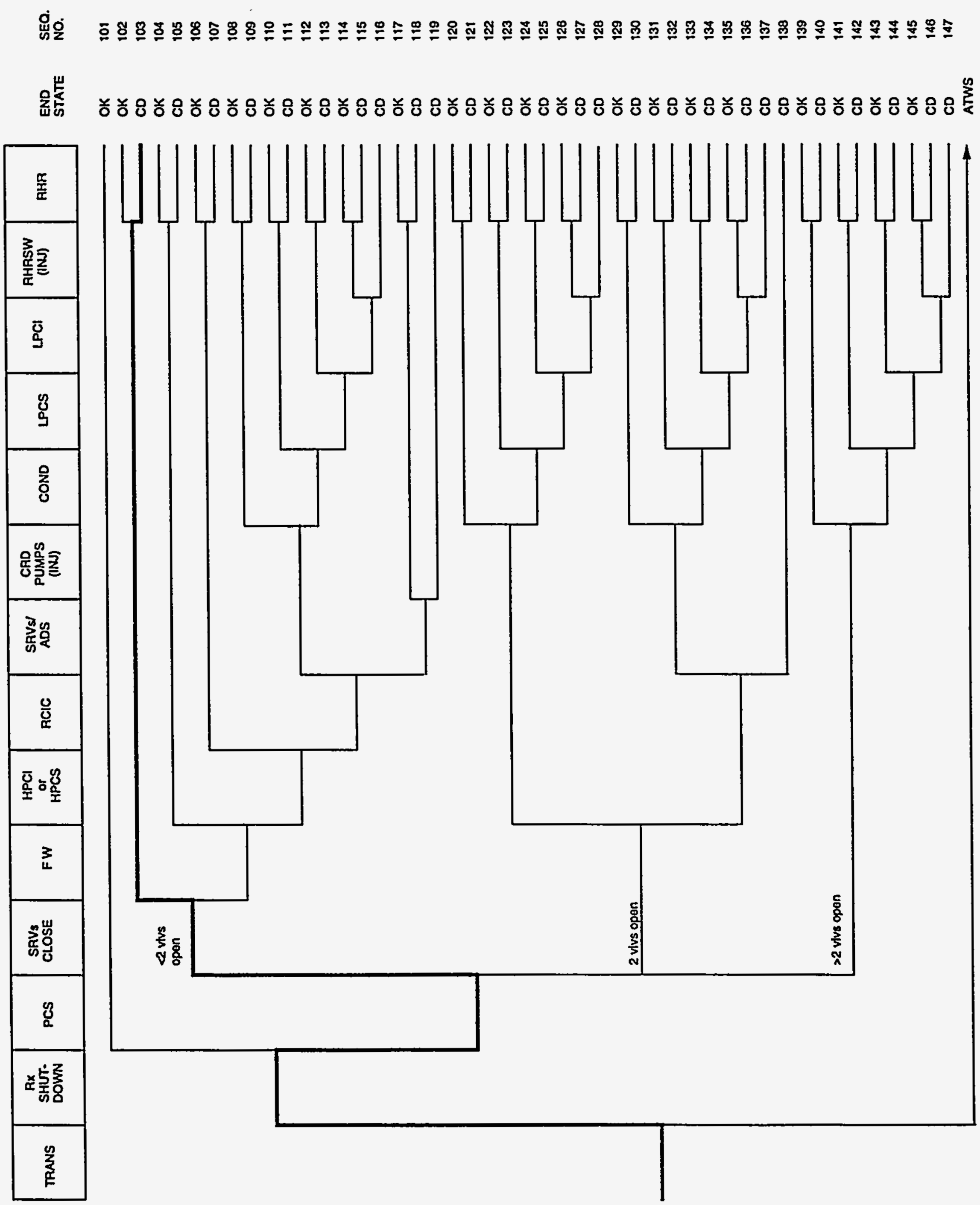

Figure B.42.1 Dominant core damage sequence for LER 366/82-084

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 366/82-084
Event Description: RHRSW pumps A. C. D failed
Event Date: August 13. 1982
Plant: Hatch 2

UNAVAILABILITY. DURATION= 360

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	5.5E-01
LOOP	2.1E-03
LOCA	8.0E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	2.2E-04
LOOP	1.3E-05
LOCA	8.0E-07
Total	2.4E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.9E-04	2.5E-02
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	3.4E-05	9.5E-03
202	loop -rx.shutdown -ep srv.ftc.<2 -hpci RHR	CD	1.2E-05	1.9E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.9E-04	2.5E-02
105	trans -rx.shutdown pcs srv.ftc.<2 3.4E-05	9.5E-03		
202	loop -rx.shutdown -ep srv.ftc.<2 -hpci RHR	CD	1.2E-05	1.9E-02

** non-recovery credit for edited case

Note: For unavailabilities. conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\hatch2.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	4.6E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.1E-01 **	1.6E-02 > 5.4E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.1E-01 **	8.3E-03 > 2.8E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.1E-01	1.0E+00 > 5.4E-02	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00 > 1.1E-01		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.1E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.1E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file

** forced

B.43 LER No. 366/83-042, -055, and -056

Event Description: Reactor Trip with RCIC and RHR Loop A Unavailable

Date of Event: July 14, 1983

Plant: Hatch 2

B.43.1 Summary

On July 14, 1983, during startup following a refueling outage, Hatch 2 started losing condenser vacuum and the turbine was tripped. The reactor was manually scrammed when one control rod was found in an "out of sequence" position. The reactor core isolation cooling system (RCIC) was unavailable at the time. While placing the A loop of residual heat removal (RHR) in the shutdown cooling mode to achieve a cold shutdown condition, the A loop heat exchanger outlet valve failed to open. The conditional core damage probability estimated for the event is 1.5×10^{-4}.

B.43.2 Event Description

On July 13, 1983, with the plant at approximately 14% power, and again on July 21, 1983, with the plant at approximately 60% power, the RCIC pump failed to deliver the minimum required flow of 400 gpm, and was declared unavailable. On July 14, 1983, the plant was starting up from a refueling outage and was at approximately 7% power, when the unit started losing condenser vacuum. The turbine was tripped, and control room personnel scrammed individual rods with the scram switches at the scram timing panel in an effort to quickly reduce power. The rod worth minimizer (RWM) was bypassed, and at one point the "emergency rod in" control was used to achieve the greatest possible insertion rate. After several rods had been inserted, one rod was found in an out-of-sequence position, and the reactor was manually scrammed.

RCIC failed to meet its minimum required flow because its electric governor remote (EGR) actuator was out of adjustment, which caused the governor valve to fail to open completely. The EGR actuator was adjusted and RCIC was returned to operability on July 21, 1983.

Following the turbine trip on July 14, 1983, control rods were being rapidly inserted to reduce power. Control room personnel attempted to lower reactor power quickly so that the mechanical vacuum pump could be placed in service before the decreasing vacuum reached the reactor feed pump low vacuum trip point. A reactor feed pump low vacuum trip results in a loss of feedwater flow to the vessel. Since RCIC was unavailable, operators were trying to avoid losing feedwater flow.

On July 15, 1983, while placing the A loop of RHR in the shutdown cooling mode to achieve a cold shutdown condition, the A loop heat exchanger outlet valve failed to open because the valve motor was faulted.

The failed RHR loop A heat exchanger outlet valve motor was replaced, and RHR loop A was returned to service on July 15, 1983.

B.43.3 Additional Event-Related Information

On July 14, 1983, during normal startup activities from a refueling outage, the plant was operating at about 25% power. Problems with the main condenser vacuum had occurred and air ejector troubleshooting had been in progress. Condenser vacuum began to decrease and the turbine was unloaded and tripped. Control rods were inserted in an attempt to reduce reactor power to within the limit of the mechanical vacuum pump so that it could be placed in service in order to maintain vacuum above the trip set point of the reactor feed pumps. A reactor feed pump low vacuum trip would cause a loss of feedwater flow to the vessel.

To reduce power more quickly, the licensee bypassed the RWM and assigned a second licensed operator to verify control rod movement as permitted by the Technical Specifications. At one point, the "emergency rod in" control was used to achieve the greatest possible insertion rate.

When the operator reached groups of low worth peripheral rods in the sequence, a discussion among the licensed operators and supervisors in the control room resulted in a decision to scram individual rods by using the individual scram switches at the scram timing panel which was already set up for scram time testing. This was not an approved procedure and resulted in the insertion of rods in an out-of-sequence manner. Vacuum at the time was about 0.5 inches above the trip point.

While the plant operator continued inserting rods at the front panel, two other operators began to insert rods at the scram timing panel with the individual scram switches. When the front panel operator observed those rods going in, he stopped inserting and verified further insertions from the scram panel. A process computer printout indicated that several rods were not fully inserted (i.e., scram toggle switches were not held down sufficiently long). These rods were subsequently rescrammed. One rod was also found in a position which was not expected based upon the rod manipulations performed by the operators. The reason for this was not determined. At this point, the vacuum pump was placed in service and vacuum stabilized at a low level. Because the one rod was improperly positioned, the reactor was scrammed as required by procedure.

During this event, the rod sequence control system (RSCS) was effectively bypassed. The RSCS is a backup system to the RWM and independently imposes restrictions on control rod movements to mitigate the effects of a control rod drop accident. The plant's Technical Specifications require the RSCS to be operable when reactor power is below 20%. However, the use of the "emergency rod in" control and the scram switches on the scram timing panel circumvented the RSCS.

B.43.4 Modeling Assumptions

This event was modeled as a reactor scram with RCIC and one train (2 pumps) of RHR unavailable. The ASP model assumes the dominant failure mode is a common cause failure of the RHR pumps. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event. To address the failure of the heat exchanger outlet valve, the model was modified. Failure of the heat exchanger outlet valve disables all functions of a complete train of RHR. For this analysis, one train of RHR (all modes) was assumed failed. The probability of the second train failing was assumed to be 0.1. Pretrip actions related to rod insertion are not addressed in this analysis.

B.43.5 Analysis Results

The conditional core damage probability estimated for this event is 1.5×10^{-4}. The dominant core damage sequences, highlighted on the event tree in Figure B.43.1, involves the observed transient, failure of the power conversion system, main feedwater system success, and failure of the residual heat removal system.

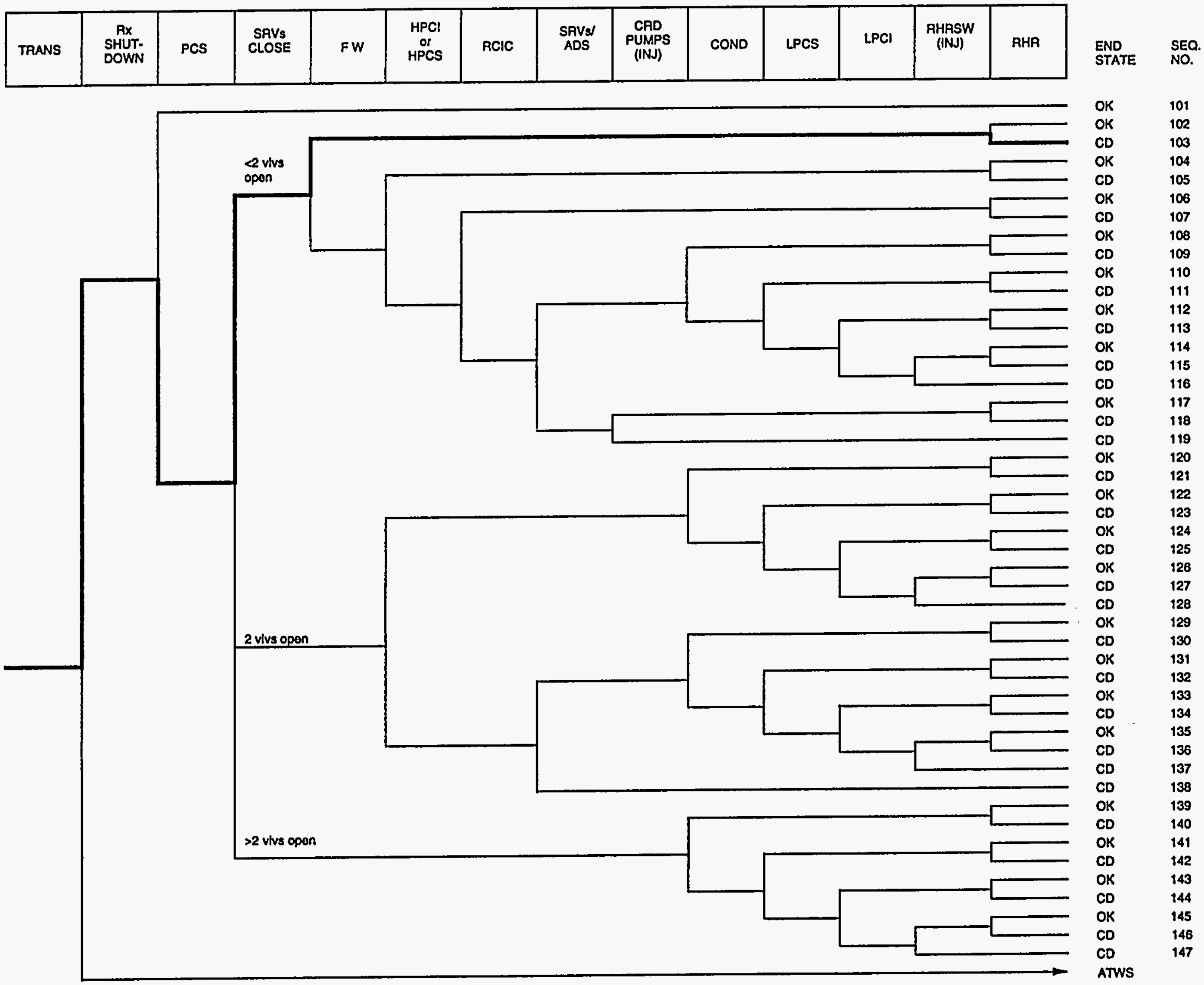

Figure B.43.1 Dominant core damage sequence for LER 366/83-042, -055, and -056

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 366/83-042
Event Description: Scram with RCIC and RHR loop A unavailable
Event Date: July 14. 1983
Plant: Hatch 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.5E-04
Total	1.5E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.2E-04	7.3E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	2.2E-05	2.8E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.2E-04	7.3E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	2.2E-05	2.8E-03

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\hatch2.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	

rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	4.6E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > Failed		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.0E-01 **	1.6E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.0E-01 **	8.3E-03	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.0E-01 **	1.0E+00	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.0E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.0E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.44 LER No. 366/83-084

Event Description: Reactor Scram with RHR Loop A Unavailable

Date of Event: August 17, 1983

Plant: Hatch 2

B.44.1 Summary

On August 16, 1983, Hatch 2 experienced a reactor scram. While proceeding to cold shutdown, the A loop heat exchanger outlet valve failed to open. The conditional core damage probability estimated for the event is 1.4×10^{-4}.

B.44.2 Event Description

On August 16, 1983, Hatch 2 experienced a reactor scram on low water level due to a reactor feed pump turbine control signal spike. On August 17, 1983, as the unit was going from hot shutdown to cold shutdown, the residual heat removal system (RHR) A loop heat exchanger outlet valve (2E11-F003A) failed to open because of a burned-out motor.

When plant personnel attempted to open 2E11-F003A, its position indication was lost and personnel received a "valve overload" alarm. An investigation of the valve revealed that its motor suffered an electrical fault when personnel tried to open the valve.

B.44.3 Additional Event-Related Information

The cause of the valve motor fault is unknown; however, a similar incident occurred one month earlier (see the analysis of LERs 366/83-042, -055, and -056). The valve motor was replaced and the torque switch setting was calibrated. The valve was returned to service on August 10, 1983.

B.44.4 Modeling Assumptions

This event was modeled as a reactor scram with one train of RHR unavailable. The ASP model assumes the dominant failure mode is a common cause failure of the RHR pumps. The potential for common cause failure exists even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that failed as part of the event. To address the failure of the heat exchanger outlet valve, the model was modified. Failure of the heat exchanger outlet valve disables all functions of a complete train of RHR. For this analysis, one train of RHR (all modes) was assumed failed. The probability of the second train failing, given failure of the first train, was assumed to be 0.1.

B.44.5 Analysis Results

The conditional core damage probability estimated for this event is 1.4×10^{-4}. The dominant core damage sequence, highlighted on the event tree in Figure B.44.1, involves the observed transient, failure of the power conversion system, main feedwater system success, and failure of the RHR system.

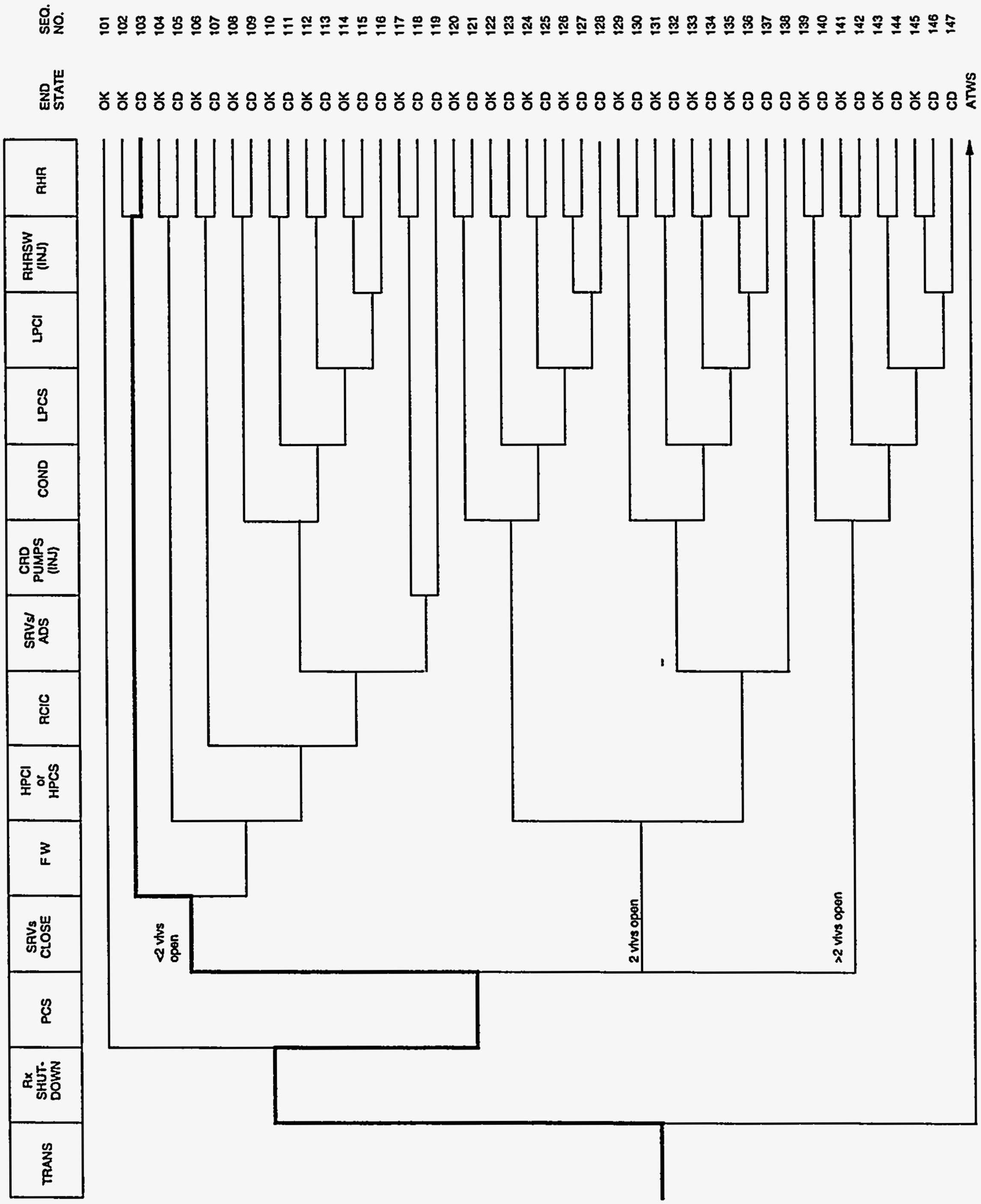

Figure B.44.1 Dominant core damage sequence for LER 366/83-084

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 366/83-084
Event Description: Scram with RHR loop A unavailable
Event Date: August 17. 1983
Plant: Hatch 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.4E-04
Total	1.4E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.2E-04	7.3E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	2.2E-05	2.8E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.2E-04	7.3E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	2.2E-05	2.8E-03

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\hatch2.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	

pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	4.6E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.0E-01 **	1.6E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.0E-01 **	8.3E-03	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.0E-01 **	1.0E+00	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.0E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.0E-01 **	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	
ep.rec	1.6E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.45 LER No. 368/83-007, -011, and -012

Event Description: Transient with One Train of EFW Inoperable

Date of Event: February 14, 1983

Plant: ANO 2

B.45.1 Summary

On February 22, 1983, emergency feedwater (EFW) turbine-driven pump 2P-7A would not attain rated speed or discharge pressure during surveillance testing. A plant trip had occurred on February 14 (NUREG-0020). Between February 3 and February 18, several EFW system valve failures also occurred. The estimated conditional core damage probability for this event is 4.1×10^{-5}.

B.45.2 Event Description

On February 3, 1983, during power operations, the hydraulic pump for the operator for the EFW control valve 2CV-1075-1 was determined to be in need of replacement during the performance of preventive maintenance. On February 4 and 17, 2CV-1075-1 failed to close when attempts were made to decrease flow to steam generator B. The failure of 2CV-1075-1 to close on the 17th of February was attributed to improperly set manifold block relief valves. The relief valve settings were lower than 1000 psi, which caused the relief valves to be challenged by the normal hydraulic operating pressure.

On February 18 ,1983, while in Mode 3 with pump 2P-7B feeding the steam generators, control valve 2CV-1036 failed to close. Control valve 2CV-1075-1 was inoperable for repairs. The ability to feed the steam generators with pump 2P-7B through a cross-connect through valves 2EFW-11B and 2EFW-8A was maintained. Investigation revealed that the failure of the valve to close was due to a shorted relay coil which caused the control power fuse in control panel 2C-17 for 2CV-1036 to open. Both the relay and fuse were replaced.

On February 22, 1983 while in Mode 1, the steam-driven EFW pump 2P-7A would not attain rated speed or discharge pressure during a surveillance test. The cause was attributed to setpoint drift in the speed controller. The converter gain on the speed controller was adjusted so the pump would run at the desired speed.

B.45.3 Additional Event-Related Information

The EFW system at Arkansas Nuclear One (ANO) 1 is a two-train system. One motor-driven (2P-7B) and one turbine-driven pump (2P-7A) provide coolant flow to two steam generators (A and B). Successful accident mitigation under most circumstances requires the use of one pump supplying water to one steam generator. Control valves 2CV-1075-1 and 2CV-1036 allow pump 2P-7B to supply coolant to steam generator B. There are two separate flow paths for pump 2P-7B to supply coolant to steam generator B (through 2CV-1075-1 and 2CV-1036 or through 2EFW-11B and 2EFW-8A).

B.45.4 Modeling Assumptions

A plant trip occurred on February 14, 1983. Assuming that both valves and EFW pump 2P-7A would have been inoperable at the time of the trip, this event was modeled as a transient with one train (the turbine-driven train) of EFW failed. Since pump 2P-7B could still supply both steam generators with both control valves failed, it was assumed that train B would still function as needed. To reflect the inoperability of the turbine-driven pump, one train of EFW was set to failed. EFW/anticipated transient without scram (ATWS) was also set to failed to reflect the fact that only one pump remained operable, and flow from two pumps is required for mitigation given ATWS.

B.45.5 Analysis Results

The estimated conditional core damage probability for this event is 4.1×10^{-5}. The dominant sequence highlighted on the event tree in Figure B.45.1 is an ATWS sequence involving the failure to trip, and the failure of EFW given ATWS.

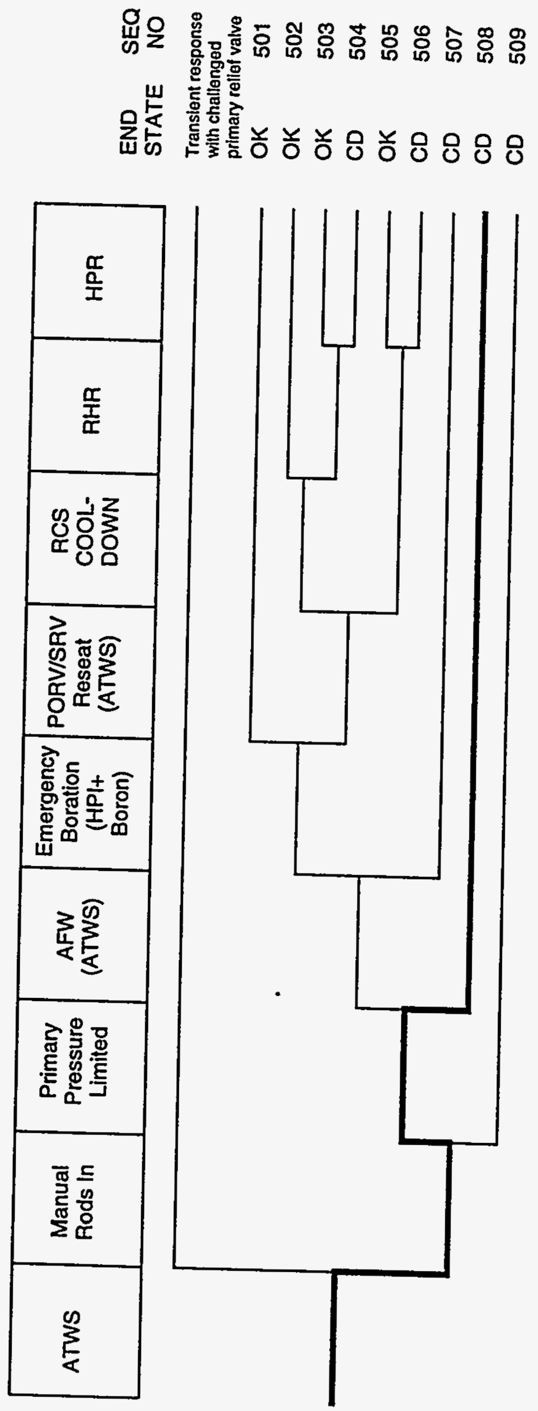

Figure B.45.1 Dominant core damage sequence for LER 368/83-007, -011, and -012

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 368/83-007. -011. and -012
Event Description: Transient with one train of EFW inoperable
Event Date: February 14. 1983
Plant: ANO 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	4.1E-05
Total	4.1E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01
121	trans -rt AFW mfw feed.bleed	CD	1.3E-05	1.5E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
121	trans -rt AFW mfw feed.bleed	CD	1.3E-05	1.5E-01
508	trans rt -prim.press.limited AFW/ATWS	CD	2.8E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrg8283.cmp
BRANCH MODEL: c:\aspcode\models\ano2.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	2.4E-03	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	5.4E-01	

sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
AFW	1.3E-03 > 2.0E-02	4.5E-01	
Branch Model: 1.OF.2+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
AFW/ATWS	7.0E-02 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	7.0E-02 > Failed		
AFW/EP	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	1.0E-02	1.1E-02	
porv.reseat/ep	1.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	1.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	3.1E-02	7.0E-02	1.0E-03
csr	4.0E-03	1.0E+00	1.0E-03
hpr	1.5E-04	1.0E+00	
ep	2.9E-03	8.9E-01	
seal.loca	4.0E-02	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.1E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	9.9E-02	1.0E+00	
offsite.pwr.rec/seal.loca	5.9E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	2.1E-02	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
** forced

B.46 LER No. 373/82-093, 373/82-094

Event Description: Scram and Multiple Failures

Date of Event: August 21, 1982

Plant: LaSalle 1

B.46.1 Summary

A controlled shutdown of LaSalle was initiated from an unspecified power level, due to an inadequate condensate inventory. During the course of the shutdown, the condensate inventory shortage became acute, forcing operators to scram the unit. In addition to the condensate and feedwater systems, the control rod drive (CRD) hydraulic system and the reactor core isolation cooling (RCIC) system were compromised or unavailable after the scram. In addition, a gross failure of the 1A recirculation pump seal was experienced. The conditional core damage probability estimated for this event is 1.1×10^{-4}.

B.46.2 Event Description

At approximately 0300, a controlled shutdown was initiated because insufficient water inventory was available for normal plant operation. Later, numerous condensate system alarms were received. Because of concern about condensate pump cavitation and about the adequacy of the CRD condensate supply, the unit was manually scrammed at 0536.

At an unspecified time on the same date, the RCIC system was inspected and it was discovered that the RCIC turbine was leaking oil from its sight glass and that the oil level could not be maintained in the turbine. Accordingly, RCIC was declared inoperable.

Initially after the scram, reactor makeup was supplied by the CRD system, but high CRD suction and discharge filter differential pressures developed and the CRD pump was tripped at 0745. Loss of CRD purge flow to the recirc pump seals meant that the seals were cooled only by the reactor building closed cooling water (RBCCW) system. Seal temperature on recirc pump 1B rose to 150°F and stabilized; however seal temperature on recirc pump 1A continued to rise. By 0828, 1A recirc pump seal temperature reached 175°F and the pump was tripped. Subsequently, the seal temperature rose to 235°F. At that time, around 0910, a drywell entry was made and the RBCCW flow to the seal was found to be low. The operations foreman increased flow from below 13 gpm to about 25 gpm over a period of about 1 minute and seal temperature dropped abruptly to about 100°F. The resulting thermal stress completely fractured both the number 1 and number 2 (backup) seals, and water and steam began blowing out directly to the drywell around the seal assembly. The flow rate increased over time, eventually reaching about 27 gpm, based on one indication which was averaged over a two-hour period.

Around 1000, operators attempted to close the recirc pump suction and discharge valves, but were unable to fully close the suction valve. Recirc pump 1A seal temperature continued rising, exceeding 300°F. Around

1225, an operator entered the drywell again and manually closed the recirc pump suction valve, stopping the leak.

B.46.3 Additional Event-Related Information

None.

B.46.4 Modeling Assumptions

This event was modeled as a scram with condensate, feedwater, CRD, and RCIC inoperable. Condensate and feedwater were assumed failed, due to the inadequate condensate inventory. The CRD and RCIC systems were assumed failed due to the equipment failures previously described.

