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Precursors To Potential Severe Core Damage Accidents: 1992 A Status Report

Main Report and Appendix A

Program Managed by
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Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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FOREWORD

This report provides the 1992 results of the Nuclear Regulatory Commission's ongoing Accident Sequence Precursor (ASP) Program. The ASP Program provides a safety significance perspective of nuclear plant operational experience. The program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include initiators, degradations of plant conditions, and safety equipment failures that could increase the probability of postulated accident sequences.

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

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, are not currently incorporated in the basic ASP models. Consequently, the ASP estimates of core damage probabilities could be conservative for certain accident sequences. To address this issue, the 1992 preliminary ASP analyses were transmitted to the pertinent nuclear plant licensees and to the NRC staff for Peer Review. These 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 Peer Review comments was evaluated for reasonableness and pertinence to the ASP analysis in an attempt to use best-estimate values. All of the preliminary precursor events were reviewed, and the conditional core damage probability calculations were revised where necessary to consider information provided during the review. The objective of the Peer Review process was to provide as realistic an analysis of the significance of the event as possible. As a result, the 1992 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 the year-to-year trending of risk somewhat more difficult, we believe it is an important step towards more realistic identification of significant events and conditions.

The most important precursor events of 1992 (with one exception) involved electrical problems, including the reliability of the electrical transmission lines (the grid) serving the plant, and plant electrical problems, such as failure of equipment in the switchyard. One of these precursors involved Hurricane Andrew, which affected southern Florida on August 24, 1992. This hurricane caused extensive damage to the electric transmission lines serving the Turkey Point nuclear plant units, requiring these units to rely on their own onsite emergency a.c. power sources for several days. The one important 1992 precursor event which did not involve electrical problems, involved a partially stuck open pressurizer safety valve at the Fort Calhoun plant.

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PREFACE

The Accident Sequence Precursor (ASP) Program was established at the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory in the summer of 1979. The first major report of that program was published in June 1982 and received extensive review. A total of ten reports documenting the review of operational events for precursors have been previously published in this program (see Sect. 1.3, Reference Nos. 1-10). These reports, which began in 1982, are for events that occurred from 1969 through 1991, 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 Operation Data (AEOD) of the Nuclear Regulatory Commission (NRC). The NRC Technical Monitor for the project is F. M. Manning.

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 is a continuation, for 1992, of the assessment undertaken in the previous reports for operational events that occurred in 1969-1981 and 1984-1991.

Normally, comments regarding the preliminary precursor analysis for each year are solicited from ORNL peer reviews as well as from NRC AEOD headquarters staff. This year, however, the preliminary analyses for 1992 events were also sent for review to the licensees and the NRC regional offices for those plants for which potential ASP events were identified. Essentially all the potential precursors were reanalyzed as result of comments received and calculations revised as appropriate. Primarily, the reanalyses focused on and gave credit for equipment and procedures recently added by the licensees that provided more protection against core damage. These additional features were beyond what is usually included in the ASP models. Therefore, comparing and trending results from prior years are more difficult since results from the 1992 analyses are likely somewhat lower after considering information provided by the licensees. Judgement should also be exercised in comparing results from one plant to another within the same class of plants given the incorporation of plant specific information in the analyses beyond that contained in the ASP models. The overall objective in soliciting and considering responses from licensees was an attempt to provide a more realistic assessment of significant events.

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 probabilistic risk assessment-related information.

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LIST OF ACRONYMS

ADS	automatic depressurization system
AEOD	NRC Office for Analysis and Evaluation of Operational Data
AFW	auxiliary feedwater
AIT	augmented inspection team
ASP	accident sequence precursor (program)
ATWS	anticipated transient without scram
BWR	boiling-water reactor
BWST	borated water storage tank
CAR	containment air recirculation
CCP	centrifugal charging pump
CCW	component cooling water
CRD	control rod drive
CSR	containment spray recirculation
CST	condensate storage tank
DG	diesel generator
DHR	decay heat removal
DSDG	dedicated shutdown diesel generator
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFW	emergency feedwater
EHC	electrohydraulic control
EOP	emergency operating procedures
EPS	emergency power system
ESF	engineered safety feature
FWCI	feedwater coolant injection
FSAR	final safety analysis report
HHSI	high-head safety injection
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HPI	high-pressure injection
HPR	high-pressure recirculation
IA	instrument air
IC	isolation condenser
IEEE	Institute of Electrical and Electronics Engineers
IIT	incident investigation team
INPO	Institute of Nuclear Power Operation
IPE	Individual Plant Examination
LER	licensee event report
LOCA	loss-of-coolant accident
LOFW	loss of main feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LPI	low-pressure injection
LPR	low-pressure recirculation
LPS	liquid poison system
LWR	light-water reactor
MDAFWP	motor-driven auxiliary feedwater pump
MDEFWP	motor-driven emergency feedwater pump
MFW	main feedwater

MOV	motor-operated valve
MSIV	main steam isolation valve
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
PCB	power circuit breaker
PCS	power conversion system
PORV	pilot- or power-operated relief valve
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RHRSW	residual heat removal service water
RPS	reactor protection system
RPV	reactor pressure vessel
RV	relief valve or reactor vessel
RWCU	reactor water cleanup
RWST	refueling water storage tank
RY	reactor year
SCSS	sequence coding and search system
SDC	shutdown cooling
SG	steam generator
SI	safety injection
SLC	standby liquid control
SRO	senior reactor operator
SRV	safety relief valve
SSF	safe shutdown facility
SSGFW	standby steam generator feedwater
SSMP	safe shutdown makeup pump
STS	standard technical specifications
SW	service water
TBS	turbine bypass system
TDAFWP	turbine-driven auxiliary feedwater pump
TDEFWP	turbine-driven emergency feedwater pump
USFAR	updated final safety analysis report

PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1992, A STATUS REPORT

ABSTRACT

Twenty-seven operational events with conditional probabilities of subsequent severe core damage of 1.0×10^{-6} or higher occurring at commercial light-water reactors during 1992 are considered to be precursors to potential severe core damage. These are described along with associated significance estimates, categorization, and subsequent analyses. This study is a continuation of earlier work, which evaluated 1969-1981 and 1984-1991 events. The report discusses (1) the general rationale for this study, (2) the selection and documentation of events as precursors, (3) the estimation and use of conditional probabilities of subsequent severe core damage to rank precursor events, and (4) the plant models used in the analysis process.

1.0 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) beginning in 1969 to identify and categorize 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] and NUREG/CR-3591, *Precursors to Potential Severe Core Damage Accidents: 1980-1981, A Status Report*,[2] as well as in earlier volumes of this document.[3-10] This report details the work of the ASP Program in its review and evaluation of operational events that occurred in 1992. The requirements for LERs are described in NUREG-1022, *Licensee Event Report System, Description of System and Guidelines for Reporting*,[11] as well as in the supplements to NUREG-1022.[12-13] LERs reviewed for precursors are described in Chapter 2.

1.1 Background

The ASP Program owes its genesis to the Risk Assessment Review Group,[14] 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."[15] Evaluations done for the 1969-1981 period were the first efforts in this type of analysis.

Accident sequences of interest in this study are those that, if additional failures had occurred, would have resulted in inadequate core cooling and that could have resulted in severe core damage. Accident sequence precursors are events that are important elements in such accident sequences. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition leading to severe core damage. Precursors were selected and evaluated by an evaluation process and significance quantification methodology similar to that used in previous yearly assessments. All 1992 LERs were computer-screened to identify events that could be precursors. Such events were subjected to an engineering evaluation that identified, analyzed, and documented the precursors, as described in Chapter 2.

In addition to the events selected as accident sequence precursors, events involving loss of containment function and other events that are considered serious but that are not modeled in the ASP Program were identified during the 1992 LER review. These events are also documented in this repo.

A study of this nature is subject to certain inherent limitations. The results were based on limited data, and the study may be biased by many of the decisions inherent in the process as well as in the methodology itself. A determined effort is being made in this program to address these limitations. Although uncertainties exist in the numeric probability estimates associated with each event addressed in the report, the identification of the more serious events from a core damage standpoint is considered reasonably certain.

1.2 Organization of the Report

This effort has been divided into several tasks, the results of which may be found in the sections indicated:

Section	Task
Chapter 2	Detailed review of 1992 LERs for accident sequence precursors and quantification of precursor significance
Chapter 3	Discussion of results
Appendix A	ASP analysis methodology and plant models
Appendix B	Precursors
Appendix C	Containment-related events
Appendix D	Interesting or "other" events
Appendix E	Events that were considered impractical to analyze
Appendix F	Licensee Event Reports and Augmented Inspection Team Reports
Appendix G	Responses to review comments from licensees and NRC

In addition, a list of acronyms and a glossary are provided.

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

2.0 ACCIDENT SEQUENCE PRECURSOR IDENTIFICATION AND QUANTIFICATION

2.1 Accident Sequence Precursor Identification

The 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 frequencies and probabilities associated with them.

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. For core damage to occur, fuel temperature must increase. Such an increase requires the heat generation rate in the core to exceed the heat removal rate. This can result from either a loss of core cooling or excessive core power. The following functions are provided at all plants to protect against these two conditions:

- Reactor subcriticality. The reactor must be placed in a subcritical condition, normally by inserting control rods into the core to terminate the chain reaction.
- Reactor coolant inventory makeup. Sufficient water must be provided to the reactor coolant system (RCS) to prevent core uncovery.
- RCS integrity. Loss of RCS integrity requires the addition of a significant quantity of water to prevent core uncovery.
- Decay heat removal (DHR). Heat generated in the core by fission product decay must be removed.
- Containment integrity. Containment integrity (containment heat removal, isolation, and hydrogen control) is not addressed in the precursor analyses unless core DHR capability is impacted.

System-based event trees were developed to model potential sequences to core damage. The event trees are specific to eight plant classes so as to reflect differences in design among plants in the U.S. LWR population. Three initiators are addressed in the event trees: 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. [Excessive core power associated with anticipated transient without scram (ATWS) is represented by a failure-to-trip sequence but is not developed.] Based on previous experience with reactor plant operational events, it is known that most operational events can be directly or indirectly associated with these initiators. Detailed descriptions of the plant classification scheme and the event tree models are included in Appendix A. Operational events that cannot be associated with one of these initiators are accommodated by unique modeling.

Armed with a knowledge of the primary core damage initiator types plus the systems that provide protection against core damage (based on the event tree models), ASP Program staff members examine LERs to determine the impact of operational events on potential core damage sequences. While the sequences detailed on the event tree models do not describe all possible paths to core damage, they form a primary basis for selecting an operational event as a precursor. Operational events are also reviewed in a more general sense for their impact on the protective functions described above.

Identification of precursors within a set of LERs involved a two-step process. First, each LER was reviewed by two experienced engineers to determine if the reported event should be examined in 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 was done by eliminating events that satisfied pre-defined criteria for rejection and accepting all others as potentially significant and requiring analysis. 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. 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. Such events included single failures in redundant systems and uncomplicated reactor trips and LOFWs. Any event with an impact that can be mapped onto the ASP core damage models can, in principle, be assessed.

LERs were eliminated from further consideration as precursors if they involved at most 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,
- a structural degradation,
- an event that occurred prior to initial criticality (since the core is not considered vulnerable to core damage at this time and since distinguishing initial testing failures from operational failures is difficult),
- 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 (selected containment-related events are documented).

Events identified for further consideration typically included

- 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 where 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.

Operational events that were not eliminated in the first review received a more extensive analysis 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), their amendments, and other information available at the Nuclear Operations Analysis Center.

The detailed review of each event considered (l) the immediate impact of an initiating event or (2) 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 evaluated according to 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, then its impact on continued DHR 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 they included one of the following attributes that impacted core damage sequences and if the conditional probability of subsequent core damage (described later) was at least 1.0×10^{-6}

- an unexpected core damage initiator (such as a LOOP, steam-line break (SLB), or small-break LOCA);
- a failure of a system (all trains of a multiple train system) required to mitigate the consequences of a core damage initiator,
- concurrent degradation in more than one system required to mitigate the consequences of a core damage initiator, or
- a transient or LOFW with a degraded mitigating system.

Events of low significance are thus excluded, allowing the reader to concentrate on the more important events. This approach is consistent with the approach used to define 1987-1991 precursors, but is different from that of earlier ASP reports, which addressed all events meeting the precursor selection criteria, regardless of conditional core damage probability.

Events that occurred in 1992 were reviewed for precursors only if they satisfied an initial significance screening. This approach, which was similar to that used in the review of 1988-1991 events, eliminated many insignificant events from review and permitted some increase in the amount of documentation provided for precursors. Two approaches were used to select events to be reviewed for precursors.

First, events were reviewed for precursors if they were identified as significant by the Nuclear Regulatory Commission's (NRCs) Office for Analysis and Evaluation of Operational Data (AEOD). AEOD's screening process identifies operating occurrences involving, in part,

- violation of a safety limit;
- an alert or higher emergency classification;
- an on-demand failure of a safety system (except surveillance failures);
- events involving unexpected system or component performance with serious safety significance or generic implications;
- events where improper operation, maintenance, or design causes a common-mode/common-cause failure of a safety system or component, with safety significance or generic implications;
- safety-significant system interactions;
- events involving cognitive human errors with safety significance or generic implications;
- safety-significant events involving earthquakes, tornadoes, floods, and fires;
- a scram, transient, or engineered safety features (ESF) actuation with failure or inoperability of required equipment;
- on-site work-related or nuclear-incident-related death, serious injury, or exposure that exceeds administrative limits;
- unplanned or unmonitored releases of radioactivity, or planned releases that exceed Technical Specification limits; and
- infrequent or moderate frequency events.

AEOD-designated significant events also involve operating conditions, where a failure or accident has not occurred but where the potential for such an event is identified.

Second, LERs were also reviewed if they were identified through a computerized search using the sequence coding and search system (SCSS) data base of LERs. This computerized search identified LERs potentially involving (1) failures in plant systems that provided the protective functions described earlier and (2) initiating events addressed in the ASP models. Based on a review of the 1984-87 precursor evaluations, this computerized search successfully identifies almost all precursors within a subset of approximately one-third of all LERs.

While review of LERs identified by AEOD and through the use of SCSS 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 1992 LERs had been reviewed may not have been identified. Because of this (plus modeling changes that impact precursor probability somewhat), it should not be assumed that the set of 1988-92 precursors is consistent with precursors identified in 1984-87.

Following AEOD and SCSS computerized screening, 1022 LERs from 1992 were reviewed for precursors. Twenty-seven operational events with conditional probabilities of subsequent severe core damage greater than 1.0×10^{-6} were identified as accident sequence precursors.

Individual failures of boiling-water reactor (BWR) high-pressure coolant injection (HPCI), high-pressure core spray (HPCS), and reactor core isolation cooling (RCIC) systems (all single-train systems), and trips and LOFWs without additional mitigating system failures were not selected as precursors. The impact of such events was determined on a plant-class basis. The results of these evaluations are provided in Appendix A.

In addition to accident sequence precursors, events involving loss of containment functions — containment cooling, containment spray, containment isolation (direct paths to the environment only), and hydrogen control — were identified in the review of 1992 LERs. Other events that were not selected as precursors but that provided insight into unusual failure modes with the potential to compromise continued core cooling are also identified. Events identified as precursors are documented in Appendix B, the containment-related events are documented in Appendix C, events considered "interesting" are documented in Appendix D, and events that were determined to be impractical to analyze are documented in Appendix E.

2.2 Estimation of Precursor Significance

Quantification of ASP 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.

In the precursor quantification process, it is assumed that the failure probabilities for systems observed to have failed during an event are equal to the likelihood of not recovering from the failure or fault that actually occurred. Failure probabilities for systems observed to have been degraded during an operational event are assumed equal to the conditional probability that the system would fail (given that it was observed degraded) and the probability that it would not be recovered within the required time period. The failure probabilities associated with observed successes and with systems unchallenged during the actual occurrence are assumed equal to a failure probability estimated from either system failure data (when available) or by the use of system success criteria and typical train and common-mode failure probabilities, with consideration of the potential for recovery. 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 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 consider and, where appropriate, give credit for additional equipment or recovery procedures the plants have recently added. Accordingly, the evaluations this year may not be directly comparable to the results of prior years. Examples of additional equipment and recovery procedures addressed in the 1992 analyses, when information was available, include use of supplemental diesel generators (DGs) for station blackout mitigation, alternate systems for steam generator (SG) and RCS makeup, and depressurization of the primary with low pressure injection (LPI) in lieu of high pressure injection (HPI).

The ASP calculational process is described in detail in Appendix A. This appendix documents the event trees used in the 1988-1992 precursor analyses, changes to these trees from prior years, the approach used to estimate event tree branch and sequence probabilities, and sample calculations; it also provides probability values used in the calculations. The overall precursor selection process is illustrated in Fig. 1.

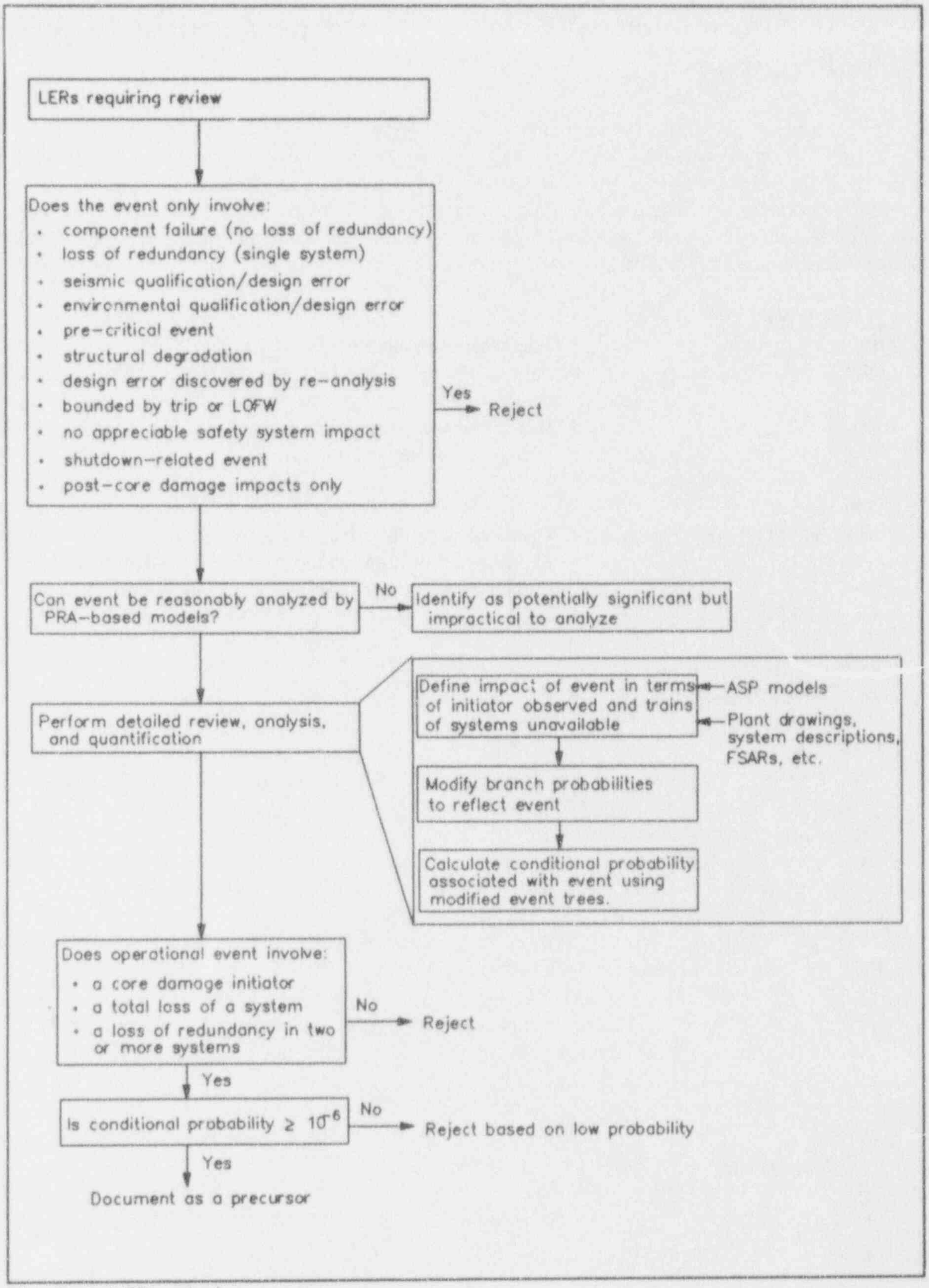

Fig. 1. ASP analysis process.

2.3 Documentation of Events Selected as Accident Sequence Precursors

Each 1992 precursor is documented in Appendix B. A description of the operational event is provided along 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 that (1) visually describe the dominant core damage sequence postulated for the event and (2) present a graph of the relative significance of the event compared with other potential events at the 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, grid-centered, severe weather and extreme severe weather-related initiators.
360h EP	• 360 h without emergency power sources (normally on-site emergency diesel generators).

PWR

LOFW + 1MTR AFW	• Transient with loss of main feedwater and one motor driven AFW (or EFW pump failed (turbine driven pump substituted if plant does not have any motor driven pumps).
360h w/o AFW	• 360 hours with all AFW (or EFW) pumps failed.

BWR

360 h w/o HPCI and RCIC	• 360 hours with HPCI and 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 Condensor is run instead for Type A BWRs).

An additional item, the conditional core damage 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 event are also provided in Appendix F, listed in docket number order.

Appendices C, D and E include similar documentation for other events selected in the ASP Program (containment-related, other, and impractical events). No probabilistic analysis was performed on these events.

2.4 Tabulation of Selected Events

The 1992 events selected as precursors are listed in Table 1. The precursors have been arranged in numerical order by event identifier and the following information is included:

1. docket/LER number associated with the event (Event Identifier);
2. name of plant where the event occurred (Plant);
3. a brief description of the event (Description);
4. date of the event (Event Date);
5. conditional probability of potential severe core damage associated with the event (C_D Probability);
6. initiator associated with the event or unavailability if no initiator was involved (TRANS).
7. abbreviations for the primary system and component involved in the event (System, Component);
8. plant operating status at the time of the event (O);
9. discovery method associated with the event (operational or testing) (D);
10. whether the event involved human error (E);
11. plant power rating, type, vendor, architect-engineer, and licensee (MWE, T, V, AE, Operator);

The information in Table 1 has been sorted in several ways to provide additional perspectives.

	Sorted by
Table 2	Plant name and LER number
Table 3	Event date
Table 4	Initiator or unavailability
Table 5	System
Table 6	Component
Table 7	Plant operating status
Table 8	Discovery method
Table 9	Conditional core damage probability
Table 10	Plant type and vendor

Abbreviations used in Tables 1—10 are defined in Tables 11a—11f.

2.5 Potentially Significant Events That Could Not Be Analyzed

A number of LERs identified as potentially significant were considered impractical to analyze. Examples of such events include component degradations where the extent of degradation could not be determined (for example, biological fouling of room coolers) or where a realistic estimate of plant response could not be made (for example, high energy line break concerns). Other events of this type include cable routing not in accordance with Appendix R requirements for fire protection, and inoperability of flood barriers. For both of these situations, detailed plant design information, and preferably an existing fire or flood PRA analysis, are required to reasonably estimate the significance of the event.

For many events classified as impractical to analyze, an assumption that the impacted component or function was unavailable over a 1-year period (as would be done using a bounding analysis) would result in a conclusion that a very significant condition 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 ASP Program. A reasonable estimate of significance for such events requires far more analysis resources than can be applied in the ASP Program.

Brief descriptions of events considered impractical to analyze are provided in Appendix E.

2.6 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.

1. *Evaluation of only a subset of 1992 LERs.* For 1969-81 and 1984-87, all LERs reported during the year were evaluated for precursors. For 1988-92, only a subset of LERs were evaluated in the ASP Program following a computerized search of the SCSS data base and screening by NRC personnel. While this subset is believed 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 was screened.

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

3. *Lack of appropriate information in the LER.* The accuracy and completeness of the LERs in reflecting pertinent operational information is 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 (e.g., 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.

4. *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 pressurized-water reactors (PWRs), long-term DHR for BWRs, and 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 (this is difficult to ascertain without extensive operating experience), the actual conditional probability for an event could be higher or lower than that calculated in the analysis.

   Known plant-specific equipment and procedures that can provide additional protection against core damage beyond the plant-class features included in the ASP event tree models were addressed in the 1992 precursor analysis. This information was not uniformly available – much of it was provided in licensee comments on preliminary analyses and in Individual Plant Examination (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 were uniformly developed.

5. *Difficulty in determining the potential for recovery of failed equipment.* Assignment of recovery credit for an event can have a significant impact on the assessment of the event. The approach used to assign recovery credit is described in detail in Appendix A. The actual likelihood of failing to recover from an event at a particular plant 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, etc., concerning the likelihood of recovering from specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event

   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 believed 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 1980-81 report.[1] These calculations demonstrated only a small impact on the relative ranking of events from changes in the numeric values used for each recovery class.

6. *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.

   If the test interval is longer than this, on the average, for a particular system, then the calculated probability will be lower than that calculated using the actual test interval. Examples of longer test intervals would be situations in which (1) system valves are operated monthly but a system pump is started only quarterly or (2) valves are partially stroked monthly but fully operated only during refueling. Conversely, more frequent testing will result in a higher calculated failure probability than that calculated using the actual, shorter test interval. Test interval assumptions can also impact system failure probabilities estimated from precursor events, as described in Ref. 1.

2.7 Reference

1. W. B. Cottrell, J. W. Minarick, P. N. Austin, E. W. Hagen, and J. D. Harris, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., *Precursors to Potential Severe Core Damage Accidents: 1980-81, A Status Report*, USNRC Report NUREG/CR-3591, Vols. 1 and 2 (ORNL/NSIC-217/V1 and V2), July 1984.*

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

Table 1. Precursors Listed by Identifier

Event Identifier	Plant	Description	Event Date	C_p Probability	TRANS	System	Component	O	D	E	MWE	T	V	AE	Operator
219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	7.1E-5	LOOP	EA	ELECON	E	O	N	650	B	G	BG	GPU
247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	3.6E-6	TRIP	HH	CKTBRK	E	O	N	873	P	W	UE	CEC
250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	8/24/92	1.6E-4	LOOP	EA	ELECON	G	O	N	693	P	W	BX	FPL
251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	8/24/92	1.6E-4	LOOP	EA	ELECON	G	O	N	693	P	W	BX	FPL
251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	3.1E-6	LOFW	HH	PUMPXX	C	C	Y	693	P	W	BX	FPL
254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	6.9E-6	TRIP	CC	VALVOP	E	O	N	789	B	G	SL	CWE
261/92-013	Robinson 2	SI PUMP OOS	07/10/92	3.5E-5	UNAVL	SF	PUMPXX	E	T	Y	700	P	W	EX	CPL
261/92-017	Robinson 2	LOOP	08/22/92	2.1E-4	LOOP	EA	RELAYX	E	O	N	700	P	W	EX	CPL
269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	4.0E-6	TRIP	SF	VALVOP	C	O	Y	887	P	B	UX	DPC
269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
270/92-004	Oconee 2	LOOP WITH FAILED EP	10/19/92	2.1E-4	LOOP	EA	ELECON	E	M	Y	887	P	B	UX	DPC
285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	2.5E-4	TRIP	IB	INSTRU	E	M	Y	478	P	C	GH	OPP
286/92-011	Indian Point 3	MULTIPLE EDGs INOP	07/06/92	1.2E-6	UNAVL	EC	ELECON	D	T	Y	965	P	W	UE	PNY
301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	9.9E-6	UNAVL	SF	PUMPXX	E	T	Y	497	P	W	BX	WEP
302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	1.7E-5	LOOP	EA	ELECON	E	M	Y	825	P	B	GX	FPC
327/92-027	Sequoyah 1	LOOP	12/31/92	1.8E-4	LOOP	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
327/92-027	Sequoyah 2	LOOP	12/31/92	1.8E-4	LOOP	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	1.9E-6	UNAVL	CF	VALVOP	E	T	Y	1148	P	W	UX	TVA
344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	5.9E-6	TRIP	HH	INSTRU	E	O	N	1130	P	W	BX	PGC
374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	6.1E-6	TRIP	CE	VALVOP	F	O	N	1078	B	G	SL	CWE
388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	6.6E-6	TRIP	EA	ELECON	E	T	N	1050	B	G	BX	PPL
483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	1.3E-5	UNAVL	IF	ANNUNC	E	T	Y	1171	P	W	BX	UEC

Table 2. Precursors Listed by Plant

Plant	Event Identifier	Description	Event Date	C_p Probability	TRANS	System	Component	O	D	E	MWE	T	V	AE	Operator
Callaway	483/92-011	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	1.3E-5	UNAVL	IF	ANNUNC	E	T	Y	1171	P	W	BX	UEC
Crystal River 3	302/92-001	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	1.7E-5	LOOP	EA	ELECON	E	M	Y	825	P	B	GX	FPC
Fort Calhoun 1	285/92-023	RX TRIP WITH FAULTY PSV	07/03/92	2.5E-4	TRIP	IB	INSTRU	E	M	Y	478	P	C	GH	OPP
Indian Point 2	247/92-007	RX TRIP & AFW PUMP PROBLEMS	04/13/92	3.6E-6	TRIP	HH	CKTBRK	E	O	N	873	P	W	UE	CEC
Indian Point 3	286/92-011	MULTIPLE EDGs INOP	07/06/92	1.2E-6	UNAVL	EC	ELECON	D	T	Y	965	P	W	UE	PNY
La Salle 2	374/92-012	RX TRIP WITH DEGRADED RCIC	08/27/92	6.1E-6	TRIP	CE	VALVOP	F	O	N	1078	B	G	SL	CWE
Oconee 1	269/92-004	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	4.0E-6	TRIP	SF	VALVOP	C	O	Y	887	P	B	UX	DPC
Oconee 1	269/92-008	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
Oconee 1	269/92-018	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
Oconee 2	269/92-008	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
Oconee 2	269/92-018	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
Oconee 2	270/92-004	LOOP WITH FAILED EP	10/19/92	2.1E-4	LOOP	EA	ELECON	E	M	Y	887	P	B	UX	DPC
Oconee 3	269/92-008	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
Oconee 3	269/92-018	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
Oyster Creek	219/92-005	LOOP DUE TO FOREST FIRE	05/03/92	7.1E-5	LOOP	EA	ELECON	E	O	N	650	B	G	BG	GPU
Point Beach 2	301/92-003	PLUGGED SI PUMP SUCTION	09/18/92	9.9E-6	UNAVL	SF	PUMPXX	E	T	Y	497	P	W	BX	WEP
Quad Cities 1	254/92-004	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	6.9E-6	TRIP	CC	VALVOP	E	O	N	789	B	G	SL	CWE
Robinson 2	261/92-013	SI PUMP OOS	07/10/92	3.5E-5	UNAVL	SF	PUMPXX	E	T	Y	700	P	W	EX	CPL
Robinson 2	261/92-017	LOOP	08/22/92	2.1E-4	LOOP	EA	RELAYX	E	C	N	700	P	W	EX	CPL
Sequoyah 1	327/92-027	LOOP	12/31/92	1.8E-4	LOOP	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
Sequoyah 2	327/92-027	SAME EVENT AS FOR UNIT 1 ABOVE	12/31/92	1.8E-4	LOOP	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
Sequoyah 2	328/92-010	EDG & RHR PUMP INOP	07/17/92	1.9E-6	UNAVL	CF	VALVOP	E	T	Y	1148	P	W	UX	TVA
Susquehanna 2	388/92-001	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	6.6E-6	TRIP	EA	ELECON	E	T	N	1050	B	G	BX	PPL
Trojan	344/92-020	RX TRIP & AFW PUMP FAIL TO START	07/22/92	5.9E-6	TRIP	HH	INSTRU	E	O	N	1130	P	W	BX	PGC
Turkey Point 3	250/92-SO1	LOOP DUE TO HURRICANE ANDREW	08/24/92	1.6E-4	LOOP	EA	ELECON	G	O	N	693	P	W	BK	FPL
Turkey Point 4	251/92-007	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	3.1E-6	LOFW	HH	PUMPXX	C	O	Y	693	P	W	BX	FPL
Turkey Point 4	251/92-SO1	LOOP DUE TO HURRICANE ANDREW	08/24/92	1.6E-4	LOOP	EA	ELECON	G	O	N	693	P	W	BK	FPL

