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Precursors to Potential Severe Core Damage Accidents: 1980–1981

A Status Report

W. B. Cottrell J. W. Minarick

P. N. Austin
E. W. Hagen
J. D. Harris

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

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*Science Applications, Inc.

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NOTE

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FOREWORD

The Accident Sequence Precursor (ASP) Program has attempted to provide a data base of nuclear power plant potentially severe accident experience. This program is complementary to the probabilistic risk assessments (PRAs) currently being performed by the Nuclear Regulatory Commission (NRC) and industry. The information now available in the ASP data base can provide added assurance that the models, data, and assumptions used in PRAs are trustworthy. Nevertheless, the relative rarity of severe events in this data base requires the exercise of great caution in its use and interpretation.

This second report on accident sequence precursors evaluates 1980–81 operational data. Thus, it represents plant experience for the period immediately following the Three Mile Island (TMI) accident. Although many operational and plant configurational changes ensued from the TMI-2 accident on March 28, 1979, these changes have been mandated and implemented in a gradual manner. Consequently, we feel that the 1980–81 period covered by this report is transitional in representing post-TMI-2 plant configuration and operation.

The ASP Program has identified ~230 potentially significant precursors from the operational experience data for the years 1969 through 1981. Some tentative conclusions are drawn in this report from the 1969–81 operational data. For example, there appears to be a downward trend in estimated overall core damage frequency from the 1980–81 data compared with the 1969–79 data. Extensive reviews undertaken by both the NRC and industry have identified the strengths and weaknesses associated with the methods and data presented in this report. It is recognized that a significant amount of subjective judgment was used in the selection of accident sequence precursors. Also, the methods used herein will undergo some modification as a result of the reviews. When the methods are finalized, the 1969–81 data will be reanalyzed and combined with an analysis of the 1982–84 period. The result of this effort should be the most trustworthy portrayal available from operational experience data for accident sequence likelihood and trends over that interval. One might then make stronger inferences concerning overall nuclear industry safety.

Robert M. Bernero, Director
Division of Risk Analysis and Operations
Office of Nuclear Regulatory Research
Nuclear Regulatory Commission

PREFACE

The Accident Sequence Precursor Program was established at the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory in the summer of 1979. The first major report of that program, *Precursors to Potential Severe Core Damage Accidents: 1969—1979 A Status Report* (NUREG/CR-2497), was formally released in June 1982. The second major report, *Pressure Vessel Thermal Shock at U.S. Pressurized Water Reactors: Events and Precursors, 1963—1981* (NUREG/CR-2789), was formally released in April 1983. The present document is a continuation, for the 1980—81 period, of the assessment undertaken in NUREG/CR-2497 for those events that occurred from 1969 through 1979.

The first document (NUREG/CR-2497) was widely reviewed both before and after publication, including a 2-d industry workshop at the Electric Power Research Institute (February 28—March 1, 1983), a critical review by the Institute of Nuclear Power Operations (INPO 82-025), a Research Information Letter (RIL 136) by the Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research, and a review by the Advisory Committee on Reactor Safeguards (ACRS). The ACRS report, *ACRS Report on the Accident Sequence Precursor Study and the Use of Operational Experience* (May 18, 1983), commented favorably on the report and contained many suggestions for additional work — both endorsing and supplementing the many proposals that evolved from the review of the report. In general, industry comments tended to be critical of the use of standardized event trees, which they contended overlooked plant-specific features that would have made some events less significant. Other reviewers commented on the lack of any uncertainty estimate, the mixing of actual statistics and conditional probabilities, and the apparent lack of objectives for the study.

Although the objectives of the first report were necessarily rather vague, that exploratory work has resulted in a better definition of the objectives for subsequent studies. The project objectives were addressed most recently in the NRC *Interim Research Information Letter for the Accident Sequence Precursor Program* (dated October 14, 1983), which lists them as follows:

a. From operational events identify significant or important sequences that, more likely than others, could have led to severe core damage.
b. Search operational events for the elements or precursors of severe core damage accident sequences which are not predicted or poorly predicted in current probabilistic risk analyses (PRA).
c. Analyze operational events to estimate the frequencies and trends of system failures, function failures, and overall frequency of severe core damage as an alternate data source to compare to frequencies estimated in PRAs.

The present report accomplished for the 1980—81 period most of all objectives except the last. Subsequent reports will focus on the remaining objectives and will integrate the data from the 1969—79 and 1980—81 reports. In addition to the three objectives expressly stated in the

Interim Research Information Letter, the data obtained from the evaluation of accident sequence precursors (e.g., function failures, initiating events, human errors, unique sequences, and systems interactions) could be extremely useful to many other safety assessments and safety-related activities.

Initial selection of the 1980–81 precursor events was undertaken in the fall of 1982, and a draft compilation of these events was selectively distributed for comments early in 1983. Among those asked to comment were all of the affected nuclear utilities, the Institute of Nuclear Power Operations (INPO), and many NRC offices. Over 80% of the nuclear utilities responded; their comments, together with those from INPO and the NRC, resulted in some event additions and deletions plus much plant-specific information on the events themselves. Listings of the organizations and individuals who have reviewed all or a portion of the draft of this document are presented in Appendix F. Resolution of these comments required judgment, and it is possible that this judgment is not shared by the NRC, INPO, and each utility.

Among the many reviewers of this document and its various drafts was the ASP Review Team, which was established by the ASP Project to review project work. This team consists of the following three probabilistic risk assessment experts:

Kenneth S. Canady, Duke Power Company;
Norman C. Rasmussen, Massachusetts Institute of Technology;
William E. Vesely, Jr., Battelle Columbus Laboratories.

Although this report is the sole responsibility of the authors and not of its many reviewers, including the Review Team, we are pleased to relate that the members of the Review Team are in essential agreement with the contents of this document.

As noted above, the Accident Sequence Precursor Program is a responsibility of the Nuclear Operations Analysis Center at Oak Ridge National Laboratory. In addition to the NOAC personnel (myself, E. W. Hagen, and J. D. Harris), two subcontract personnel (J. W. Minarick and P. N. Austin) played a major role. I particularly wish to express my appreciation to J. W. Minarick for directing this effort. Both Minarick and Austin are experienced engineers who work for Science Applications, Inc., Oak Ridge. The authors of this report also wish to acknowledge N. B. Gove of the Computer Sciences Division for his valuable support in the development of the computer codes used extensively in this study.

This report must again be viewed as part of a continuing effort. Although the data are improved, the more meaningful results await the identification of significant trends and the determination of the implication of the accident sequence precursor results. Both of these subjects will be topics of future reports.

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LIST OF ACRONYMS AND INITIALISMS

ACRS	Advisory Committee on Reactor Safety
ADS	automatic depressurization system
AFW	auxiliary feedwater
ASP	Accident Sequence Precursor
BWR	boiling-water reactor
BWST	borated water storage tank
CRDM	control rod drive mechanism
CVCS	chemical and volume control system
CS	core spray
DG	diesel generator
DH	decay heat
ECCS	emergency core cooling system
EFW	emergency feedwater
ESFAS	engineered safety features actuation system
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
HPC	high-pressure cooling
HPCI	high-pressure coolant injection
HPI	high-pressure injection
IC	isolation condenser
ICS	integrated control system
INPO	Institute of Nuclear Power Operations
LER	Licensee Event Report
LOCA	loss-of-coolant accident
LOFW	loss of main feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPR	low-pressure recirculation
LWR	light-water reactor
MSIV	main steam isolation valve
MSLB	main steam line break
NNI	nonnuclear instrumentation
NOAC	Nuclear Operations Analysis Center
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
NSSS	nuclear steam supply system
ORNL	Oak Ridge National Laboratory
PORV	pilot-operated relief valve
PRA	probabilistic risk assessment
PTS	Pressurized Thermal Shock Program (at ORNL)
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RT	reactor trip
RWST	refueling water storage tank
SAI	Science Applications, Inc.

SBLC	standby liquid control
SFAS	safety features actuation system
SG	steam generator
SI	safety injection
SLB	steam line break
TMI-2	Three Mile Island, Unit 2

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EXECUTIVE SUMMARY

The Accident Sequence Precursor Study involves the review of Licensee Event Reports of operational events that have occurred at light-water power reactors 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 in which core cooling was not adequate.

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 under the sponsorship of the Nuclear Regulatory Commission Office of Nuclear Regulatory Research. The first major report by the program (NUREG/CR-2497) involved the assessment of events that occurred from 1969 through 1979. The present report involves the assessment of events that occurred during 1980 and 1981.

A nuclear plant has safety systems for mitigating accidents or off-normal initiating events that may occur during the course of plant operation. These safety 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 to calculate the unavailability of plant safety functions (grouped systems that perform the same task). It then uses these calculated unavailabilities, the expected average frequency of initiating events (loss of feedwater, loss of offsite power, loss-of-coolant accidents, and steam line breaks, also determined when possible from the precursors), and event details to evaluate the potential impact of the following two situations:

<u>Safety function unavailability</u>. Given an LER-reported failure of a safety function or partial failures in several functions, the report uses expected initiating event occurrence rates to determine the number of initiating events that will challenge the failed and backup functions during the period associated with the failure. It multiplies the expected challenges by function failure probabilities, using event trees to evaluate the likelihood of the overall event sequence occurring.

<u>Initiating event occurrences</u>. Although standby safety functions are ideally always available, there is a probability that they will fail when called on to mitigate expected accidents or transient-initiating events. The report calculates the likelihood of potential severe core damage for precursors that included initiating events based on expected response of the safety functions. Failed or degraded functions existing at the time of the initiating event are accounted for in the calculations.

For this study, events were selected as precursors if they met one of the following requirements:

— if the event involved the failure of at least one function required to mitigate a loss of main feedwater, loss of offsite power, small-break LOCA, or steam line break;
— if the event involved the degradation of more than one function required to mitigate one of the above initiating events;
— if the event involved an actual initiating event that required safety function response.

Approximately 8400 LERs concerning events that occurred during 1980—81 were screened for accident sequence precursors. Of these, over 390 LERs (4.6%) were selected for detailed review. All LERs selected for detailed review were 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.

As a result of this detailed review, 58 events were selected as accident sequence precursors.

The failure information contained in the precursors was used to estimate average frequencies and failure probabilities for initiating events and functions observed in the study. These estimates are provided in the table shown on the following page.

Initiating event frequency and function failure probability estimates were 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 (unavailability of core cooling), given that the precursor event occurred in the manner it did, and can be considered a measure of the residual protection against potential severe core damage available during the event.

The conditional probabilities associated with each precursor were used to rank precursors as to significance, to identify dominant sequences among all postulated sequences to potential severe core damage, to rank functions as to their importance in maintaining the current level of protection against potential severe core damage and providing additional protection, and last to estimate an industry-average potential severe core damage frequency.

The following limited and tentative conclusions have been drawn from the findings detailed in this report (differences in industry performance between the 1969—79 and 1980—81 periods will be the subject of future work).

— Approximately the same total number of precursors per reactor year were seen in 1980—81 as in 1969—79, but their significance is less. The reduced significance appears to be the result of some improvement in system reliability, the availability of alternate features that can provide additional protection against potential severe core damage, and a decrease in the degree of coupling observed in the precursors.

Initiating event frequencies and function failure probabilities determined from precursor data[a]

	Point estimate
BWR functions	
HPCI/RCIC failure	2.2E-3
Emergency power failure	2.2E-3
Automatic depressurization system failure	6.7E-3
Reactor scram failure	1.9E-4
Reactor isolation failure (large SLB)	2.3E-3
Long-term core cooling failure	1.0E-4
PWR functions	
AFW failure given reactor trip success	2.7E-4
HPI failure given AFW success	6.0E-4
Long-term core cooling failure	2.6E-4
Emergency power failure	3.7E-4
Steam generator isolation failure (large SLB)	6.4E-4
Concentrated boric acid addition failure given HPI success (large SLB)	8.3E-4
Initiators (value per reactor-year)	
BWR LOOP	1.9E-2
BWR LOCA	2.1E-2
PWR LOOP	2.8E-2
PWR LOCA	8.9E-3

[a]These estimates are based on amalgamated 1969—81 failure data and are average estimates across the reactor population. Plant-specific estimates can vary substantially from these values.

— Precursors involving coupled failures were still observed, primarily caused by electrical faults. Furthermore, failures in continuously operating cooling water systems were also observed in 1980—81, whereas these failures were not observed to the same extent in the 1969—79 period.

— In 1980—81 the effective number of PWR initiating events and function failures (with potential recovery considered) was less than the expected number (based on 1969—79 data) in almost all cases. For a particular function or initiator, this result is probably not significant because of the large variance of the estimates; however, the systematic effect over all the items is believed to be a demonstration of improved performance. Boiling-water-reactor initiating events and function failures do not show this same trend.

— The estimated industry-average potential severe core damage frequency based on the 1980–81 precursors (1.6E-4/reactor-year) decreased from the revised estimate (of 2.3E-3/reactor-year) for 1969–79 precursors by an order of magnitude. This is a result of the decreased conditional probabilities associated with the 1980–81 precursors, as discussed above. The revised 1969–79 estimate is calculated on the same basis as the 1980–81 estimate in a separate report which is in preparation. This estimate could change slightly as the report is finalized. Significant uncertainty is associated with this average estimate, and the observed decrease in the later time period should only be interpreted as indicative of a downward trend.

— The dominant potential severe core damage sequences identified in the 1980–81 precursors are generally consistent with those identified to date in PRAs, although some unique failure modes and system interactions were observed. As a result, ~40% of the PWR estimate was attributable to precursor sequences that were not easily modeled using the standardized event trees (event trees for loss of main feedwater, loss of offsite power, small-break LOCA, and steam line break) developed for use in this program. Dominant sequences for PWRs were split between those associated with small-break LOCAs and transients and for BWRs were associated predominantly with transients.

— Importance analyses were performed to identify those functions providing the greatest present protection against potential severe core damage. These functions are BWR long-term core cooling, PWR auxiliary feedwater, BWR scram, PWR high-pressure injection, and PWR long-term core cooling. From the standpoint of additional risk reduction, feed and bleed for PWRs and long-term core cooling for BWRs were most significant.

The estimates developed in this report are subject to considerable uncertainty due to the limited data available, the assumptions that had to be made, and the analysis approach itself. (Specific sources of underestimation and overestimation are discussed in Sect. 3.2.4. Chapter 6 presents the results of sensitivity and uncertainty analyses.)

Various aspects of the ASP methodology and the results obtained for the 1980–81 period are discussed in this report both where the topics first arise and subsequently in Chaps. 7 and 8. Finally, Chap. 9 provides an overview of report conclusions.

PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1980—1981 A STATUS REPORT

W. B. Cottrell J. W. Minarick*

P. N. Austin*
E. W. Hagen
J. D. Harris

ABSTRACT

Descriptions of 58 operational events, reported in Licensee Event Reports (LERs), that occurred at commercial light-water reactors during 1980—81 and are considered to be precursors to potential severe core damage are presented along with associated event trees, categorization, and subsequent analyses. This study is a continuation of the work presented in NUREG/CR-2497, which somewhat similarly evaluated the 1969—79 events. The current study incorporates improvements that evolved from an assessment of the comments on the earlier report and applies these in the assessment of the LERs that occurred during 1980—81. The report sequentially discusses (1) the general rationale for this study, (2) the program methods for LER review and documentation, (3) the calculation of function failure probabilities and initiating event frequencies based on precursor data, (4) the use of the conditional probability of subsequent severe core damage estimates to rank precursor events and estimate an average industrywide risk of severe core damage, (5) the conduct of sensitivity analyses on these results, and (6) the application of program results to current concerns pertaining to nuclear power plant risk assessment. There was some apparent decrease in most initiating event frequencies and function failure probabilities in the 1980—81 period as compared with those reported in NUREG/CR-2497. Although it was not possible to conclude that all decreases were statistically significant, in conjunction with other factors they indicate a reduction in the average estimated severe core damage frequency for the nuclear industry for 1980—81 as compared with the 1969—79 period.

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1. INTRODUCTION

The Accident Sequence Precursor Study involves the review of Licensee Event Reports on operational events that have occurred at light-water power reactors beginning in 1969 to identify and categorize precursors to

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potential severe core damage accident sequences. The present report is a continuation of the work published in 1982, *Precursors to Potential Severe Core Damage Accidents: 1969—1979 A Status Report*.[1] This report details the work of the ASP Study in its review and evaluation of operational events that occurred in 1980—81 and were reported by LERs. The requirements for LERs are described in Regulatory Guide 1.16 (Ref. 2). Work on the present document began in 1982 and was essentially completed by mid-1983, except for the incorporation of comments.

1.1 Background

The ASP Study owes its genesis to the conclusions of the Risk Assessment Review Group, which states in its report[3] that "unidentified event sequences significant to risk might contribute . . . a small increment . . . [to the overall risk]." The report continues, "It is important, in our view, that potentially significant (accident) sequences, and precursors, as they occur, be subjected to the kind of analysis contained in WASH-1400 [Ref. 4]." Such an evaluation was done for the 1969—79 period (NUREG/CR-2497), which was the first effort in this type of analysis.

In many ways the present report — except for the time period covered — is similar to the first.

Accident sequences of interest in this study are those that if completed, would have resulted in inadequate core cooling in the short term, up to typically 20—30 min, and potentially resulted in severe core damage. Accident sequence precursors of interest 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 the same screening processes and a similar quantification methodology as those used in NUREG/CR-2497 (Ref. 1). Discussed in more detail in Chap. 3, this methodology permits a reasonable quantification 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.

At the same time, there are significant differences between this report and the first report which reflect the incorporation of some of the more substantive comments from the review of the first report. These differences include but are not necessarily limited to the following:

- elaboration of objectives,
- elaboration of ASP methodology,
- incorporation of more plant-specific data into the assessment of events,
- a reassessment of recovery factors,
- inclusion of sensitivity analyses,
- more precise definitions and use of terms (including a glossary).

A study of this nature is subject to certain inherent deficiencies and biases. The results are no better than the data from which they were derived (and LERs have many problems), and the study may be biased by many of the decisions inherent to the process as well as to the methodology itself. However, a determined effort has been made in this report to identify and discuss (and to quantify where possible) these problems as they come up. In any event it should be recognized by all concerned that this study, based as it is on historical data, is an assessment of past likelihood of potential severe core damage (during the period 1980 to 1981, inclusive) and does not attempt to predict what that likelihood is at the present time or what it will be at some future time. Such extrapolations can, of course, be made, but they involve an assessment of plant differences in the time periods being compared.

1.2 Objectives

The objectives of the ASP Study have been a source of concern since the publication of the first report. The objectives over the 4 years of the project's existence have become progressively more specific. The *Interim Research Information Letter*[5] published by the NRC Office of Nuclear Regulatory Research in October 1983 describes the objectives as follows:

a. From operational events identify significant or important sequences that, more likely than others, could have led to severe core damage.
b. Search operational events for the elements or precursors of severe core damage accident sequences which are not predicted or poorly predicted in current probabilistic risk analyses (PRA).
c. Analyze operational events to estimate the frequencies and trends of system failures, function failures, and overall frequency of severe core damage as an alternate data source to compare to frequencies estimated in PRAs.

The program is expected to attain these objectives by using the light-water-reactor LER file to identify those events that are actual or potential accident sequence precursors, to perform analysis and modeling of the selected events for trends, and to estimate an overall severe core damage frequency for the population of operating U.S. commercial nuclear power plants.

Although the objectives of this report are somewhat more modest than those of the entire program, the work in this report is of course intended to help fulfill the program objectives. With this in mind, the more limited objectives of this report are

1. to examine the 1980—81 LERs for potential severe core damage accident precursors;
2. to incorporate the major and/or significant comments received from the review of the previous report,[1] particularly as regards the methodology and the incorporation of plant-specific information;

3. to estimate event frequencies and failure probabilities for certain classes of events;
4. to estimate the conditional core damage probabilities for the selected events as well as the industrywide core damage frequency for the 1980—81 time period;
5. to include sensitivity analyses and parametric studies to help assess the relevancy of precursor information in light of the uncertainties involved.

1.3 Organization of the Report

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

Chaps. 2 and 3	—	description of the ASP methodology;
Chaps. 3 and 4	—	detailed review of LERs selected as significant precursors;
Appendix B	—	identification, description, and categorization of events considered to be accident sequence precursors;
Chap. 5	—	quantification of precursors;
Chap. 6	—	results of sensitivity analysis;
Chap. 7	—	discussion of program methods and limitations;
Chap. 8	—	discussion of results;
Chap. 9	—	conclusions.

In addition, a List of Acronyms and Initialisms, an Executive Summary, and a Glossary are provided.

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), Oak Ridge National Laboratory, June 1982.

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.16, Rev. 4, *Reporting of Operating Information, Appendix A: Technical Specifications*, August 1975.

3. U.S. Nuclear Regulatory Commission, *Risk Assessment Review Group Report*, NUREG/CR-0400, September 1978, p. 15.

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

5. R. B. Minogue, Director, Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, *Interim Research Information Letter for the Accident Sequence Precursor Program*, RIL 136, Oct. 14, 1983.

2. PROGRAM OVERVIEW

2.1 Public Risk and Potential Severe Core Damage

Nuclear plant risk assessments have concluded that public health risk is dominated by accidents involving severe damage to the reactor core as well as prompt failure of containment systems. If both of these occur, then significant quantities of radioactive materials can be released to the environment. However, such an event would require the failure of a series of protective features designed first to prevent radiation release from the fuel, then from the reactor primary system, and finally from the reactor building to the environment. A brief consideration of these events and the protective features designed to prevent or mitigate them is helpful in placing the ASP Study effort in perspective.

Operational occurrences at nuclear power plants which can initiate a sequence of events potentially leading to radiation release are known as "initiating events." Initiating events range from expected events such as reactor trips caused by out-of-tolerance instruments (reactor trips occur with a frequency of about six per reactor-year) to unexpected events such as a large loss-of-coolant accident (with an expected frequency on the order of once in ~10,000 reactor-years). Different initiating events require different functional response by various plant systems to shut down the reactor, provide inventory makeup, and remove decay heat. If these systems or their backups do not perform adequately, core damage could occur. If significant core damage occurs and containment systems also fail, then large amounts of radioactive material could be released to the environment. When such a release occurs, characteristics at the plant site, such as the weather and emergency response, will further affect the extent, if any, to which public safety will be impacted. This sequence of events is shown in Fig. 2.1. To minimize the chance that core damage will occur following an initiating event, redundant systems are provided to perform the required safety functions. In such systems, component failures (to a certain extent) can be tolerated and the function will still be successful. Furthermore, depending on such things as the plant power level, the combinations of systems that are operating, and the nature of the initiating event, reactor shutdown and core cooling systems can operate below their design-basis levels and still be effective. However, below some minimum level of performance core damage will begin to occur, and at some further reduced level of performance (when core cooling is completely inadequate for a period of time) core melting will occur.

The difference between the operation of a system at design-basis level and at degraded levels where core damage begins or at further degraded levels where core melting results for a particular initiating event is very difficult to determine; this determination is typically not done in defining minimum operability for analysis purposes. Instead, minimum system operability (success criteria) is usually defined in terms of <u>proper</u> operation of a minimum <u>set</u> of components in the system (for example, if a system is a two-train redundant system, then minimum system operability may require components in one of the two trains to operate correctly). Then, if minimum system operability is not provided by each

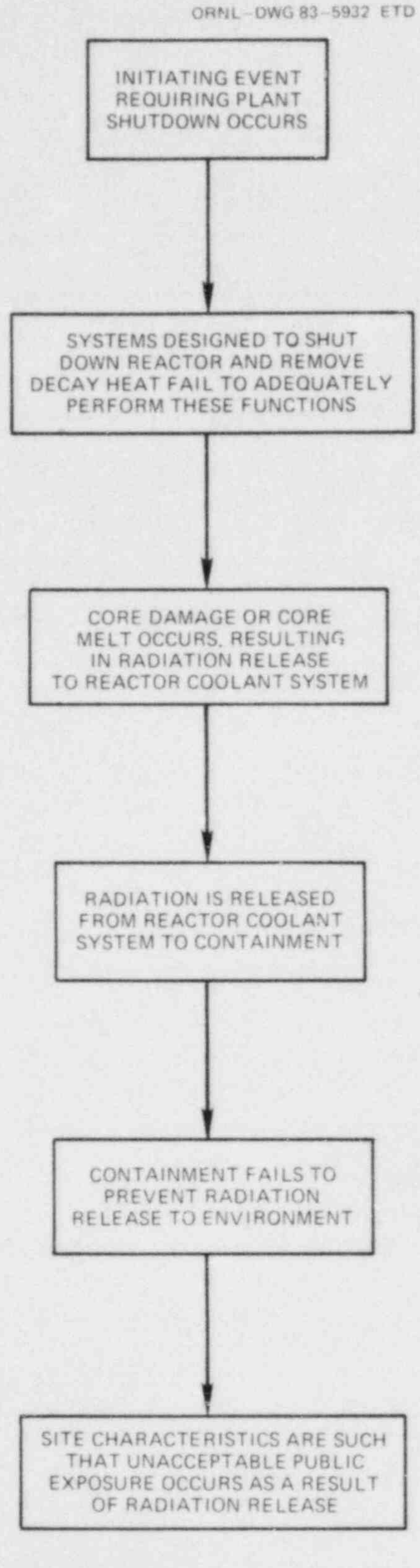


Fig. 2.1. Combination of steps required for public exposure from reactor plant initiating events.

of the systems required for reactor shutdown and core cooling following an initiating event, an unacceptable core state is assumed to have occurred. This state is variously called "core melt," "core damage," or "severe core damage," even though actual core damage may not result unless further system degradation occurs.

If actual core damage does occur, then progression to actual core melt is possible. However, such a progression, if it occurs, could take some time, during which the progression may cease because of the availability of some core cooling or because of recovery actions taken at the plant. In any event, core melt is a less probable condition than core damage, which is in turn less probable than failing to meet minimum system operability requirements. The difference in probabilities between these states is extremely difficult to quantify and estimates vary widely.

In the Accident Sequence Precursor Study, failure to meet an established set of minimum system operability requirements following an initiating event is defined as "potential severe core damage."

2.2 Estimating Potential Severe Core Damage

If a very large number of reactor-years (e.g., 50,000) of commercial nuclear power plant experience existed, it would be possible to better estimate the frequency of the state previously defined as "potential severe core damage" by counting the number of actual events that had occurred and dividing by the number of reactor-years. Experience with such events would also provide information concerning the likelihood of proceeding to actual core damage and core melt and possibly concerning the extent of expected radiation release. Because potential severe core damage events are rare and the number of reactor-years with respect to these events is relatively small, frequency estimates must be made in other ways.

As previously discussed, probabilistic risk assessment has been the method used to estimate the probability of potential severe core damage. PRA provides an integrated set of methodology areas to model and assess plant reliability and to estimate the potential for severe core damage. These areas can be grouped into five major categories as shown in Table 2.1. The individual tasks within each category are explained in detail in NUREG/CR-2300, *PRA Procedure Guide*.[1]

Many of the models, data, and assumptions used in the PRA tasks shown in Table 2.1 are subjectively based and have large uncertainties associated with them. The ASP Program can provide a better basis for some of the PRA methodology areas and tasks by validating the methods and ensuring that the data and assumptions are consistent with the relevent experience. For example, task 1.1, initiating event identification, requires that a comprehensive list of initiating events be developed based on evaluation of past experience and on plant-specific evaluations. The ASP Program provides detailed analyses of events that could verify the lists of initiating events used in PRAs. If significant events are occurring that cannot be grouped into the more standard initiating event categories, then this aspect of the PRA methodology would need to be modified to better reflect these important observations.

Table 2.1. Probabilistic risk assessment methodology areas and outline of tasks

1. Initiating events
    - 1.1 Initiating event identification
    - 1.2 Initiating event frequencies
2. Accident sequence delineation
    - 2.1 Aggregation of initiating events
    - 2.2 Event tree construction
    - 2.3 Initiating event—event tree coupling
    - 2.4 System dependencies
    - 2.5 System interactions
    - 2.6 Boolean reduction/sequence analysis
3. Plant systems modeling
    - 3.1 System success criteria
    - 3.2 Top event—fault tree coupling
    - 3.3 Postaccident heat removal transitions
    - 3.4 Treatment of human error
    - 3.5 Electrical and logic systems impact
    - 3.6 Treatment of recovery
4. Accident sequence analysis
    - 4.1 Data base development
    - 4.2 Parameter estimation
    - 4.3 Data manipulation
    - 4.4 Estimation of human error probabilities
    - 4.5 Treatment of recovery
    - 4.6 Postaccident cutoff time
    - 4.7 Common-cause analysis
    - 4.8 Component environmental qualification
5. General considerations
    - 5.1 Assumptions
    - 5.2 Completeness

2.3 Accident Sequence Precursor Identification

The ASP Program is concerned with identifying and documenting parts of potential severe core damage accident sequences that have been historically observed and with estimating probabilities associated with them. Sequences possibly leading to potential severe core damage were developed by first identifying types of initiating events that require response by plant systems to provide continued core cooling and then identifying the combinations of plant systems that can provide that cooling. Once this was done, system failures that could prevent adequate core cooling were identified. Because of the variety of detailed system designs among plants and the fact that support systems are frequently required for operability, systems were functionally described. This permitted grouping of like systems in a generic way.

System failure combinations, together with the applicable initiating events, were considered potential severe core damage accident sequences. Historic operational events were reviewed, and those that impacted one or more steps in such sequences were selected as accident sequence precursors if (1) the operational event involved the failure of at least one of the systems included in an accident sequence, (2) the operational event involved degradation of more than one system included in the set of accident sequences associated with a particular initiating event, or (3) the operational event included an initiating event that required response by plant systems to provide continued core cooling.

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 one of a series of protective features designed to prevent core damage was compromised. The likelihood of potential severe core damage while an accident sequence precursor existed depends on the effectiveness of the remaining protective features and, in the case of precursors that did not include initiating events, the chance of such an initiator.

Reference

1. Nuclear Regulatory Commission, *PRA Procedure Guide*, NUREG/CR-2300, Vol. 1, January 1983.

3. ACCIDENT SEQUENCE PRECURSOR IDENTIFICATION AND QUANTIFICATION

3.1 Accident Sequence Precursor Identification

The Accident Sequence Precursor Program is concerned with identification and documentation of those portions of potential severe core damage sequences that have been historically observed and with the estimation of frequencies and probabilities associated with them.

For core damage to occur, fuel temperature must increase. Such an increase requires the heat generation rate in the core to exceed the heat removal rate. This can result from either a loss of core cooling or excessive core power. Most initiating events that can potentially result in these conditions can be associated with three initiating event classes — loss of normal feedwater (loss of heat removal from the primary coolant), loss-of-coolant accidents (loss of primary coolant inventory), and steam line breaks (excessive heat removal from primary coolant). An additional initiating event, loss of offsite power, is frequently considered because of the required emergency power system response, although in other respects the initiator is similar to a loss of normal feedwater. Anticipated transients without scram or overpower events are modeled within the framework of these event classes if they involve such events. Otherwise they are depicted using unique event sequences.

Functionally based mitigation sequences (standardized event trees) were developed for the above four initiating event classes. Based on previous experience with reactor plant operational events, it was felt that most initiating events could be directly or indirectly associated with these initiators. Detailed descriptions of the four event trees for both PWRs and BWRs are included in Appendix A. Initiating events that could not be associated with one of these initiators could be accommodated by developing unique sequences for the event. (Approximately 20 unique event trees have been used to date in the program.)

The sequences leading to potential severe core damage were functionally described because of the limited amount of operational experience. This permitted functions from somewhat different plant designs to be considered together and resulted in a greater number of observed functional events, albeit with less confidence that the events were representative of all the plant designs.* With the primary sequences to potential severe core damage established in functional terms, operational events could be reviewed with respect to them.

Descriptions of operational events provided through the LER system were used in the study. Although LERs were not required until mid-1975, event reports comparable to LERs existed before the inception of the LER

*In this study compensation is made for the omission of plant-specific detail in the event trees through the assignment of the recovery factors associated with function failures and through the use of plant-specific data tailoring.

system and are considered to be LERs for the purpose of this study. [The requirements of LERs are described in Regulatory Guide 1.16 (Ref. 1).]

Identification of precursors involved a two-step process. First, an abstract of each LER was reviewed to determine if the event should be reviewed in detail. Events selected then were subjected to an in-depth review to identify those events considered to be precursors to potential severe core damage accidents.

The 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.

Specific LERs were chosen for detailed review if any of the following criteria were met:

- any failure to function of a system that should have functioned as a consequence of an off-normal event or accident,
- any instance where two or more failures occurred,
- all events that resulted in or required initiation of safety-related equipment (except events that only required reactor trip and when trip was successful),
- any event or operating condition that was not enveloped by or proceeded differently from the plant design basis, and
- any other event that, based on the reviewer's experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

For two reasons, only events that occurred after initial criticality were selected for detailed review: (1) a core was considered vulnerable to severe core damage only after initial criticality and (2) in the precritical period, distinguishing initial testing (system-checkout) failures from operational failures is sometimes difficult. Additionally, because the study was concerned only with operational failures, design errors discovered by reanalysis were not considered. (Design errors that caused an operational event were considered in the study.)

Four 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 subsequent identification of an operational event as a precursor, once the event is selected for detailed review, is fairly well defined, the selection of an LER for review is somewhat judgmental. Events selected in the study were more serious than most, and it is expected that the majority of the LERs selected for detailed review would 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 abstracts. The LER abstracts stored in the Nuclear Safety Information Center data file are frequently based on a written abstract of the event developed from the LER rather than on a detailed review of each LER event. If the abstract of a potentially important LER does not show that the LER deserves review, then it will likely be missed.

3. Lack of appropriate information in the LER itself. Licensee Event Reports are frequently written to fulfill a legal commitment rather

than to provide engineering data. Because of this, an LER may not provide a complete description of an event of interest.

4. Specificity of the LER reporting system. Licensee Event Reports are required to be filed when plant Technical Specifications are violated or limiting conditions of operation are entered. These requirements are described in Regulatory Guide 1.16 and are dependent on the detailed wording of each plant's Technical Specifications. Because of this, certain events of interest may not be reported. The scope of this study included only events reported via the LER system. (In particular, reporting of LOFW events is not required.)

The operational events selected in the initial screening and selection process were then subjected to an in-depth review 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 but also used Final Safety Analysis Reports, 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 was immediately detectable and occurred while the plant was at power, then it was evaluated according to the likelihood that the event and the ensuing plant response could lead to severe core damage.

2. If the event or failure was immediately detectable but occurred while the plant was not at power, then it was evaluated according to the likelihood that the event plus the plant response could have led to severe core damage if it had occurred while at power or at hot shutdown immediately following power operation.

3. If the event or failure had no immediate effect on plant operation (e.g., if no initiating event occurred), then it was evaluated based on the likelihood of severe core damage from a postulated initiating event (during the failure period) that would require the failed items for mitigation.

For each actual occurrence or postulated initiating event associated with an LER event, the sequence of operation of various mitigating functions required to prevent potential severe core damage was considered. Events were selected as precursors to potential severe core damage accidents (accident sequence precursors) if they met one of the following criteria:

1. The operational event included the total failure of at least one function included on an event tree modeling potential plant response to a loss of feedwater, loss of offsite power, small-break LOCA, or steam line break (applicable event trees are described in Appendix A).

2. The operational event included degradation of more than one of the functions included on the event trees.

3. The operational event included an initiator that required safety system response.

Operational events that could not be accommodated on the event trees described in Appendix A were selected if they met equivalent requirements.

3.2 Accident Sequence Precursor Quantification

A conditional probability of subsequent potential severe core damage was calculated for each precursor. This calculation assumed that the failure probabilities associated with observed failures were equal to a recovery factor that measures the likelihood of failing to recover from the failure; the failure probabilities associated with observed successes and with functions unchallenged during a historic occurrence were equal to an industrywide average failure probability calculated from the totality of precursors. The precursor conditional probability calculated in the study assumes that failures observed during an event were coupled, but observed successes were not. This calculation is useful in ranking because it permits estimation of the measure of protection remaining once the failures have occurred. These calculations are described in the following sections.

3.2.1 Determination of recovery factors, initiating event frequencies, and function failure probabilities

As a consequence of the study selection criteria, certain initiating events (see Sect. 3.1) and all complete failures of mitigating functions included in the event trees developed for loss of feedwater, loss of offsite power, small-break LOCA, and steam line break were selected as precursors. Initiating event frequencies were calculated based on the aggregate event occurrence experience during the number of reactor-years of operation in the observation period, assuming such frequencies can be used to describe frequencies at individual plants for the purposes of this study. Failure probabilities were calculated based on observed failure occurrences and on an estimate of the total number of demands (including tests and additional nontest demands) to which the function would be expected to respond. The failure information obtained from the precursors was qualified in several ways to provide reasonable frequency and probability estimates.

Determination of recovery factors. The chance of recovery was included by considering each failure to be composed of the observed failure and a subsequent recovery step. Four recovery classes were defined 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, considering the event specifics. The assignment of an event to a class and the numeric value assigned to each class were based on engineering judgment. The likelihood of recovery 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. The effective number of nonrecoverable events associated with an initiating event or failed function was then determined by summing the recovery class numeric value assigned to each applicable event. Sensitivity and uncertainty analyses performed as

Table 3.1. Description and quantification of recovery classes

Recovery class	Description	Likelihood of failing to recover from event	
		Numeric value range	Average numeric value[a]
R1	Failure did not appear to be recoverable in required period, either from control room or at failed equipment	0.3–1.0	0.58
R2	Failure appeared recoverable in required period at failed equipment, and equipment was accessible; recovery from control room did not appear possible	0.1–0.8	0.34
R3	Failure appeared recoverable in required period from control room, but recovery was not routine	0.03–0.3	0.12
R4	Failure appeared recoverable in required period from control room and was considered routine	0.01–0.1	0.04

[a]See Chap. 6 for development of average values from recovery value ranges. These average values are used for consistency of analysis. The actual likelihood of failing to recover from an event is difficult to assess and may vary substantially from the values listed above.

part of this study (described in Chap. 6) considered the impact of alternate numeric values for the different recovery classes.

Determination of initiating event frequencies and function failure probabilities. Initiating event frequencies were determined by dividing the effective number of nonrecoverable initiating events by the number of reactor-years under observation. Function failure probabilities were determined by dividing the observed number of nonrecoverable failures by the estimated number of aggregated demands for the function. The number of demands was estimated based on an assumed test frequency (typically once per month per plant) plus consideration of other situations (such as a normal shutdown or startup) that could require operation.

The nature of the operational data base is such that the following initiating event frequency and function failure probability information has been estimated:

Initiating event frequencies

PWR loss of offsite power (>30 min)
BWR loss of offsite power (>30 min)
PWR small-break LOCA
BWR small-break LOCA

Function failure probabilities

PWR AFW failure
PWR HPI failure
PWR long-term core cooling (sump recirculation) failure
PWR emergency power failure
PWR steam generator isolation failure
PWR HPI for steam line break mitigation (concentrated boric acid addition) failure
BWR scram failure
BWR HPCI/RCIC failure
BWR ADS failure
BWR emergency power failure
BWR reactor vessel isolation failure
BWR long-term core cooling failure

Note that the method employed in this study to estimate unavailabilities (number of failures per number of demands) is one of several that can be used. Alternate methods (e.g., downtime divided by total time or number of failures divided by twice the number of tests) would provide other estimates.

Because of the number of unique system designs in plants that went critical before 1969, only failure and demand data associated with plants that went critical after January 1, 1969,* were considered in the industry-average initiating event frequency and demand failure probability calculations. In addition, certain initiating event frequencies and demand failure probabilities used in the severe core damage probability calculations could not be determined from information in the precursor data. In such cases, plant design specifics, previous experience, and engineering judgment were used in the estimation of these values.

Sample calculation. One example involving the estimation of the failure-on-demand probability for PWR auxiliary feedwater illustrates this calculational process. Eight events concerning auxiliary feedwater failures were identified in a 288 reactor-year period (from 1969 through 1981) and were assigned to recovery classes (Table 3.2), based on reviews of each event. Use of the previously identified numeric values (Table 3.1) for the recovery classes results in an effective number of nonrecoverable

*The following plants were critical prior to January 1, 1969: Big Rock Point, Dresden 1, Haddam Neck, Humboldt Bay, Indian Point 1, LaCrosse, San Onofre 1, and Yankee Rowe.

Table 3.2. Illustration of sample calculation showing recovery classes for AFW failures

Date of event	Plant	Failure description	Recovery class
06-18-73	Turkey Point 4	Failure of pumps to auto-start due to failure to install fuses	R4
04-07-74	Point Beach 1	Failure to deliver flow due to clogged suction strainers	R2
05-08-74	Turkey Point 3	Failure of pumps to start due to overtightened packings and controller malfunction	R1
11-05-75	Kewaunee	Failure to deliver flow due to clogged suction strainers	R2
12-11-77	Davis-Besse 1	Loss of AFW pump control due to mechanical binding and blown control power fuses	R2
03-25-78	Farley	Failure of turbine-driven pumps to start plus open bypass valves	R2
03-28-79	TMI-2	Failure of AFW to deliver flow due to closed valves	R3
12-11-81	Zion 2	Failure of two AFW pumps to start with third pump unavailable	R4

failures on demand of 0.04 + 0.34 + 0.58 + 0.34 + 0.34 + 0.34 + 0.12 + 0.04 = 2.14. For the auxiliary feedwater function, the number of demands was estimated to be 27.3/reactor-year, or 7862 for the 288 reactor-year period. The probability of failure on demand is then estimated as 2.14 failures per 7862 demands, or 2.7E-4.

3.2.2 Calculation of conditional probability associated with each precursor

A conditional probability of potential severe core damage associated with each precursor was calculated using applicable sequences from the sequence of interest event trees developed for each precursor event by applying appropriate failure probabilities to each event tree branch. A typical sequence of interest tree developed for an event is shown in Fig. 3.1.

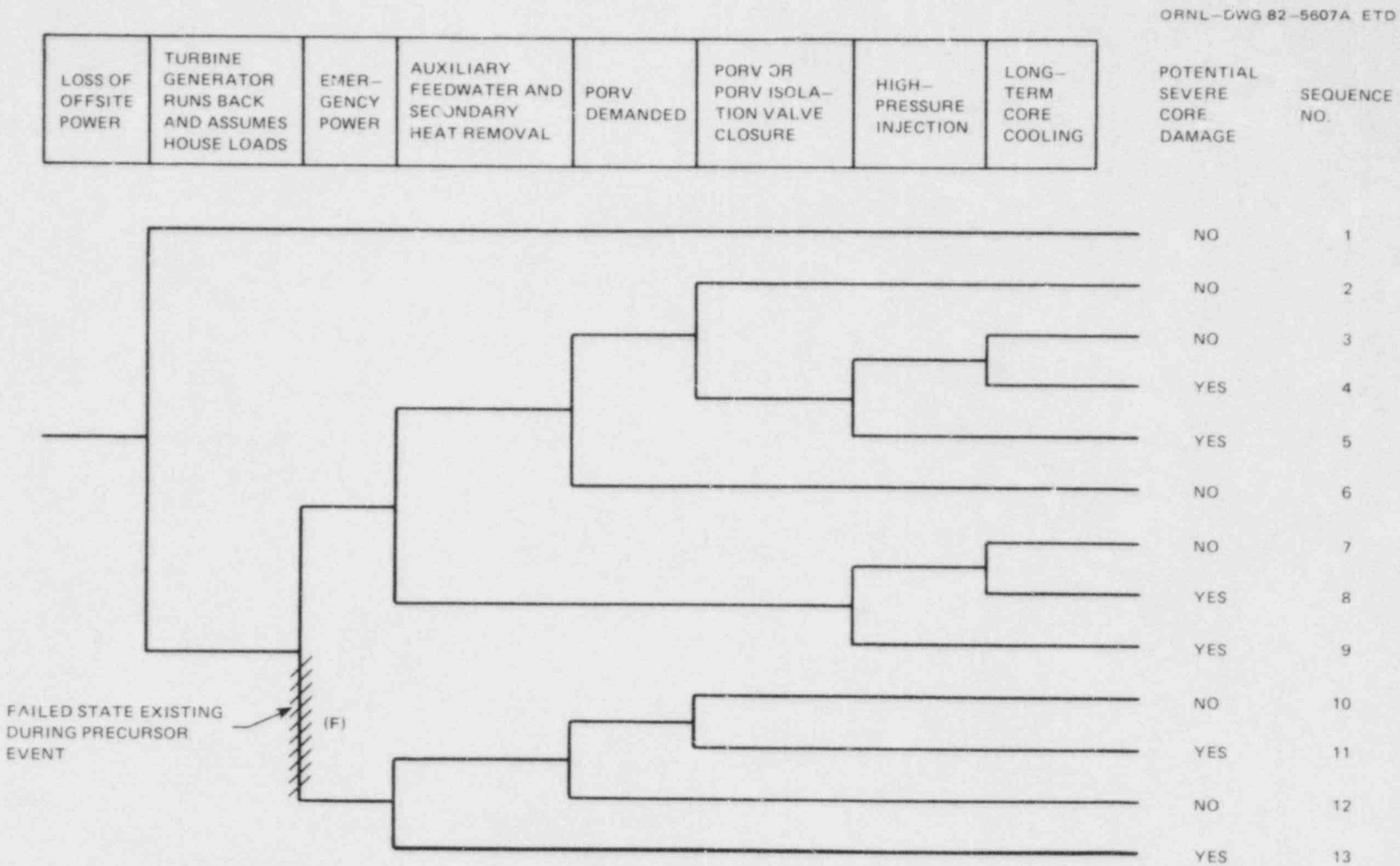


Fig. 3.1. Example sequence of interest event tree.

