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Precursors to Potential Severe Core Damage Accidents: 1991 A Status Report

Main Report and Appendix A

Prepared by
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Prepared for
U.S. Nuclear Regulatory Commission

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This document is bound in two volumes: Volume 15 contains the main report and Appendix A; Volume 16 contains Appendices B-D.

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FOREWORD

This report provides the 1991 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 event tree models and 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.

Several precursors involving phenomenological problems were experienced during 1991. Two precursor events involved the potential for gas binding of high-pressure injection pumps due to hydrogen gas entrainment. Another precursor concerned the potential hydraulic lockup of motorized valves in low-pressure coolant injection systems. The motorized valves are required for operation given a loss-of-coolant accident.

The precursor estimated to have the highest significance involved undetected common-cause water hammer damage to two relief valves in the high-pressure injection pump "miniflow" lines. The relief valves are intended to allow a minimum recirculation flow and prevent pump overheating while operating under deadhead (no-flow) conditions. The damaged valves opened at a much lower than normal pressure. If safety injection had been required, as it is for responding to a loss-of-coolant accident, sufficient flow may have been diverted through the failed relief valves to cause the loss of the safety function.

Several other precursors involved the common-cause failure of a safety function. One such precursor involved potential failure of a high-pressure injection system because the pump output relief valve setpoints were too close to the normal operating pressure. Another common-cause precursor involved undetected failure of logic power supplies and subsequent loss of uninterruptible power supplies. One other significant precursor of this type involved damage to a boiling-water reactor's automatic depressurization system due to improperly installed thermal insulation.

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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 nine reports documenting the review of operational events for precursors have been previously published in this program:

1969-1979	*Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report*, NUREG/CR-2497, June 1982
1980-1981	*Precursors to Potential Severe Core Damage Accidents: 1980-1981, A Status Report*, NUREG/CR-3591, July 1984
1984	*Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report*, NUREG/CR-4674, Vols. 3 and 4, May 1987
1985	*Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report*, NUREG/CR-4674, Vols. 1 and 2, December 1986
1986	*Precursors to Potential Severe Core Damage Accidents: 1986, A Status Report*, NUREG/CR-4674, Vols. 5 and 6, May 1988
1987	*Precursors to Potential Severe Core Damage Accidents: 1987, A Status Report*, NUREG/CR-4674, Vols. 7 and 8, July 1989
1988	*Precursors to Potential Severe Core Damage Accidents: 1988, A Status Report*, NUREG/CR-4674, Vols. 9 and 10, February 1990
1989	*Precursors to Potential Severe Core Damage Accidents: 1989, A Status Report*, NUREG/CR-4674, Vols. 11 and 12, September 1990
1990	*Precursors to Potential Severe Core Damage Accidents: 1990, A Status Report*, NUREG/CR-4674, Vols. 13 and 14, August 1991

The current effort was undertaken on behalf of the Office of Analysis and Evaluation of Operational Data 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 1991, of the assessment undertaken in the previous reports for operational events that occurred in 1969-1981 and 1984-1990.

The operational events selected in the ASP Program form a unique database 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
CC	containment cooling
CCW	component cooling water
CD	core damage
CRD	control rod drive
CSR	containment spray recirculation
DG	diesel generator
DHR	decay heat removal
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPS	emergency power system
ESF	engineered safety feature
FWCI	feedwater coolant injection
FSAR	final safety analysis report
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HPI	high-pressure injection
HPR	high-pressure recirculation
IC	isolation condenser
IIT	incident investigation team
LER	licensee event report
LOCA	loss-of-coolant accident
LOFW	loss of main feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPI	low-pressure injection
LPR	low-pressure recirculation
LWR	light-water reactor
MFW	main feedwater
MOV	motor-operated valve
MSIV	main steam isolation valve
NRC	Nuclear Regulatory Commission
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
ROAB	Reactor Operations Analysis Branch of AEOD, NRC
RPS	reactor protection system
RV	relief valve or reactor vessel
RWCU	reactor water cleanup
RWST	refueling water storage tank
SCSS	Sequence Coding and Search System database
SDC	shutdown cooling
SG	steam generator
SI	safety injection
SLB	steam-line break
SLC	standby liquid control
SP	suppression pool
SRV	safety relief valve
SW	service water
TBS	turbine bypass system

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PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1991, A STATUS REPORT

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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 1991 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 1984-1990 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. 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.[3-9] This report details the work of the ASP Program in its review and evaluation of operational events that occurred in 1991. The requirements for LERs are described in NUREG-1022, *Licensee Event Report System, Description of System and Guidelines for Reporting*[10] as well as in the supplements to NUREG-1022 (Refs. 11,12). LERs reviewed for precursors are described in Chap. 2.

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1.1 Background

The ASP Program owes its genesis to the Risk Assessment Review Group,[13] 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" (Ref. 14). 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 would have potentially 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 using an evaluation process and significance quantification methodology similar to that used in previous yearly assessments. All 1991 LERs were computer-screened to identify events that could potentially be precursors. Such events were subjected to an engineering evaluation that identified, analyzed, and documented the precursors, as described in Chap. 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 1991 LER review. These events are also documented in this report.

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
Chap. 2	Detailed review of 1991 LERs for accident sequence precursors and quantification of precursor significance

Chap. 3	Discussion of results
Appendix A	ASP analysis methodology and plant models
Appendix B	Precursors (including copies of applicable LERs)
Appendix C	Containment-related and other event documentation (including copies of applicable LERs)
Appendix D	Events that were considered impractical to analyze

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. 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. 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 decay heat removal 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 that reflect differences in design among plants in the U.S. LWR population. The initiators addressed in the event trees are: 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 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 involves 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 but 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. It must also be noted that elimination of events from further review was sometimes dictated by programmatic constraints. 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 at most involved:

- 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 (1) 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 were 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 decay heat removal 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:

- an unexpected core damage initiator (such as a LOOP, steam-line break, 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;

and if the conditional probability of subsequent core damage (described later) was at least 1.0×10^{-6}.

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 1987-1990 precursors, but is different from earlier ASP reports, which addressed all events meeting the precursor selection criteria, regardless of conditional core damage probability.

Events that occurred in 1991 were only reviewed for precursors if they satisfied an initial significance screening. This approach, which was similar to that used in the review of 1988-1990 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.

LERs were reviewed for precursors if they were identified as significant by

NRC's 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.

In addition to review if they were identified as significant by AEOD, 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-quarter 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, which would have been found if all 1991 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 1988-91 precursors is consistent with precursors identified in 1984-87.

Following AEOD and SCSS computerized screening, 616 LERs from 1991 were reviewed for precursors. Twenty-seven operational events with conditional probabilities of subsequent severe core damage $\geq 1.0 \times 10^{-6}$ 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 1991 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 remaining events are documented in Appendix C.

2.2 Estimation of Precursor Significance

Quantification of accident sequence precursor significance involves determination of a conditional probability of subsequent severe core damage given the failures observed during an operational event. This is estimated by mapping failures observed during the event onto the ASP event trees, which depict potential paths to severe core damage, and calculating a conditional probability of core damage through the use of event tree branch probabilities modified to reflect the event. The effect of a precursor on event tree branches is assessed by reviewing the operational event specifics against system design information and translating the results of the review into a revised conditional probability of system failure given the operational event.

In the precursor quantification process, it is assumed that the 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. 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. 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 remaining once the observed failures have occurred.

Because the frequencies and failure probabilities used in the calculations are derived in part from data obtained across the LWR population, the conditional probability estimated for a precursor may not be equivalent to the actual probability of severe core damage associated with the event at the 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.

The ASP calculational process is described in detail in Appendix A. This appendix documents the event trees used in the 1988-1991 precursor analyses, changes to these trees from prior years, the approach used to estimate event tree branch and sequence probabilities, and example calculations, as well as provides probability values used in the calculations. The overall precursor selection process is illustrated in Fig. 2.1.

2.3 Documentation of Events Selected as Accident Sequence Precursors

Each 1991 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) provide a graph of the relative significance of the event compared with other potential events at the plant.

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 relevant to the event are also provided.

Appendix C includes similar documentation for other events selected in the ASP Program (containment-related and other events). No probabilistic analysis is performed on these events.

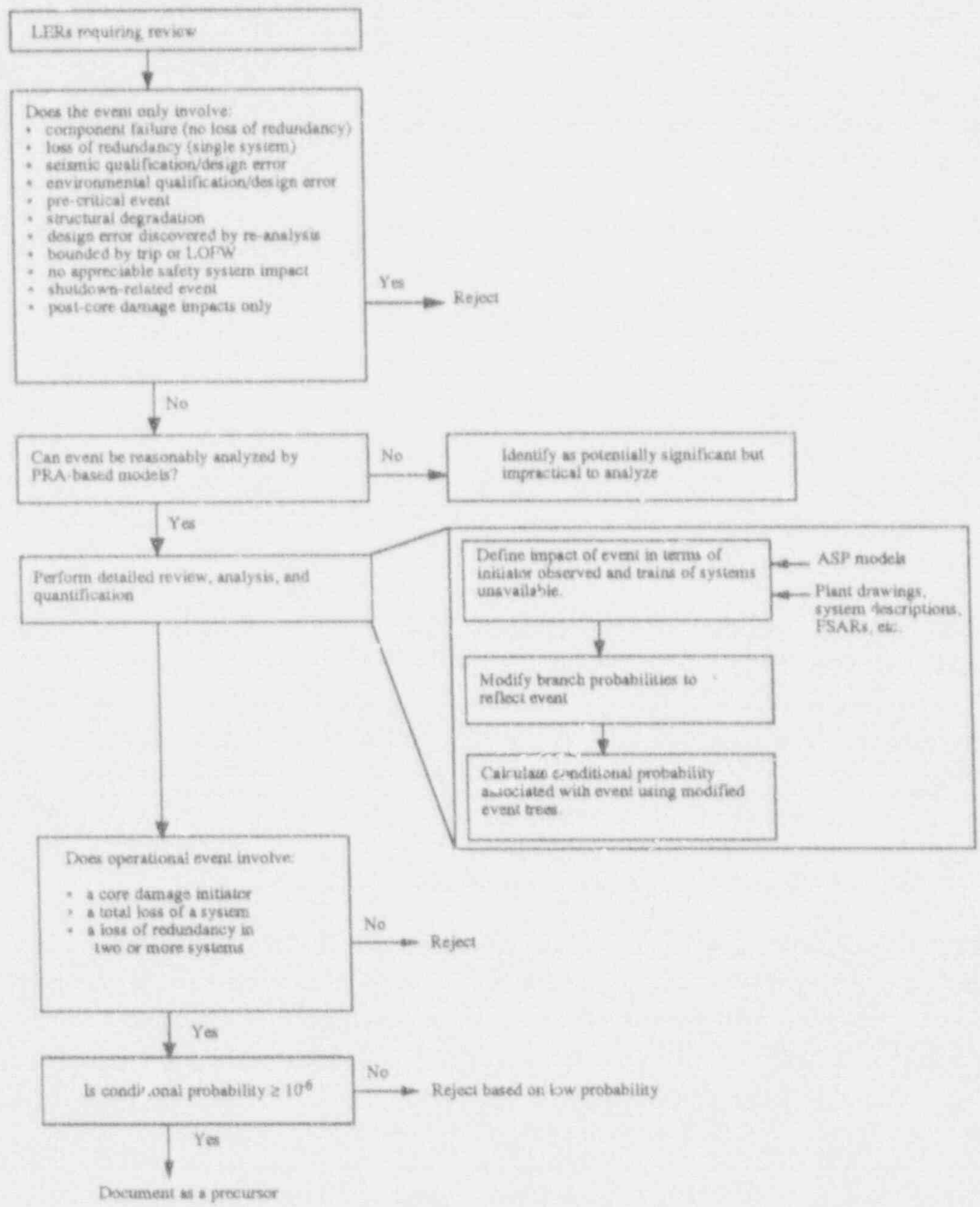


Figure 2.1 ASP analysis process

2.4 Tabulation of Selected Events

The 1991 events selected as accident sequence precursors are listed in Table 2.1. The precursors have been arranged in numerical order by plant docket and LER numbers, and the following information is included:

1. docket/LER number associated with the event (DOC/LER NO);
2. date of the event (E DATE);
3. a brief description of the event (DESCRIPTION);
4. plant name where the event occurred (PLANT);
5. abbreviations for the primary system and component involved in the event (SY, COMP);
6. plant operating status at the time of the event (O);
7. discovery method associated with the event (operational or testing) (D);
8. whether the event involved human error (E);
9. age (in years) of the plant from criticality at the time of the event (AGE);
10. conditional probability of potential severe core damage associated with the event (CDPROB);
11. plant power rating, type, vendor, architect-engineer, and licensee (RATE, T, V, AE, OPR);
12. plant criticality date (CRITICAL); and
13. initiator associated with the event or unavailability if no initiator was involved (TRANS).

The information in Table 2.1 has been sorted in several ways to provide additional perspective.

Table	Sorted by
2.2	Plant name and LER number
2.3	Event date
2.4	Initiator or unavailability
2.5	System
2.6	Component
2.7	Plant operating status
2.8	Discovery method
2.9	Conditional core damage probability
2.10	Plant type and vendor
2.11	Architect-engineer
2.12	Operating utility

Abbreviations used in each table (Tables 2.1 - 2.12) are defined in Table 2.13.

TABLE 2.1. PRECURSORS LISTED B: DOCKET AND LER NUMBER

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
029/	.15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	N	SW	YAC	08/19/60	LOOP
206/? J14	8/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
280/91-017	07/15/91	Both Unit 2 emergency diesol generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
304/91-002	03/21/91	Loss of offsite powe. with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
321/91-001	01/18/91	Loss of fe water, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
333/91-006	05/07/91	tp with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	0 /05/91	ydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
336/91-009	0 / /91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
368/91-012	04/16/91	Both normal service water trains fo led by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
369/91-001	02/11/91	Switchyard brc test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
400/91-008	04/03/91	Alternate mini- w valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1090	B	G	SW	NMP	05/23/87	TRIP
423/91-011	04/10/91	Relief valve failure, trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GH	TUG	04/01/90	UNAVL

*A similar condition existed at Oconee 2 and 3.

TABLE 2.2. PRECURSORS LISTED BY PLANT NAME AND LER NUMBER

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GH	TUG	04/01/90	UNAVL
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTR	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.3	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1090	B	G	SW	NMP	05/23/87	TRIP
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	YAC	08/19/60	LOOP
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	FA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP

*A similar condition existed at Oconee 2 and 3.

TABLE 2.3. PRECURSORS LISTED SEQUENTIALLY BY EVENT DATE

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UK	DPC	08/08/81	LOOP
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GK	CEI	06/04/86	UNAVL
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
445/91-012	03/28/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.7E-5	1150	P	W	GH	TUG	04/03/90	UNAVL
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	E	T	N	4.2	6.8E-3	900	P	W	EX	CPL	01/03/87	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	17.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	7.8E-4	514	B	G	EX	VYC	03/24/72	LOOP
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	L	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	171	P	W	SW	YAC	08/19/60	LOOP
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1090	B	G	SW	NMP	05/23/87	TRIP
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	LF	CKTBRK	O	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	07/19/85	UNAVL
269/91-010*	09/10/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	O	T	N	14.9	4.4E-6	1090	P	W	UX	PSG	12/11/76	UNAVL
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP

*A similar condition existed at Oconee 2 and 3.

TABLE 2.4. PRECURSORS LISTED BY INITIATOR OR UNAVAILABILITY

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPS	CRITICAL	TRANS
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	YTC	08/19/60	LOOP
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
400/91-010	06/23/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	F	O	Y	4.2	3.8E-4	1080	B	G	SW	NMP	05/23/87	TRIP
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	E	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
336/91-009	08/21/91	Both EDGs unavailable and unit shut down	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GX	TUG	04/01/90	UNAVL

*A similiar condition existed at Oconee 2 and 3.

TABLE 2.5. PRECURSORS LISTED BY PLANT SYSTEM

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	U	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALPOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	YAC	08/19/60	LOOP
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 12 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1080	B	C	SW	NMP	05/23/87	TRIP
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
287/91-007	07 .3/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.9E-5	887	P	B	UX	DPC	09/05/74	LOFW
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.8E-6	900	P	W	EX	CPL	01/03/87	TRIP
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-4	436	P	W	BX	SCE	06/14/67	UNAVL
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	E	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	2.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	3.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GX	TUC	04/03/90	UNAVL
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL

*A similar condition existed at Oconee 2 and 3.

