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Precursors to Potential Severe Core Damage Accidents: 1993 A Status Report

Main Report and Appendices A–D

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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

Sixteen operational events that affected sixteen commercial light-water reactors during 1993 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 1993 licensee event reports from commercial light-water reactors to identify those that could potentially be precursors. Candidate precursors were then selected and evaluated in a process similar to that used in previous assessments. Selected events underwent engineering evaluation that identified, analyzed, and documented the precursors. Other events designated by the Nuclear Regulatory Commission (NRC) also underwent a similar evaluation. Finally, documented precursors were submitted for review by licensees and NRC headquarters and regional offices to ensure the plant design and its response to the precursor were correctly characterized. This study is a continuation of earlier work, which evaluated 1969–1981 and 1984–1992 events. The report discusses the general rationale for this study, the selection and documentation of events as precursors, and the estimation of conditional probabilities of subsequent severe core damage for events. This document is bound in two volumes: Volume 19 contains the main report and Appendixes A–D; Volume 20 contains Appendixes E and F.

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PREFACE

The Accident Sequence Precursor (ASP) Program was established by the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory (ORNL) in the summer of 1979. The first major report of that program was published in June 1982 and received extensive review. Eleven reports documenting the review of operational events for precursors have been published in this program (see Sect. 4.0, Refs. 1-11). These reports describe events that occurred from 1969 through 1992, excluding 1982 and 1983. They have been completed on a yearly basis since 1987.

The current effort was undertaken on behalf of the Office for Analysis and Evaluation of Operational Data (AEOD) of the Nuclear Regulatory Commission (NRC). The NRC Project Manager for the project is P. D. O'Reilly.

The methodology developed and utilized in the ASP Program permits a reasonable estimate of the significance of operational events without the laborious detail associated with evaluation using event trees and fault trees down to the component level, while including observed human and system interactions. The present effort for 1993 is a continuation of the assessment undertaken in the previous reports for operational events that occurred in 1969–1981 and 1984–1992.

The preliminary analyses of the 1993 events were sent for review to the NRC headquarters staff, and the NRC regional staffs and licensees for those plants for which potential ASP events were identified. This is similar to the review process used for the 1992 events. In addition, the 1993 events were also independently reviewed as part of NRC's policy regarding probabilistic risk assessment (PRA) activities. All comments were evaluated, and analyses were revised as appropriate.

Reanalyses typically focused on and gave credit for equipment and procedures that provided additional protection against core damage. These additional features were beyond what has been normally included in past ASP analyses of events. Therefore, comparing and trending analysis results from prior years is more difficult because analysis results before 1992 are likely to have been different if additional information had been solicited from the licensees and incorporated. For 1993 the total number of precursors identified is less than that of past years. This is due at least in part to incorporating feedback on equipment, systems, procedures, etc., such that events initially identified as potential precursors with a conditional core damage probability somewhat greater than 10^{-6} were reanalyzed resulting in a value less than 10^{-6}, which is the threshold for rejection.

The operational events selected in the ASP Program form a unique data base of historical system failures, multiple losses of redundancy, and infrequent core damage initiators. These events are useful in identifying significant weaknesses in design and operation, for trends analysis concerning industry performance and the impact of regulatory actions, and for PRA-related information.

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FOREWORD

This report provides the 1993 results of the Nuclear Regulatory Commission's (NRC'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 that were 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 that can be used to trend nuclear plant core damage risk, and (3) to provide a partial check on PRA predicted dominant core damage scenarios.

In recent years, licensees of U.S. nuclear plants have added safety equipment and have improved plant and emergency operating procedures. Some of these changes, particularly those involving use of alternate equipment or recovery actions in response to specific accident scenarios, can have a significant effect on the calculated conditional core damage probabilities for certain accident sequences. In keeping with the practice initiated last year, the 1993 preliminary ASP analyses were transmitted to the pertinent nuclear plant licensees and to the NRC staff for review. The licensees were requested to review and comment on the technical adequacy of the analyses, including a depiction of their plant equipment and equipment capabilities. Each of the review comments received from licensees and the NRC staff was evaluated for reasonableness and pertinence to the ASP analysis in an attempt to use realistic models and data. All of the preliminary precursor events were reviewed, and the conditional core damage probability calculations were revised where appropriate. The objective of this review process was to provide as realistic an analysis of the significance of the event as possible. As a result, the 1993 ASP significant precursor conditional core damage probability results are somewhat lower than would have been calculated with the methods used in previous years. Although this will make year-to-year comparisons somewhat more difficult, it is an important step toward more realistic identification of significant events and conditions. In addition, consistent with the recommendations of the NRC's interoffice PRA Working Group, each of the analyses has been independently peer reviewed. This review provided a quality check of the analysis, ensured consistency with the ASP analysis guidelines, and verified the adequacy of the modeling approach and appropriateness of the assumptions used in the analysis.

The total number of precursors (16) identified for 1993 is less than previous years. This decrease is due in part to consideration of additional plant-specific mitigating equipment and recovery measures that were not considered in the previous years' analyses.

The four most important precursor events for 1993 involved failure of equipment in the plant switchyard (auxiliary transformer), failures of multiple service water system valves (two units), and clogged suppression pool strainers at a boiling-water reactor.

Charles E. Rossi, Director
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of Operational Data

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1 Introduction

The Accident Sequence Precursor (ASP) Program involves the review of licensee event reports (LERs) of operational events that have occurred at light-water reactors (LWRs). The ASP Program identifies and categorizes precursors to potential severe core damage accident sequences. The present report is a continuation of the work published in NUREG/CR-2497, *Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report,*[1] as well as in earlier versions of this document.[2-11] This report details the review and evaluation of operational events that occurred in 1993. The requirements for LERs are described in NUREG-1022, *Licensee Event Report System, Description of System and Guidelines for Reporting.*[12-14]

1.1 Background

The ASP Program owes its genesis to the Risk Assessment Review Group,[15] 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."[16] Evaluations done for the 1969–1981 period were the first efforts in 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. For example, a postulated loss-of-coolant accident with a failure of a high-pressure injection (HPI) system may be examined or studied. In this simple example, the precursor would be the HPI failure.

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 event trees. Those events with conditional probabilities of subsequent severe core damage $\geq 1.0 \times 10^{-6}$ are identified and documented as precursors. For more information, see Chapter 2 of this report or last year's report, NUREG/CR-4674, Vols. 17 and 18.[11]

1.2 Current Process

The current process for identifying, analyzing, and documenting precursors is described in detail in Chap. 2. Documented precursors were transmitted for review by licensees and Nuclear Regulatory Commission (NRC) headquarters and regional office staff. Each documented precursor analysis also received an independent review by an NRC contractor.

In addition to the events selected as accident sequence precursors, those involving loss of containment function and others considered serious but which are not modeled in the ASP Program were identified during the 1993 LER review. These events are also documented in this report.

The NRC's sequence coding and search system (SCSS) data base contained 1472 LERs for 1993. The ASP computer search algorithm selected 650 of these for two-engineer review as potential precursors. The NRC identified 39 events from other sources for review. As a result of the two-engineer review process, 68 LERs were determined to be potentially significant. Of these 68, 28 were rejected after detailed review, 19 LERs were determined to be impractical to analyze, and 1 LER was documented as an "interesting" event. The remaining LERs were analyzed and are documented as identified in Tables 3.1–3.4.

Chapter 2 describes the selection and analysis process used for the review of 1993 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 five appendixes: Appendix A contains all the precursors with appropriate documentation, Appendix B includes containment-related events, Appendix C consists of interesting events, Appendix D describes events that are potentially significant but impractical to analyze, Appendix E contains the resolution of licensee and NRC staff review comments, and Appendix F includes the LERs and Augmented Inspection Team reports cited in Appendixes A, B, and C.

2 Selection Criteria and Quantification

2.1 Accident Sequence Precursor Selection Criteria

The Accident Sequence Precursor (ASP) Program is concerned with the identification and documentation of operational events that have involved portions of core damage sequences and with the estimation of associated frequencies and probabilities.

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 three initiators: trip [which includes loss of main feedwater (LOFW) within its sequences], loss-of-offsite power (LOOP), and small-break loss-of-coolant accident (LOCA). These three initiators are primarily associated with loss of core cooling. ASP Program staff members examine licensee event reports (LERs) 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 for the plant and core damage-related initiating events. Based on a review of the 1984–1987 precursor evaluations, this computerized search successfully identifies almost all precursors within a subset of approximately one-third to one-half of all LERs.

LERs were also selected for review if an Augmented Inspection Team (AIT) or Incident Investigation Team (IIT) report was written regarding the event. The Nuclear Regulatory Commission (NRC) may designate other events for inclusion in the review process.

After the ASP computer search, those events selected underwent two independent reviews by different NOAC staff members. Each LER was reviewed to determine if the reported event should be examined in greater detail. This initial review 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 potentially significant and requiring analysis. Events also were 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, single failures in redundant systems, uncomplicated reactor trips, and LOFW events.

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 loss of redundancy in only one system,
- a seismic design or qualification error,
- an environmental design or qualification error,

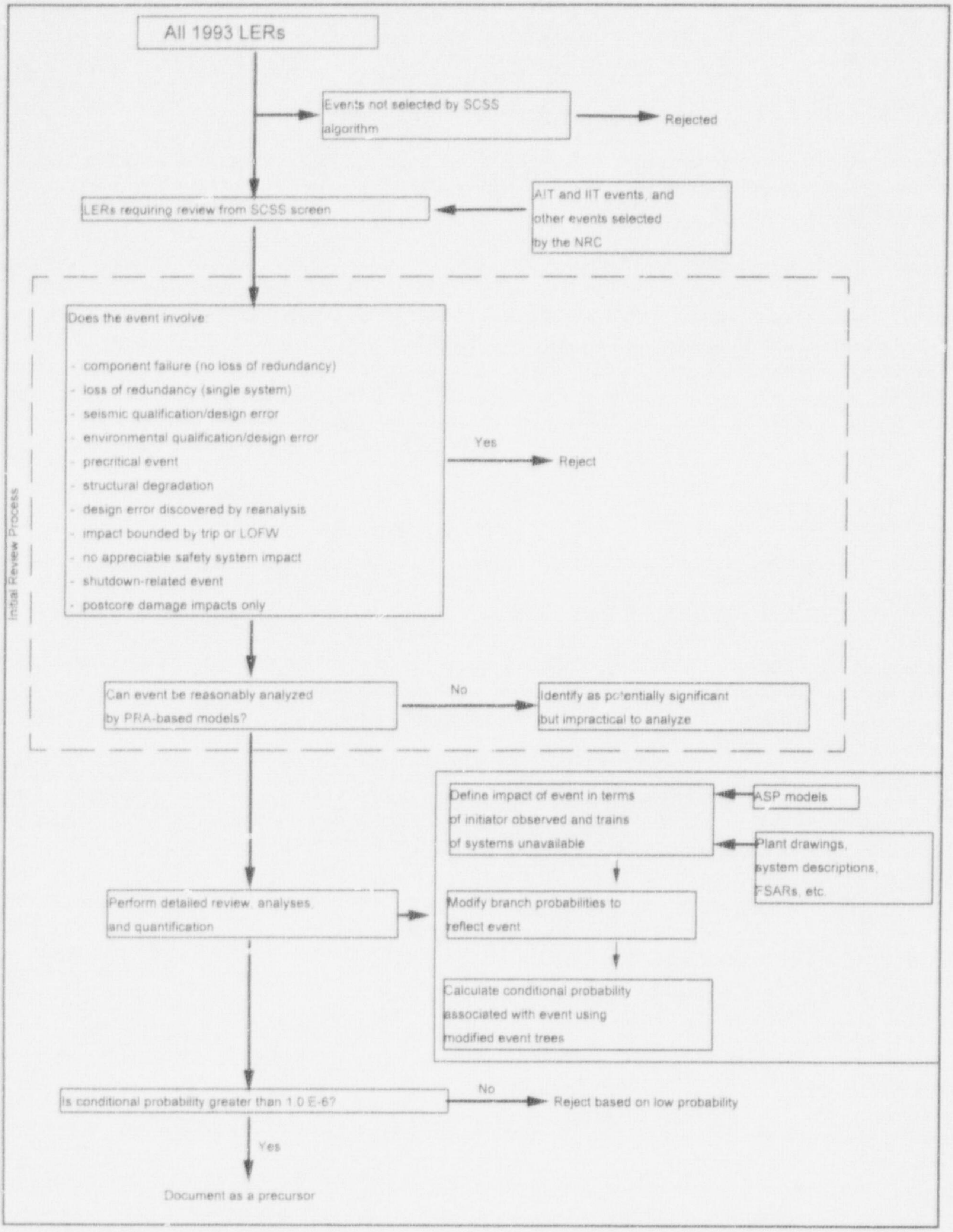

Fig. 2.1 ASP Analysis Process

Selection Criteria and Quantification

- a structural degradation,
- an event that occurred prior to initial criticality,
- a design error discovered by reanalysis,
- an event impact 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 and small-break LOCA);
- all events in which 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 available at NOAC and from the NRC, 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 the equipment failures or operator errors on readiness of systems in the plant for mitigation of 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 could have occurred while at power or at hot shutdown immediately following power operation. If the event could only occur at cold shutdown or refueling shutdown, then its impact on continued decay heat removal during shutdown was assessed.

For each actual occurrence or postulated initiating event associated with an operational event reported in an LER, 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 Sect. 2.2). Events of low significance are thus excluded, allowing attention to be focused on the more important events.

This approach is consistent with the approach used to define 1987–1992 precursors, but differs from that of earlier ASP reports, which addressed all events meeting the precursor selection criteria regardless of conditional core damage probability. While review of LERs identified by this process is expected to identify almost all precursors, it is possible that a few precursors exist within the set of unreviewed LERs. Some potential precursors that would have been found if all 1993 LERs had been reviewed may not have been identified. Because of this, it should not be assumed that the set of !988–1993 precursors is consistent with precursors identified in 1984–1987.

Sixteen 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 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, identified in the yearly review of 1993 LERs are documented in Appendix B. No such events were identified in 1993.

2.1.3 "Interesting" Events

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

2.1.4 Potentially Significant Events Considered Impractical to Analyze

In some cases, events are impractical to analyze due to lack of information or inability to reasonably model 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.

Several LERs identified as potentially significant were considered impractical to analyze. It is thought that such events are capable of impacting core damage sequences. However, the events usually involve 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.

For many events classified as impractical to analyze, an assumption that the affected component or function was unavailable over a 1-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 or by the limited engineering evaluation performed in the ASP Program. Brief descriptions of events considered impractical to analyze are provided in Appendix D.

2.2 Precursor Quantification

Quantification of accident sequence precursor significance 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 event trees, which depict potential paths to severe core damage, and calculating a conditional probability of core damage through the use of event tree branch probabilities modified to reflect the event. The effect of a precursor on event tree branches is assessed by reviewing the operational event specifics against system design information and translating the results of the review into a revised conditional probability of system 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.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the light-water reactor (LWR) population, even though they are applied to sequences that are plant-class specific in nature. 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.

The evaluation of precursor events in this report considers and, where appropriate, gives credit for additional equipment or recovery procedures at the plants. Accordingly, the evaluations for this year may not be directly comparable to the results of prior years. Examples of additional equipment and recovery procedures addressed in the 1993 analyses, when information was available, include use of supplemental diesel generators (DGs) for station blackout mitigation, alternate systems for steam generator (SG) and reactor coolant system (RCS) makeup, depressurization of the primary coolant system at pressurized-water reactors (PWRs) and the use of low-pressure injection (LPI) in lieu of high-pressure injection (HPI), and the use of suppression pool venting for boiling-water reactors (BWRs).

The ASP calculational process is described in detail in Appendix A of Ref. 11. This appendix documents the event trees used in the 1988–1993 precursor analyses, changes to these trees from previous years, the approach used to estimate event tree branch and sequence probabilities, and sample calculations; it also provides probability values used in the calculations.

2.3 Review of Precursor Documentation

After completion of the initial analyses of the precursors, the analyses were transmitted to the pertinent nuclear plant licensees for review and to the NRC staff for review. The licensees were requested to review and comment on the technical adequacy of the analyses, including the depiction of their plant equipment and equipment capabilities. Each of the review comments was evaluated for reasonableness and pertinence to the ASP analysis in an attempt to use best-estimate models and data. Although all of the preliminary precursor events were sent out for review, comments were not received from all the licensees. Each of the comments received was reviewed to determine the effect on the modeling of the events.

This year, for the first time, the 1993 precursor analyses were also sent to an NRC contractor, Sandia National Laboratories, for an independent review. The review was intended to (1) provide an independent quality check of the analysis, (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 some cases the analysis results were significantly affected as a result of comments received. In general, this was the result of incorporation of plant-specific equipment or strategies for mitigating events. Incorporation of these factors for a subset of the analyses reduces the validity of ranking the events by conditional core damage probability. Consistent incorporation of these mitigation strategies across all of the events would affect the conditional core damage probability of some events and may affect the ranking of the events.

A summary of the comments received from the licensees and the NRC staff, as well as a response to each comment, can be found in Appendix E.

2.4 Precursor Documentation Format

Each 1993 precursor is documented in Appendix A. A description of the operational event is provided with additional information relevant to the assessment of the event, the ASP modeling assumptions and approach used in the analysis, and analysis results. Two figures are also provided. The first graphically represents the relative significance of the event compared with other potential events at the plant (using a log scale). The second displays the dominant core damage sequence postulated for the event. The other potential events at the same plant, used for comparison purposes, are briefly described in Table 2.1.

Table 2.1 Description of Reference Events

PWR & BWR	
Trip	Trip with equipment nominally operable
LOOP	Loss of offsite power. Includes plant-centered, grid-centered, severe-weather and extreme severe-weather-related initiators
360 h EP	360 h without emergency power sources (normally on-site emergency DGs)
PWR	
LOFW + 1 MTR AFW	Transient with loss of main feedwater and one motor-driven auxiliary feedwater (AFW) [or emergency feedwater (EFW)] pump failed (turbine-driven pump substituted if plant does not have any motor-driven pumps)
360 h AFW	360 h without AFW
BWR	
360 h HPCI and RCIC	360 h with high-pressure coolant injection (HPCI) and reactor core isolation cooling system (RCIC) failed (not applicable for Type A BWRs)
LOFW and HPCI	Transient with loss of main feedwater and HPCI (loss of main feedwater and loss of isolation condenser for Type A BWRs)

An additional item, the conditional core damage probability calculation, documents the calculations performed to estimate the conditional core damage probability associated with the precursor and includes probability summaries for end states, the conditional probability for the more important sequences, and the branch probabilities used. Copies of the LERs and AIT reports relevant to the events are contained in Appendix F.

2.5 Potential Sources of Error

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

Evaluation of only a subset of 1993 LERs. For 1969–1981 and 1984–1987, all LERs reported during the year were evaluated for precursors. For 1988–1993, only a subset of the LERs was evaluated in the ASP Program after a computerized search of the SCSS data base and screening by NRC personnel. 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 included in the subset that resulted from the screening process.

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.

Lack of appropriate information in the LER. The accuracy and completeness of the LERs in reflecting pertinent operational information are questionable in some cases. Requirements associated with LER reporting (i.e., 10 CFR 50.73), plus the approach to event reporting practiced at particular plants, can result in variation in the extent of events reported and report details among plants. Although the LER rule of 1984 has reduced the variation in reported details, some variation still exists. In addition, only details of the sequence (or partial sequences for failures discovered during testing) that actually occurred are usually provided; details concerning potential alternate sequences of interest in this study must often be inferred.

Accuracy of the ASP models and probability data. The event trees used in the analysis are plant-class specific and reflect differences between plants in the eight plant classes that have been defined. While major differences between plants are represented in this way, the plant models utilized in the analysis may not adequately reflect all important differences. Known problems concern the representation of HPI for some PWRs, ac power recovery following a LOOP and battery depletion (station blackout issues). 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 modeled response 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 (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 in the 1993 precursor analysis. This information was not uniformly available; much of it was provided in licensee comments on preliminary analyses and in 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, analyses of multiple events that occurred at an individual plant or at similar units at the same site have been consistently analyzed.

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 of Ref. 11. 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 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. Programmatic constraints have prevented substantial efforts in estimating actual recovery class distributions. The values currently used are based on a review of recovery actions during historic events and also include consideration of human error during recovery. These values have been reviewed both within and outside the ASP Program. While it is acknowledged that substantial uncertainty exists in them, they are thought adequate for ranking purposes which is the primary goal of the current precursor calculations. This assessment is supported by the sensitivity and uncertainty calculations documented in the 1980–1981 report (see Ref. 2). These calculations demonstrated only a small impact on the relative ranking of events from changes in the numeric values used for each recovery class.

Assumption of a 1-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 1 month was assumed unless another interval was specified in the LER. See Ref. 2 for a more comprehensive discussion of test interval assumptions.

3 Results

This chapter summarizes results of the review and evaluation of 1993 operational events. The primary result 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 precursor selection criteria: (1) a core damage initiator 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. Sixteen such events identified for 1993 are documented in Appendix A.

Direct comparison of results with those of earlier years is not possible without substantial effort to reconcile analysis differences. Additional equipment and procedures (beyond those addressed in the ASP models described in Appendix A of Vol. 17) were incorporated into the analysis of 1992 and 1993 events. The models used in the analysis of 1988–1993 events differ from those used in 1984–1987 analyses. Starting in 1988, the project team evaluated only a portion of the licensee event reports (LERs) (as described in Sect. 2.1.1). Before 1988, all LERs were reviewed by members of the project team. Because of the differences in analysis methods, only limited observations are provided here. Refer to the 1986 precursor report[5] for a discussion of observations for 1984–1986 results and to the 1987–1991 reports[6-10] for the results of those years.

3.1 Tabulation of Precursor Events

The 1993 accident sequence precursor events are listed in Tables 1–4. The following information is included on each table:

- Docket/LER number associated with the event (Event Identifier)
- Name of the plant where the event occurred (Plant)
- A brief description of the event (Description)
- Conditional probability of potential core damage associated with the event [p(cd)]
- Date(s) of the event (Event Date)
- Plant type (Plant Type)
- Initiator associated with the event or unavailability if no initiator was involved (TRANS)

The tables are sorted as follows:

- Table 3.1–Precursors involving unavailabilities sorted by plant
- Table 3.2–Precursors involving initiating events sorted by plant
- Table 3.3–Precursors involving unavailabilities sorted by conditional core damage probability
- Table 3.4–Precursors involving initiating events sorted by conditional core damage probability
- Table 3.5–Abbreviations used in Tables 3.1–3.4.

Table 3.1 Precursors Involving Unavailabilities Sorted By Plant

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	TRANS
Arkansas Nuclear One, Unit 1	313/93-003	Both trains of recirculation inoperable for 14 h	PWR	9/30/93	5.1×10^{-5}	UNAVAIL
Beaver Valley 2	412/93-012	Failure of both EDG load sequencers	PWR	10/4/93–10/6/93	2.1×10^{-6}	UNAVAIL
Catawba 1 and 2	413/93-002	Essential service water potentially unavailable	PWR	2/25/93	1.5×10^{-4}	UNAVAIL
Haddam Neck	213/93-S01,* 213/93-006, 213/93-007	Degradation of MCC-5, pressurizer PORV, and both emergency diesel generators	PWR	6/27/93	6.5×10^{-5}	UNAVAIL
Quad Cities 2	265/93-010, 265/93-012	Degradation of both emergency diesel generators	BWR	4/22/93	6.0×10^{-5}	UNAVAIL
South Texas Project, Unit 1	498/93-005, 498/93-007	Unavailability of one EDG and the turbine driven AFW pump	PWR	12/29/92–1/22/93	1.2×10^{-5}	UNAVAIL
Three Mile Island 1	289/93-002	Both RHR heat exchangers unavailable	PWR	1/29/93	3.1×10^{-6}	UNAVAIL

*AIT Report 213/93-80.

Table 3.2 Precursors Involving Initiating Events Sorted By Plant

Plant	Event Identifier	Description	Plant Type	Event Date	p(cd)	TRANS
Beaver Valley 1	334/93-013	Dual-unit loss-of-offsite power	PWR	10/12/93	5.5×10^{-5}	LOOP
Cook 2	316/93-007	Reactor trip with degraded AFW	PWR	8/2/93	2.4×10^{-6}	TRIP
LaSalle 1	373/93-015	Scram and loss-of-offsite power	BWR	9/14/93	1.3×10^{-4}	LOOP
McGuire 2	370/93-008	Loss-of-offsite power and failure of an MSIV to close	PWR	12/27/93	9.3×10^{-5}	LOOP
North Anna 2	339/93-002	AFW disabled after reactor trip	PWR	4/16/93	1.1×10^{-6}	TRIP
Palo Verde 2	529/93-001	Steam generator tube rupture	PWR	3/14/93	4.7×10^{-5}	SGTR
Perry	440/93-011, 440/93-010	Clogged suppression pool strainers and service water flood	BWR	3/26/93	1.2×10^{-4}	TRIP
Pilgrim	293/93-004	Weather-induced LOOP, vessel pressure/temperature limits violated	BWR	3/13/93	4.6×10^{-6}	LOOP

Table 3.3 Precursors Involving Unavailabilities Sorted By Conditional Core Damage Probability

p(cd)	Plant	Plant Type	Event Identifier	Description	Event Date	TRANS
1.5×10^{-4}	Catawba 1 and 2	PWR	413/93-002	Essential service water potentially unavailable	2/25/93	UNAVAIL
6.5×10^{-5}	Haddam Neck	PWR	213/93-S01,* 213/93-006, 213/93-007	Degradation of MCC-5, pressurizer PORV, and both emergency diesel generators	6/27/93	UNAVAIL
6.0×10^{-5}	Quad Cities 2	BWR	265/93/010, 265/93-012	Degradation of both emergency diesel generators	4/22/93	UNAVAIL
5.1×10^{-5}	Arkansas Nuclear One, Unit 1	PWR	313/93-003	Both trains of recirculation inoperable for 14 hours	9/30/93	UNAVAIL
1.2×10^{-5}	South Texas Project, Unit 1	PWR	498/93-005, 498/93-007	Unavailability of one EDG and the turbine-driven AFW pump	12/29/92–1/22/93	UNAVAIL
3.1×10^{-6}	Three Mile Island 1	PWR	289/93-002	Both RHR heat exchangers unavailable	1/29/93	UNAVAIL
2.1×10^{-6}	Beaver Valley 2	PWR	412/93-012	Failure of both EDG load sequencers	10/4/93–10/6/93	UNAVAIL

*AIT Report 213/93-80.

Table 3.4 Precursors Involving Initiating Events Sorted By Conditional Core Damage Probability

p(cd)	Plant	Plant Type	Event Identifier	Description	Event Date	TRANS
1.3×10^{-4}	LaSalle 1	BWR	373/93-015	Scram and loss-of-offsite power	9/14/93	LOOP
1.2×10^{-4}	Perry	BWR	440/93-011, 440/93-010	Clogged suppression pool strainers and service water flood	3/26/93	TRIP
9.3×10^{-5}	McGuire 2	PWR	370/93-008	Loss-of-offsite power and failure of an MSIV to close	12/27/93	LOOP
5.5×10^{-5}	Beaver Valley 1	PWR	334/93-013	Dual unit loss-of-offsite power	10/12/93	LOOP
4.7×10^{-5}	Palo Verde 2	PWR	529/93-001	Steam generator tube rupture	3/14/93	SGTR
4.6×10^{-6}	Pilgrim	BWR	293/93-004	Weather-induced LOOP, vessel pressure/temperature limits violated	3/13/93	LOOP
2.4×10^{-6}	Cook 2	PWR	316/93-007	Reactor trip with degraded AFW	8/2/93	TRIP
1.1×10^{-6}	North Anna 2	PWR	339/93-002	AFW disabled after reactor trip	4/16/93	TRIP

Table 3.5 Event Initiator or Unavailability Abbreviations

Abbreviation	Definition
LOFW	Loss of feedwater
LOOP	Loss-of-offsite power
LOCA	Loss-of-coolant accident
LSDC	Loss of shutdown cooling
MSLB	Main steam-line break
SGTR	Steam generator tube rupture
TRIP	Reactor trip
UNAVAIL	System(s) unavailable
UNIQ	Unique sequence

3.1.1 Containment-Related Events

No containment-related events were found for 1993. This event category includes losses of containment function, such as containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control.

3.1.2 "Interesting" Events

One "interesting" event was found for 1993 and is documented in Appendix C of this report. This event category includes events that were not selected as precursors but that provided insight into unusual failure modes with the potential to compromise continued core cooling.

3.1.3 Potentially Significant Events That Were Impractical to Analyze

Nineteen potentially significant events were considered impractical to analyze for 1993. Typically, this event category includes events that are impractical to analyze due to lack of information or inability to reasonably model the event within a probabilistic risk assessment framework, considering the level of detail typically available in probabilistic risk analysis models. These potentially significant events are documented in Appendix D of this report.

3.2 Important Precursors

Four precursors with conditional core damage probabilities of $\geq 10^{-4}$ were identified for 1993. Events with such conditional probabilities have traditionally been considered significant in the ASP Program. For 1993, these events, in alphabetical order, include the following:

3.2.1 Catawba Units 1 and 2 (LER 413/93-002)

On February 25, 1993, with Catawba 1 at 100% power and Catawba 2 in refueling shutdown, three of the four essential service water (ESW) pump discharge valves failed to open during a surveillance test. It was later determined that the torque switch settings (TSSs) for all four of the ESW pump discharge valves were improperly set.

In 1989, the "open" torque switch settings (TSSs) for 56 butterfly valves were to be set to the maximum value to address problems with opening these valves under high differential pressure. The four ESW pump discharge valves were included in these 56 valves. The "open" TSSs for the Unit 1 ESW pump discharge valves were set to the maximum value (3.0). However, the "open" TSSs for the Unit 2 valves were incorrectly left at 1.5. The "close" TSS was adjusted to the maximum value instead.

In August 1992, the Unit 1 ESW pump discharge valves were set-up per Generic Letter (GL) 89-10 criteria. This resulted in the "open" TSSs being reduced from the maximum value of 3.0 to 2.0. The Unit 2 valves were not reset to the GL 89-10 criteria at the time of the event.

Following the failure of the B train valves on February 25, 1993, the licensee realized that the TSSs for the Unit 2 ESW valves had been mistakenly reversed in 1989. The TSSs for the Unit 2 valves were changed to the maximum setting. The discharge valves for the Unit 1 pumps were set to 20 deg open. Following these changes, all valves were successfully opened against maximum differential pressure.

The licensee conducted a study of the history of TSSs for the ESW valves and discovered that (1) ESW pump 1A was affected between August 1992 and February 1993, (2) ESW pump 1B was affected between November 1985 and July 1989 and between August 1992 and February 1993, and (3) ESW pump 2B was affected between November 1985 and February 1993. As a result, from August 1992 through February 1993, three of the four ESW pump discharge valves (1A, 1B, and 2B) were unable to open against full differential pressure.

With three of the four valves unable to open under full differential pressure conditions, the failure of the ESW pump associated with the operable valve would result in a loss of ESW to both Catawba units.

The event was modeled as a potential failure of the "2A" ESW pump to run. Following the failure of the "2A" ESW pump, two mitigation strategies were considered possible. The first involves the recovery of one ESW pump before an RCP seal LOCA (50 min). Recovery of the one ESW pump would supply sufficient cooling water for both units, assuming that a LOCA did not occur on either unit. A LOCA concurrent with a trip of the running ESW pump was considered unlikely. Even if a LOCA did occur, once the first ESW pump was running, the second could be started from the control room because the discharge valves would not have to open against full system differential pressure. Once ESW is recovered, systems cooled by ESW would become operable. The other recovery strategy involves placing the safe shutdown facility (SSF) in service to provide RCP seal cooling and starting the turbine-driven AFW pump to provide secondary-side heat removal. This would allow the plant to achieve a hot shutdown condition even without the recovery of the ESW system.

The conditional probability of core damage estimated for this event is 1.2×10^{-4}. The dominant core damage sequence involves a failure of the running ESW pump, failure to recover ESW within 50 min, and failure of the SSF. A second important core damage sequence involves a failure of the operating ESW pump, failure to recover ESW within 50 min, successful SSF operation, and failure of the turbine-driven AFW pump for secondary-side heat removal.

This event was considered to be two precursor events since it affected both Catawba 1 and 2.

3.2.2 LaSalle 1 (LER 373/93-015)

LaSalle 1 was operating at 100% power on September 14, 1993, when a fault occurred in the buswork associated with the system auxiliary transformer (SAT). The resulting electrical system perturbations caused the loss of one main feed pump and a reactor scram on low vessel level. Reactor makeup after the scram was initially supplied by the motor-driven reactor feed pump, but the vessel overfilled, resulting in feed pump and main turbine/generator trips. Once the main generator separated from the grid, the unit auxiliary transformer was deenergized and the plant experienced a LOOP. The emergency diesel generators (EDGs) started and loaded to supply the emergency buses. The high-pressure core spray (HPCS) diesel also started. safety relief valves (SRVs) were operated to reduce pressure by relieving steam to the suppression pool. Suppression pool cooling (SPC) was initiated and reactor core isolation cooling (RCIC) was aligned for vessel makeup. After about 75 min, offsite ac power was restored to Unit 1 by connecting Unit 1 buses to Unit 2. Late in the event, one SRV failed to operate on demand. When reactor pressure decreased to 500 psig, the low-pressure core spray (LPCS) system was aligned to provide makeup, and the reactor was then placed in shutdown cooling (SDC).

The conditional probability of subsequent core damage for this event is estimated to be 1.3×10^{-4}. The dominant core damage sequence involves the plant-centered LOOP, a postulated failure of emergency power, successful reactor shutdown, and postulated failure to recover emergency power before battery depletion.

3.2.3 Perry (LER 440/93-010)

When the Perry suppression pool was inspected in May 1992, an accumulation of dirt and debris was noticed on the suction strainers for residual heat removal (RHR) trains A and B. Strainer cleaning was scheduled for a later date, because RHR system performance was considered acceptable based on surveillance testing.

The suppression pool strainers were inspected again and cleaned during a maintenance outage in January 1993. RHR train A and B suction strainers were found to be deformed, with the area of the strainer surface between internal stiffeners partially collapsed inward, in the direction of system flow. It was determined that the strainers were deformed by excessive differential pressure caused by strainer fouling during normal pump operation. Review of a videotape taken during the May 1992 inspection revealed evidence of deformation that had not been noticed at the time of the taping. The containment side of the suppression pool was inspected and cleaned in February 1993, and the deformed strainers were replaced.

On March 26, 1993, the reactor was scrammed following the rupture of a 30-in. SW line. Condenser vacuum was lost after the loss of SW, which required closure of the main steam isolation valves. The RCIC system was used for pressure vessel makeup, and both trains of the RHR system were started for suppression pool cooling (SPC). After shutdown cooling was established using RHR train A, RHR train B continued to provide SPC for an additional 5 h. Then, RCIC was secured and the control rod drive system was used for level control. The A CRD pump experienced minor cavitation due to loss of suction.

The total volume discharged through the SW break was ~1.7 million gallons. About 5% of the total leakage entered numerous plant buildings, accumulating in the lowest level of the auxiliary building and control complex, where safety-related equipment is located. Although no safety-related equipment was impacted by the flood, water entered multiple plant buildings that would normally be considered independent structures in an internal flooding analysis.

If SW had not been secured, continued flooding of the auxiliary building and control complex could have resulted in damage to ECCS components. During the actual event, the HPCS pump motor was wetted by water dripping from a ceiling hatch plug; however, the pump was not damaged. The lack of detailed information concerning equipment locations and flood pathways prevented consideration of potential flooding effects in the analysis. However, sensitivity analyses were performed to bound the potential effects of the flood.

On April 14, 1993, all emergency core cooling system (ECCS) strainers were inspected. The RHR train B strainer was fouled and deformed in a manner similar to that observed during the January 1993 inspection. The remaining strainers showed no signs of fouling. Without disturbing the debris on the strainer, a test run of RHR pump B was performed. The pump running suction pressure decreased to 0 psig after operating for 8 h, and the pump was secured. The pump suction strainer was then inspected. The debris from the strainer was analyzed, and it was determined that the debris contained fibrous material and corrosion products. The predominant fibrous material was glass fiber from roughing filter material used in the drywell air cooler system. The RHR strainer provided a structural framework for a uniform covering of the fibrous material, which in turn acted as a filter for suspended solids that would have otherwise passed through the strainer.

The licensee inspected and cleaned the containment and the suppression pool following the discovery of the clogged strainers and did not identify large quantities of the fibrous material. Based on this, the licensee concluded that there was no chronic degradation of the properly installed filter media. Instead, the licensee concluded that the fibrous material entered the suppression pool as intact pieces as a result of installation or maintenance activities (the roughing filters are normally replaced before startup from refueling outages). These pieces subsequently broke down to fibers once in the suppression pool. The actual time when the material entered the suppression pool could not be determined.

Excessive differential pressure across the RHR strainers from debris accumulation would fail SPC and could fail LPCI if it was required to operate for long periods of time. The event was modeled as an unavailability of RHR/SPC following (1) postulated initiators in the 1-year period before discovery of the clogged strainers and (2) the reactor trip following the SW pipe rupture on March 26, 1993.

The conditional core damage probability estimated for this event is 1.2×10^{-4}. The dominant core damage sequence involves a scram with PCS and FW unavailable following the SW pipe rupture, HPCS success, failure of long-term decay heat removal via the RHR system, and failure to vent the containment. The results of the sensitivity analysis to address potential flooding effects indicate that potential flooding effects do not significantly contribute to the overall event. If the HPCS pump motor had been damaged by the water that dripped from the ceiling hatch, the estimated core damage probability could have been substantially greater than the 1.2×10^{-4} estimated for the event.

3.3 Number of Precursors Identified

Sixteen precursors [p(core damage) $\geq 10^{-6}$] affecting 16 units were identified in 1993. The distribution of precursors as a function of conditional probability is shown in Table 3.6. The distribution of 1988–1992 precursors is also shown for comparison purposes.

Table 3.6 Number of Precursors

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
1988	0	7	14	11	32
1989	0	7	11	12	30
1990	0	6	11	11	28
1991	1	27	8	6	27
1992	0	7	7	13	27
1993	0	4	7	5	16

As described previously, differences in the ASP models and the analysis methods from year to year preclude a direct comparison between the number of events identified for different calendar years. In particular, the conditional core damage probabilities estimated for the 1992 and 1993 events are lower for equivalent events in earlier years because supplemental and plant-specific mitigating systems beyond those included in the ASP models were incorporated into the analyses.

3.4 Insights

3.4.1 Likely Sequences

Precursors with conditional probabilities of $\ge 10^{-4}$ that were identified for 1993 were reviewed to determine the most likely core damage sequences associated with each event. These sequences include the observed plant state plus additional postulated failures required for core damage. For the events that occurred or could have occurred at power and with core damage probabilities of $\ge 10^{-4}$, the following dominant core damage sequences were identified:

PWRs —based on two events (the one Catawba event that affects both units)

- Failure of the running ESW pump, failure to recover ESW within 50 min., and failure of the safe shutdown facility

BWRs— based on two events

- plant-centered LOOP with failure of emergency power that is not recovered before battery depletion
- transient with FW and PCS unavailable, failure of long-term decay heat removal, and failure to vent containment

3.4.2 Observations

A review of the analyses for all 16 precursors for 1993 revealed the following trends and patterns across the different analyses.

- As can be seen in Tables 3.3 and 3.4, two of the four precursors with p(cd) $\geq 10^{-4}$ selected for 1993 are pressurized-water reactor (PWR) events. For all 1993 precursors, 12 were associated with PWRs and 4 with boiling-water reactors (BWRs).

- A number of the events involved problems with electrical systems. LOOP events occurred at four plants. These four events had conditional core damage probabilities that ranged from 1.3×10^{-4} to 4.6×10^{-6}. The range in the conditional core damage probabilities for these events is primarily due to the type and number of mitigating systems incorporated into the models beyond the normal ASP models. For example, in the Pilgrim LOOP (293/93-004) with a conditional core damage probability of 4.6×10^{-6}, the inclusion of a blackout diesel generator and an offsite power line that is used only after EDG failure resulted in a significant decrease in the conditional core damage probability from the base ASP value.

  Three of the precursors associated with unavailabilities involved the degradation or unavailability of electrical equipment: (1) the degradation of the bus transfer scheme for motor control center 5 and the EDGs at Haddam Neck, (2) the degradation of the emergency load sequencers at Beaver Valley 2, and (3) the loss of the diesel generator cooling water pumps at Quad Cities Unit 2.

- The precursors are evenly divided between unavailabilities and initiators. The distribution of the events by conditional core damage probability in the two categories is roughly the same.

Number of Precursors

Event Category	$10^{-4} \leq p(cd) < 10^{-3}$	$10^{-5} \leq p(cd) < 10^{-4}$	$10^{-6} \leq p(cd) < 10^{-5}$	Total
Unavailabilities	2	4	2	8
Initiators	2	3	3	8

- Seven of the eight precursors associated with unavailabilities occurred at PWRs. The precursors associated with the initiating events were roughly evenly divided between the PWRs (5 events) and BWRs (3 events).

- Twelve of the sixteen events (75%) occurred at multi-unit sites. This is about the same as the percentage of units at multi-unit sites (71%). Only one of the precursor events affected both units at a dual-unit site.

A review of the ASP reports for 1990–1993 indicates the following trends and patterns.

- Long-term unavailabilities and LOOP initiators typically dominate the events with the highest conditional core damage probabilities.

- The events with the highest conditional core damage probabilities are dominated by PWRs.

- The number of precursors identified for 1993 is lower than for previous years. This decrease is due in part to the differences in the ASP models for 1993. In particular, the conditional core damage probabilities estimated for the 1993 events are lower than equivalent events in earlier years because of consideration of supplemental and plant-specific mitigating systems beyond those modeled in the ASP models. A number of events that would have met the precursor criteria for prior years were rejected on low probability following the incorporation of additional mitigating systems.

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 the occurrence of an accident sequence precursor, 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. Availability is the complement of unavailability.

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 certain conditions.

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 systems. 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 *nonrecovery 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 term used to describe a failure resulting in a plant response that is apparent at the time of the failure.

Independence. A condition existing when two or more entities 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 NRC by utilities who operate nuclear plants as described in NUREG-1022. LERs describe abnormal operating occurrences that generally involve violation of the plant's Technical Specifications.

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

Nonrecovery factor (recovery class). See *Recovery factor.* Recovery and nonrecovery are used interchangeably throughout this report.

Operational event. An event that occurs in a plant and generally constitutes a reportable occurrence under NUREG-1022 as an LER.

Postulated event. An event that may happen at some time in the course of a plant's operation.

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 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 system, its associated turbine generator, auxiliaries, and engineered safety features.

5 References

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Appendix A:

Precursors

A.0 Precursors

A.0.1 Accident Sequence Precursor Program Event Analyses for 1993

This report documents 1993 operational events selected as accident sequence precursors.

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.

Details of the precursor review, analysis, and documentation process are provided in Vol. 19 of this report (*Precursors to Potential Severe Core Damage Accidents: 1993, A Status Report*, NUREG/CR-4674, Vol. 19).

A.0.2 Precursors Identified

Sixteen precursors were identified among the 1993 LERs reviewed at the Nuclear Operations Analysis Center. Events 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 at least 10^{-6}:

- 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 precursors identified are listed in Table A.1.

Table A.1 List of ASP Events

Event No.	Plant	Event description	Page
213/93-S01,* 213/93-006, 213/93-007	Haddam Neck	Degradation of MCC-5, Pressurizer PORV, and Emergency Diesel Generators	A.1-1
265/93-010, 265/93-012	Quad Cities 2	Emergency Power System Unavailable	A.2-1
289/93-002	Three Mile Island 1	Both Residual Heat Removal Heat Exchangers Unavailable	A.3-1
293/93-004	Pilgrim	Weather-Induced Loss-of-Offsite Power, Pressure Vessel Pressure/Temperature Limits Violated	A.4-1
313/93-003	Arkansas Nuclear One, Unit 1	Both Trains of Recirculation Inoperable for 14 h	A.5-1
316/93-007	Cook 2	Reactor Trip with Degraded Auxiliary Feedwater	A.6-1
334/93-013	Beaver Valley 1	Dual-Unit Loss-of-Offsite Power	A.7-1
339/93-002	North Anna 2	Auxiliary Feedwater Disabled After Reactor Trip	A.8-1
370/93-008	McGuire 2	Loss-of-Offsite Power and Failure of an MSIV to Close	A.9-1
373/93-015	LaSalle 1	Scram and Loss-of-Offsite Power	A.10-1
412/93-012	Beaver Valley 2	Failure of Both Emergency Diesel Generator Load Sequencers	A.11-1
413/93-002	Catawba 1 & 2	Essential Service Water Potentially Unavailable	A.12-1
440/93-011, 440/93-010	Perry	Clogged Suppression Pool Strainers and Service Water Flood	A.13-1
498/93-005, 498/93-007	South Texas Project 1	Unavailability of One Emergency Diesel Generator and the Turbine-Driven Auxiliary Feedwater Pump	A.14-1
529/93-001	Palo Verde 2	Steam Generator Tube Rupture	A.15-1

*AIT Report 213/93-80.

A.0.3 Event Documentation

Analysis documentation and precursor calculation sheets (if applicable) for each precursor are attached. The precursors are in docket/LER number order.

For each precursor, 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. Two figures are [illegible] included. The first figure compares the significance of the event from a core damage standpoint with other potential events at the same plant. The other potential events at the same plant are briefly described below:

	PWR & BWR
Trip	Trip with equipment operable
LOOP	Loss-of-offsite power. Includes plant-centered
360 h EP	360 h without emergency power sources (normally on-site emergency diesel generators)
	PWR
LOFW + 1MTR AFW	Transient with loss of main feedwater (LOFW) and one motor-driven auxiliary feedwater (AFW) or emergency feedwater (EFW) pump failed (turbine-driven pump substituted if plant does not have any motor-driven pumps)
360 h w/o AFW	360 h with all AFW (or EFW) pumps failed
	BWR
360 h w/o HPCI and RCIC	360 h with high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) failed (not applicable for Type A BWRs)
LOFW and HPCI	Transient with loss of main feedwater and HPCI (loss of main FW and loss of Isolation Condenser is run instead for Type A BWRs)

The second figure highlights the dominant core damage sequence associated with the event. A conditional core damage calculation is also provided if applicable.

A.1 AIT No. 213/93-80, LER Nos. 213/93-006 and -007

Event Description: Degradation of Motor Control Center 5, Pressurizer Power-Operated Relief Valve, and Emergency Diesel Generators

Date of Event: June 27, 1993

Plant: Haddam Neck

A.1.1 Summary

In part as a result of the Haddam Neck Individual Plant Examination (IPE), a decision was made by the licensee to perform an integrated test of the automatic bus transfer (ABT) scheme for motor control center (MCC) MCC-5. Such a test had never been performed, although some of the individual components had been tested. Several important systems are directly dependent on MCC-5 for their operation, including both trains of high- and low-pressure safety injection (HPSI and LPSI) and both normally closed pressurizer power-operated relief valve (PORV) block valves. On June 27, 1993, the first test of the ABT scheme for MCC-5 was unsuccessful. MCC-5 was without power until an operator took local action to close a breaker. On May 25, 1993, one of the pressurizer PORVs was found to have an air leak that would drain the air receiver if feed-and-bleed were initiated. Also on May 25, 1993, the licensee was performing a 24-h endurance run of the "A" emergency diesel generator (EDG). After 22 h the EDG exhibited abnormal kilowatt, kilovar, and ampere indications that led to the termination of the test. Components in the exciter control cabinet had failed due to overheating. The exciter control cabinet for the "B" EDG was also susceptible to this failure mode. The conditional core damage probability of this combined event is 6.5×10^{-5}. The relative significance of this event compared to other postulated events at Haddam Neck is shown in Fig. A.1.1.

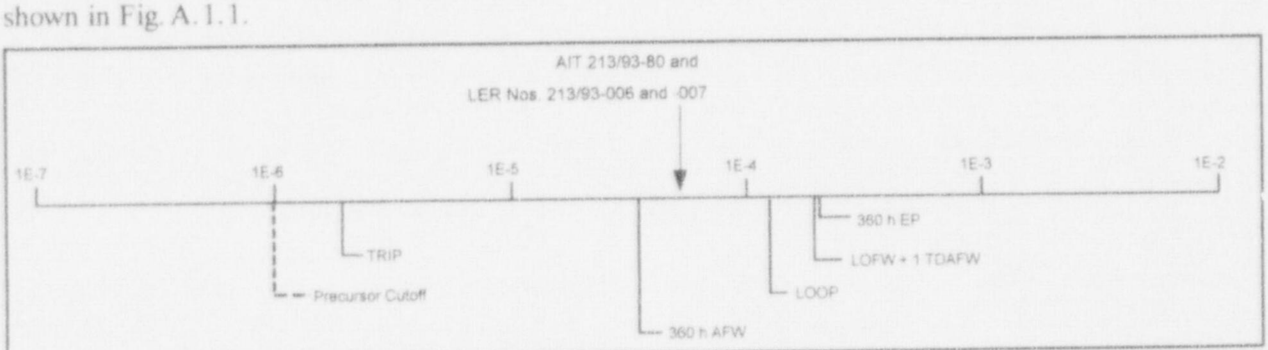


Fig. A.1.1 Relative event significance of AIT 213/93-80 and LER Nos. 213/93-006 and -007 compared with other potential events at Haddam Heck

A.1.2 Event Description

This analysis includes the effects of several failures and equipment degradations noted in separate licensee event reports (LERs). Each event is described separately. Additional event-related information is included for each event individually. The modeling assumptions and results include the effects of all of the events.

A.1.2.1 Loss of MCC-5 (AIT Report 213/93-80)

In part as a result of the Haddam Neck IPE, a decision was made by the licensee to perform an integrated test of the ABT scheme for MCC-5. Such a test had never been performed, although some of the individual components were previously tested. At the time of the event, MCC-5 supplied electrical power for all of the HPSI and LPSI valves needed to perform emergency core cooling. Failure of this single MCC to provide power to these valves would result in a total loss of safety injection (SI) capabilities. Many other important systems also relied on the operation of MCC-5, such as the normally closed PORV block valves and instrument air (see the Additional Event-Related Information section for a description of the loads served by MCC-5 at the time the event occurred). Changes to the plant design since this event are described in Section A.1.3.3.

Because of its importance, MCC-5 is provided with an ABT scheme to ensure power to the MCC. At the time of this event, this scheme aligned the bus to the preferred power source (manually selected) if it was available. If the preferred source was not available, it would switch to the alternate supply. If power was subsequently restored to the preferred source, the scheme would realign the supply back to the preferred source even if the alternate power supply was still available. Before the first integral test described below, portions of the ABT scheme had never been tested.

On June 27, 1993, the first integral test of the ABT scheme for MCC-5 was being conducted. The test was divided into two portions. During the first portion of the test, bus 5 was selected as the preferred source (see Fig. A.1.2 for an electrical distribution diagram). Bus 5 was always selected as the preferred bus unless the "A" EDG and its associated safeguards train were inoperable for an extended period of time. This first portion of the test was successful. When the preferred source (bus 5) was de-energized, its supply breaker to MCC-5 (9C) opened and the supply breaker (11C) from the alternate source (bus 6) closed as expected. When power was restored to the preferred source (bus 5), breaker 11C opened, and breaker 9C closed. For the second portion of the test, bus 6 was the preferred power source. When power was removed from bus 6 at 1848 hours, MCC-5 switched to the alternate source (bus 5) that was being supplied from offsite power via T-389. This was the expected system response. When the 2B EDG restored power to the preferred source (bus 6), the bus 5 breaker (9C) opened as expected, but the bus 6 breaker (11C) did not close. This left MCC-5 without a power source. In an attempt to restore power to the MCC from bus 5, the operator located at the ABT in the switchgear room moved the preferred power source selector switch (PPSSS) to the bus 5 position. However, breaker 9C did not close. The operator attempted to close the breaker by pushing the manual close button on the breaker, but this was also unsuccessful. Subsequently, the operator was able to mechanically close the breaker using a portable operating handle that re-energized MCC-5 from bus 5 at 1852 hours. The root cause of the failure was initially suspected to be either an Agastat timing relay or a relay for breaker 11C, which is an integral part of the Westinghouse DB 25 breakers.

