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Vol. 5

Precursors to Potential Severe Core Damage Accidents: 1986 A Status Report

Main Report and
Appendixes A, B, and C

Prepared by J. W. Minarick, J. D. Harris, P. N. Austin,
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Prepared for
U.S. Nuclear Regulatory
Commission

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NOTE

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FOREWORD

The Accident Sequence Precursor (ASP) Program was established at the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory in summer 1979. The first major report of that program was formally published in June 1982 and received extensive review. Since then three other reports documenting the review of operational events for precursors have been 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)
1985	*Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report* (NUREG/CR-4674, Vols. 1 and 2), December 1986

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 present document is a continuation, for 1986, of the assessment undertaken in the previous reports for operational events that occurred in 1969—1981, 1984 and 1985. A preliminary assessment of all precursors identified in 1984—1986 is also provided.

These models and analyses may in some instances be conservative, particularly regarding operator actions or recovery given certain events. A further review of the models and recovery actions is planned for subsequent ASP analyses of LERs.

As noted above, the ASP Program is the responsibility of NOAC. In addition to NOAC personnel (J. D. Harris and E. W. Hagen), personnel from two subcontractors, Science Applications International Corporation (J. W. Minarick and P. N. Austin) and Professional Analysis, Inc. (J. D. Cletcher), played a major role.

NOAC has designed and developed a number of major data bases that it operates and maintains for NRC. These data bases collect diverse types of information on nuclear power reactors from the construction phase through routine and off-normal operation. These data bases make extensive use of reactor-operator-submitted reports, such as the licensee event reports.

NOAC also publishes staff studies and bibliographies, disseminates monthly nuclear power plant operating event reports, and prepares the Technical Progress Review Journal *Nuclear Safety*.

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LIST OF ACRONYMS

ADS	automatic depressurization system
AFW	auxiliary feedwater
ATWS	anticipated transient without scram
BWR	boiling-water reactor
CC	component cooling
CRD	control rod drive
DHR	decay heat removal
FSAR	final safety analysis report
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HPI	high-pressure injection
IC	isolation condenser
LER	licensee event report
LOCA	loss-of-coolant accident
LOFW	loss of main feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LWR	light-water reactor
MFWP	main feedwater pump
MSIV	main steam isolation valve
PCS	power conversion system
PMG	permanent magnet generator
PORV	pilot- or power-operated relief valve
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RHR	residual heat removal
RV	relief valve or reactor vessel
SG	steam generator
SI	safety injection
SLB	steam-line break

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EXECUTIVE SUMMARY

The Accident Sequence Precursor Program reviews licensee event reports of operational events that have occurred at LWRs to identify and categorize precursors to potential severe core-damage accidents. Accident sequences considered in the study are those associated with inadequate core cooling. Accident sequence precursors are events that are important elements in such 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 with inadequate core cooling.

Originally proposed in the *Risk Assessment Review Group Report* (Lewis Committee report) in 1978, the study — subsequently named the Accident Sequence Precursor Program — was initiated at the Nuclear Operations Analysis Center in 1979. Earlier reports by the program involved assessment of events that occurred in 1969–1981 and 1984–1985. The present report involves the assessment of events that occurred during 1986.

A nuclear plant has safety systems for mitigating the consequences of accidents or off-normal initiating events that may occur during the course of plant operation. These systems are built to high-quality standards and are redundant; nonetheless, they have a nonzero probability of failing or being in a failed state when required to operate. This report uses LERs and other plant data, estimated system unavailabilities, the expected average frequency of initiating events (LOFWs, LOOPs, and LOCAs), and event details to evaluate the potential impact of the following two situations.

1. *Safety system unavailability*. Given an LER-reported failure of a safety system or partial failures in two or more systems, the report uses expected initiating event occurrence rates to determine the number of initiating events that may challenge the failed and backup systems during the period associated with the failure. It multiplies the expected challenges by system failure probabilities, using event trees, to evaluate the likelihood that the overall event sequence will occur.
2. *Initiating event occurrences*. Although standby safety systems are ideally always available, the probability exists that they may fail when called on to mitigate the consequences of expected accidents or transient-initiating events. Based on expected response of the safety systems, the report calculates the likelihood of potential severe core damage for precursors that included initiating events. Failed or degraded systems existing at the time of the initiating event are accounted for in the calculations.

All LERs are screened for accident sequence precursors and selected for detailed review if they included a reactor trip or more serious initiator, included two or more component failures or unavailabilities, or described an event that proceeded differently than expected. All LERs selected for detailed review are subjected to an in-depth evaluation, including

- a review of the accident sequence (if there was one) as described in the LER,
- a review of the design of systems in the reactor plant reporting the LER to determine the impact of the failure on the operation of these systems, and
- a review of the plant accident analyses to determine the extent to which affected systems would be required to function for different off-normal and accident conditions.

Based on this detailed review, events were selected as precursors if they met one of the following requirements:

- involved the failure of at least one system required to mitigate the consequences of a LOFW, LOOP, small-break LOCA, or steam-line break;
- involved the degradation of more than one system required to mitigate the effects of one of the above initiating events; or
- involved an actual initiating event that required safety system response.

Because LOFWs occur frequently within the reactor population, they are documented as precursors only if other failures also occurred. (Representative calculations of the significance of LOFWs without additional failures, however, are performed.)

Initiating event-frequency and system-failure probability estimates are used, in conjunction with precursor event trees, to estimate a conditional probability of potential severe core damage associated with each precursor. This probability is an estimate of the chance of potential severe core damage (inadequate core cooling), given that the precursor event occurred in the manner it did, and can be considered a measure of the residual protection against severe core damage available during the event.

The conditional probabilities associated with each precursor are used to rank precursors as to significance and to identify dominant sequences among all postulated sequences to potential severe core damage for the more highly ranked events.

Approximately 2900 LERs from 1986 were screened for precursors. Thirty-four precursors were identified for 1986, approximately the same number per reactor year (0.4) as identified in 1969—1981 and somewhat fewer than identified in 1984—1986 (0.6 per reactor year). Six events with conditional core-damage probabilities of $>10^{-4}$ were observed in 1986, compared with 10 in 1985 and 17 in 1984. The two most significant events involved small-break LOCA initiators.

Initiating-event frequencies and branch-failure probabilities utilized in the 1986 calculations were based on failures identified in the 1984—1986 period in the Accident Sequence Precursor Program. An overall reduction in estimated initiator frequencies and failure probabilities compared with those estimated in 1969—1979 was observed for PWRs and to a lesser extent for BWRs.

Likely core-damage accident sequences associated with the more important 1986 precursors were generally consistent with sequences associated with 1984—1985 events.

As part of the current effort, more serious precursors observed in 1984—1986 were qualitatively compared with those observed in 1969—1981. Based on this comparison, the more serious events currently being identified appear more consistent with events typically modeled in probabilistic risk assessments than was the case in 1969—1981. Complicated events involving electric power and instrumentation and control interactions were not seen nearly to the extent they previously were. Performance of the PWR AFW systems and the BWR combined high-pressure coolant injection and reactor-core-isolation cooling systems appears improved compared with 1969—1981, and they both exhibit failure probabilities consistent with PRA models.

The estimates developed in this report are subject to considerable uncertainty because of the limited data available, the assumptions that had to be made, and the analysis approach itself.

An overview of the methodology is provided in Chap. 2 of Vols. 1 and 3 of this document; in Chapter 5 they also address program limitations and sources of error. Chapter 4 of this volume provides a more comprehensive discussion of results for 1986 precursors, plus an initial assessment of 1984—1986 precursors compared with those observed in 1969—1981.

PRECURSORS TO POTENTIAL SEVERE CORE-DAMAGE ACCIDENTS: 1986, A STATUS REPORT

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ABSTRACT

Thirty-four operational events, reported in licensee event reports and occurring at commercial LWRs during 1986, 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—1985 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 initial conclusions from the assessment of 1986 events and from the collective assessment of 1984—1986 events.

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1. INTRODUCTION

The Accident Sequence Precursor Program involves the review of licensee event reports (LERs) on operational events that have occurred at 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.[3,4] This report details the work of the Accident Sequence Precursor Program in its review and evaluation of operational events that occurred in 1986 and were reported in LERs. The requirements for LERs are described in NUREG-1022, *Licensee Event Report System, Description of System and Guidelines for Reporting*,[5] as well as in the supplements to NUREG-1022 (Refs. 6, 7).

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1.1 Background

The Accident Sequence Precursor Program owes its genesis to the Risk Assessment Review Group,[8] which concluded that "unidentified event sequences significant to risk might contribute . . . a small increment . . . [to the overall risk]."[8] 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. 9). 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 completed, would have resulted in inadequate core cooling in the short term (typically up to 20–30 min) 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 a screening process and significance quantification methodology similar to that used for 1984–1985 events.[3] Discussed in more detail in Chap. 2 of Refs. 3 and 4, this methodology permits a reasonable quantification of the significance of an event 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.

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. However, a determined effort has been made in this program to address these problems. 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 1986 LERs for accident sequence precursors
Appendix D	Identification, description, and categorization of events considered to be accident sequence precursors
Chap. 3	Quantification of precursor significance
Chap. 4	Discussion of results

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

Because of its similarity with the 1984 and 1985 efforts, this report is somewhat abbreviated compared with those reports. In particular, Ref. 3 contains additional detail concerning the event-tree and branch-probability models employed in the analysis, and Refs. 3 and 4 provide a more detailed overview of the accident sequence precursor methodology, potential sources of error, and program limitations. However, a preliminary analysis of events selected as precursors during the 1984–1986 period is provided herein. References 3 and 4 only address results for individual years.

1.3 References

1. J. W. Minarick and C. A. Kukielka, *Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report*, NUREG/CR-2497 Vols. 1 and 2 (ORNL/NSIC-182/V1 and V2), Union Carbide Corp., Nuclear Div., Oak Ridge Natl. Lab., June 1982.

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

3. J. W. Minarick, J. D. Harris, P. N. Austin, E. W. Hagen, and J. W. Cletcher, *Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report*, NUREG-4674, Vols. 1 and 2 (ORNL/NOAC-4674/V1 and V2), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1986.

4. J. W. Minarick, J. D. Harris, P. N. Austin, E. W. Hagen, and J. W. Cletcher, *Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report*, NUREG-4674, Vols. 3 and 4 (ORNL/NOAC-4674/V3 and V4), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., May 1987.

5. *Licensee Event Report System, Description of System and Guidelines for Reporting*, NUREG-1022, U.S. Nuclear Regulatory Commission, September 1983.

6. *Licensee Event Report System, Description of System and Guidelines for Reporting*, NUREG-1022, Supplement 1, U.S. Nuclear Regulatory Commission, February 1984.

7. *Licensee Event Report System, Evaluation of First Year Results, and Recommendations for Improvements*, NUREG-1022, Supplement 2, U.S. Nuclear Regulatory Commission, September 1985.

8. *Risk Assessment Review Group Report*, NUREG/CR-0400, U.S. Nuclear Regulatory Commission, September 1978.

9. *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400 (NUREG-75/014), U.S. Nuclear Regulatory Commission, October 1975.

2. SELECTION OF 1986 OPERATIONAL EVENTS AS ACCIDENT SEQUENCE PRECURSORS

The identification of precursors within the licensee event report (LER) data base involved a two-step process. First, all 1986 LERs, including supplemental information, were reviewed to determine if the reported event should be reviewed 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 did not appear important. Events selected for detailed review included:

- core-damage initiators (including LOFWs, LOOPs, and small-break LOCAs);
- all events in which reactor trip was demanded;
- 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 occur;
- any event or operating condition that is not predicted or proceeds 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.

Over 2800 LERs were examined, and 1320 LERs (46%) from 1986 were selected for detailed review.

These operational events were reviewed 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 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 given a postulated initiating event during the failure period.

3. If the event or failure occurred while the plant was not at power, then the event was 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 during shutdown conditions, it was not selected as a precursor.

Thus, for each actual occurrence or postulated initiating event associated with an LER event, the sequence of operation of various mitigating systems required to prevent severe 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:

- a core-damage initiator [such as a LOOP, steam-like 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; or
- degradation in more than one system required to mitigate the consequences of a core-damage initiator.

Of the 1320 LERs selected for detailed review, 34 operational events were selected as accident sequence precursors:

- LOOP, small-break LOCA, and small SLB initiators (8 events);
- LOFW initiators with failures in systems required for LOFW mitigation (2 events);
- failures of redundant systems required to mitigate postulated core-damage initiators (18 events);
- degradation in multiple systems required to mitigate postulated core-damage initiators (2 events); and
- reactor trips with failures of redundant systems required to mitigate core damage following a reactor trip (4 events).

The review process is summarized in Fig. 2.1. Individual failures of BWR high-pressure-coolant injection, HPCS, and RCIC systems and total LOFW events without additional mitigating system failures were identified during the detailed review (84 events) but not selected as precursors. The impact of such events was determined on a plant-class basis. The impact of a nonspecific reactor trip without additional failures was also determined on a plant-class basis, to provide an estimate of core-damage likelihood for a typical trip, based on the event sequence models employed in the study. The results of these evaluations are provided in Chap. 3.

All reactor-trip events were reviewed as a part of the 1986 LER review. These are listed in Appendix F by LER number. Although only those involving core-damage initiators, total system failures, and multiple degraded systems were individually analyzed, the remaining reactor-trip events (particularly those involving partial LOFW and single degraded systems) provide important information on the likelihood of system/component failure following a true demand and are listed for reference. Unavailabilities of decay-heat-removal (DHR) systems while those systems were in use (typically short-term trips of the operating

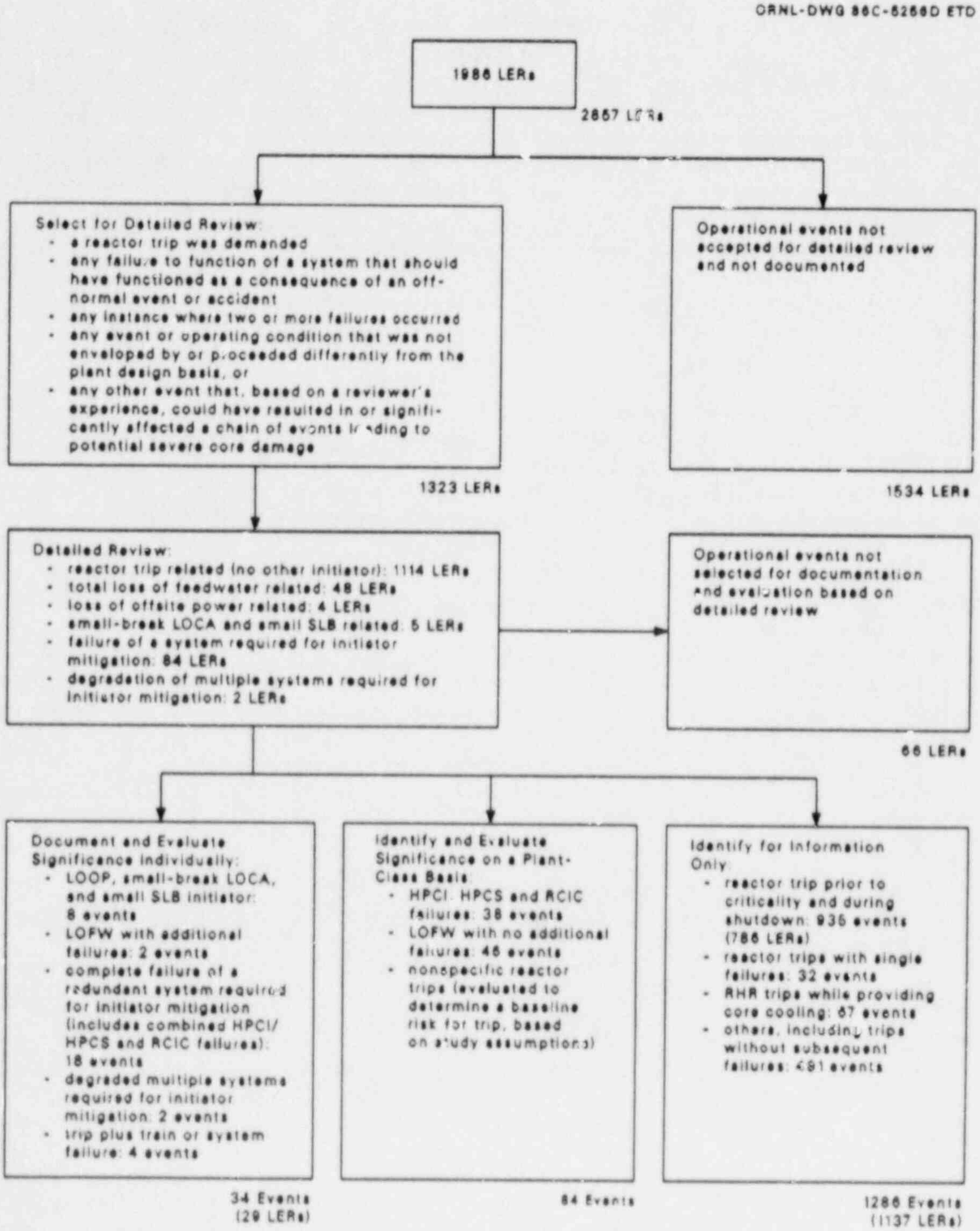

Fig. 2.1. Operational review process for 1986.

train or trains) were also identified during the detailed review process but were not, in themselves, considered precursors. However, failures of the DHR system in test or on demand would have been considered precursors. Although the Accident Sequence Precursor Program is concerned with initiators and failures of core-damage mitigating systems that occurred or could have occurred at power, the DHR system unavailabilities were listed for information (see Appendix F).

Two potential sources of error in selecting events for detailed review from the LER data base must be recognized.

1. *Inherent biases in the selection process.* Although the criteria for identification of an operational event as a precursor, once the event is selected in the initial review, are fairly well defined, the selection of an LER for 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.

2. *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),[1] 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 to the study must often be inferred.

2.1 Documentation of Events Selected as Accident Sequence Precursors

For each of the precursors, two items were prepared. The first, Precursor Description and Analysis Sheet (Fig. 2.2), briefly describes the event sequence, identifies the corrective action taken after the event, provides selected plant and event data, and documents the impact of the event on event sequence models used in the study.

The second item, Conditional Core-Damage Calculations (Fig. 2.3), documents the calculations performed to estimate the conditional core-damage probability associated with the precursor and includes probability summaries for each end state, the dominant sequence associated with each end state, the conditional probability for the more important sequences,* and the branch probabilities used. The results of the conditional probability calculations are described in Chap. 3.

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*Sequences with a conditional probability equal to at least 0.03 of the dominant sequence associated with the end state.

ORNL-DWG 87-3686 ETD

PRECURSOR DESCRIPTION SHEET

LER No.:
Event Description:
Date of Event:
Plant:

EVENT DESCRIPTION

Sequence

Corrective Action

Plant/Event Data

Systems Involved:

Components and Failure Modes Involved:

Component Unavailability Duration:
Plant Operating Mode:
Discovery Method:
Reactor Age:
Plant Type:

Comments

Event Identifier:

Fig. 2.2. Precursor description and analysis sheet.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Branches Impacted and Branch Nonrecovery Estimate

Plant Models Utilized

Event Identifier:

Fig. 2.2 (continued)

ORNL-DWG 87-3687 ETD

CONDITIONAL CORE DAMAGE CALCULATIONS

LER Number:
Event Description:
Event Date:
Plant:

UNAVAILABILITY, DURATION=

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

SEQUENCE CONDITIONAL PROBABILITY SUMS

DOMINANT SEQUENCES

SEQUENCE CONDITIONAL PROBABILITIES

Note:

Conditional probability values are differential values which reflect the added risk due to observed failures.
Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

MODEL:
DATA:

BRANCH FREQUENCIES/PROBABILITIES

Event Identifier:

Fig. 2.3. Conditional core-damage calculations.

The Precursor Description and Analysis sheets and Conditional Core-Damage Calculations are included in Appendix D. The LERs associated with each precursor are included in Appendix E. Appendix F contains various listings of the events identified in the review process but not selected and documented as precursors, such as the DHR system unavailabilities described above. Appendixes D, E, and F are bound separately.

2.2 Tabulation of Selected Events

The 1986 events selected as precursors to potential severe core-damage accidents are listed in Table 2.1 at the end of this section. The precursor events 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 (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 NAME);
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 (CD PROB) as well as the sum of CD PROB and conditional probability of core vulnerability associated with the events (SUM PROB) (defined in Chap. 3);
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 perspectives.

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	Plant type and vendor
2.10	Architect-engineer
2.11	Operating utility

Abbreviations used in each table (Tables 2.1—2.11) are defined in Table 2.12. The information in the above tables is sorted by conditional probability in Chap. 3.

Table 2.1. Precursors listed by docket and LER number

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEI	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPII	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPII	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPII	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEI	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPII	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEI	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
366/86-025	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EI	FPL	06/02/83	UNAVL
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEII	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL

Table 2.2. Precursors listed by plant name and LER number

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	D	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEII	E	D	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	D	N	12.9	4.1E-5	4.2E-5	478	[illegible]	C	GH	OPP	08/06/73	TRIP
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	D	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	D	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	D	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEI	E	D	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPII	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPII	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPII	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEI	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	D	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEI	G	D	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	D	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	D	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	D	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EI	FPL	06/02/83	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	D	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPII	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	D	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL

Table 2.3. Precursors listed sequentially by plant event date

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EF	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEI	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEI	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEII	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWB: 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EI	FPL	06/02/83	UNAVL
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPII	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEI	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPII	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPII	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPII	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
250/86-039	12/27/36	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL

Table 2.4. Precursors listed by initiator or transient

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	DPR	CRITICAL	TRANS
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	D	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	D	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	D	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	D	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	D	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	D	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	D	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	D	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	D	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	D	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	D	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	D	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	D	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EX	FPL	06/02/83	UNAVL
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	D	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL

Table 2.5. Precursors listed by plant system

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEI	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEI	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEI	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	K	12.9	4.1E-5	4.2E-5	[illegible]	P	C	GH	OPP	08/06/73	TRIP
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	[illegible]/08/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EI	FPL	06/02/83	UNAVL
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	D	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	D	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	D	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	D	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEII	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPII	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPII	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPII	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPII	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	D	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL

Table 2.6. Precursors listed by component

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABIL[illegible]	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EI	FPL	06/02/83	UNAVL
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UX	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UX	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UX	DPC	09/05/74	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UX	DPC	04/19/73	TRIP
277/86-003	01/24/86	DC TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BX	PEC	09/16/73	TRIP
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BX	WMP	05/30/72	MSLB
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BX	FPL	10/20/72	TRIP
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HC	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BX	BGE	11/30/76	TRIP
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL

Table 2.7. Precursors listed by plant operating status

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
250/86-03[illegible]	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	B	UI	CPL	09/20/70	LOOP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EI	FPL	06/02/83	UNAVL
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL

Table 2.8. Precursors listed by discovery method

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEI	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPII	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEI	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXI	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPII	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPII	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPII	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEI	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPII	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EI	FPL	06/02/83	UNAVL
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB

Table 2.9. Precursors listed by plant type and vendor

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	D	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOT*2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BX	PEC	09/16/73	TRIP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	D	N	14.4	7.7E-6	7.7E-6	655	B	G	BX	BEC	06/16/72	LOOP
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	D	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UX	DPC	04/19/73	TRIP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UX	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UX	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UX	DPC	09/05/74	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BX	BGE	11/30/76	TRIP
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BX	SCE	08/29/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EX	FPL	06/02/83	UNAVL
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BX	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BX	FPL	10/20/72	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UX	CPL	09/20/70	LOOP
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	D	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/17/74	UNAVL
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	D	N	14.3	4.8E-7	4.8E-7	497	P	W	BX	WMP	05/30/72	MSLB
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB

Table 2.10. Precursors listed by architect-engineer

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-036	11/06/06	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BI	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BI	FPL	10/20/72	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BI	FPL	10/20/72	TRIP
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BI	PEC	09/16/73	TRIP
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BI	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BI	NSP	12/17/74	UNAVL
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	D	N	14.4	7.7E-6	7.7E-6	655	B	G	BI	BEC	06/16/72	LOOP
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	D	N	14.3	4.8E-7	4.8E-7	497	P	W	BI	WMP	05/30/72	MSLB
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BI	BGE	11/30/76	TRIP
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BI	SCE	08/29/83	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EX	FPL	06/02/83	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	D	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	D	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
366/66-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	D	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	D	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UI	CPL	09/20/70	LOOP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UI	DPC	04/19/73	TRIP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UI	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UI	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UI	DPC	09/05/74	UNAVL

Table 2.11. Precursors listed by operating utility

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BX	BEC	06/16/72	LOOP
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BX	BGE	11/30/76	TRIP
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UX	CPL	09/20/70	LOOP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UX	DPC	04/19/73	TRIP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UX	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UX	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UX	DPC	09/05/74	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	D	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BX	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	D	N	14.1	1.4E-3	2.1E-3	693	P	W	BX	FPL	10/20/72	TRIP
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EX	FPL	06/02/83	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	D	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/17/74	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BX	PEC	09/16/73	TRIP
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	D	N	3.0	2.6E-7	8.9E-7	1080	P	C	BX	SCE	08/29/83	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	D	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	798	P	W	SW	VEP	03/07/73	UNAVL
301/86-004	09/20/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	O	N	14.3	4.0E-7	4.8E-7	497	P	W	BX	WMP	05/30/72	MSLB

Table 2.12. Abbreviations used in precursor lists

LER NO: DOCKET NUMBER/LICENSEE EVENT REPORT NUMBER
E DATE: EVENT DATE
DESCRIPTION: DESCRIPTION OF EVENT
PLANT NAME: 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 COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS
CJ	OTHER COOLANT SUBSYSTEMS AND THEIR CONTROLS

ENGINEERED SAFETY FEATURES

SA	REACTOR CONTAINMENT SYSTEMS
SB	CONTAINMENT HEAT REMOVAL SYSTEMS AND CONTROLS
SC	CONTAINMENT AIR PURIFICATION AND CLEANUP SYSTEMS AND CONTROLS
SD	CONTAINMENT ISOLATION SYSTEMS AND CONTROLS
SE	CONTAINMENT COMBUSTIBLE GAS CONTROL SYSTEMS AND CONTROLS
SF	EMERGENCY CORE COOLING SYSTEMS AND CONTROLS
SG	CONTROL ROOM HABITABILITY SYSTEMS AND CONTROLS
SH	OTHER ENGINEERED SAFETY FEATURE SYSTEMS AND THEIR CONTROLS

INSTRUMENTATION AND CONTROLS

IA	REACTOR TRIP SYSTEMS
IB	ENGINEERING SAFETY FEATURE INSTRUMENT SYSTEMS
IC	SYSTEMS REQUIRED FOR SAFE SHUTDOWN
ID	SAFETY RELATED DISPLAY INSTRUMENTATION
IE	OTHER INSTRUMENT SYSTEMS REQUIRED FOR SAFETY
IF	OTHER INSTRUMENT SYSTEMS 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 (COMPOSITE AC AND DC)
EE	EMERGENCY GENERATOR SYSTEMS AND CONTROLS
EF	EMERGENCY LIGHTING SYSTEMS AND CONTROLS
EG	OTHER ELECTRIC 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

AUXILIARY PROCESS SYSTEMS

PA	COMPRESSED AIR SYSTEMS AND CONTROLS
PB	PROCESS SAMPLING SYSTEMS
PC	CHEMICAL, VOLUME CONTROL AND LIQUID POISON SYSTEMS AND CONTROLS
PD	FAILED FUEL DETECTION SYSTEMS
PE	OTHER AUXILIARY PROCESS SYSTEMS AND CONTROLS

Table 2.12 (continued)

OTHER AUXILIARY SYSTEMS

AA	AIR CONDITIONING, HEATING, COOLING AND VENTILATION SYSTEMS AND CONTROLS
AB	FIRE PROTECTION SYSTEMS AND CONTROLS
AC	COMMUNICATION SYSTEMS
AD	OTHER AUXILIARY SYSTEMS AND THEIR CONTROLS

STEAM AND POWER CONVERSION SYSTEMS

HA	TURBINE-GENERATORS AND CONTROLS
HB	MAIN STEAM SUPPLY SYSTEM 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 CLEAN-UP SYSTEMS AND CONTROLS
HH	CONDENSATE AND FEEDWATER SYSTEMS AND CONTROLS (OTHER THAN CH)
HI	STEAM GENERATOR BLOWDOWN SYSTEMS AND CONTROLS
HJ	OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEMS (NOT INCLUDED ELSEWHERE)

RADIOACTIVE WASTE MANAGEMENT SYSTEMS

MA	LIQUID RADIOACTIVE WASTE MANAGEMENT SYSTEMS
MB	GASEOUS RADIOACTIVE WASTE MANAGEMENT SYSTEMS
MC	PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS
MD	SOLID RADIOACTIVE WASTE MANAGEMENT SYSTEMS

RADIATION PROTECTION SYSTEMS

BA	AREA MONITORING SYSTEMS
BB	AIRBORNE RADIOACTIVITY MONITORING SYSTEMS
XX	OTHER SYSTEMS
ZZ	SYSTEM CODE NOT APPLICABLE

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 (CRDRVE)	
AIR DRYERS (AIRDRY)		DEMINERALIZERS (DEMINX)	ION EXCHANGERS
ANNUNCIATOR MODULES (ANNUNC)	ALARMS BUZZERS CLAXONS HORNS GONGS SIRENS	ELECTRICAL CONDUCTORS (ELECON)	BUS CABLE WIRE
		ENGINES, INTERNAL COMBUSTION (ENGINE)	DIESEL ENGINES GASOLINE ENGINES NATURAL GAS ENGINES PROPANE ENGINES STRAINERS SCREENS
		FILTERS (FILTER)	
BATTERIES AND CHARGERS (BATTRY)	CHARGERS DRY CELL WET CELLS STORAGE CELLS	FUEL ELEMENTS (FUELXX)	
		GENERATORS (GENERA)	INVERTERS
BLOWERS (BLOWER)	COMPRESSORS GAS CIRCULATORS FANS VENTILATORS	HEATERS, ELECTRIC (HEATER)	HEAT TRACERS

Table 2.12 (continued)

CIRCUIT CLOSERS/ INTERRUPTERS (CKTBRK)	CIRCUIT BREAKERS CONTRACTORS 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 DIFFERENTIALS INTEGRATORS (TOTALIZERS) POWER SUPPLIES RECORDERS SWITCHES TRANSMITTERS COMPUTATION MODULES	RELAYS (RELAYX)	SWITCHGEAR
		SHOCK SUPPRESSORS AND SUPPORT (SUPORT)	HANGERS SUPPORTS SWAY BRACES/STABILIZERS SNUBBERS ANTI-VIBRATION DEVICES
		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 CONTAIN. (PENETR)	AIR LOCKS PERSONNEL ACCESS FUEL HANDLING EQUIPMENT ACCESS ELECTRICAL INSTRUMENT LINE PROCESS PIPING	VESSELS, PRESSURE (VESSEL)	CONTAINMENT VESSELS DRYWELLS PRESSURE SUPPRESSION PRESSURIZERS REACTOR VESSELS
PIPES, FITTINGS (PIPEXX)		OTHER COMPONENTS (XXXXXX)	
PUMPS (PUMPXX)		CODES NOT APPLICABLE (ZZZZZZ)	
RECOMBINERS (RECOMB)			

O: PLANT OPERATING STATUS:

CODE	STATUS
A	(UNDER) CONSTRUCTION
B	PREOPERATIONAL, STARTUP OR POWER ASCENSION TESTS (IN PROGRESS)
C	ROUTINE STARTUP OPERATIONS
D	ROUTINE SHUTDOWN OPERATIONS
E	STEADY STATE OPERATION
F	LOAD CHANGES DURING ROUTINE POWER OPERATION
G	SHUTDOWN (HOT OR COLD) EXCEPT FOR REFUELING
H	REFUELING
U	UNKNOWN
X	OTHER (INCLUDING SPECIAL TESTS, EMERGENCY SHUTDOWN OPERATIONS, ETC.)
Z	ITEM NOT APPLICABLE

D: DISCOVERY METHOD (O-OPERATIONAL EVENT, T-TESTING, M-MAINTENANCE)
E: HUMAN ERROR INVOLVED (N-NO, Y-YES)
AGE: PLANT AGE AT THE TIME OF THE EVENT IN YEARS
CD PROB: CONDITIONAL CORE DAMAGE PROBABILITY
SUM PROB: CONDITIONAL PROBABILITY OF CORE DAMAGE AND CORE VULNERABILITY
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

Table 2.12 (continued)

AE: PLANT ARCHITECT ENGINEER

AE-AMERICAN ELECTRIC POWER	GH-GIBBS AND HILL	SS-SOUTHERN SERVICES
BR-BURNS AND ROE	GX-GILBERT	SW-STONE AND WEBSTER
BX-BECHTEL	PX-PIONEER	UE-UNITED ENGINEERS
DPC-DUKE POWER COMPANY	RT-BROWN AND ROOT	UX-UTILITY
EX-EBASCO	SL-SARGENT AND LUNDY	XX-OTHER
FP-FLOUR POWER		

OPR: PLANT LICENSEE ABBREVIATIONS

LICENSEE ABBREV.	LICENSEE	LICENSEE ABBREV.	LICENSEE
APC	ALABAMA POWER COMPANY	NMP	NIAGRA MOHAWK POWER CORPORATION
APL	ARKANSAS POWER AND LIGHT COMPANY	NNE	NORTHEAST NUCLEAR ENERGY COMPANY
APS	ARIZONA PUBLIC SERVICE COMPANY	NPP	NEBRASKA PUBLIC POWER DISTRICT
BEC	BOSTON ELECTRIC COMPANY	NSP	NORTHERN STATES POWER COMPANY
BGE	BALTIMORE GAS AND ELECTRI COMPANY	PEC	PHILADELPHIA ELECTRIC COMPANY
CEC	CONSOLIDATED EDISON COMPANY	PEG	PUBLIC SERVICE ELECTRIC AND GAS COMPANY
CEI	CLEVELAND ELECTRIC ILLUMINATING COMPANY	PEP	POTOMAC ELECTRIC POWER COMPANY
CGE	CINCINNATI GAS AND ELECTRIC COMPANY	PGC	PORTLAND GENERAL ELECTRIC COMPANY
CPC	CONSUMERS POWER COMPANY	PGE	PACIFIC GAS AND ELECTRIC COMPANY
CPL	CAROLINA POWER AND LIGHT COMPANY	PNY	POWER AUTHORITY OF THE STATE OF NEW YORK
CWE	COMMONWEALTH EDISON COMPANY	PPL	PENNSYLVANIA POWER AND LIGHT COMPANY
CYA	CONNECTICUT YANKEE ATOMIC POWER COMPANY	PSC	PUBLIC SERVICE COMPANY OF COLORADO
DPL	DAIRYLAND POWER COOPERATIVE	PSI	PUBLIC SERVICE OF INDIANA
DLC	DUQUENSE LIGHT COMPANY	PSN	PUBLIC SERVICE OF NEW HAMPSHIRE
DPC	DUKE POWER COMPANY	PSO	PUBLIC SERVICE COMPANY OF OKLAHOMA
DPP	OMAHA POWER COMPANY	PSP	PUGET SOUND POWER AND LIGHT COMPANY
FPC	FLORIDA POWER CORPORATION	RGE	ROCHESTER GAS AND ELECTRIC CORPORATION
FPL	FLORIDA POWER AND LIGHT COMPANY	SCC	SOUTH CAROLINA ELECTRI AND GAS COMPANY
GPC	GEORGIA POWER COMPANY	SCE	SOUTHERN CALIFORNIA EDISON COMPANY
GSU	GULF STATES UTILITIES	SMU	SACRAMENTO MUNICIPAL UTILITIES DISTRICT
HLP	HOUSTON LIGHTING AND POWER COMPANY	TEC	TOLEDO EDISON COMPANY
IEL	IOWA ELECTRIC LIGHT AND POWER COMPANY	TUG	TEXAS UTILITIES GENERATING COMPANY
IME	INDIANA AND MICHIGAN ELECTRIC COMPANY	TVA	TENNESSEE VALLEY AUTHORITY
IPC	ILLINOIS POWER COMPANY	UEC	UNION ELECTRIC COMPANY
JCP	JERSEY CENTRAL POWER AND LIGHT COMPANY	VEP	VIRGINIA ELECTRIC AND POWER COMPANY
KGE	KANSAS GAS AND ELECTRIC COMPANY	VYC	VERMONT YANKEE NUCLEAR POWER CORPORATION
LIL	LONG ISLAND LIGHTING COMPANY	WEP	WISCONSIN ELECTRIC POWER COMPANY
LPL	LOUSIANA POWER AND LIGHT COMPANY	WMP	WISCONSIN-MICHIGAN POWER COMPANY
MEC	METROPOLITAN EDISON COMPANY	WPP	WASHINGTON PUBLIC POWER SUPPLY SYSTEM
MPL	MISSISSIPPI POWER AND LIGHT COMPANY	WPS	WISCONSIN PUBLIC SERVICE CORPORATION
MYA	MAINE YANKEE ATOMIC POWER COMPANY	YAC	YANKEE ATOMIC ELECTRIC COMPANY
NIC	NORTHERN INDIANA PUBLIC SERVICE COMPANY		

CRITICAL: PLANT CRITICALITY DATE

TRANS: EVENT INITIATOR OR UNAVAILABILITY

ECIT - EXCESSIVE COOLANT INVENTORY
EQUK - EARTHQUAKE
INAA - INADVERTANT ADS ACTUATION
LOFW - LOSS OF FEEDWATER
LOOP - LOSS OF OFFSITE POWER
LOCA - LOSS OF COOLANT ACCIDENT
LRTR - LOCKED ROTOR ACCIDENT
MSLB - MAIN STEAM LINE BREAK
SGTR - STEAM GENERATOR TUBE RUPTURE
TRIP - REACTOR TRIP
UNAVAIL - SYSTEM(S) UNAVAILABLE
UNIQ - A UNIQUE SEQUENCE

2.3 Reference

1. 10 CFR Pt. 50.73.