TABLE 2.6. PRECURSORS LISTED BY COMPONENT

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SF	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1090	B	G	SW	NMP	05/23/87	TRIP
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.PO'NT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DI?CANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	YAC	08/19/60	LOOP
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
368/91-012	04/16/91	Both normal service water tra'ns fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded LFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	7.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
325/91-018	07/10/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GE	TUG	04/01/90	UNAVL
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RE'AYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	E	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	[illegible]	VALVEX	H	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL

*A similar condition existed at Oconee 2 and 3.

TABLE 2.7. PRECURSORS LISTED BY PLANT OPERATING STATUS

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	GW	YAC	08/19/60	LOOP
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
271/91-009	[illegible]/23/91	Extended loss of offsite power	VERMON.YNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
400/91-[illegible]	[illegible]	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
410/91-017	08/13/[illegible]	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1080	B	G	SW	NMP	05/23/87	TRIP
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.1E-5	1200	P	W	UE	PSN	06/13/89	LOOP
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GH	TUG	04/01/90	UNAVL
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	R	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	R	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL

*A similar condition existed at Oconee 2 and 3.

TABLE 2.8. PRECURSORS LISTED BY DISCOVERY METHOD

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	YAC	08/19/60	LOOP
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	C	EX	VYC	03/24/72	LOOP
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	4.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1090	B	G	SW	NMP	05/23/87	TRIP
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	O	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	L	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GX	TUG	04/01/90	UNAVL

*A similar condition existed at Oconee 2 and 3.

TABLE 2.9. PRECURSORS LISTED BY CONDITIONAL CORE DAMAGE PROBABILITY

DOC/LER NO	E DATE	DESCRIPTION	PLANT	ST	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	A	30.8	6.1E-4	175	P	W	SW	YAC	08/19/60	LOOP
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1090	B	G	SW	NMP	05/23/87	TRIP
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	AX	NNE	10/17/75	UNAVL
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	[illegible]	CIZZZZ	E	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GX	TUG	04/01/90	UNAVL
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UX	CPL	10/08/76	LOFW
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PSG	12/11/76	UNAVL
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	[illegible]	PGE	08/19/85	UNAVL
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP

*A similar condition existed at Oconee 2 and 3.

TABLE 2.10. PRECURSORS LISTED BY PLANT TYPE AND VENDOR

DOC/LER NO	I DATE	DESCRIPTION	PLANT	SY	COMP	C	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	E	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	F	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.-	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.8E-4	1090	B	G	SW	NMP	05/23/87	TRIP
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	[illegible].3E-4	1191	B	G	CX	CEI	06/06/86	UNAVL
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	[illegible].7E-4	887	P	B	UX	DPC	04/19/73	UNAVL
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/1[illegible]/75	UNAVL
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	[illegible]	08/19/60	LOOP
206/91-016	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	1[illegible].6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-4	1090	P	W	UX	PEG	12/11/76	UNAVL
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EC	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
304/91-004	06/[illegible]/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACALYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	[illegible]	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
400/91-010	0[illegible]/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.6E-6	900	P	W	EX	CPL	01/03/87	TRIP
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GX	TUG	04/01/90	UNAVL

*A similar condition existed at Oconee 2 and 3.

TABLE 2.11. PRECURSORS LISTED BY ARCHITECT-ENGINEER

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-4	436	P	W	CX	SCE	06/14/67	UNAVL
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	R	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BCC	06/16/72	LOOP
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	R	T	N	4.2	6.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.4E-6	900	P	W	EX	CPL	01/03/87	TRIP
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GX	TUG	04/01/90	UNAVL
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.3E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	YAC	08/19/60	LOOP
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.3E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	3.5E-4	1090	B	G	SW	NMP	05/23/87	TRIP
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-5	1154	P	W	SW	NNE	01/23/86	UNAVL
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-5	873	P	W	UE	CEC	05/22/73	TRIP
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-7	821	B	G	UE	CPL	10/08/76	LOFW
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	E	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.3	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.4E-4	1180	P	W	UX	DPC	08/08/81	LOOP

*A similar condition existed at Oconee 2 and 3.

TABLE 2.12. PRECURSORS LISTED BY OPERATING UTILITY

DOC/LER NO	E DATE	DESCRIPTION	PLANT	SY	COMP	O	D	E	AGE	CDPROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
368/91-012	04/16/91	Both normal service water trains fouled by debris	ARKANSAS 2	WA	FILTER	C	M	Y	12.4	4.8E-4	912	P	C	BX	APL	12/05/78	UNAVL
293/91-024	10/30/91	Loss of offsite power and RCIC trip	PILGRIM 1	EA	ELECON	G	O	N	19.4	1.2E-4	655	B	G	BX	BEC	06/16/72	LOOP
247/91-001	01/07/91	Reactor trip and auxiliary feedwater pump failure	IND.POINT2	HH	CKTBRK	E	O	Y	17.6	2.0E-6	873	P	W	UE	CEC	05/22/73	TRIP
440/91-009	03/14/91	Two EDGs inoperable	PERRY 1	EB	INSTRU	E	T	N	4.8	5.7E-4	1191	B	G	GX	CEI	06/06/86	UNAVL
325/91-018	07/18/91	Loss of feedwater with degraded HPCI system	BRUNSWICK1	SF	PIPEXX	E	O	N	14.8	6.0E-5	821	B	G	UE	CPL	10/08/76	LOFW
400/91-008	04/03/91	Alternate mini-flow valves inop, HPI unavailable	HARRIS 1	SF	VALVEX	H	T	N	4.2	4.3E-3	900	P	W	EX	CPL	01/03/87	UNAVL
400/91-010	06/03/91	Reactor trip breaker fails to open on trip	HARRIS 1	IA	CKTBRK	E	O	Y	4.4	6.8E-6	900	P	W	EX	CPL	01/03/87	TRIP
304/91-002	03/21/91	Loss of offsite power with 1 EDG out of service	ZION 2	EA	TRANSF	E	O	Y	17.2	2.1E-4	1040	P	W	SL	CWE	12/24/73	LOOP
304/91-004	06/11/91	Main feedwater pump trip with one AFW pump failed	ZION 2	HH	INSTRU	C	O	N	17.5	1.0E-5	1040	P	W	SL	CWE	12/24/73	TRIP
269/91-010*	09/19/91	Potential for hydrogen entrainment in HPI pumps	OCONEE 1	SF	ZZZZZZ	Z	T	Y	18.4	1.2E-4	887	P	B	UX	DPC	04/19/73	UNAVL
287/91-007	07/03/91	Reactor trip due to LOFW plus degraded EFW	OCONEE 3	HH	INSTRU	E	O	Y	16.8	1.8E-5	887	P	B	UX	DPC	09/05/74	LOFW
369/91-001	02/11/91	Switchyard breaker test causes LOOP	MCGUIRE 1	EA	INSTRU	E	T	Y	9.5	2.6E-4	1180	P	W	UX	DPC	08/08/81	LOOP
321/91-001	01/18/91	Loss of feedwater, HPCI degraded and RCIC failed	HATCH 1	SF	INSTRU	E	O	N	16.4	1.1E-5	776	B	G	SS	GPC	09/12/74	LOFW
410/91-017	08/13/91	Loss of 5 non-safety uninterruptible pwr supplies	NINE MILE2	ED	BATTRY	E	O	Y	4.2	7.8E-6	1090	B	G	SW	NMP	05/23/87	TRIP
336/91-009	08/21/91	Both EDGs unavailable and unit shutdown	MILLSTN 2	EB	INSTRU	E	T	N	15.8	2.1E-4	870	P	C	BX	NNE	10/17/75	UNAVL
423/91-011	04/10/91	Relief valve failure, both trains of HPSI inop	MILLSTN 3	SF	VALVEX	D	T	N	5.2	8.1E-4	1154	P	W	SW	NNE	01/23/86	UNAVL
278/91-017	09/24/91	Control wiring for ADS/relief valves found damaged	PEACHBOTM3	SF	INSTRU	H	M	Y	17.1	3.3E-4	1065	B	G	BX	PEC	08/07/74	UNAVL
272/91-030	09/20/91	Both PORVs failed due to leaking actuators	SALEM 1	CA	VALVOP	D	T	N	14.8	4.4E-6	1090	P	W	UX	PEG	12/11/76	UNAVL
323/91-003	09/01/91	Containment sump and spray unavail at hot shutdown	DIACANYON2	SF	CKTBRK	D	T	Y	6.0	2.1E-6	1119	P	W	UX	PGE	08/19/85	UNAVL
333/91-006	05/07/91	Trip with both LPCI trains inoperable	FITZPATRIC	SF	VALVEX	E	T	N	16.5	2.0E-5	821	B	G	SW	PNY	11/17/74	TRIP
333/91-014	08/05/91	Hydraulic locking of 2 ECCS injection valves	FITZPATRIC	SF	VALVEX	G	T	N	16.7	9.5E-5	821	B	G	SW	PNY	11/17/74	UNAVL
443/91-008	06/27/91	Loss of offsite power	SEABROOK 1	EA	RELAYX	E	M	Y	2.0	4.4E-5	1200	P	W	UE	PSN	06/13/89	LOOP
206/91-014	08/07/91	Inoperable volume control tank level transmitters	SANONOFRE1	IB	INSTRU	E	O	N	24.1	2.1E-6	436	P	W	BX	SCE	06/14/67	UNAVL
445/91-012	03/26/91	Potential charging pump unavail due to hydrogen	COMANCHE 1	SF	PUMPXX	E	T	N	1.0	6.2E-5	1150	P	W	GH	TUG	04/03/90	UNAVL
280/91-017	07/15/91	Both Unit 2 emergency diesel generators inop 13 h	SURRY 2	EB	INSTRU	E	T	Y	18.4	2.9E-6	788	P	W	SW	VEP	03/07/73	UNAVL
271/91-009	04/23/91	Extended loss of offsite power	VERMONTYNK	EA	INSTRU	E	O	Y	19.1	2.9E-4	514	B	G	EX	VYC	03/24/72	LOOP
029/91-002	06/15/91	Loss of offsite power caused by lightning strike	YANKEEROWE	EA	ELECON	E	O	N	30.8	6.1E-4	175	P	W	SW	YAC	08/19/60	LOOP

*A similar condition existed at Oconee 2 and 3.

TABLE 2.13. ABBREVIATIONS USED IN PRECURSOR LISTS

DOC/LER NO: DOCKET NUMBER/LICENSEE EVENT REPORT NUMBER
E DATE: EVENT DATE
DESCRIPTION: DESCRIPTION OF EVENT
PLANT: NAME OF PLANT AND UNIT NUMBER
SY: SYSTEM ABBREVIATION

SYSTEM CODE DESCRIPTION

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 SYSTEMS AND CONTROLS
CH	FEEDWATER SYSTEMS AND CONTROLS
CI	REACTOR COOL' ' PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS
CJ	OTHER COOLA' ' 'YSTEMS AND THEIR CONTROLS

-GINEERED 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
SH	OTHER ENGINEERED SAFETY FEATURE SYSTEMS AND THEIR CONTROLS

INSTRUMENTATION AND CONTROLS

IA	REACTOR TRIP SYSTEMS
IB	ENGINEERED SAFETY FEATURE INSTRUMENT SYSTEMS
IC	SYSTEMS REQU RED FOR SAFE SHUTDOWN
ID	SAFETY RELATED DISPLAY INSTRUMENTATION
IE	OTHER INSTRUMENT SYSTEMS REQUIRED FOR SAFETY
IF	OTHER INSTRUMENT SYSTEMS NOT 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 (COMPOSIT 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

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

'UXILIARY 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 VENTILATI N SYSTEMS AND CONTROLS
AB	IRE PROTECT'ON 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 ST_AM 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 MANAGMENT SYSTEMS
MC PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS
MD SOLID RADIOACTIVE WASTE MANAGEMENT SYSTEMS

RADIATION PROTECTION SYSTEMS

RA AREA MONITORING SYSTEMS
RB AIRBORNE RADIOACTIVITY MONITORING SYSTEMS

XX OTHER SYSTEMS

ZZ SYSTEM CODE NOT APPLICABLE

COMP: SYSTEM COMPONENT CODE

COMPONENT TYPE (COMPONENT CODE)	COMPONENT TYPE INCLUDES	COMPONENT TYPE (COMPONENT CODE)	COMPONENT TYPE INCLUDES
ACCUMULATOR (ACCUMU)	SCRAM ACCUMULATORS SAFETY INJECTION TANKS SURGE TANKS HOLDUP/STORAGE TANKS	CONTROL DRIVE MECHANISMS (CRDRVS)	
		DEMINERALIZERS (DEMINX)	ION EXCHANGERS
AIR DRYERS (AIRDRY)		ELECTRICAL CONDUCTIONS (ELECON)	BUS CABLE WIRE
ANNUNCIATOR MODULES (ANNUNC)	ALARMS BUZZERS CLAXONS HORNS GONGS SIRENS	ENGINES, INTERNAL COMBUSTION (ENGINE)	DIESEL ENGINES GASOLINE ENGINES NATURAL GAS ENGINES PROPANE ENGINES STRAINERS SCREENS
		FILTERS (FILTER)	
BATTERIES AND CHARGERS (BATTRY)	CHARGERS DRY CELLS WET CELLS STORAGE CELLS	FUEL ELEMENTS (FUELXX)	
		GENERATORS (GENERA)	INVERTERS
BLOWERS (BLOWER)	COMPRESSORS GAS CIRCULATORS FANS VENTILATORS	HEATERS, ELECTRIC (HEATER)	HEAT TRACING
CIRCUIT CLOSERS/ INTERUPTERS (CKTBRK)	CIRCUIT BREAKERS CONTACTORS CONTROLLERS STARTERS SWITCHES (OTHER THAN SENSORS) SWITCHGEAR	HEAT EXCHANGERS (HTEXCH)	CONDENSERS COOLERS EVAPORATORS REGENERATIVE HEAT EXCHANGERS STEAM GENERATORS FAN COIL UNITS
CONTROL RODS (CONROD)	POISON CURTAINS		
INSTRUMENTATION AND CONTROLS (INSTRU)	CONTROLLERS SENSORS/DETECTORS/ELEMENTS INDICATORS	RELAYS (RELAYX)	SWITCHGEAR
	DIFFERENTIALS INTEGRATORS (TOTALIZERS) POWER SUPPLIES RECORDERS SWITCHES	SHOCK SUPPRESSORS AND SUPPORT (SUPORT)	HANGERS SUPPORTS SWAY BRACES/STABILIZERS SNUBBERS ANTI-VIBRATION DEVICES
	TRANSMITTERS COMPUTATION MODULES	TRANSFORMERS (TRANSF)	
MECHANICAL FUNCTION UNITS (MECFUN)	MECHANICAL CONTROLLERS GOVERNORS GEAR BOXES VARIDRIVES	TURBINES (TURBIN)	STEAM TURBINES GAS TURBINES HYDRO TURBINES
ELECTRIC MOTORS (MOTORX)	VALVES HYDRAULIC MOTORS PNEUMATIC (AIR) MOTORS SERVO MOTORS	VALVES (VALVEX)	DAMPERS
		VALVE OPERATORS (VALVOP)	EXPLOSIVE, SQUIB
PENETRATIONS, PRIMARY	AIR LOCKS	VESSELS, PRESSURE	CONTAINMENT VESSELS

CONTAIN (PENETR)	PERSONNEL ACCESS FUEL HANDLING EQUIPMENT ACCESS ELECTRICAL INSTRUMENT LINE PROCESS PIPING	(VESSEL)	DRYWELLS PRESSURE SUPPRESSION PRESSURIZERS REACTOR VESSELS
PIPES, FITTINGS (PIPEXX)	OTHER COMPONENTS (XXXXXX)		
PUMPS (PUMPXX)	CODES NOT APPLICABLE (ZZZZZZ)		
RECOMBINERS (RECOM)			

C: 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

D: DISC METHOD (O-OPERATONAL EVENT, T-TESTING, M-MAINTENANCE)
E: HUMA H INVOLVED (N-NO, Y-YES)
AGE: PLANT AGE AT THE TIME OF THE EVENT IN YEARS
CDPROB: CONDITIONAL CORE DAMAGE PROBABILITY
SRCE: LER SUBSET WHICH INCLUDED PRECURSORS

AEOD - EVENT DETERMINED TO BE SIGNIFICANT BY AEOD
NRC - OTHER EVENTS REVIEWED AT NRC REQUEST
SCSS - EVENT IDENTIFIED FOR REVIEW BY SEQUENCE CODING AND SEARCH SYSTEM SEARCH STRATEGY