B.46.5 Analysis Results

The conditional core damage probability estimated for this event is 1.1×10^{-4}. The dominant core damage sequence, highlighted on the event tree in Figure B.46.1, involves the observed scram, failure of the power conversion system, feedwater failure, HPCI and RCIC failure, and failure of automatic depressurization (ADS) and CRD injection.

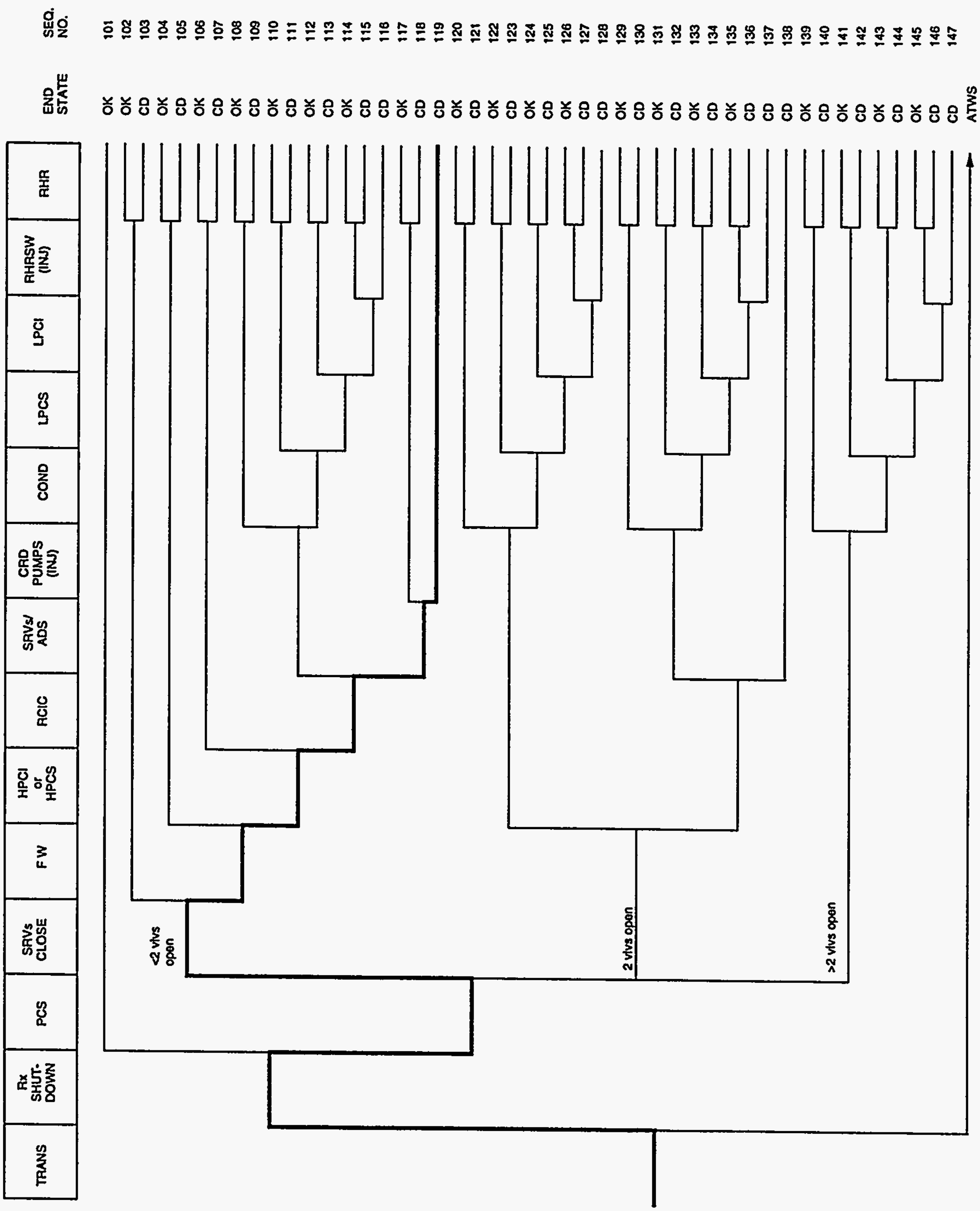

Figure B.46.1 Dominant core damage sequence for LER 373/82-093, 373/82-094

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 373/82-093
Event Description: Scram and multiple failures
Event Date: August 21, 1982
Plant: LaSalle 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.1E-04
Total	1.1E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads C RD(INJ)	CD	8.5E-05	2.4E-01
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci rhr.and.pcs.nrec	CD	1.8E-05	8.3E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci rhr.and.pcs.nrec	CD	1.8E-05	8.3E-03
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads C RD(INJ)	CD	8.5E-05	2.4E-01

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\lasalle1.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail

trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	2.9E-01 > 1.0E+00	3.4E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-01 > 1.0E+00		
hpci	2.0E-02	3.4E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
CRD(INJ)	1.0E-02 > 1.0E+00	1.0E+00	1.0E-02
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
cond	1.0E+00	1.0E-01	1.0E-03
lpcs	2.0E-02	1.0E+00	
lpci	6.0E-04	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.0E-03	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.0E-03	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.0E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	1.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.7E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.47 LER No. 373/83-057

Event Description: Scram, LOFW with RCIC Inoperable

Date of Event: June 1, 1983

Plant: LaSalle 1

B.47.1 Summary

Following a loss of feedwater and scram, the operators at LaSalle 1 had difficulty providing adequate makeup to the reactor vessel. As the reactor core isolation cooling (RCIC) system and the motor-driven feedpump were inoperable, operators blew down reactor pressure to 250 psig to allow makeup using the condensate pumps. The status of the high pressure core spray (HPCS) system during the event is unclear. The conditional core damage probability estimated for this event is 2.1 x 10^{-5}.

B.47.2 Event Description

LaSalle 1 experienced trips of the condensate booster pumps and the reactor feed pumps around 0230 on June 1, 1983. Very shortly thereafter, the reactor scrammed on low level. Operators were apparently unable to restore the turbine-driven reactor feed pumps to service and the motor-driven pump was unavailable, as was the RCIC system. The status of the high-pressure core spray system during the event is not given.

Because of the difficulties experienced in providing adequate makeup to the vessel, operators blew down the vessel to 250 psig. This permitted them to make up with the condensate system.

Reactor pressure dropped from about 920 psig to about 250 psig in about one-half hour. Within a one-hour period, reactor vessel temperature dropped from 536°F to 418°F, thus exceeding Technical Specification cooldown limits. Subsequently, General Electric (GE) performed an assessment of design information provided by the reactor vessel supplier, Combustion Engineering. GE concluded that since the limiting components for thermal stress were the vessel flange bolts and, since the vessel water level never reached the flange, the vessel stresses experienced during the event were bounded by those experienced in normal shutdown. Stresses were apparently not calculated for other points in the vessel.

B.47.3 Additional Event-Related Information

None.

B.47.4 Modeling Assumptions

This event was modeled as a scram and loss of feedwater, with RCIC unavailable. HPCS was assumed to be available, although it is odd that it was not used during the event. (An additional calculation was performed assuming HPCS was unavailable, to explore the sensitivity of the model to this.)

B.47.5 Analysis Results

The conditional core damage probability estimated for this event is 2.1×10^{-5}. The dominant core damage sequence, highlighted on the event tree in Figure B.47.1, involves the observed scram, failure of the power conversion system, main feedwater failure, HPCS success and failure of residual heat removal (RHR). If HPCS is assumed to have been unavailable, the conditional core damage probability estimate is increased to 1.3×10^{-4}.

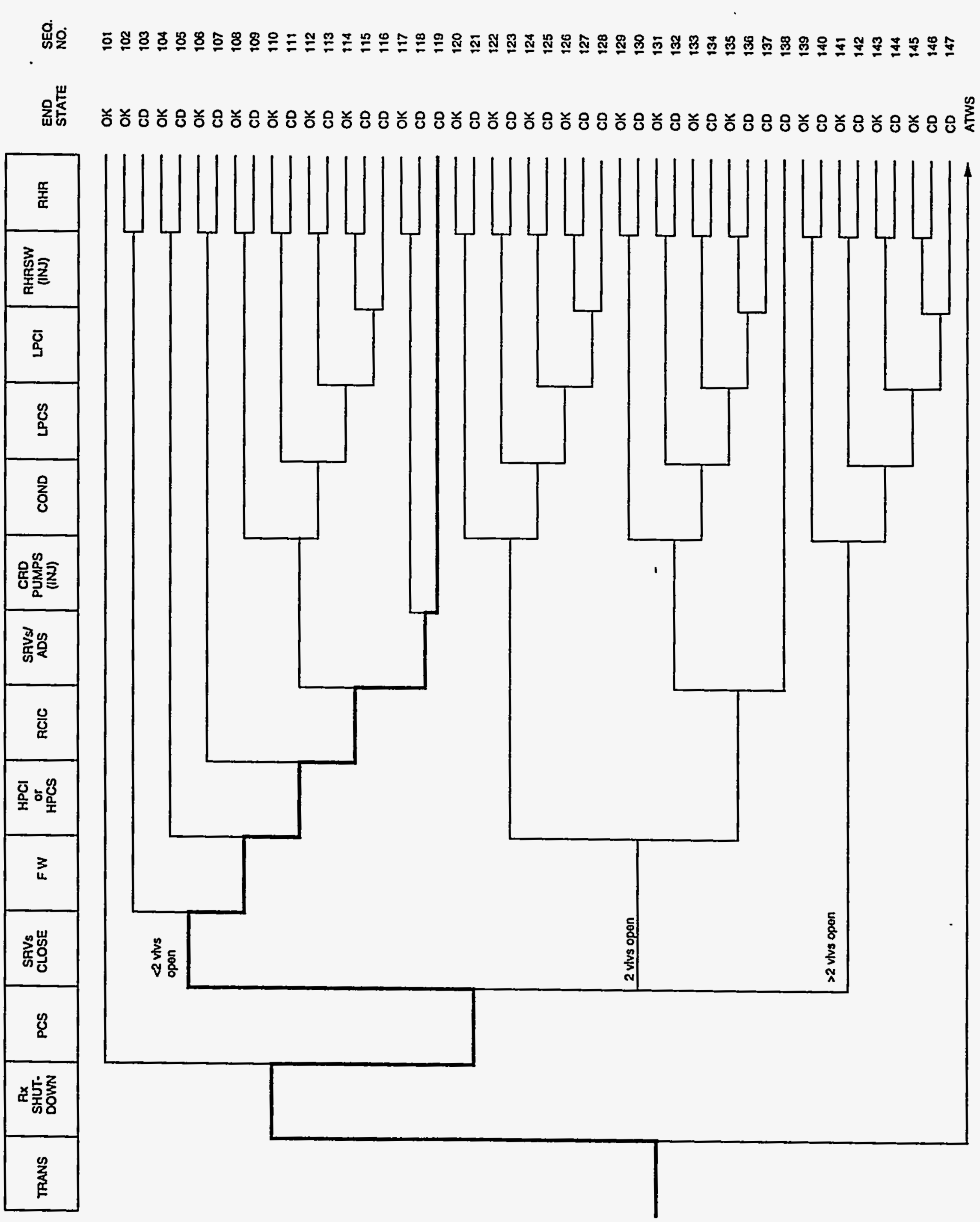

Figure B.47.1 Dominant core damage sequence for LER 373/83-057

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 373/83-057
Event Description: Scram. LOFW with RCIC inop
Event Date: June 1, 1983
Plant: LaSalle 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	2.1E-05
Total	2.1E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW rhr.and.pcs.nrec	CD	1.2E-05	5.5E-03
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci rhr.and.pcs.nrec	CD	6.2E-06	2.8E-03
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	5.8E-07	8.1E-02
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW rhr.and.pcs.nrec	CD	1.2E-05	5.5E-03
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci rhr.and.pcs.nrec	CD	6.2E-06	2.8E-03
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	5.8E-07	8.1E-02
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\lasalle1.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	2.9E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-01 > 1.0E+00		
hpci	2.0E-02	3.4E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-02	1.0E+00	
lpci	6.0E-04	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.0E-03	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.0E-03	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.0E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	1.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.7E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 373/83-057
Event Description: Scram, LOFW with RCIC inop
Event Date: June 1, 1983
Plant: LaSalle 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.3E-04
Total	1.3E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
119	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI RCIC srv.ads c rd(inj)	CD	8.5E-05	2.4E-01
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.6E-05	7.0E-01
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW rhr.and.pcs.nrec	CD	1.2E-05	5.5E-03
109	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI RCIC -srv.ads -c ond rhr	CD	5.7E-06	3.6E-03
111	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI RCIC -srv.ads c ond -lpcs rhr	CD	2.9E-06	1.8E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW rhr.and.pcs.nrec	CD	1.2E-05	5.5E-03
109	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI RCIC -srv.ads -c ond rhr	CD	5.7E-06	3.6E-03
111	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI RCIC -srv.ads c ond -lpcs rhr	CD	2.9E-06	1.8E-03
119	trans -rx.shutdown PCS srv.ftc.<2 MFW HPCI RCIC srv.ads c rd(inj)	CD	8.5E-05	2.4E-01
138	trans -rx.shutdown PCS srv.ftc.2 HPCI srv.ads	CD	1.6E-05	7.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\lasalle1.82

PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	2.9E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.9E-01 > 1.0E+00		
HPCI	2.0E-02 > 1.0E+00	3.4E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-02 > 1.0E+00		
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-02	1.0E+00	
lpci	6.0E-04	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.0E-03	1.6E-02	1.0E-05
rhr.and.pcs.nrec	1.0E-03	8.3E-03	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.0E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	1.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.7E-01	
ep.rec	1.7E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.48 LER No. 373/83-117, -147

Event Description: B RHR Heat Exchanger Outlet Valve Failure

Date of Event: September 30, 1983

Plant: LaSalle 1

B.48.1 Summary

Both licensee event reports indicate that the B train residual heat removal (RHR) heat exchanger outlet valve failed to open on demand, due to hydraulic locking. Manual attempts to open the valve were unsuccessful on both occasions. A reactor scram occurred on November 3, during the time that the B train of RHR was potentially unavailable. The conditional core damage probability estimated for this event is 1.4×10^{-4}.

B.48.2 Event Description

On September 20, 1983, while LaSalle 1 was in cold shutdown, operators at LaSalle attempted to open the B RHR heat exchanger outlet valve, but were unable to do so. Attempts using the valve motor operator and manual attempts were both unsuccessful. Subsequent inspection determined that the valve was experiencing hydraulic locking when water became trapped in the bonnet cavity. Since the bonnet cavity did not have any means to vent off water trapped within, the valve could become locked in the closed position. Licensee event report 373/83-117 indicates that the problem was recurring.

After the event, the motor operator for the valve was inspected. The motor windings were found to be burned and the motor was replaced. A plan was then formulated to check the performance of the valve over the next two complete shutdowns before making a permanent fix.

On November 12, 1983, a few days after a scram, operators attempted to open the B RHR heat exchanger outlet valve but, again, were unable to do so. The licensee event report for this event, 373/83-147, refers to the test plan of the prior event but does not make it clear if this was the first or second complete shutdown.

Again, the cause of the valve failure was found to be hydraulic locking. The valve motor operator was again found to be burned, and the motor was replaced.

B.48.3 Additional Event-Related Information

None.

B.48.4 Modeling Assumptions

This event was modeled as a scram with the B RHR heat exchanger assumed inoperable. Train B of RHR was assumed to be inoperable for all modes requiring RHR service water (SW) cooling. Although the specific failure discovered was not reported to be present in redundant portions of the system, other potential common cause failure modes remained and could have affected system performance. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as a part of the postulated event.

The plant power conversion system was assumed to have remained available but vulnerable to failure during this event.

It would also have been appropriate to have modeled the period that the B RHR heat exchanger was unavailable, thereby analyzing the risk impact of potential initiators which could have occurred. However, while the problems with the B train RHR heat exchanger outlet valve were recurring and apparently existed during much of 1983, limited information was available concerning other failures and plant operating history during that time. Therefore, the unavailability was not modeled.

B.48.5 Analysis Results

The conditional core damage probability estimated for this event is 1.4×10^{-4}. The dominant core damage sequence for this event, highlighted on the event tree in Figure B.48.1, involves scram, failure of the power conversion system, main feedwater success and RHR failure.

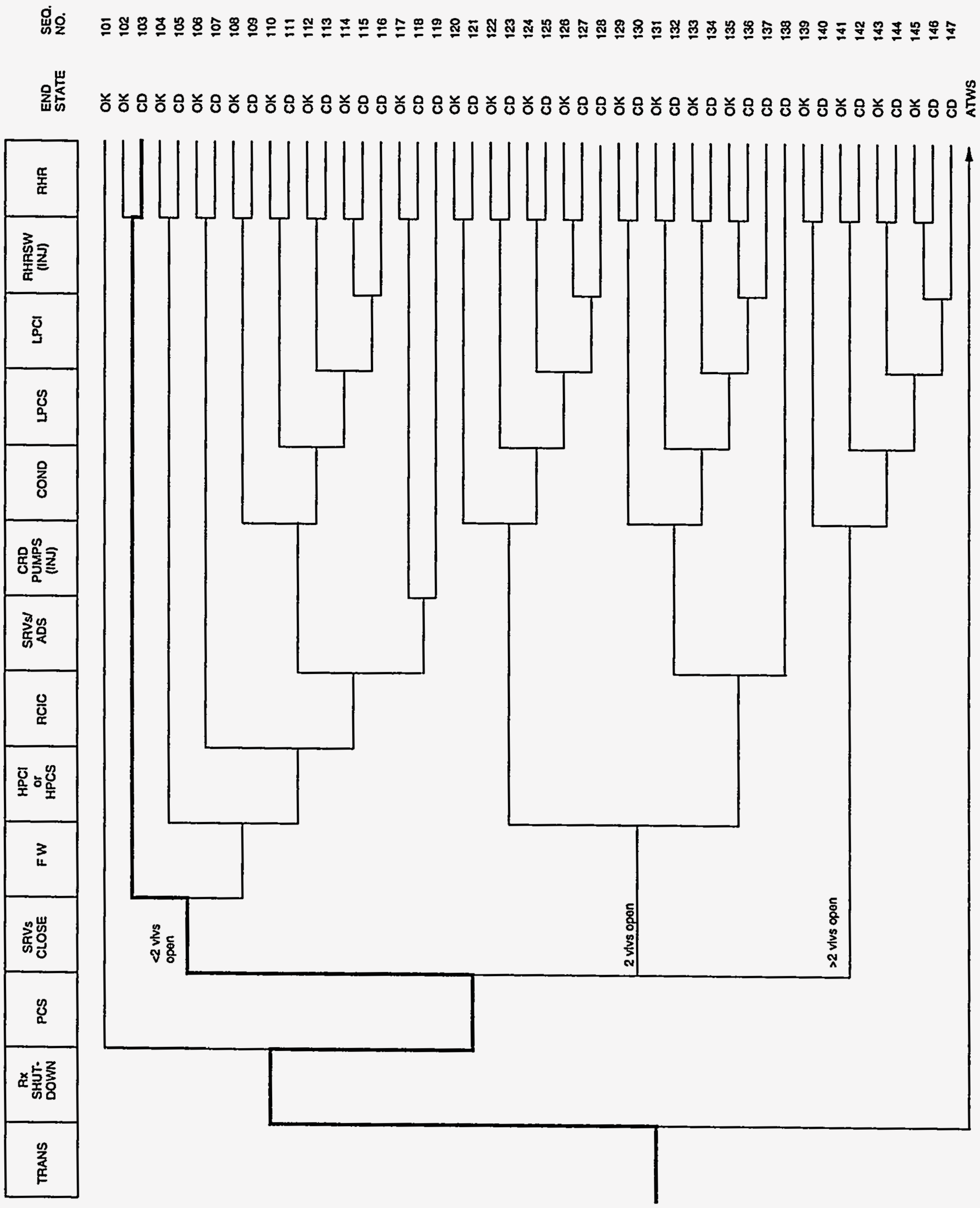

Figure B.48.1 Dominant core damage sequence for LER 373/83-117, -147

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 373/83-117
Event Description: B RHR heat exchanger outlet valve failure
Event Date: September 30. 1983
Plant: LaSalle 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.4E-04
Total	1.4E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.3E-04	7.7E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	1.4E-05	2.8E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown pcs srv.ftc.<2 -mfw RHR.AND.PCS.NREC	CD	1.3E-04	7.7E-03
105	trans -rx.shutdown pcs srv.ftc.<2 mfw -hpci RHR.AND.PCS.NREC	CD	1.4E-05	2.8E-03

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\lasalle1.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
hpci	2.0E-02	3.4E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-02	1.0E+00	
lpci	6.0E-04	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.0E-03 > 1.0E-01	1.6E-02	1.0E-05
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01		
RHR.AND.PCS.NREC	1.0E-03 > 1.0E-01	8.3E-03	1.0E-05
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01		
RHR/-LPCI	0.0E+00 > 1.0E-01 **	1.0E+00	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.0E-03 > 1.0E-01	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	2.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01		
RHR(SPCOOL)/-LPCI	1.0E-03 > 1.0E-01	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02 > 1.0E+00		
Train 2 Cond Prob:	1.0E-01		
ep	2.9E-03	8.7E-01	
ep.rec	1.7E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.49 LER No. 387/82-061

Event Description: ESW Pumps B and D Fail to Start

Date of Event: December 22, 1982

Plant: Susquehanna 1

B.49.1 Summary

On December 22, 1982, while performing the loss of offsite power (LOOP) test, the B and D emergency service water (ESW) pumps failed to start. This resulted in a loss of train B of ESW, which would have subsequently failed residual heat removal (RHR) pumps B and C. Earlier in the day, the reactor scrammed following turbine valve fast closure. The conditional core damage probability estimated for this event is 4.3 x 10^{-5}.

B.49.2 Event Description

On December 22, 1982, while performing the LOOP test, the B and D ESW pumps failed to start. This resulted in a loss of train B of ESW. The operators manually started the pumps prior to overheating of the serviced equipment (i.e., residual heat removal (RHR) pumps B and C, etc.). An investigation revealed that the pump B failure was the result of loose wires on a relay terminal, while the pump D failure was the result of loose wires on relay terminals, a loose states link, and an out-of-adjustment instantaneous contact. These problems were corrected, train A equipment was examined to determine whether the same failures were present (they were not), and the pumps retested.

Earlier in the day, as part of scheduled startup testing, generator output breakers were opened, causing a reactor scram on turbine control valve fast closure trip.

B.49.3 Additional Event-Related Information

Susquehanna's emergency service water system consists of two independent divisions (trains A and B), each of which is designed to supply 100 percent of the flow required by one division in both units plus cooling for four emergency diesel generators (i.e., DGs A, B, C, and D). Each division has two motor-driven pumps, each of which is capable of providing sufficient flow to remove the heat from the loads cooled by the division. ESW pumps A and C comprise train A and pumps B and D comprise train B. Train B provides cooling for diesel generators A, B, C, and D; pump cooling for RHR pumps B and C; plus cooling for other loads.

Susquehanna's RHR pumps can be operated in several modes. These include low-pressure coolant injection (LPCI), suppression pool cooling, shutdown cooling, containment spray, reactor head spray, and fuel pool cooling. Susquehanna's individual plant examination (IPE) submittal states that the RHR pumps can be operated 30 minutes without pump cooling.

B.49.4 Modeling Assumptions

The event was modeled as a transient with two ESW pumps (train B) failed. This failure results in the loss of the B and C RHR pumps owing to loss of pump cooling. Unavailability of these two pumps affects RHR. To reflect the potential failure of the other two pumps due to the same failure mode, trains 1 and 2 of RHR, and RHR(SPCOOL) model were set to failed. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common cause failure was included in the analysis for those components that were assumed to have been failed as a part of the postulated event.

Because the scram was a part of the startup test program, the analysis assumed the unit was operating normally and was stable prior to the scram. The failure probability for power conversion system (PCS) was revised to only address potential failures after the scram occurred. A value of 0.01 was utilized, consistent with the Susquehanna IPE. The nonrecovery probability for RHR was revised to 0.054 to reflect the RHRSW failures (see Appendix A). For sequences involving potential RHR or PCS recovery, the nonrecovery estimate was revised to 0.054 x 0.52 (PCS nonrecovery), or 0.028.

B.49.5 Analysis Results

The estimated conditional core damage probability for the event is 4.3×10^{-5}. The dominant sequence highlighted on the event tree in Figure B.49.1 involved a transient initiator followed by successful reactor shutdown, failure of the power conversion system, successful feedwater recovery, and failure of the residual heat removal system.

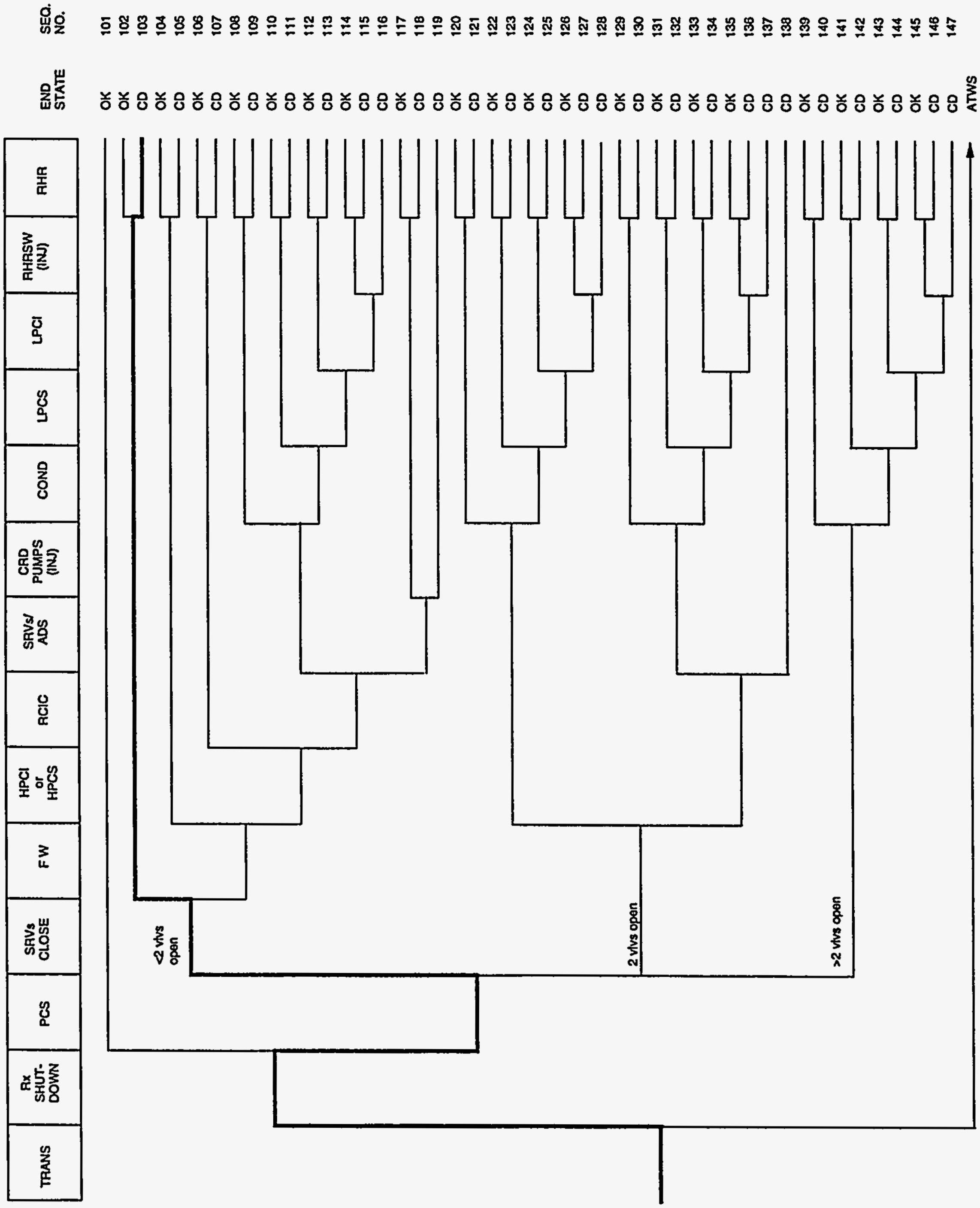

Figure B.49.1 Dominant core damage sequence for LER 387/82-061

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 387/82-061
Event Description: ESW pumps B and D fail to start
Event Date: December 22, 1982
Plant: Susquehanna 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	4.3E-05
Total	4.3E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	2.8E-05	1.8E-02
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	1.4E-05	9.5E-03

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	2.8E-05	1.8E-02
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	1.4E-05	9.5E-03

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\susque.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	

PCS	1.7E-01 > 1.0E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E-02		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.5E-01	1.6E-02 > 5.4E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-01	8.3E-03 > 2.8E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.5E-01	1.0E+00 > 5.4E-02	1.0E-05
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	0.0E+00 > 1.5E-01		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.5E-01	1.0E+00	1.0E-03
Branch Model: 1.OF.4+ser+opr			
Train 1 Cond Prob:	1.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.5E-01	1.0E+00	1.0E-03
Branch Model: 1.OF.1+ser+opr			
Train 1 Cond Prob:	0.0E+00 > 1.5E-01		
Serial Component Prob:	2.0E-03		
ep	1.4E-03	8.7E-01	
ep.rec	2.1E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.50 LER No. 387/83-051

Event Description: RCIC System Unavailable Owing to Governor Valve Problem

Date of Event: March 22, 1983

Plant: Susquehanna 1

B.50.1 Summary

On March 22, 1983, in response to a low reactor pressure vessel (RPV) water level signal following a scram, the reactor core isolation cooling (RCIC) system initiated and then tripped on turbine overspeed. The conditional core damage probability estimated for the event is 1.2×10^{-5}.

B.50.2 Event Description

On March 22, 1983, in response to a low RPV water level signal following a scram, the RCIC system initiated and then tripped on turbine overspeed. Operations personnel manually started the RCIC immediately after the overspeed trip; the high-pressure injection system started; and vessel level was recovered and maintained. Investigations revealed the overspeed trip was caused by the slow response of the governor valve during system start. The slow response was caused by dirt deposited in the opening of the pilot valve. This was corrected on May 17, 1984 by installing a new upgraded governor in which the pilot valve opening was enlarged.