The plant returned to power on July 17, 1993, and ran continuously until February 13, 1994. During this time, periodic tests of one of the relays (52X) for breaker 9C were conducted. This was done by removing control power from the breaker and visually verifying the operation of the relay. All of these tests were successful.

Following the plant shutdown on February 13, 1994, the second integrated test of the ABT was conducted on February 16, 1994, at 0130 hours. The ABT failed again during this test. In this case, the first portion of the test was again successful. With bus 5 as the preferred source, bus 5 was de-energized; breaker 9C opened, and breaker 11C closed as expected. When power was restored to bus 5, breaker 11C opened, and breaker 9C closed. To begin the second part of the test, the PPSSS was switched from the bus 5 position to the bus 6 position. When the ABT realigned the supply to the new preferred source, breaker 9C opened, but breaker 11C did not close as expected. An operator manually closed breaker 9C to restore power to MCC-5.

Following the February 1994 failure, a root cause investigation determined that the failure was attributable to a mispositioned snap ring on the breaker manual closing shaft. The snap ring was found ~ 0.5 in. from its groove. With the snap ring out of position, the manual operating shaft was unrestrained from travel. The cam-end of the manual closing shaft was found resting on the edge of the trip plate. This condition results in a trip if the breaker is closed, and if the breaker is already open, it prevents it from closing.

With the snap ring out of position, the breaker may operate differently under different circumstances. In addition, the manual closing shaft is free to move when the breaker is in the closed position. If the breaker is manually closed with the closing handle, this action moves the cam-end of the closing shaft such that it no longer rests on the trip plate. Following this, the breaker can perform an undetermined number of successful operations electrically before the closing handle cam-end begins to interfere with the trip plate and places the breaker in a trip-free condition once again. Therefore, the failure mechanism is intermittent. Additional cycling of the breaker during testing increases the likelihood that the breaker will fail in subsequent demands.

As a result of the root cause investigation following the February 1994 failure, the root cause investigation of the June 1993 failure was reviewed. It was concluded that the root cause of the June 1993 event was also the mispositioned snap ring and not the Agastat relay and the 52X closing relay for breaker 11C as earlier thought. This new failure mechanism (the mispositioned snap ring) accounted for the observed operation of the breaker during the testing in June 1993.

During the June 1993 test, the failure of breaker 11C to reclose also prevented breaker 9C from reclosing. However, under actual loss-of-offsite power (LOOP) conditions, breaker 9C would reclose following a failure of breaker 11C. During the June 1993 test failure, the preferred power source was bus 6. Following the failure of breaker 11C to close, the operator moved the PPSSS to the bus 5 position, but breaker 9C did not close. This was due to the fact that between the opening of breaker 9C and the attempted reclosure, breaker 11C did not close, and bus 5 had not lost power. This prevented the 52X closing coil for breaker 9C from resetting and prevented the breaker from closing on antipump protection. However, under actual LOOP conditions, the PPSSS would be aligned to the bus 5 position, and both buses (5 and 6) would lose power. Following the opening of both breakers 9C and 11C, 11C will attempt to close if bus 6 is repowered before bus 5. If 11C fails to close, 9C will close if power is subsequently restored to bus 5. In this case the 52X relay for breaker 9C will have been reset by the loss of power to bus 5. Therefore, the root cause of the event is consistent with the observed behavior of the system during both the 1993 and 1994 events.

A.1.2.2 Pressurizer PORV Air Receiver Leak (LER 213/93-007)

On May 25, 1993, the licensee discovered that the air receiver pressure for the PORVs decayed faster than allowed by Technical Specifications. The air receiver leak was traced to a leak in the diaphragm assembly of one of the PORVs. The leak was noticed during a visual inspection of the valve. The licensee did not measure the leak rate, but estimated that the accumulator would deplete within an hour. The leak was caused by both the inadequate sealing of the PORV diaphragm assembly and the failure of the PORV air supply pressure regulating valve. The root cause of the problem was not conclusively identified. However, the licensee stated that the problem probably existed since the last refueling outage and that the valve would have failed during the current operating cycle.

The air receiver is used to provide air pressure for the PORVs for feed-and-bleed cooling after the loss of the nonsafety-related containment air compressors. These air compressors are located in containment. The licensee indicated that the containment air compressors would be expected to run for some period of time following the initiation of bleed-and-feed. But, since the compressors are not qualified for the post-feed-and-bleed environment, it is not known how long they would operate.

A.1.2.3 EDG Fails During 24-h Endurance Run (LER 213/93-006)

On May 25, 1993, the licensee was performing a 24-h endurance run of the "A" EDG. After 22 h the EDG exhibited abnormal behavior. The EDG was shut down, and the causes were investigated. The investigation revealed a lack of cooling to the EDG exciter control cabinet because (1) the cooling fan for the cabinet had failed and (2) the fan had run continuously with no filter on the input air, resulting in a large accumulation of dust that reduced the heat transfer abilities of the components in the cabinet. The lack of cooling led to the overheating of two rectifiers in the control circuitry. This, in turn, caused abnormal field voltage and ultimately a loss of generator field. The exciter control cabinet for the "B" EDG was also examined, and it, too, could have failed due to lack of cooling. The cooling fan for its cabinet was also inoperable, and the dust level in the "B" control cabinet was similar to that found in the cabinet for the "A" EDG. The "B" EDG was declared inoperable until the rear exciter control cabinet covers were removed to provide adequate ventilation. The control cabinet for the "B" EDG was cleaned before the endurance run for the "B" EDG.

The LER noted that the 24-h test exceeded the accident requirements in both real and reactive power levels. After an SI actuation condition coincident with a LOOP, the EDG load is expected to be 2850 kW. After a LOOP only, the EDG load is expected to be 2200 kW. This test was conducted with an EDG load of 2850 kW. It was noted that the power factor was always much better during actual operation (i.e., closer to one) than during the 24-h test. Under accident conditions, the power factor is expected to be 0.9; during testing, the power factor was 0.85. A better (higher) power factor requires less field current and thus generates less heat in the exciter control cabinet. Due to the lower cabinet temperatures expected during actual operation, the licensee stated that the EDG was expected to perform its intended function well beyond the successful test period of 22 h. If both EDGs operate post-loss-of-coolant accident (post-LOCA), then the load on the EDGs would be even lower and, therefore, would result in lower exciter control cabinet heat loads.

A.1.3 Additional Event-Related Information

A.1.3.1 Loss of MCC-5

A simplified diagram of the MCC-5 power supplies is shown in Fig. A.1.2. There are two sources of offsite power to the safeguards buses via transformers T-389 and T-399. A switchyard breaker, 389T399, allows cross connection of the two incoming lines. Power to the 4160-V ac safeguards buses, buses 8 and 9, is from buses 1-2 and 1-3. These can be cross connected to allow feed from a single offsite power transformer. In addition to the feeds from buses 1-2 and 1-3, each safeguards bus has its own EDG. The 480-V ac buses (buses 5 and 6) are supplied from buses 8 and 9, respectively. They, in turn, supply MCC-5 via breakers 9C and 11C.

MCC-5 is important to the operation of a number of mitigation systems. The following is a list of the functions that would have been completely inoperable without power supplied to MCC-5.

Front-Line Systems:

- HPSI

  Normally closed loop injection valves

- LPSI

  Normally closed injection (core deluge) valves

- Sump Recirculation

Normally closed sump suction valve (parallel manual valve would be operable)

- Long-Term Cooling

  Normally closed motor-operated valves (MOVs) inside containment and in the auxiliary building

- Containment Spray
- Reactor Coolant System (RCS) Loop Isolation Valves
- Pressurizer Relief

  Both PORV block valves

- Emergency Boration
- Feedwater Isolation (feedwater regulating valves are not affected)
- Reactor Coolant Pump (RCP) Seal Cooling

  MOVs on the charging line would have to be manually throttled to ensure sufficient flow to the RCP seals. The main lube oil pumps for the charging pumps would be lost, and the auxiliary lube oil pumps would have to be manually started. Cooling water is also lost to the RCPs seals unless instrument air is restored via service air (this is not possible following a LOOP).

Support Systems

- Service Water (SW)

  When MCC-5 fails and only one of two EDGs starts, insufficient SW flow to the running EDG may cause EDG failure unless two SW MOVs are manually closed to isolate SW flow to nonsafety-related loads.

- Control Air Compressors
- Closed Cooling Water

Systems that require the operation of the support systems dependent on MCC-5 would also fail or be degraded. For example, the auxiliary feedwater (AFW) flow control valves could not be operated remotely after the loss of instrument air.

Because both trains of several safety systems are dependent upon MCC-5 for power, an ABT scheme was installed to ensure a continuous power supply for MCC-5. One of the 480-V ac buses (5 or 6) is designated as the preferred source and the other the alternate. With bus 6 as the preferred source (as it was during the June 1993 test failure), loss of voltage on bus 6 will trip open breaker 11C. If bus 5 is energized, breaker 9C will close after breaker 11C opens to supply power to the MCC from bus 5. If there is no voltage on bus 5, breaker 9C will not close. When power is restored to bus 6, breaker 9C opens (if closed), and breaker 11C recloses. In the loss of MCC-5 event on July 27, 1993, breaker 11C failed to reclose. However, the PPSSS is normally in the bus 5 position. In this case, the preceding description of the switching scheme is the same except the breakers and buses are switched. If the preferred bus is re-energized before the alternate source, then the breaker for the preferred source simply recloses. The breaker for the alternate source will not cycle in this case.

The ABT would be expected to operate in any situation where power is lost to the preferred power source. The ABT will close in on the first bus that is energized. Because no substantial bias is built into the system to favor one bus over the other, there is an essentially equal likelihood that the system will close in on the alternate power supply or the preferred power supply, provided restoration of power to both buses occurs at about the same time (as would be expected following EDG start and load).

Postevent investigations, although inconclusive, initially attributed the failure of the ABT in June 1993 to relays associated with breakers 9C and 11C. After the February 1994 event, the root cause of that event was attributed to a split ring in breaker 11C. The Nuclear Regulatory Commission (NRC) also noted that the split ring was apparently mispositioned for quite a period of time based on the amount of dust on the manual close linkage lubrication. It appears that the June 1993 event was also caused by the mispositioned split ring. Most likely, the ring was moved out of position during vendor breaker maintenance.

Following the failure of the ABT, one possible recovery action would involve manually closing the supply breaker from an energized bus. For example, following the failure of breaker 11C to automatically close and the failure of the "A" EDG, power could be restored to MCC-5 by manually closing breaker 11C.

A review of the Haddam Neck LERs back to 1984 was conducted to determine if the ABT had successfully switched from bus 5 to bus 6 and back (the bus 6 breaker cycled) in prior events. Four LOOP events were found (LER Nos. 213/93-010, 93-009, 84-014, 84-009). Two of the events occurred shortly before the unsuccessful ABT test in June 1993. Both of these events occurred with the plant at refueling shutdown conditions during testing of relay schemes for offsite power restoration. It is unclear from the LERs whether the breaker for bus 6 cycled in these events because the order of EDG breaker closures is not mentioned. Discussions with the licensee indicated that it would be not be possible to determine the sequence of operation for the ABT during these transients. The two other events occurred in 1984. In one event (LER 213/84-014), a LOOP occurred during refueling shutdown when a circulating water pump was started. In this case one of the EDGs did not close in on its associated bus for 20 min. However, the LER does not state which EDG this was, so it is not possible to determine if the bus 6 breaker cycled. In the remaining event, the bus 6 breaker must have cycled because all offsite power was lost, and power was restored to bus 6 a few seconds before power was restored to bus 5. From this information it appears that the system successfully cycled the bus 6 breaker in 1984, but it is unclear if the subsequent LOOPs demanded a cycling of the bus 6 breaker. It should be noted that a loss of power to bus 5 with bus 6 remaining energized would also have required the ABT to switch power supplies from bus 5 to bus 6 and back to bus 5. However, this type of event would not typically be reportable to the NRC.

A.1.3.2 Pressurizer PORV Air Receiver Leak

The pressurizer PORV air receivers are required to be used for feed-and-bleed cooling after the loss of the nonsafety-related containment air compressors. Loss of the containment air compressors is expected in the environment created by the feed-and-bleed process, because they are not qualified for that environment. Feed-and-bleed is critical to meeting the probabilistic risk assessment core melt frequency goal and is credited following the loss of main feedwater and AFW.

The procedures for the feed-and-bleed process direct the operator to open one of the two pressurizer PORVs. However, it does not specify which one to open. If the operator opened the faulty valve first, the relief valve for the PORV would fail open due to a faulty pressure regulator. The air in the accumulator would then be bled down until the low accumulator air pressure signal alarmed. When the annunciator for low accumulator air pressure alarms, a caution in the operating procedures indicates that this may be indicative of an accumulator leak. The operator is directed to close the open PORV and open the remaining PORV. This action would isolate the leak path. If the operator faied to do this, the air accumulator would be vented through the faulty relief valve until it was below the pressure required to maintain the valve in an open position or to open the redundant valve. In addition, this would allow for a reduced number of cycles of the remaining valve.

The pressurizer block valves are normally kept in the closed position at Haddam Neck. Therefore, feed-and-bleed success also depends on the successful opening of the block valves. The PORV block valves were both powered from MCC-5 at the time of the event. Therefore, failure of MCC-5 would also prevent initiation of bleed-and-feed. Because the block valves are normally closed, the PORVs would also not be available if required for RCS pressure relief in the first few moments of a transient such as a LOOP.

A.1.3.3 Plant Design Changes After the Event

The licensee implemented several design changes that reduced the dependence on MCC-5 after the events addressed in this analysis. These changes included powering one residual heat removal (RHR) to charging pump suction valve, the main lube oil pump for charging pump A, and one PORV block valve from MCC-12 instead of from MCC-5. In addition, PORV PR-AOV-570 was repowered from a semivital panel to a vital panel. The preferred source selector switch was also removed from the ABT.

A.1.4 Modeling Assumptions

Events of concern in this analysis are situations in which (1) the ABT for MCC-5 is required to operate and the systems that rely directly or indirectly on this MCC are required to function, (2) the EDGs are required to operate for an extended period of time, or (3) the PORVs are required for feed-and-bleed. During trip and LOCA scenarios, the EDGs are not required, and the equipment that is dependent on MCC-5 is expected to operate, because ABT operation will not be required. For these initiating events, only the potential unavailability of the PORVs is of concern. During a partial loss of power to one of the safeguards buses, the ABT will be challenged, but it was assumed that the dependent equipment would not have to operate because the plant would not trip under these conditions. Only under a LOOP initiator would both the ABT and the dependent equipment need to operate. For a postulated LOOP, the potential failure of MCC-5, the EDGs after 22 h, and feed-and-bleed must be addressed.

Long-term unavailabilities have typically been modeled in the ASP Program for a 1-year period, assuming that the plant was at-power for 70% of the time; this is equal to 6132 h (365 d × 24 h/d × 0.7). This value was utilized for these combined events, although the exact time and cause of the failures are unknown in a number of cases. It is unclear when the MCC-5 ABT failure occurred. However, it appeared that the snap ring was mispositioned for quite some time as the result of vendor breaker maintenance. It was not known how long the pressurizer PORV air supply leak had existed. However, the licensee believes that the leak had existed since the previous refueling outage. The duration of the degraded EDG condition could not be determined. However, the failure was due to an accumulation of dust over time and the inoperability of the cooling fan.

The effects of the failures described above on ASP model event tree branches are discussed below.

A.1.4.1 MCC-5 Failure and Restoration

Based on the condition of breaker 11C and the unpredictability of its observed failures, breaker 11C was assumed to be failed in this analysis. In addition to the failure of breaker 11C, one additional failure must occur for MCC-5 to lose power. Either breaker 9C must fail to reclose, or EDG "A" must fail to start and run.

After a LOOP with the PPSSS in the bus 5 (normal) position, power would be lost to buses 5 and 6. Two cases could then occur.

- Bus 5 is re-energized before bus 6. In this case, breaker 9C will attempt to reclose. If 9C fails to close, the ABT will automatically try to close breaker 11C once bus 6 is energized. However, since breaker 11C is assumed to be failed, manual operator action is required to restore power to MCC-5.

- Bus 6 is re-energized before bus 5. In this case, breaker 11C will attempt to close. Assuming 11C fails to close, the ABT will attempt to automatically reclose breaker 9C after power is restored to bus 5. If breaker 9C fails to reclose, manual operator action is required to restore power to MCC-5. Data collected by the licensee on EDG performance indicate that the time to rated speed and voltage for both of the EDGs was essentially the same. This would mean that bus 6 would reach rated voltage first about 50% of the time. (Circuit time delays may affect this value somewhat, but would have little impact on the analysis results.) Assuming that breaker 11C will fail to close on demand, and a beta factor of 0.1 for breaker 9C, the probability of failure of the ABT given a LOOP is:

$$[p(5before6) \times p(9C \mid 11C)] + [p(\overline{5before6}) \times \{p(EDG\ A) + p(9C \mid 11C)\}] =$$

$$[p(5before6) \times p(9C \mid 11C)] + p(\overline{5before6}) \times p(EDG\ A) + p(\overline{5before6}) \times p(9C \mid 11C) =$$

$$p(9C \mid 11C) + p(\overline{5before6}) \times p(EDG\ A) =$$

$$0.1 + (0.5 \times 0.05) = 0.125\ *$$

* For situations where offsite power is recovered in the short term, the probability for MCC-5 failure is 0.1.

The licensee performed a detailed analysis of this event. Their assessment indicates that the probability that MCC-5 fails to supply power is 0.059 for LOOP events. However, this assumed a nominal failure rate for breaker 11C.

To recover MCC-5 following a failure of the ABT, an operator must proceed to MCC-5, diagnose the situation, and manually close one of the MCC-5 feeder breakers. During the June 1993 event, it took the operator 4 min to complete this action. However, the operator was already stationed at the selector switch, was immediately aware of the ABT failure, and had a minimum of other distractions and stresses.

Following a postulated LOOP with failure of MCC-5, additional delays would be introduced, including detection time (unavailability of power on MCC-5 is not directly addressed in procedure E-0, "Reactor Trip or Safety Injection," until step 16), delays for the control room to contact an auxiliary operator and describe the problem, and operator transit time. A median value of 10 min was used in the analysis; this assumes 6 min for diagnosis and transit time and the observed 4 min for recovery at the equipment. A 6-min diagnosis and transit time is considered possible because of the proximity of MCC-5 to the control room. [The 10-min median value is somewhat longer than the licensee's estimate of 5 to 6 min (2 to 3 min diagnostic time, 1 min transit, and 2 min to operate breakers), and somewhat shorter than a 16-min value that can be estimated based on a distribution of transit times in response to a faulted EDG (another important component) included in "Electric Power Recovery Models," J. W. Reed and K. N. Fleming, *Proceedings of the International Topical Meeting On Probabilistic Safety Assessment*, PSA'93, January 26–29, 1993.]

The probability of not recovering MCC-5 was estimated by assuming that the 10-min period was the median of a lognormal distribution with an error factor of 3.2 (see Dougherty and Fragola, *Human Reliability Analysis*, John Wiley and Sons, New York, 1988, Chap. 10). This is the error factor for time-reliability correlations (TRC) for actions without hesitancy, which is considered appropriate based on the recognized importance of MCC-5. Three primary time intervals for MCC-5 recovery were considered in this analysis. These intervals and the associated MCC-5 nonrecovery probability follow:

Time interval	p(MCC-5 not recovered)
0.5 h	6.0×10^{-2}
1.0 h	5.6×10^{-3}
1.5 h	9.5×10^{-4}

A.1.4.2 EP—Emergency Power

Based on the observed failure of the "A" EDG 22 h into the surveillance test and the similar condition of the "B" EDG, it was assumed that both EDGs were vulnerable to failure due to exciter cabinet overheating after starting in response to a LOOP. Failure of the running EDGs would result in loss of emergency power if offsite power had not been recovered by that time.

Because of the nature of the observed failures, the precise time to failure cannot be estimated. It is also likely that the EDGs will not fail at the same time. If the EDG failures are reasonably separated in time, then the cause of the first failure would be expected to be found and corrective actions taken to prevent failure of the remaining EDG.

Based on information provided by the licensee, the EDGs are run monthly for 2 h. Thermal equilibrium would be expected to be reached on the exciter cabinets in about 30 min. Assuming that the "A" EDG diode failures were the result of thermal aging and that the "A" EDG was operable at the start of the 1-year observation period, a mean-time-to-failure of 38 h is estimated. Assuming further that failure of the second EDG will only occur if there is insufficient time to discover and correct the problem following failure of the first EDG, the probability of emergency power failure can be represented as

p(ac power not recovered) × p(first EDG fails)
× p(failure cause not found and corrected before second EDG fails).

In this analysis, the failure cause was assumed not to be repairable if the second EDG failed within 2 h of the first. Assuming an exponential distribution for EDG failure and the LOOP recovery distributions for Haddam Neck described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTE-89/11, August 1989, a probability of emergency power failure due to EDG exciter cabinet overheating of 2.1×10^{-3} is estimated. This value was added to the emergency power failure on demand probability (2.9×10^{-3}) used in the analysis. This is somewhat conservative because all emergency power failures are assumed to occur at the time of the LOOP, when the potential failure of MCC-5 must also be addressed. However, based on the probabilities estimated for core damage following a blackout with and without MCC-5 failure (see Seal LOCA later in this section), the difference is not significant.

Failure of MCC-5 requires the manual closure of two valves to isolate SW flow to nonsafety turbine-building loads. If both EDGs start, adequate SW flow is maintained without isolation of turbine building loads. If only one EDG starts, however, isolation of these loads may be required to ensure adequate SW flow to the operating EDG, unless a second SW pump can be started. The time required to isolate turbine building loads before EDG cooling problems could not be determined; isolation may only be required for high SW temperatures or if the containment air coolers are required to remove post-LOCA decay heat loads. Because of the uncertainty in the need for isolation of turbine building loads, this was not addressed in the analysis.

A.1.4.3 AFW—Auxiliary Feedwater

Normal AFW flow control is dependent on MCC-5. However, flow control is also possible using the hydraulically powered turbine pump steam admission valves. AFW flow is controlled using these valves during startup and shutdown, so operators are familiar with their use. Therefore, nominal AFW response was assumed following the postulated loss of MCC-5.

A.1.4.4 PORV/SRV CHALL—Challenge Rate for Pressurizer PORVs and Safety Relief Valves (SRVs)

The PORV block valves are maintained in a closed position at Haddam Neck. Therefore, this probability is the probability that the SRVs lift after a LOOP with MCC-5 unavailable. It was assumed that the lift rate for the SRVs is the same as when both the PORVs and SRVs are available. Therefore, this value was not modified.

A.1.4.5 PORV/SRV RESEAT—Reseat of Challenged Pressurizer PORVs and SRVs

It was assumed that the failure to reseat probability for the SRVs is the same as for the PORVs. The nonrecovery value was set to 1.0 because the safety valves do not have block valves.

A.1.4.6 SEAL LOCA—RCP Seal LOCA Probability

Operator action is required to recover both means of RCP seal cooling (seal injection and thermal barrier cooling) following a LOOP and the loss of MCC-5. Component cooling water, which provides thermal barrier cooling, is lost following the LOOP due to the loss of instrument air. The charging pumps, which provide seal injection, also trip following a LOOP due to an automatic tripping feature that had recently been installed. Because the main lube oil pumps for the charging pumps were powered from MCC-5, the charging pumps could not be restarted without first recovering MCC-5 or providing lube oil pressure from the alternate lube oil pumps. The alternate lube oil pumps are powered from MCC-8, which is load shed following a LOOP (MCC-8 can be repowered using EDG 23, if required). As a result, seal injection would only be available if MCC-5 is recovered or MCC-8 is repowered.

The potential impact of an RCP seal LOCA following loss of MCC-5 but with emergency power available was addressed in the event tree model shown in Fig. A.1.3. This model is applicable to sequences involving emergency power and AFW success and the SRV closed. In this model, seal injection andhigh-pressure injection (HPI) were assumed to be unavailable unless MCC-5 is recovered. Recovery of charging by repowering MCC-8 was assumed to also be addressed within the MCC-5 recovery model. (Licensee comments in a telephone conversation on August 26, 1994, with NRC and ORNL personnel indicated that resources would be diverted from MCC-5 recovery to recovery of the charging pumps if MCC-5 could not be recovered in 5 to 10 min. This supports consideration of both actions within a single time reliability correlation-based recovery model.)

To simplify the analysis, an RCP seal LOCA was assumed likely in nonblackout sequences if MCC-5 is not recovered at 1 h. The probability of not recovering MCC-5 at 1 h, given that it was not recovered at 0.5 h (this probability is addressed in a conditioning event tree branch), was estimated to be 9.4×10^{-2}. The probability of seal LOCA occurring at this time was assumed to be 0.7, consistent with other ASP analyses.

For blackout sequences, both ac power and MCC-5 must be recovered to prevent an RCP seal LOCA. The probability of not recovering both in 1 h (the time at which seal LOCAs are assumed to begin) is estimated to be 0.17, based on a convolution approach. The event tree model used to address potential seal LOCAs following a station blackout is shown in Fig. A.1.4. This model utilizes the same assumption regarding the onset of a seal LOCA and recovery of HPI as the nonblackout case.

A.1.4.7 HPI—High Pressure Injection

Following the loss of MCC-5, HPI is lost (Haddam Neck IPE Table B-1, System/Function 2. Sump Recirculation). Restoration of power to MCC-5 is required to regain HPI function. The charging pumps are also unavailable until MCC-5 is recovered or MCC-8 is repowered and the auxiliary lube oil pumps are started.

For a stuck-open SRV, the probability of HPI failure, given that MCC-5 was not recovered at 0.5 h, was assumed to be 1.0. For an RCP seal LOCA with emergency power initially available, the failure probability for HPI was estimated to be 0.17 [p(MCC-5 not recovered 0.5 h after a potential seal LOCA | MCC-5 not recovered at 1 h {onset of seal LOCA}—see MCC-5 Failure and Restoration)]. For an RCP seal LOCA following a station blackout, the HPI failure probability was estimated to be 0.57.

A.1.4.8 EP REC (LONG)—Long-Term Recovery of Offsite Power

The probability of failing to recover offsite power before battery depletion at 6 h was estimated to be 0.037, based on LOOP recovery models described in ORNL/NRC/LTR-89/11. These models are based on the results of the data distributions contained in NUREG-1032.

A.1.4.9 Feed-and-Bleed

Feed-and-bleed requires the operation of HPI or the charging pumps, the high-pressure recirculation system (HPR), and the pressurizer PORVs. One HPI or charging pump and one PORV are required for success. Because the normally closed PORV block valves are powered by MCC-5, feed-and-bleed would be unavailable until MCC-5 is recovered. For sequences with MCC-5 failed, the unavailability of feed-and-bleed was represented in the model by setting PORV OPEN to failed.

If MCC-5 does not fail or is recovered, HPI and both PORVs can be utilized; however, one of the PORVs was unavailable due to the air leak in the PORV relief valve. This air leak would deplete the backup accumulator that is required for PORV operability during feed-and-bleed. The accumulator is required because the containment air compressors are not qualified for a feed-and-bleed environment and would be expected to fail. The procedures do not specify which valve the operator should open first. If the operator opens the faulty valve first, then he must recognize the PORV fault, close the faulty valve, and open the remaining PORV and block valve. Given the additional operator actions during this scenario, this was considered a burdened operator action. ASP Recovery Class R3 (NUREG/CR-4674, Vol.17, Sec. A.1.3, Recovery Class R3, failure recoverable in the required period from the control room recovery not routine or involved substantial operator burden) was used with a operator failure probability of 0.12. A beta factor of 0.1 was assumed for failure of the second valve given the failure of the first valve. Therefore the probability of PORV failure is given by:

p(oper opens faulty valve first) ×

[p(oper fails to recognize and respond to existing PORV fault)

+ p(second valve leaks| first valve leaks) + p(second PORV fails to open)]

+ p$\overline{\text{(oper opens faulty valve first)}}$ × [p(second valve leaks | first valve s)

+ p(second valve fails to open)] =

= 0.5 [0.12 + 0.1 + 0.01] + 0.5 [0.1 + 0.01] = 0.17 .

A.1.4.10 Core Damage Probability Calculation

The impact of the leaking PORV on feed-and-bleed following postulated transients and small-break LOCAs was assessed by modifying the PORV failure probability during feed-and-bleed to reflect the leaking valve (p = 0.17), and calculating an increase in core damage probability over the 6132-h assumed unavailability period (calculation 1).

To address the loss of MCC-5, the degraded EDGs, and the leaking PORV following a postulated LOOP, a conditioning event tree was used. This event tree characterized potential plant conditions involving EDG success and failure, short-term (30-min) LOOP recovery, and short-term MCC-5 recovery. The event tree, shown in Fig. A.1.5, includes the following conditioning sequences:

Sequence	Description
1	Initial EP success with short-term recovery of offsite power and MCC-5 following the postulated LOOP. This is similar to a loss of feedwater, but with a higher probability of a transient-induced LOCA, because the SRVs would lift (if necessary), because of the inoperable PORV block valves. Feed and bleed is degraded because of the leaking relief valve.
2	Initial EP success and short-term recovery of offsite power but with MCC-5 not recovered at 30 min. This is similar to sequence 1, but with the potential for an RCP seal LOCA if MCC-5 is not recovered at 1 h. HPI is assumed unavailable if MCC-5 is not recovered ~0.5 h following a seal LOCA. HPI following a stuck-open SRV and feed-and-bleed are also assumed to be unavailable, since MCC-5 is unavailable at 30 min.
3	LOOP with EP initially successful, MCC-5 recovered, and feed-and-bleed degraded. Higher probability of a transient-induced LOCA.
4	LOOP with EP initially successful but neither MCC-5 nor offsite power recovered at 30 min. Potential for RCP seal LOCA if MCC-5 is not recovered. Higher probability of a transient-induced LOCA. HPI following a stuck-open SRV and feed-and-bleed are also assumed to be unavailable, since MCC-5 is unavailable at 30 min.
5	Station blackout.
6	Anticipated transient without scram (sequence not described in ASP models).

Figure A.1.6 also includes the relevant branch and conditioning sequence probabilities and identifies the calculation sheet associated with each sequence. Specific branch probability modifications are indicated in the Branch Frequencies/Probabilities section at the end of each calculation sheet.

The conditional probabilities estimated in calculations 1a (feed and bleed degraded during transients), 1b (feed and bleed degraded following a small-break LOCA), 2.1 (conditioning sequences 1 and 3), 2.2 (conditioning sequences 2 and 4), Fig. A.1.3 (seal LOCA for nonblackout sequences), and Fig. A.1.4 (station blackout) were combined with the probabilities of such sequences occurring in the 6132-h observation period to estimate the conditional probability for the combined event:

Sequence	p(sequence)	p(cd| sequence)	p(cd)
1a	6.8×10^{-1}	1.6×10^{-5} (calc 1a)	1.1×10^{-5}
1b	1.5×10^{-2}	8.6×10^{-4} (calc 1b)	1.3×10^{-5}
2.1	7.4×10^{-2}	2.4×10^{-4} (calc 2.1)	1.8×10^{-5}
2.2	4.4×10^{-4}	2.1×10^{-3} (calc 2.2)	9.2×10^{-7}
		1.1×10^{-2} (seal LOCA, Fig. A.1.3)	4.8×10^{-6}
2.3	2.3×10^{-2}	2.4×10^{-4} (calc 2.1)	5.5×10^{-6}
2.4	1.7×10^{-4}	2.1×10^{-3}(calc 2.2)	3.6×10^{-7}
		1.1×10^{-2} (seal LOCA, Fig. A.1.3)	1.9×10^{-6}
2.5	4.9×10^{-4}	7.0×10^{-2} (blackout, Fig. A.1.4)	3.4×10^{-5}

The sum of the probabilities for the sequences is 8.9×10^{-5}.

For operational events involving unavailabilities, such as this event, the ASP program estimates the core damage probability for the event by calculating the probability of core damage during the unavailability period conditioned on the failures observed during the event and subtracting a base-case probability for the same period, assuming plant equipment performs nominally. Because a conditioning event tree was used to perform some of the sequences associated with a postulated LOOP, the ASP computer code was not used to perform this differential calculation. Instead, the ASP code was used to calculate the probability of core damage given the conditions observed during the event and a postulated initiating event. This probability was then multiplied by the probability of the initiator during the unavailability period. The nominal core damage probability was estimated in the same way. For this analysis, subtracting the nominal core damage probability for the 6132-h period (2.4×10^{-5}) results in an overall estimate of 6.5×10^{-5}. The base-case calculations are not included.

A.1.5 Analysis Results

The conditional core damage probability estimated for the combined MCC-5, EDG and PORV degradation is 6.5×10^{-5}. The dominant sequence is a postulated LOOP, failure of both EDGs, a subsequent RCP seal LOCA, and failure to recover AC power before core uncovery. The failure of the EDGs because of a lack of exciter cabinet ventilation contributes to this sequence more than the other failures observed during the event. This is primarily because of the assumption made in the analysis that MCC-5 would only fail about 10% of the time following a LOOP (only one of the two MCC-5 circuit breakers was failed). If both MCC-5 circuit breakers had failed, sequences involving failure to recover MCC-5, combined with the increased failure to run probability for the EDGs would dominate. The degraded PORV for feed and bleed contributes to a lesser degree.

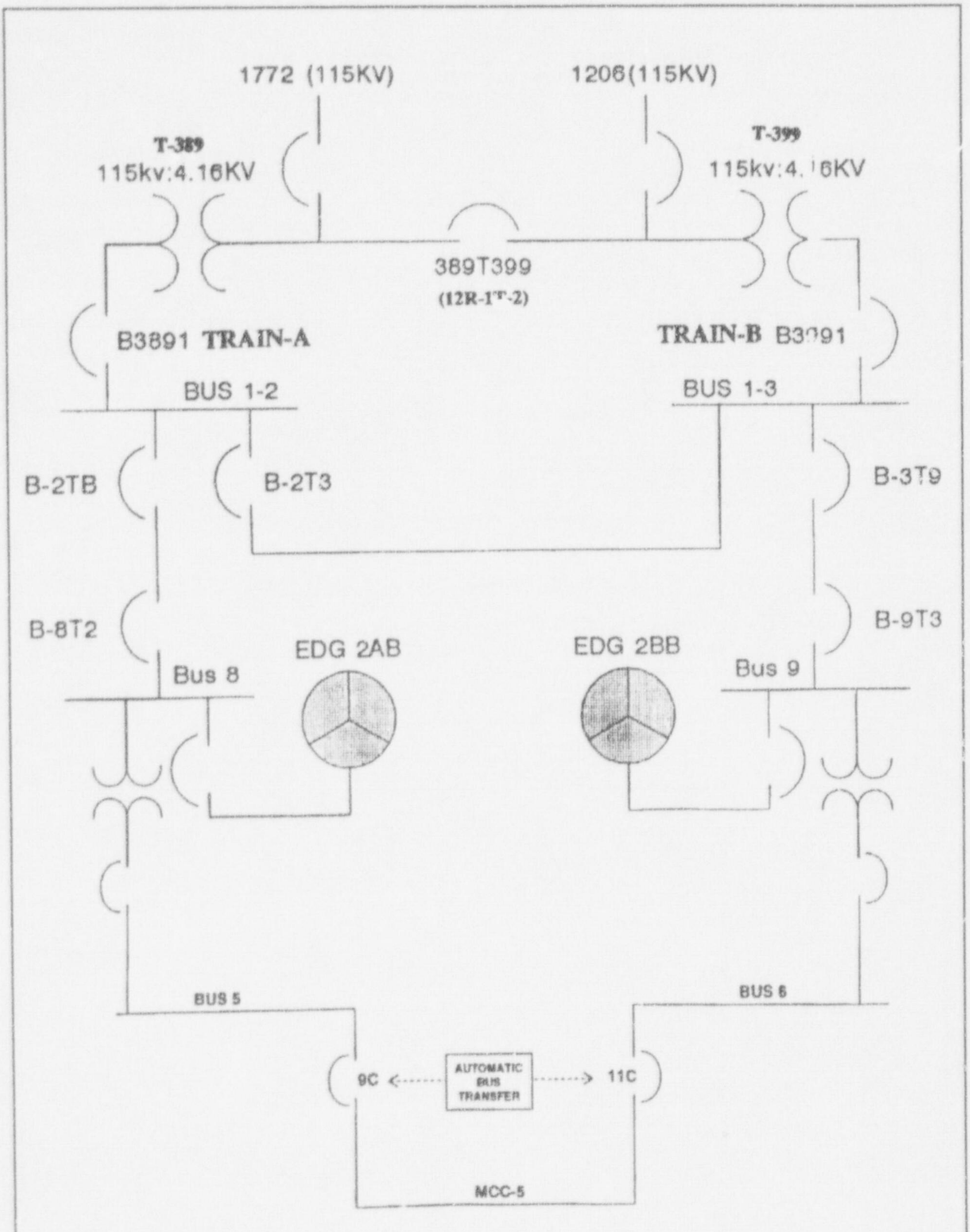


Fig. A.1.2 Simplified diagram of MCC-5 power supplies

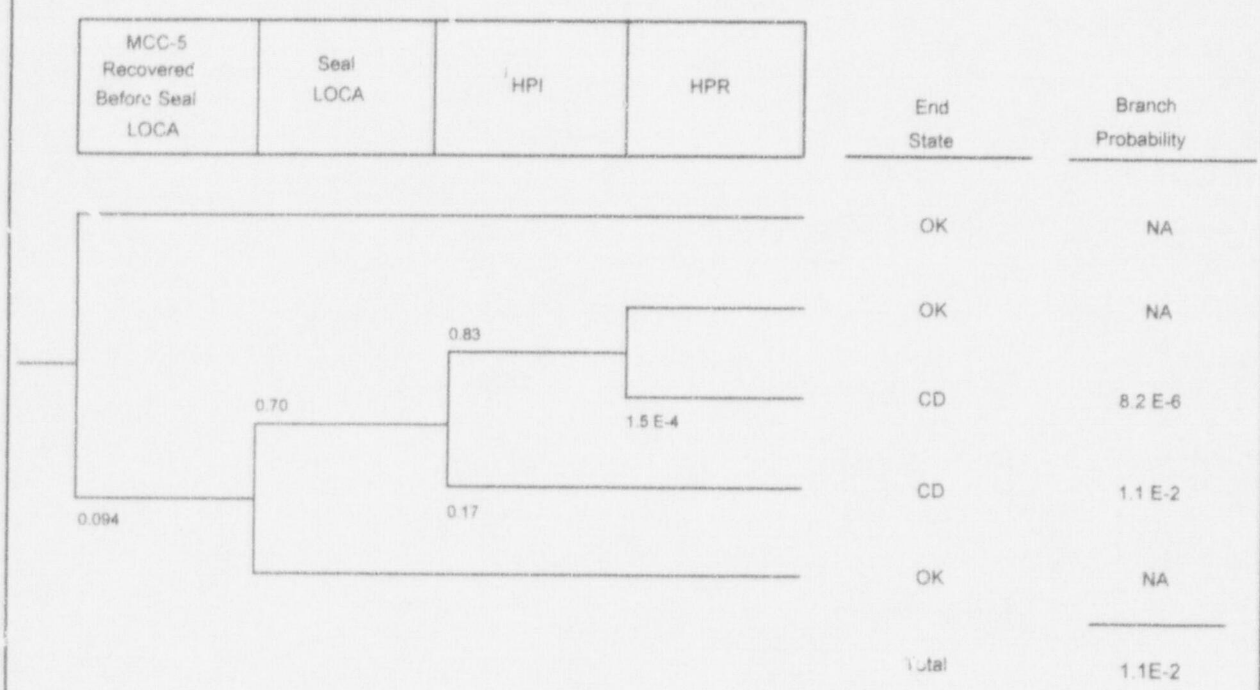


Fig. A.1.3 Event tree for RCP seal LOCA (nonblackout sequences)

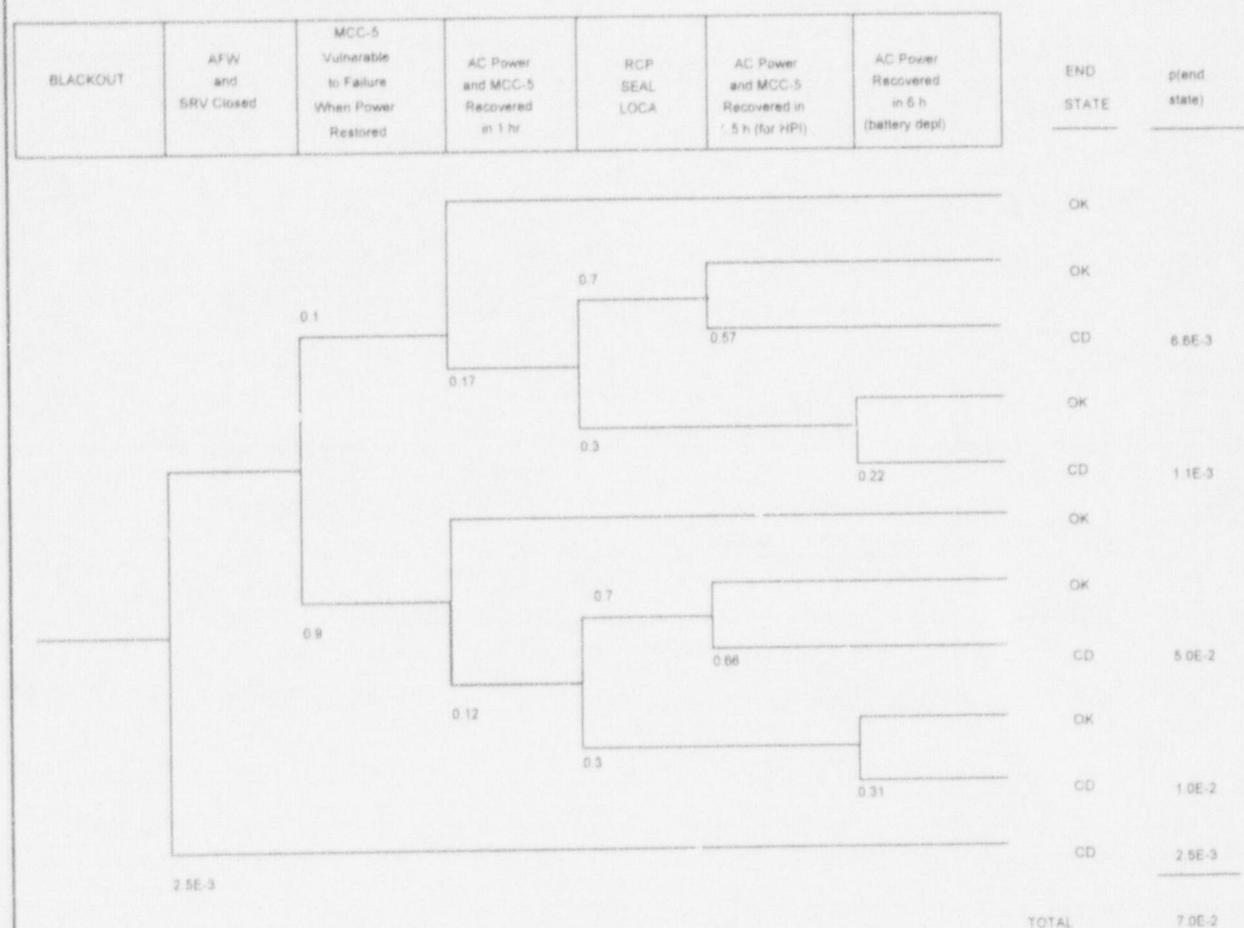
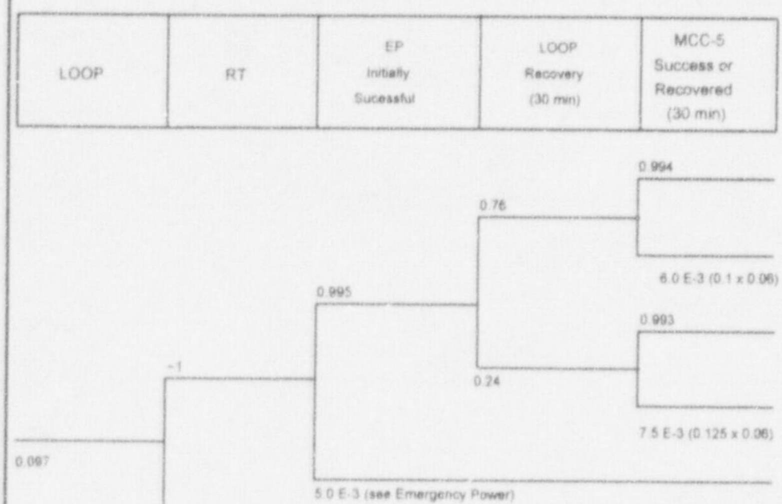


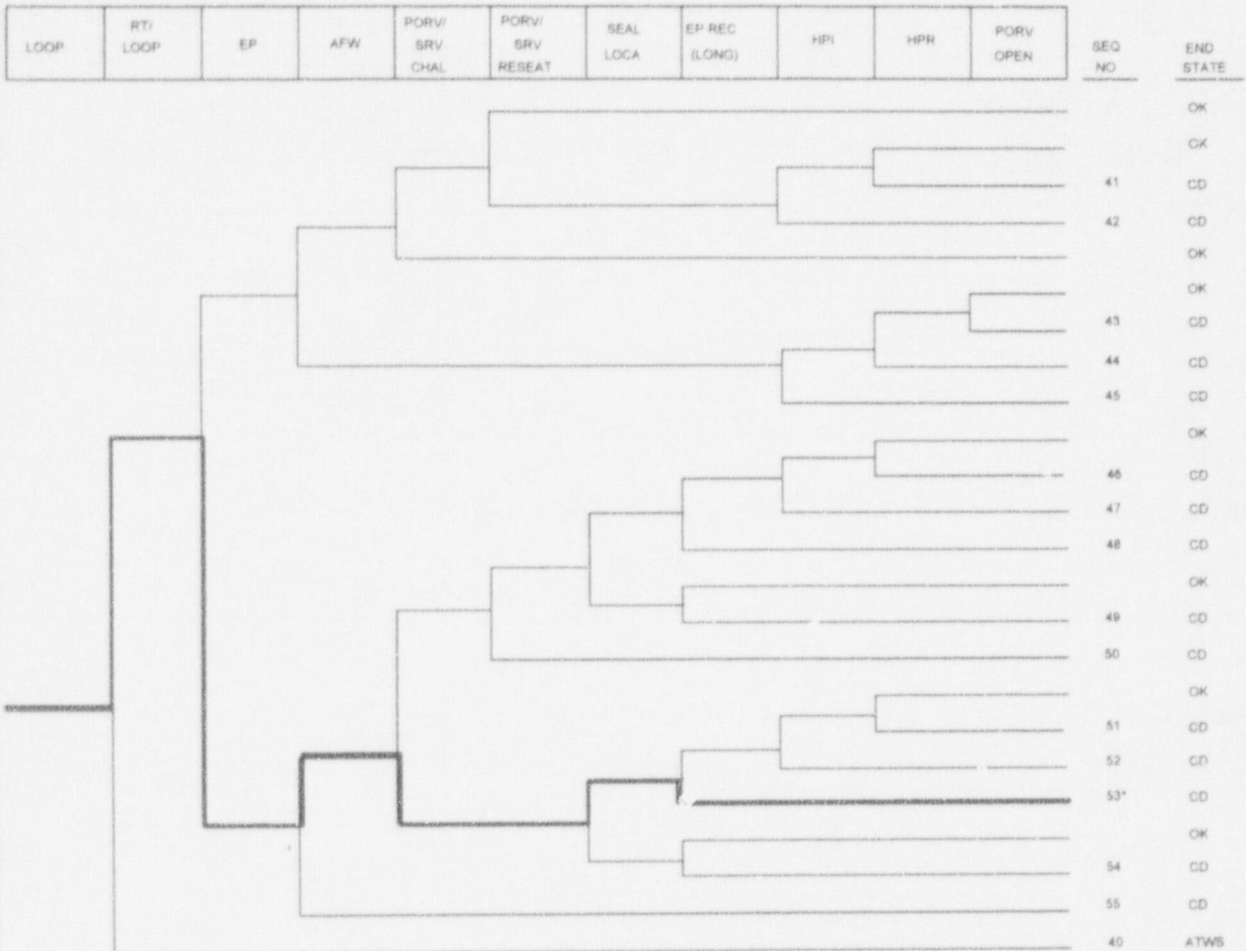
Fig. A.1.4 Event tree model for blackout sequences



Conditioning Sequence	Conditioning Sequence Probability	Calculations
1	7.4 E-2	2.1
2	4.4 E-4	2.2
3	2.3 E-2	2.1
4	1.7 E-4	2.2
5	4.9 E-4*	Fig. A.1.4
6	~0	NA

* LOOP and MCC-5 recovery addressed in Fig. A.1.4.

Fig. A.1.5 Conditioning event tree for postulated LOOP



* The dominant sequence consists of partial sequences from Figs. 3 and 4. The sequence shown on this event tree is representative of those partial sequences.