3. QUANTIFICATION OF PRECURSORS

Operational events selected as 1986 precursors were quantified for ranking purposes. This quantification involved determination of a conditional probability of subsequent severe core damage given the failures observed during the event. The calculation assumed that the failure probabilities for systems observed failed during the event were equal to the likelihood of failing to recover from the failure or fault that actually occurred. Failure probabilities used for systems observed degraded during an operational event were assumed equal to the conditional probability that the system would fail (given that is was observed degraded) and the probability that is would not be recovered within a ~30-min period. The failure probability associated with observed successes and with systems unchallenged during the actual occurrence was assumed equal to a failure probability determined, based on either available system failure data or based on system success criteria and typical train and common-mode-failure probabilities. The conditional core-damage probability is useful in ranking because it permits estimation of the measure of protection remaining once the failures have occurred.

The likelihood of recovery associated with the event failure(s) was described using a process equivalent to that employed in the 1980—1981 and 1984—1985 event reviews.[1-3] This process considered each failure to be composed of an observed failure and a subsequent recovery step. Four recovery classes were used to describe the different types of recovery that could be involved. Events were assigned to a particular class based on an assessment of likelihood that recovery would *not* be effected in the required period of time, considering the event specifics. The assignment of an event to a recovery class and the numeric value assigned to each recovery class were based on engineering judgment, which considered whether such recovery would be required in a moderate- to high-stress situation following a postulated initiating event. The four recovery classes are described in Table 3.1.

3.1 Estimation of Initiating-Event Frequencies and Branch-Failure Probabilities

A set of initiating event frequencies and system failure probabilities was developed for application in the quantification of the event-tree models associated with the precursors. This set includes initiating-event frequencies and failure probabilities applicable to the branches of each event tree included in Appendix B, which were used to classify and quantify the majority of precursors. Frequencies and failure probabilities for unique initiators and other plant functions were also estimated, when required, using the same approach.

The approach used to develop frequency and probability estimates employed failure data in the precursors themselves, as was done in the 1980—1981 review.[3] When precursor data were available for a system or initiating event, its probability or frequency was estimated by counting the effective number of nonrecoverable failures in the observation

Table 3.1. Description and quantification of recovery classes

Recovery class	Description	Likelihood of failing to recover from event[a]
R1	Failure did not appear to be recoverable in required period, either from control room or at failed equipment	1.00[b]
R2	Failure appeared recoverable in required period at failed equipment, and equipment was accessible; recovery from control room did not appear possible	0.34
R3	Failure appeared recoverable in required period from control room, but recovery was not routine or involved substantial stress	0.12
R4	Failure appeared recoverable in required period from control room and was considered routine or procedurally based	0.04

[a]These values are used for consistency of analysis. 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 above.

[b]Note that a value of 0.58 was used in NUREG/CR-3591 (Ref. 3) in lieu of 1.00 to facilitate uncertainty analyses.

period, making appropriate demand assumptions, and then calculating the effective number of failures per demand or initiating events per reactor year as described in Sect. 3.2. of Ref. 3. For demand failure probabilities, the number of demands was calculated based on the estimated number of tests per reactor year plus any additional demands to which the function would be expected to respond. This estimate was then multiplied by the number of applicable reactor years in the observation period to determine the total number of demands. The observation period used was 1984—1986, and precursors identified during the period formed the basis for the estimates. This information was then used to tailor the component probabilities associated with the train-based system models used in the 1984—1985 event reviews[1,2] such that the overall system probability estimates were consistent with the failures observed in 1984—1986.

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.

Probability values employed in the precursor conditional probability calculations are developed in Appendix C. Probabilities applicable to each significance calculation are also listed at the end of each calculation in Appendix D. Average initiating event frequencies and system-failure probabilities developed from 1984—1986 precursor data are listed in Table 3.2. These values are compared with previous precursor data in Chap. 4.

3.2 Conditional Probabilities Associated with Each Precursor

Failure events identified in the detailed review of each precursor were mapped onto plant-class event trees (included in Appendix B and described in Ref. 1) to estimate a conditional probability of subsequent severe core damage for each precursor. This probability can be considered as a measure of residual protection remaining, given the failures observed in an event.

Each event tree includes three nondesired end states designated core damage (CD), in which inadequate core cooling is believed to exist for a period greater than ~30 min; core vulnerability (CV), in which core protection is believed to be provided but for which no specific analytic basis generally is available; and ATWS, for the failure-to-trip sequence. The end states are distinct; sequences associated with core damage and ATWS are not subsets of core vulnerability sequences. Except for the fact that detailed analysis information does not generally exist, core vulnerability sequences are expected to end in successful core cooling. The ATWS sequence, if fully developed, would consist of a number of sequences ending in either success, core vulnerability, or core damage.

Conditional probabilities for each end state associated with a precursor were calculated by applying appropriate failure probabilities to each event-tree branch and summing the resulting conditional sequence probabilities for the given end states.

Because the frequencies and failure probabilities used in these calculations are derived in part from data obtained across the LWR population, even though they are applied to sequences that are plant-class-specific in nature, the conditional probabilities determined for each precursor should not be directly associated with the probability of potential severe core damage resulting from the actual precursor event at the specific reactor plant at which it occurred. The probabilities calculated in this study are homogenized probabilities considered representative of probabilities resulting from the occurrence of the selected events at plants representative of the plant class.

3.2.1 Event Sequences Requiring Calculation

1. If an initiating event occurred as part of a precursor (i.e., the precursor consisted of an initiating event plus possible additional failures), then the conditional probability of potential severe core damage was calculated based on the event tree associated with the initiator.

Table 3.2. Average initiating event-frequency and branch-failure probability estimates developed from 1984—1986 precursors

Initiator/branch	Initial failure likelihood	Nonrecovery estimate	Total
	PWRs		
LOOP	4.1×10^{-2}/year	0.39	1.6×10^{-2}/year
Small 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 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}

2. If an initiating event did not occur as part of a precursor (i.e., the precursor consisted of an unavailability), then the conditional probability of potential severe core damage was calculated considering potential initiating events, their expected frequency, and the estimated or actual (if reported) duration of the unavailability. Only sequences associated with each potential initiator impacted by the precursor were included in the calculated probability.

3.2.2 Initiating Event Probability Determination

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

2. If an initiating event did not occur as part of a selected precursor, then the probability used for the initiating event was developed assuming a constant hazard rate. For the frequencies and durations

associated with most precursors, this value is approximately equal to the product of the estimated initiating event frequency and the time during which the precursor existed. As described previously, the initiating event frequency estimates include the potential for recovery. Event durations (the period of time during which the failure existed) were based on information included in each licensee event report (LER), if provided. If the event was discovered during testing, then one-half of the test period (15 d for a typical 30-d test interval) was assumed, unless specific failure durations were available.

3.2.3 Branch Probability Determination

1. For event-tree branches for which no failed or degraded condition existed, a probability equal to the branch-failure probability described previously was assigned.
2. For event-tree branches associated with a failed system, a probability equal to the numeric value associated with the recovery class was assigned. This permitted consideration of potential recovery for observed failures.
3. For event-tree branches that included a degraded system (i.e., a system that still met minimum operability requirements but with reduced or no redundancy), the estimated failure probability was modified to reflect the loss of redundancy. To estimate the system's conditional-failure probability under these conditions, train probabilities were modified to reflect the train failures or unavailabilities observed in the event. The calculational method employed recognized the change in system success criteria required given the observed failures or unavailabilities. For example, a system that required two of three trains to be operable for success was modeled as a "two-out-of-two" system if one train was observed failed. The calculations also addressed the difference between a failed train, which would imply a higher likelihood of failure for the next train because of common mode effects, and a train rendered unavailable because of surveillance testing or because of support system failures, which would imply a failure probability for the next train equal to the normally expected failure probability for the first train in the system.
4. Systems or trains rendered unavailable as a result of support system failures were modeled recognizing that, as long as the affected support system remained failed, all impacted systems (or trains) were unavailable; but if the support system were recovered, all the affected systems were recovered.

3.2.4 Event Calculation

Once the branch probabilities that reflected the conditions of the precursor event were established, the sequences leading to modeled end states (core vulnerability, core damage, and ATWS) were calculated and summed to produce an estimate of the conditional probability of each end state for the precursor.

3.2.3 Sample Calculations

Two hypothetical events are used to illustrate this calculational process. The first event assumes an LOFW but no other observed failures during mitigation. The (simplified) event tree for this event is shown in Fig. 3.1. This hypothetical precursor involved an initiating event that was assigned to recovery class R2 (the numeric value associated with this recovery class is 0.34). Systems assumed available were assigned failure probabilities developed as described previously. The estimated conditional probabilities for nondesirable end states associated with the event are then

(1) core vulnerability:

$$P_{CV} = P\,[\text{seq. } 3] = 0.34 \times (1 - 3.5 \times 10^{-5}) \times 3 \times 10^{-4} \times 0.04 \times (1 - 0.3)$$

$$= 2.9 \times 10^{-6}\ .$$

(2) core damage:

$$P_{CD} = P\,[\text{seq. } 4] = 0.34 \times (1 - 3.5 \times 10^{-5}) \times 3 \times 10^{-4} \times 0.04 \times 0.3$$

$$= 1.2 \times 10^{-6}\ .$$

(3) ATWS:

$$P_{ATWS} = P\,[\text{seq. } 5] = 0.34 \times 3.5 \times 10^{-5} = 1.2 \times 10^{-5}\ .$$

If more than one sequence were associated with an end state (as is

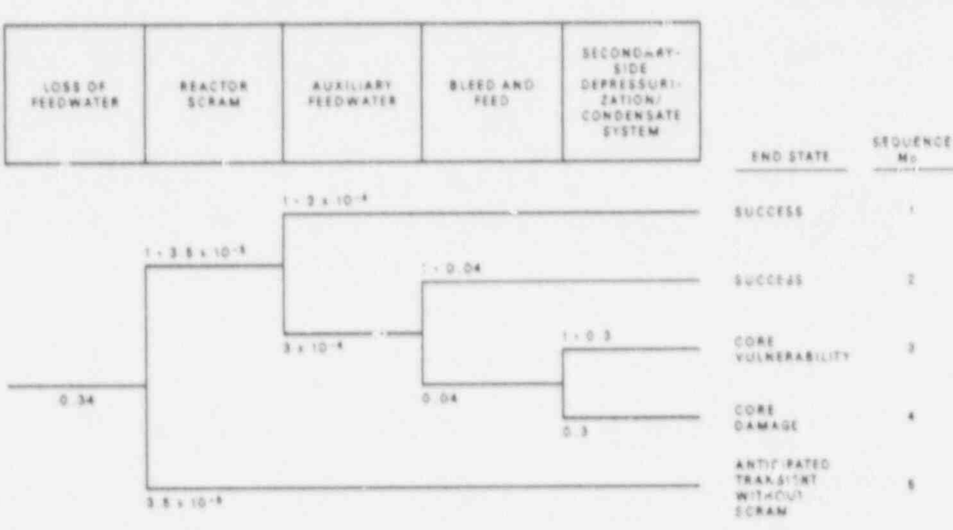

Fig. 3.1. Example event tree for initiator calculation. Bleed and feed is assumed capable of removing adequate decay heat for this example.

usually the case), the probabilities calculated for each of the sequences would be summed to estimate an overall conditional probability for the end state.

The second example event involves failures that would prevent secondary-side depressurization if required to prevent core damage following an LOFW with subsequent AFW and bleed-and-feed failure. Assume these failures were discovered during testing. The event tree for this example precursor is shown in Fig. 3.2. The failure probability associated with the precursor event (secondary-side depressurization failure) would be assigned based on the recovery class associated with the event. No initiating event occurred with the precursor; however, a failure duration of 360 h was estimated based on one-half of a monthly test interval. The estimated nonrecoverable LOFW frequency (assumed to be 0.3/reactor year in this example), combined with this failure interval (360 h), results in an estimated initiating event probability of 1.2×10^{-2}. So that event tree branches not involved with the precursor were eliminated and only the additional contribution (incremental risk) associated with the precursor was estimated, the event tree was calculated a second time using the same initiating-event probability but with all branches assigned normal failure probabilities (no failed or degraded states). This value was subtracted from the value obtained in the first calculation to obtain the conditional probability associated with the precursor. The probabilities for sequences involving undesirable end states (employing the same calculational method as above and subtracting the normal risk during the time interval) are -1×10^{-7} for core vulnerability, 1×10^{-7} for core damage, and 0.0 for ATWS. Note that the impact of the postulated failure on the core-vulnerability sequence is negative, indicating an effective decrease in the likelihood of this sequence during the failure period compared with the same period

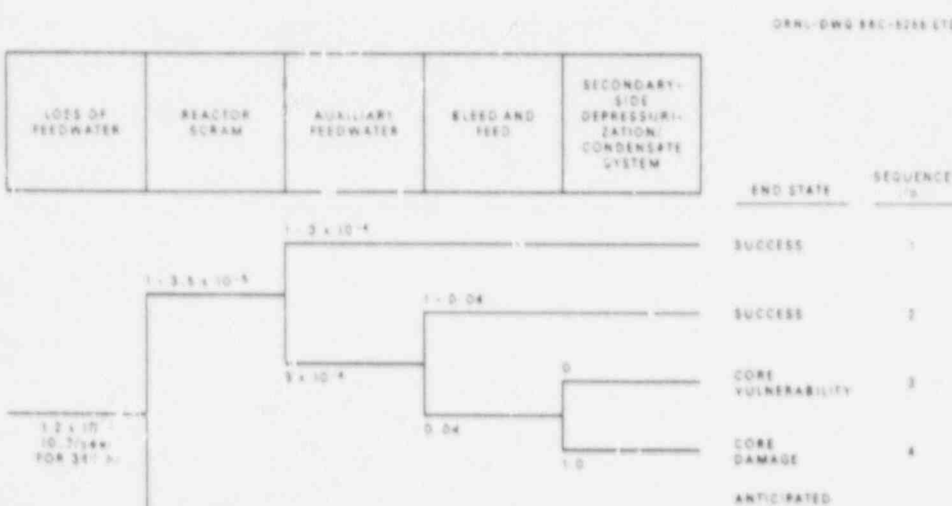

Fig. 3.2. Example event tree for unavailability calculation.

without the failure (1×10^{-7}). Note also that the impact of the postulated failure on the ATWS sequence is zero because secondary-side depressurization success or failure does not impact that sequence as modeled.

3.2.6 1986 Precursor Calculations

The conditional probability of potential severe core damage associated with each precursor (calculated in Appendix D) is identified under the heading CD PROB in Table 3.3.

A combined conditional probability for non-ATWS undesirable sequences was estimated by adding conditional probabilities for core-damage and core-vulnerability sequences and is listed under the heading SUM PROB. As noted in Chap. 2 of Refs. 1 and 2, this estimate is conservative because the core-vulnerability end state includes sequences [such as uncontrolled cooldown without high-pressure injection (HPI)] not believed significant from a core-damage standpoint.

As discussed in Sect. 3.1, the conditional probabilities determined for each precursor were based in part on industrywide data and therefore should not be directly associated with the probability of potential severe core damage resulting from the actual precursor event at the specific reactor plant at which it occurred.

The distribution of precursors as a function of conditional probability of core damage is shown in Fig. 3.3. A corresponding distribution as a function of core damage plus core vulnerability is shown in Fig. 3.4. The shape of this distribution is somewhat similar to that in Fig. 3.3 — an indication that the conservative use of the sum of the probabilities of core damage and core vulnerability does not substantially impact the ranking of operational events in 1986.

3.3 Reference Event Calculations

Conditional core-damage probability estimates were also calculated for nonspecific reactor trip, LOFW, and unavailabilities in certain single-train BWR systems (HPCI, high-pressure core spray, reactor-core isolation cooling and control-rod drive-cooling). These calculations provide a reference to the relative importance of these events, which are too numerous to warrant individual calculations. The results of these calculations are listed in Table 3.4.

Table 3.4 shows that nonspecific reactor trips without additional observed failures have conditional core-damage probabilities in the range of 10^{-5} to 10^{-7} 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 range from 10^{-6} to mid 10^{-5} per LOFW event. 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.

Table 3.3. Precursors listed by docket and LER number

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	E	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	D	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	D	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BX	FPL	06/11/73	UNAVL
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	D	N	14.1	1.4E-3	2.1E-3	693	P	W	BX	FPL	10/20/72	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UX	CPL	09/20/70	LOOP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	D	N	12.8	2.1E-6	3.4E-5	887	P	B	UX	DPC	04/19/73	TRIP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	[illegible]E-5	887	P	B	UX	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	[illegible]E-5	887	P	B	UX	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UX	DPC	09/05/74	UNAVL
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BX	PEC	09/16/73	TRIP
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	D	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/17/74	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/17/74	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	D	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	D	N	14.4	7.7E-6	7.7E-6	655	B	G	BX	BEC	06/16/72	LOOP
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	D	N	14.3	4.8E-7	4.8E-7	497	P	W	BX	WMP	05/30/72	MSLB
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	D	N	9.8	1.8E-6	2.5E-4	845	P	C	BX	BGE	11/30/76	TRIP
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	D	N	3.0	2.6E-7	8.9E-7	1080	P	C	BX	SCE	08/29/83	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EX	FPL	06/02/83	UNAVL
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	D	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	D	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	D	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	[illegible]	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL

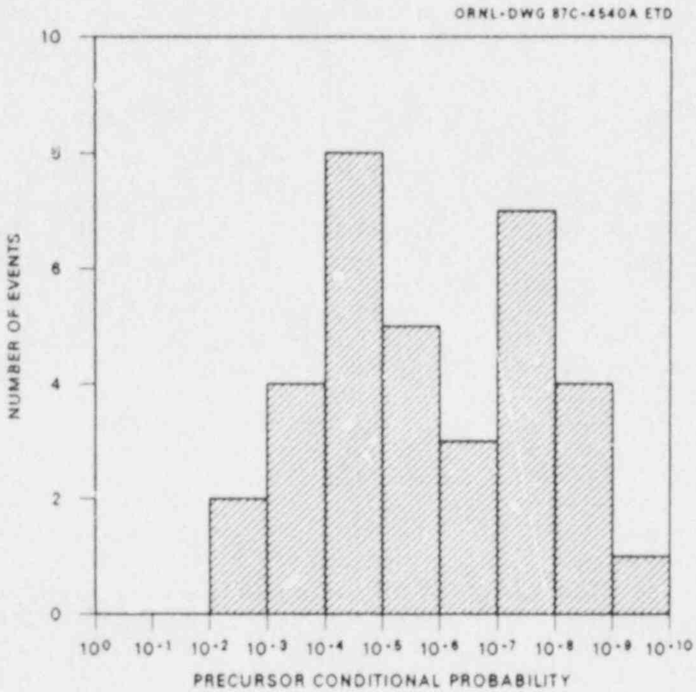

Fig. 3.3. Distribution of 1986 precursors as a function of core-damage probability.

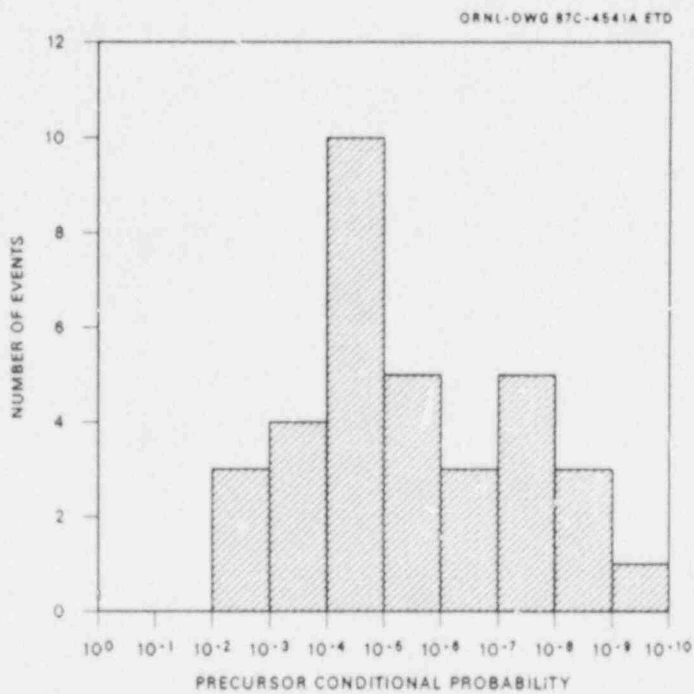

Fig. 3.4. Distribution of 1986 precursors as a function of core-damage plus core-vulnerability probability.

Table 3.4. Reference conditional event probability values

Calculation[a]	Core-damage probability	Core-damage plus core vulnerability probability
BWR Class A nonspecific reactor trip	6.0×10^{-7}	5.9×10^{-6}
BWR Class A LOFW	1.2×10^{-5}	1.2×10^{-4}
BWR Class B nonspecific reactor trip	5.3×10^{-8}	1.6×10^{-7}
BWR Class B LOFW	5.8×10^{-7}	2.8×10^{-6}
BWR Class C (turbine-driven feed pumps) nonspecific reactor trip	1.0×10^{-5}	1.0×10^{-5}
BWR Class C (turbine-driven feed pumps) LOFW	6.0×10^{-5}	6.0×10^{-5}
BWR Class C (motor-driven feed pumps) nonspecific reactor trip	8.3×10^{-6}	8.3×10^{-6}
BWR Class C (motor-driven feed pumps) LOFW	4.9×10^{-5}	4.9×10^{-5}
PWR Class A nonspecific reactor trip	1.4×10^{-6}	1.8×10^{-5}
PWR Class A LOFW	2.6×10^{-6}	2.2×10^{-5}
PWR Class B, C, E, and F nonspecific reactor trip	1.3×10^{-6}	8.7×10^{-6}
PWR Class B, C, E, and F LOFW	2.2×10^{-6}	1.1×10^{-5}
PWR Class D nonspecific reactor trip	1.3×10^{-6}	3.4×10^{-6}
PWR Class D LOFW	2.1×10^{-6}	5.6×10^{-6}
PWR Class G (with PORV) nonspecific reactor trip	1.8×10^{-6}	3.5×10^{-6}
PWR Class G (with PORV) LOFW	2.7×10^{-6}	5.9×10^{-6}
PWR Class G (without PORV) nonspecific reactor trip	2.8×10^{-6}	8.4×10^{-6}
PWR Class G (without PORV) LOFW	1.3×10^{-5}	3.6×10^{-5}
BWR Class C HPCI unavailability (turbine-driven feed pumps, 360-h unavailability)	2.5×10^{-5}	2.5×10^{-5}
BWR Class C HPCS unavailability (turbine-driven feed pumps, 360-h unavailability)	1.8×10^{-5}	1.8×10^{-5}
BWR Class C RCIC unavailability (turbine-driven feed pumps, 360-h unavailability)	2.6×10^{-7}	3.0×10^{-7}
BWR Class C CRD cooling unavailability (turbine-driven feed pumps, 360-h unavailability)	3.0×10^{-7}	3.0×10^{-7}

[a]Multiple calculations were performed for BWR Class C because plants in this class use turbine- and motor-driven feed pumps. Closure of the MSIVs on low reactor water level results in an LOFW for BWRs that use turbine-driven pumps. Multiple calculations were also done for PWR Class G plants with and without PORVs.

3.4 Precursor Rankings

The conditional probability of severe core damage was used to rank each event selected as a precursor. This ranking is related to the impact of an event under conditions consistent with those observed during the actual event. Table 3.5 ranks events in order from the highest conditional core-damage probability to the lowest. Table 3.6 ranks the precursors based on the sum of core-damage and core-vulnerability probabilities.

The conditional probabilities of severe core damage listed in Table 3.5 range from 3.3×10^{-3} to below 1.0×10^{-9}, with many events below 1×10^{-4}. Because of the uncertainties inherent in the calculations, ranking events on an event-by-event basis is not considered desirable. Therefore, events were binned into conditional probability ranges to identify the more significant events. Table 3.7 lists the events identified in Tables 3.5 and 3.6 by order of magnitude.

3.5 Uses of Results

A comparison of the current results with 1969—1981 accident-sequence-precursor efforts cannot be made by simply adding all core-damage and core-vulnerability sequences. Those efforts modeled secondary-side cooldowns in terms of success or core damage (using a steam-line break event tree), did not address potential recovery after main- and auxiliary-feedwater failure and failure of feed and bleed, and provided a limited ATWS sequence development, with defined success and core-damage end states. However, the 1986 results can be compared with the 1984—1985 results.

For the current results, a more conservative measure of event significance can be obtained by considering both the core-damage and core-vulnerability end states as undesirable and adding them, as is done in the column SUM PROB in Tables 3.3 and 3.4. Caution must be used when doing this, however, since relatively minor sequences (for example, unavailability of HPI following a trip with a non-isolable stuck open secondary side relief valve in a PWR) are included within the set of core vulnerability sequences. Note that for ranking, however, use of CD PROB or SUM PROB give similar results — see Figs. 3.3 and 3.4. A sequence-by-sequence assessment could also be used to eliminate those core-vulnerability sequences considered less serious, such as minor overcooling sequences. Because ATWS sequences were not developed in this effort, they should not be added to other sequences for event ranking.

Table 3.5. Precursors listed by conditional core-damage probability

LER NO.	E DATE	DESCRIPTION	PLANT NAME	SY	COMP	O	D	:	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX	E	O	N	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	E	O	N	14.1	1.4E-3	2.1E-3	693	P	W	BX	FPL	10/20/72	TRIP
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	N	15.4	3.0E-4	5.6E-3	700	B	G	UX	CPL	09/20/70	LOOP
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BX	PEC	09/16/73	TRIP
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BX	FPL	10/20/72	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UX	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UX	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UX	DPC	09/05/74	UNAVL
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	EE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BX	BEC	06/16/72	LOOP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EX	FPL	06/02/83	UNAVL
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UX	DPC	04/19/73	TRIP
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BX	BGE	11/30/76	TRIP
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BX	WMP	05/30/72	MSLB
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BX	SCE	08/29/83	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.9	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/17/74	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/17/74	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BX	FPL	06/11/73	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL

Table 3.6. Precursors listed by sum of core-damage and core-vulnerability probabilities

LER NO.	E DATE	DESCRIPTION	PLANT NAME	S/	COMP	O	D	[illegible]	AGE	CD PROB	SUM PROB	RATE	T	V	AE	OPR	CRITICAL	TRANS
261/86-005	01/28/86	BUS FAILS WITH LOOP	ROBINSON 2	EB	ELECON	E	M	[illegible]	15.4	3.0E-4	5.6E-3	700	B	G	UX	CPL	09/20/70	LOOP
413/86-031	06/13/86	SBLOCA PLUS TRIP	CATAWBA 1	PC	PIPEXX		O	[illegible]	1.4	3.3E-3	4.9E-3	1145	P	W	DPC	DPC	01/07/85	LOCA
250/86-039	12/27/86	TRIP & OPEN PORV	TKY.POINT3	CA	VALVOP	[illegible]	[illegible]	[illegible]	14.1	1.4E-3	2.1E-3	693	P	W	BX	FPL	10/20/72	TRIP
247/86-035	10/20/86	TRIP,LOFW & AFW TRN	IND.POINT2	IA	CKTBRK	E	O	N	13.4	2.9E-4	8.0E-4	873	P	W	UE	CEC	05/22/73	TRIP
318/86-006	09/05/86	TRIP AND OPEN ASDV	CALCLIFFS2	HE	VALVOP	E	O	N	9.8	1.8E-6	2.5E-4	845	P	C	BX	BGE	11/30/76	TRIP
414/86-028	06/27/86	OPEN MSRV PLUS TRIP	CATAWBA 2	CC	INSTRU	E	T	Y	0.1	1.1E-4	1.1E-4	1145	P	W	DPC	DPC	05/08/86	MSLB
247/86-017	05/28/86	OPEN TBSV AND TRIP	IND.POINT2	HE	VALVOP	E	O	N	13.0	1.0E-4	1.0E-4	873	P	W	UE	CEC	05/22/73	MSLB
277/86-003	01/24/86	DG TRIP CAUSES SCRAM	PEACHBOTM2	CD	VALVEX	E	T	N	12.3	8.1E-5	8.1E-5	1065	B	G	BX	PEC	09/16/73	TRIP
458/86-002	01/01/86	UNCOMPLICATED LOOP	RIVERBEND1	EE	TRANSF	G	O	Y	0.2	7.0E-5	7.0E-5	936	B	G	SW	GSU	10/31/85	LOOP
250/86-038	12/04/86	UNAVAILABILITY AFW	TKY.POINT3	HH	INSTRU	E	T	N	14.2	5.8E-5	5.8E-5	693	P	W	BX	FPL	10/20/72	UNAVL
285/86-001	07/02/86	TRIP & ADS/TBS FAIL	FTCALHOUN	EB	GENERA	E	O	N	12.9	4.1E-5	4.2E-5	478	P	C	GH	OPP	08/06/73	TRIP
269/86-001	01/31/86	LOFW, OPEN MSRV	OCONEE 1	CC	VALVEX	E	O	N	12.8	2.1E-6	3.4E-5	887	P	B	UX	DPC	04/19/73	TRIP
409/86-023	07/10/86	UNCOMPLICATED LOOP	LACROSSE	EE	CKTBRK	G	O	N	19.0	2.0E-5	2.0E-5	50	B	A	SL	DPL	07/11/67	LOOP
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 1	WA	PUMPXX	E	T	N	13.5	1.1E-5	1.1E-5	887	P	B	UX	DPC	04/19/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 2	WA	PUMPXX	E	T	N	12.9	1.1E-5	1.1E-5	887	P	B	UX	DPC	11/11/73	UNAVL
269/86-011	10/01/86	ECCW IS UNAVAILABLE	OCONEE 3	WA	PUMPXX	E	T	N	12.1	1.1E-5	1.1E-5	887	P	B	UX	DPC	09/05/74	UNAVL
281/86-010	07/11/86	HHIS IS UNAVAILABLE	SURRY 2	SF	PUMPXX	E	T	Y	13.3	3.1E-8	1.0E-5	788	P	W	SW	VEP	03/07/73	UNAVL
293/86-027	11/19/86	UNCOMPLICATED LOOP	PILGRIM 1	CE	ELECON	E	O	N	14.4	7.7E-6	7.7E-6	655	B	G	BX	BEC	06/16/72	LOOP
249/86-013	08/27/86	HPCI, CSS & DG UNAVL	DRESDEN 3	SF	VALVOP	E	T	N	15.6	2.7E-6	4.7E-6	794	B	G	SL	CWE	01/31/71	UNAVL
280/86-029	09/29/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	O	N	14.2	1.0E-8	3.4E-6	788	P	W	SW	VEP	07/01/72	UNAVL
389/86-011	07/09/86	EPS UNAVAILABILITY	ST.LUCIE 2	EE	ENGINE	E	T	N	3.1	2.6E-6	3.4E-6	830	P	C	EX	FPL	06/02/83	UNAVL
280/86-031	10/30/86	HHIS IS UNAVAILABLE	SURRY 1	SF	PUMPXX	E	M	Y	14.2	3.1E-9	1.0E-6	788	P	W	SW	VEP	07/01/72	UNAVL
362/86-011	08/04/86	SWS/CCWS UNAVAILABLE	SANONOFRE3	WA	HTEXCH	E	O	N	3.0	2.6E-7	8.9E-7	1080	P	C	BX	SCE	08/29/83	UNAVL
341/86-048	12/24/86	RCIC/HPCI UNAVAIL	FERMI 2	SF	MECFUN	C	T	N	1.5	6.5E-7	6.5E-7	1093	B	G	SL	DEC	06/21/85	UNAVL
301/86-004	09/28/86	MSIVS FAIL TO CLOSE	PT.BEACH 2	CD	VALVEX	G	O	N	14.3	4.8E-7	4.8E-7	497	P	W	BX	WMP	05/30/72	MSLB
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	13.0	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-011	12/27/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	[illegible]	[illegible]	11.9	4.0E-8	5.1E-8	530	P	W	BX	NSP	12/17/74	UNAVL
370/86-006	03/29/86	MULTIPLE HHIS TRAINS	MCGUIRE 2	EE	ENGINE	H	T	Y	2.9	3.4E-8	4.8E-8	1180	P	W	DPC	DPC	05/08/83	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS1	EE	ENGINE	E	T	N	12.8	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/01/73	UNAVL
282/86-006	09/08/86	EPS UNAVAILABILITY	PRAIRIEIS2	EE	ENGINE	E	T	N	11.7	1.9E-8	2.4E-8	530	P	W	BX	NSP	12/17/74	UNAVL
458/86-047	07/31/86	MULTIPLE TRAINS FAIL	RIVERBEND1	EE	ENGINE	E	O	N	0.8	7.1E-9	7.1E-9	936	B	G	SW	GSU	10/31/85	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT3	EE	INSTRU	E	T	Y	14.1	1.1E-9	5.7E-9	693	P	W	BX	FPL	10/20/72	UNAVL
250/86-036	11/06/86	UNAVAILABILITY EPS	TKY.POINT4	EE	INSTRU	E	T	Y	13.4	1.1E-9	5.7E-9	693	P	W	BX	FPL	06/11/73	UNAVL
366/86-035	11/13/86	LPCS IS UNAVAILABLE	HATCH 2	SF	VALVOP	G	T	Y	8.3	3.6E-10	3.6E-10	784	B	G	SS	GPC	07/04/78	UNAVL

Table 3.7. Precursors for 1986 ranked by order of magnitude

Conditional probability range	Events ranked by probability of core damage	Events ranked by probability of core damage and core vulnerability
10^{-1} to 1	None	None
10^{-2} to 10^{-1}	None	None
10^{-3} to 10^{-2}	Small LOCA from letdown-line rupture at Catawba 1 (413/86-031)	Small LOCA from letdown-line rupture at Catawba 1 (413/86-031)
	Reactor trip with stuck-open PORV at Turkey Point 3 (250/86-039)	Reactor trip with stuck-open PORV at Turkey Point 3 (250/86-039)
	LOOP with one DG out of service at Robinson 2 (261/86-005)	LOOP with one DG out of service at Robinson 2 (261/86-005)
10^{-4} to 10^{-3}	Reactor trip, LOFW, and and failure of two AFW trains at Indian Point 2 (247/86-035)	Reactor trip, LOFW, and failure of two AFW trains at Indian Point 2 (247/86-035)
	Inadvertent opening of SG PORVs during a test, followed by uncontrolled letdown, failure to provide HPI, and failure of one MFW pump at Catawba 2 (414/86-028)	Inadvertent opening of SG PORVs during a test, followed by uncontrolled letdown, failure to provide HPI, and failure of one MFW pump at Catawba 2 (414/86-028)
	Steam dump valves inadvertently open, and one safeguards train fails to actuate at Indian Point 2 (247/86-017)	Steam dump valves inadvertently open, and one safeguards train fails to actuate at Indian Point 2 (247/86-017)
		Reactor trip and one atmospheric steam dump valve stuck open at Calvert Cliffs 2 (318/86-006)
10^{-5} to 10^{-4}	8 events	10 events
10^{-6} to 10^{-5}	5 events	5 events
$<10^{-6}$	15 events	12 events

3.6 References

1. J. W. Minarick, J. D. Harris, P. N. Austin, E. W. Hagen, and J. W. Cletcher, *Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report*, NUREG/CR-4674, Vol. 1 (ORNL/NOAC-232/V1), December 1986.

2. J. W. Minarick, J. D. Harris, P. N. Austin, J. W. Cletcher, and E. W. Hagen, *Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report*, NUREG/CR-4674, Vol. 3 (ORNL/ NOAC-232/V3), May 1987.

3. 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., *Precursors to Potential Severe Core Damage Accidents: 1980—81, A Status Report*, NUREG/CR-3591, Vols. 1 and 2 (ORNL/NSIC-217/V1 and V2), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., July 1984.

4. RESULTS

This chapter describes results of the 1986 effort plus a preliminary qualitative assessment of differences in more serious precursors in 1969–1981 and 1984–1986. The body of the report and the precursor documentation in Appendix D contain additional insights and findings and serve to place the following comments in perspective.

4.1 Important 1986 Precursors

The following 1986 precursors were ranked high by the ranking method described in Chap 3. These events primarily involve stuck-open secondary-side valves, reactor trip with failure of mitigating systems, LOOP, as well as a small LOCA from a letdown line rupture.

At Catawba 1 [licensee event report (LER) 413/86-031] a small LOCA occurred, initiated by a loss of control power to the letdown orifice valve, which caused the valve to fail open. Following the flow surge, a line rupture occurred downstream of the failed valve's flange. Letdown isolation valves were subsequently closed to contain the LOCA.