The scram was caused by an operator error that allowed air to be injected into the reactor vessel via the condensate demineralizers, resulting in high main steam radiation signals.

B.50.3 Additional Event-Related Information

The RCIC system consists of a single turbine-driven pump that can provide primary coolant makeup at a maximum rate of 600 gpm. The RCIC pump is provided with two suction sources. The primary source is the condensate storage tank (CST), with the suppression pool providing the secondary source. The system is designed to swap suction from the CST to the suppression pool on low CST level.

B.50.4 Modeling Assumptions

Given that a plant trip occurred, this event was modeled as a transient initiator. The main steam isolation valves (MSIVs) were assumed to have closed as a result of the high main steam radiation signals. This would result in unavailability of the power conversion system (PCS) and the feedwater (FW) system, since Susquehanna uses turbine-driven FW pumps. In addition, Susquehanna's individual plant examination (IPE) submittal states that flow through the MSIVs is needed for the turbine-driven FW pumps; thus, it was assumed that the use of the MSIV bypass valves to supply steam for the FW pumps was not appropriate. RCIC was assumed failed, owing to the governor valve problem. Short-term recovery of PCS or FW was not considered, since the MSIVs had closed. Recovery of RCIC was considered since the control room operator manually started RCIC immediately after the overspeed trip. This action was assumed to take place in the control room with a failure probability of 0.01. Thus, the probability of nonrecovery of RCIC was set to 0.052 [p(nrec) = 0.01 + 0.06 * 0.7] to account for the fact that RCIC might also fail from other causes. The nonrecovery probability for PCS was revised to 0.017 to reflect the MSIV closure (see Appendix A). Combining this value with the estimated long-term residual heat removal (RHR) nonrecovery probability of 0.016 results in a combined nonrecovery probability for RHR and PCS of 2.7E-4.

B.50.5 Analysis Results

The estimated conditional core damage probability for the event is 1.2×10^{-5}. The dominant sequence, highlighted on the event tree in Figure B.50.1, involved a transient initiator followed by successful reactor shutdown, failure of the power conversion system, failure of the feedwater system and failure of the residual heat removal system.

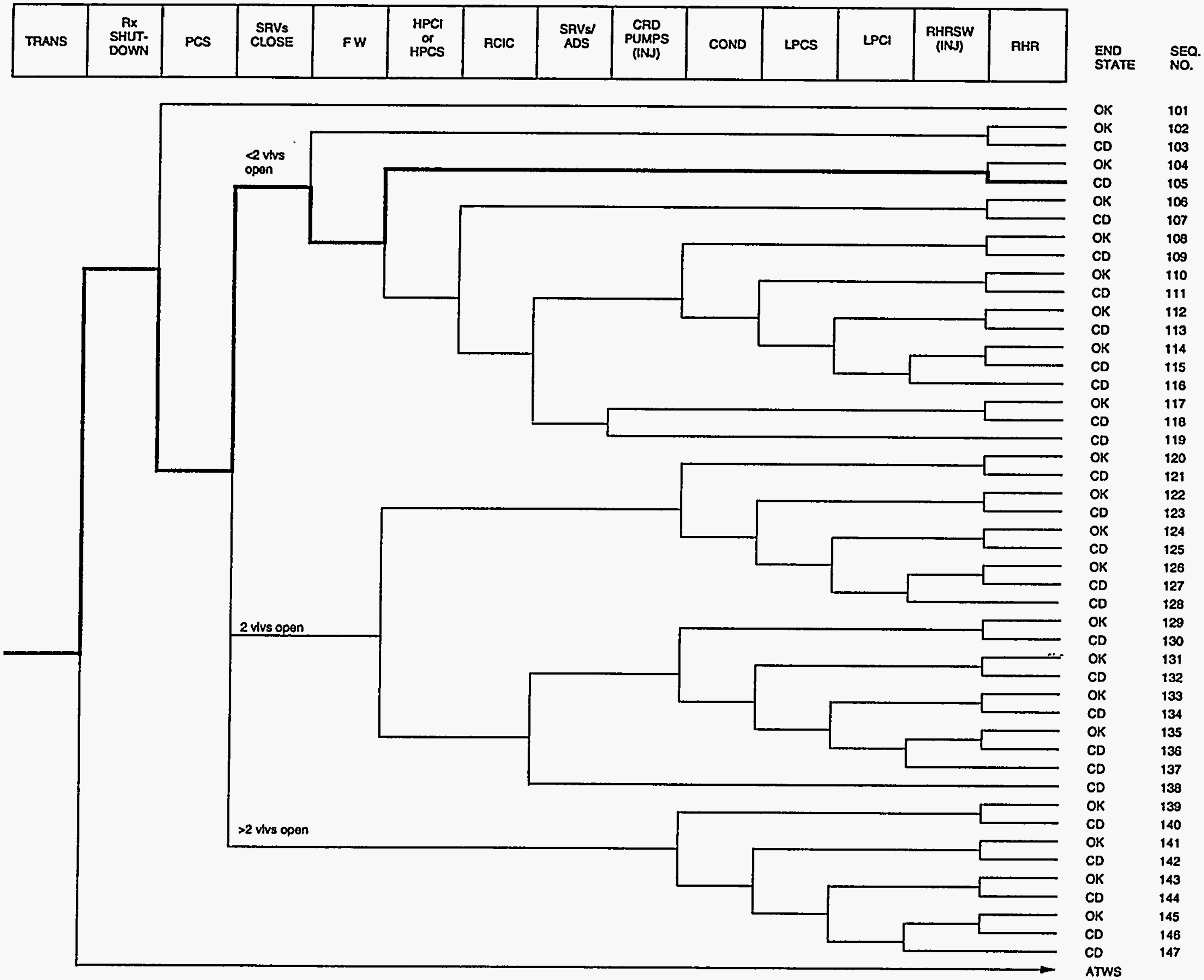

Figure B.50.1 Dominant core damage sequence for LER 387/83-051

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 387/83-103
Event Description: Scram. MSIV isolation and RCIC failure
Event Date: July 7, 1983
Plant: Susquehanna 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.4E-05
Total	1.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\susque.82

PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	1.4E-03	8.7E-01	
ep.rec	2.1E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.51 LER No. 387/83-103

Event Description: RCIC System Unavailable Owing to Governor Valve Problem

Date of Event: July 7, 1983

Plant: Susquehanna 1

B.51.1 Summary

On July 7, 1983, during testing to demonstrate the operability of the reactor core isolation cooling (RCIC) system, the RCIC turbine tripped. RCIC had also tripped two days earlier, during response to a scram. The conditional core damage probability estimated for the event is 1.4×10^{-5}.

B.51.2 Event Description

On July 7, 1983, during testing to demonstrate the operability of the RCIC system, the RCIC turbine tripped. Prior to the test, on July 5, a plant trip had occurred, RCIC was demanded, and subsequently tripped. Based on vendor recommendations, clearances between the governor valve and bonnet guide sleeve were measured and found restrictive. The governor valve was reworked to updated vendor specifications and the system successfully retested.

The scram on July 5, 1983 was caused by main steam line radiation spikes associated with placing condensate demineralizers in service.

B.51.3 Additional Event-Related Information

The RCIC system consists of a single turbine-driven pump that can provide primary coolant makeup at a maximum rate of 600 gpm. The RCIC pump is provided with two suction sources. The primary source is the condensate storage tank (CST), with the suppression pool providing the secondary source. The system is designed to swap from the CST to the suppression pool on low CST level.

B.51.4 Modeling Assumptions

Given that a plant trip had occurred on July 5 with a demand for RCIC, this event was modeled as a transient initiator. The main steam isolation valves (MSIVs) are assumed to have closed as a result of the radiation spikes. This will result in unavailability of the power conversion system (PCS) and the feedwater (FW) system since Susquehanna uses turbine-driven FW pumps. In addition, Susquehanna's IPE submittal states that flow through the MSIVs is needed for the turbine-driven FW pumps; thus, it is assumed that the use of the MSIV bypass valves to supply steam for the FW pumps is not appropriate. RCIC was assumed failed, owing to the governor valve problem. Short-term recovery of PCS or FW was not considered since the MSIVs had closed. Recovery of RCIC was not considered since RCIC had tripped twice in two days. The nonrecovery probability for PCS was revised to 0.017 to reflect the MSIV closure (see Appendix A). Combining this value with the estimated long-term RHR nonrecovery probability of 0.016 results in a combined probability for RHR and PCS of 2.7E-4.

B.51.5 Analysis Results

The estimated conditional core damage probability for the event is 1.4×10^{-5}. The dominant sequence, highlighted on the event tree in Figure B.51.1, involves a transient initiator followed by successful reactor shutdown, failure of the power conversion system, failure of the feedwater system, and failure of the residual heat removal system.

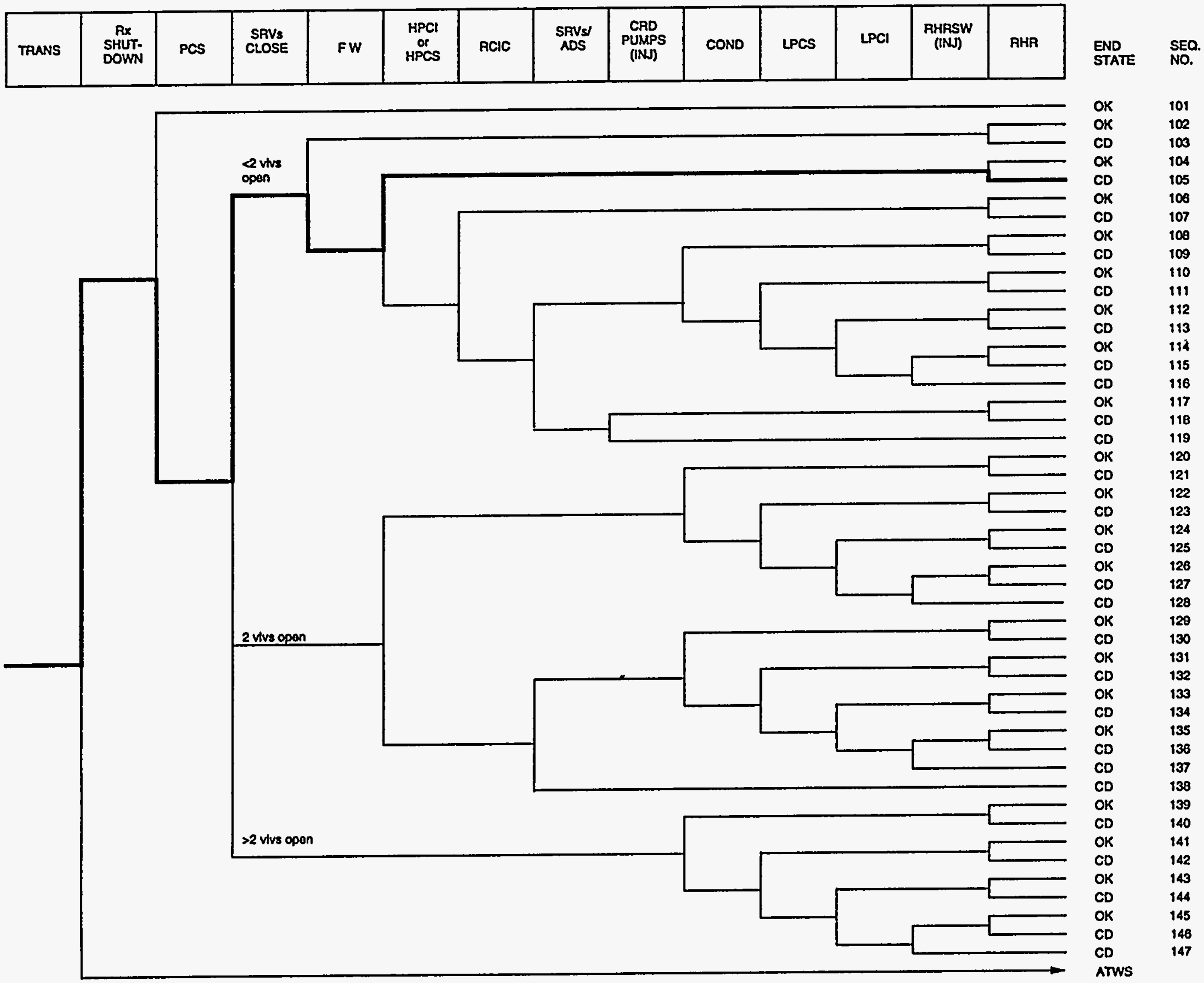

Figure B.51.1 Dominant core damage sequence for LER 387/83-103

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 387/83-103
Event Description: Scram, MSIV isolation and RCIC failure
Event Date: July 7, 1983
Plant: Susquehanna 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.4E-05
Total	1.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
119	trans -rx.shutdown PCS srv.ftc.<2 MFW hpci RCIC srv.ads c rd(inj)	CD	1.7E-06	1.7E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\susque.82

PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	1.4E-03	8.7E-01	
ep.rec	2.1E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

B.52 LER No. 387/83-120

Event Description: RCIC System Unavailable Owing to Governor Valve Problem

Date of Event: August 28, 1983

Plant: Susquehanna 1

B.52.1 Summary

During a post-scram vessel level fluctuation on August 28, 1983, the reactor core isolation cooling (RCIC) system initiated and then tripped on turbine overspeed 3 seconds later. The conditional core damage probability estimated for the event is 1.2×10^{-5}.

B.52.2 Event Description

During a post-scram vessel level fluctuation on August 28, 1983, the RCIC system initiated and then tripped on turbine overspeed 3 seconds later. Operations personnel established manual control of RCIC and adjusted turbine speed to maintain proper vessel level. Investigations revealed the overspeed trip was caused by slow response of the governor valve during system start. The governor valve linkage travel was reduced by one-quarter inch and the system successfully retested.

The scram occurred when a main turbine stop valve opened, causing a main steam isolation valve (MSIV) isolation to occur. A scram followed owing to the MSIVs being less than 94% open. Spurious actuation of main steam line pressure switches is considered to be the cause of the scram.

B.52.3 Additional Event-Related Information

The RCIC system consists of a single turbine-driven pump that can provide primary coolant makeup at a maximum rate of 600 gpm. The RCIC pump is provided with two suction sources. The primary source is the condensate storage tank (CST), with the suppression pool providing the secondary source. The system is designed to swap suction from the CST to the suppression pool on low CST level.

B.52.4 Modeling Assumptions

Given that a plant trip occurred, this event was modeled as a transient initiator. The main steam isolation valves were closed as a result of the MSIV isolation. This will result in unavailability of the power conversion system (PCS) and the feedwater (FW) system since Susquehanna uses turbine-driven FW pumps. In addition, Susquehanna's individual plant examination (IPE) submittal states that flow through the MSIVs is needed for the turbine-driven FW pumps; thus, it is assumed that the use of the MSIV bypass valves to supply steam for the FW pumps is not appropriate. RCIC was assumed failed owing to the governor valve problem. Short-term recovery of PCS or FW was not considered, since the MSIVs had closed. Recovery of RCIC was considered since manual control of RCIC was established after the overspeed trip. This action was assumed to take place in the control room with a failure probability of 0.01. Thus, the probability of nonrecovery of RCIC was set to 0.052 [p(nrec) = 0.01 + 0.06 * 0.7] to account for the fact that RCIC might also fail from other causes. The nonrecovery probability for PCS was revised to 0.017 to reflect the MSIV closure (see Appendix A). Combining this value with the estimated long-term residual heat removal system (RHR) nonrecovery probability of 0.016 results in a combined nonrecovery probability for RHR and PCS of 2.7E-4.

B.52.5 Analysis Results

The estimated conditional core damage probability for the event is 1.2×10^{-5}. The dominant sequence, highlighted on the event tree in Figure B.52.1, involved a transient initiator followed by successful reactor shutdown, failure of the power conversion system, failure of the feedwater system, and failure of the residual heat removal system.

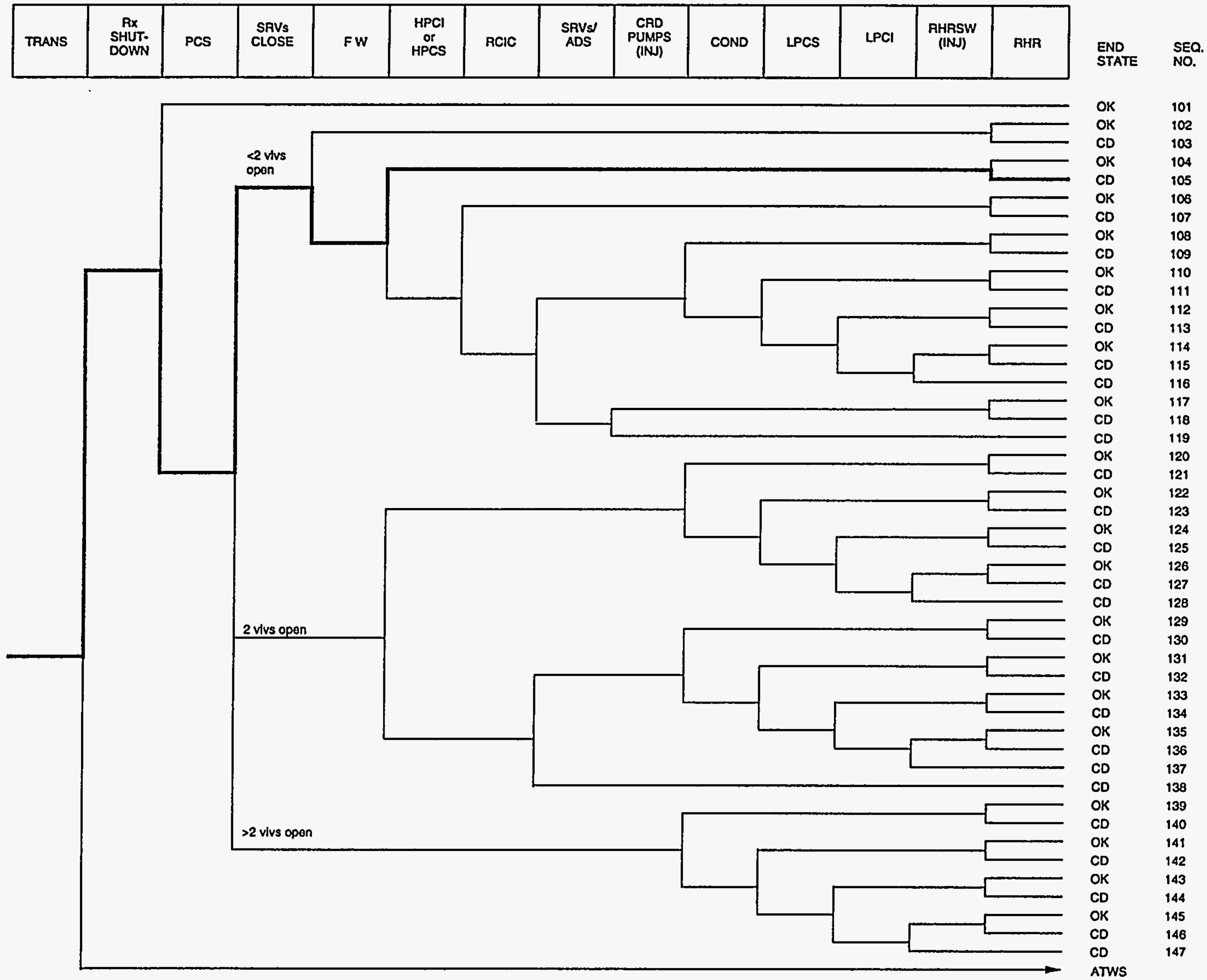

Figure B.52.1 Dominant core damage sequence for LER 387/83-120

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 387/83-120
Event Description: Scram, MSIV isolation and RCIC failure
Event Date: August 28, 1983
Plant: Susquehanna 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.2E-05
Total	1.2E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
103	trans -rx.shutdown PCS srv.ftc.<2 -MFW RHR.AND.PCS.NREC	CD	6.6E-06	1.8E-04
105	trans -rx.shutdown PCS srv.ftc.<2 MFW -hpci RHR.AND.PCS.NREC	CD	3.3E-06	9.1E-05
138	trans -rx.shutdown PCS srv.ftc.2 hpci srv.ads	CD	3.3E-07	4.9E-01
412	trans rx.shutdown -rpt -slcs PCS ads.inhibit	CD	3.4E-07	1.0E-01
413	trans rx.shutdown -rpt slcs	CD	4.1E-07	1.0E-01
414	trans rx.shutdown rpt	CD	6.7E-07	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\asp\1982-83\bwrc8283.cmp
BRANCH MODEL: c:\asp\1982-83\susque.82
PROBABILITY FILE: c:\asp\1982-83\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
PCS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 1.0E+00		
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
MFW	4.6E-01 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 5.2E-02	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > 1.0E+00		
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	1.7E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	1.6E-02	1.0E-05
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-04	8.3E-03 > 2.7E-04	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
ep	1.4E-03	8.7E-01	
ep.rec	2.1E-01	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

Appendix C: Precursors with Conditional Core Damage Probabilities Between 1.0×10^{-5} and 1.0×10^{-6}

C.0 Precursors with Conditional Core Damage Probabilities Between 1.0 x 10^{-5} and 1.0 x 10^{-6}

C.0.1 Accident Sequence Precursor Program Event Analyses for 1982-83

This appendix documents 1982 and 1983 operational events selected as accident sequence precursors with conditional core damage probabilities (CCDPs) between 1.0 x 10^{-5} and 1.0 x 10^{-6}.

Licensee event reports (LERs) describing operational events at commercial nuclear power plants were reviewed for potential precursors if

- the LER was identified as requiring review based on a computerized search of the Sequence Coding and Search System data base maintained at Oak Ridge National Laboratory, or
- the LER was identified as requiring review by the NRC Office for Analysis and Evaluation of Operational Data, or
- the LER was discussed in NUREG-0900 (*Report to Congress on Abnormal Occurrences)* or in issues of *Nuclear News* and appeared to be a potential precursor.

C.0.2 Precursors Identified

Fifty-six precursors with CCDPs between 1.0 x 10^{-5} and 1.0 x 10^{-6} were identified from the 1982-1983 LERs reviewed. Events in this group were identified as precursors if they met one of the following precursor selection criteria and the conditional core damage probability estimated for the event was between 1.0 x 10^{-5} and 1.0 x 10^{-6}.

- the event involved the total failure of a system required to mitigate the effects of a core damage initiator,
- the event involved the degradation of two or more systems required to mitigate the effects of a core damage initiator,
- the event involved a core damage initiator such as a loss of offsite power or small-break loss-of-coolant accident, or
- the event involved a reactor trip or loss of feedwater with a degraded safety system. The precursors identified are listed in Table C.1.

C.0.3 Event Documentation

This appendix provides summaries for 56 precursor events with CCDPs between 1.0 x 10^{-5} and 1.0 x 10^{-6}. The precursors are in docket/LER number order.

Summarized Precursors

Table C.1 List of ASP Events with CCDPs Between 1.0 x 10^{-5} and 1.0 x 10^{-6} [(a)]

Event Identifier	Plant	Description	Page
237/83-045 237/83-046 237/83-052	Dresden 2	Core Spray A, LPCI A, and SDC A Inoperable; Scram	C-6
247/83-005	Indian Point 2	Trip with Turbine-Driven AFW Pump Inoperable	C-7
250/82-008	Turkey Point 3	Trip with High-Head Safety Injection Pump Failure	C-7
255-82-002	Palisades	Reactor Shutdown with AFW Auto-Initiation Inoperable	C-8
265/82-010	Quad Cities 2	Trip with HPCI Inoperable	C-8
265/82-017 265/82-018	Quad Cities 2	HPCI and One EDG Inoperable	C-9
271/82-019	Vermont Yankee	Trip with HPCI Inoperable	C-10
272/82-041	Salem 1	Trip with Two Charging Pumps Inoperable	C-11
272/82-056 272/82-053	Salem 1	Trip with One AFW Pump and One EDG Inoperable	C-12
272/82-069	Salem 1	Trip with One Charging Pump Inoperable	C-12
277/83-028	Peach Bottom 2	Trip with Two HPSW Pumps Inoperable	C-13
278/82-004	Peach Bottom 3	Trip with One LPCS and RHR Pump Inoperable	C-13
281/83-005	Surry 2	Trip with AFW Pump Inoperable	C-14
282/82-015	Prairie Island 1	Two EDGs Simultaneously Inoperable for 1.5 Hours	C-14
285/82-009	Fort Calhoun	Three of Four CCW Heat Exchangers Inoperable	C-15
293/82-043 293/82-042	Pilgrim	RCIC and HPCI Suction Valves Inoperable	C-16
293/83-039	Pilgrim	Trip with HPCI Inoperable	C-16

Event Identifier	Plant	Description	Page
293/83-052	Pilgrim	Trip with HPCI Inoperable	C-17
295/82-025	Zion 1	Postulated Grid/Weather-Related LOOP with Two EDGs Inoperable for 24 hours	C-17
295/82-033	Zion 1	Postulated Grid/Weather-Related LOOP with Two EDGs Inoperable for 24 hours	C-19
298/83-014	Cooper	Reactor Trip with HPCI Unavailable	C-21
302/82-041 302/82-051 302/83-037	Crystal River 3	Trip with One RHR Train Inoperable	C-21
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(a) Acronyms used in table are defined as follows: low pressure coolant injection (LPCI), shutdown cooling (SDC), auxiliary feedwater (AFW), high-pressure coolant injection (HPCI), emergency diesel generator (EDG), high-pressure service water (HPSW), low-pressure core spray (LPCS), residual heat removal (RHR), component cooling water (CCW) emergency service water (ESW), emergency core cooling system (ECCS), reactor core isolation cooling (RCIC), control rod drive (CRD), turbine driven auxiliary feedwater (TDAFW), high-pressure injection (HPI), low-pressure safety injection (LPSI)

C.1 LER No. 237/83-045, -046, -052

Event Description: Core Spray A, LPCI A, and SDC A Inoperable, Scram

Date of Event: June 8, 1983

Plant: Dresden 2

Summary

On June 8, 1983, Dresden Unit 2 was operating at approximately 100% power when the circuit breaker for core spray (CS) injection valve M02-1402-25B was found tripped after the valve had been exercised. The breaker trip setting was found to have been adjusted incorrectly some time before. On June 15, 1983, Dresden was operating at approximately 70% power when low-pressure coolant injection (LPCI) train A was aligned to pump down the suppression pool. As the system was being secured, it was discovered that the bolts attaching the train A minimum flow bypass valve motor to the valve (MO 2-1501-13A) had broken, allowing the motor to separate from the valve. On June 21, 1983, operators were attempting to align shutdown cooling (SDC) following a scram, when the A train SDC return valve failed to open. An investigation revealed that a packing leak on another valve allowed water to enter the SDC return valve motor, causing a fault and burning out the motor. Dresden 2 was returned to service after an extended outage on April 25, 1983. Two scrams occurred during the time that the CS, LPCI, and SDC systems were inoperable, one on June 11 and the other on June 20. The unit was returned to service immediately after the June 11 scram but was shut down after the June 20 scram.

This event is complicated by multiple overlapping equipment unavailabilities. The CS train B failure identified on June 8 was assumed to have existed for half of a one-month surveillance interval. The LPCI A minimum flow valve failure identified on June 15 was assumed to have resulted in failure of the A LPCI train and to have existed for half of a one-month surveillance interval. The SDC A return valve failure identified when the system was demanded on June 21 was assumed to have existed for half of the interval since the last

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known demand on the system, at the end of the outage on April 25. The potential impact of these events was therefore evaluated as the sum of the effects of the following: a scram with SDC A return inoperable (June 20), a scram with LPCI A and SDC A inoperable (June 11), 7 days of unavailability of LPCI A, CS B, and SDC A; 7 days of unavailability of LPCI A and SDC A, 8 days of unavailability of CS B and SDC A; and 6 days of unavailability of SDC.

Calculated conditional core damage probabilities for the events involving unavailabilities were small relative to those for the scrams and were neglected. The conditional core damage probability estimated for the combined scram events is 3.3×10^{-6}. The dominant core damage sequence involves the observed transient, failure to scram, and failure of the recirculation pump trip breakers to operate.

C.2 LER No. 247/83-005

Event Description: Transient with the Turbine-Driven AFW Pump Inoperable

Date of Event: March 8, 1983

Plant: Indian Point 2

Summary

During normal operation on March 8, 1983, while the No. 22 turbine-driven auxiliary feedwater (AFW) pump was being brought up to operating speed for a bimonthly surveillance test, the outboard bearing began smoking and the pump was removed from service for maintenance. Inspection of the pump indicated that repacking was required. A trip had occurred less than a month prior to this event on February 13 *(Licensed Operating Reactors, Status Summary Report* (NUREG-0020), published monthly, U.S. Nuclear Regulatory Commission, hereafter referred to as NUREG-0020). Assuming that a bimonthly test indicates that a test was performed every two months, half the surveillance period of the turbine-driven AFW pump is a month.

Since the trip occurred less than a month earlier, it was assumed that the problem with the AFW pump existed at the time of the trip. This event was modeled as a transient with the turbine-driven train of AFW failed. The model for the failure of AFW given an anticipated transient without scram (ATWS) requires the use of 2 of 3 pumps. The AFW/ATWS probability was set to 0.04 to reflect the probability of failure for motor-driven pump 22 or motor-driven pump 23. The conditional core damage probability of this event is 3.9×10^{-6}. The dominant core damage sequence involves a successful reactor trip, the failure of AFW, the failure of main feedwater, and failure of feed and bleed.

C.3 LER Number 250/82-008

Event Description: Failure of High-Head Safety Injection Pump

Date of Event: June 9, 1982

Plant: Turkey Point 3

Summary

On June 9, 1982 during power operation, the 3A high-head safety injection (HHSI) pump failed to start when the normal Unit 3 control switch was manually operated. The three remaining HHSI pumps were available. The breaker control panel fuses were replaced and the breaker was racked out and then back in. The 3A pump was then successfully started. No cause for the failure could be determined. The event was analyzed as a HHSI pump failure with a loss of feedwater (LOFW) transient.

A trip and apparent LOFW occurred on June 1, 1982. If a test interval of one month is assumed for the HHSI pumps and if HHSI pump 3A is assumed to have been unavailable for half of an interval, this event can be modeled as an HHSI pump failure in conjunction with an LOFW transient. One HHSI pump was assumed to be failed for both high-pressure injection (HPI) and feed-and-bleed functions. Since the cause of the failure was not identified, it was assumed that the remaining HHSI pumps were vulnerable to a similar failure. The conditional core damage probability estimated for this event is 3.3×10^{-6}. The dominant core damage sequence involves the observed transient, successful reactor trip, loss of auxiliary and main feedwater, and failure of feed and bleed.