Fig. A.1.6 Dominant core damage sequence for AIT Report 213/93-80

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 213/93-80
Event Description: Degradation of MCC-5 ABT and other components (calc 1a)
Event Date: 06/27/93
Plant: Haddam Neck

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.6E-05
Total	1.6E-05
ATWS	
TRANS	3.4E-05
Total	3.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
15	trans -rt afw mfw -hpi(f/b) -hpr/-hpi PORV.OPEN	CD	1.5E-05	9.2E-02
18	trans rt	ATWS	3.4E-05	1.2E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
15	trans -rt afw mfw -hpi(f/b) -hpr/-hpi PORV.OPEN	CD	1.5E-05	9.2E-02
18	trans rt	ATWS	3.4E-05	1.2E-01

** non-recovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\1989\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\1989\haddem.sl1
PROBABILITY FILE: s:\asp\prog\models\1989\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.8E-04	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
afw	5.0E-03	2.7E-01	
afw/emerg.power	5.0E-03	3.4E-01	
mfw	1.9E-01	3.4E-01	
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	2.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	

seal.loca	2.5E-01	1.0E+00	
ep.rec(sl)	6.9E-01	1.0E+00	
ep.rec	2.2E-02	1.0E+00	
hpi	1.0E-03	8.4E-01	
hpi(f/b)	1.0E-03	8.4E-01	1.0E-02
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
PORV.OPEN	1.0E-02 > 1.7E-01	1.0E+00	4.0E-04
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	1.0E-02 > 1.7E-01		

* branch model file
** forced

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 213/93-80
Event Description: Degradation of MCC-5 ABT and other components (calc 1b)
Event Date: 06/27/93
Plant: Haddam Neck

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOCA	4.3E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOCA	8.6E-04
Total	8.6E-04
ATWS	
LOCA	1.4E-05
Total	1.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
71	loca -rt -afw -hpi hpr/-hpi	CD	4.9E-04	4.3E-01
72	loca -rt -afw hpi	CD	3.6E-04	3.6E-01
78	loca rt	ATWS	1.4E-05	5.2E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
71	loca -rt -afw -hpi hpr/-hpi	CD	4.9E-04	4.3E-01
72	loca -rt -afw hpi	CD	3.6E-04	3.6E-01
78	loca rt	ATWS	1.4E-05	5.2E-02

** non-recovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\1989\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\1989\haddem.sl1
PROBABILITY FILE: s:\asp\prog\models\1989\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.8E-04	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
afw	5.0E-03	2.7E-01	
afw/emerg.power	5.0E-03	3.4E-01	
mfw	1.9E-01	3.4E-01	
porv.or.srv.chall	4.0E-02	1.0E+00	

porv.or.srv.reseat	2.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	2.5E-01	1.0E+00	
ep.rec(sl)	6.9E-01	1.0E+00	
ep.rec	2.2E-02	1.0E+00	
hpi	1.0E-03	8.4E-01	
hpi(f/b)	1.0E-03	8.4E-01	1.0E-02
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
PORV.OPEN	1.0E-02 > 1.7E-01	1.0E+00	4.0E-04
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	1.0E-02 > 1.7E-01		

* branch model file
** forced

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 213/93-80
Event Description: Degradation of MCC-5 ABT and other components (calc 2.1)
Event Date: 06/27/93
Plant: Haddam Neck

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	2.4E-04
Total	2.4E-04
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
43	LOOP -rt/loop -EMERG.POWER afw -hpi(f/b) -hpr/-hpi PORV.OPEN	CD	2.3E-04	2.7E-01
45	LOOP -rt/loop -EMERG.POWER afw hpi(f/b)	CD	1.5E-05	2.3E-01

** non-recovery credit for edited case

SEQUEN(?ONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
43	LOOP -rt/loop -EMERG.POWER afw -hpi(f/b) -hpr/-hpi PORV.OPEN	CD	2.3E-04	2.7E-01
45	LOOP -rt/loop -EMERG.POWER afw hpi(f/b)	CD	1.5E-05	2.3E-01

** non-recovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\1989\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\1989\haddem.sl1
PROBABILITY FILE: s:\asp\prog\models\1989\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.8E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	2.4E-01 > 1.0E+00	
Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
EMERG.POWER	2.9E-03 > 0.0E+00 **	8.0E-01 > 1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02		
afw	5.0E-03	2.7E-01	
afw/emerg.power	5.0E-03	3.4E-01	

mfw	1.9E-01	3.4E-01	
porv.or.srv.chall	4.0E-02	1.0E+00	
PORV.OR.SRV.RESEAT	2.0E-02 > 2.0E-02	1.1E-02 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-02		
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	2.5E-01	1.0E+00	
ep.rec(sl)	6.9E-01	1.0E+00	
ep.rec	2.2E-02	1.0E+00	
hpi	1.0E-03	8.4E-01	
hpi(f/b)	1.0E-03	8.4E-01	1.0E-02
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
PORV.OPEN	1.0E-02 > 1.7E-01	1.0E+00	4.0E-04
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	1.0E-02 > 1.7E-01		

* branch model file
** forced

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 213/93-80
Event Description: Degradation of MCC-5 ABT and other components (calc 2.2)
Event Date: 06/27/93
Plant: Haddam Neck

I‹ TIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	2.1E-03
Total	2.1E-03
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
45	LOOP -rt/loop -EMERG.POWER afw HPI(F/B)	CD	1.4E-03	2.7E-01
42	LOOP -rt/loop -EMERG.POWER -afw porv.or.srv.chall PORV.OR.SRV. RESEAT HPI	CD	8.0E-04	1.0E+00

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
42	LOOP -rt/loop -EMERG.POWER -afw porv.or.srv.chall PORV.OR.SRV. RESEAT HPI	CD	8.0E-04	1.0E+00
45	LOOP -rt/loop -EMERG.POWER afw HPI(F/B)	CD	1.4E-03	2.7E-01

** non-recovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\1989\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\1989\haddem.sl1
PROBABILITY FILE: s:\asp\prog\models\1989\pwr_bsl1.pro

No Recovery Limit

RANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.8E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	2.4E-01 > 1.0E+00	
Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
EMERG.POWER	2.9E-03 > 0.0E+00 **	8.0E-01 > 1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02		

afw	5.0E-03	2.7E-01	
afw/emerg.power	5.0E-03	3.4E-01	
mfw	1.9E-01	3.4E-01	
porv.or.srv.chall	4.0E-02	1.0E+00	
PORV.OR.SRV.RESEAT	2.0E-02 > 2.0E-02	1.1E-02 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-02		
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	2.5E-01	1.0E+00	
ep.rec(sl)	6.9E-01	1.0E+00	
ep.rec	2.2E-02	1.0E+00	
HPI	1.0E-03 > 1.0E+00	8.4E-01 > 1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	1.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HPI(F/B)	1.0E-03 > 1.0E+00	8.4E-01 > 1.0E+00	1.0E-02
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
PORV.OPEN	1.0E-02 > 1.0E+00	1.0E+00	4.0E-04
Branch Model: 1.OF.1+opr			
Train 1 Cond Prob:	1.0E-02 > Unavailable		

* branch model file
** forced

A.2 LER Nos. 265/93-010 and -012

Event Description: Emergency Power System Unavailable

Date of Event: April 22, 1993

Plant: Quad Cities 2

A.2.1 Summary

During a surveillance test on April 22, 1993, the Quad Cities swing diesel generator cooling water pump (DGCWP) breaker locked-up on antipump protection. The licensee determined that the potential for lock-up existed since the initial plant startup if the pump power source was aligned to Unit 2. A 1992 modification ensured that the cooling water pump would be powered from Unit 2 if a loss-of-offsite power (LOOP) occurred on that unit. Unavailability of cooling water for ~ 5 to 10 min is sufficient to damage the DG.

About one month earlier, inadequate bearing oil level had been found in the Unit 2 dedicated diesel cooling water pump, the result of an incorrectly reassembled oiler. The pump would have been expected to fail if it had been required to run for more than a short period of time. The Unit 2 emergency power system was vulnerable to failure for a 7-month period beginning in August 1992.

The conditional core damage probability estimated for the event is 6.0×10^{-5}. The relative significance of this event compared to other potential events at Quad Cities 2 is shown in Fig. A.2.1.

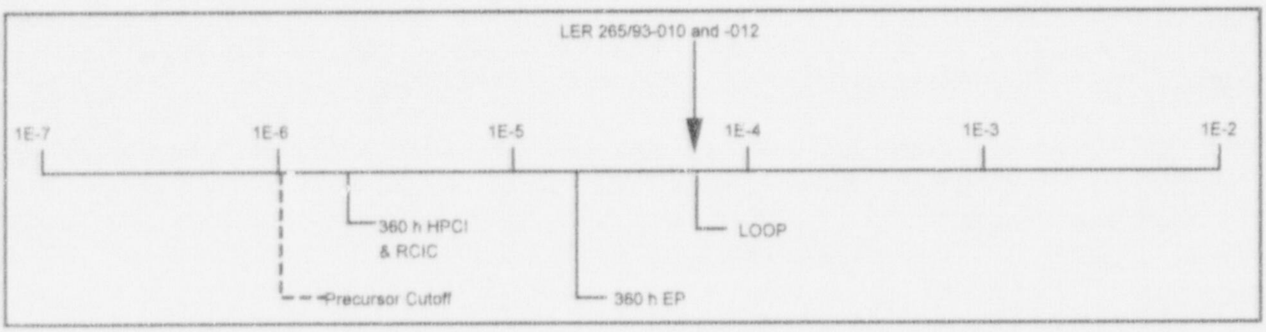


Fig. A.2.1 Relative event significance of LER Nos. 265/93-010 and -012 compared with other potential events at Quad Cities 2

A.2.2 Event Description

A.2.2.1 1/2 DGCWP Failure

On April 22, 1993, Quad Cities 2 was shut down and performing 4-kV bus 23-1 Undervoltage Functional Test QOS-6500-4. During the performance of this surveillance test, the 1/2 (swing) DGCWP failed to start as required. The pump was manually started at the DG ~ 2 min later by repositioning the feed power selector switch for the pump.

The 1/2 DGCWP breaker had locked-up on antipump protection. The licensee determined that the potential for lock-up of the 1/2 DGCWP if its power source was aligned to Unit 2 had existed since the initial design of the plant. However, before April 1992, the control logic aligned the 1/2 DGCWP to Unit 1 if power was available to the Unit 1 bus, even if the 1/2 DG was aligned to supply power to Unit 2. An undervoltage logic modification in April 1992 revised the control logic for the 1/2 DGCWP so that the pump would receive power from the same bus that the 1/2 DG was powering, which ensured that the lock-up condition would occur in the event of a LOOP on Unit 2. Postmodification testing failed to identify the lock-up problem. A similar problem existed if the 1/2 DGCWP was powered from Unit 1, but only if the power selector switch was aligned to Unit 1 bus 18. Normally this switch is set to the "normal" position, which aligns 1/2 DGCWP power to the 1/2 DG powered-bus.

In the event of a LOOP on Unit 2, the 1/2 DG would be expected to start and load but would fail after ~5 min because of the loss of cooling water caused by the locked-up 1/2 DGCWP. A loss of DG cooling is not specifically annunciated in the control room—receipt of panel 902-8 annunciator A-4, "Diesel Generator 1/2 Trouble" would require operator response in the DG room. An operator is routinely dispatched to the DG rooms following an auto-start at Quad Cities. In the DG room, annunciator C-3, "Diesel Cooling Water Pump Failure or Diesel Cooling Water Pump Locked Out" would be alarmed. Its alarm procedure requires the 1/2 DGCWP to be manually started. Based on training, the operator would be expected to accomplish this by repositioning the pump feed power selector switch to the "Bus 28-1" position. Quad Cities procedures prohibit tripping a DG that has autostarted; recovery must, therefore, be accomplished by restoring cooling water flow before the DG overheats.

A.2.2.2 Unit 2 DGCWP Failure

On March 25, 1993, the Unit 2 dedicated DG (2 DG) had been taken out of service for scheduled maintenance. Four days later an operator questioned the height of the oiler for the 2 DGCWP. Upon disassembly of the pump, approximately one tablespoon of oil was found in the oil reservoir. This was the expected level based on the height of the oiler. The bearing retainer ring, which provides spaces between the ball bearings, was found in pieces. The races and ball bearings were intact, but the bearing and pump shaft had apparent heat damage, and the balls were coated with a heavy, grease-like film.

The 2 DGCWP bearing degradation was caused by the incorrect reassembly of the oiler piping during pump maintenance in January 1992. Although the pump bearing oil level was very low and the bearing showed evidence of heat damage, the 2 DGCWP was run and successfully passed its surveillance tests through February 16, 1993. This included a 56-h run between August 5–7, 1992, and six 2.5-h monthly surveillance tests.

Unit 2 entered a refueling outage on March 6, 1993. The emergency power system was vulnerable to failure if required to operate for longer than the limited 2 DGCWP lifetime from August 1992.

A.2.3 Additional Event-Related Information

A simplified diagram of the emergency power system at Quad Cities is provided in Fig. A.2.2. Three DGs provide emergency power to the two units: the 1 DG provides power to Unit 1 bus 14-1, the 2 DG provides power to Unit 2 bus 24-1, and the 1/2 DG provides power to either Unit 1 bus 13-1 or Unit 2 bus 23-1 in the event of a LOOP on Unit 1 or Unit 2, respectively. In the event of a dual-unit LOOP with a loss-of-coolant accident (LOCA) on one unit, the 1/2 DG provides power to the unit with the LOCA. In the event of a dual-unit LOOP without a LOCA, the 1/2 DG powers the unit that first suffers the LOOP. Unit 1 bus 14-1 and Unit 2 bus 24-1 can be cross-tied by closing two normally open breakers.

Two 250-V dc and two 125-V dc batteries are shared between both units. The 250-V dc batteries primarily power large loads, such as dc-powered pumps and valves, while the 125-V dc batteries provide control power to components such as circuit breakers. Each battery is sized to power its respective loads for 4 h.

Normally, Unit 1 batteries are charged using power from bus 14-1 (via bus 19); Unit 2 batteries are charged from bus 24-1 (via bus 29). An alternate charger can be powered from buses 13-1 and 23-1 and can charge either unit's battery. The 480-V ac buses that power the battery chargers on each unit can also be cross-tied.

In addition to high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC), Quad Cities can utilize a shared safe shutdown makeup pump (SSMP) to provide high-pressure makeup in the event of a loss of feedwater (FW). The motor-driven pump is capable of supplying 400 gal/min at essentially all reactor pressures. It is powered from safe shutdown bus 31, which is supplied from Unit 1 bus 14-1 (preferentially) or Unit 2 bus 24-1, and can provide makeup to either of the two units. The pump and associated valves can be operated from the control room. Utilization of the SSMP requires opening a test return valve, starting the pump, opening the injection valve, and closing the test return valve. The SSMP would be used if both HPCI and RCIC failed.

Thermal-hydraulic analyses performed in support of the individual plant examination (IPE) indicated that RCIC or the SSMP, in addition to HPCI and FW, can provide sufficient makeup to prevent core damage in the event of a single stuck-open relief valve.

Additional information concerning this event is provided in NRC inspection report 50-254/93016; 50-265/93-016 dated June 9, 1993.

A.2.4 Modeling Assumptions

The event was modeled as a potential LOOP during the 7-month (211-d) period in which both DGCWPs were vulnerable to failure. Consistent with information provided by the licensee, the analysis assumed that the 1/2 DGCWP breaker would lock-up on antipump protection if a LOOP occurred at Unit 2 at any time following the undervoltage logic modification in April 1992. The analysis also assumed that the 2 DGCWP was vulnerable to failure due to low bearing oil level following the successful 56-h run on August 7, 1992. After that, the 2 DGCWP was run 2.5 h each month until February 16, 1993, for DG surveillance testing. Following the February 16, 1993, test, the oil level was assumed to be inadequate, and the pump was assumed to be failed. The average lifetime of the 2 DGCWP in the 7-month period is, therefore, 7.5 h. The emergency power system would fail, on average, 7.5 h after a postulated LOOP; battery depletion would occur 4 h later unless power was recovered to the battery chargers.

Due to the nature of the DGCWP failures, no restoration of emergency power through DG recovery was assumed possible. Cooling water would have to be restored to the 1/2 DG within ~ 5 to 10 min of receipt of the control room 1/2 DG trouble alarm to prevent damage to the DG. At this time, both DGs would be running and powering their safety-related buses. Although an operator is dispatched to the EDGs following auto-start and could reach the 1/2 DG room in 10 min under these circumstances, diagnosis and recovery of cooling water within that time was considered unlikely.* Failure of the 2 DG after the 2 DGCWP fails from lack of bearing oil was also considered nonrecoverable.

The analysis considered potential plant-centered LOOPs that would impact only Unit 2 and potential grid- and weather-related LOOPs that would impact both units. Although multiple-unit, plant-centered LOOPs have been historically observed, their impact is small compared to grid- and weather-related LOOPs.

*In "Electric Power Recovery Models," J. W. Read and K. N. Fleming, *Proceedings of the International Topical Meeting on Probabilistic Safety Assessment, PSA '93*, January 26-29, 1993, the probability of first operator arrival at a failed DG is estimated. For times up to 10 min, this probability is 0.26 (p_{fail} = 0.74). In the dominant sequence for this event (developed later), the EDG is also failed. This failure would compete with the 1/2 EDG for response/recovery resources.

Recovery of power to Unit 2 buses was assumed to occur following recovery of offsite power. Recovery of power to bus 24-1 was also assumed to be possible from Unit 1 bus 14-1 through closure of normally open breakers 1421 and 2429.

The ASP model frequency for LOOP and 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/NRC/LTR-89/11, August 1989. These models are based on the data distributions described in *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, June 1988.

For operational events involving unavailabilities, such as this event, the ASP program estimates the core damage probability for the event by calculating the probability of core damage during the unavailability period conditioned on the failures observed during the event, and subtracting a base-case probability for the same period, assuming plant equipment performs nominally. Because potential plant-centered LOOPs were addressed separately from potential grid- and weather-related LOOPs in this analysis, the ASP computer code could not be used to perform this differential calculation. Instead, the ASP code was used to calculate the probability of core damage given the conditions observed during the event and a postulated LOOP. This probability was then multiplied by the probability of a LOOP during the 211-d multiple DG unavailability. The nominal core damage probability was estimated in the same way. For this analysis, subtracting the nominal core damage probability did not significantly affect the overall results. Because of this, those calculations are not included here.

The analysis addressed the potential use of the SSMP, RCIC, and containment venting in providing core protection for certain sequences at Quad Cities. The SSMP was considered the primary backup for HPCI and RCIC. Because the pump can be operated from the control room, it was assumed that no effort would be made to recover HPCI or RCIC before using the SSMP. Two motor-operated valves plus the pump itself must be remote-manually operated for SSMP success. A failure probability of 0.04 was estimated, based on nominal failure probabilities used in the ASP Program (0.01 for pumps and motor-operated valves) and an assumed operator error probability of 0.01. This operator error probability is typically used for failure to utilize the control rod drive (CRD) pumps for reactor pressure vessel makeup following HPCI and RCIC failure (see Appendix A, Sect. A.3.2 and Table A.14, *Precursors to Potential Severe Core Damage Accidents: 1992, A Status Report*, NUREG/CR-4674, Vol. 17). At Quad Cities, however, the operators are directed to use the CRD pumps only if HPCI, RCIC, and the SSMP all fail. (This would require the operator error probability associated with the CRD pumps to be increased from the nominal ASP value. However, because of the observed DG unavailabilities, CRD pump availability for injection does not impact this analysis, and the CRD branch failure probability was not revised.) Sequences with successful SRV closure and HPCI and RCIC failure were modified to include failure of the SSMP by multiplying their sequence probabilities by p(SSMP).

To address the potential use of RCIC or the SSMP to provide core cooling in the event of a single stuck-open relief valve, the conditional probabilities for sequences involving a stuck-open safety-relief valve with HPCI failure were multiplied by

$$p(\text{two or more SRVs open} \mid \text{one SRV open}) + p(\text{RCIC}) \times p(\text{SSMP}).$$

Because only one SRV is manually opened at Quad Cities for most transients, p(two or more SRVs open | one SRV open) ~ 0.

The existing ASP model was modified to include the potential use of venting for decay heat removal in the event that both the shutdown cooling and suppression pool cooling modes of RHR fail. This was done by revising sequences involving failure of both RHR cooling modes to also include failure to vent the containment. The probability of failing to vent was assumed to be dominated by human error. A probability of 0.01 is utilized for sequences in which the injection source operates at low pressure and the source of water is separate from the suppression pool.

For sequences in which the injection source takes suction from the suppression pool (such as LPCS or LPCI), an alternate injection source, the CRD pumps or RHR service water, must be aligned for injection following venting. Venting is considered much less reliable in such cases, and an operator error probability of 0.5 was utilized (see NRR Daily Events Evaluation Manual, 1-275-03-?36-01, January 31, 1992). Because of the expected delay in recovering ac power in this event, HPCI was assumed to have transferred to the suppression pool prior to venting.

Case 1. Plant-Centered LOOP at Unit 2. For a postulated plant-centered LOOP at Unit 2 only, offsite power remains available at Unit 1. Power can be recovered to bus 24-1 after the failure of the 2 DG by recovering offsite power or by closing the cross-tie from Unit 1 bus 14-1. Because of the shared dc system at Quad Cities, dc power will remain available for instrumentation even after the Unit 2 batteries are depleted (on average 11.5 h after the postulated LOOP in this event, if offsite power is not recovered by then). Because the SSMP is preferentially powered from Unit 1, it will also be available without operator action to align a power source.

The frequency of plant-centered LOOP and the probability of failing to recover offsite power in the short-term and before battery depletion were estimated to be 8.5×10^{-2}/y, 0.50, and 1.3×10^{-6}, respectively. Because one train of Unit 2 instrumentation is powered from the Unit 1 batteries, sequences involving SSMP success will not proceed to core damage when the Unit 2 batteries are depleted.

As described previously, ac power can also be recovered to bus 24-1 through closure of cross-tie breakers 1421 and 2429. The probability of failing to perform this action before battery depletion was assumed to be 0.12 (ASP nonrecovery class R3, see Appendix A, Sect. A.1 to the 1992 precursor report). This value was chosen because recovery appeared possible in the required time from the control room, but was not considered routine (the value chosen for this failure probability for this case is considered a bounding probability and does not impact the final analysis results).

Modifications to blackout sequence conditional probabilities indicated on the Conditional Core Damage Probability Calculation sheets to reflect the above considerations for this case follow:

Sequence	p(LOOP)	p(RCIC)	p(SSMP)	p(vent)
65	0.07	n/a	n/a	0.5
66	0.07	included	n/a	0.5
67	0.07	included	0.04	n/a
68	0.07	n/a	n/a	0.5
69	0.07	0.042	0.04	n/a
83	invalid sequence*			

*Sequence 83 is not a valid sequence because one train of Quad Cities batteries is charged from Unit 1. Provided power is available to charge the Unit 1 batteries, one train of dc power will remain available at Unit 2 for instrumentation and control functions.

For the dominant sequences shown on the calculation sheets, the above modifications result in the following revised probabilities:

Sequence	Calculation sheet probability	Revised probability
65	2.6×10^{-6}	9.1×10^{-8}
66	5.1×10^{-8}	1.8×10^{-9}
67	4.2×10^{-4}	1.2×10^{-6}
68	2.6×10^{-8}	9.1×10^{-10}
69	1.0×10^{-4}	1.2×10^{-8}
83	7.8×10^{-8}	0*

*Sequence 83 is not a valid sequence because one train of Quad Cities batteries is charged from Unit 1. Provided power is available to charge the Unit 1 batteries, one train of dc power will remain available at Unit 2 for instrumentation and control functions.

The conditional probability estimated for this case is 1.3×10^{-6}.

Case 2a. Dual-unit LOOP with 1 DG Success. For a postulated dual-unit LOOP (primarily grid- and weather-related LOOPs), offsite power is unavailable to both units. The potential availability and unavailability of the 1 DG must be separately considered to correctly address the potential use of the SSMP and recovery of bus 24-1 from bus 14-1.

For this subcase, 1 DG is assumed to have started and provided power to bus 14-1. Because bus 14-1 is powered, power is available to the SSMP and for recovery of bus 24-1 via the cross-tie, similar to case 1. The same failure probability, 0.12, was utilized for failure to power bus 24-1 via the cross-tie. However, unlike case 1, in which offsite power is assumed to exist at bus 14-1, considerable care would be required to shed loads before closing the cross-tie breakers and then to selectively repower bus 14-1 and 24-1 loads to prevent tripping the 1 DG.

For a dual-unit LOOP, the 1/2 DG will power bus 13-1 or 23-1, depending on which first experiences a loss of power (assuming a LOCA does occur simultaneously). If bus 13-1 experiences the LOOP first, the 1/2 DG will be available to power that bus, and it may be possible to power bus 23-1 from the 1/2 DG. However, this analysis assumed that the 1/2 DG would fail due to loss of cooling water flow when its output was transferred from bus 13-1 to 23-1, because the 1/2 DGCWP would also switch to Unit 2 and lock out.

The frequency of a dual-unit LOOP with 1 DG successful ($p_{DG\ success}$ = 0.95) and the probability of failing to recover offsite power in the short-term and before battery depletion were estimated to be 1.6×10^{-2}/year, 0.66, and 0.23, respectively, using the approach described for case 1. Because one train of Unit 2 instrumentation is powered from the Unit 1 batteries, sequences involving SSMP success will not proceed to core damage when the Unit 2 batteries are depleted.

Modifications to blackout sequence conditional probabilities indicated on the Conditional Core Damage Probability Calculation sheets to reflect the above considerations for this subcase follow:

Sequence	p(LOOP)	p(RCIC)	p(SSMP)	p(vent)
65	0.013	n/a	n/a	0.5
66	0.013	included	n/a	0.5
67	0.013	included	0.04	n/a
68	0.013	n/a	n/a	0.5
69	0.013	0.042	0.04	n/a
83	invalid sequence*			

*Sequence 83 is not a valid sequence because one train of Quad Cities batteries is charged from Unit 1. Provided power is available to charge the Unit 1 batteries, one train of dc power will remain available at Unit 2 for instrumentation and control functions.

For the dominant sequences shown on the calculation sheets, the above modifications result in the following revised probabilities:

Sequence	Calculation sheet probability	Revised probability
65	3.3×10^{-6}	2.2×10^{-8}
66	6.5×10^{-8}	4.2×10^{-10}
67	5.4×10^{-4}	2.8×10^{-7}
68	3.3×10^{-8}	2.2×10^{-10}
69	1.3×10^{-4}	2.8×10^{-9}
83	1.8×10^{-2}	0*

*Sequence 83 is not a valid sequence because one train of Quad Cities batteries is charged from Unit 1. Provided power is available to charge the Unit 1 batteries, one train of dc power will remain available at Unit 2 for instrumentation and control functions.

The conditional probability estimated for this subcase is 3.1×10^{-7}.

Case 2b. Dual-unit LOOP with 1 DG Failure. This subcase assumes that the 1 DG fails to provide power to bus 14-1. Because bus 14-1 is not powered, the SSMP is unavailable to provide makeup to the core and the cross-tie cannot be used to restore power to bus 24-1.

As in case 2a, the 1/2 DG may be available for powering battery chargers and RHR pumps if bus 13-1 experiences the LOOP before bus 23-1. Because it is equally likely that the LOOP will occur first on either bus, the probability that the 1/2 DG will fail to provide power is 0.55 (assuming a nominal DG failure probability of 0.05). The probability of failing to recover offsite power before battery depletion was multiplied by this factor to address the potential availability of the 1/2 DG to power instrumentation and control loads via the Unit 1/2 battery chargers.

The frequency of a dual-unit LOOP with 1 DG failed ($p_{DG\ fail} = 0.05$) and the probability of failing to recover offsite power in the short-term and before battery depletion were estimated to be 8.5×10^{-4}/year, 0.66, and 0.23, respectively, using the approach described in Case 1.

Modifications to blackout sequence conditional probabilities indicated on the Conditional Core Damage Probability Calculation sheets to reflect the above considerations for this subcase follow:

Sequence	p(LOOP)	p(RCIC)	p(SSMP)	p(vent)
65	7.0×10^{-4}	n/a	n/a	0.5
66	7.0×10^{-4}	included	n/a	0.5
67	7.0×10^{-4}	included	unavail	n/a
68	7.0×10^{-4}	n/a	n/a	0.5
69	7.0×10^{-4}	0.042	unavail	n/a
83	7.0×10^{-4}	n/a	n/a	n/a

For the dominant sequences shown on the calculation sheets, the previous modifications result in the following revised probabilities:

Sequence	Calculation sheet probability	Revised probability
65	3.0×10^{-6}	1.1×10^{-9}
66	5.9×10^{-8}	2.1×10^{-11}
67	4.9×10^{-4}	3.5×10^{-7}
68	3.0×10^{-8}	1.1×10^{-11}
69	1.2×10^{-4}	3.5×10^{-9}
83	8.3×10^{-2}	5.8×10^{-5}

The conditional probability estimated for this subcase is 5.8×10^{-5}.

Combining the probabilities for cases 1, 2a, and 2b results in an estimated overall conditional core damage probability of 6.0×10^{-5}.

A.2.5 Analysis Results

The conditional core damage probability estimated for this event is 6.0×10^{-5}. The dominant core damage sequence, highlighted on the event tree shown in Fig. A.2.3, involves a postulated dual-unit LOOP (primarily grid- or weather-related) with subsequent failure of all three Quad Cities DGs (see case 2b, sequence 83), and failure to recover offsite power before battery depletion. In the dominant sequence, the 1/2 DG is postulated to fail due to a loss of cooling water following its alignment to Unit 2 (the postulated dual-unit LOOP affects Unit 2 first), the 2 DG also fails due to loss of cooling water after its DGCWP bearing oil is depleted, and the 1 DG fails for unspecified reasons (random failure).

The core damage probability for the event is strongly influenced by the probability of the dominant sequence. The next highest conditional probability sequence has a probability of 1.2×10^{-6} and involves a postulated LOOP at Unit 2 only, with failure of high-pressure injection before recovery of ac power. The dominant sequence probability is dictated by the LOOP probability distributions described in NUREG-1032 and assumptions concerning the expected lifetime of the 2 DGCWP.

The continued availability of dc power from the Unit 1 batteries in Cases 1 and 2a significantly affects the results of the analysis. Because Unit 1 dc power remains available in these cases, the potential use of the bus 14-1-24-1 cross-tie does not contribute to the analysis results.

The potential use of the SSMP also influences the results of the analysis. Consideration of the SSMP along with the shared dc system, without consideration of RCIC and containment venting, results in a core damage probability estimate within 2% of the value calculated for the event. Assumptions concerning the potential use of RCIC following a stuck-open relief valve and containment venting have essentially no impact on the analysis results.

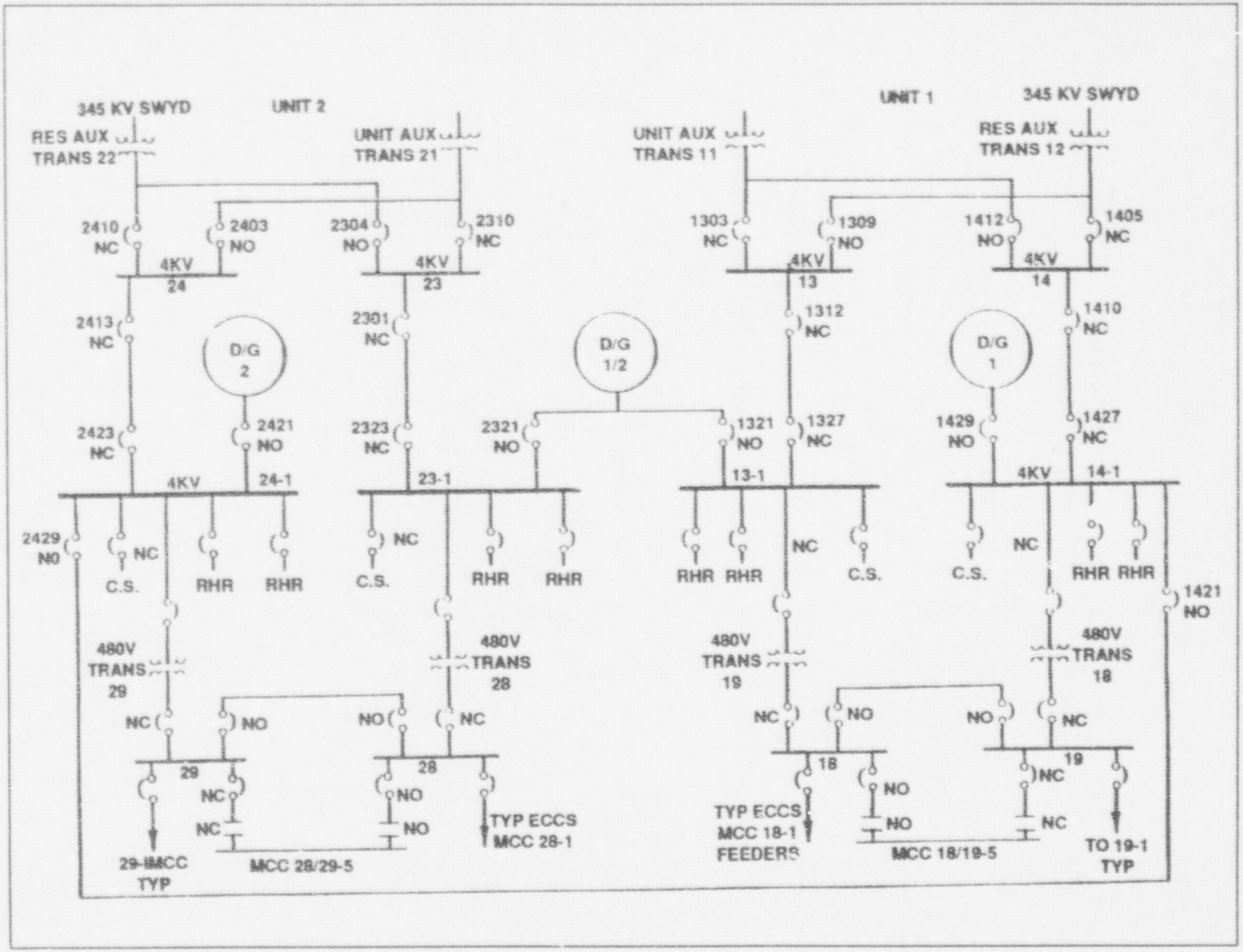


Fig. A.2.2 Simplified diagram of Quad Cities emergency power system

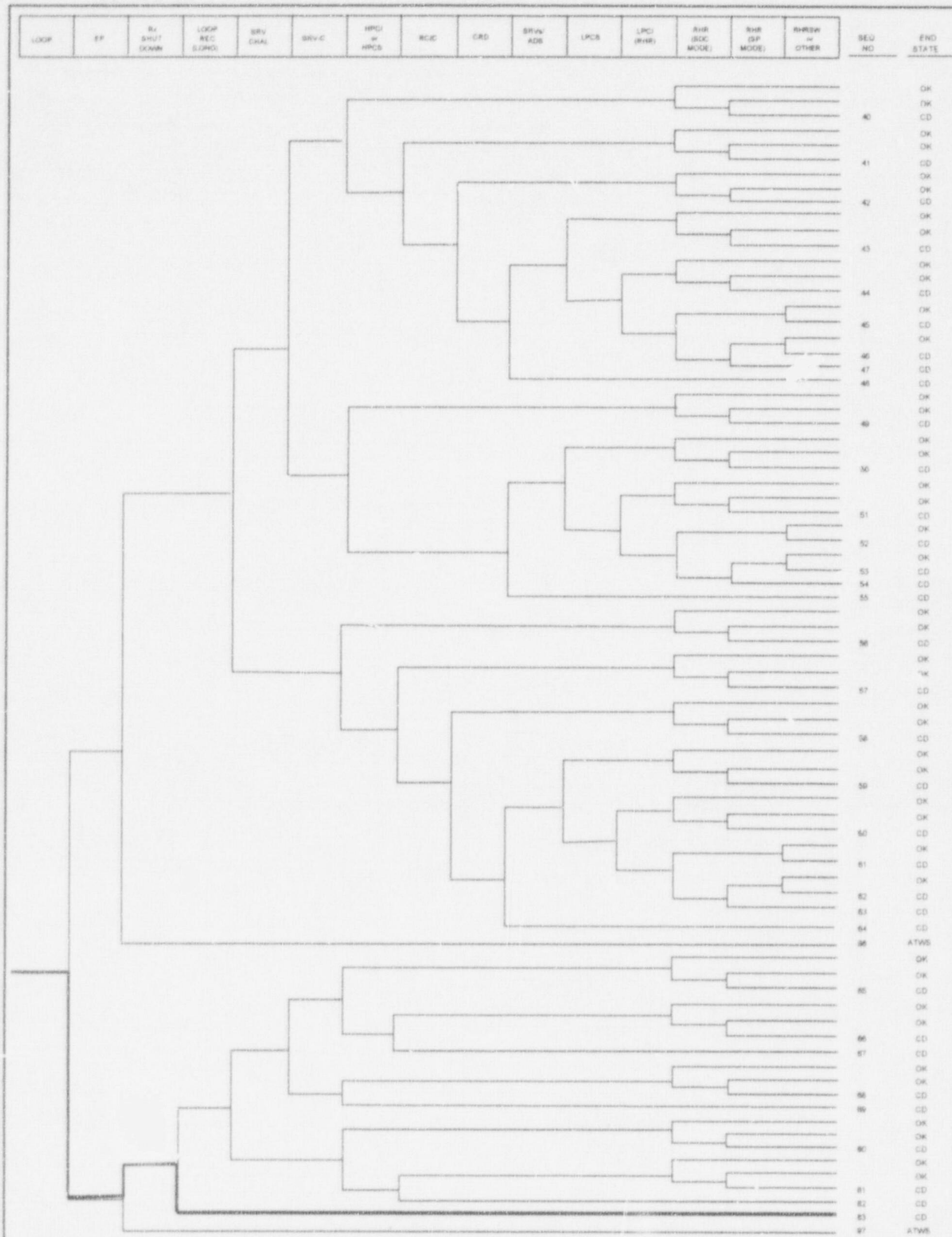


Fig. A.2.3 Dominant core damage sequence for LER Nos. 265/93-010 and -012

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 265/93-010
Event Description: Emergency power system unavailable (case 1)
Event Date: 04/22/93
Plant: Quad Cities 2

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	5.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	5.3E-04(1)
Total	5.3E-04(1)
ATWS	
LOOP	1.5E-05
Total	1.5E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	4.2E-04(1)	2.4E-01
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	1.0E-04(1)	3.5E-01
65	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	2.6E-06(1)	5.7E-02
83	LOOP EMERG.POWER -rx.shutdown/ep EP.REC	CD	7.8E-08(1)	6.0E-02
66	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	5.1E-08(1)	4.0E-02
68	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	2.6E-08(1)	5.7E-02
97	LOOP EMERG.POWER rx.shutdown	ATWS	1.5E-05	5.0E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
65	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	2.6E-06	5.7E-02
66	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	5.1E-08	4.0E-02
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	4.2E-04	2.4E-01
68	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	2.6E-08	5.7E-02
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	1.0E-04	3.5E-01
83	LOOP EMERG.POWER -rx.shutdown/ep EP.REC	CD	7.8E-08	6.0E-02
97	LOOP EMERG.POWER rx.shutdown	ATWS	1.5E-05	5.0E-01

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\bwrcseal.cmp
BRANCH MODEL: s:\asp\prog\models\quadcit2.sl1
PROBABILITY FILE: s:\asp\prog\models\bwr_csl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	5.3E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	5.3E-01 > 5.0E-01(2)	
Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
pcs/trans	1.7E-01	1.0E+00	
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	1.0E-02	1.0E+00	
EMERG.POWER	2.9E-03 > 1.0E+00	8.0E-01 > 1.0E+00(2)	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed(2)		
Train 2 Cond Prob:	5.7E-02 > Failed(2)		
EP.REC	4.9E-02 > 1.3E-06	1.0E+00 > 1.2E-01(2)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.9E-02 > 1.3E-06(2)		
fw/pcs.trans	2.9E-01	3.4E-01	
fw/pcs.loca	4.0E-02	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	3.0E-03	3.4E-01	
lpci(rhr)/lpcs	1.0E-03	7.1E-01	
rhr(sdc)	2.1E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
rhr(spcool)/rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/-lpci.rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/lpci.rhr(sdc)	9.3E-02	1.0E+00	
rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file
** forced

Notes:

1. See Modeling Assumptions for modifications to this sequence conditional probabilities.
2. See Modeling Assumptions for the development of this probability value.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 265/93-010
Event Description: Emergency power system unavailable (case 2a)
Event Date: 04/22/93
Plant: Quad Cities 2

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	6.6E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	1.9E-02(1)
Total	1.9E-02(1)
ATWS	
LOOP	2.0E-05
Total	2.0E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
83	LOOP EMERG.POWER -rx.shutdown/ep EP.REC	CD	1.8E-02(1)	7.9E-02
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	5.4E-04(1)	3.2E-01
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	1.3E-04(1)	4.5E-01
65	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.3E-06(1)	7.4E-02
66	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	6.5E-08(1)	5.1E-02
68	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.3E-08(1)	7.4E-02
97	LOOP EMERG.POWER rx.shutdown	ATWS	2.0E-05	6.6E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
65	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.3E-06	7.4E-02
66	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	6.5E-08	5.1E-02
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	5.4E-04	3.2E-01
68	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.3E-08	7.4E-02
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	1.3E-04	4.5E-01
83	LOOP EMERG.POWER -rx.shutdown/ep EP.REC	CD	1.8E-02	7.9E-02
97	LOOP EMERG.POWER rx.shutdown	ATWS	2.0E-05	6.6E-01

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\bwrcseal.cmp
BRANCH MODEL: s:\asp\prog\models\quadcit2.sl1
PROBABILITY FILE: s:\asp\prog\models\bwr_csil.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	5.3E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	5.3E-01 > 6.6E-01(2)	
Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
pcs/trans	1.7E-01	1.0E+00	
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	1.0E-02	1.0E+00	
EMERG.POWER	2.9E-03 > 1.0E+00	8.0E-01 > 1.0E+00(2)	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed(2)		
Train 2 Cond Prob:	5.7E-02 > Failed(2)		
EP.REC	4.9E-02 > 2.3E-01	1.0E+00 > 1.2E-01(2)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.9E-02 > 2.3E-01(2)		
fw/pcs.trans	2.9E-01	3.4E-01	
fw/pcs.loca	4.0E-02	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	3.0E-03	3.4E-01	
lpci(rhr)/lpcs	1.0E-03	7.1E-01	
rhr(sdc)	2.1E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
rhr(spcool)/rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/-lpci.rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/lpci.rhr(sdc)	9.3E-02	1.0E+00	
rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file
** forced

Notes:

1. See Modeling Assumptions for modifications to this sequence conditional probabilities.
2. See Modeling Assumptions for the development of this probability value.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 265/93-010
Event Description: Emergency power system unavailable (case 2b)
Event Date: 04/22/93
Plant: Quad Cities 2

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	6.6E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	8.4E-02(1)
Total	8.4E-02(1)
ATWS	
LOOP	2.0E-05
Total	2.0E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
83	LOOP EMERG.POWER -rx.shutdown/ep EP.REC	CD	8.3E-02(1)	3.6E-01
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	4.9E-04(1)	3.0E-01
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	1.2E-04(1)	4.3E-01
65	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.0E-06(1)	7.1E-02
66	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	5.9E-08(1)	4.9E-02
68	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.0E-08(1)	7.1E-02
97	LOOP EMERG.POWER rx.shutdown	ATWS	2.0E-05	6.6E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
65	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.0E-06	7.1E-02
66	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	5.9E-08	4.9E-02
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	4.9E-04	3.0E-01
68	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close -hpci rhr(sdc)/-lpci rhr(spcool)/-lpci.rhr(sdc)	CD	3.0E-08	7.1E-02
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	1.2E-04	4.3E-01
83	LOOP EMERG.POWER -rx.shutdown/ep EP.REC	CD	8.3E-02	3.6E-01
97	LOOP EMERG.POWER rx.shutdown	ATWS	2.0E-05	6.6E-01

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\bwrcseal.cmp
BRANCH MODEL: s:\asp\prog\models\quadcit2.sl1
PROBABILITY FILE: s:\asp\prog\models\bwr_csl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	5.3E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	5.3E-01 > 6.6E-01(2)	
Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
pcs/trans	1.7E-01	1.0E+00	
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv close	1.0E-02	1.0E+00	
EMERG.POWER	2.9E-03 > 1.0E+00	8.0E-01 > 1.0E+00(2)	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed(2)		
Train 2 Cond Prob:	5.7E-02 > Failed(2)		
EP.REC	4.9E-02 > 2.3E-01	1.0E+00 > 5.5E-01(2)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.9E-02 > 2.3E-01(2)		
fw/pcs.trans	2.9E-01	3.4E-01	
fw/pcs.loca	4.0E-02	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	3.0E-03	3.4E-01	
lpci(rhr)/lpcs	1.0E-03	7.1E-01	
rhr(sdc)	2.1E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
rhr(spcool)/rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/-lpci.rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/lpci.rhr(sdc)	9.3E-02	1.0E+00	
rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file
** forced

Notes:

1. See Modeling Assumptions for modifications to this sequence conditional probabilities.
2. See Modeling Assumptions for the development of this probability value.

A.3 LER No. 289/93-002

Event Description: Both Residual Heat Removal Heat Exchangers Unavailable

Date of Event: January 29, 1993

Plant: Three Mile Island 1

A.3.1 Summary

Three Mile Island 1 (TMI-1) was operating at 100% power on January 29, 1993, when an operator aligned river water system valves to bypass both decay heat service (DHS) coolers. The coolers remained unavailable for about 3 h. With the DHS coolers unavailable, it would not have been possible to remove heat from several safety-related systems had they been demanded. The conditional core damage probability estimated for this event is 3.1×10^{-6}. The relative significance of this event compared to other postulated events at Three Mile Island 1 is shown in Fig. A.3.1.

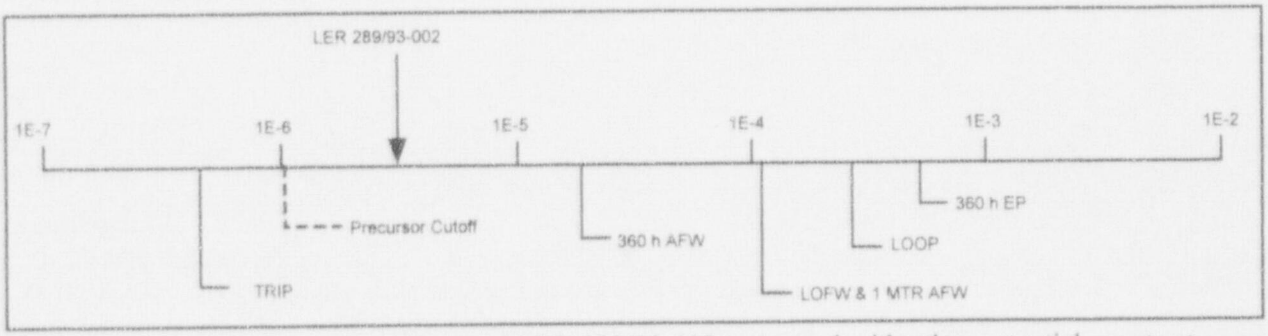


Fig. A.3.1 Relative event significance of LER 289/93-002 compared with other potential events at Three Mile Island 1

A.3.2 Event Description

During execution of a surveillance instruction involving operation of decay heat river water (DHRW) pumps, an auxiliary operator simultaneously bypassed DHS coolers DC-C-2A and DC-C-2B. The DHS coolers serve as the heat sink for the decay heat closed cooling water (DCCW) system. Loads on the DCCW system include decay heat removal (DHR) coolers, DHR pump motor and bearing coolers, DCCW pump bearing coolers, reactor building spray (BS) pump motor and bearing coolers, and two of three makeup [charging/high-pressure injection (HPI)] pump motor, bearing, and gear reducer coolers.

After ~2.5 h, a control room operator discovered the error while evaluating the steps taken for the surveillance instruction. The DHS coolers were returned to service ~0.5 h later.

In the LER, the licensee discussed the potential plant response to a large-break loss-of-coolant accident (LOCA) with the DHS coolers isolated. They concluded that core and containment response would be unaffected before sump recirculation. Following initiation of sump recirculation, DHR would be provided by the reactor building emergency cooling fan coolers in conjunction with the recirculation flow from the low-pressure injection (LPI) and reactor BS pumps. They also concluded, based on the licensee's engineering judgement, that at least 30 min was available to restore cooling to the LPI and spray pumps. The impact of the isolated DHS coolers on sump recirculation following a small-break LOCA was not discussed in the LER.

A.3.3 Modeling Assumptions

In the sump-recirculation phase following a small-break LOCA, flow from the discharge of the DHR coolers is directed to the suction of the makeup HPI pumps to provide adequate net positive suction head for HPI pump operation. This water is cooled to prevent damaging the makeup pumps (the TMI-1 final safety analysis report indicates that the design temperature of the makeup pumps is 200°F). With the DHS coolers isolated, makeup and LPI pump cooling water temperatures would exceed design temperatures during sump recirculation following a small-break LOCA, resulting in failure of high-pressure recirculation (HPR). The time to pump failure cannot be accurately estimated based on available data, although it may be as long as several hours.

The event was modeled as a 3-h unavailability of HPR. Because of the uncertainty in the available time before pump damage and the potential radiological conditions at the closed valves following initiation of HPR, recovery of the isolated DHS coolers (through operation of the two 18-in. manual valves in each train) was assumed not to be possible in the analysis. The low temperature of the borated water storage tank (BWST) fluid before sump recirculation was assumed not to impact HPI pump operation in the injection phase.

A.3.4 Analysis Results

The conditional core damage probability estimated for this event is 3.1×10^{-6}. The dominant sequence, highlighted on the event tree in Fig. A.3.3 , involves a postulated small-break LOCA, success of reactor trip, auxiliary feedwater, and HPI functions followed by failure of HPR.

The core damage probability estimated for this event is strongly influenced by the probability of not recovering the DHR service coolers used in the analysis. For example, if a nonrecovery probability of 0.34 (ASP Recovery Class R2, the failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from the central room did not appear possible) is assumed, the conditional core damage probability is reduced to 1.1×10^{-6}.