At Turkey Point 3 (LER 250/86-039), following a loss of turbine governor oil pressure and subsequent rapid load decrease, the unit was tripped. During the transient, a primary-side PORV opened but failed to close fully. The operators closed the PORV block valve, and the unit was stabilized.

A LOOP occurred at Robinson 2 (LER 261/86-005) following a transient when a west bus lockout occurred in the 115-kV switchyard. The B emergency diesel generator (DG) was out of service at the time. DG B was subsequently started manually and loaded to restore power to its emergency bus.

At Indian Point 2 (LER 247/86-035) an inadvertent reactor trip from 100% power occurred, and in the ensuing transient AFW was demanded to recover dropping steam generator (SG) levels. However, one motor-driven AFW pump tripped and the turbine-driven AFW pump failed when the steam supply line became overpressurized, resulting in a relief valve lift. SG levels were maintained by the remaining AFW pump.

At Catawba 2 (LER 414/86-028) all four atmospheric dump valves inadvertently opened during a test for loss of control room function. A transient ensued with SG depressurization, and a main feedwater pump tripped on low suction pressure. Loss of letdown-flow control occurred and high-pressure-injection (HPI) flow from the charging pumps was demanded. Because of the test configuration and valve labeling errors, HPI flow requirements were not met. The test was terminated, allowing HPI to actuate.

At Indian Point 2 (LER 247/86-017) all 12 condenser steam dump valves inadvertently opened, resulting in a transient and safety injection (SI) actuation. SI train B failed to actuate, but train A actuation closed the main steam isolation valves (MSIVs), ending the high-steam-flow condition.

4.2 Number of Precursors Identified

Of the 96 reactor years of experience in 1986, 34 accident sequence precursors were identified. This is ~0.4 per reactor year, the same number per reactor year identified in the 1969—1981 period. For the 1984 and 1985 periods, ~0.6 precursor per reactor year was identified.

With the revised LER rule that went into effect in 1984, requiring more detailed reporting, and the review of all reactor-trip events for precursors, some increase in the number precursors per reactor year might be expected over the number found in previous years. The fact that no increase has occurred for the 1986 period may indicate a slight improvement in operations over the 1984—1985 period. (Note, however, that in 1986 a number of plants that in the past consistently reported precursors were shut down for a substantial amount of time.)

The number of events with high conditional-core-damage probability is also somewhat lower than in previous periods. For events with conditional-core-damage probability of $>10^{-4}$, 17 were observed in 1984, 10 in 1985, and 6 in 1986. Sixteen were observed in the 2-year 1980—1981 period. The number of events with conditional-core-damage probability of $>10^{-3}$ appears consistent with one observed in 1984, two in 1985, and two in 1986. Six such events were observed in the 2-year 1980—1981 period.

When core-vulnerability-sequence probabilities are conservatively added to core-damage-sequence probabilities, the number of events at $>10^{-4}$ are 21 in 1984, 11 in 1985, and 7 in 1986.

The frequency of events (per reactor year) with conditional-core-damage probabilities of $>10^{-4}$ for all periods reviewed in the Accident Sequence Precursor Program are

Period	Frequency of events	
	With P (core damage) $>10^{-3}$	With P (core damage) $>10^{-4}$
1969—1979	0.039	0.15
1980—1981	0.045	0.12
1984—1986	0.022	0.13

Based on these frequencies, the incidence of accident sequence precursors of $>10^{-4}$ has remained essentially constant in the three periods, but the number of events that were $>10^{-3}$ apparently decreased in 1984—1986 compared with those in 1969—1981. However, if 90% Poisson confidence bounds are applied to the number of precursors observed in either 1986 or 1984—1986 that were $>10^{-3}$, the expected number of precursors based on the number observed in 1969—1979 and 1980—1981 still falls within these bounds. Because of this, the apparent decrease in highly significant events may be the result of random fluctuations in the data.

BWR plants currently contribute a greater number of precursors with conditional-core-damage probability of $>10^{-4}$ than would be expected if the likelihood of such events was proportional to the number of BWR and PWR reactor years in the observation period. Of such events in 1984—1986, 56% occurred at BWRs; yet these plants make up ~37% of the reactor population.

4.3 Initiating-Event Frequencies and System-Failure Probabilities

In Accident Sequence Precursor Program efforts concerning 1969—1981 operational events, initiating-event frequencies and branch-failure probabilities used in the quantitative precursor assessments were developed from the precursors themselves when at all possible. This development used the effective number of nonrecoverable events seen in the observation period, combined with appropriate demand assumptions, to estimate branch probabilities used in calculating sequence frequencies.

For 1984—1985 because of increased model specificity and the limited observation period, initiating-event frequencies and branch-failure probabilities were developed in most cases from train-based system models. Probabilities used to quantify the system models were based on data developed from 1969—1981 events, if applicable, or from typically assumed train-failure probabilities. Most system-failure probabilities employed in the calculation were developed by first estimating train and serial component (such as a tank) failure probabilities and then using these to estimate the failure probability of the entire system. Such an approach resulted in system-failure probability estimates that reflected the degree of redundancy actually available and permitted easy revision of these probabilities based on train failures and unavailabilities observed during an operational event. However, the probabilities used in the calculations did not reflect later precursor information.

The precursors identified in 1984—1986 were used to develop branch probabilities for 1986 precursor conditional-probability estimates. The specific events utilized in this development, the nonrecovery likelihoods assigned to each event, and the demand assumptions utilized for each estimate are listed in Appendix C.

Table 4.1 identifies the number of nonrecoverable events expected in 1984—1986 based on frequency and probability values used in the 1969—1979 and 1980—1981 efforts. The values listed are the number of nonrecoverable events. Figures 4.1 and 4.2 show these values for BWRs and PWRs separately and include 90% confidence bounds on the observed number of events. (Confidence bounds were estimated by assuming the failures could be described using a Poisson process and by interpolating between Poisson 90% bounds integer values.)

Although the confidence bounds associated with the small number of events in each category are large (in fact in every instance any reasonable confidence bound on the observed number of events in 1984—1986 overlaps the expected number of events), the number of categories with fewer events than expected can be used to draw conclusions concerning a

Table 4.1. Comparison of nonrecoverable failures seen in 1984–1986 with those estimated based on 1969–1981 precursors

Initiating event/branch	Expected events in 1984–1986: Based on 1969–1979 data/estimates	Based on 1980–1981 data/estimates	Based on combined 1969–1981 estimates	Observed number of events in 1984–1986
		PWRs		
LOOP	5.1	3.0	4.6	2.6
Small LOCA	1.4	2.2	1.5	1.1
Auxiliary feedwater	1.7	0.1	1.2	0.4
High-pressure injection	1.6	0.7	1.2	1.0
Long-term core cooling	1.3	0	0.6	0
Emergency power	1.7	0	0.7	1.0
SG isolation	2.0	2.0	1.3	1.0
		BWRs		
LOOP	2.0	0.2	1.7	2.9
Small LOCA	1.9	4.0	1.9	0.9
HPCI/RCIC	3.1	3.8	1.8	0.7
RV isolation	0.6	0	0.4	0.3
Long-term core cooling		2.0		1.1
Emergency power	4.8	1.0	2.4	0
Automatic depressurization	1.2	0.2	0.6	0

Fig. 4.1. Observed vs expected nonrecoverable failures for BWR initiators and branches.

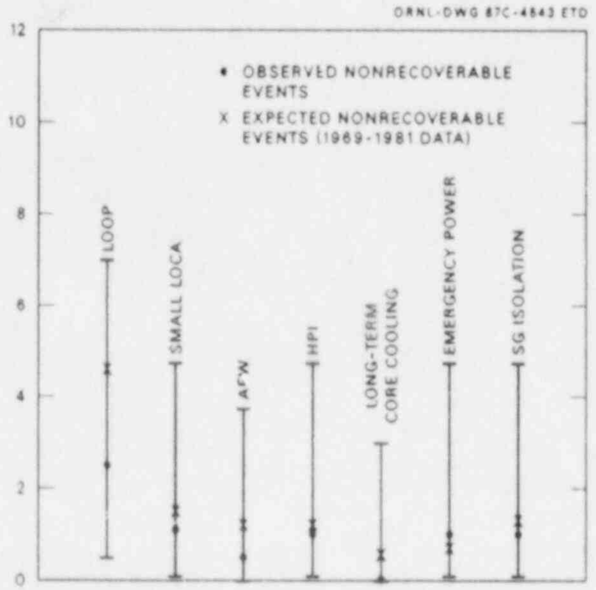

Fig. 4.2. Observed vs expected nonrecoverable failures for PWR initiators and branches.

general reduction in initiating-event frequencies and demand-failure probabilities compared with earlier observation periods (using the Chi-square test). For PWRs this reduction can be demonstrated with >90% confidence. For BWRs the confidence in such a reduction is less (80%), partly because BWR long-term core-cooling failures were not the subject of detailed review in 1969—1979.

Initiating-event frequencies and branch-failure probabilities estimated using precursor data for the 1969—1979, 1980—1981, and 1984—1986 periods are shown in Table 4.2. Values for 1969—1979 and 1980—1981 were developed using nonrecovery likelihoods consistent with those used with 1984—1986 precursors and reflected resolution of comments on the earlier Accident Sequence Precursor Program reports.

4.4 Likely Sequences

Precursors with conditional core-damage probabilities of $>10^{-4}$ that occurred in 1984—1986 were reviewed to identify the more likely severe-core-damage sequences associated with the precursors. The most likely core-damage sequences associated with these events include the observed plant state plus additional postulated failures beyond the operational event, required for core damage. These sequences can generally be categorized as

- failure of secondary-side cooling, plus failure to initiate condensate cooling successfully following SG depressurization (64% of PWR events $>10^{-4}$);
- station blackout (14% of PWR events $>10^{-4}$);
- failure to initiate recirculation cooling following a small-break LOCA (21% of PWR events $>10^{-4}$);
- failure of all high-pressure cooling and failure to depressurize following transients, LOOPs, and small-break LOCAs (72% of BWR events $>10^{-4}$);
- failure of long-term decay heat removal following a transient (17% of BWR events $>10^{-4}$); and
- Failure of high-pressure cooling following a LOOP plus unavailability of emergency power for low-pressure core cooling (11% of BWR events $>10^{-4}$).

As has been also noted in earlier volumes in this report, these sequences are generally consistent with those predicted in earlier probabilistic risk assessment (PRAs). Results of the revised PRAs for the five reference plants reported in the Reactor Risk Reference Document [NUREG 1150, February 1987 (draft)],[2] estimate a more substantial contribution from station blackout sequences than observed in BWR precursors.

In addition, loss of component cooling water is an important contributor to sequences in two of the three PWRs analyzed in NUREG 1150. Although two precursors associated with loss of cooling water systems were observed in the 1980—1981 period (plus one precursor associated with a non-safety-related cooling water system in 1985), only safety-related cooling water train unavailabilities have been observed in the

Table 4.2. Comparison of average initiating-event frequency, system-failure probability, and nonrecovery likelihood point estimates for 1969–1979, 1980–1981, and 1984–1986

Initiating event/branch	Average nonrecovery likelihood (1969–1981, 1984–1986)	Average initiating event frequency frequency/system failure probability		
		1969–1979[a]	1980–1981[a]	1984–1986
		PWRs		
LOOP	0.39	3.1×10^{-2}	1.9×10^{-2}	1.6×10^{-2}
Small LOCA	0.43	8.3×10^{-3}	1.4×10^{-2}	6.4×10^{-3}
Auxiliary feedwater	0.26	3.9×10^{-4}	1.8×10^{-5}	9.9×10^{-5}
High-pressure injection	0.84	8.1×10^{-4}	3.5×10^{-4}	5.1×10^{-4}
Long-term core cooling	1.00	6.2×10^{-4}	2.6×10^{-4}[b]	1.5×10^{-4}[b]
Emergency power	0.78	8.5×10^{-4}	2.6×10^{-4}[b]	5.0×10^{-4}
SG isolation	0.64	1.0×10^{-3}	1.0×10^{-3}	5.3×10^{-4}
		BWRs		
LOOP	0.32	2.2×10^{-2}	2.7×10^{-3}	3.3×10^{-2}
Small LOCA	0.50	2.1×10^{-2}	4.6×10^{-2}[c]	1.0×10^{-2}
HPCI/RCIC	0.49	3.8×10^{-3}	4.7×10^{-3}	8.4×10^{-4}
RV isolation	1.00	3.3×10^{-3}	3.8×10^{-3}[b]	1.7×10^{-3}
Long-term core cooling	0.71	1.1×10^{-4}[d]	4.5×10^{-4}	7.4×10^{-4}
Emergency power	0.85	4.5×10^{-3}	8.9×10^{-4}	8.9×10^{-5}[b]
Automatic depressurization system	0.71	1.4×10^{-2}	2.7×10^{-3}	2.6×10^{-3}[b]

[a]With nonrecovery numeric values consistent with those used in the assessment of 1984–1986 events (see Table 3.1).

[b]No events were observed in the time period. The estimate was developed based on assumption of 0.33 event in observation period × average nonrecovery likelihood.

[c]Three small LOCA-related events were observed at Pilgrim during this period. Two of these three events have been assumed to be specific to that plant.

[d]Value assumed in NUREG/CR-2497 (Ref. 1). Based on zero observations in the observation period (see note *c*), a value of 9.2×10^{-5} is estimated.

other periods. Two reactor trips were identified in 1984—1986 in which malfunction of the component cooling water system resulted in the need to repair RCP seals:

Plant	LER	Plant
Calvert Cliffs 2	318/85-001	Degradation in RCP seals due to CCW pressure fluctuations
St. Lucie 2	389/84-016	Loss of bus B following trip and subsequent unavailability of non-safety-related B loads, resulting in seal degradation of the reactor coolant pump

In both cases safety-related cooling water was maintained.

4.5 Qualitative Comparison of 1984—1986 and Earlier Precursors

The more significant precursors identified in 1969—1981 and 1984—1986 were reviewed to develop a preliminary, qualitative understanding of differences in the types of events observed in the two periods. Events chosen for this review were primarily those with conditional core-damage probabilities of $>10^{-3}$. However, because the number of such events in 1984—1986 is very small, events with conditional probabilities of $>10^{-4}$ were also utilized to some extent. Although the event sequence models and probability values used in the assessment of 1969—1981 precursors are somewhat different from those used in later analyses, this is not expected to substantially bias the results of the review.

Events that occurred in 1969—1981 and were assessed at $>10^{-3}$ were used to define a limited number of event classifications: transients driven by electrical and instrumentation interactions, precursors involving AFW or HPCI/RCIC inoperability, events related to small-break LOCAs, and miscellaneous events. Precursors identified in 1984—1986 were then reviewed against these categories to determine changes in the number and nature of events currently being observed.

Transients driven by electrical and instrumentation interactions. Eight events with conditional probabilities of $>10^{-3}$ were identified in 1969—1981, including the Rancho Seco nonnuclear instrumentation bus failure (March 20, 1978), loss of power to safety-related buses at Millstone 2 resulting from incorrect undervoltage set points (July 20, 1976), the loss of a dc bus at Millstone 2 (January 2, 1981), the Crystal River 3 nonnuclear instrumentation bus failure (February 26, 1980), and the installation of dummy instrument signals at Zion 2 and subsequent draining of the pressurizer (July 12, 1977). In many of these events, the observed plant response was not anticipated by the operators (although a detailed analysis could have predicted it), and restoration of stable plant conditions was haphazard. No events of this type with conditional probabilities of $>10^{-4}$ were observed in 1984—1986.

Precursors involving AFW or HPCI/RCIC inoperability. Six events involving AFW system inoperability and two events with combined HPCI/RCIC inoperability with conditional probabilities of $>10^{-3}$ were observed in 1969—1981. Included in this set is the Three Mile Island Nuclear Station Unit 2 (TMI-2) accident and two events involving clogged AFW pump suction strainers. For 1984—1986 only one event of $>10^{-3}$ was associated with AFW system failure (Davis-Besse, June 9, 1985). AFW-related events with probabilities of $>10^{-4}$ were also observed in 1984—1986, but to a lesser extent than in 1969—1981. One HPCI/RCIC unavailability with conditional probability of $>10^{-3}$ was observed in 1984—1986, and it is described below.

Small-break LOCA-related events. In addition to the TMI accident, two additional LOCA-related events with conditional probabilities $>10^{-3}$ were observed in 1969—1981: the stuck-open PORV at Davis-Besse (September 24, 1977) and a stuck-open safety valve with RCIC inoperable and residual heat removal (RHR) degraded at Brunswick (April 29, 1975). For 1984—1986, three LOCA-related events were also observed, an open relief valve (caused by water dripping from a heating, ventilation, and air-conditiong duct onto control room instrumentation) with both RCIC and HPCI unavailable at Hatch 1 (May 15, 1985), a LOCA associated with a letdown line drain valve at Catawba 1 (June 13, 1986), and a stuck-open PORV at Turkey Point 3 (December 27, 1986). (Note: High conditional probabilities for the latter two events may be driven by the particular model used in the event assessments and may be overly conservative.)

Miscellaneous events. Events placed in this category for 1969—1981 include the Browns Ferry fire (March 22, 1975), unavailability of both RHR heat exchangers at Brunswick 1 due to oyster-shell plugging (April 19, 1981), an LOFW and subsequent low reactor vessel level due to incorrectly closed recirculation valves at Oyster Creek (May 2, 1979), and the top-head steam bubble incident during natural circulation cool-down at St. Lucie 1 (June 25, 1980). In both the Oyster Creek and St. Lucie 1 events, a misunderstanding of expected plant response was exhibited. For 1984—1986 only one event was considered applicable: the LOFW combined with the potential for RCIC and shutdown cooling failure within 15 d at LaSalle (September 21, 1984).

As a result of this review, it appears that the more serious current events being identified in the Accident Sequence Precursor Program are more consistent with events typically modeled in PRAs than was the case in 1969—1981. (This is not totally the case; the open relief valve at Hatch 1 caused by water dripping onto control room instrumentation and control equipment raises the potential for complicated system interactions only marginally addressed in contemporary PRAs.) Complicated events involving electric power and instrumentation interactions were not seen in 1984—1986 nearly to the extent that they were seen in 1969—1981. Auxiliary feedwater and HPCI/RCIC system performance both appear improved compared with 1969—1981 and, in fact, exhibit failure probabilities consistent with PRA models.

4.6 References

1. J. W. Minarick and C. A. Kukielka, *Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report*, NUREG/CR-2497, Vols. 1 and 2 (ORNL/NSIC-182/V1 and V2), Union Carbide Corp., Nuclear Div., Oak Ridge Natl. Lab., June 1982.

2. *Reactor Risk Reference Document*, NUREG-1150, U.S. Nuclear Regulatory Commission, February 1987 (Draft).

GLOSSARY

GLOSSARY

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

Accident sequence precursor. A historically observed element in a postulated sequence of events leading to some undesirable consequence. For purposes of the ASP Study, the undesirable consequence is usually potential 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 potential 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 chance 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 failure. 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.

Consequently degraded system. A system was considered consequently degraded if a component failure external to the system resulted in loss of system redundancy (e.g., if an AFW train was rendered unavailable during a potential LOOP because of a DG failure).

Consequently failed system. A system was considered consequently failed if it failed because of (1) the failure of another system or (2) an internal fault that would have rendered it degraded plus an external fault that eliminated the remaining operability. [An example of the second case is a failed HPI system during a postulated LOOP due to the unavailability of one of two HPI pumps plus the unavailability of the DG that would provide power to the operable HPI pump.]

Core damage. See *severe core damage.*

Core-melt accident. An event in a nuclear power plant in which core cooling is insufficient to prevent the core from heating up to a temperature at 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 failure*.

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

Demand. A test or an operating condition that requires the availability of a component or a system. In this study, it includes actuations required during testing and because of the initiating events that were accounted for. One demand consisted 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 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. See *occurrence*.

Event sequence. A particular path on an event tree.

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

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

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

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

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

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

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

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

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

Licensee Event Reports. Those reports submitted to NRC by utilities who operate nuclear plants as required by 10 CFR 50 and NUREG-0161. 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 effected, 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.

Uncertainty analysis. Analysis that provides a measure of the overall uncertainty in a result because of known uncertainties that influence the overall result.

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

APPENDIX A

PLANT CATEGORIZATION

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PLANT CATEGORIZATION

The models used for evaluation of the selected accident sequence precursors were based on the realization that many plants have a similar response to a transient at the system or functional level, and they were developed to reflect these similarities.

Detailed categorization of the plants by functional response was initially performed by the University of Maryland for the Accident Sequence Precursor Program.[1] The Accident Sequence Precursor Program has generally employed these categorizations; however, some modifications were required to reflect more closely the specific needs of the accident sequence precursor evaluations. Tables A.1 and A.2 identify system similarities and differences important to the plant groupings. Table A.3 lists plant-specific information previously included with the individual precursor documentation.

For the BWRs, three general and one plant-specific class were necessary. BWR Class A consists of the older plants, which are characterized by ICs and FWCI systems that employ the MFWPs. BWR Class B consists of plants that have ICs but a separate HPCI system instead of FWCI. BWR Class C includes the modern plants that have neither ICs nor FWCI. However, they have an RCIC system that Classes A and B lack. The Class C plants could be separated into two subgroups, those plants with motor-driven MFWPs and those with steam-turbine-driven pumps. This difference is addressed in the probabilities assigned to branches impacted by the use of motor- vs turbine-driven pumps. A fourth, unique class includes only the LaCrosse Boiling Water Reactor.

The PWRs are separated into eight classes. Two of the classes represent most Babcock and Wilcox Company (Class D) and Combustion Engineering (Class G) plants.*

Westinghouse plants require six classes due to inherent design differences. One class comprises some unique plants that must be represented individually. The other five are characterized by differences in their HPI systems (high- vs intermediate-head), their use of CC for core protection, and their employment of low-pressure systems. Several of these classes have similar responses to the point of core damage but differ in containment response. Because post-core-damage sequences are

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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 into 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.

not currently addressed in the Accident Sequence Precursor program, these classes have been additionally grouped for analysis.

Reference

1. M. Modarres, E. Lois, and P. Amico, *LWR Categorization Report*, University of Maryland, November 13, 1984.

Table A.1. BWR pl

Plant	BWR type	RPS	SBLC'	PCS	SRV	MFW	IC/ICMUP	FWCI	RCIC	HPCI	CR
Big Rock Point	1	X	X	X	X	X	X				X
Millstone 1	3	X	X	X	X	X	X	X			X
Nine Mile Point 1	2	X	X	X	X	X	X	X			X
Oyster Creek	2	X	X	X	X	X	X				X
Dresden 2	3	X	X	X	X	X	X			X	X
Dresden 3	3	X	X	X	X	X	X			X	X
Browns Ferry 1, 2, 3	4	X	X	X	X	T			X	X	X
Brunswick 1, 2	4	X	X	X	X	T			X	X	X
Cooper	4	X	X	X	X	T			X	X	X
Duane Arnold		X	X	X	X	M			X		X
Fermi 2		X	X	X	X				X	X	X
Fitzpatrick	4	X	X	X	X	T			X	X	X
Grand Gulf 1	5	X	X	X	X	T			X		X
Hatch 1, 2	4	X	X	X	X	T			X	X	X
Hope Creek 1											
LaSalle 1, 2	5	X	X	X	X	2-T, 1-M			X		X
Limerick 1	4	X	X	X	X	T			X	X	X
Monticello	3	X	X	X	X	M			X	X	X
Peach Bottom 2, 3	4	X	X	X	X	T			X	X	X
Perry 1		X	X	X	X	2-T, 1-M			X		X
Pilgrim 1		X	X	X	X				X	X	X
Quad Cities 1, 2	3	X	X	X	X	M			X		X
River Bend 1		X	X	X	X				X	X	X
Shoreham	4	X	X	X	X				X		X
Susquehanna 1, 2		X	X	X	X				X	X	X
Vermont Yankee		X	X	X	X				X		X
Washington Public Power 2		X	X	X	X	T			X		X
LaCrosse		X	X	X	X	X					

nt classes

System													
DS	HPCS	ADS	LPCI	LPCS	COND	RHR (LPCI)	RHRSW	SDC	CC	RHR (SDC)	RHR (SPCOOL)	CI&V	Class
		X		X				X	X				A
		X	X	X				X	X				A
		X		X				X	X				A
		X		X				X	X				A
		X	X	X				X	X				B
		X	X	X				X	X				B
		X	X	X	X	X			X				C
		X	X	X	X	X			X				C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
	X	X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
	X	X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
	X	X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
		X	X	X	X	X			X	X	X		C
	X	X	X	X	X	X			X	X	X		C
													Unique

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Table A.2. PWR plant classes[a]

Plant	U-tube SG	OTSG	AFWS	ICE COND	ACC	UHI	CS	CSR	CS/CHR	CSR/CHR	CS/LPR	CVCS	CARC	HP/RPR	HP/LPR	HPI	LPI/LPR	LPI/CSS	LPR/CHR	RHR	RHR/LP	LPI	HPI/LPR	LPR/CSR	HPI/CSR	RECIRC	RECIRC/RHR	Class
														Systems														
Beaver Valley 1	X		X		X		X	X		X					X		X			X								A
Millstone 3	X		X		X		X	X		X					X		X			X								A
North Anna 1, 2	X		X		X		X	X		X					X		X			X								A
Surry 1, 2	X		X		X		X	X		X					X		X			X								A
Byron 1	X		X		X		X				X		X		X		X		X		X							B
Callaway 1	X		X		X						X		X		X		X		X		X							B
Diablo Canyon 1, 2	X		X		X						X		X		X		X		X		X							B
Ginna	X		X		X		X				X		X		X		X		X		X							B
Kewaunee	X		X		X		X				X	X	X		X		X		X		X							B
Prairie Island 1, 2	X		X		X		X				X	X	X		X		X		X		X							B
R[illegible]	X		X		X		X				X		X		X		X		X		X							B
Salem 1, 2	X		X		X		X				X		X		X		X		X		X							B
Vogtle 1, 2	X		X		X		X				X		X		X		X		X		X							B
Wolf Creek	X		X		X		X				X		X		X		X		X		X							B
Zion 1, 2	X		X		X		X				X		X		X		X		X		X							B
[illegible]arley 1, 2	X		X		X				X			X	X		X		X		X		X							C
Maine Yankee	X		X		X										X	X		X			X			X				C
Summer	X		X		X				X			X	X		X				X		X							C
Trojan	X		X		X				X			X	X		X				X		X							C
Arkansas Nuclear One, 1		X	X		X				X				X		X		X		X		X							D
Crystal River 3		X	X		X				X				X		X		X		X		X							D
Davis-Besse 1		X	X		X				X			X	X		X		X		X		X							B
Oconee 1, 2, 3		X	X		X				X				X		X		X		X		X							D
Rancho Seco		X	X		X				X				X		X		X		X		X							D
Three Mile Island 1		X	X		X				X				X		X		X		X		X							D

8805260278-02

Table A.2 (c

Plant	U-tube SC	OTSG	AFWS	ICE COND	ACC	UHI	CS	CSR	CS/CHR	CSR/CHR	CS/LPR	CVCS	CARC	HP/HPR
Point Beach 1, 2	X		X		X					X	X	X		
Turkey Point 3, 4	X		X		X					X	X	X		
Catawaba 1, 2	X		X	X	X	X			X	X	X			
Cook 1, 2	X		X	X	X	X			X	X	X			
McGuire 1, 2	X		X	X	X	X			X	X	X			
Sequoyah 1, 2	X		X	X	X	X			X	X	X			
Arkansas Nuclear One, 2	X		X		X				X	X		X	X	X
Calvert Cliffs 1, 2	X		X		X				X	X	X		X	X
Fort Calhoun	X		X		X				X	X		X	X	X
Millstone 2	X		X		X				X	X		X	X	X
Palisades	X		X		X				X	X		X	X	X
Palo Verde 1, 2	X		X		X				X	X		X	X	X
St. Lucie 1, 2	X		X		X				X	X		X	X	X
San Onofre 2, 3	X		X		X				X	X		X	X	X
Haddam Neck	X		X		X								X	
Indian Point 2, 3	X		X		X		X				X	X	X	
San Onofre 1	X		X		X		X							
Yankee Rowe	X		X		X							X	X	

ontinued)

ystem													
HP/LPR	HPI	LPI/LPR	LPI/CSS	LPR/CHR	RHR	RHR/LP	LPI	HPI/LPR	LPR/CSR	HPI/CSR	RECIRC	RECIRC/RHR	Class
X		X		X		X							E
X		X		X		X							E
X		X		X		X							F
X		X		X		y							F
X		X		X		X							F
X		X		X		X							F
						X	X						G
						X	X						G
						X	X						G
						X	X						G
						X	X						G
						X	X						G
						X	X						G
						X	X						G
X		X	X	X		X							Unique
X		X		X		X		X	X				Unique
	X					X					X		Unique
	X						X					X	Unique

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Table A.3. Generic plant data as of September 30, 1986, sorted by plant name

PLANT IS ARKANSAS UNIT 1
DOCKET IS 313
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 850 MWE
CORE THERMAL POWER IS 2568 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS ARKANSAS POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 6 MILES WNW of RUSSELLVILLE, AR
INITIAL CRITICALITY DATE IS AUGUST 6, 1974
COMMERCIAL OPERATING DATE IS DECEMBER 19, 1974
NRC REGION IS 4

PLANT IS ARKANSAS UNIT 2
DOCKET IS 368
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 912 MWE
CORE THERMAL POWER IS 2815 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS ARKANSAS POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 6 MILES WNW of RUSSELLVILLE, AR
INITIAL CRITICALITY DATE IS DECEMBER 5, 1978
COMMERCIAL OPERATING DATE IS MARCH 26, 1980
NRC REGION IS 4

PLANT IS BEAVER VALLEY UNIT 1
DOCKET IS 334
REACTOR TYPE IS PWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 835 MWE
CORE THERMAL POWER IS 2652 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS DUQUESNE LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 30 MILES NW of PITTSBURGH, PA
INITIAL CRITICALITY DATE IS MAY 10, 1976
COMMERCIAL OPERATING DATE IS OCTOBER 1, 1976
NRC REGION IS 1

Table A.3 (continued)

PLANT IS BEAVER VALLEY UNIT 2
DOCKET IS 412
REACTOR TYPE IS PWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 833 MWE
CORE THERMAL POWER IS 2660 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS DUQUESNE LIGHT COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 5 MILES E of E. LIVERPOOL, OH, PA
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 1

PLANT IS BIG ROCK POINT
DOCKET IS 155
REACTOR TYPE IS BWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 72 MWE
CORE THERMAL POWER IS 240 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS CONSUMERS POWER COMPANY
CONTAINMENT TYPE IS 1
PLANT LOCATION IS 4 MILES NE of CHARLEVOIX, MI
INITIAL CRITICALITY DATE IS SEPTEMBER 27, 1962
COMMERCIAL OPERATING DATE IS MARCH 29, 1963
NRC REGION IS 3

PLANT IS BRAIDWOOD UNIT 1
DOCKET IS 456
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1120 MWE
CORE THERMAL POWER IS 3425 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 24 MILES SSW of JOLIET, IL
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

Table A.3 (continued)

PLANT IS BRAIDWOOD UNIT 2
DOCKET IS 457
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1120 MWE
CORE THERMAL POWER IS 3425 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 24 MILES SSW of JOLIET, IL
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

PLANT IS BROWNS FERRY UNIT 1
DOCKET IS 259
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1065 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS TENNESSEE VALLEY AUTHORITY
PLANT OPERATOR IS TENNESSEE VALLEY AUTHORITY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 10 MILES NW of DECATUR, AL
INITIAL CRITICALITY DATE IS AUGUST 17, 1973
COMMERCIAL OPERATING DATE IS AUGUST 1, 1974
NRC REGION IS 2

PLANT IS BROWNS FERRY UNIT 2
DOCKET IS 260
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1065 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS TENNESSEE VALLEY AUTHORITY
PLANT OPERATOR IS TENNESSEE VALLEY AUTHORITY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 10 MILES NW of DECATUR, AL
INITIAL CRITICALITY DATE IS JULY 20, 1974
COMMERCIAL OPERATING DATE IS MARCH 1, 1975
NRC REGION IS 2

Table A.3 (continued)

PLANT IS BROWNS FERRY UNIT 3
DOCKET IS 296
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1065 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS TENNESSEE VALLEY AUTHORITY
PLANT OPERATOR IS TENNESSEE VALLEY AUTHORITY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 10 MILES NW of DECATUR, AL
INITIAL CRITICALITY DATE IS AUGUST 8, 1976
COMMERCIAL OPERATING DATE IS MARCH 1, 1977
NRC REGION IS 2

PLANT IS BRUNSWICK UNIT 1
DOCKET IS 325
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 821 MWE
CORE THERMAL POWER IS 2436 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS UNITED ENGINEERS AND CONTRACTORS
PLANT OPERATOR IS CAROLINA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 3 MILES N of SOUTHPORT, NC
INITIAL CRITICALITY DATE IS OCTOBER 8, 1976
COMMERCIAL OPERATING DATE IS MARCH 18, 1977
NRC REGION IS 2

PLANT IS BRUNSWICK UNIT 2
DOCKET IS 324
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 821 MWE
CORE THERMAL POWER IS 2436 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS UNITED ENGINEERS AND CONTRACTORS
PLANT OPERATOR IS CAROLINA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 3 MILES N of SOUTHPORT, NC
INITIAL CRITICALITY DATE IS MARCH 20, 1975
COMMERCIAL OPERATING DATE IS NOVEMBER 3, 1975
NRC REGION IS 2

PLANT IS BYRON UNIT 1
DOCKET IS 454
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1120 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 17 MILES SW of ROCKFORD, IL
INITIAL CRITICALITY DATE IS FEBRUARY 2, 1985
COMMERCIAL OPERATING DATE IS SEPTEMBER 16, 1985
NRC REGION IS 3

PLANT IS BYRON UNIT 2
DOCKET IS 455
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1120 MWE
CORE THERMAL POWER IS 3425 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 17 MILES SW of ROCKFORD, IL
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

PLANT IS CALLAWAY UNIT 1
DOCKET IS 483
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1171 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS UNION ELECTRIC COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 10 MILES SE of FULTON, MO
INITIAL CRITICALITY DATE IS OCTOBER 2, 1984
COMMERCIAL OPERATING DATE IS DECEMBER 19, 1984
NRC REGION IS 3

Table A.3 (continued)

PLANT IS CALLAWAY UNIT 2
DOCKET IS 486
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1120 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS UNION ELECTRIC COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 35 MILES WNW of COLUMBIA, MO
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

PLANT IS CALVERT CLIFFS UNIT 1
DOCKET IS 317
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 845 MWE
CORE THERMAL POWER IS 2700 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS BALTIMORE GAS & ELECTRIC
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 40 MILES S of ANNAPOLIS, MD
INITIAL CRITICALITY DATE IS OCTOBER 7, 1974
COMMERCIAL OPERATING DATE IS MAY 8, 1975
NRC REGION IS 1

PLANT IS CALVERT CLIFFS UNIT 2
DOCKET IS 318
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 845 MWE
CORE THERMAL POWER IS 2700 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS BALTIMORE GAS & ELECTRIC
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 40 MILES S of ANNAPOLIS, MD
INITIAL CRITICALITY DATE IS NOVEMBER 30, 1976
COMMERCIAL OPERATING DATE IS APRIL 1, 1977
NRC REGION IS 1

Table A.3 (continued)

PLANT IS CATAWBA UNIT 1
DOCKET IS 413
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1145 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS DUKE POWER COMPANY
PLANT OPERATOR IS DUKE POWER COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 6 MILES NNW of ROCK HILL, SC
INITIAL CRITICALITY DATE IS JANUARY 7, 1985
COMMERCIAL OPERATING DATE IS JUNE 29, 1985
NRC REGION IS 2

PLANT IS CATAWBA UNIT 2
DOCKET IS 414
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1145 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS DUKE POWER COMPANY
PLANT OPERATOR IS DUKE POWER COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 6 MILES NNW of ROCK HILL, SC
INITIAL CRITICALITY DATE IS MAY 8, 1986
COMMERCIAL OPERATING DATE IS AUGUST 30, 1986
NRC REGION IS 2

PLANT IS CLINTON UNIT 1
DOCKET IS 461
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 933 MWE
CORE THERMAL POWER IS 2894 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS ILLINOIS POWER COMPANY
CONTAINMENT TYPE IS 5AE
PLANT LOCATION IS 6 MILES E of CLINTON, IL
INITIAL CRITICALITY DATE IS FEBRUARY 27, 1987
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

Table A.3 (continued)

PLANT IS CLINTON UNIT 2
DOCKET IS 462
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 933 MWE
CORE THERMAL POWER IS 2894 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS ILLINOIS POWER COMPANY
CONTAINMENT TYPE IS 5AE
PLANT LOCATION IS 6 MILES E of CLINTON, IL
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

PLANT IS COMANCHE PEAK UNIT 1
DOCKET IS 445
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1150 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS GIBBS AND HILL
PLANT OPERATOR IS TEXAS UTILITIES GENERATING COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 4 MILES N of GLEN ROSE, TX
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 4

PLANT IS COMANCHE PEAK UNIT 2
DOCKET IS 446
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1150 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS GIBBS AND HILL
PLANT OPERATOR IS TEXAS UTILITIES GENERATING COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 4 MILES N of GLEN ROSE, TX
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 4