RATE: PLANT ELECTRICAL RATING IN MEGAWATTS ELECTRIC
T: PLANT TYPE (B-BWR, P-PWR)
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
BE-BECHTEL
BR-BURNS AND ROE
EB-EBASCO
FP-FLUOR POWER
GH-GIBBS AND HILL
GE-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

OPR: PLANT LICENSEE ABBREVIATIONS

LICENSEE ABBREV.	LICENSEE	LICENSEE ABBREV.	LICENSEE
APC	ALABAMA POWER COMPANY	OEC	OHIO EDISON COMPANY
APL	ARKANSAS POWER AND LIGHT COMPANTY	OPP	OMAHA PUBLIC POWER DISTRICT
APS	ARIZONA PUBLIC SERVICE COMPANY	PEC	PHILADELPHIA ELECTRIC COMANY
BEC	BOSTON EDISON COMPANY	PEG	PUBLIC SERVICE ELECTRIC AND GAS COMPANY
BGE	BALTIMORE GAS AND ELECTRIC COMPANY	PEP	POTOMAC ELECTRIC POWER COMPANY
CEC	CONSOLIDATED EDISON COMPANY	[illegible]	PORTLAND GENERAL ELECTRIC COMPANY
CEI	CLEVELAND ELECTRIC ILLUMINATING COMPANY	PGE	PACIFIC GAS AND ELECTRIC COMPANY
CGE	CINCINNATI GAS AND ELECTRIC COMPANY	PNY	NEW YORK POWER AUTHORITY
COY	CONNECTICUT YANKEE ATOMIC POWER AND LIGHT COMPANY	PPL	PENNSYLVANIA POWER AND LIGHT COMPANY
CPC	CONSUMERS POWER COMPANY	PSC	PUBLIC SERVICE COMPANY OF COLORADO
CPL	CAROLINA POWER AND LIGHT COMPANY	PSI	PUBLIC SERVICE OF INDIANA
CWE	COMMONWEALTH EDISON COMPANY	PSN	PUBLIC SERVICE OF NEW HAMPSHIRE
DEC	DETROIT EDISON COMPANY	PSO	PUBLIC SERVICE OF OKLAHOMA
DLP	DAIRYLAND POWER COOPERATIVE	PUG	PUGET SOUND POWER AND LIGHT COMPANY
DPC	DUKE POWER COMANY	RGE	ROCHESTER GAS AND ELECTRIC CORPORATION
DUQ	DUQUESNE LIGHT COMPANY	SCL	SOUTH CAROLINA ELECTRIC AND GAS COMPANY
FPC	FLORIDA POWER CORPORATION	SCE	SOUTHERN CALIFORNIA EDISON COMPANY
FPL	FLORIDA POWER AND LIGHT COMPANY	SMU	SACRAMENTO MUNICIPAL UTILITIES DISTRICT
GPC	GEORGIA POWER COMPANY	TEC	TOLEDO EDISON COMPANY
GSU	GULF STATES UTILITIES	TUG	TEXAS UTILITIES GENERATING COMPANY
HLP	HOUSTON LIGHTING AND POWER COMPANY	TVA	TENNESSEE VALLEY AUTHORITY
IEL	IOWA ELECTRIC LIGHT AND POWER COMPANY	UEC	UNION ELECTRIC COMPANY

IME	INDIANA AND MICHIGAN ELECTRIC COMPANY
IPC	ILLINOIS POWER COMPANY
JCP	JERSEY CENTRAL POWER AND LIGHT COMPANY
KGE	KANSAS GAS AND ELECTRIC COMPANY
LIL	LONG ISLAND LIGHTING COMPANY
LPL	LOUISIANA POWER AND LIGHT COMPANY
MEC	METROPOLITAN E[illegible]N COMPANY
MPL	MISSISSIPPI POWER AND LIGHT COMPANY
MYA	MAINE YANKEE ATOMIC POWER COMPANY
NEP	NEW ENGLAND POWER COMPANY
NMP	NIAGARA MOHAWK POWER CORPORATION
NNE	NORTHEAST NUCLEAR ENERGY COMPANY
NPC	NORTHERN INDIANA PUBLIC SERVICE COMPANY
NPP	NEBRASKA PUBLIC POWER DISTRICT
NSP	NORTHERN STATES POWER COMPANY
VEP	VIRGINIA ELECTRIC AND POWER COMPANY
VYC	VERMONT YANKEE NUCLEAR POWER CORPORATION
WEP	WISCONSIN ELECTRIC POWER COMPANY
WMP	WISCONSIN MICHIGAN POWER COMPANY
WPP	WASHINGTON PUBLIC POWER SUPPLY SYSTEM
WPS	WISCONSIN PUBLIC SERVICE CORPORATION
YAE	YANKEE ATOMIC ELECTRIC COMPANY

CRIT: PLANT CRITICALITY DAYS

TRANS: EVENT INITIATOR OR UNAVAILABILITY

ECIT	-	EXCESSIVE COOLANT INVENTORY
EQUK	-	EARTHQUAKE
INRA	-	INADVERTANT ADS 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

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-yr 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 D.

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 1991 LERs.* For 1969-81 and 1984-87, all LERs reported during the year were evaluated for precursors. For 1988-91, only a subset of LERs were evaluated in the ASP Program following a computerized search of the SCSS database 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, even though criteria for that selection are established. Events selected in the study were more serious than most, so the majority of the LERs selected for detailed review would likely 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 revised LER rule has reduced the variation in reported details, some variation still exists. In addition, only details of the sequence (or partial sequences for failures discovered during testing) that actually occurred are usually provided; details concerning potential alternate sequences of interest in this study must often be inferred.

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 high-pressure injection for some PWRs, long-term decay heat removal 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 multi-train 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.

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 were developed based on a review of recovery actions during historic events, 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.[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 based on 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 only started 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.*

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

3. RESULTS

This chapter summarizes results of the 1991 effort. The primary result of the ASP Program for 1991 is the identification of operational events satisfying one of the three precursor selection criteria: a core damage initiator requiring safety system response, the failure of a system required to mitigate the consequences of a core damage initiator, or degradation of more than one system required for mitigation. These events are documented in Appendix B. Twenty-seven such events were identified for 1991.

Because of the changes in the plant models used in the analysis of 1988-91 events, plus the evaluation of only a portion of 1991 LERs by the project team (as described in section 2.1), the set of 1991 precursors may be different from the set that would have resulted if the 1984-87 plant models had been used and if all 1991 LERs had been reviewed. These differences may bias comparisons between results for 1988-91 and 1984-87. Because of this, only limited observations are provided herein. Refer to the 1986 and 1987 precursor reports[1,2] for a discussion of observations for 1984-86 and 1987 events.

3.1 Important Precursors

Thirteen precursors with conditional core damage probabilities equal to or greater than 10^{-4} were identified for 1991. Events with such conditional probabilities have traditionally been considered significant in the ASP Program. For 1991, these events include:

At Yankee Rowe (LER 029/91-002), offsite power was lost because of a lightning strike. As a result of the lightning, fuses for the normal DC supplies to both 120 VAC instrument bus inverters blew. Both inverters transferred to their alternate emergency diesel generator (EDG) sources. However, if the EDGs had failed, 120 VAC instrument power would also have been lost.

At Oconee (LER 269/91-010), analysis of maximum letdown storage tank pressure determined that hydrogen in the tank would enter the high-pressure injection (HPI) pump suctions and gas bind the pumps for small LOCA scenarios involving the failure of a borated water storage tank (BWST) outlet valve to open. This problem has affected each Oconee unit since criticality.

At Vermont Yankee (LER 271/91-009), a loss of offsite power (LOOP) occurred during switchyard maintenance activities. Recovery of offsite power, which took ~13 h, was complicated by communications and organizational difficulties and travel time for support personnel from Providence, RI.

At Peach Bottom 3 (LER 278/91-017), improperly installed insulation on the automatic depressurization system (ADS)/safety relief valves (SRVs) resulted in damage to the SRV control wiring and SRV unavailability under conditions of high containment temperatures. This condition existed throughout the refueling cycle; high-pressure coolant injection (HPCI) was unavailable for significant periods of time during this interval as well.

At Pilgrim (LER 293/91-024), a LOOP occurred 2-1/2 h after the plant was shut down during a storm. RCIC tripped twice during mitigation of the LOOP. One of these trips was caused by a trip of the RCIC inverter when an RHR pump was started. Start of the RHR pump caused an AC voltage transient, which in turn caused a DC voltage transient because of poor battery charger regulation. The DC voltage transient exceeded the inverter overvoltage setpoint and tripped the inverter.

At Zion 2 (LER 304/91-002), multiple deluge system actuations sprayed a station auxiliary transformer and resulted in a LOOP. One EDG was out of service for maintenance at the time of the event. In addition, feed and bleed capability was degraded when one power-operated relief valve (PORV) was unavailable because of a failed air line.

At Millstone 2 (LER 336/91-009), both EDGs were found to exhibit erratic load control, a result of either a resistance change in the "droop" potentiometers in the electronic governor controls or contaminated oil in the hydraulic actuator units.

At Arkansas Nuclear One, Unit 2 (LER 368/91-012), errors during traveling screen maintenance caused significant quantities of debris to carry over into the service water (SW) pump suction pits. Pump discharge strainers became fouled, resulting in inoperability of both SW trains.

At McGuire 1 (LER 369/91-001), errors and equipment failures during installation of new switchyard relay protection resulted in the opening of all switchyard breakers connecting the unit to the grid. An excessive cooldown rate resulted in safety injection (SI) actuation and main steam isolation valve (MSIV) closure.

At Harris (LER 400/91-008), relief valves and associated piping in the alternate minimum recirculation lines for the HPI pumps, which are used following SI, were found failed. Had high-head SI been demanded, sufficient flow would have been diverted to fail the injection function and also cause loss of emergency sump inventory during high-pressure recirculation (HPR).

At Nine Mile Point 2 (LER 410/91-017), a main transformer fault caused a turbine trip and reactor scram. Following the transformer fault, five uninterruptible power supplies deenergized, removing power from nonsafety-related instrumentation and equipment and affecting rod position indicators, control room annunciators, lighting, and communications systems. Since rod position indication could not be verified, ADS was inhibited. Two of three trains of the low-pressure coolant injection (LPCI) system were initially unavailable, having previously been removed from service for maintenance.

At Millstone 3 (LER 423/91-011), relief valves in the high-pressure SI system were found to lift at a pressure only slightly above normal operating pressures. Perturbations in system pressure, including those resulting from operation at minimum flow conditions, would result in lifting the valves. Flow from the relief valves could result in a loss of ~160 gpm in injection flow. In addition, flow from these valves would result in the loss of emergency sump inventory outside containment after switchover to HPR.

At Perry 1 (LER 440/91-009), both EDGs failed their surveillance test. One EDG failed because its field contactor failed to close. The second EDG failed to synchronize to the grid because of governor speed control problems.

3.2 Number of Precursors Identified

Twenty-seven precursors [p(core damage) $\geq 10^{-6}$] were identified in 1991. The distribution of precursors as a function of conditional probability is shown in Table 3.1. This distribution compares as follows with events identified in 1988-90:

	Number of precursors		
	p(cd)$\geq 10^{-4}$	p(cd)$\geq 10^{-5}$	p(cd)$\geq 10^{-6}$
1988	7	21	32
1989	7	18	[illegible]
1990	6 *	17 **	28 **
1991	13	21	27

*including one event at cold shutdown

**including two events at cold shutdown

As can be seen in Table 3.1, 8 of the 13 precursors with p(cd) $\geq 10^{-4}$ selected for 1991 are PWR events. This is roughly equivalent to the fraction of each type in the U.S. LWR population. This differs from the results for 1988-90, where almost all of the more significant events occurred at PWRs. For all 1991 precursors, 9 were associated with BWRs and 18 with PWRs, also roughly equivalent to the fraction of each type of plant in the U.S. LWR population.

Table 3.1. Precursors for 1991 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}	High-head SI unavailable at Harris due to failed relief valves and associated piping in the minimum recirculation flow lines used during SI (400/91-008).
10^{-4} to 10^{-3}	LOOP at Yankee Rowe and failure of DC fuses to the 120-VAC instrument bus inverters because of a lightning strike (029/91-002).
	Potential for gas binding of the HPI pumps with hydrogen from the letdown storage tank on all three Oconee units for small-break LOCA scenarios with failure of a borated water storage tank (BWST) isolation valve to open (269/91-010).
	LOOP at Vermont Yankee with delayed recovery caused by communications and organization difficulties and travel time for support personnel (271/91-009).
	Improperly installed insulation on ADS valves at Peach Bottom 3 resulted in damaged control wiring and valve inoperability under certain containment conditions (278/91-017).
	LOOP at Pilgrim with a subsequent RCIC inverter trip caused by an electrical system disturbance when an RHR pump was started (293/91-024).
	LOOP at Zion 2 when a deluge system sprayed a transformer. One EDG was out of service for maintenance and one PORV was failed, which degraded feed and bleed (304/91-002).
	Unavailability of both EDGs at Millstone 2. Both EDGs exhibited erratic load control (336/91-009).
	Loss of service water (SW) at Arkansas Nuclear One, Unit 2. Errors during traveling screen maintenance resulted in debris clogging both operating SW pump strainers (368/91-012).
	LOOP at McGuire 1 (369/91-001).
	Failure of five uninterruptible power supplies at Nine Mile Point 2 following a transformer fault and scram, resulting in loss of rod position indication, control room annunciators, lighting, and communications. Because rod position could not be verified, ADS was inhibited (410/91-017).

	Relief valves in the high-pressure SI system at Millstone 3 were found to lift at normal system pressures. Flow through the relief valves could result in loss of injection and would result in loss of emergency sump inventory outside containment (423/91-011).
	Both EDGs failed surveillance testing at Perry due to unrelated causes (440/91-009).
10^{-5} to 10^{-4}	8 events
10^{-6} to 10^{-5}	6 events

3.3 Likely Sequences

Precursors with conditional probabilities of $\geq 10^{-4}$ that were identified for 1991 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
- Small-break LOCA with failure of HPR
- LOOP with failure of secondary-side cooling and feed and bleed
- LOOP with failure of emergency power, a resulting RCP seal LOCA, and failure to recover AC power prior to core uncovery
- LOOP with failure of emergency power and failure to recover AC power prior to battery depletion

BWRs
- LOOP with failure of emergency power and failure to recover AC power prior to battery depletion
- Small-break LOCA with failure of HPCI and ADS
- LOFW and failure of long-term core cooling

3.4 References

1. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., *Precursors to Potential Severe Core Damage Accidents; 1986, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 5 and 6), May 1988.*

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. *

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

GLOSSARY

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 an accident sequence precursor occurred, depends on the effectiveness of the remaining protective features and, in the case of precursors that do not include initiating events, the probability of such an initiator.

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

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

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

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

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

Core damage. See *severe core damage.*

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

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

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

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

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

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

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

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

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

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

Event sequence. A particular path on an event tree.

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

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

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

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

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

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

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

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

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

nsee Event Reports. Those reports submitted to NRC by utilities who operate nuclear plants as described in NUREG-1022. LERs describe abnormal operating occurrences at plants where, generally, the Technical Specifications have been violated.

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

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

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

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

Precursor. See *accident sequence precursor.*

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

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

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

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

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

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

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

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

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

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

Appendix A

ASP MODELS

Appendix A

ASP MODELS

Note: The models used in the analysis of 1991 precursors are the same as those used for 1989-90 precursors. Information concerning both the ASP modeling approach and the event trees is included in this report for the convenience of the reader.

This appendix provides information concerning the methods and models used to estimate event significance in the Accident Sequence Precursor (ASP) Program.

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.

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 loss of main feedwater (LOFW) within the model], loss of offsite power (LOOP), and small-break loss-of-coolant accident (LOCA). The event tree models are system-based and include a model applicable to each of eight plant classes: three for boiling-water reactors (BWRs) and five for pressurized-water reactors (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) anticipated transient without scram (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.

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.

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 (R.Y) 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 control room or at failed equipment.
R2	0.34	The failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from control room did not appear possible.
R3	0.12	The failure appeared recoverable in the required period from 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 or 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.*

* 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.

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 steam generator 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.

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.

Because the frequencies and failure probabilities used in the calculations are derived in part from data obtained across the light-water reactor (LWR) population, even though they are applied to sequences that are plant-class specific in nature, *the conditional probabilities determined for each precursor cannot be directly 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.