C.4 LER No. 255/82-002

Event Description:	Reactor Shutdown with AFW Auto-Initiation Inoperable
Date of Event:	January 6, 1982
Plant:	Palisades

Summary

On January 6, 1982 during monthly testing of the auxiliary feedwater (AFW) system, the flow control valves failed to function properly. One valve did not open until 15 minutes after auto-initiation. The second valve had flow oscillations varying from 120 gpm to 170 gpm. Normal flow should be 150 gpm. The malfunction of these valves rendered the AFW auto-initiation inoperable. The valve controls were placed in manual, and the valves were positioned to deliver the required flow. Investigation revealed that the flow controllers were out of adjustment. Adjustments were made and operability was restored.

The unit was shut down on January 3 to repair several secondary-side leaks. It was assumed that AFW was manually initiated during the shutdown and that, had the unit tripped, operator action to initiate AFW would have been required. This event was modeled as a potential trip during plant shutdown with AFW inoperable. The malfunction of the AFW auto-initiation initially fails the AFW system when it is called for. By placing the valves in manual control, AFW can be recovered. This analysis assumes that both trains of AFW were inoperable without some operator action due to the failure of the auto-initiation feature. To reflect the initial failure of AFW, both trains of AFW were set to failed, and AFW given anticipated transient without scram (ATWS) (AFW/ATWS) was set to failed. The nonrecovery probability for AFW was modified to reflect the manual control capabilities that would recover AFW. The nonrecovery probability for AFW was set to 0.01 to reflect possible routine recovery capability from the control room (see Appendix A). The nonrecovery probability for AFW/ATWS was left at 1.0 due to the lack of time available for recovery given an ATWS. The probability of a reactor trip during the shutdown was assumed to be 0.1. The estimated conditional core damage probability for this event is 5.0×10^{-6}. The dominant sequence involved a postulated ATWS sequence with AFW failed.

C.5 LER No. 265/82-010

Event Description: Trip with HPCI Inoperable

Date of Event: June 24, 1982

Plant: Quad Cities 2

Summary

On June 24, 1982, with the plant increasing power in preparation for rolling the turbine and placing the unit online, HPCI pump discharge motor-operated valve 2-2301-8 failed to open when given a signal from the control room during an HPCI valve operability surveillance test. The HPCI was declared inoperable. The valve was manually opened and taken out of service. Investigation revealed that the open torque switch in the motor operator had a broken arm. The arm was replaced and the valve reassembled. The valve was opened successfully three times and high-pressure coolant injection (HPCI) was returned to service the next day.

A plant trip occurred approximately two days prior to the discovery of the faulty HPCI pump discharge valve. Thus, this event was modeled as a transient with HPCI assumed inoperable. The HPCI train probability was set to failed and the HPCI nonrecovery probability was set to 0.55 to reflect the ability of the operators to recover HPCI locally within the allowable recovery time (see Appendix A). The estimated conditional core damage probability for this event is 4.7×10^{-6}. The dominant sequence involves the trip with a postulated failure of the power conversion system, successful operation of main feedwater, and the failure of the residual heat removal system.

C.6 LER No. 265/82-017 and -018

Event Description: HPCI and One EDG Inoperable

Date of Event: October 1, 1982

Plant: Quad Cities 2

Summary

On October 1, 1982, during routine surveillance a small leak was discovered in the steam line break flange of the high-pressure coolant injection (HPCI) system supply due to a failed flange gasket. The licensee stated that the steam leakage may have been sufficient to cause HPCI isolation on a high HPCI area temperature following prolonged operation. A few days later on October 6, 1982, following monthly preventive maintenance on emergency diesel generator (EDG) 2, the EDG tripped on high temperature 10 minutes after loading due to fouled heat exchangers in the EDG cooling water system. Thus, this event was modeled as an unavailability of HPCI and one EDG. Assuming that both HPCI and the EDG were faulted for a period of half their surveillance periods prior to the discovery of the faults, the duration of the unavailability was estimated to be 10 days (240 hours). To reflect the failure of EDG 2, one train of emergency power was set to failed and all system trains that relied on EDG 2 (bus 24-1) given a loss of offsite power (LOOP) were set to unavailable. Since Unit 2 bus 24-1 can be fed by Unit 1 bus 14-1 through cross-connection, recovery of power to bus 24-1 was assumed possible from Unit 1 bus 14-1 by the closure of the normally open breakers 2429 and 1421 for plant-centered LOOPs.

This event was modeled as two cases. The first case examines the likelihood of the occurrence of a plant-centered LOOP during the unavailability with credit given for the ability to recover power through the use of the cross-connect. In this case, the LOOP frequency was revised to 1.39×10^{-5} with a short-term nonrecovery probability of 0.5, and offsite power recovery prior to battery depletion (EP.REC) was modified to 6.4×10^{-3} to reflect values for plant-centered LOOPs determined from the models described in *Revised LOOP Frequency and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989. The probability of failing to close breakers before battery depletion was assumed to be 0.10 (see Appendix A) and reflects the probability of the operators performing the required nonroutine actions in the required time from the control room. The probability of failing to recover power prior to battery depletion was revised to 0.29 (0.10 nonrecovery probability for closing the breakers + 0.19 probability of EDG1 failing given that EDG 2 and the swing EDG were failed). To reflect the inoperability of HPCI, HPCI was set to failed, and the nonrecovery probability for HPCI was set to 1.0 to reflect the likelihood that operators would not be able to recover HPCI within the allotted recovery time.

The second case examines the likelihood of the occurrence of a dual unit LOOP from grid or weather-related LOOPs. In this case, the LOOP frequency was revised to 2.78×10^{-6} with a short-term nonrecovery probability of 0.66, and offsite power recovery prior to battery depletion (EP.REC) was modified to 0.21 to reflect values for grid and weather-related LOOPs determined from the models described in *Revised LOOP Frequency and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989. Since both units would need their designated EDGs, no credit was given for recovery using the breakers, and the probability of failing to recover power prior to battery depletion was left at 1.0. To reflect the inoperability of HPCI, HPCI was set to failed, and the nonrecovery probability for HPCI was set to 1.0 to reflect the likelihood that operators would not be able to recover HPCI within the allotted recovery time.

The increase in core damage probability (CDP), or importance, over the event duration for the first case is 3.6×10^{-6}. The base-case CDP for the same period is 9.1×10^{-7}, resulting in an estimated conditional core damage probability of 4.5×10^{-6}. The dominant sequence involved a postulated plant-centered LOOP with the failure of emergency power, recovery of offsite power, the failure of HPCI, and the failure of reactor core isolation cooling (RCIC). The increase in core damage probability over the event duration for the second case is 5.1×10^{-6}. The dominant sequence involved a postulated grid/weather-related LOOP with the failure of emergency power and failure to recover offsite power prior to battery depletion.

C.7 LER No. 271/82-019

Event Description: Trip with HPCI Inoperable

Date of Event: August 19, 1982

Plant: Vermont Yankee

Summarized Precursors

Summary

On August 19, 1982 with the plant operating at 100% power, the high-pressure coolant injection (HPCI) governor valve opened once for timing but did not open on system initiation during a monthly HPCI valve and pump operability test. Alternative testing was started. Investigation revealed that the failure of the governor valve to open on system initiation was due to a faulty ramp generator. The ramp generator was replaced and calibrated, and HPCI was returned to service.

A plant trip occurred four days prior to the discovery of the faulty HPCI governor. Assuming that the governor fault was present during the trip, this event was modeled as a transient with HPCI failed. The branch probability for HPCI was set to failed, and the nonrecovery probability for HPCI was set to 1.0 to reflect the fact that operators most likely would not have been able to repair the HPCI within the allowable recovery time. The estimated conditional core damage probability for this event is 6.1×10^{-6}. The dominant sequence involves the observed transient, failure of the power conversion system, two relief valves sticking open, the failure of HPCI, and failure of the automatic depressurization system.

C.8 LER No. 272/82-041

Event Description:	Trip with Two Charging Pumps Inoperable
Date of Event:	June 26, 1982
Plant:	Salem 1

Summary

On June 26, 1982, during a routine inspection, water was discovered in the No. 12 charging pump gear oil reservoir, and the pump was declared inoperable. A leak in the gear oil cooler allowed service water to mix with the gear oil of the No. 12 pump. Charging pump No. 13 was caution tagged and already inoperable at the time of the discovery due to an unidentified noise that occurred during its operation. The No. 11 charging pump was operable during this event. Charging pump No. 13 is a hydro pump and is not modeled in the accident sequence precursor (ASP) models. The ASP models assume that in order for the HPI system and feed and bleed to function properly using the charging pumps, both centrifugal charging pumps (Nos. 11 and 12) are needed.

Since it is not known when the leak in the gear oil cooler occurred, it was assumed that the condition was present during a trip that occurred five days prior to this event. This event was modeled as a transient with train 3 of high-pressure injection (HPI) and feed and bleed failed. The conditional core damage probability estimate is 1.1×10^{-6}. The dominant sequence involved a successful reactor trip, failure of auxiliary feedwater (AFW), failure of main feedwater (MFW), and failure of feed and bleed.

C.9 LER No. 272/82-056 and -053

Event Description: Trip with One AFW MDP and One EDG Inoperable

Date of Event: July 31, 1982

Plant: Salem 1

Summary

On August 2, 1983, with Salem 1 operating at 84% power, an operator discovered that the auxiliary feedwater (AFW) pump No. 11 recirculation valve 11AF40 failed to open as required during the performance of surveillance procedure SP(O) 4.0.5-P. The No. 11 AFW pump was then declared inoperable. Investigation led to the discovery that the low side of the flow transmitter had been left isolated, resulting in the closure of the recirculation valve. An operator had moved the low side valve of the flow transmitter earlier to facilitate maintenance. This valve had not been tagged out and was therefore not repositioned after maintenance was complete. Several days earlier on July 31, a leak in the 1C EDG jacket cooling hose was discovered during routine surveillance. The 1C emergency diesel generator (EDG) was then declared inoperable. The leak in the hose was due to deterioration from age. A few days earlier, on July 28, Salem 1 tripped (NUREG-0020).

Since Salem 1 tripped only a few days prior to the discovery of the fault in the AFW pump recirculation valve and the leak in the water jacket cooling system for the EDG, it was assumed that both faults existed during the trip, and the event was modeled as a transient initiating event with one train of AFW failed and one EDG failed, with the nonrecovery values for emergency power and auxiliary feedwater left as their default values. The conditional core damage probability calculated for this event is 9.8×10^{-6}. The dominant core damage sequence involves a successful trip, failure of AFW, failure of main feedwater, and failure of feed and bleed.

C.10 LER No. 272/82-069

Event Description: Trip with One Charging Pump Inoperable

Date of Event: August 31, 1982

Plant: Salem 1

Summary

During routine operation on August 31, 1982, the primary equipment operator discovered a service water leak on the lubrication oil cooler for the No. 12 centrifugal charging pump and the pump was declared inoperable. The cause of the leak was determined to be erosion of the pipe by silt in the service water. An inoperable charging pump renders one train of both high-pressure injection (HPI) and feed and bleed failed in the accident sequence precursor (ASP) models.

Since it is not known how long the leak existed prior to discovery, it was assumed that it existed during a plant trip that occurred three days earlier and that the leak was serious enough to fault the charging pump. The event was modeled as a transient with one train of HPI and one train of feed and bleed failed. The conditional core damage probabilities (CCDP) was estimated to be 1.1×10^{-6}. The dominant sequence involved a successful reactor trip, failure of auxiliary feedwater (AFW), failure of main feedwater (MFW), and failure of feed and bleed.

C.11 LER No. 277/83-028

Event Description: Trip with Two HPSW Pumps Inoperable

Date of Event: December 23, 1983

Plant: Peach Bottom 2

Summary

During normal operation on December 23, 1983, while surveillance testing was being performed, high-pressure service water (HPSW) pump 2B was declared inoperable due to low flow caused by a stuck-open discharge check valve on the 2D HPSW pump. The HPSW 2A pump was removed from service on July 20, 1983 for an overhaul. The HPSW 2D discharge check valve was inspected and the cause of the check valve failure was determined to be internal wear to the valve disk pin and arm. These parts were replaced and pump 2B was returned to service on December 30th. The HPSW system provides cooling to the residual heat removal (RHR) system heat exchangers. Without HPSW cooling water flow to the RHR heat exchangers, RHR cannot adequately remove decay heat. There was a plant trip on December 25 before the HPSW pump 2B was returned to service (NUREG-0020).

This event was modeled as a transient with degraded RHR and low-pressure coolant injection (LPCI). HPSW pump 2B would continue to work with the failed 2D check valve as long as pump 2D was working, but if pump 2D failed, pump 2B would also fail. The probability that the remaining HPSW pumps would fail was estimated to be 0.01 x 0.1, or 0.001. The conditional core damage probability estimated for this event is 7.7×10^{-6}. The dominant sequence involves a successful reactor shutdown, failure of the power conversion system, failure of one safety relief valve (SRV) to close, successful main feedwater, and failure of RHR.

C.12 LER No. 278/82-004

Event Description: Trip with One Pump of RHR Inoperable

Date of Event: April 10, 1982

Plant: Peach Bottom 3

Summary

During normal operation on April 10 , 1982, while surveillance tests were being performed on the D low-pressure core spray (LPCS) and D residual heat removal (RHR) pumps, the room cooler fans failed to start with the control switches placed in the auto or run position. The fans were placed in continuous service until the auto and run positions could be verified as operable. Investigation revealed that a control circuit had been inadvertently de-energized. The circuit was re-energized and surveillance tests were successfully completed. A plant trip occurred on March 17 (NUREG-0020) a few weeks prior to the discovery of the failure of the room cooler fans to start. The length of time in which the room cooler fans were inoperable is not known. It is assumed in this analysis that the room cooler fans were inoperable at the time of the plant trip. NUREG/CR 4550 Vol. 4, Rev. 1, Part 1 *Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events* assumes that RHR will fail within 10 hours if room cooling fails.

Since NUREG/CR 4550 assumed that RHR would fail within 10 hours given a loss of room cooling, LPCI and LPCS were assumed to be unaffected by the loss or room cooling and one train of RHR and RHR(SPCOOL) was set to failed. It was assumed unlikely that the inadvertent de-energization of the control circuits would have occurred on other LPCS and RHR room cooler fans. This event was modeled as a transient with one train of RHR and RHR(SPCOOL) unavailable, with no increase in the failure probability for other RHR trains. The estimated conditional core damage probability for this event is 3.3×10^{-6}. The dominant sequence involved a successful reactor shutdown, failure of the power conversion system, successful main feedwater, and failure of RHR.

C.13 LER No. 281/83-005

Event Description: Trip with AFW Pump Inoperable

Date of Event: February 11, 1983

Plant: Surry 2

Summary

Surry Unit 2 was operating at full power on February 11, 1983 when the turbine-driven auxiliary feedwater (AFW) pump tripped during testing. The turbine governor was found to be defective, causing the turbine to trip on overspeed. Corrosion in the regulator piston in the pump governor valve prevented the piston from moving freely, which led to a pump overspeed trip.

Since a trip was reported on February 8, 1983, this event was modeled as a trip with the turbine-driven AFW pump assumed to be inoperable. The conditional core damage probability estimated for this event is 3.8×10^{-6}. The dominant sequence involves a transient with reactor trip success, failure of main and auxiliary feedwater, and failure of feed-and-bleed cooling.

C.14 LER No. 282/82-015

Event Description: Two EDGs Simultaneously Inoperable for 1.5 Hours

Date of Event: August 27, 1982

Plant: Prairie Island 1

Summarized Precursors

Summary

On August 27, 1982, during normal operation, emergency diesel generator (EDG) D1 was out of service for preventive maintenance when an operability test was done on EDG D2. A procedural step was missed, which resulted in the auto/manual switch for the D2 supply to bus 16 being left in the manual position. This would have prevented EDG D2 from automatically closing onto bus 16 in the event that offsite power was lost to the bus. The operator error was discovered and corrected within one hour and 25 minutes. During this time, both EDGs would have been unavailable given the loss of offsite power.

This event was modeled as an unavailability of both trains of emergency power given a postulated loss of offsite power (LOOP). The nonrecovery factor for emergency power was modified to 0.55 to reflect the ability of the operators to recover EDG D2 locally (see Appendix A). The increase in core damage probability, or importance, over the duration of the event is 2.3×10^{-6}. The base-case CDP over the duration of the event is 2.1×10^{-8}, resulting in an estimated conditional core damage probability of 2.3×10^{-6}. The dominant sequence involved a postulated LOOP with the failure of emergency power (station blackout) and failure to recover offsite power prior to battery depletion.

C.15 LER No. 285/82-009

Event Description:	Three of Four CCW Heat Exchangers Inoperable
Date of Event:	April 11, 1982
Plant:	Fort Calhoun

Summary

During normal power operation on April 11, 1982, while the component cooling water (CCW) heat exchangers were being exchanged, three of the four outlet valves (HCV-490B, HCV-491B, and HCV-492B) failed to open. (HCV-491B did open partially.) Only one CCW heat exchanger was operational. Within minutes, an operator was dispatched and arrived at the three subject valves. He manually tapped on the actuator parts of all three valves. Valves HCV-490B and HCV-492B opened after they were tapped. Again, HCV-491B opened only slightly. HCV-491B was disassembled. No apparent problem could be determined so the valve packing was loosened and the valve was tested for operation. All valves were cycled several times and operated successfully. The CCW system at Fort Calhoun consists of three CCW pumps and four CCW heat exchangers. One pump and at least two heat exchangers are normally operating. Pumps are rotated once a week. The CCW system provides cooling to the high-pressure safety injection (HPSI) pumps, the low-pressure safety injection (LPSI) pumps, the shutdown cooling heat exchangers, the control air conditioners, the containment air cooling coils, and the containment spray system pumps. According to the Fort Calhoun individual plant examination document, three of four CCW heat exchangers and one CCW pump are sufficient to provide cooling to these systems for all plant modes, and CCW cooling is only needed during recirculation modes of residual heat removal (RHR) and high-pressure recirculation system (HPR). It was assumed in this analysis that three of the four CCW heat exchangers are needed for decay heat removal. This may be conservative for colder weather.

The Licensee event report (LER) states that the CCW heat exchanger valves are cycled two to three times a week. Thus, this event was modeled as an unavailability of three of the four CCW heat exchangers for a period of three days (72 hours). To reflect the loss of CCW cooling to the residual heat removal system, containment spray recirculation (CSR), and the high pressure recirculation system, all trains for each system were set to failed and the corresponding nonrecovery probabilities for each system were set to 1.0. To determine an estimate of the conditional core damage probability that reflects the ability of the operators to locally recover the CCW valves and therefore recover the failed systems, the estimated conditional core damage probability for the failed systems was multiplied by the nonrecovery probability for CCW. The CCW nonrecovery probability was assumed to be 0.054, based on recovery times for service water-related failures included in "Faulted Systems Recovery Experience," Nuclear Safety Analysis Center (NSAC)-161, May 1992. The nominal conditional core damage probability given CCW is recovered and all recirculation systems function normally would not contribute significantly to the total estimated conditional core damage probability compared with that of the CCW not being recovered and was therefore not subtracted from the estimate. The increase in core damage probability (CDP), or importance, over the duration of the event is 5.7×10^{-6}. The base-case CDP over the duration of the event is 3.3×10^{-7}, resulting in an estimated conditional core damage probability of 6.0×10^{-6}. The dominant sequence involves a postulated small loss-of-coolant accident (LOCA) with the failure of RHR and HPR.

C.16 LER 293/82-043 and -042

Event Description: RCIC and HPCI Suction Valves Inoperable

Date of Event: September 30, 1982

Plant: Pilgrim

Summary

On September 30, 1982 at 0600 hours, position indication was lost for reactor core isolation cooling (RCIC) torus suction valve 1301-25 during a surveillance test. Prior to this, on September 29, high-pressure coolant injection (HPCI) had been declared inoperable, owing to the failure of HPCI torus suction valve 2301-35 during a timing surveillance (licensee event report (LER) 293/82-42).

The increase in core damage probability (CDP), or importance, over the duration of the event is 5.8×10^{-6}. The base-case CDP over the duration of the event is 1.8×10^{-6}, resulting in an estimated conditional core damage probability of 7.6×10^{-6}. The dominant sequence involves a postulated loss of offsite power with successful reactor shutdown, failure of the emergency power system, successful restoration of offsite power before battery depletion, no more than one safety relief valve failing to close, failure of HPCI, and failure of RCIC.

C.17 LER No. 293/83-039

Event Description: Trip with HPCI Inoperable

Date of Event: July 2, 1983

Plant: Pilgrim

Summarized Precursors

Summary

During reactor startup surveillance tests on July 2, 1983, the high-pressure coolant injection (HPCI) system was declared inoperable when the HPCI turbine stop valve required more than the normal amount of time to open. A manual scram occurred six days earlier.

Assuming HPCI was inoperable during the scram, the conditional core damage probability estimated for this event is 5.2×10^{-6}. The dominant sequence involved a transient initiator followed by successful reactor shutdown, failure of the power conversion system, failure of two safety relief valves (SRVs) to close, unavailability of HPCI, and failure of the automatic depressurization system (ADS).

C.18 LER No. 293/83-052

Event Description: Trip with HPCI Inoperable

Date of Event: September 23, 1983

Plant: Pilgrim

Summary

While monthly surveillance of the high-pressure coolant injection (HPCI) system was being performed during steady-state operation on September 23, 1983, the HPCI was declared inoperable when the HPCI motor-operated valve 2302-3 failed to open. Fourteen days earlier, on September 9, 1983, a scram occurred during surveillance testing.

Assuming HPCI was failed at the time of the scram, a conditional core damage probability of 5.2×10^{-6} is estimated. The dominant sequence involved a transient initiator followed by successful reactor shutdown, failure of the power conversion system, failure of two safety relief valves (SRVs) to close, unavailability of HPCI, and failure of the automatic depressurization system (ADS).

C.19 LER No. 295/82-025

Event Description: Postulated Grid/Weather-Related LOOP with Two EDGs Inoperable

Date of Event: August 11, 1982

Plant: Zion 1

Summary

During normal operation on August 11, 1982, while the 0 emergency diesel generator (EDG) was being tested as a daily requirement for EDG 1A being out of service, a small fire was observed near the turbocharger on the 0 EDG. The diesel was shut down and declared inoperable. A unit shutdown commenced due to the two out of three EDGs being inoperable. The 1A EDG was made operable prior to reaching hot shutdown and the load decrease was terminated. Investigation revealed that the mounting screw for the turbocharger lubrication oil filter canister vibrated loose, allowing oil to spray onto the hot exhaust manifold through an O-ring seal, which caused the lubrication oil to flash. The O-ring was replaced and the canister was retightened. Zion 1 has three emergency diesel generators, two of which are specifically dedicated to Zion 1. Diesel generator 1A feeds 4-kV bus 148, and diesel generator 2B feeds 4-kV bus 149. One diesel generator (diesel generator 0) is connected to both Zion 1 bus 147 and Zion 2 bus 247. The buses are electrically interlocked to prevent the operation of both buses at the same time. Diesel generator 1A bus 148 supplies auxiliary power to auxiliary feedwater pump 1B, residual heat removal (RHR) pump B, and safety injection pump B. Diesel generator 0 bus 147 supplies auxiliary power to safety injection pump A and charging pump B. In addition to the diesel generators, power from the Unit 2 station auxiliary transformer (SAT) can be manually aligned to supply power to Unit 1. Since auxiliary power can be supplied from Unit 2, plant-centered loss of offsite power (LOOPs) would not be of particular importance in this event. LOOPs that affected both units (i.e., Unit 2 could not provide auxiliary power to Unit 1 given both EDGs were inoperable), such as grid-related and weather-related LOOPs, would be of importance given both dedicated EDGs were inoperable.

This event was modeled as a postulated grid-related/weather-related LOOP with two EDGs inoperable. The LOOP frequency, the offsite power recovery probabilities, and the probability of seal loss-of-coolant-accident (LOCA) were modified as shown in Table C.18.1 to reflect those values associated with grid-related and weather-related LOOPs (see ORNL/NRC/LTR 89/11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989). The first train of emergency power was set to failed to reflect the failed EDG since it was assumed that the fault discovered in EDG 0 could also have occurred in the other EDGs. The second train of emergency power was set to failed to reflect the fact that EDG 1A was out of service for repairs. The corresponding system trains that rely on these diesels for power given the loss of offsite power were also modified to reflect their unavailability. Since the test done on EDG 0 that resulted in the fire was performed on a daily basis while EDG 1A was out for service, the length of time in which both faults were present was assumed to be 24 hours. The increase in core damage probability (CDP), or importance, over the duration of the event is 3.8×10^{-6}. The base-case CDP over the duration of the event is 1.2×10^{-7}, resulting in an estimated conditional core damage probability of 3.9×10^{-6}. The dominant sequence involved a postulated LOOP with emergency power failure (station blackout), a reactor coolant pump (RCP) seal LOCA, and failure to recover offsite power before core uncovery.

Table C.18.1 Revised LOOP Probabilities

Event	Default Probability	Revised Probability
LOOP frequency	1.6×10^{-5}	2.8×10^{-6}
LOOP short-term nonrecovery	0.53	0.66
Seal LOCA probability	0.27	0.42
Offsite power recovery prior to battery depletion given no seal LOCA	0.031	0.14

Summarized Precursors

Event	Default Probability	Revised Probability
Offsite power recovery prior to battery depletion given seal LOCA	0.57	0.77
Offsite power recovery within twp hours (OFFSITE.PWR.REC/-EP.AND.-AFW)	0.22	0.52
Offsite power recovery within six hours (OFFSITE.PWR.REC/-EP.AND.AFW)	0.067	0.32

C.20 LER No. 295/82-033

Event Description: Postulated Grid-Related/Weather-Related LOOP with Two EDGs Inoperable

Date of Event: October 15, 1982

Plant: Zion 1

Summary

During normal operation on October 15, 1982 while the 1B EDG was being tested as a requirement for emergency diesel generator (EDG) 1A being out of service, the EDG tripped on low turbo lubrication oil pressure. The 0 EDG was operable. A unit shutdown commenced due to the two out of three EDGs being inoperable. Investigation revealed that the low pressure was due to clogged filters. The oil and filters were changed and 1B EDG was returned to service. EDG 1B would have started on a safety injection (SI) signal but would have been expected to fail due to a lack of turbocharger lubrication. Zion 1 has three emergency diesel generators, each rated at 4,000 kW and cooled by service water. Two diesel generators are specifically dedicated to Zion 1. Diesel generator 1A feeds 4-kV bus 148, and diesel generator 2B feeds 4-kV bus 149. One diesel generator (diesel generator 0) is connected to both Zion 1 bus 147 and Zion 2 bus 247. The buses are electrically interlocked to prevent the operation of both buses at the same time. Diesel generator 1A bus 148 supplies auxiliary power to auxiliary feedwater pump 1B, residual heat removal (RHR) pump B, and safety injection pump B. Diesel generator 1B bus 149 supplies auxiliary power to auxiliary feedwater pump 1C, RHR pump A and charging pump A. In addition to the diesel generators, power from the Unit 2 station auxiliary transformer (SAT) can be manually aligned to supply power to Unit 1.

Since auxiliary power can be supplied from Unit 2, plant-centered loss of offsite power (LOOPs) would not be of particular importance in this event. LOOPs that affected both units (i.e., Unit 2 could not provide auxiliary power to Unit 1 given both EDGs were inoperable) such as grid-related and weather-related LOOPs would be of importance given that both dedicated EDGs were inoperable. This event was modeled as a postulated grid-related/weather-related LOOP with two EDGs inoperable. The LOOP frequency, the offsite power recovery probabilities, and the probability of seal loss-of-coolant accident (LOCA) were modified as shown in Table C.19.1 to reflect the values associated with grid-related and weather-related LOOPs (see ORNL/NRC/LTR 89/11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989). The first train of emergency power was set to failed to reflect the failed EDG since it was assumed that the lack of lubrication found in EDG 1B could also have occurred in the other EDGs. The third train of emergency power was set to unavailable to reflect the unavailability of EDG 1A due to maintenance. The corresponding system trains that rely on these diesels for power given the loss of offsite power were also modified to reflect their unavailability. Since the test done on EDG 1B that resulted in the EDG trip was performed daily while EDG 1A was out of service, the length of time in which both faults were present was assumed to be 24 hours. The increase in core damage probability (CDP), or importance, over the duration of the event is 1.4×10^{-6}. The base-case CDP over the duration of the event is 1.2×10^{-7}, resulting in an estimated conditional core damage probability of 1.5×10^{-6}. The dominant sequence involved a postulated LOOP with emergency power failure (station blackout), a reactor coolant pump (RCP) seal LOCA, and failure to recover offsite power before core uncovery.

Table C.19.1 Revised LOOP Probabilities

Event	Default Probability	Revised Probability
LOOP frequency	1.6×10^{-5}	2.8×10^{-6}
LOOP short-term nonrecovery	0.53	0.66
Seal LOCA probability	0.27	0.42
Offsite power recovery prior to battery depletion given no seal LOCA	0.031	0.14
Offsite power recovery prior to battery depletion given seal LOCA	0.57	0.77
Offsite power recovery within two hours (OFFSITE.PWR.REC/-EP.AND.-AFW)	0.22	0.52
Offsite power recovery within six hours (OFFSITE.PWR.REC/-EP.AND.AFW)	0.067	0.32

C.21 LER No. 298/83-014

Event Description: Reactor Trip with High-Pressure Coolant Injection Unavailable

Date of Event: September 15, 1983

Plant: Cooper

Summary

On September 15, 1983, the high-pressure coolant injection (HPCI) system was declared inoperable due to pressurization of the suppression chamber air space while the HPCI turbine was undergoing surveillance testing. The cause of the event was traced to a vacuum breaker jammed in the open position, allowing the HPCI turbine exhaust to enter the air space of the suppression chamber. The plant was shut down, interim repairs were made to the vacuum breaker, and the HPCI system was restored to service. It was found that the vacuum breaker was damaged because it had been installed in the wrong size of pipe. Other core standby cooling systems and the reactor core isolation cooling (RCIC) system were operable at the time of the HPCI failure.