Table A.3 (continued)

PLANT IS COOK UNIT 1
DOCKET IS 315
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1030 MWE
CORE THERMAL POWER IS 3250 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS AMERICAN ELECTRIC POWER SERVICES COMPANY
PLANT OPERATOR IS INDIANA & MICHIGAN ELECTRIC COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 11 MILES S of BENTON HARBOR, MI
INITIAL CRITICALITY DATE IS JANUARY 18, 1975
COMMERCIAL OPERATING DATE IS AUGUST 27, 1975
NRC REGION IS 3

PLANT IS COOK UNIT 2
DOCKET IS 316
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1100 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS AMERICAN ELECTRIC POWER SERVICES COMPANY
PLANT OPERATOR IS INDIANA & MICHIGAN ELECTRIC COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 11 MILES S of BENTON HARBOR, MI
INITIAL CRITICALITY DATE IS MARCH 10, 1978
COMMERCIAL OPERATING DATE IS JULY 1, 1978
NRC REGION IS 3

PLANT IS COOPER STATION
DOCKET IS 298
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 778 MWE
CORE THERMAL POWER IS 2381 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BURNS AND ROE
PLANT OPERATOR IS NEBRASKA PUBLIC POWER DISTRICT
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 23 MILES S of NEBRASKA CITY, NE
INITIAL CRITICALITY DATE IS FEBRUARY 21, 1974
COMMERCIAL OPERATING DATE IS JULY 1, 1974
NRC REGION IS 4

Table A.3 (continued)

PLANT IS CRYSTAL RIVER UNIT 3
DOCKET IS 302
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 825 MWE
CORE THERMAL POWER IS 2544 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS GILBERT ASSOCIATES
PLANT OPERATOR IS FLORIDA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 7 MILES NW of CRYSTAL RIVER, FL
INITIAL CRITICALITY DATE IS JANUARY 14, 1977
COMMERCIAL OPERATING DATE IS MARCH 13, 1977
NRC REGION IS 2

PLANT IS DAVIS-BESSE UNIT 1
DOCKET IS 346
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 906 MWE
CORE THERMAL POWER IS 2772 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS TOLEDO EDISON COMPANY
CONTAINMENT TYPE IS 2A
PLANT LOCATION IS 21 MILES E of TOLEDO, OH
INITIAL CRITICALITY DATE IS AUGUST 12, 1977
COMMERCIAL OPERATING DATE IS JULY 31, 1978
NRC REGION IS 3

PLANT IS DIABLO CANYON UNIT 1
DOCKET IS 275
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1086 MWE
CORE THERMAL POWER IS 3338 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS PACIFIC GAS & ELECTRIC COMPANY
PLANT OPERATOR IS PACIFIC GAS & ELECTRIC COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 12 MILES WSW of SAN LUIS OBISPO, CA
INITIAL CRITICALITY DATE IS APRIL 29, 1984
COMMERCIAL OPERATING DATE IS MAY 7, 1985
NRC REGION IS 5

Table A.3 (continued)

PLANT IS DIABLO CANYON UNIT 2
DOCKET IS 323
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1119 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS PACIFIC GAS & ELECTRIC COMPANY
PLANT OPERATOR IS PACIFIC GAS & ELECTRIC COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 12 MILES WSW of SAN LUIS OBISPO, CA
INITIAL CRITICALITY DATE IS AUGUST 19, 1985
COMMERCIAL OPERATING DATE IS MARCH 13, 1986
NRC REGION IS 5

PLANT IS DRESDEN UNIT 1
DOCKET IS 10
REACTOR TYPE IS BWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 180 MWE
CORE THERMAL POWER IS 700 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 1A
PLANT LOCATION IS 47 MILES NE of CHICAGO, IL
INITIAL CRITICALITY DATE IS OCTOBER 31, 1959
COMMERCIAL OPERATING DATE IS AUGUST 1, 1960
NRC REGION IS 3

PLANT IS DRESDEN UNIT 2
DOCKET IS 237
REACTOR TYPE IS BWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 794 MWE
CORE THERMAL POWER IS 2527 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 9 MILES E of MORRIS, IL
INITIAL CRITICALITY DATE IS JANUARY 7, 1970
COMMERCIAL OPERATING DATE IS JUNE 9, 1970
NRC REGION IS 3

Table A.3 (continued)

PLANT IS DRESDEN UNIT 3
DOCKET IS 249
REACTOR TYPE IS BWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 794 MWE
CORE THERMAL POWER IS 2527 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 9 MILES E of MORRIS, IL
INITIAL CRITICALITY DATE IS JANUARY 31, 1971
COMMERCIAL OPERATING DATE IS NOVEMBER 16, 1971
NRC REGION IS 3

PLANT IS DUANE ARNOLD
DOCKET IS 331
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 538 MWE
CORE THERMAL POWER IS 1658 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS IOWA ELECTRIC LIGHT & POWER COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 8 MILES NW of CEDAR RAPIDS, IA
INITIAL CRITICALITY DATE IS MARCH 23, 1974
COMMERCIAL OPERATING DATE IS FEBRUARY 1, 1975
NRC REGION IS 3

PLANT IS ENRICO FERMI UNIT 2
DOCKET IS 341
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1093 MWE
CORE THERMAL POWER IS 3292 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS DETROIT EDISON COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS LAGUNA BEACH, MI
INITIAL CRITICALITY DATE IS JUNE 21, 1985
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

Table A.3 (continued)

PLANT IS FARLEY UNIT 1
DOCKET IS 348
REACTOR TYPE IS PWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 829 MWE
CORE THERMAL POWER IS 2652 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SOUTHERN SERVICES, INCORPORATED
PLANT OPERATOR IS ALABAMA POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 18 MILES SE of DOTHAN, AL
INITIAL CRITICALITY DATE IS AUGUST 9, 1977
COMMERCIAL OPERATING DATE IS DECEMBER 1, 1977
NRC REGION IS 2

PLANT IS FARLEY UNIT 2
DOCKET IS 364
REACTOR TYPE IS PWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 829 MWE
CORE THERMAL POWER IS 2652 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SOUTHERN SERVICES, INCORPORATED
PLANT OPERATOR IS ALABAMA POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 18 MILES SE of DOTHAN, AL
INITIAL CRITICALITY DATE IS MAY 5, 1981
COMMERCIAL OPERATING DATE IS JULY 30, 1981
NRC REGION IS 2

PLANT IS FITZPATRICK
DOCKET IS 333
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 821 MWE
CORE THERMAL POWER IS 2436 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS POWER AUTHORITY OF THE STATE OF NEW YORK
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 8 MILES NE of OSWEGO, NY
INITIAL CRITICALITY DATE IS NOVEMBER 17, 1974
COMMERCIAL OPERATING DATE IS JULY 28, 1975
NRC REGION IS 1

Table A.3 (continued)

PLANT IS FORT CALHOUN UNIT 1
DOCKET IS 285
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 478 MWE
CORE THERMAL POWER IS 1500 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS GIBBS AND HILL
PLANT OPERATOR IS OMAHA PUBLIC POWER DISTRICT
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 19 MILES N of OMAHA, NB
INITIAL CRITICALITY DATE IS AUGUST 6, 1973
COMMERCIAL OPERATING DATE IS JUNE 20, 1974
NRC REGION IS 4

PLANT IS GINNA
DOCKET IS 244
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 470 MWE
CORE THERMAL POWER IS 1520 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS GILBERT ASSOCIATES
PLANT OPERATOR IS ROCHESTER GAS & ELECTRIC CORPORATION
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 15 MILES NE of ROCHESTER, NY
INITIAL CRITICALITY DATE IS NOVEMBER 8, 1969
COMMERCIAL OPERATING DATE IS JULY 1, 1970
NRC REGION IS 1

PLANT IS GRAND GULF UNIT 1
DOCKET IS 416
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1250 MWE
CORE THERMAL POWER IS 3833 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS MISSISSIPPI POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 5AE
PLANT LOCATION IS 25 MILES S of VICKSBURG, MS
INITIAL CRITICALITY DATE IS AUGUST 18, 1982
COMMERCIAL OPERATING DATE IS JULY 1, 1985
NRC REGION IS 2

Table A.3 (continued)

PLANT IS HADDAM NECK
DOCKET IS 213
REACTOR TYPE IS PWR
REACTOR CLASS IS H
DESIGN ELECTRICAL RATING IS 582 MWE
CORE THERMAL POWER IS 1825 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS CONNECTICUT YANKEE ATOMIC POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 13 MILES E of MERIDEN, CT
INITIAL CRITICALITY DATE IS JULY 24, 1967
COMMERCIAL OPERATING DATE IS JANUARY 1, 1968
NRC REGION IS 1

PLANT IS HARRIS UNIT 1
DOCKET IS 400
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 900 MWE
CORE THERMAL POWER IS 2775 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS EBASCO SERVICES, INCORPORATED
PLANT OPERATOR IS CAROLINA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 20 MILES SW of RALEIGH, NC
INITIAL CRITICALITY DATE IS JANUARY 3, 1987
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 2

PLANT IS HATCH UNIT 1
DOCKET IS 321
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 777 MWE
CORE THERMAL POWER IS 2436 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SOUTHERN SERVICES, INCORPORATED
PLANT OPERATOR IS GEORGIA POWER COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 11 MILES N of BAXLEY, GA
INITIAL CRITICALITY DATE IS SEPTEMBER 12, 1974
COMMERCIAL OPERATING DATE IS DECEMBER 31, 1975
NRC REGION IS 2

Table A.3 (continued)

PLANT IS HATCH UNIT 2
DOCKET IS 366
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 784 MWE
CORE THERMAL POWER IS 2436 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SOUTHERN SERVICES, INCORPORATED
PLANT OPERATOR IS GEORGIA POWER COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 11 MILES N of BAXLEY, GA
INITIAL CRITICALITY DATE IS JULY 4, 1978
COMMERCIAL OPERATING DATE IS SEPTEMBER 5, 1979
NRC REGION IS 2

PLANT IS HOPE CREEK UNIT 1
DOCKET IS 354
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1067 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PUBLIC SERVICE ELECTRIC & GAS COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 10 MILES SW of SALEM, NJ
INITIAL CRITICALITY DATE IS JUNE 28, 1986
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 1

PLANT IS HUMBOLDT BAY
DOCKET IS 133
REACTOR TYPE IS BWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 63 MWE
CORE THERMAL POWER IS 163 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PACIFIC GAS & ELECTRIC COMPANY
CONTAINMENT TYPE IS C
PLANT LOCATION IS EUREKA, CA
INITIAL CRITICALITY DATE IS FEBRUARY 16, 1963
COMMERCIAL OPERATING DATE IS JANUARY 8, 1963
NRC REGION IS 5

Table A.3 (continued)

PLANT IS INDIAN POINT UNIT 1
DOCKET IS 3
REACTOR TYPE IS PWR
REACTOR CLASS IS ** UNKNOWN **
DESIGN ELECTRICAL RATING IS 265 MWE
CORE THERMAL POWER IS 585 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS
PLANT OPERATOR IS CONSOLIDATED EDISON COMPANY
CONTAINMENT TYPE IS 1A
PLANT LOCATION IS 25 MILES S of NEW YORK CITY, NY
INITIAL CRITICALITY DATE IS AUGUST 2, 1962
COMMERCIAL OPERATING DATE IS JANUARY 31, 1963
NRC REGION IS 1

PLANT IS INDIAN POINT UNIT 2
DOCKET IS 247
REACTOR TYPE IS PWR
REACTOR CLASS IS H
DESIGN ELECTRICAL RATING IS 873 MWE
CORE THERMAL POWER IS 2758 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS UNITED ENGINEERS AND CONTRACTORS
PLANT OPERATOR IS CONSOLIDATED EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 25 MILES N of NEW YORK CITY, NY
INITIAL CRITICALITY DATE IS MAY 22, 1973
COMMERCIAL OPERATING DATE IS AUGUST 1, 1974
NRC REGION IS 1

PLANT IS INDIAN POINT UNIT 3
DOCKET IS 286
REACTOR TYPE IS PWR
REACTOR CLASS IS H
DESIGN ELECTRICAL RATING IS 965 MWE
CORE THERMAL POWER IS 3025 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS UNITED ENGINEERS AND CONTRACTORS
PLANT OPERATOR IS POWER AUTHORITY OF THE STATE OF NEW YORK
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 25 MILES N of NEW YORK CITY, NY
INITIAL CRITICALITY DATE IS APRIL 6, 1976
COMMERCIAL OPERATING DATE IS AUGUST 30, 1976
NRC REGION IS 1

Table A.3 (continued)

PLANT IS KEWAUNEE
DOCKET IS 305
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 535 MWE
CORE THERMAL POWER IS 1650 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS PIONEER SERVICES
PLANT OPERATOR IS WISCONSIN PUBLIC SERVICE CORPORATION
CONTAINMENT TYPE IS 2A
PLANT LOCATION IS 27 MILES SE of GREEN BAY, WI
INITIAL CRITICALITY DATE IS MARCH 7, 1974
COMMERCIAL OPERATING DATE IS JUNE 16, 1974
NRC REGION IS 3

PLANT IS LA CROSSE
DOCKET IS 409
REACTOR TYPE IS BWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 50 MWE
CORE THERMAL POWER IS 165 MWT
PLANT VENDOR IS ALLIS CHALMERS
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS DAIRYLAND POWER COOPERATIVE
CONTAINMENT TYPE IS 2
PLANT LOCATION IS 19 MILES S of LA CROSSE, WI
INITIAL CRITICALITY DATE IS JULY 11, 1967
COMMERCIAL OPERATING DATE IS NOVEMBER 1, 1969
NRC REGION IS 3

PLANT IS LA SALLE UNIT 1
DOCKET IS 373
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1078 MWE
CORE THERMAL POWER IS 3323 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 11 MILES SE of OTTAWA, IL
INITIAL CRITICALITY DATE IS JUNE 21, 1982
COMMERCIAL OPERATING DATE IS JANUARY 1, 1984
NRC REGION IS 3

Table A.3 (continued)

PLANT IS LA SALLE UNIT 2
DOCKET IS 374
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1078 MWE
CORE THERMAL POWER IS 3323 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 11 MILES SE of OTTAWA, IL
INITIAL CRITICALITY DATE IS MARCH 10, 1984
COMMERCIAL OPERATING DATE IS OCTOBER 19, 1984
NRC REGION IS 3

PLANT IS LIMERICK UNIT 1
DOCKET IS 352
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1055 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PHILADELPHIA ELECTRIC COMPANY
CONTAINMENT TYPE IS 5AC
PLANT LOCATION IS 21 MILES NW of PHILADELPHIA, PA
INITIAL CRITICALITY DATE IS DECEMBER 22, 1984
COMMERCIAL OPERATING DATE IS FEBRUARY 1, 1986
NRC REGION IS 1

PLANT IS MAINE YANKEE
DOCKET IS 309
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 825 MWE
CORE THERMAL POWER IS 2630 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS MAINE YANKEE ATOMIC POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 10 MILES N of BATH, ME
INITIAL CRITICALITY DATE IS OCTOBER 23, 1972
COMMERCIAL OPERATING DATE IS DECEMBER 28, 1972
NRC REGION IS 1

Table A.3 (continued)

PLANT IS MC GUIRE UNIT 1
DOCKET IS 369
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1180 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS DUKE POWER COMPANY
PLANT OPERATOR IS DUKE POWER COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 17 MILES N of CHARLOTTE, NC
INITIAL CRITICALITY DATE IS AUGUST 8, 1981
COMMERCIAL OPERATING DATE IS DECEMBER 1, 1981
NRC REGION IS 2

PLANT IS MC GUIRE UNIT 2
DOCKET IS 370
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1180 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS DUKE POWER COMPANY
PLANT OPERATOR IS DUKE POWER COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 17 MILES N of CHARLOTTE, NC
INITIAL CRITICALITY DATE IS MAY 8, 1983
COMMERCIAL OPERATING DATE IS MARCH 1, 1984
NRC REGION IS 2

PLANT IS MILLSTONE POINT UNIT 1
DOCKET IS 245
REACTOR TYPE IS BWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 660 MWE
CORE THERMAL POWER IS 2011 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS EBASCO SERVICES, INCORPORATED
PLANT OPERATOR IS NORTHEAST NUCLEAR ENERGY COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 5 MILES SW of NEW LONDON, CT
INITIAL CRITICALITY DATE IS OCTOBER 26, 1970
COMMERCIAL OPERATING DATE IS MARCH 1, 1971
NRC REGION IS 1

Table A.3 (continued)

PLANT IS MILLSTONE POINT UNIT 2
DOCKET IS 336
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 870 MWE
CORE THERMAL POWER IS 2700 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS NORTHEAST NUCLEAR ENERGY COMPANY
CONTAINMENT TYPE IS 3B
PLANT LOCATION IS 5 MILES SW of NEW LONDON, CT
INITIAL CRITICALITY DATE IS OCTOBER 17, 1975
COMMERCIAL OPERATING DATE IS DECEMBER 26, 1975
NRC REGION IS 1

PLANT IS MILLSTONE POINT UNIT 3
DOCKET IS 423
REACTOR TYPE IS PWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 1154 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS NORTHEAST NUCLEAR ENERGY COMPANY
CONTAINMENT TYPE IS 3BC
PLANT LOCATION IS 3.2 MILES WSW of NEW LONDON, CT
INITIAL CRITICALITY DATE IS JANUARY 23, 1986
COMMERCIAL OPERATING DATE IS APRIL 23, 1986
NRC REGION IS 1

PLANT IS MONTICELLO
DOCKET IS 263
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 545 MWE
CORE THERMAL POWER IS 1670 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS NORTHERN STATES POWER COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 30 MILES NW of MINNEAPOLIS, MN
INITIAL CRITICALITY DATE IS DECEMBER 10, 1970
COMMERCIAL OPERATING DATE IS JUNE 30, 1971
NRC REGION IS 3

Table A.3 (continued)

PLANT IS NINE MILE POINT UNIT 1
DOCKET IS 220
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 620 MWE
CORE THERMAL POWER IS 1850 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS NIAGARA MOHAWK POWER COMPANY
PLANT OPERATOR IS NIAGARA MOHAWK POWER COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 8 MILES NE of OSWEGO, NY
INITIAL CRITICALITY DATE IS SEPTEMBER 15, 1969
COMMERCIAL OPERATING DATE IS DECEMBER 1, 1969
NRC REGION IS 1

PLANT IS NINE MILE POINT UNIT 2
DOCKET IS 410
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1090 MWE
CORE THERMAL POWER IS 3323 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS NI[illegible]RA MOHAWK POWER COMPANY
CONTAINMENT TYPE IS [illegible]
PLANT LOCATION IS 8 MILES NE of OSWEGO, NY
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 1

PLANT IS NORTH ANNA UNIT 1
DOCKET IS 338
REACTOR TYPE IS PWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 907 MWE
CORE THERMAL POWER IS 2775 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS VIRGINIA ELECTRIC & POWER CORPORATION
CONTAINMENT TYPE IS 3D
PLANT LOCATION IS 40 MILES NW of RICHMOND, VA
INITIAL CRITICALITY DATE IS APRIL 5, 1978
COMMERCIAL OPERATING DATE IS JUNE 6, 1978
NRC REGION IS 2

Table A.3 (continued)

PLANT IS NORTH ANNA UNIT 2
DOCKET IS 339
REACTOR TYPE IS PWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 907 MWE
CORE THERMAL POWER IS 2275 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS VIRGINIA ELECTRIC & POWER CORPORATION
CONTAINMENT TYPE IS 3D
PLANT LOCATION IS 40 MILES NW of RICHMOND, VA
INITIAL CRITICALITY DATE IS JUNE 12, 1980
COMMERCIAL OPERATING DATE IS DECEMBER 14, 1980
NRC REGION IS 2

PLANT IS OCONEE UNIT 1
DOCKET IS 269
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 887 MWE
CORE THERMAL POWER IS 2568 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS DUKE POWER COMPANY
PLANT OPERATOR IS DUKE POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 26 MILES W of GREENVILLE, SC
INITIAL CRITICALITY DATE IS APRIL 19, 1973
COMMERCIAL OPERATING DATE IS JULY 15, 1973
NRC REGION IS 2

PLANT IS OCONEE UNIT 2
DOCKET IS 270
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 887 MWE
CORE THERMAL POWER IS 2568 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS DUKE POWER COMPANY
PLANT OPERATOR IS DUKE POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 26 MILES W of GREENVILLE, SC
INITIAL CRITICALITY DATE IS NOVEMBER 11, 1973
COMMERCIAL OPERATING DATE IS SEPTEMBER 9, 1974
NRC REGION IS 2

Table A.3 (continued)

PLANT IS OCONEE UNIT 3
DOCKET IS 287
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 887 MWE
CORE THERMAL POWER IS 2568 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS DUKE POWER COMPANY
PLANT OPERATOR IS DUKE POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 26 MILES W of GREENVILLE, SC
INITIAL CRITICALITY DATE IS SEPTEMBER 5, 1974
COMMERCIAL OPERATING DATE IS DECEMBER 16, 1974
NRC REGION IS 2

PLANT IS OYSTER CREEK
DOCKET IS 219
REACTOR TYPE IS BWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 650 MWE
CORE THERMAL POWER IS 1930 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BURNS AND ROE
PLANT OPERATOR IS JERSEY CENTRAL POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 9 MILES S of TOMS RIVER, NJ
INITIAL CRITICALITY DATE IS MAY 3, 1969
COMMERCIAL OPERATING DATE IS DECEMBER 1, 1969
NRC REGION IS 1

PLANT IS PALISADES
DOCKET IS 255
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 805 MWE
CORE THERMAL POWER IS 2530 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS CONSUMERS POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 5 MILES S of SOUTH HAVEN, MI
INITIAL CRITICALITY DATE IS MAY 4, 1971
COMMERCIAL OPERATING DATE IS DECEMBER 31, 1971
NRC REGION IS 1

Table A.3 (continued)

PLANT IS PALO VERDE UNIT 1
DOCKET IS 528
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 1221 MWE
CORE THERMAL POWER IS 3800 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS ARIZONA PUBLIC SERVICE COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 36 MILES W of PHOENIX, AZ
INITIAL CRITICALITY DATE IS MARCH 25, 1985
COMMERCIAL OPERATING DATE IS FEBRUARY 13, 1986
NRC REGION IS 5

PLANT IS PALO VERDE UNIT 2
DOCKET IS 529
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 1221 MWE
CORE THERMAL POWER IS 3800 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS ARIZONA PUBLIC SERVICE COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 36 MILES W of PHOENIX, AZ
INITIAL CRITICALITY DATE IS APRIL 18, 1986
COMMERCIAL OPERATING DATE IS SEPTEMBER 30, 1986
NRC REGION IS 5

PLANT IS PALO VERDE UNIT 3
DOCKET IS 530
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 1250 MWE
CORE THERMAL POWER IS 3800 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS ARIZONA PUBLIC SERVICE COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 36 MILES W of PHOENIX, AZ
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 5

Table A.3 (continued)

PLANT IS PEACH BOTTOM UNIT 2
DOCKET IS 277
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1065 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PHILADELPHIA ELECTRIC COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 19 MILES S of LANCASTER, PA
INITIAL CRITICALITY DATE IS SEPTEMBER 16, 1973
COMMERCIAL OPERATING DATE IS JULY 5, 1974
NRC REGION IS 1

PLANT IS PEACH BOTTOM UNIT 3
DOCKET IS 278
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1065 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PHILADELPHIA ELECTRIC COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 19 MILES S of LANCASTER, PA
INITIAL CRITICALITY DATE IS AUGUST 7, 1974
COMMERCIAL OPERATING DATE IS DECEMBER 23, 1974
NRC REGION IS 1

PLANT IS PERRY UNIT 1
DOCKET IS 440
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1205 MWE
CORE THERMAL POWER IS 3579 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS GILBERT ASSOCIATES
PLANT OPERATOR IS CLEVELAND ELECTRIC ILLUMINATING COMPANY
CONTAINMENT TYPE IS 5AE
PLANT LOCATION IS 7 MILES NE of PAINESVILLE, OH
INITIAL CRITICALITY DATE IS JUNE 10, 1986
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 3

Table A.3 (continued)

PLANT IS PILGRIM UNIT 1
DOCKET IS 293
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 655 MWE
CORE THERMAL POWER IS 1998 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS BOSTON EDISON COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 4 MILES SE of PLYMOUTH, MA
INITIAL CRITICALITY DATE IS JUNE 16, 1972
COMMERCIAL OPERATING DATE IS DECEMBER 1, 1972
NRC REGION IS 1

PLANT IS POINT BEACH UNIT 1
DOCKET IS 266
REACTOR TYPE IS PWR
REACTOR CLASS IS E
DESIGN ELECTRICAL RATING IS 497 MWE
CORE THERMAL POWER IS 1518 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS WISCONSIN-MICHIGAN POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 15 MILES N of MANITOWOC, WI
INITIAL CRITICALITY DATE IS NOVEMBER 2, 1970
COMMERCIAL OPERATING DATE IS DECEMBER 21, 1970
NRC REGION IS 3

PLANT IS POINT BEACH UNIT 2
DOCKET IS 301
REACTOR TYPE IS PWR
REACTOR CLASS IS E
DESIGN ELECTRICAL RATING IS 497 MWE
CORE THERMAL POWER IS 1518 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS WESTINGHOUSE ELECTRIC COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 15 MILES N of MANITOWOC, WI
INITIAL CRITICALITY DATE IS MAY 30, 1972
COMMERCIAL OPERATING DATE IS OCTOBER 1, 1972
NRC REGION IS 3

Table A.3 (continued)

PLANT IS PRAIRIE ISLAND UNIT 1
DOCKET IS 282
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 530 MWE
CORE THERMAL POWER IS 1650 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS
PLANT OPERATOR IS NORTHERN STATES POWER COMPANY
CONTAINMENT TYPE IS 2A
PLANT LOCATION IS 28 MILES SE of MINNEAPOLIS, MN
INITIAL CRITICALITY DATE IS DECEMBER 1, 1973
COMMERCIAL OPERATING DATE IS DECEMBER 16, 1973
NRC REGION IS 3

PLANT IS PRAIRIE ISLAND UNIT 2
DOCKET IS 306
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 530 MWE
CORE THERMAL POWER IS 1650 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS
PLANT OPERATOR IS NORTHERN STATES POWER COMPANY
CONTAINMENT TYPE IS 2A
PLANT LOCATION IS 28 MILES SE of MINNEAPOLIS, MN
INITIAL CRITICALITY DATE IS DECEMBER 17, 1974
COMMERCIAL OPERATING DATE IS DECEMBER 21, 1974
NRC REGION IS 3

PLANT IS QUAD CITIES UNIT 1
DOCKET IS 254
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 789 MWE
CORE THERMAL POWER IS 2511 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 2[illegible] MILES NE of MOLINE, IL
INITIAL CRITICALITY DATE IS OCTOBER 18, 1971
COMMERCIAL OPERATING DATE IS FEBRUARY 18, 1973
NRC REGION IS 3

Table A.3 (continued)

PLANT IS QUAD CITIES UNIT 2
DOCKET IS 265
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 789 MWE
CORE THERMAL POWER IS 2511 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 20 MILES NE of MOLINE, IL
INITIAL CRITICALITY DATE IS APRIL 26, 1972
COMMERCIAL OPERATING DATE IS MARCH 10, 1973
NRC REGION IS 3

PLANT IS RANCHO SECO
DOCKET IS 312
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 918 MWE
CORE THERMAL POWER IS 2772 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS SACRAMENTO MUNICIPAL UTILITY DISTRICT
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 26 MILES SE of SACRAMENTO, CA
INITIAL CRITICALITY DATE IS SEPTEMBER 16, 1974
COMMERCIAL OPERATING DATE IS APRIL 17, 1975
NRC REGION IS 5

PLANT IS RIVER BEND UNIT 1
DOCKET IS 458
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 936 MWE
CORE THERMAL POWER IS 2894 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS GULF STATES UTILITIES
CONTAINMENT TYPE IS 5AE
PLANT LOCATION IS 24 MILES NNW of BATON ROUGE, LA
INITIAL CRITICALITY DATE IS OCTOBER 31, 1985
COMMERCIAL OPERATING DATE IS JUNE 30, 1986
NRC REGION IS 4

Table A.3 (continued)

PLANT IS ROBINSON UNIT 2
DOCKET IS 261
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 700 MWE
CORE THERMAL POWER IS 2300 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS EBASCO SERVICES, INCORPORATED
PLANT OPERATOR IS CAROLINA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 5 MILES NW of HARTSVILLE, SC
INITIAL CRITICALITY DATE IS SEPTEMBER 20, 1970
COMMERCIAL OPERATING DATE IS MARCH 7, 1971
NRC REGION IS 2

PLANT IS SALEM UNIT 1
DOCKET IS 272
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1090 MWE
CORE THERMAL POWER IS 3338 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS PUBLIC SERVICE ELECTRIC & GAS COMPANY
PLANT OPERATOR IS PUBLIC SERVICE ELECTRIC & GAS COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 10 MILES S of SALEM, NJ
INITIAL CRITICALITY DATE IS DECEMBER 11, 1976
COMMERCIAL OPERATING DATE IS JUNE 30, 1977
NRC REGION IS 1

PLANT IS SALEM UNIT 2
DOCKET IS 311
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1115 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS PUBLIC SERVICE ELECTRIC & GAS COMPANY
PLANT OPERATOR IS PUBLIC SERVICE ELECTRIC & GAS COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 10 MILES S of SALEM, NJ
INITIAL CRITICALITY DATE IS AUGUST 8, 1980
COMMERCIAL OPERATING DATE IS OCTOBER 13, 1981
NRC REGION IS 1

Table A.3 (continued)

PLANT IS SAN ONOFRE UNIT 1
DOCKET IS 206
REACTOR TYPE IS PWR
REACTOR CLASS IS H
DESIGN ELECTRICAL RATING IS 436 MWE
CORE THERMAL POWER IS 1347 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS SOUTHER CALIFORNIA EDISON COMPANY
CONTAINMENT TYPE IS 1
PLANT LOCATION IS 5 MILES S of SAN CLEMENTE, CA
INITIAL CRITICALITY DATE IS JUNE 14, 1967
COMMERCIAL OPERATING DATE IS JANUARY 1, 1968
NRC REGION IS 5

PLANT IS SAN ONOFRE UNIT 2
DOCKET IS 361
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 1070 MWE
CORE THERMAL POWER IS 3410 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS SOUTHER CALIFORNIA EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 5 MILES S of SAN CLEMENTE, CA
INITIAL CRITICALITY DATE IS JULY 26, 1982
COMMERCIAL OPERATING DATE IS AUGUST 8, 1983
NRC REGION IS 5

PLANT IS SAN ONOFRE UNIT 3
DOCKET IS 362
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 1080 MWE
CORE THERMAL POWER IS 3310 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS SOUTHER CALIFORNIA EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 5 MILES S of SAN CLEMENTE, CA
INITIAL CRITICALITY DATE IS AUGUST 29, 1983
COMMERCIAL OPERATING DATE IS APRIL 1, 1984
NRC REGION IS 5

Table A.3 (continued)

PLANT IS SEABROOK UNIT 1
DOCKET IS 443
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1200 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS UNITED ENGINEERS AND CONTRACTORS
PLANT OPERATOR IS PUBLIC SERVICE OF NEW HAMPSHIRE
CONTAINMENT TYPE IS 3AC
PLANT LOCATION IS 10 MILES S of PORTSMOUTH, NH
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 1

PLANT IS SEQUOYAH UNIT 1
DOCKET IS 327
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1148 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS TENNESSEE VALLEY AUTHORITY
PLANT OPERATOR IS TENNESSEE VALLEY AUTHORITY
CONTAINMENT TYPE IS 2AC
PLANT LOCATION IS 9.5 MILES NE of CHATTANOOGA, TN
INITIAL CRITICALITY DATE IS JULY 5, 1980
COMMERCIAL OPERATING DATE IS JULY 1, 1981
NRC REGION IS 2

PLANT IS SEQUOYAH UNIT 2
DOCKET IS 328
REACTOR TYPE IS PWR
REACTOR CLASS IS F
DESIGN ELECTRICAL RATING IS 1148 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS TENNESSEE VALLEY AUTHORITY
PLANT OPERATOR IS TENNESSEE VALLEY AUTHORITY
CONTAINMENT TYPE IS 2AC
PLANT LOCATION IS 9.5 MILES NE of CHATTANOOGA, TN
INITIAL CRITICALITY DATE IS NOVEMBER 5, 1981
COMMERCIAL OPERATING DATE IS JUNE 1, 1982
NRC REGION IS 2

Table A.3 (continued)

PLANT IS SHOREHAM
DOCKET IS 322
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 819 MWE
CORE THERMAL POWER IS 2436 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS LONG ISLAND LIGHTING COMPANY
CONTAINMENT TYPE IS 5AC
PLANT LOCATION IS BROOKHAVEN, NY
INITIAL CRITICALITY DATE IS JUNE 1, 1986
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 1

PLANT IS SOUTH TEXAS UNIT 1
DOCKET IS 498
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1250 MWE
CORE THERMAL POWER IS 3800 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BURNS AND ROE
PLANT OPERATOR IS HOUSTON LIGHTING & POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 12 MILES SSW of BAY CITY, TX
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 4

PLANT IS SOUTH TEXAS UNIT 2
DOCKET IS 499
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1250 MWE
CORE THERMAL POWER IS 3800 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BURNS AND ROE
PLANT OPERATOR IS HOUSTON LIGHTING & POWER COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 12 MILES SSW of BAY CITY, TX
INITIAL CRITICALITY DATE IS ** UNKNOWN **
COMMERCIAL OPERATING DATE IS ** UNKNOWN **
NRC REGION IS 4

Table A.3 (continued)

PLANT IS ST LUCIE UNIT 1
DOCKET IS 335
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 830 MWE
CORE THERMAL POWER IS 2700 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS EBASCO SERVICES, INCORPORATED
PLANT OPERATOR IS FLORIDA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 2A
PLANT LOCATION IS 12 MILES SE of FT. PIERCE, FL
INITIAL CRITICALITY DATE IS APRIL 22, 1976
COMMERCIAL OPERATING DATE IS DECEMBER 21, 1976
NRC REGION IS 2

PLANT IS ST LUCIE UNIT 2
DOCKET IS 389
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 830 MWE
CORE THERMAL POWER IS 2700 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS EBASCO SERVICES, INCORPORATED
PLANT OPERATOR IS FLORIDA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 2A
PLANT LOCATION IS 12 MILES SE of FT. PIERCE, FL
INITIAL CRITICALITY DATE IS JUNE 2, 1983
COMMERCIAL OPERATING DATE IS AUGUST 8, 1983
NRC REGION IS 2

PLANT IS SUMMER UNIT 1
DOCKET IS 395
REACTOR TYPE IS PWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 900 MWE
CORE THERMAL POWER IS 2775 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS GILBERT ASSOCIATES
PLANT OPERATOR IS SOUTH CAROLINA ELECTRIC & GAS COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 26 MILES NW of COLUMBIA, SC
INITIAL CRITICALITY DATE IS OCTOBER 22, 1982
COMMERCIAL OPERATING DATE IS JANUARY 1, 1984
NRC REGION IS 2

Table A.3 (continued)

PLANT IS SURRY UNIT 1
DOCKET IS 280
REACTOR TYPE IS PWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 788 MWE
CORE THERMAL POWER IS 2441 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS VIRGINIA ELECTRIC & POWER CORPORATION
CONTAINMENT TYPE IS 3D
PLANT LOCATION IS 17 MILES NW of NEWPORT NEWS, VA
INITIAL CRITICALITY DATE IS JULY 1, 1972
COMMERCIAL OPERATING DATE IS DECEMBER 22, 1972
NRC REGION IS 2

PLANT IS SURRY UNIT 2
DOCKET IS 281
REACTOR TYPE IS PWR
REACTOR CLASS IS A
DESIGN ELECTRICAL RATING IS 788 MWE
CORE THERMAL POWER IS 2441 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS VIRGINIA ELECTRIC & POWER CORPORATION
CONTAINMENT TYPE IS 3D
PLANT LOCATION IS 17 MILES NW of NEWPORT NEWS, VA
INITIAL CRITICALITY DATE IS MARCH 7, 1973
COMMERCIAL OPERATING DATE IS MAY 1, 1973
NRC REGION IS 2

PLANT IS SUSQUEHANNA UNIT 1
DOCKET IS 387
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1065 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PENNSYLVANIA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 7 MILES NE of BERWICK, PA
INITIAL CRITICALITY DATE IS SEPTEMBER 10, 1982
COMMERCIAL OPERATING DATE IS JUNE 8, 1983
NRC REGION IS 1

Table A.3 (continued)

PLANT IS SUSQUEHANNA UNIT 2
DOCKET IS 388
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1065 MWE
CORE THERMAL POWER IS 3293 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PENNSYLVANIA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 5A
PLANT LOCATION IS 7 MILES NE of BERWICK, PA
INITIAL CRITICALITY DATE IS MAY 8, 1984
COMMERCIAL OPERATING DATE IS FEBRUARY 12, 1985
NRC REGION IS 1