Table 3. Precursors Listed by Event Date

Event Date	Event Identifier	Plant	Description	C_p Probability	TRANS	System	Component	O	D	E	MWE	T	V	AE	Operator
02/06/92	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	6.9E-6	TRIP	CC	VALVOP	E	O	N	789	B	G	SL	CWE
03/18/92	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	6.6E-6	TRIP	EA	ELECON	E	T	N	1050	B	G	BX	PPL
03/27/92	302/92-001	Crystal River 3	LOOP WITN INOP VITAL BUS INVERTER	1.7E-5	LOOP	EA	ELECON	E	M	Y	825	P	B	GX	FPC
04/13/92	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	3.6E-6	TRIP	HH	CKTBRK	E	O	N	873	P	W	UE	CEC
05/03/92	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	7.1E-5	LOOP	EA	ELECON	E	O	N	650	B	G	BG	GPU
05/08/92	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	4.0E-6	TRIP	SF	VALVOP	C	O	Y	887	P	B	UX	DPC
07/03/92	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	2.5E-4	TRIP	IB	INSTRU	E	M	Y	478	P	C	GH	OPP
07/06/92	286/92-011	Indian Point 3	MULTIPLE EDGs INOP	1.2E-6	UNAVL	EC	ELECON	D	T	Y	965	P	W	UE	PNY
07/10/92	261/92-013	Robinson 2	SI PUMP OOS	3.5E-5	UNAVL	SF	PUMPXX	E	T	Y	700	P	W	EX	CPL
07/17/92	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
07/17/92	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
07/17/92	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	2.8E-6	UNAVL	EA	ELECON	E	O	N	887	P	B	UX	DPC
07/17/92	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	1.9E-6	UNAVL	CF	VALVOP	E	T	Y	1148	P	W	UX	TVA
07/22/92	344/92-020	Trojan	RX TRP & AFW PUMP FAIL TO START	5.9E-6	TRIP	HH	INSTRU	E	O	N	1130	P	W	BX	PGC
08/22/92	261/92-017	Robinson 2	LOOP	2.1E-4	LOOP	EA	RELAYX	E	O	N	700	P	W	EX	CPL
08/24/92	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	1.6E-4	LOOP	EA	ELECON	G	O	N	693	P	W	BK	FPL
08/24/92	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	1.6E-4	LOOP	EA	ELECON	G	O	N	693	P	W	BK	FPL
08/27/92	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	6.1E-6	TRIP	CE	VALVOP	F	O	N	1078	B	G	SL	CWE
09/18/92	301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	9.9E-6	UNAVL	SF	PUMPXX	E	T	Y	497	P	W	BX	WEP
09/29/92	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	3.1E-6	LOFW	HH	PUMPXX	C	O	Y	693	P	W	BX	FPL
10/17/92	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	1.3E-5	UNAVL	IF	ANNUNC	E	T	Y	1171	P	W	BX	UEC
10/19/92	270/92-004	Oconee 2	LOOP WITH FAILED EP	2.1E-4	LOOP	EA	ELECON	E	M	Y	887	P	B	UX	DPC
12/02/92	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
12/02/92	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
12/02/92	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	3.2E-5	UNAVL	EA	ELECON	E	T	N	887	P	B	UX	DPC
12/31/92	327/92-027	Sequoyah 1	LOOP	1.8E-4	LOOP	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
12/31/92	327/92-027	Sequoyah 2	LOOP	1.8E-4	LOOP	EA	ELECON	E	T	Y	1148	P	W	UX	TVA

Table 4. Precursors Listed by Initiator or Unavailability

TRANS	C_p Probability	Event Identifier	Plant	Description	Event Date	System	Component	O	D	E	MWE	T	V	AE	Operator
LOFW	3.1E-6	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	H[illegible]	[illegible]MPXX	C	O	Y	693	P	W	BX	FPL
LOOP	7.1E-5	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	[illegible]	ELECON	E	O	N	650	B	G	BG	GPU
LOOP	1.6E-4	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	08/24/92	[illegible]	ELECON	G	O	N	693	P	W	BK	FPL
LOOP	1.6E-4	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	08/24/92	EA	ELECON	G	O	N	693	P	W	BK	FPL
LOOP	2.1E-4	261/92-017	Robinson 2	LOOP	08/22/[illegible]	EA	RELAYX	E	O	N	700	P	W	EX	CPL
LOOP	2.1E-4	270/92-004	Oconee 2	LOOP WITH FAILED EP	10/1[illegible]	EA	ELECON	E	M	Y	887	P	B	UX	DPC
LOOP	1.7E-5	302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	EA	ELECON	E	M	Y	825	P	B	GX	FPC
LOOP	1.8E-4	327/92-027	Sequoyah 1	LOOP	12/31/92	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
LOOP	1.8E-4	327/92-027	Sequoyah 2	LOOP	12/31/92	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
TRIP	3.6E-6	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	HH	CKTBRK	E	O	N	873	P	W	UE	CEC
TRIP	6.9E-6	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	CC	VALVOP	E	O	N	789	B	G	SL	CWE
TRIP	4.0E-6	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	SF	VALVOP	C	O	Y	887	P	B	UX	DPC
TRIP	2.5E-4	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	IB	INSTRU	E	M	Y	478	P	C	GH	OPP
TRIP	5.9E-6	344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	HH	INSTRU	E	O	N	1130	P	W	BX	PGC
TRIP	6.1E-6	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	CE	VALVOP	F	O	N	1078	B	G	SL	CWE
TRIP	6.6E-6	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	EA	ELECON	E	T	N	1050	B	G	BX	PPL
UNAVL	3.5E-5	261/92-013	Robinson 2	SI PUMP OOS	07/10/92	SF	PUMPXX	E	T	Y	700	P	W	EX	CPL
UNAVL	2.8E-6	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	EA	ELECON	E	O	N	887	P	B	UX	DPC
UNAVL	2.8E-6	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	EA	ELECON	E	O	N	887	P	B	UX	DPC
UNAVL	2.8E-6	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	EA	ELEC[illegible]	E	O	N	887	P	B	UX	DPC
UNAVL	3.2E-5	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	EA	ELECON	E	T	N	887	P	B	UX	DPC
UNAVL	3.2E-5	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	EA	ELECON	E	T	N	887	P	B	UX	DPC
UNAVL	3.2E-5	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	EA	ELECON	E	T	N	887	P	B	UX	DPC
UNAVL	1.2E-6	286/92-011	Indian Point 3	MULTIPE EDGs INOP	07/06/92	EC	ELECON	D	T	Y	965	P	W	UE	PNY
UNAVL	9.9E-6	301/92-003	[illegible] Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	SF	PUMPXX	E	T	Y	497	P	W	BX	WEP
UNAVL	1.9E-6	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	CF	VALVOP	E	T	Y	1148	P	W	UX	TVA
UNAVL	1.3E-5	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	IF	ANNUNC	E	T	Y	1171	P	W	BX	UEC

Table 5. Precursors Listed by System

System	Component	O	D	E	C_p Probability	TRANS	Event Identifier	Plant	Description	Event Date	MWE	T	V	AE	Operator
CC	VALVOP	E	O	N	6.9E-6	TRIP	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	789	B	G	SL	CWE
CE	VALVOP	F	O	N	6.1E-6	TRIP	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	1078	B	G	SL	CWE
CF	VALVOP	E	T	Y	1.9E-6	UNAVL	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	1148	P	W	UX	TVA
EA	ELECON	E	O	N	7.1E-5	LOOP	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	650	B	G	BG	GPU
EA	ELECON	G	O	N	1.6E-4	LOOP	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL
EA	ELECON	G	O	N	1.6E-4	LOOP	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL
EA	RELAYX	E	O	N	2.1E-4	LOOP	261/92-017	Robinson 2	LOOP	08/22/92	700	P	W	EX	CPL
EA	ELECON	E	O	N	2.8E-6	UNAVL	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
EA	ELECON	E	O	N	2.8E-6	UNAVL	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
EA	ELECON	E	O	N	2.8E-6	UNAVL	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
EA	ELECON	E	T	N	3.2E-5	UNAVL	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
EA	ELECON	E	T	N	3.2E-5	UNAVL	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
EA	ELECON	E	T	N	3.2E-5	UNAVL	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
EA	ELECON	E	M	Y	2.1E-4	LOOP	270/92-004	Oconee 2	LOOP WITH FAILED EP	10/19/92	887	P	B	UX	DPC
EA	ELECON	E	M	Y	1.7E-5	LOOP	302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	825	P	B	GX	FPC
EA	ELECON	E	T	Y	1.8E-4	LOOP	327/92-027	Sequoyah 1	LOOP	12/31/92	1148	P	W	UX	TVA
EA	ELECON	E	T	Y	1.8E-4	LOOP	327/92-027	Sequoyah 2	LOOP	12/31/92	1148	P	W	UX	TVA
EA	ELECON	E	T	N	6.6E-6	TRIP	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	1050	B	G	BX	PPL
EC	ELECON	D	T	Y	1.2E-6	UNAVL	286/92-011	Indian Point 3	MULTIPLE EDGs INOP	07/06/92	965	P	W	UE	PNY
HH	CKTBRK	E	O	N	3.6E-6	TRIP	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	873	P	W	UE	CEC
HH	PUMPXX	C	O	Y	3.1E-6	LOFW	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	693	P	W	BX	FPL
HH	INSTRU	E	O	N	5.9E-6	TRIP	344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	1130	P	W	BX	PGC
IB	INSTRU	E	M	Y	2.5E-4	TRIP	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	478	P	C	GH	OPP
IF	ANNUNC	E	T	Y	1.3E-5	UNAVL	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	1171	P	W	BX	UEC
SF	PUMPXX	E	T	Y	3.5E-5	UNAVL	261/92-013	Robinson 2	SI PUMP OOS	07/10/92	700	P	W	EX	CPL
SF	VALVOP	C	O	Y	4.0E-6	TRIP	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	887	P	B	UX	DPC
SF	PUMPXX	E	T	Y	9.9E-6	UNAVL	301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	497	P	W	BX	WEP

Table 6. Precursors Listed by Component

Component	System	O	D	E	C_p Probability	TRANS	Event Identifier	Plant	Description	Event Date	MWE	T	V	AE	Operator
ANNUNC	IF	E	T	Y	1.3E-5	UNAVL	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	1171	P	W	BX	UEC
CKTBRK	HH	E	O	N	3.6E-6	TRIP	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	873	P	W	UE	CEC
ELECON	EA	E	O	N	7.1E-5	LOOP	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	650	B	G	BG	GPU
ELECON	EA	G	O	N	1.6E-4	LOOP	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL
ELECON	EA	G	O	N	1.6E-4	LOOP	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL
ELECON	EA	E	O	N	2.8E-6	UNAVL	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
ELECON	EA	E	O	N	2.8E-6	UNAVL	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
ELECON	EA	E	O	N	2.8E-6	UNAVL	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
ELECON	EA	E	T	N	3.2E-5	UNAVL	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
ELECON	EA	E	T	N	3.2E-5	UNAVL	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
ELECON	EA	E	T	N	3.2E 5	UNAVL	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
ELECON	EA	E	M	Y	2.1E-4	LOOP	270/92-004	Oconee 2	LOOP WITH FAILED EP	10/19/92	887	P	B	UX	DPC
ELECON	EC	D	T	Y	1.2E-6	UNAVL	286/92-011	Indian Point 3	MULTIPLE EDGs INOP	07/06/92	965	P	W	UE	PNY
ELECON	EA	E	M	Y	1.7E-5	LOOP	302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	825	P	B	GX	FPC
ELECON	EA	E	T	Y	1.8E-4	LOOP	327/92-027	Sequoyah 1	LOOP	12/31/92	1148	P	W	UX	TVA
ELECON	EA	E	T	Y	1.8E-4	LOOP	327/92-027	Sequoyah 2	LOOP	12/31/92	1148	P	W	UX	TVA
ELECON	EA	E	T	Y	6.6E-6	UNAVL	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	1050	B	G	BX	PPL
INSTRU	IB	E	M	Y	2.5E-4	TRIP	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	478	P	C	GH	OPP
INSTRU	HH	E	O	N	5.9E-6	TRIP	344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	1130	P	W	BX	PGC
PUMPXX	HH	C	O	Y	3.1E-6	LOFW	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	693	P	W	BX	FPL
PUMPXX	SF	E	T	Y	3.5E-5	UNAVL	261/92-013	Robinson 2	SI PUMP OOS	07/10/92	700	P	W	EX	CPL
PUMPXX	SF	E	T	Y	9.9E-6	UNAVL	301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	497	P	W	BX	WEP
RELAYX	EA	E	O	N	2.1E-4	LOOP	261/92-017	Robinson 2	LOOP	08/22/92	700	P	W	EX	CPL
VALVOP	CC	E	O	N	6.9E-6	TRIP	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	789	B	G	SL	CWE
VALVOP	SF	C	O	Y	4.0E-6	TRIP	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	887	P	B	UX	DPC
VALVOP	CF	E	T	Y	1.9E-6	UNAVL	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	1148	P	W	UX	TVA
VALVOP	CE	F	O	N	6.1E-6	TRIP	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	1078	B	G	SL	CWE

Table 7. Precursors Listed by Operating Status

O	System	Component	D	E	C_p Probability	TRANS	Event Identifier	Plant	Description	Event Date	MWE	T	V	AE	Operator
C	HH	PUMPXX	O	Y	3.1E-6	LOFW	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	693	P	W	BX	FPL
C	SF	VALVOP	O	Y	4.0E-6	TRIP	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	887	P	B	UX	DPC
D	EC	ELECON	T	Y	1.2E-6	UNAVL	286/92-011	Indian Point 3	MULTIPLE EDGs INOP	07/06/92	965	P	W	UE	PNY
E	EA	ELECON	O	N	7.1E-5	LOOP	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	650	B	G	BG	GPU
E	HH	CKTBRK	O	N	3.6E-6	TRIP	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	873	P	W	UE	CEC
E	CC	VALVOP	O	N	6.9E-6	TRIP	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	789	B	G	SL	CWE
E	SF	PUMPXX	T	Y	3.5E-5	UNAVL	261/92-013	Robinson 2	SI PUMP OOS	07/10/92	700	P	W	EX	CPL
E	EA	RELAYX	O	N	2.1E-4	LOOP	261/92-017	Robinson 2	LOOP	08/22/92	700	P	W	EX	CPL
E	EA	ELECON	O	N	2.8E-6	UNAVL	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
E	EA	ELECON	O	N	2.8E-6	UNAVL	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
E	EA	ELECON	O	N	2.8E-6	UNAVL	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
E	EA	ELECON	T	N	3.2E-5	UNAVL	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
E	EA	ELECON	T	N	3.2E-5	UNAVL	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
E	EA	ELECON	T	N	3.2E-5	UNAVL	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
E	EA	ELECON	M	Y	2.1E-4	LOOP	270/92-004	Oconee 2	LOOP WITH FAILED EP	10/19/92	887	P	B	UX	DPC
E	IB	INSTRU	M	Y	2.5E-4	TRIP	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	478	P	C	GH	OPP
E	SF	PUMPXX	T	Y	9.9E-6	UNAVL	301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	497	P	W	BX	WEP
E	EA	ELECON	M	Y	1.7E-5	LOOP	302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	825	P	B	GX	FPC
E	EA	ELECON	T	Y	1.8E-4	LOOP	327/92-027	Sequoyah 1	LOOP	12/31/92	1148	P	W	UX	TVA
E	EA	ELECON	T	Y	1.8E-4	LOOP	327/92-027	Sequoyah 2	LOOP	12/31/92	1148	P	W	UX	TVA
E	CF	VALVOP	T	Y	1.9E-6	UNAVL	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	1148	P	W	UX	TVA
E	HH	INSTRU	O	N	5.9E-6	TRIP	344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	1130	P	W	BX	PGC
E	EA	ELECON	T	N	6.6E-6	TRIP	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	1050	B	G	BX	PPL
E	IF	ANNUNC	T	Y	1.3E-5	UNAVL	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	1171	P	W	BX	UEC
F	CE	VALVOP	O	N	6.1E-6	TRIP	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	1078	B	G	SL	CWE
G	EA	ELECON	O	N	1.6E-4	LOOP	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL
G	EA	ELECON	O	N	1.6E-4	LOOP	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL

Table 8. Precursors Listed by Discovery Method

D	System	Component	O	E	C_p Probability	TRANS	Event Identifier	Plant	Description	Event Date	MWE	T	V	AE	Operator
M	EA	ELECON	E	Y	2.1E-4	LOOP	270/92-004	Oconee 2	LOOP WITH FAILED EP	10/19/92	887	P	B	UX	DPC
M	IB	INSTRU	E	Y	2.5E-4	TRIP	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	478	P	C	GH	OPP
M	EA	ELECON	E	Y	1.7E-5	LOOP	302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	825	P	B	GX	FPC
O	EA	ELECON	E	N	7.1E-5	LOOP	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	650	B	G	BG	GPC
O	HH	CKTBRK	E	N	3.6E-6	TRIP	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	873	P	W	UE	CEC
O	EA	ELECON	G	N	1.6E-4	LOOP	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL
O	HH	PUMPXX	C	Y	3.6E-6	LOFW	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	693	P	W	BX	FPL
O	EA	ELECON	G	N	1.6E-4	LOOP	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	08/24/92	693	P	W	BK	FPL
O	CC	VALVOP	E	N	6.9E-6	TRIP	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	789	B	G	SL	CWE
O	EA	RELAYX	E	N	2.1E-4	LOOP	261/92-017	Robinson 2	LOOP	08/22/92	700	P	W	EX	CPL
O	SF	VALVOP	C	Y	4.0E-6	TRIP	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	887	P	B	UX	DPC
O	EA	ELECON	E	N	2.8E-6	UNAVL	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
O	EA	ELECON	E	N	2.8E-6	UNAVL	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
O	EA	ELECON	E	N	2.8E-6	UNAVL	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	887	P	B	UX	DPC
O	HH	INSTRU	E	N	5.9E-6	TRIP	344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	1130	P	W	BX	PGC
O	CE	VALVOP	F	N	6.1E-6	TRIP	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	1078	B	G	SL	CWE
T	SF	PUMPXX	E	Y	[illegible].5E-5	UNAVL	261/92-015	Robinson 2	SI PUMP OOS	07/10/92	700	P	W	EX	CPL
T	EA	ELECON	E	N	3.2E-5	UNAVL	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
T	EA	ELECON	E	N	3.2E-5	UNAVL	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
T	EA	ELECON	E	N	3.2E-5	UNAVL	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	887	P	B	UX	DPC
T	EC	ELECON	D	Y	1.2E-6	UNAVL	286/92-011	Indian Point 3	MULTIPLE EDGs INOP	07/06/92	965	P	W	UE	PNY
T	SF	PUMPXX	E	Y	9.9E-6	UNAVL	301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	497	P	W	BX	WEP
T	EA	ELECON	E	Y	1.8E-4	LOOP	327/92-027	Sequoyah 1	LOOP	12/31/92	1148	P	W	UX	TVA
T	EA	ELECON	E	Y	1.8E-4	LOOP	327/92-027	Sequoyah 2	LOOP	12/31/92	1148	P	W	UX	TVA
T	CF	VALVOP	E	Y	1.9E-6	UNAVL	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	1148	P	W	UX	TVA
T	EA	ELECON	E	N	6.6E-6	TRIP	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	1050	B	G	BX	PPL
T	IF	ANNUNC	E	Y	1.3E-5	UNAVL	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	1171	P	W	BX	UEC

Table 9. Precursors Listed by Conditional Core Damage Probability

C_p Probability	TRANS	Event Identifier	Plant	Description	Event Date	System	Component	O	D	E	MWE	T	V	AE	Operator
2.5E-4	TRIP	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	IB	INSTRU	E	M	Y	478	P	C	GH	OPP
2.1E-4	LOOP	261/92-017	Robinson 2	LOOP	08/22/92	EA	RELAYX	E	O	N	700	P	W	EX	CPL
2.1E-4	LOOP	270/92-004	Oconee 2	LOOP WITH FAILED EP	10/19/92	EA	ELECON	E	M	Y	887	P	B	UX	DPC
1.8E-4	LOOP	327/92-027	Sequoyah 1	LOOP	12/31/92	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
1.8E-4	LOOP	327/92-027	Sequoyah 2	LOOP	12/31/92	EA	ELECON	E	T	Y	1148	P	W	UX	TVA
1.6E-4	LOOP	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	08/24/92	EA	ELECON	G	O	N	693	P	W	BK	FPL
1.6E-4	LOOP	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	08/24/92	EA	ELECON	G	O	N	693	P	W	BK	FPL
7.1E-5	LOOP	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	EA	ELECON	E	O	N	650	B	G	BG	GPU
3.5E-5	UNAVL	261/92-013	Robinson 2	SI PUMP OOS	07/10/92	SF	PUMPXX	E	T	Y	700	P	W	EX	CPL
3.2E-5	UNAVL	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	EA	ELECON	E	T	N	887	P	B	UX	DPC
3.2E-5	UNAVL	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	EA	ELECON	E	T	N	887	P	B	UX	DPC
3.2E-5	UNAVL	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	EA	ELECON	E	T	N	887	P	B	UX	DPC
1.7E-5	LOOP	302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	EA	ELECON	E	M	Y	825	P	B	GX	FPC
1.3E-5	UNAVL	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	IF	ANNUNC	E	T	Y	1171	P	W	BX	UEC
9.9E-6	UNAVL	301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	SF	PUMPXX	E	T	Y	497	P	W	BX	WEP
6.9E-6	TRIP	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	CC	VALVOP	E	[illegible]	N	789	B	G	SL	CWE
6.6E-6	TRIP	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/1[illegible]/92	EA	ELECON	E	T	N	1050	B	G	BX	PPL
6.1E-6	TRIP	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	CE	VALVOP	F	O	N	1078	B	G	SL	CWE
5.9E-6	TRIP	344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	HH	INSTRU	E	O	N	1130	P	W	BX	PGC
4.0E-6	TRIP	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	SF	VALVOP	C	O	Y	887	P	B	UX	DPC
3.6E-6	TRIP	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	HH	CKTBRK	E	O	N	873	P	W	UE	CEC
3.1E-6	LOFW	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	HH	PUMPXX	C	O	Y	693	P	W	BX	FPL
2.8E-6	UNAVL	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	EA	ELECON	E	O	N	887	P	B	UX	DPC
2.8E-6	UNAVL	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	EA	ELECON	E	O	N	887	P	B	UX	DPC
2.8E-6	UNAVL	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	EA	ELECON	E	O	N	887	P	B	UX	DPC
1.9E-6	UNAVL	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	CF	VALVOP	E	T	Y	1148	P	W	UX	TVA
1.2E-6	UNAVL	286/92-011	Indian Point 3	MULTIPLE EDGs INOP	07/06/92	EC	ELECON	D	T	Y	965	P	W	UE	PNY

Table 10. Precursors Listed by Plant Type and Vendor

T	V	MWE	Event Identifier	Plant	Description	Event Date	C_p Probability	TRANS	System	Component	O	D	E	AE	Operator
B	G	650	219/92-005	Oyster Creek	LOOP DUE TO FOREST FIRE	05/03/92	7.1E-5	LOOP	EA	ELECON	E	O	N	BG	GPU
B	G	789	254/92-004	Quad Cities 1	RX TRIP WITH HPCI & ONE SRV UNAVAIL	02/06/92	6.9E-6	TRIP	CC	VALVOP	E	O	N	SL	CWE
B	G	1078	374/92-012	La Salle 2	RX TRIP WITH DEGRADED RCIC	08/27/92	6.1E-5	TRIP	CE	VALVOP	F	O	N	SL	CWE
B	G	1050	388/92-001	Susquehanna 2	RX TRIP WITH EDG & VITAL BUS UNAVAIL	03/18/92	6.6E-6	TRIP	EA	ELECON	E	T	N	BX	PPL
P	W	873	247/92-007	Indian Point 2	RX TRIP & AFW PUMP PROBLEMS	04/13/92	3.6E-6	TRIP	HH	CKTBRK	E	O	N	UE	CEC
P	W	693	250/92-SO1	Turkey Point 3	LOOP DUE TO HURRICANE ANDREW	08/24/92	1.6E-4	LOOP	EA	ELECON	G	O	N	BK	FPL
P	W	693	251/92-007	Turkey Point 4	MFW PUMP TRIP WITH ONE AFW PUMP OOS	09/29/92	3.1E-6	LOFW	HH	PUMPXX	C	O	Y	BX	FPL
P	W	693	251/92-SO1	Turkey Point 4	LOOP DUE TO HURRICANE ANDREW	08/24/92	1.6E-4	LOOP	EA	ELECON	G	O	N	BK	FPL
P	W	700	261/92-013	Robinson 2	SI PUMP OOS	07/10/92	3.5E-5	UNAVL	SF	PUMPXX	E	T	Y	EX	CPL
P	W	700	261/92-017	Robinson 2	LOOP	08/22/92	2.1E-4	LOOP	EA	RELAYX	E	O	N	EX	CPL
P	B	887	269/92-004	Oconee 1	RX TRIP WITH ONE EFW TRAIN INOP	05/08/92	4.0E-6	TRIP	SF	VALVOP	C	O	Y	UX	DPC
P	B	887	269/92-008	Oconee 1	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	UX	DPC
P	B	887	269/92-008	Oconee 2	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	UX	DPC
P	B	887	269/92-008	Oconee 3	BOTH KEOWEE UNITS UNAVAIL	07/17/92	2.8E-6	UNAVL	EA	ELECON	E	O	N	UX	DPC
P	B	887	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	UX	DPC
P	B	887	269/92-018	Oconee 2	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	UX	DPC
P	B	887	269/92-018	Oconee 3	BOTH KEOWEE UNITS POTENT UNAVAIL	12/02/92	3.2E-5	UNAVL	EA	ELECON	E	T	N	UX	DPC
P	B	887	270/92-004	Oconee 2	LOOP WITH FAILED EP	10/19/92	2.1E-4	LOOP	EA	ELECON	E	M	Y	UX	DPC
P	B	825	302/92-001	Crystal River 3	LOOP WITH INOP VITAL BUS INVERTER	03/27/92	1.7E-5	LOOP	EA	ELECON	E	M	Y	GX	FPC
P	C	478	285/92-023	Fort Calhoun 1	RX TRIP WITH FAULTY PSV	07/03/92	2.5E-4	TRIP	IB	INSTRU	E	M	Y	GH	OPP
P	W	965	286/92-011	Indian Point 3	MULTIPLE EDGs INOP	07/06/92	1.2E-6	UNAVL	EC	ELECON	D	T	Y	UE	PNY
P	W	497	301/92-003	Point Beach 2	PLUGGED SI PUMP SUCTION	09/18/92	9.9E-6	UNAVL	SF	PUMPXX	E	T	Y	BX	WEP
P	W	1148	327/92-027	Sequoyah 1	LOOP	12/31/92	1.8E-4	LOOP	EA	ELECON	E	T	Y	UX	TVA
P	W	1148	327/92-027	Sequoyah 2	LOOP	12/31/92	1.8E-4	LOOP	EA	ELECON	E	T	Y	UX	TVA
P	W	1148	328/92-010	Sequoyah 2	EDG & RHR PUMP INOP	07/17/92	1.9E-6	UNAVL	CF	VALVOP	E	T	Y	UX	TVA
P	W	1130	344/92-020	Trojan	RX TRIP & AFW PUMP FAIL TO START	07/22/92	5.9E-6	TRIP	HH	INSTRU	E	O	N	BX	PGC
P	W	1171	483/92-011	Callaway	LOSS OF MN CONT BOARD ANNUNCIATORS	10/17/92	1.3E-5	UNAVL	IF	ANNUNC	E	T	Y	BX	UEC

Table 11.a Abbreviations Used in Precursor Lists

Event Identifier	Docket Number/Licensee Event Report Number
Plant	Name of Plant and Unit Number
Description	Description of Event
Event Date	Event Date
C_D Probability	Conditional core damage probability
TRANS	Event initiator or unavailability (see Table 11.b)
System	System Abbreviation (see Table 11.c)
COMP	System Component Code (see Table 11.d)
O	Plant Operating Status (see Table 11.e)
D	Discovery Method (O-operational event, T-testing, M-maintenance)
E	Human error involved (N-no, Y-yes)
MWE	Plant electrical rating in megawatts electric
T	Plant type (B - boiling-water reactor, P - pressurized-water reactor)
V	Plant NSS vendor: A - Allis Chalmers B - Babcock and Wilcox C - Combustion Engineering G - General Electric W - Westinghouse
AE:	Plant architect engineer: AE - American Electric Power BR - Burns and Roe BX - Bechtel EX - Ebasco FP - Fluor Power GH - Gibbs and Hill GX - Gilbert PX - Pioneer RT - Brown and Root SL - Sargent and Lundy SS - Southern Services SW - Stone and Webster UE - United Engineers UX - Utility XX - Other
Operator	Plant licensee abbreviations (see Table 11.f)

Table 11.b Event initiator or unavailability abbreviations

ECIT	-	Excessive coolant inventory
EQUK	-	Earthquake
INAA	-	Inadvertent automatic depressurization system actuation
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
UNAVL	-	System(s) unavailable
UNIQ	-	Unique sequence

Table 11.c System Codes and Abbreviations

	Reactor
RA	Reactor vessel internals
RB	Reactivity control systems
RC	Reactor core
	Reactor Coolant System and Connected Systems
CA	Reactor vessels and appurtenances
CB	Coolant recirculation systems and controls
CC	Main steam systems and controls
CD	Main steam isolation systems and controls
CE	Reactor core isolation cooling systems and controls
CF	Residual heat removal systems and controls
CG	Reactor coolant cleanup and systems and controls
CH	Feedwater systems and controls
CI	Reactor coolant pressure boundary leakage detection systems
CJ	Other coolant subsystems and their controls
	Engineered Safety Features
SA	Reactor containment systems
SB	Containment heat removal systems and controls
SC	Containment air purification and cleanup systems and controls
SD	Containment isolation systems and controls
SE	Containment combustible gas control systems and controls
SF	Emergency core cooling systems and controls
SG	Control room habitability systems and controls
SM	Other engineered safety feature systems and their controls
	Instrumentation and Controls
IA	Reactor trip systems
IB	Engineered safety feature instrument systems
IC	Systems required for safe shutdown
ID	Safety related display instrumentation
IE	Other instrument systems required for safety
IF	Other instrument systems no required for safety
	Electric Power Systems
EA	Offsite power systems and controls
EB	AC onsite power systems and controls
EC	DC onsite power systems and controls
ED	Onsite power systems and controls (composite AC and DC)
EE	Emergency generator systems and controls
EF	Emergency lighting systems and controls
EG	Other electrical power systems and controls
	Fuel Storage and Handling Systems
FA	New fuel storage facilities
FB	Spent fuel storage facilities
FC	Spent fuel pool cooling and cleanup systems and controls
FD	Fuel handling systems

Table 11.c System Codes and Abbreviations

	Auxiliary Water Systems
WA	Station service water systems and controls
WB	Cooling systems for reactor auxiliaries and controls
WC	Demineralized water make-up systems and controls
WD	Potable and sanitary water systems and controls
WE	Ultimate heat sink facilities
WF	Condensate storage facilities
WG	Other auxiliary water systems and their controls
	Auxiliary Process Systems
PA	Compressed air systems and controls
PB	Process sampling systems
PC	Chemical, volume control and liquid poison systems and controls
PD	Failed fuel detection systems
PE	Other auxiliary process systems and controls
	Other Auxiliary Systems
AA	Air conditioning, heating, cooling and ventilation systems and controls
AB	Fire protection systems and controls
AC	Communication systems
AD	Other auxiliary systems and their controls
	Steam and Power Conversion Systems
HA	Turbine-generators and controls
HB	Main steam supply systems and controls (other than CC)
HC	Main condenser systems and controls
HD	Turbine gland sealing systems and controls
HE	Turbine bypass systems and controls
HF	Circulating water systems and controls
HG	Condensate cleanup systems and controls
HH	Condensate and feedwater systems and controls (other than CH)
HI	Steam generator blowdown systems and controls
HJ	Other features of steam and power conversion systems (not included elsewhere)
	Radioactive Waste Management Systems
MA	Liquid radioactive waste management systems
MB	Gaseous radioactive waste management systems
MC	Process and effluent radiological monitoring systems
MD	Solid radioactive waste management systems
	Radiation Protection Systems
BA	Area monitoring systems
BB	Airborne radioactivity monitoring systems
XX	Other Systems
ZZ	System code not applicable

Table 11.d System Component Codes

Component Type	Component Code	Includes:
Accumulator	ACCUMU	Scram accumulators, Safety injection tanks, Surge tanks, Holdup/storage tanks
Air dryers	AIRDRY	
Annunciator modules	ANNUNC	Alarms, Buzzers, Claxons, Horns, Gongs, Sirens
Batteries and chargers	BATTRY	Chargers, Dry cells, Wet cells, Storage cells
Blowers	BLOWER	Compressors, Gas circulators, Fans, Ventilators
Circuit closers/interrupters	CKTBRK	Circuit breakers, Contactors, Controllers, Starters, Switches (other than sensors), Switchgear
Control drive mechanisms	CRDRVE	
Control rods	CONROD	Poison curtains
Demineralizers	DEMINX	Ion exchangers
Electrical conductions	ELECON	Bus, Cable, Wire
Electric motors	MOTORX	Valves, Hydraulic motors, Pneumatic (air) motors, Servo motors
Engines, internal combustion	ENGINE	Diesel, Gasoline, Natural gas, and Propane engines, Strainers, Screens
Filters	FILTER	
Fuel elements	FUELXX	
Generators	GENERA	Inverters
Heaters, electric	HEATER	Heat tracing
Heat exchangers	HTEXCH	Condensers, Coolers, Evaporators, Regenerative heat exchangers, Steam generators, Fan coil units
Instrumentation and controls	INSTRU	Controllers, Sensors/detectors/elements, Indicators, Differentials, Integrators (totalizers), Power supplies, Recorders, Switches, Transmitters, Computation modules
Mechanical function units	MECFUN	Mechanical controllers, Governors, Gear boxes, Varidrives
Penetrations, primary contain.	PENETR	Air locks, Personnel access, Fuel handling, Equipment access, Electrical, Instrument line, Process piping
Pipes, fittings	PIPEXX	Other components, (XXXXXX)
Pumps	PUMPXX	Codes not applicable, (ZZZZZZ)
Recombiners	RECOM	
Relays	RELAYX	Switchgear
Shock suppressors and support	SUPORT	Hangers, Supports, Sway braces/stabilizers, Snubbers, Anti-vibration devices
Transformers	TRANSF	
Turbines	TURBIN	Steam, Gas, and Hydro turbines
Valves	VALVEX	Dampers
Valve Operators	VALVOP	Explosive, Squib
Vessels, pressure	VESSEL	Containment vessels, Drywells, Pressure suppression, Pressurizers, Reactor vessels

Table 11.e Plant Operating Status

Code	Status
B	Startup or power ascension tests (in progress)
C	Routine startup
D	Routine shutdown
E	Steady state operation
F	Load changes during routine power operation
G	Shutdown (hot or cold) except for refueling
H	Refueling
X	Other
Z	Unknown/not applicable

Table 11.f Plant licensee abbreviations

Licensee Abbrev.	Licensee	Licensee Abbrev.	Licensee
APC	Alabama Power Company	NNE	Northeast Nuclear Energy Company
APL	Arkansas Power and Light Company	NPC	Northern Indiana Public Service Company
APS	Arizona Public Service Company	NPP	Nebraska Public Power District
BEC	Boston Edison Company	NSP	Northern States Power Company
BGE	Baltimore Gas and Electric Company	OEC	Ohio Edison Company
CEC	Consolidated Edison Company	OPP	Omaha Public Power District
CEI	Cleveland Electric Illuminating Company	PEC	Philadelphia Electric Company
CGE	Cincinnati Gas and Electric Company	PEG	Public Service Electric & Gas Company
COY	Connecticut Yankee Atomic Power & Light Company	PEP	Potomac Electric Power Company
CPC	Consumers Power Company	PGC	Portland General Electric Company
CPL	Carolina Power and Light Company	PGE	Pacific Gas and Electric Company
CWE	Commonwealth Edison Company	PNY	New York Power Authority
DEC	Detroit Edison Company	PPL	Pennsylvania Power and Light Company
DLP	Dairyland Power Corporation	PSC	Public Service Company of Colorado
DPC	Duke Power Company	PSI	Public Service of Indiana
DUQ	Duquesne Light Company	PSN	Public Service of New Hampshire
FPC	Florida Power Corporation	PSO	Public Service of Oklahoma
FPL	Florida Power and Light Company	PUG	Puget Sound Power and Light Company
GPC	Georgia Power Company	RGE	Rochester Gas & Electric Corporation
GSU	Gulf States Utilities	SCC	South Carolina Electric & Gas Company
HLP	Houston Lighting and Power Company	SCE	Southern California Edison Company
IEL	Iowa Electric Light and Power Company	SMU	Sacramento Municipal Utilities District
IME	Indiana and Michigan Electric Company	TEC	Toledo Edison Company
IPC	Illinois Power Company	TUG	Texas Utilities Generating Company
JCP	Jersey Central Power and Light Company	TVA	Tennessee Valley Authority
KGE	Kansas Gas and Electric Company	UEC	Union Electric Company
LIL	Long Island Lighting Company	VEP	Virginia Electric and Power Company
LPL	Louisiana Power and Light Company	VYC	Vermont Yankee Nuclear Power Corp.
MEC	Metropolitan Edison Company	WEP	Wisconsin Electric Power Company
MPL	Mississippi Power and Light Company	WMP	Wisconsin-Michigan Power Company
MYA	Maine Yankee Atomic Power Company	WPP	Washington Public Power supply System
NEP	New England Power Company	WPS	Wisconsin Public Service Corporation
NMP	Niagara Mohawk Power Corporation	YAE	Yankee Atomic Electric Company

3.0 RESULTS

This chapter summarizes results of the 1992 effort. The primary result of the ASP Program for 1992 is the identification of operational events satisfying 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 LOFW with a degraded mitigating system. These events are documented in Appendix B. Twenty-seven such events were identified for 1992.