This event was modeled as a reactor trip with the HPCI system assumed to be unavailable. This approach was based on the fact that a scram occurred three days earlier. The conditional core damage probability estimated for this event is 6.2×10^{-6}. The dominant core damage sequence involves the observed transient, successful reactor trip, failure of the power conversion system, two safety relief valves failing to close, unavailability of HPCI, and automatic depressurization system (ADS) failure.

C.22 LER No. 302/82-041, 302/82-051, and 302/83-037

Event Description: Transient with One RHR Pump Flow Control Valve Inoperable

Date of Event: June 8, 1982

Plant: Crystal River 3

Summary

On June 8, 1982 while the operability of the boron injection system was being verified, discharge throttle valve DHV-111 of the decay heat removal (DHR) system failed to control flow automatically, as required in Technical Specification 3.5.2. DHR train B was declared inoperable. Maintenance was initiated, and DHR train B was restored later on June 8. A plant trip occurred nine days earlier, on May 30. The valve also malfunctioned on June 22, 1982. Maintenance was performed on the valve and the train was restored to service on June 23. A stuck high-flow switch was determined to be the cause for the malfunctioning valve. A plant trip also occurred six days prior to the valve failure (NUREG-0020). A similar event involving DHR train A occurred on July 28, 1982. While the borated water storage tank (BWST) was being recirculated with DHR pump 1A on July 28, the pump discharge throttle valve DHV-110 on DHR train A failed to operate

correctly and was declared inoperable. Maintenance was initiated and the train was restored to operability later that day. The cause of the improper operation of DHV-110 was determined to be air in the flow indicating switch sensing lines that regulate the valve. A plant trip occurred on July 15, 13 days prior to the discovery of the valve malfunction (NUREG-0020). On September 7, 1983, another similar event occurred. While surveillance tests on the operability of the emergency core cooling system pumps were being performed, the breaker for valve DHV-110 tripped while cycling the valve from open to closed. The cause of the breaker trip was unknown. The breaker was reset and the valve cycled successfully. A plant trip occurred 11 days earlier on August 26 (NUREG-0020).

Although these events were separate, they were analyzed as one event since all involved a transient with one DHR pump flow control valve inoperable. It was assumed in this analysis that the throttle valve failed to control flow in such a way that there was insufficient flow from the DHR pump train. The first train of the residual heat removal (RHR) was set to failed to reflect the failure of the pump train due caused by failed flow control. Since high-pressure recirculation (HPR) uses the RHR pumps, the first train of HPR/RHR.AND.HPR were set to failed as well. The estimated conditional core damage probability for these events is 4.8×10^{-6}. The dominant sequence was a postulated anticipated transient without scram (ATWS) sequence involving the failure to trip and the failure of auxiliary feedwater and did not involve any modified branches.

C.23 LER No. 302/83-056 and -057

Event Description: Trip with AFW Turbine-Driven Pump Inoperable

Date of Event: November 22, 1983

Plant: Crystal River 3

Summary

On November 22, 1983, while a routine surveillance procedure [SP-349B, emergency feedwater (EFW) system operability demonstration] was being performed, the governor on the turbine-driven EFW pump was found to be set incorrectly. The control knob was adjusted and the surveillance test proceeded. Later that day, the motor actuator for the turbine-driven pump's steam admission valve was discovered to be inoperable. Investigation revealed that the failure of the motor actuator was due to a contact failure. The actuator was repaired, tested, and returned to service. A plant trip occurred ten days earlier on November 12 (NUREG-0020) because of a loss of feedwater control, which resulted in a scram on high reactor coolant system (RCS) pressure (NUREG-0020). This could have resulted in the loss of main feedwater. If main feedwater had been lost, the EFW pump would have been demanded. If it had been demanded, it would have had to function in such a manner that the misadjusted valve went unnoticed. If main feedwater control was lost during the trip but flow remained sufficient for steam generator cooling, the EFW pump would not have been demanded and the misadjusted valve would not have been detected.

This analysis assumed that the degraded condition on the EFW turbine-driven pump would have prevented the pump from working at the time of the November 12 trip and that main feedwater was not completely lost during the trip. This event was modeled as a transient with the turbine-driven EFW pump inoperable. The third train of the auxiliary feedwater (AFW) in the model was set to failed to reflect the failure of the turbine-driven EFW pump. The estimated conditional core damage probability for this event is 9.5×10^{-6}. The dominant sequence involves a successful reactor trip, failure of EFW, failure of main feedwater, and failure of the operator to initiate feed and bleed.

C.24 LER No. 311/82-126

Event Description: Trip with One Charging Pump Inoperable

Date of Event: October 19, 1982

Plant: Salem 2

Summary

On October 19, 1982 during a routine operation, analysis of a sample of the No. 21 charging pump lubrication oil revealed that water was mixed with oil in the gear oil reservoir, and the No. 21 pump was declared inoperable. A leak in the gear oil cooler from erosion and corrosion of the cooler tubes was allowing service water to mix with the gear oil. The No. 22 charging pump was operable during this event. The accident sequence precursor (ASP) models assume that in order for the high-pressure injection (HPI) system and feed and bleed to function properly using the charging pumps, both pumps (Nos. 21 and 22) are needed. Since it is not known when the leak in the gear oil cooler occurred, it was assumed that the condition was present during a trip that occurred eight days prior to this event (NUREG-0020). The trip was an automatic scram due to control rod drive (CRD) problems. Main feedwater was not affected. Thus, this event was modeled as a transient with train 3 of HPI and feed and bleed failed. The conditional core damage probability estimated is 1.1×10^{-6}. The dominant sequence involved a successful reactor trip, failure of auxiliary feedwater (AFW), failure of main feedwater (MFW), and failure of feed and bleed.

C.25 LER No. 311/83-041

Event Description: Trip with Number 2A Vital Bus De-energized

Date of Event: August 1, 1983

Plant: Salem 2

Summary

On August 1, 1983 during routine power operation, a low-component cooling flow alarm was received in the control room. Upon entering the control room, the shift supervisor observed No. 2A vital bus infeed breaker 22ASD trip without an automatic transfer, thus de-energizing the bus. Shortly after the bus de-energized, the reactor tripped on a power range neutron flux high negative rate signal. The bus was declared inoperable, Technical Specification Action Statement 3.8.2.1 was entered, and the plant was placed in a stable shutdown condition. The bus was re-energized within the time allowed in the Technical Specification action statement. Investigation revealed that the alternative 24-Vdc power supply for rod control cabinet 2SCD had failed prior to the occurrence, and a spurious channel A actuation of the safeguards equipment control (SEC) system caused the loss of the 2A bus, which resulted in an attempt of the rod control cabinet to transfer to the failed alternative power supply. The rod control cabinet power supply failure led to the dropping of rod banks C and D, which resulted in the negative flux rate and thus the reactor trip. The failed alternative power supply was replaced and the unit was returned to power operation on August 2. On August 9, another spurious SEC channel A actuation occurred and the No. 2A vital bus was de-energized once again. The reactor apparently did not trip following vital bus de-energization. The bus was again declared inoperable, and was again re-energized within the time specified in the Technical Specification action statement. Further investigation revealed that monitoring equipment connected to the SEC channel 2A circuit at several locations led to near-short conditions on the terminals of the output test panel and was possibly the cause of the spurious SEC signals.

The August 1, 1983 trip was modeled as a transient with the No. 2A vital bus failed. The licensee event report (LER) states that when the No. 2A vital bus was de-energized, the No. 21 boric acid transfer pump, the No. 21 component cooling water pump, the No. 21 containment fan coil unit, the No. 21 fuel handling exhaust fan, the No. 22 service water pump, and the No. 21 shield ventilation fan were also de-energized. The service water pump is one of six and any two pumps can fully supply all service water needs. The component cooling water pump provides seal cooling to SI pump 11, charging pump 12, and residual heat removal (RHR) pump 11. Since pump cooling is not needed for injection but is most likely needed in the recirculation modes, this event was modeled as a transient with one train of RHR, high-pressure recirculation (HPR) and RHR.AND.HPR failed. The estimated conditional core damage probability for this event is 1.2×10^{-6}. The dominant sequence involved a successful reactor trip, failure of auxiliary feedwater (AFW), failure of main feedwater (MFW), and failure of feed and bleed. The dominant sequence did not include any modified events.

C.26 LER No. 313/83-014

Event Description: Transient with Loss of Feedwater and One AFW Pump Inoperable

Date of Event: June 9, 1983

Plant: ANO 1

Summary

On June 9, 1983 following a reactor trip caused by the trip of both main feedwater (MFW) pumps on a spurious low-suction pressure signal, auxiliary feedwater (AFW) pump P-75 became inoperable due to a break in the seal supply piping, which resulted in a seal failure. Both emergency feedwater (EFW) pumps were operable and available to feed the steam generators. The cause of the piping failure was attributed to misalignment of the seal supply piping. Secondary pressure was lowered until the steam generators could be fed with the condensate pumps. The broken portion of the seal supply piping was replaced, and the piping was realigned to reduce stresses. The pump was tested satisfactorily and returned to service.

ANO 1 has one AFW motor-driven pump that is used to provide cooling to two steam generators during startup and shutdown. ANO 1 also has an emergency feedwater system that can be used to provide cooling to the steam generators during normal operation in the event that the MFW is unavailable. The EFW system consists of two trains that can feed either or both of the steam generators. One train has a motor-driven pump, and the other has a turbine-driven pump. One pump train supplying flow to one steam generator is sufficient for secondary-side cooling.

This event was modeled as a transient with main feedwater (MFW) failed due to the spurious low-suction pressure signal trip, and the motor-driven auxiliary feedwater (AFW) train failed. The accident sequence precursor (ASP) models incorporate the single AFW pump and the two EFW pumps in the AFW branch of the model. The second train of AFW in the model was set to failed to reflect the failure of the AFW pump and the assumption that the observed failure was most likely not common cause. The MFW train was set to failed to reflect the loss of main feedwater that initiated the plant trip. The estimated conditional core damage probability for this event is 4.7×10^{-6}. The dominant sequence highlighted in the event tree in Figure B.26.1 involved a successful reactor shutdown, failure of AFW, failure of MFW, and failure of feed and bleed.

C.27 LER No. 313/83-015

Event Description: Transient with One HPI Injection Valve Failed Closed

Date of Event: June 16, 1983

Plant: ANO 1

Summary

On June 16, 1983 while in hot shutdown following a main turbine generator exciter failure reactor trip, high-pressure injection (HPI) system control valve CV-1219 failed to open on demand from the control room. The valve was being opened to allow additional makeup flow following the reactor trip. Redundant HPI valves CV-1220. CV-1227, and CV-1228 were available and operable. A redundant valve was immediately used and CV-1219 was opened manually. The root cause could not be determined, and subsequent valve testing could not duplicate the failure. Contacts on the torque switch were found to be slightly corroded, but investigation of the circuit revealed that this should not have prevented CV-1219 from opening. The HPI system at ANO 1 has three pumps. Two pumps provide coolant to the reactor coolant cold legs through four injection lines. The third pump can be used in the event that one of the normally used pumps is unavailable. When valve CV-1219 failed to open, only one injection line was inoperable. All pumps were still operable and three injection lines remained.

This event was modeled as a transient with one HPI injection valve inoperable. The HPI model consists of three trains that are assumed to be dominated by pump failures. Thus, the HPI model does not directly address the failure of the injection valves. To address the failure of one of the four injection valves, the branch probability for HPI was modified by adding the probability of failure of the three remaining injection valves given one injection valve failed, i.e.,

$$\begin{aligned} p(\text{HPI})_{new} = {} & p(\text{HPI})_{old} + p(\text{second inj. valve fails} \mid \text{one inj.valve failed}) \\ & + p(\text{third inj. valve fails} \mid \text{two inj. valves failed}) \\ & + p(\text{fourth inj. valve fails} \mid \text{three inj. valves failed}). \end{aligned}$$

The conditional failure probabilities for the injection valves are shown in Table C.27.1. Since feed and bleed (FEED.BLEED) utilizes the HPI pumps and injection valves, the FEED.BLEED branch probability was modified in the same manner as the HPI branch probability. The estimated conditional core damage probability for this event is 2.9 x 10^{-6}. The dominant sequence was an anticipated transient without scram (ATWS) sequence which involved the failure to trip and the failure of auxiliary feedwater (AFW) given ATWS. The second highest contributing sequence involved a successful reactor trip, failure of the auxiliary feedwater system, failure of main feedwater, and failure of feed and bleed.

Table C.27.1 Conditional Failure Probabilities for HPI Injection Valves

Event	Failure Probability
Second inj. valve fails given the first inj. valve failed	0.1
Third inj. valve fails given two inj. valves failed	0.3
Fourth inj. valve fails given three inj. valves failed	0.5

C.28 LER No. 316/82-011

Event Description: ESW Header and ECCS Train A Inoperable

Date of Event: January 28, 1982

Plant: Cook 2

Summary

On January 28, 1982, while Cook was operating at 100% power, a leak was discovered in the emergency service water (ESW) system piping, downstream of the outlet valve from the east component cooling water (CCW) heat exchanger. In order to effect repairs, the ESW piping was isolated, rendering equipment supplied by that train of ESW inoperable. In addition, emergency core cooling system (ECCS) train A was declared inoperable. At the same time, the B train of high-pressure injection (HPI)/high-pressure recirculation (HPR) was rendered inoperable by an obstruction in the system piping. This failure is discussed in licensee event report 316/82-113 and the associated analysis.

This event was modeled as an unavailability of ECCS train A and other systems dependent on the affected train of ESW. Since the duration of the unavailability was not given, a 24-hour duration was assumed. Systems assumed to be affected included high-pressure recirculation, residual heat removal, and the A train emergency diesel generator. The B train of HPI/HPR was also assumed to be inoperable. Two calculations were performed: one for the case of a potential loss of offsite power (LOOP), and another for all other initiators. For the LOOP case, the equipment associated with the inoperable A emergency diesel generator (EDG) was considered inoperable, including the A motor-driven AFW pump, HPI/HPR train A, and RHR train A. The B train of HPI/HPR was assumed to be inoperable also. The charging pumps at Cook can provide a redundant source of high-pressure injection but, since flow from two pumps may be required to equal the flow of an HPI pump, this redundant source was assumed to be unavailable because the A charging pump would be de-energized. For all other initiators, train B of HPI/HPR was assumed to be unavailable, along with train A of HPR and RHR.

The increase in core damage probability (CDP), or importance, over the duration of the event is 1.5×10^{-6}. The base-case CDP over the duration of the event is 3.5×10^{-7}, resulting in an estimated conditional core damage probability of 1.9×10^{-6}. The dominant core damage sequence involves a postulated loss of offsite power (LOOP), failure of emergency ac power, seal LOCA, and failure to recover ac power before core uncovery.

C.29 LER No. 316/82-072

Event Description: Control Room Instrument Distribution Bus IV Fails, Trip

Date of Event: August 24, 1982

Plant: Cook 2

Summary

During normal operation at 100% power, Cook Unit 2 suddenly tripped when a component failure resulted in the loss of the control room instrument distribution (CRID)-IV 120-V ac vital bus. In addition to causing a reactor trip, the loss of the CRID-IV bus resulted in loss of power in the control room to several instrument and control circuits. Power was also lost to the solid state protection system (SSPS) channel B slave relays.

The four CRID trains provide power to channels in the reactor protection system (RPS), the solid-state protection system, and various instrumentation panels. Licensee event report (LER) 316/82-072 does not specifically identify the affected system; however, such a listing may be found in LER 316/89-014, which reports a similar failure of CRID-IV. These systems include the protection system status lights, No. 24 reactor coolant pump (RCP) operating parameter indication, steam generator wide-range level indication, loop 4 indication of auxiliary feedwater flow, two main steam pressure indicators, one channel of steam generator narrow-range level indication, and the steam dump control system.

The RPS is designed to fail safe on loss of CRID power, so loss of a CRID train will not prevent a trip. Redundant indications not dependent on CRID-IV exist to ensure that operators can monitor and control all necessary safety functions. The SSPS is designed so that two of four channels are generally sufficient to initiate a trip, so loss of certain CRID trains will not render either SSPS train inoperable. However, CRID-IV also provides power to the SSPS train B slave relays. Concurrent with the failures described in this event, the B train of HPI/HPR was rendered inoperable by an obstruction in the system piping. This failure is discussed in LER 316/82-113 and the associated analysis.

This event was modeled as a transient with unavailability of auto-initiation of HPI by SSPS. Train B of HPI/HPR was also assumed unavailable due to the failure described in LER 316/82-113. The conditional core damage probability estimated for this event is 1.3×10^{-6}. The dominant core damage sequence involves the observed transient, and failure of auxiliary feedwater, main feedwater, and feed and bleed.

C.30 LER No. 316/83-052

Event Description: Control Room Instrument Distribution Bus IV Fails, Trip

Date of Event: June 23, 1983

Plant: Cook 2

Summary

During normal operation at 100% power, Cook Unit 2 suddenly tripped when a component failure resulted in the loss of the control room instrument distribution (CRID)-IV 120-V ac vital bus. In addition to causing a reactor trip, the loss of the CRID-IV bus resulted in loss of power in the control room to several instrument and control circuits. Power was also lost to the solid-state protection system (SSPS) channel B slave relays.

The four CRID trains provide power to channels in the reactor protection system (RPS), the solid-state protection system, and various instrumentation panels. Licensee event report (LER) 316/82-072 does not specifically identify the affected system; however, such a listing may be found in LER 316/89-0014, which reports a similar failure of CRID-IV. These systems include the protection system status lights, No. 24 reactor coolant pump (RCP) operating parameter indication, steam generator wide-range level indication, loop 4 indication of auxiliary feedwater flow, two main steam pressure indicators, one channel of steam generator narrow-range level indication, and the steam dump control system.

The RPS is designed to fail safe on loss of CRID power, so loss of a CRID train will not prevent a trip. Redundant indications not dependent on CRID-IV exist to ensure that operators can monitor and control all necessary safety functions. The SSPS is designed so that two of four channels are generally sufficient to initiate a trip, so loss of certain CRID trains will not render either SSPS train inoperable. However, CRID-IV also provides power to the SSPS train B slave relays.

This event was modeled as a transient with (because of the unavailability of power to the SSPS train B slave relays) failure of auto-initiation of one train of HPI. The conditional core damage probability estimated for this event is 1.0×10^{-6}. The dominant core damage sequence involves the observed transient and failure of auxiliary feedwater (AFW), main feedwater (MFW), and feed and bleed.

Summarized Precursors

C.31 LER No. 317/82-054

Event Description: Transient with One Turbine-Driven AFW Pump Inoperable

Date of Event: August 31, 1982

Plant: Calvert Cliffs 1

Summary

On August 31, 1982, while a surveillance test was being performed, the governor linkage of auxiliary feedwater (AFW) pump No. 12 vibrated loose, rendering the pump inoperable. A loose nut connecting the governor lever to the connecting rod backed off its pin, causing the pin to come out. The linkage was repaired and returned to service approximately 45 minutes later. Other AFW pumps were checked for loose governor linkage nuts, but all were tight.

A plant trip occurred on August 22, 1982, nine days prior to the discovery of the faulted AFW pump. This event was modeled as a transient with one AFW pump assumed to be inoperable. The estimated conditional core damage probability for this event is 2.9×10^{-6}. The dominant sequence was an ATWS sequence involving the failure of AFW given ATWS. The second highest contributor involved a successful reactor trip, failure of AFW, failure of main feedwater, and failure of feed and bleed.

C.32 LER No. 317/83-046 and -049

Event Description: One EDG and One AFW Turbine-Driven Pump Inoperable

Date of Event: August 16, 1983

Plant: Calvert Cliffs 1

Summary

On August 16, 1983 during normal operation, emergency diesel generator (EDG) 12 shut down on a loss of fuel oil during the performance of a surveillance test. It was determined that this condition existed from August 10-16, 1983. The cause of the fault was a personnel error which occurred on August 10. Procedures were not followed, which resulted in the lower level switch isolation valve being left shut after testing. On August 18, 1983 while postmaintenance testing was being carried out on the flow path from the condensate storage tank, auxiliary feedwater (AFW) pump 11 was found to be inoperable due to bound-up stage piece and shaft piece balances.

Since it is not known how long the AFW pump was inoperable, and the EDG was determined to have been faulty for six days, this event was modeled as an unavailability of the turbine-driven AFW pump 11 and EDG 12 with a postulated loss of offsite power (LOOP) for six days (144 hours). All associated system trains that rely on EDG 12 for emergency power were set to unavailable. Since the motor-operated AFW pump does not rely on EDG 12 but relies on EDG 11, it was assumed to be operable. The increase in core damage probability (CDP), or importance, over the duration of the event is 5.8×10^{-6}. The base-case CDP over the duration of the event is 7.1×10^{-7}, resulting in an estimated conditional core damage probability of 6.5×10^{-6}. The dominant sequence involved a postulated LOOP with failure of emergency power (station blackout) and failure of AFW given the loss of emergency power.

C.33 LER No. 317/83-076

Event Description: Transient with the Motor-Driven AFW Pump Inoperable

Date of Event: December 30, 1983

Plant: Calvert Cliffs 1

Summary

On December 30, 1983, the No. 13 auxiliary feedwater (AFW) pump was removed from service to repair an oil leak on the outboard pump bearing. When an attempt was made to drain the oil from the bearings, the bearing drain was found plugged with metal filings. This indicated that bearing damage had occurred. The damage was attributed to improper lubrication caused by a portion of an O-ring left by the manufacturer in the oil return passage in the bearing housing. The O-ring was removed and the bearings were replaced. The pump was returned to service approximately 11 days after it was initially removed from service. A plant trip occurred on December 28, 1983, two days prior to the discovery of the faulty AFW pump.

This event was modeled as a transient with the motor-driven train of AFW assumed to be failed. The estimated conditional core damage probability is 7.7×10^{-6}. The dominant sequence involved a successful reactor trip, failure of AFW, failure of main feedwater (MFW), and failure of feed and bleed.

C.34 LER No. 318/83-061

Event Description: Transient with One LPSI Pump Inoperable

Date of Event: November 7, 1983

Plant: Calvert Cliffs 2

Summary

On November 7, 1983, during monthly surveillance testing of the engineered safety feature actuation system (ESFAS) logic, the No. 22 low-pressure safety injection (LPSI) pump could not be restarted after being stopped on a recirculation actuation signal (RAS). The trip mechanism on the pump breaker was out of adjustment, causing the breaker to trip free. A plant trip occurred on October 26, 1983, approximately 13 days prior to the discovery of the failed LPSI breaker.

This event was modeled as a transient with one LPSI residual heat removal (RHR) train failed. The estimated conditional core damage probability for this event is 2.5×10^{-6}. The dominant sequence was an anticipated transient without scram (ATWS) sequence with the failure of auxiliary feedwater (AFW). None of the highest-ranking sequences involved the modified branch probability, residual heat removal (RHR).

C.35 LER No. 321/82-011, -012

Event Description: Trip with RCIC Unavailable

Date of Event: February 12, 1982

Plant: Hatch 1

Summary

On February 11, 1982, during testing of the high-pressure coolant injection (HPCI) system, the HPCI auxiliary oil pump failed to perform as required. The oil pump rapidly cycled on and off approximately five times before sealing in and running normally. On February 12, 1982, a reactor scram occurred and the reactor core isolation cooling (RCIC) system was manually initiated to maintain reactor vessel level. Following RCIC initiation, plant personnel discovered that smoke was coming from the RCIC space and the RCIC system was declared inoperable.

The cause of the HPCI auxiliary oil pump cycling was not positively identified. Upon investigation, the condition could not be reproduced and therefore the identification of the exact component causing the problem was not possible. The two components deemed most probably the cause of the failure were replaced to prevent recurrence. The smoke coming from the RCIC space was found to be caused by a 4-oz/hour oil leak in the RCIC lubrication system leaking onto the hot turbine casing.

Although the HPCI pump had some trouble initially starting, it eventually ran satisfactorily. Therefore, for the purposes of this evaluation, HPCI was assumed to be available. This event was modeled as a reactor trip. It was assumed that RCIC would fail without adequate bearing lubrication, so RCIC was assumed to be failed and not recoverable in the model. The estimated conditional core damage probability is 3.3×10^{-6}. The dominant sequence involves the observed transient, failure of the power conversion system, main feedwater system success, and failure of the residual heat removal system. This event was also evaluated assuming that the feedwater and power conversion systems were inoperable, since their actual status during the event is unknown. Assuming main feedwater (MFW) and power conversion system (PCS) are initially unavailable, the conditional core damage probability estimated for this event would be 1.5×10^{-5}.

C.36 LER No. 321/82-070, Rev. 1

Event Description: HPCI and RCIC Simultaneously Unavailable

Date of Event: August 5, 1982

Plant: Hatch 1

Summary

On August 5, 1982, the reactor core isolation cooling (RCIC) minimum flow bypass valve was found to be cycling open and closed during full RCIC pump discharge flow. On August 20, 1982, the high-pressure coolant injection (HPCI) system was tagged out of service for maintenance. When RCIC was tested on August 20, 1982 to verify its availability, the minimum flow bypass valve experienced the same malfunction.

This event was modeled as an unavailability of both RCIC and HPCI. RCIC was assumed to be unavailable for 30 days (half the time between tests plus the 15 days involved in this event.) HPCI was assumed to be unavailable for seven days (half the assumed LCO period) because the cause and duration of the maintenance activities are unknown. Therefore both systems were assumed to have been simultaneously unavailable for seven days. HPCI was assumed to not be recoverable and a nonrecovery probability of 1.0 was assigned to the system. The increase in core damage probability (CDP), or importance, over the duration of the event is 3.4×10^{-6}. The base-case CDP over the duration of the event is 1.3×10^{-6}, resulting in an estimated conditional core damage probability of 4.7×10^{-6}. The dominant core damage sequence involves a postulated loss of offsite power, failure and recovery of emergency power, failure of HPCI, and failure of RCIC.

C.37 LER No. 321/82-088

Event Description: HPCI and RCIC Unavailable

Date of Event: September 24, 1982

Plant: Hatch 1

Summary

On September 24, 1982, the the high-pressure coolant injection (HPCI) inboard discharge valve's motor operator failed due to a failure of the dc motor windings. At the time, the reactor core isolation cooling (RCIC) system was unavailable due to maintenance. Since both RCIC and HPCI were unavailable, a reactor shutdown was initiated. The plant was not shut down since the RCIC system was returned to service and was demonstrated operable within the required 24-hour time period. HPCI is assumed to have failed 15 days prior to the discovery of the failed valve operator (half a test interval). It is not known how long RCIC was unavailable due to maintenance; however, the limited condition of operation (LCO) time associated with an unavailability of RCIC is 14 days. RCIC is assumed to have been unavailable for half of its allowable LCO time, seven days.

Therefore, this event was modeled as a loss of RCIC and HPCI for seven days (168 hours). RCIC was assumed to have been nonrecoverable, and a nonrecovery probability of 1.0 was assigned to it. A nonrecovery value of 0.55 was used for the HPCI system, since the valve could have been manually operated locally, in accordance with the approach outlined in the *Methods Incorporated into the SAPHIRE ASP Models* paper published in NUREG/CP-0140, *Proceedings of the USNRC Twenty-Second Water Reactor Safety Information Meeting*. The increase in core damage probability (CDP), or importance, over the duration of the event is 2.5×10^{-6}. The base-case CDP over the duration of the event is 1.3×10^{-6}, resulting in an estimated conditional core damage probability of 3.8×10^{-6}. The dominant core damage sequence involves a postulated loss of offsite power, a failure of the emergency diesel generators, a failure of HPCI, and a failure of RCIC.

Summarized Precursors

C.38 LER No. 321/83-122

Event Description: Trip with HPCI Inoperable

Date of Event: December 28, 1983

Plant: Hatch 1

Summary

On December 28, 1983 while Hatch 1 was at 12% power during startup, a surveillance test was performed on the HPCI system. After high-pressure coolant injection (HPCI) was started, it exhibited an "erratic response" and was tripped and declared inoperable. The event was found to have been caused by faulty components in the electronic control system for HPCI. In addition, the control oil pressure regulator was determined to be set wrong. On December 27, 1983, Hatch 1 experienced a reactor scram following a condensate booster pump trip and, presumably, a loss of feedwater.

This event was modeled as a scram and loss of feedwater (LOFW) with HPCI assumed to be unavailable. The conditional core damage probability estimated for this event is 6.5×10^{-6}. The dominant core damage sequence involves the observed trip, failure of the power conversion system, failure of two safety relief valve (SRVs) to close, unavailability of HPCI, and failure of the automatic depressurization system (ADS).

C.39 LER No. 325/82-069

Event Description: Scram with RCIC Inoperable

Date of Event: July 15, 1982

Plant: Brunswick 1

Summary

Brunswick Unit 1 was operating at approximately 55% power when a system operability test was performed on the reactor core isolation cooling (RCIC) system. When the RCIC turbine was started, it immediately tripped on high exhaust pressure. The ultimate cause of the trip was attributed to a control circuit design error. There was a scram five days prior to the RCIC failure, and RCIC was presumably inoperable at the time of the scram. The conditional core damage probability estimated for the event is 3.3×10^{-6}.

C.40 LER Number 327/83-077

Event Description: Reactor Trip with Train A of Auxiliary Feedwater Unavailable

Date of Event: May 31, 1983

Plant: Sequoyah 1

Summary

Two days after a reactor trip, automatic control valve 1-PCV-3-122 in train A of the auxiliary feedwater (AFW) system was declared inoperable due to failure to operate correctly. The failure was due to a clogged hydraulic valve and a worn hydraulic pump, which caused the motor to fail due to excessive current. Valve 1-PCV-3-122 is located in the discharge line of the motor-driven pump (MDP) in train A. Since this was inoperable, the train A MDP could not supply water to the steam generators.

The analysis assumed the valve was inoperable at the time of the trip, and train A was modeled in the analysis as failed. The analysis showed that the dominant accident sequence was the result of AFW failure. The conditional core damage probability was 9.8×10^{-6}. The dominant accident sequence comprised 4.9×10^{-6} of this total, with four other sequences contributing an additional 4.9×10^{-6}.