PLANT IS THREE MILE ISL UNIT 1
DOCKET IS 289
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 819 MWE
CORE THERMAL POWER IS 2535 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS GILBERT ASSOCIATES
PLANT OPERATOR IS METROPOLITAN EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 10 MILES SE of HARRISBURG, PA
INITIAL CRITICALITY DATE IS JUNE 5, 1974
COMMERCIAL OPERATING DATE IS SEPTEMBER 2, 1974
NRC REGION IS 1

PLANT IS THREE MILE ISL UNIT 2
DOCKET IS 320
REACTOR TYPE IS PWR
REACTOR CLASS IS D
DESIGN ELECTRICAL RATING IS 906 MWE
CORE THERMAL POWER IS 2772 MWT
PLANT VENDOR IS BABCOCK AND WILCOX
ARCHITECT / ENGINEER IS BURNS AND ROE
PLANT OPERATOR IS METROPOLITAN EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 10 MILES SE of HARRISBURG, PA
INITIAL CRITICALITY DATE IS MARCH 28, 1978
COMMERCIAL OPERATING DATE IS DECEMBER 1, 1978
NRC REGION IS 1

Table A.3 (continued)

PLANT IS TROJAN
DOCKET IS 344
REACTOR TYPE IS PWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1130 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS PORTLAND GENERAL ELECTRIC COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 32 MILES NW of PORTLAND, OR
INITIAL CRITICALITY DATE IS DECEMBER 15, 1975
COMMERCIAL OPERATING DATE IS MAY 20, 1976
NRC REGION IS 5

PLANT IS TURKEY POINT UNIT 3
DOCKET IS 250
REACTOR TYPE IS PWR
REACTOR CLASS IS E
DESIGN ELECTRICAL RATING IS 693 MWE
CORE THERMAL POWER IS 2200 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS FLORIDA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 25 MILES S of MIAMI, FL
INITIAL CRITICALITY DATE IS OCTOBER 20, 1972
COMMERCIAL OPERATING DATE IS DECEMBER 14, 1972
NRC REGION IS 2

PLANT IS TURKEY POINT UNIT 4
DOCKET IS 251
REACTOR TYPE IS PWR
REACTOR CLASS IS E
DESIGN ELECTRICAL RATING IS 693 MWE
CORE THERMAL POWER IS 2200 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS FLORIDA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 25 MILES S of MIAMI, FL
INITIAL CRITICALITY DATE IS JUNE 11, 1973
COMMERCIAL OPERATING DATE IS SEPTEMBER 7, 1973
NRC REGION IS 2

Table A.3 (continued)

PLANT IS VERMONT YANKEE
DOCKET IS 271
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 514 MWE
CORE THERMAL POWER IS 1593 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS EBASCO SERVICES, INCORPORATED
PLANT OPERATOR IS VERMONT YANKEE NUCLEAR POWER CORPORATION
CONTAINMENT TYPE IS 4A
PLANT LOCATION IS 5 MILES S of BRATTLEBORO, VT
INITIAL CRITICALITY DATE IS MARCH 24, 1972
COMMERCIAL OPERATING DATE IS NOVEMBER 30, 1972
NRC REGION IS 1

PLANT IS WASHINGTON NP UNIT 2
DOCKET IS 397
REACTOR TYPE IS BWR
REACTOR CLASS IS C
DESIGN ELECTRICAL RATING IS 1100 MWE
CORE THERMAL POWER IS 3323 MWT
PLANT VENDOR IS GENERAL ELECTRIC COMPANY
ARCHITECT / ENGINEER IS BURNS AND ROE
PLANT OPERATOR IS WASHINGTON PUBLIC POWER SUPPLY SYSTEM
CONTAINMENT TYPE IS 4AC
PLANT LOCATION IS 12 MILES NW of RICHLAND, WA
INITIAL CRITICALITY DATE IS JANUARY 19, 1984
COMMERCIAL OPERATING DATE IS DECEMBER 13, 1984
NRC REGION IS 5

PLANT IS WATERFORD UNIT 3
DOCKET IS 382
REACTOR TYPE IS PWR
REACTOR CLASS IS G
DESIGN ELECTRICAL RATING IS 1104 MWE
CORE THERMAL POWER IS 3410 MWT
PLANT VENDOR IS COMBUSTION ENGINEERING
ARCHITECT / ENGINEER IS EBASCO SERVICES, INCORPORATED
PLANT OPERATOR IS LOUISIANA POWER & LIGHT COMPANY
CONTAINMENT TYPE IS 2A
PLANT LOCATION IS 20 MILES W of NEW ORLEANS, LA
INITIAL CRITICALITY DATE IS MARCH 4, 1985
COMMERCIAL OPERATING DATE IS SEPTEMBER 24, 1985
NRC REGION IS 4

PLANT IS WOLF CREEK UNIT 1
DOCKET IS 482
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1170 MWE
CORE THERMAL POWER IS 3411 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS BECHTEL CORPORATION
PLANT OPERATOR IS KANSAS GAS & ELECTRIC COMPANY
CONTAINMENT TYPE IS 3C
PLANT LOCATION IS 3.5 MILES NE of BURLINGTON, KS
INITIAL CRITICALITY DATE IS MAY 22, 1985
COMMERCIAL OPERATING DATE IS SEPTEMBER 3, 1985
NRC REGION IS 4

PLANT IS YANKEE ROWE
DOCKET IS 29
REACTOR TYPE IS PWR
REACTOR CLASS IS H
DESIGN ELECTRICAL RATING IS 175 MWE
CORE THERMAL POWER IS 600 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS STONE AND WEBSTER
PLANT OPERATOR IS YANKEE ATOMIC ELECTRIC COMPANY
CONTAINMENT TYPE IS 1
PLANT LOCATION IS 25 MILES NE of PITTSFIELD, MA
INITIAL CRITICALITY DATE IS AUGUST 19, 1960
COMMERCIAL OPERATING DATE IS JULY 1, 1961
NRC REGION IS 1

PLANT IS ZION UNIT 1
DOCKET IS 295
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1040 MWE
CORE THERMAL POWER IS 3250 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 40 MILES N of CHICAGO, IL
INITIAL CRITICALITY DATE IS JUNE 19, 1973
COMMERCIAL OPERATING DATE IS DECEMBER 31, 1973
NRC REGION IS 3

Table A.3 (continued)

PLANT IS ZION UNIT 2
DOCKET IS 304
REACTOR TYPE IS PWR
REACTOR CLASS IS B
DESIGN ELECTRICAL RATING IS 1040 MWE
CORE THERMAL POWER IS 3250 MWT
PLANT VENDOR IS WESTINGHOUSE ELECTRIC CORPORATION
ARCHITECT / ENGINEER IS SARGENT AND LUNDY
PLANT OPERATOR IS COMMONWEALTH EDISON COMPANY
CONTAINMENT TYPE IS 3
PLANT LOCATION IS 40 MILES N of CHICAGO, IL
INITIAL CRITICALITY DATE IS DECEMBER 24, 1973
COMMERCIAL OPERATING DATE IS SEPTEMBER 17, 1974
NRC REGION IS 3

APPENDIX B

EVENT-TREE MODELS

APPENDIX B

EVENT-TREE MODELS

Event trees were constructed to describe the core-damage mitigation sequences for three initiating events: a nonspecific reactor trip, a LOOP, and a small-break LOCA. These event trees are system based and include an event tree applicable to each plant class defined. Plant classes were defined (Appendix A) that allowed grouping of plants with similar systemic response to these initiating events. A detailed discussion of the development of these trees is included in Appendix B of Ref. 1.

System designs and specific nomenclature may differ among plants included in a particular class, but functionally they are considered to be 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. The specific branch probability estimates used in evaluating the 1986 precursors are listed with the calculations in Appendix D. The development of these estimates is described in Appendix C.

Certain events could not be described using any of the plant class event trees developed. In these cases, unique event trees were developed to describe the sequences of interest.

This appendix presents the event trees used in the study for the three initiating events described above for each of the plant classes. For PWR Classes B, C, E, and F, one set of event trees adequately describes the plant responses to the three initiating events. A slight modification to this same set of event trees also depicts the models applicable to PWR Class D plants. The event trees for the combined group apply to the greatest number of operating PWRs.

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. On the event trees, the sequences that do not result in successful transient mitigation are numbered sequentially, utilizing series 100 numbers for nonspecific reactor trips, series 200 numbers for LOOPs, and series 300 for LOCAs. Abbreviations appearing on the event trees are defined in Table B.1. The trees are presented in the following order:

Table B.1. Abbreviations used in event trees

Abbreviation	Definition
	PWR event trees
AFW	auxiliary feedwater fails
ATWS	anticipated transient without scram end state
CORE DAMAGE	core damage end state
CORE VULN	core vulnerability end state
CSR	containment spray recirculation fails
EP	emergency power fails
HPI	high-pressure injection fails
HPR	high-pressure recirculation fails
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
LPI	low-pressure injection fails
LPR	low-pressure recirculation fails
MFW	main feedwater fails
PORV OPEN	power-operated relief valve fails to open for bleed-and-feed cooling
PORV/SRV CHALL	power-operated relief valve or safety relief valve is 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
SEC SIDE DEPRESS	secondary-side depressurization fails
SEC SIDE REL TERM	secondary-side relief is terminated
SEQ NO	sequence number
TRANS	nonspecific reactor-trip transient
	BWR event trees
CC	containment cooling fails
CI&V	containment injection and venting fail
COND	failure of condensate system
CRD	control-rod-drive cooling fails
EP	emergency power fails
FIRWTR or OTHER	firewater or other equivalent water source fails
FW	unavailability of main feedwater
FWCI	failure of feedwater coolant injection system
FWCI or HPCI	feedwater coolant injection or high-pressure coolant injection fails
HPCI or HPCS	high-pressure coolant injection or high-pressure core spray fails
IC/IC MUP	isolation condenser or isolation condenser makeup fails

Table B.1 (continued)

Abbreviation	Definition
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
LPI	low-pressure injection
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
LPR	low-pressure recirculation
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 (SP MODE)	residual-heat-removal suppression pool cooling mode fails
RHR SW or OTHER	residual-heat-removal service water or other water source fails
RT/LOOP	reactor trip or loss of offsite power
RX SCRAM	reactor fails to scram
SDC	shutdown cooling system fails
SLC/ROD INSERT	standby liquid control system fails or manual rod insertion fails
SRVs/ADS	safety relief valve(s) fail to open for depressurization or automatic depressurization system fails
SRV C	safety relief valve(s) fail to close
SRV CHAL	safety relief valve(s) challenged (challenge rate)
TRANSIENT	nonspecific reactor-trip transient

Figure No.	Event tree
B.1	PWR Class A nonspecific reactor trip
B.2	PWR Class A LOOP
B.3	PWR Class A small LOCA
B.4	PWR Classes B, C, D, E, and F nonspecific reactor trip
B.5	PWR Classes B, C, D, E, and F LOOP
B.6	PWR Classes B, C, D, E, and F small LOCA
B.7	PWR Class G nonspecific reactor trip
B.8	PWR Class G LOOP
B.9	PWR Class G small-break LOCA
B.10	BWR Class A nonspecific reactor trip
B.11	BWR Class A LOOP
B.12	BWR Class A small LOCA
B.13	BWR Class B nonspecific reactor trip
B.14	BWR Class B LOOP
B.15	BWR Class B small-break LOCA
B.16	BWR Class C nonspecific reactor trip
B.17	BWR Class C LOOP
B.18	BWR Class C small LOCA

The event-tree models shown in this appendix are the same ones used in Ref. 1. That report provides a detailed description of the mitigation sequences for each of the three initiators at each plant class (i.e., each event tree) as well as success criteria required for the event-tree branches.

Reference

1. J. W. Minarick, J. D. Harris, P. N. Austin, E. W. Hagen, and J. W. Cletcher, *Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report*, NUREG-4674, Vols. 1 and 2 (ORNL/NOAC-4674/V1 and V2), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1986.

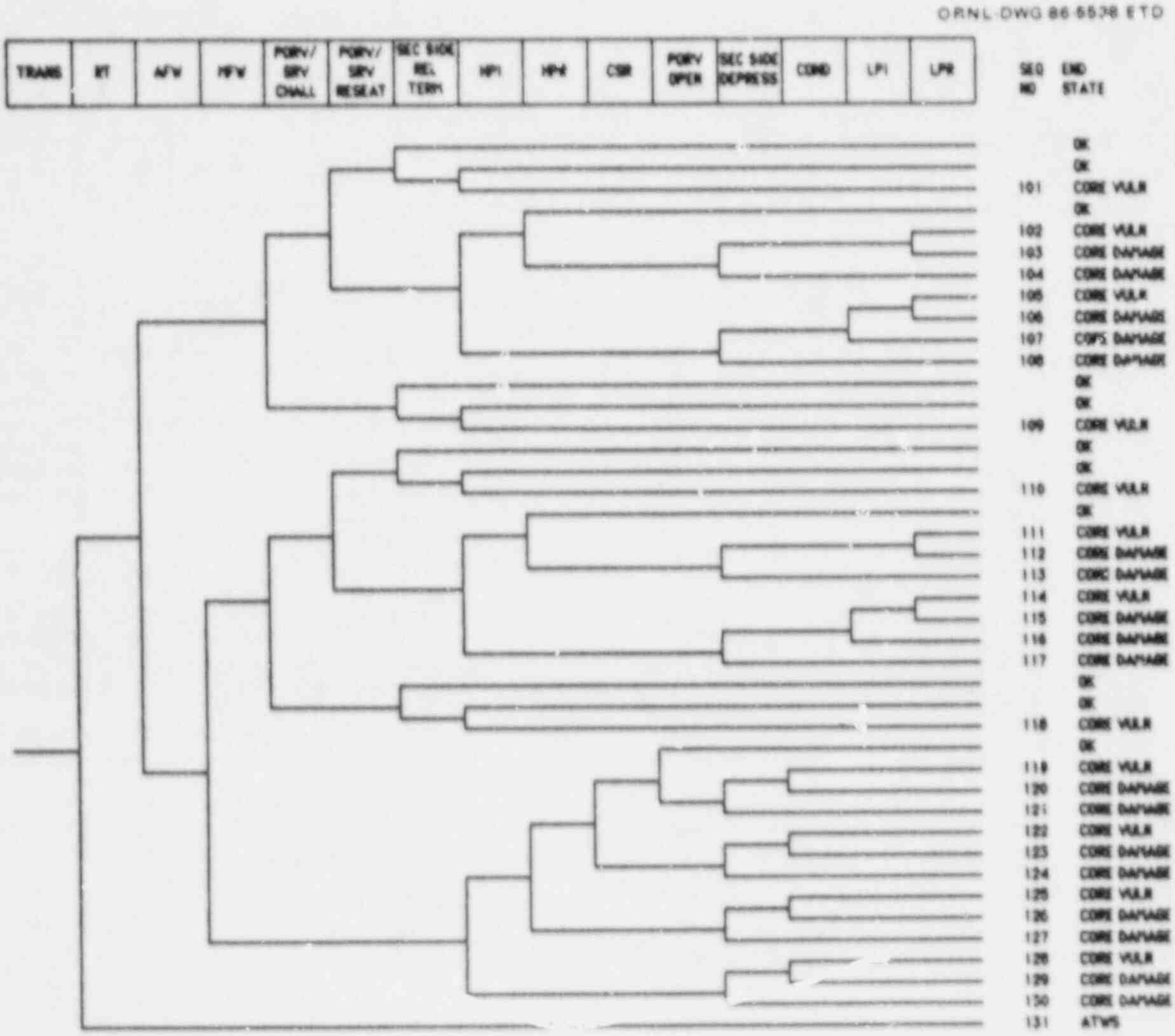


Fig. B.1. PWR Class A nonspecific reactor-trip event tree.

ORNL-DWG 86-5539 ETD

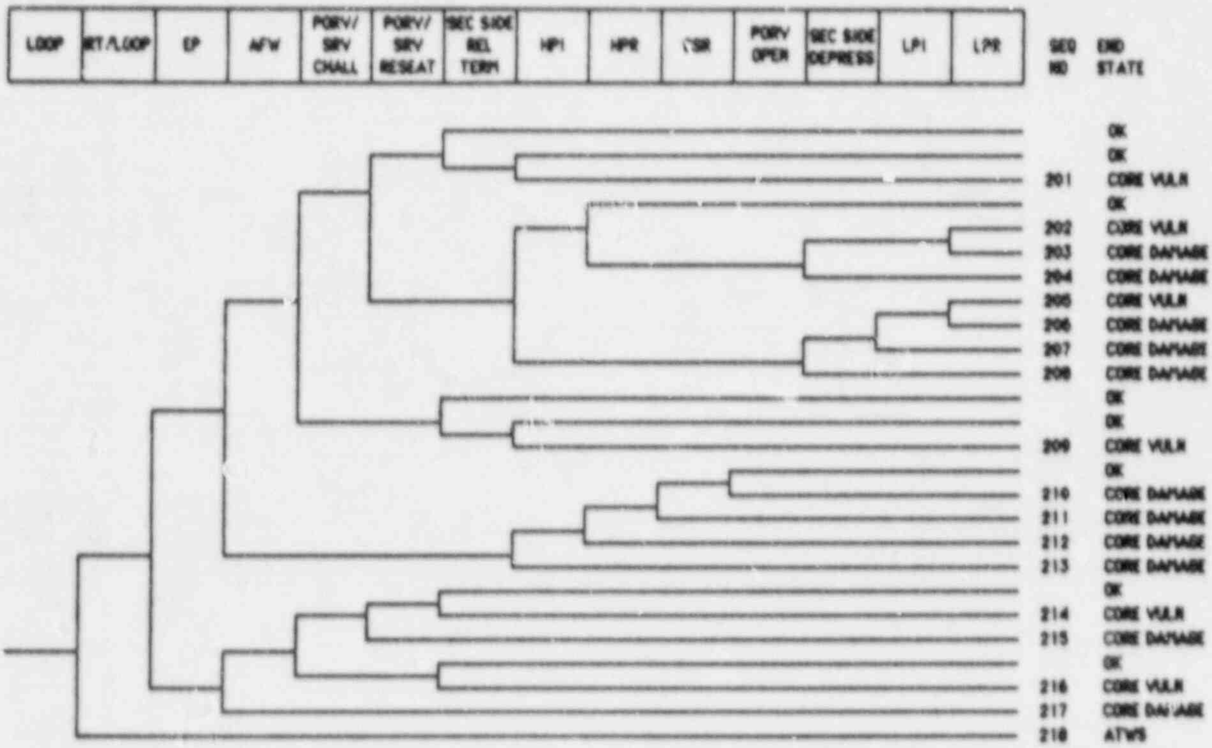


Fig. B.2. PWR Class A LOOP event tree.

ORNL-DWG 86-5540 ETD

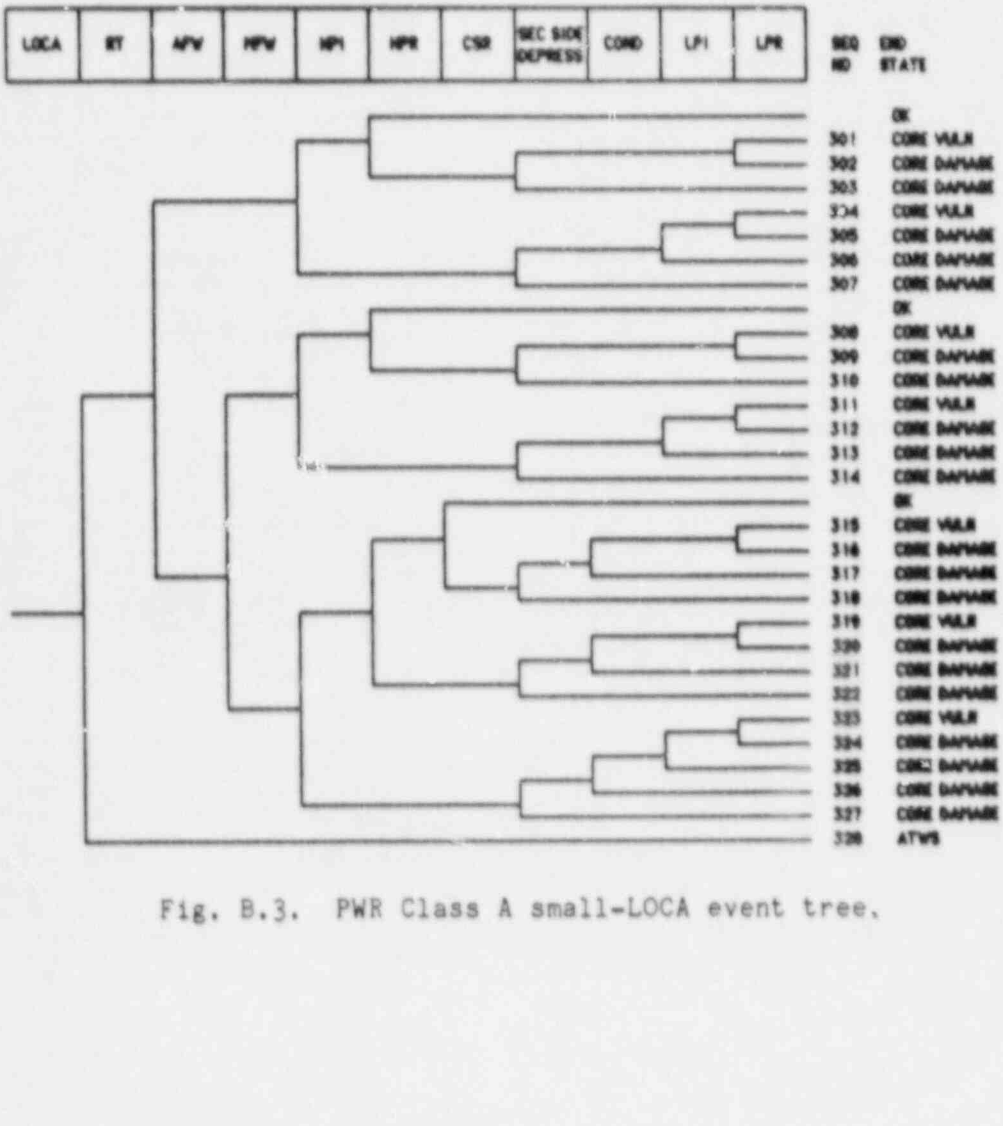


Fig. B.3. PWR Class A small-LOCA event tree.

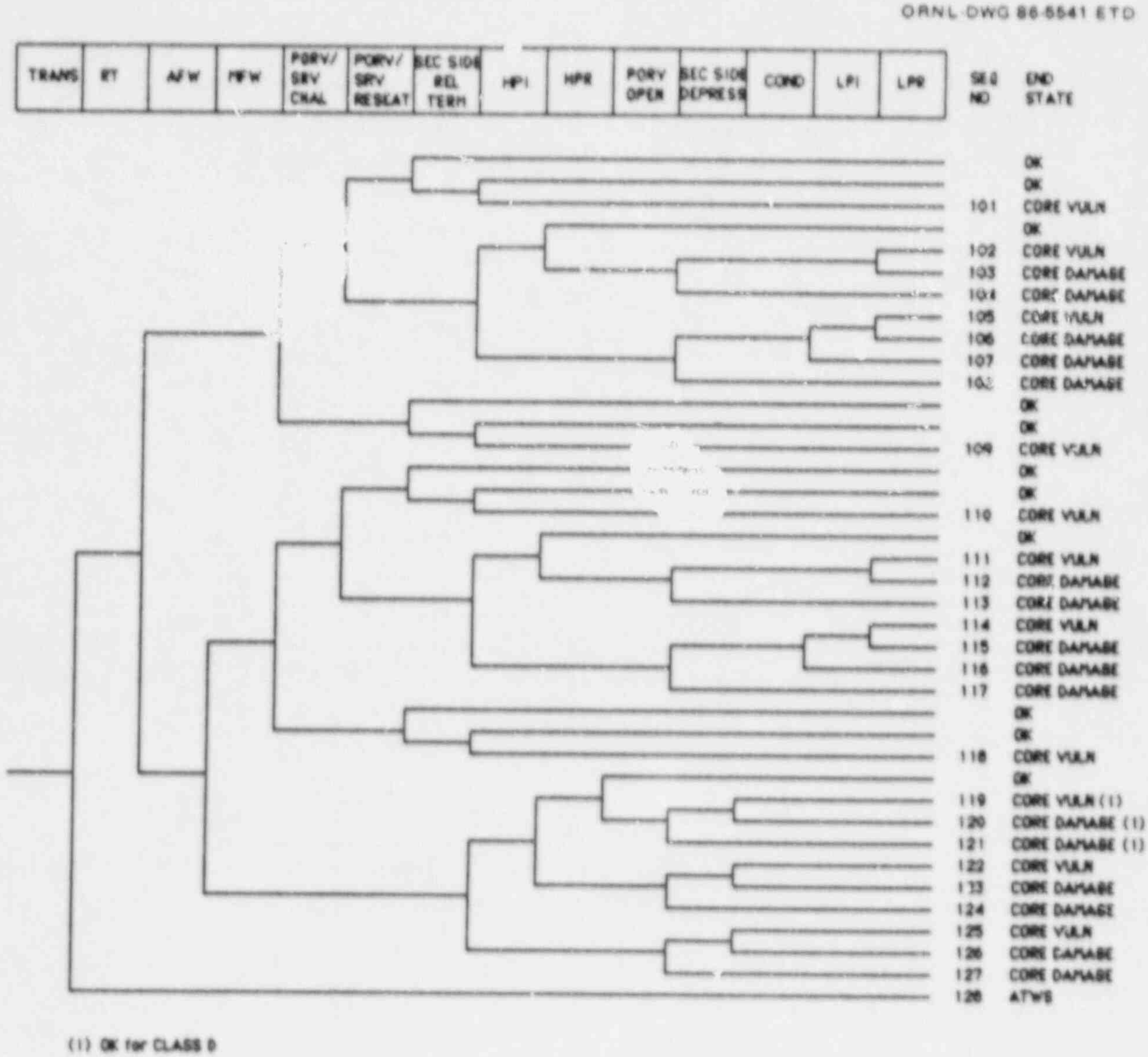
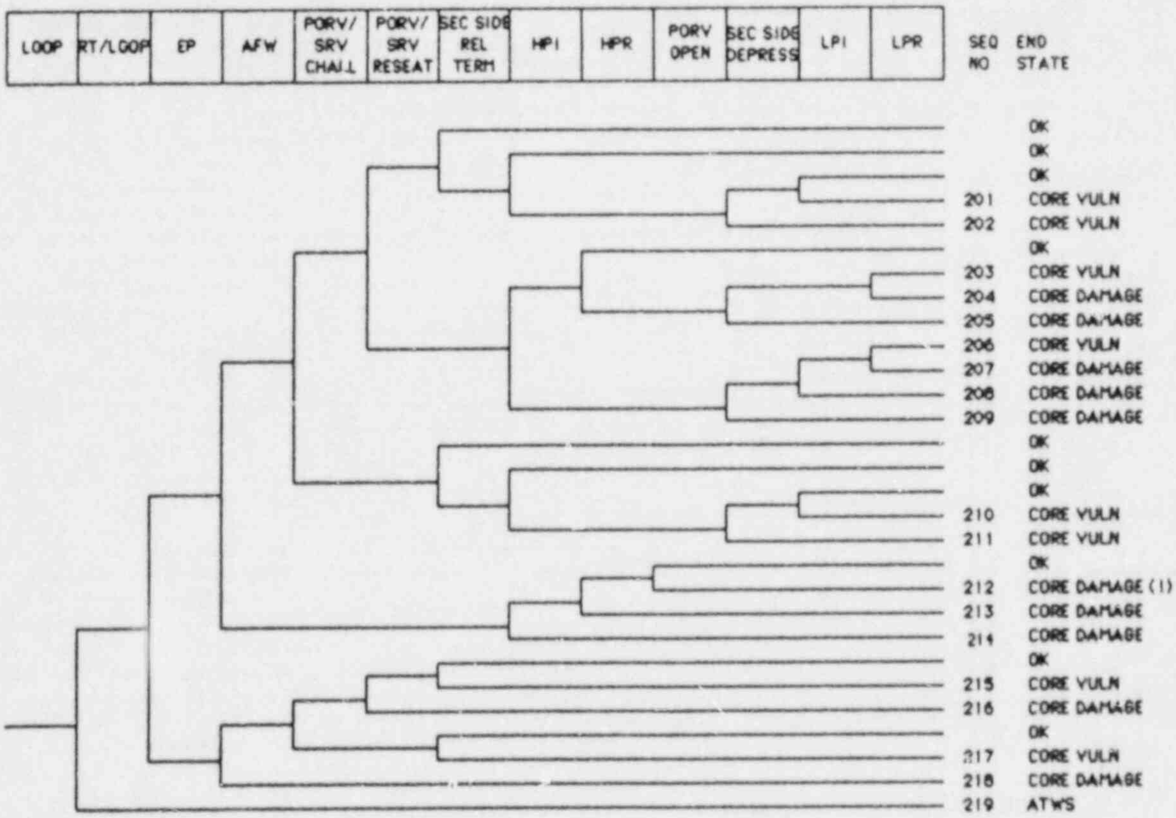


Fig. B.4. PWR Classes B, C, D, E, and F nonspecific reactor-trip event tree.

ORNL-DWG 86-5542 ETD



(1) OK FOR CLASS D

Fig. B.5. PWR Classes B, C, D, E, and F LOOP event tree.

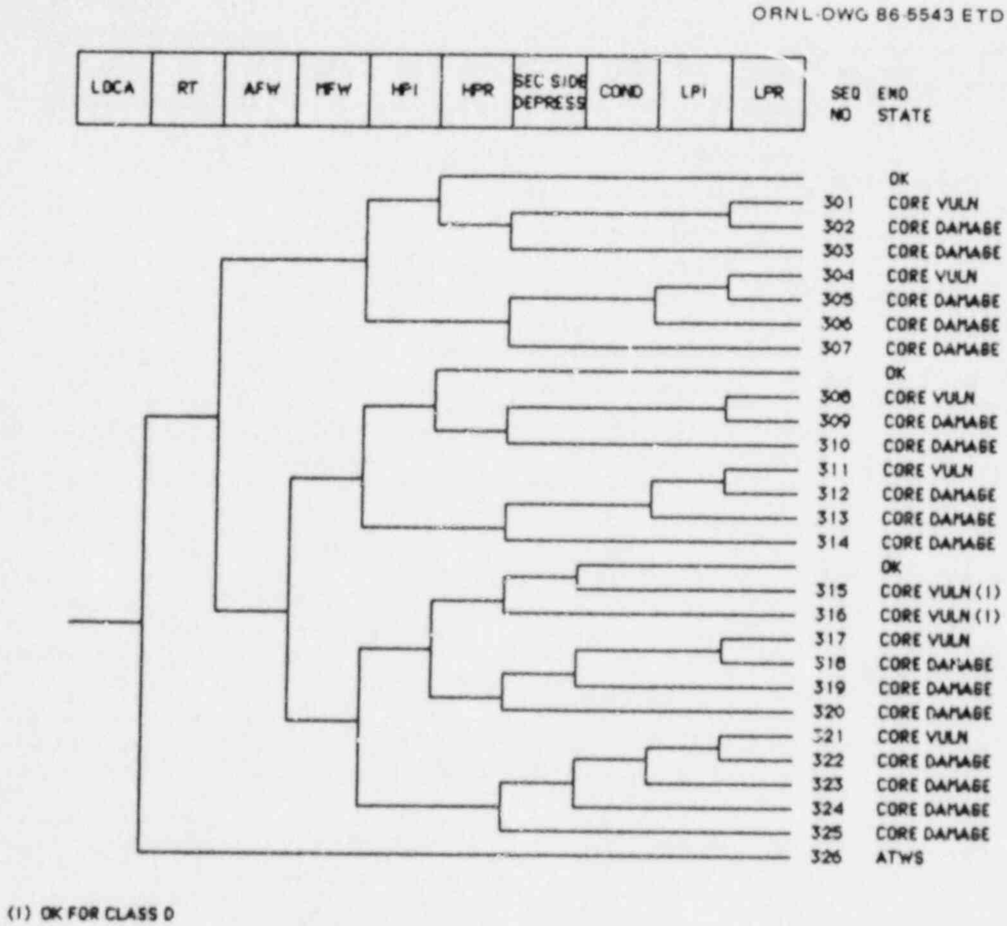


Fig. B.6. PWR Classes B, C, D, E, and F small-LOCA event tree.

ORNL-DWG 86-5544 ETD

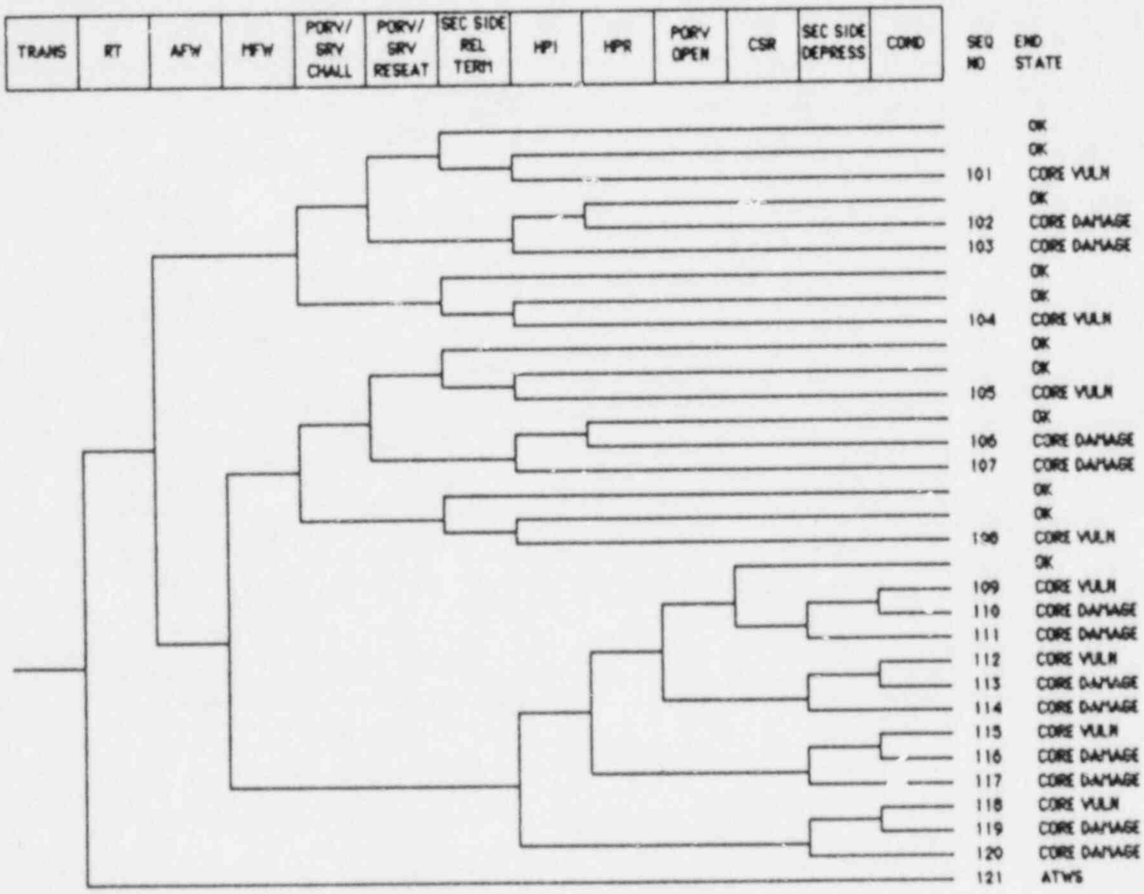


Fig. B.7. PWR Class G nonspecific reactor-trip event tree.

ORNL-DWG 86-5545 ETD

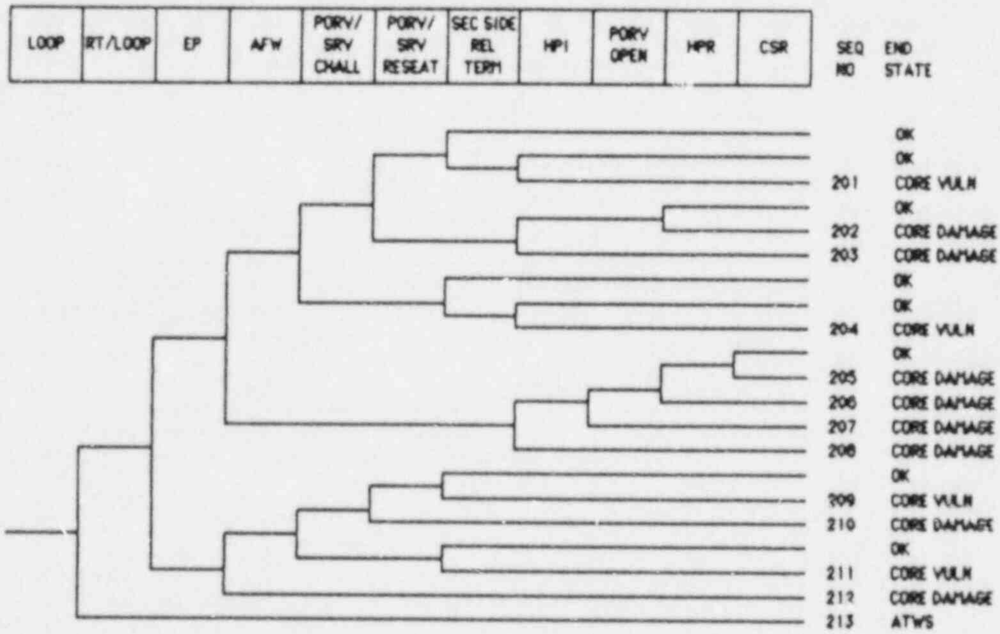


Fig. B.8. PWR Class G LOOP event tree.