Because of (1) the consideration of additional equipment and procedures (beyond those addressed in the ASP models described in Appendix A) in the analysis of 1992 events, (2) changes in the models used in the analysis of 1988−91 events from those used in 1984−87 analyses, and (3) the evaluation of only a portion of 1988−92 LERs by the project team (as described in Sect. 2.1), comparison of results with those of earlier years is not possible without substantial effort to reconcile analysis differences. Because of this, only limited observations are provided here. Refer to the 1986 precursor report[1] for a discussion of observations for 1984−86 and to the 1987−91[2−6] reports for observations for those years.

To "count" precursors, certain conventions have been followed. The following examples clarify the counting process. Four events occurred in 1992 that affected more than one plant. The first event was at Sequoyah (Precursor 327/92-027) and caused a LOOP at both Unit 1 and Unit 2. This event is listed as two precursors because each plant experienced the LOOP and the ASP Program is not able to analyze dual plant trips because the ASP models do not account for systems that are cross-tied between plants. Similarly, Hurricane Andrew caused a LOOP at both Turkey Point 3 & 4 (Precursors 350/92-SO1 and 251/92-SO1) which again "counts" as two precursors. The other events occurred at Oconee (Precursor 269/92-008 and 269/92-018); since all three Oconee units were susceptible to a system unavailability, both of these events were listed as three precursors, one for each Oconee unit. In other instances, there were multiple LERs at one plant that were analyzed as one precursor (e.g., Precursor 302/92-001); however, there were instances of multiple events at one plant that were analyzed as a group, and multiple precursors emerged. For example, four events occurred at Robinson over a period of one month that were examined individually as well as collectively. The results of this study indicated two separate precursors (Precursors 261/92-013 and 261/92-017) had occurred at Robinson.

3.1 Important Precursors

Seven precursors with conditional core damage probabilities equal to or greater than 10^{-4} were identified for 1992. Events with such conditional probabilities have traditionally been considered significant in the ASP Program. For 1992, these events include:

Fort Calhoun (LER 285/92-023)

> Fort Calhoun tripped from 100% power on July 3, 1992. The reactor tripped on high pressure following the closure of all turbine control valves. Two pressurizer power-operated relief valves (PORVs) and one pressurizer safety valve opened to relieve RCS pressure. After an initial pressure decrease in the RCS, the safety valve opened again. When RCS pressure reached 1000 psia, the valve closed but continued to leak.

Robinson (LER 261/92-017)

On August 22, 1992, with the plant operating at 100% power, the loss of the startup transformer resulted in loss of one of the two emergency buses and an instrument bus. Following a subsequent reactor/turbine trip, the transfer of the other emergency bus to offsite power failed and resulted in a total LOOP.

Oconee 2 (LER 270/92-004)

Use of a poorly designed procedure for switchyard battery replacement resulted in a lockout of the Oconee 230-kV switchyard, a reactor trip, and a LOOP at Unit 2, and unavailability of power to the startup transformers for Units 1 and 3. An operator error at the Keowee Hydro Station, the emergency power source for the three Oconee units, caused a loss of all auxiliary power to both hydro units. Auxiliary power was recovered 0.5 h later. Problems were also experienced with the emergency feedwater (EFW) system because of water in the turbine-driven pump steam line.

Sequoyah 1 (LER 327/92-027)

Shortly after a switchyard breaker was installed, it faulted and caused an undervoltage condition in the switchyard. This resulted in the tripping of Sequoyah 1 from 100% power on LOOP. Because of the momentary undervoltage condition on the safeguards buses, the EDGs started and loaded.

Sequoyah 2 (LER 327/92-027)

Shortly after a switchyard breaker was installed, it faulted and caused an undervoltage condition in the switchyard. This resulted in the tripping of Sequoyah 2 from 100% power on LOOP. Because of the momentary undervoltage condition on the safeguards buses, the EDGs started and loaded.

Turkey Point 3 (LER 250/92-SO1)

On August 24, 1992, Hurricane Andrew struck Turkey Point 3. The storm caused a LOOP which required the use of the emergency diesel generators (EDGs) for 6.5 d. The plant had been shut down prior to the arrival of the storm. Damage to non-class 1 structures and equipment, including the offsite power supplies, offsite communications, on-site electrical distribution systems, fire protection system, and miscellaneous plant structures, complicated the recovery from the event.

Turkey Point 4 (LER 251/92-SO1)

On August 24, 1992, Hurricane Andrew struck Turkey Point 4. The storm caused a LOOP which required the use of the emergency diesel generators (EDGs) for 6.5 d. The plant had been shut down prior to the arrival of the storm. Damage to non-class 1 structures and equipment, including the offsite power supplies, offsite communications, on-site electrical distribution systems, fire protection system, and miscellaneous plant structures, complicated the recovery from the event.

3.2 Number of Precursors Identified

Twenty-seven precursors [p(core damage) $\geq 10^{-6}$] were identified in 1992. The distribution of precursors as a function of conditional probability is shown in Table 12. This distribution compares as follows with events identified in 1988-91:

Number of Precursors

	$10^{-4} \leq p(cd) \leq 1$	$10^{-5} \leq p(cd) < 10^{-4}$	$10^{-6} \leq p(cd) < 10^{-5}$
1988	7	14	11
1989	7	11	12
1990	6	11	11
1991	13	8	6
1992	7	7	13

Table 12. Precursors for 1992 ranked by order of magnitude

Conditional probability range	Events ranked by conditional probability of subsequent core damage
10^{-1} to 1	None
10^{-2} to 10^{-1}	None
10^{-3} to 10^{-2}	None
10^{-4} to 10^{-3}	Reactor trip on high pressure at Fort Calhoun with two pressurizer power-operated relief valves and one pressurized safety valve opening. The safety valve opened twice and failed to reseat properly (285/92-023).
	LOOP at Robinson with one SI pump inoperable (261/92-017).
	LOOP at Oconee 2 and loss of all auxiliary power to both Keowee Hydro Station units (270/92-004).
	LOOP at Sequoyah 1 (327/92-027).
	LOOP at Sequoyah 2 (327/92-027).
	LOOP at Turkey Point 3 due to Hurricane Andrew. Plant was at shutdown and required the use of EDGs for 6.5 d (250/92-SO1).
	LOOP at Turkey Point 4 due to Hurricane Andrew. Plant was at shutdown and required the use of EDGs for 6.5 d (251/92-SO1).

Table 12. Precursors for 1992 ranked by order of magnitude

Conditional probability range	Events ranked by conditional probability of subsequent core damage
10^{-5} to 10^{-4}	7 events
10^{-6} to 10^{-5}	13 events

As can be seen in Table 12, all seven precursors with p(cd) $\geq 10^{-4}$ selected for 1992 are PWR events. This is similar to the results for 1988-91, where almost all of the more significant events occurred at PWRs. For all 1992 precursors, four were associated with BWRs and 23 with PWRs.

3.3 Likely Sequences

Precursors with conditional probabilities of $\geq 10^{-4}$ that were identified for 1992 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, beyond the operational event, required for core damage. For the events that occurred or could have occurred at power and with core damage probabilities $\geq 10^{-4}$, the following dominant core damage sequences were identified:

PWRs Small-break LOCA with failure of HPI

LOOP with failure of emergency power and failure to recover ac power prior to battery depletion

LOOP with failure to recover emergency power, failure to utilize the SSF for RCS and SG makeup, and failure to recover ac power before battery depletion

Postulated failure of emergency power, failure to load the DSDG, and failure to restore ac power prior to core uncovery

Failure of emergency power restoration resulting in an RCP seal LOCA.

3.4 References

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2. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp.; and Professional Analysis, Inc.; *Precursors to Potential Severe Core Damage Accidents: 1987, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 7 and 8), July 1989.*

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3. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp.; and Professional Analysis, Inc.; *Precursors to Potential Severe Core Damage Accidents: 1998, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 9 and 10), February 1990.*
4. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp.; and Professional Analysis, Inc.; *Precursors to Potential Severe Core Damage Accidents: 1989, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 11 and 12), September 1990.*
5. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp.; and Professional Analysis, Inc.; *Precursors to Potential Severe Core Damage Accidents: 1990, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 13 and 14), August 1991.*
6. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp.; and Professional Analysis, Inc.; *Precursors to Potential Severe Core Damage Accidents: 1991, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 15 and 16), August 1992.*

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

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

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

Event sequence. A particular path on an event tree.

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

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

APPENDIX A. ASP MODELS

A. ASP MODELS

This appendix provides information concerning the methods and models used to estimate event significance in the ASP Program. The basic models used in the analysis of 1992 precursors are the same as those used for 1989-91 precursors. However, the analysis of 1992 precursors considered the potential use of alternate equipment and procedures, beyond that addressed in the basic models, that recently have been added by the licensees to provide additional protection against core damage, if information regarding this equipment was available. This equipment is described in Sect. A.3.

A.1 Precursor Significance Estimation

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 event trees depicting 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. In the quantification processes, it is assumed that the event tree branch failure probabilities for systems observed failed during an event are equal to the likelihood of not recovering from the failure or fault that actually occurred. Event tree branch failure probabilities for systems observed degraded during an operational event are assumed equal to the conditional probability that the system would fail (given that it was observed degraded) and the probability that it would not be recovered within the required time period. Event tree branch failure probabilities used for systems observed to be successful and systems unchallenged during the actual occurrence are assumed equal to a failure probability estimated from either system failure data (when available) or by the use of system success criteria and typical train and common-mode failure probabilities. The conditional probability estimated for each precursor is useful in ranking because it provides an estimate of the measure of protection against core damage remaining once the observed failures have occurred.

A.1.1 ASP Event Tree Models

Models used to rank precursors as to significance consist of plant-class specific event trees that are linked to simplified plant-specific system models. These models describe mitigation sequences for three initiating events: a nonspecific reactor trip [which includes LOFW within the model], LOOP, and small-break LOCA. The event tree models are system-based and include a model applicable to each of eight plant classes: three for BWRs and five for PWRs.

Plant classes are defined based on the use of similar systems in providing protective functions in response to transients, LOOPs, and small-break LOCAs. System designs and specific nomenclature may differ among plants included in a particular class; but functionally, they are similar in response. Plants where certain mitigating systems do not exist, but which are largely analogous in their initiator response, are grouped into the appropriate plant class. In modeling events at such plants, the event tree branch probabilities are modified to reflect the actual systems available at the plant. For operational events that cannot be described using the plant-class specific event trees, unique models are developed to describe the potential sequences to severe core damage.

Each event tree includes two undesired end states. The undesired end states are designated as (1) core damage (CD), in which inadequate core cooling is believed to exist; and (2) ATWS, for the failure-to-scram sequence. The end states are distinct; sequences associated with ATWS are not subsets of core damage sequences. The ATWS sequence, if fully developed, would consist of a number of sequences ending in either success or core damage. Successful operation is designated "OK" in the event trees included in this appendix.

A.1.2 Precursor Impact on Event Tree Branches

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. This translation process is simplified in many cases through the use of train-based models that represent an event tree branch. If a train-based model exists, then the impact of the operational event need only be determined at the train level, and not at the system level.

Once the impact of an operational event on systems included in the ASP event tree models has been determined, branch probability values are modified to reflect the event, and the event trees are then used to estimate a conditional probability of subsequent core damage, given the precursor.

A.1.3 Estimation of Initiating Event Frequencies and Branch Failure Probabilities Used with the Event Tree Models

A set of initiating event frequencies and system failure probabilities was developed for use in the quantification of the event tree models associated with the precursors. The approach used to develop frequency and probability estimates employs failure or initiator data in the precursors themselves when sufficient data exists. When precursor data are available for a system, its failure probability is estimated by counting the effective number of nonrecoverable failures in the observation period, making appropriate demand assumptions, and then calculating the effective number of failures per demand. The number of demands is calculated based on the estimated number of tests per reactor year plus any additional demands to which a system would be expected to respond. This estimate is then multiplied by the number of applicable reactor years in the observation period to determine the total number of demands. A similar approach is employed to estimate initiator frequencies per reactor year from observed initiating events.

The potential for recovery is addressed by assigning a recovery action to each system failure and initiating event. Four classes are currently used to describe the different types of recovery that could be involved:

Recovery class	Likelihood of nonrecovery	Recovery characteristic
R1	1.00	The failure did not appear to be recoverable in the required period, either from the control room or at the failed equipment.
R2	0.34	The failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from the control room did not appear possible.
R3	0.12	The failure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial operator burden.
R4	0.04	The failure appeared recoverable in the required period from the control room and was considered routine and procedurally based.

The assignment of an event to a recovery class is based on engineering judgment, which considers the specifics of each operational event and the likelihood of not recovering from the observed failure in a moderate to high-stress situation following an initiating event. For analysis purposes, consistent probabilities of failing to recover an observed failure are assigned to each event in a particular recovery class. It must be noted that the actual likelihood of failing to recover from an event at a particular plant is difficult to assess and may vary substantially from the values listed. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, etc., concerning the likelihood of recovering specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.*

The branch probability estimation process is illustrated in Table A.1. Table A.1 lists two operational events that occurred in 1984-86 involving failure of SG isolation. For each event, the likelihood of failing to recover from the failure is listed (Column 3). The effective number of nonrecoverable events (1.04 in this case) is then divided by an estimate of the total number of demands in the 1984-86 observation period (1968) to calculate a failure on demand probability of 5.3×10^{-4}.

The likelihood of system failure as a result of hardware faults is combined with the likelihood that the system could not be recovered, if failed, and with an estimate of the likelihood of the operator failing to initiate the system, if manual initiation were required, to estimate the overall failure probability for an event-tree branch. Calculated failure probabilities are then used to tailor the probabilities associated with train-based system models. Such an approach results in system failure probability estimates that reflect, to a certain extent, the degree of redundancy actually available and permits easy revision of these probabilities based on train failures and unavailabilities observed during an operational event.

*Programmatic constraints have prevented substantial efforts in estimating actual recovery class distributions. The values currently used were developed based on a review of events with the potential for short-term recovery, in addition to 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 believed 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 1980-81 report. These calculations demonstrated little impact on the relative ranking of events from variance in recovery class values.

A.1.4 Conditional Probability Associated with Each Precursor

The calculation process for each precursor involves a determination of initiators that must be modeled and their probability, plus any modifications to system probabilities necessitated by failures observed in an operational event. Once the branch probabilities that reflect the conditions of the precursor are established, the sequences leading to the modeled end states (core damage and ATWS) are calculated and summed to produce an estimate of the conditional probability of each end state for the precursor. So that only the additional contribution to risk (incremental risk) associated with a precursor is calculated, conditional probabilities for precursors associated with equipment unavailabilities (during which no initiating event occurred) are calculated a second time using the same initiating event probability but with all branches assigned normal failure probabilities (no failed or degraded states) and subtracted from the initially calculated values. This eliminates the contribution for sequences unimpacted by the precursor, plus the normal risk contribution for impacted sequences during the unavailability. This calculational process is summarized in Table A.2.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the 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 probabilities calculated in the ASP study are homogenized probabilities considered representative of probabilities resulting from the occurrence of the selected events at plants representative of the plant class.

A.1.5 Sample Calculations

Three hypothetical events are used to illustrate the calculational process.

1. The first event assumes a trip and LOFW but no other observed failures during mitigation. An event tree for this event is shown in Fig. A.1. On the event tree, successful operation is indicated by the upper branch and failure by the lower branch. With the exception of relief valve lift, failure probabilities for branches are indicated. For HPI, the lowest branch includes operator action to initiate feed and bleed. Success probabilities are 1 - p(failure). The likelihood of not recovering the initiator (trip) is assumed to be 1.0, and the likelihood of not recovering MFW is assumed to be 0.34 in this example. Systems assumed available were assigned failure probabilities currently used in the ASP Program. The estimated conditional probabilities for undesirable end states associated with the event are then:

$$
\begin{aligned}
p(cd) \quad &= p[\text{seq. } 11] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times (1 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}] \\
&+ p[\text{seq. } 12] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times 8.4 \times 10^{-4}] \\
&+ p[\text{seq. } 13] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times 9.9 \times 10^{-5} \times (1 - 0.34) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times (1.0 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}]
\end{aligned}
$$

$$+ \ p[\text{seq. } 14] + p[\text{seq. } 15] + p[\text{seq. } 16] + p[\text{seq. } 17]$$

$$= 7.7 \times 10^{-7}$$

$$p(\text{ATWS}) = p[\text{seq. } 18]$$

$$= 3.0 \times 10^{-5}$$

2. The second example event involves failures that would prevent HPI if required to mitigate a small-break LOCA or if required for feed and bleed. Assume such failures were discovered during testing. This event impacts mitigation of a small-break LOCA initiator and potentially impacts mitigation of a trip and LOOP, should a transient-induced LOCA occur or should feed and bleed be required upon loss of AFW and MFW. The event tree for a postulated small-break LOCA associated with this example precursor is shown in Fig. A.2. The failure probability associated with the precursor event (unavailability of HPI) is assigned based on the likelihood of not recovering from the failure in a 20-30 min time frame (assumed to be 1.0 in this case). No initiating event occurred with the example precursor; however, a failure duration of 360 h was estimated based on one-half of a monthly test interval. The estimated small-break LOCA frequency (assumed to be 1.0×10^{-6}/h in this example), combined with this failure duration, results in an estimated initiating event probability of 3.6×10^{-4} during the unavailability. The probabilities for small-LOCA sequences involving undesirable end states (employing the same calculational method as above and subtracting the nominal risk during the time interval) are 3.6×10^{-4} for core damage and 0.0 for ATWS. Note that the impact of the postulated failure on the ATWS sequence is zero because HPI success or failure does not impact that sequence as modeled.

   For most unavailabilities, similar calculations would be required using the trip and LOOP event trees, since these postulated initiators could also occur. In this example, neither of these two initiators contributes substantially to the core damage probability associated with the event.

3. The third example event involves a trip with unavailability of one of two trains of service water (SW). Assumed unavailability of the SW train results in unavailability of one train of HPI, high-pressure recirculation (HPR), and AFW, all because of unavailability of cooling to the respective pumps. In this example, SW cooling of two motor-driven AFW pumps is assumed. An additional turbine-driven pump is assumed to be self-cooled. Since SW is not explicitly addressed in the ASP event trees, the probabilities of front-line systems impacted by the loss of SW are instead modified.

   Figure A.3 shows a transient event tree with branch failure probabilities modified to reflect unavailability of one train of service water. The likelihoods of not recovering failed front line systems are assumed to be unchanged, since the failure mechanisms for (observed) non-faulted trains are expected to be consistent with historically observed failures. The conditional probability of core damage given the trip and one service water train unavailable is 1.1×10^{-6}. If the second train of service water were to fail, HPI and HPR (and hence feed and bleed) would be rendered unavailable; however, the turbine-driven AFW pump would still be operable. In this case, the likelihood of not recovering HPI and HPR is assumed to be 1.0 until service water is recovered. Sequences associated with loss of both service water trains increase the core damage probability associated with the event. The extent of this increase is dependent in PWRs on the likelihood of a reactor coolant pump seal failure following the loss of service water (since seal injection and seal cooling would be typically lost). Assuming that the conditional probability of loss of the second service water train is 0.01, that the likelihood of not recovering SW is 0.34, and that the failure probability of the turbine-driven AFW pump is 0.05, the increase in core damage probability is 1.7×10^{-4} if no RCP seal failure occurs, and 3.4×10^{-3} if the likelihood of seal failure is 1.0.

A.1.6 Event Tree Changes Made to 1988–1991 Event Models

Two changes were made to the event trees used in the 1988-91 precursor assessments: core vulnerability sequences on trees used for 1984-87 assessments were reassigned as success or core damage sequences, and the likelihood of PWR RCP seal LOCA following station blackout was explicitly modeled.

In the prior models, the core vulnerability end state was assigned to sequences in which core protection was expected to be provided but for which no specific analytic basis was generally available or which involved non-proceduralized operator actions. Core vulnerability sequences were assigned to either success or core damage end states in the current models, as follows:

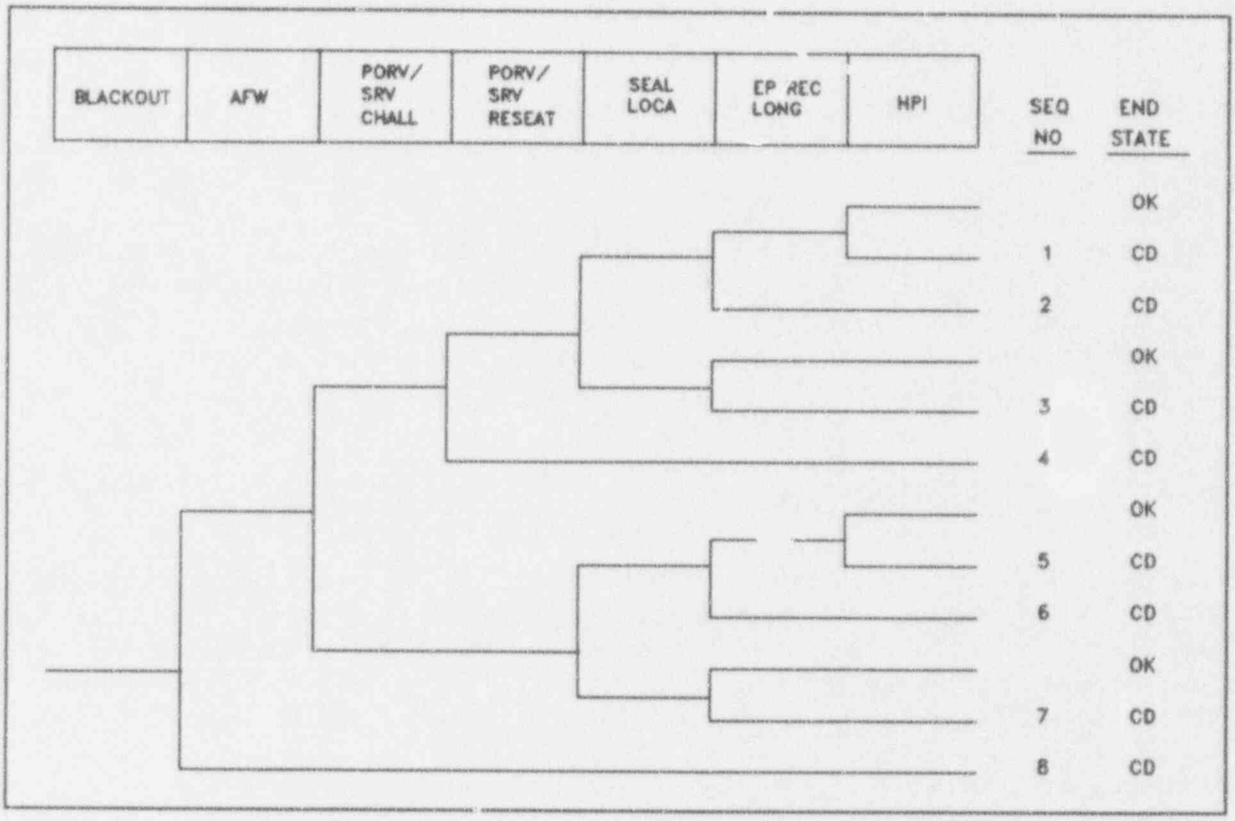
Core vulnerability sequence type	Revised end state
Stuck-open secondary-side relief valve with a failure of HPI in a PWR	Success
Steam generator (SG) depressurization and use of condensate system following failure of AFW, MFW, and feed and bleed in a PWR	Core damage (except for PWR Class H)
Use of containment venting as an alternate core cooling method in a BWR	Core damage

The net effect of this change is a significant reduction in the complexity of the event trees, with little impact on the relative significance estimated for each precursor. The impact of this modeling change on conditional probability estimates for 1987 precursors is described in Sect. 3.6 of Ref. 1. (Alternate calculations using models with the above changes were performed on 1987 events.) As illustrated in Ref. 1, modest differences existed between the core damage, core damage plus core vulnerability, and revised core damage model conditional probability estimates for most of the more significant events. Where differences did exist, the sum of probabilities of core damage and core vulnerability (all non-ATWS undesirable end states in the earlier models) was closer to the core damage probability estimated with the revised models.

Three 1987 events had substantially higher "sum" probabilities—these events involved trips with single safety-related train unavailabilities, for which the dominant core vulnerability sequence was a stuck-open secondary-side relief valve with HPI failure (assigned to success in the revised models).

The second modeling change was the inclusion of PWR RCP seal LOCA in blackout sequences. The impact of such a seal LOCA on the core damage probability estimated for an event had previously been bounded by the use of a conservative value for failure to recover ac power prior to battery depletion following a LOOP and loss of emergency power

The PWR event trees have been revised to address potential seal LOCA during station blackout through the use of seal LOCA and electric power recovery branches, as shown below:



Two time periods are represented in the sequences in the above figure. Auxiliary feedwater, power-operated relief valve/safety relief valve (PORV/SRV) challenge, and PORV/SRV reseat are shot-term responses following loss of the diesel generators. If turbine-driven AFW is unavailable, or if an open PORV/SRV fails to close, then core damage is assumed to occur, since no high-pressure injection is available as an alternate means of core cooling or for RCS makeup. SEAL LOCA, EP REC LONG, and HPI are branches applicable in the long term. SEAL LOCA represents the likelihood of a seal LOCA prior to restoration of ac power. EP REC LONG represents the likelihood of not restoring ac power prior to core uncovery (if a seal LOCA exists) or prior to battery depletion (in the case of no seal LOCA). Once the batteries are depleted, core damage is assumed to occur, since control of turbine-driven pumps and the ability to monitor core and RCS conditions are lost. HPI represents the likelihood of failing to provide HPI following a seal LOCA to prevent core damage. The ASP models have been simplified somewhat by assuming that HPI is always adequate to make up for flow from a failed seal or seals.

The three seal LOCA-related sequences are illustrated in sequences 1, 2, and 3. In sequence 1, a seal LOCA occurs prior to restoration of ac power, ac power is successfully restored prior to core uncovery, but HPI fails to provide makeup flow. In sequence 2, a seal LOCA also occurs, and ac power is not restored prior to core uncovery. In sequence 3, no seal LOCA occurs, but ac power is not recovered prior to battery depletion. The likelihood of seal LOCA prior to ac power restoration and the likelihood of ac power recovery are time-dependent, and this time-dependency is accounted for in the analysis. A more detailed description of the changes associated with explicitly modeling RCP seal LOCA is included in Ref. 2.

In addition to elimination of core vulnerability sequences, two other changes were made to simplify the previously complex BWR event trees:

- Failure to trip with soluble boron injection success was previously developed in detail and involved a large number of low probability sequences. All failure to trip sequences are now assigned to the ATWS end state.

- The condensate system was previously modeled as an alternate source of low-pressure injection water. This use of the condensate system is now considered a recovery action. this reduces the number of sequences on the event trees without substantially impacting the core damage probability estimates developed using the trees. Systems addressed on the event trees for low-pressure injection include LPCS, LPCI, and RHRSW.

A.2 Plant Categorization

Both the 1969−79 and 1980−81 precursor reports (Refs. 1 and 2) used simplified, functionally based event trees to model potential event sequences. One set of event trees was used to model for PWR initiating events: LOFW, LOOP, small-break LOCA, and steam line break. A separate set of event trees was used to model BWR response to the same initiators. Operational events that could not be modeled using these "standardized" event trees were addressed using models specifically developed for the event.

It was recognized during the review of the 1969-79 precursor report that plant designs were sufficiently different that multiple models would be required to more correctly describe the impact of an operational event in different plants. In 1985, substantial effort was expended to develop a categorization scheme for all U.S. LWRs that would permit grouping of plants with similar response to a transient or accident at the system or functional level, and to subsequently develop eight sets of plant-class specific event tree models. Much of the categorization and early event sequence work was done at the University of Maryland (Refs. 3 and 4). The ASP Program has generally employed these categorizations; however, some modifications have been required to reflect more closely the specific needs of the precursor evaluations.

In developing the plant categorizations, each reactor plant was examined to determine the systems used to perform the following plant functions required in response to reactor trip, LOOP, and small-break LOCA initiators to prevent core damage: reactor subcriticality, RCS integrity, reactor coolant inventory, short-term core heat removal, and long-term core heat removal.

Functions related to containment integrity (containment overpressure protection and containment heat removal) and post-accident reactivity removal are not included on the present ASP event trees (which only concern core damage sequences) and are not addressed in the categorization scheme.

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

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

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

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

Plants in which initiator response cannot be described using plant-class models are addressed using unique models, for example, the now deactivated LaCrosse BWR.

Table A.17 lists the class associated with each plant.

A.3 Event Tree Models

The plant class event trees describe core damage sequences for three initiating events: a nonspecific reactor trip, a LOOP, and a small-break LOCA. The event trees constructed are system-based and include an event tree applicable to each plant class defined.

System designs and specific nomenclature may differ among plants included in a particular class; but functionally, they are similar. Plants where certain mitigating systems do not exist, but which are largely analogous in their transient response, were grouped into the plant classes accordingly. In modeling events at such plants, the event tree branch probabilities were modified to reflect the systems available at the plant. Certain events (such as a postulated steam line break) could not be described using the plant-class event trees presented in this appendix. In these cases, unique event trees were developed to describe the sequences of interest.

**Maine Yankee Atomic Power Plant was built by Combustion Engineering but has a response to initiating events more akin to the Westinghouse Electric Corporation design, so it is grouped in a class with other Westinghouse plants. Davis-Besse Nuclear Power Station was also placed in a Westinghouse plant class because its HPI system design requires the operator to open the PORV for feed and bleed, as in most Westinghouse plants. The requirement to open the PORV for feed and bleed is a primary difference between event trees for Westinghouse and Babcock and Wilcox plants. Plant response differences resulting from the use of different SG designs are not addressed in the models.

This section (1) describes the potential plant response to the three initiating events described above, (2) identifies the combinations of systems required for the successful mitigation of each initiator, and (3) briefly describes the criteria for success of each system-based function. The sequences are considered first for PWRs and then separately for BWRs. PWR Class B event trees are described first, along with those for Class D, which are similar. (The major difference between Class B and Class D plants is that PORV operability is not required for feed and bleed on Class D plants.) The event trees for the combined group apply to the greatest number of operating PWRs. Therefore, these are discussed first, followed by those for PWR Classes G, H, and then A. For the BWR event trees, the plant Class C models are described first, because these are applicable to the majority of the BWRs, followed by discussions for the A and B BWR classes, respectively. The event trees are constructed with branch (event or system) success as the upper branch and failure as the lower branch. Each sequence path is read from left to right, beginning with the initiator followed by subsequent systems required to preclude or mitigate core damage.