C.41 LER No. 327/83-100

Event Description: Reactor Trip with Unavailability of Auxiliary Feedwater Motor-Driven Pump Train

Date of Event: July 11, 1983

Plant: Sequoyah 1

Summary

On July 11, 1983, a reactor trip occurred during Unit 1 operation. The auxiliary feedwater (AFW) turbine-driven pump (TDP) was removed from service to check a leaking valve. Subsequently, and less than an hour after the trip, automatic control valve 1-PCV-3-122 in train A of the AFW system was declared inoperable due to failure to open. It was found that the manual control switch for the valve had been placed in the closed position. The valve was opened and returned to service. The cause of the mispositioned switch could not be determined.

These events were modeled as an unavailability of one train of AFW due to the control valve failure. Train 1 was therefore failed in the AFW corresponding to motor-driven pump (MDP) A. Assuming the leaking valve would not have made the TDP inoperable during the trip, no change was made to this pump in the AFW model. With two of the three AFW pumps operable, the AFW system would have been effective if an ATWS occurred so the AFW/ATWS model was not changed. A transient was selected as the event initiator for the analysis. The conditional core damage probability for this event is 5.7×10^{-6}. The dominant core damage sequence involves a transient, successful trip, failure of AFW, failure of main feedwater, and failure of feed and bleed.

C.42 LER Nos. 333/82-009

Event Description: Reactor Trip with High-Pressure Coolant Injection System Inoperable

Date of Event: February 23, 1982 through March 9, 1982

Plant: Fitzpatrick

Summary

On February 10, 1982 during shutdown for refueling, it was discovered that the high-pressure coolant injection (HPCI) system high steam flow set point in the plant Technical Specifications was in error. On March 10, a trip occurred. Two days later, the high-pressure coolant injection system isolated during a startup test due to a high steam flow signal.

It is assumed that the HPCI system was unavailable during the trip since it isolated two days after the trip occurred. In the analysis, the HPCI system was therefore assumed unavailable. The nonrecovery probability was set equal to 0.55 to reflect the possibility that the failure might be recovered at the pump. The conditional core damage probability estimated for this event is 4.8×10^{-6}. The dominant core damage sequence involves the transient, successful reactor shutdown, failure of the power conversion system, success of main feedwater, and failure of the residual heat removal (RHR) system.

C.43 LER No. 334/82-024

Event Description: Transient with Two CCW Pumps Inoperable

Date of Event: July 18, 1982

Plant: Beaver Valley 1

Summary

During plant startup on July 18, 1982, a high-temperature alarm came in on the B component cooling water (CCW) system pump bearing. Steam generator blowdown was isolated to decrease the load on the system and pump B was shut down. Pump C was out for maintenance as well. Pump A remained operable. The B pump bearing had failed due to an apparent motor/pump misalignment. Pump B was restored to service 43 hours later. On the same day, an auto scram occurred due to a high B steam generator level.

This event was modeled as a transient with one train of the CCW inoperable. The CCW has three pumps and two trains. Two pumps provide flow to two trains. The third pump can supply flow to either train in the event that a pump fails. One pump is sufficient to supply CCW to one train of the residual heat removal (RHR) system heat exchangers and pump seal coolers as well as the reactor coolant pump (RCP) thermal barriers, motor lubrication oil coolers and motor air coolers. One train of RHR and HPR was set to failed to reflect the failure of the two CCW pumps on a single RHR pump train. The main feedwater (MFW) train was set to failed and the nonrecovery probability was set to 0.1 to reflect the isolation of MFW due to the high B steam generator level trip. The estimated conditional core damage probability for this event is 3.5×10^{-6}. The dominant sequence did not involve the loss of HPR or RHR, but the failure of auxiliary feedwater, the failure of main feedwater, and the failure of feed and bleed.

C.44 LER No. 334/83-008

Event Description:	Transient with the Turbine-Driven Auxiliary Feedwater Pump Inoperable
Date of Event:	February 18, 1983
Plant:	Beaver Valley 1

Summary

On February 18, 1983 during routine surveillance testing of the turbine-driven auxiliary feedwater (AFW) pump, excessive pump heating at the inboard packing follower was observed. The pump was removed from service. Investigation revealed that the packing follower heatup was caused by a packing failure. Two outer rings were found dry. The pump was repacked and returned to service the next day. A plant trip occurred on February 12, 1983 due to low steam line pressure. A fitting on the air supply line to the trip valve on the B main steam line TV-MS-101B had separated, allowing TV-MS-101B to close (NUREG-0020).

Since the plant trip occurred only six days prior to the discovery of the failed AFW pump packing, this event was modeled as a transient with main feedwater (MFW) assumed unavailable and the turbine-driven AFW pump assumed failed. The MFW train was set to failed and the nonrecovery probability was set to 0.1 to reflect the trip due to low steam line pressure, which would most likely isolate MFW. The third train of AFW was set to failed to reflect the inoperable turbine-driven pump. The estimated conditional core damage probability for this event is 5.9×10^{-6}. The dominant sequence involved the failure of AFW, the failure of MFW, and the failure of feed and bleed.

C.45 LER Number 335/82-062

Event Description:	Inadvertent Safety Injection and Loss of Vital Power Supplies
Date of Event:	November 26, 1982
Plant:	St. Lucie 1

Summary

On November 26, 1982 during full-power operation, safety injection actuation signals (SIAS) for channels A and B of the emergency safety features actuation system (ESFAS) were actuated because a trip test switch was incorrectly positioned by maintenance personnel performing a monthly preventive maintenance test. All appropriate automatic actions occurred; however, the static uninterruptable power supply (SUPS) was lost due to an incorrectly set time delay for the 480-V emergency bus undervoltage relay. The reactor was manually tripped. The diesel generators (DGs) automatically loaded to provide ac power to the plant vital loads. Vital power and normal plant status were restored in approximately 45 minutes.

The event was modeled using three different scenarios. In the baseline case it was assumed that both diesels operated as they did during the actual event. In the second case it was assumed that one diesel failed to start or was unavailable, i.e., only one train of emergency power (EP) was available. Finally, the case of both diesels failing (both trains of EP) was considered. For the second two cases, the appropriate numbers of trains of auxiliary feedwater (AFW), feed and bleed, residual heat removal (RHR), containment spray recirculation (CSR), and high-pressure recirculation (HPR) were made unavailable in the event tree branches. The conditional core damage probabilities (CCDPs) were then calculated for each case using a transient as the potential initiator. The results of the three cases were then weighted by multiplying the CCDP by the probability of the corresponding number of failed diesels. Summing these values provided a final weighted average of 5.6×10^{-6}. Based on the weighted probabilities, the dominant accident sequence consisted of a successful reactor trip, loss of AFW, loss of main feedwater, and failure of feed and bleed.

C.46 LER No. 338/82-021

Event Description: Transient with One AFW Pump Inoperable

Date of Event: April 16, 1982

Plant: North Anna 1

Summary

On April 16, 1982 following a manual reactor trip due to a loss of the circulating water pumps (NUREG-0020), auxiliary feedwater (AFW) pump 3B was inoperable for a period of 78 minutes. The control switch slipped when an attempt was made to place the switch in the pull-to-lock position. The pump momentarily re-energized while it was slowing down, which resulted in an overcurrent trip. AFW pump 3A was immediately lined up to supply steam generator B. The pump motor was bridged, meggered, and verified operable.

This event was modeled as transient with one AFW pump inoperable. The estimated conditional core damage probability is 1.8×10^{-6}. The dominant sequence involved the failure of AFW, the failure of main feedwater, and the failure of feed and bleed.

C.47 LER No. 339/82-061

Event Description: Transient with One Low-Pressure Safety Injection Pump Inoperable

Date of Event: September 5, 1982

Plant: North Anna 2

Summary

On September 5, 1982 during normal operation, the containment sump suction valve of low-pressure safety injection (LPSI) pump 1A failed to open during a periodic surveillance test. The valve handwheel, which is connected to the valve with a flexible extension from the first level of the safeguards area to the bottom of the valve pit, could not be moved. Investigation revealed that the flexible extension had been wrapped with a power cord by maintenance personnel, which prevented the extension's rotation. A plant trip occurred on August 22, 1982, 14 days prior to the discovery of the valve fault (NUREG-0020).

This event was modeled as a transient with one LPSI pump residual heat removal (RHR train) assumed to be failed. Since high-pressure recirculation (HPR) relies on these pumps as well, one train of HPR was also set to failed. The estimated conditional core damage probability for this event is 1.1×10^{-6}. The dominant sequence was not affected by the branch probability modifications. The dominant sequence involved the failure of auxiliary feedwater, main feedwater, and feed and bleed.

C.48 LER No. 364/82-022

Event Description: Transient with One HPI Pump Inoperable

Date of Event: May 19, 1982

Plant: Farley 2

Summary

On May 19, 1982 during the monthly operability check of Charging Pump 2A, the pump was declared inoperable when it would not start. The charging pump's feeder breaker DF06 failed to close due to a misaligned microswitch arm. The arm was repaired and the charging pump was declared operable approximately 1.5 hours later. A plant trip occurred one week before the discovery of the faulted charging pump. Three charging pumps are used for high-pressure injection and feed and bleed.

This event was modeled as a transient with one charging pump (one train of high-pressure injection (HPI) and FEED.AND.BLEED) inoperable. The estimated conditional core damage probability is 1.6×10^{-6}. The dominant sequence involved the failure of auxiliary feedwater, the failure of main feedwater, and the failure of feed and bleed.

C.49 LER No. 366/82-095

Event Description: RHRSW Loops A and B Unavailable

Date of Event: August 17, 1982

Plant: Hatch 2

Summarized Precursors

Summary

On August 17, 1982 while residual heat removal service water (RHRSW) loop B was out of service for maintenance, the personnel responsible for closing the B loop strainer inlet valve inadvertently closed the A loop strainer inlet valve instead. This blocked the only open flow path in the A loop and resulted in both trains of RHRSW being simultaneously unavailable. The A loop strainer valve was opened and the A loop of RHRSW returned to operable status.

The RHRSW system provides cooling water from the ultimate heat sink (the Altamaha River) to remove decay heat via the residual heat removal (RHR) heat exchangers. By means of a crosstie with the RHR system, the RHRSW system can also supply makeup to the reactor coolant system (RCS) when all emergency core cooling systems have failed. The RHRSW system consists of two independent trains consisting of two pumps each. Each train supplies cooling water from the intake structure to one RHR heat exchanger. During power operation, the RHRSW system is not operating. When required, the system is placed in operation by remote manual means. Pressure control is achieved by a flow control valve on the RHR heat exchanger outlet. Each RHRSW train has a rated decay heat removal capacity of 100%. This implies that two RHRSW pumps supplying a single RHR heat exchanger can provide adequate decay heat removal capacity.

The duration of the combined unavailability of both trains of RHRSW is not clearly indicated in the licensee event report (LER), but it is reported that one train was recovered before an 8-hour Technical Specification time limit was exceeded. A one-hour duration was assumed and this event was modeled as a one-hour unavailability of both trains of RHR. The nonrecovery probability for RHR was revised to 0.054 to reflect the RHRSW failures (based on data included in "Faulted Systems Recovery Experience," Nuclear Safety Analysis Center (NSAC)-161, May 1992). For sequences involving potential RHR or power conversion system (PCS) recovery, the nonrecovery estimate was revised to 0.054 x 0.52 (PCS nonrecovery), or 0.028.

The increase in core damage probability (CDP), or importance, over the duration of the event is 7.7×10^{-6}. The base-case CDP over the duration of the event is 7.7×10^{-9}, resulting in an estimated conditional core damage probability of 7.7×10^{-6}. The dominant sequence involves a postulated transient with a successful reactor shutdown, failure of the power conversion system, success of main feedwater, and failure of all three modes of RHR.

C.50 LER No. 366/83-069, Rev. 1

Event Description: Reactor Scram with HPCI Unavailable

Date of Event: July 22, 1983

Plant: Hatch 2

Summary

On July 22, 1983, one of the two reactor feed pump turbines at Hatch caught on fire. While operators were reducing load in response to the fire, a reactor scram occurred. During testing on July 31, 1983, the high-

pressure coolant injection (HPCI) pump controller was determined to be failed and HPCI was declared inoperable. HPCI was repaired and returned to service on August 2, 1983.

It is assumed that the second reactor feed pump operated successfully during the reactor trip on July 22. HPCI is assumed to be have been unavailable for half of the one-month surveillance interval prior to July 31, rendering it unavailable at the time of the scram. This event was modeled as a reactor scram with HPCI unavailable, and HPCI was assumed to be not recoverable. The conditional core damage probability estimated for this event is 6.2×10^{-6}. The dominant core damage sequence involves the observed transient, failure of the power conversion system, failure of two safety relief valves (SRVs) to close, the HPCI unavailability, and failure of the automatic depressurization system (ADS).

C.51 LER No. 369/82-052

Event Description: Loss of Vital I and C Bus and Trip

Date of Event: June 13, 1982

Plant: McGuire 1

Summary

McGuire 1 was operating at about 75% power on June 13, 1982, when a loss of power occurred on vital instrumentation and control bus 1EKVA, causing failure of channel 1 of the reactor protection and instrument systems and the engineered safety feature (ESF) systems. Channel III of the reactor coolant system loop C flow instrumentat: was already tripped and a 2 out of 3 coincidence low-flow reactor trip occurred. The ac vital bus power supply to bus 1EKVA was declared inoperable and the bus was repowered from its maintenance source. During the post-trip transient, the condensate-feedwater system was overpressurized, several reheater relief valves lifted, and reheater D-1 relief line ruptured. Typical channel-A vital 120 V ac loads expected on a 4-loop Westinghouse pressurized water reactor such as McGuire would include one train of nuclear instrumentation, various process indications, one isolation channel, inputs to both ESF trains and, possibly, one train of ESF slave relays.

This event was modeled as a loss of feedwater with unavailability of auto-initiation for one train of systems initiated by the ESF actuation system. Insufficient information was available to confirm the accuracy of the assumption that the ESF actuation system was affected, so this assumption may be conservative. Because of the assumed impacts of the loss of one channel of ESF output relays, one train of high-pressure injection was assumed to be unavailable when those systems would be auto-initiated [transient-induced loss-of-coolant accident]. Main feedwater was assumed to be unavailable, due to the effects of the system overpressurization. The conditional core damage probability estimated for this event is 3.1×10^{-6}. The dominant core damage sequence includes the observed trip, failure of main and auxiliary feedwater, and failure of feed-and-bleed cooling.

Summarized Precursors

C.52 LER No. 373/82-107, -099

Event Description: Scram with RCIC and CRD Inoperable

Date of Event: August 12, 1982

Plant: LaSalle 1

Summary

At 1835 on August 12, 1982, LaSalle was critical and at low power in startup mode when control rod drive (CRD) pump A tripped on low suction pressure. Operators attempted to start the B CRD pump but were unable to do so. At that point it was discovered that the level in the condensate storage tank, the suction supply to the CRD pumps, was below the low-level alarm set point. At that point, operators scrammed the reactor by taking the mode switch to shutdown. On August 15, 1982, a surveillance test of the reactor core isolation cooling (RCIC) system was performed. RCIC was found to be inoperable when it tripped on overspeed during the test and the governor valve was discovered to be binding. This latent failure presumably existed during the unit scram that occurred on August 12.

This event was modeled as a trip with the RCIC and CRD systems assumed to be inoperable. Reduced condensate inventory has the potential to render the condensate and feedwater systems inoperable but, since it was not indicated that these systems were inoperable during the event, they were assumed to be available in this analysis. The conditional core damage probability estimated for this event is 5.7×10^{-6}. The dominant core damage sequence for this event involves the observed scram, failure of the power conversion system, feedwater success, and failure of the residual heat removal system.

C.53 LER No. 387/83-106

Event Description: HPCI Pump Fails to Deliver Required Flow

Date of Event: August 2, 1983

Plant: Susquehanna 1

Summary

On August 2, 1983 during quarterly surveillance for high-pressure coolant injection (HPCI) verification, the HPCI pump failed to reach required speed and discharge pressure for 5,000 gpm flow. A scram occurred during July, within half of the apparently quarterly surveillance intervals.

This event was analyzed as a scram with HPCI assumed unavailable. The conditional core damage probability estimated for this event is 6.2×10^{-6}. The dominant sequence involves the transient initiator followed by successful reactor shutdown, failure of the power conversion system, no more than one safety relief valve failing to close, success of the main feedwater system, and failure of the residual heat removal system.

C.54 LER Nos. 389/83-037 and -039

Event Description: Trip with Emergency Diesel B and AFW Pump C Inoperable

Date of Event: July 28, 1983

Plant: St. Lucie 2

Summary

On July 28, 1983, with the unit at 0% power, diesel 2B failed to load onto its 4,160-V bus during a loss-of-offsite (LOOP) power test. The cause of the failure was traced to a broken electrical lug, which prevented the output breaker in the diesel generator circuit from closing. The 2A diesel started and loaded normally. Both offsite power sources were available. The 2B diesel was returned to service within five hours. This event was reported in licensee event report (LER) 389/83-037. On the same date, after the LOOP test, the 2C auxiliary feedwater (AFW) turbine-driven pump tripped three times during attempts to start it manually. No cause for the pump failure to start could be determined and it was returned to service. The unit tripped twice on July 26, 1983, two days prior to the diesel generator and AFW pump failures, as well as on July 28.

This event was modeled as a trip on July 28 with one diesel generator (DG) and the turbine-driven AFW pump unavailable. These same equipment unavailabilities may have existed during the trips that occurred on July 26. The conditional core damage probability estimated for this event is 3.7×10^{-6}. The dominant core damage sequence consists of a transient, successful reactor trip, failure of AFW, failure of main feedwater, and failure of feed and bleed.

C.55 LER No. 395/83-019

Event Description: Both Residual Heat Removal Trains and One High-Pressure Injection Train Inoperable

Date of Event: March 17, 1983

Plant: Summer

Summary

On March 17, 1983, with the unit at full power, both trains of the residual heat removal (RHR) and one train of high-head safety injection (HHSI) were inoperable for approximately 45 minutes. RHR train B had been removed from service for preventive maintenance on March 16, 1983. The next day, train A of the chilled water system (CWS) failed to start after a surveillance test. The CWS provides cooling water to the component cooling water (CCW) pump motor bearings. The CCW system, in turn, supplies cooling water to the RHR pump seals and the RHR heat exchanger. As a result, RHR train A was inoperable. The CWS also provides cooling water to the HHSI pump oil coolers, so one train of this system also was considered inoperable per the

criteria contained in the Summer individual plant examination (IPE). The CWS failure was traced to a problem in the starting circuitry of the chiller unit.

Since one train of HHSI was inoperable during this event, one train of high-pressure injection (HPI) and one train of feed and bleed were assumed unavailable. With both RHR trains inoperable, no RHR or high-pressure recirculation (HPR) was available. Since RHR train B was out of service when the failure of train A of the CWS occurred on March 17, it was assumed that both RHR trains and the HHSI train were unavailable for the 45-minute repair period. Transient, loss-of-offsite power (LOOP), loss-of-coolant accident (LOCA), and steam generator tube rupture (SGTR) were used as potential initiators in the unavailability analysis. The increase in core damage probability (CDP), or importance, over the duration of the event is 1.0×10^{-6}. The base-case CDP over the duration of the event is 8.1×10^{-9}, resulting in an estimated conditional core damage probability of 1.0×10^{-6}. The increase in CDP is due almost entirely to the postulated LOCA. The LOCA sequence consists of a postulated LOCA in the 45-minute period, successful reactor trip followed by successful operation of the auxiliary feedwater and remaining HPI trains. Subsequent reactor coolant system cooldown and depressurization to allow RHR initiation are successful, but the unavailable RHR and HPR systems then lead to core damage.

C.56 LER No. 395/83-045

Event Description: Turbine-Driven Auxiliary Feedwater Pump Inoperable Due to Incorrectly Set Speed Control

Date of Event: May 31, 1983

Plant: Summer

Summary

On May 31, 1983, while operating at 100% power, the speed control for the turbine-driven auxiliary feedwater (TDAFW) pump was found to be in the "slow" (minimum) speed position rather than the "fast" position. The controller was immediately placed in the "fast" position. The cause of this incorrect setting was operator error and procedural deficiencies. At the time of the event, the two motor-driven auxiliary feedwater (AFW) pumps were operable. With the controller in the "slow" position, the TDAFW pump will not deliver sufficient flow to meet operability requirements. Six and three days prior to this event, the plant experienced trips.

If the TDAFW pump is assumed to be tested on a monthly basis and the failure is assumed to have existed for half of a test interval, then this event may be analyzed as a trip with the AFW pump turbine unavailable. The conditional core damage probability estimated for this event is 4.6×10^{-6}. The same calculation is applicable to the earlier trip as well. The dominant accident sequence consists of a successful trip following the transient, failure of the remaining AFW trains, failure of main feedwater, and failure of feed and bleed.

Appendix D: Shutdown-Related Events

D.0 Shutdown-Related Events

No cold shutdown events were analyzed in this study because the lack of information concerning plant status at the time of the event [e.g., systems unavailable, decay heat loads, reactor coolant system (RCS) heat-up rates, etc.] prevented the development of models for such events. However, cold shutdown events such as a prolonged loss of residual heat removal (RHR) during conditions of high decay heat can be risk significant. Sixteen shutdown-related events that may have potential risk significance are listed in Table D.1 and summarized in this appendix.

Table D.1 List of Shutdown-Related Events

Event Identifier	Plant	Description	Page
155/82-019	Big Rock Point	Emergency ac Power Unavailable while Shut Down	D-3
155/83-009	Big Rock Point	Emergency ac Power Unavailable while Shut Down	D-4
206/82-015	San Onofre 1	Flood in Saltwater Cooling Pump Intake Structure	D-4
281/82-030	Surry 2	Boron Dilution Event	D-5
311/83-032	Salem 2	Large Leak in Service Water System Bay Leaves Service Water System Inoperable during Shutdown	D-5
312/82-015	Rancho Seco	Loss of Power to B Vital Bus Results in Short-Duration Loss of Shutdown Cooling	D-5
312/83-028	Rancho Seco	Power-Operated Relief Valve Fails Open while in Shutdown	D-6
325/83-007	Brunswick 1	Four Fuel Bundles Inserted in Reactor with Adjacent Control Rods Withdrawn	D-6
327/83-112	Sequoyah 1	Both Trains of Reactor Trip Logic Discovered Inoperable during Shutdown	D-7
328/83-101	Sequoyah 2	Cavitation of Both RHR Pumps during Pumpdown of Refueling Cavity	D-8
336/82-002	Millstone 2	Loss of Shutdown Cooling Due to Electrical Component Failure	D-8

Event Identifier	Plant	Description	Page
338/82-067, 338/83-009, 339/82-026, 339/82-049, 339/83-038	North Anna 1 & 2	Loss of RHR Suction during Shutdown	D-9
369/82-024	McGuire 1	Loss of Shutdown Cooling, Air Binding of RHR Pump during RCS Draindown	D-9
369/83-017	McGuire 1	Both RHR Pumps Cavitate during RCS Draindown	D-10
373/83-105	LaSalle 1	Reactor Vessel Draindown while Shut Down	D-10
387/83-172	Susquehanna 1	Shutdown Cooling Lost Due to Containment Isolation	D-11

D.1 LER No. 155/82-019

Event Description: Emergency ac Power Unavailable while Shut Down

Date of Event: June 18, 1982

Plant: Big Rock Point

Summary

Big Rock Point was shut down when a routine test of the emergency diesel generator (EDG) found it inoperable. Since Big Rock Point is equipped with only one EDG, this event represents a complete unavailability of emergency ac power.

Big Rock Point was shut down by manual scram on June 11, 1982, when a fire occurred in the main generator exciter. After the shutdown, the onsite emergency ac power source was tested every three days. On June 18, when the EDG was tested, a start failure occurred which was attributed to dirty contacts in the starting circuit control relays. The event represents a complete unavailablity of emergency ac power for a period of time. A typical assumption of the unavailability period would be that it existed for half of a test interval, or 1.5 days.

D.2 LER No. 155/83-009

Event Description: Emergency ac Power Unavailable while Shut Down

Date of Event: August 10, 1983

Plant: Big Rock Point

Summary

Big Rock Point was shut down when a routine test of the emergency diesel generator (EDG) found it inoperable. Since Big Rock Point is equipped with only one EDG, this event represents a complete unavailability of emergency ac power.

Big Rock Point was shut down for refueling and modifications in May 1983. On August 10, 1983, the plant was still shut down when the EDG failed during testing. The EDG was being used to power emergency 480-V ac bus 2B when the diesel's fuel pump failed, rendering the EDG inoperable. The event represents a complete unavailablity of emergency ac power for a period of time.

D.3 LER No. 206/82-015

Event Description: Flood in Salt Water Cooling Pump Intake Structure

Date of Event: May 13, 1982

Plant: San Onofre 1

Summary

Errors during maintenance resulted in an internal flood of the salt water intake structure and loss of all salt water cooling during an outage. At 0805 on May 13, 1982, an intake structure high level alarm sounded. Shortly thereafter it was determined that the salt water intake structure was filling with water flowing from the south salt water cooling pump (SWCP), which had been pulled from its foundation flange for maintenance. A short time later, the north SWCP was stopped when its discharge valve malfunctioned and excessive motor loading was observed. The water level in the intake structure rose to approximately 6 inches below the SWCP motor flange elevation. Operators began to align an alternative salt water source, the screen wash pumps, when it was realized that the auxiliary salt water cooling pump would supply the same salt water header that was flooding the intake structure. Maintenance personnel then began to replace the south SWCP. At 0836, both screen wash pumps were started to supply one component cooling water (CCW) heat exchanger. The CCW heat exchanger outlet temperature reached 76°F, and by 0900, the temperature was down to 66°F. A portable submersible pump was dropped into the salt water intake structure, and by 1040, the flood level was observed to be decreasing. Since the unit had been shut down for approximately three months, decay heat loads were minimal and the outlet temperature of the residual heat removal heat exchanger increased only 2°F during the event. Substantial time was thus available to recover RHR cooling.

D.4 LER No. 281/82-030

Event Description: Boron Dilution Event

Date of Event: May 26, 1982

Plant: Surry 2

Summary

While Surry Unit 2 was in cold shutdown, operators began refilling the reactor coolant system (RCS) loop A after steam generator repairs. When makeup water to RCS was aligned, the boric acid flow controller malfunctioned and the makeup water was insufficiently borated. The final boron concentration was not provided; however, the actual shutdown margin was verified to be greater than that required by Technical Specifications. A boron dilution event also occurred on April 15, 1982, on Surry Unit 1.

D.5 LER No. 311/83-032

Event Description: Large Leak in Service Water System Bay Leaves Service Water System Inoperable during Shutdown

Date of Event: June 23, 1983

Plant: Salem 2

Summary

During routine shutdown on June 23, 1983, an equipment operator discovered a large leak in the No. 2 service water bay. An attempt was made to isolate the leak by shutting the No. 21 nuclear header supply valve. Water accumulation was approximately six feet and continued to rise after the header supply valve was closed. To protect the service water pump motors, all pumps were stopped. Loss of service water flow to the charging pumps, residual heat removal (RHR) pumps and heat exchangers, and diesel generator coolers renders these systems inoperable. The flooding was stopped before the level reached the service water motors. The leak was determined to be from the flange of check valve 22SW5. The leak was manually isolated and service water was restored within an hour.

D.6 LER No. 312/82-015

Event Description: Loss of Power to B Vital Bus Results in Short-Duration Loss of Shutdown Cooling

Date of Event: June 24 , 1982

Plant: Rancho Seco

Summary

On June 24 ,1982, while the plant was shut down, preventive maintenance was being performed on the B inverter. The B bus was being powered by a temporary source during the maintenance. When the technicians attempted to return the inverter to service, it tripped to a standby state, causing the B bus to lose power. This caused pressure transmitter PT-21009 to fail high. The high transmitter reading triggered the overpressure protection feature of the decay heat removal system, which closed valve HV-20002 and tripped the running B decay heat removal system pump. Operators reopened the valve and restarted the pump. The same sequence of events occurred twice before the inverter was successfully returned to service.

D.7 LER No. 312/83-028

Event Description: Power-Operated Relief Valve Fails Open while in Shutdown

Date of Event: September 19, 1983

Plant: Rancho Seco

Summary

On September 19, 1983, during cooldown following a steam generator tube leak, a power-operated relief valve (PORV) failed open and could not be shut. The reactor pressure was at 29 psig with the A loop of the decay heat removal system in operation. The block valve upstream of the PORV was closed and pressure relief was terminated. The operational mode of the PORV at the time of the event was to open for cold overpressure protection at a setpoint of 500 psi. A metal pointer was added to the valve actuating lever to show the position of the pilot valve plug in order to indicate power to the solenoid in fulfillment of a Technical Specification requirement. The pilot valve actuating arm was chosen since it is the only external, accessible feature of the valve that could indicate whether the solenoid was energized properly. When the reactor pressure fell to approximately 30 psig, the added weight of the pointer caused the actuating arm to drop, which opened the pilot valve, which in turn opened the PORV. The pointer was removed from the arm, and the actuating arm was used to determine valve position.

D.8 LER No. 325/83-007

Event Description: Four Fuel Bundles Inserted in Reactor with Adjacent Control Rods Withdrawn

Date of Event: January 23, 1983

Plant: Brunswick 1

Summary

While control rod drives were being rebuilt during a refueling outage, it was decided to work on drives in cells that had not been defueled. The fuel bundles were moved from these cells to other locations in the core. Four of the relocated fuel bundles were subsequently noted to have been placed adjacent to withdrawn control rods. Boiling water reactors do not use borated reactor coolant as a neutron poison, but rely entirely on boron-filled control rods to keep the reactor subcritical during refueling operations. In order to prevent inadvertent criticalities, refueling procedures require that fuel be loaded only into cells with inserted control rods. A shutdown margin calculation for the incorrect core configuration determined that an adequate shutdown margin still existed.

D.9 LER Number 327/83-112

Event Description: Both Trains of Reactor Trip Logic Discovered Inoperable during Shutdown

Date of Event: September 11, 1983

Plant: Sequoyah 1

Summary

On September 11, 1983, during hot shutdown with all control rods fully inserted, manual reactor trip channel tests were performed. During this period, both trains of automatic actuation logic were disabled. This would have prevented an automatic reactor trip if there had been an uncontrolled rod withdrawal. The cause of this event was a procedural error which allowed both trains of the automatic trip logic to be disabled when the reactor trip breakers were closed and the control rods were capable of being withdrawn. The procedure instructions called for the installation of a jumper in both trains of the automatic trip actuation logic from an energized bus bar to the undervoltage (UV) coils. These jumpers caused the UV coil to remain energized, preventing any automatic trip actuation. This condition existed for at least 30 minutes and was not discovered until after the manual reactor trip channel tests had been completed.