Because the frequencies and function demand failure probabilities used in these calculations are derived from data obtained across the entire LWR population and are applied to sequences that are generic in nature, the conditional probabilities determined for each precursor cannot 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, averaged over the plants, and are considered representative of potential severe core damage probabilities resulting from the occurrence of the selected events at plants representative of the general reactor population.

The sequence of interest tree probabilities were calculated in the following manner.

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 sequence of interest event tree associated with the initiator.

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 or expected initiating events. Only sequences associated with each potential initiator that were impacted by the precursor were included in the calculated probability.

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 times 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, th initiating event frequency estimates included the potential for recovery. Event durations (the period of time during which the failure existed) were based on information included in each 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. If a precursor occurred when the plant was not at power, then the probability of the event occurring while at power or shortly after shutdown (while decay heat was still significant) was multiplied by the initiating event probability determined in steps 1 or 2, in order to estimate

the probability of a nonrecoverable initiator while the plant was most vulnerable. The potential impact of the event at less vulnerable times was not considered in the analysis.

Sequence branch probability determination

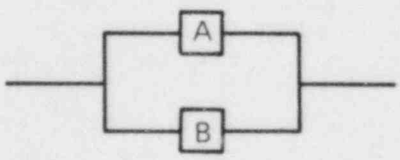
1. For event tree branches for which no failed or degraded condition existed, a probability equal to the function failure probability developed as in Sect. 3.2.1 was assigned.

2. For event tree branches associated with a failed function, a probability equal to the numeric value associated with the recovery class was assigned. This permitted consideration of potential recovery for observed failures, in the same manner as was done in Sect. 3.2.1.

3. For event tree branches that included a degraded function (a function with internal faults that still met minimum operability requirements but was reduced to no redundancy), the estimated failure probability was modified to reflect the loss of redundancy. To estimate a revised failure probability, the function was assumed to behave as a two-train system that could be modeled using the β-factor method.* This method was employed because it permitted estimates of the impact of degraded functions without identification of single failures, which were beyond the scope of this program. In the β-factor method, failures in redundant functions are separated into independent and dependent (coupled) contributors to total failure:

$$\lambda = \lambda_i + \lambda_c ,$$

where i is independent and c is coupled. Beta is defined as the fraction of the total failure rate attributable to dependent failures, so that $\lambda_c = \beta\lambda$ for $0 \leq \beta \leq 1$. Then, for a one-out-of-two function with identical components A and B,



*Development of the β-factor method is described in more detail in Sect. 3.7 of the *PRA Procedures Guide*.[2]

the failure on demand probability U can be written as

$$U = P(A) \times P(B/A)$$

$$= P(A)P(B/A)\Big|_{\substack{\text{independent}\\\text{failures}}} + P(A)P(B/A)\Big|_{\substack{\text{dependent}\\\text{failures}}}$$

$$= (1 - \beta)P(A)P(B) + \beta P(A) \times 1$$

$$= (1 - \beta)\lambda^2 + \beta\lambda \ ,$$

because $\lambda_A = \lambda_B$. Degraded functions were defined, for the purposes of this study, as those in which operability, but no additional redundancy, exists. (This permitted consideration of multiple-redundant functions as two-component functions required by the β-factor model.) Because the demand failure probability for each function has been previously determined, assumption of a representative β permits the calculation of the single-component (train) failure probability:

$$U = (1 - \beta)\lambda^2 + \beta\lambda \ ,$$

where

U = probability of overall system failure on demand,
β = fraction of failures on demand attributable to common cause,
λ = failure-on-demand probability for a single component (train).

Solving for λ,

$$\lambda = \frac{-\beta \pm \sqrt{\beta^2 + 4 \times (1 - \beta) \times U}}{2 \times (1 - \beta)} \ ;$$

for λ to be positive, the sign before the square root symbol must be positive. With λ estimated, the probability of the second component (train) failing, given that the first has been observed as failed, can be estimated based on the fact that the overall function failure probability U is related to the component failure probabilities by U = P(A) × P(B/A). The probability of the second component failing, given that the first has failed, is then P(B/A) = U/P(A) or function failure probability divided by λ. Degraded functions may in certain instances be returned to complete operability through some recovery action in the same manner as discussed in Sect. 3.2.1 for function failures. To account for this, degraded functions were also assigned to recovery classes. The probability of function failure for a degraded system can then be estimated as the probability of the second component (train) failing multiplied by the likelihood of the previously failed component not being recovered (the likelihood of the recovered train failing a second time has been assumed to be negligible):

$$P(\text{degraded function}) = R_{\substack{\text{degraded}\\ \text{function}}} U_{\substack{\text{degraded}\\ \text{function}}} / \lambda_{\substack{\text{degraded}\\ \text{function}}} ,$$

where R is the numeric value of the recovery class, U is the function failure probability, and λ is the train failure probability.

4. For event tree branches consequently degraded because of another degraded function (for example, if an auxiliary feedwater train was rendered unavailable during a loss of offsite power because of a diesel generator failure), the remaining train would be available with a failure probability equal to the train failure probability, λ, associated with the consequently degraded function, because the unavailable train had not itself failed. The approach to calculating the demand failure probability in this case is the same as with a degraded function, except for the substitution of λ for U/λ and the use of the recovery class associated with the function causing the consequential degrading (once the degraded function is recovered, the consequentially degraded function is restored to complete operability):

$$P(\text{consequently degraded}) = R_{\substack{\text{support}\\ \text{system}}} \lambda_{\substack{\text{consequently}\\ \text{degraded}\\ \text{function}}}$$

$$+ (1 - R_{\substack{\text{support}\\ \text{system}}}) U_{\substack{\text{consequently}\\ \text{degraded}\\ \text{function}}} ,$$

where R, λ, and U are as previously defined.

5. For event tree branches consequently failed as a result of another degraded support function and an internal failure (for example, a failed HPI system due to the unavailability of one diesel generator and the opposite train pump), the potential exists to recover either the internal failure or the degraded support function. However, to simplify modeling, the internal failure was assumed unrecoverable, and thus the consequently failed function was assumed failed for as long as the support function remained degraded. If the support function was recovered, then the previous consequently failed function was assumed to be degraded as described in item 4 above. This results in a failure probability estimate

for the consequently failed function of

$$P(\text{consequently failed}) = 1 \times R_{\substack{\text{support}\\\text{function}}}$$

$$+ (1 - R_{\substack{\text{support}\\\text{function}}})\, U_{\substack{\text{consequently}\\\text{failed}\\\text{function}}} / \lambda_{\substack{\text{consequently}\\\text{failed}\\\text{function}}} .$$

6. Functions that were consequently failed as a result of support system failures were modeled recognizing that, as long as the affected support system remained failed, all consequently failed functions were failed; but if the support system were recovered, all the affected functions were recovered.

7. Certain event tree branches were modified to reflect the design of systems at the plant at which an event occurred. Event tree branches impacted by this modification include:

— PWR turbine runback (which does not exist on many plants),
— PWR auxiliary feedwater given emergency power failure (which is very dependent on system design, particularly in the pre-1980 time period),
— PWR PORV demanded (certain plants do not use PORVs),
— PWR high-pressure injection given auxiliary feedwater failure (bleed and feed) (which is very dependent on system design and is assumed not to exist prior to the TMI-2 accident.)

To permit tailoring for these design differences, five plant-specific tailoring classes were defined. These are shown in Table 3.3. Numeric values associated with these classes were assigned to applicable event tree branches during individual event quantification.

For precursors occurring in plants that went critical before 1969, the design of mitigating systems was considered before failures were assessed using probability calculations. Function failure probabilities were revised when necessary to reflect unusual system designs.

Event calculation. Once the branch probabilities that reflect the conditions of the precursor event have been established, the sequences leading to potential severe core damage are calculated and summed to produce an estimate of the conditional probability of potential severe core damage for the precursor.

Sample calculations. Two events are used to illustrate this calculational process. The first event involved a loss of offsite power caused by a turbine-generator trip and an isolated startup transformer. The sequence of interest tree for this event is shown in Fig. 3.2. The probabilities included on each branch have been obtained as previously described. The precursor involved an initiating event that was assigned to recovery class R2 (the numeric value associated with this recovery class

Table 3.3. Description and quantification of plant-specific tailoring classes

Plant-specific tailoring class	Description	Numeric range	Average numeric value[a]
P1	Function does not exist at plant; event tree branch is faulted	1.0	1.0
P2	Function exists	0.1–0.8	0.34
P3	Function exists	0.03–0.3	0.12
P4	Function exists	0.01–0.1	0.04
P5	Function does not exist at plant; event tree branch not faulted	0.0	0.0

[a]See Chap. 6 for development of average values from numeric ranges. These average values are used for consistency of analysis.

is 0.34). Plant-specific tailoring values were used to describe probabilities for turbine-generator runback, auxiliary feedwater given emergency power failure, PORV challenge rate, and high-pressure injection following AFW failure. Probabilities for the remaining functions, which were assumed available based on the report of the event, were assigned industrywide average failure probabilities developed as described in Sect. 3.1. The estimated conditional probability of severe core damage, P_{SCD}, associated with the event is then

$$
\begin{aligned}
P_{SCD} = {} & P_{SCD}\ [\text{seq. } 4] + P_{SCD}\ [\text{seq. } 5] + P_{SCD}\ [\text{seq. } 8] \\
& + P_{SCD}\ [\text{seq. } 9] + P_{SCD}\ [\text{seq. } 11] + P_{SCD}\ [\text{seq. } 13] \\
= {} & [(2.5E\text{-}4)(\sim 1)(2.9E\text{-}3)(0.12)(\sim 1)(\sim 1)(1.0)(0.34)]\ [\text{seq. } 4] \\
& + [(5.9E\text{-}4)(2.9E\text{-}3)(0.12)(\sim 1)(\sim 1)(1.0)(0.34)]\ [\text{seq. } 5] \\
& + [(2.5E\text{-}4)(0)(2.7E\text{-}4)(\sim 1)(1.0)(0.34)]\ [\text{seq. } 8] \\
& + [(1.0)(2.7E\text{-}4)(\sim 1)(1.0)(0.34)\ [\text{seq. } 9] \\
& + [(2.9E\text{-}3)(0.12)(0.88)(3.6E\text{-}4)(1.0)(0.34)]\ [\text{seq. } 11] \\
& + [(0.12)(3.6E\text{-}4)(1.0)(0.34)]\ [\text{seq. } 13]
\end{aligned}
$$

$$P_{SCD} \cong 1.1E\text{-}4 \ .$$

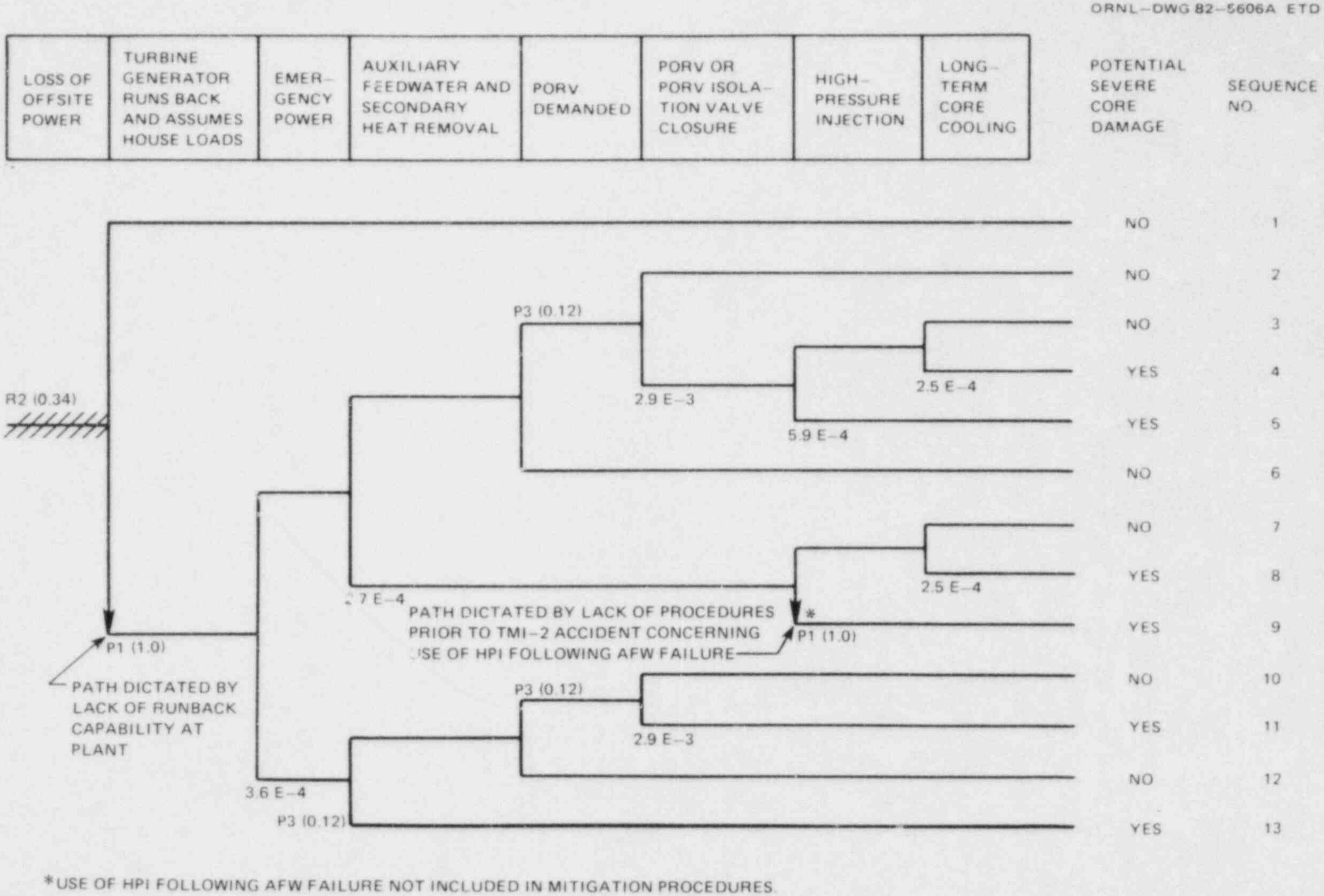


Fig. 3.2. Example calculation of initiating event conditional probability.

The second example event involved an unavailability of the undervoltage trip circuitry for the engineered safety features actuation system, which was discovered during troubleshooting. The sequence of interest event tree for this precursor is shown in Fig. 3.3. The failure probability associated with the precursor event (the emergency power failure) was assigned based on the recovery class associated with the event. No initiating event occurred with the precursor; however, a failure duration of 7.5 h was specified. The estimated nonrecoverable loss-of-offsite-power frequency, 2.75E-2/reactor-year, combined with this failure interval, results in an estimated initiating event probability of 2.4E-5. The combined branch probability for branches leading to potential severe core damage is 7.2E-7 (employing the same calculational method as above). So that event tree branches not involved with the precursor were eliminated and only the additional contribution associated with the precursor was calculated, the sequence of interest event tree was calculated a second time using the same initiating event probability but with all branches assigned demand 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. For this example, this second value (2.4E-9) had no effect on the probability calculated for the precursor.

3.2.3 Calculation of average potential severe core damage frequency

An estimate of the frequency of potential severe core damage over the observation period is calculated based on an approach discussed by Apostolakis and Mosleh.[3] In this approach, an estimate of the potential severe core damage frequency $\langle\lambda\rangle$ is given by

$$\langle\lambda\rangle = \frac{\sum_i P_i}{T} ,$$

where P_i is the conditional probability for event i and T is the observation period. (Reference 3 provides a development of the equation as well as some problems involved in the use of the Poisson model in this application.)

The precursors of interest in this estimate are those that included initiating events, because these precursors define the expected number of initiators and the response to initiators over the period of observation. The sum of the applicable conditional probabilities was divided by the observation period to obtain the frequency estimate for observed events.

Because certain initiating events are not reportable in the LER system, the frequency estimate was increased to account for these events. The primary contributors within this unreported set are losses of feedwater. (The frequency of loss of feedwater is over one per year at some plants.) The potential severe core damage frequency for losses of feedwater was used as an estimate of the frequency associated with nonreported, and thus, unobserved, events. A consistent approach for losses of feedwater would require their identification, along with any degraded and

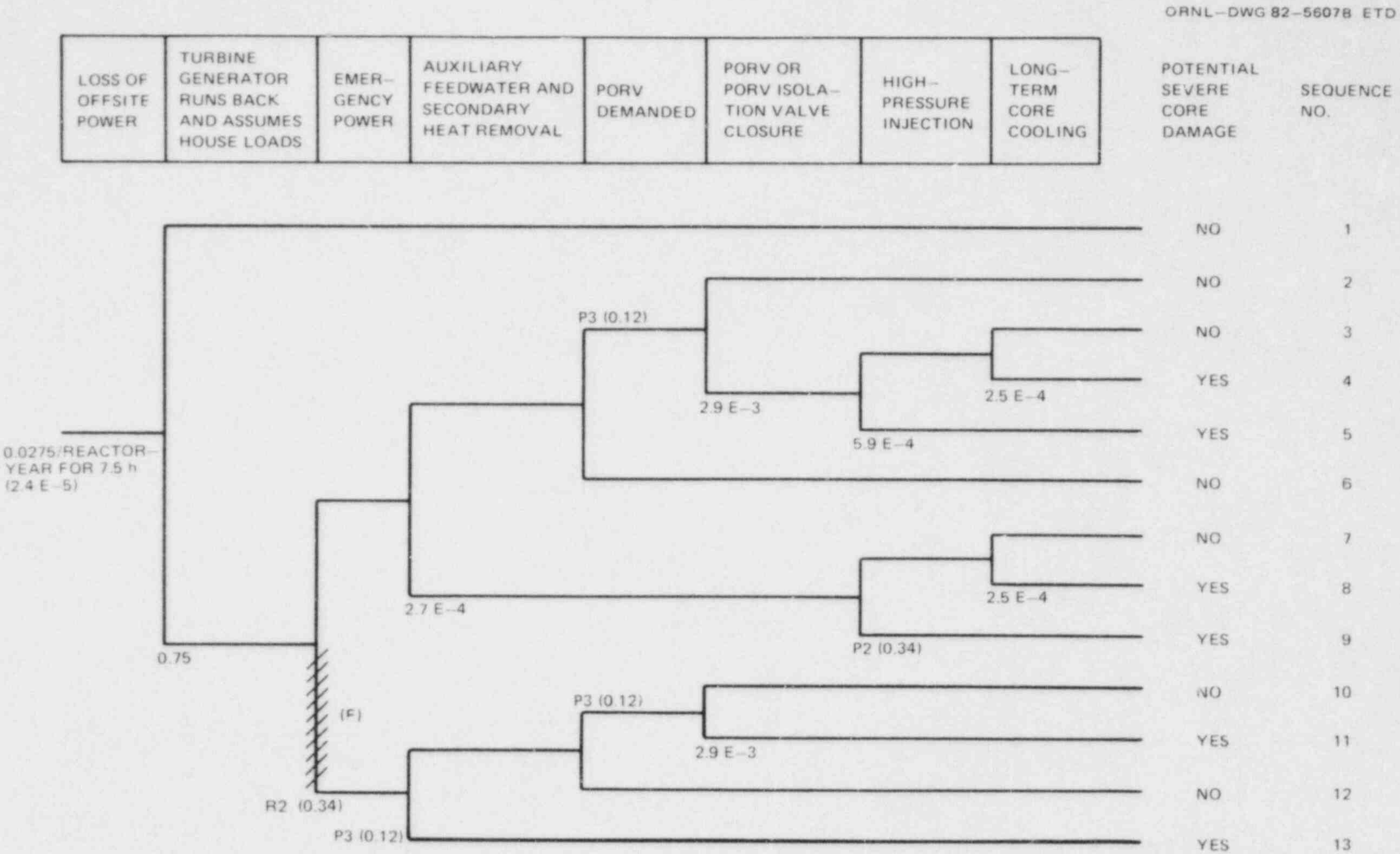


Fig. 3.3. Example calculation of unavailability conditional probability.

failed mitigating functions that occurred with them, in the same manner used with such events as a loss of offsite power. Because losses of feedwater are not usually reportable via the LER system, this could not be accomplished in this study. In lieu of this, a typical event tree approach was used to predict the frequency of severe core damage for these events.

This process can be represented in equation form as follows.

$$\begin{array}{c}\text{Frequency of}\\\text{severe core}\\\text{damage}\end{array} = \sum_{\substack{\text{precursors}\\\text{associated}\\\text{w/initiat-}\\\text{ing events}}} \begin{pmatrix}\text{frequency}\\\text{of precursor}\\\text{event}\end{pmatrix} \times \begin{pmatrix}\text{probability}\\\text{of subsequent}\\\text{severe core}\\\text{damage associ-}\\\text{ated w/event}\end{pmatrix}$$

$$+ \sum_{\substack{\text{expected initi-}\\\text{ating events}\\\text{not included in}\\\text{LER data base}}} \begin{pmatrix}\text{estimated}\\\text{frequency}\\\text{of event}\end{pmatrix} \times \begin{pmatrix}\text{probability}\\\text{of subsequent}\\\text{severe core}\\\text{damage associ-}\\\text{ated w/event}\end{pmatrix}$$

$$\begin{array}{c}\text{Frequency of}\\\text{severe core}\\\text{damage}\end{array} \cong \sum_{\substack{\text{precursors}\\\text{associated}\\\text{w/initiat-}\\\text{ing events}}} \begin{pmatrix}\text{1/observation}\\\text{period}\end{pmatrix} \times \begin{pmatrix}\text{probability of}\\\text{subsequent severe}\\\text{core damage asso-}\\\text{ciated w/precursor}\end{pmatrix}$$

$$+ \begin{array}{c}\text{estimated}\\\text{frequency}\\\text{of LOFW}\end{array} \times \begin{pmatrix}\text{probability of}\\\text{severe core damage}\\\text{given LOFW occurs}\end{pmatrix} .$$

The above approach includes considerable uncertainty and does not result in an unbiased estimate of the frequency of severe core damage.

Specific sources of underestimation and overestimation are discussed in the following sections.

3.2.4 Potential sources of error

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

In addition to the introduction of known nonconservative or conservative biases (which are discussed later in this section), certain program methods may introduce biases that are either conservative or nonconservative:

— The accuracy and completeness of the LERs in reflecting pertinent operational information is questionable in some cases. As discussed previously, LERs are required to be filed when plant Technical Specifications are violated or limiting conditions of operation are entered. These requirements, plus the approach to event reporting practiced at particular plants, result in wide variation in the extent of events reported and report details among plants. In addition, only details of the sequence (or partial sequence for failures discovered during testing) that actually occurred are usually provided; details concerning potential alternate sequences of interest to the study must be inferred.

— The event trees used for most precursors are functionally based and generic in nature. Although the trees were tailored to individual plant designs to the extent practical and unique trees were used when the standardized trees appeared inappropriate, the final event tree associated with the precursors may not adequately reflect differences between plants. This is particularly so in the cases where alternate systems (frequently not safety-related and therefore not described in FSARs) can be used for mitigation. (However, information concerning the use of alternate systems was solicited and reviewed during the event review process.)

— Average or generic data are combined with plant-specific operational occurrences in calculating the conditional probability of potential severe core damage. Because of this, modeled response for each event will tend toward an average response. If the systems at the plant at which the event occurred are better or worse than the average (this is difficult to ascertain without extensive operational experience or reliance on fault tree models), the actual conditional probability could be lower or higher than that calculated in the analysis.

— The recovery credit for system failure involves engineering judgment. Assignment of different recovery credit for an event can have a significant impact on the assessment of that event.* The impact of alternate numeric values for recovery classes is addressed in the sensitivity and uncertainty analyses described in Chap. 6.

— A test interval of once per month was assumed in the study for calculation of the probability of function failure on demand. If the test interval is longer than this, on the average, for a particular function and this is not compensated by overestimation in the number of nontest demands, then the calculated probability will be lower than that calculated using the actual test interval. Examples of longer test intervals would be situations in which (1) system valves are operated monthly but a system pump is only started quarterly or (2) valves are partially stroked monthly but fully operated only during refueling. Conversely, testing more frequently than assumed will result in a higher calculated failure probability than that calculated using the actual, shorter test interval. The impact of variation in test intervals is addressed in the uncertainty analysis described in Chap. 6.

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*There was considerable disagreement among NUREG/CR-2497 (Ref. 4) reviewers on the capability to recover certain operational failures.

— Events that occurred while the plant was shut down were accepted as precursors if it appeared that the events could have occurred while at power (i.e., if it appeared that they were not specifically shutdown related). The number of demands assumed in the analysis was based on an entire year of testing, whether the plant was at power or not. If the number of events occurring while the plant was at power is not consistent with the number of events occurring while the plant was shut down, then the probability estimates developed using at-power and shutdown data may not correctly estimate the probability of a particular failure while at power. Using only at-power data would decrease the number of failure events considered but would not necessarily decrease the probability estimate because of the lower number of demands while at power.

Potential sources of underestimation. A major source of underestimation in the program comes from the nonreporting or underreporting of operational events. Many events are known to have occurred that have not been reported through the LER system. Function failures and initiators included in such events are not accounted for in the ASP Program. This results in potential underestimation in the function failure probabilities and initiating event frequencies calculated in the program.

A second source of underestimation results from the lack of consideration in the study of certain events that nevertheless can be associated with potential severe core damage accident sequences. Programmatic constraints dictated that certain less important events, such as single failures found in redundant systems during testing, not be reviewed in detail. In addition, events known to be risk contributors, but which have not been operationally observed, such as large-break LOCAs, are not accounted for in the program.

Table 3.4 provides a list of potential risk contributors not selected in this study, along with reasons why they were not selected. This table is not intended to be all inclusive, but it is provided to clarify the reasons why certain types of events were not selected. As can be seen in Table 3.4, certain events were not selected because only the potential for failure existed, rather than an actual function failure. Others were not selected because they concerned failures associated with containment cooling or occurred prior to initial criticality. This is not to say that such events are not of concern. However, it is believed that those events identified in the ASP Study are the dominant historic operational events of importance from the standpoint of potential severe core damage.

A third potential source of underestimation comes from the manner in which the probability of failure on demand for event tree functions is determined in the study. These are determined by dividing an observed number of nonrecoverable failures by an estimated number of demands. Potential conditionality between functions is not considered in these calculations. (It is, of course, considered in the calculations of conditional probability of potential severe core damage.)

In an event sequence in which two functions must fail to achieve an undesired state,

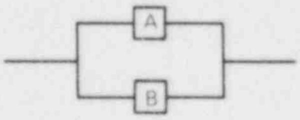


Table 3.4. Potential risk contributors not selected in the ASP Study

Item number	Plant, LER number, date	Event description	Reason for exclusion
1	Arkansas 1, 81-006, 4-8-81	Electrical short resulted in erroneous signal to ICS, resulting in an undercooling transient followed by a reactor trip and an overcooling transient	The significant portion of this event involved the subsequent overcooling transient. The impact of overcooling transients is being addressed as part of the pressurized thermal shock (PTS) program being performed separately at ORNL. The ASP Program has specifically not addressed the impact of potential PTS sequences pending the outcome of that work. This program has identified potential PTS-related operational events, which are documented in NUREG/CR-2789 (Ref. 5).
2	Rancho Seco, 81-037, 7-7-81	Air system degradation, sluggish valve behavior caused by contaminants in control air system	This event did not result in actual system failures, although the potential for such failure existed. Historically observed failures are used in the ASP Program to measure function failure likelihood.
3	Indian Point 2, 80-016, 10-17-80	Leakage of service water from containment fan cooling units (LER did not describe event adequately)	Although the potential for system failure existed, no actual failures occurred in systems associated with mitigation of potential severe core damage sequences.
4	Failed valves and operators in many different plants, multiple LERs	Survey of valve-operator-related events occurring during 1978—80 period revealed potential for common-mode failure of motor operator valves due to improper setting of torque switches, motor burnout, and changes in valve assembly characteristics	Although the potential for common-mode failure existed, no total function failures occurred. Total failures of functions are identified in the ASP Study as precursors and are used to develop estimates of the likelihood of failure of a particular function.
5	Westinghouse, B&W and CE PWRs, four LERs from different plants and numerous IE resident inspector reports	Steam generator overfill events	See response to item 1, above.

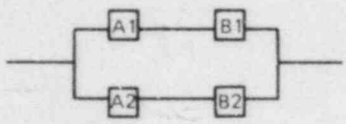
Table 3.4 (continued)

Item number	Plant, LER number, date	Event description	Reason for exclusion
6	Loose parts found at many plants, multiple LERs 1977—80	Internal appurtenances found to be loose in LWRs	If a loose part caused an initiating event or resulted in failure of a function or degradation of more than one function, it would be identified in this study. The potential for causing an initiating event or loss of function is typically not identified, because historic occurrences are considered better indicators of the likelihood of events of interest.
7	Turkey Point 4, 81-015, 11-28-81 and 11-29-81	RCS was overpressurized during startup following a refueling outage; overpressure mitigation system failed to operate	See response to item 1, above.
8	Surry 2, 81-033, 5-10-81 and 5-28-81	A routine sample of an underground fuel oil tank revealed water in the tank; this oil is fed to the emergency DGs	Because the diesels were still operable and only the potential for common-mode failure existed, the event was not selected as a precursor. Other actual multiple DG failures are used in the program to assess the likelihood of emergency power failing to respond when required.
9	Sequoyah 2, 81-094, 8-6-81	Inadvertent loss of reactor coolant; opening of a single valve in the RHR system allowed reactor coolant to leak into containment; plant was in shutdown cooling mode and RHR system was in operation	Sequoyah 2 was not critical at the time of this event. Only events occurring after criticality are considered in the program for two reasons: (1) a core was considered vulnerable to severe core damage only after initial criticality and (2) in the precritical period, distinguishing initial testing (system-checkout) failures from operational failures was sometimes difficult.
10	Arkansas 2, 80-072, 9-3-80	Blockage of coolant flow to safety-related systems and components	Based on the information available at NOAC, the containment building coolers were impacted by the blockages. Containment systems are not considered in this study.

Table 3.4 (continued)

Item number	Plant, LER number, date	Event description	Reason for exclusion
11	Pilgrim 1, 80-049, 8-28-81	Reactor building component cooling water — service water heat exchanger baffle plate damage caused by mussel fouling; potential for loss of both trains of service water resulting in the loss of cooling to redundant trains of safety equipment	Only the potential existed for loss of both trains of service water cooling. Events in which multiple trains of service water were unavailable have been selected in the study.
12	Beaver Valley 1, 12-25-81	Manual scram due to leaking diverter valve in the air dryer	No information was available in the LERs on file at NOAC concerning these events.
	Haddam Neck, 12-22-81 Source: prompt notification reports	Loss of control air caused a feedwater trip and lifting of PORV	
13	Calvert Cliffs 1 and 2, 81-079 and 81-047, 11-5-81	Unavailability of backflow protection of floor drain systems/ service water system	Although the potential for loss of multiple train existed, no such operational event actually occurred. Flooding events that impact multiple equipment are identified in the study.
14	Fort Calhoun, 80-010, 5-15-80	Stud failure in RCS pressure boundary (RCP), due to the corrosive action of primary system fluid on RCP carbon steel bolts	Although this event concerned the degradation of several RCP studs and the potential existed for sequential degradation to the point that a large-break LOCA might have occurred, an LOCA in actuality did not occur. In addition, it was not possible in this study to estimate the probability of a large-break LOCA occurring, given the observed degradation; thus, the event was not selected.

but neither are observed failed, the probability of the unacceptable state, $P(A)\ P(B/A)$, is estimated as $P(A)\ P(B)$. Any potential coupling between A and B is not considered. This coupling can result from improper maintenance actions; for example, an improperly trained maintenance team is made responsible for functions A and B. Train separation can also result in $P(A)\ P(B/A)$ being higher than $P(A)\ P(B)$ even if there is no common-mode coupling. This can be seen by considering a situation where either of two functions can fail, resulting in an undesired state. The two functions, A and B, each consist of two separated trains, 1 and 2, under the constraint that train 1 of function A cannot be associated with train 2 of function B and vice versa:



The probability of an unacceptable state in this case is, for small failure probabilities, approximately $P(B1)\ P(B2/B1) + P(A2)\ P(B1/A2) + P(A1)\ P(B2/A1) + P(A1)\ P(A2/A1)$. In this sum, $P(A1)\ P(A2/A1)$ and $P(B1)\ P(B2/B1)$ are the failure on demand probabilities as determined in the study. The other two terms are not accounted for.

Potential sources of overestimation. Besides the items listed at the beginning of this section, which may in certain circumstances contribute to overestimation, two items directly result in some degree of overestimation.

The first is the use of observed failures on demand (primarily from testing) for estimating failure probability. Such a method can result in overestimation by a factor of 2 for each function estimated. However, since unavailabilities at other times are not included in these estimates, the overestimation is not expected to be as serious as a factor of 2.* Numerous unavailabilities other than failures on demand were identified in this study.

A major source of overestimation results from the potential to overcount the number of failures actually observed. This is a result of the way in which failures are accounted for on each event tree. If a function fails following an initiating event, it is possible that the failure was unrelated to the initiator and just happened to have occurred at the same time. In this case, the probability of the function failing, given that the initiating event occurred, would be the probability of failure on demand associated with the function. In such a case, the observed failure is simply a manifestation of the existing function failure probability. It is also possible that the failure of a function following some initiating event is strongly coupled to that initiating event. This could occur if the effect was causal, and it could also occur due to coupled maintenance actions; for example, an improperly trained maintenance group could have been responsible both for the equipment related to the initiating event and for the function that failed. In such a case, the probability

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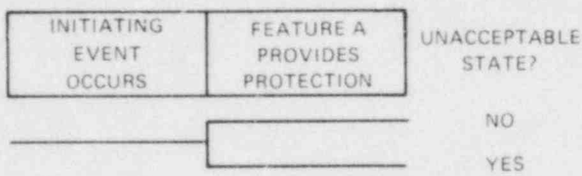
*A mean failure-to-recover probability of 0.58 was also used for recovery class R1 (the class associated with failures that did not appear recoverable), which also minimizes this overestimation.

of the function failing, given that the initiating event occurred, could be as high as 1.0, depending on the degree of coupling involved. In actuality, the degree of coupling is most likely somewhere between these two cases.

In the ASP Study, all failures observed at the same time are represented as given conditions on the event trees. It is possible to account for the potential impact of this representation. Because failures are included on applicable event trees and are also used to calculate a probability of function failure on demand (used on event trees when no failed or degraded function was observed), the number of expected function failures included in the event trees will be greater than the observed number of failures. This can be seen in the following example. Assume a single protective feature A provides protection for some initiating event. This can be described in the following event tree:



Consider that N such initiating events occur. During these events, i failures of A occur. Because A failed i times following the initiating event, i unacceptable states would be observed. This can be represented on N event trees as in Fig. 3.4.

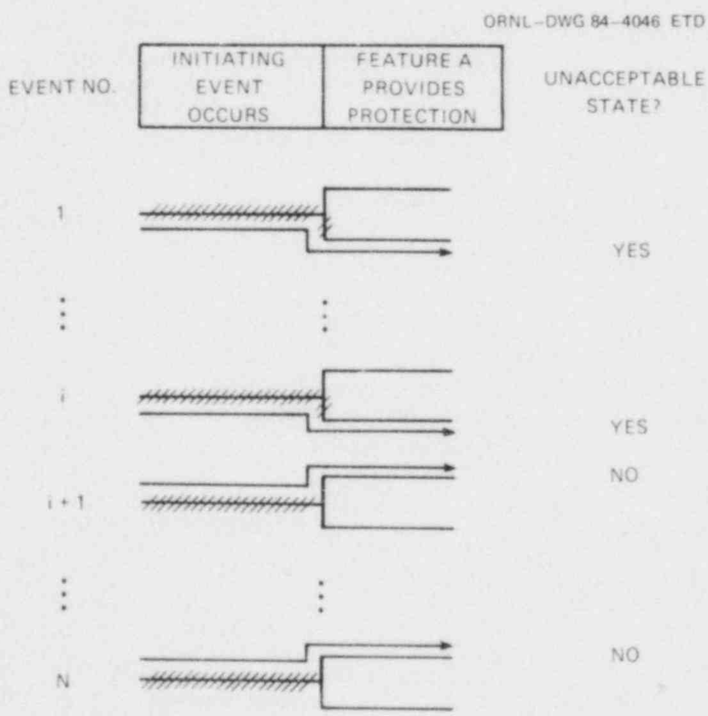


Fig. 3.4. Example treatment of N initiating events.

To recognize that an event tree function could have failed even though successful operation was observed or inferred, the ASP Program assigned a failure probability to functions that operated correctly. This was the average failure probability for the function developed by dividing the total number of observed failures by the total number of demands. Failed functions were assigned a failure probability representing the likelihood of failing to recover from the particular failure. In the example in Fig. 3.4, if N events were used as the basis for estimating the probability of A failing and the failure of A was considered nonrecoverable, the value (i/N) would have been assigned to each event tree where success was actually observed. A value of 1.0 would have been assigned where a failure was observed. This would result in an effective number of failures of protective feature A accounted for on the event trees equal to

[No. of trees with observed failures (i)] × [probability of failure (1)]

+ [No. of trees with observed successes (N − i)]

× [probability of failure assigned in this case (i/N)]

or

$$i + (N - i) \times i/N \approx 2i \quad \text{for } i/N \ll 1 .$$

In the ASP Study, the probability of failure on demand was actually determined by counting the effective number of actual demand failures and failures observed during testing (the majority of failures were observed during testing). In this situation, the impact on each event tree branch is dependent on the relationship of the test-related failures to the failures observed during actual demands and can be between one and two, depending on the degree of coupling.*

The impact of this on the overall frequency estimate is dependent on the structure of each tree and the degree of coupling that actually existed during each event. The range of impact can be estimated through the application of actual probabilities of function failure to the event trees utilized. This impact is addressed for the precursors selected in the 1980–81 period in Chap. 5.

It is recognized that the procedure used in the ASP Study results in a biased estimate of the industrywide potential severe core damage frequency. Effort is currently under way to develop a practical nonbiased estimation procedure.

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*Consider again a situation in which N initiating events occur and i protective function failures are observed. If j failures had been observed in T tests, the probability of failure on demand for the function would have been calculated as $(i + j)/(T + N)$ per demand. The expected number of failures in N demands is then $N \times (i + j)/(T + N)$. The effective

(Continued on p. 3-27)

number of failures accounted for on the N event trees is $i + (N - i) \times [(i + j)/(T + N)]$. If the number of failures during actual demands is consistent with the number of failures during testing (i.e., $i/N \approx j/T$), if the number of tests is much greater than the number of actual demands, and if the number of failures is small compared with the number of demands, then the expected number of failures in N demands is $N \times (i + i \times T/N)/(T + N) \approx (Ni/T) + (NiT/NT) \approx i$. The number of failures accounted for on ASP event trees is

$$i + (N - i)(i + i \times T/N)/(T + N) \approx i + \frac{(N - i)}{T + N} + \frac{(N - i)iT}{N(T + N)} \approx 2i ,$$

the same as before. However, if the number of failures during actual demands is greater than expected based on testing (i.e., if there is some dependence between the initiating event and the function failure) such that $i/N \gg j/T$, then the expected number of failures, obtained by combining test and demand data, is the same as before, but the number of failures accounted for on the ASP event trees is

$$i + (N - i) \times i/(T + N) + (N - i) \times j/(T + N)$$

$$\approx i \times \left(1 + \frac{N - i}{T + N}\right) + \frac{(N - i)j}{T + N} \approx i .$$

In this case, the number of events accounted for on the study event trees is more representative.[6]

References

1. Nuclear Regulatory Commission, Regulatory Guide 1.16, *Reporting of Operating Information, Appendix A: Technical Specifications*, Rev. 4, August 1975.

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3. G. Apostolakis and A. Mosleh, "Expert Opinion and Statistical Evidence: An Application to Reactor Core Damage Melt Frequency," *Nucl. Sci. Eng.* 70: 135-149 (1979).

4. 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), Oak Ridge National Laboratory, June 1982.

5. D. L. Phung, *Pressure Vessel Thermal Shock at U.S. Pressurized-Water Reactors: Events and Precursors, 1963—1981*, NUREG/CR-2789, (ORNL/NSIC-210), Oak Ridge National Laboratory, April 1983.

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4. SELECTION OF 1980—81 OPERATIONAL EVENTS AS ACCIDENT SEQUENCE PRECURSORS

The identification of precursors within the LER data base involved a two-step process. First, the abstract of each 1980—81 LER was 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. Over 8400 LER abstracts were examined, and approximately 390 LERs (4.6%) from the 1980—81 period were selected for detailed review, which considered (1) the immediate impact of each event and (2) the potential impact of equipment failures or operator errors on the readiness of systems in the plant for mitigation of off-normal or accident conditions. This review and selection process is described in Sect. 3.1. Fifty-eight 1980—81 operational events were selected as accident sequence precursors.

Following identification of the 1980—81 precursors, the LER data base was sampled to determine the completeness of the precursor selection process. The sample consisted of a random selection of 10% of the 1981 LERs. This sample was separately reviewed, using all information available in the LERs, for events that should have been selected as precursors but were missed. In this check, no additional precursors were identified. Based on this and assuming that failure to select precursors can be described using a Poisson process, it is estimated that at least 81% of the events for the years of interest were identified. No increases were made in any numerical value subsequently calculated in the study because of nonselected events.

4.1 Documentation of Selected Events

For each of the precursors, four items were prepared: (1) a precursor description and data sheet, (2) a categorization of accident sequence precursors, (3) an event tree describing the actual occurrence, and (4) an event tree for a postulated sequence of interest that incorporates the significant aspects of the actual occurrence. A description of these items follows.

1. Precursor description and data sheet (Fig. 4.1) briefly describes the LER event, identifies what corrective action was taken after the event, and describes the purpose of the failed system or component.

2. Categorization of accident sequence precursors (Fig. 4.2) includes selected characteristics of the event and of the plant where the event occurred. These categorizations were used in subsequent analyses.

3. Event tree describing the actual occurrence is based on reported event details. Failed equipment is identified by hatch marks, and the sequence of the event is indicated by arrows. These actual-occurrence event trees do not usually identify all reactor plant functions available to prevent severe core damage potentially stemming from the reported event,

ORNL-DWG 83-5934 ETD

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number:

Date:

Title:

The failure sequence was:

Corrective action:

Design purpose of failed system or component:

Fig. 4.1. Precursor description and data sheet.

ORNL-DWG 83-5933 ETD

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER:

LER NO:

DATE OF LER:

DATE OF EVENT:

SYSTEM INVOLVED:

COMPONENT INVOLVED:

CAUSE:

SEQUENCE OF INTEREST:

ACTUAL OCCURRENCE:

REACTOR NAME:

DOCKET NUMBER:

REACTOR TYPE:

DESIGN ELECTRICAL RATING: MWe

REACTOR AGE: years

VENDOR:

ARCHITECT-ENGINEERS:

OPERATORS:

LOCATION:

DURATION:

PLANT OPERATING CONDITION:

TYPE OF FAILURE: (a) Inadequate performance;
(b) failed to start;
(c) made inoperable;
(d) ____________________

DISCOVERY METHOD:

COMMENT:

Fig. 4.2. Form for categorization of accident sequence precursors.

but they do specify the major functions associated with the reported event which maintained the plant in . safe condition.

4. Event tree describing the sequence of interest considers the impact of the reported event on the mitigation sequence for the actual initiating event (if there was one) or for a postulated initiating event (if the reported event concerned equipment unavailability only). Failed and degraded functions related to the operational event are identified. These failed and degraded functions are based on the effect of the event on the safety-related systems of the plant at which the failure occurred.