Example 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 main feedwater 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:

$$\begin{aligned}
p(cd) &= p[\text{seq. } 11] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times (1 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}] \\
&+ p[\text{seq. } 12] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times 8.4 \times 10^{-4}] \\
&+ p[\text{seq. } 13] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times 9.9 \times 10^{-5} \times (1 - 0.34) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times (1.0 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}] \\
&+ p[\text{seq. } 14] + p[\text{seq. } 15] + p[\text{seq. } 16] + p[\text{seq. } 17] \\
&= 7.7 \times 10^{-7}
\end{aligned}$$

$$\begin{aligned}
p(ATWS) &= p[\text{seq. } 18] \\
&= 3.0 \times 10^{-5}
\end{aligned}$$

2. The second example event involves failures that would prevent high-pressure injection (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 auxiliary feedwater (AFW) and main feedwater (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. Assumed unavailability of the service water train results in unavailability of one train of high-pressure injection, high-pressure recirculation, and AFW, all because of unavailability of cooling to the respective pumps. In this example, service water cooling of two motor-driven AFW pumps is assumed. An additional turbine-driven pump is assumed to be self-cooled. Since service water is not explicitly addressed in the ASP event trees, the probabilities of front-line systems impacted by the loss of service water are instead modified.

   Fig. 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 high-pressure recirculation (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 cooling pump (RCP) 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 service water 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.

Event Tree Changes from Pre-1988 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:

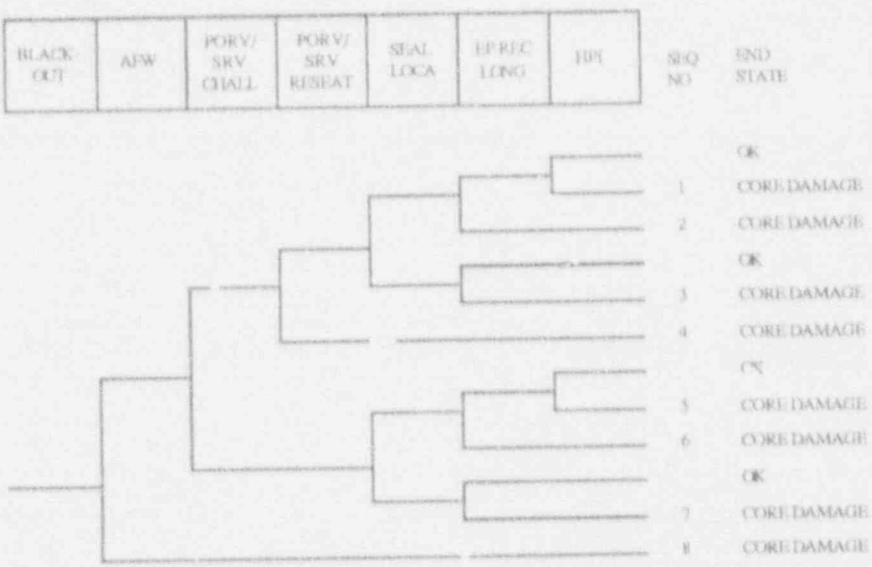
Core vulnerability sequence type	Revised end state
Stuck-open secondary-side relief valve with a failure of HPI in a PWR	Success
Steam generator (SG) depressurization and use of condensate system following failure of AFW, MFW, and feed and bleed in a PWR	Core damage (except for PWR Class H)
Use of containment 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 section 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 short-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. 3 and 4) 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. 5 and 6). 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, reactor coolant system 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 main feedwater pumps. BWR Class B consists of plants that have ICs but a separate high-pressure coolant injection (HPCI) system instead of FWCI. BWR Class C includes the modern plants that have neither ICs nor FWCI. However, they have a reactor core isolation cooling (RCIC) system that Classes A and B lack. The Class C plants could be separated into two subgroups, those plants with turbine-driven HPCI systems and those with motor-driven high-pressure core spray (HPCS) systems. This difference is addressed instead in the 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).*

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* 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.

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-/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.3 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.

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 difference 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 (RT) success, 11-39; LOOP with RT success, 40-69; small-break LOCA with RT success, 71-79, ATWS sequences, 91-99.

The trees are presented in the following order:

Figure No.	Event tree
A.4	PWR Class A nonspecific reactor trip
A.5	PWR Class A loss of offsite power
A.6	PWR Class A small-break loss-of-coolant accident
A.7	PWR Classes B and D nonspecific reactor trip
A.8	PWR Classes B and D loss of offsite power
A.9	PWR Classes B and D small-break loss-of-coolant accident
A.10	PWR Class G nonspecific reactor trip
A.11	PWR Class G loss of offsite power
A.12	PWR Class G small-break loss-of-coolant accident
A.13	PWR Class H nonspecific reactor trip
A.14	PWR Class H loss of offsite power
A.15	PWR Class H small-break loss-of-coolant accident
A.16	BWR Class A nonspecific reactor trip
A.17	BWR Class A loss of offsite power
A.18	BWR Class A small-break loss-of-coolant accident
A.19	BWR Class B nonspecific reactor trip
A.20	BWR Class B loss of offsite power
A.21	BWR Class B small-break loss-of-coolant accident
A.22	BWR Class C nonspecific reactor trip
A.23	BWR Class C loss of offsite power
A.24	BWR Class C small-break loss-of-coolant accident

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 Sec. 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 Reactor Trip

The PWR nonspecific reactor trip event tree constructed for plant Classes B and D is shown in Fig. A.7. The event-tree 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 steam-line break (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 steam generators. Successful AFW operation requires flow from one or more AFW pumps to one or more steam generators over a period of time ranging from 12 to 24 h (typically, one pump to one steam generator 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 auxiliary feedwater 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 steam generators (MFW and AFW) are unavailable, core cooling can be accomplished on most plants by establishing a feed and bleed operation. This operation (1) allows heat removal via discharge of reactor coolant to the containment through the PORVs and (2) reactor coolant system (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 low-pressure injection 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 function, taking suction directly from the containment sump without the aid of the low-pressure pumps. Decay heat removal 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, decay heat removal 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 high-pressure injection 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 auxiliary feedwater unavailability. If main or auxiliary feedwater cannot be recovered, the atmospheric dump valves can be used to depressurize the steam generators 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 containment cooling 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 auxiliary feedwater is available, these valves do not lift. In the case where both main and auxiliary feedwater 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 auxiliary feedwater 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 steam generators to the point that the condensate pumps can be used for steam generator cooling. In the event of main and auxiliary feedwater unavailability, failure to depressurize one steam generator 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 steam generator is assumed adequate. Unavailability of the condensate pumps in the event of failure to

recover main and auxiliary feedwater 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 decay heat removal during high-pressure recirculation. Use of CSR for decay heat removal was assumed to be required if AFW and MFW were unavailable. LPI pumps are required to provide suction to the HPI pumps during recirculation. The event-tree branches and sequences are discussed further below.

1. Initiating event (transient). 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 decay heat removal during high-pressure recirculation 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 decay heat removal 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 like 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 of Offsite Power

The event trees constructed define representative plant responses to a LOOP. A LOOP (without turbine runback on plants with this feature) will result in reactor trip due to unavailability of power to the control rod drive 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 are addressed in the ASP Program.

2. Reactor trip given LOOP. Unavailability of power to the control rod drive 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, diesel generators 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 diesel generators 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 steam generator 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 emergency power system 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 service water 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 diesel generator (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 emergency power system.

10. PORV open (for feed and bleed). The success requirements for this branch are similar to those following a nons,ecific 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 . ORV 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, decay heat removal 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 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 decay heat removal for sequences in which secondary-side cooling is unavailable.

The event tree constructed for a PWR Class H LOOP is shown in 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 decay heat removal method. The condensate system is assumed unavailable following a LOOP, which limits the diversity of decay heat removal 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 lants with the addition of the CSR branch, which is required for successful feed and bleed. At Class A plants, decay heat removal during high-pressure recirculation is accomplished by the CSR system; whereas at Class B and D plants, decay heat removal 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 high-pressure injection 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 high-pressure injection 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 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 steam generator 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).

Two alternate recovery actions can potentially mitigate the effects of an initiating event, if normal and alternate mitigation systems are unavailable. The first of these is the use of SG depressurization and condensate pumps if AFW, MFW, and feed and bleed are unavailable on PWR Classes A, B, D, and G. This recovery action requires that the condensate system be available (even though AFW and MFW are unavailable), and that adequate depressurization capability exist on a plant. Procedures to support this action are known to exist on some plants.

The second recovery action is depressurization follow[illegible] all-break LOCA to the initiation pressure of the decay heat removal 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 two alternate recovery actions are qualitatively considered for high probability sequences when analyzing precursors, although they are not modeled on the event trees.

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 Sec. 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), control rod drive (CRD) (non-LOCA situations), feedwater coolant injection (FWCI)] Main feedwater Low-pressure injection systems following blowdown [low-pressure coolant injection (LPCI) (BWR Classes B and C), low-pressure core spray (LPCS), residual heat removal (RHR) service water or equivalent]

Short-term core heat removal	Power conversion system (PCS) 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 PCS is faulted) requires use of the RHR system for containment heat removal in the long term.
Long-term core heat removal	PCS Isolation condenser (BWR Class A) RHR [shutdown cooling (SDC or suppression pool (SP) cooling modes (BWR Class C)] Shutdown cooling (BWR Classes A and B) Containment cooling (BWR Class A) LPCI [containment cooling 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 reactor protection system (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. Upon successful reactor scram, continued operation

of the power conversion system (PCS) would allow continued heat removal via the main condenser. This was considered successful mitigation of the transient. Continued operation of the PCS requires the main steam isolation valves (MSIVs) to remain open and the operation of the condenser, the turbine bypass system, 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. SRVs 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 (FW) to the RPV will keep the core from becoming uncovered. This, in combination with successful long-term decay heat removal, will mitigate the transient, preventing core damage. For plants with turbine-driven feed pumps, the PCS failure with subsequent FW success cannot involve MSIV closure, or loss of condenser vacuum, because this would disable the feed pumps.

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

8. Reactor core isolation cooling. The reactor core isolation cooling 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 manually 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. Control rod drive 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 SRVs or the automatic depressurization system (ADS). In the event that short-term decay heat removal 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. Low-pressure core spray. Low-pressure injection can be provided by the LPCS system if required. The LPCS system performs the same functions as the LPCI system (described below) except that the coolant, which is drawn from the suppression pool or the condensate storage tank, is sprayed over the core.

12. Low-pressure coolant injection. The LPCI system can provide short-term heat removal and cooling water makeup if the reactor has been depressurized to the operating range of the low-head RHR pumps. At Class C plants, LPCI is a mode of the RHR system; thus, the RHR pumps operate during LPCI. LPCI takes suction from the suppression pool or the condensate storage tank 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 mode. In this mode, the RHR system provides normal long-term decay heat removal. 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 shutdown cooling (SDC) success following successful reactor scram and high- or low-pressure injection of water to the RPV will prevent core damage.

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

15. Residual heat removal 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 service water 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 isolation condensers 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 shutdown cooling 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 isolation condenser (IC) system can provide for decay heat removal 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. Feedwater or feedwater coolant injection. Either FW or FWCI can provide short-term transient mitigation. When FW or FWCI is required and is successful, long-term decay heat removal is required for complete transient mitigation. (PCS unavailability is assumed prior to FW or FWCI demand.) FWCI or FW 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. Control rod drive pumps.

8. Depressurization via SRVs or ADS.

9. Low-pressure core spray.

10. Fire water or other. Fire water or other raw water systems can provide a capability similar to that provided by the service water/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. Shutdown cooling. Like the RHR system at Class C plants, the SDC system is a closed-loop system that performs the long-term decay heat removal function by circulating primary coolant from the reactor through the system's heat exchangers and back to the reactor vessel. Success requires the operation of at least one SDC loop. Long-term decay heat removal 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 decay heat removal, the containment cooling system can remove decay heat. The system utilizes dedicated containment cooling pumps, drawing suction from the suppression pool, passing it through heat exchangers where heat is rejected to the service water system and then either returning it directly to the suppression pool or spraying it into the dry well.

The event tree constructed for a BWR 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 low-pressure coolant injection. Also, at Class B BWRs, the containment cooling 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.)

<u>BWR Loss of Offsite Power</u>

The event trees 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 diesel generators at almost all plants. The diesel generators receive an initiation signal when an undervoltage condition is detected. Emergency power success requires the starting and loading of a sufficient number of diesel generators 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. High-pressure coolant injection (or high-pressure core spray) or reactor core isolation cooling. 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 emergency power system. HPCI and RCIC only require DC power and sufficient steam to operate the pump turbines. HPCS systems utilize a motor-driven pump but are diesel-backed and utilize dedicated service water cooling.

7. Control rod drive 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 SRVs or the ADS.

9. LPCS, LPCI, or RHR service water.

10. RHR shutdown cooling mode or RHR suppression pool 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 decay heat removal required for transient mitigation. If emergency power fails, it must be recovered to power long-term decay heat removal equipment. However, long-term decay heat removal is not required until 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 isolation condensers 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 isolation condenser success, as long as no transient-induced LOCA is initiated. In the emergency power failure sequences, the isolation condenser 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. Isolation condensers. Following successful reactor scram, the IC system can provide enough decay heat removal, 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. Control rod drive pumps.

9. Depressurization via SRVs or ADS.

10. Low-pressure core spray, 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. Shutdown cooling 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 low-pressure coolant injection. At Class B BWRs the containment cooling 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 shutdown cooling-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.)

<u>BWR Loss-of-Coolant Accident</u>

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 high-pressure injection 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 low-pressure injection 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. High-pressure coolant injection or high-pressure core spray. HPCI (or HPCS, depending on the plant) can provide the required inventory makeup.

4. Depressurization via SRVs 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.

6. Residual heat removal (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 decay heat removal. Long-term decay heat removal 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, suppression pool and containment cooling systems are independent of the shutdown cooling 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. Feedwater coolant injection. 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 SRVs 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 low-pressure injection 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 decay heat removal 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 low-pressure coolant injection. At Class B BWRs the containment cooling 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 isolation condensers on BWR Classes A and B) and cannot be recovered in the short term, the use of the control rod drive 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.

Two alternate recovery actions can potentially mitigate the effects of an initiating event, if normal and alternate mitigation systems are unavailable. The first of these is the use of the condensate system for low-pressure injection. 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 second recovery action is the use of containment venting for long-term decay heat removal, provided an injection source is available. This core cooling method has been addressed in some PRAs.

The potential use of these two alternate recovery actions are qualitatively considered for high probability sequences when analyzing precursors, although they are not modeled on the event trees.

A.4 Branch Probability Estimates

Branch probability estimates used in the 1988-1991 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. 7 and in Ref. 2.

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 decay heat removal 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 1988-1991 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 should be assumed to be higher. This assumption, plus a revised approach for actions outside the control room, have not yet been incorporated into the ASP models.) Operator action failure probabilities used in the 1988-1991 calculations are shown in Table A.14.

A.5 Reference Event Calculations

Conditional core damage probability estimates were also calculated for nonspecific reactor trip, nonrecoverable LOFW, and unavailabilities in certain single-train BWR systems (HPCI, HPCS, RCIC, and control rod drive cooling). These calculations indicate the relative importance of these events, which are too numerous to warrant individual calculation. The results of these calculations 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 an 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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2. J. W. Minarick, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp., *Revised LOOP Recovery and PWR Seal LOCA Models*, Technical Letter Report ORNL/NRC/LTR-89/11, August 1989.*

3. J. W. Minarick and C. A. Kukielka, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., and Science Applications, Inc., *Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report*, USNRC Report NUREG/CR-2497 (ORNL/NOAC-232, Vol. 1 and 2), 1982.*

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5. M. Modarres, E. Lois, and P. Amico, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *LER Categorization Report*, University of Maryland, College Park, MD, Nov. 13, 1984.

6. E. Lois, *A Class-Specific Approach to Nuclear Power Plant Safety Studies with Applications*, PhD Dissertation, University of Maryland, College Park, MD, 1985.

7. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., *Precursors to Potential Severe Core Damage Accidents; 1986, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 5 and 6), May 1988.*

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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 x 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
ANO - Unit 1	PWR Class D
ANO - Unit 2	PWR Class G
Beaver Valley 1	PWR Class A
Beaver Valley 2	PWR Class A
Big Rock Point	BWR Class A
Browns Ferry 1	BWR Class C
Browns Ferry 2	BWR Class C
Browns Ferry 3	BWR Class C
Braidwood 1	PWR Class B
Braidwood 2	PWR Class B
Brunswick 1	BWR Class C
Brunswick 2	BWR Class C
Byron 1	PWR Class B
Byron 2	PWR Class B
Callaway 1	PWR Class B
Calvert Cliffs 1	PWR Class G
Calvert Cliffs 2	PWR Class G
Catawba 1	PWR Class B
Catawba 2	PWR Class B
Clinton 1	BWR Class C
Comanche Peak	PWR Class B
Cook 1	PWR Class B
Cook 2	PWR Class B
Cooper Station	BWR Class C
Crystal River 3	PWR Class D
Davis-Besse	PWR Class B
Diablo Canyon 1	PWR Class B
Diablo Canyon 2	PWR Class B
Dresden 2	BWR Class B
Dresden 3	BWR Class B
Duane Arnold	BWR Class C
Farley 1	PWR Class B
Farley 2	PWR Class B
Fermi 2	BWR Class C
Fitzpatrick	BWR Class C
Fort Calhoun	PWR Class G
Ginna	PWR Class B
Grand Gulf 1	BWR Class C

Table A.3 ASP reactor plant classes (cont.)