ORNL-DWG 86-5546 ETD

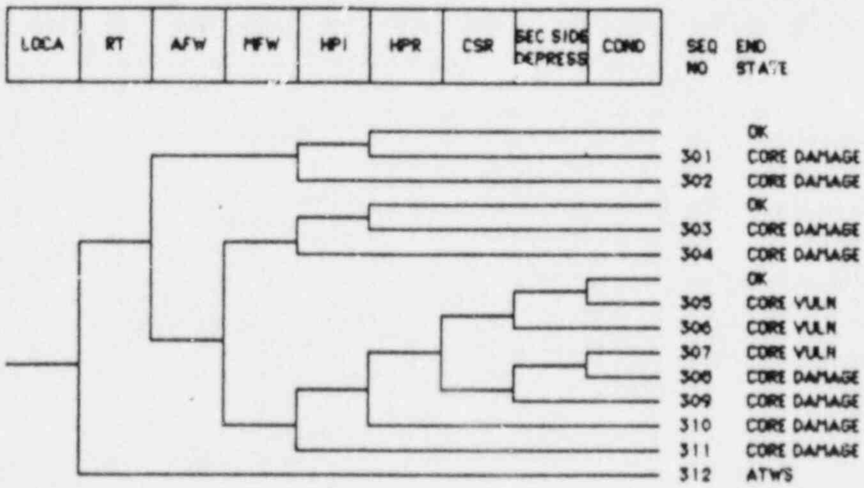
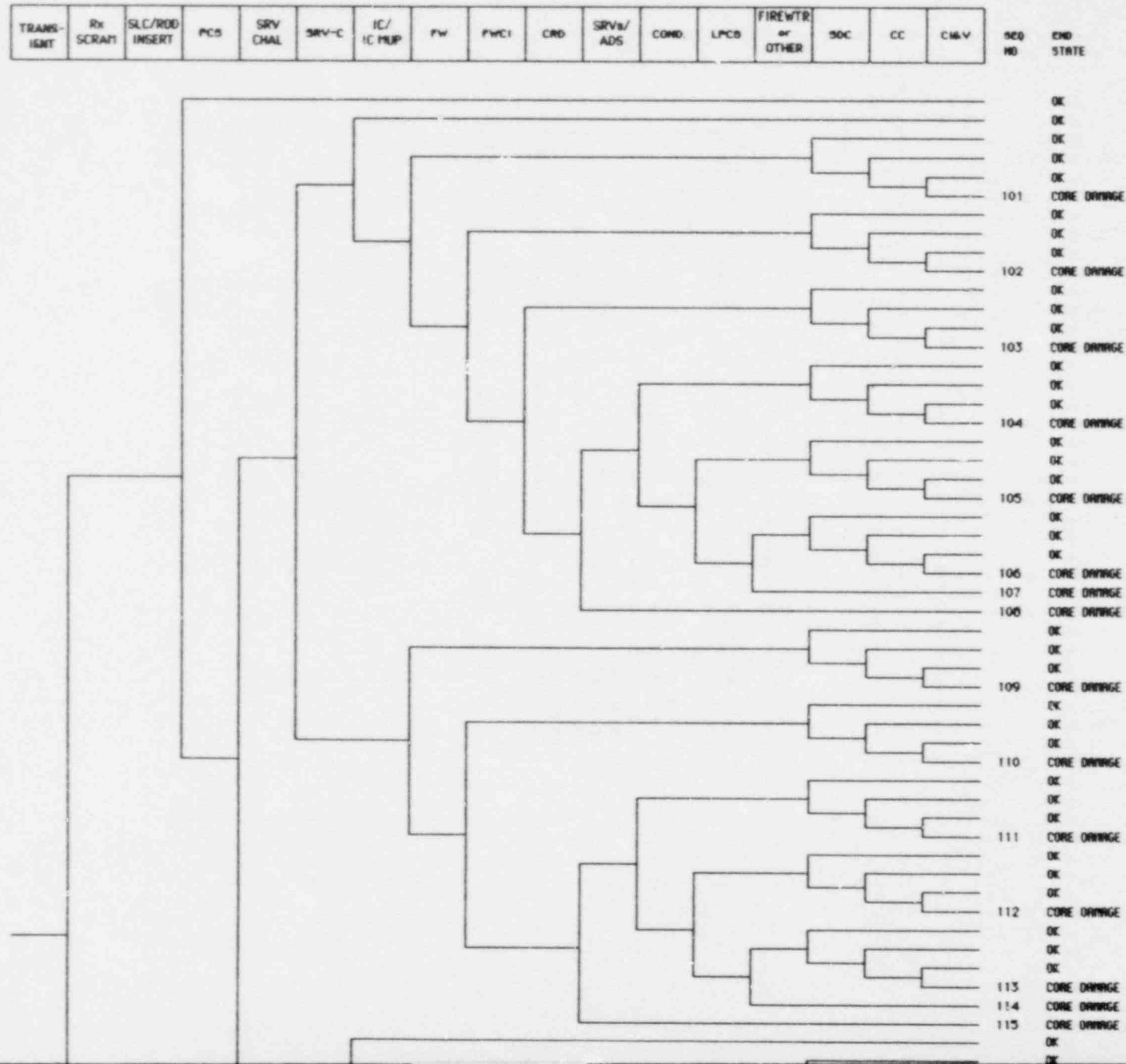


Fig. B.9. PWR Class G small-LOCA event tree.


TRANS-IENT
Rx SCRAM
SLC/ROD INSERT
PCS
SRV CHAL
SRV-C
IC/ IC MUP
FW
FWCI
CRD
SRVs/ ADS
COND
LPCS
FIREWTR or OTHER
SDC
CC
CH&V
SEQ NO
END STATE
OK
OK
OK
OK
OK
101 CORE DAMAGE
OK
OK
OK
102 CORE DAMAGE
OK
OK
OK
103 CORE DAMAGE
OK
OK
OK
104 CORE DAMAGE
OK
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105 CORE DAMAGE
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106 CORE DAMAGE
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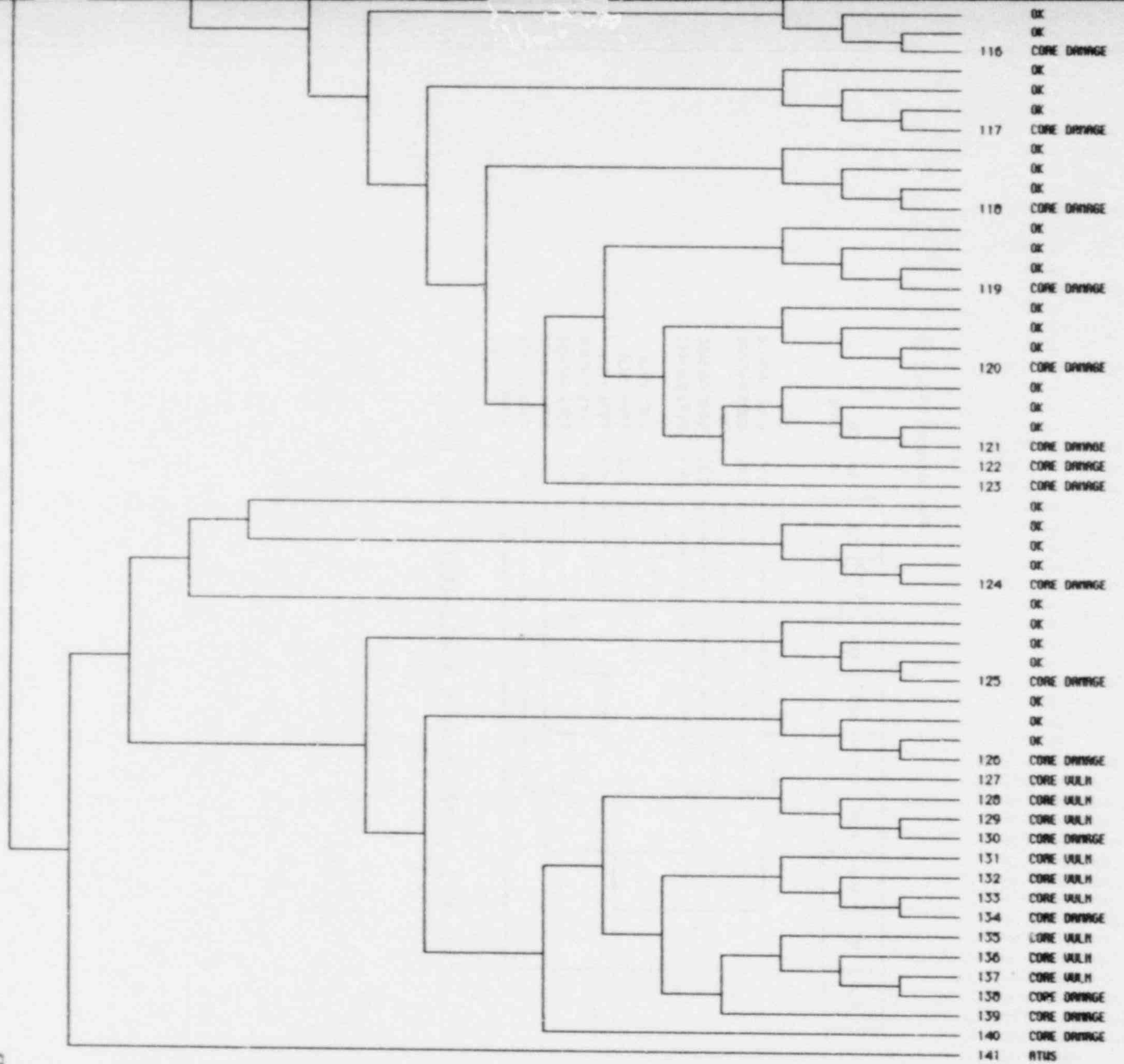

Fig. B.10. BWR Class A nonspecific reactor-trip event tree.

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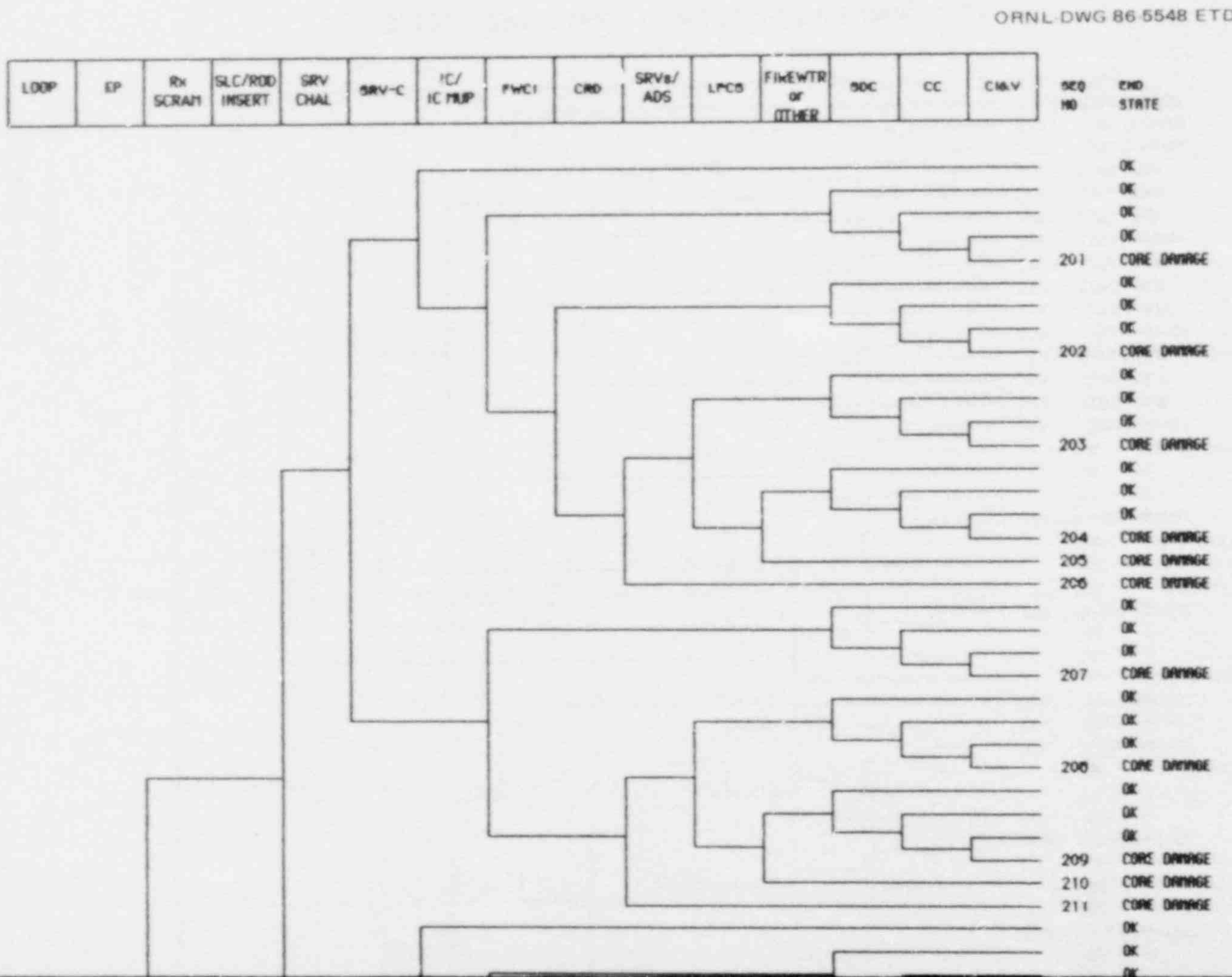

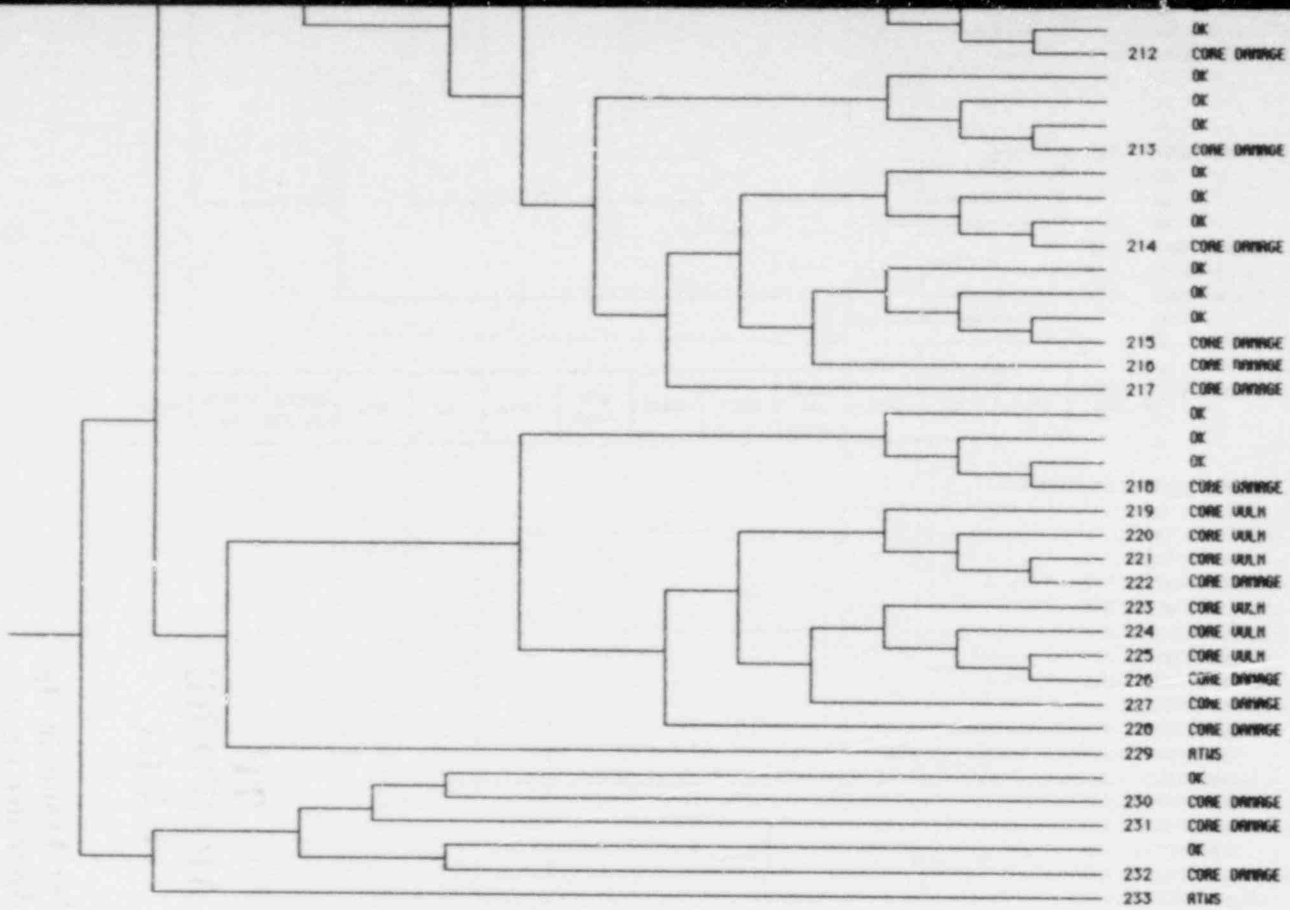

Fig. B.11. BWR Class A LOOP event tree.

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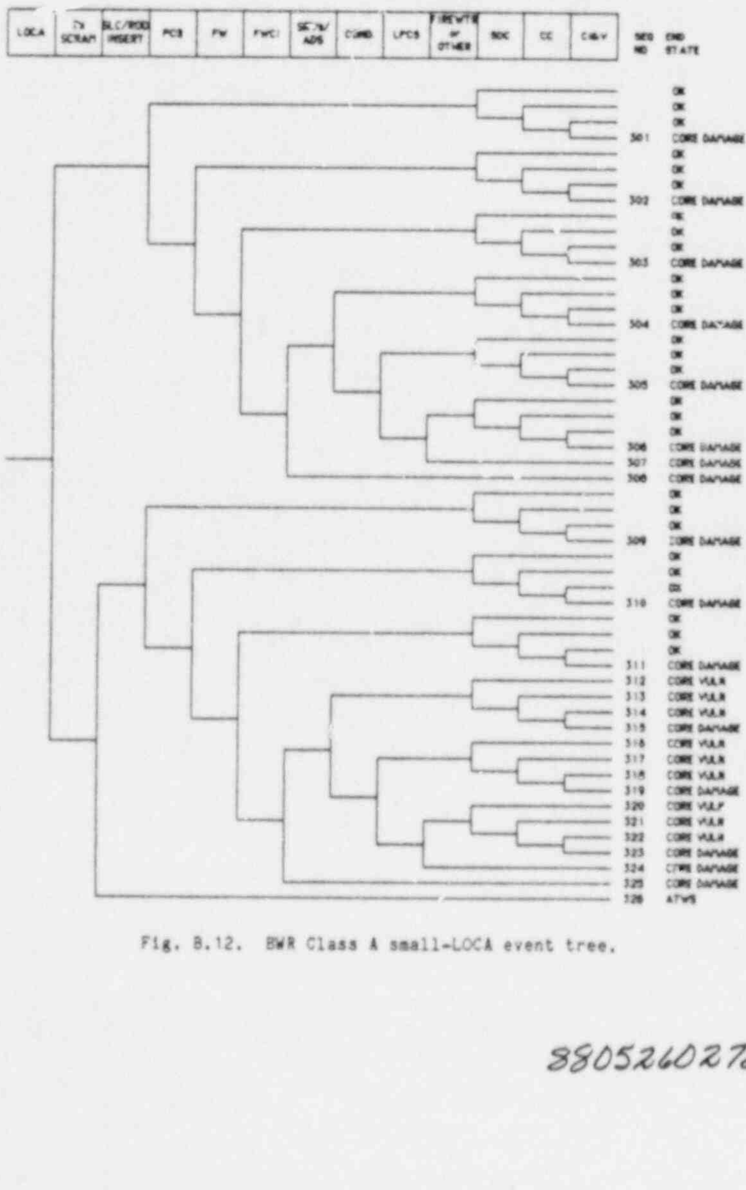

Fig. B.12. BWR Class A small-LOCA event tree.

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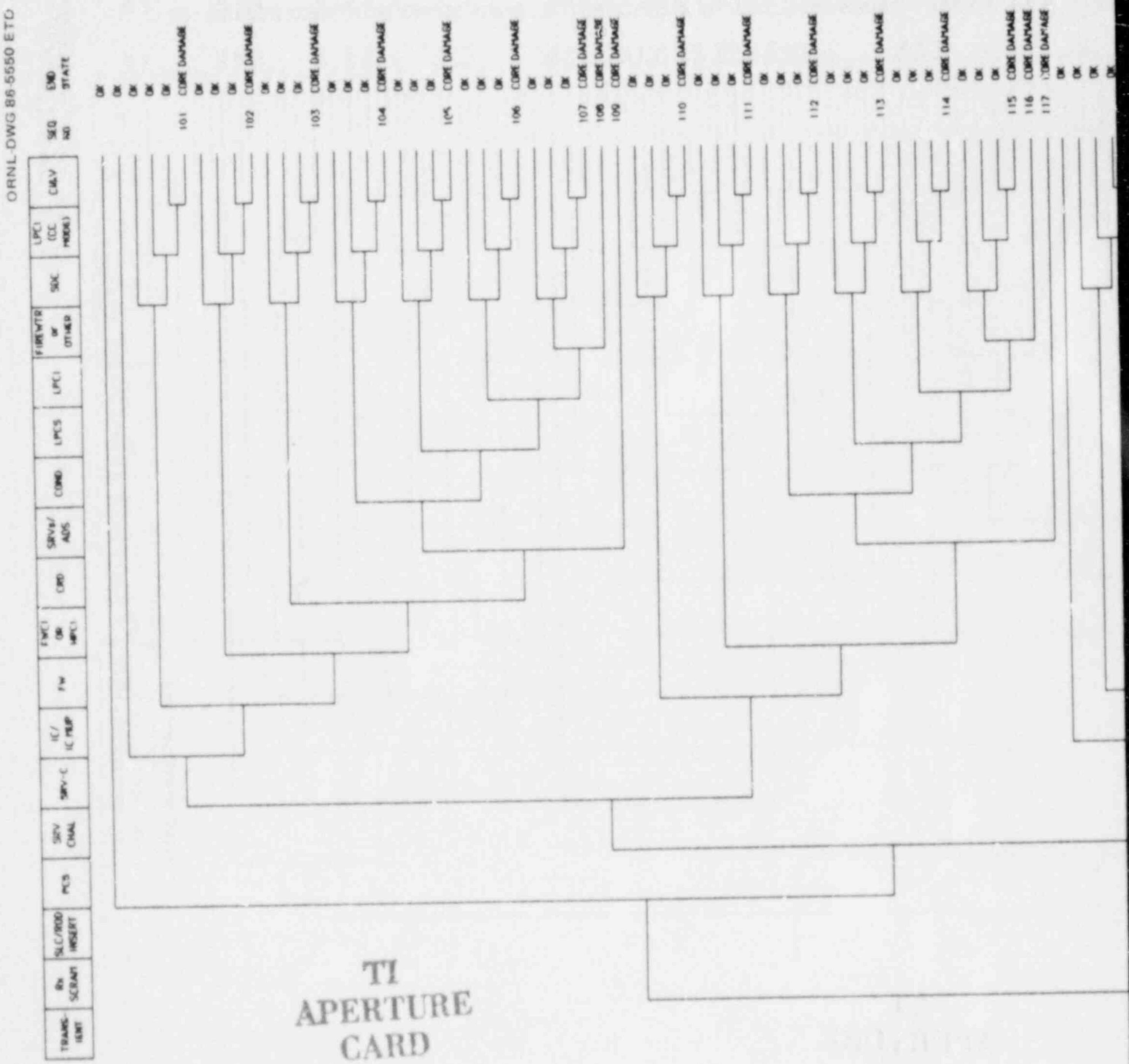

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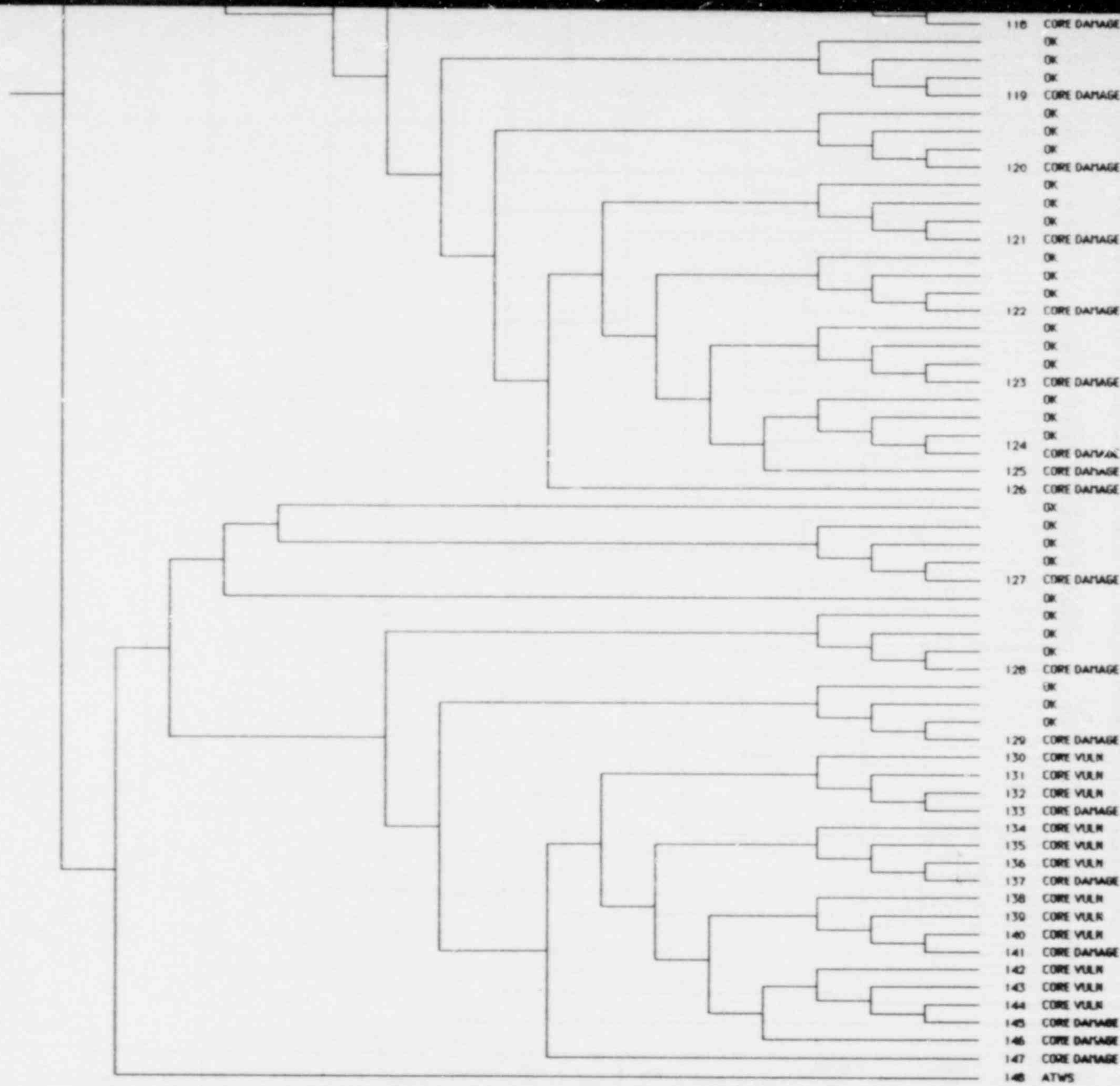

Fig. B.13. BWR Class B nonspecific reactor-trip event tree.

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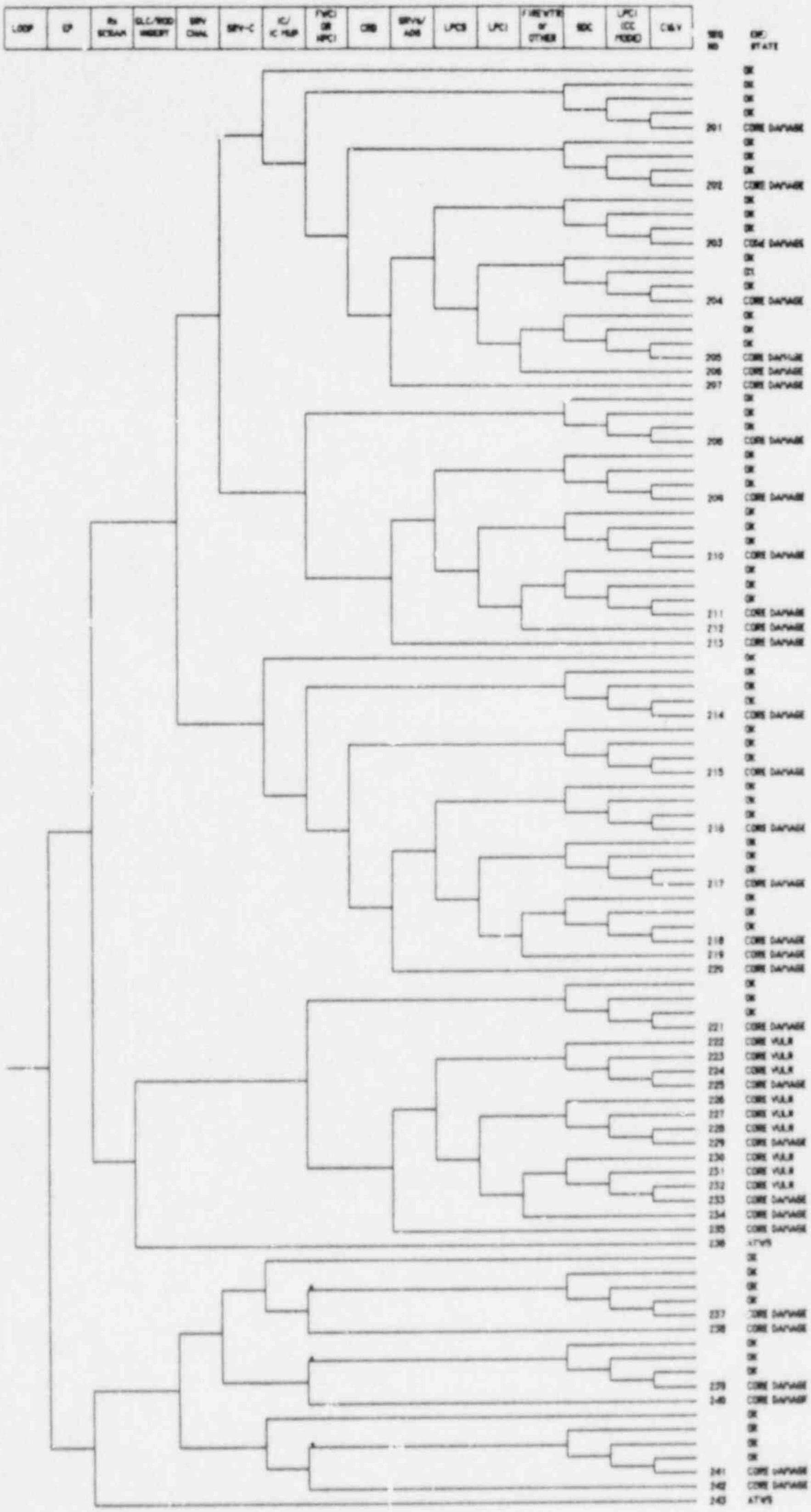
ORNL-DWG 86-5551 ETD



* SUCCESS FOR HPCI PLANTS ONLY

Fig. B.14. BWR Class B LOOP event tree.