The description and data sheet, categorization sheet, and two event trees developed for each precursor are included in Appendix B.

4.2 Tabulation of Selected Events

The 1980—81 events that were selected as precursors to potential severe core damage accidents are listed in Table 4.1. The precursor events have been arranged in numerical order by NSIC accession number, and the following information is included:

1. NSIC accession number associated with each precursor (ACCESS);
2. date of the event (E DATE);
3. postulated sequence of interest associated with the event (an initiating event or transient only occurred if I, item 10, was identified as Yes) (SEQ);
4. a brief description of the event (ACTUAL OCCURRENCE);
5. plant name and docket number where the event occurred (PLANT NAME, DOC);
6. abbreviations for the system and component involved in the event (SY, COMPXX);
7. plant operating status at the time of the event (O);
8. discovery method associated with the event (operational or testing) (D);
9. involvement of human error (E);
10. association of a transient or accident with the actual event (I);
11. age of the plant from criticality (in days) at the time of the event (AGEX);
12. conditional probability of potential severe core damage associated with the event (PROB) (defined in Chap. 5);
13. plant electrical rating, plant type, vendor, architect-engineer, and licensee (RATE, T, V, AE, OPR); and
14. plant criticality date (CRITXX).

Abbreviations used within the table are defined at the end of the table.

The information in Table 4.1 has been sorted in several ways to provide additional clarification. These sorts, included in Appendix C, are as follows:

Table	Sorted by
C.1	NSIC accession number
C.2	Event date
C.3	Plant name
C.4	System
C.5	Component
C.6	Plant operating status
C.7	Discovery method
C.8	Events involving human error
C.9	Events involving a transient or accident
C.10	Plant type and vendor
C.11	Architect-engineer
C.12	Operating utility

Abbreviations used in each table are defined in Table C.13 and at the end of Table 4.1.

Table 4.1. Precursors listed by NSIC accession number

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	C	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	C	UE	CPL	750320
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323

Table 4.1 (continued)

ACCESS: 6 DIGIT NSIC ACCESSION NUMBER
E DATE: EVENT DATE
SEQ: SEQUENCE OF INTEREST FOR THE EVENT

- 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
- RCPT - REACTOR COOLANT PUMP TRIP
- SGTR - STEAM GENERATOR TUBE RUPTURE
- UNIQ - A UNIQUE SEQUENCE

ACTUAL OCCURRENCE: DESCRIPTION OF EVENT
PLANT NAME: NAME OF PLANT AND UNIT NUMBER
DOC: PLANT DOCKET NUMBER
SY:SYSTEM ABBREVIATION:

STANDARD GENERIC CODE	SYSTEM 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	ENGINEERED SAFETY FEATURE INSTRUMENT SYSTEMS
IC	SYSTEMS REQUIRED FOR SAFE SHUTDOWN
ID	SAFETY RELATED DISPLAY INSTRUMENTATION
IE	OTHER INSTRUMENT SYSTEMS REQUIRED FOR SAFETY
IF	OTHER INSTRUMENT SYSTEMS NOT REQUIRED FOR SAFETY

Table 4.1 (continued)

	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 THEIR CONTROLS
	OTHER AUXILIARY SYSTEMS
AA	AIR CONDITIONING, HEATING, COOLING AND VENTILATION SYSTEMS AND CONTROLS
AB	FIRE PROTECTION SYSTEMS AND CONTROLS
AC	COMMUNICATION SYSTEMS
AD	OTHER AUXILIARY SYSTEMS AND THEIR CONTROLS
	STEAM AND POWER CONVERSION SYSTEMS
HA	TURBINE-GENERATORS AND CONTROLS
HB	MAIN STEAM SUPPLY 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)

Table 4.1 (continued)

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

COMPXX: SYSTEM COMPONENT CODE:

COMPONENT TYPE (COMPONENT CODE)	COMPONENT TYPE INCLUDES
ACCUMULATORS (ACCUMU)	SCRAM ACCUMULATORS SAFETY INJECTION TANKS SURGE TANKS HOLDUP/STORAGE TANKS
AIR DRYERS (AIRDRY)	
ANNUNCIATOR MODULES (ANNUNC)	ALARMS BELLS BUZZERS CLAXONS HORNS GONGS SIRENS
BATTERIES AND CHARGERS (BATTRY)	CHARGERS DRY CELLS WET CELLS STORAGE CELLS
BLOWERS (BLOWER)	COMPRESSORS GAS CIRCULATORS FANS VENTILATORS
CIRCUIT CLOSERS/INTERRUPTERS (CKTBRK)	CIRCUIT BREAKERS CONTACTORS CONTROLLERS STARTERS SWITCHES (OTHER THAN SENSORS) SWITCHGEAR
CONTROL RODS (CONROD)	POISON CURTAINS
CONTROL ROD DRIVE MECHANISMS (CRDRVE)	
DEMINERALIZERS (DEMINX)	ION EXCHANGERS
ELECTRICAL CONDUCTORS (ELECON)	BUS CABLE WIRE
ENGINES, INTERNAL COMBUSTION (ENGINE)	BUTANE ENGINES DIESEL ENGINES GASOLINE ENGINES NATURAL GAS ENGINES PROPANE ENGINES
FILTERS (FILTER)	STRAINERS SCREENS
FUEL ELEMENTS (FUELXX)	
GENERATORS (GENERA)	INVERTERS
HEATERS, ELECTRIC (HEATER)	HEAT TRACERS
HEAT EXCHANGERS (HTEXCH)	CONDENSERS COOLERS EVAPORATORS REGENERATIVE HEAT EXCHANGERS STEAM GENERATORS FAN COIL UNITS

Table 4.1 (continued)

INSTRUMENTATION AND CONTROLS (INSTRU)	CONTROLLERS SENSORS/DETECTORS/ELEMENTS INDICATORS DIFFERENTIALS INTEGRATORS (TOTALIZERS) POWER SUPPLIES RECORDERS SWITCHES TRANSMITTERS COMPUTATION MODULES
MECHANICAL FUNCTION UNITS (MECFUN)	MECHANICAL CONTROLLERS GOVERNORS GEAR BOXES VARIDRIVES COUPLINGS
MOTORS (MOTORX)	ELECTRIC MOTORS HYDRAULIC MOTORS PNEUMATIC (AIR) MOTORS SERVO MOTORS
PENETRATIONS, PRIMARY CONTAIN. (PENETR)	AIR LOCKS PERSONNEL ACCESS FUEL HANDLING EQUIPMENT ACCESS ELECTRICAL INSTRUMENT LINE PROCESS PIPING
PIPES, FITTINGS (PIPEXX)	
PUMPS (PUMPXX)	
RECOMBINERS (RECOMB)	
RELAYS (RELAYX)	SWITCHGEAR
SHOCK SUPPRESSORS AND SUPPORT (SUPORT)	HANGERS SUPPORTS SWAY BRACES/STABILIZERS SNUBBERS ANTI-VIBRATION DEVICES
TRANSFORMERS (TRANSF)	
TURBINES (TURBIN)	STEAM TURBINES GAS TURBINES HYDRO TURBINES
VALVES (VALVEX)	VALVES DAMPERS
VALVE OPERATORS (VALVOP)	EXPLOSIVE, SQUIB
VESSELS, PRESSURE (VESSEL)	CONTAINMENT VESSELS DRYWELLS PRESSURE SUPPRESSION PRESSURIZERS REACTOR VESSELS
OTHER COMPONENTS (XXXXXX)	
CODES NOT APPLICABLE (ZZZZZZ)	

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 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)
E: HUMAN ERROR INVOLVED (N-NO, Y-YES)
I: TRANSIENT/ACCIDENT INDUCED BY ACTUAL OCCURRENCE (N-NO, Y-YES)
AGEX: PLANT AGE AT THE TIME OF THE EVENT IN DAYS

Table 4.1 (cont...ued)

PROB: PROBABILITY
RATE: PLANT ELECTRICAL RATING IN MEGAWATTS ELECTRIC
T: PLANT TYPE (B=BWR, P=PWR)
V: PLANT NSSS VENDOR
- A-ALLIS CHALMERS
- B-BABCOCK AND WILCOX
- C-COMBUSTION ENGINEERING
- G-GENERAL ELECTRIC
- W-WESTINGHOUSE

AE: PLANT ARCHITECT ENGINEER

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

OPR: PLANT LICENSEE ABBREVIATION:

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

5. QUANTIFICATION OF PRECURSORS

The operational events selected as 1980–81 precursors were quantified to accomplish three things: (1) development of a set of industry-average initiating event frequencies and function failure probabilities, (2) calculation of a conditional severe core damage probability for each precursor for use in ranking of the precursors, and (3) estimation of an industrywide frequency of potential severe core damage based on the 1980–81 events. The detailed procedure followed in the quantification of precursors is given in Chap. 3.

For each precursor selected, the likelihood of recovery associated with the event failure(s) was described using one of four recovery classes and was quantified according to the mean numeric value associated with the assigned recovery class. These numeric values were determined by first defining a minimum and a maximum value for each of the four recovery classes and then assuming that between these extremes each could be described using a truncated log-uniform distribution (this distribution is flat on the log scale). The range of values assigned to each recovery class was sufficiently large to ensure reasonable overlap between ranges. The choice of this distribution is conservative in its uncertainty representation, in that it gives no preference to any set of values within its range on the log scale. As a contrast, the log-normal distribution, if it were used, would weight the values in the center of the region more than those in the tails. A description of the derivation of the mean value from the extremes using this distribution is provided in Chap. 6. The range of values assigned and the corresponding mean values are shown in Table 5.1.

5.1 Estimation of Initiating Event Frequencies and Function Failure Probabilities

A set of industry-average initiating event frequencies and function failure probabilities was developed to apply in the quantification of event trees associated with the precursors. The set includes initiating event frequencies and failure probabilities applicable to the branches of the standardized event trees described in Appendix A, which were used to classify and quantify the majority of possible precursors. Frequencies and failure probabilities for unique initiators and other plant functions were estimated, when required, based on plant design specifics, previous experience, and engineering judgment.

When precursor data were available for a function or initiating event, its probability or frequency was estimated by counting the effective number of nonrecoverable failures in a given observation 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.

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.

Table 5.1. Numeric values for recovery classes

Recovery class	Description	Range of numeric value assumed	Average value used in calculation[a]
R1	Failure did not appear to be recoverable in required period, either from control room or at failed equipment	0.3—1.0	0.58
R2	Failure appeared recoverable in required period at failed equipment, and equipment was accessible; recovery from control room did not appear possible	0.1—0.8	0.34
R3	Failure appeared recoverable in required period from control room, but recovery was not routine	0.03—0.3	0.12
R4	Failure appeared recoverable in required period from control room and was considered routine	0.01—0.1	0.04

[a]See Chap. 6 for development of average values from recovery value ranges. These average values are used for consistency of analysis. The actual likelihood of failing to recover from an event is difficult to assess and may vary substantially from the values listed above.

This estimate was then multiplied by the number of applicable reactor-years in the observation period to determine the total number of demands. [Failure and demand data were included only for plants that went critical after January 1, 1969 (see Sect. 3.2)].

Because of the small number of failures observed in 1980—81 for many of the reactor plant functions and initiators, it was felt that failure estimates based on a larger data base would provide better industrywide (homogenized) estimates if no clear chronological trend could be demonstrated for each function or initiator. Preliminary regression analyses indicated that the failure data were too sparse and scattered to demonstrate trends on an individual function basis.

In lieu of regression analysis, the effective number of observed failures or initiators in the 1980—81 period was compared with the number that would be expected for the 1980—81 period based on the number of events observed in 1969—79. The expected number was calculated by multiplying the number of events observed in 1969—79 by the ratio of the effective number of reactor-years in each period. These comparisons are shown in Figs. 5.1 and 5.2. In addition to the observed and expected number of

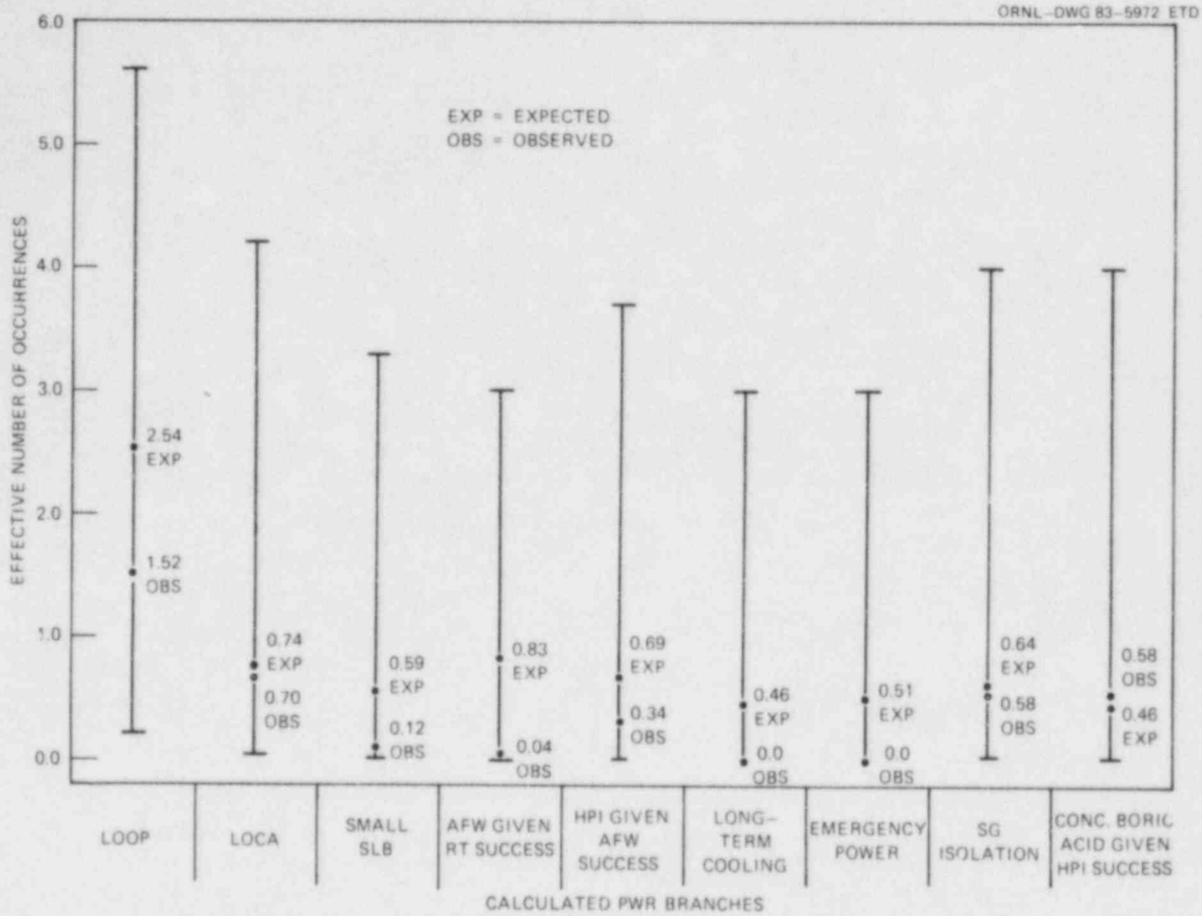

Fig. 5.1. Observed vs expected PWR effective number of occurrences with confidence bounds on the 1980—81 occurrences.

events, these figures include 90% confidence bounds* on the observed number of events.

As can be seen in Figs. 5.1 and 5.2, the observed number of nonrecoverable events in 1980—81 is consistent with the expected number based on the 1969—79 data, with two exceptions: BWR scram failure and BWR long-term core cooling, both high-reliability functions for which no failures were observed in 1969—79 but for which failures were observed in 1980—81. (It is interesting to note that Fig. 5.1 demonstrates a consistent decrease in the number of observed events in 1980—81 for PWRs compared with those in 1969—79. This effect is not observed for BWRs. Potential implications of this are discussed in Chap. 8.)

Because of the relative consistency of 1969—79 and 1980—81 data, the failures and initiating events in both periods were combined to estimate

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*Confidence bounds were estimated by assuming that the failures could be described using a Poisson process and by interpolating between Poisson 90% bounds integer values.

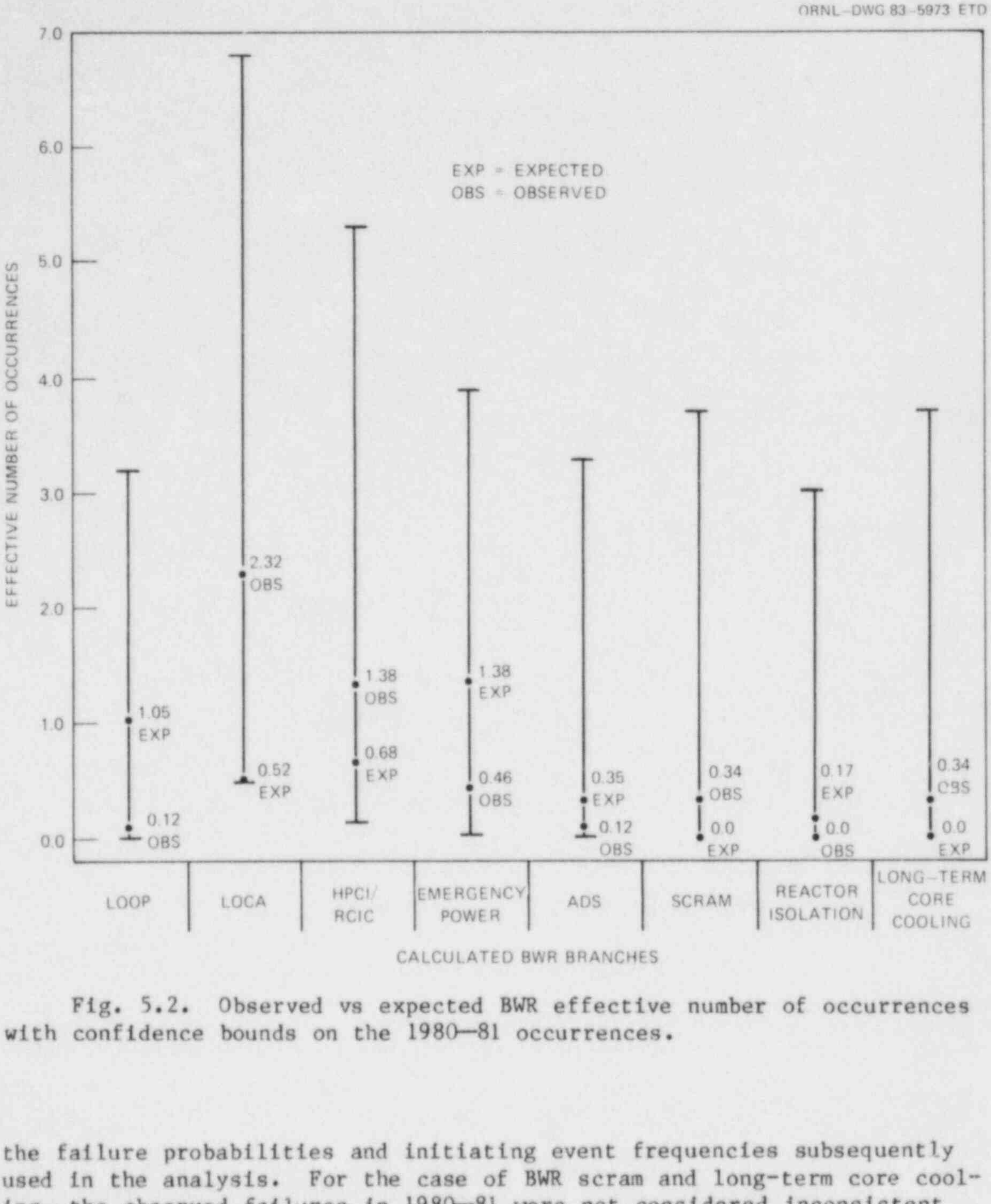

Fig. 5.2. Observed vs expected BWR effective number of occurrences with confidence bounds on the 1980–81 occurrences.

the failure probabilities and initiating event frequencies subsequently used in the analysis. For the case of BWR scram and long-term core cooling, the observed failures in 1980–81 were not considered inconsistent with the zero failures observed earlier, and data for these two functions were also combined. It must be noted, however, that if aging is involved in these failures, this approach may underestimate the expected number of failures in 1980–81.

If the observed number of failures in 1980–81 had been considered inconsistent with that expected based on 1969–79 information such that amalgamation of the bases could not be reconciled, the values would have been determined solely from 1980–81 information. In general, the estimated probabilities that were used in the subsequent quantification of the precursor event trees were based on the entire 1969–81 precursor data base and observation period.

Initiating event frequencies and function failure probabilities developed from the combined 1969–79 and 1980–81 periods are summarized in Tables 5.2 and 5.3. These tables also list the estimated range associated with each value. (These ranges are developed in Chap. 6.)

Table 5.4 details the individual 1980–81 contributing events, brief event descriptions, recovery classes assigned, effective number of nonrecoverable events from 1969–79, demand/test interval/reactor-year assumptions, and the subsequent initiating event frequencies and function failure probabilities calculated. Table 5.4 also includes the values and

Table 5.2. Initiating event frequencies and function failure probabilities determined from precursor data

	Point estimate	Estimated range[a]
BWR functions		
HPCI/RCIC failure	2.2E-3	1.0E-4 to 1.0E-2
Emergency power failure	2.2E-3	9.4E-5 to 1.1E-2
Automatic depressurization system failure	6.7E-3	1.9E-7 to 8.7E-2
Reactor scram failure	1.9E-4	4.5E-7 to 1.5E-3
Reactor isolation failure (large SLB)	2.3E-3	2.8E-7 to 2.7E-2
Long-term core cooling failure	1.0E-4	1.7E-6 to 6.0E-4
PWR functions		
AFW failure given reactor trip success	2.7E-4	5.8E-5 to 7.6E-4
HPI failure given AFW success	6.0E-4	2.0E-5 to 3.0E-3
Long-term core cooling failure	2.6E-4	1.2E-5 to 1.2E-3
Emergency power failure	3.7E-4	9.3E-6 to 2.0E-3
Steam generator isolation failure (large SLB)	6.4E-4	2.2E-5 to 3.2E-3
Concentrated boric acid addition failure given HPI success (large SLB)	8.3E-4	2.3E-5 to 4.4E-3
Initiators (values per reactor-year)		
BWR LOOP	1.9E-2	5.2E-3 to 4.7E-2
BWR LOCA	2.1E-2	7.5E-3 to 4.6E-2
PWR LOOP	2.8E-2	9.0E-3 to 6.2E-2
PWR LOCA	8.9E-3	3.9E-3 to 1.7E-2

[a]See Chap. 6 for development of these ranges. These estimates are based on amalgamated 1969–81 failure data and are average estimates across the reactor population. Plant-specific estimates can vary substantially from these values.

assumptions employed in the estimations that could not be determined from the precursor data.

5.2 Conditional Probabilities Associated with Each Precursor

As described in Sect. 3.2 and shown in Appendix B, the events associated with each precursor were analyzed and then mapped onto the applicable sequence of interest event tree for the precursor. If the precursor did not involve an initiator and was associated with an unavailability of a function, more than one event tree was typically applicable, and the duration of the event was used to estimate initiator probability.

Table 5.3. Initiating event frequencies and function failure probabilities determined from other than precursor data

	Point estimate	Estimated range[a]
BWR functions		
High-pressure cooling failure	1.7E-2	3.0E-3 to 5.0E-2
SBLC failure given scram failure	1.2E-1	3.0E-2 to 3.0E-1
LPCI/core spray failure	1.5E-4	6.8E-8 to 1.5E-3
PWR functions		
Reactor trip failure	3.6E-5	~0.0 to 1.5E-3
AFW failure given reactor trip failure	2.7E-3	1.2E-6 to 2.7E-2
PORV demanded	4.0E-2[b]	1.0E-2 to 1.0E-1[b]
PORV closure failure	2.9E-3	3.6E-4 to 1.0E-2
HPI failure given AFW failure	[c]	[c]
Turbine generator runback failure	[c]	[c]
AFW failure given emergency power failure	[c]	[c]
PORV opened due to HPI (large SLB)	8.0E-1[b]	6.0E-1 to 1.0[b]
PORV closure failure (large SLB)	6.0E-3	8.0E-4 to 2.0E-2
Initiators (values per reactor-year)		
LOFW (BWR and PWR)	3.0E-1	4.0E-2 to 1.0
Large SLB (BWR and PWR)	1.0E-3	4.5E-7 to 1.0E-2

[a]See Chap. 6 for development of these ranges, which are average estimates. Plant-specific estimates can vary substantially from these values.

[b]Lift rate.

[c]Plant dependent, see Table 5.4.

Table 5.4. Initiating event frequency and function failure probability estimates

Initiating event/ function under consideration	Event description: NSIC accession number	Event description: Event date	Event description: Plant name	Event description: Event description	Value or recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
				BWR initiating events				
Loss of feedwater	The frequency estimate in NUREG/CR-2497 was 0.56/reactor-year, based on 81 LOFW events reported in NUREG-0626 that occurred in a 3-year 66-reactor-year period in BWRs critical after 1968 and an estimated recovery probability of 0.5. Based on comments on NUREG/CR-2497 and revision of the R2 recovery class value from 0.5 to 0.34, a value of 0.30/reactor-year is the estimated LOFW frequency.							3.0×10^{-1}/ reactor-year
Loss of offsite power	166384	04-27-81	Monticello	Loss of entire 4.16-kV essential bus during refueling outage due to operator racking out breaker under load.	R3 (0.12)	3.64	191.79 post-1968 BWR reactor-years	1.9×10^{-2}/ reactor-year
	1969-79 events		Post-1968 BWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of loops occurring from 1969 to 1979 was 3.52.	3.52			
Small loss of coolant accident	160497	10-07-80	Pilgrim 1	At full power (96%), reactor vessel relief valve opened spuriously and could not be closed until the reactor depressurized to 20 psig.	R1 (0.58)	4.06	191.79 post-1968 BWR reactor-years	2.1×10^{-2}/ reactor-year
	160559	10-31-80	Pilgrim 1	At full power, reactor vessel relief valve opened spuriously and could not be closed until later in the event, when reactor pressure had reached 340 psig.	R1 (0.58)			
	160926	10-01-80	Pilgrim 1	A reactor vessel relief valve was opened to reduce reactor pressure but remained stuck open until the plant depressurized to 20 psig.	R1 (0.58)			
	174073	01-11-80	Duane Arnold	Stuck-open relief valve following manual lift.	R1 (0.58)			
	1969-79 events		Post-1968 BWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, the tota' effective number of small LOCAs occurring from 1969 to 1979 was 1.74.	1.74			
Large SLB	Value used was assumed to be an order of magnitude larger than the WASH-1400 value of 10^{-4}/reactor-year for a large LOCA. This value was used in lieu of the large LOCA value to reflect the greater lengths of steam-side piping and the fact that much of it is not reactor grade.							1.0×10^{-3}/ reactor-year

Table 5.4 (continued)

Initiating event/ function under consideration	NSIC accession number	Event date	Plant name	Event description	Value or recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
	Event description							
				BWR function failures				
Failure of HPCI/ RCIC	161906	11-16-80	Quad Cities 2	During performance testing, RCIC and then HPCI were declared inoperable.	R2 (0.34)	3.66	Twelve demands per reactor-year due to testing plus 0.30 LOFW demands per reactor-year x 132.73 years for post-1968 plants with HPCI/RCIC systems results in 1633 demands.	2.2×10^{-3}
	163478	06-26-80	Hatch 1	HPCI failed to inject following scram with RCIC inoperable as well.	R3 (0.12)			
	164995	06-28-81	Hatch 1	HPCI became inoperable during test while RCIC was removed from service.	R2 (0.34)			
	166082	04-10-81	Brunswick 2	HPCI and RCIC found inoperable during test.	R1 (0.58)			
	1969-79 events		Post-1968 BWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of HPCI/RCIC failures occurring from 1969 to 1979 was 2.28.	2.28			
Failure to provide high-pressure cooling	The value estimated here is based on the NUREG/CR-2497 value for HPCI alone, 5.7×10^{-2} times an estimated probability of 0.30 for failure to provide and control feedwater given the failure or unavailability of HPCI.							1.7×10^{-2}
Failure of emergency power	165438	04-05-81	Hatch 1	Emergency diesel circuit breakers failed to auto-close and energize emergency buses during a LOOP test.	R3 (0.12)	5.10	Twelve demands per reactor-year due to testing in 191.79 post-1968 BWR reactor-years results in 2301 demands. (Demands due to the small number of actual LOOPs are neglected here.)	2.2×10^{-3}
	171842	10-23-81	Dresden 3	Unit 2/3 and unit 3 diesel generators tripped during testing.	R2 (0.34)			
	1969-79 events		Post-1968 BWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of emergency power failures from 1969 to 1979 was 4.64.	4.64			
Failure of ADS	158231	04-25-80	Dresden 3	Several ADS valves failed to open on test during startup.	R3 (0.12)	1.28	One demand per reactor-year due to testing in 191.79 post-1968 BWR reactor-years results in 191.79 demands.	6.7×10^{-3}
	1969-79 events		Post-1968 BWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of ADS failure was 1.16.	1.16			

Table 5.4 (continued)

Initiating event/ function under consideration	Event description: NSIC accession number	Event description: Event date	Event description: Plant name	Event description: Event description	Value of recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
Failure of reactor scram	163405	06-28-80	Browns Ferry 3	Seventy-six control rods failed to insert initially on several attempts to manually scram.	R2 (0.34)	0.34	From NUREG/CR-2378, 180 auto and manual scrams were initiated and 87 normal shut-downs took place in 1980 for the 22 post-1968 BWRs. Assuming one-third of the normal shutdowns utilized scram, the total number of demands in 1980 was 29 + 180 = 209, or 209 ÷ 22 = 9.5 scrams per reactor year. For the 191.79 post-1968 BWR reactor-years, the total number of scram demands is 1822.	1.9×10^{-4}
Failure to initiate standby liquid control given scram failure	Demand of the SBLC system given a failure of the reactor to scram is a highly stressful situation, to which the operator must respond. The conditional probability assumed is 0.12.							1.2×10^{-1}
Failure of LPCI/ core spray	No simultaneous failures of LPCI and core spray were observed during the 191.79 post-1968 BWR reactor-years. Assuming a simultaneous failure on the next demand with a recovery value of 0.34 (the expected recovery value for systems dominated by pump failures) and assuming 12 test demands per year results in a failure probability estimate of $0.34/(191.79 \times 12) = 1.5 \times 10^{-4}$.							1.5×10^{-4}
Failure of reactor vessel isolation	1969-79 events		Post-1968 BWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, the total effective number of vessel isolation failures from 1969-79 was 0.58.	0.58	0.58	One demand per reactor-year for the full closure test plus one 0.30 LOFW event per reactor-year in 191.79 post-1968 BWR reactor-years results in 249 demands.	2.3×10^{-3}

Table 5.4 (continued)

Initiating event/ function under consideration	Event description				Value or recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
	NSIC accession number	Event date	Plant name	Event description				
Failure of long-term core cooling	166072	04-19-80	Brunswick 1	Loss of RHR cooling was experienced due to fouling in both RHR heat exchangers.	R2 (0.34)	0.34	The function is demanded on test, during LOFW, and on shutdowns of longer duration (>48 h assumed). NUREG/CR-2378 showed 110 such shutdowns for the 22 post-1968 BWRs, yielding 5 shutdowns per reactor-year. With 12 tests and 0.30 LOFWs assumed per reactor-year, 17.30 demands per reactor-year x 191.79 reactor-years results in 3318 demands.	1.0×10^{-4}
				<u>PWR initiating events</u>				
Loss of feedwater	The frequency estimate from the 1969-79 report was 0.56/reactor-year, based on a review of main feedwater events reported in NUREG-0611 (pp. II-15 through II-27), where 80 LOFWs occurred over a 3-year period (72 reactor-years) and 50% appeared rectifiable. Based on comments on the 1969-79 report and revision of recovery class values, a value of 0.30/reactor-year is the estimated LOFW frequency.							3.0×10^{-1}/ reactor-year
Loss of offsite power	158228	07-15-80	Prairie Island 2	LOOP due to severe electrical storm and subsequent No. 10 transformer lockout.	R2 (0.34)	7.92	287.63 post-1968 PWR reactor-years	2.8×10^{-2}/ reactor-year
	158232	06-03-80	Indian Point 2	LOOP due to lightning striking a transmission tower and tower lines falling and faulting feeder lines.	R3 (0.12)			
	158279	07-14-80	Arkansas 2	LOOP due to tornado activity and protective relaying. Offsite power available through manual connection from 161-kV transmission system.	R3 (0.12)			
	159134	06-24-80	Arkansas 1	LOOP due to ground fault that initiated a sequence of overload trips and voltage upsets.	R3 (0.12)			
	159136	06-24-80	Arkansas 2	LOOP due to ground fault that initiated a sequence of overload trips and voltage upsets.	R3 (0.12)			

Table 5.4 (continued)

Initiating event/ function under consideration	Event description: NSIC accession number	Event description: Event date	Event description: Plant name	Event description: Event description	Value or recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
Loss of offsite power (cont'd)	167117	06-19-81	Rancho Seco	Excessive electrical demand resulted in switchyard voltage drop below limit.	R3 (0.12)			
	167636	06-16-81	Crystal River 3	LOOP due to feeder line struck by lightning and lightning arrester system failure.	R2 (0.34)			
	168548	08-07-81	Rancho Seco	Excessive electrical demand resulted in switchyard voltage drop below limit.	R3 (0.12)			
	169042	08-13-81	Arkansas 1	LOOP due to tornado activity.	R3 (0.12)			
	1969-79 events		Post-1968 PWRs	LOOPs that occurred in the 1969-79 observation period, revised in evaluation based on comments on NUREG/CR-2497. Effective number of failures with revised recovery classes is 6.40.	6.40			
Small loss of coolant accident	163499	05-10-80	Arkansas 1	Reactor coolant pump seal failure	R1 (0.58)	2.56	287.5 post-1968 PWR reactor-years	8.9×10^{-3}/ reactor-year
	172198	12-19-81	St. Lucie 1	Safety valve fails to correctly reseat.	R3 (0.12)			
	1969-79 events		Post-1968 PWRs	LOCAs that occurred in the 1969-79 observation period, revised in evaluation based on NUREG/CR-2497 comments and revised recovery class values. Effective number of failures with revised recovery classes is 1.86.	1.86			
Large SLB	Value used for this function was assumed to be an order of magnitude larger than the WASH-1400 value of 10^{-4}/reactor-year. See BWR large SLB.							1.0×10^{-3}/ reactor year
				PWR function failures				
Reactor trip failure	The probability for failure to trip was assumed to be equal to the value calculated in WASH-1400 (p. II-97).							3.6×10^{-5}

Table 5.4 (continued)

Initiating event/ function under consideration	Event description				Value or recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
	NSIC accession number	Event date	Plant name	Event description				
AFW and secondary heat removal failure given reactor trip success	171733	12-11-81	Zion 2	Two AFW pumps fail to start on demand with the third pump unavailable.	R4 (0.04)	2.14	Twelve demands per reactor-year due to testing plus one per shutdown of <48 h plus two per shutdown of >48 h. Based on 1979 operational data from NUREG/CR-1496, an average 7.9 outages of <48 h and 3.7 outages of >48 h occurred per plant. This results in 27.3/reactor-year x 287.63 post-1968 PWR reactor-years, yielding 7852 demands.	2.7×10^{-4}
	1969-79 events		Post-1968 PWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of failures occurring from 1969 to 1979 was 2.10.	2.10			
AFW and secondary heat removal failure given failure to trip	Probability for failure of AFW and secondary heat removal given failure to trip is assumed to be ten times the failure probability calculated for AFW and secondary heat removal given trip success.							2.7×10^{-3}
PORV demanded	Probability that the PORV would be demanded following a reactor trip with successful AFW initiation was assumed to be 0.04. (This value is a lift frequency.)							4.0×10^{-2}
Failure of open PORV to close and failure of operator to detect failure and close isolation valve	Probability that the PORV would stick open once it was open was assumed equal to 0.01. Probability of the operator failing to isolate the open valve was based on 1969-79 precursors: seven events involving a failed-open PORV with two of these events involving a failure to isolate the valve. This results in a combined failure probability of 0.0029.							2.9×10^{-3}
Failure of HPI given AFW success	169587	10-21-81	Turkey Point 4	Safety injection path found obstructed during test.	R2 (0.34)	2.08	Twelve demands per reactor-year due to testing in 287.63 post-1968 reactor-years results in 3452 demands.	6.0×10^{-4}
	1969-79 events		Post-1968 PWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, the total effective number of failures occurring from 1969 to 1979 was 1.74.	1.74			

Table 5.4 (continued)

Initiating event/ function under consideration	Event description: NSIC accession number	Event date	Plant name	Event description	Value or recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
Failure of HPI given AFW failure	Probability value used for failure of HPI function to provide core cooling given failure of AFW and secondary heat removal was dependent on the type of HPI function provided at the plant. This was accomplished with the plant-specific tailoring factors, P1-P5. A value of 0.12 (P3) was assumed for plants with HPI pumps capable of providing flow at the relief valve set point, and a value of 0.34 (P2) was assumed for plants that required manual depressurization to achieve injection flow and as a general value for this function and condition in lieu of plant specifics.							See Appendix D for individual events
Failure of long-term core cooling	1969-79 events		Post-1968 PWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of failures occurring from 1969 to 1979 was 1.16.	1.16	1.16	Twelve demands per reactor-year due to testing plus one per shutdown >48 h (3.7 such outages per reactor-year, see AFW probability estimate). This results in 4516 demands in 287.63 post-1968 PWR reactor-years.	2.6×10^{-4}
Failure of turbine generator to run back and assume house loads	Probability value used for failure to run back was based on availability of the function at a particular plant. For plants without a runback feature, the value was assumed to be 1.0. For plants with a runback feature, the value was assumed to be 0.75.							See Appendix D for individual events
Failure of emergency power	1969-79 events		Post-1968 PWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, the total effective number of failures occurring from 1969 to 1979 was 1.28	1.28	1.28	Twelve demands per reactor-year due to testing plus 0.1 LOOP demand per reactor-year results in 3480 demands.	3.7×10^{-4}
Failure of AFW and secondary heat removal given failure of emergency power	Probability value assumed for this function was based on the design of the AFW system in the plants in which the precursor events occurred. For units with only electrically operated pumps or with requirements for service water for turbine-driven pump cooling, the failure probability was assumed to be 1.0 (P1).							See Appendix D for individual events

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Table 5.4 (continued)

Initiating event/ function under consideration	Event description				Value or recovery class	Total effective number of events	Observation period or demand assumptions	Frequency or probability estimate
	NSIC accession number	Event date	Plant name	Event description				
Failure of AFW and secondary heat removal (cont'd)	For units with self-cooled turbine-driven pumps but with electrically operated valves capable of manual operation, the failure probability was assumed to be 0.12 (P3). This value was based on sufficient time being available for manual initiation of AFW following LOFW or LOOP and operating procedures that are assumed to be sufficiently specific to allow for manual initiation within the available time period.							
	For units with self-cooled turbine-driven pumps and with valves without ac power (except vital power) dependency, the failure probability was assumed to be 0.04 (P4). This number was based on the fact that, with few exceptions, all AFW systems having turbine-driven pumps use only one such pump and that, based on a review of LERs related to pump failures, 0.04 appears to be a representative failure-on-demand number for turbine-driven pumps.							
Failure of steam generator isolation	156204	04-11-80	Trojan	Failure of three of four MSIVs to close on demand	R1 (0.58)	2.20	Twelve demands per reactor-year due to testing resulted in 3452 demands.	6.4×10^{-4}
	1969-79 events		Post-1968 PWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of failures occurring fro[illegible] 1969 to 1979 was 1.62.	1.62			
Failure of concentrated boric acid addition given HPI success	170098	11-06-81	Salem 1	[illegible] boron injection tank inlet valves failed to open on demand.	R1 (0.58)	1.75	Twelve demands per reactor-year for Westinghouse plants (176 post-1968 PWR reactor-years) resulted in 2112 demands.	8.3×10^{-4}
	1969-79 events		Post-1968 Westinghouse PWRs	With revisions based on NUREG/CR-2497 comments and revised recovery class numeric values, total effective number of failures occurring from 1969 to 1979 was 1.17.	1.17			
PORV opened due to continued HPI following a large SLB	Value of this probability was assumed to be 0.8.							8.0×10^{-1}
PORV or PORV isolation valve failure to close following continued HPI following a large SLB	Value for this probability was assumed to be equal to a combined failure probability for the PORV isolation valve ($\sim 10^{-3}$) and for an operator error in failing to close the valve ($\sim 5 \times 10^{-3}$). This results in a failure probability estimate of 6×10^{-3}.							6.0×10^{-3}

Appropriate failure probabilities were applied to each event tree branch to reflect plant conditions and the precursor failures. The sequences leading to potential severe core damage were then calculated and summed to produce an estimate of the conditional probability of potential severe core damage for the precursor. When none of the standardized sequence of interest trees (in Appendix A) were applicable to a precursor, a unique sequence of interest event tree was developed and applied. These unique event trees are included in Appendix B.

Determination of probability values used on each branch of the event trees depended on the plant conditions during the precursor event. Branches for observed or expected successful function states were quantified using the probability values determined in Sect. 5.1. Branch probability modification to reflect partially or totally faulted states is described in detail in Sect. 3.2 and included changes to account for (1) observed function failures, (2) consequent failures (function inoperability as a consequence of other plant failures), (3) degraded functions (in which redundancy was lost but the function still met minimum operability requirements), and (4) consequently degraded functions.

If the conditional state was determined to be consequently failed, degraded, or consequently degraded, the impact of potential common-mode coupling was taken into account by modeling the function as a two-train redundant system using the β-factor method as described in Sect. 3.2. In this assessment, a value of 0.12 was used for β. This value may be conservative or nonconservative, depending on which components were associated with the failure.* Sensitivity and uncertainty analyses described in Chap. 6 consider the impact of assuming alternate values for β.

In addition, certain event tree branches were tailored when necessary to reflect conditions that existed at the plant. This tailoring primarily involved PWR events in which (1) certain event tree branches were forced in a particular direction to reflect the nonexistence of a function (such as the lack of turbine runback at many plants) or (2) certain plant-specific probability estimates were used [e.g., for auxiliary feedwater given emergency power failure and bleed and feed (strongly design dependent)]. Five classes of plant-specific tailoring values were used. These are described in detail in Sect. 3.2.

Numeric values used in the conditional probability calculation associated with each precursor are identified on event trees included in Appendix D. Event durations (or duration estimates) are also included where applicable.

The conditional probability of potential severe core damage associated with each precursor is identified under the heading PROB in Table 5.5. (Comments provided in Chap. 4 pertaining to the headings and abbreviations associated with Table 4.1 are applicable to Table 5.5 as well.) As discussed in Chap. 3, the conditional probabilities determined for each precursor were based on industrywide data, not plant-specific data, and therefore should not be directly associated with the probability of potential severe core damage resulting from the actual precursor event

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*For example, NUREG/CR-2098 (Ref. 1) identifies a β-factor of 0.2 for motor-driven AFW pump failure to start, 0.03 for turbine-driven BWR HPCI/RCIC pump failure to start, and 0.3 for PWR HPI and CVCS pump (pooled) failure to start.

Table 5.5. Precursors for 1980—81 sorted by accession number

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	345	P	C	BX	BGE	761130
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	C	BX	IEL	740323

at the specific reactor plant at which it occurred. The conditional probabilities represent an average over the industry.

The distribution of events selected as precursors as a function of conditional probability is shown in Fig. 5.3. As can be seen, the number of events is maximum in the 10^{-4} to 10^{-5} probability range and decreases both above and below these probability values. This is to be expected and is a result of two factors: (1) the number of more serious events is known to be less than the number of less serious events and (2) the criteria used in the study emphasized the selection of more serious events. The latter resulted in selection of comparatively fewer less significant events.

5.3 Quantification of Industry-Average Potential Severe Core Damage Frequency

An estimate of the industry-average potential severe core damage frequency was developed based on the method described in Sect. 3.2. This method sums the conditional probabilities of subsequent potential severe core damage associated with initiating events and divides this sum by the number of reactor-years in the observation period (87.6 for PWRs and 48.0 for BWRs) to obtain a frequency estimate.