Plant name	Plant class
Haddam Neck	PWR Class B
Harris 1	PWR Class B
Hatch 1	BWR Class C
Hatch 2	BWR Class C
Hope Creek 1	BWR Class C
Indian Point 2	PWR Class B
Indian Point 3	PWR Class B
Kewaunee	PWR Class B
LaCrosse	Unique
LaSalle 1	BWR Class C
LaSalle 2	BWR Class C
Limerick 1	BWR Class C
Limerick 2	BWR Class C
Maine Yankee	PWR Class B
McGuire 1	PWR Class B
McGuire 2	PWR Class B
Millstone 1	BWR Class A
Millstone 2	PWR Class G
Millstone 3	PWR Class A
Monticello	BWR Class C
Nine Mile Point 1	BWR Class A
Nine Mile Point 2	BWR Class C
North Anna 1	PWR Class A
North Anna 2	PWR Class A
Oconee 1	PWR Class D
Oconee 2	PWR Class D
Oconee 3	PWR Class D
Oyster Creek	BWR Class A
Palisades	PWR Class G
Palo Verde 1	PWR Class H
Palo Verde 2	PWR Class H
Palo Verde 3	PWR Class H
Peach Bottom 2	BWR Class C
Peach Bottom 3	BWR Class C
Perry 1	BWR Class C
Pilgrim 1	BWR Class C
Point Beach 1	PWR Class B
Point Beach 2	PWR Class B

Table A.3 ASP reactor plant classes (cont.)

Plant name	Plant class
Prairie Island 1	PWR Class B
Prairie Island 2	PWR Class B
Quad Cities 1	BWR Class C
Quad Cities 2	BWR Class C
Rancho Seco	PWR Class D
River Bend 1	BWR Class C
Robinson 2	PWR Class B
Salem 1	PWR Class B
Salem 2	PWR Class B
San Onofre 1	Unique
San Onofre 2	PWR Class H
San Onofre 3	PWR Class H
Seabrook 1	PWR Class B
Sequoyah 1	PWR Class B
Sequoyah 2	PWR Class B
South Texas 1	PWR Class B
St. Lucie 1	PWR Class G
St. Lucie 2	PWR Class G
Summer 1	PWR Class B
Surry 1	PWR Class A
Surry 2	PWR Class A
Susquehanna 1	BWR Class C
Susquehanna 2	BWR Class C
Three Mile Island 1	PWR Class D
Trojan	PWR Class B
Turkey Point 3	PWR Class B
Turkey Point 4	PWR Class B
Vermont Yankee	BWR Class C
Vogtle 1	PWR Class B
Vogtle 2	PWR Class B
WNPSS 2	BWR Class C
Waterford 3	PWR Class H
Wolf Creek 1	PWR Class B
Yankee Rowe	PWR Class B
Zion 1	PWR Class B
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

Table A.4. PWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
		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)
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 asubsequent 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)

Table A.6. PWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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 train(s) success. (PWR Classes A, B, D, G, and H)
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)

Table A.6. PWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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)

Table A.7. PWR LOOP sequences summary

Seq. No.	End State	RT/ LOOP	EP	AFW	RV Chall	RV Reseat	Seal LOCA	RCP 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				

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 [illegible] 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 if 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 if 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)

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

Sequence No.	End state	Description
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.

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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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 valve challenge and unsuccessful reseat, unsuccessful main feedwater and followed by successful vessel depressurization and low-pressure core spray.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
29	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.
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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
		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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
		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.
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.

Table A.10. BWR transient core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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, and 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 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 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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
46	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 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 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, and successful feedwater coolant injection.
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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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.

Table A 11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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) followinga 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 (cont.)

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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
63	Core damage	Similar to Sequence 48 except the safety relief valves are not challenged.
64	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 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.
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) followinga 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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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

Table A 11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
		with successful emergency power, reactor scram, and 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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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 sytem 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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.
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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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.

Table A.11. BWR LOOP core damage and ATWS sequences (cont.)

Sequence No.	End state	Description
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 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 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.

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

Sequence No.	End state	Description
		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.
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.

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

Sequence No.	End state	Description
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.

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

Sequence No.	End state	Description
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*
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}

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

Table A.14 Operator action failure probabilities

Operator 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	1.3×10^{-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 section 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 [illegible]ction fails
HPR	[illegible]ion fails
LOCA	[illegible] accident
LOOP	loss of [illegible] 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
	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	unavailability of main feedwater
FWCI	failure of feedwater coolant injection system
HPCI or HPCS	high-pressure coolant injection or high-pressure core spray fails

Table A.16 Abbreviations used in event trees (cont.)

Abbreviation	Description
IC/IC 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 system 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