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ORNL-DWG 86-5552 ETD

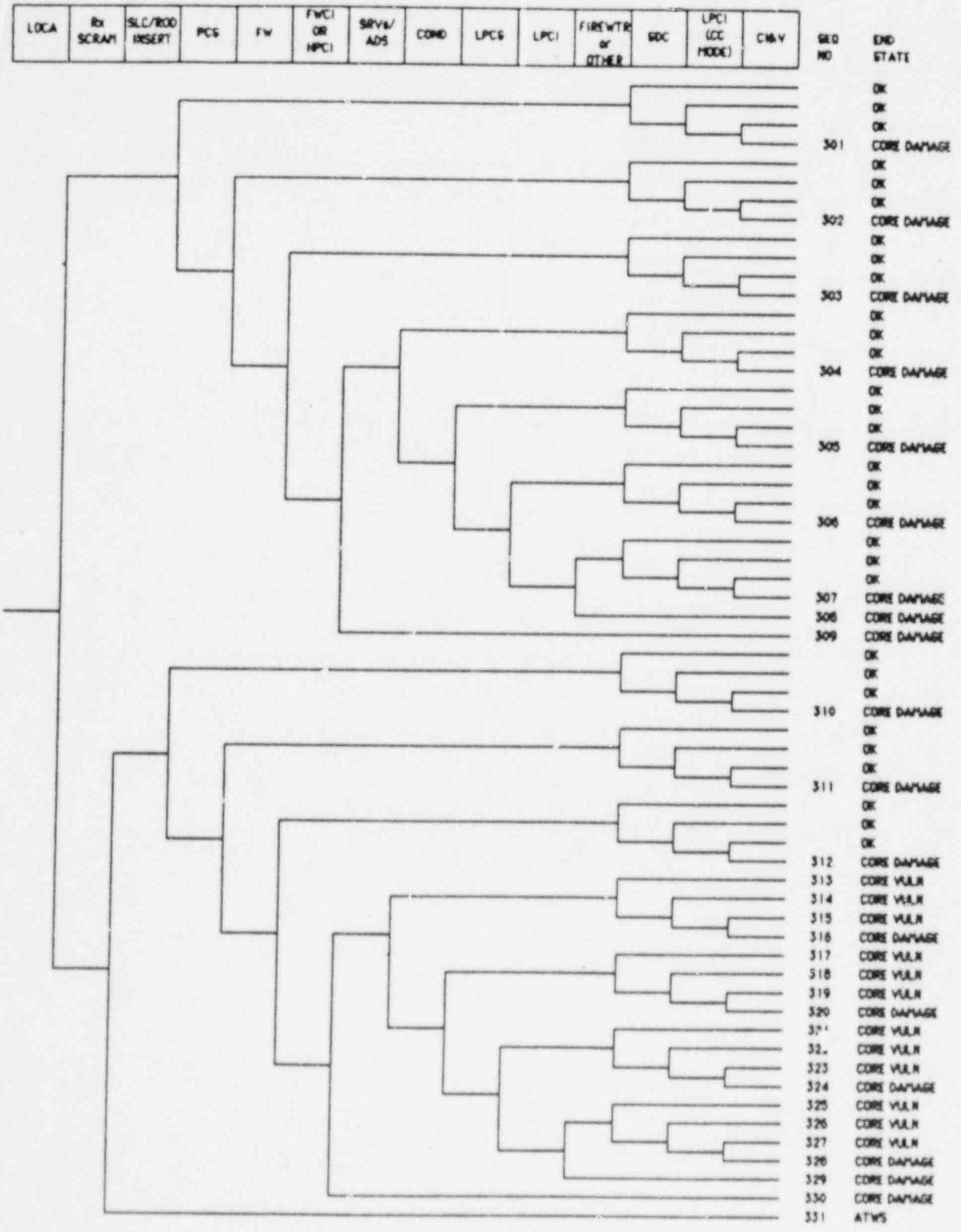
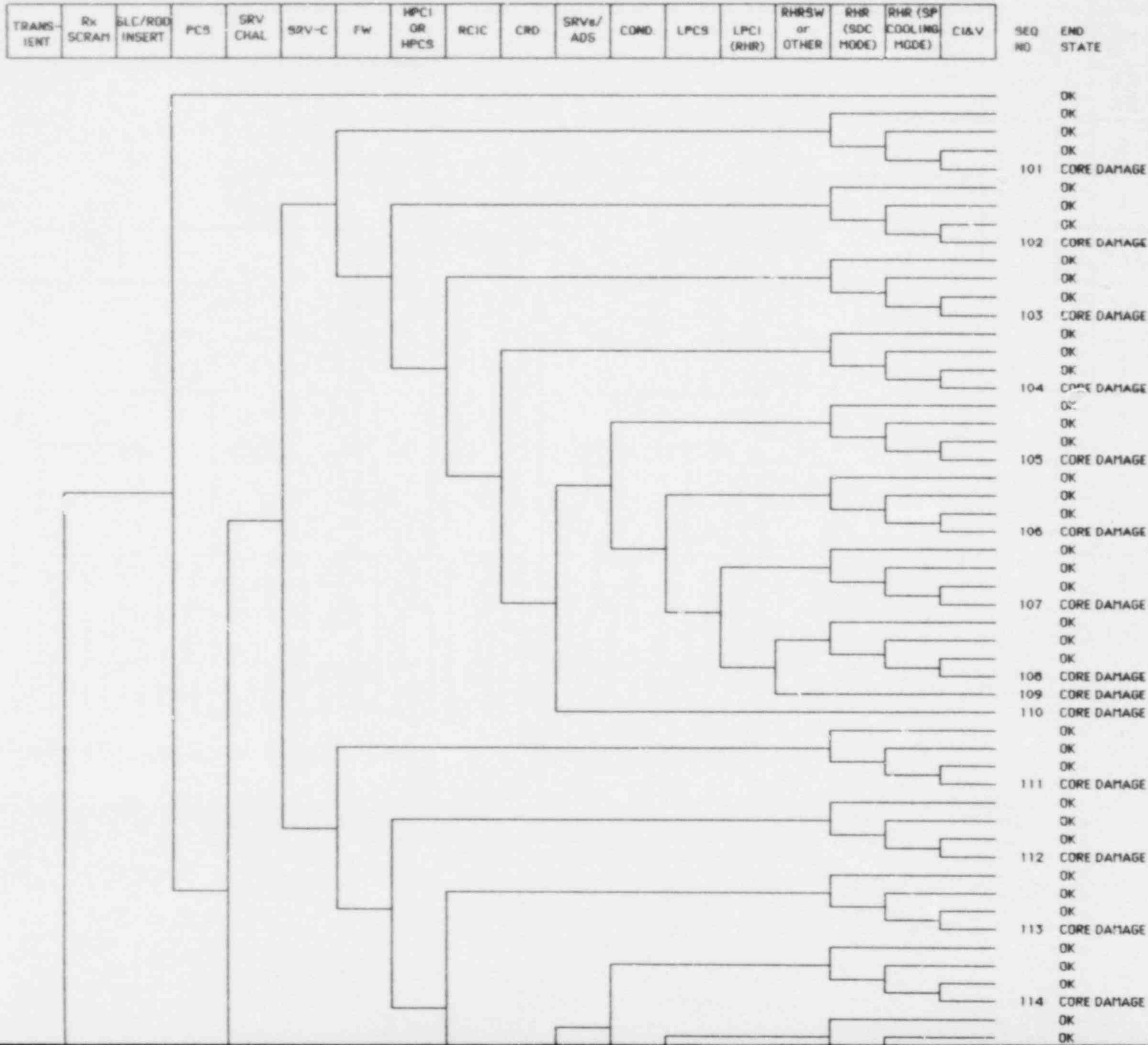
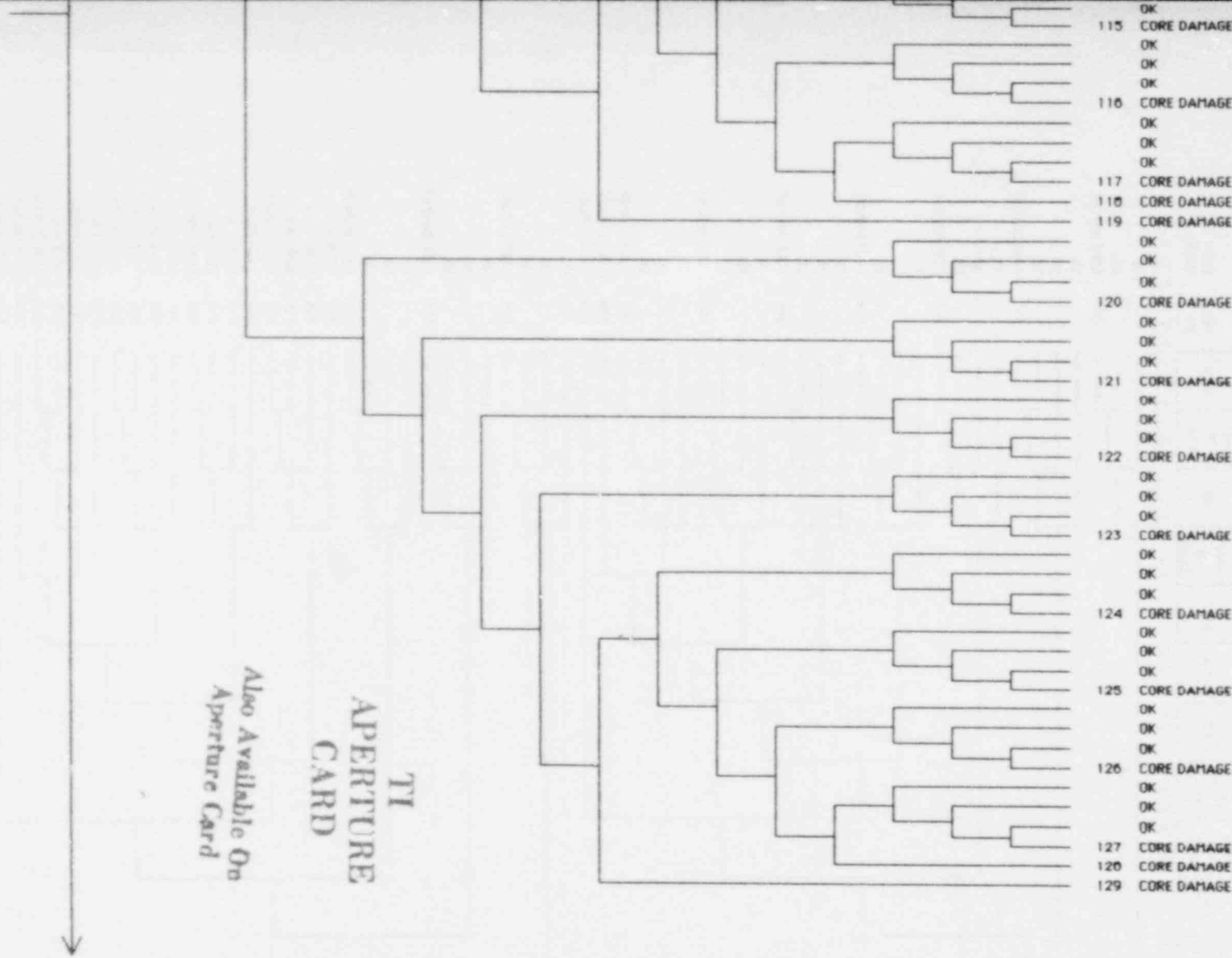


Fig. B.15. BWR Class B small-LOCA event tree.

ORNL-DWG 86-5553 ETD





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* Break size dependent
** If SRV Fails Open, Break Size Dependent

Fig. B.16. BWR Class C nonspecific reactor-trip event tree.

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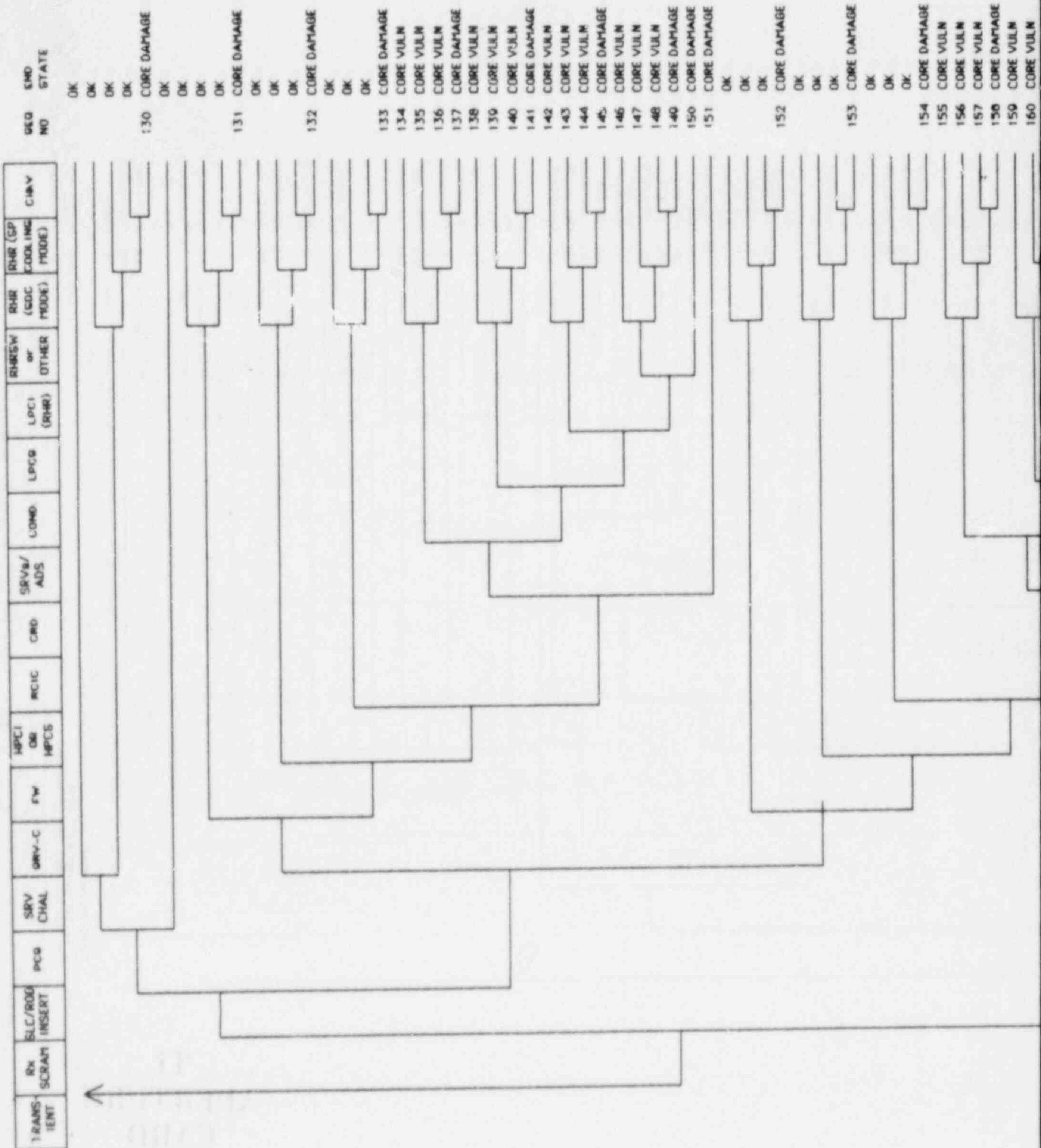

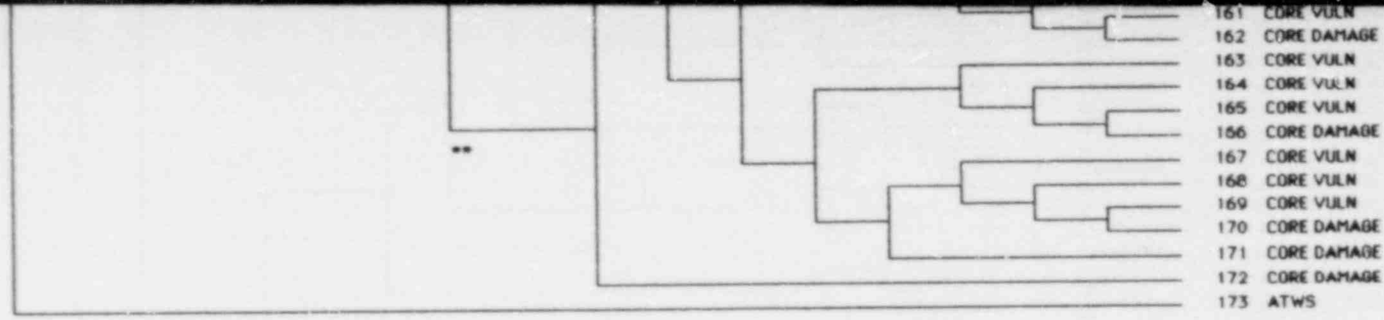

Fig. B.16 (continued)

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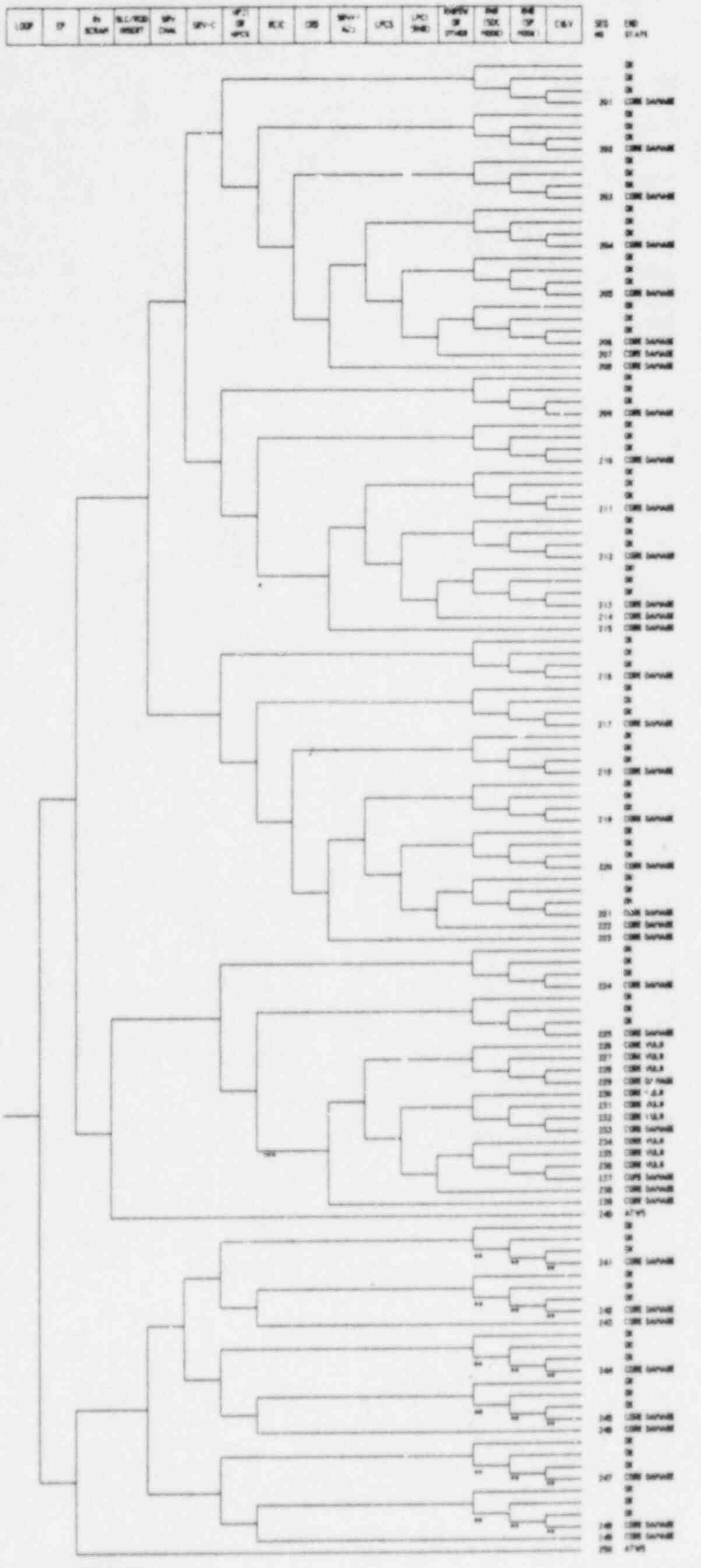

* Break Size Dependent
** EP Recovery Required for Success
*** If SRV Failed Open, Break Size Dependent

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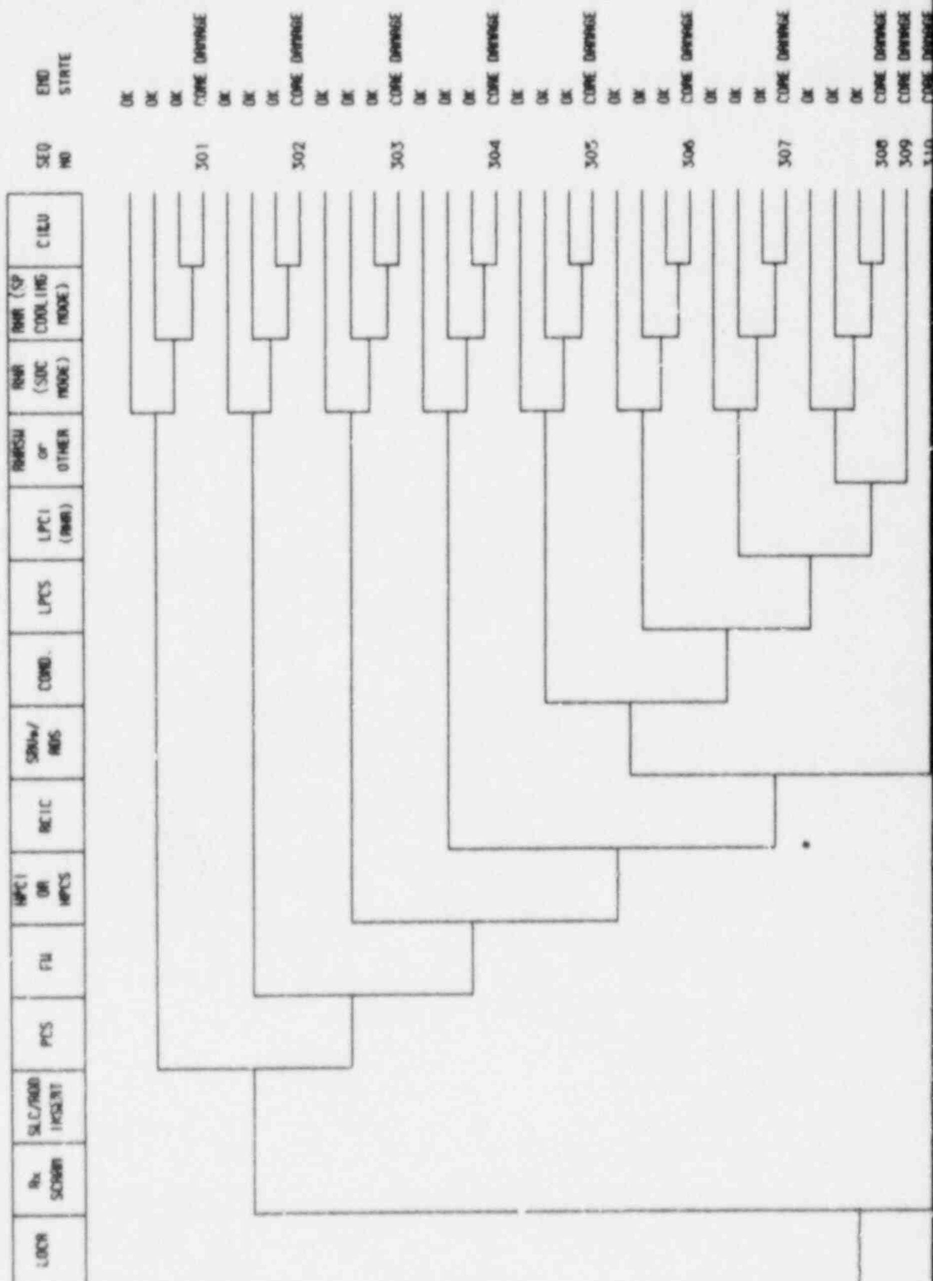
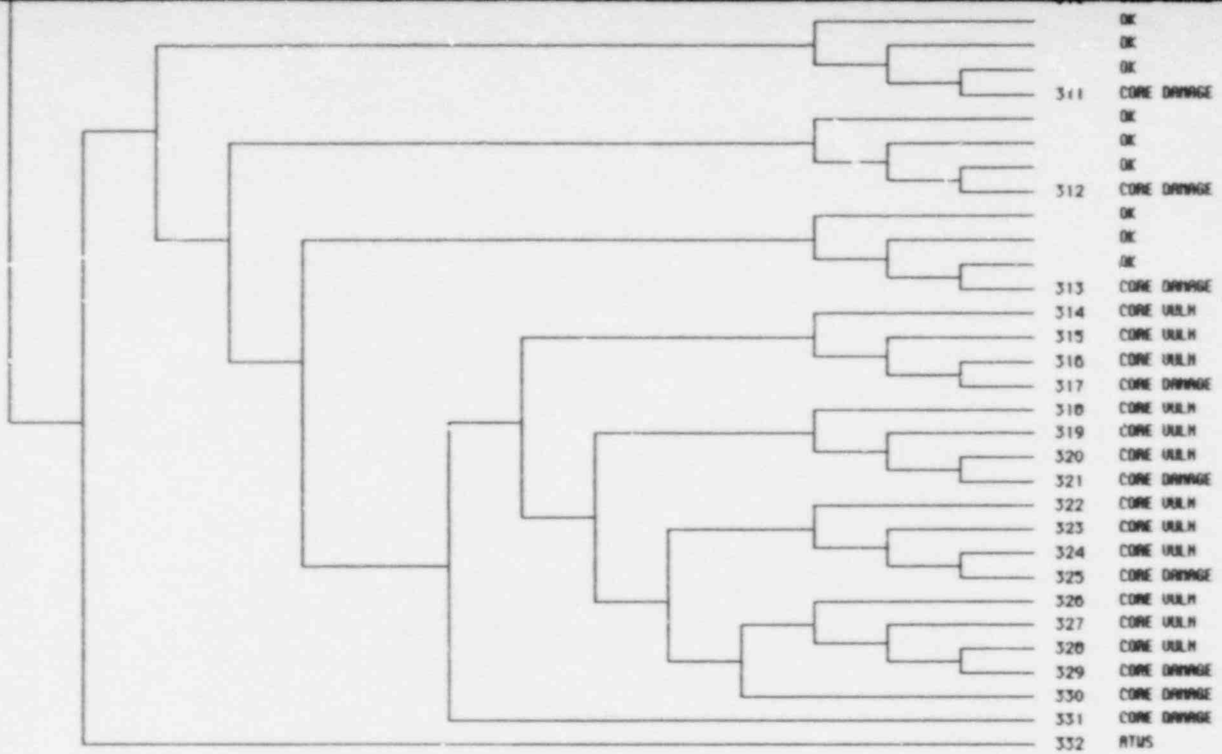
Fig. B.17. BWR Class C LOOP event tree.

ORNL-DWG 86-5555 ETD





*Break Size Dependent

Fig. B.18. BWR Class C small-LOCA event tree.

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APPENDIX C

BRANCH PROBABILITY ESTIMATES

APPENDIX C

BRANCH PROBABILITY ESTIMATES

This appendix provides information concerning the probabilities used in the core damage models as they pertain to the estimated failure rates of systems included on the event trees. Branch probabilities are estimated for each plant according to the methodology described in Volumes 1 (Ref. 1) and 3 (Ref. 2) of this document and the development provided here. Table C.1 develops average estimates based on 1984–86 precursors. These estimates are then used to develop plant-class estimates in Tables C.2–C.4 for BWRs and C.5–C.8 for PWRs. Notes applicable to the tables follow Table C.8.

Table C.1 Initiating event frequency

Initiating event/ function under consideration	Event description			
	LER number	Date	Plant	Event
				PWR initi
LOOP	369/84-024	08-21-84	McGuire 1	At 100% power, 30 power opened, resulting in a ac power; this was cau deficiency in the swit which led to failure t put control circuits f tenance
	206/85-017	11-21-85	San Onofre 1	Ground on 4160-V bus, w tied to main generator ators to trip the plan LOOP had occurred; pow all 4160-V buses
	247/85-016	12-12-85	Indian Point 2	During reactor trip rec error resulted in trip offsite power breakers
	251/85-011	05-17-85	Turkey Point 4	LOOP occurred due to mu on the high-voltage po
	287/85-002	08-28-85	Oconee 3	With main feeder buses energized during a ref the remaining powered energized by actuation power relay
	528/85-058	10-03-85	Palo Verde 1	LOOP occurred due to pl failure from 52% power
	528/85-076	10-07-85	Palo Verde 1	During troubleshooting multiplexer, a false s a LOOP
	261/86-005	01-28-86	Robinson 2	With plant loads on the former following a tur reactor trip from 80% transformer was deener bus lockout occurring switchyard
Small LOCA	250/86-039	12-27-86	Turkey Point 3	During a transient caus turbine governor oil s a PORV opened and faile
	413/86-031	06-13-86	Catawba 1	At power, the variable valve failed open, and downstream of the valv result of vibration-in

nd function failure probability estimates

	Value or recovery class	Total effective number of events	Observation period on demand assumptions	Frequency or probability estimate
ating events				
circuit breakers loss of offsite ed by a de ign hyard computer, reset the out- llowing main-	R2 (0.34)	2.56	164 PWR reactor-years during 1984—1986 period	1.6E-2/ reactor year
th other buses output led oper- , believing a r was lost to	R4 (0.04)			
very, operator ing of the	R2 (0.34)			
tiple faults er system	R1 (1.0)			
and 2 de- eling outage, us was de- of its fault	0.26			
nt multiplexor	R3 (0.12)			
f the plant gnal caused	R3 (0.12)			
startup trans- ine runback and ower, the startup ized by a west n the 115-kV	R2 (0.34)			
d by a loss of stem pressure, d to close fully	0.05	1.05	164 PWR reactor-years during 1984—1986 period	6.4E-3/ reactor year
etdown orifice the letdown line ruptured as a uced fatigue	R1 (1.0)			

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Table C.1. (

Initiating event/ function under consideration	Event description			
	LER number	Date	Plant	Event
				PWR branch
AFW failure	206/85-017	11-21-85	San Onofre 1	Turbine pump flow delayed pump warm-up period; moto failed to actuate due to of diesel generators to l AFW flow was degraded bec fee/water-line check valv
	344/85-009	07-20-85	Trojan	AFW pumps started on deman on low suction pressure.
	346/85-013	06-09-85	Davis Besse 1	Operator error in actuatio feedwater rupture control for AFW flow, following a feedwater, resulted in AF independently, both AFW p overspeed
HPI	272/84-017	07-16-84	Salem 1	With the redundant chargin able, pump 12 ceased whi. surveillance testing; met found in the casings of a on investigation
Failure of long-term core				No events were observed; 0 average nonrecovery of 1.

(continued)

	Value or recovery class	Total effective number of events	Observation period on demand assumptions	Frequency or probability estimate
failures				
because of a -driven pumps LOOP and failure oad as required; ause of failed es	R4 (0.04)	0.42	Twelve demands per reactor year due to testing plus one per shutdown of <48 h plus two per shutdown of >48 h (Based on 1985 operational data from NUREG-0020, Vol. 9, Nos. 6—10, an average 6.0 outages of <48 h and 4.0 outages of >48 h occurred per plant. This results in 26/reactor year × 164 PWR reactor years, yielding 4265 demands	9.9E-5
d but tripped	R4 (0.04)			
n of steam and system (SFRCS) loss of main W isolation; umps tripped on	R2 (0.34)			
g pumps inoper- e performing al filings were ll the pumps	R1 (1.0)	1.0	Twelve demands per reactor year due to testing in 164 reactor years, resulting in 1968 demands	5.1E-4
.33 event at an) was assumed			Twelve demands per reactor year due to testing plus one per shutdown >48 h (4.0 such outages per reactor year — see AFW probability estimate)— resulting in 2132 demands in 164 reactor years	1.5E-4

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Table C.1. (continued)

Initiating event/ function under consideration	Event description: LER number	Event description: Date	Event description: Plant	Event description: Event	Value or recovery class	Total effective number of events	Observation period on demand assumptions	Frequency or probability estimate
				PWR function failures				
Failure of emergency power	389/86-011	07-09-86	St. Lucie 2	At 100% power, one DG was declared inoperable following failure of the other DG in a surveillance test	R1 (1.0)	1.0	Twelve demands per reactor year due to testing plus 0.1 LOOP demand per reactor year, resulting in 1984 demands	5.0E-4
Failure of SG	309/85-009	08-07-85	Maine Yankee	Steam-line pressure transmitters (9 of 12) were found in faulty alignment, which would have prevented automatic steam-line isolation on demand	R4 (0.04)	1.04	Twelve demands per reactor year due to testing resulting in 1968 demands	5.3E-4
	301/86-004	09-28-86	Point Beach 2	All MSIVs failed to close from the control room for refueling	R1 (1.0)			
				BWR initiating events				
LOOP	331/84-028	07-14-84	Duane Arnold	LOOP occurred due to degraded offsite voltage	R2 (0.34)	2.92	Eighty-nine BWR reactor years during 1984–1986 period	3.3E-2/ reactor year
	409/84-011	09-16-84	LaCrosse	Potential transformer in switchyard shorted out during a storm, causing a LOOP.	R2 (0.34)			
	237/85-034	08-16-85	Dresden 2	LOOP was caused by a fault on the Unit 1 auxiliary transformer, which was sensed on the Unit 2 reserve transformer	R2 (0.34)			
	245/85-018	09-27-85	Millstone 1	LOOP was caused by a hurricane.	0.10			
	409/85-017	10-22-85	LaCrosse	69-kV tie line breaker to the grid opened due to a maintenance error, causing a LOOP	R2 (0.34)			
	293/86-027	11-19-86	Pilgrim 1	LOOP occurred due to faults on the transmission lines during severe winter storm.	R3 (0.12)			
	409/86-023	07-10-86	LaCrosse	LOOP occurred during severe thunderstorm	R2 (0.34)			
	458/86-002	01-01-86	River Bend 1	Station transformers were de-energized by spurious signals believed to be initiated by radio frequency interference	R1 (1.0)			

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Table C.1.

Initiating event/ function under consideration	Event description			
	LER number	Date	Plant	Event
				BWR initiat
Small LOCA	331/84-001	06-07-84	Duane Arnold	SRV opened spuriously bec nician error during rout procedure
	259/84-027	06-27-84	Browns Ferry 1	SRV unseated and continue startup following a shor
	259/84-032	08-14-84	Browns Ferry 1	Operator and installation in overpressurization an the low-pressure core sp valve during surveillanc the valve
	321/85-018	05-15-85	Hatch 1	Relief valve stucks open
				BWR branch
Failure of HPCI and RCIC	321/85-010	01-06-85	Hatch 1	RCIC was inoperable follo HPCI failed on demand
	321/85-018	05-15-85	Hatch 1	HPCI was inoperable with service for maintenance
Failure of reactor vessel isolation	324/85-008	09-27-85	Brunswick 2	Three of eight MSIVs faile
Failure of long-term core cooling	254/84-014	08-08-84	Quad Cities 1	LPCI valves failed to ope during refueling
	324/84-014	11-27-84	Brunswick 2	RHR loop A was rendered i to water hammer, and loo able due to leakage past mary containment isolati

(continued)

	Value or recovery class	Total effective number of events	Observation period on demand assumptions	Frequency or probability estimate
...ing events				
...ise of tech- ...ne surveillance	R2 (0.34)	0.90	Eighty-nine BWR reactor years during 1984—1986 period	1.0E-2/ reactor year
... to leak in ... outage	(0.1)			
...errors resulted ... leakage through ...ay isolation ... testing of	R3 (0.12)			
	R2 (0.34)			
... failures				
...ing a demand;	R2 (0.34)	0.68	Twelve demands per reactor year due to testing plus one LOFW demand per reactor year × 62 reactor years for plants with HPCI/RCIC systems, resulting in 806 demands	8.4E-4
...CIC out of	R2 (0.34)			
...d to fast close	0.30	0.30	One demand per reactor year for the full closure test plus one LOFW event per reactor year in 89 BWR reactor years, resulting in 178 demands	1.7E-3
... on demand	R1 (1.0)	1.12	Function demanded on test, during LOFW, and on shutdowns of longer duration (>48 h assumed)	7.4E-4
...operable due ... B was inoper- ...the LPCI pri- ...n valve.	R3 (0.12)		NUREG-0020, Vol. 9, Nos. 6—10 showed 54 such shutdowns for BWRs, yielding 3.9 shutdowns per reactor year; with 12 tests and one LOFW assumed per reactor year, 16.9 demands per reactor . × 89 reactor years results in 1504 demands.	

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Table C.1. (continued)

Initiating event/ function under consideration	Event description				Value or recovery class	Total effective number of events	Observation period on demand assumptions	Frequency or probability estimate
	LER number	Date	Plant	Event				
				BWR [illegible] Failures				
Failure of emergency power				No events were observed; 0.33 event at an average nonrecovery of 0.3 assumed			Twelve demands per reactor year due to testing in 89 BWR reactor years plus five demands due to actual LOOPs resulting in 1073 demands.	8.9E-5
Failure of ADS				No events were observed; 0.33 event at an average nonrecovery of 0.71 was assumed			One demand per reactor year due to testing in 89 reactor years, resulting in 89 demands	2.6E-3

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Table C.2. 1986 BWR Class A

Plant	SCRAM (Note 1)	SLC.OR.RODS (Note 2)	PCS/TRANS (Note 3)	PCS/LOCA (Note 4)	SRV.CHALL/TRANS.SCRAM (Note 5)	SRV.CHALL/TRANS.SCRAM (Note 6)	SRV.CHALL/LOOP.-SCRAM (Note 7)	SRV.CHALL/LOOP.SCRAM (Note 8)	SRV.CLOSE (Note 9)	EMERG.POWER (Note 10)	FW/PCS.TRANS (Note 37)	FW/PCS.LOCA (Note 37)	ISOL.COND (Note 36)	FWCI/FW.TRANS
Oyster Creek	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	16	1 of 2	1.0	1.0	0.01	0.
Big Rock Pt	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	6	1 of 1	1.0	1.0	0.01	0.
Millstone 1	3.5E-4	0.05	0.17	1.0	0.3	1.0	0.3	1.0	5	1 of 2	1.0	1.0	0.01	0.
Nine Mi Pt 1	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	16	1 of 2	1.0	1.0	0.01	0.

[a]Branch probability abbreviations:

ADS	automatic depressurization system
AFW	auxiliary feedwater
BIT	boron injection tank
CC	containment cooling
CHALL	challenged
C.I.AND.V	containment injection and venting
COND	condensate pumps
CRD	control rod drive pump cooling
CSR	containment spray recirculation
DEPRESS	depressurization
EMERG	emergency
F/B	feed and bleed
FW	feedwater
FWCI	feedwater coolant injection
HPCI	high-pressure coolant injection
HPI	high-pressure injection
HPR	high-pressure recirculation
ISOL.COND	isolation condenser
LOCA	loss-of-coolant accident
LOOP	loss of offsite power

LPCI
LPCS
LPI
LPR
M
MFW
PCS
PORV
RHR
RHRSW
RT
SDC
SLC.OR.Rods
SPCOOL
SRV
SS
T
TERM
TRANS

[b]Branch notations:

/	given
—	branch success
( )	(no "—") branch failure
.	(before "/") separates words
.	(after "/") means and

[c]Not yet developed in the program.

branch probabilities[a,b]

(Note 38)	FWCI/LOOP (Note 39)	FWCI/FW.LOCA (Note 40)	CRD (Note 16)	SRV.ADS (Note 41)	COND/FW.PCS (Note 18)	LPCS (Note 42)	FIREWATER.OR.OTHER/LPCS.LOCA (Note 43)	FIREWATER.OR.OTHER/LPCS.TRANS (Note 43)	FIREWATER.OR.OTHER/LPCS.LOOP (Note 43)	SDC (Note 44)	CC/SDC (Note 45)	C.I.AND.V (Note 28)
10	1.0	3.4E-4	0.05	0.031	0.34	1E-3	1.0	1.0	1.0	7.1E-3	1E-3	0.34
10	1.0	3.4E-4	0.05	1.0	0.34	[c]	1.0	1.0	1.0	[c]	[c]	0.34
10	0.1	3.4E-4	0.05	0.031	0.34	6.8E-4	1.0	1.0	1.0	3.7E-3	3.4E-4	0.34
10	1.0	3.4E-4	0.05	0.031	0.34	6.8E-4	1.0	1.0	1.0	9.9E-4	1.0	0.34

low-pressure coolant injection
low-pressure core spray
low-pressure injection
low-pressure recirculation
motor-driven
main feedwater
power conversion system
power-operated relief valve
residual heat removal
residual heat removal system service water convoction
reactor trip
shutdown cooling
standby liquid control or manual rod insertion
suppression pool cooling
safety relief valve
secondary side
turbine-driven
terminated
transient

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Table C.5. 1986 BWR Class B branch probabilities[a]

PLANT	SCRAM (Note 1)	SLC.OR.RODS (Note 2)	PCS.TRANS (Note 3)	PCS/LOCA (Note 4)	SRV.CHALL/TRANS.-SCRAM (Note 5)	SRV.CHALL/TRANS.-SCRAM (Note 6)	SRV.CHALL/LOOP.-SCRAM (Note 7)	SRV.CHALL/LOOP.SCRAM (Note 8)	SRV.CLOSE (Note 9)	EMERG.POWER (Note 10)	FW/PCS.TRANS (Note 11)	FW/PCS.LOCA (Note 12)	HPCI (Note 14)	ISOL.COND (Note 29)	CRD (Note 16)	SRV.ADE (Note 17)	COND/FW.PCS (Note 18)	LPCS (Note 30)	LPCI (Note 31)	FIREWATER.OR.OTHER/LPCS.LPCI/TRANS (Note 32)	FIREWATER.OR.OTHER/LPCS.LPCI/LOOP (Note 32)	FIREWATER.OR.OTHER/LPCS.LPCI/LOCA (Note 32)	SDC (Note 13)	LPCI(CC) (Note 34)	LPCI(CC)/LPCI (Note 35)	C.I.AND.V/LPCI (Note 28)
Dresden 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	9	1 of 2	M	M	0.020	0.02	0.05	0.031	0.34	6.8E-4	7.1E-4	1.0	1.0	1.0	2.9E-3	7.1E-4	1.0	0.34
Dresden 3	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	9	1 of 2	M	M	0.020	0.02	0.05	0.031	0.34	6.8E-4	7.1E-4	1.0	1.0	1.0	2.9E-3	7.1E-4	1.0	0.34

[a]See Table C.2, note d, for abbreviations and branch notations.