Of the 1980–81 precursors, 30 involved initiating events. In addition to these observed and analyzed events, some events not reportable via the LER system (and hence unanalyzed in this program) are known to have occurred. The primary contributors in this set of unobserved events are

ORNL-DWG 84-4171 ETD

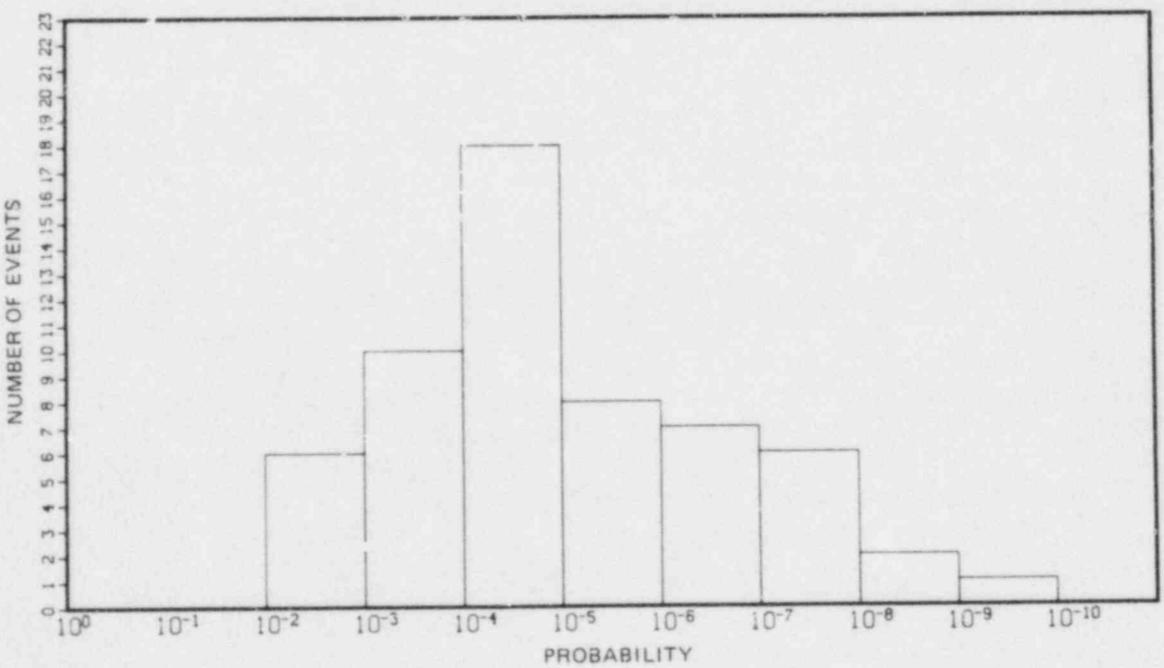

Fig. 5.3. Distribution of precursors as a function of conditional probability.

believed to be losses of feedwater, and a frequency of potential severe core damage associated with typical losses of feedwater was developed using the LOFW event trees included in Appendix A and the failure probability estimates developed in Sect. 5.1 to represent this set of events.

Table 5.6 lists the potential severe core damage frequency estimates for analyzed and unobserved PWR and BWR events in the 1980—81 time period, not corrected for potential overestimation. The caution included at the end of Sect. 5.2 must be reiterated here: these estimates are based on industrywide data homogenized over most of the plants, not plant-specific data; therefore, the estimates should not be directly associated with the frequency of potential severe core damage (unavailability of required core cooling) at any specific plant.

Table 5.6. Potential severe core damage frequency estimates by reactor type for 1980—81[a]

	Potential severe core damage frequency estimate (per reactor-year)		
	BWRs	PWRs	BWRs and PWRs
Frequency estimate based on analyzed precursors	1.7E-5	1.9E-4	1.3E-4
Frequency estimate for non-analyzed events (based on estimate for LOFW initiators)	4.2E-5	2.8E-5	3.3E-5
Combined estimate for analyzed and nonobserved events	5.9E-5	2.2E-4	1.6E-4

[a]Not corrected for overestimation and with 1969—81 amalgamated failure probabilities. See Table 8.2 for alternate estimates based solely on 1980—81 failure data.

The fact that the method employed in this program may overestimate the potential severe core damage frequency has been discussed in general terms in Sect. 3.2.4. Two methods have been used to approximate the degree of potential overestimation associated with this approach for 1980—81 events. In the first method, standardized event trees were used to calculate an alternate frequency estimate. The second method utilized revision of the conditional probabilities associated with failures observed in conjunction with initiating events, based on event details.

In the first method, an upper limit on the degree of overestimation was approximated by calculating the average potential severe core damage frequency using the initiating event frequencies and function failure probabilities developed in Sect. 5.1 and applying these to the standardized event trees described in Appendix A without using event tree models of individual precursors. Use of this method results in a frequency estimate of 4.3E-5, a factor of 3.7 less than the value estimated using the method employed in this study.

This calculation assumes that all observed events were independent, that the number of failures observed in conjunction with initiating events was consistent with the number observed in testing, and that unique events (those observed in the ASP Program that could not be modeled using standardized event trees) did not impact risk.

Precursors modeled using unique sequences contributed 6.3E-5/reactor-year to the overall estimate; precursors modeled using standardized event trees contributed 1.0E-4/reactor-year. Considering only precursors modeled using the standardized trees, the overestimation is a factor of 2.4. The extent of this overestimation would be decreased by a greater number of function failures following initiating events than those expected based on testing and any coupling between event tree functions.

The second method used to estimate the degree of overcounting assessed the likelihood that the multiple failures observed during initiating events were actually related, either because of incorrect maintenance actions, plant status during the event, or other causes. This estimate of the degree of overcounting is considered a reasonable estimate for 1980–81 precursors. Table 5.7 lists those 1980–81 precursors that involved initiating events and, in addition, other faulted functions. These failures have been classified, based on a review of each event, as being most likely unrelated, possibly related, or strongly related. Based on this assessment, the conditional probabilities for events not considered strongly related were reassessed.

For events considered most likely unrelated, subsequent failures were considered to be manifestations of the already determined industry-average failure probability, and the probabilities developed in Sect. 5.1 were used in the calculation. For events considered possibly related, the probability assigned to a faulted branch was estimated using a logarithmic average of the faulted and unfaulted probabilities. This permitted a moderate degree of coupling to be represented on the event tree.

The revised conditional probabilities for events with subsequent failures were then summed with those conditional probabilities associated with initiators that did not involve subsequent failures to correct the estimate of potential severe core damage frequency for possible overestimation. The revised frequency estimate is 1.1E-4/reactor-year, a factor of 1.5 lower than that determined previously.

Beyond the potential for overestimation discussed above, the industry-average frequency estimate is influenced by other uncertainties in both the approach and the numeric values used. Specific sources of underestimation and overestimation related to the approach are discussed in Sect. 3.2.4. Chapter 6 presents preliminary results of uncertainty and sensitivity analyses concerning factors and variables of interest.

5.4 Precursor Rankings

Two schemes have been used to rank the 1980–81 events by significance. The first ranking is simply by conditional probability of potential severe core damage. This ranking, which is related to the impact of

Table 5.7. Precursors for 1980–81 involving initiating events and additional faulted functions

NSIC accession number	Description of event	Estimated degree of coupling between failures	Conditional probability[a]	Revised probability[b]
158279	Loss of offsite power and cavitation of emergency feedwater pumps at Arkansas 2	Unrelated	6.0E-4	1.6E-5
160846	Loss of 24-V dc to nonnuclear instrumentation at Crystal River 3	Possibly related	5.0E-3	4.7E-4
163478	HPCI turbine isolated and loss of feedwater at Hatch 1	Possibly related	3.3E-4	2.3E-4
164453	Loss of offsite power and degraded load shed at San Onofre 1	Possibly related	6.1E-5	2.2E-5
164617	Loss of dc bus and diesel generator trip at Millstone 2	Possibly related	5.1E-3	5.1E-3
167611	Inadvertent spray initiation and draining of reactor coolant at Sequoyah 1	Possibly related	8.7E-4	8.7E-4
167624	Loss of offsite power and failure of one diesel to start at Crystal River 3	Unrelated	3.7E-4	2.1E-5
171667	Loss of vital bus at Davis-Besse 1	Possibly related	1.7E-3	1.4E-4

[a]As calculated in Sect. 5.2.

[b]Based on assessment of extent of coupling during event.

the observed event at some industry-average plant, is shown in Table 5.8, with events ranked in order from highest conditional probability to lowest. (Comments provided in Chap. 4 pertaining to the headings and abbreviations associated with Table 4.1 are applicable to Tables 5.8 and 5.9 as well.)

The ranking in Table 5.8 includes all precursors but tends to mask the implications of certain events — particularly those that involved significant function failures at times when the function was not required. An alternate ranking method was used to identify these events.

In the alternate ranking method, the entire 1969–81 data base was used to provide comprehensive failure data. The 1980–81 events were ranked on the basis of their impact on the potential severe core damage frequency estimates when each event was individually excluded. Because initiating events and function failures on demand are used to estimate initiating event frequencies and demand failure probabilities, which are subsequently used in assessing individual event conditional probabilities, events involving such failures can impact the conditional probabilities associated with many events.

Table 5.8. Precursors for 1980–81 sorted by conditional probability

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	32.	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	VYA	670724
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
158232	800603	LOOP	LIGNTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	FE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
161649	901016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307

Table 5.9. Precursors for 1980—81 sorted by impact on industry-average potential severe core damage frequency

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB†	RATE	T	V	AE	OPR	CRITXX
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	1.6E-5	1065	B	G	UX	TVA	760808
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	1.5E-5	821	B	G	UE	CPL	761008
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	9.0E-6	870	P	C	BX	NNE	751017
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	8.9E-6	825	P	B	GX	FPC	770114
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	3.0E-6	906	P	B	BX	TEC	770812
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	2.4E-6	906	P	B	BX	TEC	770812
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	2.0E-6	802	P	C	EX	FPL	760422
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	1.5E-6	1148	P	W	UX	TVA	800705
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.2E-6	794	B	G	SL	CWE	710131
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	1.1E-6	912	P	C	BX	APL	781205
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	8.3E-7	850	P	B	BX	APL	740806
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	6.5E-7	777	B	G	SS	GPC	740912
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	6.3E-7	825	P	B	GX	FPC	770114
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	4.7E-7	693	P	W	BX	FPL	730611
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	4.2E-7	1040	P	W	SL	CWE	731224
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	3.9E-7	821	B	G	UE	CPL	750320
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	2.4E-7	789	B	G	SL	CWE	720426
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	2.4E-7	777	B	G	SS	GPC	740912
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	2.3E-7	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	2.3E-7	655	B	G	BX	BEC	720616
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	2.3E-7	655	B	G	BX	BEC	720616
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	1.3E-7	802	P	C	EX	FPL	760422
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	1.1E-7	436	P	W	BX	SCE	670614
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	8.1E-8	530	P	W	PX	NSP	741217
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	6.6E-8	794	B	G	SL	CWE	710131
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	5.9E-8	580	P	W	SW	CYA	670724
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	5.9E-8	580	P	W	SW	CYA	670724
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	3.2E-8	545	B	G	BX	NSP	701210
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	2.8E-8	912	P	C	BX	APL	781205
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	2.7E-8	436	P	W	BX	SCE	670614
165438	810405	LOOP	DC CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.3E-8	777	B	G	SS	GPC	740912
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	2.3E-8	538	B	G	BX	IEL	740323
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	2.2E-8	873	P	W	UE	CEC	730522
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.8E-8	50	B	A	SL	DLP	670711
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	1.5E-8	700	P	W	EX	CPL	700920
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	1.2E-8	918	P	B	BX	SMU	740916
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	9.0E-9	850	P	B	BX	APL	740806
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	9.0E-9	918	P	B	BX	SMU	740916
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	9.0E-9	850	P	B	BX	APL	740806
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	7.2E-9	907	P	W	SW	VEP	800612
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	*	845	P	C	BX	BGE	761130
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	*	1130	P	W	BX	PGC	751215
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	*	794	B	G	SL	CWE	710131
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	*	845	P	C	BX	BGE	741007
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	*	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	*	906	P	B	BX	TEC	770812
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	*	655	B	G	BX	BEC	720616
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	*	1090	P	W	UX	PEG	761211
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	*	1148	P	W	UX	TVA	800705
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	*	912	P	C	BX	APL	781205
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	*	805	P	C	BX	CPC	710524
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	*	852	P	W	SW	DLC	760510
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DC UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	*	805	P	C	BX	CPC	710524
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	*	436	P	W	BX	SCE	670614
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	*	1090	P	W	UX	PEG	761211
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	*	535	P	W	FP	WPS	740307
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	*	693	P	W	BX	FPL	721020
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	*	436	P	W	BX	SCE	670614

†Industry-average frequency differential.

*Event cannot be ranked by this method.

The 1980–81 precursors, ranked in order of significance by this method from highest to lowest, are shown in Table 5.9. The numeric value associated with each event is the potential severe core damage frequency differential when the particular event is removed from the data base. Because the potential severe core damage frequency estimate developed in this program is based on conditional probabilities associated with initiating events, precursors that only involve unavailabilities (no initiating events or failures on demand) cannot be ranked by this method.

5.5 Dominant Sequences and Function Importance

The dominant sequences among those leading to potential severe core damage in the 1980–81 precursors were identified. Sequences to potential severe core damage are described in detail in Appendix A for LOFW, LOOP, small-break LOCA, and SLB initiating events. Sequences applicable to unique events are shown on the event trees developed for those events in Appendix B.

In the development of the relative contributions of the sequences, the contribution of each sequence to the overall potential severe core damage frequency estimate was compared with the total estimate. Dominant sequences and their percentage contributions are shown in Table 5.10. It must be noted that a large percentage of the contribution due to loss of feedwater sequences is a result of unobserved events (the details of which were not available). The relative contribution of the sequences identified in Table 5.10 are compared in Chap. 8 with contributions estimated in other studies.

The importance of various functions identified on the standardized event trees to prevent potential severe core damage was investigated using methods described by Vesely et al. Risk importance measures are defined to evaluate a function's importance in further reducing the risk and its importance in maintaining the present risk level. One defined importance measure, called the risk reduction worth of a function, is useful for prioritizing feature improvements that can most reduce the risk. The other defined importance, called the risk achievement worth of a function, is useful for prioritizing features that maintain the existing risk level and are most important in reliability assurance and maintenance activities.

The risk achievement worth is defined as the increase in risk if the function was assumed not to exist (or was assumed to be failed). If R_i^+ equals the increased risk of potential severe core damage without function i and R_0 equals the present risk level, then, on an interval scale, the risk achievement worth A_i is defined as:

$$A_i = R_i^+ - R_0 .$$

Risk reduction worth is similarly defined to be the decrease in risk if the function were assumed to be optimized or were assumed to be made perfectly reliable. If R_i^- equals the decreased risk level with the function optimized (or assumed to be perfectly reliable) then, on an interval

Table 5.10. Dominant potential severe core damage accident sequences observed in 1980–81 precursors

Sequence description	Contribution (%)
PWR sequences	
Small-break LOCA with subsequent recirculation failure	27
Failure of dc bus at Millstone 2 (NSIC 164617) with postulated nonrestoration of the dc bus and AFW failure[a]	25
LOFW with subsequent AFW and feed and bleed failure (8.7% from reported events; 12.6% from nonreported events)	21
Vital bus failure at Davis-Besse 1 (NSIC 158860), which resulted in loss of decay heat removal with postulated nonrestoration of either DH loop and failure of other means of DH removal[a]	7
Loss of RCP cooling at St. Lucie 1 (NSIC 158233) with postulated termination of natural circulation due to top-head bubble growth and failure of bleed and feed[a]	6
LOOP with emergency power system success, AFW and bleed and feed failure	4
Small LOCA due to opened containment spray valve at Sequoyah 1 (NSIC 167611) with failure to close open spray valve and inadequate makeup to the RCS[a]	3
Small-break LOCA with subsequent HPI failure	3
LOOP with subsequent emergency power and AFW failure	2
Remaining sequences (25 sequences)	2
BWR sequences	
LOFW with subsequent long-term core cooling failure (1.0% from reported events; 52.0% from nonreported events)	53
LOFW with subsequent HPCI/RCIC and ADS failure (9.7% from reported events; 7.7% from nonreported events)	17
LOFW with subsequent failure of scram and SBLC system (0.3% from reported events; 11.5% from nonreported events)	12
Small-break LOCA with subsequent failure of high-pressure cooling and ADS	7
Small-break LOCA with subsequent long-term core cooling failure	6
Small-break LOCA with subsequent failure of scram and SBLC system	1
Remaining sequences (29 sequences)	4

[a]These events were modeled using unique event trees.

scale, the risk reduction worth D_i is defined as:

$$D_i = R_0 - R_i^- .$$

Risk achievement and reduction worths were calculated for each function represented on the event trees described in Appendix A. These values for functions ranked by risk achievement are shown in Table 5.11 and those for risk reduction worth are shown in Table 5.12.

Table 5.11. Functions ranked by risk achievement worth

Function	Risk achievement worth	Risk reduction worth
PWRs		
AFW given reactor trip success	1.1E-1	5.6E-5
HPI given AFW success	1.0E-2	5.8E-6
Long-term core cooling	1.0E-2	5.9E-5
Reactor trip	4.4E-3	7.8E-8
Emergency power	1.5E-3	4.9E-6
AFW given emergency power failure	4.5E-4	6.1E-5
HPI given AFW failure	2.4E-4	6.8E-5
PORV closure	1.2E-5	3.5E-8
AFW given reactor trip failure	1.1E-5	2.9E-8
Turbine generator runback	3.3E-6	4.3E-6
PORV demanded	8.4E-7	3.5E-8
PORV closure given SLB	2.7E-7	1.6E-9
PORV opened due to HPI (large SLB)	4.1E-10	1.6E-9
SG isolation (large SLB)	*a*	*a*
Concentrated boric acid addition given HPI success (large SLB)	*a*	*a*
BWRs		
Long-term core cooling	3.5E-1	3.5E-5
Scram only	3.7E-2	6.9E-6
Automatic depressurization	2.1E-3	1.4E-5
HPCI/RCIC	2.1E-3	1.0E-5
LPCI/core spray	2.1E-3	3.2E-7
High-pressure cooling provided following LOCA	2.4E-4	4.2E-6
SBLC given scram failure	5.8E-5	6.5E-6
Emergency power	9.8E-6	4.9E-8
Reactor isolation (large SLB)	*a*	*a*

[a]Because no initiating events requiring operation of these functions have occurred, the importance values associated with these functions could not be determined by the methods used in this study.

Table 5.12. Functions ranked by risk reduction worth

Function	Risk achievement worth	Risk reduction worth
PWRs		
HPI given AFW failure	2.4E-4	6.8E-5
AFW given emergency power failure	4.5E-4	6.1E-5
Long-term core cooling	1.0E-2	5.9E-5
AFW given reactor trip success	1.1E-1	5.6E-5
HPI given AFW success	1.0E-2	5.8E-6
Emergency power	1.5E-3	4.9E-6
Turbine generator runback	3.3E-6	4.3E-6
Reactor trip	4.3E-3	7.8E-8
PORV demanded	8.4E-7	3.5E-8
PORV closure given SLB	1.2E-5	3.5E-8
AFW given reactor trip failure	1.1E-5	2.9E-8
PORV opened due to HPI (large SLB)	4.1E-10	1.6E-9
PORV closure given SLB	2.7E-7	1.6E-9
SG isolation (large SLB)	*a*	*a*
Concentrated boric acid addition given HPI success (large SLB)	*a*	*a*
BWRs		
Long-term core cooling	3.5E-1	3.5E-5
Automatic depressurization	2.1E-3	1.4E-5
HPCI/RCIC	2.1E-3	1.0E-5
Scram only	3.7E-2	6.9E-6
SBLC given scram failure	5.8E-5	6.5E-6
High-pressure cooling provided following LOCA	2.4E-4	4.2E-6
LPCI/core spray	2.1E-3	3.2E-7
Emergency power	9.8E-6	4.9E-8
Reactor isolation (large SLB)	*a*	*a*

[a]Because no initiating events requiring operation of these functions have occurred, the importance values associated with these functions could not be determined by the methods used in this study.

References

1. C. L. Atwood, *Common-Cause Fault Rates for Pumps, Estimates Based on Licensee Event Reports at U.S. Commercial Nuclear Power Plants, Jan. 1, 1972—Sept. 30, 1980*, NUREG/CR-2098 (EGG-EA-5289), EG&G Idaho, Inc., February 1983.

2. W. E. Vesely, T. C. Davis, R. S. Denning, and N. Saltos, *Measures of Risk Importance and Their Applications*, NUREG/CR-3385 (BMI-2103), Battelle-Columbus Laboratories, July 1983.

6. SENSITIVITY AND UNCERTAINTY ANALYSES

The initiating event frequencies, function failure probabilities, conditional core damage probabilities, and event rankings developed in the previous sections were based in part on observed events, recovery values assigned by engineering judgment to failures, and test frequency assumptions. In actuality, factors used in the analyses are not fixed and could vary over a significant range. For example, for some failure assigned to recovery class R3 (see Table 3.1), the likelihood of failing to recover could be 0.25 or 0.07 instead of the 0.12 value used in the analysis. Even the number of events of a particular type observed is not absolute; the data are a sample of actual reactor population operation.

In an attempt to bound this variability, sensitivity and uncertainty analyses were performed (1) to determine the impact of different recovery class numeric values, initiating event frequencies, and function failure probabilities on event ranking and the distribution of event probabilities and (2) to determine uncertainty bounds on initiating event frequencies, function failure probabilities, and the estimated severe core damage frequency. In these calculations, the following distribution assumptions were made:

- Variables for which failures on demand or initiating events were observed were described using binomial or Poisson distributions, respectively.
- Variables estimated using engineering judgment and for which distribution characteristics were unknown (for example, the recovery factors and plant-specific tailoring factors) were described using a truncated log-uniform distribution. The end points for these variables were determined using engineering judgment. Such a distribution assumes that any set of values between the extremes, on a logarithmic scale, is equally likely.

In both cases, the distribution characteristics were used to estimate the mean and variance for each variable. These values, plus calculated or estimated extreme values, were then used to determine the potential impact of analysis assumptions and data variability on event ranking, the distribution of events, and the estimated potential severe core damage frequency.

6.1 Development of Mean and Variance Estimates

For variables for which point estimates were developed from failures or initiating events, the variance was estimated based on a binomial failure model for demand failure occurrences and a Poisson model for initiating event occurrences.[1] For f failures in D demands, the failure per demand estimate, $\hat{p}$, is

$$\hat{p} = f/D \ .$$

The variance σ^2 associated with $\hat{p}$ can be estimated as

$$\sigma^2_{\hat{p}} = \hat{p}(1 - \hat{p})/D \ .$$

For f initiating events in time T, the initiating event frequency estimate is

$$\hat{\lambda} = f/T \ ,$$

and the associated variance estimate is

$$\sigma^2_{\hat{\lambda}} = \hat{\lambda}/T = f/T^2 \ .$$

For variables for which observational data were not obtained in the ASP Program (such as the numeric values associated with recovery classes), the values the variable could assume were based on other estimates and on engineering judgment. Between extreme values, the variable was assumed to be capable of being modeled using a truncated log-uniform distribution for the purpose of developing estimates of mean and variance.

With a truncated log-uniform distribution between end points x_{min} and x_{max},

$$F(x) = P(\underline{X} \leq x) = \frac{\ln x - \ln x_{min}}{\ln x_{max} - \ln x_{min}} \ ,$$

and

$$f(x) = \frac{dF}{dx} = \frac{1}{x(\ln x_{max} - \ln x_{min})} \ .$$

The mean or expected value of $\underline{\overline{X}}$, $E(\underline{\overline{X}})$, is then

$$\overline{x} = E(\underline{\overline{X}}) = \int_{x_{min}}^{x_{max}} x \ f(x) \ dx$$

$$= \int_{x_{min}}^{x_{max}} x \cdot \frac{1}{x(\ln x_{max} - \ln x_{min})} \ dx$$

$$= \frac{x_{max} - x_{min}}{\ln x_{max} - \ln x_{min}} \ .$$

The variance σ_x^2 is

$$\sigma_x^2 = E(\underline{\overline{X}^2}) - [E(\underline{\overline{X}})]^2$$

$$= \int_{x_{min}}^{x_{max}} x^2 \ f(x) \ dx - [E(\underline{\overline{X}})]^2$$

$$= \frac{x_{max}^2 - x_{min}^2}{2(\ln x_{max} - \ln x_{min})} - \frac{(x_{max} - x_{min})^2}{(\ln x_{max} - \ln x_{min})^2} .$$

In the study, both the failure-on-demand probabilities and the resulting potential severe core damage frequency were obtained through mathematical operations on observed and estimated variables. For such operations, the mean value for each variable can be used to approximately calculate the mean value of the resulting operation. The variance associated with such a result can be approximated as follows.

For a general function

$$y = F(x_1, x_2, \ldots, x_N) ,$$

the first-order Taylor approximation for the variance of y, σ_y^2, is

$$\sigma_y^2 \approx \sum_{i=1}^{N} \left(\frac{\partial F}{\partial x_i}\right)^2 \sigma_{x_i}^2 + \sum_{i=1}^{N} \sum_{\substack{j=1 \\ i \neq j}}^{N} \left(\frac{\partial F}{\partial x_i} \frac{\partial F}{\partial x_j}\right) \text{cov}(x_i, x_j) ,$$

where $\sigma_{x_i}^2$ is the variance of x_i and $\text{cov}(x_i, x_j)$ is the covariance of x_i and x_j. In terms of the expectation E,

$$\text{cov}(x_i, x_j) = E[(x_i - \overline{x}_i) (x_j - \overline{x}_j)] ,$$

where $\overline{x}_i$ is the mean of x_i and $\overline{x}_j$ is the mean of x_j. If x_i and x_j are treated as independent random variables, then

$$\text{cov}(x_i, x_j) = 0 .$$

This results in an approximation for the variance of y of

$$\sigma_y^2 \cong \sum_{i=1}^{N} \left(\frac{\partial F}{\partial x_i}\right)^2 \sigma_{x_i}^2 \ ,$$

where the partial derivatives $\partial F/\partial x_i$ are evaluated at the means.

6.1.1 Application to initiating event frequency and function failure probability estimates

The ASP Program considered the potential for failure recovery. For such partial failure estimates, one approach is to explicitly separate occurrences from recovery estimates and associate variances with each factor. In this study, four recovery classes were used (see Table 3.1). In the case of initiating events, the initiating event frequency was calculated by summing the product of the number of observed events in each recovery class and the numeric value for the recovery class (i.e., summing the "fractional" events) and dividing by the time interval as follows:

$$\lambda = \sum_{i=1}^{4} \frac{N_i R_i}{T} \ ,$$

where T is the time interval, N_i is the number of events observed in recovery class i, and R_i is the numeric value of recovery class i. Splitting λ into terms that address recovery factors and Poisson occurrence rates (N_i/T) separately,

$$\hat{\lambda} = R_1 \frac{N_1}{T} + R_2 \frac{N_2}{T} + R_3 \frac{N_3}{T} + R_4 \frac{N_4}{T} \ .$$

Letting $Z_i = N_i/T$,

$$\hat{\lambda} = R_1 Z_1 + R_2 Z_2 + R_3 Z_3 + R_4 Z_4 \ .$$

Using the Taylor Series approximation for the variance and assuming independence results in an estimate for $\sigma_{\hat{\lambda}}^2$ of

$$\sigma_{\hat{\lambda}}^2 = Z_1^2 \sigma_{R_1}^2 + Z_2^2 \sigma_{R_2}^2 + \ldots + R_1^2 \sigma_{Z_1}^2 + R_2^2 \sigma_{Z_2}^2 + \ldots$$

$$= \frac{N_1^2}{T^2} \sigma_{R_1}^2 + \frac{N_2^2}{T^2} \sigma_{R_2}^2 + \ldots + R_1^2 \frac{N_1}{T^2} + R_2^2 \frac{N_2}{T^2} + \ldots \ .$$

For failures on demand, the variability in the number of demands must also be considered. In this case, the function failure probability estimate, $\hat{p}$, was calculated as follows:

$$\hat{p} = \frac{1}{D} \sum_{i=1}^{4} N_i R_i \ .$$

Recognizing that p is small for functions considered in the ASP Study and thus $\sigma_{\hat{p}}^2 = \hat{p}/D = N/D^2$, the variance of $\hat{p}$ can be approximated by

$$\sigma_{\hat{p}}^2 = \frac{N_1^2}{D^2} \sigma_{R_1}^2 + \frac{N_2^2}{D^2} \sigma_{R_2}^2 + \ldots + R_1^2 \frac{N_1}{D^2} + R_2^2 \frac{N_2}{D^2} + \ldots + \left(\frac{\partial \hat{p}}{\partial D}\right)^2 \sigma_D^2 \ ,$$

where the last term provides a variance contribution due to errors in estimating demands. Continuing,

$$\sigma_{\hat{p}}^2 = \frac{N_1^2}{D^2} \sigma_{R_1}^2 + \frac{N_2^2}{D^2} \sigma_{R_2}^2 + \ldots + R_1^2 \frac{N_1}{D^2} + R_2^2 \frac{N_2}{D^2} + \ldots$$

$$+ \frac{1}{D^4} (N_1 R_1 + N_2 R_2 + N_3 R_3 + N_4 R_4)^2 \sigma_D^2 \ .$$

Note that the developments for both initiating events and function failures include terms associated with recovery value variation and terms associated with variation of other parameters.

6.1.2 Application to industrywide average potential severe core damage frequency estimate

The industrywide potential severe core damage frequency estimate was developed by summing the conditional probabilities of subsequent potential severe core damage associated with precursors that involved initiating events, adding a contribution from unobserved but expected initiators (i.e., unreported losses of feedwater), and dividing by the number of reactor-years in the observation period. This calculation involved application of recovery factors in the calculation of failure probabilities, in the event trees, and in application of failure probabilities for event tree branches for which failures were not observed.

The mean value for the core damage frequency estimate was approximated using estimated mean values for each variable employed in the calculation. A variance estimate was generated again through the use of the Taylor Series expansion. Accordingly, the variances estimated for each variable used in the overall calculation were combined with the square of the partial derivative of the frequency estimate associated with the variable. The partial derivatives were calculated by finite differences using the computer code and data base employed in the program to model the failure sequences. The following variables were considered in this estimation:

- recovery class values,
- function failure probabilities,
- loss-of-main-feedwater frequency [used in calculation of contribution from unanalyzed (unobserved) but expected initiators],
- β factor (used in modeling of degraded functions),
- plant-specific tailoring values.

In addition, the variability in the number of observed initiating events was considered. The contribution of each initiating event j was considered as a product of its conditional probability P_j and its occurrence rate 1/T. The mean potential severe core damage frequency estimate is then:

$$\hat{\lambda}_{PSCD} = \frac{1}{T} P_1 + \frac{1}{T} P_2 + \dots + \frac{N_{LOFW}}{T} P_{LOFW}$$

$$= Z_1 P_1 + Z_2 P_2 + \dots + Z_{LOFW} P_{LOFW}$$

$$= \sum_{\substack{j \\ INIT}} Z_j P_j + Z_{LOFW} P_{LOFW} ,$$

where Z_j is the Poisson estimate of the occurrence rate for initiator j, (1/T); PSCD is potential severe core damage; and N_{LOFW} is the number of loss-of-feedwater events. The overall variance is then estimated as

$$\sigma^2_{\hat{\lambda}} = \sum_{\substack{i \\ VARI}} \left[\frac{\partial}{\partial x_i} \left(\sum_{\substack{j \\ INIT}} Z_j P_j + Z_{LOFW} P_{LOFW} \right) \right]^2 \sigma^2_{x_i}$$

$$= \frac{1}{T^2} \sum_{\substack{i \neq Z,\ N_{LOFW} \\ VARI}} \left[\frac{\partial}{\partial x_i} \left(\sum_{\substack{j \\ INIT}} P_j + N_{LOFW} P_{LOFW} \right) \right]^2 \sigma^2_{x_i}$$

$$+ \frac{1}{T^2} \sum_{\substack{j \\ INIT}} P_j^2 + \frac{1}{T^2} P^2_{LOFW} \sigma^2_{N_{LOFW}} ,$$

where VARI stands for variables and INIT stands for initiators.

The uncertainty associated with potential overcounting (see Sects. 3.2.4 and 5.3) would add to this estimate but has not been included in this analysis.

Once a mean and variance have been estimated for a particular function F, distribution assumptions can again be used to then approximate the range of F. The truncated log-uniform distribution was used to estimate a range for each initiator and function included on the standardized event trees (the range was used in subsequent sensitivity analyses) as well as to estimate a range for the industrywide average potential severe core damage frequency.

In the use of the above formulas, it must be emphasized that (1) the formulas are approximate and should only be interpreted as indicating gross values, and (2) the Taylor Series expansion gives approximate variances for the associated estimator. These can be used as indications of the uncertainty associated with the estimates but are not in the strict sense related to classical confidence intervals.

6.2 Calculation of Specific Extremes, Means, and Variances for ASP Variables

Estimates of the mean, variance, and maximum and minimum values for each variable used in the initiating event frequency, function failure probability, and potential severe core damage frequency calculations were developed based on the formulas included in Sect. 6.1.

Table 6.1 summarizes these results and indicates the type of information used in developing the estimates. Table 6.2 provides supporting data used in the development of Table 6.1.

6.3 Sensitivity Analyses

Sensitivity analyses were undertaken to determine the potential impact on event rankings and on the distribution of event conditional probabilities because of variation in individual variables used in the calculations. The impact on the potential severe core damage frequency estimate of variables that caused large variation in event ranking or event distribution was also addressed. [The impact of variation of each variable on this frequency estimate can also be inferred from the value of its partial derivative (see Sect. 6.4)]. The potential impact of variation in multiple variables has not been addressed to date except in the uncertainty analysis.

The estimated maximum and minimum values for each variable developed in Sect. 6.2 were used to estimate alternate conditional probabilities of potential severe core damage associated with each precursor. These revised values were then compared with the point estimates developed in Chap. 5 by comparing changes in event probability distribution and event ranking. An example of such a comparison for recovery class R1 is shown in Fig. 6.1. Equivalent figures for applicable variables are included in Appendix E.

Table 6.1. Estimated mean, variance, and extreme values for ASP variables

ASP variables	x_{min}	x_{max}	$\overline{x}_{mean}$	σ_x^2	Note number
Recovery classes					
R1	0.3	1.0	0.58	4.0E-2	1
R2	0.1	0.8	0.34	3.8E-2	1
R3	0.03	0.3	0.12	5.6E-3	1
R4	0.01	0.1	0.04	6.2E-4	1
Plant-specific tailoring factors					
P1	1.0	1.0	1.0	0.0	2
P2	0.1	0.8	0.34	3.8E-2	1
P3	0.03	0.3	0.12	5.6E-3	1
P4	0.01	0.1	0.04	6.2E-4	1
P5	0.0	0.0	0.0	0.0	2
β factor	0.03	0.3	0.12	5.6E-3	1
BWR functions					
HPCI/RCIC	1.04E-4	1.04E-2	2.24E-3	6.78E-6	4,6
High-pressure cooling provided following LOCA	3.0E-3	5.0E-2	1.70E-2	1.63E-4	3
Emergency power	9.38E-5	1.06E-2	2.22E-3	6.91E-6	4,6
Automatic depressurization	1.94E-7	8.68E-2	6.67E-3	2.4E-4	4,6
SBLC only	3.0E-2	3.0E-1	1.2E-1	5.6E-3	1
Scram only	4.45E-7	1.52E-3	1.87E-4	1.07E-7	4,6
LPCI/core spray	6.79E-8	1.50E-3	1.50E-4	8.90E-8	3
Reactor isolation (large SLB)	2.78E-7	2.67E-2	2.33E-3	2.57E-5	4,6
Long-term core cooling	1.71E-6	5.98E-4	1.02E-4	2.02E-8	4,6
BWR initiators					
LOFW	4.1E-2	1.0	3.0E-1	6.61E-2	3
LOOP	5.16E-3	4.72E-2	1.9E-2	1.36E-4	5,6
LOCA	7.46E-3	4.60E-2	2.12E-2	1.17E-4	5,6
Large SLB	4.53E-7	1.0E-2	1.0E-3	4.00E-6	3
PWR functions					
Reactor trip	~0.0	1.50E-3	3.60E-5	2.57E-8	3
AFW given reactor trip success	5.84E-5	7.57E-4	2.73E-4	3.68E-8	4,6
AFW given reactor trip failure	1.24E-6	2.73E-2	2.73E-3	2.98E-5	3
PORV demanded	1.0E-2	1.0E-1	4.00E-2	6.22E-4	3
PORV closure given SLB	3.59E-4	1.0E-2	2.90E-3	6.61E-6	3
HPI given AFW success	2.01E-5	3.04E-3	6.03E-4	5.61E-7	4,6
HPI given AFW failure	1.0E-1	8.0E-1	3.4E-1	3.8E-2	3
Long-term core cooling	1.20E-5	1.20E-3	2.57E-4	8.87E-8	4,6
Turbine generator runback	1.0	1.0	1.0	0.0	2
Emergency power	9.34E-6	1.98E-3	3.68E-4	2.30E-7	4,6
AFW given emergency power failure	3.0E-2	3.0E-1	1.2E-1	5.6E-3	1
SG isolation (large SLB)	2.23E-5	3.18E-3	6.37E-4	6.13E-7	4,6
Concentrated boric acid addition given HPI success (large SLB)	2.30E-5	4.35E-3	8.27E-4	1.13E-6	4,6
PORV opened due to HPI (large SLB)	6.4E-1	1.0	8.00E-1	1.1E-2	3
PORV closure given SLB	8.0E-4	2.0E-2	6.00E-3	2.65E-5	3

Table 6.1 (continued)

ASP variables	x_{min}	x_{max}	$\overline{x}_{mean}$	σ_x^2	Note number
PWR initiators					
LOFW	4.1E-2	1.0	3.0E-1	6.61E-2	3
LOOP	9.02E-3	6.19E-2	2.75E-2	2.19E-4	5,6
LOCA	3.90E-3	1.70E-2	8.90E-3	1.37E-5	5,6

Notes:

1. Means and variances were developed from estimated extreme values assuming the variable could be modeled using a truncated log-uniform distribution.

2. Variances and ranges are not appropriate for these plant-specific factors.

3. Minimum values and variances were developed from an estimated maximum value and mean, assuming the variable could be modeled using a truncated log-uniform distribution.

4. Means and variances were developed from failures observed in the ASP Program using a binomial failure model. The variance developed reflects the number of occurrences in each recovery class, the value and variance associated with each recovery class, and the variance associated with the number of demands estimated.

5. Means and variances were developed from initiating events observed in the ASP Program using a Poisson failure model. Variances for initiating event frequencies calculated from precursor data reflect recovery factors and the number of occurrences in each recovery class.

6. See Table 6.2 for demand variances, mean number of demands, maximum number of demands assumed, and number of occurrences in each recovery class (N_i) calculated from precursor data.

Based on a review of the distributions included in Appendix E, it is concluded that the largest impacts resulted from changes in the values of variables associated with recovery classes R1, R2, and R3 and plant tailoring class P3. Changes in these variables produced changes in a large number of precursor conditional probabilities and across the entire range of probabilities with respect to the mean-value (reference) ranking. Changes in the β factor, the numeric value associated with plant tailoring class P2, the failure probabilities for BWR ADS, PWR emergency power and HPI (given AFW success), and the frequencies for BWR LOFW and PWR LOOP impacted a number of precursors; but the impact occurred typically only in a limited part of the overall range. Impacts of changes on the remaining variables were either negligible or nonexistent.

The following observations are noted regarding the impact of changes in the numeric values for recovery classes R1, R2, and R3 and plant tailoring class P3:

Recovery class R1 — Change in ranking for a substantial number of events; noticeable increase in the number of events in the 10^{-2} to 10^{-3} probability range and a decrease in the 10^{-5} to 10^{-8} range when the value of R1 is maximum; changes in event probabilities widespread although not substantial.

Table 6.2. Supporting data for Table 6.1

ASP variables	N_i, number of occurences in recovery class R_i				Function demands					
	N_1	N_2	N_3	N_4	Mean per reactor-year	Max. per reactor-year	Number of reactor-years	Mean	Max.	Variance, σ^2
BWR functions										
HPCI/RCIC	4.0	3.0	2.0	2.0	12.3	52	132.7	1,632	6,900	3.05E+6
Emergency power	8.0	1.0	1.0	0.0	12.0	52	191.8	2,301	9,974	6.34E+6
Automatic depressurization	2.0	0.0	1.0	0.0	1.0	12	191.8	192	2,301	1.84E+5
Scram	0.0	1.0	0.0	0.0	9.5	52	191.8	1,822	9,974	5.80E+6
Reactor isolation (large SLB)	1.0	0.0	0.0	0.0	1.3	12	191.8	249	2,301	2.24E+5
Long-term core cooling	0.0	1.0	0.0	0.0	17.3	52	191.8	3,318	9,974	6.52E+6
PWR functions										
AFW given reactor trip success	1.0	4.0	1.0	2.0	27.3	52	287.6	7,852	14,956	1.07E+7
HPI given AFW success	3.0	1.0	0.0	0.0	12.0	52	287.6	3,452	14,956	1.42E+7
Long-term core cooling	2.0	0.0	0.0	0.0	15.7	52	287.6	4,516	14,956	1.48E+7
Emergency power	2.0	0.0	1.0	0.0	12.1	52	287.6	3,480	14,956	1.43E+7
SG isolation (large SLB)	3.0	1.0	1.0	0.0	12.0	52	287.6	3,452	14,956	1.42E+7
Concentrated boric acid addition given HPI success (large SLB)	3.0	0.0	0.0	0.0	12.0	52	175.5	2,106	9,126	5.31E+6
Initiators										
BWR										
LOOP	0.0	10.0	2.0	0.0						
LOCA	7.0	0.0	0.0	0.0						
PWR										
LOOP	0.0	20.0	9.0	1.0						
LOCA	2.0	1.0	4.0	0.0						

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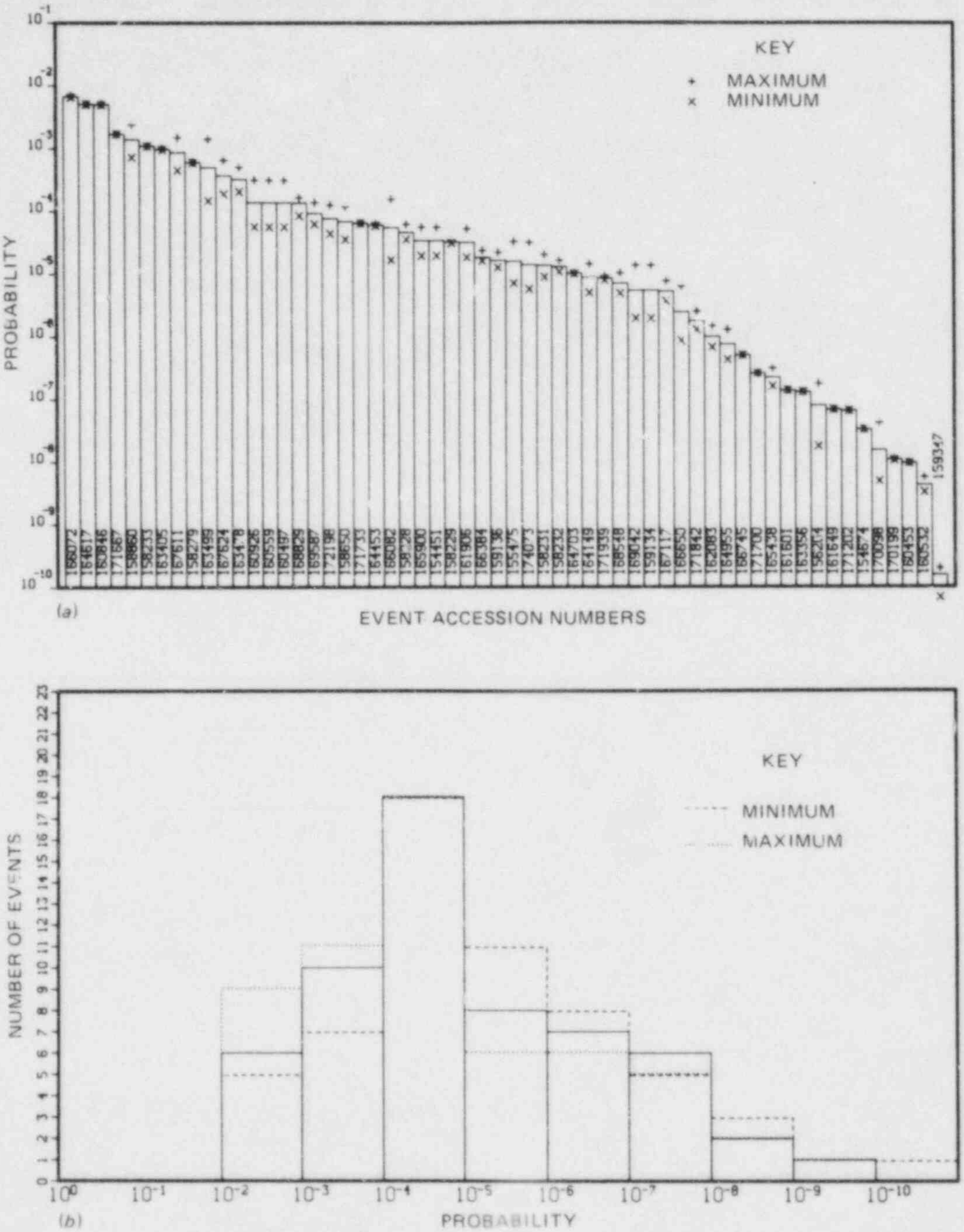

Fig. 6.1. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in recovery class R1 numeric values.