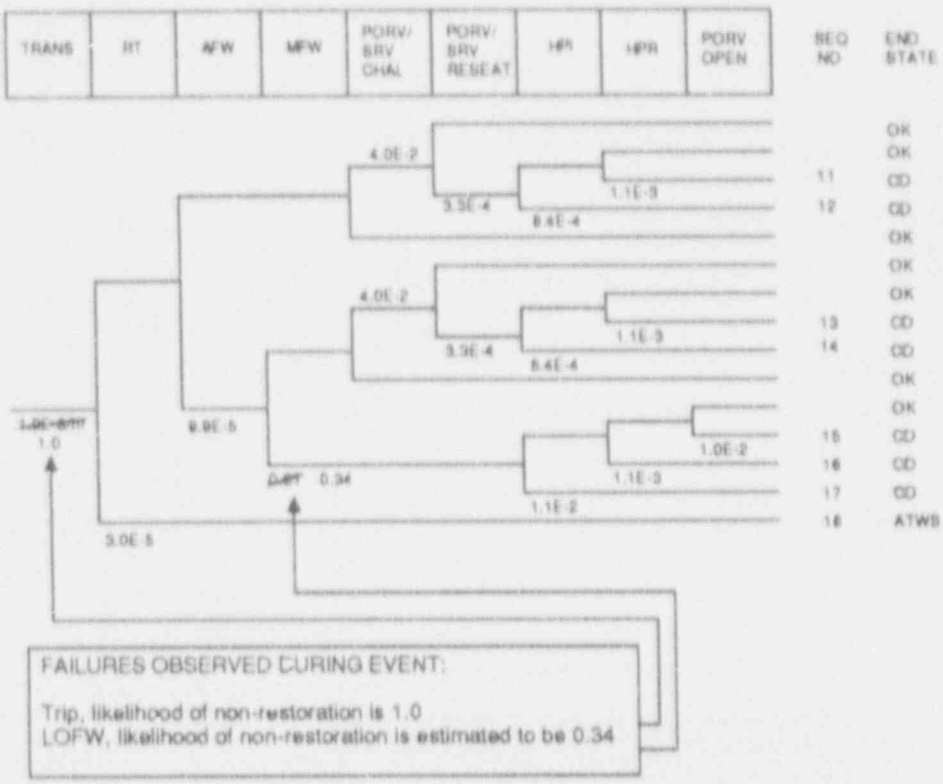

Figure A.1. Example initiator calculation

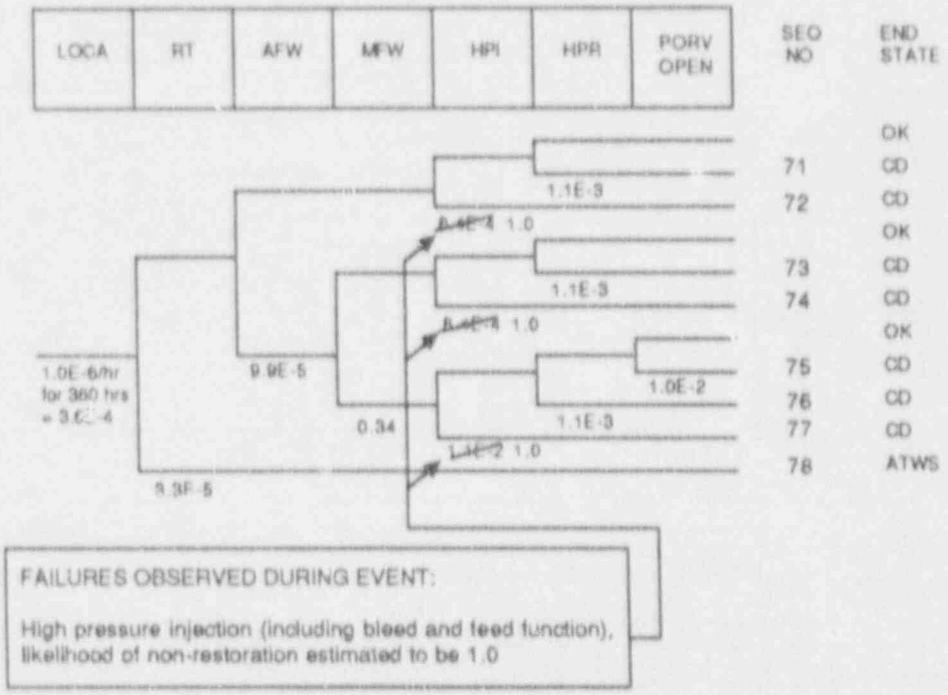

Figure A.2. Example unavailability calculation

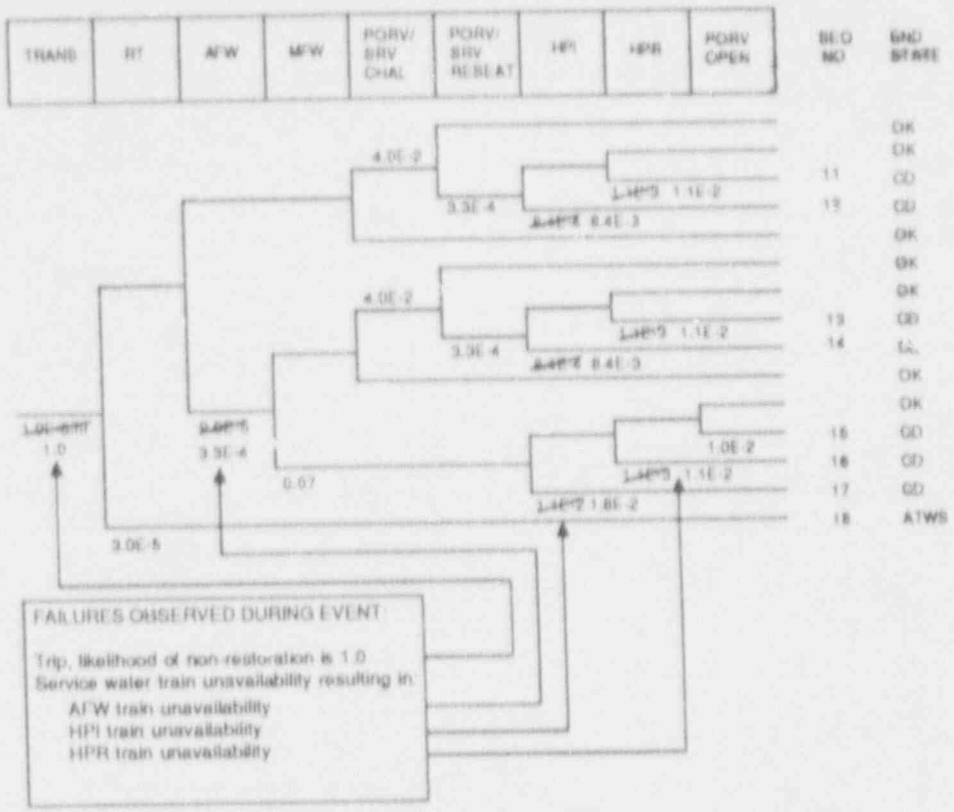

Figure A.3. Example trip with support system degraded

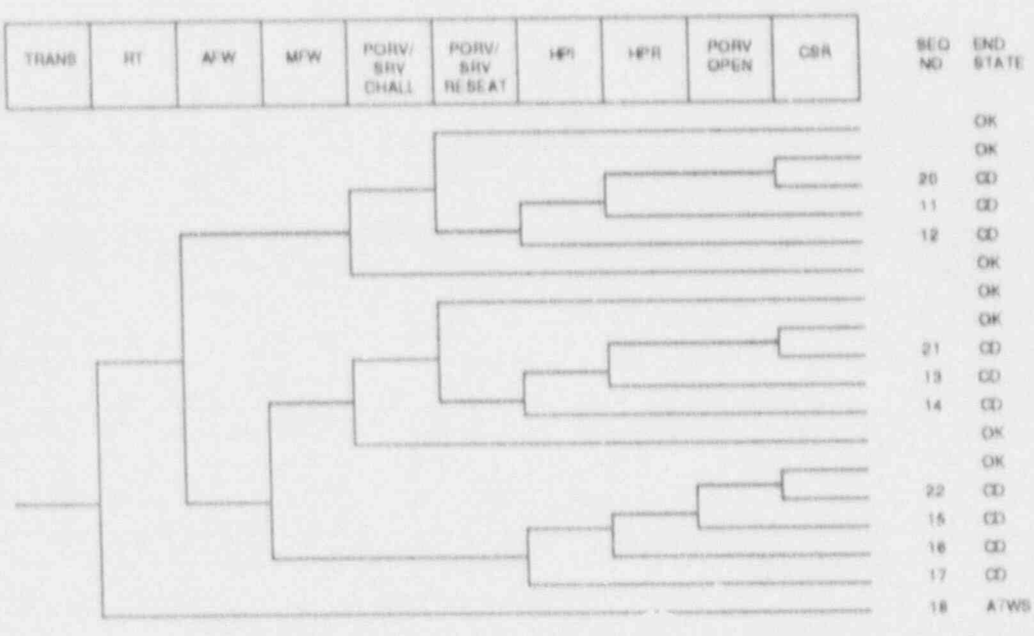

Figure A.4. PWR class A nonsp... ic reactor trip

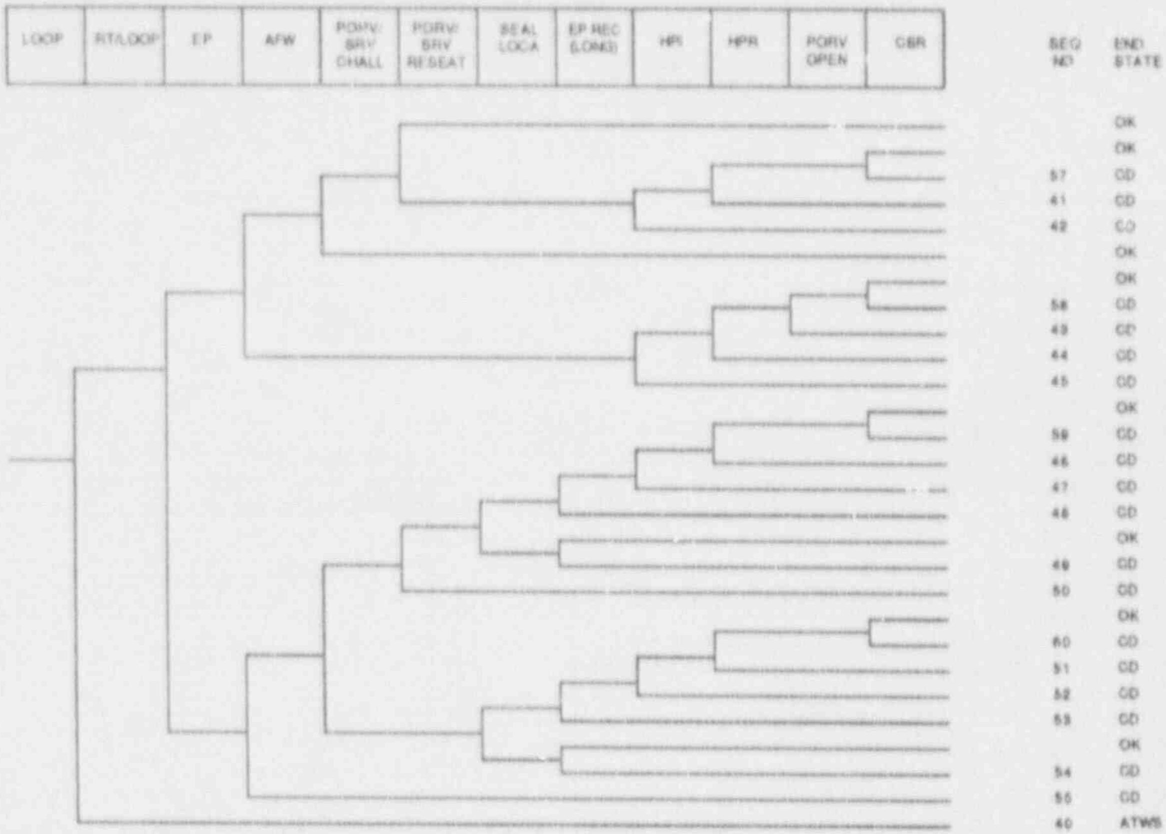

Figure A.5. PWR class A loss of offsite power

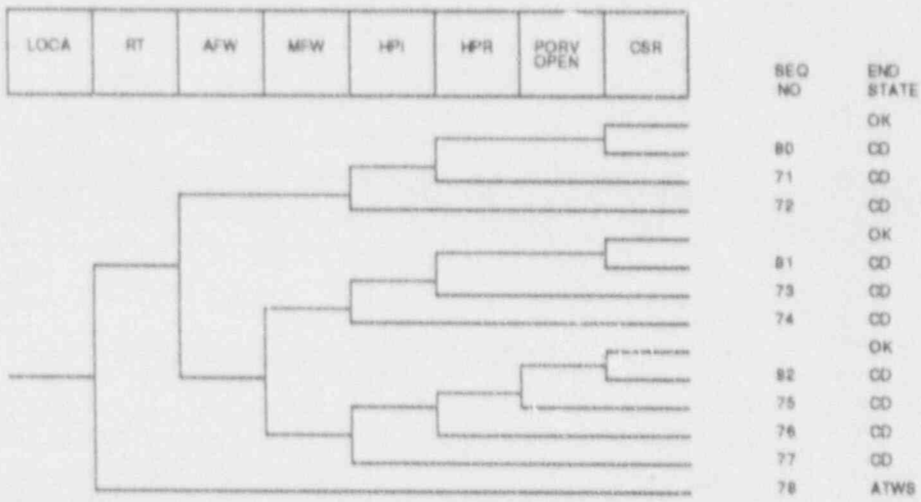

Figure A.6. PWR class A small-break loss-of-coolant accident

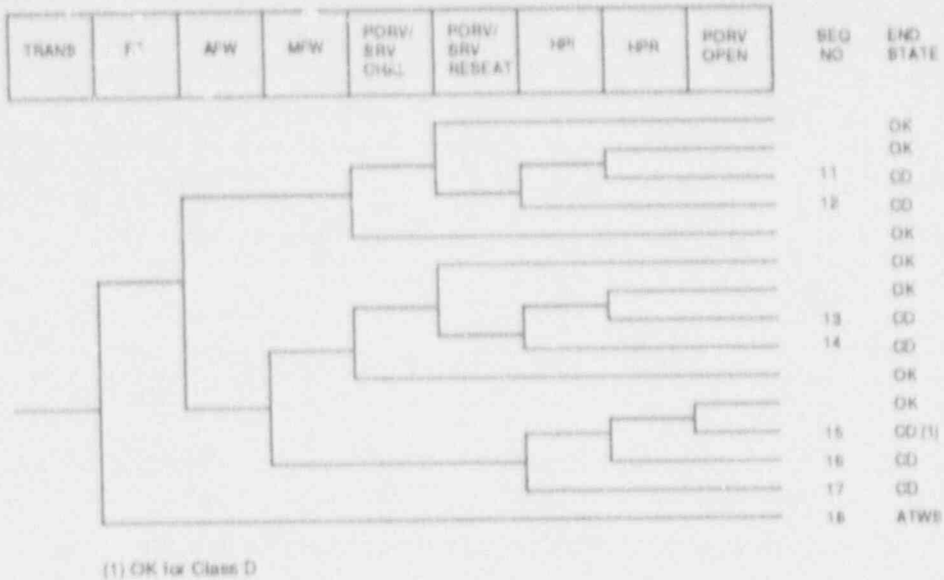

Figure A.7. PWR class B and D nonspecific reactor trip

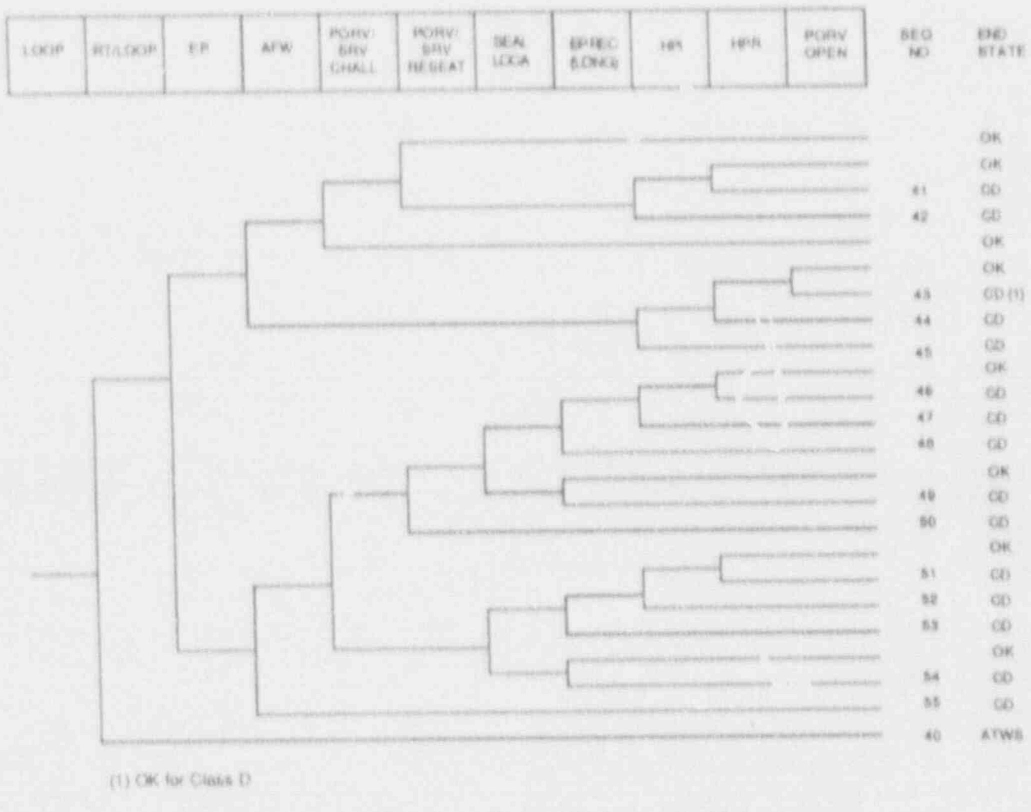

Figure A.8. PWR class B and D loss of offsite power

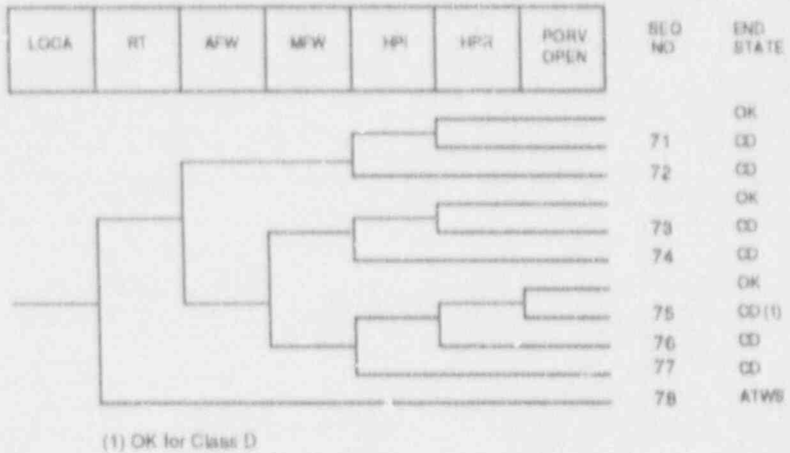

Figure A.9. PWR class B and D small-break loss-of-coolant accident

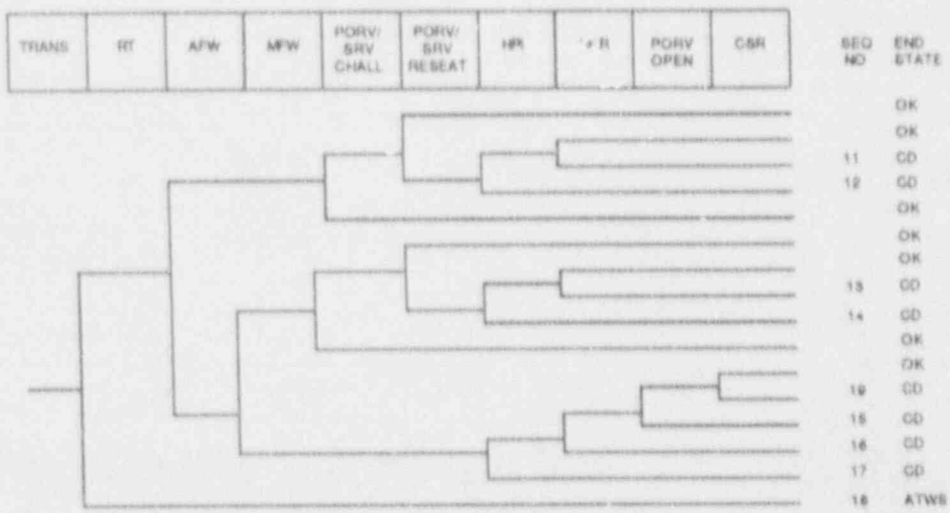