This is a shutdown event that is impractical to analyze in detail. Therefore, it was not modeled as an accident sequence precursor. However, an estimate of the probability of the event can be made. NUREG/CR-3862, Development of Initiating Event Frequencies for Use in Probabilistic Risk Assessments, states that the uncontrolled rod withdrawal frequency for a pressurized water reactor is 0.01/yr. Assuming this frequency applies to hot shutdown and that core damage would occur if scram failed, the core damage frequency can be estimated as:

$$0.01/\text{yr} \times (1/6132)\text{yr/reactor hr} \times 0.5\ \text{hr} \times 0.12\ (\text{manual scram prob.}) = 9.8 \times 10^{-8}$$

Based on this estimate, this is a low-probability event.

D.10 LER Number 328/83-101

Event Description: Cavitation of Both Residual Heat Removal Pumps during Pumpdown of Refueling Cavity

Date of Event: August 6, 1983

Plant: Sequoyah 2

Summary

On August 6, 1983, during refueling, the refueling cavity was being pumped down so that a leak in the inspection plate of the loop 4 reactor coolant system cold leg nozzle could be repaired. During the pumpdown the pump in train B of the residual heat removal (RHR) system began to cavitate. The train B RHR pump was stopped and the train A RHR pump was started. It also began to cavitate. The unit operator (UO) then began charging the reactor coolant system (RCS) from the refueling water storage tank (RWST) at 0902, using a centrifugal charging pump. The RHR pump suction piping was vented to remove any entrained air. The RHR pumps were restarted at 0957, but the pumps cavitated again. The RWST to RHR pump flow control valve, 2-FCV-63-1, was then opened to reflood the RCS and recirculation was established. At 1103, a second attempt to pump down the refueling cavity was started at a lower pumping rate. At 1200 the train A RHR pump again began cavitating. Valve 2-FCV-63-1 was reopened and the RCS level was allowed to increase until the pump stabilized.

During the event, the average RCS hot-leg temperature increased from 140°F at 0920, to a maximum value of about 195°F at 0955. By 1145 the temperature was below 140°F. The cause of the pump cavitation was attributed to the RCS water level being pumped below the center line of the RCS loop 4 hot leg. This apparently happened because a temporary tube installed to monitor RCS water level indicated a level that was higher than the actual RCS water level. This false indication was probably due to a flow restriction in the tube. An excessive pumpdown rate may have been a contributing factor. This event occurred with the reactor vessel head removed, allowing gravity feed to be used for makeup, if required.

D. 11 LER No. 336/82-002

Event Description: Loss of Shutdown Cooling Due to Electrical Component Failure

Date of Event: January 6 , 1982

Plant: Millstone 2

Summary

On January 6, 1982, during a refueling outage, preventive maintenance was being performed on static switch VS2. A short-circuited test lead was connected to the printed circuit card of the switch. The card failed and the silicon-controlled rectifier became continually gated. The load on inverter number 2 switched to inverter

number 6. Since the rectifier was on continuously, both inverter outlets were connected in parallel. The frequency differences and a large current flow caused the input fuses to blow. This resulted in the loss of vital instrument ac panel VA20. The loss of the panel caused contact closure in the overpressure circuit for shutdown cooling which in turn caused the 2-S1-652 valve to close. The valve was reopened and shutdown cooling was returned to service within seven minutes. During that time, reactor temperature rose from 84°F to 114°F. At the time of the event, the unit was apparently at midloop. Approximately a half hour was available for recovery before core boiling would have occurred.

D.12 LER No. 338/82-067, 338/83-009, 339/82-026, 339/82-049, and 339/83-038

Event Description: Loss of Residual Heat Removal Suction during Shutdown

Date of Event: October 19, 1982, February 18, 1983, May 20, 1982, July 17, 1982, and May 3, 1983

Plant: North Anna 1 and 2

Summary

On several occasions during the 1982-1983 period, both units at North Anna experienced a loss of residual heat removal (RHR) pump suction, which resulted in fluctuating pump amps and low RHR flow. The loss of RHR flow during each event was the result of ambiguous RCS level readings. On each occasion RCS was supposed to be drained down to the centerline of the nozzles, but the level was actually lower than the centerline, resulting in RHR suction problems. Each time the malfunctioning pump or pumps would be removed from service for venting, and the RCS level was raised. After the pumps were vented and the RCS level raised, the RHR pumps would function properly. On each occasion, RHR was inoperable for less than one hour. Following the July 17, 1982 occurrence at North Anna 2 (LER 339/82-049), procedures were changed to avoid draining the RCS below 10 inches above the nozzle centerline. In spite of the change in procedure, on May 3, 1983, North Anna 2 experienced another loss of RHR due to the RCS being drained below the operating limits of 10 inches above the nozzle centerline to the refueling water storage tank. The RCS level and thus the RHR pumps were again restored.

D.13 LER No. 369/82-024

Event Description: Loss of Shutdown Cooling, Air-Binding of Residual Heat Removal Pump during Reactor Coolant System Drain Down

Date of Event: March 2, 1982

Plant: McGuire 1

Summary

During an outage, the McGuire Unit 1 reactor coolant system (RCS) was being drained down to permit inspection of the steam generator when the operating residual heat removal (RHR) system pump, pump 1A, became air bound and was shut down by operators. Pump 1B was out of service for maintenance at the time. Operators investigated and determined that the control room RCS level instrumentation was not indicating accurately and that the RCS level had been pumped down below that required to ensure adequate net postitive suction head to the RHR pumps. Over a period of about 50 minutes, operators opened the refueling water storage tank supply to the RHR system in an attempt to raise the level sufficiently to prevent futher air binding of RHR pump 1A. Pump 1A was restarted and again displayed symptoms of air binding. The RCS level was again determined to be below that required for RHR pump operation. The RCS level was increased further and the RHR pump finally began providing about 3,000 gpm of flow. The RCS temperature increased from 105°F. to 130°F. during the time that RHR was unavailable. An investigation determined that improper correction for nitrogen cover gas pressure caused a misleading level indication. A discrepancy of up to 80 inches was found to be possible between indicated and actual levels.

D.14 LER No. 369/83-017

Event Description:	Both Residual Heat Removal Pumps Cavitate during Reactor Coolant System Draindown
Date of Event:	April 5, 1983
Plant:	McGuire 1

Summary

During a refueling outage, the refueling cavity was being drained in preparation for installing the reactor vessel head when both residual heat removal (RHR) pumps began to cavitate. Presumably this was because the reactor was pumped down to the bottom of the reactor coolant system (RCS) loops. Both pumps were stopped for an indefinite period while the level was restored. The pumps were filled and vented and successfully restarted. The RCS temperature increased by 28°F during the time that the RHR pumps were unavailable, however the duration of the unavailability and the final RCS temperature were not given. The event was attributed to the inadvertent isolation of the reactor vessel-level gauge used by operators to monitor the cavity level. In addition, no one was assigned to visually monitor level during the pumping-down operation.

D.15 LER No. 373/83-105

Event Description:	Reactor Vessel Draindown while Shut Down
Date of Event:	September 14, 1983
Plant:	LaSalle 1

Summary

While at cold shutdown, a system relay logic test was performed on the residual heat removal (RHR) system. At one point in the test, a valve lineup was established that relied on the B train RHR low-pressure coolant

injection (LPCI) check valve to prevent backflow from the vessel. This valve was stuck open at the time and a large backflow occurred, draining from the LPCI injection line to the B train containment spray and torus return lines. This flow path was apparently not protected by automatic isolation, but operators recognized and promptly isolated the draindown path.

D.16 LER No. 387/83-172

Event Description:	Shutdown Cooling Lost Due to Containment Isolation
Date of Event:	December 30, 1983
Plant:	Susquehanna 1

Summary

On December 30, 1983, during the Unit 1 - Unit 2 tie-in outage, one of the breakers of the motor generator set for the electrical protection assembly of the B reactor protection system (RPS) tripped. A primary containment isolation occurred which resulted in the loss of shutdown cooling (SDC) due to the closure of the SDC suction inboard and outboard isolation valves. Reactor coolant circulation was established through the fuel pool cooling system and core spray was available. The failed breaker was replaced and SDC was re-established within three hours.

Appendix E:
Potentially Significant Events Considered Impractical to Analyze

E.0 Potentially Significant Events Considered Impractical to Analyze

Forty-six events [some involving more than a single licensee event report (LER)] were identified as potentially significant but impractical to analyze. It is believed that such events are capable of affecting core damage sequences. However, the events usually involved degradation of components in which the extent of the degradation could not be determined or the impact of the degradation on plant response could not be ascertained.

For several events classified as impractical to analyze, an assumption that the affected component or function was unavailable over a one-year period (as would be done using a bounding analysis) resulted in the conclusion that a very significant event existed. This conclusion was not supported by the specifics of the event as reported in the LER, or by the limited engineering evaluation performed in the Accident Sequence Precursor (ASP) Program. A reasonable estimate of significance for these events requires far more analytical resources than can be applied in the ASP Program. Brief descriptions of these events are provided in Table E.1.

Table E.1 Events Identified as Potentially Significant But Impractical to Analyze

Plant Name	LER Number	Title/Summary
Big Rock Point	155/82-021	The backup core spray supply was found inoperable. The plant was shut down at the time; however, it had been manually scrammed on June 11, 1982, when a fire broke out in the main generator exciter. It is assumed that the core spray supply was inoperable for half of a test interval, the core spray system was then inoperable at the time of the scram.
Dresden 2	237/82-050	High river level caused flooding in the service water pump room and both units were shut down. Intake water level increased to a maximum of 5.5 inches above the floor. The performance of service water pumps may have been degraded.
Dresden 2	237/82-055	Leaks were found at both ends of a feedwater heater emergency spill drain line. Shutdown was initiated to allow repairs.
Millstone 1	245/82-004	Isolation condenser condensate return valve failed to open on demand. Failure was attributed to a surveillance performed on February 8, 1982. At that time, the valve was manually tightened while warm. As the valve cooled, it contracted and the valve disk became tightly wedged. Subsequently, the condensate return valve body was externally heated and then it opened freely.

Plant Name	LER Number	Title/Summary
Millstone 1	245/82-014, -013	A surveillance test of emergency service water (ESW) system determined that the D ESW pump was inoperable. On the following day, a surveillance test of the emergency gas turbine generator, one of two emergency power sources at Millstone, determined that it was inoperable.
Quad Cities 1	254/83-012	Control rods were inserted in reverse order while preparing for a Unit 1 outage.
Palisades	255/83-007	One cooling fan in each engineered safety feature (ESF) room was found to be connected to an alternative emergency diesel generator (EDGs). If one EDG failed following a loss of offsite power (LOOP), then only one fan would be available for room cooling. One fan may have been inadequate for this.
Palisades	255/83-008	One of two cooling fans in both ESF rooms was found failed. A second fan in one room was found to be wired incorrectly, resulting in the fan running backward when operated.
Palisades	255/83-070	Low-temperature overpressure protection (LTOP) was found to be unavailable when in cold shutdown between 325°F (when the shutdown cooling system isolation valves MOV-3015 and MOV-3016 are opened) and 260°F [when the power-operated relief valves (PORVs) begin to provide relief protection]. Procedural inadequacy was cited.
Quad Cities 2	265/82-022	During the monthly suppression chamber to drywell vacuum breaker test, vacuum breaker 2-1601-33A stuck in its open position. Differential pressure between the drywell and suppression chamber could not be maintained, creating the potential for large slugs of water to damage the suppression chamber during a loss-of-coolant (LOCA) or safety valve actuation. Shutdown was initiated immediately.

Plant Name	LER Number	Title/Summary
Point Beach 1	266/82-026	Following the replacement of 62 Westinghouse relays in the reactor protection system (RPS), two were noted to have excessive dropout times after being de-energized. Coil filler epoxy had apparently leaked into the plunger cavity. All the Unit 1 RPS relays were replaced and two more were found with coil filler leakage. The leakage was apparently caused by improper mixing of epoxy.
Oconee 1	269/82-017	Five of six instrumentation and control batteries were declared inoperable because deficiencies identified during monthly surveillance testing had not been corrected. Various cells had been identified with low specific gravity or individual cell voltage out-of-tolerance. It was stated that the battery conditions would not necessarily indicate a loss of capacity. Test discharges on the batteries with out-of-tolerance cells indicted that battery capacity was not degraded.
Salem 1	272/82-001	Pressurizer overpressure protection system (POPS) relief valve 1PR1 leaked through and was isolated for approximately 24 hours. One day later, POPS valve 1PR2 failed to open on demand. Thus, two POPS valves were simultaneously inoperable.
Salem 1	272/82-028	While refilling the spent fuel pool, the refueling water storage tank (RWST) was drained below the minimum allowable level. The unit was at 62% power. The actual RWST level was not indicated, but level was restored in approximately 1.5 hours.
Surry 1	280/82-001	With the unit in hot shutdown, a spurious safety injection caused a phase 1 containment isolation. A containment component cooling water (CCW) return isolation valve failed to close, resulting in only the CCW piping serving as a barrier to separate the atmosphere inside and outside containment.
Surry 1	280/82-106	With Unit 1 in cold shutdown and the overpressure mitigating system in enable, both pressurizer PORVs were declared inoperable. One PORV was inoperable due to a leaking diaphragm and the other was declared inoperable due to low air pressure in its backup bottle air supply.

Potentially Significant Events Considered Impractical to Analyze

Plant Name	LER Number	Title/Summary
Salem 2	311/83-005	Both POPS channels were declared inoperable due to leakage in the pressurizer relief lines. In addition, due to leakage problems with PORVs, both PORV block valves had been closed. The problems led to the redesign of the PORV control circuitry to allow the PORV valves to function as POPS valves.
Salem 2	311/83-010	POPS was inoperable due to the problems noted in LER 311/83-005, which is discussed above. During this time period, the reactor coolant system vent path was lost due to inadvertent closure of PORV 2PR1.
Cook 1	315/83-102	A 10 CFR 21 notification was received at the plant from Westinghouse which identified a non-conservatism in Barton transmitter model 762, Lot 2. These transmitters were being used for low pressurizer pressure ESF actuation and thereby created a possible nonconservative low pressurizer pressure ESF actuation setpoint. The extent of the nonconservatism was not discussed.
Cook 2	316/83-080 316/83-078	During testing, EDG AB failed to trip as required and was declared inoperable to replace a solenoid valve coil. The outage time needed to repair and restore EDG AB was not given. Three days later EDG CD was removed from service due to an air leak. The joint outage time for the two EDGs, if it existed, could not be determined.
Calvert Cliffs 1	317/83-028	Both emergency core cooling system (ECCS) pump room air coolers were out of service for approximately 22 hours. One was out for maintenance. The other was isolated to facilitate draining the saltwater piping on the cooler that was out for maintenance. These coolers can be necessary to support long-term operation of the ESF pumps during accident conditions.
Hatch 1	321/83-011	The essential control power needed to trip the circulating water pump feeder breakers on a division 2 condenser bay flooding signal was found to never have been connected. The event was attributed to improper installation during construction.

Plant Name	LER Number	Title/Summary
Brunswick 2	324/83-069	Neither of the residual heat removal (RHR) room coolers would start when demanded. In one case, the fan start limit switch was out of adjustment and in the other the airflow dampers were mechanically binding. These coolers can be necessary to support long-term cooling for the low-power coolant injection (LPCI) and suppression cooling modes of RHR.
Brunswick 1	325/82-038	The 125-V dc battery charger output breaker for battery 1A-1 was opened while the DIV-1 battery was isolated. This de-energized the 1A-1 dc bus, causing a scram. Limited information was available regarding the impact of the loss of dc bus 1A-1.
Sequoyah 2	328/82-002 328/82-004	The turbine-driven auxiliary feedwater pump failed to start on a safety injection signal. It was determined that a pump stop valve would not open because the overspeed trip latch function had inadvertently not been reset. Three days later it was discovered that vital battery bank II had an average battery cell temperature of less than 60°. The battery bank was declared inoperable. If these conditions overlapped, a non-trivial conditional core damage probability (CCDP) would be obtained, assuming a LOOP occurred.
Sequoyah 2	328/83-114	One safety injection pump and both centrifugal charging pumps failed to meet minimum head requirements during refueling testing. Information was not provided regarding the degree to which the performance of the pumps was degraded.
Duane Arnold	331/82-036 331/82-037	During surveillance testing, high-pressure coolant injection (HPCI) initially failed to come up to speed in test mode. In a manual control mode, HPCI fast started successfully. On the same day, reactor core isolation cooling (RCIC) failed to reach design flow, but the decrease in flow was less than 10%. Too many assumptions would be necessary to model this event.

Plant Name	LER Number	Title/Summary
Fitzpatrick	333/83-030 333/83-041	Standby liquid control (SLC) was declared inoperable to repair squib valve firing circuits after they were inadvertently actuated. On the same day, it was determined that the EDGs might not have remained operable if the diesel exhaust systems were subjected to tornado wind loads.
Beaver Valley 1	334/82-010	Instrument inaccuracies following a high-energy line break could have led to early termination of safety injection. Inadequate reactor coolant system (RCS) cooling could have resulted.
St. Lucie 1	335/82-037	After an inadvertent loss of one 4.16-kV bus and subsequent EDG start and load, plant operators tripped the unit because of unspecified conflicting indications. It was unclear whether the confusion would be expected to continue after the trip.
Davis-Besse 1	346/82-024	During shutdown (mode 5), both boric acid addition tank (BAAT) pumps were defeated when operators failed to restore a manual valve to its normally open position after maintenance. This event would be of concern during a boron-dilution accident, but potential causes of such an initiator and the potential mitigating responses could not be determined without knowledge of system/plant status during the shutdown.
San Onofre 2	361/82-073	A review of reactor water storage tank (RWST) Technical Specification levels determined that the minimum allowable level should have been 87.5%, rather than the 75% indicated in procedures. While RWST levels were generally kept at 90% since initial criticality, the level had apparently been as low as 80% on several occasions. The length of time the level was low was not specified.
San Onofre 2	361/82-081	Channel A and channel C departure from nuclear boiling ratio (DNBR) margin monitors failed channel check tests and were declared inoperable. The channels were apparently simultaneously inoperable for 8 days.

Plant Name	LER Number	Title/Summary
San Onofre 2	361/82-089	Qualification testing of the core protection calculator (CPC) revealed a discrepancy between the functional requirements for the CPC and the CPC software. The discrepancy involved a lack of application of a proper local power density penalty factor under certain low-power conditions.
San Onofre 2	361/82-136	At 20% power, a momentary loss of power to the feedwater control system occurred. The reactor manually tripped and ECCS automatically occurred. The plant cooldown rate was greater than 100° per hour and the temperature may have dropped as much as 135° in a six-minute period. With the ECCS actuation, pressurized thermal shock is a concern.
San Onofre 2	361/82-167	With Units 2 and 3 in Mode 5 and operating their shutdown cooling systems (SDC), control room emergency chiller E-336 tripped on high bearing temperature and was declared inoperable. This rendered the train A shutdown cooling system inoperable for both units. Train B SDC for Unit 3 was out of service, but train B for Unit 2 was operating. Thus, a potential problem existed for Unit 3 in the context of a LOOP while in shutdown.
San Onofre 2	361/82-168	In mode 1, control element assembly (CEA) 12 went from 148 inches to 120 inches. The reactor tripped, a concurrent turbine trip occurred, and a motor control center failed to energize. Excessive steam demand and rapid RCS cooldown then occurred, making pressurized thermal shock a concern.
San Onofre 2	361/83-024	Loss of emergency chiller E-336 (train A) rendered high-pressure safety injection (HPSI) pump 2P-017 inoperable. HPSI 2P-018 was not aligned to the train B bus and HPSI 2P-019 had dc control power to its related bus turned off.

Potentially Significant Events Considered Impractical to Analyze

Plant Name	LER Number	Title/Summary
San Onofre 2	361/83-039	In Mode 5, EDG 2G002 was declared inoperable when its starting time exceeded the allowable time, and when successfully restarted, it could not be loaded. EDG 2G003 was also out of service during the event for a design change. The slow start may have been related to a procedural ambiguity, rather than an actual slow start.
San Onofre 2	361/83-073	Core protection calculator channels B and D were found to have incorrect addressable constants. The incorrect constants were apparently nonconservative default values that had been left in following the last 18-month channel calibration and functional test. The potential impact of incorrect constants was not clear.
Hatch 2	366/82-067	The A and B service water supply valves to the RHR and core spray pump room cooler failed to open when the cooler was started during a test. Failure was caused by a lack of lubrication.
Hatch 2	366/83-006	During testing, the reactor water low low level instruments (2B21-N024 A and B and 2B21-NO25A and B) double actuated. That is, the switches actuated correctly at -50 inches, but then deactuated at -80 inches.
ANO 2	368/82-042 368/83-009	In the first event, incorrect data were loaded into the core protection calculators, possibly resulting in nonconservative DNBR and related calculations on all four channels. In the other event, 2 of 4 CPC channels were inoperable due to failed resistance temperature detectors. The impact on the plant's trip and scram capabilities was not clear.
McGuire 2	370/83-089	Following quarterly preventive maintenance, 120-V ac vital instrumentation and control (I&C) battery inverter 2EVI.3 tripped, and the unit was shut down. Limited information was available regarding the loss of the 120-V I&C bus, and no information was available on how the plant shutdown proceeded.

Plant Name	LER Number	Title/Summary
LaSalle 1	373/83-075	A watertight door between the Unit 2 high pressure core spray (HPCS) diesel cooling water pump room and the 2A EDG cooling water pump room was found open and damaged. A flood in one room could have caused both systems to be unavailable.
Susquehanna	387/82-028	Emergency service water loop A was rendered inoperable after a fire started when modifications were being made to the ESW system. The fire started as a result of touching a grounded cable to a live bus bar. The fire was extinguished immediately.

Potentially Significant Events Considered Impractical to Analyze

Appendix F:
Containment-Related Events

F.0 Containment-Related Events

Three containment-related events were found for 1982-83 and are listed in Table F.1. This event category includes losses of containment functions, such as containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control. It should be noted that the search algorithm of the sequence coding and search system (SCSS) does not specifically search for containment-related events. If these events are identified for other reasons during the search, they are then examined and documented. For each event, a summary, an event description, and any additional event-related plant information are provided.

Table F.1 List of Containment-Related Events

Event Identifier	Plant	Description	Page
251/83-016	Turkey Point 4	Both Trains of Containment Spray Inoperable	F.1-1
305/82-030	Kewaunee	All Six Containment Pressure Transmitters were Found Capped Inside Containment	F.2-1
328/82-141	Sequoyah 2	Both Trains of Automatic Actuation for Containment Ventilation Isolation Inoperable	F.3-1

F.1 LER Number 251/83-016

Event Description: Both Trains of Containment Spray Inoperable

Date of Event: October 4, 1983

Plant: Turkey Point 4

F.1.1 Summary

During preparations for a periodic test of the containment spray pumps, the manual discharge valves on both pumps were found locked in the closed position. In this configuration, the containment spray system would have been unavailable if demanded. The valves were misaligned when an operator assigned to close valves on Unit 3 inadvertently closed valves on Unit 4. When the error was discovered, the discharge valve for the 4B pump was opened. The other valve was opened following completion of surveillance testing of the 4A pump.

F.1.2 Event Description

With the plant at 100% power, a configuration verification in preparation for a periodic test of the containment spray pumps revealed that the manual discharge valves on both the 4A and 4B containment spray pumps were locked in the closed position. This isolated the containment spray system and would have made it unavailable in the event of an accident that required this system. The cause was determined to be personnel error. The person assigned to close the identical valves on Unit 3, which was in cold shutdown, closed the valves on Unit 4 instead. When the error was discovered, the discharge valve for the 4B pump was opened and the other valve was returned to the open position following completion of the monthly surveillance test of the 4A pump.

F.1.3 Additional Event-Related Information

During the duration of the event, all of the normal and emergency containment coolers were available. According to the Turkey Point individual plant examination (IPE), these coolers can provide necessary containment heat removal during accidents.

F.1.4 Modeling Assumptions

This event was not modeled as an accident sequence precursor.

F.2 LER No. 305/82-030

Event Description: All Six Containment Pressure Transmitters were Found Capped Inside Containment

Date of Event: October 4, 1982

Plant: Kewaunee

F.2.1 Summary

On October 4, 1982, during full power operation, sensing lines on all six safety-related containment pressure transmitters were found capped off inside containment, thus rendering the transmitters inoperable. The minimum number of channels required for high-containment pressure safety injection actuation and high-high containment pressure for containment spray and steam line isolation were not available. Redundant safety injection and steam line isolation instrumentation remained operable.

F.2.2 Event Description

On October 4, 1982, sensing lines on all six safety-related containment pressure transmitters were found capped off inside containment. Ten caps were placed on the ends of the sensing lines during the performance of a local leak rate test during the 1982 refueling outage. The ten caps rendered twelve pressure-sensing instruments inoperable. The inoperable instrumentation is listed in Table F.2.1. The inoperable pressure sensing instruments left no reliable indicator of containment pressure or containment shield building differential pressure in the control room. The containment vacuum breaker function was rendered inoperable. Engineered safety feature (ESF) actuation signals derived from containment pressure were inoperable. These included safety injection actuation at 4 psig, main steam line isolation at 17 psig, and containment spray actuation at 23 psig. However, redundant ESF auto-actuation signals were available for safety injection (low steam line pressure and low pressurizer pressure) and main steam line isolation (high-high steam flow coincident with safety injection and high steam flow coincident with low T_{avg} (average temperature) and safety injection). Manual initiation was also available for safety injection, main steam line isolation, and containment spray. The licensee stated that the loss of the automatic ESF actuation signals derived from containment pressure has significance for those accidents involving a ruptured steam pipe or a loss of coolant and that safety injection and main steam line isolation could have been auto-initiated by a low steam line pressure signal and a high-high steam flow with safety injection signal or a high steam flow with low T_{avg} and safety injection signal. The licensee also states that containment cooling from the containment fan coil units can provide sufficient containment cooling during a loss-of-coolant accident or a ruptured steam pipe.

Table F.2.1 Pressure Sensing Instrumentation Rendered Inoperable

Instrument No.	Description
P16427	Containment Vacuum Breaker Control
P16428	Containment Vacuum Breaker Control
P21100	Containment Pressure - ESF Actuation
P21101	Containment Pressure - ESF Actuation
P21102	Containment Pressure - ESF Actuation
P21105	Narrow Range Containment Pressure Indication
P21117	Containment Pressure - ESF Actuation
P21118	Containment Pressure - ESF Actuation
P21119	Containment Pressure - ESF Actuation
P21122	Containment - Shield Building Differential Pressure Indication
P21132	Containment Wide Range Pressure
P21133	Containment Wide Range Pressure

F.2.3 Additional Event-Related Information

None

F.2.4 Modeling Assumptions

This event was not modeled as an accident sequence precursor.

F.3 LER Number 328/82-141

Event Description: Both Trains of Automatic Actuation for Containment Ventilation Isolation Inoperable

Date of Event: December 14, 1982

Plant: Sequoyah 2

F.3.1 Summary

Sequoyah 2 was at 0% power during performance of a surveillance test of the slave relays in both trains of the containment ventilation logic. Relays in both trains A and B failed to latch and hold in the energized position. The relays are required to remain latched after receipt of a safety injection signal, which is simulated as part of the test. Evaluation and testing of the relays and latching mechanisms did not reveal the cause of the failures.

F.3.2 Event Description

With the plant at 0% power, a surveillance test of the slave relays in both trains of the containment ventilation logic indicated that relay K615 in train A and relays K615 and K622 in train B failed to latch and hold in the energized position. Both trains were declared inoperable. The relays and latching mechanisms were evaluated and tested extensively, but no additional failures were produced and no failure mode could be identified. The latching mechanisms for all three relays were replaced and tested to verify operability on December 16, 1982.

F.3.3 Additional Event-Related Information

The purpose of the containment ventilation logic is to ensure that the valves in the containment ventilation system are closed when a containment ventilation isolation signal is generated by an automatic or manual safety injection signal, high radiation in the containment, or manual actuation of the phase A or B containment isolation switches. The test of the relays is performed by simulating the safety injection signal. No failures similar to the ones documented in this licensee event report had occurred previously.

F.3.4 Modeling Assumptions

This event was not modeled as an accident sequence precursor.

Appendix G:
Interesting Events

G.0 Interesting Events

Fifteen "interesting" events were found for 1982-83. This event category includes events that were not selected as precursors, but that provided insight into unusual failure modes with the potential for compromising continued core cooling. These events are listed in Table G.1 and summarized in this appendix.

Table G.1 List of Interesting Events

Event Identifier	Plant	Description	Page
259/82-096	Browns Ferry 1	Reactor Period Less than 5 Seconds on Startup	G-3
280/82-048	Surry 1	Shutdown Boron Dilution not Detected until Approach to Criticality	G-4
280/82-049 and Others	Surry 1 and 2	Charging Pump Support Degradation	G-4
293/83-048	Pilgrim	High-Pressure Coolant Injection (HPCI) Suction Piping Subjected to Reactor Pressure and HPCI Inoperable	G-8
302/82-061 302/83-016	Crystal River 3	Fire Service Water in Instrument Air System	G-9
311/83-059	Salem 2	Emergency Diesel Generators (EDGs) 2A and 2B Unavailable Along with 2B Equipment Safeguards Control during Shutdown	G-10
313/82-025	ANO 1	Shutdown Bypass High-Pressure Trips Found Bypassed	G-11
318/83-007	Calvert Cliffs 2	120-V ac Bus Failure 2 and Operator Error Cause RCS Blowdown	G-11
321/82-040, -43, -54, -62, 83-081	Hatch 1	Multiple Service Water System Failures	G-12
321/82-60	Hatch 1	Scram, Isolation, all 11 SRVs Fail to Operate as Required	G-13

Event Identifier	Plant	Description	Page
331/82-057, -058, -059, and -061	Duane Arnold	River Water Supply System (RWSS) B Inoperable, Reactor Core Isolation Cooling (RCIC) Room Cooling Failed, Two Residual Heat Removal Service Water (RHRSW) Pumps Fail Technical Specifications Flow Requirements during a Surveillance Test	G-13
335/82-050	St. Lucie	All Three Charging Pumps Became Gas Bound when Operating to Recover Pressurizer Level after Reactor Trip	G-14
366/82-089	Hatch 2	Control Rod Movement Procedure Violated	G-14
369/82-007	McGuire 1	Cold Weather Complication, Trip	G-15
369/82-015	McGuire 1	Both Charging Pumps Gas-Bound with Hydrogen	G-16

G.1 LER No. 259/82-096

Event Description: Reactor Period Less than 5 Seconds on Startup

Date of Event: December 6, 1982

Plant: Browns Ferry 1

Summary

On December 6, 1982, Browns Ferry Unit 1 was critical, on range 5 of the intermediate range monitor (IRMs). The operator was pulling rods to maintain heatup rate when he pulled control rod 18-51, a known high-worth rod. When rod 18-51 was withdrawn, it withdrew more than a single notch. By the time it had withdrawn two notches, a high-flux IRM scram was received, shutting down the reactor. Subsequent investigation attributed the continuous double-notch withdrawal to defective seals in the control rod drive unit associated with rod 18-51. The reactor period during the event was calculated to be less than five seconds. A reactor period of less than five seconds is sufficiently short that it can be assumed only automatic scrams will operate quickly enough to avoid exceeding fuel limits. Operator manual scram would be very slow relative to the rate of power increase.