Recovery class R2 — Change in ranking for a substantial number of events; movement of three events into the 10^{-1} to 10^{-2} probability range when the value of R2 is maximum, although the general shape of the distribution remains unchanged; minimum value of R2 results in a flattening of the distribution.

Recovery class R3 — Change in ranking for a substantial number of events; movement of two events into the 10^{-1} to 10^{-2} probability range when value of R3 is maximum; changes in event probabilities not substantial.

Plant tailoring class P3 — Change in ranking for a substantial number of events, but with small changes in event probabilities and with many event probabilities not changing at all; movement of one event into the 10^{-1} to 10^{-2} probability range.

The following observations are noted for those variables for which changes moderately impact event distribution and ranking:

β factor — Change in probabilities for some high probability events, but minimal changes elsewhere in the range; movement of one event into the 10^{-1} to 10^{-2} range for maximum value of β.

Plant tailoring class P2 — Change in ranking of a few events affected in the middle and higher probability ranges; a flattening of the distribution over the 10^{-2} to 10^{-4} range for maximum values of P2.

BWR ADS failure probability — Change in ranking for several events in the middle probability range; changes in a few probabilities appear substantial.

BWR loss-of-main-feedwater frequency — Change in ranking of a few events affected in the middle probability range; movement of one event into the 10^{-1} to 10^{-2} range.

PWR emergency power failure probability — Change in ranking for a few events in low and middle probability ranges; distribution unchanged by minimum value, distribution more peaked for maximum value.

HPI (given AFW success) failure probability — Change in ranking of a few events in upper and middle probability ranges but with none moving into the 10^{-1} to 10^{-2} range.

PWR LOOP frequency — Change in event probabilities of many events in lower probability range; shape of distribution for lower probability events changed for minimum frequency value.

In general, the overall ranking for the set of precursors did not change substantially with variation of any of the parameters; that is, no event changed from a high probability to a low probability event or vice versa. Furthermore, the general shape of the distribution remained fairly constant with the mode in the 10^{-4} to 10^{-5} range, and the extremes moved only one order of magnitude beyond the reference distribution extremes.

As a further illustration of the effect of changes in the variables appearing to impact the sensitivity results the most, estimates of the average potential severe core damage frequency were calculated over the ranges of five variables: recovery classes R1, R2, R3, plant tailoring class P3, and the β factor. The β factor was chosen in addition to the others because changes in some high-probability events were observed with its variation. The results of these calculations, ranked in order of greatest impact, are shown in Table 6.3.

Table 6.3. Estimates of average potential severe core damage frequency calculated over the ranges of five variables

Variable, range	Average potential severe core damage frequency (per reactor-year)	
	Using minimum variable value	Using maximum variable value
Numeric value for recovery class R2, 0.1–0.8	6.7E-5	3.9E-4
Numeric value for recovery class R3, 0.03–0.3	9.8E-5	3.1E-4
β factor, 0.03–0.3	1.3E-4	2.5E-4
Numeric value for plant tailoring class P3, 0.03–0.3	1.3E-4	2.4E-4
Numeric value for recovery class R1, 0.3–1.0	1.5E-4	2.0E-4

Note that variation in R2 produces the greatest change in the average frequency, but the results differ from the mean frequency estimate of 1.6E-4/reactor-year by less than a factor of 2.5.

In summary, the sensitivity analysis shows that relatively few of the variables individually impact the ranking, distribution, and conditional probability of potential severe core damage associated with the precursors. Although some impact on ranking and distribution can be seen from a few variables, no variables substantially impact the industry-average frequency estimate.

6.4 Calculation of Variance and Range Values for Industrywide Average Potential Severe Core Damage Frequency Estimate

The estimated variance on the average potential severe core damage frequency was calculated according to the approach described in Sect. 6.1. Partial derivatives for each variable used in the potential severe core damage frequency estimate were approximated by finite differences and calculated using the computer code employed in the program to model potential failure sequences. This computer program permits variation of all variables employed in the calculations.

The partial derivatives so determined are listed in Table 6.4, together with the variance contribution associated with each variable. These contributions, together with those associated with the occurrence rate for each event, were summed to estimate the overall variance. The

Table 6.4. Partial derivatives, variance estimates, and overall variance contribution for ASP variables

ASP variables	Partial derivatives[a] ($\partial P/\partial x_i$)	Variance on x_i ($\sigma^2_{x_i}$)	Contribution of x_i to overall variance[a] $[(\partial P/\partial x_i)^2 \times \sigma^2_{x_i}]$
Recovery classes			
R1	1.04E-2	4.0E-2	4.33E-6
R2	5.90E-2	3.8E-2	1.32E-4
R3	1.03E-1	5.6E-3	5.94E-5
R4	6.07E-3	6.2E-4	2.28E-8
Plant-specific tailoring factors			
P1	0.0	0.0	0.0
P2	5.34E-3	3.8E-2	1.08E-6
P3	5.90E-2	5.6E-3	1.95E-5
P4	9.41E-3	6.2E-4	5.49E-8
P5	0.0	0.0	0.0
β factor	6.2E-2	5.6E-3	2.15E-5
BWR functions			
HPCI/RCIC	1.02E-1	6.41E-6[b]	6.67E-8
High-pressure cooling provided following LOCA	1.26E-2	1.63E-4	2.59E-8
Emergency power	1.06E-3	6.41E-6[b]	7.20E-12
Automatic depressurization	1.04E-1	2.40E-4[b]	2.60E-6
SBLC only	3.12E-3	5.6E-3	5.45E-8
Scram only	2.10	9.6E-8[b]	4.23E-7
LPCI/core spray	1.03E-1	8.90E-8	9.44E-10
Reactor isolation (large SLB)	0.0	2.50E-5[b]	0.0
Long-term core cooling	1.67E+1	1.68E-8[b]	4.69E-6
BWR initiators			
LOFW	6.72E-3	NA[c]	NA[d]
LOOP	0.0	NA	NA
LOCA	0.0	NA	NA
Large SLB	0.0	NA	NA
PWR functions			
Reactor trip	1.9E-1	2.57E-8	9.18E-10
AFW given reactor trip success	9.77	2.61E-8[b]	2.49E-6
AFW given reactor trip failure	9.5E-4	2.98E-5	2.69E-11
PORV demanded	7.9E-5	6.2E-4	3.87E-12
PORV closure given SLB	1.09E-3	6.61E-6	7.85E-12
HPI given AFW success	8.95E-1	5.27E-7[b]	4.22E-7
HPI given AFW failure	7.17E-3	3.8E-2	1.95E-6
Long-term core cooling	1.07	8.08E-8[b]	9.25E-8
Turbine generator runback	0.0	0.0	0.0
Emergency power	1.33E-1	2.16E-7[b]	3.82E-9
AFW given emergency power failure	0.0	5.6E-3	0.0
SG isolation (large SLB)	0.0	5.80E-7[b]	0.0
Concentrated boric acid addition given HPI success (large SLB)	0.0	1.04E-6[b]	0.0
PORV opened due to HPI	2.0E-7	1.1E-2	4.4E-16
PORV closure given SLB	3.0E-5	2.65E-5	2.39E-14
PWR initiators			
LOFW	8.15E-3	NA	NA[d]
LOOP	0.0	NA	NA
LOCA	0.0	NA	NA

[a] $P = \sum_{j \in INIT} P_j + N_{LOFW} P_{LOFW}$; i.e., P is not a frequency estimate.

[b] These variances have the recovery class variances removed, because variance contribution associated with each recovery class has been developed independently in this table.

[c] Not applicable.

[d] The variance contribution for the number of LOFW events is developed as a part of the variance on the number of events. This contribution is 7.41E-6 and 5.90E-5 for the number of LOFW events and the number of initiators, respectively.

contribution of these variables to the overall variance estimate follows:

Description	Contribution (%)
Recovery class numeric values	~61
Event occurrences	19
Plant tailoring values	7
β factor	7
BWR functions	2
PWR functions	2
Number of LOFW occurrences	2

The resulting variance estimate is 1.7E-8 (standard deviation of 1.3E-4).

By assuming that the industrywide potential severe core damage frequency can be represented by a known distribution, one can develop an estimate of the range on the frequency. This has been done for the two frequency estimates developed in Sect. 5.3 (without removal of potential overcounting, 1.6E-4/reactor-year, and with potential overcounting removed, 1.1E-4/reactor-year) using two distributions, the truncated log-uniform distribution discussed previously and the lognormal distribution:

Frequency estimate (per reactor-year)	Range based on distribution (per reactor-year)	
	Log uniform	Lognormal (5 to 95%)
1.6E-4	2.5E-5 to 5.1E-4	8.3E-6 to 5.8E-4
1.1E-4	4.7E-6 to 5.2E-4	1.5E-5 to 3.3E-4

These values are estimates of the range on the industrywide average frequency and do not bound plant-to-plant variations. Individual plant potential severe core damage frequency estimates would be expected to vary widely from the industry average calculated in this study.

An alternate estimate of a maximum value can be developed through the use of Chebyshev's Inequality.[2] This inequality can be used to describe the percentage of all values of a variable that are within a specified number of standard deviations from the mean. The bounds defined by Chebyshev's Inequality are valid for any distribution and are typically wider than those associated with specific distributions. The 95% upper bound estimated using this inequality is 5.8E-4/reactor-year for the frequency estimate without consideration of potential overcounting (mean of 1.6E-4/reactor-year) and 5.2E-4/reactor-year when overcounting is addressed (mean of 1.1E-4/reactor-year).

As stated previously, the variance calculation is approximate and requires independence assumptions, which are most likely not completely correct. As such, the variance estimate should be considered as an indication of uncertainty associated with the frequency estimate. In addition, the ranges computed based on the use of assumed distributions should not be associated directly with classical confidence bounds.

References

1. Nuclear Regulatory Commission, *PRA Procedure Guide*, NUREG/CR-2300, Vol. 1, January 1983, p. 5-23.

2. H. F. Martz and R. A. Waller, *Bayesian Reliability Analysis*, John Wiley & Sons, Inc., New York, 1982, pp. 42–43.

7. DISCUSSION OF PROGRAM METHODS AND LIMITATIONS

This chapter provides further discussion and amplification of ASP Program methods and limitations. The calculations and conclusions in this report were derived from operational data. However, application of the data in arriving at the results involved interpretation of the written record, determination of probable operator response, and application of this information in mathematical models. This inherently involves subjective or engineering judgment and other modeling limitations. These limitations as well as other important steps in this process are addressed in this chapter.

7.1 Potential Severe Core Damage vs Core Melt

This report involves the assessment of the impact of selected events with regard to potential severe core damage. It is therefore most important that the reader understand what is meant by potential severe core damage. For the purposes of this study, potential severe core damage is defined as a situation in which, given certain initiating events, the performance of one or more functions required to shut down the reactor or cool the core falls below that level of performance known to be effective to prevent core damage. Criteria for equipment performance levels are generally consistent with plant Technical Specifications and with analyses reported in Safety Analysis Reports.

Some further reduction in performance (below that described above) is generally required to actually achieve core damage. If actual core damage does occur, then progression to actual core melt is possible. However, such a progression, if it occurs, would be expected to take some time, during which the progression might cease because of recovery actions taken at the plant. In any event, core melt is a less probable condition than core damage, which is in turn less probable than potential severe core damage as used in this study. The difference in probabilities between these states is extremely difficult to quantify, and estimates vary widely.

7.2 Event Screening and Selection

The ASP Study uses a two-phase LER screening and selection process to review a large licensee event data base to identify a relatively small set of precursors to potential severe core damage. The basic steps in the ASP screening process are described in Chap. 3, where it is noted that the selection process might lead to some significant events being missed and conversely to the selection of some inappropriate or marginally significant events. Inappropriate events are eliminated in the detailed review. Marginally significant events are identified in the subsequent ranking process.

It probably is not possible to define an "absolute" system for selection and ranking of LERs, but alternate screening criteria have been proposed.* The selection criteria are not derived mathematically, and therefore, subjective judgments are made in establishing a set of selection criteria. An individual criterion may also require additional subjective judgments during actual screening of LERs, although the intent of the criterion is to minimize this phenomenon. This subjectivity makes it difficult to obtain absolute agreement even on the application of a single set of selection criteria.

The primary importance of any selection criterion is its ability to extract from a very large data base a manageable set of potentially significant items to be analyzed in more detail. With any screening criteria, some potentially significant events may be overlooked. This, however, is not considered a fatal flaw in a screening and selection process that is being applied to an incomplete data base.

The impact of missed events is dependent on the type of event missed. A missed event of a type already included in the data base (e.g., a failure on demand during testing of a function for which a number of failures on demand have already been observed) would not be expected to impact analysis results to the degree that a missed event of a previously unobserved type might. An example of the latter would be a BWR LPCI/core spray failure (none were observed through 1981).

Certain events that impact risk are currently not addressed in the ASP Program. Some of these involve only a potential for causing failures in functions or initiating events. These events are excluded, for the most part, because the actual historic failures and initiating events are considered better indicators of the number of such events. Other events (such as those associated with pressurized-thermal-shock sequences) have not been addressed pending resolution of their risk significance. The exclusion of these events is described in greater detail in Sect. 3.2.4.

Other historic events that have not been selected are failures in single trains of support systems that do not induce a transient. If not repaired quickly, such a failure often requires a plant shutdown — an action that can further tax the degraded support system and potentially initiate a transient. Such events would not affect the initiating event frequency, the function failure probability, and (because there was no actual initiator) the potential severe core damage frequency estimates developed in this study; but they could be significant from the standpoint of conditional probability of potential severe core damage.

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*For example, an alternate method might involve the use of a selection and ranking process whereby the number of remaining operable mitigating systems (or "barriers") was considered as the primary selection criterion. The highest ranked LERs would be those in which no additional systems were available to prevent severe core damage in an event sequence of interest. Other LERs would be ranked lower based on the availability of one, two, or more additional mitigating systems or functions. This selection method would likely define a set of significant LERs that is somewhat different than the set identified in the ASP Study.

7.3 Methodology and Data Refinement

No substantial methodological differences exist between the 1969–79 study effort reported in NUREG/CR-2497 (Ref. 1) and the current effort. Both efforts (1) used the same selection criteria; (2) estimated initiating event frequencies and demand failure probabilities from observed events; and (3) applied these estimates, in conjunction with recovery probabilities for observed failures, to event trees to develop a conditional probability of potential severe core damage associated with each event. The estimated industrywide frequency of potential severe core damage was calculated in the same way in both efforts.

However, the 1980–81 results are not directly comparable with the 1969–79 results as reported in NUREG/CR-2497. The results of the current study reflect the following refinements to the ASP methodology and models over those described in NUREG/CR-2497:

- Changes were made in the point estimates for recovery classes (in part to accommodate sensitivity and uncertainty analyses).
- An additional recovery class was introduced to model procedural actions in the control room during moderately stressed situations.
- The degraded function model was improved.
- The PWR steam line break event tree was revised to more correctly model the requirements for steam generator isolation and concentrated boric acid addition.
- The BWR event trees were revised to more correctly model expected response to ADS actuation following scram failure and HPCI/RCIC failure.
- Certain functions (particularly PWR bleed and feed and AFW following emergency power failure) were tailored to better reflect plant conditions.
- Uncertainty and sensitivity analyses were introduced.

In addition, the results herein are dependent in part on 1969–79 data, which were first revised to reflect resolution of comments received on NUREG/CR-2497 and then were used together with 1980–81 data in certain calculations.

7.4 Recovery Factors

In the course of studying operating experience, it becomes apparent that the control room and plant operating staff play a major role in mitigating events through performance of recovery actions when systems or components malfunction. Such actions can range from the manual actuation of systems at the appropriate time or the manual compensation for an auto-start failure according to routine procedures to the rapid in-plant repair of essential components. An operator must also make decisions pertaining to the use of manually actuated alternate systems if primary systems are discovered to be unavailable.

Each precursor was assessed to account for the possibility that observed failures could have been corrected in sufficient time following an actual or postulated initiating event. When failures were accounted for, the chance of recovery was included by considering each event to be composed of the observed failure and a subsequent potential recovery step. Four recovery classes were defined to describe the potential for nonrecovery in an event (see Table 3.1). The likelihood of recovery considered whether such recovery would be required in a moderate- to high-stress situation following a postulated initiating event.

Although recovery classes were assigned based on the specifics of each event as reported, they are admittedly subjective. The likelihood of effecting recovery (following, in many cases, postulated initiating events) is difficult to assess. Interviews with plant maintenance and operations personnel could have provided additional information but were not within the scope of the study. Even with detailed information concerning a specific failure, estimates of repairability can vary widely.[2]

Very little data currently exist concerning effective recovery from failures in nuclear power plants, particularly during conditions of moderate or high stress; yet assumptions concerning such recovery are important to results in this program.

As an example of the impact of alternate recovery assumptions, consider Table 7.1. Although the frequency estimates for the two cases associated with recovery values at their estimated lower and upper limits lie close to the estimated bounds for the calculated 1980–81 industry-average potential severe core damage frequency, they do differ by a factor of 26. This impact is also supported by the uncertainty analysis described in Chap. 6, in which the recovery factors contributed 62% of the overall variance estimate.

The impact of variation of numeric values for recovery classes R1, R2, and R3 on event ranking and probability distribution is generally greater than for all other ASP variables. Variation in the numeric value

Table 7.1. Potential severe core damage frequency estimates for 1980–81[a]

Description	Recovery class numeric values				Average potential severe core damage frequency (per reactor-year)
	R1	R2	R3	R4	
Estimate based on lower end of assumed ranges for each recovery class	0.30	0.10	0.03	0.01	2.9E-5
Estimate based on upper end of assumed ranges for each recovery class	1.0	0.80	0.30	0.10	7.6E-4
Estimate based on program point estimates for each recovery class	0.56	0.34	0.12	0.04	1.6E-4
Estimate based on assumption that no recovery is attempted	1.0	1.0	1.0	1.0	2.8E-3

[a]The above calculations use failure probability estimates for plant functions based on amalgamated 1969–81 data (see Sect. 5.1). These are average estimates across the reactor population. Potential severe core damage frequency estimates for individual plants may vary substantially from these estimates.

of recovery class R4, however, produces negligible impact. Changes in the event ranking, distribution, and the potential severe core damage frequency estimate resulting from individual variation in the numeric values for R1, R2, and R3 are discussed further in Sect. 8.7 and in Chap. 6.

7.5 Standardized vs Plant-Specific Event Trees

Event trees are used in the program to model each precursor. An event tree is a logic model that represents existing dependencies and combinations of actions required to achieve defined end states following an initiating event.

In contrast to typical PRA studies, here the event trees are developed only to the function level. This process was facilitated for a large number of events by the development of standardized event trees for four commonly considered initiating events: loss of main feedwater, loss of offsite power, loss-of-coolant accident, and steam line break. Specific plant systems were associated with the functions included on the event trees. These standardized trees were used with the majority of events selected as precursors. The trees were tailored, to a certain extent, to more accurately represent existing systems at specific plants (see Sect. 3.2 and Table 3.3). This tailoring, using plant-specific tailoring factors, permitted assignment of a unique failure probability to certain functions. However, certain events could not be described using the generic trees. In such cases event-specific (unique) trees were developed.

The use of standardized event trees in the ASP Study is frequently identified as a major source of error. The extent to which it is reasonable to employ standardized event trees as surrogates is highly dependent on the goal desired and the amount of failure data available. The use of such trees provides a versatile means of data evaluation in which failures indicative of an overall failure probability may be observed, although with a corresponding loss of resolution for specific plants. More generalized forms of event trees, not specialized to particular plant hardware, may more effectively model unusual failures, including coupled events associated with human error.

Development of functionally based event trees for different classes of plants offers possible improvement, although this may be accompanied by a potential loss of failure data for some class-specific functions. Functions for which no failures were observed in the ASP Program are assigned failure probabilities based on other estimates, such as PRA models. If the number of cases of plant class-specific functions in which no failures are observed (because of the reduced number of reactor-years in each plant class) is large, then gain in event tree specificity may not compensate for the reduced availability of necessary failure information.

7.6 Modeling of Degraded Functions

The failure probability for event tree branches that included a degraded function (a function that met minimum operability requirements but included no redundancy) was modified to reflect the observed loss of redundancy. The β-factor method for common-cause failures was used to estimate the probability that the remaining portion of a degraded system would fail. In the β-factor method, failures in redundant functions are separated into independent and dependent contributors to total failure, and β is defined as the fraction of total failures attributable to dependent failures.

Alternate methods that some analysts consider more accurately model system performance have been suggested to account for failures in redundant systems (for example, the binomial failure rate model). The β-factor method was used in this study for the following reasons.

— It was considered a reasonable method for estimating the conditional probability of function failure given an observed degraded state, considering the types of failures observed in the study. (Single failures were typically not selected for review.)

— It permitted a degraded function to be clearly defined as one in which no redundancy existed. Functions in which a failure had occurred but which still included some redundancy were not considered as degraded in this analysis.

The impact of variation in the value of β was considered in the uncertainty and sensitivity analyses described in Chap. 6. The range of values assumed for β in this study encompassed values estimated in unrelated studies concerning common-cause fault rates for various equipment (see Sect. 5.2). Based on the results of these analyses, it is concluded that the overall study results for 1980–81 were sensitive to the value of β to a certain degree. This sensitivity is addressed in Chap. 6 and summarized in Sect. 8.7.

7.7 Conservatisms and Nonconservatisms

As with any analytic procedure, the availability of information and modeling assumptions can bias results. Potential sources of error in this study are addressed in detail in Sect. 3.2.4. In that section, the following potential error sources are identified, along with their impact:

- Nonreporting or underreporting of operational events: underestimation.
- Exclusion of certain risk-related events: underestimation.
- Lack of consideration of potential coupling between postulated failures on event trees: underestimation.
- Use of observed failures on demand, primarily due to testing, in estimating the probability of failure on demand: overestimation.
- Lack of accuracy and completeness of the LERs in reflecting pertinent operational information in all cases: impact not known.
- Use of standardized event trees: either overestimation or underestimation.
- Use of average or homogenized data with plant-specific operational occurrences in calculating conditional probabilities of potential severe core damage: impact not known.

— Use of recovery classes to model potential failure recovery: impact not known.
— Assumption of monthly test interval for many functions: impact not known, but some overestimation likely.
— Accounting for observed combined failures as coupled failures and assigning estimated failure probabilities to branches for which success was observed (overcounting): overestimation.

The potential to overestimate ("overcount") the potential severe core damage frequency calculated in the study has been recognized since NUREG/CR-2497. This overestimation results from the potential to overcount the number of failures actually observed and from differences between a statistic associated with an observed failure and the probability of the failure. This problem is discussed in Sects. 3.2.4 and 5.3.

Section 5.3 describes two methods for estimating the degree of overcounting. One was based on the use of standardized event trees and the initiating event frequencies and function failure probabilities developed in the study to calculate an average potential severe core damage frequency. This method yielded an overestimation factor of 3.7. This factor is considered conservative because it assumes independence between functions and ignores unique initiators (i.e., those observed in the ASP Program that could not be modeled using standardized event trees). The second method to estimate overcounting was based on an assessment of the likelihood that multiple failures observed during initiating events were actually related. This method yielded an estimated overcounting factor of 1.5, which is considered more realistic.

Alternate methods of eliminating overcounting have been proposed and are being investigated. The resolution of the overcounting problem is complicated by the small amount of failure and initiating event data actually available and the number of unusual sequences seen. The result of the small amount of data is the lack of observation of portions of many postulated sequences, making the application of the proposed techniques difficult. As stated in the ACRS review letter,[3]

> Concerning the possible factor of three overcounting acknowledged in the report [NUREG/CR-2497], other methods of analyzing the data have been suggested but none appears to be unequivocally "the right one" and the actual degree of "overcounting" remains difficult to quantify.

In addition to the above items, this analysis, as all others, potentially suffers from biases on the part of the analysts. While the program staff has benefited from the extensive review of NUREG/CR-2497, the utility reviews of the events selected for 1980–81, the reviews of drafts of this document, plus the staff's many years of experience in reactor design, reactor operations, and systems evaluations, the process is still subjective; not all specialists will necessarily agree with every event selected and/or omitted or with subsequent treatment of those events.

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 2), Oak Ridge National Laboratory, June 1982.

2. Advisory Committee on Reactor Safeguards, *Official Transcript Proceedings Before Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards*, Subcommittee on Reliability and Probabilistic Risk Assessment, March 9, 1983, pp. 69–78.

3. J. C. Ebersole, Acting Chairman of Advisory Committee on Reactor Safeguards, letter to W. J. Dircks, NRC Executive Director for Operations, May 18, 1983, Subject: ACRS Report on the ASP Study and the Use of Operational Experience Data.

8. DISCUSSION OF RESULTS

8.1 Important Precursors

The following 1980–81 precursors were ranked high by the ranking methods described in Chap. 5. These events involve (1) significant failures in important functions for which no failures had been observed in 1969–79 and (2) system interactions, primarily in electrical systems.

At Browns Ferry 3 (NSIC 163405), 76 control rods failed to insert (75 of 88 from the east bank) during a shutdown. Subsequent draining of the scram discharge volume and initiation of additional scrams resulted in insertion of all control rods 14 min after the first scram attempt. Core power could have been excessively high following a design-basis transient such as a loss of feedwater.

At Brunswick 1 (NSIC 166072), oyster shells accumulated in both RHR heat exchangers because the service water chlorination system was out of service for an extended period of time. During repair of heat exchanger 1B, a second service water pump was started in loop A; this resulted in displacement of the 1A heat exchanger baffle plate due to high-pressure drop caused by the shells and unavailability of both heat exchangers.

At Millstone 2 (NSIC 164617), because of an operator error a 125-V dc bus was deenergized, resulting in a plant transient. Loss of the dc bus, which supplied control power to a number of systems, resulted in loss of the control room annunciators, failure to trip the turbine, failure to transfer to the reserve station transformer, loss of power to emergency bus B, and failure of the main generator switchyard breakers to open. Diesel generator A would have failed to close onto its bus if it had been needed. Diesel generator B tripped because of a service water leak 10 min into the event. Had dc bus A not been restored prior to completion of the main generator reverse-power time-delay relay operation (dc power was restored at 50 s, and the relay operated at 60 s), a station blackout would have occurred when diesel generator B tripped. Power could have been made available to affected buses by manual operation of breakers.

Crystal River 3 (NSIC 160846) experienced a loss of the nonnuclear instrumentation power supply X, which resulted in a reactor power increase, feedwater runback, and opening of the PORV. The reactor scrammed on high pressure, HPI was initiated, and the PORV block valve was manually shut within 5 min. Power was restored to the NNI about 21 min into the transient, but HPI was continued for 84 min.

At Davis-Besse 1 (NSIC 171667), personnel error caused a loss of several buses (including one supplied by its alternate source), a reactor trip, and loss of feedwater. Numerous systems were deenergized. One AFW pump failed to start, and a main steam safety valve lifted and failed to reseat properly.

Also at Davis-Besse 1 (NSIC 158860), the reactor was in cold shutdown in preparation for refueling with the head detensioned, both steam generator manways opened, and one DH loop drained for maintenance. A breaker actuation in a maintenance-revised electrical lineup resulted in loss of two distribution panels, which resulted in full SFAS actuation, isolation of DH letdown, stroking of BWST and sump isolation valves, and air being drawn into the operating pump suction. The pump was stopped to prevent damage. Decay heat removal was unavailable for 2.5 h.

At St. Lucie 1 (NSIC 158233), the failure of a valve in the common component cooling water return line from the reactor coolant pumps resulted in the loss of component cooling to all RCPs. The plant and RCPs were tripped, and a natural circulation cooldown initiated. During the cooldown, it was discovered that a steam bubble had formed in the reactor vessel head due to slower cooldown of the top head compared with the rest of the system. The steam bubble, if it had grown larger, could have impacted natural circulation cooling.

In addition to the above events, other events occurred that involved unexpected failures and interactions. Many of these were ranked less important because of the conditions at the plant where they occurred; they could have been more serious at another plant. Examples of such events follow.

At Crystal River 3 (NSIC 167624), a diesel generator failed to start following a loss of offsite power. The failed emergency bus was powered from an adjacent fossil plant startup transformer via a manual connection.

At Calvert Cliffs 1 (NSIC 158650), an instrument air cooler, cooled by a nonsafety portion of the service water system, leaked air and caused a loss of both safety-related service water trains. The reactor was tripped. An operator error, after cooldown, resulted in isolation of the AFW supply.

At Palisades (NSIC 163356), personnel error resulted in both station batteries being disconnected for 1 h.

Three depressurizations occurred at Pilgrim 1 (NSIC 160497, 160559, 160926) because of stuck-open relief valves. Two of these occurred because of nitrogen system problems.

8.2 Initiating Event Frequencies and Function Failure Probabilities

Initiating event frequencies were estimated by dividing the effective number of nonrecoverable initiating events by the number of reactor-years under observation. Function failure probabilities were estimated by dividing the observed number of nonrecoverable failures by an estimated number of demands for the function.

Frequencies and failure probabilities estimated in this study using this method are listed in Table 8.1. These are homogenized values across the light-water-reactor population, reflecting events in the entire 1969–81 period.

A comparison of the number of initiators and function failures expected in 1980–81, based on the number observed in 1969–79, is provided in Sect. 5.1. This comparison indicated fewer PWR events in 1980–81 for all but one of the initiating event and function failure categories. A corresponding effect was not seen in BWRs.

Although the confidence bounds associated with the small number of events in each category are large [in fact in almost every instance any reasonable confidence bound on the observed number of events in 1980–81 overlaps the expected number of events (see Figs. 5.1 and 5.2)], the fact

Table 8.1. Initiating event frequencies and function failure probabilities determined from 1969–81 precursor data

	Point estimate[a]
BWR functions	
HPCI/RCIC failure	2.?E-3
Emergency power failure	2.2E-3
Automatic depressurization system failure	6.7E-3
Reactor scram failure	1.9E-4
Reactor isolation failure (large SLB)	2.3E-3
Long-term core cooling failure	1.0E-4
PWR functions	
AFW failure given reactor trip success	2.7E-4
HPI failure given AFW success	6.0E-4
Long-term core cooling failure	2.6E-4
Emergency power failure	3.7E-4
Steam generator isolation failure (large SLB)	6.4E-4
Concentrated boric acid addition failure given HPI success (large SLB)	8.3E-4
Initiators (value per reactor-year)	
BWR LOOP	1.9E-2
BWR LOCA	2.1E-2
PWR LOOP	2.8E-2
PWR LOCA	8.9E-3

[a]These estimates are based on amalgamated 1969–81 failure data and are average estimates across the reactor population. Plant-specific estimates can vary substantially from these values.

that the observed number of events in eight of the nine PWR categories is less than expected can be used to demonstrate with 95% confidence (using a χ-square test) a general reduction in PWR initiating event frequencies and demand failure probabilities observed in this study compared with those in the 1969–79 study.

A number of observations can be made concerning the frequency and probability estimates and the precursors contributing to them:

— The LOOP frequencies estimated in this study consider potential recovery within a period up to ~30 min. Because of this, the estimated frequencies are lower than LOOP frequencies estimated in other studies that do not include this consideration. The number of LOOPs observed in 1980–81 was less than that expected based on 1969–79 data for both PWRs and BWRs.

— The number of PWR small-break LOCA initiating events was consistent with that expected based on 1969–79 observations. The number of

BWR small-break LOCA events increased; however, three of these events occurred at one plant (Pilgrim 1). This experience may be atypical and may bias the frequency estimate high. BWR small-break LOCAs are dominated by stuck-open relief valves. PWR small-break LOCAs are dominated by seal failures (45%) and open relief valves (55%).

— Probability estimates for failure of BWR scram, long-term core cooling, and reactor isolation are based on one observation over the entire 1969—81 time period. The uncertainty in these estimates is therefore high. No failures were observed in 1969—81 for PWR scram, and uncertainty in the failure probability assumed [which was based on WASH-1400 (Ref. 1)] is also high.

— Common-mode or common-cause failures were involved in the majority of function failures observed in 1980—81. All 1980—81 failures on demand for BWR emergency power and long-term core cooling and for PWR auxiliary feedwater, high-pressure injection, steam generator isolation, and concentrated boric acid addition involved common-mode or common-cause elements. The single automatic depressurization system failure involved multiple causes, but all were valve related. Four failures were observed for HPCI/RCIC, but none appeared to be related to common-mode elements. The specific cause of the Browns Ferry failure to scram was not determined.

8.3 Human Error Associated with Precursors

During the survey of LERs as a part of this study, it was not uncommon to find LERs in which the event cause was presented as a component/equipment failure when, in reality, the event more accurately reflected human error. If human error either caused or substantively contributed to a precursor, based on this program's review of the event, this is identified under column "E" in Table 4.1 and associated tables. A review of precursors associated with human error for trends over the two study periods, and for different types of human error, is the subject of continuing work. One general observation, however, is that human error remains a substantial contributor (greater than 25% of precursors) to precursors to potential severe core damage.

8.4 Industry-Average Potential Severe Core Damage Frequency

The estimated industry-average frequency of potential severe core damage was developed in this study by summing the conditional probabilities associated with initiating events, dividing by the number of reactor-years in the observation period (87.6 for PWRs and 48.0 for BWRs), and adding an estimated contribution to account for unreported transients (primarily losses of feedwater). The method is described in Sect. 3.2. This frequency estimate should not be directly associated with the frequency of potential severe core damage at a particular plant because the frequencies and failure probabilities used in the calculations were based on data obtained across the entire light-water-reactor population and were applied to sequences that are generic in nature. Even though event trees were plant-tailored, failure probabilities may be higher or lower at a particular plant than the averages developed here. However, the frequency calculated is considered to be an average frequency across the population of light-water reactors.

The industry-average frequency estimate based on 1980–81 events is 1.6E-4/reactor-year without consideration of potential overcounting and 1.1E-4/reactor-year with potential overcounting removed. A revised estimate for the 1969–79 period, developed on the same basis as the 1980–81 estimate (i.e., revised recovery class numeric values, degraded function model, etc.) and with revisions to the 1969–79 data base to reflect comments on NUREG/CR-2497 (Ref. 2), is 2.3E-3/reactor-year (not corrected for potential overcounting). The 1980–81 frequency estimate is lower than the 1969–79 estimate because the conditional probabilities associated with the 1980–81 precursors are generally lower — a result of lower failure probabilities, fewer coupled-failure events of consequence, the availability of alternate mitigation paths (such as bleed and feed), and removal of dependencies (such as removal of ac dependency from at least one auxiliary feedwater train). However, considerable uncertainty, as well as the potential for overestimation and underestimation (as described in this report), is associated with both period estimates.

The frequency estimates developed in the ASP Program are mean estimates and are not directly comparable with median estimates developed in many PRAs. Also note that the uncertainty estimate developed in the study is associated with the mean frequency estimate, and thus the range indicated in the analysis does not necessarily bound the potential severe core damage frequencies for individual plants.

Because of the paucity of the ASP data base, it is difficult to draw definitive conclusions concerning the difference between the 1969–79 and 1980–81 estimates at this time. However, the following points should be recognized in an attempt to put the 1980–81 estimate in perspective:

- Branch failure probabilities used on the event trees were based on an amalgamated 1969–81 data set.
- When no failures were observed in the data for some functions, the failure probabilities used were based on other estimates.
- Except for their contribution to branch failure probabilities, events observed in the 1969–79 period were not considered in the 1980–81 period estimate.
- The number of reactor-years associated with the 1980–81 period is approximately one-third of that associated with the 1969–79 period (23% of the total 1969–81 period).

The impact of the first two points can be assessed by developing additional frequency estimates based only on 1980–81 events: (1) using assumed probabilities for functions with no observed failures and (2) using a failure probability of zero when no failures were identified (Table 8.2).*

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*In the ASP Program some function probabilities are developed independently of observed failures because of program methodology (such as the probability of feed and bleed failure). These probabilities were not changed in the above calculations.

Table 8.2. Alternate 1980–81 potential severe core damage frequency estimates[a]

Calculation description	Estimates (per reactor-year)		
	BWR	PWR	Combined BWR and PWR
Estimate based on 1980–81 precursors, 1980–81 failure data, and assumed probability estimates when no failures were observed	2.0E-4	1.9E-4	1.9E-4
Estimate based on 1980–81 precursors, 1980–81 failure data, and a failure probability of 0.0 when no failures were observed	2.0E-4	1.3E-4	1.5E-4
Estimate developed in this study based on 1980–81 precursors, amalgamated 1969–81 failure data, and assumed probability estimates when no failures were observed	5.9E-5	2.2E-4	1.6E-4

[a]See Table 5.6 for additional information.

As can be seen, the BWR and PWR combined estimates for 1980–81 do not vary substantially, although the separate frequency estimates for BWRs and PWRs do vary. Use of an amalgamated failure data base results in a higher frequency estimate for PWRs and a lower estimate for BWRs in 1980–81 than would be the case if only a 1980–81 failure data base were employed.

The impact of the relatively small number of reactor-years in 1980–81 and the exclusion of major 1969–79 events in developing the frequency estimate for 1980–81 is more difficult to assess. Three major events (the TMI-2 accident, the Browns Ferry fire, and the Rancho Seco NNI failure) were observed in 1969–79. If there were no reduction in the number of such events, slightly less than one event would be expected in 1980–81. The fact that no events of equivalent magnitude were observed is not necessarily a strong indication of a reduction in their number. (Although observation periods later than 1981 have not been addressed in detail in this study, the fact that no such events have occurred through 1983 provides additional substantiation that there may be a reduction.)

Substantial efforts have been expended for improvements following the TMI-2 accident, and initiating event frequencies, function failure probabilities, and precursor conditional probabilities have in general decreased compared with those in the 1969–79 period. Yet, it is doubtful that major events such as TMI-2, the Browns Ferry fire, or Rancho Seco NNI failure have been completely prevented from possible recurrence. Because of this, these events should still be considered to contribute something to the potential severe core damage frequency estimate. This contribution has not been explored to date but would tend to raise the 1980–81 estimate developed in this study.

Overall, based on the extensive "lessons learned" efforts and the demonstrated reduction in event probabilities and coupling in 1980–81, the industry-average potential severe core damage frequency is believed to have decreased from that in the 1969–79 period; how significant the reduction is cannot be conclusively determined.

8.5 Dominant Sequences

Dominant sequences among all sequences leading to potential severe core damage were identified for the 1980–81 precursors. These sequences are those that contribute significantly to the average frequency estimate.

The majority of sequences modeled in the program concern response to loss of main feedwater, loss of offsite power, small-break LOCA, or steam line break. However, certain events could not be easily modeled using event trees developed for the above four initiators. In these cases, unique event trees were developed to describe postulated sequences to potential severe core damage.

The major sequences contributing to the potential severe core damage estimate are listed in Table 5.10. Sequences associated with events not typically modeled in past PRAs contributed over 40% to the PWR estimate. Four PWR sequences that fall in this category include failure of the dc bus at Millstone, vital bus failure at Davis-Besse 1, loss of RCP cooling at St. Lucie 1, and a small-break LOCA due to an opened containment spray valve at Sequoyah 1. (These are also listed and described in Table 5.10.) Although this percentage may be the result of specific events seen in 1980–81, it is indicative of the many events at reactor plants that may not be adequately captured through the usual consideration of LOFW, LOOP, LOCA, and steam line break initiators.

Two recent papers, by Young and Asselin[3] and by Joksimovich, Frank, and Worledge,[4] describe dominant accident sequences developed in PRAs. Combinations of these sequences are compared with those identified in this program in Table 8.3. Because of differences among this study and those reported in the two papers, only certain categories of events could be compared. In addition, it is likely that work described in both papers used the same PRAs in part, and therefore the results should not be considered independent.

A review of the percentage contributions from each group of sequences in Table 8.3 indicates relative agreement in many cases, particularly with loss of main feedwater and small-break LOCA sequences. The relative contribution of sequences associated with loss of offsite power appears lower in the 1980–81 ASP Study than in the other papers. This may be a result of the fact that fewer LOOPs were observed in 1980–81 than were expected. In addition, potential RCP seal failure following a LOOP plus emergency power failure has not been considered to date in the ASP Study, which may contribute to higher estimates developed in some PRAs.

Table 8.3. Comparison of dominant potential severe core damage accident sequences

Sequence description	Contribution (%)		
	ASP study[a]	Young (Ref. 3)	Joksimovich (Ref. 4)
	PWR sequences		
Small LOCA with failure of recirculation	27 (47)	32[b]	32[c]
LOOP and subsequent failures	>6 (11)	20[d]	19[c]
Small LOCA with failure of safety injection	3 (5)	12[b]	16[c]
Loss of main feedwater and subsequent failures	>21 (37)	22[e]	14[f]
Uniquely modeled events	43	Not considered	Not considered
	BWR sequences		
Loss of main feedwater and subsequent failures	82	69[g]	61[h]
LOOP and subsequent failures	<1	11[d]	23[c]
Small LOCA with subsequent failures	14	15[i]	7[c]

[a] Number in parentheses represents percent contribution without consideration of unique sequences.

[b] Ref. 3, Table 2.

[c] Ref. 4, Fig. 1.

[d] Ref. 3, Table 2, station blackout.

[e] Ref. 3, Table 2, loss of feedwater, loss of feedwater with failure of feed and bleed, plus anticipated transients without scram.

[f] Ref. 4, Fig. 1, transients with failure of power conversion system and SI plus transients with failure of long-term decay heat removal.

[g] Ref. 3, Table 2, all transients except station blackout.

[h] Ref. 4, Fig. 1, transients with failure of long-term decay heat removal, transients with failures of power conversion system and SI, plus anticipated transients without scram.

[i] Ref. 3, Table 2, LOCA with failure of long-term heat removal plus LOCA with failure of injection.

The overall contribution of different initiators to potential severe core damage shown in Table 8.3 is similar for PWRs. Dominant sequences are split between small-break LOCAs and transients; for BWRs, transients strongly dominate.

8.6 Function Importance Calculations

Importance calculations to determine the risk achievement and risk reduction worths associated with the event tree functions are described in Chap. 5. These calculations highlight the impact of a given function in providing the current degree of protection against potential severe core damage and that function's potential for providing further protection if it were made completely reliable (see Tables 5.11 and 5.12).

If it is determined that the present risk level is satisfactory, then the functions having the highest risk achievement worths are of most concern in order to maintain that level. The risk achievement worths are thus of special interest in reliability assurance programs and inspection and enforcement activities. On the other hand, if one wishes to reduce the risk, the features having the highest risk reduction worths are of most interest. The risk reduction worths are of particular interest in plant-upgrading programs and backfitting activities. If further risk reduction is desired, attention should be focused on functions having high risk reduction worth without diverting attention from functions having high risk achievement worth. The two risk worth measures thus complement one another with regard to their characterization of what is important in providing protection against potential severe core damage.

The following functions were ranked high with respect to risk achievement worth:

- BWR long-term core cooling,
- PWR auxiliary feedwater (following reactor trip success),
- BWR scram,
- PWR high-pressure injection,
- PWR long-term core cooling.

The following functions were ranked high with respect to risk reduction worth:

- PWR bleed and feed,
- PWR auxiliary feedwater following emergency power failure,
- PWR long-term core cooling,
- PWR auxiliary feedwater (following reactor trip success),
- BWR long-term core cooling,
- BWR automatic depressurization (including manual actuation),
- BWR HPCI/RCIC.

It is interesting to note that three functions were ranked high with respect to both risk achievement and risk reduction worths — BWR long-term core cooling, PWR auxiliary feedwater, and PWR long-term core cooling.

8.7 Sensitivity Analyses

The scope of the initial ASP report, NUREG-2497 (Ref. 2), did not include a sensitivity assessment to estimate the impact of variation in individual failure probabilities or other program variables on the ranking or distribution of events. The present study does include a sensitivity evaluation of these parameters. This assessment, the results of which are discussed in Chap. 6 and Appendix E, consisted of varying each parameter individually to an estimated maximum and minimum value and then ascertaining the extent of any changes in event ranking and precursor conditional probability distribution. No attempt has been made as yet to systematically consider the impact of the simultaneous variation of several parameters in the sensitivity analysis.