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Table C.4. 1986 BWR Class

PLANT	SCRAM (Note 1)	SLC.JR.RODS (Note 2)	PCS/TRANS (Note 3)	PCS/LOCA (Note 4)	SRV.CHALL/TRANS.-SCRAM (Note 5)	SRV.CHALL/TRANS.SCRAM (Note 6)	SRV.CHALL/LOOP.-SCRAM (Note 7)	SRV.CHALL/LOOP.SCRAM (Note 8)	SRV.CLOSE (Note 9)	EMERG.POWER (Note 10)	FW/PCS.TRANS (Note 11)	FW/PCS.LOCA (Note 12)	HPCI (Note 14)	RCIC/TRANS.OR.LOOP (Note 13)	RCIC/LOCA
Arnold	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	8	1 of 2	M	M	0.020	0.042	1.
Browns Ferry 1, 2, 3	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	13	1 of 3	T	T	0.020	0.042	1.
Brunswick 1	3.5E-4	0.05	0.17	1.0	0.3	1.0	0.3	1.0	11	1 of 2	T	T	0.020	0.042	1.
Brunswick 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	11	1 of 2	T	T	0.020	0.042	1.
Cooper	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	11	1 of 2	T	T	0.020	0.042	1.
Fermi 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	15	1 of 4	T	T	0.020	0.042	1.
Fitzpatrick	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	11	1 of 2	T	T	0.020	0.042	1.
Grand Gulf 1, 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	20	1 of 2	T	T	0.020	0.042	1.
Hatch 1, 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	11	1 of 3	T	T	0.020	0.042	1.
Hope Creek 1	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	11	3 of 4	T	T	0.020	0.042	1.
LaSalle 1, 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	18	1 of 2	T	T	0.020	0.042	1.
Limerick 1	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	11	1 of 3	T	T	0.020	0.042	1.
Monticello	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	8	1 of 2	M	M	0.020	0.042	1.
Nine Mi Pt 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	18	2 of 3	T	T	0.020	0.042	1.
Peach Bottom 2, 3	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	13	1 of 4	T	T	0.020	0.042	1.
Perry 1	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	20	2 of 3	T	T	0.020	0.042	1
Pilgrim 1	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	6	1 of 2	M	M	0.020	0.042	1
Quad Cities 1, 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	13	1 of 2	M	M	0.020	0.042	1
River Bend 1	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	18	2 of 3	T	T	0.020	0.042	1
Shoreham 1	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	11	1 of 2	T	T	0.020	0.042	1
Susquehanna 1, 2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	16	1 of 3	T	T	0.020	0.042	1
Vt Yankee	3.5E-4	0.05	0.17	1.0	0.3	1.0	0.3	1.0	6	1 of 2	M	M	0.020	0.042	1
WNP-2	3.5E-4	0.05	0.17	1.0	1.0	1.0	1.0	1.0	18	2 of 3	T	T	0.020	0.042	1

[a]See Table C.2, note *a*, for abbreviations and branch notations.

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C branch probabilities[a]

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(Note 15)	CRD (Note 16)	SRV.ADS (Note 17)	COND/FW.PCS (Note 18)	LPCS (Note 19)	LPCI(RHR)/LPCS (Note 20)	RHRSW/LPCS.LPCI.TRANS (Note 21)	RHRSW/LPCS.LPCI.LOOP (Note 22)	RHRSW/LPCS.LPCI.LOCA (Note 23)	RHR(SDC) Note 24	RHR(SDC)/-LPCI Note 24	RHR(SDC)/LPCI (Note 25)	RHR(SPCOOL)/-LPCI.RHR(SDC) (Note 26)	RHR(SPCOOL)/LPCI.RHR(SDC) (Note 27)	C.I.AND.V/RHR(SDC).RHR(SPCOOL) (Note 28)
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1-E4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.01	4.3E-4	1.0	1.0	1.0	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.1E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.01	4.3E-4	1.0	1.0	1.0	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	4.3E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	4.3E-4	1.0	1.0	1.0	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	4.3E-4	1.0	1.0	1.0	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	7.4E-4	0.5	0.5	0.5	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34
0	0.05	0.031	0.34	0.001	4.3E-4	1.0	1.0	1.0	7.1E-3	6.8E-3	1.0	0.02	0.52	0.34

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Table C.5. 1986 PWR Class A branch probabilities[a]

PLANT	RT (Note 1)	IE/LOOP (Note 2)	EMERG.POWER (Note 3)	AFW (Note 4)	AFW/EMERG.POWER (Note 5)	MFW (Note 6)	PORV.OR.SRV.CHALL (Note 7)	PORV.OR.SRV.RESEAT (Note 8)	PORV.OR.SRV.RESEAT/EMERG.POWER (Note 9)	SS.RELEASE.TERM (Note 11)	HPI (Note 13)	HPR/-HPI (Note 14)	CSR (Note 27)	PORV.OPEN (Note 15)	SS.DEPRESS (Note 16)	COND/MFW (Note 17)	LPI/HPI (Note 18)	LPR/-HPI.HPR (Note 19)	LPR/HPI (Note 20)	HPI(F/B) (Note 24)
Beaver Valley	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2	2 of 3 + BIT	0.04	1.0E-3	0.0	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI
Millstone 3	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2	2 of 3 + BIT	0.04	1.0E-3	0.01	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI
North Anna	3E-5	~0	1 of 2	1 of 3	0.05	M	0.04	2	2	1.5E-2	1 of 3 + BIT	0.04	1.0E-3	0.01	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI
Surry	3E-5	~0	1 of 3	1 of 3	0.05	M	0.04	2	2	1.5E-2	1 of 3 + BIT	0.04	1.0E-3	0.01	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI

[a]See Table C.2, note a, for abbreviations and branch notations.

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Table C.6. 1986 BWR Class B, C,

PLANT	RT (Note 1)	RT/LOOP (Note 2)	EMERG.POWER (Note 3)	AFW (Note 4)	AFW/EMERG.POWER (Note 5)	MFW (Note 6)	PORV.OR.SRV.CHALL (Note 7)	PORV.OR.SRV.RESEAT (Note 8)	PORV.OR.SRV.RESEAT/ EMERG.POWER (Note 9)	SS.RELEASE.TERM (Note 11)
Byron 1	3E-5	~0	1 of 2	1 of 2	0.05	T	0.04	2	2	1.5E-2
Calloway 1	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2
Catawba 1, 2	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	3	3	1.5E-2
Cook 1, 2	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	3	3	1.5E-2
Davis-Besse	3E-5	~0	1 of 2	Note 21	Note 22	T	0.08	1	1	1.5E-2
Diablo Canyon 1, 2	3E-5	~0	1 of 3	1 of 3	0.05	T	0.04	2	2	1.5E-2
Farley 1, 2	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2
Ginna	3E-5	~0	1 of 2	1 of 3	0.05	M	0.04	2	2	1.5E-2
Indian Pt 2, 3	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2
Kewaunee	3E-5	~0	1 of 2	1 of 3	0.05	M	0.04	2	2	1.5E-2
McGuire 1, 2	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	3	3	1.5E-2
Point Beach 1, 2	3E-5	~0	1 of 2	1 of 3	0.05	M	0.04	2	2	1.5E-2
Prairie Island 1, 2	3E-5	~0	1 fo 2	1 of 2	0.05	M	0.04	2	2	1.5E-2
Robinson 2	3E-5	~0	1 of 3	1 of 3	0.05	M	0.04	2	2	1.5E-2
Salem 1, 2	3E-5	~0	1 of 3	1 of 3	0.05	T	0.04	2	2	1.5E-2
Sequoyah 1, 2	3E-5	~0	1 ot 2	1 of 3	0.05	T	0.04	2	2	1.5L-2
Summer	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2
Trojan	3E-5	~0	1 of 2	Note 21	Note 23	T	0.04	2	2	1.5E-2
Turkey Pt 3, 4	3E-5	~0	1 of 2	Note 21	Note 22	M	0.04	2	2	1.5E-2
Wolf Creek	3E-5	~0	1 of 2[b]	1 of 3	0.05	T[b]	0.04	2	2	1.5E-2
Zion 1, 2	3E-5	~0	2 of 3	1 of 3	0.05	T	0.04	2	2	1.5E-2

[a] See Table C.2, note a, for abbreviations and branch notacions.

[b] Assumed.

E, and F b. ch probabilities[a]

SS.RELEASE.TERM/-MFW (Note 12)	HPI (Note 13)	HPR/-HPI (Note 14)	PORV.OPEN (Note 15)	SS.DEPRESS (Note 16)	COND/MFW (Note 17)	LPI/HPI (Note 18)	LPR/-HPI.HPR (Note 19)	LPR/HPI (Note 20)	HPI(F/B) (Note 24)
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2 + BIT	1 of 3	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 3	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 3	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 3	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2[b]	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI
1.5E-2	1 of 2	0.04	0.01	0.036	0.34	1.5E-5	0.67	1.5E-4	0.04 + HPI

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Table C.7. 1986 PWR Class D branch probabilities[a]

PLANT	RT (Note 1)	RC/LOOP (Note 2)	EMERG.POWER (Note 3)	[illegible] (Note 4)	AFW/EMERG.POWER (Note 5)	MFW (Note 6)	PORV.OR.SRV.CHALL (Note 7)	PORV.OR.SRV.RESEAT (Note 8)	PORV.OR.SRV.RESEAT/ EMERG.POWER (Note 9)	SS.RELEASE.TERM (Note 11)	SS.RELEASE.TERM/-MFW (Note 12)	HPI (Note 13)	HPR/-HPI (Note 14)	SS.DEPRESS (Note 16)	COND/MFW (Note 17)	LPI/HPI (Note 18)	LPR/-HPI.HPR (Note 19)	LPR/HPI (Note 20)	HPI(F&B) (Note 24)
Arkansas Nucl 1	3E-5	~0	1 of 3	1 of [illegible]	0.05	T	0.08	1	1	1.5E-2	1.5E-2	1 of 3	0.04	0.036	0.34	5.1E-5	0.67	1.5E-4	3.04 + HPI
Crystal River 3	3E-5	~0	1 of 3	1 of 2	0.05	T	0.08	1	1	1.5E-2	1.5E-2	1 of 3	0.04	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI
Oconee 1, 2, 3	3E-5	~0	Note 26	1 of 3	0.05	T	0.08	1	1	1.5E-2	1.5E-2	1 of 3	0.04	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI
Rancho Seco	3E-5	~0	1 of 3	1 of 2	[illegible]	T	0.08	1	1	1.5E-2	1.5E-2	1 of 3	0.04	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI
TMI 1	3E-5	~0	1 of 3	1 of 3	0.05	T	0.08	1	1	1.5E-2	1.5E-2	1 of 3	0.04	0.036	0.34	5.1E-5	0.67	1.5E-4	0.04 + HPI

[a]See Table C.2, note a, for abbreviations and branch notations.

Table C.8. 1986 PWR Class G branch probabilities[a]

PLANT	RT (Note 1)	RT/LOOP (Note 2)	EMERG.POWER (Note 3)	AFW (Note 4)	AFW/EMERG.POWER	MFW (Note 6)	PORV.OR.SRV.CHALL (Note 7)	PORV.OR.SRV.RESEAT (Note 8)	PORV.OR.SRV.RESEAT/ EMERG.POWER (Note 9)	SS.RELEASE.TERM (Note 11)	SS.RELEASE.TERM/-MFW (Note 12)	SS.DEPRESS (Note 16)	COND/MFW (Note 17)	HPI (Note 13)	PORV.OPEN (Note 15)	HPR/-HPI (Note 14)	CSR (Note 25)	HPI(F&B) (Note 24)
Arkansas Nucl 2	3E-5	~0	1 of 3	1 of 2	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
Calvert Cliffs 1, 2	3E-5	~0	1 of 3	1 of 2	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
Fort Calhoun	3E-5	~0	1 of 2	1 of 2	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
Millstone 2	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
Palisades	3E-5	~0	1 of 2	1 of 2	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
Palo Verde 1,2	3E-5	~0	1 of 2[b]	1 of 3[b]	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
St. Lucie 1,2	3E-5	~0	1 of 2	1 of 3	0.05	M	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
San Onofre 2, 3	3E-5	~0	1 of 2	1 of 2	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI
Waterford 3	3E-5	~0	1 of 2	1 of 3	0.05	T	0.04	2	2	1.5E-2	1.5E-2	0.036	0.34	1 of 2	0.036	1.5E-4	6.8E-4	0.04 + HPI

[a]See Table C.2, note a, for abbreviations and branch notations.

[b]Assumed.

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1. BWR Branch Probability Notes (for Tables C.2—C.4)

1. Failure probabilities developed based on one observation (Browns Ferry failure to scram, NSIC 163405) through 1986. Estimated BWR scrams for this period, at 9.5 scrams per reactor year through 1983 [see NUREG/CR-3591 (Ref. 1)] and 6.3 scrams per reactor year in the f982—1985 period [based on a review of scrams documented in NUREG-0020, Vol. 9, Nos. 6-10 (Ref. 3)] is 2858. This results in a failure probability estimate of 3.5×10^{-4} per demand. Recovery is addressed in the branch for standby liquid control or manual rod insertion (SLC.OR.RODS).
2. Rod insertion or boration failure is assumed to be dominated by operator response. An equipment failure probability of 0.01, combined with an operator failure to initiate of 0.04, results in an overall failure probability of 0.05 per demand.
3. A generic value based on the number of trip initiators that would fault the PCS compared with the total number of trip initiators was developed from data included in the draft of NUREG/CR-3862 (Ref. 4), Table 7. The primary contributors to loss of PCS were assumed to be turbine bypass valve/control valve closure; MSIV closure; and loss of condenser vacuum, of all feedwater flow, and of offsite power. Comparing the frequencies of these initiators with all trip initiators described in Ref. 4, Table 7, results in a likelihood of PCS unavailability, given a reactor trip of 0.17. Note that event specifics would be expected to modify this value substantially.
4. All LOCAs analyzed are assumed sufficiently large to require closure of the MSIVs. Because of this, a failure probability of 1.0 is assumed for this branch.
5. SRVs are assumed to lift on trip except for plants with 100% turbine bypass capacity. For those plants, a 0.3 likelihood of lift has been assumed.
6. SRVs are assumed to lift on all plants.
7. SRVs are assumed to lift except for plants with 100% turbine bypass capacity. For those plants, a likelihood of lift of 0.3 has been assumed.
8. SRVs are assumed to lift on all plants.
9. A failure to close probability of 3.3×10^{-3} per valve demand [see NUREG/CR-2770 (Ref. 5), p. 119] has been assumed. (All SRVs are assumed to lift.) The column in the table lists the number of valves per plant, including ADS valves.
10. For 1984—85 estimates (see Ref. 2), emergency power train failure probabilities were developed based on the assumption that train failures are dominated by diesel generator failures. Based on NUREG-1032 (Ref. 6), pp. 4-6, a value of 0.05 per demand was used for a single diesel failure on demand. This value is consistent with an earlier estimate in NUREG/CR-2497 (Ref. 7) of 0.064 per demand. Using the range of failure probabilities for emergency power systems included in Ref. 6, and assuming a point estimate at the geometric means of these ranges resulted in the following DG failure probability estimates: first diesel, 0.05; second diesel,

0.057; third diesel, 0.19. Emergency-power-system failure probabilities associated with these diesel failure probabilities were then calculated to be: one of two trains required for success, 2.9×10^{-3}; one of three trains required for success, 5.4×10^{-4}; and two of three trains required for success, 8×10^{-3}. Considering data from the 1984—86 precursors and the fractional contribution of BWRs and PWRs, these estimates are still considered valid. The nonrecovery likelihood was revised to 0.80 to reflect all precursor data. The column in the table lists the emergency power system success criteria assumed in analysis.

11. The failure probability is a generic value based on the number of trip initiators resulting in LOFW, given that PCS is failed. Two values were developed from data included in NUREG/CR-3862 draft (Ref. 4), Table 7: one for plants with motor-driven feedwater pumps (for which MSIV closure does not result in LOFW) and one for plants with turbine-driven pumps. These two values are 0.46 for turbine-driven feedwater systems and 0.29 for motor-driven feedwater systems. A nonrecovery likelihood of 0.34 was assumed for turbine-driven feedwater systems, and a value of 0.12 was assumed for motor-driven systems. The differences in these values were intended to reflect the additional difficulty observed in restoring faulted turbine-driven systems. The column in the table lists the type of MFW system.
12. MSIV closure was assumed following a LOCA. Because of this, turbine-driven MFW systems were assumed initially unavailable following LOCA. MFW systems that utilized motor-driven pumps were assumed to trip with a probability of 0.04. Nonrecovery likelihoods assumed were consistent with those specified in note 11. The column in the table lists the type of MFW system.
13. The combined HPCI/RCIC failure probability was estimated to be 1.7×10^{-3}, with an additional nonrecovery likelihood of 0.49 (8.4×10^{-4} overall probability). Based on individual HPCI and RCIC failures observed following LOFW and LOOP, HPCI and RCIC failure on demand probabilities of 0.024 and 0.050, respectively, are estimated. Based on previous reviews of combined HPCI and RCIC failures observed in the Accident Sequence Precursor Program, these failures appeared independent. Using this fact, an estimate of combined HPCI and RCIC unavailability (assuming the independent failure probabilities developed above) is 1.2×10^{-3}. These independent values were raised to result in a combined failure probability consistent with that developed in Chap. 3. This results in an RCIC failure probability of 0.060. Assuming that nonrecovery for HPCI and RCIC could be equally apportioned between the two systems, a nonrecovery likelihood of 0.7 was estimated.
14. Consistent with the development described in note 13, an HPCI failure probability of 0.029 and a nonrecovery likelihood of 0.7 were assumed.
15. All LOCAs analyzed were assumed large enough to preclude the use of RCIC for mitigation. Because of this, a failure probability of 1.0 was used.

16. Use of the control-rod-drive pumps for core coding requires manual operator initiation for success. A nominal system failure probability of 0.01 was assumed, along with a failure to initiate of 0.04. No recovery of initially failed equipment was assumed.
17. A generic failure probability for the automatic depressurization system of 3.7×10^{-3}, as developed in Chap. 3 of this document, was employed, combined with an operator failure to initiate probability of 0.04. A nonrecovery likelihood of 0.71 was additionally assumed.
18. Because the feedwater system is assumed failed before the use of this function is demanded and the condensate system must pump through the feedwater system, it was also initially assumed failed but recoverable with a nonrecovery likelihood of 0.34.
19. The LPCS system was assumed to be a two-train system, each train consisting of two pumps (required) and one closed motor-operated valve. The first train failure probability was assumed to be 0.03, and a common mode—dominated failure probability of 0.1 was assumed for the second train. A nonrecovery likelihood of 0.34 [which is consistent with the typical system nonrecovery estimated in NUREG/CR-3591 (Ref. 1)] was utilized.
20. For 1984—85, the LPCI (RHR) failure probability was developed based on the design of the Browns Ferry LPCI system and applied to other plants with two-train systems. Each train was assumed to contain two parallel pumps with closed, motor-operated suction valves and a single closed, motor-operated discharge valve. The use of an alternate head spray line with two closed motor-operated valves was considered to result in success if the normal discharge valve failed close. Considering these active components, a train failure probability of 0.004 was estimated. This was used with a conditional probability of 0.1 for the second train and a nonrecovery likelihood of 0.34. Based on the precursors observed in 1984—86, the failure probability for the first train was revised to 0.01, and the nonrecovery likelihood was revised to 0.71, resulting in a system failure probability of 7.1×10^{-4}. For three-train systems—including those at Perry, LaSalle, Grand Gulf, River Bend, and WNP-2—the failure probability used was 0.02 for the first train, 0.1 for the second train, and 0.3 for the third train (given failure of the first two train).
21. Considering the design of the LPCI system and the location of the service water connection, the probability of failure for RHR service water is dominated by the likelihood that the LPCI failure mode was the result of the failure of the discharge valve and the alternate head spray injection path. Based on the probabilities used to develop the LPCI failure probability, this value is 0.5. No recovery is assumed given an initial failure because such recovery would be consumed in attempting to recover LPCI in the first place. At Perry, LaSalle, Grand Gulf, River Bend, and WNP-2, no service water connection exists. Instead, the LPCI system has three trains.
22. RHR service water is assumed available if the emergency buses are powered. See note 21 for development of the applicable failure probability.

23. RHR service water flow has been assumed adequate given a small-break LOCA. See note 21 for development of the applicable failure probability.
24. RHR (SDC) failure is dominated by the failure to open the two drop-line suction valves. In addition, if LPCI has not been challenged, then the same failure mechanisms that can fail LPCI are also applicable to RHR (SDC). This system was modeled on the same basis as LPCI but with the addition of a serial component (failure probability of 0.02) and a lower nonrecovery likelihood of 0.34 because RHR (SDC) is not required in the short term.
25. Based on the design of the LPCI and RHR systems on Browns Ferry, the probability of RHR (SDC) failure given LPCI failure is 1.0.
26. Based on the design of the LPCI and RHR systems on Browns Ferry, RHR (SP cooling) failure given LPCI success and RHR (SDC) failure is dominated by failure of the RHR drop-line valves.
27. Based on the design of the LPCI and RHR systems, RHR (SP cooling) success given LPCI and RHR (SDC) failure requires LPCI and RHR (SDC) failure due to injection-valve failure. The likelihood of the SP cooling valves failing to open plus the likelihood that LPCI and RHR (SDC) were failed because of failure of the injection valves result in a failure probability estimate of $0.02 + 0.5 \cong 0.52$.
28. The likelihood of containment injection and venting (C.I.AND.V) failure, given failure of RHR (SDC) and RHR (SP cooling), was assumed to be 0.34, with no likelihood of recovery.
29. The Dresden isolation condenser consists of a heat exchanger and two normally closed dc-powered inlet and outlet isolation valves. Redundant sources of shell-side cooling water are provided. The failure probability for this system has been based on the failure probability of either closed valve (0.01 each). Because these are located in containment, no likelihood of recovery of an initially failed system is assumed.
30. The Dresden core spray system is a two-train system, each consisting of a pump and a normally closed motor-operated valve that must function for train success. A failure probability of 0.02 for the first train and 0.1 for the second train and a likelihood of nonrecovery of an initially failed system of 0.34 were assumed, resulting in an overall failure probability of 6.8×10^{-4}.
31. The LPCI system on Dresden is a two-train system, each train consisting of two parallel pumps and one closed motor-operated discharge valve. The first train failure probability was assumed dominated by the motor-operated discharge valve (failure probability of 0.01). A second train conditional probability of 0.1 was assumed, along with a nonrecovery likelihood of 0.71. This results in a system failure probability estimate of 7.1×10^{-4}.
32. No firewater connection was identified on the Dresden FSAR drawings. Because of this, this function has been assumed not available.

33. The Dresden SDC system consists of three trains, each consisting of a pump, a heat exchanger, and two normally closed motor-operated valves. These three trains are connected to the recirculation loops through normally closed, parallel suction and discharge valves. A failure probability of 0.03 was assumed for the first train, 0.1 for the second, and 0.3 for the third. A failure probability of 0.001 was assumed for the suction and discharge valves, and a nonrecovery likelihood of 3.4 was employed. This resulted in a system failure probability of 2.9×10^{-3}.
34. LPCI (CC) utilizes the same components as LPCI, with the addition of a heat exchanger cooled by service water in each train. Service water is provided by parallel pumps and is not considered to dominate the train failure probability. Because of this, the same failure probability estimated for LPCI is assumed.
35. Because LPCI (CC) utilizes the same components as LPCI, the probability of failure of LPCI (CC) given LPCI failure is 1.0. No recovery likelihood for an initially failed system is assumed because recovery efforts would have been expended on the initially failed LPCI system.
36. IC failure probability for this class was developed based on Millstone 1. One motor-operated (dc power) isolation valve must open to initiate IC cooling. A failure probability of 0.01 was assumed.
37. The feedwater system is realigned to FWCI in the event of a transient. Because the success criteria for feedwater and FWCI are the same, the likelihood of failure is addressed in FWCI feedwater transient (FWCI/FW.TRANS), LOOP, and LOCA.
38. Either of the two feedwater trains provides success for this function. For nonspecific transients, a value of 0.29 was assumed for this branch, with a nonrecovery probability of 0.34. See note 11 for the development of this value.
39. For FWCI given a LOOP, the failure probability was assumed dominated by failure of the emergency power source required for the feedwater pumps. Because of the design of the event trees, this probability is the likelihood of failure of the FWCI power source, given that emergency power has succeeded. For Millstone 1, operability of the gas turbine is required; this is the only plant in the class in which FWCI is powered following LOOP. Based on an assumed gas turbine (Millstone 1) failure probability of 0.1, the failure probability of FWCI feedwater LOOP (FWCI/FW.LOOP) is p (gas turbine fails) $\cong 0.1$.
40. Either of the two feedwater trains provides success for this function. An assumed failure probability of 0.01 for the first train and 0.1 for the second train, plus a nonrecovery likelihood of 0.34, results in an overall failure probability of 3.4×10^{-4}.
41. SRV automatic depressurization system (SRV.ADS) capability is not specified for Big Rock Point. See note 17 for other plants.
42. For Oyster Creek, core spray consists of two trains, each containing two series sets of two parallel pumps plus two parallel discharge valves. Assuming a failure probability of 0.01 for the first pump and valve, 0.1 for the second pump and valve, 0.1 for the second train given failure of the first, and a nonrecovery likelihood of 0.34 results in an overall failure probability of

1.0×10^{-3}. For Millstone 1 and Nine Mile Point 1 (assumed), the system design and, hence, failure probability are similar to Dresden 2 and 3 (see note 30).

43. Firewater connections could not be identified for either Millstone 1 or Oyster Creek. They have been assumed not to exist on the other two plants in the class as well.
44. For Oyster Creek, SDC is similar to Dresden except that single-suction and discharge valves exist to connect the three parallel cooling trains to the recirculation lines. The resulting system failure probability, developed using the same component and non-recovery values, is 7.1×10^{-3}. Nine Mile Point 1 has been assumed to utilize a system similar to Oyster Creek, except only one final component is required for system success. For Millstone 1, two trains of SDC are provided. Each train consists of two closed isolation valves, a pump, and a heat exchanger. Single valves serve to isolate the two parallel trains from the recirculation loops, one of which is closed. Using an approach similar to that for Oyster Creek, the system failure probability is estimated to be 9.9×10^{-4}.
45. Containment cooling for Oyster Creek consists of two parallel pumps and heat exchangers. No closed valves exist in the two trains. Assuming a failure probability of 0.01 for the first train and 0.1 for the second train results in a system failure probability of 1×10^{-3}. For Millstone 1, containment cooling utilizes LPCI with heat exchangers in each train and is therefore unavailable given LPCI failure. The failure probability for Nine Mile Point 1 is assumed similar to Oyster Creek but with a nonrecovery probability of 0.34.

2. PWR Branch Probability Notes (for Tables C.5–C.8)

1. A generic value for failure to trip was applied to all PWRs. This was developed by assuming that the single Salem failure to trip was recoverable with a nonrecovery likelihood of 0.12. Based on an estimated average of 7.33 trips per year over the 1969–83 observation period [based on the number of PWR trips associated with shutdowns reported for 1979 in NUREG/CR-1496 (Ref. 8) plus one-third the number of manual shutdowns also reported (scram assumed as part of shutdown procedure)] and an estimated average of 4.1 trips per year [reported in NUREG-0020, Vol. 9, Nos. 6–10 (Ref. 3)] in 1984–86, a failure to trip probability of 3.4×10^{-5} is estimated.
2. Rods were assumed to trip upon loss of power to the buses powering the control-rod-drive mechanisms. Some small failure probability is acknowledged to still exist, associated with mechanical failure of a number of rods to insert, but it was considered sufficiently small to be ignored.
3. See BWR branch probability notes, note 10.
4. AFW systems are difficult to model because of the use of two different design pumps (motor-driven and turbine-driven) within most systems. For AFW systems, a common-mode failure contribution was considered as a separate serial component instead of being incorporated within each train failure probability estimate. One estimate was assumed for both two- and three-train systems; it was based on the average failure probability estimated in Chap. 3. In addition, the conditional failure probability for the second motor-driven train, if it existed, was increased above that of the first train to reflect common mode coupling between these two components. The following failure probabilities were typically assumed: first motor-driven train, 0.02; second motor-driven train (given failure of the first), 0.1; the turbine-driven train, 0.05; three-train common-mode serial component, 2.8×10^{-4}; nonrecovery, 0.26. These train probabilities result in the following system failure probabilities: one of two trains required for success with one turbine-driven train 3.3×10^{-4}, one of three trains required for success with one turbine-driven train, 9.9×10^{-5}. The closeness of these two failure probabilities is consistent with previous Accident Sequence Precursor Program observations that there was little difference in failure probabilities for two- and three-train AFW systems. The column in the table lists the AFW success criteria at the plant. (See also note 21.)
5. The majority of AFW systems include one turbine-driven train capable of operating without diesel-backed power. On these plants, a failure probability of 0.05 was assumed, with a nonrecovery likelihood of 0.34.
6. An estimate of noncontinuance of feedwater was developed based on the frequencies of types of transients listed in Table 6 of the NUREG/CR-3862 draft.[4] The following trip-related initiators were assumed to render feedwater initially unavailable: total LOFW flow, increased feedwater flow (due to expected feedwater system trip on high SG level), feedwater flow instability, closure of all

MSIVs (at turbine MFW-pump plants only), loss of all condensate pumps, loss of condenser vacuum, loss of circulating water, inadvertent safety injection signal, and LOOP. For plants with turbine-driven pumps, this value is 0.20, for plants with motor-driven pumps, 0.19. A nonrecovery likelihood of 0.34 was assumed in both cases. The column in the table lists the type of feedwater pump at the plant.

7. The challenge rate for primary relief valves was assumed to be 0.04 for plants with U-tube SGs, which is consistent with that employed in NUREG/CR-3591 (Ref. 1). For plants with once-through SGs, this value was raised to 0.08 to reflect the limited response time available before the SGs dry out.
8. A failure to close probability of 0.01 was assumed for each PORV. Failure to isolate an open PORV considered failure of the block valve to close (0.01) plus failure of the operator to initiate closure (0.04). The column in the table lists the number of PORVs on the plant. SRVs are not assumed challenged if PORVs are available.
9. All PORV block valves were assumed to require emergency power for closure. In the event of unavailability of power on these buses, the probability of PORV closure failure was based on 0.01 failure to close probability for each valve alone, without supplemental isolation capability.
10. Deleted.
11. The likelihood of failing to terminate secondary-side release, as a result of either failure to close of an atmospheric dump valve or a turbine bypass valve, was developed using values consistent with those developed in two pressurized thermal-shock studies reported in NUREG/CR-4022 (Ref. 9) and NUREG/CR-4183 (Ref. 10). Based on information included in Appendixes C and B of these two documents, a value of 1.5×10^{-2} per trip was assumed. This is dominated by atmospheric dump valve failures. A nonrecovery likelihood of 0.34 was also assumed.
12. This branch requires continued MFW operation (for DHR), given secondary-side release terminated and, hence, prohibits closure of all MSIVs as a means of isolating a secondary-side blowdown. Because the majority of secondary-side blowdowns are isolable through closure of an associated isolation valve or isolation of only the associated SG, the failure probability for this branch was assumed to be equivalent to that in note 11.
13. For plants without boron injection tanks, the system failure probability was estimated based on the number of active components requiring operation in each train, and assuming conditional failure probabilities that account for common-mode effects for additional trains. For three-train systems with one pump and closed discharge valve, a system failure probability of 3×10^{-4} is estimated, plus a nonrecovery estimate of 0.84. For plants with boron injection tanks, a separate serial component was also used to reflect system failure due to blockage at the boron injection tank. This component utilized a failure probability of 1.2×10^{-3}, again with a nonrecovery estimate of 0.84. The use of the values results in an average nonrecoverable failure probability for HPI consistent with

that developed in Chap. 3. The column in the table lists the system success criteria.

14. High-pressure recirculation given HPI success was developed based on the number of components requiring operation in the low-pressure recirculation system to provide flow from the containment sump to the HPI pump suctions. Typically three components per train were involved: the containment sump isolation valve, the LPI (RHR) pump, and the LPI/HPI cross-connect valve. Based on the values developed in Chap. 3 for long-term core cooling, a failure probability of 1.5×10^{-4} is estimated, with no likelihood of recovery. In addition, a 0.04 probability for the operator failing to open the LPI/HPI cross-connect valves was included, except for combustion plants where the HPI pumps take direct suction from the sump and automatic switchover is assumed. Service water unavailability has been assumed not to contribute significantly to the system failure probability.
15. The probability of the PORV failing to open for bleed and feed was estimated as 0.01. Note that on most plants initiating bleed and feed requires both opening the PORV and initiating HPI; both actions are included in the same procedure. On plants with high-head safety injection pumps [typically Babcock and Wilcox (B&W) plants, but some Westinghouse plants as well], only HPI need be initiated. To account for both of these cases, the operator error probability associated with bleed and feed was assigned to HPI initiation.
16. The likelihood of failing to depressurize the secondary side is dependent on the size of the atmospheric dumps, on whether the MSIVs are closed, or on the operability of the turbine bypass system and condenser if the MSIVs are open. Many plants do not include atmospheric dumps of sufficient size for depressurization, and the analysis assumed that if the MSIVs were closed, secondary-side blowdown could not be achieved. In addition, unavailability of the condenser, due to either loss of vacuum or circulating water, was also considered to fail secondary-side depressurization. Based on the initiator frequencies listed in the NUREG/CR-3862 draft,[4] the likelihood of failing secondary-side depressurization for a nonspecific reactor trip is 0.036. Note that this value is consistent with typical operator failure-to-initiate values and has therefore been applied to other initiator situations as well.
17. See BWR branch probability notes for Table C.1—C.3, note 18.
18. Low-pressure injection is typically independent of HPI, except for the refueling water storage tank (RWST) and associated isolation valves. Estimates of low-pressure injection failure probability were based on the number of trains and number of components required to operate in each train, with consideration of the probability values identified in Chap. 3 for long-term core cooling. The average system failure probability was estimated to be 1.5×10^{-4}, with nonrecovery probability of 0.34.
19. Operability of low-pressure recirculation, given HPI success and HPR failure, is dependent on the mechanism that fails HPR. Because common mode effects typically dominate system failure modes, failure combinations resulting in failure of HPR (note 14) would

consist of the system sump valves, the LPI pumps, or the LPI/HPI cross-connect valves. Of these, failure of the sump valves and pumps would also fail low-pressure recirculation. Because the failure probability for these three items has been assumed the same in this development, the likelihood of LPR failure, given high-pressure recirculation failure, is 0.67.

20. The failure probability for low-pressure recirculation is assumed consistent with the estimate for failure of long-term core cooling developed in Chap. 3.
21. AFW systems for these plants are atypical. The Davis-Besse AFW system is a two-train system utilizing turbine-driven pumps. The Turkey Point system is a three-train system shared between units and consists of three turbine-driven pumps. Trojan utilizes one diesel-driven and one turbine-driven pump. The following system failure probability estimates were employed for these plants: Davis-Besse, 1.7×10^{-3}; Turkey Point, 5×10^{-4}; and Trojan, 8.5×10^{-4}. Each of these estimates included a nonrecovery estimate of 0.34.
22. Both the Davis-Besse and Turkey Point AFW systems were assumed operable without emergency AC power.
23. Only the turbine-driven train on Trojan is operable without emergency power.
24. HPI feed and bleed [HPI(F/B)] consists of the manual initiation of the HPI system for feed and bleed. The system failure probabilities were developed based on note 13. In addition, an operator failure to initiate probability of 0.04 was assumed.
25. The containment spray recirculation system was modeled as a two-train system requiring operation of one pump and one normally closed discharge valve per train. The correct operation of the containment sump valves is also required but is addressed under high-pressure recirculation, note 14. A system failure probability of 6.8×10^{-4} was estimated, including a nonrecovery estimate of 0.34.
26. Oconee utilizes two Keowee hydroelectric units for emergency power. Their combined failure probability was assumed to be 1×10^{-3}.
27. Containment spray recirculation at Beaver Valley provides an alternate source of sump water to the HPI pumps during recirculation. During feed and bleed, containment spray recirculation is assumed required for containment sump cooling; no cooling is provided by LPR. Possible recovery is taken into account in the HPI branch. The possibility of operator failure to align the HPI pump suctions is already accounted for in the HPR branch. The system is thus modeled as a two-train system with no recovery and failure probability of 1×10^{-3}.

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Thirty-five operational events, reported in licensee event reports and occuring at commercial LWRs during 1986, 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 1984-1985 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 initial conclusions from the assessment of 1986 events and from the collective assessment of 1984-1986 events.

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