The event trees can be found following the discussion sections and are grouped according to plant classes, beginning with the PWR classes and followed by the BWR classes. The abbreviations used in the event tree models are defined in Table A.16 preceding the event trees. Sequence numbers are provided on the event trees for undesirable end states (core damage and ATWS). Because of the similarities among PWR sequences for different plant classes, common sequence numbers have been assigned when possible. PWR Class B sequences were used as a basis for this. Sequence numbers beyond those for Class B are used for uncommon sequences on other plant classes. This approach facilitates comparison of sequences among plant classes. This approach could not be used for BWRs because of the significant differences in systems used on plants in the three plant classes. For BWRs, sequences are numbered in increasing order moving down each event tree. The following sequence number groups are employed for all event trees: transient with reactor trip success, 11-39; LOOP with reactor trip success, 40-69; small-break LOCA with reactor trip success, 71-79; ATWS sequences, 91-99.

The trees are presented in the following order:

A.3.1 PWR Event Sequence Models

The PWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following three initiating events: reactor trip, LOOP, and small-break LOCA. The systems modeled in the event trees are those associated with the generic functions required in response to an initiating event, as described in Sect. A.2. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality:	Reactor trip
Reactor coolant system integrity:	Addressed in small-break LOCA models plus trip and LOOP sequences involving failure of primary relief valves to close
Reactor coolant inventory:	High-pressure injection (assumed required only following a LOCA)
Short-term core heat removal:	Auxiliary feedwater
	Main feedwater
	High-pressure injection and PORV (feed and bleed, PWR Classes A, B, D, and G)
	Secondary-side depressurization and use of condensate system (PWR Class H)
Long-term core heat removal:	Auxiliary feedwater
	Main feedwater
	High-pressure recirculation (PWR Classes B and D) (also required to support RCS inventory for all classes)
	Secondary-side depressurization and use of condensate system (PWR Class H)
	Containment spray recirculation (PWR Classes A and G)

PWR Nonspecific Rea.'or Trip

The PWR nonspecific reactor trip event tree constructed for plant Classes B and D is shown in Fig. A.7. The event-tree branches and the sequences leading to severe core damage and ATWS follow.

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

2. Reactor trip. To achieve reactor subcriticality and thus halt the fission process, the reactor protection system (RPS) is required to insert control rods into the core. If the automatically initiated RPS fails, a reactor trip may be initiated manually. Failure to trip was considered to lead to the end state ATWS and was not developed further.

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

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

5. PORV or SRV challenged. For sequences in which both reactor trip and steam generator feedwater flow (MFW or AFW) have been successful, the pressurizer PORV may or may not lift, depending on the peak pressurizer pressure following the transient. (In most transients, these valves do not lift.) The upper branch indicates that the valve or valves were challenged and opened. Because of the multiplicity of relief and safety valves, it was assumed that a sufficient number would open if the demand from a pressure transient exists.

   The lower branch indicates that the pressurizer pressure was not sufficiently high to cause opening of a relief valve. For the sequence in which both AFW and MFW fail following a reactor trip, at least one PORV or SRV was assumed to open for overpressure protection.

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

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

8. HPI and PORV open. If normal methods of achieving decay heat removal via the SGs (MFW and AFW) are unavailable, core cooling can be accomplished on most plants by establishing a feed and bleed operation. This operation (l) allows heat removal via discharge of reactor coolant to the containment through the PORVs and (2) RCS makeup via injection of borated water from the HPI system. Except at Class D plants, successful feed and bleed requires the operator to open the PORV manually. At Class D plants, the HPI discharge pressure is high enough to lift the primary-side safety valves, and feed and bleed can be accomplished without the operator manually opening the PORVs. HPI success is dependent on plant design but requires the introduction of sufficient

amounts of borated water into the RCS to remove decay heat and provide sufficient reactor coolant makeup to prevent core damage.

9. High-pressure recirculation. Following a transient-induced LOCA (a PORV or SRV fails to reseat), or failure of secondary-side cooling (AFW and MFW) and initiation of feed and bleed, continued core cooling and makeup are required. This requirement can be satisfied by using HPI in the recirculation mode. In this mode the HPI pumps recirculate reactor coolant collected in the containment sump and pass it through heat exchangers for heat removal. When MFW or AFW is available, heat removal is only required for HPI pump cooling; if AFW or MFW is not available, HPR is required to remove decay heat as well. Typically, at Class B and D plants, the LPI pumps are utilized in the HPR mode, taking suction from the containment sump, passing the pumped water through heat exchangers, and providing net positive suction head to the HPI pumps.

The event tree applicable to a PWR Class G nonspecific reactor trip is shown in Fig. A.10. Many of the event tree branches and the sequences leading to successful transient mitigation and core damage are similar to those following a nonspecific reactor trip transient for plant Class B. At Class G plants, however, the HPR system performs both the high- and low-pressure recirculation (LPR), function, taking suction directly from the containment sump without the aid of the low-pressure pumps. DHR is accomplished during recirculation by the containment spray recirculation (CSR) system. The event-tree branches and sequences are discussed further.

1. Initiating event (transient). The initiating event is a nonspecific reactor trip, similar to that described for PWR Classes B and D. The following branches have functions and success requirements similar to those following a transient at PWR Class B.

2. Reactor trip.

3. Auxiliary feedwater or main feedwater.

4. PORV or SRV challenged reseats.

5. High-pressure injection.

6. HPI and PORV open (feed and bleed). Success requirements for feed and bleed are similar to those following the plant Class B transient. Feed and bleed with operator opening of the PORV is required in the event that both AFW and MFW are unavailable for secondary-side cooling. In addition, DHR was assumed required to prevent potential core damage. This is provided by the CSR system.

7. High-pressure recirculation. In the event of a transient-induced LOCA, continued HPI via sump recirculation is needed to provide makeup to the break to prevent potential core damage. In addition, HPR is required when both AFW and MFW are unavailable following a transient, to recirculate coolant during the feed and bleed procedure. If HPR fails and normal secondary-side cooling is also failed, core damage will occur. In Class G plants, initiation of HPR realigns the HPI pumps to the containment sump. The use of LPI pumps for suction-pressure boosting is not required.

8. Containment spray recirculation. When feed and bleed (HPI, HPR, and PORV open) is required, the CSR system operates to remove decay heat from the reactor coolant being recirculated. Without the CSR system, the feed and bleed operation could not remove decay heat. Successful operation of feed and bleed and CSR was assumed to result in successful mitigation of core damage.

The event tree for PWR Class H non-specific reactor trip is shown in Fig. A.13. This class of plants is different than other PWR classes in that PORVs are not included in the plant design and feed and bleed cannot be used to remove decay heat in the event of main and AFW unavailability. If main or AFW cannot be recovered, the atmospheric dump valves can be used to depressurize the SGs to below the shutoff head of the condensate pumps, and these can be used, if available, for RCS cooling. Because of the need for secondary-side cooling for all success sequences, a requirement for CC to prevent core damage has not been modeled.

1. Initiating event (transient). The initiating event is a non-specific reactor trip, similar to that described for the previous PWR classes. The following branches have functions and success requirements similar to those following a transient at PWRs associated with previously described PWR classes.

2. Reactor trip.

3. Auxiliary feedwater.

4. Main feedwater.

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

6. SRV reseat. Success for this branch requires the closure of any open safety valve once pressurizer pressure has been reduced below the safety valve set point.

7. High-pressure injection. In the case of a transient-induced LOCA, HPI is required to provide RCS makeup to keep the core covered.

8. High-pressure recirculation. The requirement for continued core cooling during mitigation of a transient-induced LOCA and following depletion of the refueling water tank can be satisfied by using HPI in the recirculation mode. In Class H plants, initiation of HPR realigns the HPI pumps to the containment sump. The use of LPI pumps for suction-pressure boosting is not required.

9. Steam generator depressurization. In the event that main and AFW are unavailable, the atmospheric dump valves (or turbine bypass valves if the main steam isolation valves are open) may be used on Class H plants to depressurize the SGs to the point that the condensate pumps can be used for SG cooling. In the event of main and AFW unavailability, failure to depressurize one SG to the operating pressure of the condensate system is assumed to result in core damage.

10. Condensate pumps. As described above, use of the condensate pumps on Class H plants along with secondary-side depressurization can provide adequate core cooling. Flow from one condensate

pump to one SG is assumed adequate. Unavailability of the condensate pumps in the event of failure to recover main and AFW is assumed to result in core damage.

The event tree applicable to PWR plant Class A nonspecific reactor trip is shown in Fig. A.4. Many of the event-tree branches and the sequences leading to successful transient mitigation and severe core damage are similar to those following a nonspecific reactor trip transient for plant Classes B and G.

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

1. Initiating event (transient). The initiating event is a nonspecific reactor trip, similar to that described for the other PWR plant classes. The following branches have functions and success requirements similar to those following a transient at PWRs associated with plant Classes B, D, and G.

2. Reactor trip.

3. Auxiliary feedwater

4. Main feedwater.

5. PORV or SRV challenged.

6. PORV/SRV reseats.

7. High-pressure injection.

8. High-pressure recirculation. In the event of a transient-induced LOCA, HPR can provide sufficient makeup to the break to terminate the transient. The LPI pumps provide suction to the high-pressure pumps in the recirculation mode. In the event that feed and bleed is required (following a transient in which both AFW and MFW are unavailable), HPR success is required.

9. Containment spray recirculation. The CSR system provides DHR during HPR when AFW and MFW are not available. In transient-induced LOCA sequences, HPI and HPR success is required to mitigate the event. In the event that secondary-side cooling via AFW or MFW is unavailable, feed and bleed with CSR, for DHR is considered sufficient to prevent core damage.

10. PORV open. The PORV must be opened by the operator below its set point to establish feed and bleed operation in the event that secondary-side cooling via AFW or MFW is unavailable.

Sequences resulting in core damage or ATWS following a PWR transient, shown on event trees applicable to each plant class, are described in Table A.4.

Many of the sequences are the same for different plant classes, the primary differences being the use of CSR on Class G and Class A, and the use of SG depressurization and condensate pumps for RCS cooling in lieu of feed and bleed on Class H. Because of this similarity, consistent sequence numbers have been used for the sequences in different PWR plant classes. All sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.5.

PWR Loss o. .site Power

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

The PWR LOOP tree constructed for plant Classes B and D is shown in Fig. A.8. The event-tree branches and the sequences leading to core damage follow.

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

2. Reactor trip given LOOP. Unavailability of power to the CRD mechanisms is expected to result in a reactor trip and rapid shutdown of the plant. If the reactor trip does not occur, the transient was considered to proceed to ATWS and was not developed further.

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

4. Auxiliary feedwater. The AFW system functions to remove decay heat via the SG secondary side. Success requirements for this branch are equivalent to those following a nonspecific reactor trip and unavailability of MFW. Both MFW and condensate pumps would be unavailable following a LOOP. Therefore, with emergency power and AFW failed, no core cooling would be available, and core damage would be expected to occur. Because, specific AFW systems may contain different combinations of turbine-driven and motor-driven AFW pumps, the capability of the system to meet its success requirements will depend on the state of the EPS and the number of turbine-driven AFW pumps that are available.

5. PORV or SRV challenged. The upper and lower states for this branch are similar to those following a nonspecific reactor trip. The PORV or SRV may or may not lift, depending on the peak pressure following the transient.

6. PORV or SRV reseats. The success requirements for this branch are similar to those following a nonspecific reactor trip. However, for the sequence in which emergency power is failed and the PORV fails to reseat, the HPI/HPR system would be without power to mitigate potential core damage.

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

design (for some seal designs, seal failure can be prevented by isolating the seal return isolation valve).

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

8. Electric power recovered (long term). For sequences in which a seal LOCA has occurred, success requirements are the restoration of ac power [either through recovery of offsite power or recovery of a DG] prior to core uncovery. For sequences in which a seal LOCA does not occur, success requires the recovery of ac power prior to battery depletion, typically 2 to 4 h.

9. High-pressure injection and recirculation. The success requirements for this branch are similar to those following a nonspecific reactor trip. Because all HPI/HPR systems use motor-driven pumps, the capability of the HPI or HPR system to meet its success requirements depends on the success of the EPS.

10. PORV open (for feed and bleed). The success requirements for this branch are similar to those following a nonspecific reactor trip. The PORV is opened in conjunction with feed and bleed operations when secondary-side heat removal is unavailable. For Class D plants, the PORV does not have to be manually opened to establish feed and bleed because the HPI pump discharge pressure is high enough to lift the PORV or primary relief valve.

The event tree constructed for the PWR Class G LOOP is shown in Fig. A.11. Most of the event-tree branches and the sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, at Class G plants, DHR during recirculation is provided by the CSR system, not the HPR system. The event-tree branches and sequences are discussed further below.

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

2. Reactor trip given LOOP.

3. Emergency power.

4. Auxiliary feedwater.

5. PORV or SRV challenged.

6. PORV/SRV valve reseats.

7. Seal LOCA.

8. Electric power recovered (long term).

9. High-pressure injection and recirculation.

10. PORV open (for feed and bleed).

11. Containment spray recirculation. The success requirements for this branch are similar to those following a nonspecific reactor trip. The CSR system provides DHR for sequences in which secondary-side cooling is unavailable.

The event tree constructed for a PWR Class H LOOP is shown in Fig. A.14. Many of the event tree branches and sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, Class H plants do not have feed and bleed capability and rely instead on secondary-side depressurization and the condensate system as an alternate DHR method. The condensate system is assumed unavailable following a LOOP, which limits the diversity of DHR methods on this plant class following this initiator. The event branches and sequences are discussed further below.

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

2. Reactor trip given LOOP.

3. Emergency power.

4. Auxiliary feedwater.

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

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

7. Seal LOCA.

8. Electric power recovered (long-term).

9. High pressure injection and recirculation.

The event tree constructed for the plant Class A LOOP is shown in Fig. A.5. All of the event-tree branches and the sequences leading to successful transient mitigation, potential core vulnerability, and severe core damage are analogous to those following a LOOP at Class B plants with the addition of the CSR branch, which is required for successful feed and bleed. At Class A plants, DHR during HPR is accomplished by the CSR system; whereas at Class B and D plants, DHR is an integral part of the HPR system. Additional information on the use of the CSR system is provided in the discussion of the PWR Class A nonspecific reactor trip event tree.

Sequences resulting in core damage and ATWS following a PWR LOOP, shown on event trees applicable to each plant class, are described in Table A.6.

Many of the sequences are the same for different plant classes, the primary differences being the use of CSR on Class G and Class A, and the unavailability of feed and bleed on Class H. As with the PWR transient sequences, this similarity permits consistent numbering of a large number of sequences. All sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.7.

PWR Small-Break Loss-of-Coolant Accident

Event trees were constructed to define the responses of PWRs to a small-break LOCA. The LOCA chosen for consideration is one that would require a reactor trip and continued HPI for core protection. Because of the limited amount of borated water available, the mitigation sequence also includes the requirement to recirculate borated water from the containment sump.

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

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

2. Reactor trip. Reactor trip success is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition. Failure to trip was considered to lead to the end state ATWS.

3. Auxiliary feedwater or main feedwater. Use of AFW or MFW was assumed necessary for some small breaks to reduce RCS pressure to the point where HPI is effective. At Class D plants, the HPI pumps operate at a much higher discharge pressure and hence can function without secondary-side cooling from the AFW or MFW systems.

4. High-pressure injection. Adequate injection of borated water from the HPI system is required to prevent excessive core temperatures and consequent core damage.

5. High-pressure recirculation. Following a small-break LOCA, continued high pressure injection is required. This is typically accomplished with the residual heat removal (RHR) system, which takes suction from the containment sump and returns the lost reactor coolant to the core via the HPI pumps. The RHR system includes heat exchangers that remove decay heat prior to recirculating the sump water to the RCS.

6. PORV open. In the event AFW and MFW are unavailable following a small break LOCA, opening the PORV can result in core cooling using the feed and bleed mode. Depending on the size of the small break, opening the PORV may not be required for success. PORV open is not required for success for Class D.

The event tree constructed for a small-break LOCA at Class G plants is shown in Fig. A.12. The LOCA event tree for Class G plants is similar to that for Class B and D plants except that long-term cooling is provided by the CSR system rather than by the HPR system. The event-tree branches and sequences are discussed further below.

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

2. Reactor trip.

3. Auxiliary feedwater and main feedwater

4. High-pressure injection.

5. High-pressure recirculation.

6. PORV open.

7. Containment spray recirculation. In the event that normal secondary-side cooling (AFW or MFW) is unavailable following a small LOCA, cooling via the CSR system during HPR is required to mitigate the transient.

The event tree constructed for a small-break LOCA at PWR Class H plants is shown in Fig. A.15. The event tree has been developed assuming that SG depressurization and condensate pumps can provide adequate RCS pressure reduction in the event of an unavailability of AFW and MFW to permit HPI and HPR to function in these plants. The event tree branches and sequences are discussed further below.

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

2. Reactor trip.

3. Auxiliary and main feedwater.

4. High-pressure injection.

5. High-pressure recirculation.

6. SG depressurization. In the event that AFW and MFW are unavailable following a small-break LOCA, SG depressurization combined with the use of the condensate pumps can provide for RCS depressurization such that adequate HPI and HPR can be achieved. Success requirements are the same as those following a transient with unavailability of AFW and MFW.

7. Condensate pumps. Use of one condensate pump provided flow to at least one SG as required in conjunction with SG depressurization to provide for RCS depressurization and cooling.

The event tree constructed for a small LOCA at Class A plants is shown in Fig. A.6. The LOCA event tree for Class A plants is similar to that for Classes B and D except that the CSR system is required in conjunction with HPR in some sequences where secondary cooling is not provided. The sequences that follow combined AFW and MFW failure with HPR and CSR success are identical to those that follow HPR success at Class B and D plants; and sequences that follow HPR or CSR failure at Class A plants are identical to those that follow HPR failure.

Sequences resulting in core damage or ATWS following a PWR small-break LOCA, shown on event trees applicable to each plant class, are described in Table A.8.

As with the PWR transient and LOOP sequences, differences between plant classes are driven by the use of CSR on plant classes A and G, and by the use of secondary-side depressurization and condensate pumps in lieu of feed and bleed on PWR Class H. All small-break LOCA sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.9.

Alternate Recovery Actions

The PWR event trees have been developed on the basis that proceduralized recovery actions will be attempted if primary systems that provide protection from core damage are unavailable. In the event AFW and MFW are unavailable and cannot be recovered in the short term, the use of feed and bleed cooling is modeled on all plants except for Class H, where SG depressurization and use of the condensate pumps is modeled instead. In addition, the potential for short-term recovery of a faulted system is also included in appropriate branch models (AFW, MFW, and HPI, for example).

Alternate equipment and procedures, beyond the systems and functions included in the event trees, may be successful in mitigating the effects of an initiating event, provided the appropriate equipment or procedure is available at a particular plant. This may include:

- The use of supplemental DGs, beyond the normal safety-related units, to power equipment required for continued core cooling and reactor plant instrumentation. A number of plants have added such equipment, often for fire protection.

- Depressurization following a small-break LOCA to the initiation pressure of the LPI systems to provide RCS makeup in the event that HPI fails. Procedures to support this action are known to exist on some plants.

- Depressurization following a small-break LOCA to the initiation pressure of the DHR system, and then proceeding to cold shutdown. While plant procedures specify the use of sump recirculation following a small LOCA or feed and bleed, sufficient RWST inventory exists to delay this action until many hours into the event, during which recovery of faulted systems may be affected. It is likely that operators will delay sump recirculation as long as possible while trying to place the plant in a stable condition through recovery of secondary-side cooling and the use of RHR.

The potential use of these alternate recovery actions was addressed in the analysis of the 1992 precursors when information concerning their plant specific applicability was available.

A.3.2 BWR Event Sequence Models

The BWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following the same three initiating events addressed for PWRs: trip, LOOP, and small-break LOCA. The systems modeled in the event trees are those associated with the generic functions required in response to any initiating event, as described in Sect. A.2. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality:	Reactor scram
Reactor coolant system integrity:	Addressed in small-break LOCA models and in trip and LOOP sequences involving failure of primary relief valves to reseat
Reactor coolant inventory:	High-pressure injection systems [HPCI or HPCS, RCIC (non-LOCA situations), CRD (non-LOCA situations), FWCI]
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), LPCS, RHRSW or equivalent]
Short-term core heat removal:	Power conversion system
	High-pressure injection systems [HPCI, RCIC, CRD, FWCI (BWR Class A)]
	Isolation condenser (BWR Classes A and B)
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), LPCS]
	Note: Short-term core heat removal to the suppression pool (all cases where power conversion system is faulted) requires use of the RHR system for containment heat removal in the long term.
Long-term core heat removal:	Power conversion system
	Isolation condenser (BWR Class A)
	Residual heat removal [shutdown cooling or suppression pool cooling modes (BWR Class C)]
	Shutdown cooling (BWR Classes A and B)
	Containment cooling (BWR Class A)
	Low-pressure coolant injection [CC mode (BWR Class B)]

BWR Nonspecific Reactor Trip

The nonspecific reactor trip event tree constructed for BWR plant Class C is shown in Fig. A.22. The event tree branches and the sequences leading to potential severe core damage follow. The Class C plants are discussed first because all but a few of the BWRs fit into the Class C category.

1. Initiating event (transient). The initiating event is a transient or upset event that results in a rapid shutdown of the plant. Transients that are initiated by a LOOP or a small-break LOCA are modeled

in separate event trees. Transients initiated by a large-break LOCA or large SLB are not addressed in the event trees described here; trees applicable to such initiators are developed separately if required.

2. Reactor shutdown. To achieve reactor subcriticality and thus halt the fission process, the RPS commands rapid insertion of the control rods into the core. Successful scram requires rapid insertion of control rods with no more than two adjacent control rods failing to insert.

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

4. SRV challenged. Depending on the transient, one or more SRVs may open. The upper branch on the event tree indicates that the valves were challenged and opened. If the transient is followed by continued PCS operation and successful scram, the SRVs are not expected to be challenged. If the PCS is unavailable, at least some of the SRVs are assumed to be challenged and to open.

5. SRV close. Success for this branch requires the reseating of any open relief valves once the reactor pressure vessel (RPV) pressure decreases below the relief valve set point. If an SRV sticks open, a transient-induced LOCA is initiated.

6. Feedwater. Given unavailability of the PCS, continued delivery of feedwater to the RPV will keep the core from becoming uncovered. This, in combination with successful long-term DHR, will mitigate the transient, preventing core damage. For plants with turbine-driven feed pumps, the PCS failure with subsequent feedwater success cannot involve MSIV closure, or loss of condenser vacuum, because this would disable the feed pumps.

7. HPCI or HPCS. The primary function of the HPCI or HPCS system is to provide makeup following small-break LOCAs while the reactor is at high-pressure (not depressurized). The system is also used for DHR following transients involving a loss of feedwater. Some later Class C plants are equipped with HPCS systems, but the majority are equipped with HPCI systems. HPCI or HPCS can provide the required makeup and short-term DHR when DHR is unavailable from the condenser and the feedwater system cannot provide makeup.

8. RCIC. The RCIC system is designed to provide high-pressure coolant makeup for transients that result in LOFW. Both RCIC and HPCI (or HPCS) initiate when the reactor coolant inventory drops to the low-low level set point, taking suction from the condensate storage tank or the suppression pool. HPCI is normally secured after HPCI/RCIC initiation when pressure and water level are restored, to prevent tripping of HPCI and RCIC pumps on high water level. RCIC must then be operated until the RHR system can be placed in service. Following a transient, scram, and unavailability of the PCS, reactor pressure may increase, causing the relief valves to open and close periodically to maintain reactor pressure control.

9. CRD pumps. In transient-induced sequences where heat removal and minimal core makeup are required (i.e., not transient-induced LOCA sequences), the CRD pumps can deliver high-pressure coolant to the RPV.

10. Depressurization via SRV or the automatic depressurization system (ADS). In the event that short-term DHR and core makeup are required and high-pressure systems have failed to provide adequate flow, the RPV can be depressurized to allow use of the low-pressure, high-capacity injection systems. If depressurization fails in this event, core damage is expected to occur. The ADS will automatically initiate on high drywell pressure and low-low reactor water level, and the availability of one train of the LPCI or LPCS systems, following a time delay. The SRVs can be opened by the operators to speed the depressurization process or to initiate it if ADS fails and if additional, operable valves are available.

11. LPCS. LPI can be provided by the LPCS system if required. The LPCS system performs the same functions as the LPCI system (described below) except that the coolant, which is drawn from the SP or the condensate storage tank (CST), is sprayed over the core.

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

13. Residual heat removal shutdown cooling (SDC) mode. In this mode, the RHR system provides normal long-term DHR. Coolant is circulated from the reactor by the RHR pumps through the RHR heat exchangers and back to the reactor vessel. Long-term core cooling success requires that heat transfer to the environment commence within 24 h of the transient. RHR SDC success following successful reactor scram and high- or low-pressure injection of water to the RPV will prevent core damage.

14. RHR SP cooling mode. If RHR SDC is unavailable, the RHR pumps and heat exchangers can be aligned to take water from the SP, cool it via the RHR heat exchangers, and return it to the SP. This alignment can provide long-term cooling for transient mitigation.

15. RHR service water or other. This is a backup measure for providing water to the reactor to reflood the core and maintain core cooling if LPCI and LPCS are unavailable Typically, the high-pressure SW pumps are aligned to the shell side of the RHR heat exchangers for delivery of water to one of the recirculation loops.

The event tree constructed for a BWR plant Class A nonspecific reactor trip is shown in Fig. A.16. The event tree is similar to that constructed for BWR Class C plants with the following exceptions: Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. The isolation condensers can provide long-term core cooling. Class A plants do not have LPCI systems, although they are equipped with LPCS; SP cooling is provided by a system independent of the SDC system. The event tree branches and sequences are discussed further below.

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

2. Reactor shutdown.

3. Power conversion system.

4. SRV challenged and closed.

5. Isolation condensers and isolation condenser makeup. If PCS is not available and significant inventory has not been lost via the SRVs, then the IC system can provide for DHR and mitigate the transient. The IC system is an essentially passive system that condenses steam produced by the core, rejecting the heat to cooling water and returning the condensate to the reactor. Makeup is provided to the cooling water as needed. The system does not provide makeup to the reactor vessel.

6. FW or FWCI. Either FW or FWCI can provide short-term transient mitigation. When feedwater or FWCI is required and is successful, long-term DHR is required for complete transient mitigation. (PCS unavailability is assumed prior to feedwater or FWCI demand.) FWCI or feedwater is required for makeup in transient-induced LOCA sequences and for heat removal in sequences when the IC system would have mitigated the transient but was not available. FWCI is initiated automatically on low reactor level and uses the normal feedwater trains to deliver water to the reactor vessel.

7. CRD pumps.

8. Depressurization via SRV or ADS.

9. LPCS.

10. Fire water or other. Fire water or other raw water systems can provide a capability similar to that provided by the SW/RHR connection on Class C BWRs. As a backup source, if all normal core cooling is unavailable, fire water can be aligned to the LPCS injection line to provide water to the reactor vessel.

11. SDC. Like the RHR system at Class C plants, the SDC system is a closed-loop system that performs the long-term DHR function by circulating primary coolant from the reactor through the system's heat exchangers and back to the reactor vessel. Success requires the operation of at least one SDC loop. Long-term DHR is required to terminate transients in which high- or low-pressure injection is required to mitigate the transient.

12. Containment cooling. If the SDC system fails to provide long-term DHR, the CC system can remove decay heat. The system utilizes dedicated CC pumps, drawing suction from the SP, passing it through heat exchangers where heat is rejected to the SW system and then either returning it directly to the SP or spraying it into the dry well.

The event tree constructed for a BWR plant Class B nonspecific reactor trip is shown in Fig. A.19. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same except that Class B plants are equipped with HPCI systems instead of FWCI systems, and they are equipped with a LPCI system that represents an additional capability for providing LPCI. Also, at Class B BWRs, the CC system considered in the event tree utilizes the LPCI pumps rather than having its own dedicated pumps.

Sequences resulting in core damage following a BWR transient, shown on event trees applicable to each plant class, are described in Table A.10. Because of differences in the mitigation systems used in the three BWR classes, it is not possible to associate most sequences among different plant classes. Because of this, similar sequence numbers used for sequences in different plant classes do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

BWR Loss of Offsite Power

The event cores constructed define responses of BWRs to a LOOP in terms of sequences representing success and failure of plant systems. A LOOP condition will result in a generator load rejection that would trip the turbine control valves and initiate a reactor scram.

The event tree constructed for a LOOP at BWR Class C plants is shown in Fig. A.23. The event-tree branches and the sequences leading to core damage follow.

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

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

3. Reactor shutdown. Given a load rejection, a scram signal is generated. Successful scram is the same as for the transient trees: a rapid insertion of control rods with no more than two adjacent control rods failing to insert. The scram can be automatically or manually initiated.

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

5. SRV challenged and closed. If one or more SRV is challenged and fails to close, a transient-induced LOCA is initiated.

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

7. CRD pumps. Given emergency power success, CRD pump success requirements following a LOOP are identical to those following a transient. The CRD pumps can provide sufficient makeup to remove decay heat but not enough makeup to mitigate a transient-induced LOCA. Manual restart of the CRD pumps is required following the LOOP.

8. Depressurization via SRV or the ADS.

9. LPCS, LPCI, or RHR service water.

10. RHR SDC mode or RHR SP cooling mode. For emergency power success sequences, the success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class C BWRs. Success for any one of these three branches can provide the long-term DHR required for transient mitigation. If emergency power fails, it must be recovered to power long-term DHR equipment. However, long-term DHR is not required until several hours (up to 24 h) into the transient.

The event tree constructed for a LOOP at BWR Class A plants is shown in Fig. A.17. The event tree is similar to that constructed for BWR Class C plants with the major exception that Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. However, given a LOOP, FWCI would be unavailable, because it is not backed by emergency power. Also, additional long-term core cooling is not required with IC success, as long as no transient-induced LOCA is initiated. In the emergency power failure sequences, the IC system is the only system that can provide core cooling because FWCI would be without power. The event-tree branches and sequences are further discussed below.

1. Initiating event (LOOP). The initiating event is a LOOP similar to that described for Class C BWRs. The following branches have functions and success requirements similar to those following a LOOP at BWRs associated with previously described BWR classes.

2. Emergency power.

3. Reactor shutdown.

4. LOOP recovery (long-term).

5. SRV challenged and closed.

6. IC. Following successful reactor scram, the IC system can provide enough DHR, in both the short and long term, to mitigate the transient if a transient-induced LOCA has not been initiated. The IC system cannot provide coolant makeup, which would be required in a transient-induced LOCA. The IC system is an essentially passive system that does not require ac power for success.

7. FWCI. The FWCI system can provide short-term core cooling and makeup for transient mitigation. However, FWCI success requires normal power supplies and cannot be powered by emergency power following a LOOP.

8. CRD pumps.

9. Depressurization via SRV or ADS.

10. LPCS, fire water, or other water source. Success requirements for these branches are similar to those following a nonspecific reactor trip at Class A BWRs. With interim high-pressure cooling unavailable, either LPCS or, as a last resort, fire water or another water source can be used to provide low-pressure water for core makeup and cooling.

11. SDC and containment cooling. The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs.

The event tree constructed for a BWR plant Class B LOOP is shown in Fig. A.20. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that Class B plants are equipped with HPCI systems instead of FWCI systems and are equipped with a LPCI system, which represents an additional capability for providing LPCI. At Class B BWRs the CC system utilizes the LPCI pumps rather than having its own dedicated pumps. In emergency power failure sequences, either the IC or HPCI system can provide the required core cooling for short-term transient mitigation. However, if an SRV sticks open (transient-induced LOCA), the ICs cannot provide the makeup needed, and HPCI is required. The ICs can also provide long-term cooling, but when only HPCI is operable, recovery of emergency power is necessary to power SDC-related loads.

Sequences resulting in core damage following a BWR LOOP, as shown on each plant-class event tree, are described in Table A.11. As in the case of BWR transients, similar sequence numbers do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

BWR Loss-of-Coolant Accident

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

The LOCA event tree constructed for BWR Class C plants is shown in Fig. A.24. The event-tree branches and sequences leading to core damage and core vulnerability follow.

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

2. Reactor shutdown. Successful scram is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition.

3. HPCI or HPCS. HPCI (or HPCS, depending on the plant) can provide the required inventory makeup.

4. Depressurization via SRV or ADS. The success requirements for this branch are similar to those following a nonspecific reactor trip transient. SRV/ADS success allows the use of low-pressure systems to provide short-term core cooling and makeup.

5. LPCS, LPCI, or RHR service water. The success requirements for these branches are similar to those following a nonspecific reactor trip transient. Any one of these branches can provide short-term core cooling and makeup if SRV/ADS is successful.

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

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(716) 265-1600

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IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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6. RHR (SDC mode) or RHR (SP cooling mode). Success requirements for these branches are similar to those following a nonspecific reactor trip transient, except that heat rejection to the environment may be required sooner than 24 h into the transient, depending on the break size. These methods each have the capability of providing long-term DHR. Long-term DHR is required in all sequences for LOCA mitigation.