In general, individual variation of most parameters did not significantly alter the ranking of high-, low-, or middle-range probability events, nor did it affect the general shape of the distribution. In all cases the mode of the distribution remained in the 10^{-4} to 10^{-5} conditional probability range, and the extremes did not move more than one order of magnitude beyond the mean (reference) distribution extremes. However, changes in ranking and probabilities for individual events were seen with variation of several variables. The largest impact was observed with variation of the numeric values associated with recovery classes R1, R2, R3, and plant tailoring class P3, with most events over the entire conditional probability range affected. Variation in other variables produced changes primarily limited to either high-, middle-, or low-probability events. These variables included the β factor; plant tailoring class P2; failure probabilities for BWR ADS, PWR emergency power, and HPI; and frequency estimates for PWR LOFW and PWR LOOP. Specific observations regarding the variation of parameters with notable impact are provided in Sect. 6.3.

Section 6.3 contains core damage frequency estimates based on changes in individual recovery class variables R1, R2, R3, plant tailoring class variable P3, and for the β factor, using the upper and lower limits for these variables (see Sect. 7.4 for the impact of variation of multiple recovery class variables). In no case did this result in variation of the frequency estimate by more than a factor of 2.5. Thus, although some impact on ranking and distribution for individual events was observed, there is no substantial impact on the frequency estimate based on variation of any individual parameter.

8.8 Uncertainty Analyses

A limited uncertainty analysis was performed using Taylor Series to approximate a variance on the average potential severe core damage frequency estimate. Variances were estimated for each variable used in the study. These variance estimates were then employed, in conjunction with partial derivatives of the function used to estimate the average potential severe core damage frequency as described in Chap. 6, to estimate the overall variance.

Variables for which no observational data existed, in particular the numeric values associated with the four recovery classes, were modeled using a truncated log-uniform distribution between defined end points. The log-uniform distribution is flat, on a logarithmic scale, between the end points. Although in actuality a distribution describing a recovery class would be expected to be peaked, no data were available to the program to indicate the actual distribution, and the use of the log-uniform distribution was considered acceptable as a means of estimating variance. Variance estimated using this distribution is high compared with other distributions, and thus its use is considered conservative.

The overall variance was estimated as 1.7E-8. This is equivalent to a standard deviation of 1.3E-4. Assumptions were then made concerning two potential distributions of the potential severe core damage frequency, the truncated log-uniform distribution and the lognormal distribution, and the following bounds were calculated:

Frequency estimate (per reactor-year)	Range based on distribution (per reactor-year)	
	Log uniform	Lognormal (5 to 95%)
1.6E-4*	2.5E-5 to 5.1E-4	8.3E-6 to 5.8E-4
1.1E-4†	4.7E-6 to 5.2E-4	1.5E-5 to 3.3E-4

In addition, Chebyshev's inequality was used to estimate a 95% upper bound independent of distribution assumptions. This estimate is 5.8E-4/reactor-year without consideration of potential overcounting and 5.2E-4/reactor year when potential overcounting is removed.

These values are estimates of the range on the industry-average frequency and do not bound plant-to-plant variations. Individual plant potential severe core damage frequency estimates would be expected to vary widely from the average calculated in this study.

The variance calculation is approximate and requires independence assumptions that are most likely not completely valid. As such, the variance estimate should be considered an indication of uncertainty associated with the estimated potential severe core damage frequency. In addition, the ranges computed, based on the use of assumed distributions, should not be associated directly with classical confidence bounds.

References

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

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*Without removal of potential overcounting.

†With potential overcounting removed.

2. 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 2), Oak Ridge National Laboratory, June 1, 1982.

3. J. Young and S. Asselin, *Perceptions of LWR Risk for Decision Making*, presented at the Water Reactor Safety Research Meeting, Gaithersburg, Maryland, Oct. 28, 1983.

4. V. Joksimovich, M. V. Frank, and D. R. Worledge, *Dominant Accident Sequences Derived from Review of Five PRA Studies*, presented at the American Nuclear Society Meeting on Anticipated and Abnormal Plant Transients in Light-Water Reactors, Jackson, Wyoming, Sept. 26–29, 1983.

9. CONCLUSIONS

This report is the second of two reports evaluating industry performance over two sequential periods of time (1969—79 and 1980—81) with respect to potential severe core damage accident precursors. It is not the intent of this report to evaluate differences in plant or industry performance between these two time periods; that will be the scope of future work. However, certain limited and tentative conclusions can be drawn. In view of the detailed discussions on the report findings in Chap. 8, this chapter only highlights some of the more significant conclusions from the work.

Approximately the same number of precursors per reactor-year were seen in 1980—81 as in 1969—79, but the precursors are not considered to be as significant in terms of reactor safety. This appears to be the result of improvements in system reliability, the availability of alternate features that can provide further protection against potential severe core damage, and a decrease in the degree of coupling observed in the precursors.

Most precursors involved only failures of a single function or an initiating event; the number of precursors that included multiple function failures or initiating events combined with a function failure was small. Because of this, the development of conditional probabilities associated with the precursors is dependent on probabilistic techniques for estimation of the likelihood of subsequent multiple failures. Results associated with the conditional probability calculations, such as the frequency of potential severe core damage, dominant sequences, and function importance, are more uncertain than the estimated average initiating event frequencies and function failure probabilities.

Precursors involving coupled failures were still observed; those were primarily associated with electrical faults. Failures in continuously operating cooling water systems were also observed in 1980—81. These failures were not observed to the same extent in the 1969—79 period. It is not clear whether this is due to increased failures in these systems or to increased reporting emphasis, but these failures are of concern because of their potential impact on a plant.

In 1980—81 the observed number of PWR initiating events and function failures was less than the expected number (based on 1969—79 data) in almost all cases. For a particular function or initiator this result is probably not significant due to the large variance of the estimates; however, the systematic effect over all the items is believed to be a demonstration of improved performance. Boiling-water-reactor initiating events and function failures do not show this same trend.

Importance analyses were performed to identify those functions providing the greatest present protection against potential severe core damage. These functions are BWR long-term core cooling, PWR auxiliary feedwater, BWR scram, PWR high-pressure injection, and PWR long-term core cooling. From the standpoint of additional risk reduction, the following functions were determined to be most significant: feed and bleed for PWRs and long-term core cooling for BWRs.

The estimated industry-average potential severe core damage frequency based on the 1980—81 precursors (1.6E-4/reactor-year) decreased from the estimate for 1969–79 precursors by over an order of magnitude. This is a result of the decreased conditional probabilities associated with the precursors, as discussed above. Considerable uncertainty is associated with this average estimate; however, the observed decrease in the core damage frequency for the 1980–81 period is indicative of a downward trend.

The dominant potential severe core damage sequences identified in the 1980–81 precursors were generally consistent with those identified in PRAs, although some unique failure modes and system interactions were observed. The precursor sequences associated with these events could not be modeled using the standardized event trees developed in this study (event trees for loss of main feedwater, loss of offsite power, small-break LOCA, and steam line break). These events include the electric bus failures at Millstone 2 and Davis-Besse 1, the loss of RCP cooling at St. Lucie 1, and the inadvertently opened containment spray valve at Sequoyah 1 (see Table 5.10); they contributed approximately 40% to the PWR potential severe core damage frequency estimate. Dominant sequences were split between those associated with small-break LOCAs and transients for PWRs; for BWRs dominant sequences were associated predominantly with transients.

The above highlights indicate the more significant results of the study. However, the body of the report itself, the precursor event documentation in Appendix B, and the extended discussions in Chaps. 7 and 8 contain additional insights and findings and serve to place the comments included in this chapter in perspective.

Appendix A

STANDARDIZED EVENT TREES

Standardized functional event trees were constructed to describe the mitigation sequences for four initiating events used in the study: loss of main feedwater, loss of offsite power, small loss-of-coolant accident, and steam line break. These standard sequences were used with the majority of events selected as precursors. Certain events could not be described using the standardized trees presented in this appendix. In such cases, unique event trees were developed to describe the sequences of interest. The four standard sequences are considered separately for PWRs and BWRs. As discussed in the main report, the first two events are considered to be the most likely off-normal events of concern and the latter two represent bounding events for many of the safety-related systems in a reactor plant.

This appendix (1) describes the mitigation sequences utilized for the four initiating events, (2) identifies the combinations of functions required for the successful mitigation of each initiator, and (3) describes the criteria for success of each function in general terms.

The event tree associated with each mitigation sequence is constructed with function success as the upper branch and function failure as the lower branch. Each sequence path is read from left to right. As an example, a point in a sequence of events involving a PWR loss of main feedwater, a successful reactor trip, and a failure of auxiliary feedwater and secondary heat removal would be located at point A on the event tree shown in Fig. A.1.

A.1 PWR Loss of Main Feedwater

A functional event tree was constructed to define a representative response of PWRs to a loss of main feedwater in terms of the success or failure of critical safety functions. The event tree shown in Fig. A.2 includes the normal mitigation sequence for a loss of main feedwater: reactor trip and auxiliary feedwater initiation. The tree also includes (1) the use of high-pressure injection to prevent severe core damage if auxiliary feedwater is unavailable and (2) mitigation sequences for a failure to trip and for a relief valve that does not reseat following a lift after a loss of main feedwater. The event tree functions and the sequences leading to potential severe core damage follow.

1. Initiating event (loss of main feedwater). The initiating event for the tree is a loss of main feedwater to the extent that reactor trip is required.

2. Reactor trip. Reactor trip success is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition for the short term (while the reactor is maintained in a hot shutdown condition).

3. Auxiliary feedwater and secondary heat removal. Success requirements for this function depend on whether or not reactor trip has

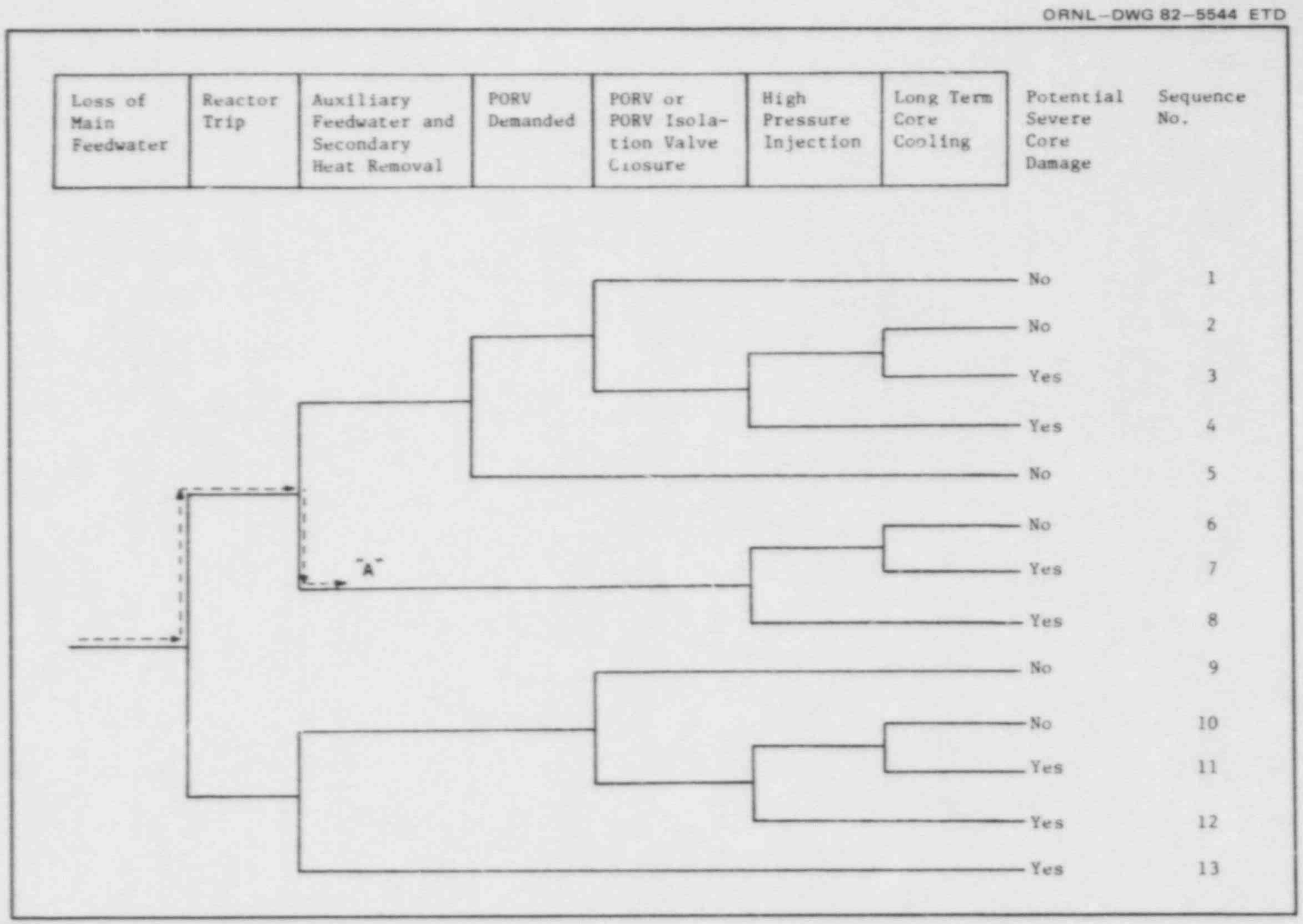

Fig. A.1. PWR standardized event tree for loss of main feedwater with successful reactor trip and failed auxiliary feedwater and secondary heat removal.

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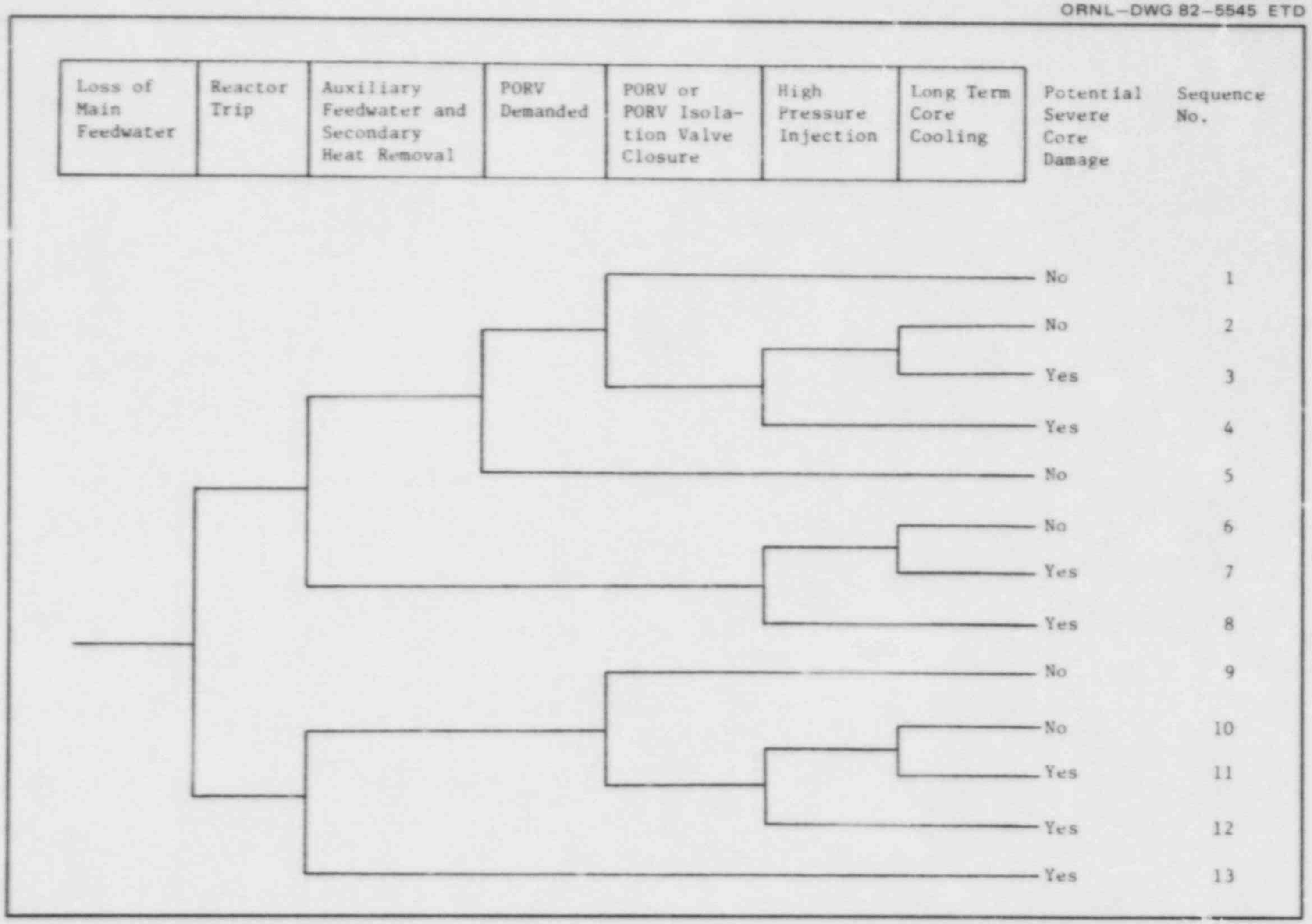

Fig. A.2. Standardized event tree for PWR loss of main feedwater.

been successful. In either case, sufficient auxiliary feedwater must be provided to remove the heat still being generated in the reactor coolant system via the steam generators and secondary side relief valves, atmospheric dump valves, and turbine bypass valves.

If reactor trip has been successful, auxiliary feedwater success usually requires flow to one or more steam generators from one train of a redundant auxiliary feedwater system over a period of time, perhaps 12–24 h.

If reactor trip has not been successful, then rapid injection of auxiliary feedwater may prevent excessive RCS pressures at some plants. The detailed requirements for successful auxiliary feedwater initiation following a failure to trip are still under discussion between the NRC and the reactor vendors, as are other plant changes that may be required for such an event. Because of this discussion of requirements, it is not possible in all cases to accurately distinguish between a failed and a degraded auxiliary feedwater system when reactor trip is unsuccessful. The event tree for a loss of main feedwater has been constructed to include the use of auxiliary feedwater following failure to trip to prevent core damage.

4. Pilot-operated relief valve demanded. For the case where both reactor trip and auxiliary feedwater initiation have been successful following a loss of main feedwater, the pressurizer pilot-operated relief valve (or valves) may or may not lift, depending on the peak pressurizer pressure following the transient. The upper branch for this function merely indicates that the valve or valves were demanded because of the peak pressure during the transient. It is assumed, because of the multiplicity of relief and safety valves, that a sufficient number will open if required for this case, as the peak pressures are not particularly high.

The lower branch indicates that the pressurizer pressure was not sufficiently high to require opening of a relief valve. For the cases in which auxiliary feedwater fails following reactor trip success or reactor trip fails, at least one pressurizer relief valve is assumed to be demanded.

5. PORV or PORV isolation valve closure. Success for this function requires the closure of any open relief valve once pressurizer pressure has decreased below the relief valve set point. In the case of pilot-operated relief valves, an isolation valve is usually provided adjacent to the valve to permit manual termination of relief valve blowdown if the valve sticks open.

6. High-pressure injection. For the case in which reactor trip is successful following a loss of main feedwater but the auxiliary feedwater and secondary heat removal function is unsuccessful, high-pressure injection may be used on some plants to provide adequate core cooling. In such a case, borated water is injected via the high-pressure injection system and discharged to the containment via the pressurizer relief valves. Depending on the design of a plant's high-pressure injection system, a PORV may have to be manually opened at a pressure considerably below its normal set point.

Success for this function is very dependent on plant design but generally requires the introduction of sufficient amounts of borated water into the reactor coolant system to remove decay heat and prevent severe core damage.

7. Long-term core cooling. Success of the auxiliary feedwater and secondary heat removal function was considered to satisfy the need for core cooling for the purposes of this study. If auxiliary feedwater and secondary heat removal was not completely successful, either because the function failed or because a PORV remained open, then the requirement for long-term core cooling (following injection of available borated water supplies) was considered satisfied by using high-pressure injection in the recirculation mode.

Because of the long-term nature of this function and the reduced heat load during its operation, the possibility of component repair and the use of alternate methods becomes much more likely. As such, a definition of function success in terms of specific system and component operability, except during the initial period, is not particularly useful. For this study, HPI recirculation was considered to be required during the initial phases of long-term cooling.

8. Loss of main feedwater sequences resulting in potential severe core damage. Seven of the thirteen sequences identified for a loss of main feedwater are considered to result in potential severe core damage. These sequences are

- failure of long-term core cooling after a PORV is demanded and fails to close (sequences 3 and 11);
- failure of high-pressure injection following failure of a demanded PORV or its isolation valve to close (sequences 4 and 12);
- failure of long-term core cooling following high-pressure injection success, auxiliary feedwater failure, and reactor trip success (sequence 7);
- failure of high-pressure injection following auxiliary feedwater failure and reactor trip success (sequence 8);
- failure of auxiliary feedwater following failure of reactor trip (sequence 13).

A.2 PWR Loss of Offsite Power

A functional event tree was constructed to define a representative response of PWRs to a loss of offsite power in terms of success or failure of critical safety functions. A loss of offsite power without turbine runback will result in reactor trip due to unavailability of power to the control rod drive mechanisms and a loss of main feedwater due to unavailability of power to components in the condensate or condenser cooling systems.

The event tree shown in Fig. A.3 includes the normal mitigation sequence for a loss of offsite power: turbine runback, if available and successful, and, if not, emergency power initiation and the mitigation sequence discussed previously for a loss of main feedwater. The event tree functions and the sequences leading to potential severe core damage follow.

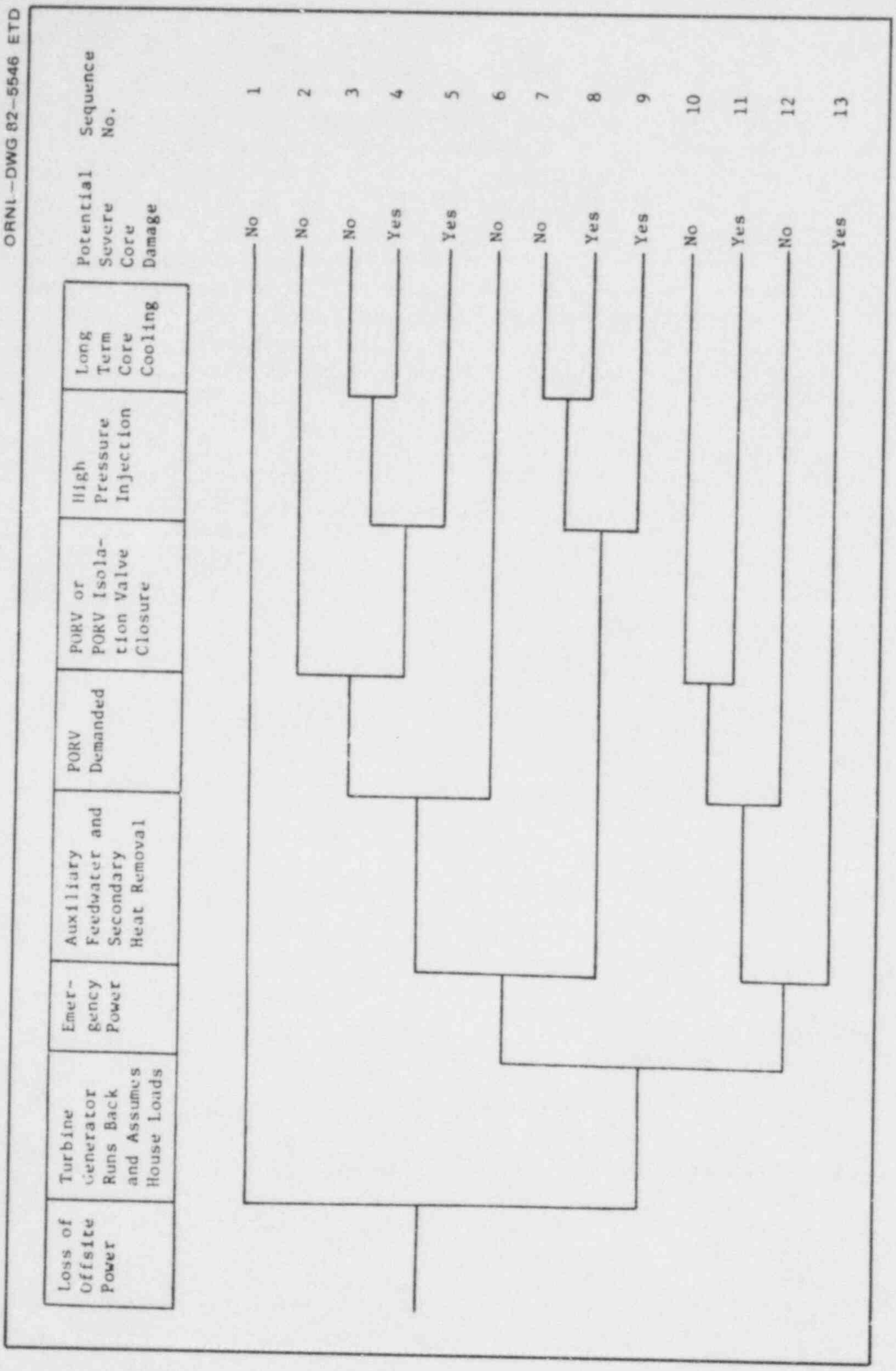

Fig. A.3. Standardized event tree for PWR loss of offsite power.

1. Initiating event (loss of offsite power). The initiating event for the tree is a grid or switchyard disturbance to the extent that the generator must be separated from the grid and all offsite power sources are unavailable to plant equipment.

2. Turbine generator runs back and assumes house loads. On certain plants the capability of a runback from full power to house loads (~10% power) is provided in the event the generator must be separated from the grid. Success for this function would be such a successful runback and continued maintenance of in-plant electric loads from the unit generator.

3. Emergency power. If a turbine runback is not successful, then electric power would be lost to all loads not backed by battery power when the unit generator trips. Reactor trip would occur at this time (because of the loss of power to the CRDMs) if it did not occur earlier due to conditions in the reactor coolant system. When power is lost, diesel generators are automatically started to provide power to the plant safety-related loads. Emergency power success requires the starting and loading of a sufficient number of diesel generators to support safety-related loads in systems required to mitigate the transient and maintain the plant in a safe shutdown condition.

4. Auxiliary feedwater and secondary heat removal. Success requirements for this function are equivalent to those following a loss of main feedwater with a subsequent reactor trip. However, since specific auxiliary feedwater systems may contain either turbine-driven or motor-driven pumps, the capability of the system to meet its success requirements will in many cases also be dependent on the success or failure of the emergency power function.

5. Pilot-operated relief valve demanded. The upper and lower branches for this function are similar to those following a loss of main feedwater with a subsequent reactor trip.

6. PORV or PORV isolation valve closure. The success requirements for this function are similar to those following a loss of main feedwater.

7. High-pressure injection. The success requirements for this function are similar to those following a loss of main feedwater. However, because all high-pressure injection systems use motor-driven pumps, the capability of the high-pressure injection function to meet its success requirements is dependent on the success of the emergency power function.

8. Long-term core cooling. The success requirements for this function are similar to those following a loss of main feedwater. Because all systems used for long-term cooling use motor-driven pumps, function success is dependent on the success of the emergency power function.

9. Loss of offsite power sequences resulting in potential severe core damage. Six of thirteen sequences identified for a loss of offsite power are considered to result in potential severe core damage. These sequences are

- failure of long-term core cooling after a PORV is demanded and fails to close (sequences 4 and 11),
- failure of high-pressure injection following failure of a demanded PORV or its isolation valve to close (sequence 5),

- failure of long-term core cooling following high-pressure injection success and auxiliary feedwater failure (sequence 8),
- failure of high-pressure injection following auxiliary feedwater failure (sequence 9),
- failure of auxiliary feedwater and emergency power (sequence 13).

A.3 PWR Small Loss-of-Coolant Accident

A functional event tree was constructed to define a representative response of PWRs to a small LOCA in terms of success or failure of critical safety functions. The LOCA chosen for consideration is one that would require a reactor trip and continued high-pressure injection for core protection. Because of the limited amount of borated water available for initial injection, the mitigation sequence also includes the capability of providing borated water from the containment sump (usually via the low-pressure recirculation system) to the high-pressure injection pump suctions.

The event tree is shown in Fig. A.4, and functions and sequences leading to potential severe core damage follow.

1. Initiating event (small loss-of-coolant accident). As described previously, the initiating event for the tree is a small LOCA that requires reactor trip and high-pressure injection for core protection.

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

3. Auxiliary feedwater and secondary heat removal. If reactor trip is not successful, then rapid injection of auxiliary feedwater may prevent core damage, as in the case of the loss of main feedwater. To an even greater extent than was the case with the loss-of-main-feedwater transient, the detailed requirements for auxiliary feedwater success following an LOCA and a failure to trip have not been determined. Because of this, it is not always possible to distinguish between a failed and a degraded auxiliary feedwater system for the case in which reactor trip has not been successful.

Certain small breaks may require auxiliary feedwater even with reactor trip success, particularly if high-pressure injection is only minimally successful. This has not been considered in the representative event tree.

4. High-pressure injection. Success for this function requires adequate injection of borated water to prevent excessive core temperatures and consequent damage. The adequacy of redundant portions of the high-pressure injection system and its associated support systems in performing this function is confirmed in each plant's safety analyses.

5. Low-pressure recirculation and LPR/HPI cross-connect. The HPI system takes suction from a tank of borated water during the initial stages of the LOCA mitigation sequence. This water is subsequently spilled from the break and collects on the containment floor and in the containment sump. Success for this function requires delivery of sump water to the suctions of the high-pressure pumps to provide continued injection of borated water. This is usually provided via the low-pressure

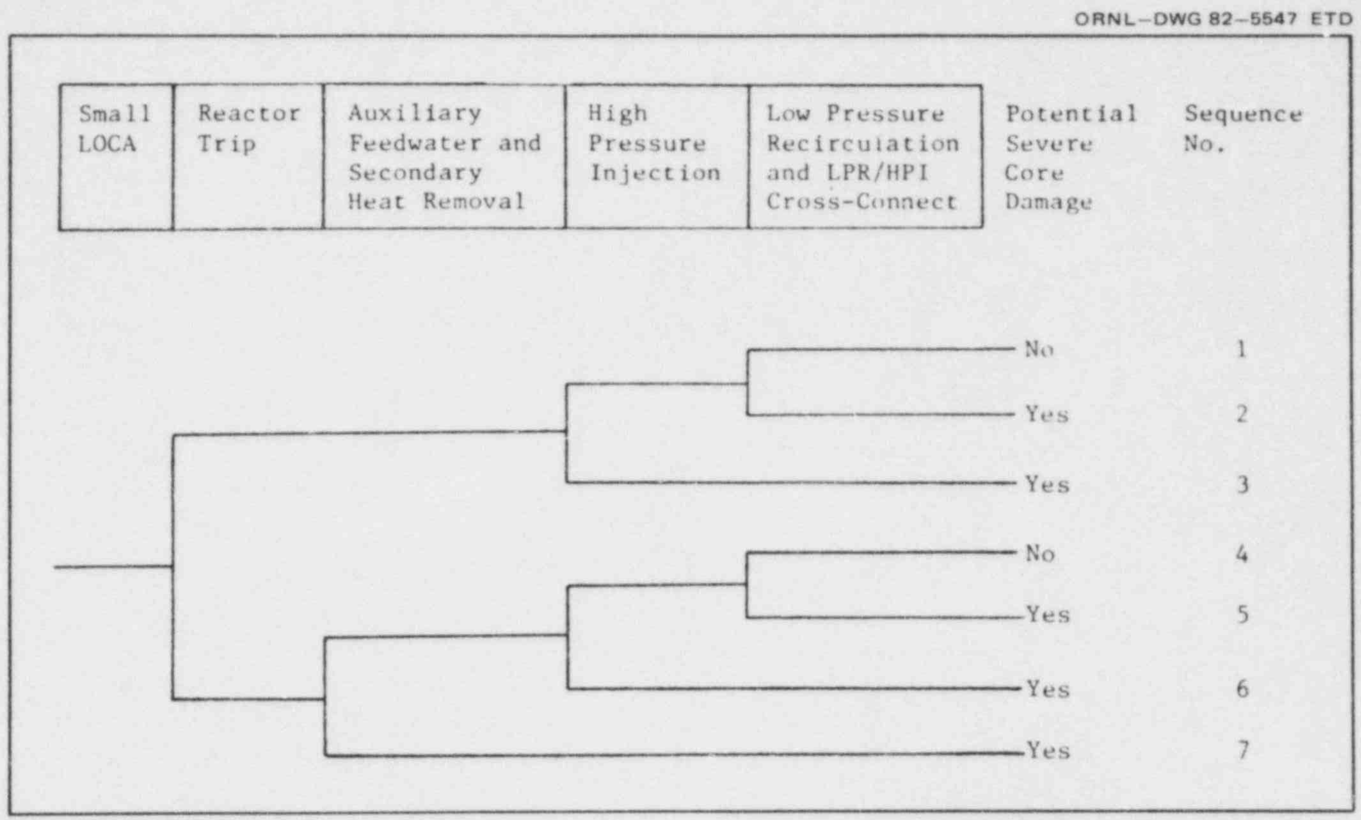

Fig. A.4. Standardized event tree for PWR small loss-of-coolant accident.

injection system realigned to take suction from the containment sump. In this manner, continued core cooling can be achieved with a limited initial volume of borated water. Because of the long-term nature of this function, the possibility of component repair (at least for support systems) and the use of alternate methods becomes more likely, and a rigid definition of function success in terms of specific system operability is difficult.

6. Small LOCA sequences resulting in potential severe core damage. Five of the seven sequences identified for a small LOCA are considered to result in potential severe core damage. These sequences are

- failure of low-pressure recirculation and LPR/HPI cross-connect (sequences 2 and 5),
- failure of high-pressure injection (sequences 3 and 6),
- failure of auxiliary feedwater following failure to trip (sequence 7).

A.4 PWR Steam Line Break

A functional event tree was constructed to define a representative response of PWRs to a steam line break in terms of success or failure of critical safety functions. The event tree shown in Fig. A.5 includes the normal mitigation sequence for a steam line break: reactor trip, steam generator isolation, use of auxiliary feedwater for reactor cooling following isolation of the break, and high-pressure injection of borated water for additional shutdown margin if required. The tree also includes a mitigation sequence for a primary relief valve that does not reseat following a lift due to continued high-pressure injection. Event tree functions and the sequences leading to potential severe core damage follow.

1. Initiating event (steam line break). The initiating event for the tree is a steam line break of the extent that reactor trip is required. The break in question is usually larger than one stuck-open relief or dump valve and may require isolation of the affected steam generator.

2. Reactor trip. Reactor trip success is defined as the rapid insertion of sufficient control rods to reduce core power to a level at which core damage will not occur and which can be reduced to subcritical, if not already subcritical, through the subsequent high-pressure injection of borated water.

Because of the excessive heat removal associated with a steam line break, failure to trip will result in excessive power levels and thus in potential severe core damage.

3. Required steam generator isolation. Depending on the size and location of the break, steam generator isolation may be required to minimize secondary side blowdown. Steam generator isolation success requires the termination of feedwater to the affected steam generator and isolation of unaffected steam generators from the steam line break. Failure to isolate the affected steam generator after a steam line break may result in excessive cooldown and consequent potential severe core damage due to overpower, even if trip occurs, depending on the size of the break

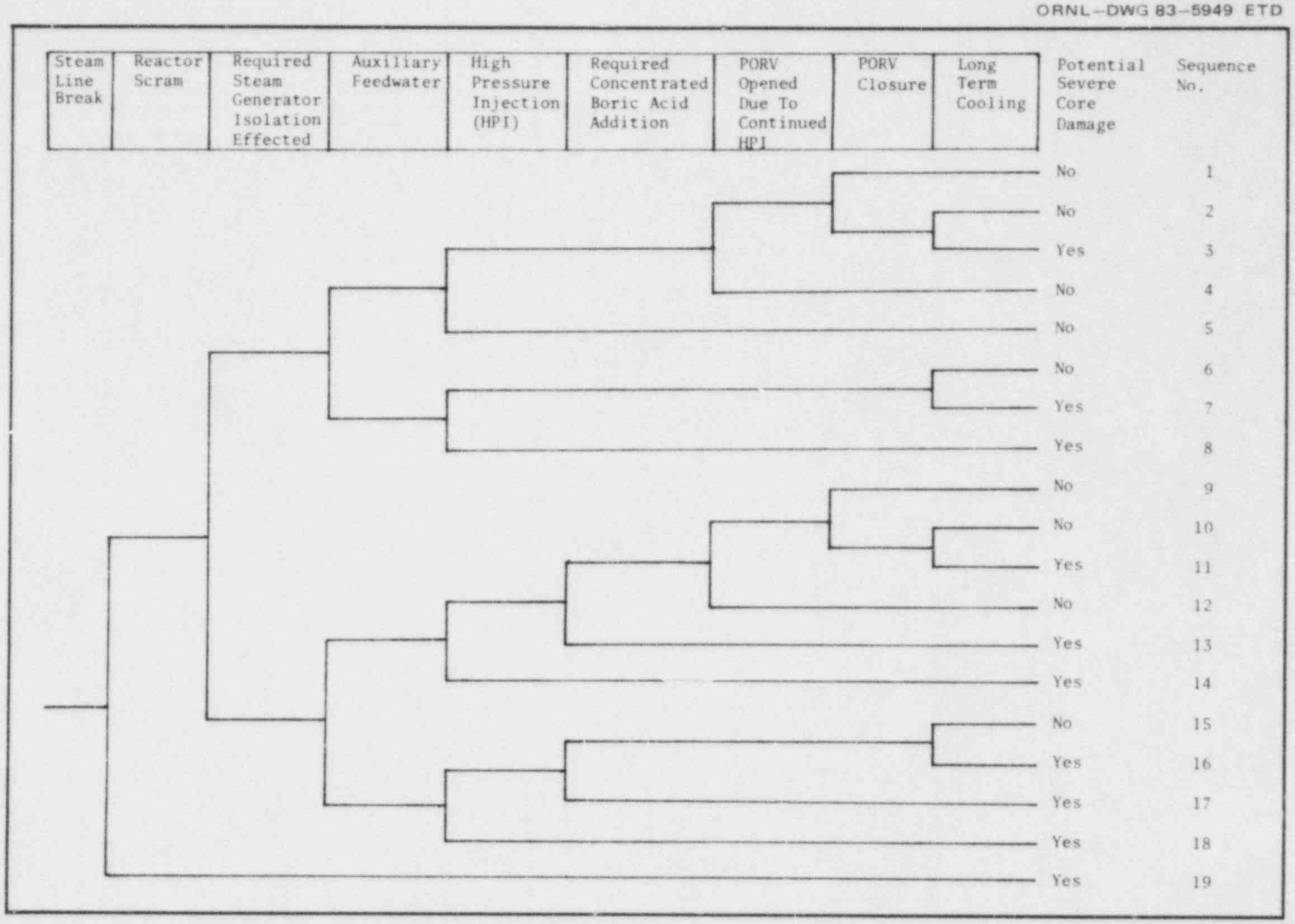

Fig. A.5. Standardized event tree for PWR steam line break.

and the specific plant. Typically steam generator isolation is only required following a large steam line break.

4. Auxiliary feedwater and secondary heat removal. After reactor trip and steam generator isolation, injection of auxiliary feedwater into an operable steam generator is required for continued core cooling. Success requirements for this function are similar to those following a loss of main feedwater with a subsequent reactor trip, with the additional requirement that auxiliary feedwater be provided only to operable steam generators.

5. High-pressure injection. Reduction in RCS pressure following a steam line break will typically initiate high-pressure injection. In the event that auxiliary feedwater provides RCS cooling and steam generator isolation is not required or if required is successful, high-pressure injection is not required for continued core cooling. In the event auxiliary feedwater fails to function following steam generator isolation, the high-pressure injection function may provide continued core cooling in a manner similar to that following a loss of feedwater. In this case, the requirements for high-pressure injection success would include those for a loss of main feedwater with subsequent reactor trip success and auxiliary feedwater failure.

6. Required addition of concentrated boric acid. Success for this function requires injection of sufficient borated water to place the core in a subcritical condition if it remains at power following trip. This could be the case at some plants if more than one steam generator blows down during a large steam line break. Certain plants require the injection of high concentration boric acid in lieu of RWST water under this circumstance.

7. Pilot-operated relief valve opened due to continued HPI. High-pressure injection is automatically initiated following detection of a steam line break. In many cases for plants with HPI capability above the pressurizer relief valve set points, continued HPI system operation is expected to result in a PORV lift. The upper branch for this function indicates that such a valve opened because of pressurizer pressure during mitigation.

The lower branch for this function indicates that high-pressure injection was secured prior to PORV lift. For the case in which high-pressure-injection success provides continued core cooling following loss of auxiliary feedwater, at least one pressurizer relief valve is assumed to open or be opened.

8. PORV or PORV isolation valve closure. Success requirements for this function are similar to those following a loss of main feedwater.

9. Long-term core cooling. The success requirements for this function are similar to those following a loss of main feedwater.

10. Steam line break sequences resulting in potential severe core damage. Ten of nineteen sequences identified for a steam line break are considered to result in potential severe core damage. These sequences are

- failure of long-term core cooling if a PORV is demanded and fails to close (sequences 3 and 11);
- failure of long-term core cooling following a failure of auxiliary feedwater (sequences 7 and 16);

- failure of high-pressure injection following a failure of auxiliary feedwater (sequences 8 and 18);
- failure of high-pressure injection following required steam generator isolation failure and AFW success (sequence 14);
- failure to inject concentrated boric acid when required following steam generator isolation failure (sequences 13 and 17);
- failure to trip (sequence 19).

A.5 BWR Loss of Main Feedwater

A functional event tree was constructed to identify a representative response of BWRs to a loss of main feedwater in terms of success or failure of critical safety functions. Figure A.6 illustrates the normal mitigation sequence for this event (reactor scram and RCIC initiation) as well as alternative methods of providing adequate core cooling (high-pressure coolant injection and the low-pressure systems, which rely on automatic depressurization for initiation). Event tree functions and the sequences leading to potential severe core damage are discussed below.

1. Initiating event (loss of main feedwater). Any trip of both reactor feed pumps, except trips induced by loss of offsite power, is considered an initiating event for the tree.

2. Reactor scram. Once the main feedwater pumps trip, flow to the reactor will decay to zero in about 5 s. The reduction in main feedwater is sensed, and the recirculation pumps are automatically throttled back, resulting in a negative insertion of reactivity. Without makeup water, the water level will drop in the reactor as coolant is boiled off. When the low-level set point is reached, a scram is initiated. If scram fails, the operator has two shutdown options available to reduce the thermal output to the decay heat load:

- Control rods not successfully inserted during scram can be manually actuated. All but two adjacent control rods must be inserted within 30 min of the scram signal.
- The standby liquid control system can be initiated. However, the SBLC system has some limitations which are discussed below.

If poisoning of the neutron chain reaction is not adequate to cause reactor shutdown within a given time frame, then the potential for severe core damage exists.

3. SBLC initiated. The SBLC system must be initiated within 10 min after the failure to scram, and the core must be subcritical within 38 min after the failure to scram. Also, the reactor recirculation pumps must be tripped in conjunction with the successful operation of the standby liquid control system.