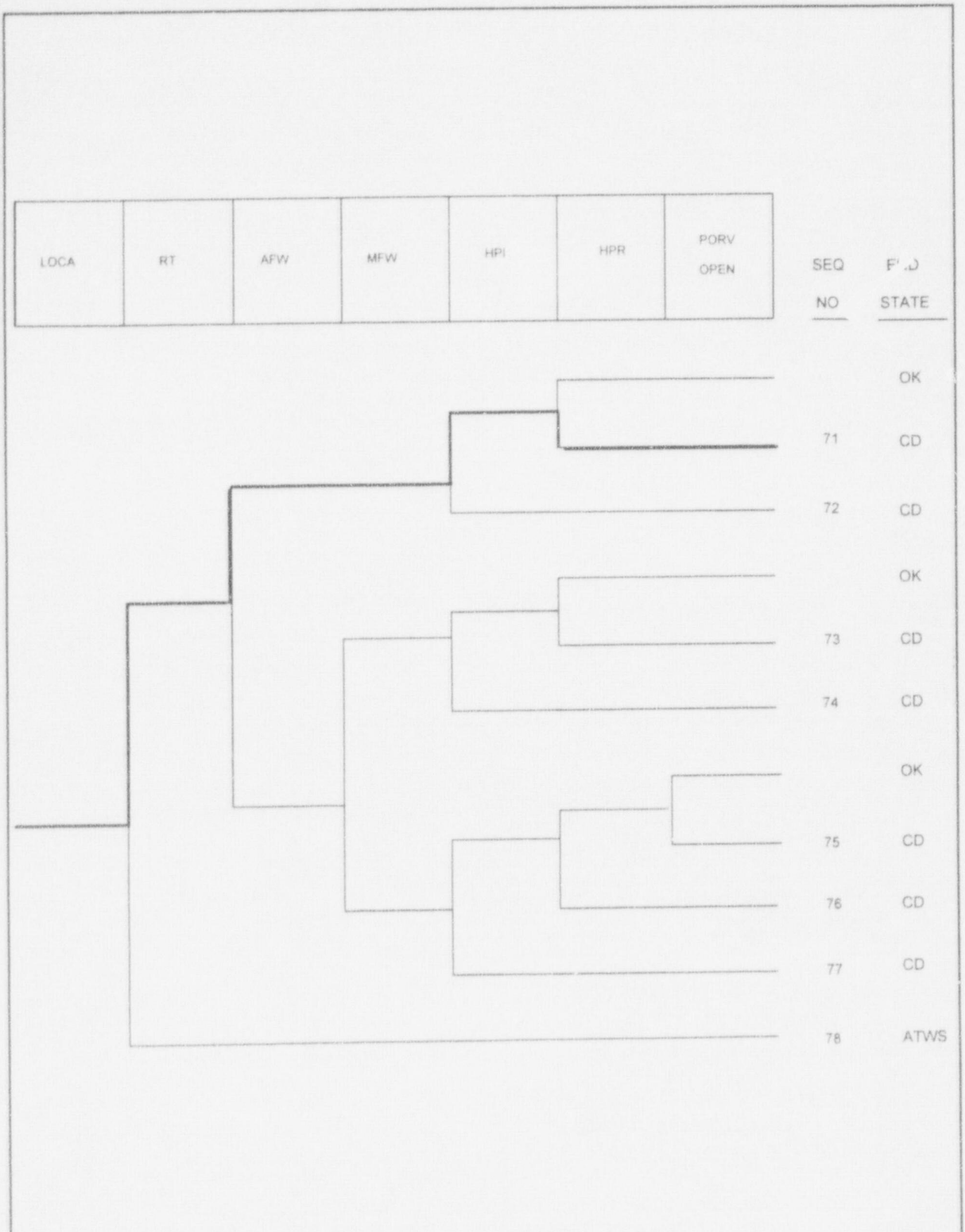


Fig. A.3.2 Dominant core damage sequence for LER 289/93-002

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 289/93-002
Event Description: Both RHR heat exchangers unavailable
Event Date: January 29, 1993
Plant: Three Mile Island 1

UNAVAILABILITY, DURATION= 3

NONRECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	3.9E-04
LOOP	2.6E-05
LOCA	3.1E-06

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.9E-08
LOOP	1.7E-08
LOCA	3.1E-06
Total	3.1E-06
ATWS	
TRANS	0.0E+00
LOOP	0.0E+00
LOCA	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
71	loca -rt -afw -hpi HPR/-HPI	CD	3.1E-06	4.3E-01
16	trans -rt afw mfw -hpi(f/b) HPR/-HPI	CD	1.5E-08	8.8E-02
44	loop -rt/loop -emerg.power afw -hpi(f/b) HPR/-HPI	CD	1.5E-08	1.4E-01
11	trans -rt -afw porv.or.srv.chall porv.or.srv.reseat -hpi HPR/-HPI	CD	3.4E-09	1.1E-02
73	loca -rt afw -mfw -hpi HPR/-HPI	CD	1.7E-09	1.1E-01
51	loop -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall seal.loca -ep.rec(sl) -hpi HPR/-HPI	CD	1.1E-09	4.2E-01
41	loop -rt/loop -emerg.power -afw porv.or.srv.chall porv.or.srv.reseat -hpi HPR/-HPI	CD	2.3E-10	5.8E-03
76	loca -rt afw mfw -hpi HPR/-HPI	CD	1.2E-10	3.8E-02
46	loop -rt/loop emerg.power -afw/emerg.power power.or.srv.chall porv.or.srv.reseat/emerg.power seal.loca -ep.rec(sl) -hpi HPR/ -HPI	CD	9.1E-11	4.2E-01
13	trans -rt afw -mfw porv.or.srv.chall porv.or.srv. reseat-hpi HPR/-HPI	CD	1.9E-12	2.7E-03

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
11	trans -rt -afw porv.or.srv.chall porv.or.srv.reseat -hpi HPR/ -HPI	CD	3.4E-09	1.1E-02
13	tra[illegible] -t afw -mfw porv.or.srv.chall porv.or.srv.reseat -hpi [illegible]/ HPI	CD	1.9E-12	2.7E-03
16	[illegible]ans -rt afw mfw -hpi(f/b) HPR/-HPI	CD	1.5E-08	8.8E-02
41	loop -rt/loop -emerg.power -afw porv.or.srv.chall porv.or.srv. reseat -hpi HPR/-HPI	CD	2.3E-10	5.8E-03
44	loop -rt/loop -emerg.power afw -hpi(f/b) HPR/-HPI	CD	1.5E-08	1.4E-01
46	loop -rt/loop emerg.power -afw/emerg.power porv.or.srv.chall -porv.or.srv.reseat/emerg.power seal.loca -ep.rec(sl) -hpi HPR/ -HPI	CD	9.1E-11	4.2E-01
51	loop -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall seal.loca -ep.rec(sl) -hpi HPR/-HPI	CD	1.1E-09	4.2E-01
71	loca -rt -afw -hpi HPR/-HPI	CD	3.1E-06	4.3E-01
73	loca -rt afw -mfw -hpi HPR/-HPI	CD	1.7E-09	1.1E-01

1

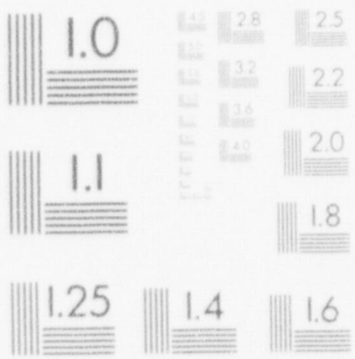
IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)



150mm

6

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

```
76   loca -rt  afw  mfw -hpi  HPR/-HPI                        CD          1.2E-10     3.8E-02
```

** nonrecovery 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:      s:\asp\prog\models\pwrdseal.cmp
BRANCH MODEL:        s:\asp\prog\models\tmi1.sl1
PROBABILITY FILE:    s:\asp\prog\models\pwr_bsl1.pro
```

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	1.3E-04	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
afw	2.3E-03	2.6E-01	
afw/emerg.power	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
porv.or.srv.chall	8.0E-02	1.0E+00	
porv.or.srv.reseat	1.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	1.0E-02	1.0E+00	
seal.loca	4.6E-02	1.0E+00	
ep.rec(sl)	5.7E-01	1.0E+00	
ep.rec	1.6E-01	1.0E+00	
hpi	1.0E-03	8.4E-01	
hpi(f/b)	1.0E-03	8.4E-01	1.0E-02
HPR/-HPI	1.5E-04 > 1.0E+00 **	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.5E-02		

* branch model file
** forced

A.4 LER No. 293/93-004

Event Description: Weather-Induced Loss-of-offsite Power, Vessel Pressure/Temperature Limits Violated

Date of Event: March 13, 1993

Plant: Pilgrim

A.4.1 Summary

Pilgrim was operating at 100% power when a severe coastal storm caused a loss-of-load scram and subsequent loss of the normal power supply to the plant. Difficulties were experienced during cooldown, when the reactor repressurized to at least 820 psig, with vessel bottom head temperature declining to ~110°F. The conditional core damage probability estimated for this event is 4.6×10^{-6}. The relative significance of this event compared to other postulated events at Pilgrim is shown in Fig. A.4.1.

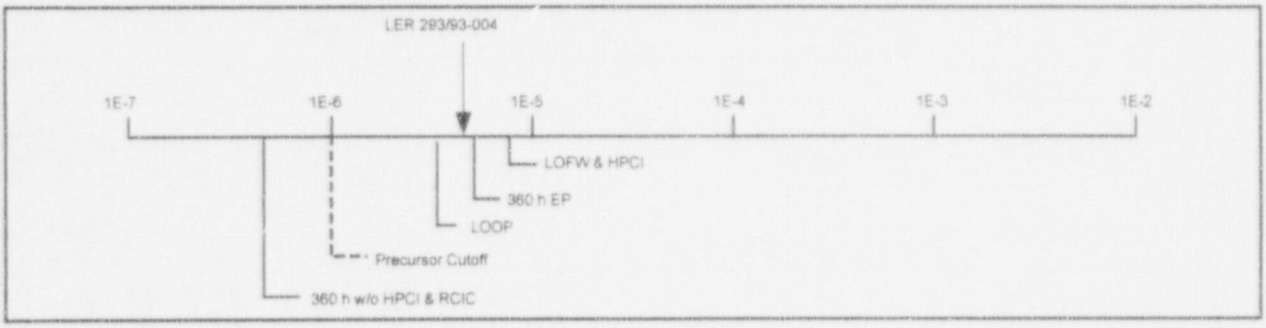


Fig. A.4.1 Relative event significance of LER 293/93-004 compared with other potential events at Pilgrim.

A.4.2 Event Description

On March 13, 1993, at 1628 hours, Pilgrim experienced a load rejection from 100% power. Wind-driven snow and ice accumulated on switchyard insulators, causing a fault that resulted in the automatic opening of switchyard circuit breakers 104 and 105 (see Fig. A.4.2). This action isolated the main transformer from the switchyard and initiated automatic turbine-generator and reactor trips. The loss of the main generator resulted in the loss of the unit auxiliary transformer (UAT). Most loads fed from the UAT fast-transferred to the alternate source, the start-up transformer (SUT). However, a breaker control failure prevented 4160-V ac bus A3 from transferring. Loads fed from bus A3, including the A recirculation pump motor-generator set, the A circulating water pump, the A main turbine auxiliary oil pump, and the 480-V ac bus B3, were deenergized. Deenergization of 480-V ac bus B3, in turn, removed power from reactor protection system (RPS) bus A.

The loss of the A main turbine auxiliary oil pump resulted in the closure of the turbine bypass valves. With these valves closed, two of the main steam relief valves opened briefly for pressure relief. Protective breakers for 120-V ac safeguards buses A and B tripped due to improper trip settings. As a result, the auto starts for pumps in the salt service water (SSW) system and the reactor building closed-cooling-water (RBCCW) system were disabled. Manual operation of these pumps was not affected.

Twelve minutes after the trip, at 1640 hours, switchyard circuit breaker 102 opened automatically due to a flashover of the energized side of switchyard circuit breaker 105. The flashover initiated an isolation and deenergization of line 355.

At 1650 hours, bus A3 was reenergized from the SUT by manually closing breaker 304. This reenergized the A turbine auxiliary lube oil pump, which enabled the turbine bypass valves to open. At 1655 hours, emergency diesel generators (EDGs) 1 and 2 were started and loaded onto their respective buses. Over the next 10 min, 120-V ac safeguards buses A and B were reenergized.

At 1710 hours, the remaining switchyard circuit breaker (105) opened, which deenergized the SUT. Buses A1, A2, A3, and A4 were deenergized. Buses A5 and A6 continued to be fed from the EDGs. At 2155 hours, switchyard breaker 103 was reclosed. The closing of breaker 103 reenergized the SUT from offsite power. Buses A3 and A4 were then realigned to the SUT. Buses A5 and A6 remained aligned to the EDGs.

By 2300 hours, the reactor coolant system (RCS) had been isolated. The main steam isolation valves (MSIVs) had been closed in response to a loss of condensate flow. The reactor-core-isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems were returned to standby service. HPCI had been providing RCS pressure control. With the RCS isolated, pressure began to rise at ~10 psig/min. By 2330 hours, the reactor vessel (RV) pressure was 510 psig, and the bottom head temperature was 110°F. It was determined 6 d later that this condition violated Technical Specifications. The RV water level was at 48 in. This is above the high-level trip set point for HPCI and RCIC; therefore, they could not be placed into service. At 2336 hours, an RPS scram signal was generated when RV pressure reached 572 psig with the MSIVs closed.

Between 0015 and 0100 hours, the operating staff discussed the use of the main steam relief valves to reduce RV pressure. During this time period, RV pressure gradually rose to 820 psig. At 0100 hours, four of the main steam relief valves were opened for about 5 min to reduce RV pressure. This also reduced RV water level below the HPCI and RCIC trip set points. By 0121 hours, HPCI had been placed into service for RV pressure control, and RCIC was placed into service for RV level control. At 0245 hours, the technical specification pressure-temperature limit for the RV was no longer exceeded. RV pressure was 345 psig, and bottom head temperature was 92°F.

At 0322 and 0345 hours, buses A5 and A6 were realigned to the SUT, and their respective EDGs were returned to standby service.

A.4.3 Additional Event-Related Information

Pilgrim has four nonsafety-related 4160-V ac buses (see Fig. A.4.2). Each of these nonsafety-related buses can be powered from the UAT, which is energized by the main generator output, or the SUT, which is connected to two offsite 345-kV lines. Upon loss of the UAT, the nonsafety-related buses are automatically fast-transferred to the SUT. The two safety-related 4160-V ac buses can also receive power from the UAT and SUT. In addition, they can be powered from a 23-kV offsite line or from the blackout diesel generator (BODG). The BODG is a nonsafety-related supply that is not dependent on any other onsite systems for its operation. It can be started manually from the control room and is capable of providing power to one of the two safeguards buses and associated loads for blackout events without a concurrent loss-of-coolant accident (LOCA) event. Upon the loss of the UAT, the safety-related buses are fast-transferred to the SUT. If the SUT is lost, the buses automatically load onto the safeguards EDGs. If an EDG fails, the breaker for the 23-kV line automatically closes 2 s later. If this should fail, the BODG can be manually aligned to one of the safeguards buses and loaded as required.

A.4.4 Modeling Assumptions

Typically the Accident Sequence Precursor (ASP) Program has selected events for analysis as a loss-of-offsite power (LOOP) event if the LOOP required the EDGs to be relied on for safeguards power for an extended period of time. In this event, only the preferred offsite power source was lost; the 23-kV line was available throughout the event. Although the 23-kV line was available, it was not used because it only closes in if an EDG fails. The plant response was typical of what most plants would experience during a total LOOP. The ASP event tree for a LOOP was used with one modification; the 23-kV line was treated as another source of emergency power.

As with all ASP Program analyses, only the observed failures were included in the modeling of the event. The observed successes were not credited. Equipment that was observed to successfully operate was modeled with nominal failure rates. This is consistent with the program objective of determining the decrease in core damage margin due to the observed events or conditions. As noted in Chap. 2, the data utilized in this analysis are not specific to the Pilgrim unit but are representative of all units of similar design. Application of plant-specific data to the model may result in an increase or decrease in the calculated conditional core damage probability.

This event was modeled as a severe-weather-induced LOOP because it was caused by a widespread ice storm. This is consistent with the categorization of LOOPs in NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*. The values for the short-term LOOP nonrecovery probability and the long-term nonrecovery of emergency power probability were both modified using the models described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89-11, August 1989. These models are based on the results of the data distributions contained in NUREG-1032. The output of this program is a short-term LOOP nonrecovery (within the first 30 min) and long-term nonrecovery (before battery depletion). These values are based on the typical duration of a LOOP caused by a severe weather condition. This results in a short-term nonrecovery of 0.9 and a long-term nonrecovery of 5.5×10^{-2}.

The BODG described in the previous section was included in the modeling. According to information provided by the licensee, the failure rate of the BODG is 0.075; this accounts for the operator failure rate to align the BODG. However, this does not account for all of the common cause failures of the two EDGs and the BODG. A calculation that incorporates all potential common cause failures results in a total failure rate for all of the DGs of 2.7×10^{-4}. To determine the failure rate for the BODG, the total failure (2.7×10^{-4}) is divided by the failure rate of the EDGs ($0.05 \times 0.057 = 2.9 \times 10^{-3}$). This results in a failure rate for the BODG of 0.093 ($2.7 \times 10^{-4}/2.9 \times 10^{-3}$). This rate is slightly higher than the value provided by the licensee, and is similar to the typical value of 0.1 used by the ASP Program.

Procedures direct the operators to place the BODG into service if the EDGs fail. It is unlikely that the operators would be able to recover the EDGs within the first half-hour after the LOOP if the BODG failed to be loaded. Therefore, the standard emergency power nonrecovery value was changed from 0.8 to 1.0.

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 345-kV lines were 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 23-kV line. Breaker failures and control system failures were assumed to be not significant given the high unavailability of the line under these conditions. The nonrecovery value for the EDGs was multiplied by 0.43 to incorporate the 23-kV line.

Although the EDGs were started and loaded 10 min before the actual loss of the 345-kV lines, the EDG probabilities were left at their nominal values. This was done because the same basic failure mechanisms are in place regardless of whether the EDGs are started before or immediately after the LOOP. In either case, the EDG has to start. The only difference between the manual start and the potential automatic start from the LOOP is the elimination of the potential failure of the automatic signal. It was assumed that this did not significantly affect the EDG failure probability. It was also assumed that insufficient time was available between the time that the EDGs were started and when the LOOP occurred (15 min) for significant recovery actions to be performed. As a result, the mean-time-to-repair of the EDGs was assumed to be unaffected by the 15-min period. Had the EDGs been started early enough that significant recovery actions could have been performed before the LOOP event, a modification of the EDG mean-time-to-repair could have decreased the conditional core damage probability for the event. However, the potential for the trip of the running and paralleled EDGs at the time of the LOOP would also need to be considered.

The loss of the A and B safeguards buses resulted in the loss of the automatic start of the SSW and the RBCCW. However, power was restored to these buses within 28 min. If power had not been restored before these two systems were needed, manual starting of the pumps would have been required. Considering the time period available to start the pumps, the operator failure probability was not modified.

The existing ASP model was modified to include the potential use of containment venting for decay heat removal if both residual heat removal (RHR)/suppression pool cooling (SPC) and RHR/shutdown cooling (SDC) fail. This was done by revising the dominant sequences involving failure of both the RHR cooling modes to also include failure to vent the containment. The probability of failing to vent was assumed to be dominated by human error. A probability of 0.01 was used for sequences in which the injection source operates at low pressure and uses a source of water that is separate from the suppression pool.

For sequences in which the injection source takes suction from the suppression pool (such as LPCS or LPCI), an alternate injection source, the control rod drive pump or the essential SW (RHRSW in the ASP models), must be aligned for injection following venting. Venting is considered much less reliable in such cases; an operator error probability of 0.5 was used (see *NRR Daily Events Evaluation Manual*, 1-275-03-336-01, January 31, 1992).

The licensee has performed a Modular Accident Analysis Program (MAAP) analysis of a trip with one stuck-open SRV and all HPI systems inoperable. The results of this analysis indicate that the RV will depressurize rapidly enough to allow LPI systems to inject and, as a result, prevent core damage. Because one or more stuck-open SRVs will perform the same function as the automatic depressurization system (ADS) system, the ADS failure rate for sequences 49 through 55 was set to zero.

The dominant core damage sequences from the calculation sheets were modified as follows to incorporate suppression pool venting and depressurization via a stuck-open SRV.

Sequence	p(sequence from calculation sheets)	p(ADS)	p(vent)	p(sequence)
40	4.8×10^{-6}	1.0	0.01	4.8×10^{-8}
48	1.9×10^{-7}	1.0	1.0	1.9×10^{-7}
55	3.0×10^{-6}	0.0	1.0	0
67	9.2×10^{-7}	1.0	1.0	9.2×10^{-7}
69	2.9×10^{-7}	1.0	1.0	2.9×10^{-7}
83	3.2×10^{-6}	1.0	1.0	3.2×10^{-6}
			Total	4.6×10^{-6}

No analytical evaluation was made of potential consequences of the RV repressurization that occurred during this event.

A.4.5 Analysis Results

The conditional core damage probability estimated for this event is 4.6×10^{-6}. The dominant core damage sequence is highlighted on the event tree in Fig. A.4.3. Sequence 83 involves a failure to recover power from the EDGs or 23-kV line following the LOOP, and failure to recover offsite power before battery depletion.

Inclusion of the BODG and the 23-kV lines in the model reduces the conditional core damage probability for the event. Inclusion of only the BODG results in a reduction of the conditional core damage probability by a factor of 7.5. Inclusion of only the 23-kV line results in a reduction of the conditional core damage probability by a factor of 2.2.

Incorporation of suppression pool venting reduces the conditional core damage probability by a factor of 1.2. It only impacts sequence 40.

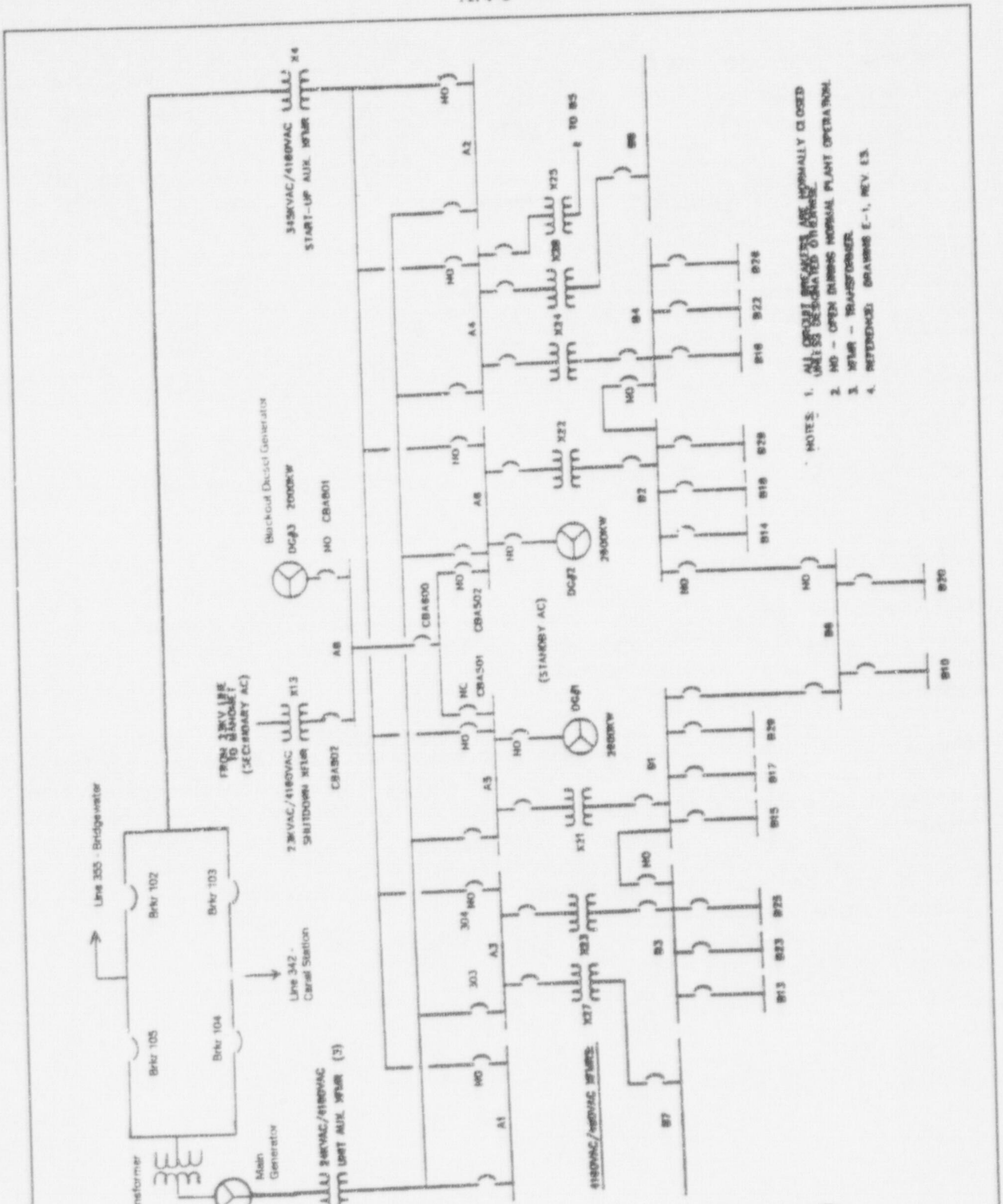


Fig. A.4.2 Simplified diagram of the Pilgrim electrical distribution system

Sequence	p(sequence from calculation sheets)	p(ADS)	p(vent)	p(sequence)
40	4.8×10^{-6}	1.0	0.01	4.8×10^{-8}
48	1.9×10^{-7}	1.0	1.0	1.9×10^{-7}
55	3.0×10^{-6}	0.0	1.0	0
67	9.2×10^{-7}	1.0	1.0	9.2×10^{-7}
69	2.9×10^{-7}	1.0	1.0	2.9×10^{-7}
83	3.2×10^{-6}	1.0	1.0	3.2×10^{-6}
			Total	4.6×10^{-6}

No analytical evaluation was made of potential consequences of the RV repressurization that occurred during this event.

A.4.5 Analysis Results

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Inclusion of the BODG and the 23-kV lines in the model reduces the conditional core damage probability for the event. Inclusion of only the BODG results in a reduction of the conditional core damage probability by a factor of 7.5. Inclusion of only the 23-kV line results in a reduction of the conditional core damage probability by a factor of 2.2.

Incorporation of suppression pool venting reduces the conditional core damage probability by a factor of 1.2. It only impacts sequence 40.

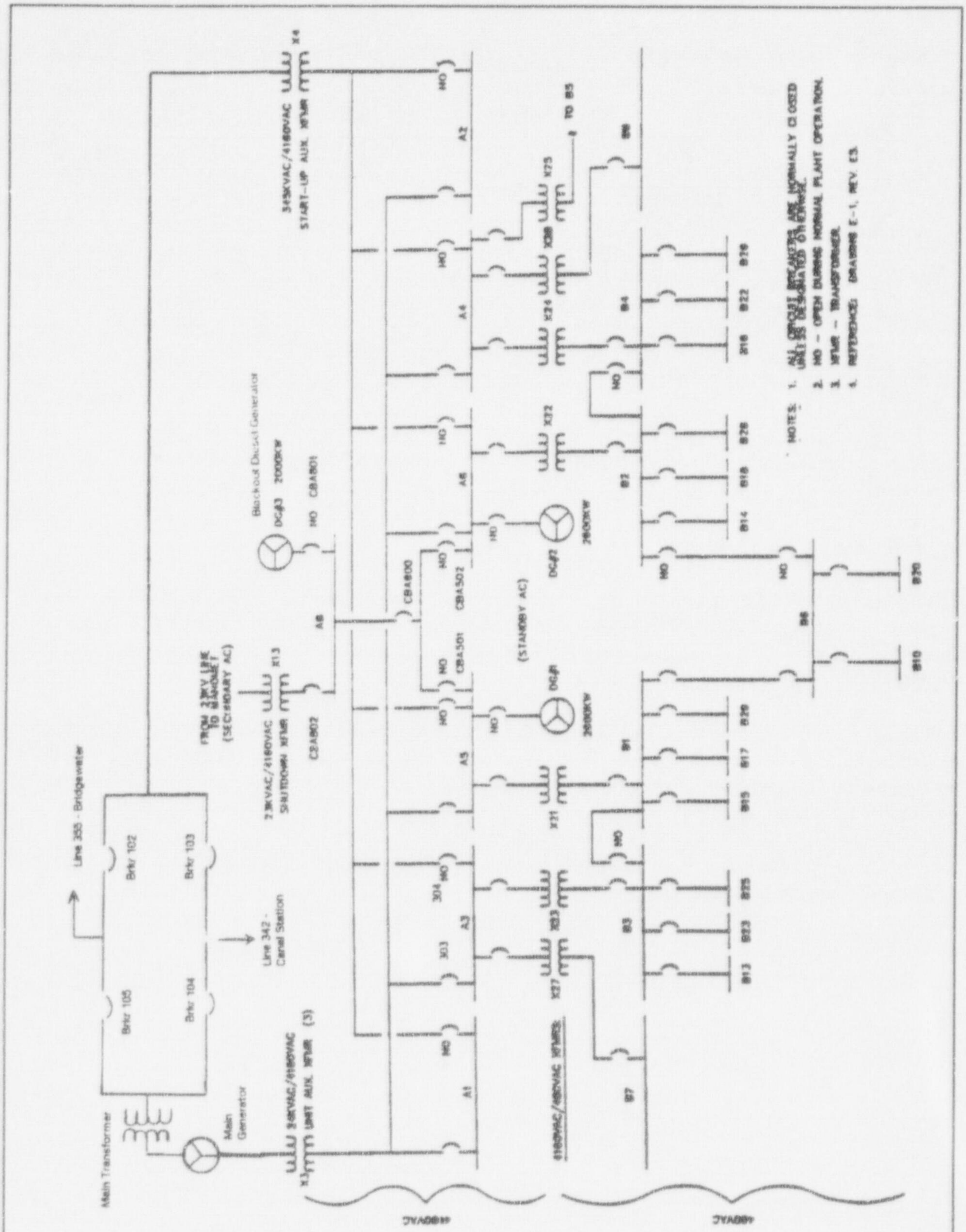


Fig. A.4.2 Simplified diagram of the Pilgrim electrical distribution system

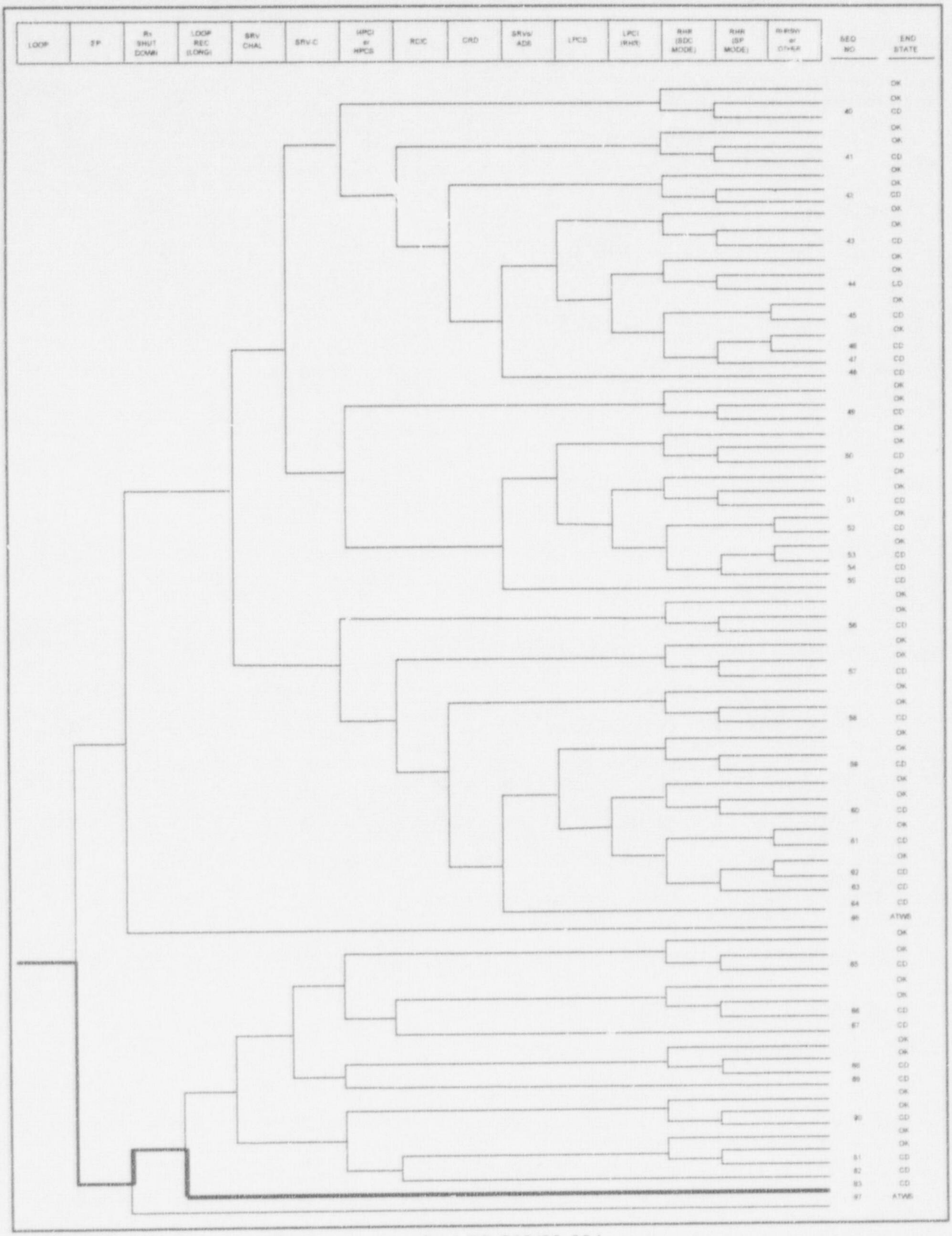


Fig. A.4.3 Dominant core damage sequence for LER 293/93-004

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 293/93-004
Event Description: Sevrre Weather Induced LOOP
Event Date: 03/13/93
Case: Includes BODG and 23-kV line
Plant: Pilgrim 1

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	9.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	1.3E-05 (1)
Total	1.3E-05 (1)
ATWS	
LOOP	2.7E-05
Total	2.7E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
40	LOOP -EMERG.POWER -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	4.8E-06(1)	1.0E-01
83	LOOP EMERG.POWER -rx.shutdown/ep EP.REC	CD	3.2E-06	3.6E-02
55	LOOP -EMERG.POWER -rx.shutdown srv.chall/loop.-scram srv.close hpci srv.ads	CD	3.0E-06(1)	4.5E-01
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpc rcic	CD	9.2E-07	1.9E-01
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram srv.close hpci	CD	2.9E-07	2.7E-01
48	LOOP -EMERG.POWER -rx.shutdown srv.chall/loop.-scram -srv.close hpci rcic crd srv.ads	CD	1.9E-07	3.1E-01
98	LOOP -EMERG.POWER rx.shutdown	ATWS	2.7E-05	9.0E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
40	LOOP -EMERG.POWER -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	4.8E-06(1)	1.0E-01
48	LOOP -EMERG.POWER -rx.shutdown srv.chall/loop.-scram -srv.close hpci rcic crd srv.ads	CD	1.9E-07	3.1E-01
55	LOOP -EMERG.POWER -rx.shutdown srv.chall/loop.-scram srv.close hpci srv.ads	CD	3.0E-06(1)	4.5E-01
98	LOOP -EMERG.POWER rx.shutdown	ATWS	2.7E-05	9.0E-01
67	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram -srv.close hpci rcic	CD	9.2E-07	1.9E-01
69	LOOP EMERG.POWER -rx.shutdown/ep -EP.REC srv.chall/loop.-scram	CD	2.9E-07	2.7E-01

```
     srv.close  hpci
 83  LOOP  EMERG.POWER -rx.shutdown/ep  EP.REC                    CD          3.2E-06     3.6E-02
```

** nonrecovery credit for edited case