The LOCA event tree constructed for BWR Class A plants is shown in Fig. A.18. The event tree is similar to the LOCA tree constructed for BWR Class C plants except that Class A plants have FWCI instead of HPCI or HPCS systems and are, in general, not equipped with LPCI systems (only LPCS systems). In addition, SP and CC systems are independent of the SDC system. The event tree branches and sequences leading to core damage follow.

1. Initiating event (small LOCA). The initiating event is a small LOCA similar to that described for BWR Class C plants. The following branches have functions and success requirements similar to those following a small LOCA at BWRs associated with the previously described BWR classes.

2. Reactor shutdown.

3. FWCI. The FWCI system has the capability to keep the core covered and provide interim core cooling. FWCI initiates automatically on low reactor water level.

4. Depressurization via SRV or ADS.

5. LPCS or fire water (or other water source). The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs. Either of these systems (branches) can provide LPI for makeup and short-term core cooling if high-pressure systems are unavailable.

6. SDC or containment cooling. The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs, except that heat rejection to the environment may be required sooner than 24 h into the transient, depending on the size of the break. Either of these methods can provide the long-term DHR required to mitigate a small LOCA.

The LOCA event tree constructed for BWR Class B plants is shown in Fig. A.21. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that some Class B plants are equipped with HPCI systems instead of FWCI systems and Class B BWRs have a LPCI system, which provides an additional capability for LPCI. At Class B BWRs the CC system uses the LPCI pumps rather than having its own dedicated pumps.

Sequences resulting in core damage following a BWR small-break LOCA, as shown on each plant-class event tree, are described in Table A.12. As in the case of BWR transients, similar sequence numbers do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

Alternate Recovery Actions

The BWR event trees have been developed on the basis that proceduralized recovery actions will be attempted if primary systems that provide protection against core damage are unavailable. If feedwater, HPCI, and RCIC are unavailable (FWCI and ICs on BWR Classes A and B) and cannot be recovered in the short term, the use of the CRD pumps (provided no LOCA exists) and the use of ADS (to depressurize below the operating pressure of low-pressure systems) are modeled. In addition, the potential for short-term recovery of a faulted system is also included in the appropriate branch model.

Alternate equipment and procedures, beyond the systems and functions included in the event tree, may be successful in mitigating the effects of an initiating event, provided the appropriate equipment or procedure is available at a particular plant. This may include:

- The use of supplemental diesel generators, beyond the normal safety-related units, to power equipment required for continued core cooling and reactor plant instrumentation. A number of plants have added such equipment, often for fire protection.

- The use of RCIC to provide RPV makeup for a single stuck-open relief valve. Thermal-hydraulic analyses performed to support a number of BWR probabilistic risk assessments have demonstrated the viability of RCIC for this purpose.

- The use of the condensate system for LPI. This recovery action requires that the condensate system be available (even though PCS and feedwater are unavailable) and that the plant has been depressurized.

- The use of containment venting for long-term DHR, provided an injection source is available. This core cooling method has been addressed in some PRAs.

The potential use of these alternate recovery actions was addressed in the analysis of the 1992 precursors when information concerning their plant specific applicability was available.

A.4 Branch Probability Estimates

Branch probability estimates used in the 1988-1992 precursor calculations were developed using information in the 1984-86 precursors when possible. Probability values developed from precursor information are shown in Table A.13. The process used to estimate branch probability values used in the precursor calculations is described in detail in Appendix C to Ref. 5 and in Ref. 6.

In addition to system failures caused by equipment failures, the likelihood of failing to actuate manually actuated systems was also included in the models. Examples of such systems are the DHR system in BWRs and feed and bleed in PWRs. For actions in the control room, revised failure to initiate probabilities consistent with those utilized for 1987 precursor calculations were also used for 1988-1992 calculations. These revised values typically assume a failure probability of 0.001 for an unburdened action and 0.01 for a burdened action. The failure probability for subsequent actions is assumed to be higher. Operator action failure probabilities used in the 1988-1992 calculations are shown in Table A.14.

A.5 Reference Event Calculations

Conditional core damage probability estimates were also calculated for nonspecific reactor trip, LOFW, and unavailabilities in certain single-train BWR systems (HPCI, HPCS, RCIC, and CRD cooling). These calculations indicate the relative importance of these events, which are too numerous to warrant individual calculation. The results of these calculations, performed without consideration of alternate recovery actions that were addressed in certain 1992 precursor assessments, are listed in Table A.15.

Table A.15 shows that nonspecific reactor trips without additional observed failures have conditional core damage probabilities below 5×10^{-6} per trip, depending on plant class. The likelihood of LOFW in conjunction with a trip is included in these calculations. LOFW conditional core damage probabilities are less than 4×10^{-5} per LOFW event, again depending on plant class, except for BWR Class A plants (1.7×10^{-4}). The conditional core damage probabilities associated with unavailabilities of HPCI and HPCS (single-train BWR systems) are also above 10^{-5}, assuming a one-half month unavailability.

A.6 References

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

Table A.1 Branch probability estimation process

Branch failure	Observed operational event	Non-recovery likelihood for event	Effective number of non-recoverable events	Observation period	Probability estimate
Steam generator isolation	Steam line pressure transmitters (9 of 12) were found in faulty alignment, which would have prevented automatic steam line isolation on demand at Maine Yankee (LER 309/85-009, 8/7/85)	0.04	1.04	12 demands per reactor year due to testing in 164 PWR reactor years (1984–86 observation period) results in 1968 demands	5.3×10^{-4}
	All MSIVs failed to close prior to entering refueling at Point Beach 2 (LER 301/86-004, 9/28/86)	1.0			

Table A.2 Rules for calculating precursor significance

1. Event sequences requiring calculation.

   If an initiating event occurs as part of a precursor (i.e., the precursor consists of an initiating event plus possible additional failures), then use the event tree associated with that initiator; otherwise, use all event trees impacted by the observed unavailability.

2. Initiating event probability.

   If an initiating event occurs as part of a precursor, then the initiator probability used in the calculation is the probability of failing to recover from the observed initiating event (i.e., the numeric value of the recovery class for the event).

   If an initiating event does not occur as part of a precursor, then the probability used for the initiating event is developed using the initiating event frequency and event duration. Event durations (the period of time during which the failure existed) are based on information included in the event report, if provided. If the event is discovered during testing, then one-half of the test period (15 days for a typical 30-day test interval) is assumed, unless a specific failure duration is identified.

3. Branch probability estimation.

   For event tree branches for which no failed or degraded condition is observed, a probability equal to the estimated branch failure probability is assigned.

   For event tree branches associated with a failed system, a probability equal to the numeric value associated with the recovery class is assigned.

   For event tree branches that include a degraded system (i.e., a system that still meets minimum operability requirements but with reduced or no redundancy), the estimated failure probability is modified to reflect the loss of redundancy.

4. Support system unavailabilities.

   Systems or trains rendered unavailable as a result of support system failures are modeled recognizing that, as long as the affected support system remains failed, all impacted systems (or trains) are unavailable; but if the support system is recovered, all the affected systems are recovered. This can be modeled through multiple calculations that address support system failure and success. Calculated core damage probabilities for each case are normalized based on the likelihood of recovering the support system. (Support systems, except emergency power, are not directly modeled in the current ASP models.)

Table A.3 ASP reactor plant classes

Plant name	Plant class	Plant name	Plant class
ANO-Unit1	PWR Class D	Millstone 3	PWR Class A
ANO-Unit	PWR Class G	Monticello	BWR Class C
Beaver Valley 1	PWR Class A	Nine Mile Point 1	BWR Class A
Beaver Valley 2	PWR Class A	Nine Mile Point 2	BWR Class C
Big Rock Point	BWR Class A	North Anna 1	PWR Class A
Browns Ferry 1	BWR Class C	North Anna 2	PWR Class A
Browns Ferry 2	BWR Class C	Oconee 1	PWR Class D
Browns Ferry 3	BWR Class C	Oconee 2	PWR Class D
Braidwood 1	PWR Class B	Oconee 3	PWR Class D
Braidwood 2	PWR Blass B	Oyster Creek	BWR Class A
Brunswick 1	BWR Class C	Palisades	PWR Class G
Brunswick 2	BWR Class C	Palo Verde 1	PWR Class H
Byron 1	PWR Class B	Palo Verde 2	PWR Class H
Byron 2	PWR Class B	Palo Verde 3	PWR Class H
Callaway 1	PWR Class B	Peach Bottom 2	BWR Class C
Calvert Cliffs 1	PWR Class G	Peach Bottom 3	BWR Class C
Calvert Cliffs 2	PWR Class G	Perry 1	BWR Class C
Catawba 1	PWR Class B	Pilgrim 1	BWR Class C
Catawba 2	PWR Class B	Point Beach 1	PWR Class B
Clinton 1	BWR Class C	Point Beach 2	PWR Class B
Comanche Peak 1	PWR Class B	Prairie Island 1	PWR Class B
Comanche Peak 2	PWR Class B	Prairie Island 2	PWR Class B
Cook 1	PWR Class B	Quad Cities 1	BWR Class C
Cook 2	PWR Class B	Quad Cities 2	BWR Class C
Cooper Station	BWR Class C	Rancho Seco	PWR Class D
Crystal River 3	PWR Class D	River Bend 1	BWR Class C
Davis-Besse	PWR Class B	Robinson 2	PWR Class B
Diablo Canyon 1	PWR Class B	Salem 1	PWR Class B
Diablo Canyon 2	PWR Class B	Salem 2	PWR Class B
Dresden 2	BWR Class B	San Onofre 1	Unique
Dresden 3	BWR Class B	San Onofre 2	PWR Class H
Duane Arnold	BWR Class C	San Onofre 3	PWR Class H
Farley 1	PWR Class B	Seabrook 1	PWR Class B
Farley 2	PWR Class B	Sequoyah 1	PWR Class B
Fermi 2	BWR Class C	Sequoyah 2	PWR Class B
Fitzpatrick	BWR Class C	South Texas 1	PWR Class B
Fort Calhoun	PWR Class G	South Texas 2	PWR Class B
Ginna	PWR Class B	St. Lucie 1	PWR Class G
Grand Gulf 1	BWR Class C	St. Lucie 2	PWR Class G
Haddam Neck	PWR Class B	Summer 1	PWR Class B
Harris 1	PWR Class B	Surry 1	PWR Class A
Hatch 1	BWR Class C	Surry 2	PWR Class A
Hatch 2	BWR Class C	Susquehanna 1	BWR Class C
Hope Creek 1	BWR Class C	Susquehanna 2	BWR Class C
Indian Point 2	PWR Class B	Three Mile Island 1	PWR Class D
Indian Point 3	PWR Class B	Trojan	PWR Class B
Kewaunee	PWR Class B	Turkey Point 3	PWR Class B
LaCrosse	Unique	Turkey Point 4	PWR Class B
LaSalle 1	BWR Class C	Vermont Yankee	BWR Class C
LaSalle 2	BWR Class C	Vogtle 1	PWR Class B
Limerick 1	BWR Class C	Vogtle 2	PWR Class B
Limerick 2	BWR Class C	WNPSS 2	BWR Class C
Maine Yankee	PWR Class B	Waterford 3	PWR Class H
McGuire 1	PWR Class B	Wolf Creek 1	PWR Class B
McGuire 2	PWR Class B	Yankee Rowe	PWR Class B
Millstone 1	BWR Class A	Zion 1	PWR Class B
Millstone 2	PWR Class G	Zion 2	PWR Class B

Table A.4 PWR transient core damage and ATWS sequences

Sequence No.	End state	Description
11	Core damage	Unavailability of HPR following successful trip and AFW initiation, primary relief valve lift and failure to reseat, and successful HPI. (PWR Classes A, B, D, G, and H)
12	Core damage	Unavailability of HPI following successful trip and AFW initiation, primary relief valve lift, and primary relief valve failure to reseat. (PWR Classes A, B, D, G, and H)
13	Core damage	Similar to sequence 11, but MFW provides SG cooling in lieu of AFW. (PWR Classes A, B, D, G, and H)
14	Core damage	Similar to sequence 12, but MFW provides SG cooling in lieu of AFW. (PWR Classes A, B, D, G, and H)
15	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is initiated, but the PORV fails to open. (PWR Classes A, B, and G)
16	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is initiated, but fails in the recirculation phase. (PWR Classes A, B, D, and G)
17	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed fails in the injection phase. (PWR Classes A, B, D, and G)
18	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models. (PWR Classes A, B, D, G, and H)
19	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is successful but CSR is unavailable. (PWR Class G)
20	Core damage	Unavailability of CSR following successful trip and AFW initiation, primary relief valve lift and failure to reseat, and successful HPI and HPR. (PWR Class A)
21	Core damage	Similar to sequence 11, but MFW provides SG cooling in lieu of AFW. (PWR Class A)
22	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is successful, but CSR is unavailable for containment heat removal. This sequence is distinguished from sequence 19 because of differences in the function of CSR on Class A and G plants. (PWR Class A)

Table A.4 PWR transient core damage and ATWS sequences

Sequence No.	End state	Description
23	Core damage	Unavailability of AFW and MFW following successful trip. The SGs are successfully depressurized, but the condensate pumps fail to provide SG cooling. (PWR Class H)
24	Core damage	Unavailability of AFW and MFW following successful trip, plus failure to depressurize the SGs to allow for the use of the condensate pumps for SG cooling. (PWR Class H)
25	Core damage	Unavailability of AFW and MFW following successful trip. At least one open SRV fails to reseat, but HPI and HPR are successful. SG depressurization is successful, but the condensate pumps fail to provide SG cooling. (PWR Class H)
26	Core damage	Similar to sequence 25 except that SG depressurization fails. (PWR Class H)
27	Core damage	Unavailability of AFW and MFW following successful trip. At least one SRV fails to reseat. HPI is initiated but HPR fails. (PWR Class H)
28	Core damage	Unavailability of AFW and MFW following successful trip. At least one SRV fails to reseat and HPI fails. (PWR Class H)

Table A.5 PWR transient sequences summary

Seq. No.	End State	RT	AFW	MFW	RV Chall	RV Reseat	HPI	HPR	PORV Open	CSR	SG Dep	Condensate Pumps	PWR Class A	B	D	G	H
11	CD	S	S		S*	F	S	F					x	x	x	x	x
12	CD	S	S		S*	F	F						x	x	x	x	x
13	CD	S	F	S	S*	F	S	F					x	x	x	x	x
14	CD	S	F	S	S*	F	F						x	x	x	x	x
15	CD	S	F	F			S	S	F				x	x		x	
16	CD	S	F	F			S	F					x	x	x	x	
17	CD	S	F	F			F						x	x	x	x	
18	ATWS	F											x	x	x	x	x
19	CD	S	F	F			S	S	S	F						x	
20	CD	S	S		S*	F	S	S		F			x				
21	CD	S	F	S	S*	F	S	S		F			x				
22	CD	S	F	F			S	S	S	F			x				
23	CD	S	F	F		S					S	F					x
24	CD	S	F	F		S					F						x
25	CD	S	F	F		F	S	S			S	F					x
26	CD	S	F	F		F	S	S			F						x
27	CD	S	F	F		F	S	F									x
28	CD	S	F	F		F	F										x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.6 PWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
40	ATWS	Failure to trip following a LOOP. (PWR Classes A, B, D, G, and H)
41	Core damage	Unavailability of HPR following a LOOP with successful trip, emergency power, and AFW; primary relief valve lift and failure to reseat; and successful HPI. (PWR Classes A, B, D, G, and H)
42	Core damage	Unavailability of HPI following LOOP with successful trip, emergency power, and AFW; primary relief valve lift and failure to reseat. (PWR Classes A, B, D, G, and H)
43	Core damage	Failure of the PORV to open for feed and bleed cooling following successful trip and emergency power, and AFW failure. (PWR Classes A, B, and G)
44	Core damage	Failure of HPR for recirculation cooling following feed and bleed initiation. Trip and emergency power are successful, but AFW fails. (PWR Classes A, B, D, and G)
45	Core damage	Unavailability of HPI for feed and bleed cooling following successful trip and emergency power and AFW failure. (PWR Classes A, B, D, and G)
46	Core damage	Unavailability of HPR following HPI success for RCP seal LOCA mitigation. AC power is recovered following successful trip, emergency power failure, turbine-driven AFW train(s) success, primary relief valve lift and reseat, and a subsequent seal LOCA. (PWR Classes A, B, D, G, and H)
47	Core damage	This sequence is similar to sequence 46 except that HPI fails for RCP seal LOCA mitigation. (PWR Classes A, B, D, G, and H)
48	Core damage	Failure to recover AC power following an RCP seal LOCA. The seal LOCA occurs following successful trip, failure of emergency power, turbine-driven AFW train(s) success, and primary relief valve lift and closure. (PWR Classes A, B, D, G, and H)
49	Core damage	Failure to recover AC power following successful trip and emergency power system failure, AFW turbine train(s) success, and primary relief valve lift and reseat. No RCP seal LOCA occurs in the sequence. (PWR Classes A, B, D, G, and H)
50	Core damage	Failure of a primary relief valve to reseat following lift subsequent to a successful trip, emergency power system failure, and AFW turbine trains(s) success. (PWR Classes A, B, D, G, and H)

Table A.6 PWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
51	Core damage	This sequence is similar to sequence 46 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
52	Core damage	This sequence is similar to sequence 47 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
53	Core damage	This sequence is similar to sequence 48 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
54	Core damage	This sequence is similar to sequence 49 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
55	Core damage	Failure of AFW following successful trip and emergency power system failure (PWR Classes A, B, D, G, and H)
56	Core damage	Failure of CSR in conjunction with successful feed and bleed following trip, emergency power system success, and AFW failure (PWR Class G)
57	Core damage	Failure of CSR following LOOP with successful trip, emergency power and AFW, primary relief valve challenge and failure to reseat, and successful HPI and HPR. (PWR Class A)
58	Core damage	Failure of CSR in conjunction with successful feed and bleed following LOOP with successful trip and emergency power initiation, and AFW failure. (PWR Class A)
59	Core damage	Failure of CSR following successful HPI and HPR required to mitigate a seal LOCA. This sequence involves a LOOP with successful trip, emergency power system failure, primary relief valve challenge and reseat, and a subsequent seal LOCA with AC power recovery prior to core uncovery. (PWR Class A)
60	Core damage	This sequence is similar to sequence 59 except that the primary relief valves are not challenged. (PWR Class A)
61	Core damage	Failure of AFW following a LOOP with successful trip and emergency power. (PWR Class H)

Table A.7 PWR LOOP sequences summary

Seq. No.	End State	RT/ LOOP	EP	AFW	RV Chall	RV Reseat	Seal LOCA	EP Recov	HPI	HPR	PORV Open	CSR	PWR Class A	B	D	G	H
40	ATWS	F											x	x	x	x	x
41	CD	S	S	S	S*	F			S	F			x	x	x	x	x
42	CD	S	S	S	S*	F			F				x	x	x	x	x
43	CD	S	S	F					S	S	F		x	x		x	
44	CD	S	S	F					S	F			x	x	x	x	
45	CD	S	S	F					F				x	x	x	x	
46	CD	S	F	S	S*	S	S*	S	S	F			x	x	x	x	x
47	CD	S	F	S	S*	S	S*	S	F				x	x	x	x	x
48	CD	S	F	S	S*	S	S*	F					x	x	x	x	x
49	CD	S	F	S	S*	S		F					x	x	x	x	x
50	CD	S	F	S	S*	F							x	x	x	x	x
51	CD	S	F	S			S*	S	S	F			x	x	x	x	x
52	CD	S	F	S			S*	S	F				x	x	x	x	x
53	CD	S	F	S			S*	F					x	x	x	x	x
54	CD	S	F	S				F					x	x	x	x	x
55	CD	S	F	F									x	x	x	x	x
56	CD	S	S	F					S	S	S	F				x	
57	CD	S	S	S	S*	F			S	S		F	x				
58	CD	S	S	F					S	S	S	F	x				
59	CD	S	F	S	S*	S	S*	S	S	S		F	x				
60	CD	S	F	S			S*	S	S	S		F	x				
61	CD	S	S	F													x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.8 PWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
71	Core damage	Unavailability of HPR following a small-break LOCA with trip, AFW and HPI success. (PWR Classes A, B, D, G, and H)
72	Core damage	Unavailability of HPI following a small-break LOCA with trip and AFW success. (PWR Classes A, B, D, G, and H)
73	Core damage	This sequence is similar to sequence 71 except that MFW is utilized for SG cooling is AFW is unavailable. (PWR Classes A, B, D, G, and H)
74	Core damage	This sequence is similar to sequence 72 except that MFW is utilized for SG cooling is AFW is unavailable. (PWR Classes A, B, D, G, and H)
75	Core damage	Unavailability of AFW and MFW following a small-break LOCA and successful trip. The PORV is unavailable to depressurize the RCS to the HPI pump discharge pressure. (PWR Classes A, B, and G)
76	Core damage	Unavailability of AFW and MFW following a small-break LOCA with trip success. HPI is successful but HPR fails. (PWR Classes A, B, D, G, and H)
77	Core damage	Unavailability of AFW and MFW following trip success. HPI fails to provide RCS makeup. (PWR Classes A, B, D, G, and H)
78	ATWS	Failure of reactor trip following a small-break LOCA. (PWR Classes A, B, D, G, and H)
79	Core damage	Unavailability of CSR for containment heat removal following a small-break LOCA with trip success. AFW and MFW failure, and feed and bleed success. (PWR Class G)
80	Core damage	Unavailability of CSR following a small-break LOCA with trip, AFW, HPI and HPR success. (PWR Class A)
81	Core damage	This sequence is similar to sequence 80 except that MFW is used for SG cooling in the event AFW is unavailable. (PWR Class A)
82	Core damage	Unavailability of CSR for containment heat removal following a small-break LOCA with trip success, AFW and MFW unavailability, and feed and bleed success. (PWR Class A)
83	Core damage	Unavailability of the condensate pumps for SG cooling following a small-break LOCA with trip success, unavailability of AFW and MFW, and successful SG depressurization. (PWR Class H)
84	Core damage	This sequence is similar to sequence 83 except that SG depressurization is unavailable. (PWR Class H)

Table A.9 PWR small-break LOCA sequences summary

Seq. No.	End State	RT	AFW	MFW	HPI	HPR	PORV Open	CSR	SG Dep	Condensate Pumps	PWR Class A	B	D	G	H
71	CD	S	S		S	F					x	x	x	x	x
72	CD	S	S		F						x	x	x	x	x
73	CD	S	F	S	S	F					x	x	x	x	x
74	CD	S	F	S	F						x	x	x	x	x
75	CD	S	F	F	S	S	F				x	x		x	
76	CD	S	F	F	S	F					x	x	x	x	x
77	CD	S	F	F	F						x	x	x	x	x
78	ATWS	F									x	x	x	x	x
79	CD	S	F	F	S	S	S	F						x	
80	CD	S	S		S	S		F			x				
81	CD	S	F	S	S	S		F			x				
82	CD	S	F	F	S	S	S	F			x				
83	CD	S	F	F	S	S			S	F					x
84	CD	S	F	F	S	S			F						x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
11	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, failure of isolation condenser, and successful main feedwater.
12	Core damage	Similar to Sequence 11 except failure of main feedwater and successful feedwater coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and feedwater coolant injection, followed by successful control rod drive cooling.
14	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, feedwater coolant injection and control rod drive cooling; followed by successful vessel depressurization and low-pressure core spray.
15	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and success of isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling. Successful vessel depressurization and failure of low-pressure core spray.
16	Core damage	Similar to Sequence 15 except the shutdown cooling system fails followed by successful containment cooling.
17	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization and failure of low-pressure core spray.
18	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, and safety relief valve challenge and successful reseat. Failure of the isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
19	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and successful main feedwater.
20	Core damage	Similar to Sequence 19 except unsuccessful main feedwater followed by successful feedwater coolant injection.
21	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief challenge and unsuccessful reseat, unsuccessful main feedwater and followed by successful vessel depressurization and low-pressure core spray.
22	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and feedwater coolant injection. Successful vessel depressurization and failure of low-pressure core spray.
23	Core damage	Similar to Sequence 22 except failure of the shutdown cooling system and successful containment spray.
24	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, unsuccessful main feedwater and feedwater coolant injection, successful vessel depressurization, and unsuccessful low-pressure core spray.
25	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and feedwater coolant injection.
26	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
27	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
28	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.
29	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
30	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
32	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
11	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, and failure of isolation condenser and successful main feedwater.
12	Core damage	Similar to Sequence 11 except failure of main feedwater followed by successful high-pressure coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and high-pressure coolant injection systems, followed by successful control rod drive cooling.
14	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, high-pressure coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization and low-pressure core spray.
15	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, high-pressure coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization, and failure of low-pressure core spray and successful low-pressure coolant injection.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
16	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; and failure of isolation condenser, main feedwater, high-pressure coolant injection, and control rod drive cooling systems. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling system.
17	Core damage	Similar to Sequence 16 except the shutdown cooling system fails followed by successful containment cooling mode of the low-pressure coolant injection system.
18	Core damage	Similar to Sequence 15 except low-pressure coolant injection system fails.
19	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, and safety relief valve challenge and successful reseat. Failure of the isolation condenser, main feedwater, high-pressure coolant injection, and control rod drive cooling.
20	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure injection) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and successful main feedwater.
21	Core damage	Similar to Sequence 20 except unsuccessful main feedwater followed by successful high-pressure coolant injection.
22	Core damage	Similar to Sequence 20 except unsuccessful main feedwater and high-pressure coolant injection, followed by successful vessel depressurization and low-pressure core spray.
23	Core damage	Similar to Sequence 20 except failure of main feedwater and high-pressure coolant injection, followed by successful vessel depressurization, failure of low-pressure core spray, and successful low-pressure coolant injection.
24	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
25	Core damage	Similar to Sequence 24 except failure of the shutdown cooling system and successful containment spray mode of low-pressure core injection.
26	Core damage	Similar to Sequence 23 except unsuccessful low-pressure coolant injection.
27	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and high-pressure coolant injection.
28	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
29	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
30	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.
32	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
34	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
35	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
36	Core damage	Similar to Sequence 19 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
11	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown cooling and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, and successful main feedwater.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
12	Core damage	Similar to Sequence 11 except failure of main feedwater with successful high-pressure coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and high-pressure coolant injection systems, with successful reactor core isolation cooling.
14	Core damage	Similar to Sequence 11 except failure of main feedwater, high-pressure coolant injection, and reactor core isolation cooling, with successful control rod drive cooling.
15	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown cooling and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, failure of main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling, with successful vessel depressurization and low-pressure core spray.
16	Core damage	Similar to Sequence 15 except failure of low-pressure core spray and successful low-pressure coolant injection.
17	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling systems. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal system in shutdown cooling mode.
18	Core damage	Similar to Sequence 17 except the residual heat removal system fails in the shutdown cooling mode and succeeds in the suppression pool cooling mode.
19	Core damage	Similar to Sequence 16 except failure of low-pressure coolant injection.
20	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat. Failure of the main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
21	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge with unsuccessful reseat, and successful main feedwater.
22	Core damage	Similar to Sequence 21 except unsuccessful main feedwater with successful high-pressure coolant injection.
23	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge with unsuccessful reseat, unsuccessful main feedwater and high-pressure coolant injection, followed by successful vessel depressurization and low-pressure core spray
24	Core damage	Similar to Sequence 23 except failure of low-pressure core spray and successful low-pressure coolant injection.
25	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal in shutdown cooling mode.
26	Core damage	Similar to Sequence 25 except the residual heat removal system fails in the shutdown cooling mode and succeeds in the suppression pool cooling mode.
27	Core damage	Similar to Sequence 24 except failure of low-pressure coolant injection.
28	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and high-pressure coolant injection systems.
29	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
30	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
32	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
34	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
35	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
36	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
37	Core damage	Similar to Sequence 19 except the safety relief valves are not challenged.
38	Core damage	Similar to Sequence 20 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
41	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat. Failure of isolation condenser and successful feedwater coolant injection.
42	Core damage	Similar to Sequence 41 except failure of the feedwater coolant injection and successful control rod drive cooling.
43	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of the feedwater coolant injection and control rod drive cooling systems, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of isolation condenser, feedwater coolant injection, and control rod drive cooling. Successful vessel depressurization and failure of low-pressure core spray.
45	Core damage	Similar to Sequence 44 except failure of the shutdown cooling system and successful containment spray.
46	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of feedwater coolant injection and control rod drive cooling, with successful vessel depressurization and failure of the low-pressure core spray.
47	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Challenge of the safety relief valves and successful reseat with unsuccessful isolation condenser, feedwater coolant injection, and control rod drive cooling.
48	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat, and successful feedwater coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
49	Core damage	Similar to Sequence 48 except failure of feedwater coolant injection followed by successful vessel depressurization and low-pressure core spray.
50	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of feedwater coolant injection. Successful vessel depressurization, failure of low-pressure core spray, and successful shutdown cooling system.
51	Core damage	Similar to Sequence 50 except failure of shutdown cooling system and successful containment cooling.
52	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat. Failure of feedwater coolant injection, successful vessel depressurization, and failure of low-pressure core spray.
53	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the feedwater coolant injection system.
54	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
55	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
56	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged..
60	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged
61	Core damage	Unavailability of the isolation condenser following a loss of offsite power, failure of emergency power, successful scram, and safety relief valve challenge and successful reseat.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
62	Core damage	Failure of an SRV to reseat following challenge after a loss of offsite power with failure of emergency power and successful reactor scram.
63	Core damage	Similar to Sequence 61 except the safety relief valves are not challenged.
64	Core damage	Failure of recovery of electric power in the long-term following a loss of offsite power, failure of emergency power, and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
41	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser and successful high-pressure coolant injection.
42	Core damage	Similar to Sequence 41 except failure of high-pressure coolant injection and successful control rod drive cooling.
43	Core damage	Similar to Sequence 41 except failure of the high-pressure coolant injection and control rod drive cooling, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of the high-pressure coolant injection and control rod drive cooling systems, with successful vessel depressurization, failure of low-pressure core spray, and successful low-pressure coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
45	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of isolation condenser, high-pressure coolant injection, and control rod drive cooling. Successful vessel depressurization, failure of low-pressure core spray, and low-pressure coolant injection with successful shutdown cooling.
46	Core damage	Similar to Sequence 45 except failure of the shutdown cooling system and successful containment spray mode low-pressure coolant injection.
47	Core damage	Similar to Sequence 44 except failure of low-pressure coolant injection.
48	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram, challenge of the safety relief valves and successful reseat with unsuccessful isolation condenser, high-pressure coolant injection, and control rod drive cooling.
49	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat, and successful high-pressure coolant injection.
50	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection followed by successful vessel depressurization and low-pressure core spray.
51	Core damage	Similar to Sequence 50 except failure of low-pressure core spray and successful low-pressure coolant injection.
52	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure core injection, and successful shutdown cooling system.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
53	Core damage	Similar to Sequence 52 except failure of shutdown cooling system and successful containment cooling mode of low-pressure coolant injection.
54	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat. Failure of high-pressure coolant injection, successful vessel depressurization and failure of low-pressure core spray and low-pressure coolant injection.
55	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the high-pressure coolant injection system.
56	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
60	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
61	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged.
62	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged.
63	Core damage	Similar to Sequence 48 except the safety relief valves are not challenged.
64	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, failed isolation condenser, and successful high-pressure coolant injection.
65	Core damage	Unavailability of high-pressure core injection following a loss of offsite power, failure of emergency power, successful reactor scram, safety relief valve challenge and reseat, and failed isolation condenser and high-pressure coolant injection systems.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
66	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and failure to reseat, and successful high-pressure coolant injection.
67	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and failure to reseat, and failure of high-pressure coolant injection.
68	Core damage	Similar to Sequence 64 except the safety relief valves are not challenged.
69	Core damage	Similar to Sequence 65 except the safety relief valves are not challenged.
84	Core damage	Failure of long-term recovery of electric power following a loss of offsite power, with failure of emergency power and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
40	Core damage	Unavailability of long-term core cooling (failure of residual heat removal in shutdown and suppression cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat, and successful high-pressure coolant injection.
41	Core damage	Similar to Sequence 40 except failure of the high-pressure coolant injection system and successful reactor core isolation cooling.
42	Core damage	Similar to Sequence 40 except failure of the high-pressure coolant injection and reactor core isolation cooling systems with successful control rod drive cooling.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
43	Core damage	Unavailability of long-term core cooling (failure of residual heat removal in shutdown and suppression cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat; failure of the high-pressure coolant injection, reactor core isolation cooling and control rod drive cooling systems, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Similar to Sequence 43 except failure of low-pressure core spray and successful low-pressure coolant injection.
45	Core damage	Unavailability of fire water or other equivalent water source for reactor makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling systems. Successful vessel depressurization, and failure of low-pressure core spray and low-pressure coolant injection with successful residual heat removal in shutdown cooling mode.
46	Core damage	Similar to Sequence 45 except failure of the residual heat removal system in shutdown cooling mode and success in suppression pool cooling mode.
47	Core damage	Similar to Sequence 44 except failure of low-pressure coolant injection.
48	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Challenge of the safety relief valves and successful reseat with high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling.
49	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and successful high-pressure coolant injection.
50	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection followed by successful vessel depressurization and low-pressure core spray
51	Core damage	Similar to Sequence 50 except failure of low-pressure core spray and successful low-pressure coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
52	Core damage	Unavailability of fire water or other equivalent water source following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal in shutdown cooling mode.
53	Core damage	Similar to Sequence 52 except failure of the residual heat removal system in shutdown cooling mode and success in suppression pool cooling mode.
54	Core damage	Similar to Sequence 51 except failure of low-pressure coolant injection.
55	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the high-pressure coolant injection system.
56	Core damage	Similar to Sequence 40 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
60	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
61	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
62	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged.
63	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged.
64	Core damage	Similar to Sequence 48 except the safety relief valves are not challenged.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
65	Core damage	Unavailability of long-term core cooling (failure of the residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, and successful high-pressure coolant injection.
66	Core damage	Similar to Sequence 65 except high-pressure coolant injection fails with successful reactor core isolation cooling.
67	Core damage	Unavailability of long-term core cooling (failure of the residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, with failures of high-pressure coolant injection and reactor core isolation cooling.
68	Core damage	Similar to Sequence 65 except the safety relief valves fail to reseat.
69	Core damage	Failure of high-pressure coolant injection following a loss of offsite power, with emergency power failure, successful reactor scram, safety relief valve challenge, and unsuccessful reseat.
80	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram and long-term recovery of electric power. The safety relief valves are not challenged, and high-pressure coolant injection is successful.
81	Core damage	Similar to Sequence 66 except the safety relief valves are not challenged.
82	Core damage	Similar to Sequence 67 except the safety relief valves are not challenged.
83	Core damage	Unable to recover long-term electric power following a loss of offsite power, failure of emergency power, and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.