Successful shutdown with SBLC alone is limited because only the high-pressure coolant injection systems (HPCI and RCIC) are compatible with SBLC for removing decay heat while maintaining the reactor subcritical. Actuation of the normal backup for HPCI/RCIC, depressurization followed by low-pressure coolant injection, will result in boron dilution, removal of boron from the system, and a potentially significant

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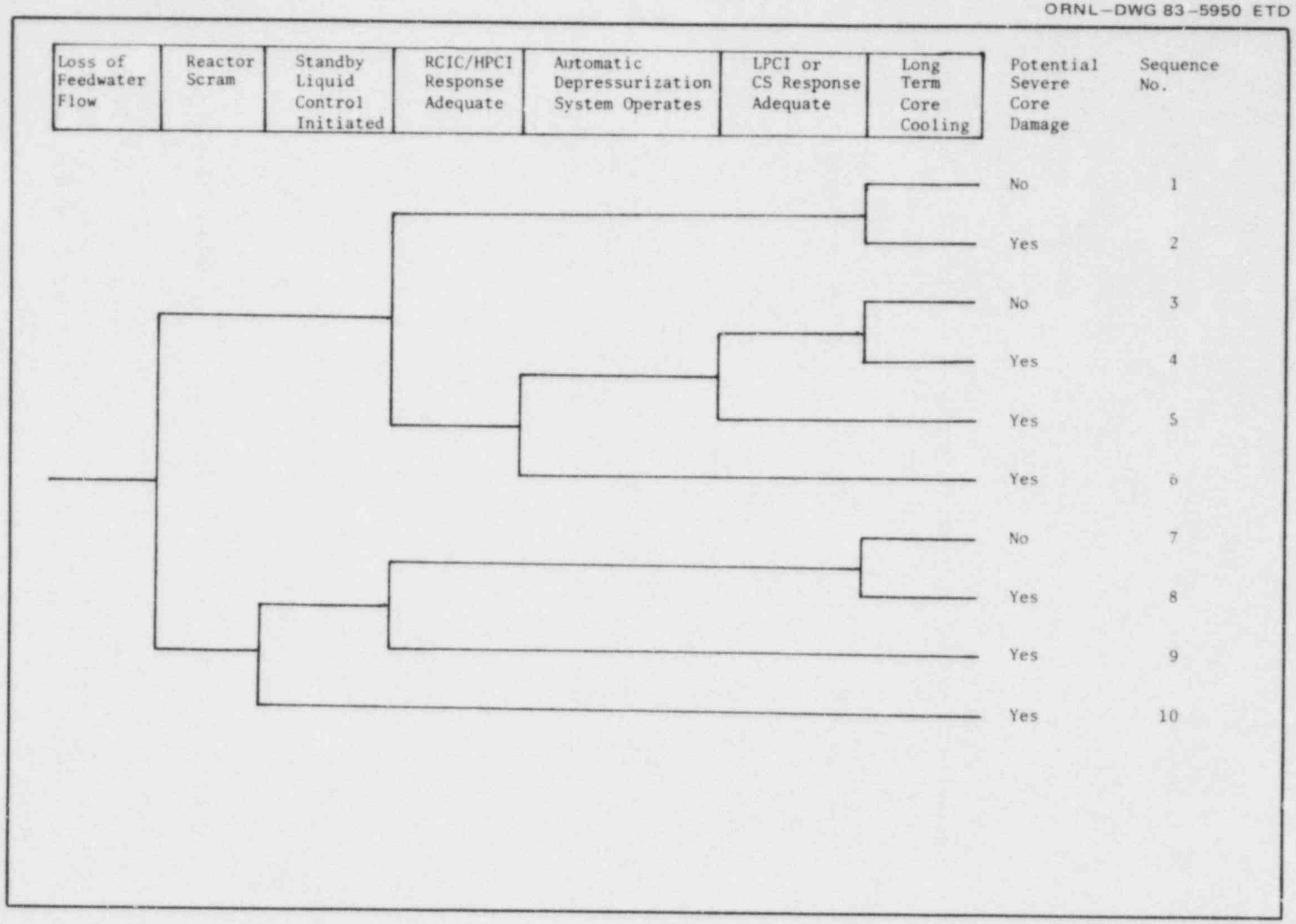

Fig. A.6. Standardized event tree for BWR loss of feedwater.

return to power. The high-volume injection will dilute the boron concentration, and the rapid depressurization will be accompanied by increased boiling and entrainment of boron with coolant relieved from the system. The drop in system temperature from the coolant injection and depressurization, combined with reduction in boron concentration, would thus promote an insertion of positive reactivity. Therefore, if HPCI/RCIC is not available, SBLC alone is not adequate to maintain the reactor at subcriticality, and the potential for severe core damage exists.

4. RCIC/HPCI response adequate. For a loss of feedwater transient, initiation and operation of reactor core isolation cooling satisfies the reactor makekup requirements. However, both the RCIC and HPCI initiate when the reactor coolant inventory drops to the low-low-level set point. The operator must secure HPCI when pressure and water level control are restored because failure to secure HPCI could result in tripping both RCIC and HPCI turbine pumps on high water level. RCIC must be operated until the residual heat removal system can be placed in service. In the event of a failure of RCIC, HPCI may be used to provide reactor makeup water.

Following reactor trip, reactor pressure will increase until the lowest set point of the electromatic relief valves is reached. These valves will open and close periodically to maintain reactor pressure control. RCIC has the capacity to maintain water level control if one relief valve fails to reset. If additional valves remain open, HPCI operation is necessary.

Some older BWRs are equipped with isolation condensers instead of RCIC. The ICs use natural circulation; their success depends on the opening of dc-powered valves, a sufficient flow path from the annulus to the core region once the IC is initiated, and sufficient coolant on the shell side of the IC heat exchanger to ensure adequate heat transfer. The majority of BWRs equipped with ICs also rely on the feedwater pumps for high-pressure injection. In this study, the term HPCI/RCIC is used where high-pressure injection systems for plants that utilize ICs are modeled.

5. Automatic depressurization system operates. If RCIC and HPCI fail to provide adequate flow to maintain reactor level control and SBLC is not employed to maintain the reactor subcriticality, the low-pressure high-capacity systems can be used to maintain level control.* However, for these systems to operate, reactor pressure must be reduced to the operating range of the low-pressure injection systems. This is accomplished by the automatic depressurization system, which can be manually or automatically initiated. Automatic initiation success requires high dry-well pressure, low-low-low reactor water level, and the availability of one train in either of the low-pressure high-capacity systems. Given this state, the ADS will autoinitiate after a time delay.

6. LPCI or CS response adequate. Given the need and successful operation of ADS, the low-pressure coolant injection or core spray can be used to ensure adequate core cooling.* Success requires operation of one train of either of the systems.

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*As discussed under SBLC system, if SBLC is employed, ADS/LPCI/CS must not be actuated, because the rapid depressurization combined with the coolant influx will cause a reduction in core boron concentration and a potentially significant return to power.

7. Long-term core cooling. Success of this function requires that heat transfer to the environment commence within about 1 d. Successful delivery of cooling water to the RHR heat exchanger satisfies this function. However, because of the long-term nature of this function, the possibility of component repair and the use of alternate methods becomes more likely.

Normally the RHR system provides for long-term core cooling. However, as discussed previously, its failure may not, under all circumstances, constitute a failure of the long-term cooling function. For example, during the Browns Ferry fire a combination of safety valves, control rod drive pumps, and the condensate and condensate booster pumps were used for several hours to maintain core cooling until normal means could be restored. Cross ties frequently exist in multiple unit plants and allow operable systems in one unit to provide for the failed function in another unit.

8. Loss-of-main-feedwater sequences resulting in potential severe core damage. Seven of the ten sequences identified are considered to result in potential severe core damage. These sequences are

- failure of long-term core cooling given success of the emergency core-cooling systems (RCIC/HPCI/CS/LPCI) (sequences 2, 4, and 8);
- complete failure of emergency core cooling systems (RCIC/HPCI/CS/LPCI) (sequence 5);
- failure of high-pressure coolant injection and automatic depressurization (sequence 6);
- failure of high-pressure coolant injection given scram failure (sequence 9);
- failure to shut down the reactor in a sufficient time period (sequence 10).

A.6 BWR Loss of Offsite Power

The functional event tree for a loss of offsite power is illustrated in Fig. A.7. The normal mitigating sequence is reactor scram and initiation and operation of the RCIC. Other available paths are reactor scram and HPCI; reactor scram, diesel power supply, and ADS/LPCI/CS; or reactor shutdown with diesel power, boron injection (SBLC), and HPCI or RCIC.

A LOOP condition would result in a generator load rejection that trips the turbine control valve and results in a scram. Event tree functions and the sequences leading to potential severe core damage follow.

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

2. Reactor scram. As previously mentioned, a scram signal is generated given a load rejection. A successful scram is taken to be a rapid insertion of control rods with no more than two adjacent control rods failing to insert. The scram can be automatically or manually initiated.

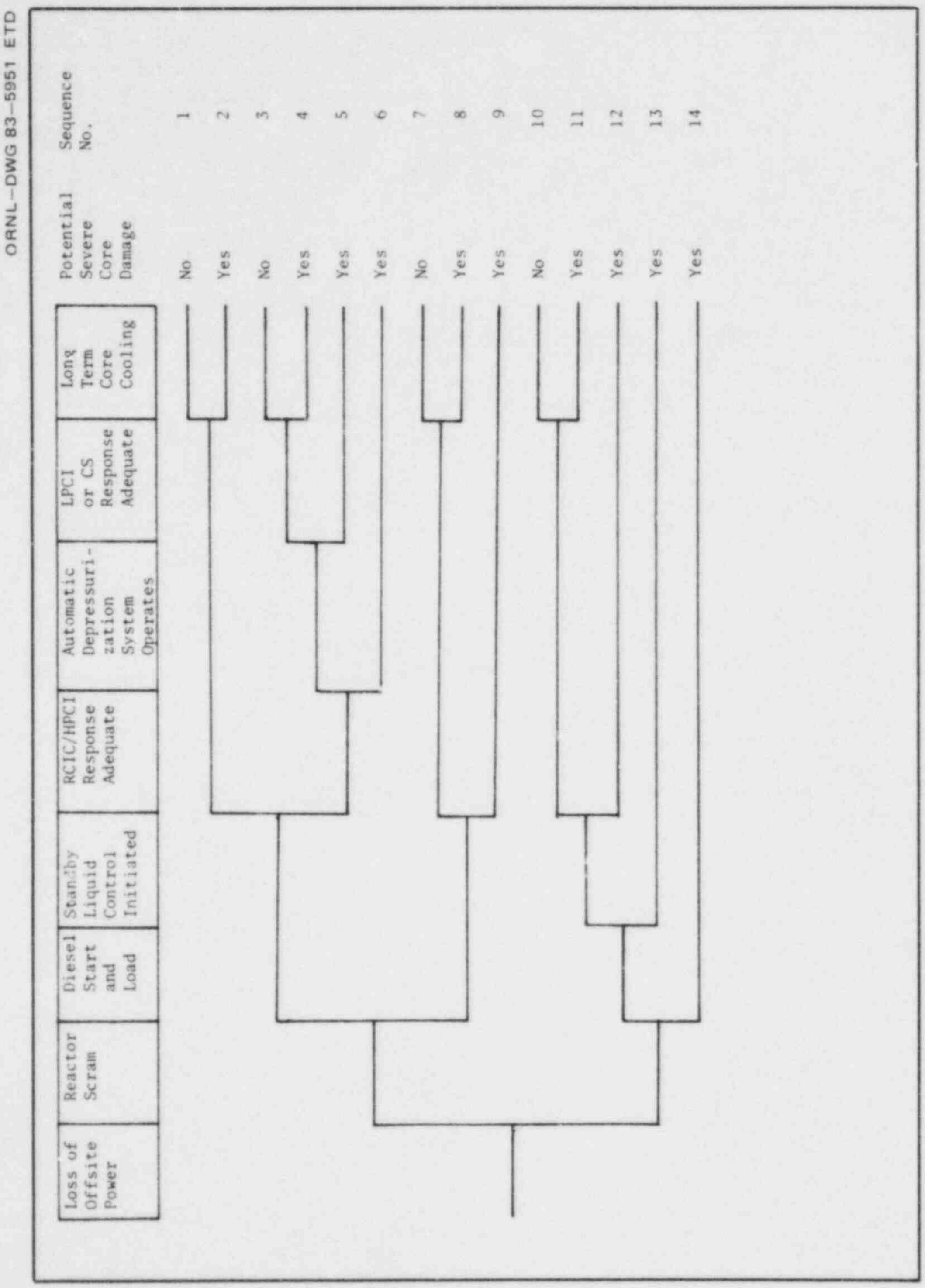

Fig. A.7. Standardized event tree for BWR loss of offsite power.

3. Diesel start and load. The diesel generators receive an initiation signal when an undervoltage condition is detected. The diesel generators must energize at least one safety-related bus, in general, for success.

4. Standby liquid control system initiated. If scram fails, SBLC can be initiated to bring the reactor to subcriticality. Also individual control rods can be driven in as in a normal shutdown by manual initiation. These functions require success of the diesel generators. Both the control rod drive pumps and the standby liquid control system pumps are driven by ac motors fed from a safety-related bus.

5. RCIC/HPCI initiates. Requirements for success of this function are identical to those discussed for a BWR loss of main feedwater. The RCIC/HPCI function relies only on dc power for initiation and turbine control purposes; thus, given a total loss of ac power, RCIC/HPCI will prevent core damage provided dc power and sufficient steam are available to the RCIC/HPIC turbines.

The amount of time batteries can supply reliable service is plant specific. Eight hours is a somewhat typical time for availability of a fully loaded battery; however, this time could be extended by load shedding.

Older BWRs equipped with isolation condensers and feedwater coolant injection instead of RCIC and HPCI have different requirements for success. These requirements are similar to those discussed under the BWR loss of main feedwater event tree description, with the additional requirement that ac power be available for the feedwater pumps. The makeup water to the shell side of the IC also requires success of a pump powered from a diesel-backed bus or a diesel-powered fire pump. Thus, the LOOP mitigation sequence (7) given diesel failure, shown in Fig. A.7, may not apply to these plants.

6. ADS/LPCI/CS initiates. Successful operation of this function is equivalent to that discussed in the BWR loss of main feedwater event tree description. At least one safety-related bus must be supplied by the diesel generators because both the core spray and LPCI pumps use ac power.

7. Long-term core cooling. Success of this function is similar to that for a loss of main feedwater flow.

8. Loss-of-offsite power sequences resulting in potential severe core damage. Ten of the fourteen sequences identified are considered to result in potential severe core damage. These sequences are

- diesel start and success of the ECCS (RCIC/HPCI or ADS and LPCI/CS) with failure of long-term core cooling (sequences 2, 4, and 11);
- complete failure of the ECCS given success of the diesel generators (sequences 5, 6, and 12);
- failure of long-term core cooling given failure of the diesel generators and success of the RCIC/HPCI (sequence 8);
- failure of the RCIC/HPCI given failure of the diesels (sequence 9);
- failure to rapidly reduce the reactor's thermal output via injection of negative reactivity given success of the diesels and a failure to scram (sequence 13);
- failure of the diesels to start and load given a failure to scram (sequence 14).

A.7 BWR Loss-of-Coolant Accident

A functional event tree was constructed to define a representative response of a BWR to an LOCA in terms of critical safety functions. The normal mitigating sequence for an LOCA depends on the break size. Small breaks of interest in this study require reactor scram and RCIC/HPCI operation. If RCIC/HPCI fails, then ADS/LPCI/CS is available. (A large LOCA requires reactor scram and operation of either LPCI or CS. The low-pressure high-capacity pumps will initiate if the capacity of the high-pressure systems is inadequate.)

The event tree shown in Fig. A.8 primarily addresses the small break LOCA, but the following discussion addresses the functions and sequences leading to potential severe core damage for both large and small breaks.

1. Initiating event (loss-of-coolant accident). Any breach in the reactor coolant system on the reactor side of the steam isolation valves that results in coolant loss in excess of the capacity of the control rod drive pumps and RCIC is considered an LOCA. A large LOCA is defined as a break in the RCS that is sufficient to reduce the system pressure, through blowdown, to the operating range of the LPCI/CS systems. All other LOCAs are considered small.

2. Reactor scram and SBLC. Successful scram is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition. For a small break, given scram failure and if high-pressure coolant is provided, initiation of SBLC can place the core in a subcritical condition. However, SBLC alone is not adequate to maintain the reactor at subcritical following a large break with subsequent scram failure. (See the SBLC discussion in the BWR loss of main feedwater event tree description concerning boron concentration reduction.)

3. High-pressure cooling provided. Success for this function requires the initiation and operation of HPCI or use and control of main feedwater to provide high-pressure cooling. The capacity provided must be adequate to replace the coolant being continually lost through the RCS break.

4. ADS/LPCI/CS response adequate. Success of this function is dependent on successful scram and whether the break is large or small. For a large LOCA (by definition), the system pressure will blow down to the operating range of the LPCI/CS systems and ADS success is not required. However, LPCI/CS success is required because HPCI cannot provide the necessary high-volume makeup. Small LOCAs require success of the ADS/LPCI/CS function in the event of HPC failure or unavailability. Successful operation of ADS/LPCI/CS systems under small LOCA conditions is identical to that discussed for a BWR loss of main feedwater.

5. Long-term core cooling. Success of this state requires that heat rejection to the environment commence within 2 to 27 h, depending on the size of the break. Long-term core cooling may be initiated when the reactor has depressurized below the RCIC/HPCI operating pressure. Functional success is difficult to specify in terms of specific component operability for the same reasons discussed under long-term core cooling following a loss of main feedwater.

6. Loss-of-coolant-accident sequences resulting in potential severe core damage. Seven of ten sequences identified are considered to result in potential severe core damage. These sequences are

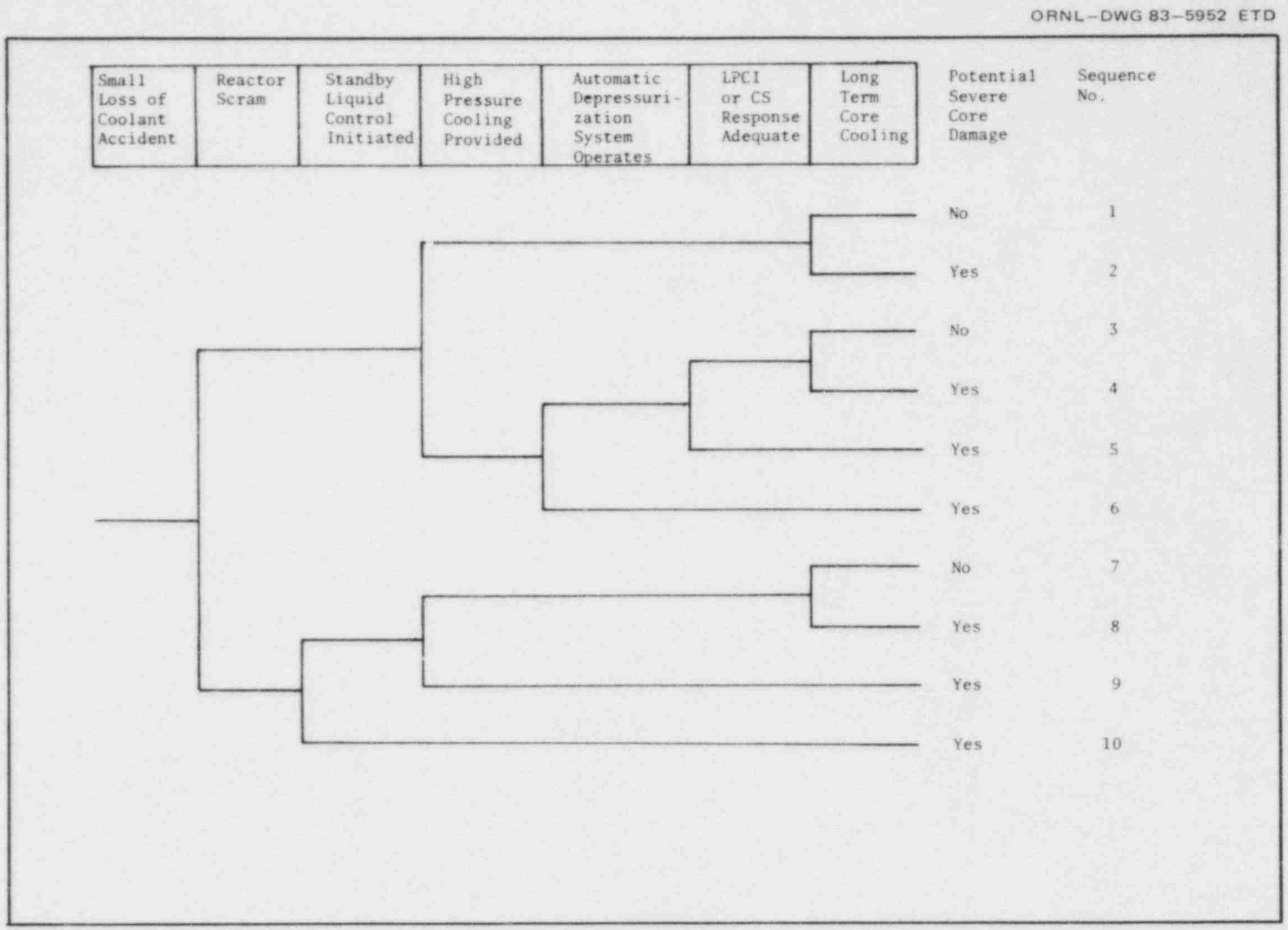

Fig. A.8. Standardized event tree for BWR loss-of-coolant accident.

- failure of long-term core cooling given success of high-pressure cooling (sequences 2 and 8),
- failure of long-term core cooling given success of ADS/LPCI/CS (sequence 4),
- failure of high-pressure cooling, given scram failure (sequence 9),
- complete failure of ECCS (sequences 5 and 6),
- failure to scram the reactor or initiate SBLC (sequence 10).

A.8 BWR Main Steam Line Break

An event tree was constructed to define a representative functional response of a BWR to a main steam line break. The event tree shown in Fig. A.9 illustrates the normal mitigating sequence and alternative sequences. The normal sequence consists of reactor vessel isolation, reactor scram, and operation of RCIC. Alternate methods exist, depending on the size and location of the break. The event tree functions and the sequences leading to potential severe core damage follow.

1. Initiating event (main steam line break). The initiating event is a break in a main steam line, which is defined in this study as a break on the turbine side of the inboard main steam isolation valves.

2. Reactor vessel isolated. Success for this function is dependent on the size of the break and on break location. For a large break, reactor vessel isolation permits control of the cooldown and depressurization rate. Failure to isolate the reactor vessel from a large break will result in uncontrolled blowdown through the broken steam line. If the break occurs between the MSIVs, then closure of the inboard MSIV will isolate the vessel. A break downstream of both MSIVs requires closure of either valve for success.

3. Reactor scrams and SBLC. Given vessel isolation, this function is identical to that for a loss of main feedwater. Success requirements for an unisolated reactor vessel are similar to those for a large-break LOCA.

4. RCIC/HPCI response adequate. Success for this function requires an isolated reactor vessel or a break sufficiently small to allow for HPCI/RCIC functionality. Beyond this, requirements for success are similar to those for loss of main feedwater.

5. ADS/LPCI/CS response adequate. Success of ADS/LPCI/CS, given reactor vessel isolation, is similar to that of a BWR loss of main feedwater. For a large break where reactor vessel isolation fails, ADS is not required to depressurize, but LPCI/CS is still required for success.

6. Long-term core cooling. Success of long-term core cooling given success of ECCS is similar to that for a BWR loss of main feedwater and for a BWR LOCA.

7. Main steam line break sequences leading to potential severe core damage. Ten of fourteen sequences are considered to lead to potential severe core damage. These sequences are

- failure of long-term core cooling given success of RCIC/HPCI or ADS/LPCI/CS (sequences 2, 4, 8, and 12);

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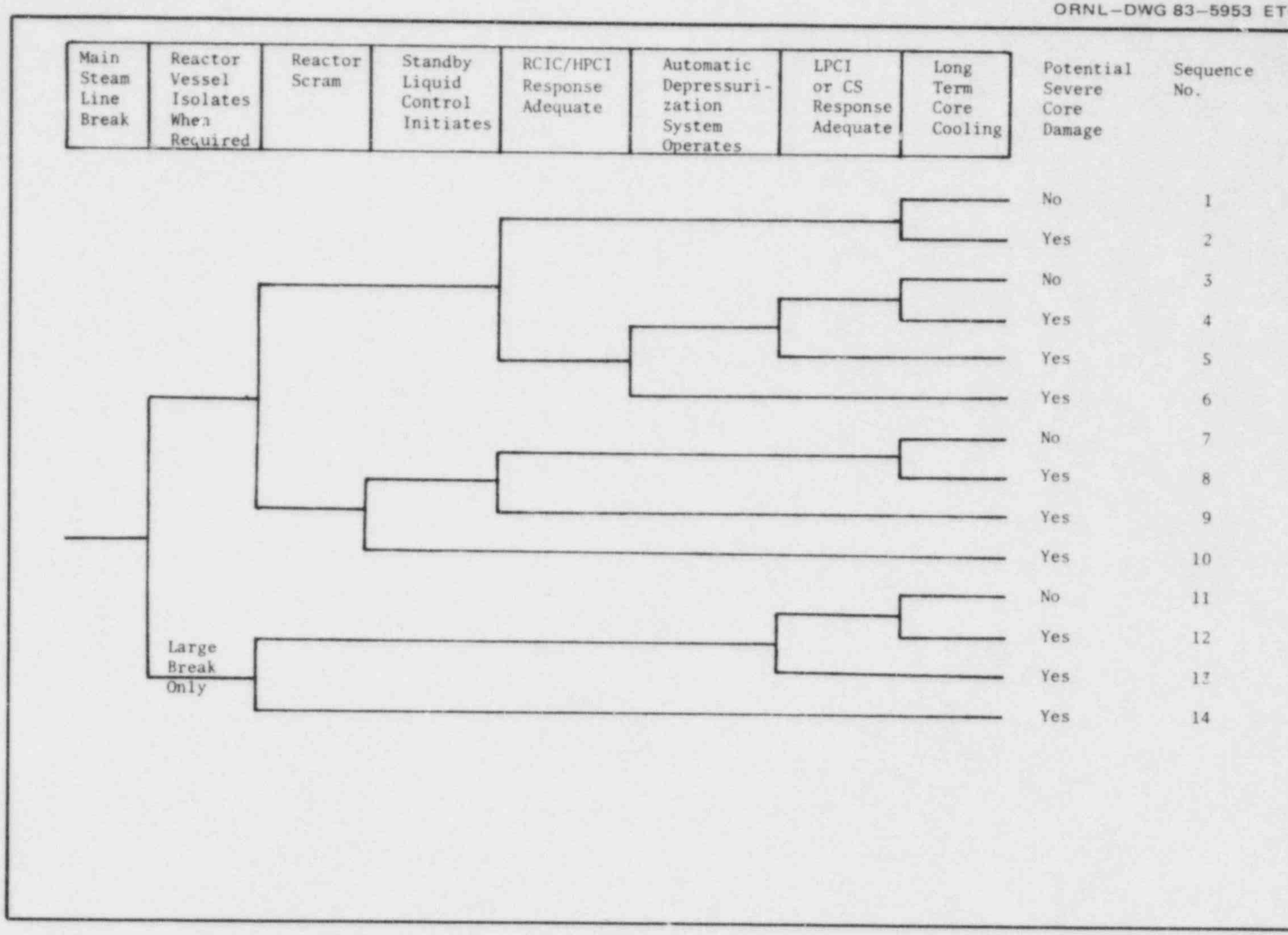


Fig. A.9. Standardized event tree for BWR main steam line break.

- complete failure of the ECCS with or without reactor isolation (sequences 5, 6, 9, and 13);
- failure of the reactor to be brought to subcriticality (sequences 10 and 14).

Appendix B

PRECURSOR SUMMARY SHEETS AND EVENT TREES

This appendix is bound separately as Volume 2 of this document.

Appendix C

PRECURSOR LISTINGS SORTED BY ATTRIBUTE

This appendix contains the 58 precursor events sorted by specific attributes, as reflected in the following list of tables.

Definitions for column headings used in Tables C.1 through C.11 are as follows:

ACCESS	Six-digit NSIC accession number
E DATE	Event date
SEQ	Abbreviation of sequence of interest for event
ACTUAL OCCURRENCE	Brief description of event
PLANT	Plant name and unit number
DOC	Plant docket number
SY	Abbreviation for system involved
COMPXX	System component code
O	Plant operating status
D	Discovery method (O = operational event; T = testing)
E	Human error involved (Y = yes; N = no)
I	Transient or accident induced by actual occurrence (Y = yes; N = no)
AGEX	Plant age from initial criticality at time of event in days
PROB	Probability of conditional core damage
RATE	Plant electrical rating [MW(e)]
T	Plant type (B = BWR; P = PWR)
V	Plant NSSS vendor code
AE	Plant architect-engineer code
OPR	Abbreviation of plant operator
CRITXX	Plant criticality date

Table C.13 contains explanations of abbreviations and codes used in Tables C.1 through C.12.

Table C.1. Precursors sorted by NSIC accession number

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
154451	800204	LOCA	PRESSURIZER RELIEF VALVE OPENS	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUC[illegible] 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2164	5.0E-4	850	P	B	BX	APL	740806
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DC UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	80070[illegible]
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	7701.4
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	C	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323

Table C.2. Precursors sorted by event date

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
154451	800204	LOCA	PRESSURIZER RELIEF VALVE OPENS	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	F	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422

Table C.3. Precursors sorted by plant name

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BY	TEC	770812
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	F	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
154451	800204	LOCA	PRESSURIZER RELIEF VALVE OPENS	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	C	BX	NSP	701210
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224

Table C.4. Precursors sorted by principal system affected

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
154451	800204	LOCA	PRESSURIZER RELIEF VALVE OPENS	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	F	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	F	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	C	UE	CPL	750320
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	3	G	SL	CWE	710131

Table C.5. Precursors sorted by principal component involved

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTEXCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTEXCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
154451	800204	LOCA	PRESSURIZER RELIEF VALVE OPENS	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	7361	7.4E-7	777	B	G	SS	GPC	740912
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916

Table C.6. Precursors sorted by plant operating status

ACCESS	E	DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
158231		800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
159347		800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
167117		810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
174073		800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	C	BX	IEL	740323
163405		800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
154451		800204	LOCA	PRESSURIZER RELIEF VALVE OPENS	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
155475		800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
158228		800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158232		800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233		800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
158279		800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
158650		800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
159134		800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136		800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
160497		801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532		801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559		801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	C	BX	BEC	720616
160846		800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
160926		801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	C	BX	BEC	720616
161906		801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
162083		801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163356		810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
163499		800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
164617		810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
165900		810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166650		810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
166745		810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
167624		810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
169042		800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
170199		811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171667		810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
171700		811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171733		811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
171842		811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
172198		811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
166082		810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
154674		800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
156204		800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158229		800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158860		800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
160453		800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
161601		801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
161649		801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
164453		801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164955		810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
166072		810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
167611		810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
168548		810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
169587		811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
170098		811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
171202		811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171939		811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
165438		810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
166384		810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
164149		810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
168829		810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
163478		800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
164703		810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711

Table C.7. Precursors sorted by discovery method

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BCE	741007
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
162083	8012?6	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163356	810105	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
163405	800626	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
163499	800510	LOCA	REA[illegible]R COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
164149	810129	LOCA	[illegible] RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
164453	801122	LOOP	[illegible] DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164617	810102	UNIQ	LOS[illegible]C POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
164703	810201	LOOP	OPR[illegible] FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
165900	810403	LOCA	PO[illegible]CK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166384	810427	LOOP	L[illegible]V BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
166650	810606	LOCA	S[illegible]VE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
167117	8106[illegible]	[illegible]OP	L[illegible]D VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
167611	810[illegible]	UNIQ	L[illegible] RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	C	BX	IEL	740323
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
15822[illegible]	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
1582[illegible]	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
16[illegible]06	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131

Table C.8. Precursors sorted by events involving human error

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEC	761211
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CF	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
168548	810[illegible]7	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
168829	810[illegible]3	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169042	8004[illegible]7	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	.EL	740323
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614

Table C.9. Precursors sorted by events involving a transient or accident

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
167611	810711	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323

Table C.10. Precursors sorted by plant type and vendor

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	FLECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DC UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UY	PEG	761211
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215

Table C.11. Precursors sorted by architect-engineer

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
160497	801007	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER INOPERABLE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	BEC	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	BEC	720616
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	3054	9.3E-5	693	P	W	BX	FPL	730611
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4F-6	850	P	B	BX	APL	740806
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
158233	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CVE	710131
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
167611	810211	UNIQ	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705

Table C.12. Precursors sorted by operating utility

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	C	D	E	I	AGEX	PROB	RATE	T	V	AE	OPR	CRITXX
158279	800407	LOOP	CAVITATION OF EFW PUMPS	ARKANSAS 2	368	EA	CKTBRK	E	O	N	Y	580	6.0E-4	912	P	C	BX	APL	781205
159136	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 2	368	EA	ELECON	E	O	N	Y	567	1.6E-5	912	P	C	BX	APL	781205
162083	801220	LOCA	ALL ESFAS RWT INSTRU INOPERABLE	ARKANSAS 2	368	IB	INSTRU	E	O	Y	N	746	9.8E-7	912	P	C	BX	APL	781205
159134	800624	LOOP	GROUND FAULT ON GRID CAUSES TRIP	ARKANSAS 1	313	EA	ELECON	E	O	N	Y	2149	5.4E-6	850	P	B	BX	APL	740806
163499	800510	LOCA	REACTOR COOLANT PUMP SEAL FAILS	ARKANSAS 1	313	CA	PUMPXX	E	O	N	Y	2104	5.0E-4	850	P	B	BX	APL	740806
169042	800407	LOOP	LOOP AND HPI VALVE FAILS TO OPEN	ARKANSAS 1	313	SF	VALVOP	E	O	N	Y	2071	5.4E-6	850	P	B	BX	APL	740806
160497	801007	LOCA	REACTOR VESSEL RELIEF VALV[illegible]S	PILGRIM 1	293	PA	VALVOP	E	O	N	Y	3035	1.4E-4	655	B	G	BX	BEC	720616
160532	801010	LOFW	COMPONENT COOLING WATER I[illegible]LE	PILGRIM 1	293	EB	CKTBRK	E	O	N	Y	3038	4.2E-9	655	B	G	BX	[illegible]	720616
160559	801031	LOCA	REACTOR VESSEL RELIEF VALVE OPENS	PILGRIM 1	293	PA	MECFUN	E	O	N	Y	3059	1.4E-4	655	B	G	BX	[illegible]	720616
160926	801001	LOCA	RELIEF VALVE STUCK OPEN	PILGRIM 1	293	SF	VALVOP	E	O	N	Y	3029	1.4E-4	655	B	G	BX	BEC	720616
154674	800202	LOOP	LOSS OF SITE EMERGENCY POWER	CALCLIFFS2	318	EE	INSTRU	G	O	Y	N	1159	3.2E-8	845	P	C	BX	BGE	761130
158650	800520	LOOP	AIR LINE LEAK FAILS SERVICE WATER	CALCLIFFS1	317	PA	HTETCH	E	O	N	Y	2052	7.0E-5	845	P	C	BX	BGE	741007
158232	800603	LOOP	LIGHTNING STRIKE TRANSMISSION TOWER	IND.POINT2	247	EA	ZZZZZZ	E	O	N	Y	2569	1.3E-5	873	P	W	UE	CEC	730522
163356	810106	LOOP	STATION BATTERY BREAKERS OPENED	PALISADES	255	EC	CKTBRK	E	O	Y	N	3515	1.3E-7	805	P	C	BX	CPC	710524
166745	810626	LOOP	TWO SHUTDWN SEQS & 1 DG UNAVAIL	PALISADES	255	EE	RELAYX	E	T	N	N	3686	5.0E-7	805	P	C	BX	CPC	710524
166082	810410	LOFW	LOSS OF RCIC AND HPCI SYSTEMS	BRUNSWICK2	324	SF	MECFUN	F	T	N	N	2213	5.6E-5	821	B	G	UE	CPL	750320
166072	810419	LOCA	RHR HEAT EXCHANGERS DAMAGED	BRUNSWICK1	325	WA	HTEXCH	G	T	N	N	1654	6.7E-3	821	B	G	UE	CPL	761008
164149	810129	LOCA	LETDOWN RELIEF VALVE LEAKS IN A LOFW	ROBINSON 2	261	PC	VALVEX		O	N	Y	3784	8.6E-6	700	P	W	EX	CPL	700920
158229	800719	LOFW	FAILURE OF SDV VENT CHECK VALVE	DRESDEN 3	249	RB	VALVEX	G	T	N	N	3457	3.3E-5	794	B	G	SL	CWE	710131
158231	800425	LOFW	ADS VALVES FAIL TO OPERATE	DRESDEN 3	249	SF	VALVOP	C	T	Y	N	3372	1.3E-5	794	B	G	SL	CWE	710131
171842	811023	LOOP	DEGRADED COOLING OF DIESEL GENS	DRESDEN 3	249	WB	VALVEX	E	T	N	N	3918	1.8E-6	794	B	G	SL	CWE	710131
161906	801116	LOFW	RCIC DISCHARGE ISOLATION FAIL TO OPEN	QUAD-CTES2	265	CE	VALVOP	E	T	N	N	3126	3.2E-5	789	B	G	SL	CWE	720426
171733	811211	LOFW	AUX FEED PUMPS FAIL TO AUTO START	ZION 2	304	SH	PUMPXX	E	O	N	N	2909	6.5E-5	1040	P	W	SL	CWE	731224
154451	800204	LOCA	PRESSURIZER PRESSURE RELIEF VALVE OPEN	HAD.NECK	213	CI	INSTRU	E	O	N	Y	4578	3.4E-5	580	P	W	SW	CYA	670724
165900	810403	LOCA	PORV AND BLOCK VALVE OPEN	HAD.NECK	213	CA	ELECON	E	O	N	Y	5002	3.4E-5	580	P	W	SW	CYA	670724
166650	810606	LOCA	SIS SUPPLY VALVE FOUND CLOSED	BVRVALLEY1	334	WB	VALVEX	E	O	Y	N	1853	2.4E-6	852	P	W	SW	DLC	760510
164703	810201	LOOP	OPR ERROR FAILS AC POWER	LACROSSE	409	EA	CKTBRK		O	Y	Y	4954	1.0E-5	50	B	A	SL	DLP	670711
160846	800226	LOCA	24 VDC TO NON-NUCLEAR INSTR LOST	CRYSTALRV3	302	IF	INSTRU	E	O	N	Y	1138	5.0E-3	825	P	B	GX	FPC	770114
167624	810616	LOOP	LOOP AND ONE DIESEL GEN FAILS	CRYSTALRV3	302	EA	XXXXXX	E	O	N	Y	1614	3.7E-4	825	P	B	GX	FPC	770114
171202	811112	LOOP	BOTH DIESEL GENERATORS UNAVAILABLE	TKY.POINT3	250	EE	VALVOP	G	T	N	N	3310	6.4E-8	693	P	W	BX	FPL	721020
158[illegible]33	800611	UNIQ	CCW LOST TO RCP SEALS	ST.LUCIE 1	335	WB	INSTRU	E	O	N	Y	1511	1.1E-3	802	P	C	EX	FPL	760422
172198	811219	LOCA	MSIV CLOSURE & SAFETY VALVE LIFT	ST.LUCIE 1	335	HB	VALVEX	E	O	N	Y	1240	7.7E-5	802	P	C	EX	FPL	760422
169587	811021	LOCA	BIT FLOW PATH TO RCS OBSTRUCTED	TKY.POINT4	251	SF	PIPEXX	G	T	N	N	30[illegible]4	9.3E-5	693	P	W	BX	FPL	730611
163478	800626	LOFW	HPCI AND RCIC FAIL TO START	HATCH 1	321	SF	MECFUN		O	N	Y	2114	3.3E-4	777	B	G	SS	GPC	740912
164955	810228	LOFW	HPCI AND RCIC INOPERABLE	HATCH 1	321	SF	PIPEXX	G	T	Y	N	2361	7.4E-7	777	B	G	SS	GPC	740912
165438	810405	LOOP	DG CIRCUIT BREAKERS FAIL TO CLOSE	HATCH 1	321	EE	RELAYX	H	T	N	N	2397	2.2E-7	777	B	G	SS	GPC	740912
174073	800111	LOCA	STUCK OPEN RELIEF VALVE	D. ARNOLD	331	SF	VALVEX	C	O	N	Y	2120	1.4E-5	538	B	G	BX	IEL	740323
164617	810102	UNIQ	LOSS OF DC POWER AND 1 DIESEL	MILLSTONE2	336	EC	CKTBRK	E	O	Y	Y	1904	5.1E-3	870	P	C	BX	NNE	751017
166384	810427	LOOP	LOSS OF 4.16KV BUS NETWORK	MONTICELLO	263	EB	CKTBRK	H	O	Y	N	3791	1.8E-5	545	B	G	BX	NSP	701210
158228	800715	LOOP	STORM SPURIOUS REACTOR TRIP	PRAIRIEIS2	306	EA	CKTBRK	E	O	N	Y	2037	4.7E-5	530	P	W	PX	NSP	741217
161601	801008	LOOP	LOSS OF SERVICE WATER TO DIESEL GENS	SALEM 1	272	WA	VALVOP	G	T	N	N	1397	1.4E-7	1090	P	W	UX	PEG	761211
170098	811106	MSLB	2 BIT INLET VALVES FAIL TO OPEN	SALEM 1	272	RB	VALVOP	G	O	N	Y	1791	1.5E-8	1090	P	W	UX	PEG	761211
156204	800411	MSLB	THREE OF FOUR MSIVS FAIL TO CLOSE	TROJAN	344	CD	VALVEX	G	T	N	N	1579	6.1E-8	1130	P	W	BX	PGC	751215
155475	800310	UNIQ	LOSS OF SERVICE WATER SYSTEM	SANONOFRE1	206	WA	PUMPXX	E	O	N	N	4653	1.6E-5	436	P	W	BX	SCE	670614
164453	801122	LOOP	LOOP AND DEGRADE LOAD SHED ABILITY	SANONOFRE1	206	EB	CKTBRK	G	O	Y	N	4910	6.1E-5	436	P	W	BX	SCE	670614
168829	810903	UNIQ	SIS VALVES FAIL IN LOFW	SANONOFRE1	206	SF	VALVEX		O	N	Y	5195	1.4E-4	436	P	W	BX	SCE	670614
171700	811119	LOOP	EMERGENCY POWER UNAVAILABLE	SANONOFRE1	206	EE	ENGINE	E	T	Y	N	5272	2.6E-7	436	P	W	BX	SCE	670614
167117	810619	LOOP	LOW SWITCHYARD VOLTAGES	RANCHOSECO	312	EA	ZZZZZZ	C	O	N	Y	2468	5.2E-6	918	P	B	BX	SMU	740916
168548	810807	LOOP	SWITCHYARD VOLTAGE IS TOO LOW	RANCHOSECO	312	EA	ELECON	G	O	N	Y	2517	6.9E-6	918	P	B	BX	SMU	740916
158860	800419	UNIQ	LOSS OF 2 ESSENTIAL BUSES	DVS-BESSE1	346	EB	RELAYX	G	O	N	Y	981	1.4E-3	906	P	B	BX	TEC	770812
160453	800826	LOOP	LOSS OF POWER TO DIESEL BREAKERS	DVS-BESSE1	346	EE	CKTBRK	G	O	Y	N	1110	9.4E-9	906	P	B	BX	TEC	770812
171667	810624	LOOP	LOSS OF VITAL BUS	DVS-BESSE1	346	EB	CKTBRK	E	O	Y	Y	1412	1.7E-3	906	P	B	BX	TEC	770812
163405	800628	LOFW	76 CONTROL RODS FAIL TO INSERT	BRN.FERRY3	296	RB	CONROD	D	O	N	N	1420	9.8E-4	1065	B	G	UX	TVA	760808
161649	801016	LOOP	TWO DIESEL GENERATORS UNAVAILABLE	SEQUOYAH 1	327	EE	INSTRU	G	O	Y	N	103	6.7E-8	1148	P	W	UX	TVA	800705
167611	810211	UNIO	LOSS OF RHRS & RCS BLOWDOWN OCCURS	SEQUOYAH 1	327	CF	VALVEX	G	O	Y	Y	221	8.7E-4	1148	P	W	UX	TVA	800705
171939	811003	MSLB	STEAM DUMP VALVES OPEN	NORTHANNA2	339	HE	INSTRU	G	O	N	Y	478	8.4E-6	907	P	W	SW	VEP	800612
159347	800819	MSLB	STEAM FLOW TRANSMITTERS ARE ISOLATED	SURRY 2	281	IB	INSTRU	C	O	Y	N	2722	1.E-10	822	P	W	SW	VEP	730307
170199	811016	LOOP	UNAVAILABILITY OF BOTH CCW TRAINS	KEWAUNEE	305	WB	HTETCH	E	O	N	N	2780	1.1E-8	535	P	W	FP	WPS	740307

Table C.13. Definitions of abbreviations and codes

ACCESS: 6 DIGIT NSIC ACCESSION NUMBER
E DATE: EVENT DATE
SEQ: SEQUENCE OF INTEREST FOR THE EVENT
- 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
- RCPT - REACTOR COOLANT PUMP TRIP
- SGTR - STEAM GENERATOR TUBE RUPTURE
- UNIQ - A UNIQUE SEQUENCE