```
SEQUENCE MODEL:      s:\asp\prog\models\bwrcseal.cmp
BRANCH MODEL:        s:\asp\prog\models\pilgrim.sl1
PROBABILITY FILE:    s:\asp\prog\models\bwr_csl1.pro
```

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	5.5E-04	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	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
pcs/trans	1.7E-01	1.0E+00	
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	1.3E-02	1.0E+00	
EMERG.POWER	2.9E-03 > 2.9E-03	8.0E-01 > 4.34E-01 (2)	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02		
EP.REC	3.1E-02 > 3.1E-02	1.0E+00 > 9.3E-02 (3)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	3.1E-02		
fw/pcs.trans	2.9E-01	7.4E-01	
fw/pcs.loca	4.0E-02	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	3.0E-03	3.4E-01	
lpci(-rhr)/lpcs	1.0E-03	7.1E-01	
rhr(sdc)	2.1E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-[illegible]
rhr(spcool)/rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/-lpci.rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/lpci.rhr(sdc)	9.3E-02	1.0E+00	
rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file
** forced

<u>NOTES</u>
(1) See Modeling Assumptions section for modifications of these values based on suppression pool venting and depressurization of the RV with one or more stuck open SRVs.
(2)Includes 23-kV line.
(3)Includes BODG.

A.5 LER No. 313/93-003

Event Description: Both Trains of Recirculation Inoperable for 14 h

Date of Event: September 30, 1993

Plant: Arkansas Nuclear One, Unit 1

A.5.1 Summary

On September 30, 1993, an engineering evaluation was completed at Arkansas Nuclear One, Unit 1, which indicated that the B decay heat removal/low-pressure injection (DHR/LPI) pump might have been incapable of performing its recirculation mode function following a loss-of-coolant accident (LOCA). This condition existed from May 24, 1993, while that plant was at power, until the plant shutdown on September 9, 1993. In addition, the A DHR/LPI pump was also inoperable for 14 h during this time period for routine maintenance and surveillance. The estimated conditional core damage probability for this event is 5.1×10^{-5}. The relative significance of this event compared to other postulated events at Arkansas Nuclear One, Unit 1, is shown in Fig. A.5.1.

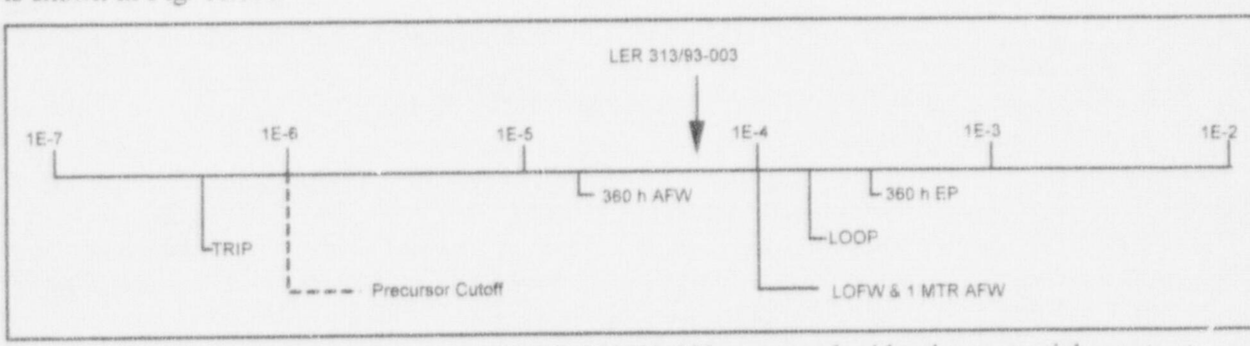


Fig. A.5.1 Relative event significance of LER 313/93-003 compared with other potential events at Arkansas Nuclear One, Unit 1

A.5.2 Event Description

On September 9, 1993, at 0432 hours, a routine plant shutdown was in progress to begin a refueling outage at Arkansas Nuclear One, Unit 1. With the reactor coolant system (RCS) temperature at 180°F, the B DHR/LPI pump was placed into service. At 0530 hours, the outboard motor bearing for the B DHR/LPI pump alarmed on high temperature, and the pump was secured. The A DHR/LPI pump remained in service. Following verification of the operation of the oil slinger and testing of the pump oil, the B pump was restarted at 1224 hours. The outboard motor bearing temperature again increased, and the pump was secured and declared inoperable at 1430 hours.

Troubleshooting efforts indicated that there was no bearing damage. However, the pump and the motor were not properly coupled. The coupling hub on the pump shaft was installed ~0.316 in. too far toward the motor. This condition caused the motor to be pushed off its magnetic center in the outward direction. Because the pump thrust bearing is on the opposite side of the pump from the motor, thermal expansion of the pump shaft while pumping hot fluids would push the shaft coupling farther in the outboard direction, creating increased thrust loading on the outboard motor bearing.

The outboard motor bearing temperature response during past surveillance tests was reviewed. Although the tests are terminated before the bearings reach stabilization temperature, the stabilization temperature can be determined from the strip chart data. This review indicated that higher than normal (but acceptable) bearing temperatures were observed after May 24, 1993. A review of maintenance records indicated that the B DHR/LPI pump coupling was greased during a system mini-outage on May 24, 1993.

The B DHR/LPI pump was considered capable of performing its LPI function throughout this period. However, from May 24, 1993, until the plant shutdown on September 9, 1993, the B pump was considered inoperable in the recirculation mode while pumping hot water from the reactor building sump. The A DHR/LPI pump was also inoperable for 14 h during this time period for routine maintenance and surveillance activities.

A.5.3 Additional Event-Related Information

The DHR/LPI pumps are used in three modes. The first is the DHR mode that is used during plant shutdowns, heatups, and outages. In this mode the pumps take suction from one of the RCS hot legs. After passing through the pumps and DHR coolers, cooled water is returned to the RCS cold legs. The B pump failed while operating in this mode. The second is the LPI mode; in this mode, the pumps take suction from the refueling water storage tank (RWST) and discharge it to the RCS loops. This LPI mode is automatically initiated following a safety injection signal. The third is the recirculation mode that is used following the depletion of the RWST. The pump suction is aligned to the containment sump. After passing through the pumps, the water is cooled in the DHR coolers. The discharge can then be aligned directly to the RCS, or if RCS pressure is above the DHR/LPI pump shutoff head, to the suction of the high-pressure injection (HPI) pumps for "piggyback" operation.

During the LPI mode, when the pumps take suction on the cooler water in the RWST, the licensee indicates that the B DHR/LPI pump will operate. However, when the warmer water from the containment sump passes through the pumps during the recirculation mode, the increased heat addition from the warmer water may cause the pump to fail due to the improper coupling of the motor.

A.5.4 Modeling Assumptions

Two cases were run. In the first case, both the A and B DHR/LPI pumps were assumed inoperable in the recirculation mode for 14 h. This calculation was performed because both trains of the system (high- and low-pressure recirculation) were inoperable. The B pump would have initially operated but would have subsequently failed as described above. The A pump was out-of-service for routine maintenance and surveillance testing. It was assumed that the A pump could potentially be recovered during the injection phase of a postulated LOCA event, making it available for the recirculation mode. A nonrecovery factor of 0.34 was assumed (NUREG/CR-4674, Vol.17, Sect. A.1.3, Recovery Class R2, 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).

The second case addresses the long-term unavailability of the B DHR/LPI pump. This calculation was performed because the long-term unavailability of a single train of recirculation significantly impacts the conditional core damage probability for the event. The pump was assumed to be inoperable in the recirculation mode for 3048 h (from May 5, 1993, to September 9, 1993). The A pump was considered operable during this period with nominal failure rates and nonrecovery values applied.

A.5.5 Analysis Results

The estimate of the conditional core damage probability for this event is 5.1×10^{-5}. This consists of a contribution of 4.9×10^{-6} for case 1 (both trains inoperable for 14 h) and 4.6×10^{-5} for case 2 (train B inoperable for 3048 h). The dominant core damage sequence for both cases, shown in Fig. A.5.2, involves a postulated LOCA, successful reactor trip, auxiliary feedwater, and HPI, followed by failure of high-pressure recirculation.

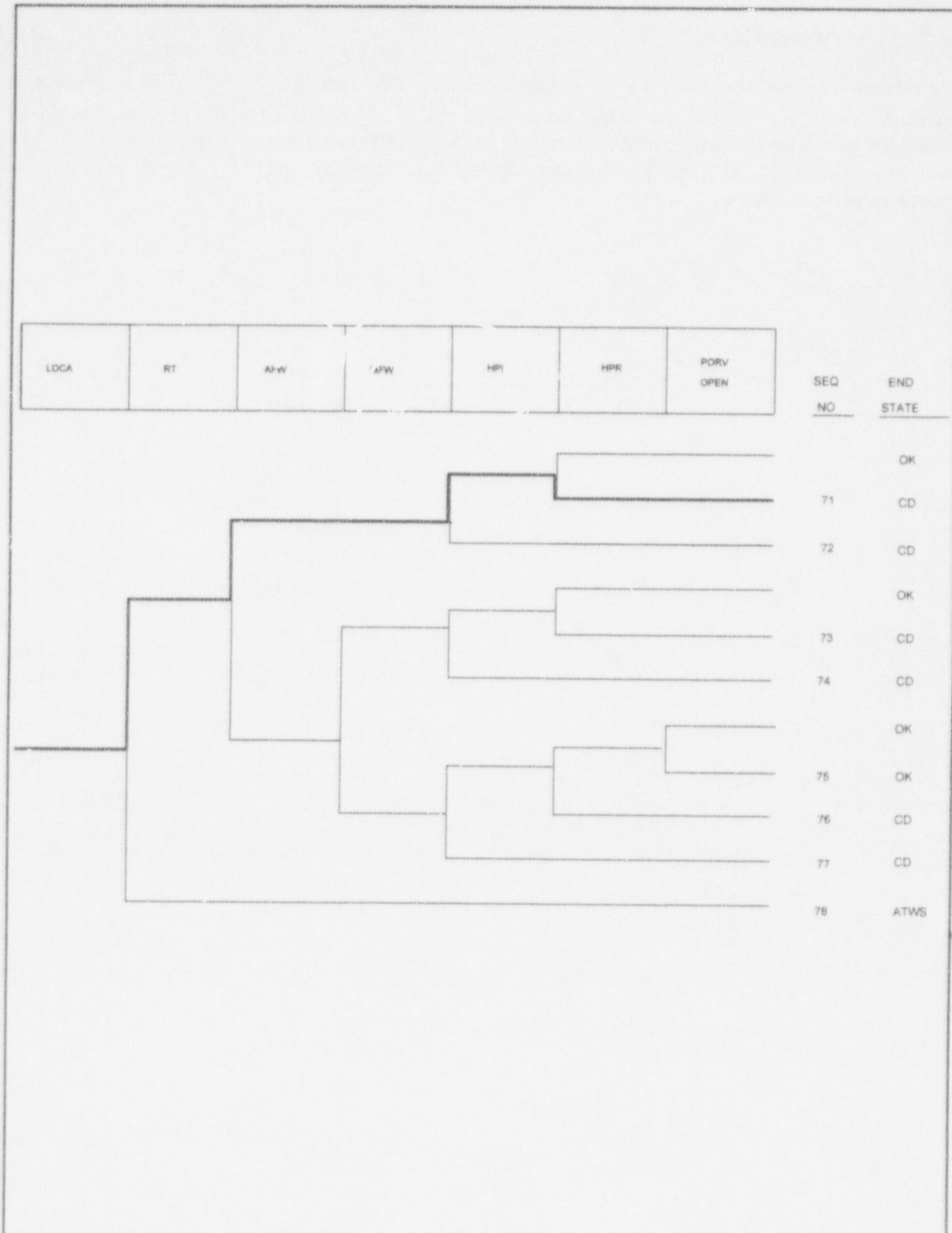


Fig. A.5.2 Dominant core damage sequence for LER 313/93-003

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 313/93-003
Event Description: Both trains of recirc inoperable
Event Date: 09/30/93
Case: Case 1 - Both trains inoperable for 14 hours, A train recoverable
Plant: ANO - Unit 1

UNAVAILABILITY, DURATION= 14

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOCA	1.4E-05

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOCA	4.9E-06
Total	4.9E-06
ATWS	
LOCA	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

Sequence	End State	Prob	N Rec**
71 loca -rt -afw -hpi HPR/-HPI	CD	4.9E-06	1.5E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

Sequence	End State	Prob	N Rec**
71 loca -rt -afw -hpi HPR/-HPI	CD	4.9E-06	1.5E-01

** nonrecovery 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: s:\asp\prog\models\pwrdseal.cmp
BRANCH MODEL: s:\asp\prog\models\ano1.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	1.4E-04	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	

emerg.power	2.9E-03	8.0E-01	
afw	2.3E-03	2.6E-01	
afw/emerg.power	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
porv.or.srv.chall	8.0E-02	1.0E+00	
porv.or.srv.reseat	1.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	1.0E-02	1.0E+00	
seal.loca	4.0E-02	1.0E+00	
ep.rec(sl)	5.9E-01	1.0E+00	
ep.rec	1.5E-01	1.0E+00	
hpi	3.0E-04	8.4E-01	
hpi(f/b)	3.0E-04	8.4E-01	1.0E-02
HPR/-HPI	1.5E-04 > 1.0E+00	1.0E+00 > 3.4E-01	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02 > Failed		
Train 2 Cond Prob:	1.5E-02 > Failed		

* branch model file
** forced

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 313/93-003
Event Description: Both trains of recirc inoperable
Event Date: 09/30/93
Case: Case 2 - Long term inop of train B, Nominal values for A train
Plant: ANO - Unit 1

UNAVAILABILITY, DURATION= 3048

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOCA	3.1E-03

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOCA	4.6E-05
Total	4.6E-05
ATWS	
LOCA	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
71	loca -rt -afw -hpi HPR/-HPI	CD	4.6E-05	4.3E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
71	loca -rt -afw -hpi HPR/-HPI	CD	4.6E-05	4.3E-01

** nonrecovery 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: s:\asp\prog\models\pwrdseal.cmp
BRANCH MODEL: s:\asp\prog\models\ano1.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	1.4E-04	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	

emerg.power	2.9E-03	8.0E-01	
afw	2.3E-03	2.6E-01	
afw/emerg.power	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
porv.or.srv.chall	8.0E-02	1.0E+00	
porv.or.srv.reseat	1.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	1.0E-02	1.0E+00	
seal.loca	4.0E-02	1.0E+00	
ep.rec(sl)	5.9E-01	1.0E+00	
ep.rec	1.5E-01	1.0E+00	
hpi	3.0E-04	8.4E-01	
hpi(f/b)	3.0E-04	8.4E-01	1.0E-02
HPR/-HPI	1.5E-04 > 1.5E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.2+opr			
Train 1 Cond Prob:	1.0E-02 > Failed		
Train 2 Cond Prob:	1.5E-02		

* branch model file
** forced

A.6 LER No. 316/93-007

Event Description: Reactor Trip with Degraded Auxiliary Feedwater

Date of Event: August 2, 1993

Plant: Cook 2

A.6.1 Summary

Cook 2 tripped from 70% power because of a spurious high-temperature signal from the main turbine exhaust hood. The auxiliary feedwater (AFW) control valves for the east motor-driven AFW pump throttled further than expected, requiring operator action to restore proper flow from that pump. Operator action was also required to reopen two main steam isolation valves (MSIVs) that started drifting closed following the trip. The conditional core damage probability estimated for the event is 2.4×10^{-6}. The relative significance of this event compared to other postulated events at Cook 2 is shown in Fig. A.6.1.

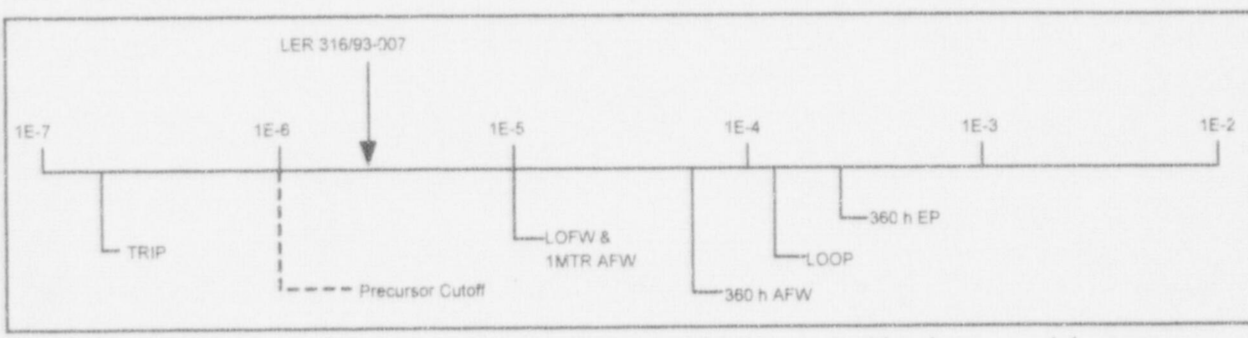


Fig. A.6.1 Relative event significance of LER 316/93-007 compared with other potential events at Cook 2

A.6.2 Event Description

On August 2, 1993, Cook 2 tripped from 70% power following a turbine trip caused by spurious actuation of the high-temperature switches on the turbine exhaust hood. Eight of the nine switch actuation set points were found to be significantly lower than the as-left condition recorded during the last calibration one year earlier.

Following the reactor trip, the AFW pumps started and provided flow to the steam generators. The feedwater control valves from the east motor-driven AFW pump throttled further than expected after receiving a flow retention signal, and operator action was required to maintain correct flow rates. AFW flow switches were subsequently recalibrated, and flow retention valve intermediate positions were reset to correct the problem.

Two MSIVs, which started to drift closed following the reactor trip, were reopened by the operators. The licensee stated that this drift was expected following a trip because of the valve actuator design.

A.6.3 Additional Event-Related Information

Cook 2 has three AFW pumps; two are motor-driven, and one is turbine-driven. The turbine-driven pump provides flow to all four SGs, and each motor-driven pump provides flow to two SGs. Flow retention valves control the flow from each pump to each SG; these valves can be controlled from the control room. A cross connect exists that can provide flow from one motor-driven pump to the other unit. The cross-connect valves are manual and normally locked closed. The main feedwater (MFW) pumps are turbine-driven.

A.6.4 Modeling Assumptions

This event was modeled as a reactor trip with degraded AFW and MFW. To reflect the reduced flow from the east motor-driven AFW pump, one of the three AFW trains was assumed to be failed in the analysis. Consistent with other precursor analyses, the probability of not recovering the potentially failed AFW system was not revised because failures were not observed in the other two trains. The accident sequence precursor (ASP) model for MFW assumes that MFW is isolated following a trip but is potentially available in the event of a failure of AFW. If all four MSIVs had drifted closed and the operators had not promptly responded and reopened the MSIVs, steam to the MFW pump turbines would have been lost, rendering MFW unavailable. Because this recovery action could be performed in the control room and the observed MSIV response was apparently not unusual at Cook, a nonrecovery probability of 0.006 was added to the nominal MFW nonrecovery probability (0.07) to estimate the overall MFW nonrecovery used in the analysis (0.076). This probability considered the potential drift of the third and fourth MSIVs ($p = 0.3$ and 0.5, respectively) and the probability that the operators would fail to reopen the drifting-closed MSIVs ($p = 0.04$). The nonrecovery probabilities used in ASP analyses are described in Sect. A.3.2 of NUREG/CR-4674, Vol. 17, *Precursors to Potential Severe Core Damage Accidents: 1992, A Status Report.*

The potential use of the locked-closed cross-connect between both units' AFW systems was not addressed in the analysis.

A.6.5 Analysis Results

The conditional core damage probability estimated for this event is 2.4×10^{-6}. The dominant core damage sequences, highlighted on the event tree in Fig. A.6.2, involves a postulated failure of AFW and MFW following the trip and subsequent failure of feed-and-bleed cooling.

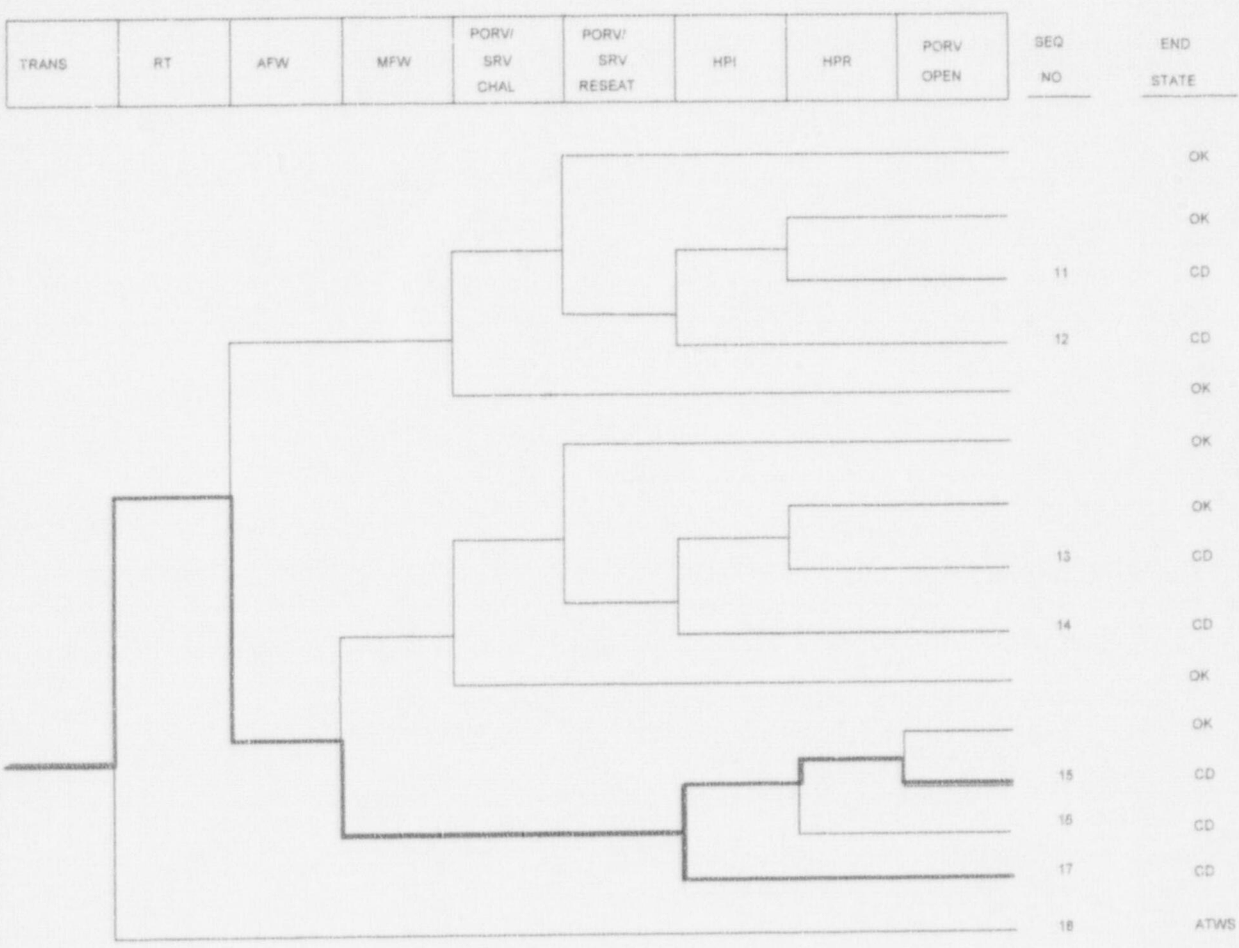


Fig. A.6.2 Dominant core damage sequence for LER 316/93-007

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 316/93-007
Event Description: Reactor trip with degraded AFW
Event Date: August 2, 1993
Plant: Cook 2

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	2.4E-06
Total	2.4E-06
ATWS	
TRANS	3.4E-05
Total	3.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
17	trans -rt AFW MFW hpi(f/b)	CD	1.1E-06	1.7E-02
15	trans -rt AFW MFW -hpi(f/b) -hpr/-hpi porv.open	CD	1.1E-06	2.0E-02
16	trans -rt AFW MFW -hpi(f/b) hpr/-hpi	CD	1.2E-07	2.0E-02
18	trans rt	ATWS	3.4E-05	1.2E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
15	trans -rt AFW MFW -hpi(f/b) -hpr/-hpi porv.open	CD	1.1E-06	2.0E-02
16	trans -rt AFW MFW -hpi(f/b) hpr/-hpi	CD	1.2E-07	2.0E-02
17	trans -rt AFW MFW hpi(f/b)	CD	1.1E-06	1.7E-02
18	trans rt	ATWS	3.4E-05	1.2E-01

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\cook.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro
No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	3.4E-04	1.0E+00	
loop	1.6E-05	2.4E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
AFW	3.8E-04 > 5.3E-03	2.6E-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/emerg.power	5.0E-02	3.4E-01	
MFW	1.0E+00 > 1.0E+00	7.0E-02 > 7.6E-02	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.0E+00		
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	3.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	3.0E-02	1.0E+00	
seal.loca	2.5E-01	1.0E+00	
ep.rec(sl)	6.9E-01	1.0E+00	
ep.rec	5.2E-02	1.0E+00	
hpi	1.0E-03	8.4E-01	
hpi(f/b)	1.0E-03	8.4E-01	1.0E-02
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
porv.open	1.0E-02	1.0E+00	4.0E-04

* branch model file
** forced

A.7 LER No. 334/93-013

Event Description: Dual-Unit Loss-of-Offsite Power

Date of Event: October 12, 1993

Plant: Beaver Valley 1

A.7.1 Summary

On October 12, 1993, the Beaver Valley site experienced a dual-unit loss-of-offsite power (LOOP). Unit 1 had been operating at 100% power and Unit 2 was in refueling shutdown at the time of the event. The conditional core damage probability estimated for this event is 5.5×10^{-5}. The LOOP was not modeled for Unit 2 because it was in refueling shutdown at the time of the event. The relative significance of this event compared to other postulated events at Beaver Valley 1 is shown in Fig. A.7.1.

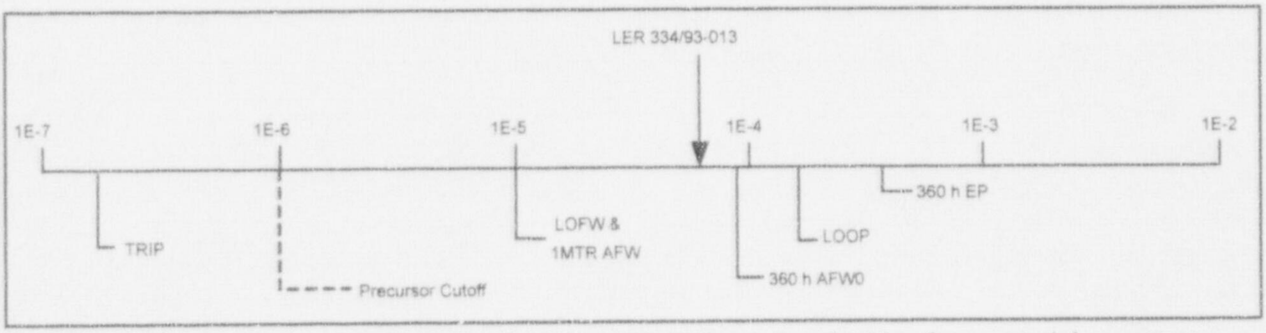


Fig. A.7.1 Relative event significance of LER 334/93-013 compared with other potential events at Beaver Valley 1

A.7.2 Event Description

On October 12, 1993, Beaver Valley Unit 1 was operating at 100% power with normal station loads being supplied from the unit station service transformers (USSTs). Unit 2 was in a refueling outage with all fuel stored in the spent fuel pool. Unit 2 loads were being supplied from a backfeed through the main unit transformer.

At 1507 hours, Unit 1 experienced a loss of the majority of its load when ten offsite feeder breakers in the switchyard opened, including the Unit 1 output breaker, PCB 341, and the Unit 2 output breaker, PCB 362. Loss of load in Unit 1 caused an overspeed trip of the turbine-generator. Generator speed peaked at 2051 rpm. This increase in generator speed caused a corresponding increase in reactor coolant pump (RCP) speed. The resulting flow transient caused a reactor trip on high flux rate.

Following the Unit 1 trip, all three auxiliary feedwater (AFW) pumps started, and the three RCPs tripped on underfrequency. Thirty seconds after the turbine trip, the generator output breakers opened as designed. The Unit 1 main generator had been the only source of power to both units following the opening of the switchyard breakers, and the trip of the Unit 1 generator caused a LOOP to both units. The Unit 1 emergency diesel generators (EDGs) sequenced loads on their buses, and a natural circulation cooldown was established using AFW and the steam generator power-operated relief valves. At 1517 hours, power was restored to the switchyard, and forced reactor coolant system (RCS) flow was reestablished. The safety-related buses were subsequently realigned to offsite power, and the EDGs were shut down.

Following the Unit 1 trip, the Unit 2 2–1 EDG sequenced all available train A safety-related loads, including the low-pressure injection (LPI) pump. However, the LPI pumps did not inject any water into the RCS because the discharge valves were closed for refueling. The 2–2 EDG and associated safeguards bus had been removed from service for outage-related maintenance at the time of the event. Offsite power was restored to Unit 2 at 1522 hours. The train A safety-related bus was repowered from offsite power at 1535 hours, and the 2–1 EDG was shutdown.

Following the Unit 1 reactor trip, a small RCS leak was noted at the loop 1A cold-leg vent valve, RC-27. Unit 1 then commenced a cooldown to cold shutdown. The leak was caused by a fillet weld failure.

The LOOP event was caused by an error during scheduled maintenance on the Unit 2 main output breaker. Continuity checks were being conducted on the auxiliary contacts for relays associated with the Unit 2 output breaker, PCB 352. During this process, underfrequency tripping relays were actuated when 125 Vdc from one set of contacts was inadvertently connected to another set of contacts in the underfrequency separation scheme via the multimeter used in the test. As a result, seven 345-kV breakers and three 138-kV breakers opened.

A.7.3 Additional Event-Related Information

Units 1 and 2 share a common 138-kV and 345-kV switchyard (see Fig. A.7.2). The 138-kV and 345-kV switchyards are connected by two auto transformers. Numerous offsite lines originate in both sections of the switchyard. The output of both main generators can be aligned to feed both of the 345-kV switchyard buses. Each main generator also feeds two USSTs. There are two system station service transformers (SSST) for each unit. One of the SSSTs for each unit is fed from each of the 138-kV substation buses—buses 1 and 2. Each SSST and associated USST feed two nonsafety-related 4160-Vac buses—buses A, B, C, and D. During power operations, buses A through D are aligned to the USST. Upon a trip of the turbine-generator, the buses fast transfer to the SSSTs. Buses A and D each feed a safety-related 4160-Vac bus, buses AE and DF. Each safety-related bus has an associated EDG that will load on a sustained loss of voltage.

The automatic loading capability of the EDGs on a safety injection (SI) signal was inoperable for both Unit 2 EDGs (see LER 412/93-012) at the time of this event. This failure would only occur when an SI signal is present coincident with a loss of the normal engineered safety feature (ESF) bus power supply. The failure mechanism had existed since November 1990. Operator actions would have been necessary to allow manual loading of equipment on the ESF buses. Because Unit 2 was in refueling at the time of the event, this does not impact the analysis.

A.7.4 Modeling Assumptions

This event was modeled as a plant-centered LOOP to Unit 1. The values for the short-term LOOP nonrecovery probability, the long-term nonrecovery of emergency power probability, and the probability of a reactor coolant pump seal loss of coolant accident (LOCA) were modified using the models described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89-11, August 1989. These models are based on the results of the data distributions contained in NUREG-1032. The models estimate the probability of short-term LOOP nonrecovery (within the first 30 min), long-term LOOP nonrecovery (before battery depletion) and seal LOCA. These values are based on the typical duration of a plant-centered LOOP. These estimates are a short-term nonrecovery of 0.3, a long-term nonrecovery of 3.8×10^{-1} given a seal LOCA, a long-term nonrecovery of 1.7×10^{-2} given no seal LOCA, and a seal LOCA probability of 0.15.

The RCS leak associated with RC-27 was small enough that it was well within the capabilities of the charging system. This was considered to have no impact on the sequence of other events or on the viability of operator recovery of other systems.

It was assumed that the maintenance work conducted on the generator output breaker relays would be done only on a unit that was shut down. In addition, the Unit 2 LOOP was of short duration, and all fuel had been moved to the spent fuel pool. As a result, the Unit 2 transient was not modeled. Because the Unit 2 transient was not modeled, the inoperability of the EDG load sequencer on a simultaneous LOOP and SI signal did not impact the analysis.

The existing ASP model was modified to include the potential use of the Appendix R Dedicated Auxiliary Feedwater Pump (DAFWP). This pump is powered by the Emergency Response Facility Diesel Generator and can provide water to the steam generators during a station blackout coincident with failure of the steam driven AFW pump. This only affects sequences where AFW has failed (i.e., sequence 55).

The probability multiplier used to adjust sequences to account for the potential use of the Appendix R Dedicated Auxiliary Feedwater Pump as provided by the licensee is 0.46. Therefore, the dominant sequences are affected as follows:

Sequence	p(sequence from calculation sheets)	p(DAFWP)	p(sequence)
48	1.5×10^{-6}	1.0	1.5×10^{-6}
53	3.7×10^{-5}	1.0	3.7×10^{-5}
54	9.3×10^{-6}	1.0	9.3×10^{-6}
55	1.2×10^{-5}	0.46	5.5×10^{-6}
Other	2.0×10^{-6}	1.0	2.0×10^{-6}
			5.5×10^{-5}

The other sequences noted in the table are below the truncation value and are therefore not displayed in the output. They account for less than 4% of the total conditional core damage probability for the event.

A.7.5 Analysis Results

The estimate of the conditional core damage probability for this event is 5.5×10^{-5}. The dominant core damage sequence, shown in Fig. A.7.3, involves a LOOP, failure of emergency power, successful AFW actuation, an RCP seal LOCA, and failure to recover offsite power prior to core unrecovery.

Unspecified (random) failures of the emergency power system are the dominant contributor to the event given that a LOOP has occurred. These contribute to all of the dominant sequences.

Incorporation of the dedicated auxiliary feedwater pump reduces the conditional core damage frequency by a factor of approximately 1.1.

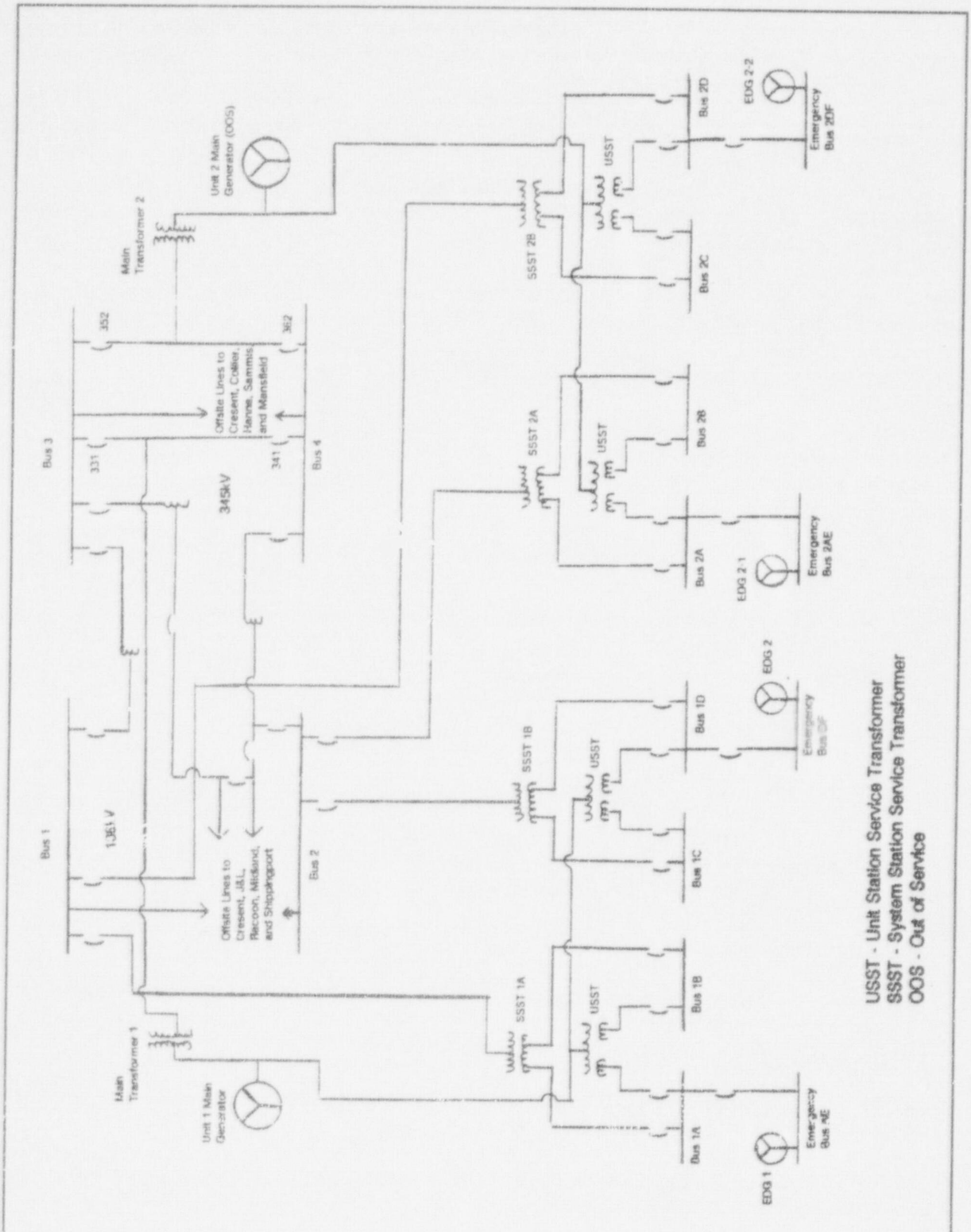


Fig. A.7.2 Simplified diagram of the Beaver Valley Unit 1 and Unit 2 switchyard

Fig. A.7.3 Dominant core damage sequence for LER 334/93-013

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 334/93-013
Event Description: LOOP at Beaver Valley 1
Event Date: 10/12/93
Plant: Beaver Valley 1

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	3.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	6.2E-05 (1)
Total	6.2E-05 (1)
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
53	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall SEAL.LOCA EP.REC(SL)	CD	3.7E-05	2.4E-01
55	LOOP -rt/loop emerg.power afw/emerg.power	CD	1.2E-05 (2)	8.2E-02
54	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall - SEAL.LOCA EP.REC	CD	9.3E-06	2.4E-01
48	LOOP -rt/loop emerg.power -afw/emerg.power porv.or.srv.chall - porv.or.srv.reseat/emerg.power SEAL.LOCA EP.REC(SL)	CD	1.5E-06	2.4E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
48	LOOP -rt/loop emerg.power -afw/emerg.power porv.or.srv.chall - porv.or.srv.reseat/emerg.power SEAL.LOCA EP.REC(SL)	CD	1.5E-06	2.4E-01
53	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall SEAL.LOCA EP.REC(SL)	CD	3.7E-05	2.4E-01
54	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall - SEAL.LOCA EP.REC	CD	9.3E-06	2.4E-01
55	LOOP -rt/loop emerg.power afw/emerg.power	CD	1.2E-05	8.2E-02

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\pwraseal.cmp
BRANCH MODEL: s:\asp\prog\models\beaver1.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	5.3E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	3.6E-01 > 3.0E-01	
Branch Model: INITOR			

Initiator Freq:	1.6E-05		
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
afw	3.8E-04	2.6E-01	
afw/emerg.power	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	3.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	3.0E-02	1.0E+00	
SEAL.LOCA	2.3E-01 > 1.5E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.3E-01 > 1.5E-01		
EP.REC(SL)	5.9E-01 > 3.8E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.9E-01 > 3.8L-01		
EP.REC	6.1E-02 > 1.7E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.1E-02 > 1.7E-02		
hpi	3.0E-04	8.4E-01	
hpi(f/b)	3.0E-04	8.4E-01	1.0E-02
porv.open	0.0E+00	1.0E+00	0.0E+00
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
csr	9.3E-05	1.0E+00	

* branch model file
** forced

Notes:
(1) See Modeling Assumptions section for modifications to the event conditional core damage probability.
(2) See Modeling Assumptions section for modifications to the sequence probability.

A.8 LER No. 339/93-002

Event Description: Auxiliary Feedwater Disabled After Reactor Trip

Date of Event: April 16, 1993

Plant: North Anna 2

A.8.1 Summary

North Anna 2 was operating at 100% power on April 16, 1993, when a malfunction in the main generator voltage regulator circuitry caused a turbine trip, which resulted in a reactor trip. Following the reactor trip, the auxiliary feedwater (AFW) pumps started on low–low steam generator (SG) level. During subsequent recovery actions the AFW pumps were disabled by placing the control switches in a "pull-to-lock" position before restoring SG levels above the automatic start set point, defeating the automatic start capability of the AFW pumps. The conditional core damage probability estimated for this event is 1.1×10^{-6}. The relative significance of this event compared to other postulated events at North Anna 2 is shown in Fig. A.8.1.

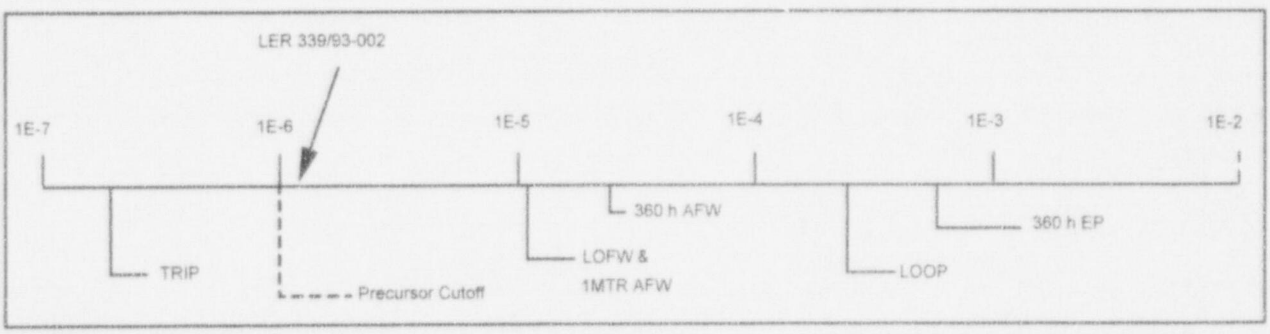


Fig. A.8.1 Relative event significance of LER 339/93-002 compared with other potential events at North Anna 2

A.8.2 Event Description

On April 16, 1993, with North Anna 2 at 100% power, an automatic reactor trip occurred from a turbine trip due to a malfunction in the main generator voltage regulator circuitry. The AFW pumps automatically started on low–low SG level. During the subsequent recovery actions of the reactor trip response procedure, it was noted that the reactor coolant system (RCS) was experiencing a cooldown due to feeding the SGs with relatively cold water from the AFW system. The operating crew became concerned with the RCS cooldown rate when RCS temperature decreased to ~ 540°F. The operator requested permission to secure AFW and to reset the ATWS Mitigation System Actuation Circuitry (AMSAC). The applicability of procedural steps used to control an excessive cooldown at this point was unclear. The first step in the procedure addressed the response to an excessive cooldown. However, the crew was currently at step 6. It was unclear if step 1 applied at this point. The Unit 2 supervisor, who was not involved in reading the procedure, requested the operator to secure AFW. As a result, the operator opened two of the main feedwater (MFW) bypass valves to establish flow to the SGs and then stopped the AFW pumps by placing the motor-driven AFW (MDAFW) pumps in pull-to-lock position and closing the two supply valves to the turbine-driven AFW (TDAFW) pump. Approximately 19 min after AFW was secured, the AFW pumps were returned to the AUTO position. At this point, SG levels had risen above 20%, and the pump automatic start signal at 18% had cleared.

A.8.3 Additional Event-Related Information

North Anna 2 is equipped with three safety-related AFW pumps. One pump is powered by a steam turbine, and the other two are motor-driven and powered from redundant 4.16-kV emergency buses. The full-capacity TDAFW pump is rated at 735 gal/min and each half-capacity MDAFW pump is rated at 370 gal/min. All three pumps are connected to two main headers. Either header may supply any of the three SGs but the headers are normally aligned so that one carries flow to a particular SG. A third header provides a flow path from the TDAFW pump to the "A" SG. This third header provides the flexibility required to dedicate a pump to each SG.

MFW isolates when T_{ave} = 540° F (LOW–LOW T_{ave}). This isolation prevents the MFW regulating valves from opening until the reactor trip breakers have been reset; however, makeup for the SGs is available through the MFW regulating bypass valves. This bypass function requires that the bypass valves be manually opened by the plant operators.

In this event, the operator disabled the AFW pumps from an auto start signal on SG low-low level by using the pull-to-lock position for the MDAFW pumps and closing the steam valves for the TDAFW pump. The first step of procedure 2-ES-0.1, Reactor Trip Response, checks for expected RCS temperatures. If the temperature is less than desired (547°F) and trending down, then AFW flow should be adjusted to 400 gal/min until at least one SG level is greater than 11%. Step 2 of the procedure has the operator check for sufficient feedwater flow. If adequate flow is not available, the operator is directed to establish AFW or MFW. No direction was provided for shutting down the AFW system and placing MFW into service. When the decision was made to secure AFW and use MFW, the task was accomplished without procedural guidance.

A.8.4 Modeling Assumptions

The AFW pumps did start automatically as required following the trip; however, the automatic capability of the system was disabled when the pumps were secured. This event was analyzed because it met the ASP precursor criteria as a reactor trip with a safety system disabled. AFW was considered disabled because manual actions would be required to restore the AFW system to operation.

This event was modeled as a reactor trip with a loss of all AFW. It was assumed that the operator would not have terminated AFW if MFW had not operated. Therefore, the failure rate for MFW was modified to be consistent with a transient with MFW initially available. The North Anna individual plant examination provides a value of 3.24×10^{-3} for this probability. The nonrecovery probability for the MFW system was left at the default value of 0.34. This assumes that failures of the MFW system following termination of AFW are recoverable in the required time period. The AFW pumps were modeled as unavailable due to the operator placing the control switches in the pull-to-lock position. A nonrecovery probability of 0.04 (ASP recovery class R4, NUREG/CR-4674, Vol.17, Appendix A, Sect. A.1.3) was assumed for the AFW pumps because the pumps could have been restarted from the control room.

A.8.5 Analysis Results

The conditional probability of subsequent core damage estimated for this event is 1.1×10^{-6}. The dominant core damage sequences, highlighted on the event tree in Fig. A.8.2, involve failure of all sources of SG makeup and failure of feed-and-bleed cooling. In all sequences both AFW and MFW fail. Feed-and-bleed fails for different reasons in the two dominant sequences. In sequence 17, feed-and-bleed fails when either the high-pressure injection system fails or the operator fails to initiate feed-and-bleed. In sequence 15,

A.8 LER No. 339/93-002

Event Description: Auxiliary Feedwater Disabled After Reactor Trip

Date of Event: April 16, 1993

Plant: North Anna 2

A.8.1 Summary

North Anna 2 was operating at 100% power on April 16, 1993, when a malfunction in the main generator voltage regulator circuitry caused a turbine trip, which resulted in a reactor trip. Following the reactor trip, the auxiliary feedwater (AFW) pumps started on low–low steam generator (SG) level. During subsequent recovery actions the AFW pumps were disabled by placing the control switches in a "pull-to-lock" position before restoring SG levels above the automatic start set point, defeating the automatic start capability of the AFW pumps. The conditional core damage probability estimated for this event is 1.1×10^{-6}. The relative significance of this event compared to other postulated events at North Anna 2 is shown in Fig. A.8.1.

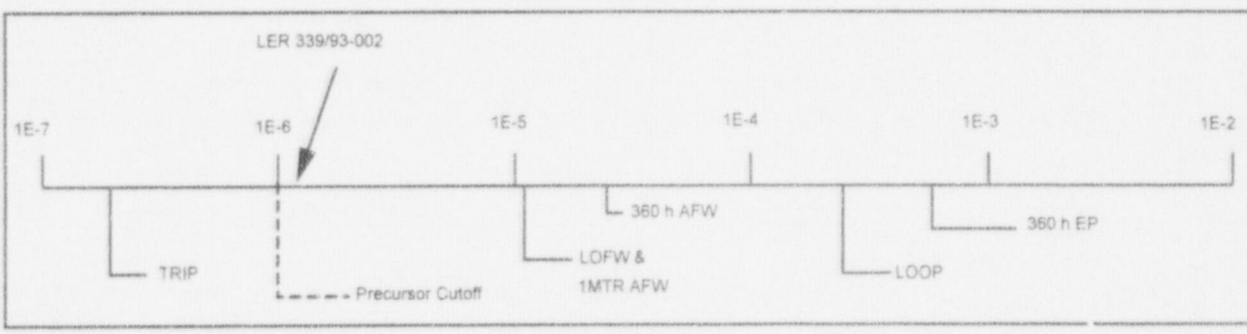


Fig. A.8.1 Relative event significance of LER 339/93-002 compared with other potential events at North Anna 2

A.8.2 Event Description

On April 16, 1993, with North Anna 2 at 100% power, an automatic reactor trip occurred from a turbine trip due to a malfunction in the main generator voltage regulator circuitry. The AFW pumps automatically started on low–low SG level. During the subsequent recovery actions of the reactor trip response procedure, it was noted that the reactor coolant system (RCS) was experiencing a cooldown due to feeding the SGs with relatively cold water from the AFW system. The operating crew became concerned with the RCS cooldown rate when RCS temperature decreased to ~ 540° F. The operator requested permission to secure AFW and to reset the ATWS Mitigation System Actuation Circuitry (AMSAC). The applicability of procedural steps used to control an excessive cooldown at this point was unclear. The first step in the procedure addressed the response to an excessive cooldown. However, the crew was currently at step 5. It was unclear if step 1 applied at this point. The Unit 2 supervisor, who was not involved in reading the procedure, requested the operator to secure AFW. As a result, the operator opened two of the main feedwater (MFW) bypass valves to establish flow to the SGs and then stopped the AFW pumps by placing the motor-driven AFW (MDAFW) pumps in pull-to-lock position and closing the two supply valves to the turbine-driven AFW (TDAFW) pump. Approximately 19 min after AFW was secured, the AFW pumps were returned to the AUTO position. At this point, SG levels had risen above 20%, and the pump automatic start signal at 18% had cleared.

A.8.3 Additional Event-Related Information

North Anna 2 is equipped with three safety-related AFW pumps. One pump is powered by a steam turbine, and the other two are motor-driven and powered from redundant 4.16-kV emergency buses. The full-capacity TDAFW pump is rated at 735 gal/min and each half-capacity MDAFW pump is rated at 370 gal/min. All three pumps are connected to two main headers. Either header may supply any of the three SGs but the headers are normally aligned so that one carries flow to a particular SG. A third header provides a flow path from the TDAFW pump to the "A" SG. This third header provides the flexibility required to dedicate a pump to each SG.

MFW isolates when T_{ave} = 540° F (LOW–LOW T_{ave}). This isolation prevents the MFW regulating valves from opening until the reactor trip breakers have been reset; however, makeup for the SGs is available through the MFW regulating bypass valves. This bypass function requires that the bypass valves be manually opened by the plant operators.

In this event, the operator disabled the AFW pumps from an auto start signal on SG low-low level by using the pull-to-lock position for the MDAFW pumps and closing the steam valves for the TDAFW pump. The first step of procedure 2-ES-0.1, Reactor Trip Response, checks for expected RCS temperatures. If the temperature is less than desired (547°F) and trending down, then AFW flow should be adjusted to 400 gal/min until at least one SG level is greater than 11%. Step 2 of the procedure has the operator check for sufficient feedwater flow. If adequate flow is not available, the operator is directed to establish AFW or MFW. No direction was provided for shutting down the AFW system and placing MFW into service. When the decision was made to secure AFW and use MFW, the task was accomplished without procedural guidance.

A.8.4 Modeling Assumptions

The AFW pumps did start automatically as required following the trip; however, the automatic capability of the system was disabled when the pumps were secured. This event was analyzed because it met the ASP precursor criteria as a reactor trip with a safety system disabled. AFW was considered disabled because manual actions would be required to restore the AFW system to operation.

This event was modeled as a reactor trip with a loss of all AFW. It was assumed that the operator would not have terminated AFW if MFW had not operated. Therefore, the failure rate for MFW was modified to be consistent with a transient with MFW initially available. The North Anna individual plant examination provides a value of 3.24×10^{-3} for this probability. The nonrecovery probability for the MFW system was left at the default value of 0.34. This assumes that failures of the MFW system following termination of AFW are recoverable in the required time period. The AFW pumps were modeled as unavailable due to the operator placing the control switches in the pull-to-lock position. A nonrecovery probability of 0.04 (ASP recovery class R4, NUREG/CR-4674, Vol.17, Appendix A, Sect. A.1.3) was assumed for the AFW pumps because the pumps could have been restarted from the control room.

A.8.5 Analysis Results

The conditional probability of subsequent core damage estimated for this event is 1.1×10^{-6}. The dominant core damage sequences, highlighted on the event tree in Fig. A.8.2, involve failure of all sources of SG makeup and failure of feed-and-bleed cooling. In all sequences both AFW and MFW fail. Feed-and-bleed fails for different reasons in the two dominant sequences. In sequence 17, feed-and-bleed fails when either the high-pressure injection system fails or the operator fails to initiate feed-and-bleed. In sequence 15, high-pressure injection is successful, but the pressurizer power-operated relief valve fails to open, resulting in failure of feed-and-bleed.

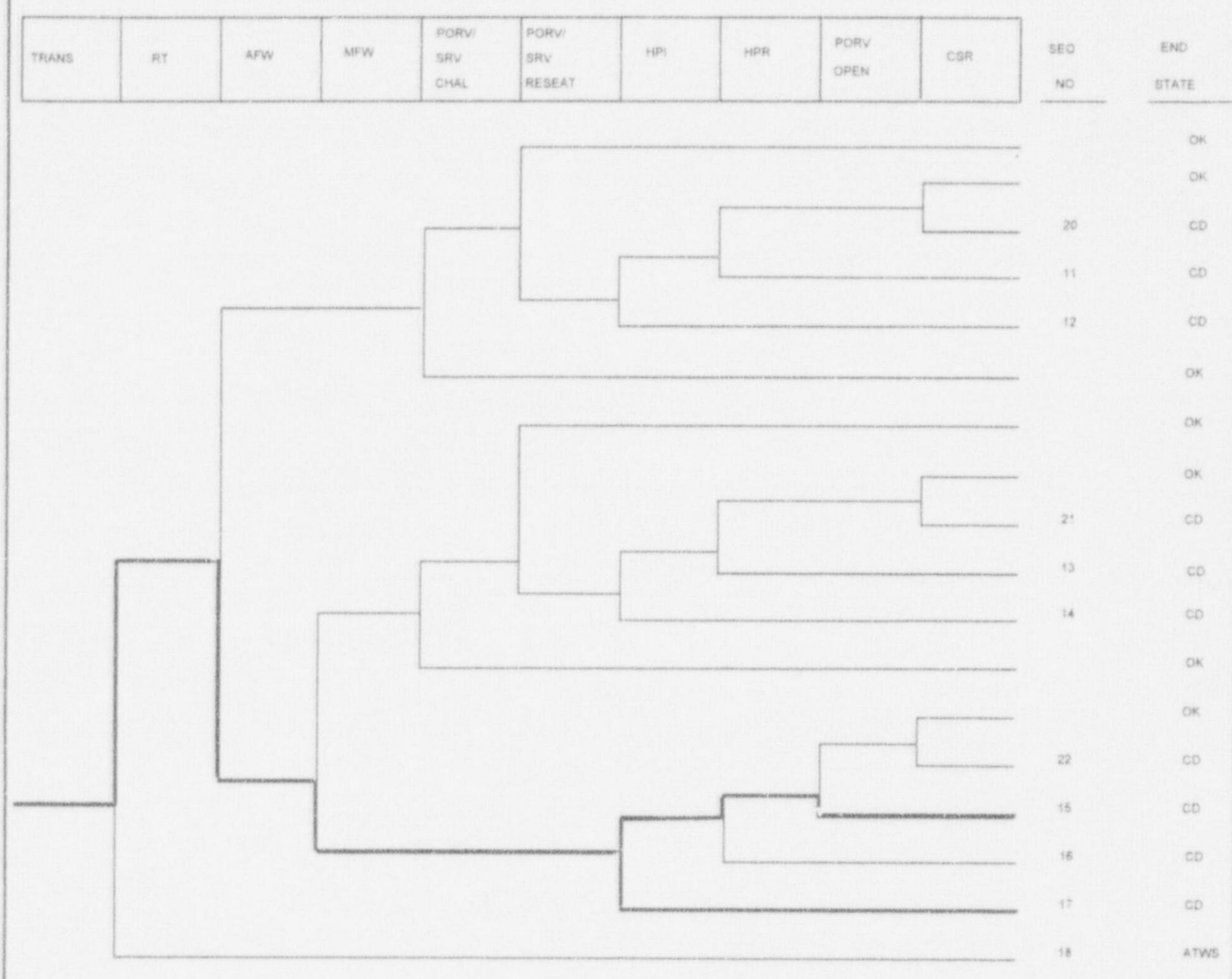


Fig. A.8.2 Dominant core damage sequence for LER 339/93-002

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 339/93-002
Event Description: AFW Disabled After Plant Trip
Event Date: 06/16/93
Plant: North Anna 2

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.1E-06
Total	1.1E-06
ATWS	
TRANS	3.4E-05
Total	3.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
17	trans -rt AFW MFW hpi(f/b)	CD	4.9E-07	1.1E-02
15	trans -rt AFW MFW -hpi(f/b) -hpr/-hpi porv.open	CD	4.5E-07	1.4E-02
16	trans -rt AFW MFW -hpi(f/b) hpr/-hpi	CD	4.9E-08	1.4E-02
22	trans -rt AFW MFW -hpi(f/b) -hpr/-hpi -porv.open csr	CD	4.6E-08	1.4E-02
12	trans -rt -AFW porv.or.srv.chall porv.or.srv.reseat hpi	CD	1.6E-08	8.9E-03
18	trans rt	ATWS	3.4E-05	1.2E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
12	trans -rt -AFW porv.or.srv.chall porv.or.srv.reseat hpi	CD	1.6E-08	8.9E-03
22	trans -rt AFW MFW -hpi(f/b) -hpr/-hpi -porv.open csr	CD	4.6E-08	1.4E-02
15	trans -rt AFW MFW -hpi(f/b) -hpr/-hpi porv.open	CD	4.5E-07	1.4E-02
16	trans -rt AFW MFW -hpi(f/b) hpr/-hpi	CD	4.9E-08	1.4E-02
17	trans -rt AFW MFW hpi(f/b)	CD	4.9E-07	1.1E-02
18	trans rt	ATWS	3.4E-05	1.2E-01

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\pwraseal.cmp
BRANCH MODEL: s:\asp\prog\models\northan2.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	1.9E-05	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	

rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
AFW	3.8E-04 > 1.0E+00	2.6E-01 > 4.0E-02	
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 > Failed		
Serial Component Prob:	2.8E-04		
afw/emerg.power	5.0E-02	3.4E-01	
MFW	1.9E-01 > 3.2E-03 **	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.9E-01		
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	3.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	3.0E-02	1.0E+00	
seal.loca	2.7E-01	1.0E+00	
ep.rec(sl)	5.7E-01	1.0E+00	
ep.rec	7.0E-02	1.0E+00	
hpi	1.5E-03	8.4E-01	
hpi(f/b)	1.5E-03	8.4E-01	1.0E-02
porv.open	1.0E-02	1.0E+00	4.0E-04
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
csr	9.3E-05	1.0E+00	

* branch model file
** forced

A.9 LER No. 370/93-008

Event Description: Loss-of-offsite Power and Failure of a Main Steam Isolation Valve to Close

Date of Event: December 27, 1993

Plant: McGuire 2

A.9.1 Summary

On December 27, 1993, with McGuire 2 operating at 100% power, the unit lost one of two half-capacity 525-kV generator output lines. The Unit 2 main turbine did not run back, resulting in the opening of the breakers on the other 525-kV bus line. The resulting loss-of-offsite power (LOOP) caused a reactor trip. Following an excessive cooldown rate, safety injection (SI) supplied water to the reactor coolant system (RCS), which caused the repeated cycling of the pressurizer (PZR) power-operated relief valves (PORVs). Recovery from the excessive cooldown rate was complicated by the failure of a main steam isolation valve (MSIV) on steam generator (SG) B to fully close, and termination of auxiliary feedwater (AFW) to the B SG resulted in B SG drying out. The plant operators then cycled the PZR PORVs to reduce the differential pressure across the tubes of the B SG in order to reduce the potential for a steam generator tube rupture (SGTR). Offsite power was restored ~ 96 min into the event. Forced circulation was restored ~ 3.5 h into the event when reactor coolant pump (RCP) A was restored to operation. The conditional core damage probability calculated for this event is 9.3×10^{-5}. The relative significance of this event compared to other postulated events at McGuire 2 is shown in Fig. A.9.1.

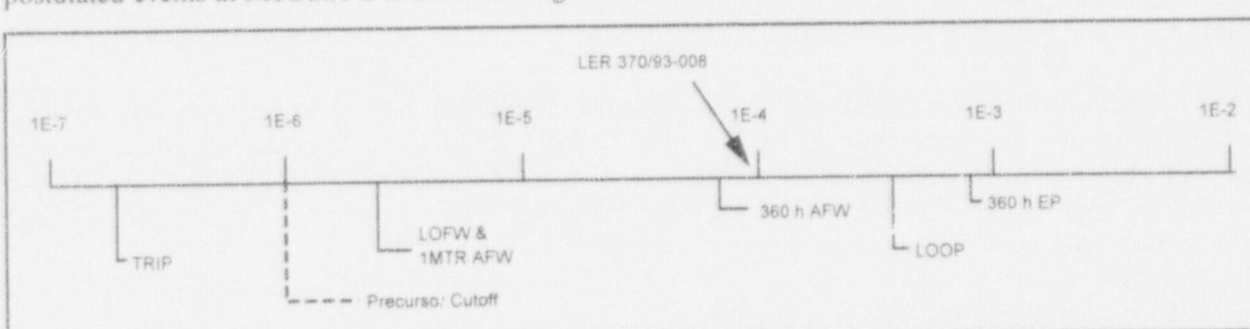


Fig. A.9.1 Relative event significance of LER 370/93-008 compared with other potential events at McGuire 2

A.9.2 Event Description

On December 27, 1993, while operating at 100% power, the 525-kV bus line 2B at McGuire 2 was lost at 2206 hours due to the failure of a 525-kV switchyard insulator. The main turbine generator did not run back because of burned resistors on a digital input slave module. Due to the failure of the turbine generator to run back, the breakers on 525-kV bus line 2A opened on overcurrent protection and caused a LOOP on Unit 2. The LOOP caused an increase in RCP speed that caused an increase in reactor power and resulted in a reactor trip at 2207 hours.

Following the LOOP, but before the reactor trip, all three PZR PORVs opened. All of the PORVs reclosed within 2 s. At 2207 hours, undervoltage on the 4.16-kV essential buses, 2ETA and 2ETB, caused both Unit 2 emergency diesel generators (EDGs) to automatically start and repower their respective essential buses. Also at 2207 hours, the turbine-driven AFW pump and both motor-driven AFW pumps automatically started, and AFW flow was initiated to all four steam generators. Plant cooldown proceeded by natural circulation. At 2210 hours, three main steam PORVs and two safety relief valves closed.

An excessive cooldown rate caused a PZR low-pressure SI signal to be generated at 2214 hours. This was immediately followed by a steam line B low-pressure SI signal. Both SI pumps started. Three of the four MSIVs closed. The B SG MSIV failed to fully close, and the RCS cooldown continued. The operators noted that the B SG MSIV failed to close, and personnel were dispatched to attempt to manually close the valve to the moisture separator reheater, the valves to the steam line drains, and the B SG MSIV.

The operators stopped AFW flow to all four SGs at 2223 hours, about 16 min after automatic initiation, because of the continued cooldown of the RCS. However, B SG continued to lose inventory, and at 2225 hours, control room personnel had transitioned to the "Safety Injection Termination Following Excessive Cooldown" procedure. AFW flow was restored to the B SG at 2227 hours.

Between 2228 and 2249 hours, the PZR PORVs automatically cycled about once a minute to control the PZR pressure increase due to the addition of SI flow into the RCS. In accordance with the emergency procedures, the PORVs were manually cycled to reduce the differential pressure across the B SG tubes to 1600 psid in order to minimize the potential for a SGTR. At 2230 hours, AFW flow for A SG was reinitiated, then terminated 2 min later. At 2236 hours, AFW flow to SG B was also stopped. AFW flow to SG B had been provided for ~9 min. AFW flow was then reinitiated to SG A and apparently continued for the duration of the event. SI flow was terminated after injecting for about 27 min at 2241 hours. At 2249 hours, the PZR PORVs automatically cycled for the last time.

Offsite power was restored at 2342 hours when the 525-kV bus line 2A was reenergized. Buses 2ETB and 2ETA were reenergized from offsite power at 0018 hours and 0032 hours, respectively, on December 28, 1993. Following this, RCP A was restarted at 0137 hours, and forced circulation was reestablished.

The LER indicates that the 525-kV bus line 2A could have been reenergized immediately following the opening of the Unit 2 generator breakers on December 27, 1993. However, the licensee decided that because the Unit 2 EDGs had successfully started and reenergized both Unit 2 essential 4.16-kV buses, a walkdown inspection of bus line 2A should be completed before returning line 2A to service. The licensee indicated that the walkdown was necessary to ensure the integrity of line 2A.

A.9.3 Additional Event-Related Information

The main generator normally feeds two 50% main transformers that feed the 530-kV offsite distribution system (see Fig. A.9.2). The main generator and one offsite line provide a feed to each of the full-size auxiliary transformers. Each auxiliary transformer normally feeds two nonsafety-related 6.9-kV buses. The two safety-related 4.16 kV buses are each supplied by one of the 6.9-kV buses. In addition, the safety-related buses can be supplied by their own EDG. If a fault occurs in one of the two independent generator output circuits (1A or 1B), the transformer breaker and the generator breaker in the affected circuit trip, and the 6.9-kV buses normally fed from the affected circuit transfers to the other auxiliary transformer. The generator then runs back to half load to maintain noninterrupted ties between the transmission system and the 6.9-kV buses.

The MSIVs automatically close on high containment pressure (> 3.0 psig), high steam line pressure rate of change (100 psig/s), or low steam line pressure (< 585 psig). The AFW system consists of two motor-driven pumps and one turbine-driven pump. The discharge from each of the motor-driven pumps can be directed to two of the four SGs. The turbine-driven pump can feed any of the SGs. Flow from each pump to the individual SGs is controlled by air-operated valves. The pumps will automatically start on low-low level in the SGs. The SI system automatically actuates on PZR low pressure (1845 psig), containment high pressure (1.2 psig), or low steam line pressure (725 psig). The PZR PORVs are designed to lift at 2335 psi. The PZR safety relief valves are set for 2485 psi. The discharge of the PORVs is routed to the PZR relief tank. The PZR relief tank has a rupture disk that actuates at 100 psi.