Table A.12 BWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
71	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss-of-coolant accident, successful scram, and successful feedwater coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss-of-coolant accident, successful scram, failure of feedwater coolant injection system, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of feedwater coolant injection. Successful vessel depressurization and failure of low-pressure core spray, and successful shutdown cooling system.
74	Core damage	Similar to Sequence 73 except failure of the shutdown cooling system and successful containment cooling.
75	Core damage	Similar to Sequence 72 except failure of the low-pressure core spray.
76	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the feedwater coolant injection system.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
71	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss-of-coolant accident, successful scram, and successful high-pressure coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss-of-coolant accident, successful scram, failure of high-pressure coolant injection, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Similar to Sequence 72 except failure of low-pressure core spray and successful low-pressure coolant injection.
74	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling system.

Table A.12 BWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
75	Core damage	Similar to Sequence 74 except failure of the shutdown cooling system and successful containment cooling mode of low-pressure coolant injection.
76	Core damage	Similar to Sequence 73 except failure of low-pressure coolant injection.
77	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
71	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss-of-coolant accident, successful scram, and successful high-pressure coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss-of-coolant accident, successful scram, failure of the high-pressure coolant injection system, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Similar to Sequence 72 except failure of low-pressure core spray, and successful low-pressure coolant injection.
74	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal system in shutdown cooling mode.
75	Core damage	Similar to Sequence 74 except failure of the residual heat removal system in the shutdown cooling mode and success in the suppression pool cooling mode.
76	Core damage	Similar to Sequence 73 except failure of low-pressure coolant injection.
77	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.

Table A.13 Average initiating event frequency and branch failure probability estimates developed from 1984-1986 precursors.

Initiator/branch	Initial estimate (no recovery attempted)	Nonrecovery estimate	Total
	PWRs		
LOOP	4.1×10^{-2}/year	0.39	1.6×10^{-2}/year[a]
Small-break LOCA	1.5×10^{-2}/year	0.43	6.4×10^{-3}/year
Auxiliary feedwater	3.8×10^{-4}	0.26	9.9×10^{-5}
High-pressure injection	6.1×10^{-4}	0.84	5.1×10^{-4}
Long-term core cooling (high-pressure recirculation)	1.5×10^{-4}	1.00	1.5×10^{-4}
Emergency power	6.4×10^{-4}	0.78	5.0×10^{-4}
SG isolation (MSIVs)	8.3×10^{-4}	0.64	5.3×10^{-4}
	BWRs		
LOOP	1.0×10^{-1}/year	0.32	3.3×10^{-2}/year
Small-break LOCA	2.0×10^{-2}/year	0.50	1.0×10^{-2}/year
HPCI/RCIC	1.7×10^{-3}	0.49	8.4×10^{-4}
RV isolation	1.7×10^{-3}	1.00	1.7×10^{-3}
LPCI	1.0×10^{-3}	0.71	7.4×10^{-4}
Emergency power	1.0×10^{-4}	0.85	8.9×10^{-5}
Automatic depressurization	3.7×10^{-3}	0.71	2.6×10^{-3}

[a]Precursor calculations utilize plant-specific LOOP frequency estimates developed from information in P.W. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, June 1988.

Table A.14 Operator action failure probabilities

Operation action	Failure probability
BWRs	
Condensate/feedwater recovery	0.001
Containment venting	0.01
Control rod drive water use	0.01
Initiation of RHR service water, fire water	0.01
Shutdown cooling	0.001
Standby liquid control initiation	0.01
PWRs	
Condensate/MFW recovery	0.01
Containment spray recirculation	0.001
Emergency core cooling recirculation	0.001
Fail to block stuck-open PORVs	0.001
Open PORVs for feed and bleed	0.0004
SG depressurization	0.001
Use feed and bleed to cool core	0.01

Table A.15 Reference event conditional probability values

Postulated operational event	Conditional core damage probability
BWR Class A nonspecific reactor trip	2.8×10^{-6}
BWR Class A LOFW	1.7×10^{-4}
BWR Class B nonspecific reactor trip	7.7×10^{-8}
BWR Class B LOFW	4.3×10^{-6}
BWR Class C (turbine-driven feed pumps) nonspecific reactor trip	1.2×10^{-6}
BWR Class C (turbine-driven feed pumps) LOFW	1.5×10^{-5}
PWR Class A nonspecific reactor trip	1.8×10^{-7}
PWR Class A LOFW	2.4×10^{-6}
PWR Class B nonspecific reactor trip	1.8×10^{-7}
PWR Class B LOFW	2.2×10^{-6}
PWR Class D nonspecific reactor trip	4.7×10^{-7}
PWR Class D LOFW	6.8×10^{-6}
PWR Class G nonspecific reactor trip	1.8×10^{-7}
PWR Class G LOFW	2.4×10^{-6}
PWR Class H nonspecific reactor trip	4.9×10^{-6}
PWR Class H LOFW	3.9×10^{-5}
BWR Class C HPCI unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	1.0×10^{-5}
BWR Class C HPCS unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	1.4×10^{-5}
BWR Class C RCIC unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	3.8×10^{-8}
BWR Class C CRD cooling unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	6.2×10^{-8}

[a]The probability of a transient, LOOP, or small-break LOCA during the 360-h unavailability was estimated as described in Sect. A.1.

Table A.16 Abbreviations used in event trees

Abbreviation	Description
	PWR event trees
AFW	auxiliary feedwater fails
ATWS	anticipated transient without scram end state
COND	condensate system fails
CD	core damage end state
CSR	containment spray recirculation fails
EP	emergency power fails
EP REC (LONG)	long-term recovery from LOOP or emergency power failure fails
HPI	high-pressure injection fails
HPR	high-pressure recirculation fails
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
MFW	main feedwater fails
PORV OPEN	power-operated relief valve fails to open for feed and bleed cooling
PORV/SRV CHALL	power-operated relief valve or safety relief valves challenged (challenge rate)
PORV/SRV RESEAT	power-operated relief valve and/or safety relief valve fails to reseat
RT	reactor trip fails
RT/LOOP	reactor trip fails given a loss of offsite power
SEAL LOCA	RCP seal LOCA occurs
SEC SIDE DEP	secondary-side depressurization fails
SEQ NO	sequence number
SRV CHALL	safety relief valves challenged
SRV RESEAT	safety relief valve fails to reseat
TRANS	nonspecific reactor-trip transient

Table A.16 Abbreviations used in event trees

Abbreviation	Description
	BWR Event Trees
CC	containment cooling fails
CRD	control-rod-drive cooling fails
EP	emergency power fails
FIREWTR or OTHER	fire water or other equivalent water source fails
FW	unavailabilty of main feedwater
FWCI	failure of feedwater coolant injection system
HPCI OR HPCS	high-pressure coolant injection or high-pressure core spray fails
IC/IP MUP	isolation condenser or isolation condenser makeup fails
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
LOOP REC (LONG)	long-term recovery from LOOP or emergency power failure fails
LPCI	low-pressure coolant injection fails
LPCI (CC MODE)	containment cooling mode of low-pressure coolant injection system fails
LPCI (RHR)	residual heat removal mode of low-pressure coolant injection core spray fails
LPCS	low-pressure core spray fails
PCS	failure of continued power conversion system operation
RCIC	reactor core isolation cooling fails
RHR (SDC MODE)	residual-heat-removal shutdown cooling mode fails
RHR (SP COOLING MODE)	residual-heat-removal suppression pool cooling mode fails
RHR SW or OTHER	residual-heat-removal service water or other water source fails
RX SHUTDOWN	reactor fails to scram
SDC	shutdown cooling system fails
SRVs/ADS	safety relief valve(s) fail to open for depressurization or automatic depressurization system fails
SRV CHAL	safety relief valve(s) challenged (challenge rate)
SRV-C	safety relief valve fails to close
TRANSIENT	nonspecific reactor-trip transient

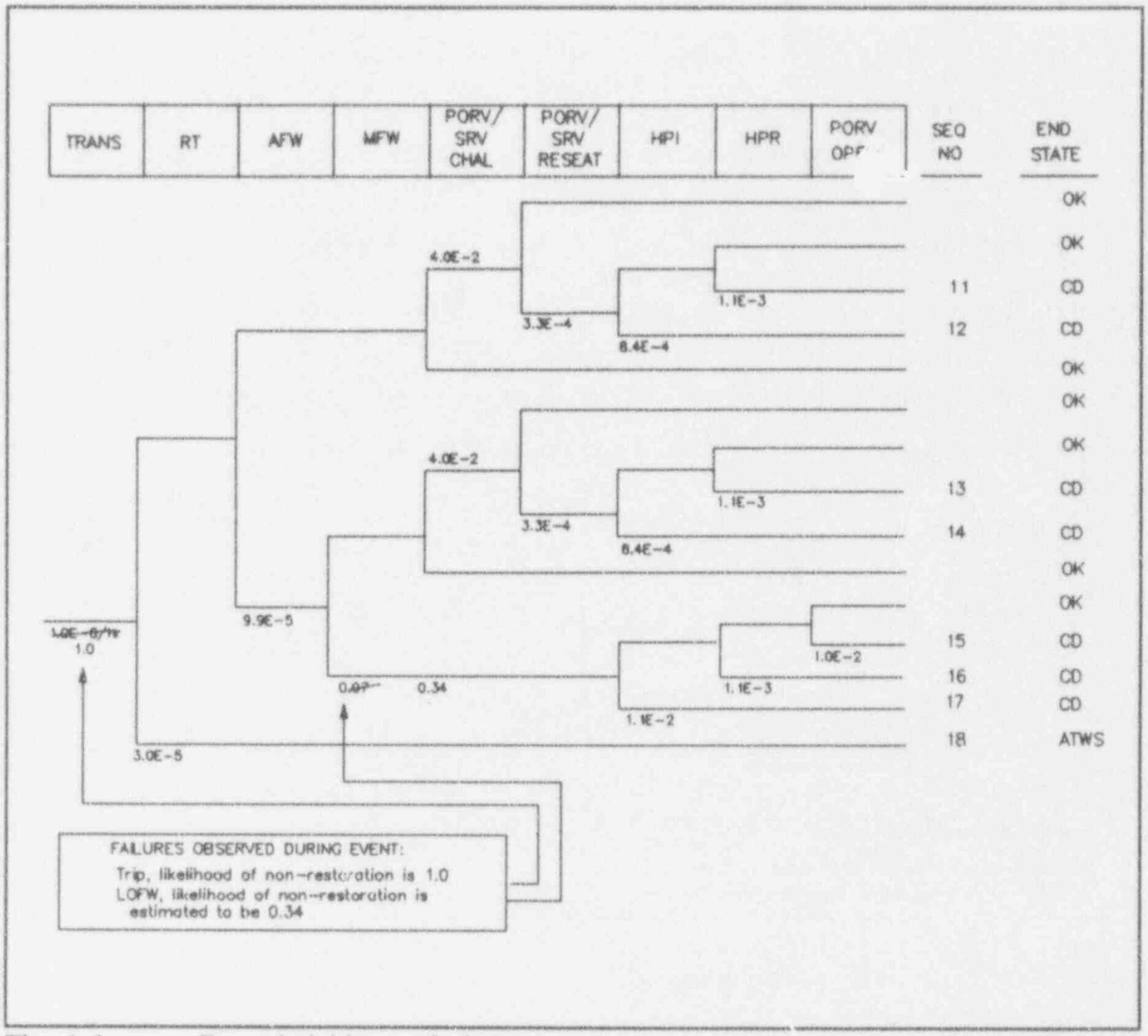

Fig. A.1. Example initiator calculation.