Figure A.10. PWR class G nonspecific reactor trip

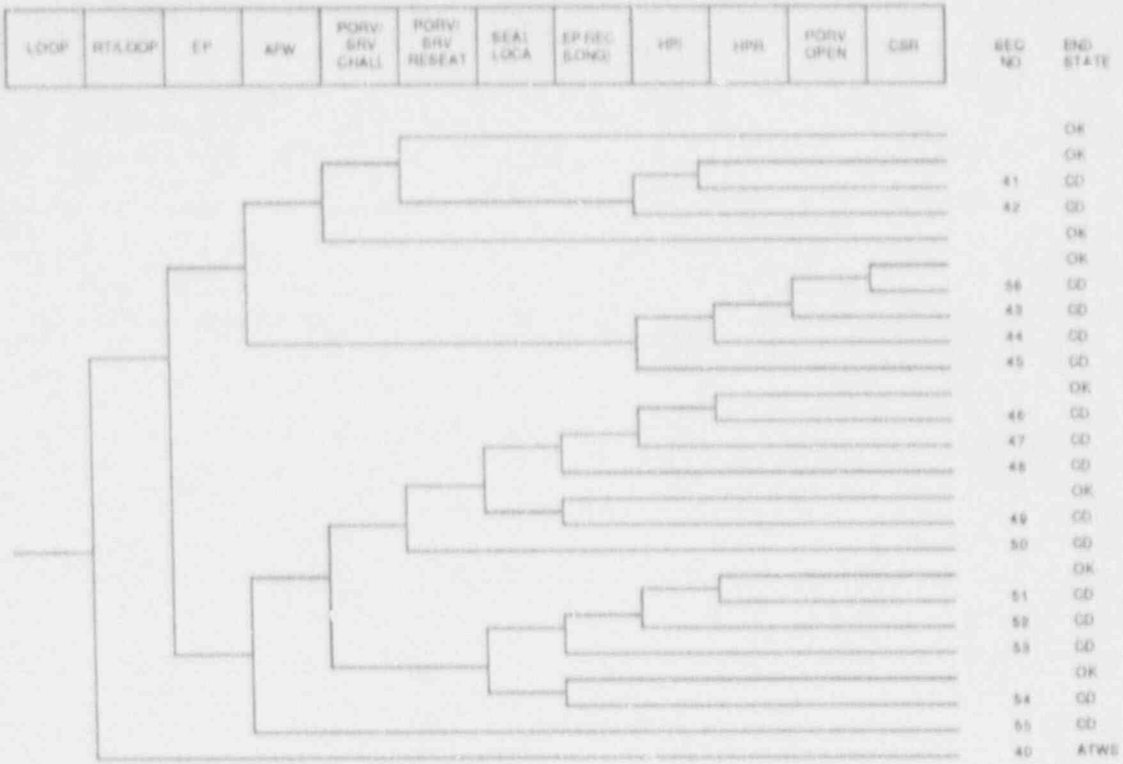

Figure A.11. PWR class G loss of offsite power

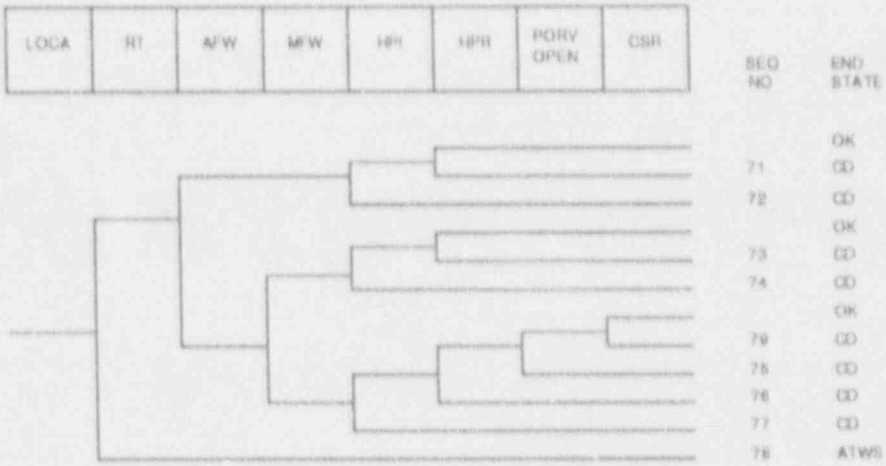

Figure A.12. PWR class G small-break loss-of-coolant accident

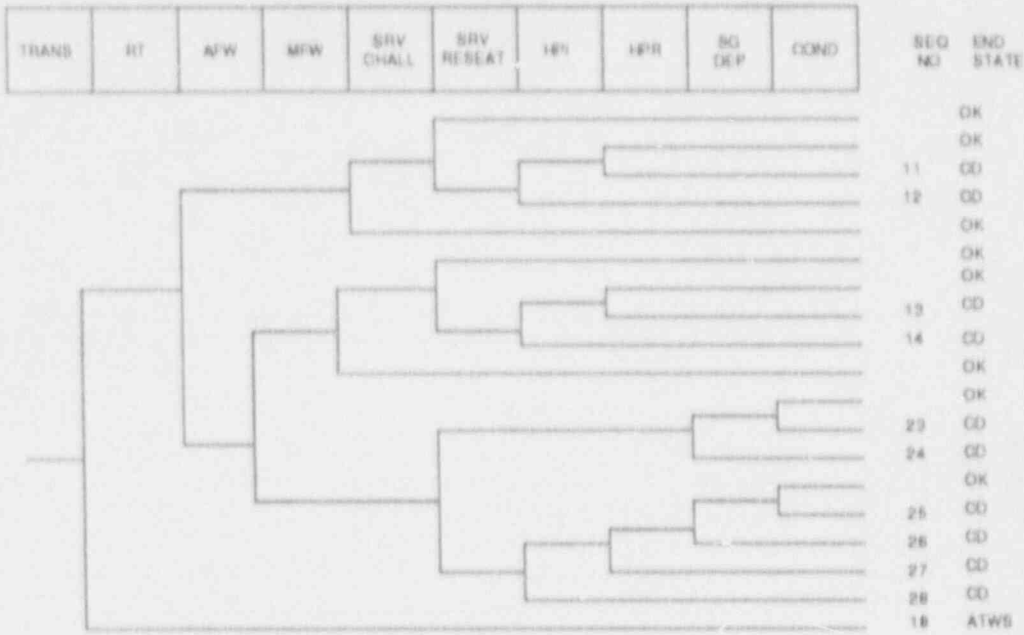

Figure A.13. PWR class H nonspecific reactor trip

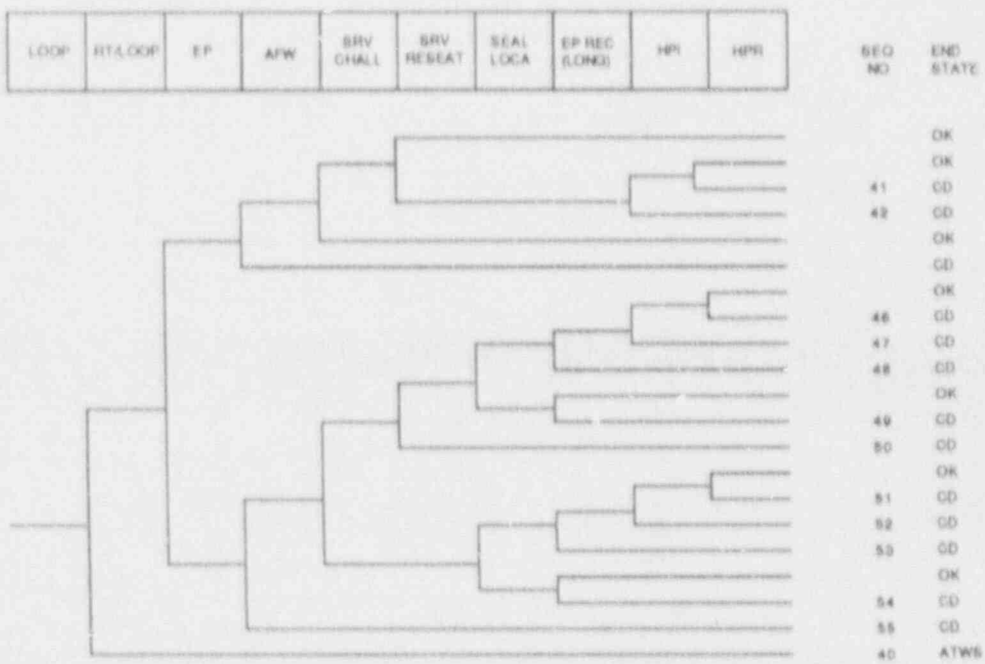

Figure A.14. PWR class H loss of offsite power

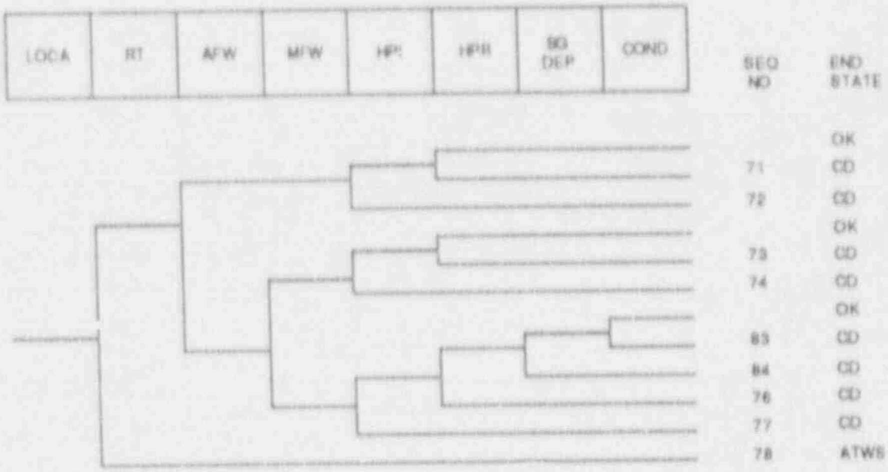

Figure A.15. PWR class H small-break loss-of-coolant accident

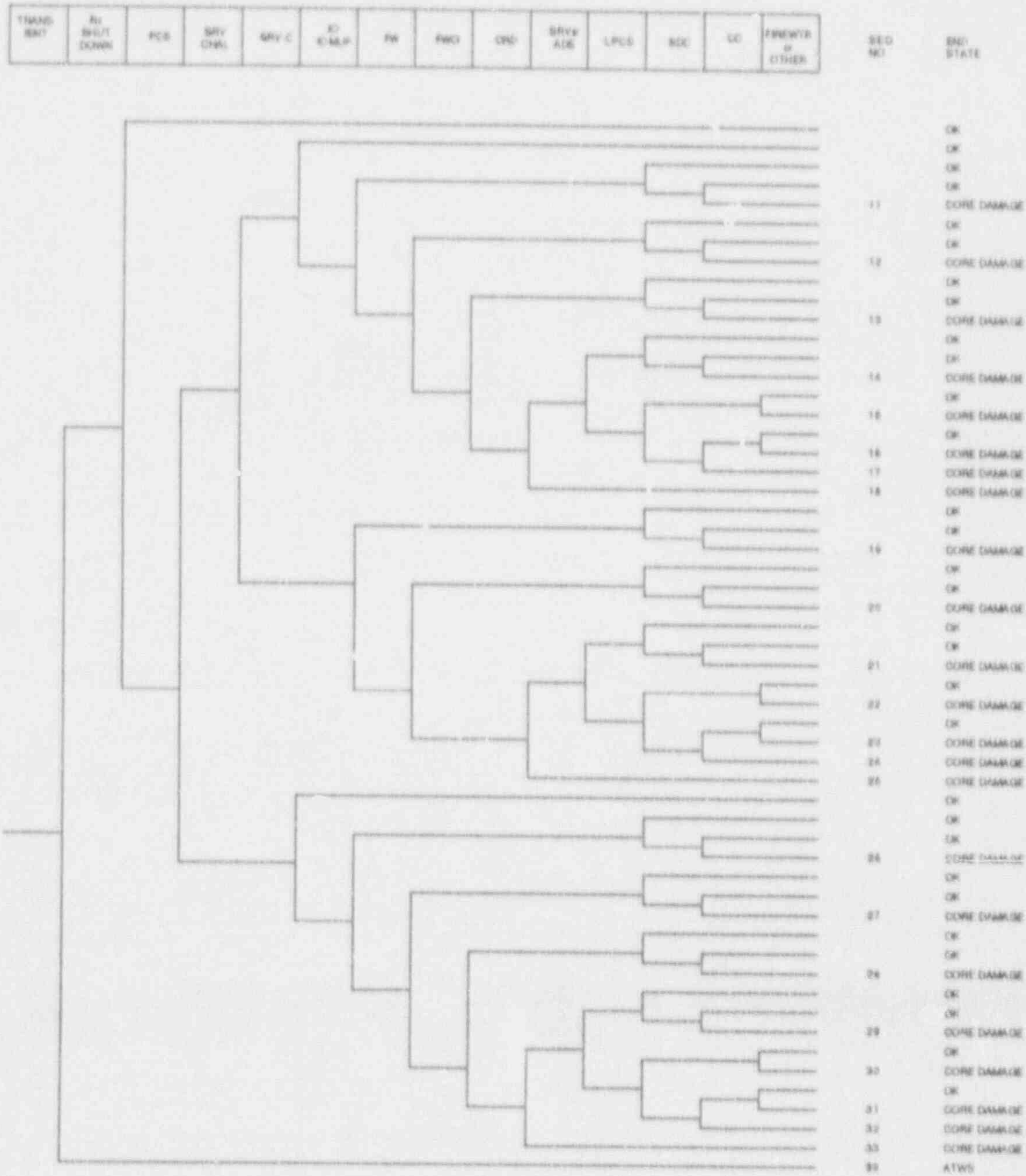

Figure A.16. BWR class A nonspecific reactor trip

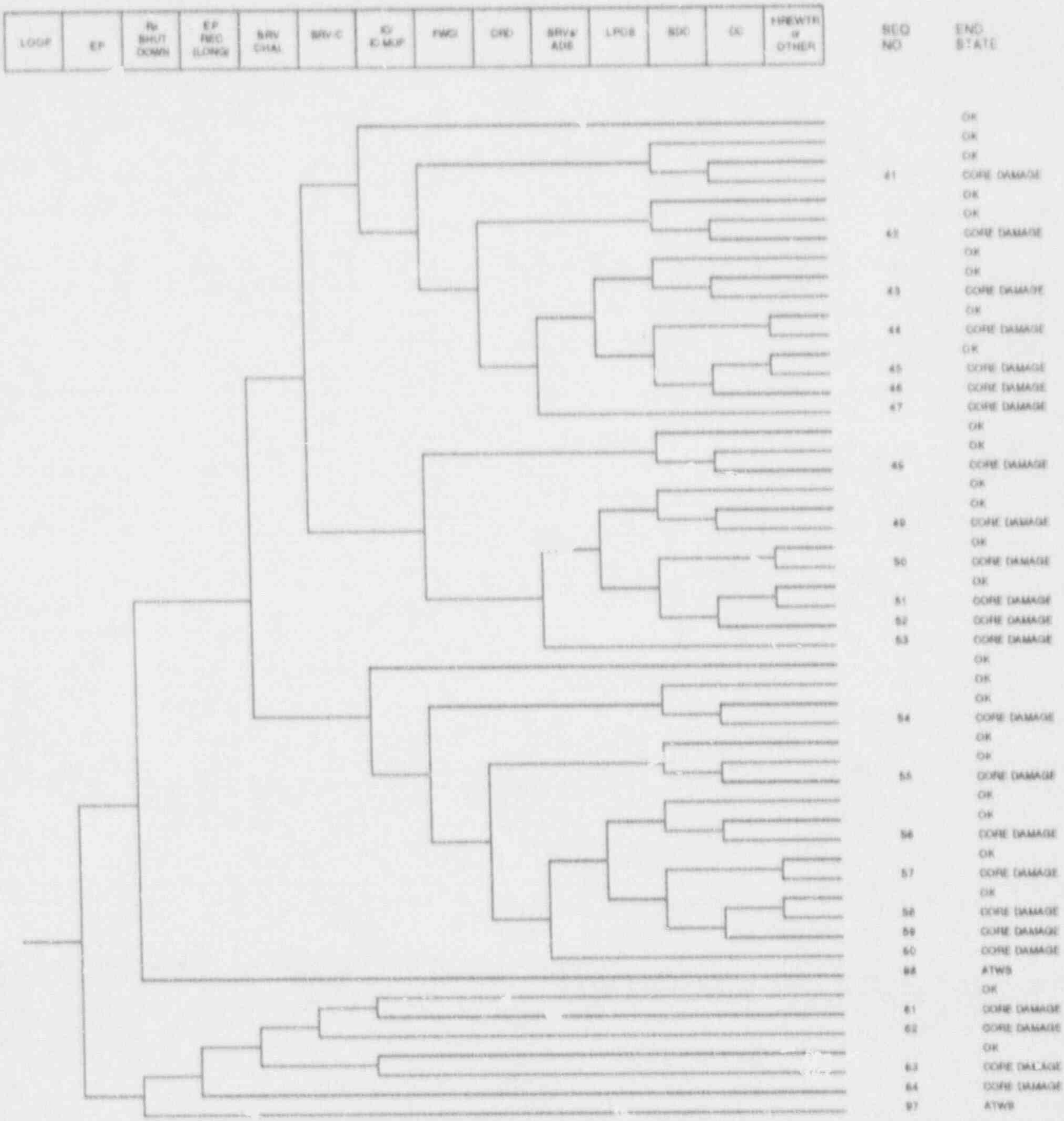

Figure A.17. BWR class A loss of offsite power

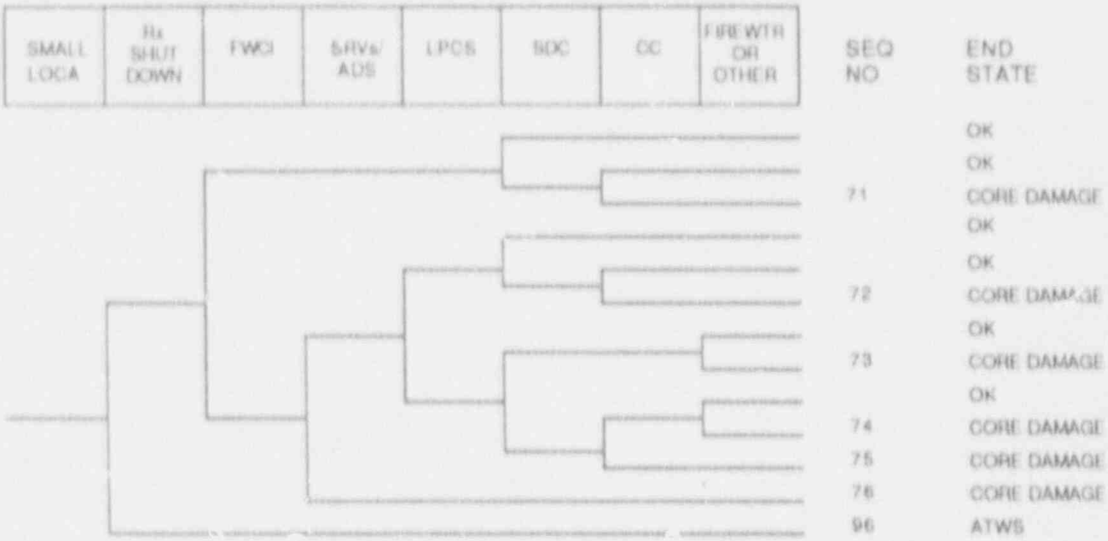

Figure A.18. BWR class A small-break loss-of-coolant accident

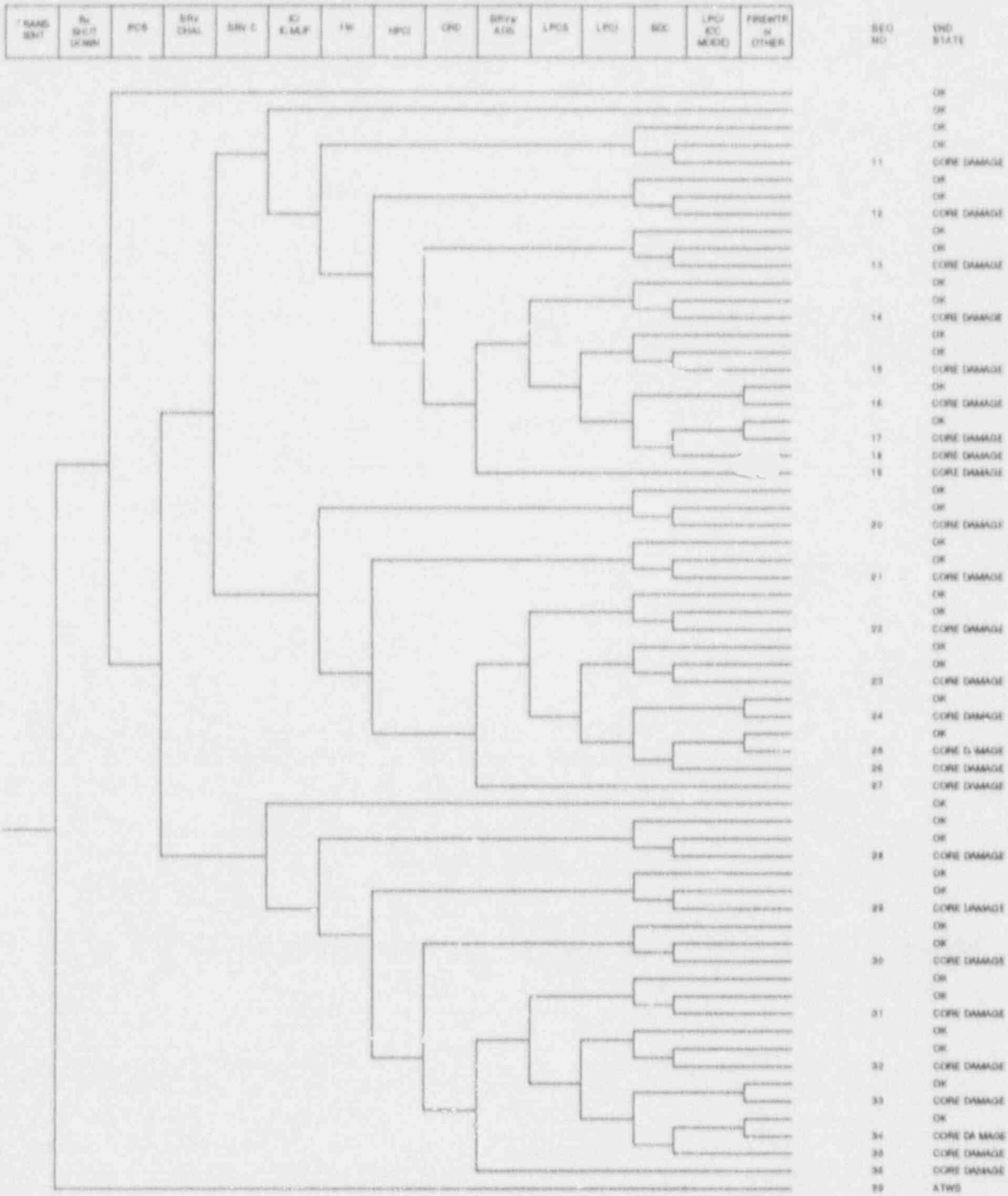

Figure A.10 BWR class B nonspecific reactor trip