The McGuire site has a Standby Shutdown Facility (SSF) that is equipped with a 700-kW diesel generator, which can power two standby makeup pumps (one for each McGuire unit). These pumps deliver water from the spent fuel pool to the RCP seals at the rate of 26 gal/min. SSF equipment does not rely on normal plant support systems. The SSF equipment, along with the turbine-driven AFW pump, provides an independent means to achieve and maintain a hot standby condition for one or both units.

A.9.4 Modeling Assumptions

This event was modeled as a plant-centered LOOP. Probabilities for LOOP nonrecovery (short-term) and failure to recover ac power before battery depletion were revised to reflect values associated with a plant-centered LOOP (see ORNL/NRC/LTR 89/11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989). The LOOP nonrecovery (short-term) probability was changed from 0.36 to 0.30. The probability of failing to recover ac power before battery depletion was changed from 0.45 to 0.35.

The event was complicated by the failure of the MSIV to close and subsequent emptying of the B SG. The likelihood of a SGTR that might lead to core damage as a result of the high-differential pressure across the SG tubes was addressed in a separate sensitivity analysis.

A.9.4.1 PORV/Safety Relief Valve (SRV) Failure to Close

Because the PORVs were repeatedly cycled, both automatically and manually, during the event, the PORV/SRV challenge rate was revised to 1.0. The PORV/SRV failure to close probability, given emergency power success, was increased by a factor of 4 (from 0.03 to 0.12). This was done to account for the numerous (~21) lifts of the PORVs caused by the mass addition to the RCS from the SI system. Based on data derived from observed events, the exact increase in the failure to close probability is not known. Further, sequences involving the failure of the PORV/SRV to close, given emergency power success, do not contribute significantly to the overall conditional core damage probability for the event. Because the additional PORV cycles were due to the mass addition from the SI system that would be unavailable if emergency power were unsuccessful, the PORV/SRV failure to close probability, given emergency power failure, was not revised.

A.9.4.2 Standby Shutdown Facility

The SSF is not included in the accident sequence precursor (ASP) model for McGuire. To reflect its impact on the modeling for this event, the sequence values determined by the ASP code were modified for sequences in which success of SSF would have prevented core damage. This was accomplished by multiplying the sequence values by the failure probability of the SSF.

The McGuire individual plant examination (IPE) provides specific information on the failure probability of the SSF. The total failure probability is estimated to be 0.26 when offsite power is available from McGuire 1 (as it was in this event). This is dominated by failure of the SSF equipment to start in time to prevent RCP seal failure. If offsite power is not available for either unit, then the SSF failure probability estimate is 0.36. If information from the IPE were not available, ASP recovery class R2, (the failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from control room did not appear possible) would normally be utilized (see *Precursors to Potential Severe Core Damage*

Accidents: 1992 A Status Report, NUREG/CR-4674, Vol. 17, December 1993). The value for recovery class R2 is 0.34; this is comparable to the value provided in the IPE. Because the IPE addresses the particular situation involved in this event and the value is similar to the typical ASP value, the IPE value of 0.26 was utilized for the SSF failure probability.

Of the three dominant sequences, SSF success will only impact sequence 49. In sequence 50, the PORV fails to reseat, resulting in a small loss-of-coolant accident (LOCA). The SSF cannot provide sufficient makeup in this case because it only provides 26 gal/min. In sequence 55, emergency ac power failure and failure of the turbine-driven AFW train result in core damage regardless of SSF success. In sequence 49, the SSF could supply emergency control power in spite of battery depletion. The result of the model calculation (see sequence 49 in the calculation sheets) for this sequence is multiplied by the SSF failure probability. The conditional core damage probability for the sequence becomes $2.3 \times 10^{-4} \times 0.26 = 6.0 \times 10^{-5}$.

A.9.4.3 SGTR

Due to the depressurization of the B SG following the failure of the associated MSIV to close and the resulting high-differential pressure across the SG tubes, the potential existed for a SGTR following the LOOP.

An event tree model for a SGTR, based on the conditions that existed during the December 27, 1993, event, was developed as shown in Fig. A.9.3. For the high-differential pressure to exist across the SG tubes, high-pressure injection (HPI) must be injecting to the reactor coolant system (RCS). Without HPI, the dryout of the B SG would cause the RCS to cool down and depressurize. Power must also be available to at least one of the safeguards buses to provide power to the HPI pump(s). Finally, the MSIV must be stuck open to allow the dryout of the B SG. This causes the depressurization of the B SG, that, combined with the elevated pressure in the RCS from HPI, allows for the development of a large pressure differential across the SG tubes of the B SG.

SGTR—The potential for having a SGTR, given the events that occurred at McGuire 2 on December 27, 1993, was developed using the methodology of NUREG-0844, *NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity*, September 1988. This report addresses the probability of SGTRs as consequences of plant transients or accidents when loadings on the steam-generator tubes are increased above normal operating loads. The conditional probability that one or more tubes will rupture during postulated accident conditions is given by:

$$C_i = C_a [(P_i - P_n) / (P_a - P_n)]^2$$

where

C_i = conditional probability for one or more tube ruptures during transient "i",

C_a = conditional probability for one or more tube ruptures during transient where the peak dP across a steam generator is 2335 psid (2.5×10^{-2}, based on McGuire SG tube inspection data),

P_i = peak dP across tubes during transient (1981 psid for this event),

P_n = normal operating pressure differential across the tubes (1275 psid for McGuire),

P_a = peak dP across tubes during a postulated MSLB transient (2335 psid, based on McGuire data).

Therefore,

$$C_i = 2.5 \times 10^{-2} \ [(1981 - 1275)/(2335\text{-}1275)]^2 = 1.1 \times 10^{-2}.$$

This formula was based on industry-wide SGTR data before 1988. A SG was considered "vulnerable" to a SGTR if a SGTR would have occurred when the SG tubes were subjected to a differential pressure (dP) of 2335 psid. For transients involving reduced differential pressures, the conditional probabilities of causing a SGTR are reduced by the square of the ratio of the pressure increase associated with the transient to the pressure increase associated with the postulated worst-case accident.

AFW—Because power is assumed to be available to the safeguards buses, it is assumed that all three AFW pumps are available. The standard ASP values for AFW system failure were utilized. This gives an overall system probability of 9.9×10^{-5}. The MFW pumps are turbine driven. Following the closure of the MSIVs, they would not be available. Therefore, they were not addressed in the model.

HPI—HPI is assumed to be operable because it is required to develop the pressure differential across the SG tubes. Without the pressure differential, the SG is not as susceptible to a SGTR.

Isolation of the Ruptured SG and RCS Cooldown—Although the MSIV for the B SG failed to close, the SG could be isolated by closing the remaining three MSIVs and the steam line drains. During the actual event, operators were dispatched to close the steam line drain valves, and the remaining MSIVs were closed. The McGuire IPE indicates that this recovery method may not be viable because radiological conditions following a SGTR may prevent operator access to the valves. In addition, it was discovered after the event that one of the remaining MSIVs (for SG A) exhibited some leakage also. As a result, it was assumed that isolation of the SG was not possible. The failure probability was set to 1.0.

Cooldown to Residual Heat Removal (RHR)—The RCS is cooled down and depressurized to the conditions required to place the RHR system in service. This is usually accomplished by use of the turbine bypass system that dumps steam to the condenser. This requires offsite power recovery to operate the circulating water pumps to enable restoration of condenser vacuum. Due to the capacity of the SG PORVs and atmospheric steam dumps, McGuire could be cooled down without the recovery of the condenser. The PORVs, which are upstream of the MSIVs, are sized for 10% full load (2.5% per valve). The atmospheric steam dumps, which are downstream of the MSIVs, are rated at 45% of full load. The failure probability was assumed to be dominated by operator actions associated with the cooldown and depressurization. An operator error probability of 0.003 was utilized (see *NRR Daily Events Evaluation Manual*, 1-275-03-336-01, January 31, 1992). Note that this cooldown may be performed under natural circulation conditions if offsite power is not recovered. If it is recovered, RCPs would restore forced flow to the RCS and facilitate the cooldown.

RHR—The RHR system is used to provide long-term core cooling. The two trains of RHR are redundant except for the hot-leg suction line. Each train consists of a common suction line with two valves in series and an RHR pump. The discharge lines to the cold legs are normally open and would not require manipulation. The suction valves from the refueling water storage tank (RWST) would have to be closed to switch to the cooldown mode, following the RSWT depletion. The RHR failure probability is, therefore, approximately {p(VLV A) + p(VLV B) + [p(PMP A) × p(PMP B | PMP A)]} × p(nrec) + p(oper). Using typical ASP screening probabilities for pump and valve failures, a nonrecovery probability of 0.34, and an operator error probability of 0.001 (see *NRR Daily Events Evaluation Manual*, 1-275-03-336-01, January 31, 1992) results in an estimated failure probability for the branch of

$$\{[(0.01 + 0.01) + (0.01 \times 0.1)] \times 0.34\} + 0.001 = 8.1 \times 10^{-3}.$$

A.9.5 Analysis Results

The estimated conditional core damage probability for the LOOP initiator for this event is 9.3×10^{-5}. This value reflects the incorporation of the SSF into the model. The conditional core damage probability without incorporation of the SSF is 2.6×10^{-4}. The dominant core damage sequence, highlighted on the event tree in Fig. A.9.4, is the same whether or not the SSF is incorporated into the modeling. It involves a LOOP, successful reactor trip, failure of the EDGs, AFW success, challenge of the PORV or SRV followed by successful reseating, and failure of long-term recovery of offsite power prior to battery depletion.

Using the SGTR event tree model as shown in Fig. A.9.3 with the initiating event frequency set to 1.1×10^{-2}, the conditional core damage probability for a postulated SGTR is 1.2×10^{-4}. The dominant sequence for the SGTR is sequence A. This sequence represents a LOOP followed by a postulated SGTR, followed by successful AFW and HPI, failure to isolate the ruptured SG, and failure of the RHR system. The conditional core damage probability for sequence A is 8.9×10^{-5}. Consideration of a potential SGTR increases the conditional core damage probability estimated for this event to 2.1×10^{-4}, a factor of 2.3 increase over the LOOP calculation alone.

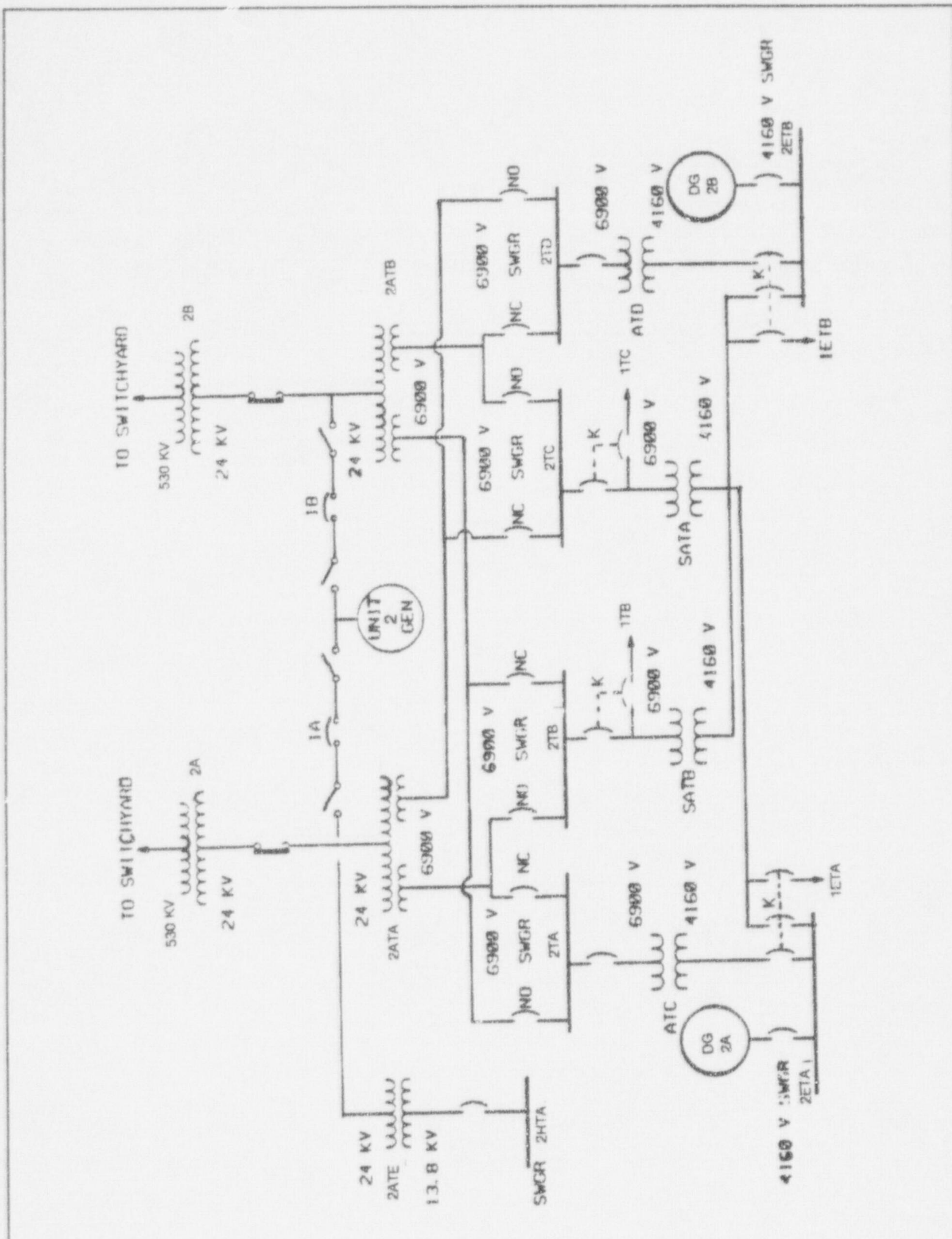


Fig. A.9.2 Simplified electrical distribution system for McGuire 2

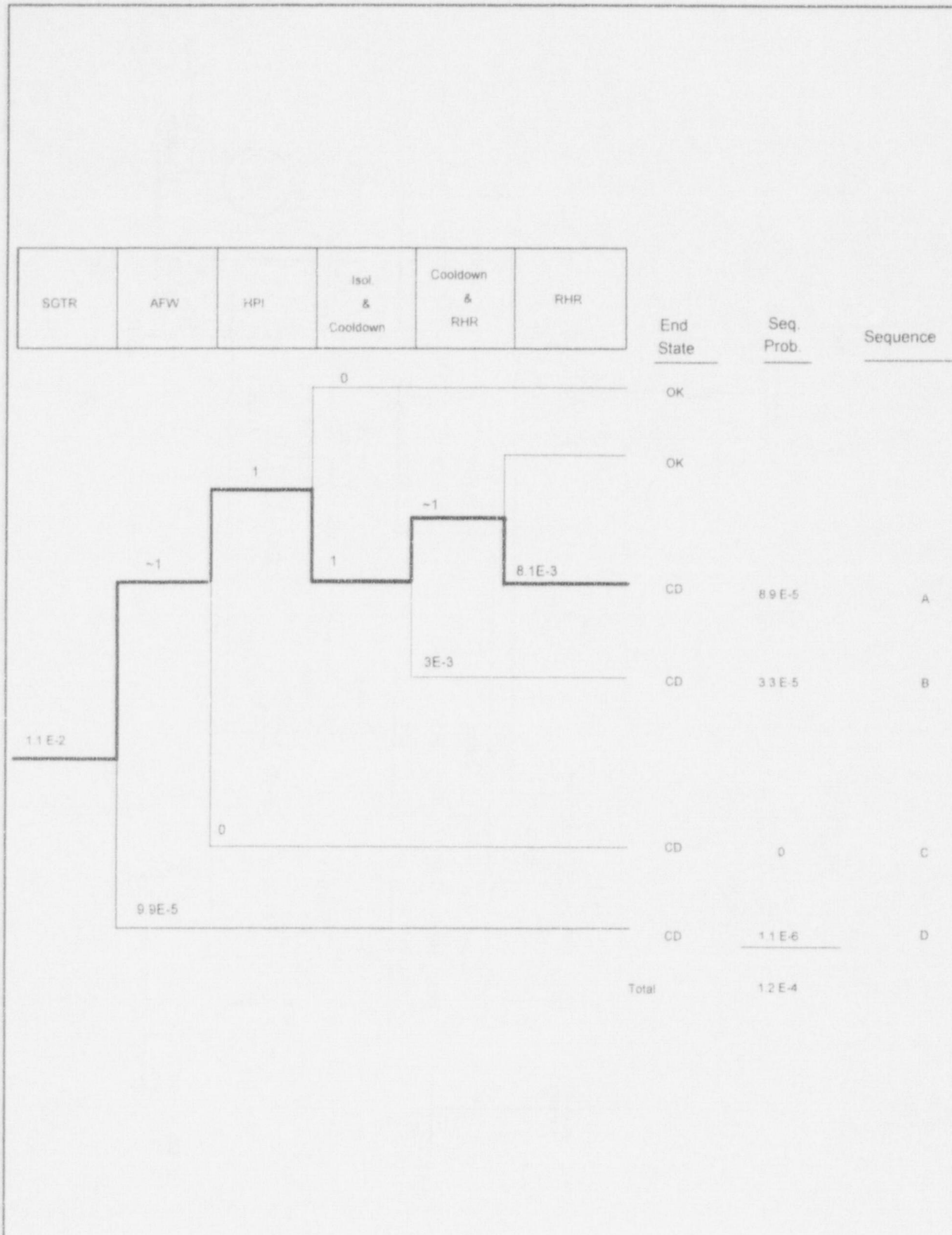


Fig. A.9.3 SGTR event tree showing dominant sequence

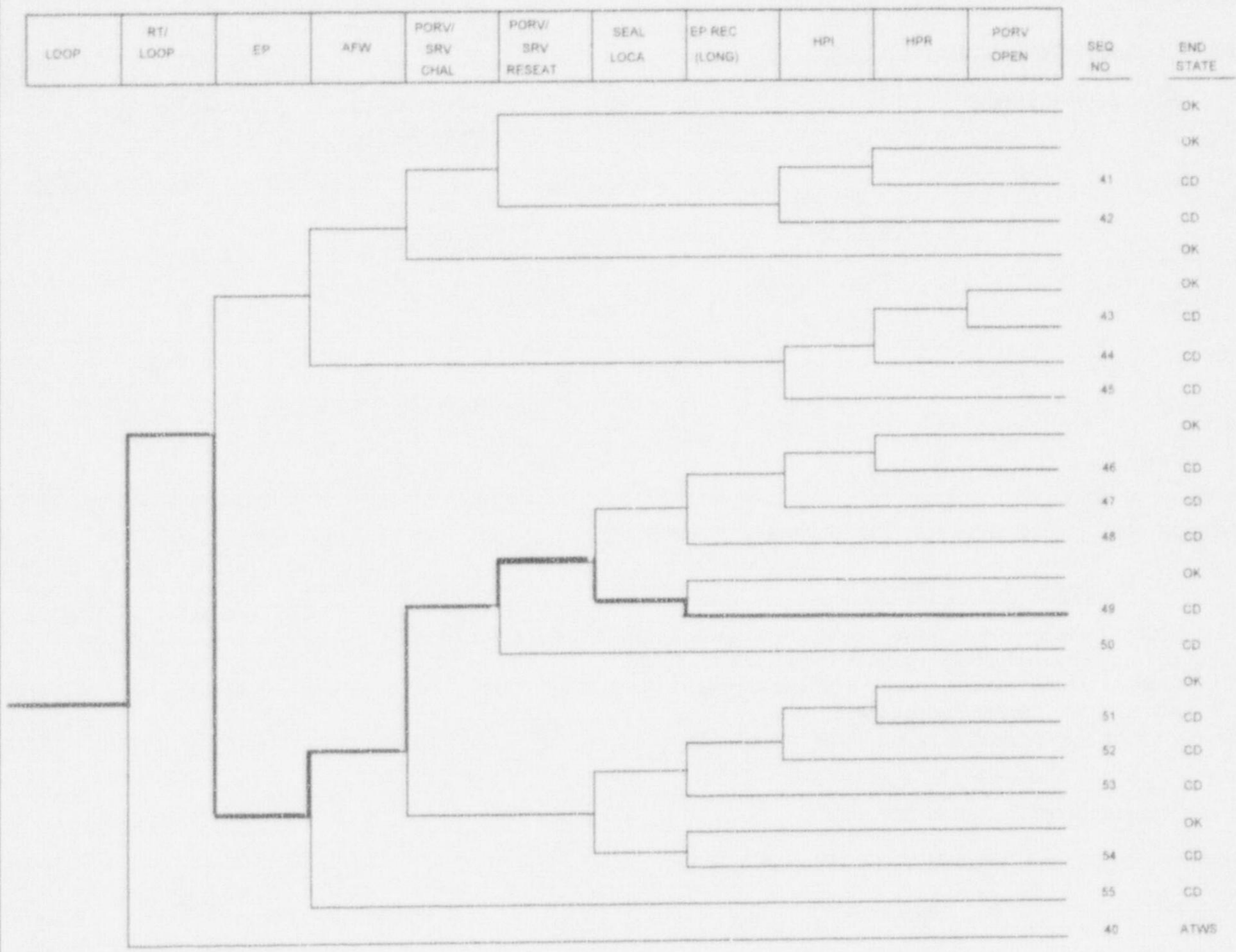


Fig. A.9.4 Dominant core damage sequence for LER 370/93-008.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 370/93-008
Event Description: LOOP
Event Date: 12/27/93
Plant: McGuire 2

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	3.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability w/o SSF	Probability w/ SSF
CD		
LOOP	2.6E-04	9.3E-05
Total	2.6E-04	9.3E-05
ATWS		
LOOP	0.0E+00	
Total	0.0E+00	

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob w/o SSF	Prob w/SSF
49	LOOP -rt/loop emerg.power -afw/emerg.power PORV.OR.SRV.CHALL -porv.or.srv.reseat/emerg.power -seal.loca EP.REC	CD	2.3E-04*(0.26)=	6.0E-05
50	LOOP -rt/loop emerg.power -afw/emerg.power PORV.OR.SRV.CHALL porv.or.srv.reseat/emerg.power	CD	2.0E-05	2.0E-05
55	LOOP -rt/loop emerg.power afw/emerg.power	CD	1.2E-05	1.2E-05

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob w/o SSF	Prob w/SSF
49	LOOP -rt/loop emerg.power -afw/emerg.power PORV.OR.SRV.CHALL -porv.or.srv.reseat/emerg.power -seal.loca EP.REC	CD	2.3E-04 *(0.26)=	6.0E-05
50	LOOP -rt/loop emerg.power -afw/emerg.power PORV.OR.SRV.CHALL porv.or.srv.reseat/emerg.power	CD	2.0E-05	2.0E-05
55	LOOP -rt/loop emerg.power afw/emerg.power	CD	1.2E-05	1.2E-05

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\mcguire.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail

trans	4.3E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	3.6E-01 > 3.0E-01	
Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
afw	3.8E-04	2.6E-01	
afw/emerg.power	5.0E-02	3.4E-01	
mfw	1.0E+00	7.0E-02	1.0E-03
PORV.OR.SRV.CHALL	4.0E-02 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.0E-02 > 1.0E+00		
PORV.OR.SRV.RESEAT	3.0E-02 > 1.2E-01	1.[illegible]E-02	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	3.0E-02 > 1.2E-01		
porv.or.srv.reseat/emerg.power	3.0E-02	1.0E+00	
seal.loca	0.0E+00	1.0E+00	
ep.rec(sl)	0.0E+00	1.0E+00	
EP.REC	4.5E-01 > 3.5E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.5E-01 > 3.5E-01		
hpi	1.0E-03	8.4E-01	
hpi(f/b)	2.2E-03	8.4E-01	1.0E-02
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
porv.open	1.0E-02	1.0E+00	4.0E-04

* branch model file
** forced

A.10 LER No. 373/93-015

Event Description: Scram and Loss-of-offsite Power

Date of Event: September 14, 1993

Plant: LaSalle 1

A.10.1 Summary

LaSalle 1 was operating at 100% power on September 14, 1993, when a fault occurred in the buswork associated with the system auxiliary transformer (SAT). The resulting electrical system perturbations caused the loss of one main feed pump and a reactor scram on low vessel level. When the main generator separated from the grid, the unit auxiliary transformer (UAT) was no longer able to provide power to plant auxiliaries, and a loss-of-offsite power (LOOP) ensued. The conditional core damage probability estimated for this event is 1.3×10^{-4}. The relative significance of this event compared to other postulated events at LaSalle 1 is shown in Fig. A.10.1.

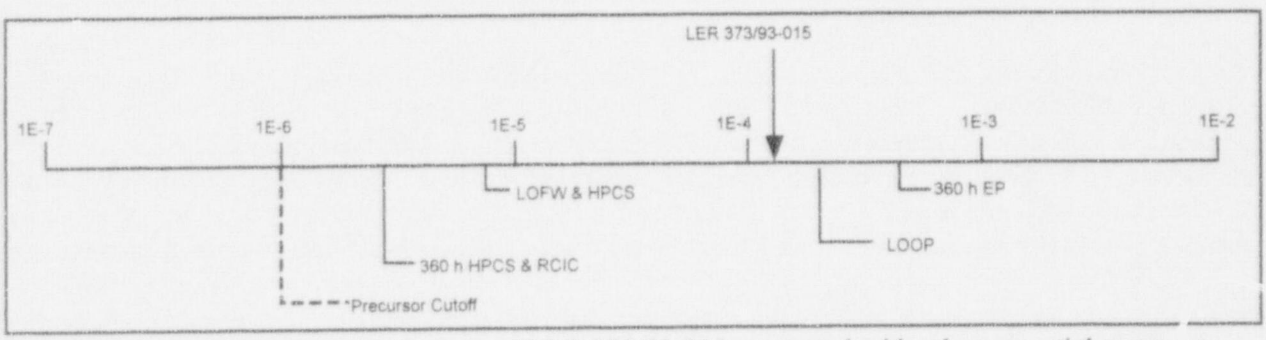


Fig. A.10.1 Relative event significance of LER 373/93-015 compared with other potential events at LaSalle 1

A.10.2 Event Description

Water leaking into the Unit 1 SAT bus duct caused a severe electrical fault at LaSalle 1 while the plant was at 100% power. SAT output voltage dropped sharply, and loads fed from the SAT transferred to the UAT. The voltage reduction caused the 1B turbine-driven reactor feed pump control circuitry to reduce pump flow to zero. Shortly thereafter, the reactor scrammed on low vessel level.

Reactor makeup after the scram was initially supplied by the motor-driven reactor feed pump, but the vessel overfilled, resulting in feed pump and main turbine/generator trips. Once the main generator separated from the grid and the UAT was deenergized, the plant experienced a LOOP. The emergency diesel generators (EDGs) started and loaded to supply emergency buses, and the high-pressure core spray (HPCS) diesel started. Safety/relief valves (SRVs) were operated to reduce pressure by relieving steam to the suppression pool. Suppression pool cooling (SPC) was initiated, and reactor core isolation cooling (RCIC) was aligned for vessel makeup. After about 75 min, offsite ac power was restored to Unit 1 by connecting Unit 1 buses to Unit 2. Late in the event, one SRV failed to operate on demand. When reactor pressure decreased to 500 psig, the low-pressure core spray (LPCS) system was aligned to provide makeup, and the reactor was then placed in shutdown cooling (SDC).

During the event, ambiguous position indication was observed for several SRVs. An investigation determined that one valve was misaligned and did not fully open. The other ambiguous position indications were attributed to miscalibrated position indicators. The safety parameter display system (SPDS) also lost power during the event and was unavailable; however, redundant control room instrumentation and the process computer remained available.

A.10.3 Additional Event-Related Information

In the event that residual heat removal (RHR) fails, the containment can be vented to remove decay heat and prevent overpressurization. To achieve this, the operator manually vents the suppression pool or the drywell. The steaming that will occur in the suppression pool may fail any injection source [such as low-pressure coolant injection (LPCI)] that draws from the suppression pool. Therefore, the feed operation associated with venting must come from an injection system that operates at low pressure and that has a source of water other than the suppression pool.

A.10.4 Modeling Assumptions

This event was modeled as a plant-centered LOOP, in accordance with the LOOP classification scheme of NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*. The short-term (within the first 30 min) LOOP nonrecovery probability was assumed to be 1.0, given that the SAT was tripped and locked out by the fault. The probability of not recovering ac power before battery depletion following the postulated failure of the EDGs was determined using the models described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89-11, August 1989. These models are based on the results of the data distributions contained in NUREG-1032. Because offsite power sources remained available throughout the event and procedures were in place to line up these sources, it was assumed that the overall recovery would be similar to a normal LOOP. That is, the probability of not recovering ac power at the point of battery depletion was assumed to be essentially equal to a normal LOOP, although the probability of not recovering ac power at 30 min was assumed to be 1.0 because of the problems with the SAT. Therefore, the long-term (before battery depletion) nonrecovery was revised to be the overall (combined short- and long-term) nonrecovery probability (5.5×10^{-2}).

The SRV failure that occurred was attributed to leakage of nitrogen control air from a nonsafety accumulator over a 2-h period. Because the automatic depressurization system (ADS) accumulator for the SRV was unaffected, ADS was not impacted, and no changes were made to its failure probability included in the accident sequence precursor (ASP) models. Also, because additional indications were available, the SRV indicator and SPDS malfunctions were not considered to impact any failure probabilities included in the ASP models.

The current ASP event trees for LaSalle do not model the potential use of RCIC to provide reactor pressure vessel makeup in the event of a single stuck-open SRV. However, the use of RCIC for this purpose was included in NUREG-1150 and the utility's individual plant examination. To address this, the conditional probabilities for the applicable sequences (sequences 50-55 and 69 in the event tree in Fig. A.10.2) were reduced by the probability of failing to successfully use RCIC for this purpose. This is the probability that either RCIC fails, two or more SRVs fail to close given one or more fails to close, or RHR long-term cooling (SDC and SPC) fails given RCIC is successful and only one SRV fails open. This probability can be approximated by p(RCIC) + p(2 or more valves fail open | 1 or more valves fail open) .

This approximation assumes that sequences involving RCIC success avoid core damage if RHR is also successful. Because the probability of RHR failure is very small relative to the probability of failing RCIC, this approximation is valid. The failure probability for RCIC during this event was estimated at 0.042, the nominal ASP value. A value of 0.027 was estimated for p(2 or more valves fail open | 1 or more valves fail open), based on an estimated probability of 0.0015 for two or more SRVs stuck open (see NUREG/CR-4550, Vol. 1, Rev. 1, *Analysis of Core Damage Frequency: Internal Events Methodology*, January 1990, p. 6–10) and an estimated probability of 0.056 for one or more SRVs stuck open (developed as described in NUREG/CR-4674, Vol. 1, *Precursors to Potential Core Damage Accidents: 1985 A Status Report*, Appendix C). The multiplier used to adjust the conditional probability for sequences 50–55 and 69 to account for this potential use of RCIC to mitigate the effects of a single stuck-open SRV is, therefore, 0.042 [p(RCIC)] + 0.027 [p(2 or more SRVs fail open | 1 or more SRVs fail open)] = 0.069.

The existing ASP model was also modified to include the potential use of containment venting for decay heat removal in the event that both RHR/SPC and RHR/ SDC fail. This was done by revising the conditional probability for sequence failure of both RHR cooling modes (sequences 40–44, 47, and equipment) to also include failure to vent the containment. The probability of failing to vent was assumed to be dominated by human error. A probability of 0.01 was utilized for sequences in which the injection source operates at low pressure and has a source of water separate from the suppression pool. For sequences in which the injection source takes suction from the suppression pool (such as LPCS or LPCI), an alternate injection source, the control rod drive (CRD) pumps or essential service water (SW) (RHR.SW in the ASP models), must be aligned for injection following venting. Venting is considered much less reliable in such cases; an operator error probability of 0.5 was utilized (see *NRR Daily Events Evaluation Manual*, 1-275-03-336-01, January 31, 1992).

The conditional probability for sequence 40 was revised from 5.6×10^{-6} to 5.6×10^{-8} to reflect the potential use of containment venting (although HPCS provides injection success in this sequence, the CRD pumps, which do not take suction from the suppression pool, are also assumed to be available for injection). The conditional probability for sequence 55 was revised from 4.8×10^{-6} to 5.3×10^{-7} to reflect the potential use of RCIC following a single stuck-open relief valve. Other sequences that were potentially impacted by these two changes had calculated probabilities below 4.0×10^{-6} (unmodified) and had minimal effect on the core damage probability estimated for the event. The probability values for these sequences, which are not shown on the calculation sheets, were not revised.

During the event, reactor protection system (RPS) motor generator set B tripped on a motor fault. Because an alternate supply to RPS bus B could not be reenergized from the Unit 2 feed or from the EDGs, certain primary containment isolations could not be immediately reset, including one affecting SDC. It was necessary to arrange a temporary power feed to RPS bus B before SDC could be placed in service. It was assumed that the failure to deal successfully with the RPS failure and recover SDC was bounded by the existing operator error value for the SDC branch, because substantial time was available for alignment of SDC.

A.10.5 Analysis Results

The conditional probability of subsequent core damage for this event is estimated to be 1.3×10^{-4}. The dominant core damage sequence, highlighted on the event tree presented in Fig. A.10.2, involves the plant-centered LOOP, a postulated failure of emergency power, and failure to recover emergency power before battery depletion.

Consideration of RCIC for makeup after a single stuck-open relief valve and containment venting, following the postulated loss of RHR shutdown and SPC have little effect on the core damage probability estimated for the event.

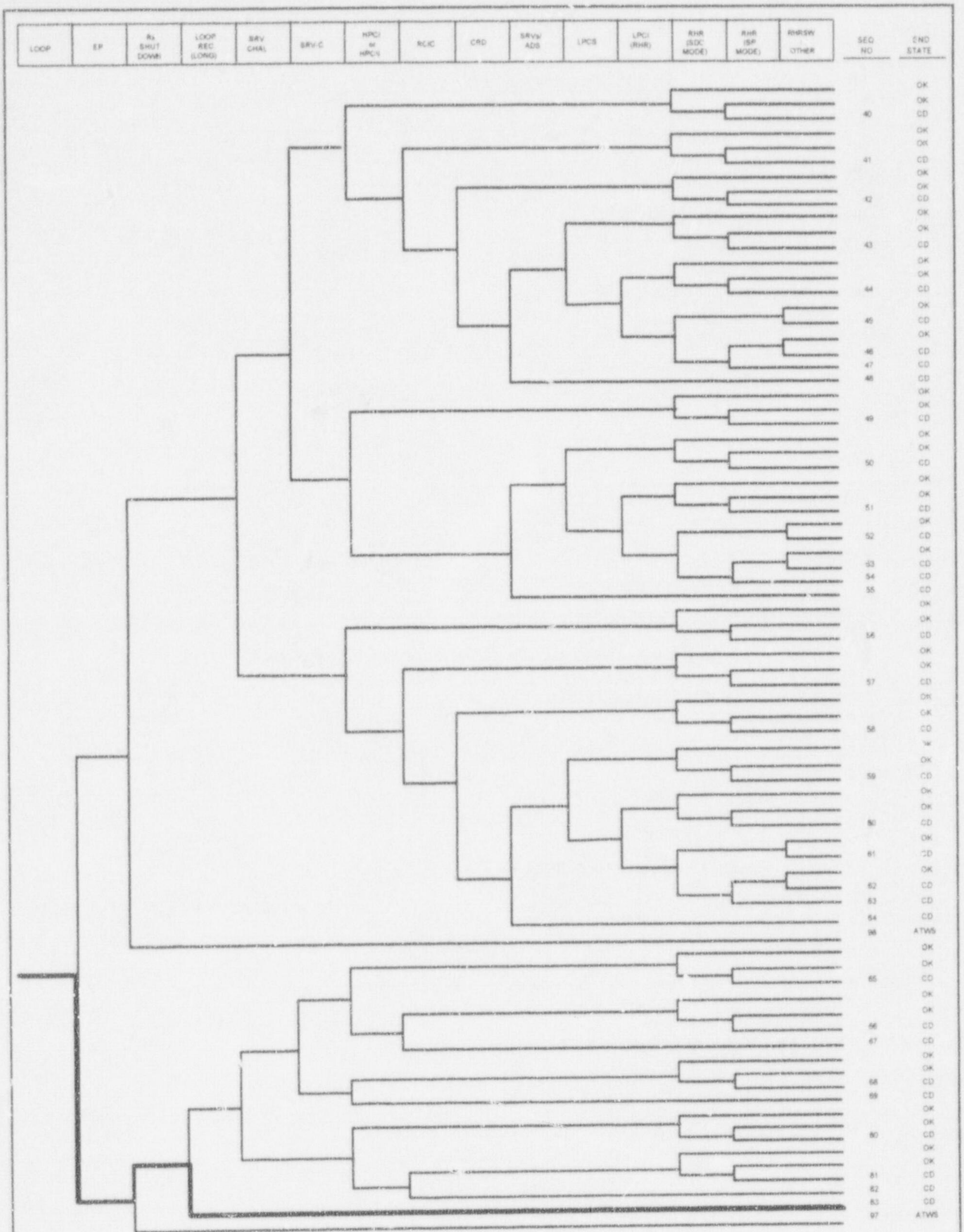


Fig. A.10.2 Dominant core damage sequence for LER 373/93-015

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 373/93-015
Event Description: Scram and Loss-of-Offsite Power
Event Date: September 14, 1993
Plant: LaSalle 1

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	1.4E-04(1)
Total	1.4E-04(1)
ATWS	
LOOP	3.0E-05
Total	3.0E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
83	LOOP emerg.power -rx.shutdown/ep EP.REC	CD	1.3E-04	8.0E-01
40	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	5.6E-06(2)	1.2E-01
55	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close hpci srv.ads	CD	4.8E-06(2)	2.4E-01
98	LOOP -emerg.power rx.shutdown	ATWS	3.0E-05	1.0E+00

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
40	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	5.6E-06(2)	1.2E-01
55	LOOP -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close hpci srv.ads	CD	4.8E-06(2)	2.4E-01
98	LOOP -emerg.power rx.shutdown	ATWS	3.0E-05	1.0E+00
83	LOOP emerg.power -rx.shutdown/ep EP.REC	CD	1.3E-04	8.0E-01

** nonrecovery credit for edited case
SEQUENCE MODEL: e:\asp\models\bwrcseal.cmp
BRANCH MODEL: e:\asp\models\lasalle.SL1
PROBABILITY FILE: e:\asp\models\BWR_CSL1.PRO

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	7.4E-05	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	5.3E-01 > 1.0E+00	

Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
pcs/trans	1.7E-01	1.0E+00	
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	5.6E-02	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
EP.REC	1.7E-01 > 5.5E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > 5.5E-02(3)		
fw/pcs.trans	4.6E-01	3.4E-01	
fw/pcs.loca	1.0E+00	3.4E-01	
hpci	2.0E-02	3.4E-01	
rcic	6.0E-02	7.0E-01	
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	2.0E-02	3.4E-01	
lpci(rhr)/lpcs	6.0E-04	7.1E-01	
rhr(sdc)	2.3E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
rhr(spcool)/rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/-lpci.rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/lpci.rhr(sdc)	9.3E-02	1.0E+00	
rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file
** forced

Notes:
1. See Analysis Results for the core damage probability estimated for this event after consideration of additional mitigating features.
2. See Modeling Assumptions for the revised conditional probability for this sequence.
3. Value modified to account for overall AC power recovery. See Modeling Assumptions for the development of this value.

A.11 LER No. 412/93-012

Event Description: Failure of Both Emergency Diesel Generator Load Sequencers

Date of Event: November 4-6, 1993

Plant: Beaver Valley 2

A.11.1 Summary

On November 4, 1993, the automatic loading capability of the 2-1 emergency diesel generator (EDG) on a safety injection (SI) signal failed during a test. Two days later, on November 6, 1993, the automatic loading capability of the 2-2 EDG on an SI signal also failed during a test. This failure would only occur when an SI signal was present coincident with a loss of the normal power supply to the engineered safety features (ESF) bus. The failure mechanism had existed since November 1990. Operator actions would have been necessary to allow manual loading of equipment on the ESF buses. The conditional core damage probability estimated for this event is 2.1×10^{-6}. The relative significance of this event compared to other postulated events at Beaver Valley 2 is shown in Fig. A.11.1.

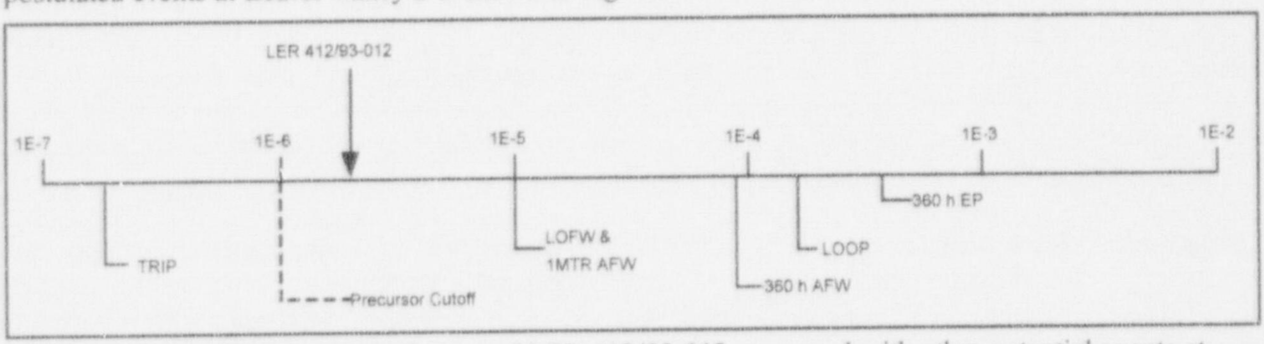


Fig. A.11.1 Relative event significance of LER 412/93-012 compared with other potential events at Beaver Valley 2

A.11.2 Event Description

On November 4, 1993, with the Beaver Valley 2 plant in cold shutdown, a test of the automatic loading capability of the 2-1 EDG on an SI signal was conducted. The test is performed during each refueling outage and verifies that the EDG circuitry will automatically load the safety-related loads on the ESF buses at the required time following the EDG start. During the test, the EDG started and reenergized the associated ESF bus, but the safety-related equipment did not automatically sequence onto the bus as expected. Approximately 2 min into the test, the SI signal was reset, and the loads began to automatically sequence on the bus. An investigation following the test indicated that two relays in the solid-state protection system had the potential to cause the observed failures. The two relays were replaced and the test was successfully rerun the following day.

On November 6, 1993, the automatic loading capability of the 2-2 EDG on SI also failed during a test. Diagnostic test equipment installed on the load sequencer identified the SI reset relay as the cause of the failure. This relay resets the sequencer if an SI signal occurs during a loss-of-bus-voltage event. Voltage spikes caused by the opening of this relay resulted in the relay reclosing. This caused the loading sequencer to "lock-up."

This failure would only occur when an SI signal was present coincident with a loss of the normal power supply to the ESF bus. The automatic loading would have functioned properly for a postulated accident without the loss of normal power or if the SI signal was actuated after the normal supply to the ESF bus was restored. The failure mechanism had existed since a modification of the sequencer relays in November 1990 (36 months). Operator actions would have been necessary to allow manual loading of equipment on the ESF buses. These actions include locally resetting the motor-control centers to restore service water (SW) to the EDGs, the high-head SI (HHSI) pump coolers, and to operate essential emergency-core-cooling system (ECCS) valves.

A.11.3 Additional Event-Related Information

Two sequencers automatically place vital safety-related equipment onto the ESF buses. The SI sequencer will operate whenever an SI signal is generated, regardless of the source of power to the ESF buses. This sequencer loads essentially all of the ESF equipment onto the ESF buses. The blackout sequencer will operate whenever the EDGs are required to supply power to the ESF buses and an SI signal is not present. Most ESF equipment is loaded by this sequencer. This includes the HHSI pumps but does not include the low-head SI (LHSI) pumps. The load sequencers are used to distribute the loads placed on the EDG in six discrete steps over a 1-min period. This prevents an overload of the EDG by spreading out the high starting currents of the motors over time. If a blackout signal and an SI signal are generated simultaneously, the SI sequence will be implemented. If an SI signal occurs after the blackout sequencer has gone to completion, the equipment loaded by the blackout sequencer remains connected to the bus. The additional equipment started by the SI sequencer would be loaded onto the bus.

During the first refueling outage in 1989, problems were encountered with obtaining the necessary set-point repeatability with the existing electromechanical timer/relays used in the sequencer circuitry. During the second refueling outage in 1990, the electromechanical relays were replaced with microprocessor-based timer/relays to improve set-point repeatability. The timers were also modified to be continuously energized to improve performance. During the third refueling outage, tests revealed that three of the eight timer/relays in each train had failed. The failures were due to overheating caused by the continuous energization. The timer/relay configuration was changed to be energized only when actuated. These previous failures were unrelated to the cause of the failures in 1993. Following the 1993 failures, diodes were installed to suppress the voltage spikes across the relays. The results of tests following the modification showed no failures after 80 cycles.

A.11.4 Modeling Assumptions

Four situations were considered: (1) a postulated loss-of-offsite power (LOOP) where SI is initiated for feed-and-bleed, (2) a postulated LOOP with an SI occurring as a result of equipment failures or operator actions, (3) a LOOP-induced loss-of-coolant accident (LOCA), and (4) a postulated LOCA that induces a LOOP as a result of the effects of the plant trip on the electrical grid.

A.11.4.1 Feed-and-Bleed following a postulated LOOP

Following the postulated LOOP, feed-and-bleed would be required after the postulated failure of main and auxiliary feedwater (AFW). The requirement for feed-and-bleed would have to occur before offsite power was recovered. Recovery of offsite power would eliminate the lock-up problem. During the initial LOOP response, the blackout sequencers would operate (because they were not affected by the relay problems) and would start the EDGs and load most ESF equipment, including the HHSI/charging pumps. When feed-and-bleed is initiated, the operators would manually actuate SI to enable the automatic switchover to containment sump recirculation. The sequencers would lock-up at this point. However, the need for the LHSI pumps and the switchover to containment sump recirculation would not occur for an extended period. In addition, procedures would require the reduction of the SI flow rate [to one pump and one power-operated relief valve (PORV)] to reduce the rate that the refueling water storage tank is depleted and extend the time to recirculation. To recover from this failure, the operator would have to reset the sequencer locally and manually start the LHSI pumps from the control room. Given the likelihood of a LOOP coincident with a failure of all AFW (motor- and turbine-driven pumps) and the extended period of time the operator has to recover from the sequencer failure, this scenario did not contribute significantly to the conditional core damage probability.

A.11.4.2 SI signal generated following a postulated LOOP

In the second situation, following the postulated LOOP, a postulated SI signal would be generated due to equipment failures or operator actions. For example, a stuck open steam generator (SG) safety valve would actuate SI due to the large cooldown and depressurization of the primary system. An operator overfeeding a SG could have a similar effect. Voltage perturbations during electrical system transients and realignments could cause spurious actuations of the SI relays.

A search of LOOP events from 1987 through 1993 revealed that of the 55 LOOPs that occurred during that time period, 9 involved the actuation of SI (four at pressurized-water reactors, five at boiling-water reactors). These SI signals were generated as the result of operator actions and equipment failures similar to those described in the examples above. In the most severe of these cases, HHSI was required initially. However, long-term use of SI that would require switchover to recirculation was not necessary in any of these events.

During the initial LOOP response, the blackout sequencers would operate (because they were not affected by the relay problems) and would start the EDGs and load most ESF equipment, including the HHSI/charging pumps. Because the switchover to the recirculation mode would not be required, no additional operator actions would be needed in this situation. Therefore the sequencer lock-up problem is not a concern in this situation. For plants where the HHSI pumps are not started by the blackout sequencer, this situation could contribute significantly to the conditional core damage probability.

A.11.4.3 LOOP-induced LOCA

In this case, the plant response to the postulated LOOP results in a postulated LOCA event. This scenario was considered highly unlikely except for the potential for a stuck-open pressurizer PORV/safety relief valve (SRV) following the LOOP. This situation is similar to that of the bleed-and-feed scenario in that the HHSI pumps are started by the blackout sequencer before the initiation of the SI sequencer. However, in this case, long-term SI operation would be required. Operator actions would be needed to start the LHSI pumps because the SI sequencer would lock-up before starting these pumps. Given the low probability for this event and the low operator failure rate (due to the extended period for LHSI recovery), this situation did not significantly contribute to the overall conditional core damage probability for this event.

A.11.4.4 LOCA with transient-induced LOOP

In this case, a postulated LOCA is the initiating event. When the plant trips in response to the LOCA, the transient results in the LOOP to the station. If offsite power is available, loads are fast-transferred to the alternate offsite power source, and the SI sequencer would operate properly. If offsite power is not available, then the EDGs will start. The normal feeder breaker to the ESF buses trips open, and load shedding occurs. The sequencer would then start and lock-up. It is assumed that one-half hour is available to establish make-up to the reactor coolant system (RCS) before core damage will occur. This event tree (Fig. A.11.2) is based on the accident sequence precursor (ASP) LOCA tree for PWR class A plants (see NUREG/CR-4674, Vol. 17, Appendix A, Section A.3-1 and Fig. A.6). Values used in the quantification of the event tree are provided in Table A.11.1.

A.11.4.5 LOCA initiation frequency

The condition existed since the 1990 refueling outage (~ 36 months). ASP analyses have typically modeled long-term unavailabilities for a 1-year period. ASP initiating event frequencies are based on operation for 70% of a year (an approximation of the percentage of the year that a typical plant spent at power). Therefore, the initiating event frequency is multiplied by 6132 h (365 d × 24 h/d × 0.7).

A.11.4.6 LOOP and short-term offsite power recovery

It is assumed that the probability of a LOOP induced by a LOCA is 1.0×10^{-3} (*Reactor Safety Study*, WASH-1400, NUREG-75/014, p. II-90). A search of the Sequence Coding and Search System for transient-induced LOOPs from 1984 to present revealed five transient-induced LOOPs out of 3985 trips. This yields a rate of 1.25×10^{-3} per trip. This provides a degree of substantiation for the WASH-1400 value. It was assumed that offsite power recovery is possible only in the first one-half hour. If the EDGs fail and a LOCA is in progress, offsite power must be restored within a half-hour to repower the HHSI pumps. The nonrecovery value of 0.48 is that associated with a grid-related LOOP (from ORNL/NRC/LTR-98-11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989), because the initiating cause of the LOOP was assumed to be grid disturbance caused by the plant trip. Note that if no LOOP occurs, the event is simply a small-break LOCA, and the sequencers will operate properly. Therefore, this branch does not contribute to the conditional core damage probability.

A.11.4.7 Emergency power

If the EDGs fail to start , it is assumed that insufficient time is available to recover the EDGs and to manually load the buses. Therefore, the nonrecovery value of the EDGs for this case is set to 1.0, and the failure of the EDGs to start leads directly to core damage (sequence 24).

A.11.4.8 Loading of the ESF equipment on the ESF buses

The operator actions necessary to load required equipment onto the ESF bus are treated as a single top event. It would be obvious to the operators that manual actions were required to load equipment on the ESF buses because none of the ESF equipment loads would be picked up by the EDGs. With both an SI and blackout signal present, the SI signal would dominate, and equipment would not be loaded by the blackout sequencers. In addition, the fact that nothing loaded onto the EDGs would probably lead the operators to suspect a sequencer failure. It is assumed that the operators would have procedural guidance to direct their actions. Equipment recovery would have to be prioritized to prevent equipment damage. SW would have to be restored to the running EDGs to cool them, and HHSI pumps would be needed to provide make-up to the RCS. Local actions would be required to reset the MCCs to restore SW to the EDGs, the HHSI pump coolers, and to operate essential ECCS valves. Because this process requires many coordinated actions and local operator actions, an operator failure rate of 0.34 was assumed (ASP Recovery Class R3, see NUREG/CR-4674, Vol. 17, Appendix A, Sect. A.1).

Failure to successfully load equipment onto the ESF bus leads to a core damage state (sequences 11 and 23). Although the turbine-driven AFW pump will operate to remove decay heat, no RCS make-up is provided. Therefore, core damage will occur. If loads are successfully loaded, then the remainder of the tree (sequences 1-10 and 13-23) is the same as the typical LOCA tree (see NUREG/CR-4674, Vol. 17, Appendix A, Sect. A.3-1 and Fig. A.6, Sequences 71-77 and 80-82) and uses standard ASP values.

A.11.5 Analysis Results

The estimate of the conditional core damage probability for this event is 2.1×10^{-6}. There are two dominant core damage sequences, shown in Table A.11.1. Sequence 11 involves a postulated LOCA, a transient-induced LOOP that is recovered in the first half-hour, and failure to load the ESF buses. The other dominant sequence (sequence 23) involves a postulated LOCA, transient-induced LOOP that is not recovered in the first half-hour, initial emergency power success, and failure to load the ESF buses.

The conditional core damage probability is directly affected by the transient-induced LOOP probability and the assumed operator failure rate for the loading of the ESF buses. If the operator failure rate for loading of the ESF buses is changed from 0.34 to 0.12 (ASP recovery class R3), the conditional core damage probability for the event is reduced by a factor of 2.8 to 7.4×10^{-7}.

Additional information concerning this event is included in Augmented Inspection Team report 50-412/93-81.

Table A.11.1. Values used in the quantification of the event tree

Top Event	Description	Value
LOCA	LOCA initiator Initiating frequency = 2.4×10^{-6}, nonrecovery = 0.43 Duration of unavailability = 6132 h	6.3×10^{-3}
LOOP	Transient-induced LOOP Frequency = 1×10^{-3}/demand	1.0×10^{-3}
LOOP	LOOP recovery (short-term)-recovery in the first half-hour for transient-induced LOOP From ORNL/NRC/LTR-98-11*Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989 Nonrecovery in the first half-hour = 0.48	4.8×10^{-1}
RT/LOOP	Reactor trip given a LOOP Failure probability = 0	0
EP	Emergency power system (LOCA and transient-induced LOOP) Failure probability (1 of 2) = Train1 × Train2 × nonrecovery Train 1 = 0.05, Train 2 = 0.057, nonrecovery = 1.0	2.9×10^{-3}
ESF LOADING	Loading of the ESF buses Operator failure probability = 0.34	3.4×10^{-1}

Top Event	Description	Value
AFW	Auxiliary feedwater system Failure probability (1 of 3 + serial failure) = [(Train1 × Train2 × Train3) + Serial] × nonrecovery Train 1 = 0.02, Train 2 = 0.1, Train 3 = 0.05, Serial = 0.00028 Nonrecovery = 0.26	9.9×10^{-5}
MFW	Main feedwater system Failure probability (1 of 1) = Train1 × nonrecovery Train 1 = 0.2, nonrecovery = 0.34	6.8×10^{-2}
HPI for feed-and-bleed	High-pressure injection initiated for feed-and-bleed Failure probability p(HPI) + operator failure p(HPI) = 2.5×10^{-4} Operator failure = 0.01	1.0×10^{-2}
HPR	High-pressure recirculation Failure probability (1 of 2 + operator action) = (Train1 × Train2 × nonrecovery) + operator action Train 1 = 0.01, Train 2 = 0.015, nonrecovery = 1.0 Operator failure = 0.001	1.1×10^{-3}
PORV OPEN	PORV open for feed and bleed Failure probability (1 of 1) = (Train1 × nonrecovery) + operator failure Train 1 = 0.01, nonrecovery = 1.0, Operator failure = 0.0004	1.0×10^{-2}
CSR	Containment sump recirculation Failure probability (2 of 4) = [4(Train 1 × Train 2 × Train 3) - 3(Train 1 × Train 2 × Train 3 × Train 4)] × nonrecovery Train 1 = 0.01, Train 2 = 0.03, Train 3 = 0.1, Train 4 = 0.3 Nonrecovery = 1.0	9.3×10^{-5}

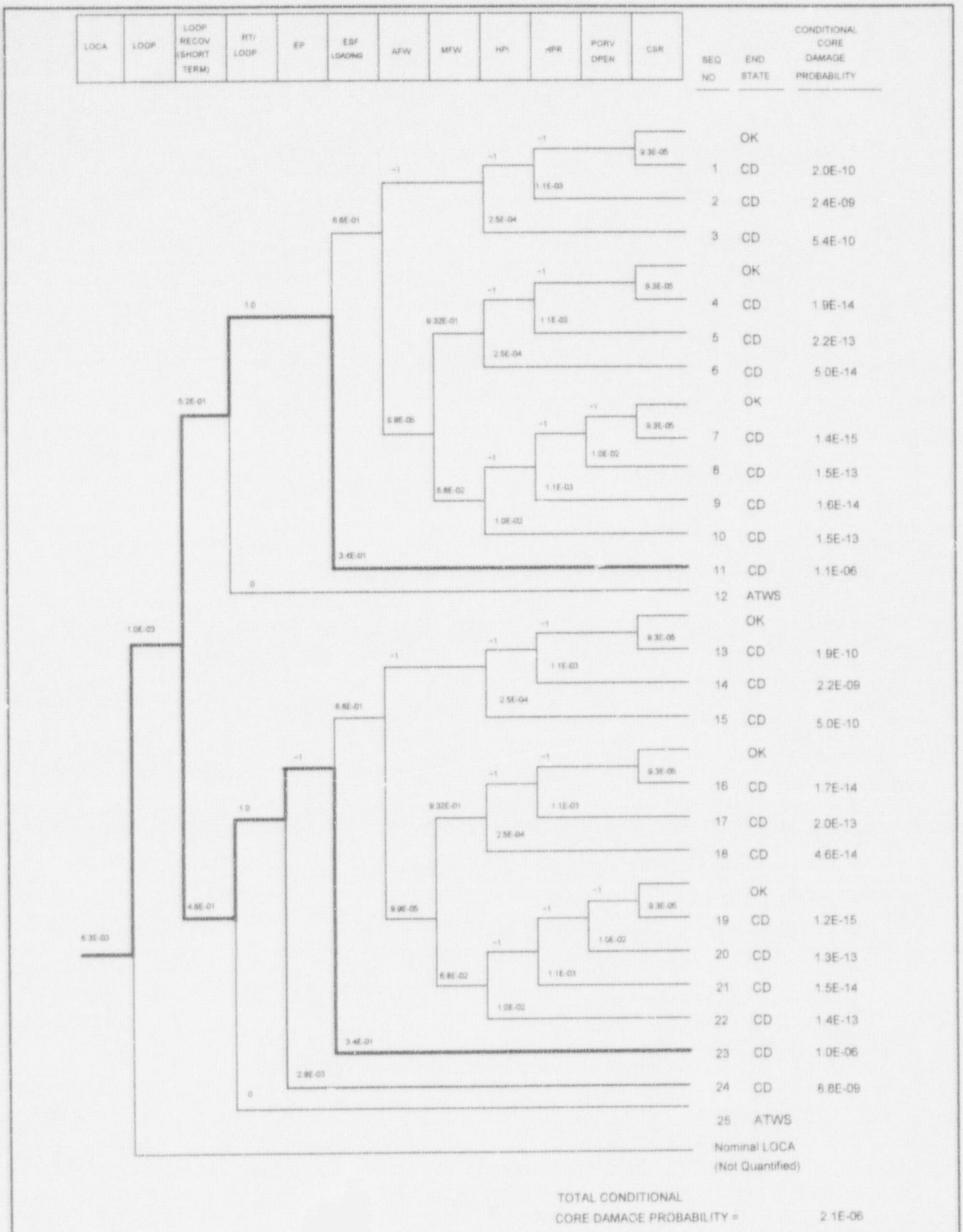


Fig. A.11.2 Dominant core damage sequence for LER 412/93-012

A.12 LER No. 413/93-002

Event Description: Essential Service Water Potentially Unavailable

Date of Event: February 25, 1993

Plant: Catawba 1 and 2

A.12.1 Summary

On February 25, 1993, with Catawba Unit 1 at 100% power and Catawba Unit 2 in a refueling shutdown, three of four essential service water (ESW) pump discharge valves failed to open during surveillance testing. Four ESW pumps serve both units. During normal operation, only one pump is used. If the pump with the operable valve tripped, it would result in the loss of ESW to both units. The conditional core damage probability estimated for this event is 1.5×10^{-4}. The relative significance of this event compared to other potential events at Catawba is shown in Fig. A.12.1.

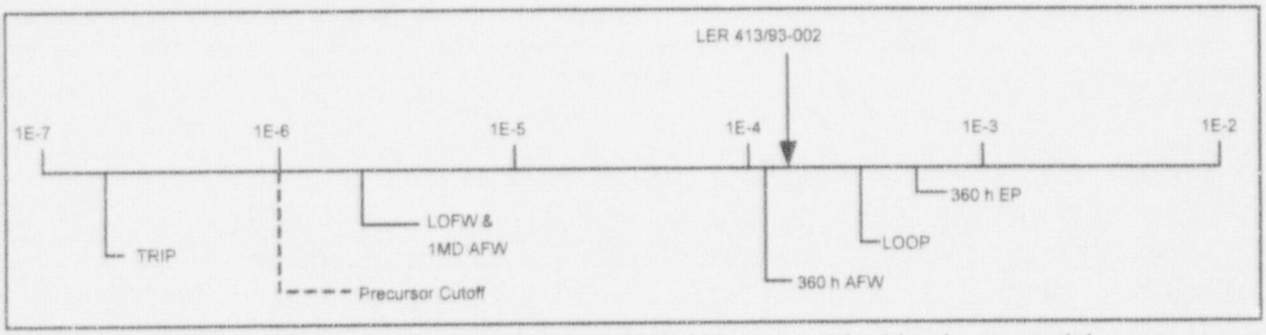


Fig. A.12.1 Relative event significance of LER 413/93-002 compared with other potential events at Catawba 1 and 2

A.12.2 Event Description

Catawba Unit 2 was in a refueling outage on February 25, 1993, when in-service pump testing was begun on the B train of ESW pumps. The cross-connect line between the trains was closed; this depressurized the ESW header downstream of the ESW motor-operated discharge valves. When the 2B pump was started at 1424 hours, the discharge valve failed to open. When the 1B pump was started at 1426 hours, its discharge valve also failed to open. The failures were attributed to higher than expected torque to open the valves when the downstream header was depressurized. Because the A train valves had a similar setup, and therefore were suspectable to the same failure, the A train of ESW was declared inoperable. When the 2A pump was started, the actuator for the discharge valve tripped and reset several times while trying to open. The valve finally opened but it took an additional 25 s.

In 1989, the "open" torque switch settings (TSSs) for 56 butterfly valves were to be set to the maximum value to address the problems of opening these valves under high differential pressure. The four ESW pump discharge valves were included in these 56 valves. The "open" TSSs for the Unit 1 ESW pump discharge valves were set to the maximum value (3.0). However, the "open" TSSs for the Unit 2 valves were incorrectly left at 1.5. The "close" TSS was adjusted to the maximum value instead on these two valves.

In August 1992, the Unit 1 ESW pump discharge valves were set-up per Generic Letter (GL) 89-10 criteria. This resulted in the "open" TSSs being reduced from the maximum value of 3.0 to 2.0. The Unit 2 valves were not reset to the GL 89-10 criteria at the time of the event.

Following the failure of the B train valves on February 25, 1993, the licensee realized that the TSSs for the Unit 2 ESW valves were mistakenly reversed. The TSSs for the Unit 2 valves were changed to the maximum setting. The discharge valves for the Unit 1 pumps were set to 20 deg open. Following these changes, all valves were successfully opened against maximum differential pressure.

The licensee conducted a study of the history of TSSs for the ESW valves and discovered that (1) ESW pump 1A was affected between August 1992 and February 1993, (2) ESW pump 1B was affected between November 1985 and July 1989 and between August 1992 and February 1993, and (3) ESW pump 2B was affected between November 1985 and February 1993. As a result, from August 1992 through February 1993, three of the four ESW pump discharge valves (1A, 1B, and 2B) were unable to open against full differential pressure. This results in a potential loss of ESW to both Catawba units, assuming a single failure of the one operable ESW pump discharge valve or failure of the single ESW pump associated with the operable valve.

A.12.3 Additional Event-Related Information

One ESW pump has sufficient capacity to supply all cooling water requirements during normal power operation of both units or during postaccident conditions if the unaffected unit is already in cold shutdown. Typically, only one pump is run at a time. The running pump is rotated to equalize run time on the four pumps such that each pump is run for ~ 1 week each month. The "2A" ESW pump had operated for 1015 h during the period from August 1992 through January 1993.

The ESW pump discharge valves are closed when their associated pump is not operating. The valves do not have to complete a closure stroke if commanded open while the valve is in midposition. The valve stroke time is about 55 s.

A standby shutdown facility (SSF) is located in a separate building on the Catawba site. This facility, which is not normally manned, is capable of providing limited high-pressure injection for reactor coolant system makeup and reactor coolant pump (RCP) seal cooling [provided an RCP seal loss-of-coolant accident (LOCA) does not occur]. It can also supply limited steam generator (SG) makeup. The facility includes a separate diesel generator that can power SSF loads in the event of a station blackout. SSF systems consist of single trains and are therefore not single-failure-proof. In conjunction with the turbine-driven auxiliary feedwater (AFW) pump and the availability of SGs, the SSF facility can maintain hot standby conditions for both units.

A.12.4 Modeling Assumptions