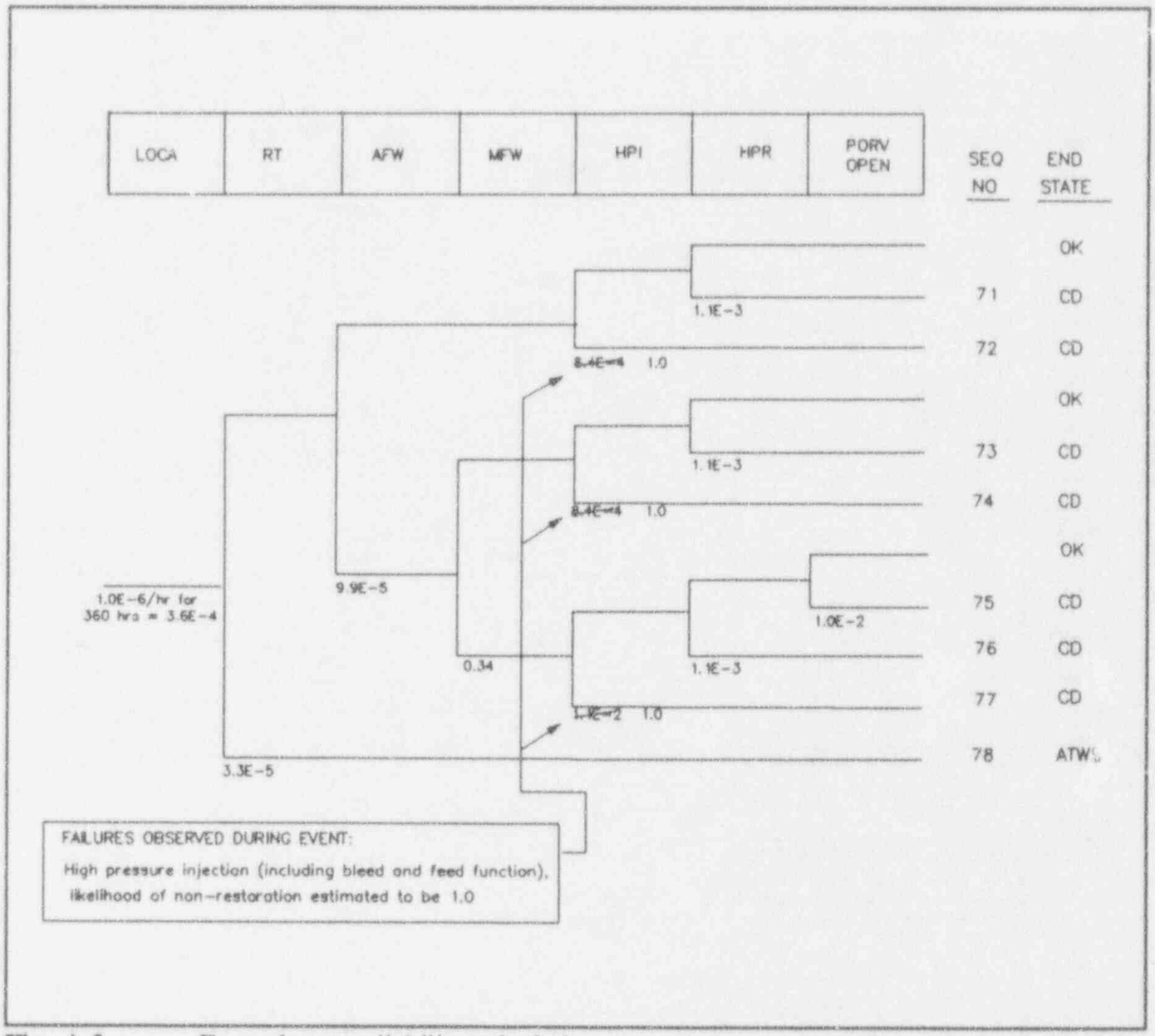

Fig. A.2. Example unavailability calculation

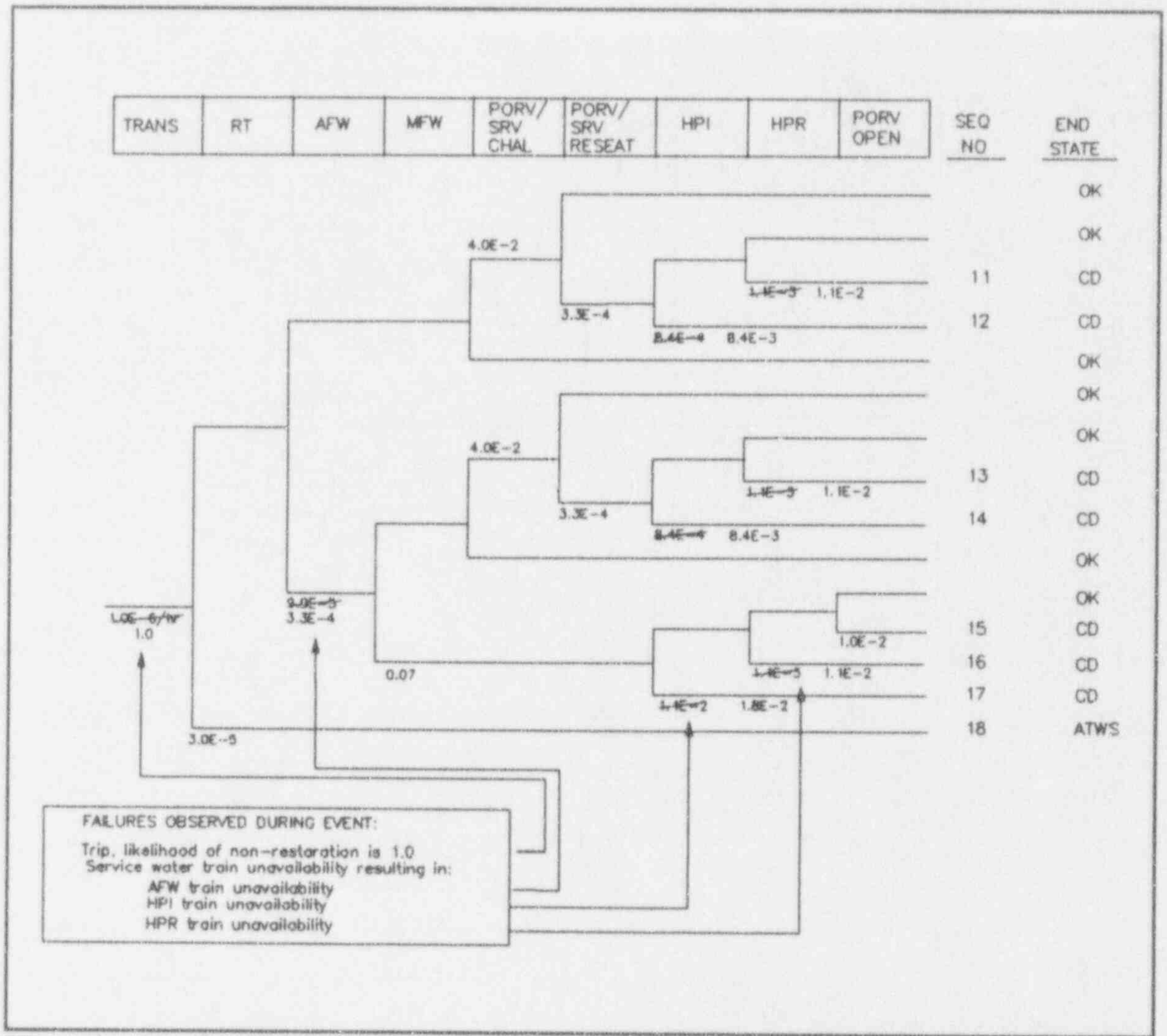

Fig. A.3. Example trip with support system degraded

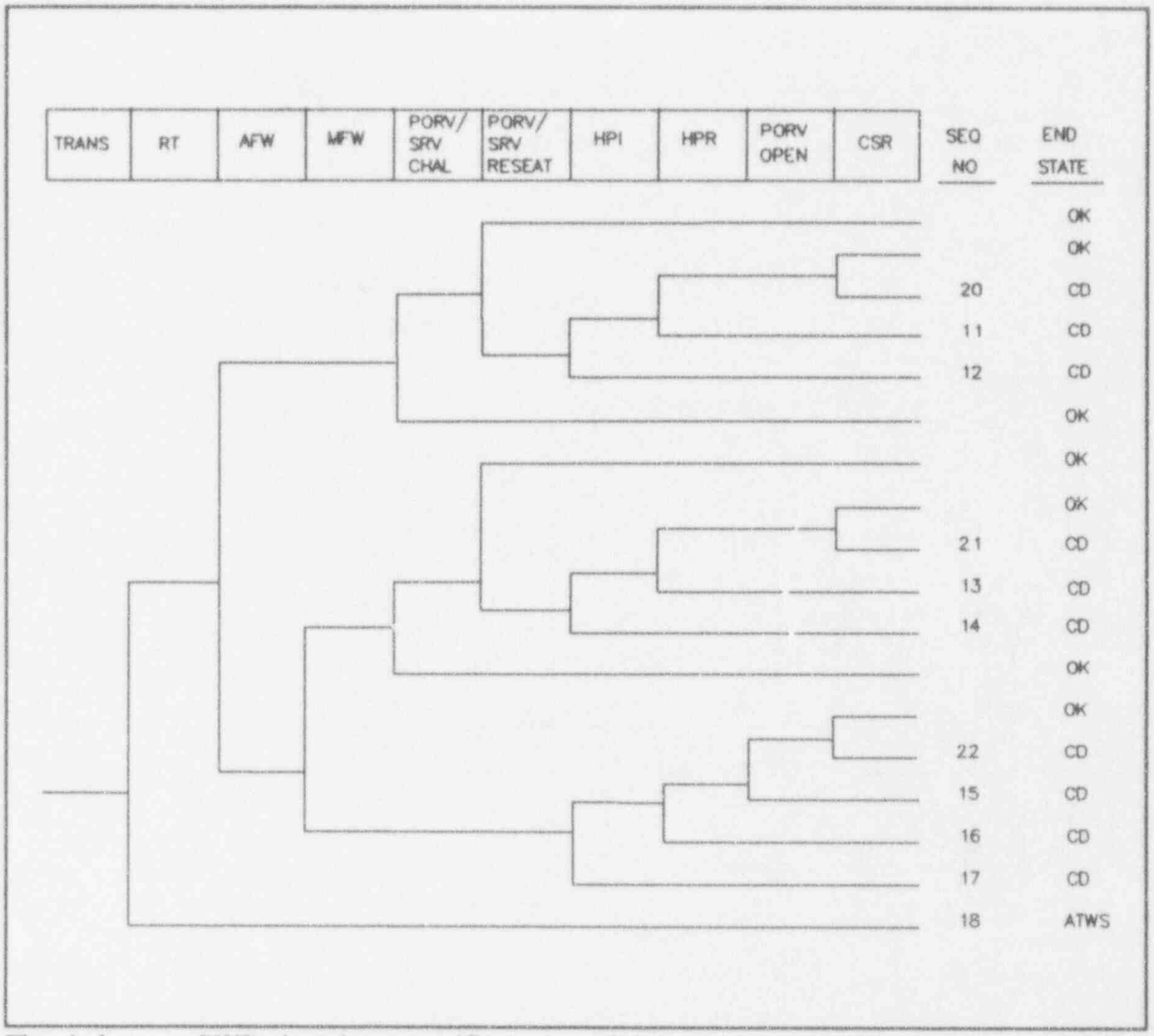

Fig. A.4. PWR class A nonspecific reactor trip

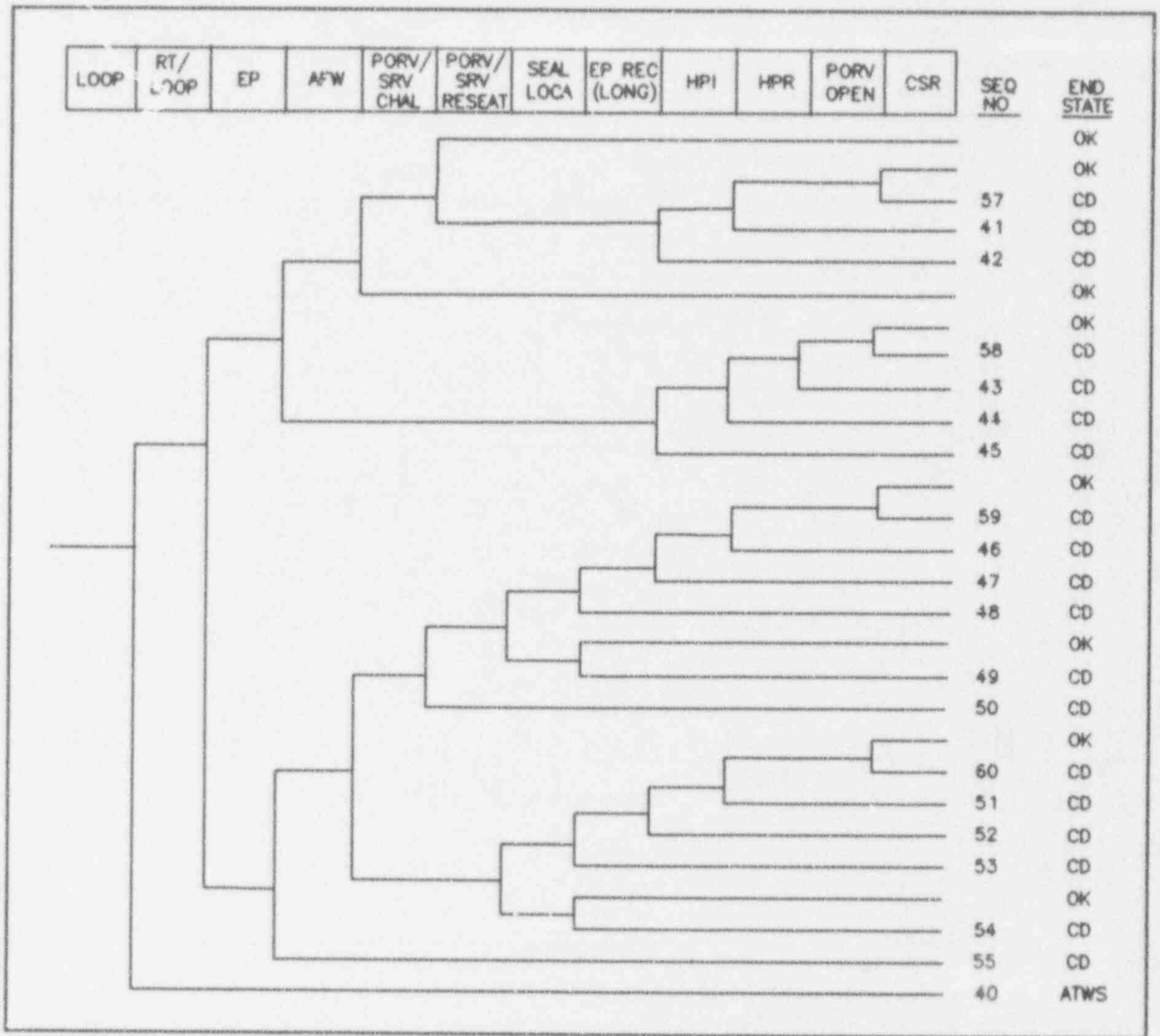

Fig. A.5. PWR class A loss of offsite power

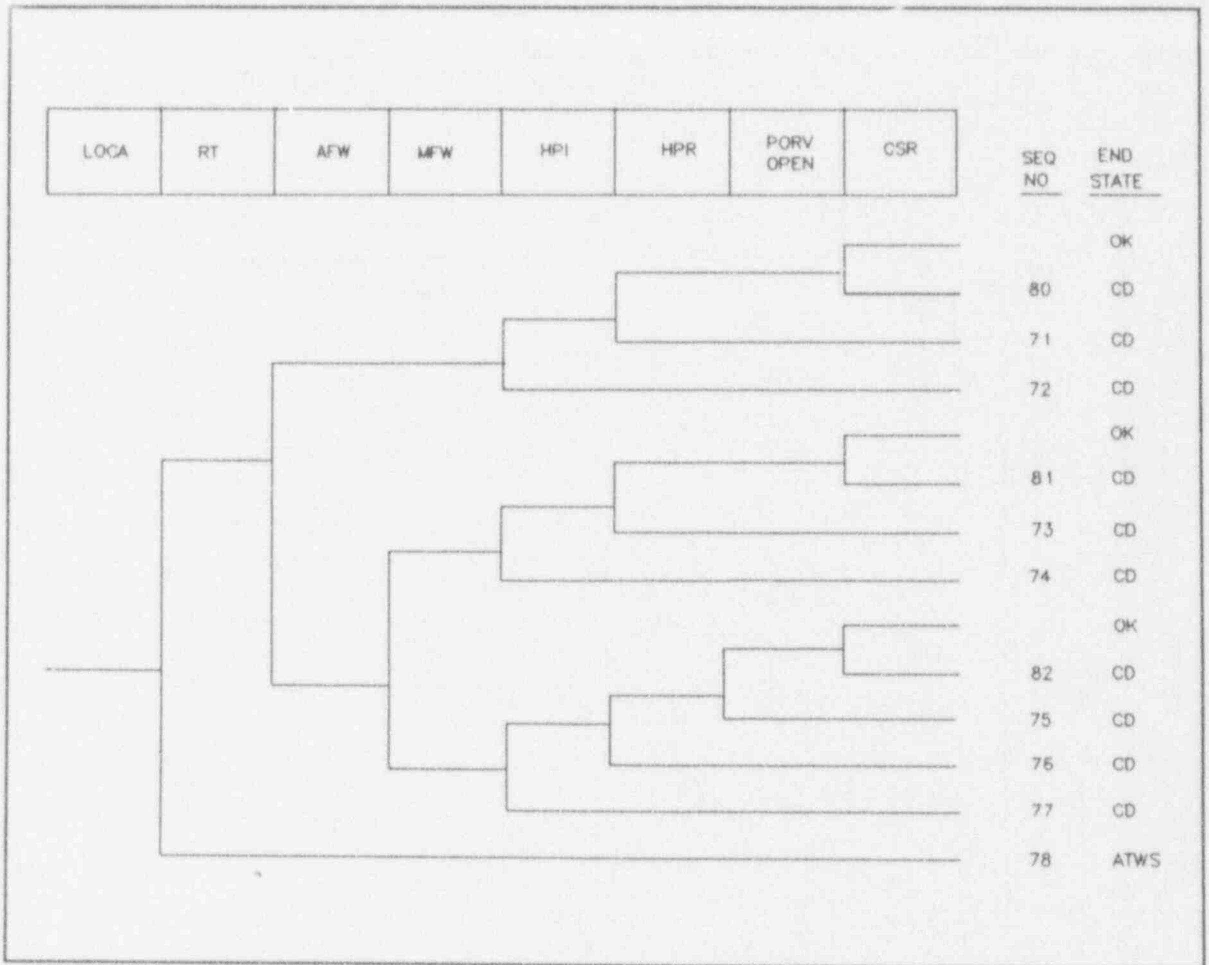

Fig. A.6. PWR class A small-break loss-of-coolant accident

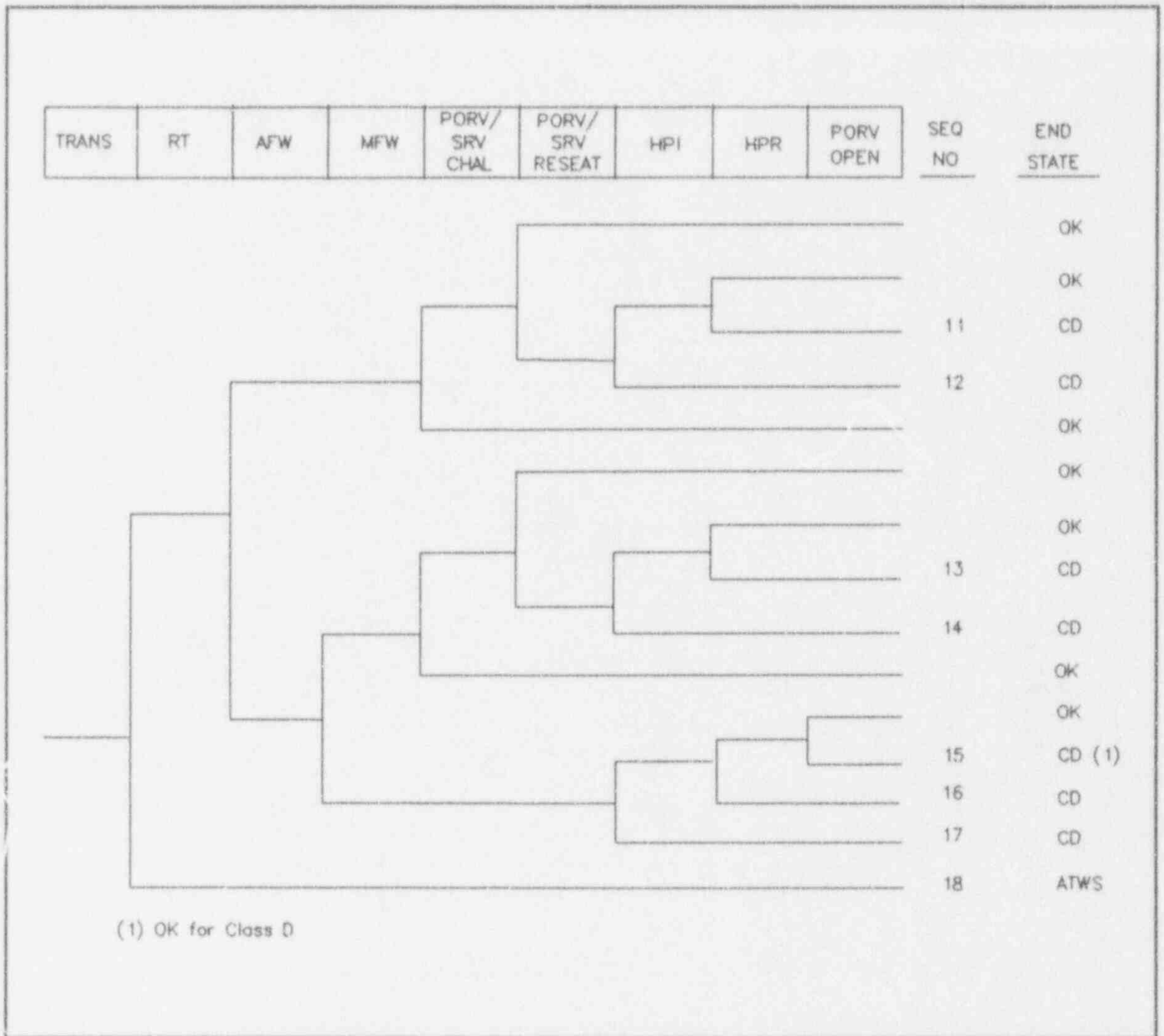

Fig. A.7. PWR class B and D nonspecific reactor trip

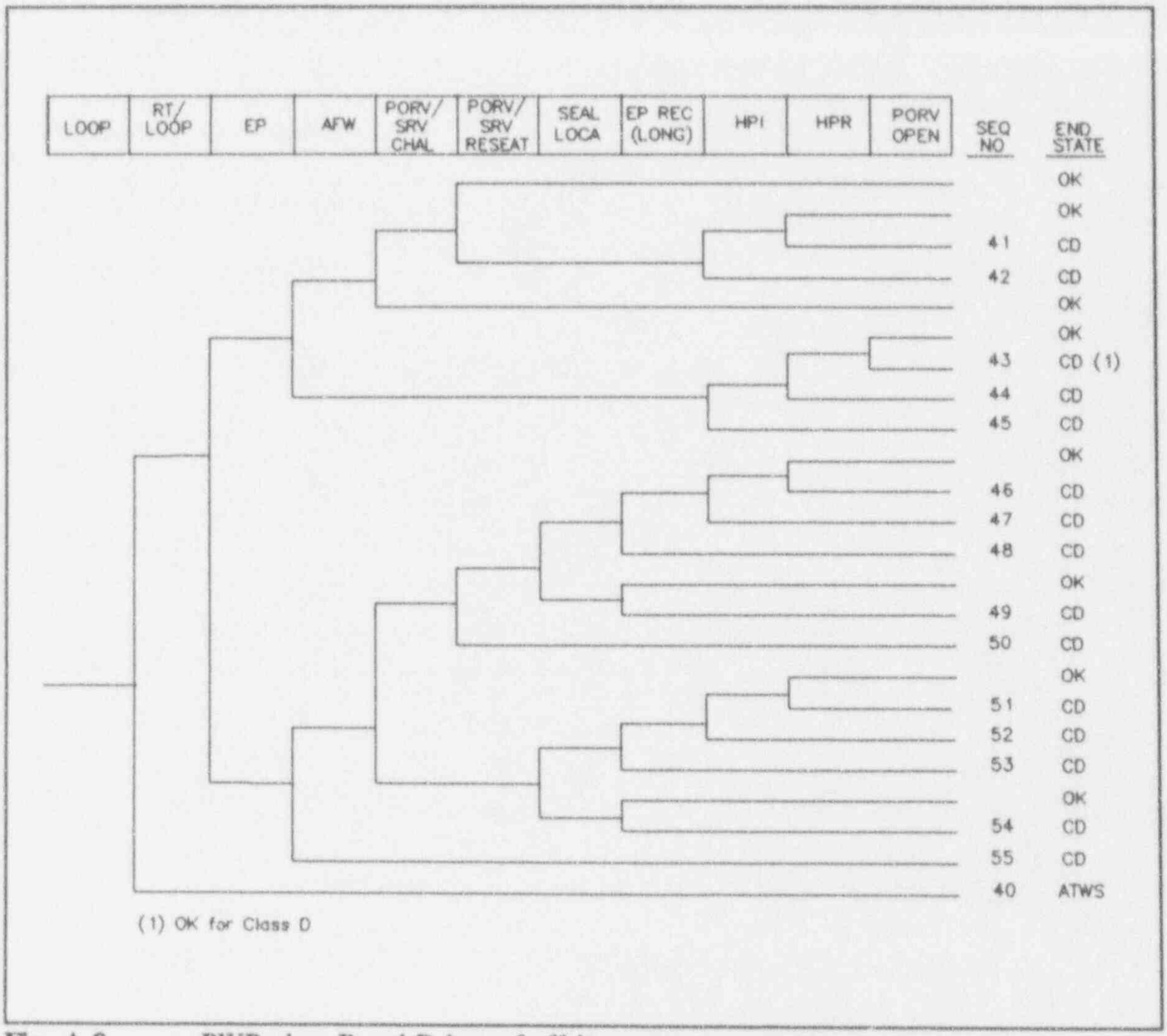

Fig. A.8. PWR class B and D loss of offsite power

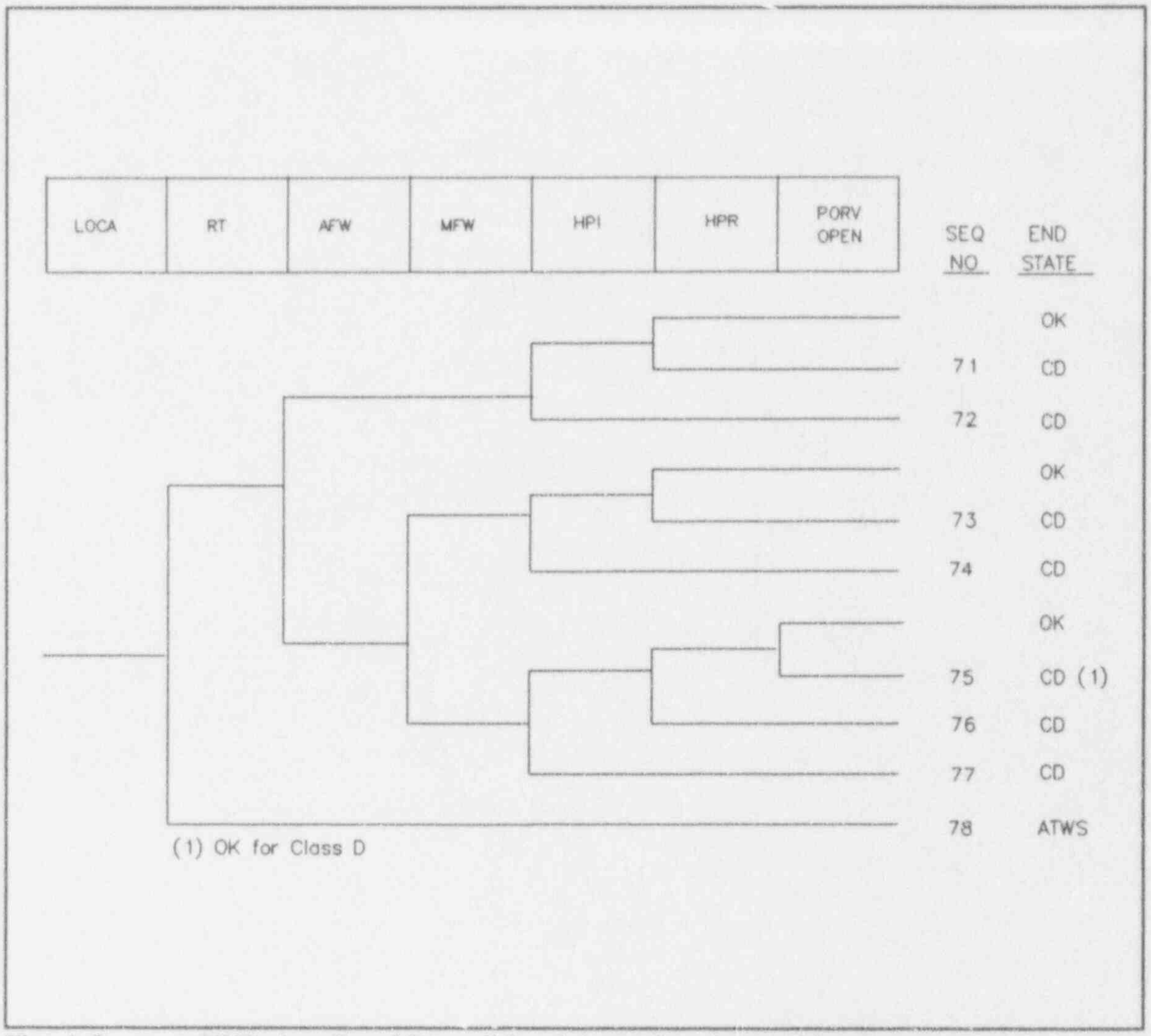

Fig. A.9. PWR class B and D small-break loss-of-coolant accident

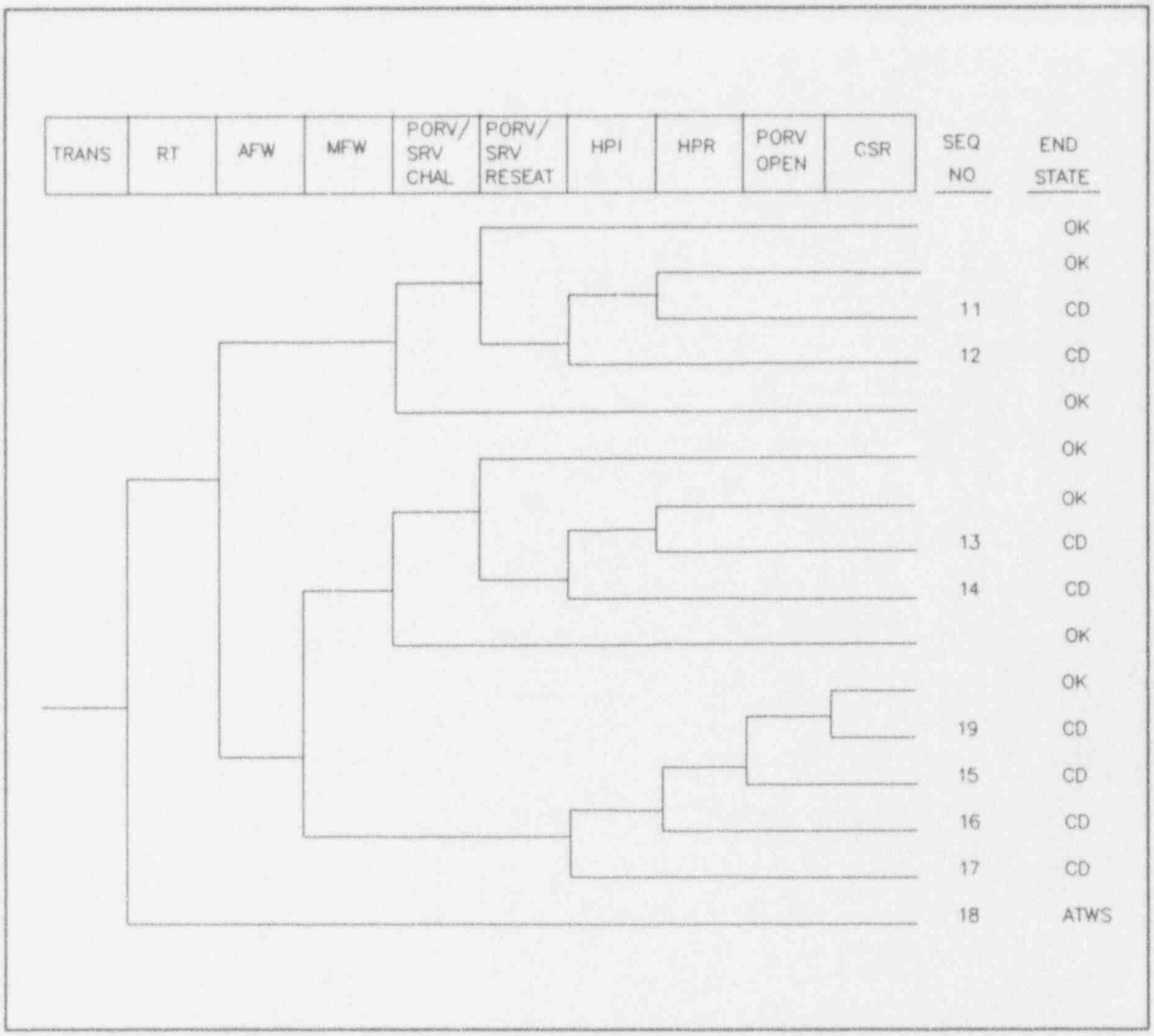

Fig. A.10. PWR class G nonspecific reactor trip

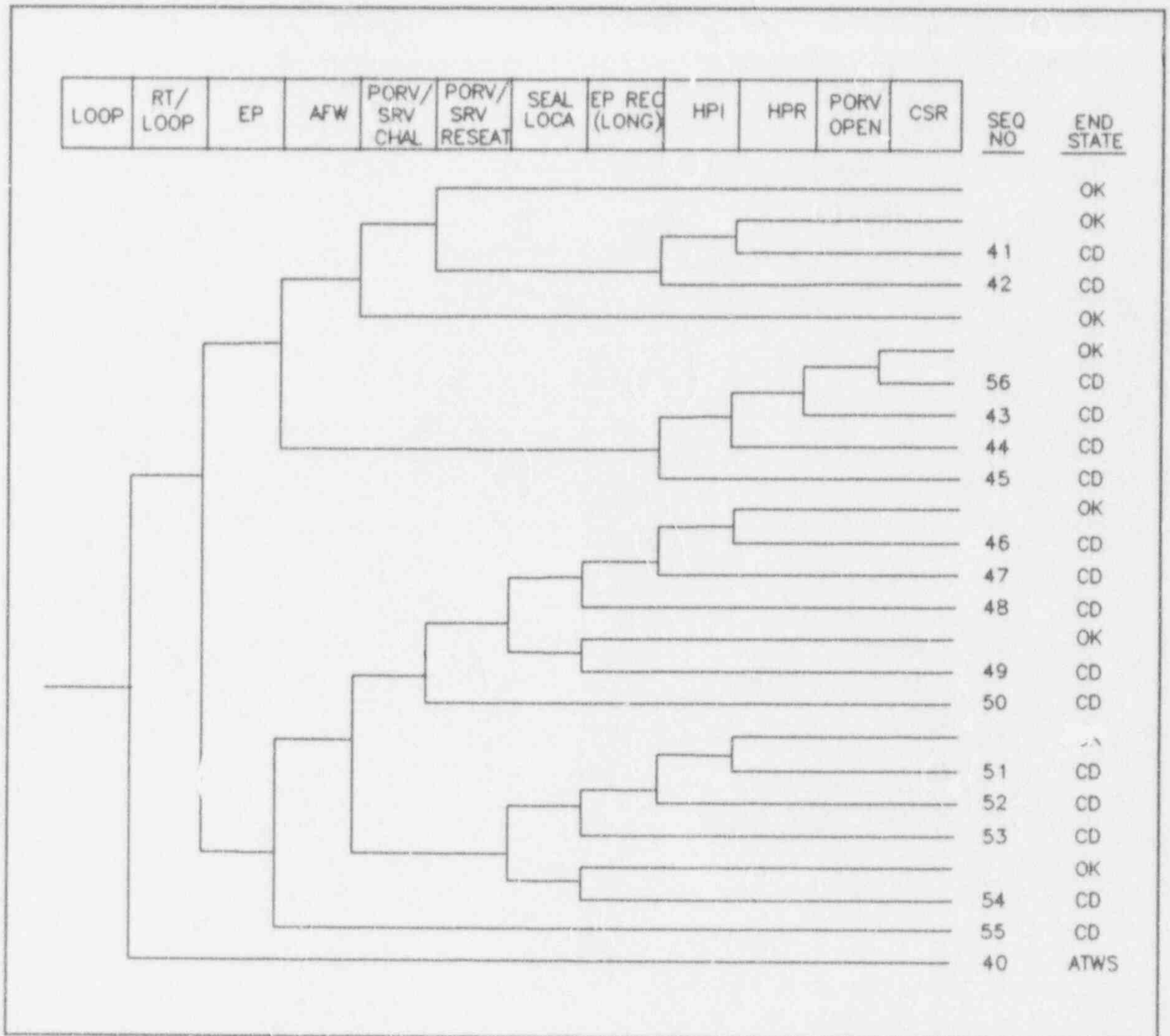

Fig. A.11. PWR class G loss of offsite power

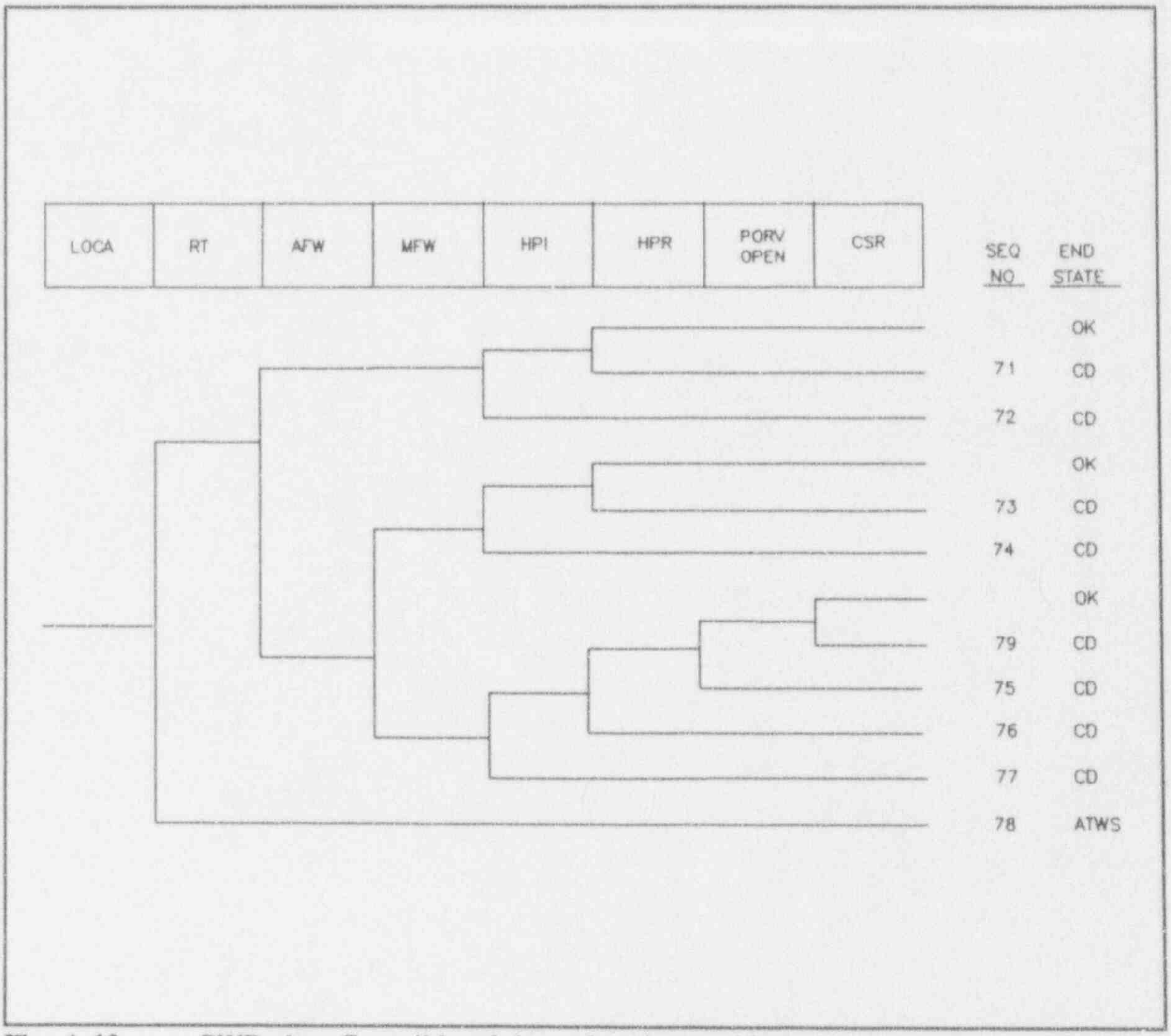

Fig. A.12. PWR class G small-break loss-of-coolant accident

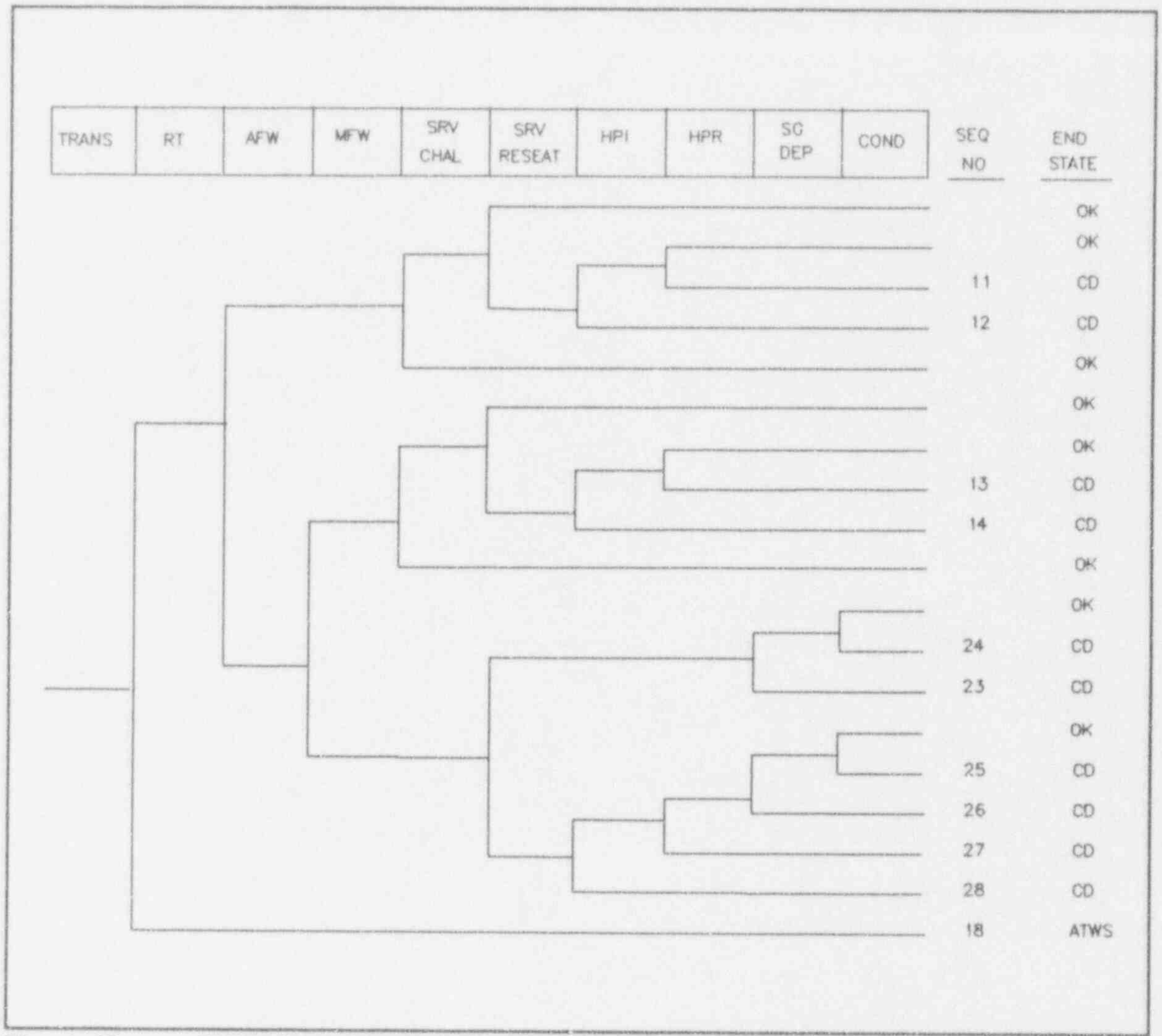

Fig. A.13. PWR class H nonspecific reactor trip

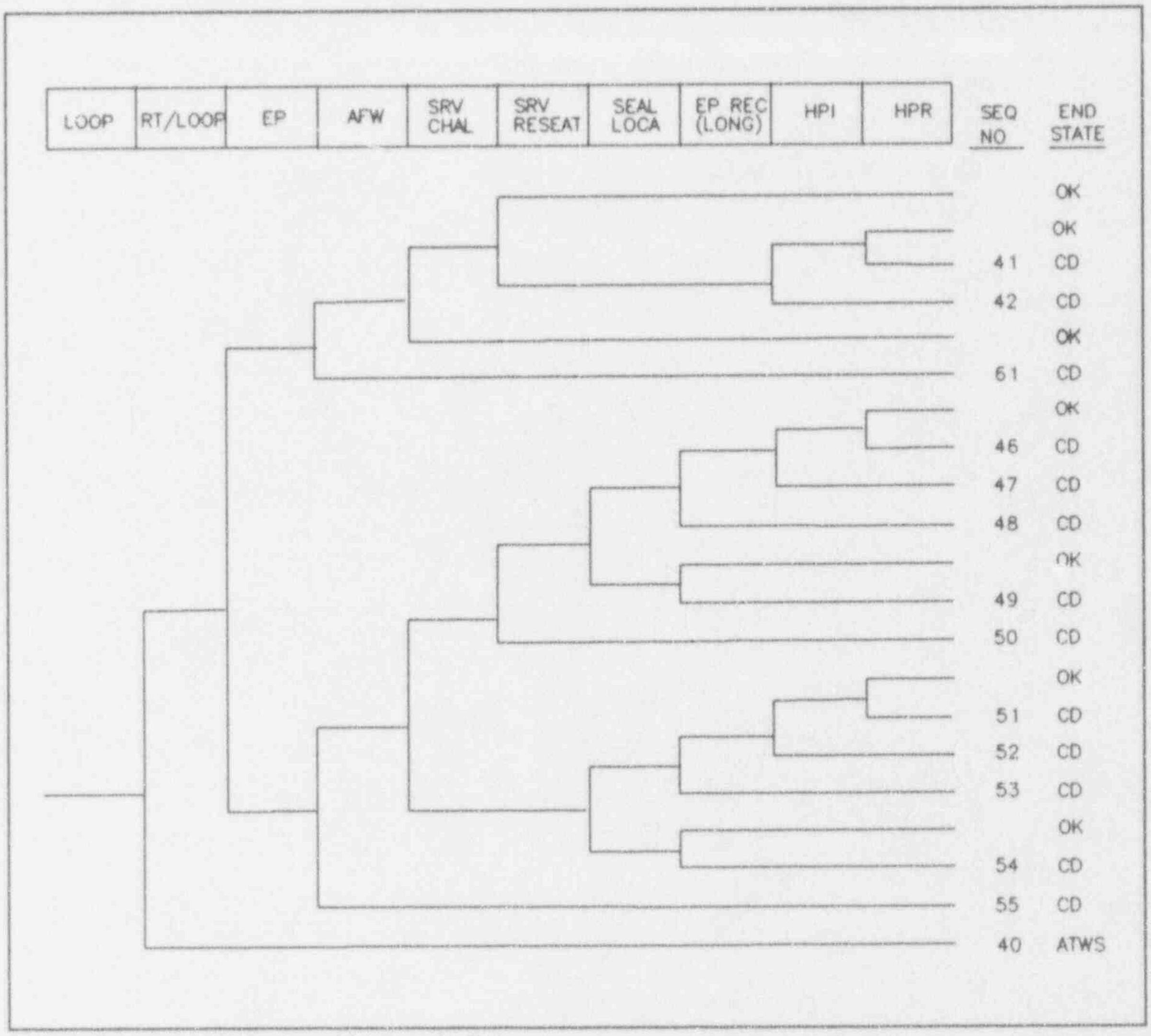

Fig. A.14. PWR class H loss of offsite power

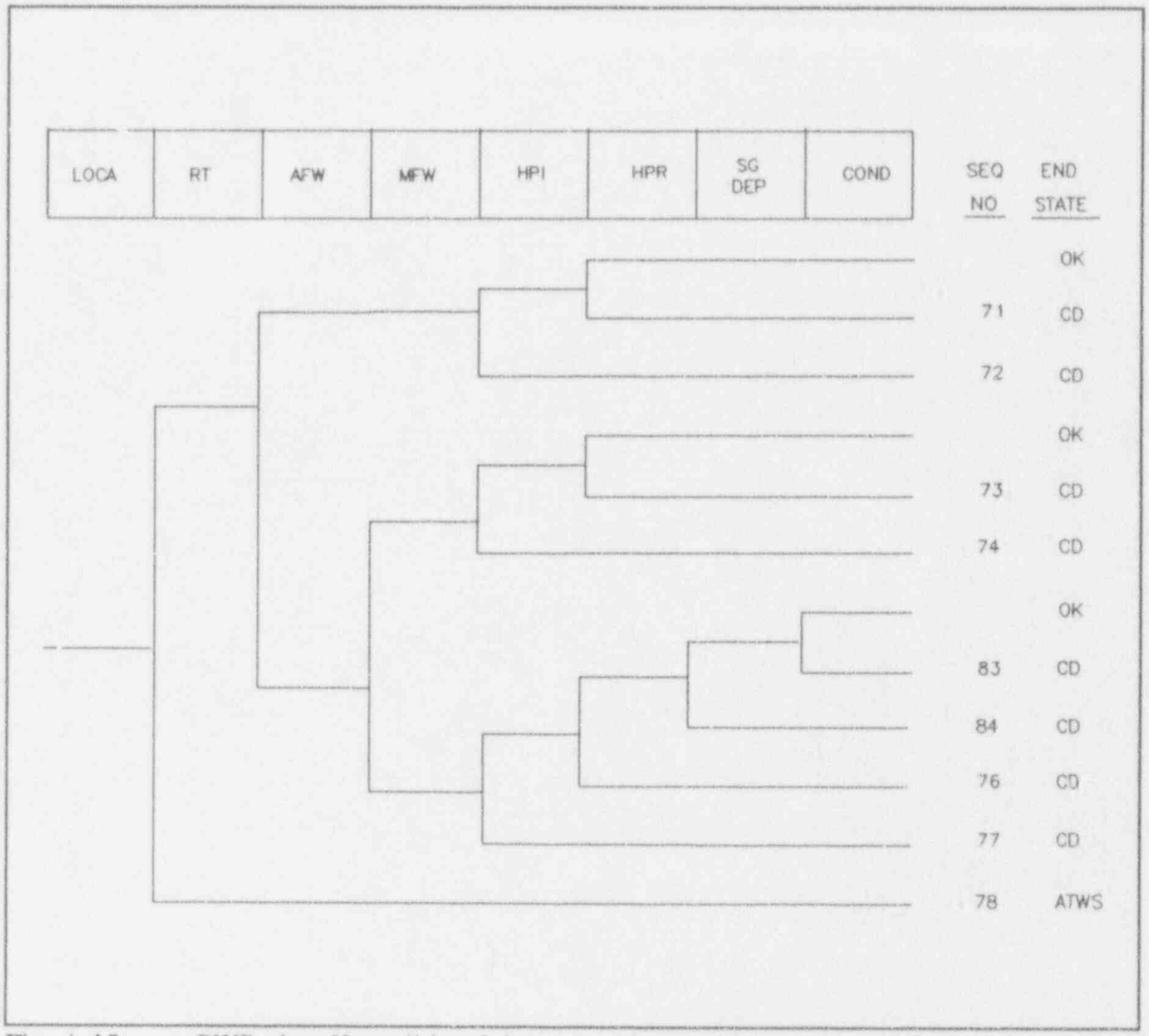

Fig. A.15. PWR class H small-break loss-of-coolant accident

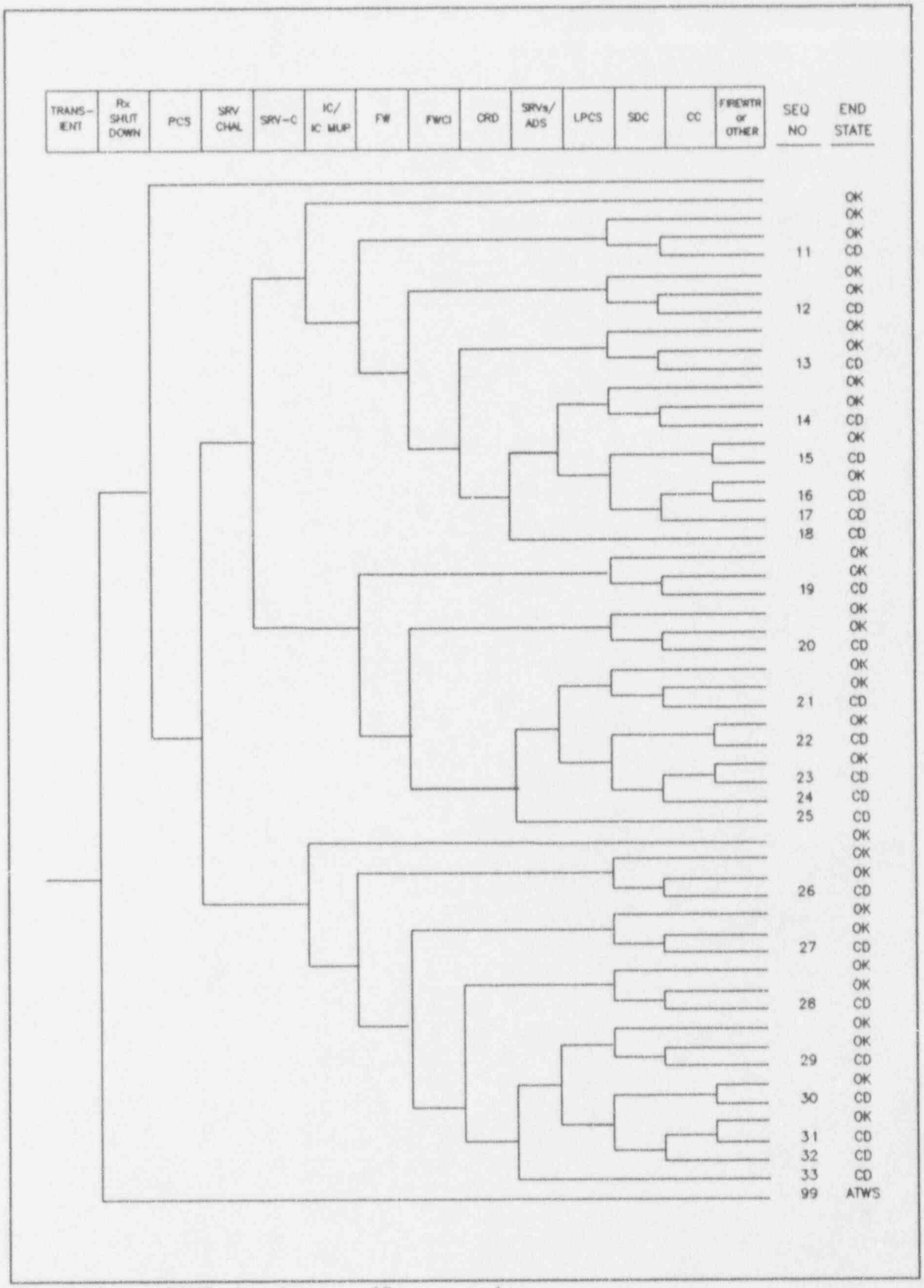

Fig. A.16. BWR class A nonspecific reactor trip

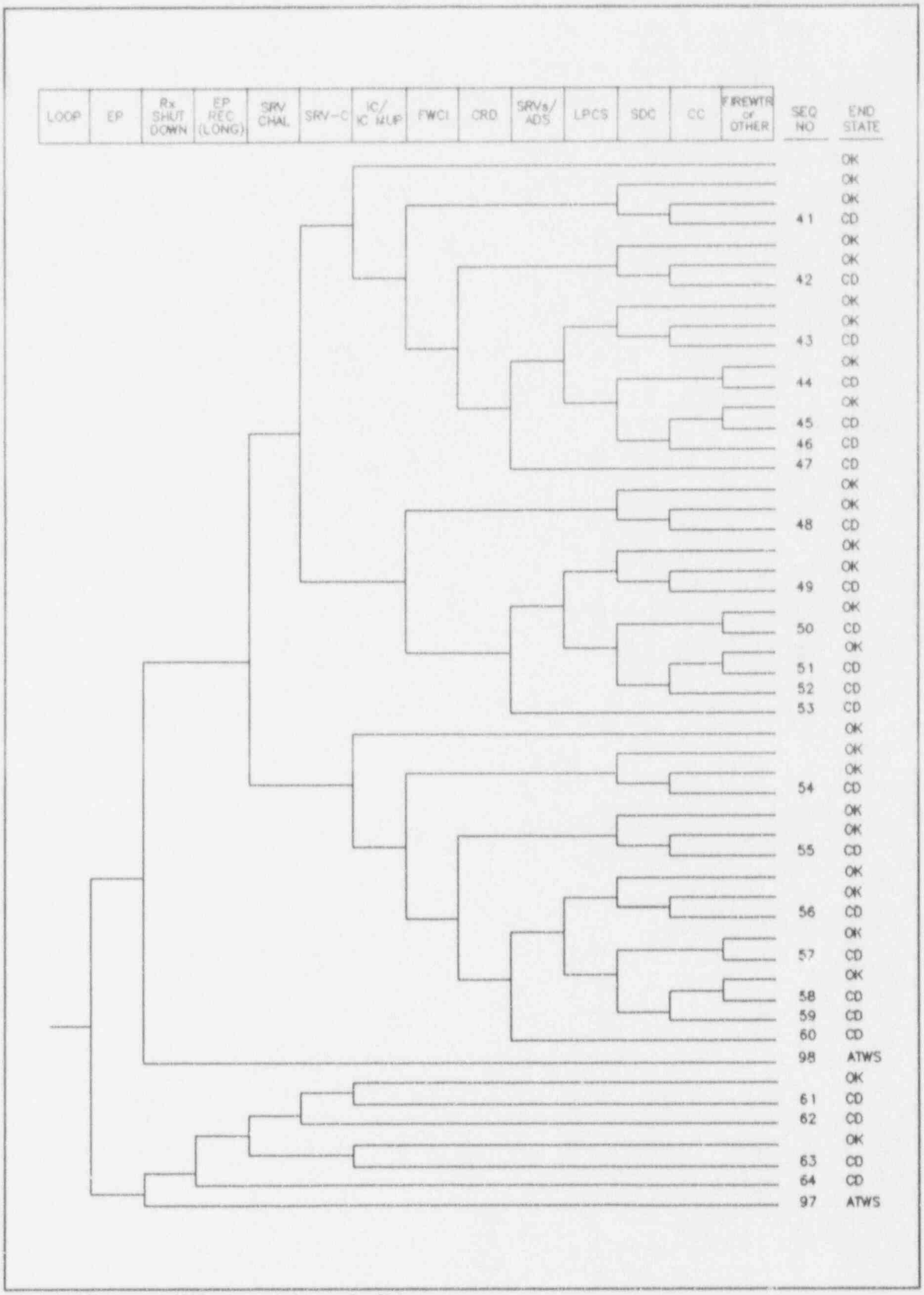