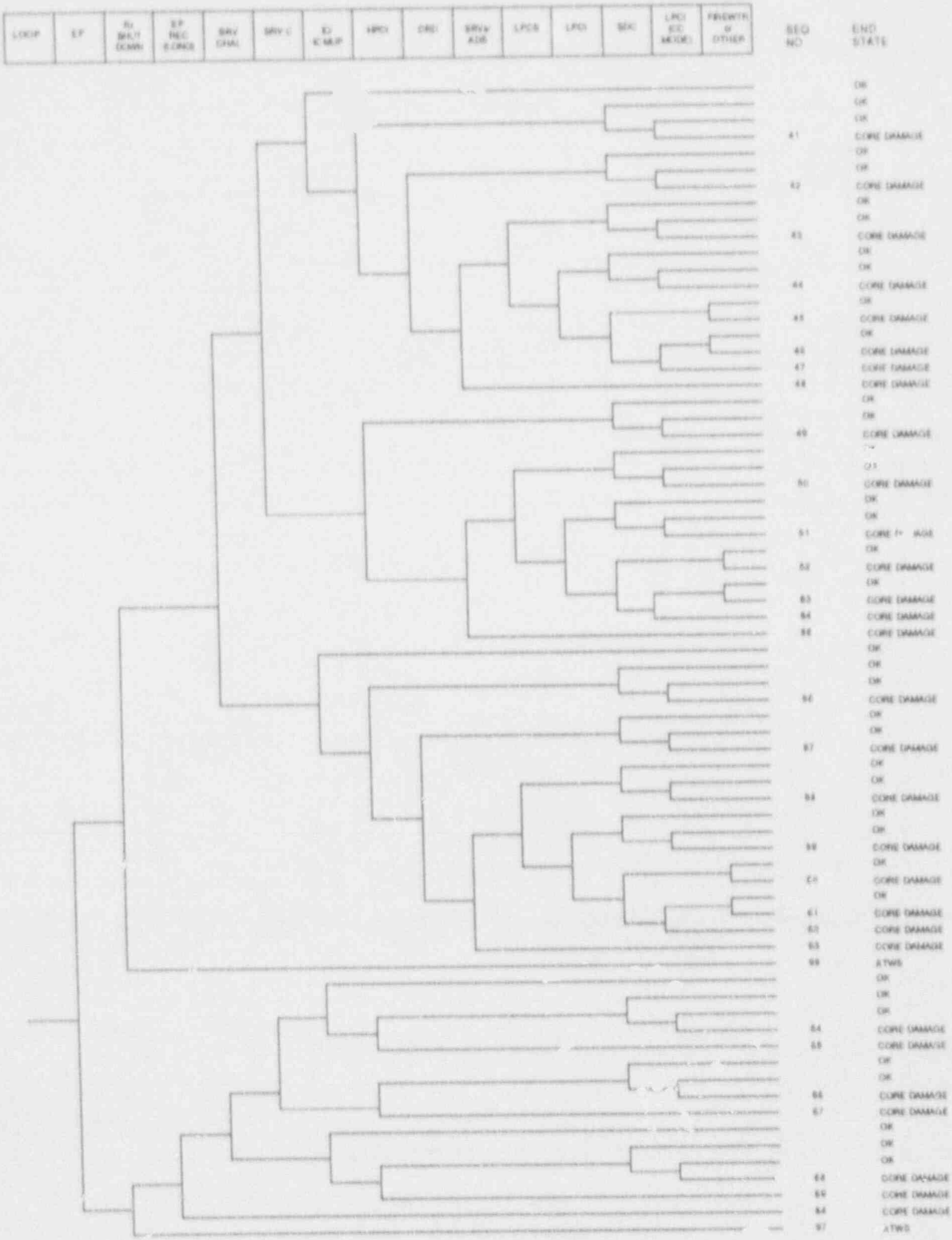

Figure A.20. BWR class B loss of offsite power

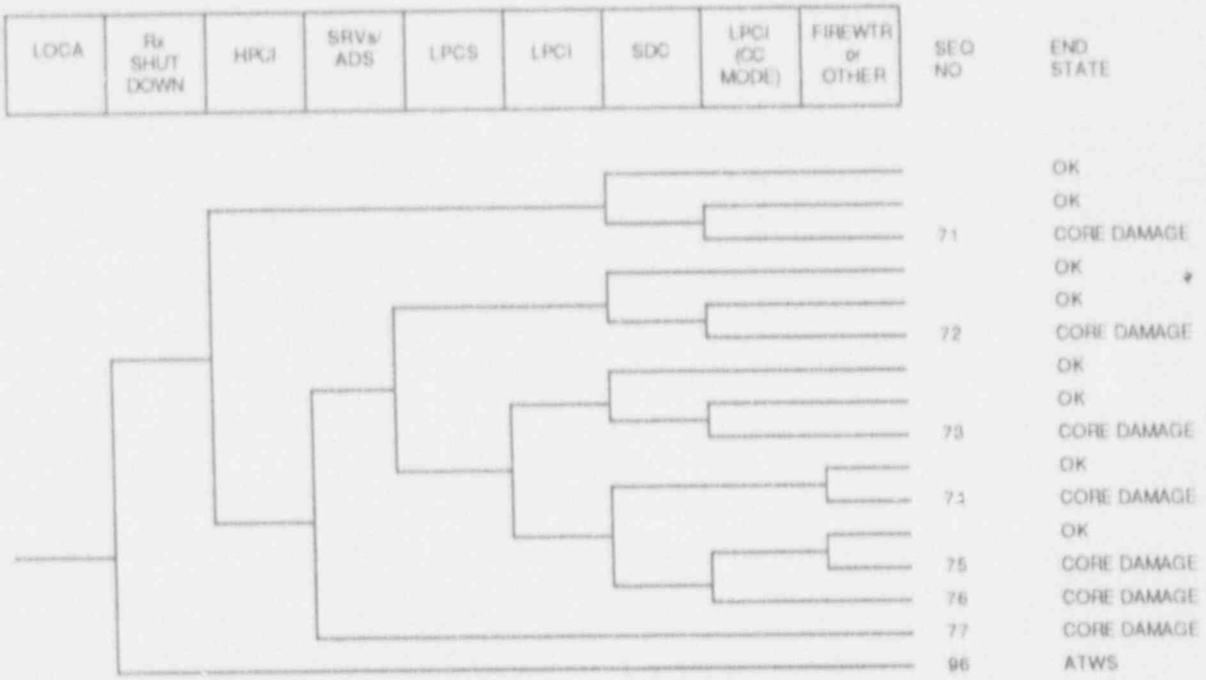

Figure A.21. BWR class B small-break loss-of-coolant accident

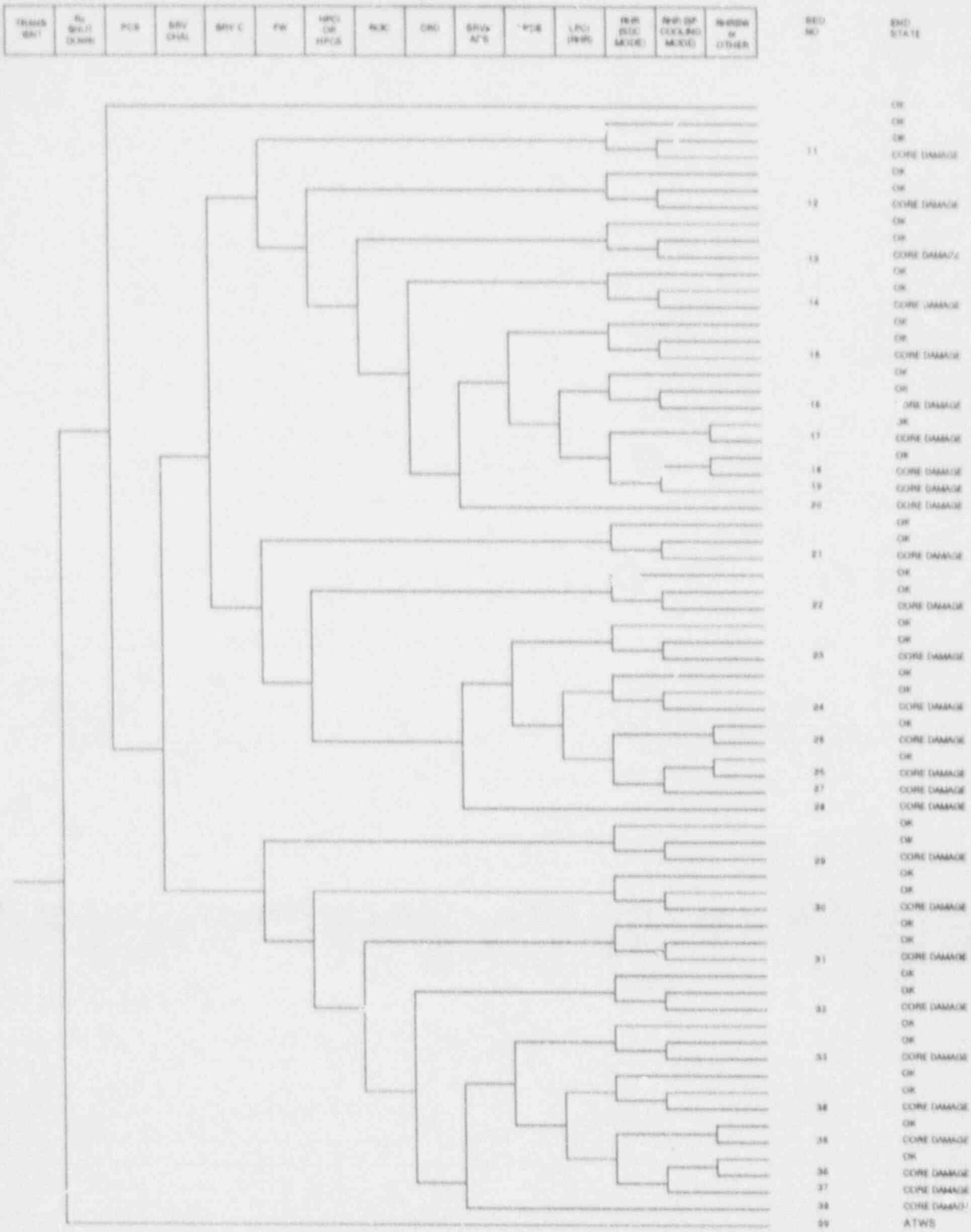

Figure A.22. BWR class C nonspecific reactor trip

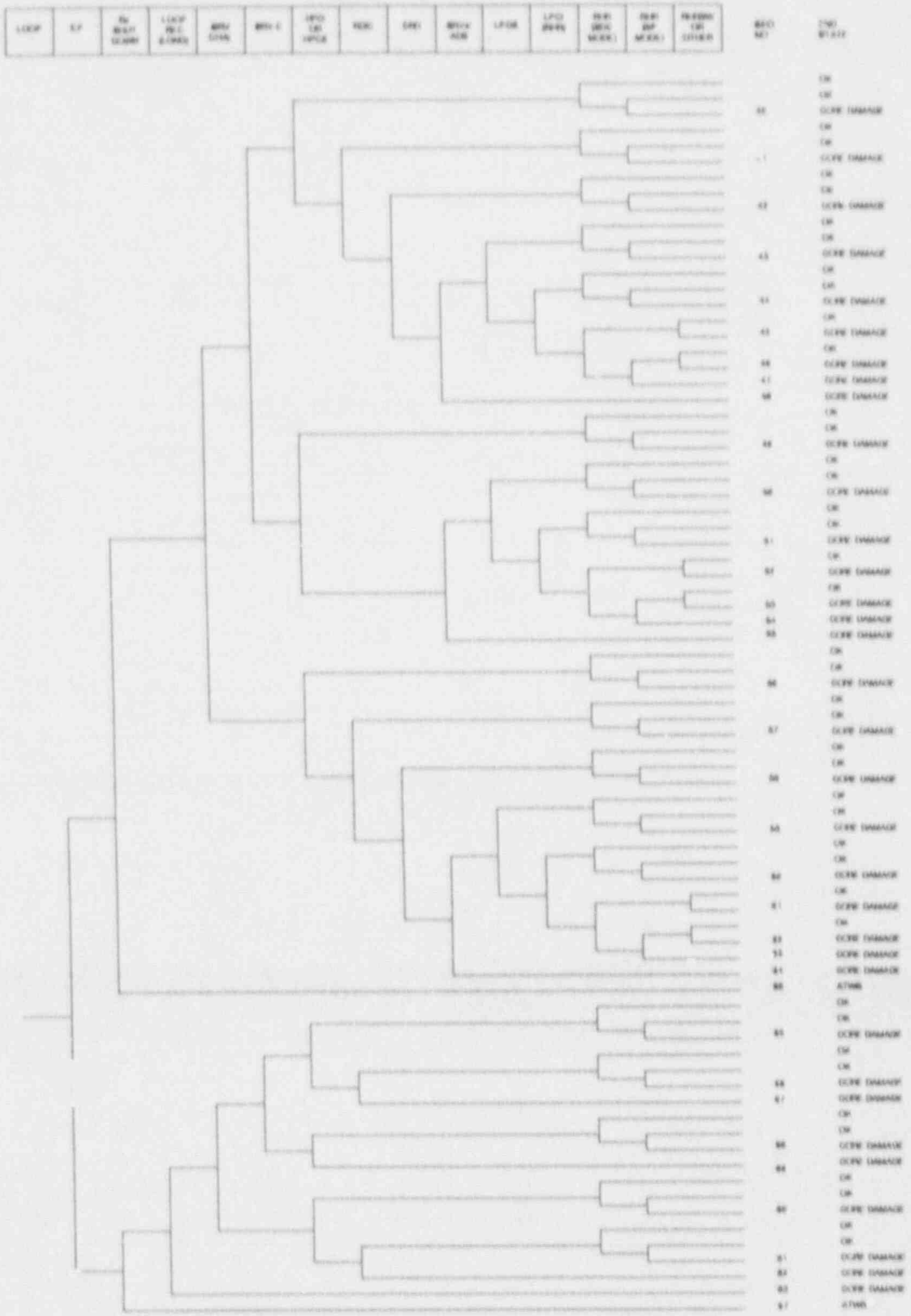

Figure A.23. BWR class C loss of offsite power

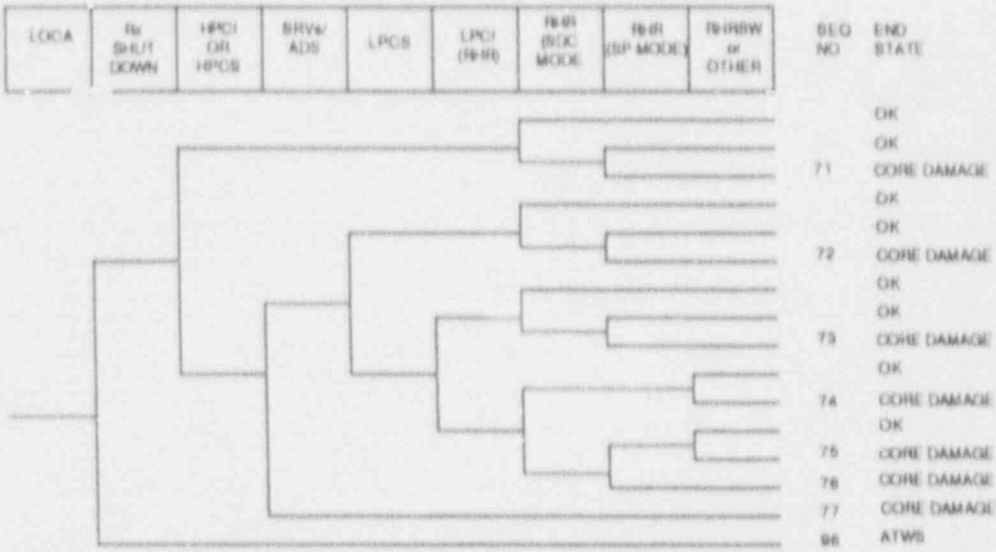

Figure A.24. BWR class C small-break loss-of-coolant accident

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11. ABSTRACT (200 words or less)

Twenty-eight operational events with conditional probabilities of core damage of 1.0 X 10^{-6} or higher occurring at commercial light-water reactors during 1991 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 the 1969-1981 and 1984-1990 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.

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