During a loss-of-offsite power (LOOP), the operating ESW pump discharge valve would be commanded closed but would not be able to close because no power was available. When the EDG was started and loaded or offsite power was recovered, the ESW pump would receive a start signal that would, in turn, command the ESW discharge valve to open. Because the valve would not have had time to close (because of the loss of power), the valve would reopen. In addition, every other ESW pump would also receive a start command. It is assumed that the successful operation of one ESW train would pressurize the header downstream of the discharge valves and reduce the differential pressure across the unopened valves. This would have allowed the other ESW pump discharge valves to open.

There are two other cases to consider during at-power conditions. The first case occurs when an ESW pump other than "2A" is providing normal service water and fails. The second case is when ESW pump "2A" is providing normal service water. The first case is not a problem because it is assumed that pump "2A" would have been started upon failure of the other pumps and its discharge valve would be able to open against the high differential pressure. This would pressurize the ESW headers and allow the other operable ESW pumps and discharge valves to operate.

The second case does present a scenario of concern. If the "2A" ESW pump were the only running pump and it tripped or failed, then the remaining ESW pump discharge valves would not be able to open against the high differential pressure. Information from the licensee indicates that the ESW header pressure would decay rapidly enough that the remaining pump discharge valves would have to open under full differential pressure conditions (conditions from which they failed to open during testing). This would result in failure of the ESW system. Upon failure of the ESW system, the RCP seals would lose cooling. All other safety-related systems would also lose cooling, with the exception of the turbine-driven AFW pump. Without the recovery of ESW or the use of SSF to provide RCP seal injection, a seal LOCA without makeup would result about 50 min after the failure of the ESW system and would lead to core damage.

Therefore this event was modeled as a potential failure of the "2A" ESW pump to run. The event tree for this event is shown in Fig. A.12.2. Following the failure of the "2A" ESW pump, two mitigation strategies are possible.

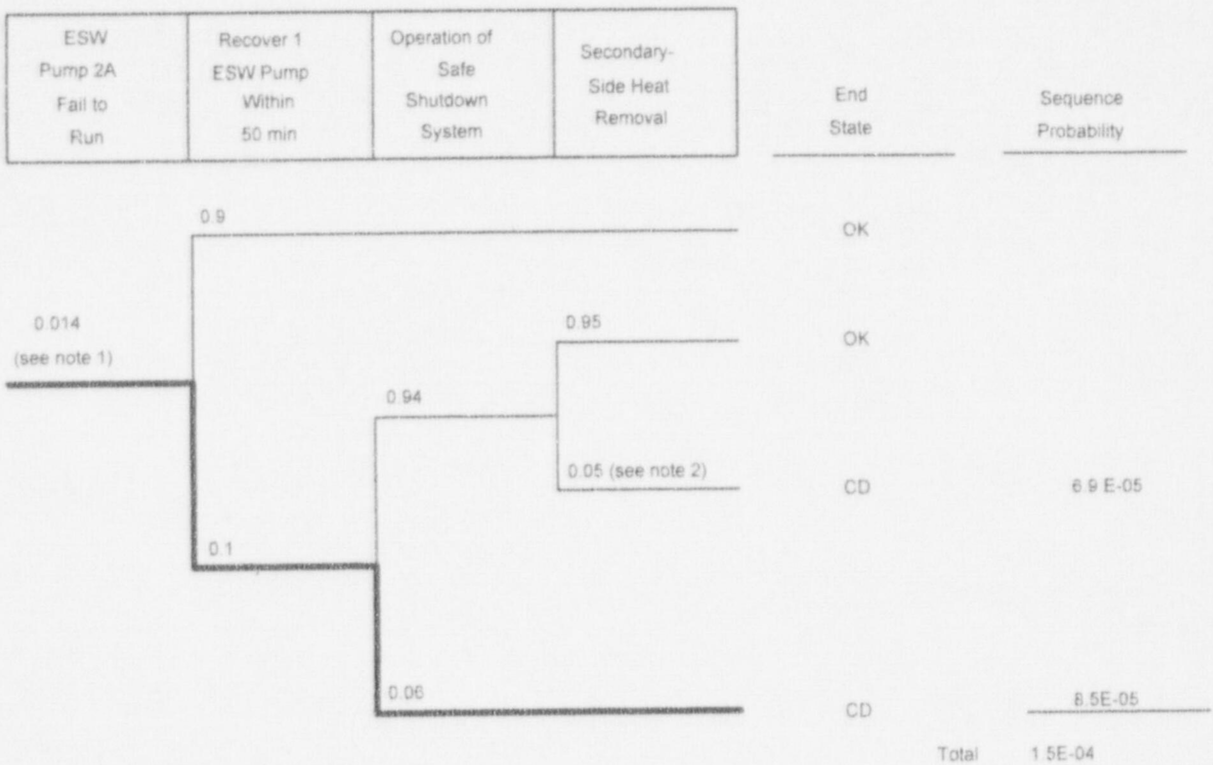
The first involves the recovery of one other ESW pump before an RCP seal LOCA (50 min). Recovery of the one ESW pump would supply sufficient cooling water for both units, assuming that a LOCA did not occur. A LOCA concurrent with a trip of the running ESW pump was considered unlikely. Even in this case, once the first ESW pump is running, the second could be started from the control room because the discharge valves would not have to open against full system differential pressure. Once ESW is recovered, the operability of the systems cooled by ESW is restored.

The other recovery strategy would be to place the SSF in service to provide RCP seal cooling and start the turbine-driven AFW pump to provide secondary-side heat removal. This would allow the plant to achieve a hot shutdown condition even without the restoration of the ESW system.

The failure probability for the ESW pump to run is obtained from Table A.6-9 of the Catawba individual plant examination (IPE) by multiplying the failure frequency (1.4×10^{-5}/h) by the number of hours the 2A pump was running during the period of concern (1015 h). The nonrecovery probability for the remaining ESW trains was estimated by the licensee using the Human Cognitive Reliability (HCR) Model with 20 min required for the recovery action within a 50-min period. Using the HCR Model, a nonrecovery probability of about 0.1 was obtained. The failure probability of the SSF with offsite power available was determined to be about 0.06 (from page 1 of Table A.18-8 of the Catawba IPE). The failure probability for secondary-side heat removal was estimated by summing the failure probability of the turbine-driven AFW pump and its corresponding turbine inlet valve. Failure probabilities for the AFW pump to start (1.2×10^{-2}) and run (1.5×10^{-3}/h) were obtained from Table A.5-8 of the Catawba IPE. A 24-h mission time was assumed in the analysis. The failure probability (4.0×10^{-3}) of the ESW pump discharge valve was also obtained from Table A.5-8 of the Catawba IPE.

A.12.5 Analysis Results

The conditional probability of core damage estimated for this event is 1.5×10^{-4}. The dominant core damage sequence highlighted on the event tree in Fig. A.12.2, involves a failure of the running ESW pump, failure to recover ESW within 50 min, and failure of the SSF. The second core damage sequence involves a failure of the operating ESW pump, failure to recover ESW within 50 min, successful SSF operation, and failure of the secondary-side heat removal.



(1) (Failure of ESW motor-driven pump to run) x (run time required) = (1.4E-05 /h) x (1,015 h) = 1.4E-02

(2) Summation of

(a) turbine-driven pump (TDP) failure to start

(b) TDP failure to run for a specified time

(c) TDP inlet valve fails to open

(1.2E-02) + (1.5E-03/h) x (1.0 d) x (24 h/d) + (4.0E-03) = 5.2E-02

Fig. A.12.2 Dominant core damage sequence for LER 413/93-002

A.13 LER Nos. 440/93-011 and -010

Event Description: Clogged Suppression Pool Strainers and Service Water Flood

Date of Event: March 26, 1993

Plant: Perry

A.13.1 Summary

During a maintenance outage in January 1993, the Perry residual heat removal (RHR) suppression pool suction strainers were found to be deformed because of excessive differential pressure caused by strainer fouling during normal RHR pump operation. The suppression pool was partially inspected and cleaned, and the deformed strainers were replaced.

On March 26, 1993, the reactor was scrammed following a rupture in a 30-in. service water (SW) line. Condenser vacuum was lost, the main steam isolation valves (MSIVs) were closed, and cavitation problems were experienced with a control-rod drive (CRD) pump. The reactor core isolation cooling (RCIC) system was used for pressure vessel makeup. Water from the break entered numerous plant buildings, accumulating in the lowest level of the auxiliary building and control complex, where safety-related equipment is located. No safety-related equipment was impacted by the flood.

Three weeks later, the RHR suppression pool strainers were again inspected. One of the strainers was fouled and deformed. Excessive differential pressures across the RHR strainers from debris accumulation would have failed suppression pool cooling (SPC) if this mode of RHR was required to operate for long periods of time. The conditional core damage probability estimated for this event is 1.2×10^{-4}. The relative significance of this event compared to other postulated events at Perry is shown in Fig. A.13.1.

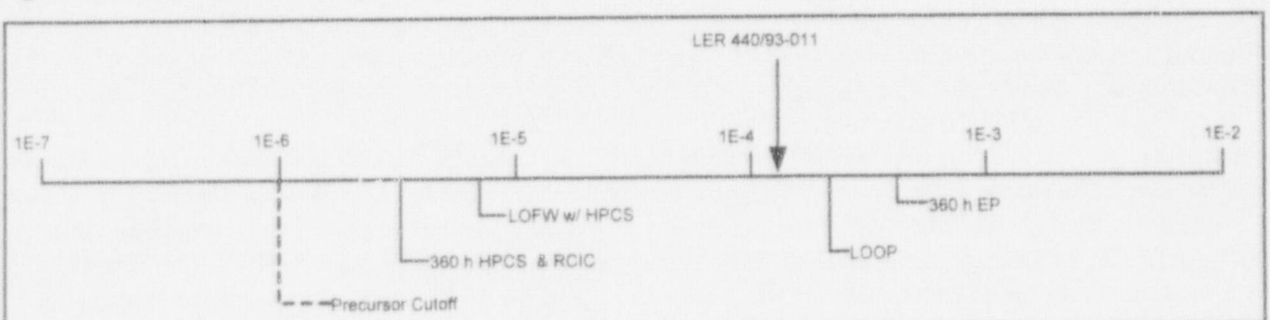


Fig. A.13.1 Relative event significance of LER 440/93-011 compared with other potential events at Perry

A.13.2 Event Description

When the Perry suppression pool was inspected in May 1992, an accumulation of dirt and debris was noticed on the suction strainers for RHR trains A and B. Strainer cleaning was scheduled for a later date, since RHR system performance was considered acceptable based on surveillance testing.

The suppression pool strainers were again inspected and cleaned during a maintenance outage in January 1993. RHR train A and B suction strainers were found to be deformed, with the area of the strainer surface between internal stiffeners partially collapsed inward, in the direction of system flow. It was determined that the strainers were deformed by excessive differential pressure caused by strainer fouling during normal pump operation. Review of a videotape taken during the May 1992 inspection revealed evidence of deformation that had not been noticed at the time of the taping. The containment side of the suppression pool was inspected and cleaned in February 1993, and the deformed strainers were replaced.

On March 26, 1993, the reactor was scrammed at 1526 hours in response to a rupture in a 30-in. SW line. A leak of unknown origin had been detected at 1314 hours, coming from under concrete slabs south of the water treatment building. At 1522 hours, a low SW discharge pressure alarm annunciated in the control room and flow from the break increased substantially. An alert was declared at 1535 hours, about 12 min after the trip and about 16 min after the rupture probably occurred. The total break volume was approximately 1.7 million gal. Approximately 5% of the total leakage entered the auxiliary, intermediate, diesel, turbine, radwaste, and offgas buildings, as well as the control complex, via electrical manway number 1 at the northwest corner of the radwaste building and by flowing under roll-up and access doors on the west side of the plant. Water levels reached during the flood did not impact safety-related equipment.

Flooded building areas included:

Auxiliary Building. A maximum of 5 in. of standing water was reported on elevation 568 ft (lowest level). Water depths of less than 20 in. on this level will not compromise the operability of safety-related equipment. Flooding on elevation 599 ft resulted in leakage into the high-pressure core spray (HPCS) room through the ceiling hatch plugs. The water dripped on the HPCS pump motor, but the motor was not damaged.

Intermediate Building. Water levels of up to 5 in. were reported on elevation 574 ft. Due to the heavy silt content of the flood water, the drains in this building backed up.

Control Complex. Water levels up to 5 in. were reported on elevation 574 ft. Equipment required for safe shutdown and control room habitability is located at 22 in.

Emergency SW Pump House. The floor of this building was wet or covered with silt. Additionally, the motor-driven fire pump controller was wet but not damaged. Water was also found in an unused Unit 2 motor control center.

Condenser vacuum was lost following the shutdown of the SW system. This required closure of the MSIVs and the use of the safety relief valves (SRVs) for reactor pressure control. The RCIC was placed in service for reactor makeup, and both trains of the RHR system were started at 1552 hours for suppression pool cooling. At 2014 hours, shutdown cooling (SDC) was established using RHR train A. RHR train B continued to provide SPC for an additional 5 h. RCIC was secured and the CRD system was used for level control. The A CRD pump experienced minor cavitation due to loss of suction. The unit reached cold shutdown at 2210 hours.

On April 14, 1993, all emergency core cooling systems (ECCS) strainers were inspected using a high-powered light and video camera. The RHR train B strainer was fouled and deformed in a manner similar to that observed during the January inspection. The remaining strainers showed no signs of fouling. Without disturbing the debris on the strainer, a test run of RHR pump B was performed. The pump running suction pressure decreased to 0 psig after operating for 8 h, and the pump was secured.

The pump suction strainer was then inspected. The debris from the strainer was analyzed, and it was determined that the debris contained fibrous material and corrosion products. The predominant fibrous material was glass fiber from roughing filter material used in the drywell air cooler system. The RHR strainer provided a structural framework for a uniform covering of the fibrous material, which in turn acted as a filter for suspended solids that would have otherwise passed through the strainer.

The licensee inspected and cleaned the containment following the discovery of the clogged strainers and did not identify large quantities of the fibrous material. Based on this, the licensee concluded that there was no chronic degradation of properly installed filter media. Instead, the licensee concluded that the fibrous material entered the suppression pool as intact pieces as a result of installation or maintenance activities (the roughing filters are normally replaced prior to startup from refueling outages). These pieces subsequently broke down to fibers once in the suppression pool. The actual time the material entered the suppression pool could not be determined.

The suppression pool was completely inspected and cleaned following the discovery of the clogged strainers. This was the first thorough inspection and cleaning since initial criticality in 1986. Previous inspection and cleanup efforts were limited to easily visible and accessible pool areas.

Additional information concerning this event are included in NRC Bulletin 93-02, Supplement 1, *Debris Plugging of Emergency Core Cooling Suction Strainers*, February 18, 1994, and Augmented Inspection Team (AIT) report 50-440/93006(DRS), *Perry Unit 1 Service Water Pipe Break*, April 15, 1993.

A.13.3 Additional Event-Related Information

Systems available at Perry for reactor vessel high-pressure makeup include RCIC, HPCS, and the CRD pumps, as well as main feedwater (MFW). In the event that these systems are unavailable, the automatic depressurization system (ADS) is used to depressurize the reactor to the point where low-pressure systems can provide makeup. Low-pressure systems include low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI).

Two of the three LPCI trains include heat exchangers and piping to remove heat from the suppression pool (suppression pool cooling mode of RHR [RHR/SPC]) and directly from the core (shutdown cooling mode of RHR [RHR/SDC]). The strainers that were found clogged during this event were associated with the two LPCI trains that can be used for RHR.

In the event that RHR fails, the containment can be vented to remove decay heat and prevent overpressurization. To achieve this, the operator manually vents the suppression pool or the drywell. The steaming that will occur in the suppression pool may fail any injection source (such as LPCI) that draws from the suppression pool. Therefore, the feed operation associated with venting must come from an injection system that operates at low pressure and whose source of water is other than the suppression pool.

Flooding of the auxiliary building 568-ft basement level will not directly affect major ECCS components, as each of the RHR, HPCS, LPCS, and RCIC pumps are located in a separate room on the 574-ft level and protected by a watertight door. However, the local panels for all these pumps are mounted in the basement corridor (20 in. above the floor, based on information in the AIT report) except for the HPCS panel, which is at the 574-ft level. Flooding of the corridor will fail the ECCS pumps once water reaches the local panels. Flooding will also lead to loss of the ADS permissive; however, this can be bypassed by the operator in the control room.

Flooding of the control complex 574-ft elevation will result in loss of the instrument air compressors (12 in. above the floor), control complex chilled water pumps which provide ventilation cooling for the battery and switchgear rooms and control room (22 in. above the floor), and emergency closed cooling (ECC) system pumps (22 in. above the floor). The ECC system provides cooling water to the RCIC, LPCS, and RHR pump room coolers and to the RHR pump seals, as well as to the control complex chillers. Although flooding did not reach 12 in. above the 574-ft elevation, instrument air was lost during the event.

A.13.4 Modeling Assumptions

Excessive differential pressure across the RHR strainers from debris accumulation would fail SPC and could fail LPCI if it was required to operate for long periods of time. The event was modeled as an unavailability of RHR/SPC following (1) postulated initiators in the 1-year period prior to discovery of the clogged strainers and (2) the reactor trip following the SW pipe rupture on March 26, 1993. The possibility of flooding damage to ECCS components was addressed in a sensitivity analysis.

Case 1. Unavailability of RHR/SPC cooling following postulated initiating events. The potential for plugging the suppression pool strainers existed prior to the May 1992 refueling outage. To estimate the relative significance of the event within a 1-year observation period (the interval between precursor reports), a 1-year observation period was used in the analysis (6132 hours, assuming the plant was critical or at hot shutdown 70% of the time). Based on the strainer deformation and clogging observed in 1992 and 1993, both trains of RHR/SPC were assumed to be failed and not recoverable for long-term decay heat removal. LPCI injection and short-term SPC prior to initiation of RHR/SDC were assumed to be operable (RHR train B suction pressure decreased to 0 psig after 17 h of operation following the SW flood). The unavailability of RHR/SPC affected sequences on each of the three ASP models: transient, loss-of-offsite power, and small-break loss of coolant accident. The reactor trip frequency utilized in the transient model was not reduced to reflect the trip following the SW pipe rupture analyzed in Case 2. A nominal reactor trip frequency was used in the analysis.

The existing ASP model was modified to include the potential use of containment venting for decay heat removal in the event that both RHR/SPC and RHR/SDC fail. This was done by revising the dominant sequences involving failure of both RHR cooling modes to also include failure to vent the containment. The probability of failing to vent was assumed to be dominated by human error. A probability of 0.01 was utilized for sequences in which the source of water for injection is separate from the suppression pool.

For sequences in which the injection source takes suction from the suppression pool (such as LPCS or LPCI), an alternate injection source, the CRD pumps or essential SW (RHRSW in the ASP models), must be aligned for injection following venting. Venting is considered much less reliable in such cases; an operator error probability of 0.5 was utilized (see NRR Daily Events Evaluation Manual, 1-275-03-336-01, January 31, 1992).

The current ASP models do not address the potential use of RCIC for reactor vessel (RV) injection in the event of a failed-open SRV. Thermal-hydraulic analyses performed in support of a number of contemporary probabilistic risk assessments indicate that RCIC can provide injection success provided only one SRV fails open. The conditional probabilities for sequences involving failed-open relief valves were revised to reflect the probability that RCIC must also fail or two or more SRVs must fail open before high-pressure RPV makeup fails. This probability was estimated as:

$$p(\text{RCIC}) + p(\text{2 or more SRVs fail open} \mid \text{1 or more SRVs fail open}).$$

This approximation assumes that sequences involving RCIC success avoid core damage if RHR is also successful. Since the probability of RHR failure is very small relative to the probability of failing RCIC, this approximation is valid. The failure probability for RCIC during this event was estimated at 0.042. A value of 0.024 was estimated for p(2 or more SRVs fail open | 1 or more SRVs fail open), based on an estimated probability for two or more SRVs failing open of 0.0015 (see NUREG/CR-4550, Vol. 1, Rev. 1, *Analysis of Core Damage Frequency: Internal Events Methodology*, January 1990, pp. 6-10) and an estimated probability of one or more SRVs failing open of 0.0627 (this is developed in Appendix C of NUREG/CR-4674, Vol 1, *Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report*, December 1986). The estimated probability of one or more SRVs failing open is dependent on the number of valves at a given plant and the probability of an SRV failing to close per demand. The probability of RCIC failure or more than one SRV failed open is then 0.042 + 0.024 = 0.066. RCIC can also provide makeup following a steam-side, small-break, loss-of-coolant accident (LOCA). Consistent with other ASP analyses, the probability of a steam-side LOCA was assumed to be 0.6. The probability of RCIC failing to provide RPV makeup following a small break LOCA is, therefore, (1-0.6) + 0.042 = 0.442.

<u>Case 2. Reactor trip, effective loss of MFW, CRD pump problems, and unavailability of RHR/SPC</u>. Following the reactor trip and SW system shutdown, condenser vacuum was lost and the MSIVs were closed. This resulted in unavailability of the power conversion system (PCS) for decay heat removal and the MFW and condensate systems for RV makeup. The CRD system was used for makeup after RCIC was secured; CRD pump A cavitated due to loss of suction. Because of the cavitation problems with the A pump, the CRD system was assumed to be unavailable for RV makeup in the short term (two-of-two CRD pumps are required for success) had it been needed in the event of failure of HPCS and RCIC. In addition, long-term RHR/SPC was also unavailable, as described in Case 1.

Analysis assumptions concerning the potential use of RCIC following a failed open SRV and containment venting were the same as for Case 1. Although CRD flow for short-term RV makeup was assumed unavailable because of cavitation problems with the pump A, CRD was assumed available for makeup following venting. One-of-two pumps provides success in this situation, since the decay heat load is lower.

If SW had not been secured, continued flooding of the auxiliary building and control complex could have resulted in damage to ECCS components. As described in Additional Event-Related Information, the LPCS, RHR, RCIC, and ECC system pumps would have been impacted had the water level reached 20-22 in. in these buildings (flood levels reached 5 in. during the actual event). The lack of detailed information concerning equipment locations and flood pathways prevents consideration of potential flooding effects in this analysis (operational events involving flooding are normally considered impractical to analyze in the ASP program because there is a lack of detailed information). However, a sensitivity analysis was performed to bound the potential effects of the flood.

The sensitivity analysis considered, in addition to the system unavailabilities described in Case 2, the unavailability of the RHR (LPCI and RHR/SDC as well as SPC already lost because of the suction strainer problems) and RCIC pumps if flooding reached 20-22 in. in the auxiliary building and control complex. To simplify the sensitivity analysis, these pumps were assumed unavailable and not recoverable if flooding reached this height. Based on information from the licensee, the LPCS pump was assumed to remain operable, although its room cooling would have been unavailable following the loss of the ECC pumps (this assumption has little affect on the sensitivity analysis results).

The probability of failing to secure SW prior to release of sufficient water to impact the RHR and RCIC pumps was estimated using the following assumptions:

- The rate of auxiliary building and control complex flooding was constant and therefore the time required before sufficient SW was released to reach 20-22 in. was approximately four times the actual flood duration. This assumption is subject to large uncertainties since details of the flooding pathways are not known.
- The compelling cue for SW shutdown was the observation of significant flooding of plant buildings at 1535 h, 14 min after the increase in break flowrate. The SW system was shut down 5 min later. Based on these times and the fact that water levels reached one-quarter of the height

required for damage, break flow must be terminated ~62 min following the cue to prevent damage to ECCS pump control panels in the auxiliary building basement corridor. Control panel flooding would fail RCIC. Damage to the ECC pumps in the control complex, which would impact RHR pump seal cooling and ECCS pump room cooling, would shortly follow.

- The observed time to secure SW (5 min) was assumed to be the median of a lognormal distribution with an error factor of 3.2 (see Dougherty and Fragola, *Human Reliability Analysis*, John Wiley and Sons, New York, 1988, Chapter 10). This is the error factor for time-reliability correlations (TRCs) for actions without hesitancy, which is considered appropriate based on the nature of the flood and the fact that the SW system is not safety-related at Perry. The resulting probability of failing to secure SW before RHR and RCIC pump impact is 1.9×10^{-4}.

During the actual event, the HPCS pump motor was wetted by water dripping from a ceiling hatch plug; however, the pump was not damaged. A separate sensitivity analysis was performed assuming the HPCS pump was unavailable and not recoverable during the actual event and during postulated flooding to understand the impact of such potential damage.

Five core damage probability calculation sheets document the analysis. Case 1 addresses unavailability of RHR/SPC for a 1-year period. Case 2 addresses the reactor trip, loss of condenser vacuum, and CRD problems following the SW pipe rupture. The conditional core damage probability for the event was estimated by modifying the sequence conditional probabilities to reflect the potential use of RCIC in the event of a single failed-open SRV and the use of containment venting for long-term decay heat removal (indicated in the notes at the end of each calculation sheet) and summing the conditional probabilities for the two cases. The three calculation sheets for the potential flooding-impacts and HPCS-unavailable sensitivity analyses are also included.

A.13.5 Analysis Results

The conditional core damage probability estimated for this event is 1.2×10^{-4}. The dominant core damage sequence, highlighted on the event tree shown in Fig. A.13.2, involves a scram with PCS and FW unavailable following the SW pipe rupture, HPCS success, failure of long-term decay heat removal via the RHR system, and failure to vent the containment.

The results of the sensitivity analysis to address potential flooding effects indicates a core damage probability of 2.5×10^{-6} given the rupture. This is small compared to the overall core damage probability for the event, indicating that potential flooding effects do not significantly contribute to the overall event, based on information available in the LER and AIT report. The flood is interesting, however, since it impacted multiple buildings that would typically be considered independent structures in an internal flooding risk analysis.

If the HPCS pump motor had been damaged by the water that dripped from the ceiling hatch, the estimated core damage probability would be 8.5×10^{-4} (including the suppression pool strainer unavailability), a much more significant event.

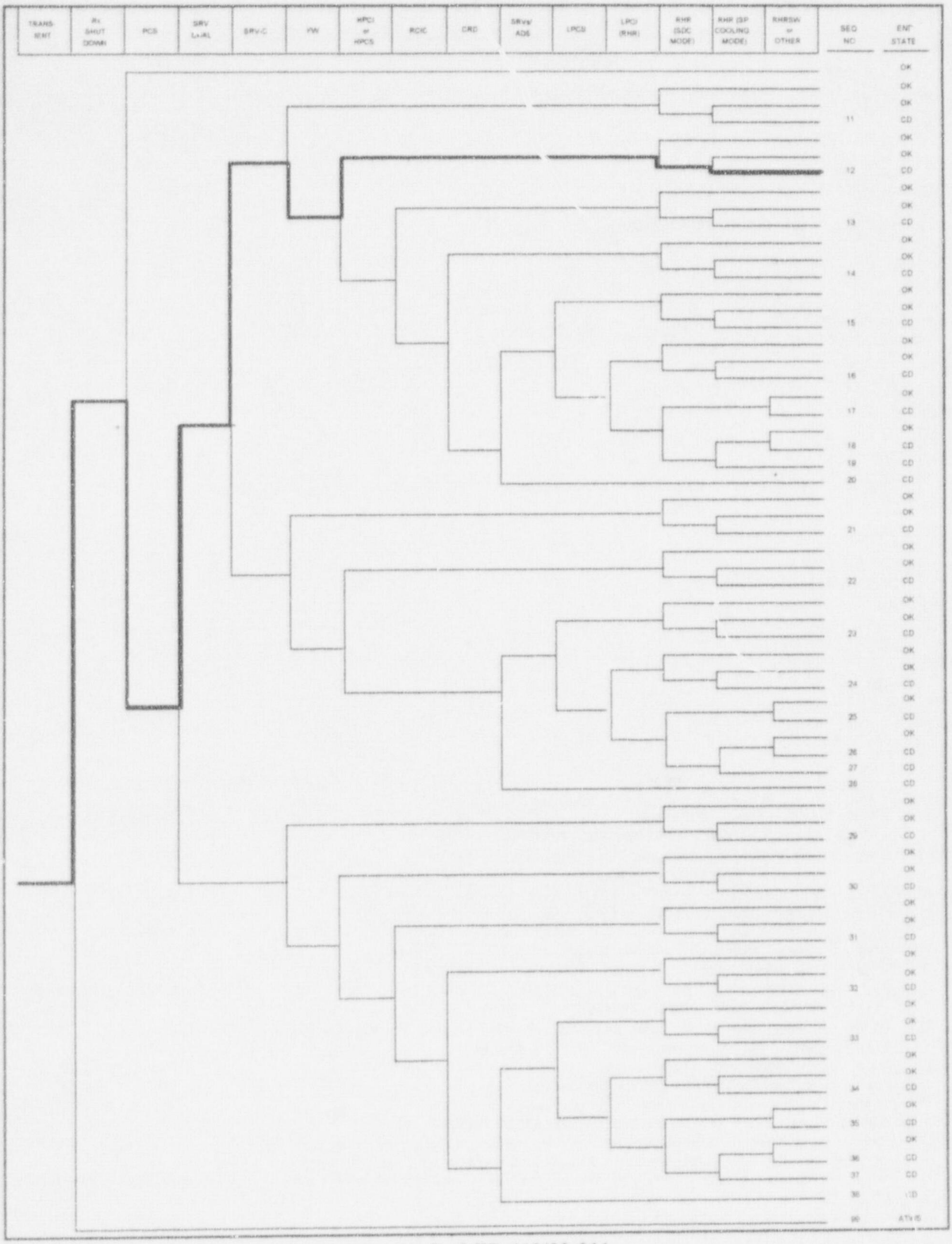


Fig. A.13.2 Dominant core damage sequence for LER 440/93-011

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 440/93-011
Event Description: Unavailability of RHR suppression pool cooling (case 1)
Event Date: 03/26/93
Plant: Perry 1

UNAVAILABILITY, DURATION= 6132

NONRECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	7.4E+00
LOOP	5.3E-02
LOCA	1.0E-02

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	2.0E-03(1)
LOOP	4.4E-04(1)
LOCA	8.8E-05(1)
Total	2.6E-03(1)
ATWS	
TRANS	0.0E+00
LOOP	0.0E+00
LOCA	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
11	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close -fw/pcs.trans rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.7E-03(1)	3.2E-01
40	loop -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	4.1E-04(1)	1.8E-01
12	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close fw/pcs.trans -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.8E-04(1)	.2E-01
21	trans -rx.shutdown pcs/trans srv.chall/trans.-scram srv.close -fw/pcs.trans rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.1E-04(1)	3.2E-01
71	loca -rx.shutdown -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	8.7E-05(1)	1.7E-01
49	loop -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	2.8E-05(1)	1.8E-01
22	trans -rx.shutdown pcs/trans srv.chall/trans.-scram srv.close fw/pcs.trans -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-05(1)	1.2E-01
41	loop -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	2.7E-06(1)	6.0E-02
13	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close fw/pcs.trans hpci rcic rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-06(1)	3.9E-02
65	loop emerg.power -rx.shutdown/ep -ep.rec srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci RHR(SPCOOL)/-LPCI.RHR(SDC)	CD	6.9E-07(1)	1.4E-01
72	loca -rx.shutdown hpci -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	5.9E-07(1)	5.7E-02
50	loop -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close hpci -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.8E-07(1)	6.1E-02

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
11	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close -fw/pcs.trans rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.7E-03(1)	3.2E-01
12	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close fw/pcs.trans -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.8E-04(1)	1.2E-01
13	trans -rx.shutdown pcs/trans srv.chall/trans.-scram -srv.close fw/pcs.trans hpci -rcic rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-06(1)	3.9E-02
21	trans -rx.shutdown pcs/trans srv.chall/trans.-scram srv.close -fw/pcs.trans rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.1E-04(1)	3.2E-01
22	trans -rx.shutdown pcs/trans srv.chall/trans.-scram srv.close fw/pcs.trans -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-05(1)	1.2E-01
40	loop -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	4.1E-04(1)	1.8E-01
41	loop -emerg.power -rx.shutdown srv.chall/loop.-scram -srv.close hpci -rcic rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	2.7E-06(1)	6.0E-02
49	loop -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	2.8E-05(1)	1.8E-01
50	loop -emerg.power -rx.shutdown srv.chall/loop.-scram srv.close hpci -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	1.8E-07(1)	6.1E-02
65	loop emerg.power -rx.shutdown/ep -ep.rec srv.chall/loop.-scram -srv.close -hpci rhr(sdc)/-lpci RHR(SPCOOL)/-LPCI.RHR(SDC)	CD	6.9E-07(1)	1.4E-01
71	loca -rx.shutdown -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	8.7E-05(1)	1.7E-01
72	loca -rx.shutdown hpci -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RH R(SDC)	CD	5.9E-07(1)	5.7E-02

** nonrecovery 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: s:\asp\prog\models\bwrcsesl.cmp
BRANCH MODEL: s:\asp\prog\models\perry.sl1
PROBABILITY FILE: s:\asp\prog\models\bwr_csl1.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	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
pcs/trans	2.3E-01	1.0E+00	
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	6.3E-02	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
ep.rec	1.7E-01	1.0E+00	
fw/pcs.trans	2.8E-01	3.4E-01	
fw/pcs.loca	1.0E+00	3.4E-01	
hpci	2.0E-02	3.4E-01	
rcic	6.0E-02	7.0E-01	
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	2.0E-02	3.4E-01	
lpci(rhr)/lpcs	6.0E-04	7.1E-01	
rhr(sdc)	2.3E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
RHR(SPCOOL)/RHR(SDC)	2.0E-03 > 1.0E+00	3.4E-01 > 1.0E+00(2)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-03 > Failed(2)		
RHR(SPCOOL)/-LPCI.RHR(SDC)	2.0E-03 > 1.0E+00	3.4E-01 > 1.0E+00(2)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-03 > Failed(2)		
RHR(SPCOOL)/LPCI.RHR(SDC)	9.3E-02 > 1.0E+00	1.0E+00	

Branch Model: 1.OF.1
Train 1 Cond Prob: 9.3E-02 > Failed(2)

rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file
** forced

Notes
1. Revised core damage probabilities reflecting the potential use of RCIC in the event of a single failed-open relief valve and containment venting for long-term decay heat removal.

Sequence	p(RCIC)	p(vent)	p(sequence)
11	n/a	0.01	1.7E-05
40	n/a	0.01	4.1E-06
12	n/a	0.01	1.8E-06
21	n/a	0.01	1.1E-06
71	n/a	0.01	8.7E-07
49	n/a	0.01	2.8E-07
22	n/a	0.01	1.2E-07
41	n/a	0.01	2.7E-08
13	n/a	0.01	1.2E-08
65	n/a	0.01	6.9E-09
72	0.466	0.5	1.4E-07
50	0.066	0.5	5.9E-09
		Total	2.5E-05

2. Nonrecoverable failure of long-term suppression pool cooling.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Description: SW break with effective LOFW and CRD problems (case 2)
Event Date: 03/26/93
Plant: Perry 1

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	8.8E-03(1)
Total	8.8E-03(1)
ATWS	
TRANS	3.0E-05
Total	3.0E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
12	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	8.2E-03(1)	3.4E-01
22	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	5.5E-04(1)	3.4E-01
13	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci -rcic rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	5.4E-05(1)	1.1E-01
28	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci srv.ads	CD	5.4E-06(1)	2.4E-01
23	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	3.7E-06(1)	1.1E-01
20	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci rcic CRD srv.ads	CD	3.4E-06(1)	1.7E-01
15	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci rcic CRD -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	2.3E-06(1)	8.0E-02
99	trans rx.shutdown	ATWS	3.0E-05	1.0E+00

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
12	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	8.2E-03(1)	3.4E-01
13	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci -rcic rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	5.4E-05(1)	1.1E-01
15	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci rcic CRD -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	2.3E-06(1)	8.0E-02
20	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci rcic CRD srv.ads	CD	3.4E-06(1)	1.7E-01
22	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS -hpci rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	5.5E-04(1)	3.4E-01

23	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci -srv.ads -lpcs rhr(sdc) RHR(SPCOOL)/RHR(SDC)	CD	3.7E-06(1)	1.1E-01
28	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci srv.ads	CD	5.4E-06(1)	2.4E-01
99	trans rx.shutdown	ATWS	3.0E-05	1.0E+00

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\bwrcseal.cmp
BRANCH MODEL: s:\asp\prog\models\perry.sl1
PROBABILITY FILE: s:\asp\prog\models\bwr_csl1.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	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
PCS/TRANS	2.3E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.3E-01 > Unavailable(3)		
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	6.3E-02	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
ep.rec	1.7E-01	1.0E+00	
FW/PCS.TRANS	2.8E-01 > 1.0E+00	3.4E-01 > 1.0E+00(3)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.8E-01 > Unavailable(3)		
FW/PCS.LOCA	1.0E+00 > 1.0E+00	3.4E-01 > 1.0E+00(3)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.0E+00		
hpci	2.0E-02	3.4E-01	
rcic	6.0E-02	7.0E-01	
CRD	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 > Failed(4)		
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	2.0E-02	3.4E-01	
lpci(rhr)/lpcs	6.0E-04	7.1E-01	
rhr(sdc)	2.3E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
RHR(SPCOOL)/RHR(SDC)	2.0E-03 > 1.0E+00	3.4E-01 > 1.0E+00(2)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-03 > Failed(2)		
RHR(SPCOOL)/-LPCI.RHR(SDC)	2.0E-03 > 1.0E+00	3.4E-01 > 1.0E+00(2)	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.0E-03 > Failed(2)		
RHR(SPCOOL)/LPCI.RHR(SDC)	9.3E-02 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	9.3E-02 > Failed(2)		
rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file
** forced

Notes

1. Revised core damage probabilities reflecting the potential use of RCIC in the event of a single failed-open relief valve and containment venting for long-term decay heat removal.

Sequence	p(RCIC)	p(vent)	p(sequence)
12	n/a	0.01	8.2E-05
22	n/a	0.01	5.5E-06
13	n/a	0.01	5.4E-07
28	0.066	1.0	3.6E-07
23	0.066	0.5	1.2E-07
20	n/a	1.0	3.4E-06
15	n/a	0.5	1.2E-06
		Total	9.3E-05

2. Nonrecoverable failure of long-term suppression pool cooling.
3. These unavailabilities result from the loss of condenser vacuum and MSIV closure.
4. CRD pump "A" cavitation.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 440/93-011
Event Description: SW break with LOFW, CRD problems and flood impacts (sensitivity)
Event Date: 03/26/93
Plant: Perry 1

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.9E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	1.9E-04(1)
Total	1.9E-04(1)
ATWS	
TRANS	5.7E-09
Total	5.7E-09

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
12	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS -hpci RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	1.8E-04(1)	1.9E-04
22	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS -hpci RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-05(1)	1.9E-04
15	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci RCIC CRD -srv.ads -lpcs RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-06(1)	6.4E-05
23	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci -srv.ads -lpcs RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	7.9E-08(1)	6.4E-05
99	TRANS rx.shutdown	ATWS	5.7E-09	1.9E-04

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
12	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS -hpci RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	1.8E-04(1)	1.9E-04
15	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS hpci RCIC CRD -srv.ads -lpcs RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-06(1)	6.4E-05
22	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS -hpci RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	1.2E-05(1)	1.9E-04
23	TRANS -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci -srv.ads -lpcs RHR(SDC) RHR(SPCOOL)/RHR(SDC)	CD	7.9E-08(1)	6.4E-05
99	TRANS rx.shutdown	ATWS	5.7E-09	1.9E-04

** nonrecovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\bwrcseal.cmp

A.14 LER Nos. 498/93-005 and -007

Event Description: Unavailability of One Emergency Diesel Generator and the Turbine-Driven Auxiliary Feedwater Pump

Date of Event: December 29, 1992, through January 22, 1993

Plant: South Texas Project, Unit 1

A.14.1 Summary

For a period of ~ 25 d, South Texas Project (STP) Unit 1 operated with one emergency diesel generator (EDG) and the turbine driven auxiliary feedwater (TDAFW) pump inoperable. The EDG was rendered inoperable because of binding of the fuel metering rods. The TDAFW pump was inoperable because of water intrusion into the turbine, which would have prevented the automatic start of the TDAFW pump. During this same period, a second EDG was removed from service for maintenance for a period of 61 h. The conditional core damage probability for this event is 1.2×10^{-5}. The relative significance of this event compared to other postulated events at STP Unit 1 is shown in Fig. A.14.1.

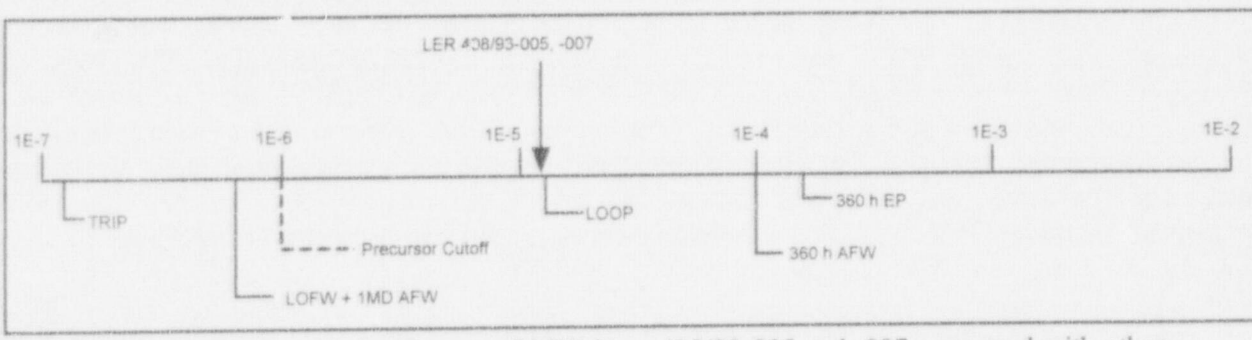


Fig. A.14.1 Relative event significance of LER Nos. 498/93-005 and -007 compared with other potential events at South Texas Project 1

A.14.2 Event Description

STP Unit 1 was operating at 95% power on January 20, 1993, when EDG 13 failed to start during a monthly surveillance test. The EDG had been painted during a 3-d period beginning December 29, 1992. Paint applied to the fuel injection pumps ran into the fuel metering ports, which caused the binding of the fuel metering rods. An operability test of the EDG was not performed after the completion of the painting. Following repair of the EDG, it was returned to service on January 22, 1993, ~ 25 d after it initially had been rendered inoperable. During the time period that EDG 13 was inoperable, EDG 12 had also been removed from service for 61 h.

The TDAFW pump was also inoperable for the 25-d period that EDG 13 was inoperable. During the fourth refueling outage, the TDAFW turbine trip/throttle valve, which had been leaking before the outage, was disassembled for repair. Although the disc and stem had steam cuts, no replacement parts could be located, so the valve was reassembled and returned to service. On December 27, 1992, the day after the end of the fourth refueling outage, the pump was tested as part of a post-maintenance test. The turbine oversped and tripped. The pump was then successfully slow started and was declared operable. Two other slow starts were successfully performed on December 31, 1992, using the anticipated transient-without-scram (ATWS) mitigation system actuation circuitry (AMSAC). The next test of the TDAFW pump was on January 28,

1993. Following maintenance, the turbine tripped during a fast start. The next day the turbine tripped during a slow start. Following repairs to the governor and a number of successful starts, the pump was returned to service on January 30, 1993. On February 1, 1993, the TDAFW pump again failed its surveillance test. Two days later the TDAFW turbine for Unit 2 oversped and tripped following a Unit 2 plant trip. A review of the maintenance history on the Unit 2 TDAFW pump also revealed problems with the overspeed trip device that rendered the pump inoperable for 4 d. This led to the decision to shut down Unit 1. The cause of the TDAFW pump overspeed events at both units was water intrusion into the turbine.

A.14.3 Additional Event-Related Information

The STP units utilize a three-train safety system arrangement. Any train of equipment is sufficient to accomplish safe shutdown of a unit for most design basis accidents. For each unit, there are three EDGs, each supplying a separate and independent load group. The AFW scheme consists of four pumps. Three of the pumps are motor driven and are supplied by their associated safeguards bus and EDG. The fourth pump is a turbine-driven pump using steam from the steam generators to provide its motive force. All four of the AFW pumps are 100% capacity pumps.

A.14.4 Modeling Assumptions

This event is modeled as a potential loss-off-offsite power (LOOP) event from December 29, 1992, to January 22, 1993. EDG 13 was assumed to be inoperable from the time that painting was begun on December 29, 1992, until the EDG was returned to service on January 22, 1993 (a total of 597 h). EDG 12 was inoperable for 61 h during this period. When an EDG is inoperable, the equipment associated with that EDG is also inoperable during a potential LOOP event where offsite power is not recovered. The model was revised to reflect this by failing variously one or two trains of equipment dependent on emergency power. The EDG failure was modeled by assuming that the other EDGs were not susceptible to the same failure mode. In this event, the EDG failure from the painting process was discovered before the other EDGs were exposed to the same failure mechanism.

The licensee determined that the TDAFW pump was inoperable from the end of the fourth refueling outage (December 26, 1992) until the plant was shut down on February 3, 1993. This encompasses the time period when the EDGs were inoperable. The recovery value for the AFW system was not changed because the failures related to the TDAFW pump were, for the most part, recoverable by starting the pump after the overspeed was reset. In these cases, the initial start attempt cleared the condensate from the steam admission line and the turbine casing and prewarmed the turbine, increasing the likelihood of successful start on subsequent attempts.

Two cases were run for the Unit 1 unavailabilities. Case 1 was calculated as a LOOP with EDG 13 and the TDAFW pump inoperable (but recoverable) for 536 h (597 - 61 h). An hourly LOOP frequency of 2×10^{-5} was multiplied by 536 h and a short-term nonrecovery probability of 0.43 (see below) to obtain a LOOP frequency for the period of interest of 4.6×10^{-3}. A similar calculation was performed (Case 2) for the 61-h period during which EDGs 12 and 13 and the TDAFW pump were inoperable, and a LOOP frequency of 5.2×10^{-4} was estimated.

Nominal battery life at South Texas is 2 h. But, by shedding unnecessary loads, battery life may be extended to perhaps 8 h. To credit this strategy, each of the two cases was further decomposed to reflect the core damage probability with and without battery load shed. (Cases 1A and 2A use 2-h battery lifetimes, and Cases 1B and 2B use 8-h battery lifetimes.) As the actions involved in load shedding appeared to fall into ASP recovery class R3, a probability of failure to shed battery loads when required of 0.12 was assumed (a description of the ASP recovery classes may be found in Appendix A, NUREG/CR-4674, Vol 17) and a weighted core damage probability was calculated for each case:

$$0.12 \times p(cd \mid \text{2-h battery life}) + (1 - 0.12) \times p(cd \mid \text{8-h battery life}).$$

The LOOP frequency and electric power recovery probabilities for South Texas were calculated according to the methods detailed in ORNL/NRC/LTR-89/11R1, *Revised LOOP Recovery and Seal LOCA Models*, October 1993. For both the 2-h and 8-h battery lifetime cases, a short-term LOOP nonrecovery probability of 0.43 was calculated. The seal LOCA probability was estimated to be 0.31 and the probability of nonrecovery of ac power in the long term given a seal LOCA was estimated to be 0.7. The probability of ac power nonrecovery given that no seal LOCA occurred was estimated to be 0.11 for the 2-h battery lifetime case. For the 8-h battery lifetime case, a probability of nonrecovery of AC power prior to battery depl.tion of 0.012 was calculated.

To credit the use of the positive displacement (PD) charging pump with power supplied by the Technical Support Center diesel, the base seal LOCA probability of 0.31 was multiplied by a nonrecovery value of 0.17 [0.12 (ASP operator nonrecovery class R3) + 0.05 (ASP probability that the EDG fails to start on demand, PD pump failure rate presumed to be small relative to EDG failure rate)] to obtain a reduced seal LOCA probability of 0.05.

The conditional core damage probability is calculated as follows:

$$0.12 \times p(cd \mid \text{2-h battery life}) + (1-0.12) \times p(cd \mid \text{8-h battery life})$$

Case 1

$$(0.12)(4.3 \times 10^{-6}) + (0.88)(3.8 \times 10^{-6}) = 3.9 \times 10^{-6}$$

Case 2

$$(0.12)(9.1 \times 10^{-6}) + (0.88)(7.9 \times 10^{-6}) = 8.0 \times 10^{-6}$$

Total

$$3.9 \times 10^{-6} + 8.0 \times 10^{-6} = 1.2 \times 10^{-5}$$

The Unit 2 plant trip with the subsequent overspeed trip of the TDAFW pump was also modeled. It was modeled as a transient with the TDAFW pump failed, but recoverable. The inoperability of the Unit 2 TDAFW pump was not modeled because it was treated as a loss of redundancy.

A.14.5 Analysis Results

The conditional core damage probability for the time period when just EDG 13 and the TDAFW pump were inoperable (Case 1), weighted to reflect the likelihood and effects of successful battery load shed, is calculated to be 3.9×10^{-6}. Similarly, the conditional core damage probability for the time period when EDGs 12 and 13 and the TDAFW pump were inoperable (Case 2) is 8.0×10^{-6}. Therefore, the total conditional core damage probability for the event is 1.2×10^{-5}. The dominant core damage sequence for both cases involves a postulated LOOP with failure of emergency power and AFW, and is highlighted on the event tree shown in Fig. A.14.2.

The modeling of the Unit 2 transient resulted in a value $< 1 \times 10^{-6}$. This is below the cutoff value for events in the ASP Program. Therefore this event is not a precursor. Additional information concerning this event is included in Augmented Inspection Team report 50-498/93-07; 50-499/93-07.

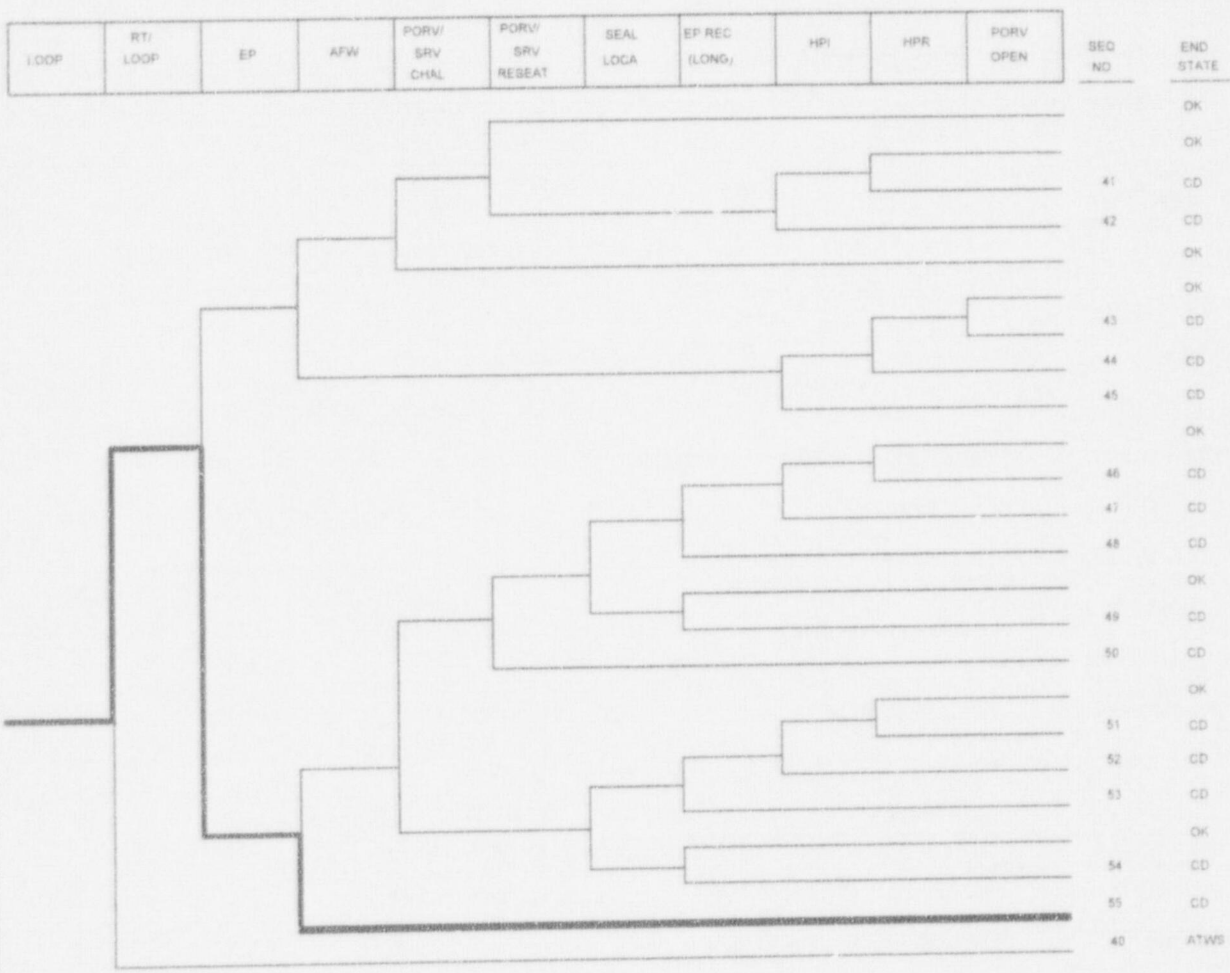


Fig. A.14.2 Dominant core damage sequence for LER 498/93-005

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 498/93-005 and 498/93-007
Event Description: EDG and AFW Pump Unavailability (Case 1A)
Event Date: 12/29/92 - 1/22/93
Case: EDG 13 and TDAFW Pump inoperable; 2 hr battery lifetime
Plant: South Texas 1

UNAVAILABILITY, DURATION= 536

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	4.6E-03

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	4.3E-06
Total	4.3E-06
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	3.5E-06	1.2E-01
54	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall -seal.loca ep.rec	CD	5.0E-07	2.3E-01
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	1.7E-07	2.3E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	1.7E-07	2.3E-01
54	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall -seal.loca ep.rec	CD	5.0E-07	2.3E-01
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	3.5E-06	1.2E-01

** nonrecovery 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: s:\asp\prog\models\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\southtex.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro
No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	6.4E-04	1.0E+00	
loop	2.0E-05	4.3E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	

EMERG.POWER	5.4E-04 > 2.9E-03	8.0E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02		
Train 3 Cond Prob:	1.9E-01 > Failed		
AFW	3.1E-04 > 2.3E-03	2.6E-01	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Failed		
Train 4 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
AFW/EMERG.POWER	5.0E-02 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.0E-02 > Failed		
mfw	1.0E+00	7.0E-02	1.0E-03
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	2.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	3.1E-01 > 5.0E-02(1)	1.0E+00	
ep.rec(sl)	7.0E-01	1.0E+00	
ep.rec	1.1E-01	1.0E+00	
HPI	3.0E-04 > 1.0E-03	8.4E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Failed		
HPI(F/B)	3.0E-04 > 1.0E-03	8.4E-01	1.0E-02
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Failed		
KPR/-HPI	1.5E-05 > 1.5E-04	1.0E+00	1.0E-03
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.5E-02		
Train 3 Cond Prob:	1.0E-01 > Failed		
porv.open	1.0E-02	1.0E+00	4.0E-04

* branch model file
** forced

NOTES
(1)Includes positive displacement pump

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