Fig. A.17. BWR class A loss of offsite power

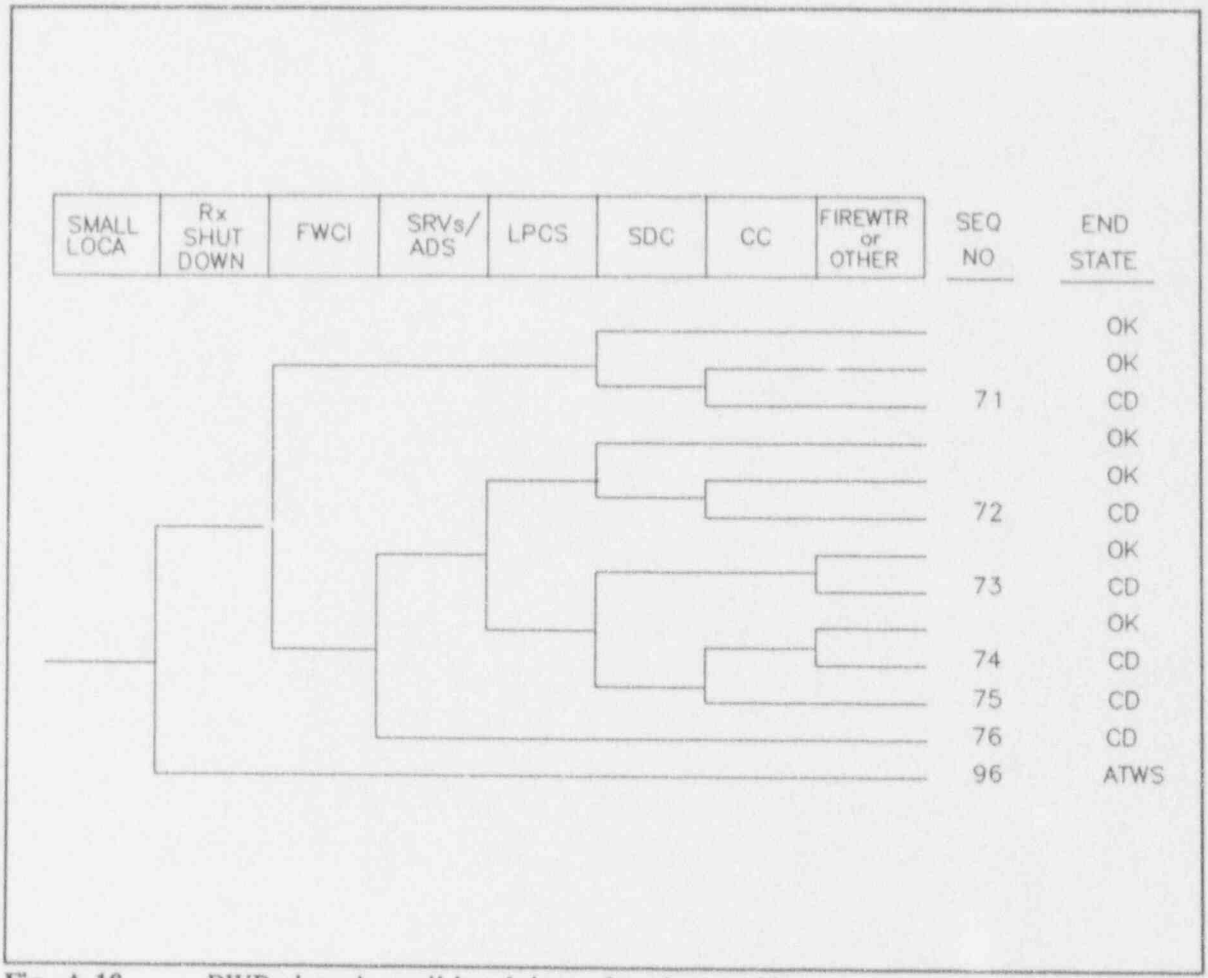

Fig. A.18. BWR class A small-break loss-of-coolant accident

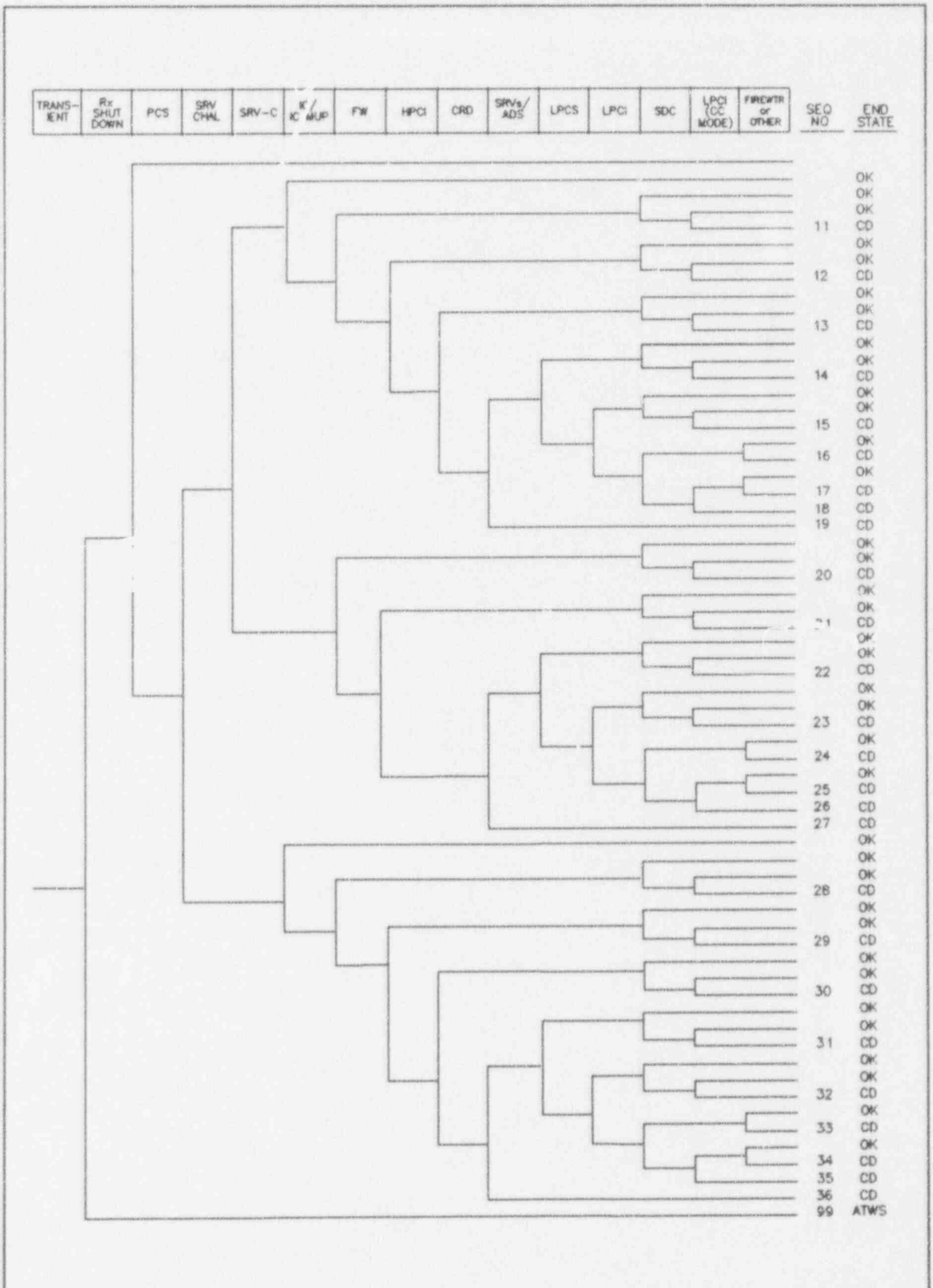

Fig. A.19. BWR class B nonspecific reactor trip

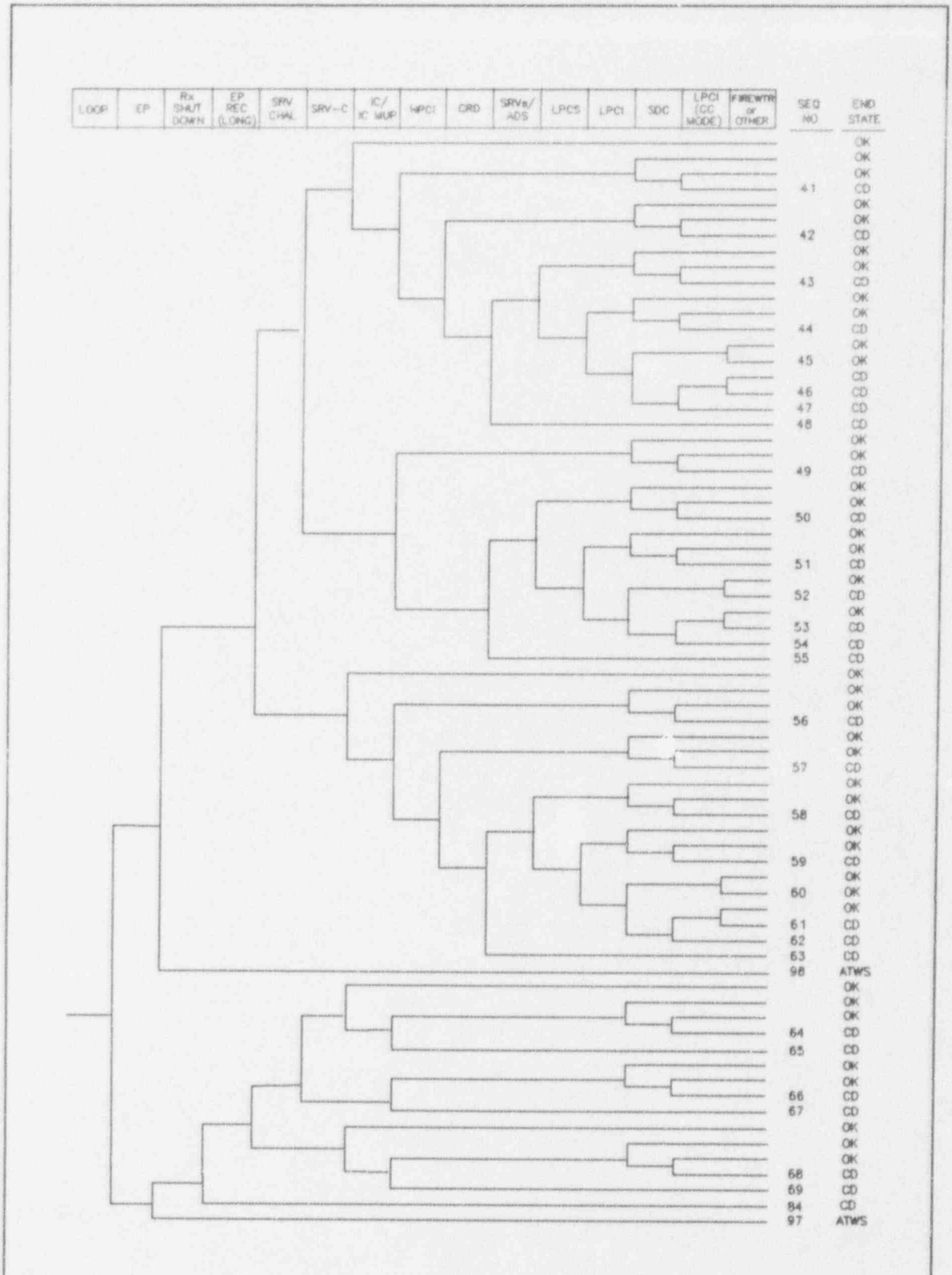

Fig. A.20. BWR class B loss of offsite power

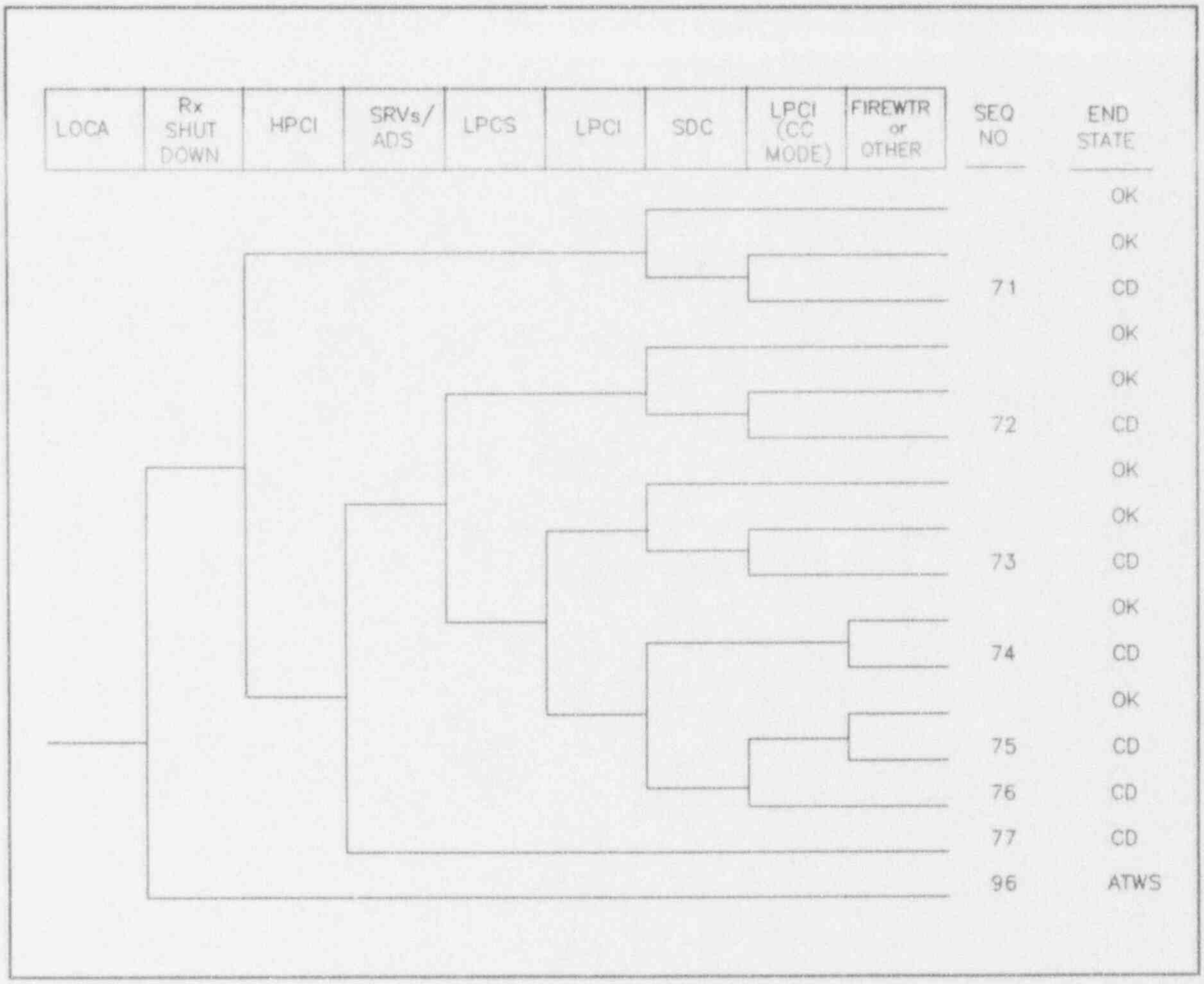

Fig. A.21. BWR class B small-break loss-of-coolant accident

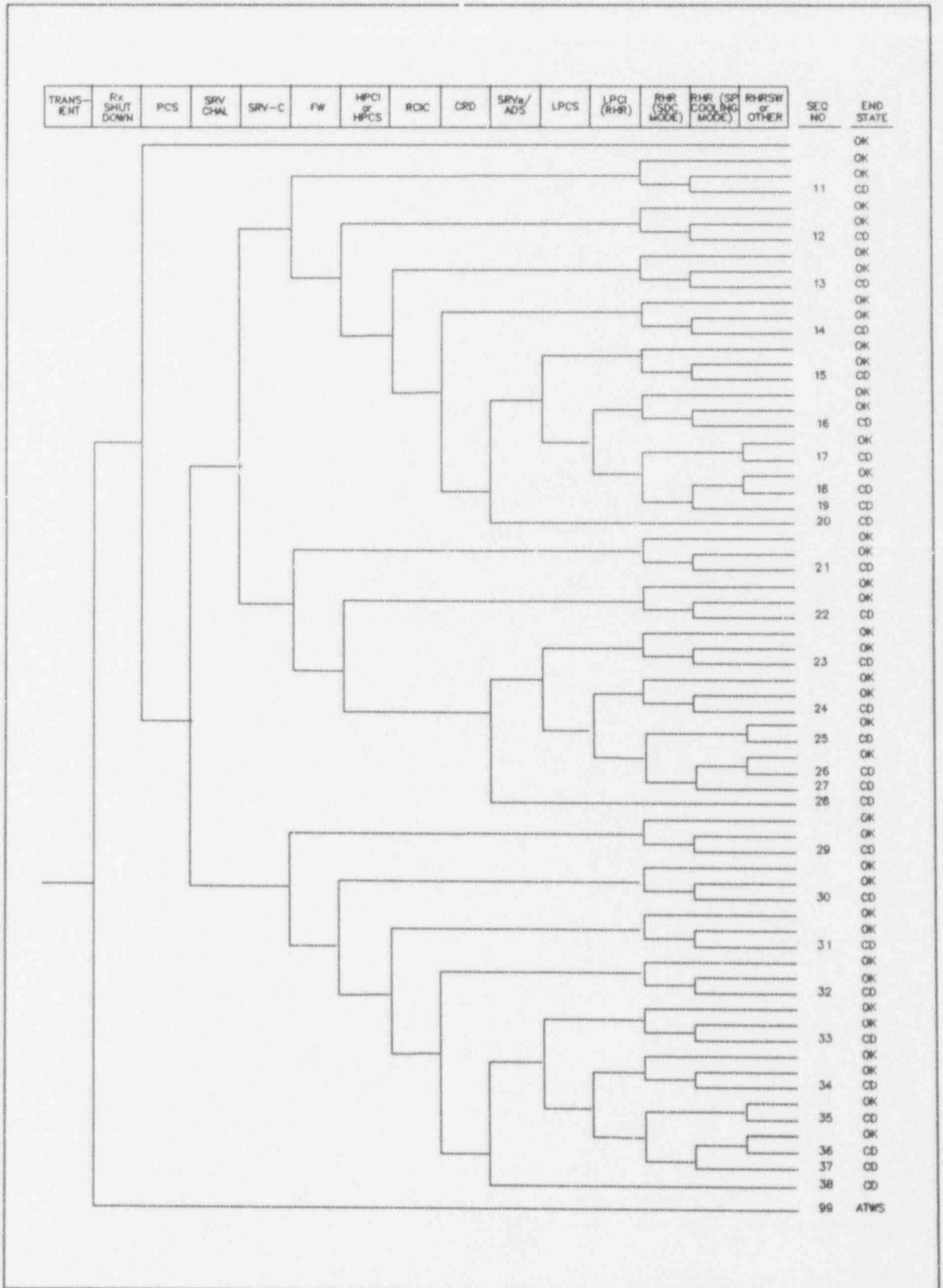

Fig. A.22. BWR class C nonspecific reactor trip

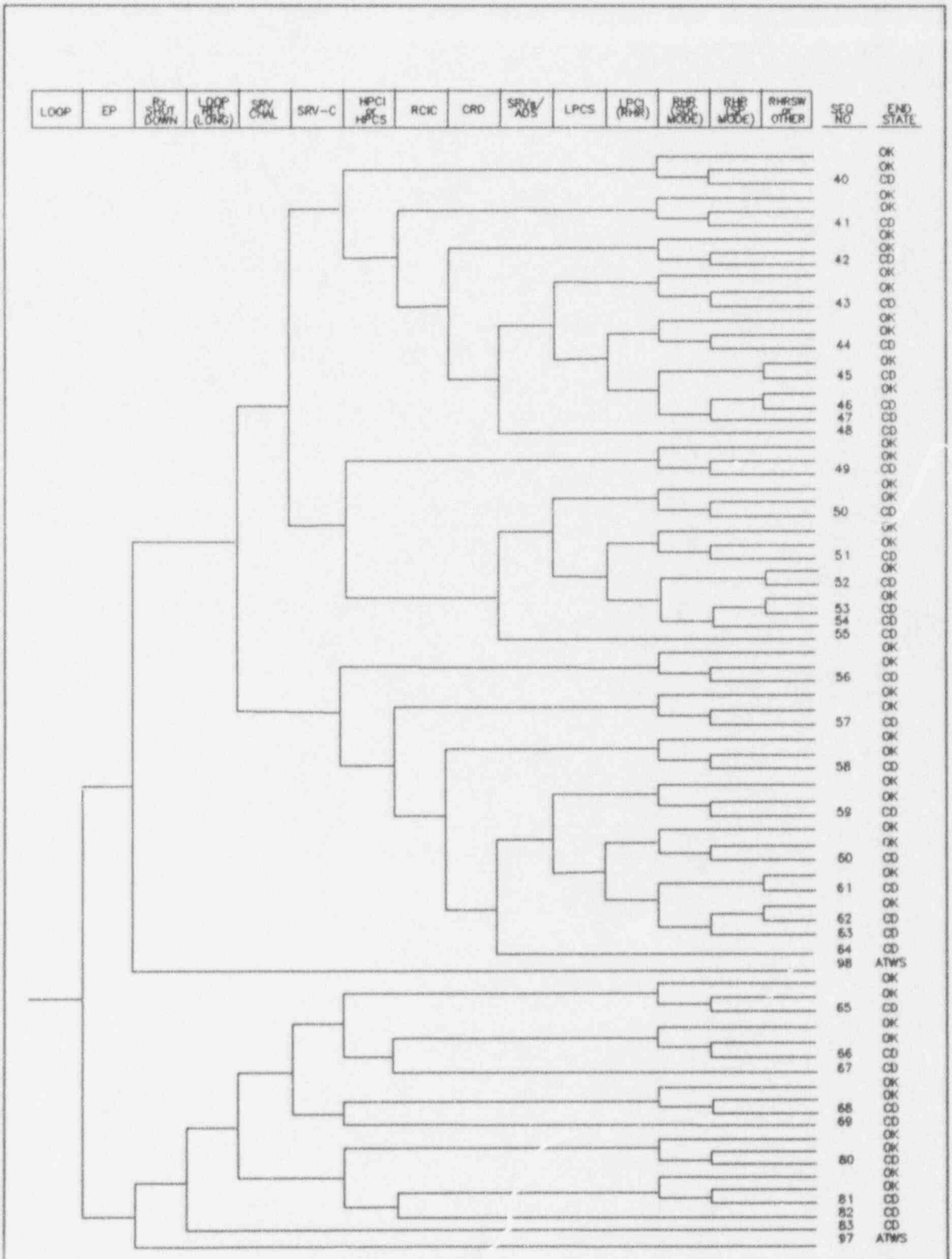

Fig. A.23. BWR class C loss of offsite power

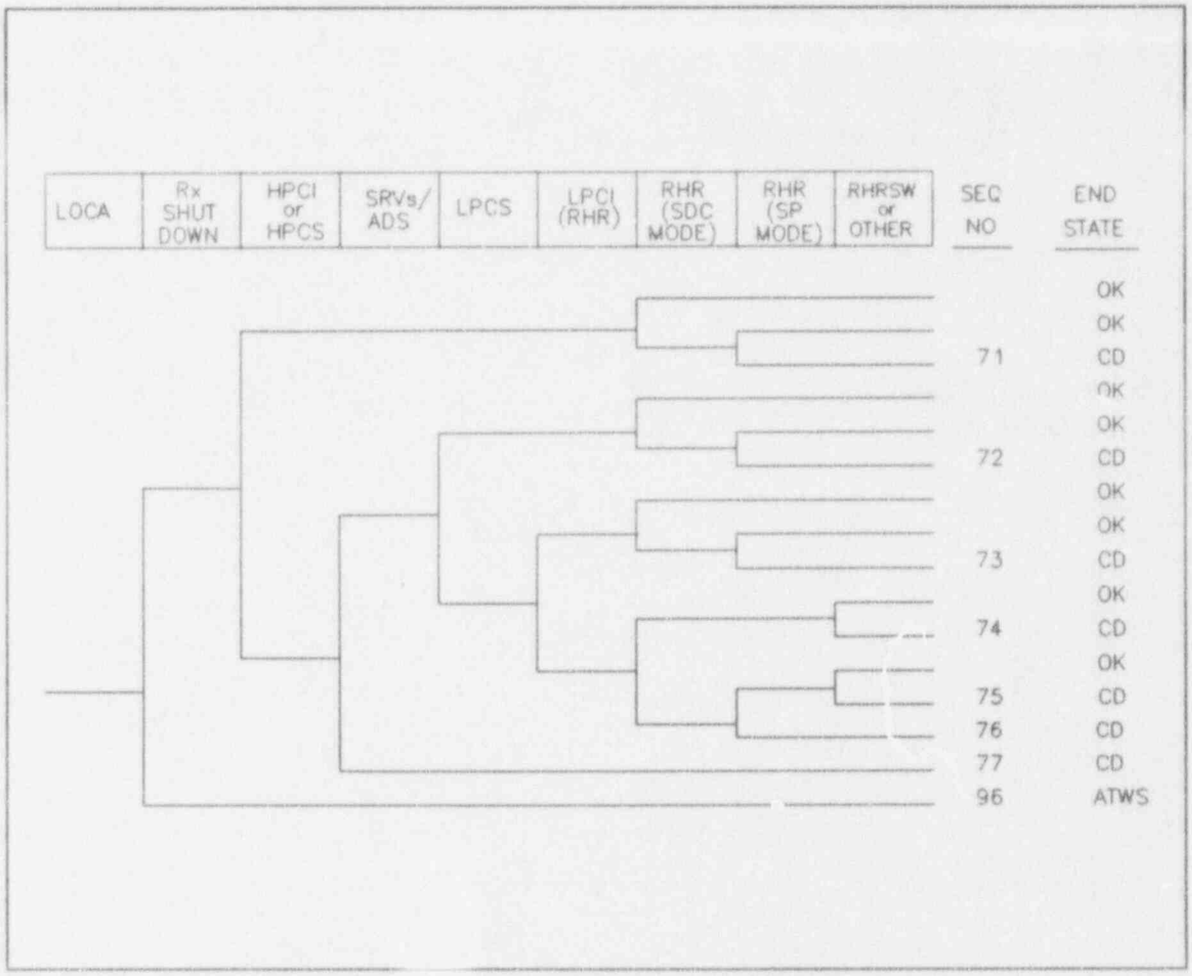

Fig. A.24. BWR class C small-break loss-of-coolant accident

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Twenty-seven operational events with conditional probabilities of subsequent severe core damage of 1.0 x 10E-06 or higher occurring at commercial light-water reactors during 1992 are considered to be precursors to potential core damage. These are described along with associated significance estimates, categorization, and subsequent analyses. The report discusses (1) the general rationale for this study, (2) the selection and documentation of events as precursors, (3) the estimation and use of conditional probabilities of subsequent severe core damage to rank precursor events, and (4) the plant models used in the analysis process.

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