Reactor period T is defined as the time for reactor power to increase by a factor of "e" or approximately 2.7. For a reactor of a given design, T is directly related to the excess multiplication factor Δk. The relation between T and Δk is approximated by the following expression:

$$T \approx \frac{l + (\beta - \Delta k)\tau}{\Delta k}$$

where l, the average neutron lifetime, is approximately 1×10^{-4} s; β, the effective delayed neutron fraction, is approximately 0.006 for an aged core; and τ, the weighted average mean life of the delayed neutrons, is about 12 s. Interpreting a reactor period of "less than five seconds," as a period of 4 s, Δk can be calculated to be about 0.005. This value is relatively large compared with the delayed neutron fraction, which is approximately 0.006.

G.2 LER No. 280/82-048

Event Description: Shutdown Boron Dilution Not Detected until Approach to Criticality

Date of Event: April 15, 1982

Plant: Surry 1

Summary

During the course of a reactor startup, the reactor began to approach criticality below the minimum control rod insertion limits. Boron concentration was increased and the reactor was made critical above the minimum insertion limits.

Subsequent analysis of the event determined that control rod position at criticality would have been 37 steps below the minimum insertion limits, had the boration not been made. Insertion limits are established to ensure adequate shutdown margin using the control rods, particularly for accidents involving reactivity additions such as a control rod ejection. Had the reactor been made critical according to the original procedure, these limits would not have been met properly.

Investigation revealed that, on the day prior to the startup, reactor coolant system makeup was increased to compensate for primary system cooldown. However, the makeup was increased by stepping up primary grade water flow to the boric acid blender without increasing boric acid flow. As a result, boron in the reactor coolant inventory was diluted below minimum acceptable limits.

G.3 LER No. 280/82-049 and Others

Event Description: Charging Pump Support System Degradations

Date of Event: April 19, 1982 and others

Plant: Surry 1 and 2

Interesting Events

Summary

Charging pump support system failures, primarily involving the charging pump service water pumps, occurred dozens of times during the 1982-83 period. Many single-train failures and several simultaneous failures of both trains of charging pump service water were reported.

Failure of both trains of charging pump service water renders the charging pumps unavailable for recirculation operation (see NUREG/CR-4550, Vol. 3, Rev. 1, Bertucio and Julius, *Analysis of Core Damage Frequency: Surry Unit 1 Internal Events*, 1990). The number of events reported involving loss of charging pump service water systems suggests a chronic concern, but a strict interpretation of the reported data indicates that the charging pumps may have been unavailable for recirculation for a period of about 24 hours during 1983, and for a period of a few hours in 1982. Since the conditional core damage probability associated with a 24-hour unavailability of recirculation is below the precursor cutoff of 1.0×10^{-6}, the events are summarized but a conditional core damage probability is not calculated.

Specific events summarized below include:

280/82-49,64,67,87,83-38,39,42
281/82-2,9,28,33,50,55,57 and
282/83-30,39,44,50,51.

280/82-049, April 19, 1982

While Surry 1 was operating at 100% power, the charging pump service water pump 1-SW-P-10B was found to have lost suction and to have no discharge pressure. Increasing service water use by the chillers was thought to be the cause. The control room chiller was shut off and the pump suction supply and discharge pressure were restored. The charging pump service water pumps supply water to the charging pump intermediate seal coolers and the charging pump lube oil coolers.

280/82-064, May 30, 1982

While Surry 1 was operating at 100% power, the charging pump service water pump 1-SW-P-10B was found to have lost suction and to have no discharge pressure. Increasing service water use by the chillers was thought to be the cause. Service water supply to the chillers was throttled and the pump suction supply and discharge pressure were restored.

280/82-067, May 24, 1982

While Surry 1 was operating at 100% power, the charging pump service water pump 1-SW-P-10A was found to be unable to develop adequate discharge pressure. Fouling of the pump suction strainer was found to be the cause. Cleaning the strainer allowed the pump to perform normally.

280/82-087, September 1, 1982

This licensee event report identifies two instances in which charging pump service water pump 1-SW-P-10A lost suction and three in which charging pump service water pump 1-SW-P-10-B lost suction. The instances in which the A pump lost suction overlap instances of pump B failing due to loss of suction.

On September 1 and 20, with Surry 1 operating at 100% power, charging pump service water pump 1-SW-P-10A experienced inadequate net positive suction head and suffered a loss of discharge pressure. On September 1, 14, and 20, pump 1-SW-P-10B experienced the same problem.

Service water supply to the chillers was throttled and the suction supply to the charging pump service water pumps was restored. Charging pump bearing temperatures remained within specifications during these events.

280/83-038, August 29, 1983

While Surry 1 was operating at 100% power and Surry 2 was in cold shutdown, maintenance work on Unit 2 required isolation of the Unit 2 service water supply to the header supplying the control room chillers and the charging pump service water pumps. As soon as the Unit 2 service water supply isolation valve, 2-SW-11, was closed, the Unit 1 charging pump service water pump 1-SW-P-10A lost suction and the alternative pump, 1-SW-P-10B, started. The B pump was also unable to develop and maintain adequate discharge pressure. The Unit 2 charging pump service water pump, 2-SW-P-10A was also unable to maintain adequate discharge pressure during the event.

During the interval that charging pump service water was unavailable, charging pump bearing temperatures were monitored and remained within acceptable limits. After a chiller was removed from service, charging pump service water pump suction supply and discharge pressure returned to adequate levels. Valve 2-SW-11 was reopened within 24 hours and the affected systems were returned to normal.

280/83-039, August 31, 1983

Surry 1 was operating at 100% power when it was discovered that charging pump service water pump 1-SW-P-10A could not maintain adequate suction supply or discharge pressure when C control room air conditioning unit was placed in service. Service water was throttled to the control room air conditioning units and 1-SW-P-10A was returned to service.

280/83-042, September 17, 1983

Surry 1 was operating at 100% power when the discharge pressure of charging pump service water pump 1-SW-P-10A dropped sufficiently to cause auto-start of 1-SW-P-10B. Pump A was found to have a clogged suction strainer.

281/82-002, January 9, 1982

During testing, while Surry 2 was operating at 86% power, the discharge check valve for charging pump service water pump 2-SW-P-10A was found to be stuck open. Had the A pump failed and the B pump auto-started, the charging pump service water supply to the charging pumps could have been compromised.

Interesting Events

281/82-009, January 28, 1982

While Surry 2 was operating at 100% power and charging pump B was out of service for service water modifications, charging pump C lubrication oil temperature began increasing due to inadequate service water flow. Pump C was declared inoperable and removed from service. The temperature control valve (TCV) in the charging pump service water piping for charging pump C was cycled several times and flow returned to normal.

One week later, a similar problem developed with charging pump C. At that time, both A and B charging pumps remained available. The TCV was disassembled and a foreign object was discovered in the valve body. The object was removed, the valve was reassembled, and the system was returned to normal.

281/82-028, May 13, 1982

While Surry 2 was operating at 96% power, both charging pump service water pumps lost suction. Service water to the control bay chillers was throttled and pump suction supply was restored in approximately 5 minutes.

281/82-033, May 30, 1982

While Surry 2 was operating at 96% power, charging pump service water pump 2-SW-P-10B was found to have lost suction. The redundant pump remained operable. Service water to the control bay chillers was throttled and the suction supply to pump B was restored.

281/82-050, August 17, 1982

Surry 2 was operating at 100% power and the 10A charging pump service water pump was in service when a low-charging pump service water pump pressure alarm was received. The 10B pump auto-started, but was unable to develop the required discharge pressure. Net positive suction head (NPSH) to the pumps was found to be inadequate and service water supply to the control bay chillers was throttled, increasing NPSH to the pumps. In addition, the suction strainer for pump 10A was found to be partially clogged. Pressure was restored after about 20 minutes; pump bearing temperatures remained within specifications during that time.

281/82-055, September 6, 1982

Surry 2 was operating at 100% power when the A charging pump auxiliary oil pump tripped. The pump and its associated motor were replaced. The replacement pump motor tripped after about 20 minutes of operation. Examination revealed that there was an open winding in the first auxiliary oil pump motor and the second motor was provided with undersized thermal overload protection. The thermal overloads were replaced and the pump was returned to service.

281/82-057, September 1, 1982

This licensee event report describes multiple failures of charging pump service water pumps 2-SW-P-10A and B. On one occasion, both pumps were inoperable on the same day.

On September 1, 13, 18, and 20, 1982, charging pump service water pump A lost suction and was unable to provide the necessary flow. On September 13 and 14, pump B experienced similar failures. In each instance, service water to the air conditioning chillers was throttled to increase net positive suction head to the charging pump service water pumps. The longest interruption of service water supply to the charging pumps was 20 minutes. During that interval, charging pump bearing temperatures did not show any significant increase.

281/83-030, June 7, 1983

Unit 2 was operating at 100% power when a surveillance test determined that neither charging pump service water pump could provide adequate flow to the charging pumps. Net positive suction head to the pumps was determined to be inadequate and the service water supply to the control room chillers was throttled to provide an improved suction supply.

281/83-039, September 26, 1983

While Surry 2 was in hot shutdown, the auxiliary oil pump for charging pump 2-CH-P-1A failed, rendering the charging pump inoperable. At the time, the B charging pump was out of service for maintenance, leaving only one charging pump available.

281/83-044, September 29, 1983

While Surry 2 was operating at 100% power, the standby charging pump service water pump 2-SW-P-10B discharge valve was found to be closed. This would have rendered the pump unavailable had it been demanded. The valve was opened and the pump was verified to be operational.

281/83-050, October 27, 1983

Surry 2 was operating at 100% power when charging pump service water pump 2-SW-P-10B lost suction and its discharge pressure dropped below 10 psig, causing the A pump to auto-start. The suction strainer for B pump was found to be obstructed.

281/83-051, October 24, 1983

While charging pump service water pump 2-SW-P-10A was out of service for maintenance, the operating B pump tripped. The thermal overload for pump B was found to be tripped. It was reset and the B pump was returned to service. Charging pump bearing temperature is reported not to have increased during the time the service water supply was unavailable.

G.4 LER No. 293/83-048

Event Description:	High-Pressure Coolant Injection (HPCI) Suction Piping Subjected to Reactor Pressure and HPCI Inoperable
Date of Event:	September 23, 1983
Plant:	Pilgrim

Interesting Events

Event Description

While surveillances were being conducted for high-pressure coolant injection team supply isolation valve logic and HPCI injection valve logic on September 23, 1983, during steady-state operations, control room annunciators were observed for HPCI turbine high suction pressure and HPCI turbine oil cooler discharge high oil temperature. While operators were responding to the high-pressure alarm, additional alarms were observed. These alarms were two smoke alarms and one battery ground alarm.

The smoke alarms were caused by vapors from heated sections of non-lagged HPCI suction pipe. The battery ground alarm was caused by water from a ruptured gland seal condenser gasket spraying a limit switch. All alarms were cleared in a short time except for the oil cooler discharge high oil temperature annunciator.

After the initial responses and investigations were completed, an HPCI operability test was initiated. During this test, the turbine stop valve failed to close on a remote signal and the HPCI was declared inoperable.

Further investigation revealed that a feedwater pressure transient had occurred in the HPCI suction piping as the result of a partially open injection check valve (2301-7) and the simultaneous opening of the HPCI pump discharge valves (2301-8 and 2301-9) during the valve logic test described above. This combination of events was assumed to subject the HPCI suction piping to a pressure of 1,100 psig. An analysis performed in response to the event determined that none of the piping exceeded yield, but stayed in the elastic region. A system walkdown failed to identify any visible damage to piping or supports.

The root cause of this event involved a combination of three items. The first was a miscommunication between station personnel that allowed both discharge valves to be opened at the same time. The second was a partially opened check valve that allowed a buildup of feedwater pressure beyond its seat, which created an instantaneous pressure transient to flow through the inadvertently opened discharge valves. The third was a failed coil for the turbine stop valve control unit which caused the HPCI system to be declared inoperable after the pressure transient.

[If the pressure transient had not occurred, the ASP analysis of this event would have yielded a conditional core damage probability estimate of 2.0E-5 given a transient had occurred within half of the test interval. Since the pressure transient did occur, this event becomes impractical to analyze as an ASP event using the existing (1982-83) ASP models. However, a preliminary analysis was performed in 1985 and documented via a draft letter report entitled "An Evaluation of BWR Over-pressure Incidents in Low Pressure Systems," by J. D. Harris of Oak Ridge National Laboratory (ORNL) and J. W. Minarick of Science Application International Corporation (SAIC). All of the difficulties described in that report still exist today.]

G.5 LER No. 302/82-061, 302/83-016

Event Description:	Fire Service Water in Instrument Air System
Date of Event:	September 29, 1982
Plant:	Crystal River 3

Summary

On September 29, 1982, water in the instrument air (IA) system caused a fan damper operator on the industrial cooler to fail. This resulted in a reduction of cooling air available for the reactor building. The lack of cooling air to the reactor building resulted in the building temperature exceeding the 130°F limit. Personnel error that occurred on September 21 was determined to be the cause. On September 21, personnel used an inappropriate flow diagram document to determine the correct position of valve FSV-250 instead of using the approved procedure (OP-207). FSV-250 was therefore left open and water from the fire service water system backed up into the IA system. Water was drained from the IA system and the valve was closed, but enough water remained to fail the fan damper operator. The potential for degraded system performance could have been increased had an initiating event occurred which called for systems that have components operated by the IA system or which called for systems that had heat-sensitive components located in the reactor building. According to the individual plant examination (IPE), the loss of IA could result in a loss of main feedwater, but is not expected to significantly affect any other safety system. However, other events have occurred in which water contamination in the IA system had significant impact on other systems. For example, in 1987, water intrusion from the fire water system at Fort Calhoun resulted in component cooling water, emergency power, and emergency core cooling system problems (see NUREG/CR-4674, Vol. 8).

On March 23, 1983, while a refueling surveillance was being performed, the air accumulators for feedwater system valves (FWVs) 39 and 40 would not maintain 27 psig for one hour as required when isolated from the air supply. Investigation revealed that the check valves that isolate the air accumulators were leaking due to water deposits from water in the IA system. The air accumulator valves were cleaned and the O-rings were replaced. The water deposits could have led to the FWVs failing closed and thus a loss of emergency feedwater. However, if the valves failed closed, FWV-161 and 162 could have supplied emergency feedwater.

G.6 LER No. 311/83-059

Event Description: Emergency Diesel Generators 2A and 2B Unavailable Along with 2B Equipment Safeguards Control during Shutdown

Date of Event: November 8, 1983

Plant: Salem 2

Summary

On November 7, 1983, during a maintenance shutdown, safeguards equipment control (SEC) 2B was removed from service in preparation for an emergency safeguards feature manual safety injection test. Emergency diesel generator (EDG) 2A was out of service for maintenance at the time of the test. Approximately 17 minutes into the test, diesel generator 2B was declared inoperable due to a problem with the shaft-driven fuel oil pump. The problem was fixed, and the test proceeded until 1800, November 8 when the accident-blackout mode was initiated. At this time, the bus was stripped and the diesel was loaded. It then unloaded and reloaded for no apparent reason. During the blackout loading, an incomplete sequence alarm was received. The test was terminated and an investigation into the spurious actuation of SEC channel 2B proceeded. Spurious actuation problems had occurred several times in the past with SEC channel 2A (LERs 83-014, 83-025, 83-031, and 83-041). Extensive testing indicated that the problems with the SEC channels were due to the SEC relays. EDG 2A was restored on November 10, and the relays were replaced on November 13. The emergency safeguards feature manual safety injection test was successfully completed on November 16. The manual safety injection test along with the EDG 2A out for service would not likely have occurred at power, but is included here for general interest.

G.7 LER No. 313/82-025

Event Description:	Shutdown Bypass High-Pressure Trips Found Bypassed
Date of Event:	August 28, 1982
Plant:	ANO 1

Summary

On August 28, 1982, while operating at 100% power during the performance of a jumper log verification, it was discovered that electrical jumpers were installed across the trip contacts of the shutdown bypass high-pressure trip bistables for the A and D reactor protection system (RPS) channels. The immediate corrective action was to remove the jumpers and perform an inspection of all RPS cabinets. No other jumpers were found. The jumpers were placed on the trip contacts during surveillance while the plant was at power. The jumpers were not removed due to personnel error in complying with procedures. The shutdown bypass does not provide protective functions while the plant is at power, but had the jumpers been in place when the plant shut down, the RPS could have been adversely affected.

G.8 LER No. 318/83-007

Event Description:	120-V ac Bus Failure and Operator Error Cause Reactor Coolant System Blowdown
Date of Event:	February 3, 1983
Plant:	Calvert Cliffs 2

Summary

While operating in Mode 3 on February 3, 1983, transfer testing of the 21 inverter led to a voltage spike in the 120-V ac system, which resulted in a blown fuse. The blown fuse led to the de-energization of the reactor protection system (RPS) channel A. A licensed operator had been instructed to de-energize RPS channel A in anticipation of the blown fuse. The operator mistakenly de-energized channel D. The 2 of 4 coincidence of the de-energized pressurizer pressure high modules caused the power-operated relief valves (PORVs) to open. An operator overrode the signal to open and the PORVs shut 30 seconds later. Pressurizer pressure decreased to 1,520 psia, which initiated a safety injection actuation signal. No water was injected into the reactor coolant system. Pressurizer pressure was returned to normal approximately an hour after the fuse blew. The pressure increase in the pressurizer quench tank caused the rupture disk to open. The PORV block valves were shut for personnel safety during the replacement of the rupture disk, and the block valves were reopened upon completion of the repair. Crossed power leads caused the fuse to blow. The leads were crossed the previous day during maintenance on the 21 inverter which required the leads to be lifted and replaced with a resistor bank for load testing.

G.9 LER No. 321/82-040, -43, -54, -62, 83-081

Event Description: Multiple Service Water System Failures

Date of Event: April 30, 1982 and others

Plant: Hatch 1

Summary

Multiple failures were reported during 1982 and 1983 in the residual heat removal service water (RHRSW) system.

Licensee event report (LER) 321/82-040, April 30, 1982, describes how RHRSW pump 1E11-C001A failed to meet Technical Specification flow requirements during an operability surveillance performed after completion of maintenance on the pump. The pump failure was attributed to a failed mechanical seal. The seal was replaced and the pump was returned to service on May 1, 1982. The LER indicates that the RHRSW pump failure was a repetitive event and refers to LER 321/81-101 for information about related failures.

LER 321/82-043, May 5, 1982, describes how RHRSW pump 1E11-C001C failed to meet Technical Specification requirements during an operability surveillance. An investigation determined that the failure was due to debris being lodged in the pump impellers. The debris was attributed to unspecified maintenance activities in the area and to a breakdown of the traveling water screens. Divers removed debris from the pump pit, the pump was rebuilt, and a successful test performed.

LER 321/-82-054, June 28, 1982, describes a failure of RHRSW pump C to meet Technical Specification flow criteria during a surveillance. The failure was attributed to normal wear on the pump impeller seal rings and casing wear rings, but the existence of sand and silt deposits around the pump suction was also noted. The pump was returned to service on July 9, 1982.

LER 321/82-062, July 14, 1982, details the failure of RHRSW pump D to meet Technical Specification flow criteria during a surveillance test. In this case, the actual flow of 4,024 gpm at 341 psig was only slightly below Technical Specification requirements of 4,000 gpm at 346 psig, so it can be assumed that the pump was still capable of performing its required safety functions. The test failure was attributed to silt buildup around the pump suction bell.

Interesting Events

LER 321/83-081, August 3, 1982, details the failure of RHRSW pump A on July 28, at which time the C pump was out of service for maintenance. In addition, on August 3, RHRSW pump A was unavailable when B pump was found to be inoperable during surveillance testing.

The significance of the RHRSW system failures depends upon assumptions regarding the ability of degraded trains to provide sufficient flow, how long the RHRSW pumps were inoperable prior to discovery, and what the likelihood of failure of other pumps from the same causes might be. Because of the number of similar events and because of the potential for common cause concerns, these events are described here for general interest. No core damage probability calculation was made.

G.10 LER No. 321/82-60

Event Description: Scram, Isolation, All 11 Safety Relief Valves Fail to Operate as Required

Date of Event: July 3, 1982

Plant: Hatch 1

Summary

Hatch Unit 1 was operating at 100% power on July 3, 1982, when a spurious high-pressure signal caused a reactor scram. The main steam isolation valves (MSIVs) closed and a pressure transient resulted. The main steam safety relief valves (SRVs) are designed to open at various pressure setpoints between 1080 and 1100 psig; however, none of the SRVs opened as pressure increased through this range. At about 1180 psig; 3 SRVs lifted, terminating the pressure increase. Subsequently, the MSIVs were opened and steam was relieved to the condenser. Evaluation of the SRVs by Wyle Laboratory failed to identify the failure mechanism. Nuclear steam supply system (NSSS) supplier General Electric (GE) and SRV manufacturer Target Rock subsequently performed analyses to try to identify the cause. GE suggested that friction in the valve labyrinth seal area or sticking of the pilot disk could have contributed to the failure.

Since this event involved a reactor trip response which did not initially proceed as expected, it is reported here for general interest. However, no core damage probability was calculated for this event.

G.11 LER No. 331/82-057, -058, -059, and -061

Event Description: River Water Supply System B Inoperable, Reactor Coolant Isolation Cooling Room Cooling Failed, Two Residual Heat Removal Service Water Pumps Fail Technical Specifications Flow Requirements during a Surveillance Test

Date of Event: September 2, 1982

Plant: Duane Arnold

Summary

During normal operation on September 2, 1982, while a trouble alarm on the river water intake system 1C-102 was being investigated, the mode switch for the B train traveling screen wash pump of the river water supply system (RWSS) was found in the off position, and train B of the RWSS was declared inoperable. On September 10, the B RWSS traveling screen failed to operate in either the "auto" or "continuous" mode. Investigation revealed that the discharge pressure for the screen wash pump was not high enough to promote screen movement. Also, on September 10, reactor core isolation cooling (RCIC) system pump 1P-226 was declared inoperable due to the failure of room cooling Unit 1V-AC-15A from a broken drive belt and the unavailability of room cooling Unit 1V-AC-15B due to maintenance. On September 14, surveillance tests revealed that residual heat removal service water (RHRSW) pumps 1P-22A and 1P-22D failed to meet the Technical Specification requirement of a total discharge head of 610 feet at a rated flow of 2,400 gpm, due to pump wear.

G.12 LER No. 335/82-050

Event Description: All Three Charging Pumps Became Gas Bound when Operating to Recover Pressurizer Level after Reactor Trip

Date of Event: October 23, 1982

Plant: St. Lucie 1

Summary

During recovery from a reactor trip with all three charging pumps operating to recover pressurizer level, all three pumps became gas bound. The condition was caused by the volume control tank (VCT) being pumped dry, allowing gas to enter the pump suctions. The VCT was pumped dry because the level indication was erroneous due to an empty reference leg. The reference leg was refilled and the VCT level instrument calibrated.

The safety significance of this event is apparently limited because the VCT is isolated during accident operation. Further, the reactor coolant pump seals at St. Lucie are not normally supplied by the charging pumps and they do not provide auxiliary spray flow to the pressurizer during normal operation. However, this event occurred because multiple trains of an important system did not respond as expected, so it is included as an interesting event.

G.13 LER No. 366/82-089

Event Description: Control Rod Movement Procedure Violated

Date of Event: August 8, 1982

Plant: Hatch 2

Interesting Events

Summary

On August 8, 1982 during a normal reactor startup, control rods 30-31 and 22-23 were not withdrawn in accordance with the control rod movement procedure. At the time, the rod worth minimizer was manually bypassed. Rod 30-31 was withdrawn to position 48 full out. Rod 22-23 was withdrawn to position 08, at which point the reactor went critical. Startup continued in a normal fashion as a heat-up rate was established.

Rods 30-31 and 22-23 were inserted to position 04 and the rest of the rods in group 3 were withdrawn to the 04 position and rod movement then continued in accordance with control rod movement procedure. The rod worth minimizer was manually bypassed at this time. Rod 30-31 is the first rod in the rod group 3 and rod 22-23 is the second rod in the rod group 3. The effect of this event on nuclear fuel thermal limits is not known.

G.14 LER No. 369/82-007

Event Description: Cold Weather Complications, Trip

Date of Event: January 11, 1982

Plant: McGuire 1

Summary

A period of extremely cold weather simultaneously caused high system load demand and equipment failures at McGuire 1, beginning on January 10, 1982. The equipment failures culminated in reactor and turbine trips, safety injection, and steamline isolation the following day.

Beginning on the morning of January 10, extremely cold weather caused several steam generator (SG) instrumentation lines to freeze at the point where they passed throught the unheated "doghouse" building adjacent to the reactor building. A steam generator A, channel I, main feedwater flow low alarm was received first, when the associated instrument lines froze. Feedwater control was placed in manual, and plant technicians began placing heaters around the sensing lines. Later, the process radiation monitoring system was declared inoperable when sample flow lines froze.

As outside air temperature approached 19°F, main feedwater flow indications began to fluctuate due to additional freezing in associated instrumentation lines. The refueling water storage tank (RWST) low-low temperature setpoint was reached, with all heaters on and the RWST inventory on continuous recirculation. A fire protection sprinkler system head burst in the turbine building, requiring isolation of a section of the fire protection system header. By mid-afternoon, all sensing lines required for steam generator level control were thawed and returned to service.

Later in the afternoon, another fire protection sprinkler head burst, and the associated header was isolated. During the evening, condensate booster pump suction pressure switches froze and failed, causing false low-suction pressure indications. Shortly thereafter, two additional fire protection lines ruptured near the feedwater heaters in the turbine building. A steam generator (SG) D auxiliary feedwater flow sensing line then froze and was declared inoperable. An SG A auxiliary feedwater flow channel then failed and was declared inoperable.

At 2148, SG A pressure indicated 1,500 psig and main steam pressure began decreasing rapidly. A power-operated relief valve (PORV) for the SG was found open and was isolated. Shortly thereafter, the SG D pressure channels were found to be frozen, and the PORV for SG D was isolated as well. Since tripping both channels as required would have resulted in safety injection (SI), efforts were made to thaw them. At the same time, operators began reducing plant power in preparation for a shutdown. The only remaining SG A pressure channel then showed signs of freezing. For a period of about five hours, automatic engineered safety feature (ESF) actuation capability was lost for inputs associated with SG A. By morning on January 12, most instrumentation channels were restored.

Around 0900 on January 12, flow was established in SG pressure transmitter lines by "bleeding" test connections. Then an SG A channel I pressure transmitter again failed, tripping its associated logic channel. At 1206, efforts to bleed instrument lines caused SG A pressure channel III to trip. The coincidence of the two failed trains caused ESF actuations, and turbine and reactor trips.

This event is described for information only; no calculation of core damage probability was performed.

G.15 LER No. 369/82-015

Event Description: Both Charging Pumps Gas Bound with Hydrogen

Date of Event: February 12, 1983

Plant: McGuire 1

Summary

A failure of the positive displacement charging pump suction dampener allowed hydrogen to enter the charging pump common suction line. Both centrifugal charging pumps became gas bound and were rendered inoperable. While filling and venting the positive displacement charging pump prior to returning it to service, operators opened the pump suction valve. Shortly thereafter, both centrifugal charging pumps (CCPs) began cavitating and were tripped. Operators were apparently familiar with the charging pump failure mode which occurred and they immediately began to fill and vent the CCPs. After perhaps 15-30 minutes, the B charging pump was restarted and returned to service. The unit remained in steady state operation during the time that the charging pumps were unavailable.

The positive displacement charging pump at McGuire is equipped with a suction pulsation dampener. This dampener consists of a vertical 12-inch section of pipe equipped with level switches and a pair of hydrogen supply solenoid valves. High water level in the pipe causes the solenoids to open to admit more hydrogen. Due to an instrumentation failure, hydrogen was added continuously for an extended period of time. It is estimated that approximately 50 cubic feet of hydrogen gas were added.

Interesting Events

The charging pumps at McGuire are not required as part of the engineered safety features system, but can provide an additional redundant source of high-pressure injection. Failure of both charging pumps represents a loss of redundancy in systems that could be used for accident mitigation. Because the charging pump suction header is connected to the high-pressure safety injection pump suction header, this event could have resulted in unavailability of the high-pressure injection system, although the licensee event report does not indicate that this actually occurred.

This event is described for information only; no calculation of core damage probability was performed.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This study is a continuation of earlier work that evaluated 1969–1981 and 1984–1994 events affecting commercial light-water reactors. One-hundred nine operational events that affected 51 reactors during 1982 and 1983 and that are considered to be precursors to potential severe core damage are described. All these events had conditional probabilities of subsequent severe core damage greater than or equal to 1.0×10^{-6}. These events were identified by first computer screening the 1982-83 licensee event reports from commercial light-water reactors to select events that could be precursors to core damage. Candidates underwent engineering evaluation that identified, analyzed, and documented the precursors. This report discusses the general rationale for the study, the selection and documentation of events as precursors, and the estimation of conditional probabilities of subsequent severe core damage for the events.

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