```
Event Identifier:  498/93-005 and 498/93-007
Event Description: EDG and AFW pump unavailability (Case 1B)
Event Date:        12/29/93 - 1/22/93
Case:              EDG 13 and TDAFW pump inoperable; 8 hr battery lifetime
Plant:             South Texas 1
```

UNAVAILABILITY, DURATION= 536

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	4.6E-03

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	3.8E-06
Total	3.8E-06
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	3.5E-06	1.2E-01
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	1.7E-07	2.3E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	1.7E-07	2.3E-01
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	3.5E-06	1.2E-01

** nonrecovery 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:      s:\asp\prog\models\pwrbseal.cmp
BRANCH MODEL:        s:\asp\prog\models\southtex.sl1
PROBABILITY FILE:    s:\asp\prog\models\pwr_bsl1.pro
No Recovery Limit
```

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Nonrecov	Opr Fail
trans	6.4E-04	1.0E+00	
loop	2.0E-05	4.3E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	

EMERG.POWER	5.4E-04 > 2.9E-03	8.0E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02		
Train 3 Cond Prob:	1.9E-01 > Failed		
AFW	3.1E-04 > 2.3E-03	2.6E-01	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Failed		
Train 4 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
AFW/EMERG.POWER	5.0E-02 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.0E-02 > Failed		
mfw	1.0E+00	7.0E-02	1.0E-03
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	2.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	3.1E-01 > 5.0E-02(1)	1.0E+00	
ep.rec(sl)	7.0E-01	1.0E+00	
ep.rec	1.1E-01 > 1.2E-02(2)	1.0E+00	
HPI	3.0E-04 > 1.0E-03	8.4E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Failed		
HPI(F/B)	3.0E-04 > 1.0E-03	8.4E-01	1.0E-02
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Failed		
HPR/-HPI	1.5E-05 > 1.5E-04	1.0E+00	1.0E-03
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.5E-02		
Train 3 Cond Prob:	1.0E-01 > Failed		
porv.open	1.0E-02	1.0E+00	4.0E-04

* branch model file
** forced

NOTES
(1)Includes positive displacement pump
(2)Reflects 8 hr battery lifetime

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 498/93-005 and 498/93-007
Event Description: EDG and AFW Pump Unavailabilities (Case 2A)
Event Date: 12/29/92 - 1/22/93
Case: EDGs 12 and 13 and TDAFW Pump inoperable; 2 hr battery lifetime
Plant: South Texas 1

UNAVAILABILITY, DURATION= 61

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	5.2E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	9.1E-06
Total	9.1E-06
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	7.1E-06	1.2E-01
54	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall -seal.loca ep.rec	CD	1.4E-06	2.3E-01
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	4.6E-07	2.3E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	4.6E-07	2.3E-01
54	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall -seal.loca ep.rec	CD	1.4E-06	2.3E-01
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	7.1E-06	1.2E-01

** nonrecovery 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: s:\asp\prog\models\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\southtex.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro
No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	6.4E-04	1.0E+00	
loop	2.0E-05	4.3E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	

EMERG.POWER	5.4E-04 > 5.0E-02	8.0E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Prob:	5.7E-02 > Failed		
Train 3 Cond Prob:	1.9E-01 > Failed		
AFW	3.1E-04 > 2.0E-02	2.6E-01	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01 > Failed		
Train 4 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
AFW/EMERG.POWER	5.0E-02 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.0E-02 > Failed		
mfw	1.0E+00	7.0E-02	1.0E-03
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	2.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	3.1E-01 > 5.0E-02(1)	1.0E+00	
ep.rec(sl)	7.0E-01	1.0E+00	
ep.rec	1.1E-01	1.0E+00	
HPI	3.0E-04 > 1.0E-02	8.4E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01 > Failed		
HPI(F/B)	3.0E-04 > 1.0E-02	8.4E-01	1.0E-02
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01 > Failed		
HPR/-HPI	1.5E-05 > 1.0E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.5E-02 > Failed		
Train 3 Cond Prob:	1.0E-01 > Failed		
porv.open	1.0E-02	1.0E+00	4.0E-04

* branch model file
** forced

NOTES
(1)Includes positive displacement pump

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 498/93-005 and 498/93-007
Event Description: EDG and AFW pump unavailabilities (Case 2B)
Event Date: 12/29/93 - 1/22/93
Case: EDGs 12 and 13 and TDAFW pump inoperable; 8 hr battery lifetime
Plant: South Texas 1

UNAVAILABILITY, DURATION= 61

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	5.2E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	7.9E-06
Total	7.9E-06
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	7.1E-06	1.2E-01
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	4.6E-07	2.3E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
53	loop -rt/loop EMERG.POWER -AFW/EMERG.POWER -porv.or.srv.chall seal.loca ep.rec(sl)	CD	4.6E-07	2.3E-01
55	loop -rt/loop EMERG.POWER AFW/EMERG.POWER	CD	7.1E-06	1.2E-01

** nonrecovery 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: s:\asp\prog\models\pwrbseal.cmp
BRANCH MODEL: s:\asp\prog\models\southtex.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	6.4E-04	1.0E+00	
loop	2.0E-05	4.3E-01	
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
EMERG.POWER	5.4E-04 > 5.0E-02	8.0E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	5.0E-02		

Train 2 Cond Prob:	5.7E-02 > Failed		
Train 3 Cond Prob:	1.9E-01 > Failed		
AFW	3.1E-04 > 2.0E-02	2.6E-01	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01 > Failed		
Train 4 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.8E-04		
AFW/EMERG.POWER	5.0E-02 > 1.0E+00	3.4E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.0E-02 > Failed		
mfw	1.0E+00	7.0E-02	1.0E-03
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	2.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	3.1E-01 > 5.0E-02(1)	1.0E+00	
ep.rec(sl)	7.0E-01	1.0E+00	
ep.rec	1.1E-01 > 1.2E-02(2)	1.0E+00	
HPI	3.0E-04 > 1.0E-02	8.4E-01	
Branch Model: 1.OF.3			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01 > Failed		
HPI(F/B)	3.0E-04 > 1.0E-02	8.4E-01	1.0E-02
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Failed		
Train 3 Cond Prob:	3.0E-01 > Failed		
HPR/-HPI	1.5E-05 > 1.0E-02	1.0E+00	1.0E-03
Branch Model: 1.OF.3+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.5E-02 > Failed		
Train 3 Cond Prob:	1.0E-01 > Failed		
porv.open	1.0E-02	1.0E+00	4.0E-04

* branch model file
** forced

NOTES
(1)Includes positive displacement pump
(2)Reflects 8 hr battery lifetime

A.15 LER No. 529/93-001

Event Description: Steam Generator Tube Rupture

Date of Event: March 14, 1993

Plant: Palo Verde 2

A.15.1 Summary

On March 14, 1993, Palo Verde 2 was at 98% power when a 240-gal/min tube rupture occurred in steam generator (SG) 2. The reactor was manually tripped, and safety injection (SI) plus containment isolation actuated on low-pressurizer pressure. As a result of a defective radiation monitor, high alert and alarm set points on two radiation monitors, isolation of the SG blowdown radiation monitors by the SI actuation, and inadequate procedure implementation, the diagnosis of the tube rupture was delayed for an hour. The ruptured generator was identified and isolated 3 h after the tube rupture occurred, and the unit was placed in cold shutdown. The conditional core damage probability estimated for this event is 4.7×10^{-5}. The relative significance of this event compared to other postulated events at Palo Verde is shown in Fig. A.15.1.

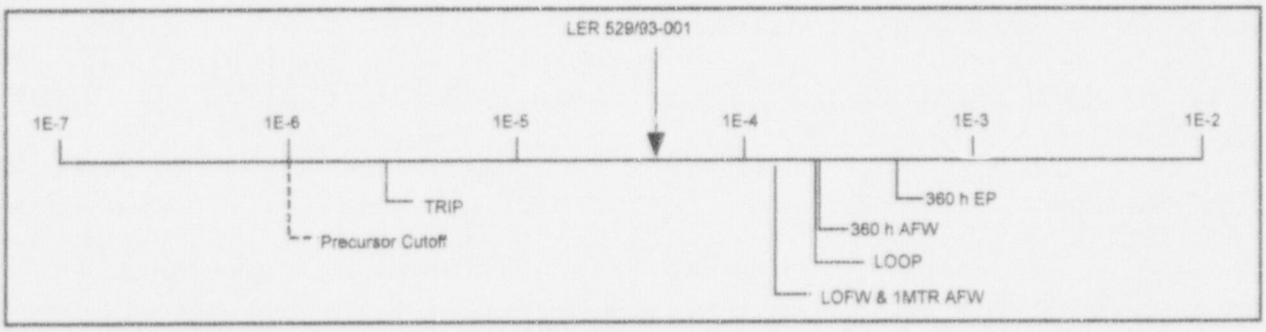


Fig. A.15.1 Relative event significance of LER 529/93-001 compared with other potential events at Palo Verde 2

A.15.2 Event Description

On March 14, 1993, at 0434 hours, Palo Verde 2 was operating at 98% power. A steam generator tube rupture (SGTR) occurred in SG 2. The rupture of the SG tube, caused by intergranular stress corrosion cracking, resulted in a reactor coolant system (RCS) leak rate of ~ 240 gal/min. Indication of SG 2 tube leakage had existed for about a month (the calculated leak rate prior to the rupture was 10 gal/d). SG 2 main steam line radiation monitor, RU-140, alarmed at the time of the rupture. A third charging pump was started, and the backup pressurizer heaters were energized in an attempt to recover pressurizer level and pressure. At 0438 hours (+ 4 min), an alarm was also received on auxiliary steam condensate receiver tank radiation monitor, RU-7.

Earlier in the evening, the gas stripper had been placed in service to degas the RCS in preparation for an upcoming refueling outage. An interfacing system loss-of-coolant accident (LOCA) through the gas stripper was recognized as a potential source of the RCS leakage, as was an SGTR; both of these would result in radiation monitor actuation. No indications existed that the LOCA was inside containment, although the tailpipe temperature on one pressurizer relief valve was high (caused by previously existing leakage). At 0440 hours (+ 6 min), the operators isolated letdown flow in an attempt to stop the leak (a leaking gas stripper would have been isolated by this action). To minimize radiation release to the environment if the leak was an SGTR, steam bypass control valves 1007 and 1008 were removed from service, and the condensate draw-off controller was disabled.

At 0447 h (+ 13 min), pressurizer level had dropped to 26%, and the pressurizer heaters deenergized. The reactor was manually tripped due to low pressurizer level and pressure. Safety injection actuation system (SIAS) and containment isolation actuation system (CIAS) actuations occurred 22 s later due to low pressurizer pressure. The RU-140 alarm cleared shortly after the trip; this was inconsistent with simulator scenarios, where RU-140 alarms late in an SGTR. (It is thought that RU-140 alarmed due to N-16. Because N-16 production ceased once the reactor was tripped, RU-140 cleared at that time.) The two SG blowdown radiation monitors, RU-4 and RU-5, were rendered ineffective when the blowdown lines were isolated by the SIAS signal. RU-4 and RU-5, along with RU-141 (the condenser vacuum exhaust monitor) are the primary indicator alarms for an SGTR. RU-141 was later determined to be reading a factor of 6 low due to a deteriorated scintillation crystal (caused by elevated temperatures from heat tracing; RU-141 had a history of operability problems before the tube rupture). The unavailability of RU-4, RU-5, and RU-141 impacted diagnosis of the SGTR. In addition, the alarm set points for RU-140 and RU-141 were based on not exceeding regulatory dose limits at the site boundary, a high value relative to the expected readings that would indicate an SGTR. This further complicated diagnosis of the event.

Following the reactor trip (RT) and SI, the operators stopped two of the four reactor coolant pumps (RCPs). High-pressure SI (HPSI) restored pressurizer level to ~ 4 to 8% at 0449 hours (+ 15 min). When operator actions to regain control of pressurizer level and pressure were not successful, the control room supervisor (CRS), using the Palo Verde emergency operations procedure diagnostic logic tree (DLT), diagnosed a RT; plant conditions did not allow diagnosis of a more specific recovery procedure. However, the entry conditions for the RT recovery procedure could not be met because pressurizer level was not greater than 10%. The event was rediagnosed; as before, a RT was indicated, but the entry conditions were still not satisfied. At 0502 hours (+ 28 min), the CRS entered the functional recovery procedure (FRP) due to inconclusive diagnosis using the DLT. The diagnosis of an SGTR was not made using the DLT (even though it was suspected) because Palo Verde used a "snap-shot" approach while proceeding through a procedure. Only the plant conditions at the specific time of a procedure step were considered, and not previous alarms or trends (the radiation monitors that had alarmed early in the event had cleared by the time the procedure steps concerning them were encountered).

The FRP directed the operators to align the charging pump suctions directly to the refueling water tank (RWT) and close the volume control tank outlet valve. Charging pump "E" tripped on low-suction pressure. Its suction was aligned to an alternate boration flow path in accordance with the FRP, and the pump was restarted. Postevent analysis concluded that inadequate charging pump suction pressure existed because three charging pumps plus a boric acid pump were taking suction from a common 3-in.-diameter pipe. At 0520 hours (+ 46 min), the operators restored SG blowdown radiation monitors RU-4 and RU-5 as directed by the FRP. These monitors had been isolated by the SIAS signal. RU-5 alarmed 9 min later, and 2 min after that RU-141 reached its alert set point. These signals allowed confirmation of the SGTR.

The CRS continued through the FRP, placing systems in normal shutdown alignments. The licensee stated in the LER that it was the CRS's intent to proceed through the FRP, depressurizing the RCS and using HPSI to restore pressurizer level. Restoration of pressurizer level would allow the FRP to be exited and the DLT to be used to diagnose the SGTR. This was different from the SGTR response strategy in the FRP, where indication of an SGTR is found at step 3.21. When step 3.21 was encountered, the radiation monitors were not alarming (although they had been 5 min earlier), and the SGTR attachment to the FRP was not utilized. At the time of the event, the Palo Verde procedures differed from the Combustion Engineering "Emergency Procedure Guidelines" (CEN-152) in two ways that also complicated diagnosis of the SGTR: (1) radiation alarm indications were used rather than secondary activity trends to aid diagnosis, and (2) a floating step to continuously check for secondary-side activity as an indication of an SGTR did not exist (the FRP checked for secondary activity only once).

At 0604 hours (+ 90 min), an RCS cooldown to 545°F and a depressurization to 1500 psia were begun. HPSI flow increased as the RCS depressurized. Pressurizer level was restored to 33%, RCS temperature and pressure were stabilized, the acceptance criteria for the FRP pressure and inventory control safety function success path were met, and the FRP was exited at 0624 hours (+ 114 min). The DLT was again performed, an SGTR was diagnosed, and the SGTR recovery procedure was entered at 0645 hours (+ 131 min). Palo Verde 2 then performed a crew turnover. At 0721 hours (+ 167 min) the RCS cooldown was restarted in accordance with the SGTR procedure. SG 2 was isolated at 0728 hours, 3 h after the tube rupture occurred. The unit was subsequently placed in cold shutdown. Use of the FRP to mitigate the event, instead of the normal SGTR procedure, resulted in significantly longer times to isolate the ruptured SG and depressurize the RCS. Recovery was delayed and complicated, following the tube rupture, because of poor procedure implementation, inappropriate radiation monitor calibration for the conditions experienced, and a degraded radiation monitor. Further complicating recovery, the qualified safety parameter display system channel "A" core exit thermocouples were reading ~25°F high, causing subcooled margin to be indicated as question marks (inconsistent data).

A.15.3 Additional Event-Related Information

Palo Verde 2 is a two-loop pressurized-water reactor (PWR) manufactured by Combustion Engineering. Each loop includes two RCPs and one U-tube SG. The Palo Verde auxiliary feedwater (AFW) system consists of two safety-related pumps (one motor- and one turbine-driven), plus one nonsafety-related motor-driven pump. Each pump can supply both SGs.

Additional information concerning this event is included in Augmented Inspection Team report No. 50-529/93-14, dated April 15, 1993.

A.15.4 Modeling Assumptions

The event has been modeled as a primary-to-secondary side LOCA (SGTR), with the potential failure to diagnose the SGTR addressed within the model. Because an SGTR is not included within the normal set of ASP models and no SGTR has been previously analyzed for the ASP plant class associated with Palo Verde (PWR Class H), a model specific to the event at Palo Verde was developed. The event tree depicting potential sequences to core damage is shown in Fig. A.15.2. The event tree includes the following branches:

INIT EVENT (SGTR). Initiating event. The initiating event is a primary-to-secondary side break with a flow rate sufficient to require HPSI for RCS makeup.

RT. Reactor trip. Failure to trip results in an anticipated transient without scram (ATWS) sequence and is not developed further.

HPSI. HPSI is required to provide RCS makeup following the break. Flow from one of the two HPSI pumps is required for success. Failure of HPSI requires rapid RCS depressurization and the use of low-pressure safety injection (LPSI) for RCS makeup.

AFW. AFW provides RCS cooling via the SGs. In the event of failure of the three AFW pumps, RCS cooling can be provided using a condensate pump following depressurization of the SGs to < 500 psi using the atmospheric dump valves (ADVs) or turbine bypass valves (TBVs).

RCS DEPRESS AND LPSI. RCS depressurization and LPSI. If HPSI fails, LPSI can provide RCS injection if the RCS is depressurized. This requires AFW flow to both SGs and the use of one-of-two ADVs on each SG or two of the eight TBVs for depressurization. In addition, two of the four SI tanks (SITs) must supply water to the RCS during the cooldown to prevent core uncovery.

SGTR IDENT. SGTR identified. This branch addresses the operator's potential success or failure in identifying the tube rupture. If the tube rupture is successfully identified, as it eventually was in this event, nominal post-SGTR response is modeled. If the operators fail to identify the tube rupture, the event tree addresses two actions that will still provide core protection: RCS depressurization and implementation of shutdown cooling (SDC), or continual HPSI with RWT makeup after ~ 40 h (based on the leak rate observed during the event).

RUPTURED SG ISOL. Ruptured SG isolated. Once the tube rupture is identified, the faulted SG is isolated by closing both main steam isolation valves (MSIVs), the AFW and main feedwater (MFW) injection valves, and the ADVs on the impacted SG. RCS pressure is reduced to below the SG relief valve set point, terminating almost all RCS flow through the break. At this point the tube rupture is considered mitigated. If the ruptured SG is not isolated, the RCS must be depressurized and placed on the SDC mode to terminate flow from the break.

DEPRESS TO SDC. RCS depressurization to the SDC initiation pressure. Either the ADVs or TBVs associated with the intact (nonfaulted) SG must be used, along with pressurizer pressure control, to depressurize the RCS to SDC entry conditions.

SDC. If the RCS is depressurized to SDC entry conditions, then the SDC system can be used to remove decay heat and cool the unit to cold shutdown conditions. Initiation of SDC (one of two LPSI pumps and its associated SDC heat exchanger) provides success.

RWT REFILL. RWT refill. If SDC initiation is unsuccessful, the RCS remains pressurized, and makeup flow must be continually provided. The RWT will have to be eventually refilled to prevent the failure of HPSI. For a break of the size observed during this event, RWT refill must occur ~ 40 h into the event.

In the event of an SGTR, the expected plant response (seen during this event) is shown on the top sequence in Fig. A.15.2. Following the tube rupture, the reactor trips. HPSI provides RCS makeup, and AFW provides core cooling via the SGs. When the ruptured SG is identified, it is isolated, and the good (intact) SG continues to be used for core cooling. Sequences that involve equipment failures or operator errors that can result in core damage are shown in Table A.15.1.

Table A.15.1. Sequence descriptions for SGTR event tree

Sequence	Description
101	Successful RT, HPSI, and AFW following the SGTR. The SGTR is identified, but the ruptured SG is not isolated. SDC fails following RCS depressurization to the SDC initiation pressure. The operators fail to make up to the RWT in the long term.
102	Similar to sequence 101 except RCS depressurization to the SDC initiation pressure fails
103	Successful RT, HPSI, and AFW following the SGTR. The SGTR remains unidentified, although the operators are aware of a LOCA outside containment and initiate RCS depressurization. SDC fails following RCS depressurization, and the operators fail to make up to the RWT in the long term.

Sequence	Description
104	Similar to sequence 103 except RCS depressurization to the SDC initiation point fails
105	Successful RT and HPSI following the SGTR. AFW (including SG depressurization and use of a condensate pump) fails.
106	HPSI failure following successful RT. AFW and the ADVs/TBVs are used to depressurize the RCS to the LPSI initiation pressure. LPSI and the SITs provide RCS makeup (at this point the RCS is at the SDC initiation pressure). SDC and long-term RWT refill fail.
107	Similar to sequence 106 except RCS depressurization or LPSI fails, resulting in a failure of RCS injection
108	Failure of HPSI and AFW following successful RT
109	ATWS sequence (not developed further); failure of RT following the SGTR

Failure probabilities assigned to the event tree branches were developed as follows (see Fig. A.15.2):

INIT EVENT (SGTR). Initiating event (SGTR). An SGTR occurred during the event. Because SGTRs cannot be recovered, a probability of 1.0 was assigned to this branch.

RT. Reactor trip fails. A probability of 3.0×10^{-5} was used, consistent with other ASP analyses.

HPSI. HPSI fails. A probability of 8.4×10^{-4} was used, consistent with other Palo Verde ASP analyses. This value was developed as described in Appendix A, Sect. A.1 of NUREG/CR-4674, Vol. 17, *Precursors to Potential Severe Core Damage Accidents: 1992, A Status Report*.

AFW. AFW fails. For sequences involving HPSI success, a probability of 1.1×10^{-5} was used. This value was developed from the failure probabilities for AFW and SG depressurization and the use of a condensate pump if AFW fails. The development of failure probabilities for AFW, SG depressurization, and condensate are described in Appendix A, Sect. A.1 of NUREG/CR-4674. The overall AFW failure probability follows:

$$p[\text{AFW}|\ \text{-HPSI}] = p[\text{AFW(nominal)}] \times p[\text{SG depress or condensate fails}] =$$

$$p[\text{AFW(nominal)}] \times \{p[\text{SG depress}] + p[\text{MFW}|\ \text{trip}] \times p[\text{condensate}|\ \text{MFW}]\} =$$

$$9.9 \times 10^{-5} \times [3.6 \times 10^{-2} + (0.2 \times 0.35)] = 1.1 \times 10^{-5}\ .$$

For sequences involving HPSI failure, RCS depressurization is addressed in conjunction with LPSI, and p[AFW(nominal)] was used for AFW fails:

$$p[\text{AFW}|\ \text{HPSI}] = 9.9 \times 10^{-5}.$$

RCS DEPRESS AND LPSI. Failure to depressurize the RCS and use the SITs and LPSI for RCS makeup given failure of HPSI. Branch failure will occur if the operators fail to initiate a secondary-side depressurization, if both LPSI trains fail, or if three of the four SITs fail to inject. Thermal-hydraulic calculations performed after the tube rupture indicate that 5 h is an acceptable time for depressurization and use of LPSI following a loss of both trains of HPSI. However, at the time that depressurization would have been required during the event, an SGTR had not been diagnosed. For some small-break LOCAs, depressurization must begin within 15 min.

To address this dichotomy, the branch failure probability was assumed to be dominated by operator failure to initiate the cooldown and depressurization. A failure probability of 0.12 was used (ASP Recovery class R3; see Sect. A.1.3 of Appendix A of NUREG/CR-4674, Vol. 17). An additional factor of 0.34 was then applied to address the potential for recovery from errors made during the initial depressurization.

SGTR IDENT. Operators fail to identify the tube rupture. Because of the problems with the radiation monitors and the event diagnosis using the DLT, the SGTR was not confirmed until 1 h after the event began. If the SGTR had not been identified, the analysis assumed that the operators would have proceeded to place the unit on SDC. Once on SDC, flow from the rupture would have been terminated, although the event would never have been correctly diagnosed. The probability of failing to identify the SGTR before SDC initiation was estimated by assuming that the observed time to identify (1 h) was the median of a lognormal distribution with an error factor of 3.2 [see Dougherty and Fragola, *Human Reliability Analysis*, John Wiley and Sons, New York, 1988, Chapter 10. This is the error factor for time reliability correlations for actions without hesitancy, which is considered appropriate based on the slowly evolving nature of the event]. The time to SDC initiation was assumed to be 3.5 h (CESSAR, Sect. 5.4.7.3), resulting in an estimated failure probability of 0.04.

RUPTURED SG ISOL. Failure to isolate the ruptured SG. Isolation requires closure of the MSIVs, isolation of MFW and AFW to the faulted SG, blocking the ADVs on that generator closed, and an RCS cooldown to reduce RCS pressure below the SG relief valve set point. A screening value of 0.01 was used in the analysis (sequences involving failure to isolate the ruptured SG do not contribute substantially to the core damage probability for the event).

DEPRESS TO SDC. Failure to depressurize the RCS to the SDC initiation pressure. The failure probability was assumed to be dominated by operator actions associated with the cooldown and depressurization (limited depressurization is previously addressed in RUPTURED SG ISO). An operator error probability of 0.001 was utilized [see Table A.14 in Appendix A of NUREG/CR-4674 and *NRR Daily Events Evaluation Manual*, 1-275-03-336-01, January 31, 1992].

SDC. Failure to provide decay heat removal via the residual heat removal portion of the LPSI system. Two redundant trains of SDC exist at Palo Verde. Each train consists of a LPSI pump, six normally closed motor-operated valves, and two parallel, normally closed LPSI injection valves. The SDC failure probability is, therefore, approximately $[p(\text{PMP A}) \times p(\text{PMP B}|\text{ PMP A}) + 6 \times p(\text{VLV A}) \times p(\text{VLV B}|\text{ VLV A})] \times p(\text{nrec}) + p(\text{opr})$. Using typical ASP screening probabilities for pump and valve failures, a nonrecovery probability of 0.34 and an operator error probability of 0.001 [see *NRR Daily Events Evaluation Manual*, 1-275-03-336-01, January 31, 1992] results in an estimated failure probability for the branch of

$$\{[(0.01 \times 0.1) + (6 \times 0.01 \times 0.1)] \times 0.34\} + 0.001 = 3.4 \times 10^{-3}.$$

RWT REFILL. Failure to refill the RWT before RWT depletion. Based on the flow rate observed during the event, RWT refill must be accomplished before 40 h following the SGTR. The Palo Verde IPE considered RWT refill in the analysis of a maximum flow rate (600-gal/min) SGTR and assumed that it would not be initiated until the RWT low-level alarm was received, ~ 2.7 h before the tank was empty. For this time period, the IPE estimated a diagnosis time of 140 min and a resulting failure probability of 8.5×10^{-3}. Although the lower flow rate that existed during this event would provide additional diagnosis time and reduce the expected failure probability, the value of 8.5×10^{-3} was used in this analysis as well (sequences involving failure of RWT refill do not substantially contribute to the core damage probability estimated for the event).

Applying the above branch probabilities to the model for the event, as shown in Fig. A.15.2, results in an estimated core damage probability of 4.7×10^{-5}.

A.15.5 Analysis Results

The conditional core damage probability estimated for the SGTR at Palo Verde is 4.7×10^{-5}. The dominant core damage sequence, shown on Fig. A.15.2, involves the tube rupture with a postulated failure of HPSI and failure to depressurize the RCS and utilize LPSI for injection.

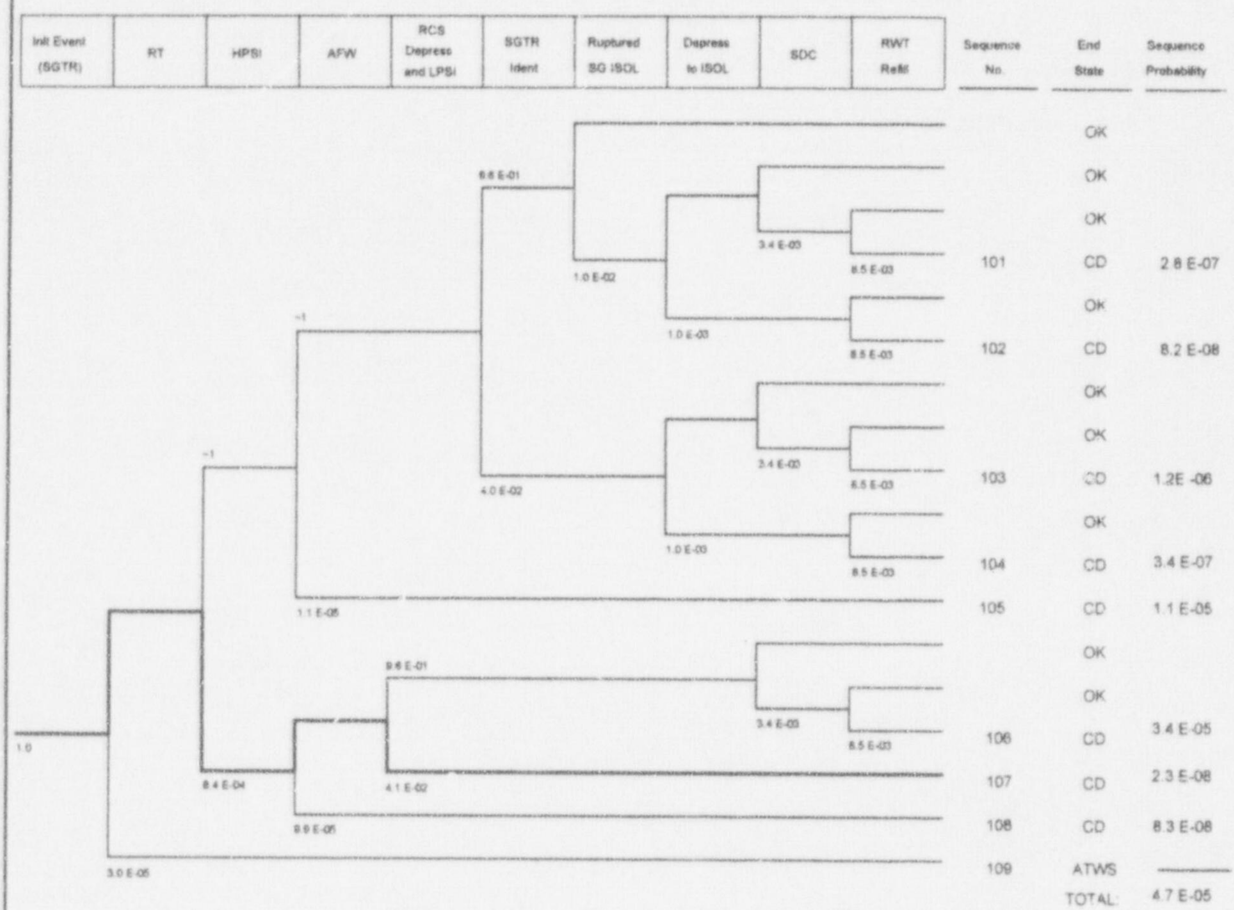


Fig. A.15.2 Dominant core damage sequence for LER 529/93-001

Appendix B:

Containment-Related Events

B.0 Containment-Related Events

None of the reactor plant operational events for 1993 were selected as containment-related events. Such events involve unavailability of containment function, containment isolation, containment cooling, containment spray, or postaccident hydrogen control. Previous reports have included a list of the events identified and, for each event, a summary, an event description, and any additional event-related plant information. Containment models have not been developed as part of the Accident Sequence Precursor Program.

Appendix C:

"Interesting" Events

C.0 "Interesting" Events

One reactor plant operational event for 1993 was selected as an "interesting" event. This event is documented in this section. "Interesting" events are not normally precursor events as defined by the Accident Sequence Precursor Program; however, they have enough unusual characteristics to warrant their inclusion in the report. The event identified for 1993 is shown in Table C.1.

A summary, event description, and any additional event-related information are provided for this event.

Table C.1 Index of "Interesting" Events

Docket/ LER No.	Description	Plant Name	Page
298/93-001	Service water system design errors	Cooper	C.1-1

C.1 LER No. 298/93-001

Event Description: Service water system design errors

Date of Event: February 25, 1993

Plant: Cooper

C.1.1 Summary

During a refueling outage at Cooper Station on February 25, 1993, a design basis review of the service water (SW) and reactor equipment cooling (REC) water systems was undertaken which identified piping configuration and other design errors. Division I service water was found to supply the division II REC heat exchanger, and division II service water was found to supply the division I REC heat exchanger. In addition, single failures were identified which could render the SW and REC systems incapable of performing their required functions.

C.1.2 Event Description

Design errors were discovered at Cooper during an engineering review of the SW and REC systems. SW division I was determined to be piped to supply the division II REC heat exchanger and SW division II was piped to supply the division I REC heat exchanger. The potential impact of this on accident sequences other than those involving floods is minimized by the fact that both SW loops are normally crosstied at Cooper. The crosstie valve between the two trains is designed to close automatically on a high water level in the control building, but is normally open. Subsequent to closure of the crosstie, many single failure concerns would exist.

The REC system at Cooper, also called the reactor building closed cooling water (RBCCW) system, is a closed-loop system providing cooling water to critical and noncritical potentially radioactive systems. Essential functions of the REC system include cooling of residual heat removal (RHR) pump seals and bearings, and room coolers in RHR, core spray (CS), and high pressure coolant injection (HPCI) pump rooms. Under accident conditions, motor-operated valves must reposition to isolate nonessential loads and to align essential loads. The nonessential REC/RBCCW loop isolation valve is fed from division I; each REC critical loop supply valve is supplied from its respective division.

The individual plant examination (IPE) for Cooper indicates that only one RBCCW pump and heat exchanger are required for RBCCW system success, given successful alignment of the motor-operated valves. In the event that the nonessential loads are not isolated, three pumps and two heat exchangers may be required. The IPE further indicates that the loss of REC will generally be of limited safety significance as service water can be provided directly to essential loads. Plans exist to provide alternate cooling to ESF equipment rooms, and other essential REC loads can operate without REC cooling for up to four hours.

Essential cooling loads on the SW system include the RBCCW heat exchangers, the diesel generators, and room cooling units in emergency diesel generator (EDG) rooms and the control bay. There are two trains of SW. Each train has two pumps. The two trains are crosstied through a normally open motor-operated valve. A normally open motor-operated valve fed from division I is provided to allow isolation of nonessential loads from the header supplying essential loads.

SW system success criteria under normal conditions require availability of two to four pumps. Under accident conditions, one SW pump is sufficient, provided it is supplying an operable loop and only supplies critical loads. Two pumps are considered adequate under normal shutdown conditions when critical and noncritical heat loads are aligned.

Given the reported design errors, had Cooper experienced a loss of offsite power (LOOP) along with a failure of EDG 1, two SW pumps would have remained available. As the nonessential service water loop isolation valve was powered from division I, remote isolation of nonessential loads apparently would have been prevented. In addition, it would not have been possible, except by manual operator actions, to isolate the non-essential REC loads, as the nonessential REC loop isolation valve was fed from division I. Further, the motor operated valve supplying critical loop "A" REC loads would not have opened, and it would not have been possible to isolate the SW flow to "A" REC heat exchanger, if it was previously aligned. With two REC pumps and one REC heat exchanger available, system success criteria would not have been met.

Consequently, for the conditions assumed, adequate cooling to the operable EDG, the functional REC heat exchanger, the RHR SW booster pump and other loads could not have been assured. Similar concerns exist for a failure of the division II EDG.

C.1.3 Analysis Results

This event was not modeled as an accident sequence precursor.

Appendix D:

Potentially Significant Events Considered Impractical to Analyze

D.0 Potentially Significant Events Considered Impractical to Analyze

Nineteen licensee event reports (LERs) have been identified as potentially significant but impractical to analyze. It is believed that such events are capable of impacting core damage sequences. However, the events usually involve 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.

For many events classified as impractical to analyze, an assumption that the affected component or function was unavailable over a 1-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 D.1.

Table D.1 Events Identified as Potentially Significant But Impractical to Analyze

Plant Name	LER Number	Title/Summary
Haddam Neck	213/93-012	All four main steam line flow transmitters were found isolated while the plant was in hot standby. The plant was returning to operation following a refueling outage. This condition effectively blocked the steam line break reactor protection function.
Indian Point 2	247/93-007	An electrical distribution system functional inspection in 1991 determined that two emergency diesel generators (EDGs) shared a "normal" 125-V dc control power supply source. The condition had existed for 10 years before being discovered. A temporary modification was made in 1991 to correct the problem. This LER reported that permanent modifications were made during the 1993 outage.
Quad Cities 2	265/93-022	Emergency core cooling system room floor drains were found inoperable. The 2B core spray/reactor core isolation-cooling (RCIC) system room drain valve failed its quarterly leak test, and debris was found lodged in the RCIC bedplate drain check valve.
Diablo Canyon 1	275/93-012	Biological fouling, which existed before a continuous chlorination process was implemented in 1992, caused a reduction in component cooling water (CCW) heat-exchanger capacity. When the CCW heat-exchanger testing was performed in response to Generic Letter No. 89-13 (GL 89-13), it was determined that design temperatures may be exceeded during the recirculation phase following a loss-of-coolant accident (LOCA).

Plant Name	LER Number	Title/Summary
Indian Point 3	286/93-045	Dampers for the control room heating, ventilation, and air conditioning (HVAC) system were determined to fail closed on loss of instrument air with unavailable backup service air. This could lead to multiple equipment failures due to overheating of control room equipment. The problem had existed since initial criticality.
Cooper	298/93-S01	Cracks were found in the seats and discs of two shutdown cooling isolation valves. The low-pressure coolant injection suction piping would be subjected to overpressurization and could possibly rupture if these two valves failed open. Also, it was possible that two residual heat removal (RHR) pumps could fail due to flooding resulting from the rupture.
Arkansas 1, Arkansas 2, Susquehanna 1, South Texas 1	313/93-005, 368/93-002, 387/93-007, 498/94-001	Breaches in the integrity of the reactor building sump were discovered. There were by-passes to the sump screens and holes in the screens that might allow debris to enter the sump and potentially block flow to certain emergency core-cooling systems. For example, on pressurized-water reactors, the high-pressure safety injection, low-pressure injection, and reactor building spray systems were vulnerable during the recirculation mode of operation following a LOCA. For Susquehanna, the licensee was assessing the potential plugging of the suppression pool strainers. This could affect the operability of the ECCS systems when their suction is aligned to the suppression pool.
Calvert Cliffs 1	317/93-007	Heat-exchanger testing and subsequent analysis in response to GL 89-13 indicated that the service water system heat exchangers may not perform as expected under certain conditions. Maximum inlet temperature was reduced 11° F to compensate.
Sequoyah 1	327/93-029	Eight check valves in the component cooling water (CCW) system supply to the reactor coolant pump thermal barriers were found stuck open. The potential existed for exposing the low-pressure piping of CCW to reactor coolant system (RCS) pressure in the event of a cooler tube failure.
McGuire 1	369/93-008	Leaf-cutter bees' nests were discovered in all the low-pressure legs (reference) of the level transmitters for the refueling water storage tank (RWST). RWST level indication during usage of the tank as a source of water could have been impaired. This, in turn, could affect longer term processes, such as sump recirculation.

Potentially Significant Events Considered Impractical to Analyze

Plant Name	LER Number	Title/Summary
McGuire 1	369/93-010	The reactor vendor disclosed a concern regarding realignment of the RHR system from service in hot shutdown to standby readiness in hot standby. There was the potential for the water in the RHR pump suction piping to flash to steam if it has not cooled enough. Following the use of RHR during a plant heatup, the RHR system is susceptible to the formation of voids in the pump section if it is used in the SI mode prior to being cooled down. The elevated RHR temperature and lower net positive suction head from the RWST could result in flashing at the pump suction.
River Bend	458/93-020	A tornado following a specific path could potentially disable certain redundant trains of equipment. Some of the diesel generator building HVAC tornado dampers may not be able to open against the discharge pressure of the exhaust fans after the tornado has passed.
Wolf Creek	482/93-014	Essential service water system fouling and long-term degradation led to low flows on various components supplied by the system. This also resulted in flow balances and throttle valve positions being set incorrectly.
Wolf Creek	482/94-001	Incorrect indicating lamps installed in various safety-related motor-control centers created the potential for common mode failures leading to multiple inoperable safety systems.
Palo Verde 1	528/93-002	A control room fire could fault control power circuits and degrade EDG operation. The affected EDG is necessary for fire mitigation. Nonessential breakers that are on a multiplexed system would also be affected.
Palo Verde 1	528/93-011	Degraded voltage protection was inadequate. Under certain conditions some safety-related 480 V components may be subjected to unacceptably low voltages.

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11. ABSTRACT (200 words or less)

Sixteen operational events that affected sixteen commercial light-water reactors during 1993 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 1993 licensee event reports from commercial light-water reactors to identify those that could potentially be precursors. Candidate precursors were then selected and evaluated in a process similar to that used in previous assessments. Selected events underwent engineering evaluation that identified, analyzed, and documented the precursors. Other events designated by the Nuclear Regulatory Commission (NRC) also underwent a similar evaluation. Finally, documented precursors were submitted for review by licensees and NRC headquarters and regional offices to ensure the plant design and its response to the precursor were correctly characterized. This study is a continuation of earlier work, which evaluated 1969-1981 and 1984-1992 events. The report discusses the general rationale for this study, the selection and documentation of events as precursors, and the estimation of conditional probabilities of subsequent severe core damage for events. This document is bound in two volumes: Volume 19 contains the main report and Appendices A-D; Volume 20 contains Appendices E and F.

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