ACTUAL OCCURRENCE: DESCRIPTION OF EVENT
PLANT NAME: NAME OF PLANT AND UNIT NUMBER
DOC: PLANT DOCKET NUMBER
SY:SYSTEM ABBREVIATION:

STANDARD GENERIC CODE	SYSTEM 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	ENGINEERED SAFETY FEATURE INSTRUMENT SYSTEMS
IC	SYSTEMS REQUIRED FOR SAFE SHUTDOWN
ID	SAFETY RELATED DISPLAY INSTRUMENTATION
IE	OTHER INSTRUMENT SYSTEMS REQUIRED FOR SAFETY
IF	OTHER INSTRUMENT SYSTEMS NOT REQUIRED FOR SAFETY

Table C.13 (continued)

	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 THEIR CONTROLS
	OTHER AUXILIARY SYSTEMS
AA	AIR CONDITIONING, HEATING, COOLING AND VENTILATION SYSTEMS AND CONTROLS
AB	FIRE PROTECTION SYSTEMS AND CONTROLS
AC	COMMUNICATION SYSTEMS
AD	OTHER AUXILIARY SYSTEMS AND THEIR CONTROLS
	STEAM AND POWER CONVERSION SYSTEMS
HA	TURBINE-GENERATORS AND CONTROLS
HB	MAIN STEAM SUPPLY 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)

Table C.13 (continued)

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

COMPXX: SYSTEM COMPONENT CODE:

COMPONENT TYPE (COMPONENT CODE)	COMPONENT TYPE INCLUDES
ACCUMULATORS (ACCUMU)	SCRAM ACCUMULATORS SAFETY INJECTION TANKS SURGE TANKS HOLDUP/STORAGE TANKS
AIR DRYERS (AIRDRY)	
ANNUNCIATOR MODULES (ANNUNC)	ALARMS BELLS BUZZERS CLAXONS HORNS GONGS SIRENS
BATTERIES AND CHARGERS (BATTRY)	CHARGERS DRY CELLS WET CELLS STORAGE CELLS
BLOWERS (BLOWER)	COMPRESSORS GAS CIRCULATORS FANS VENTILATORS
CIRCUIT CLOSERS/INTERRUPTERS (CKTBRK)	CIRCUIT BREAKERS CONTACTORS CONTROLLERS STARTERS SWITCHES (OTHER THAN SENSORS) SWITCHGEAR
CONTROL RODS (CONROD)	POISON CURTAINS
CONTROL ROD DRIVE MECHANISMS (CRDRVE)	
DEMINERALIZERS (DEMINX)	ION EXCHANGERS
ELECTRICAL CONDUCTORS (ELECON)	BUS CABLE WIRE
ENGINES, INTERNAL COMBUSTION (ENGINE)	BUTANE ENGINES DIESEL ENGINES GASOLINE ENGINES NATURAL GAS ENGINES PROPANE ENGINES
FILTERS (FILTER)	STRAINERS SCREENS
FUEL ELEMENTS (FUELXX)	
GENERATORS (GENERA)	INVERTERS
HEATERS, ELECTRIC (HEATER)	HEAT TRACERS
HEAT EXCHANGERS (HTEXCH)	CONDENSERS COOLERS EVAPORATORS REGENERATIVE HEAT EXCHANGERS STEAM GENERATORS FAN COIL UNITS

Table C.13 (continued)

INSTRUMENTATION AND CONTROLS (INSTRU)	CONTROLLERS SENSORS/DETECTORS/ELEMENTS INDICATORS DIFFERENTIALS INTEGRATORS (TOTALIZERS) POWER SUPPLIES RECORDERS SWITCHES TRANSMITTERS COMPUTATION MODULES
MECHANICAL FUNCTION UNITS (MECFUN)	MECHANICAL CONTROLLERS GOVERNORS GEAR BOXES VARIDRIVES COUPLINGS
MOTORS (MOTORX)	ELECTRIC MOTORS HYDRAULIC MOTORS PNEUMATIC (AIR) MOTORS SERVO MOTORS
PENETRATIONS, PRIMARY CONTAIN. (PENETR)	AIR LOCKS PERSONNEL ACCESS FUEL HANDLING EQUIPMENT ACCESS ELECTRICAL INSTRUMENT LINE PROCESS PIPING
PIPES, FITTINGS (PIPEXX)	
PUMPS (PUMPXX)	
RECOMBINERS (RECOMB)	
RELAYS (RELAYX)	SWITCHGEAR
SHOCK SUPPRESSORS AND SUPPORT (SUPORT)	HANGERS SUPPORTS SWAY BRACES/STABILIZERS SNUBBERS ANTI-VIBRATION DEVICES
TRANSFORMERS (TRANSF)	
TURBINES (TURBIN)	STEAM TURBINES GAS TURBINES HYDRO TURBINES
VALVES (VALVEX)	VALVES DAMPERS
VALVE OPERATORS (VALVOP)	EXPLOSIVE, SQUIB
VESSELS, PRESSURE (VESSEL)	CONTAINMENT VESSELS DRYWELLS PRESSURE SUPPRESSION PRESSURIZERS REACTOR VESSELS
OTHER COMPONENTS (XXXXXX)	
CODES NOT APPLICABLE (ZZZZZZ)	

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 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)
E: HUMAN ERROR INVOLVED (N-NO, Y-YES)
I: TRANSIENT/ACCIDENT INDUCED BY ACTUAL OCCURRENCE (N-NO, Y-YES)
AGEX: PLANT AGE AT THE TIME OF THE EVENT IN DAYS

Table C.13 (continued)

PROB: PROBABILITY
RATE: PLANT ELECTRICAL RATING IN MEGAWATTS ELECTRIC
T: PLANT TYPE (B=BWR, P=PWR)
V: PLANT NSSS VENDOR
A-ALLIS CHALMERS
B-BABCOCK AND WILCOX
C-COMBUSTION ENGINEERING
G-GENERAL ELECTRIC
W-WESTINGHOUSE

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
EX-EBASCO	RT-BROWN AND ROOT	UX-UTILITY
FP-FLOUR POWER	SL-SARGENT AND LUNDY	XX-OTHER

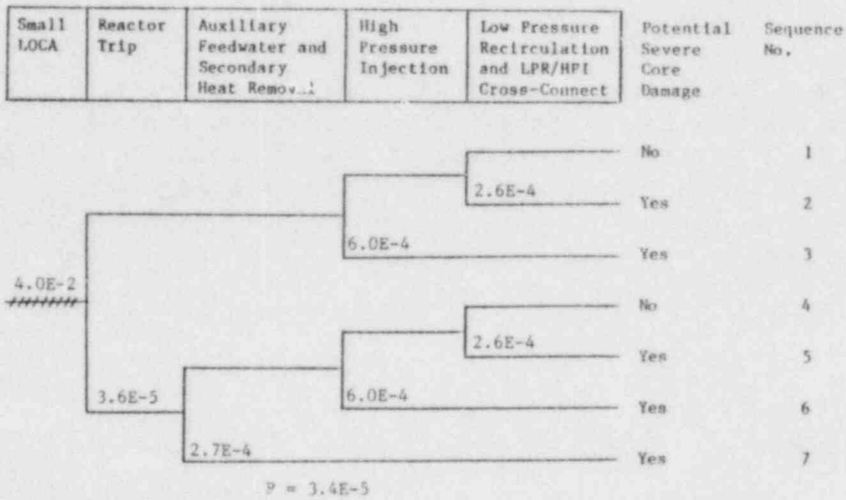
OPR: PLANT LICENSEE ABBREVIATION:

LICENSEE ABBR.	LICENSEE	LICENSEE ABBR.	LICENSEE
APC	ALABAMA POWER COMPANY	NMP	NIAGARA 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 ELECTRIC 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
DLP	DAIRYLAND POWER COOPERATIVE	PSI	PUBLIC SERVICE OF INDIANA
DLC	DUQUESNE LIGHT COMPANY	PSN	PUBLIC SERVICE OF NEW HAMPSHIRE
DPC	DUKE POWER COMPANY	PSO	PUBLIC SERVICE COMPANY OF OKLAHOMA
DPP	OMAHA PUBLIC POWER DISTRICT	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 ELECTRIC 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	LOUISIANA 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		

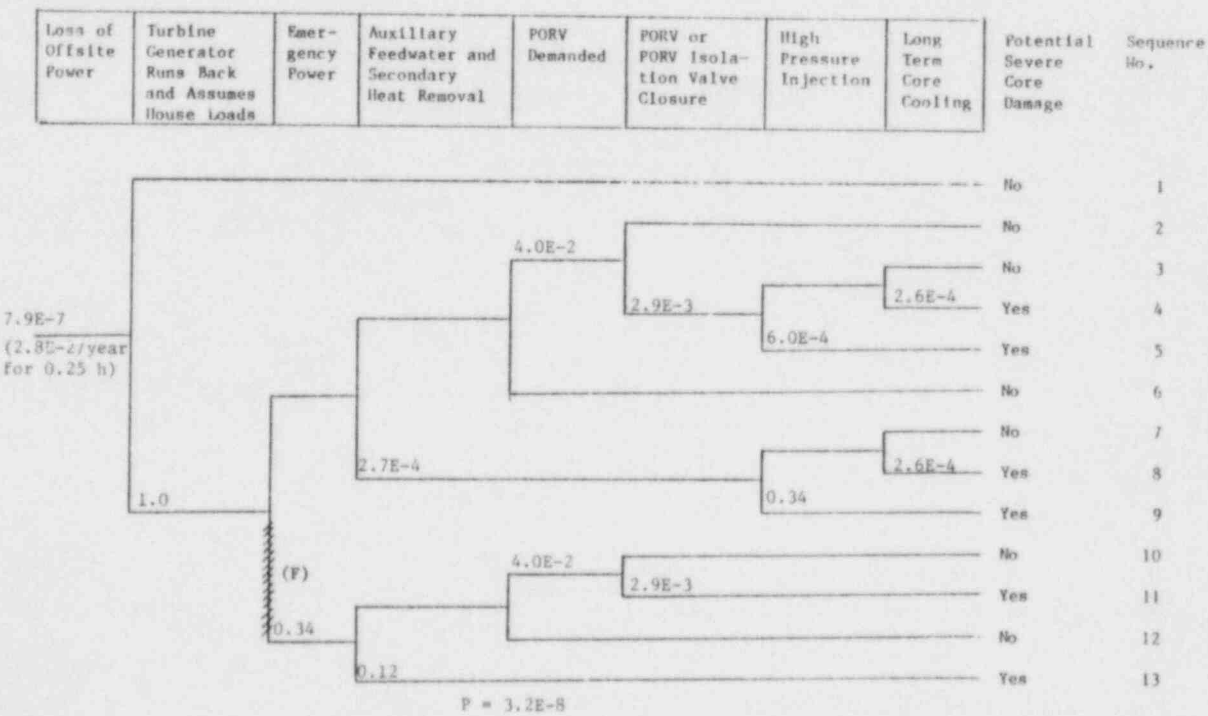
Appendix D

EVENT TREE QUANTIFICATION

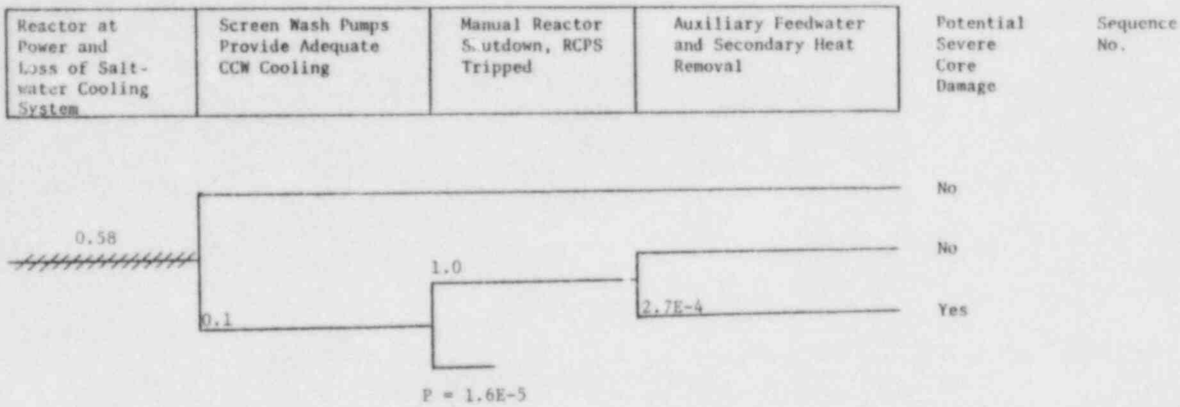
This appendix contains additional supportive material for the quantification of conditional probability for the precursor events. It includes the sequence of interest event trees for each precursor with the quantification of event tree branches, the event probability, and the faults associated with each precursor. The precursor event trees appear in order of NSIC accession number.



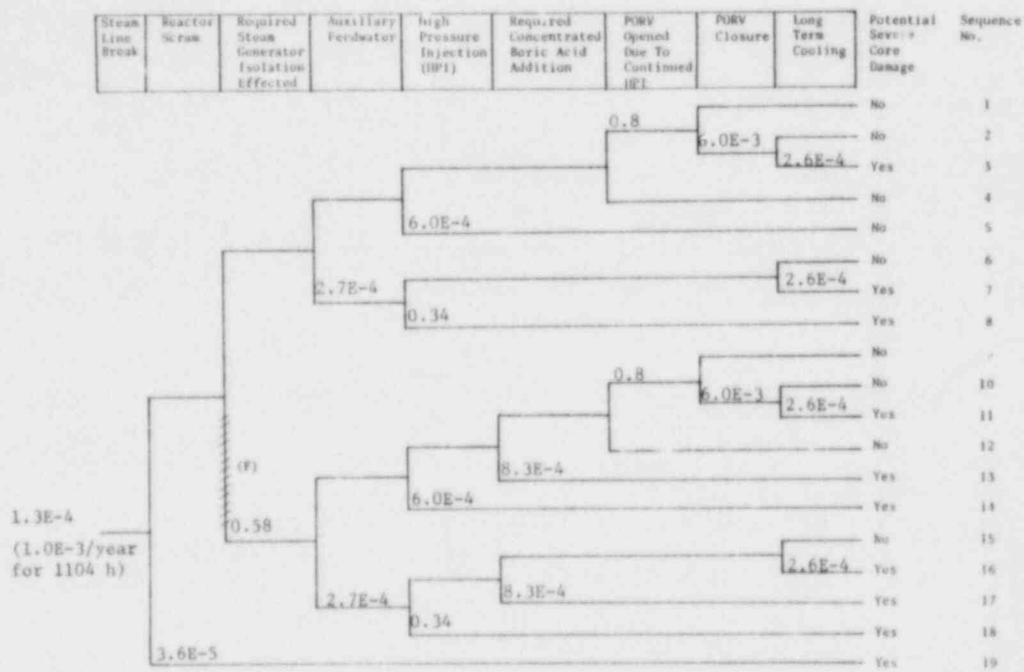
NSIC 154451 - Sequence of Interest for Inadvertent Opening of PORV at Haddam Neck



NSIC 154674 - Sequence of Interest for Emergency Power System Unavailability at Calvert Cliffs 2

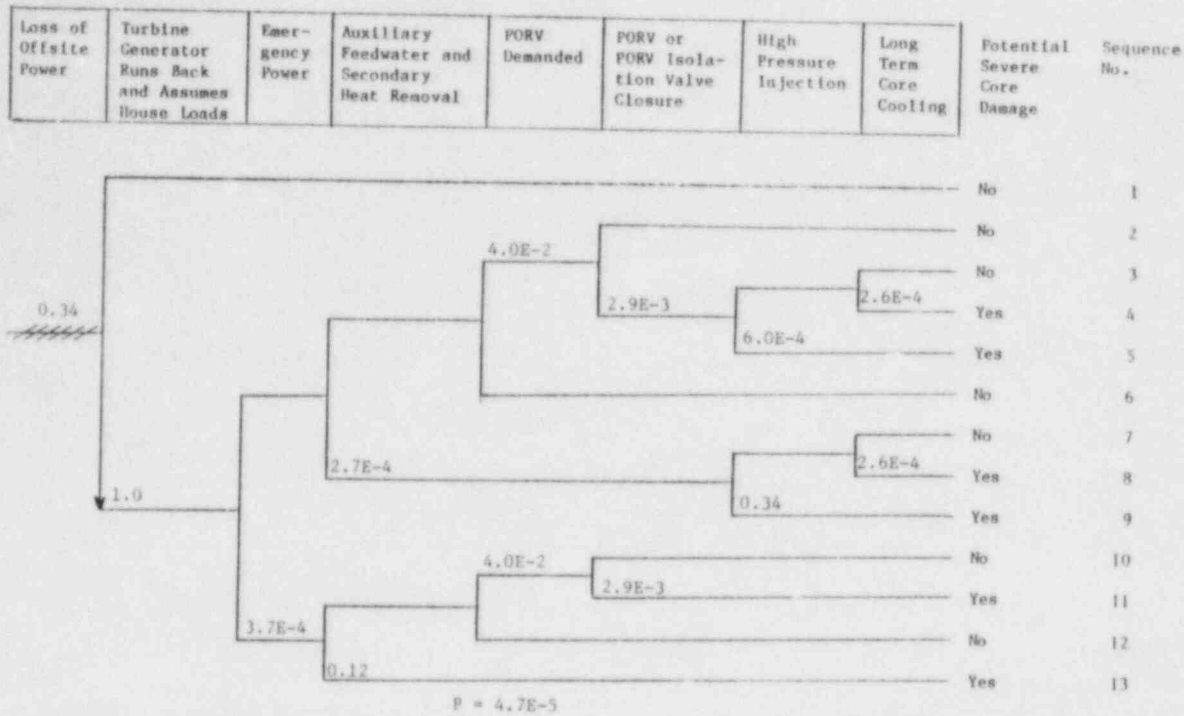


NSIC 155475 - Sequence of Interest for Loss of Saltwater Cooling System at San Onofre 1

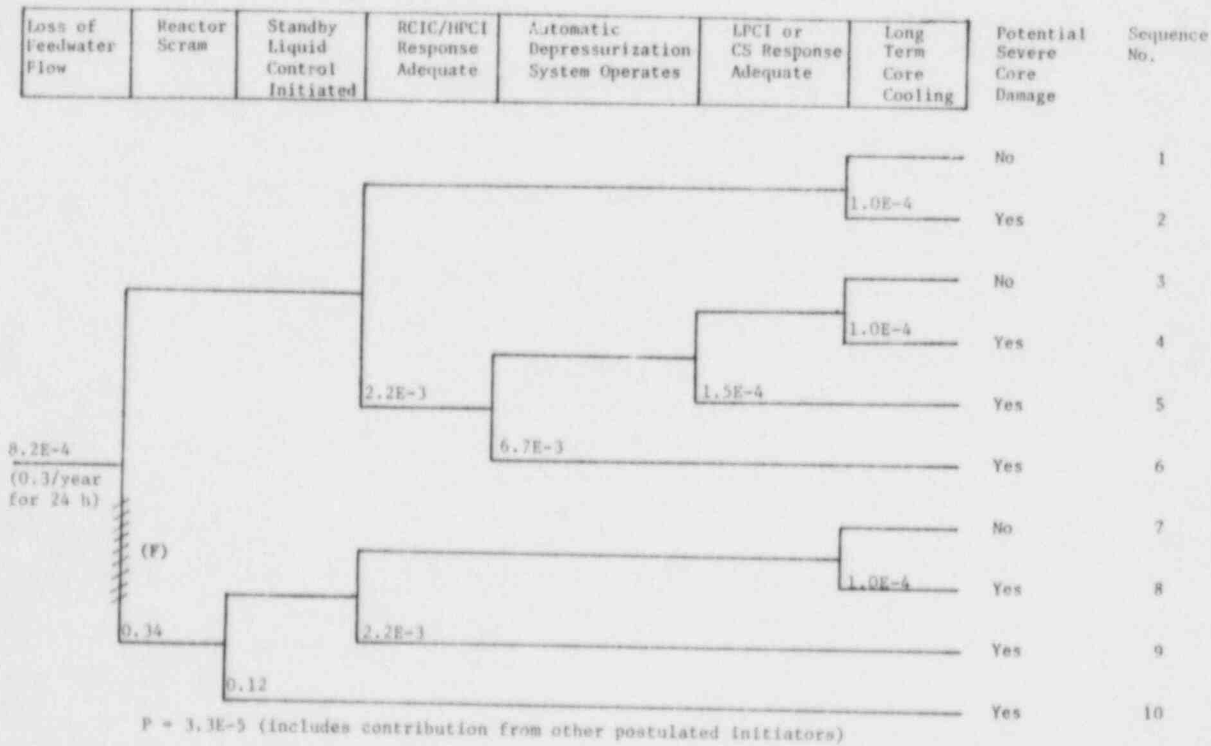


P = 6.1E-8 (includes correction of 0.75 for likelihood of occurring at power)

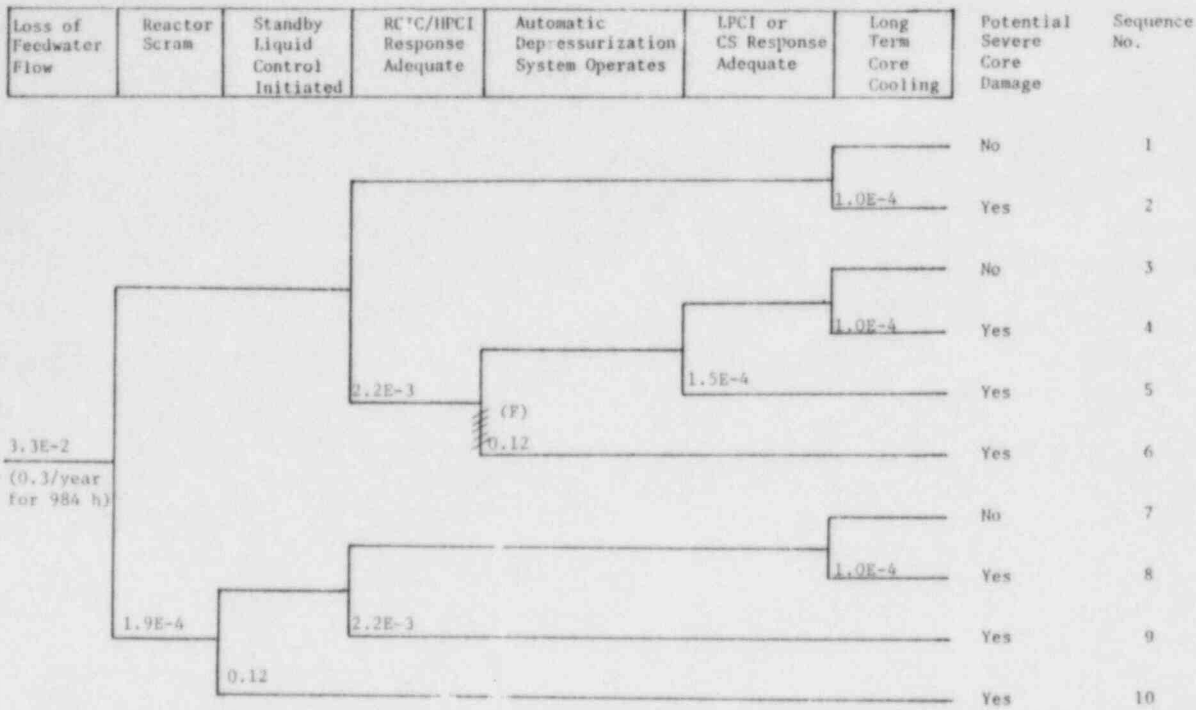
NSIC 156204 - Sequence of Interest for Failure of MSIVs to Close on Demand at Trojan



NSIC 158228 - Sequence of Interest for Loss of Offsite Power at Prairie Island 2

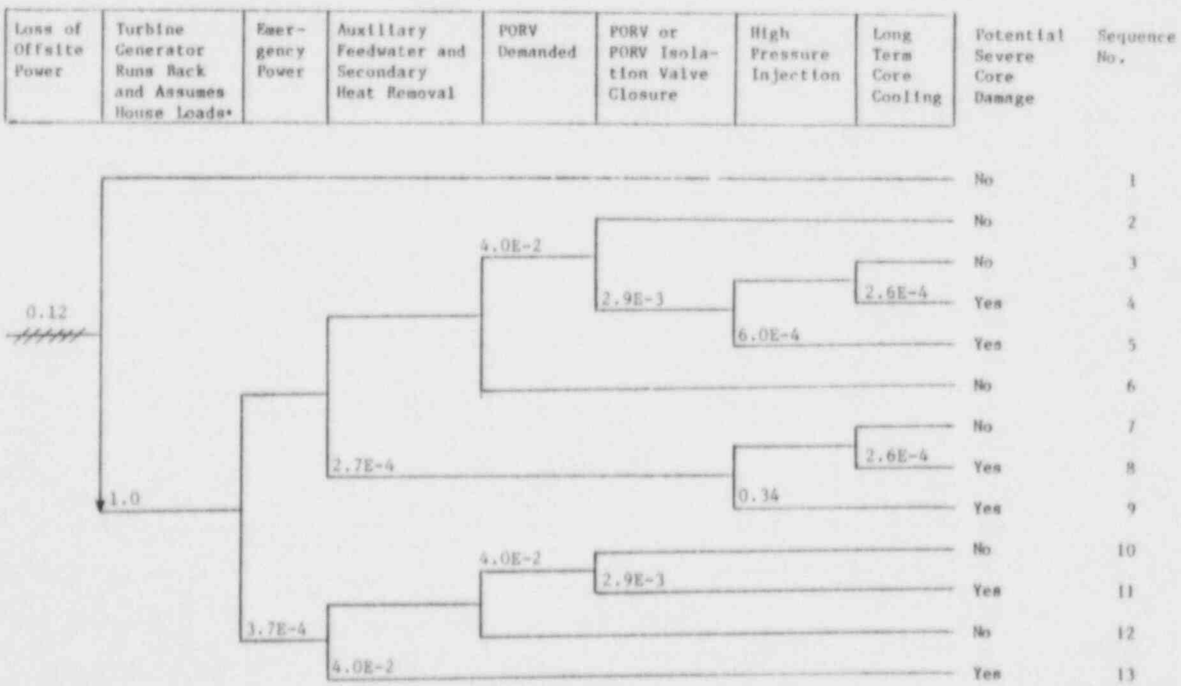


NSIC 158229 - Sequence of Interest for Scram Discharge Fails to Drain and Alarms Clear at Dresden 3



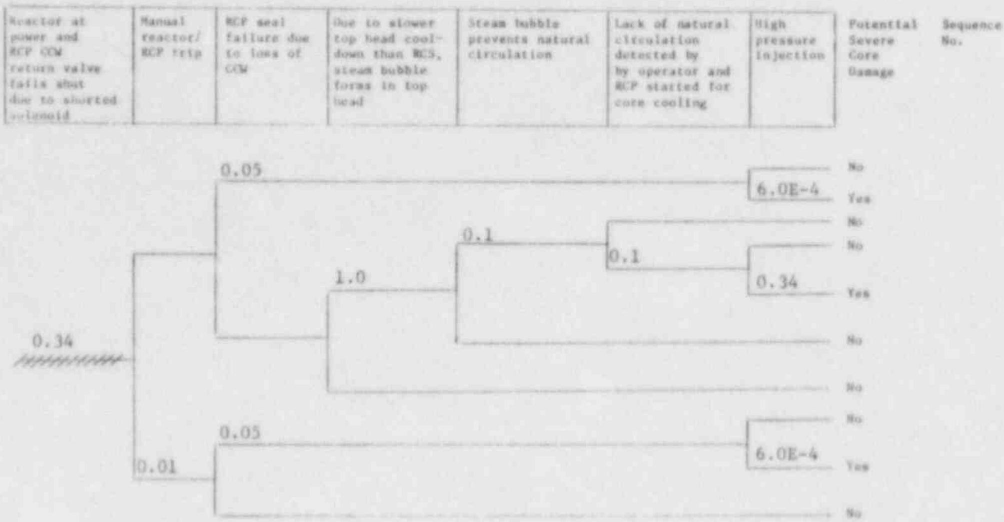
P = 1.3E-5 (includes contribution from other postulated initiators)

NSIC 158231 - Sequence of Interest for ADS Valves Fail to Open at Dresden 3



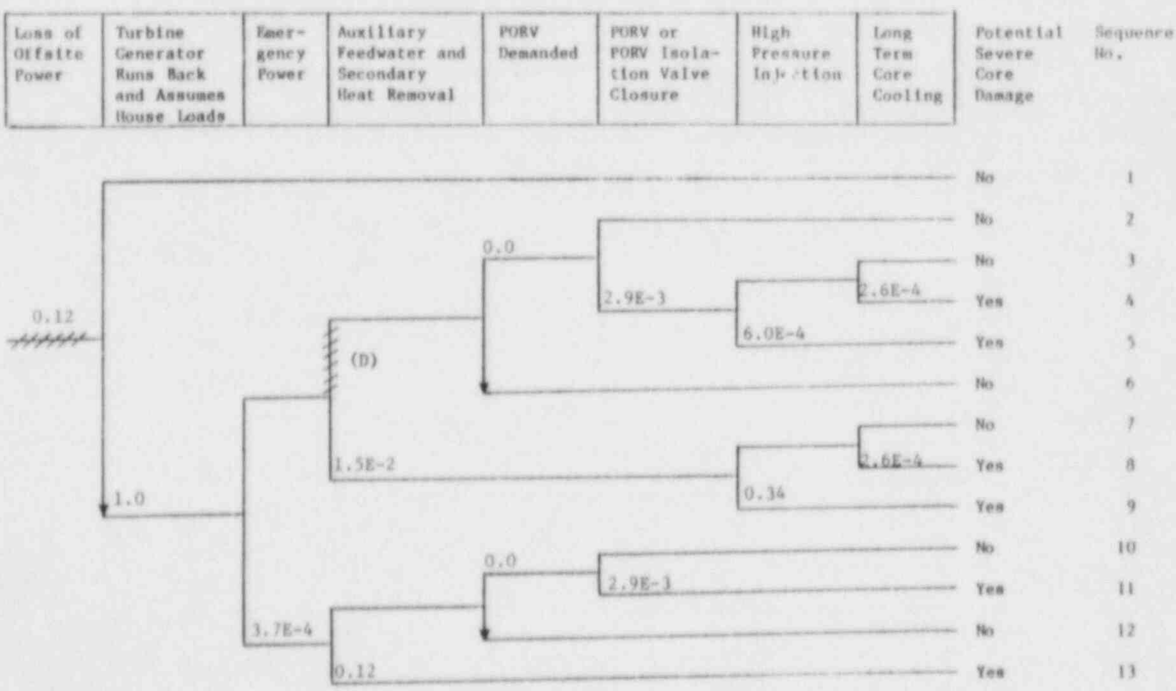
P = 1.3E-5

NSIC 158232 - Sequence of Interest for Loss of Offsite Power at Indian Point 2



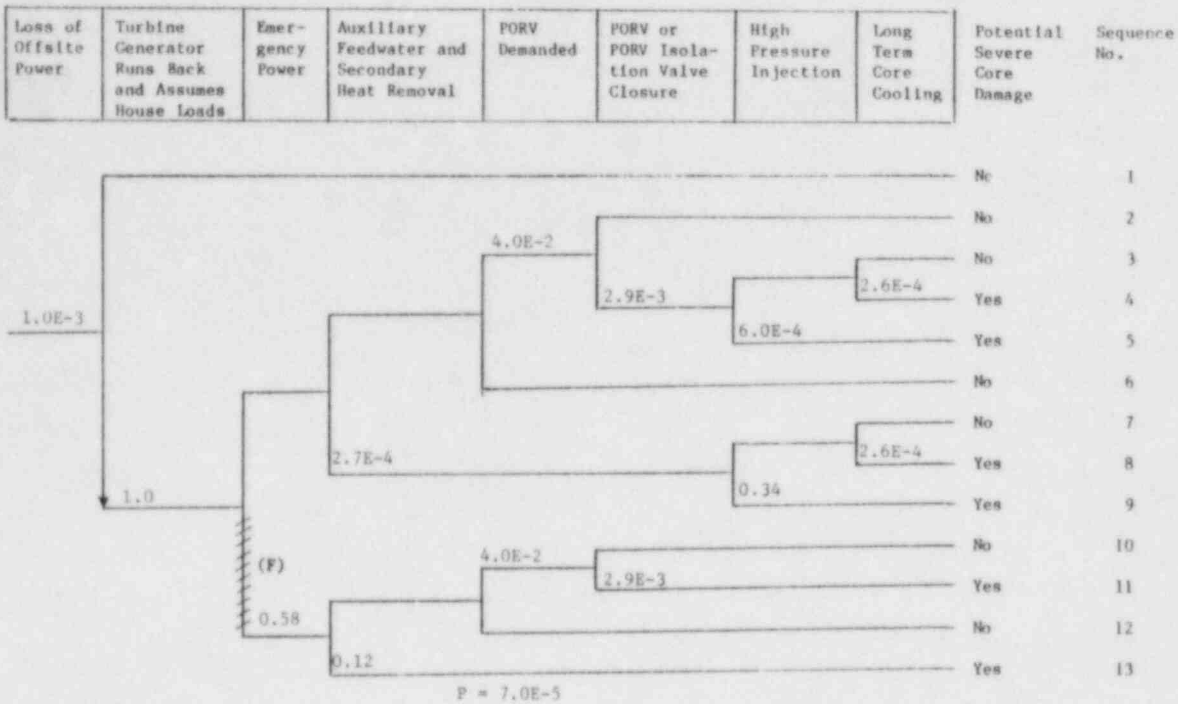
P = 1.1E-3

NSIC 158233 - Sequence of Interest for Loss of Reactor Coolant Pumps and Top Head Bubble Incident at St. Lucie 1

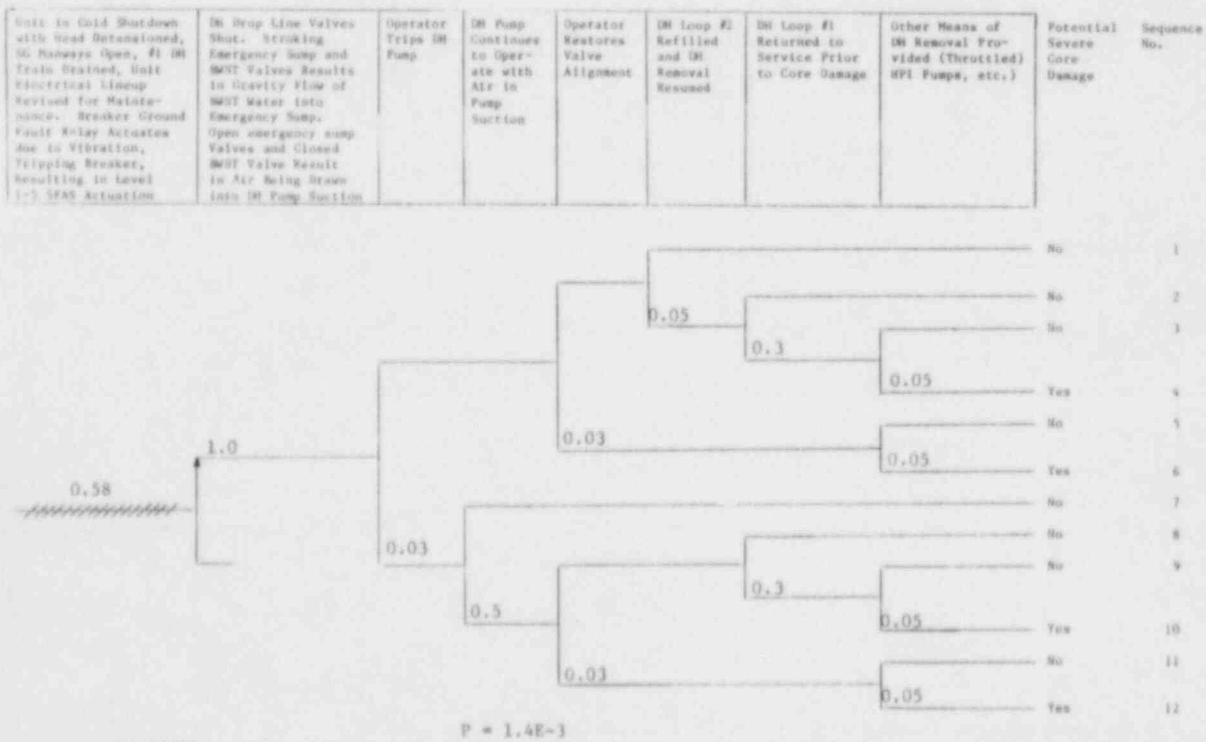


P = 6.0E-4

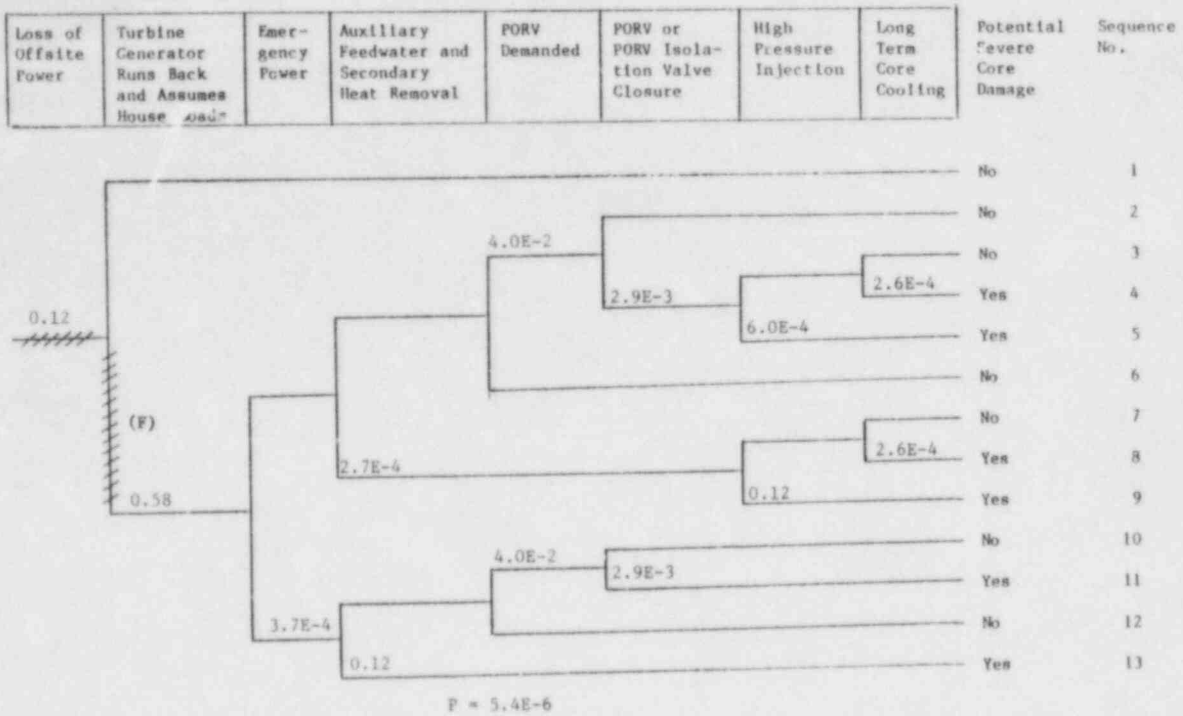
NSIC 158279 - Sequence of Interest for Degraded Emergency Feedwater System During a Loss of Offsite Power at Arkansas Unit 2



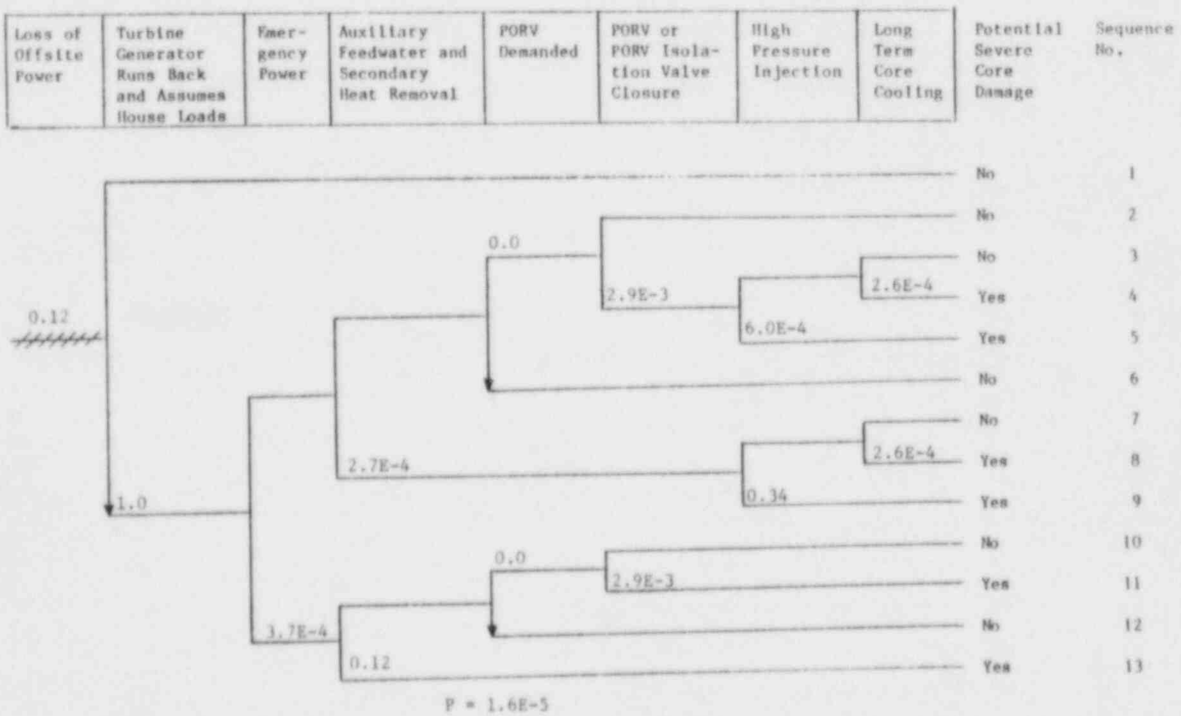
NSIC 158650 - Sequence of Interest for Failure of Service Water System and Subsequent Auxiliary Feedwater Unavailability at Calvert Cliffs 1



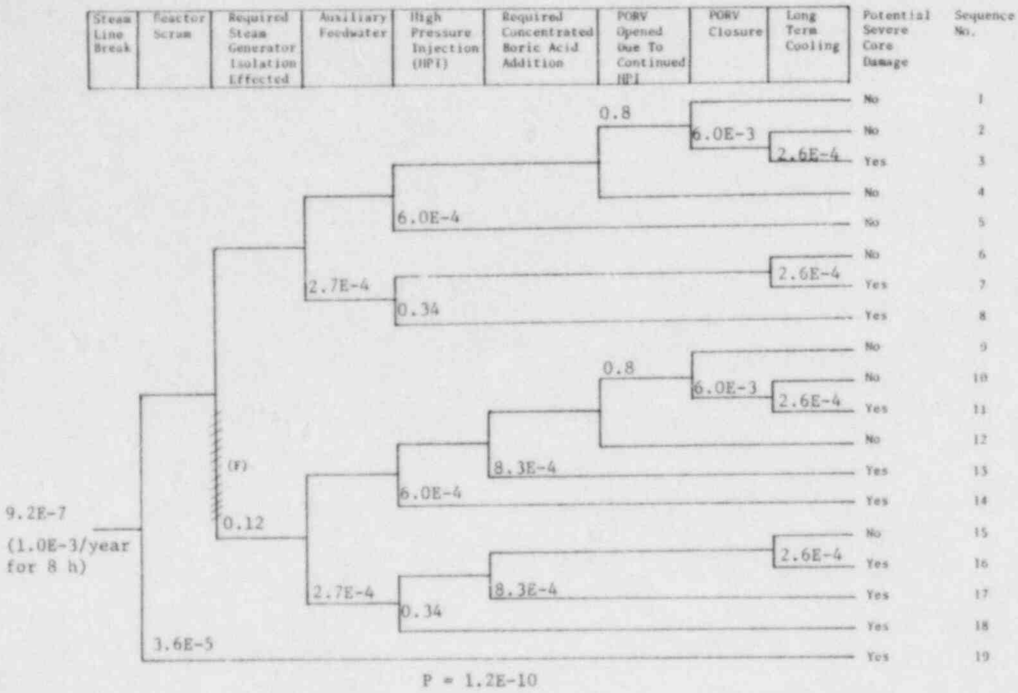
NSIC 158860 - Sequence of Interest for Loss of Two Essential Buses and Loss of Decay Heat Removal Capability at Davis-Besse 1



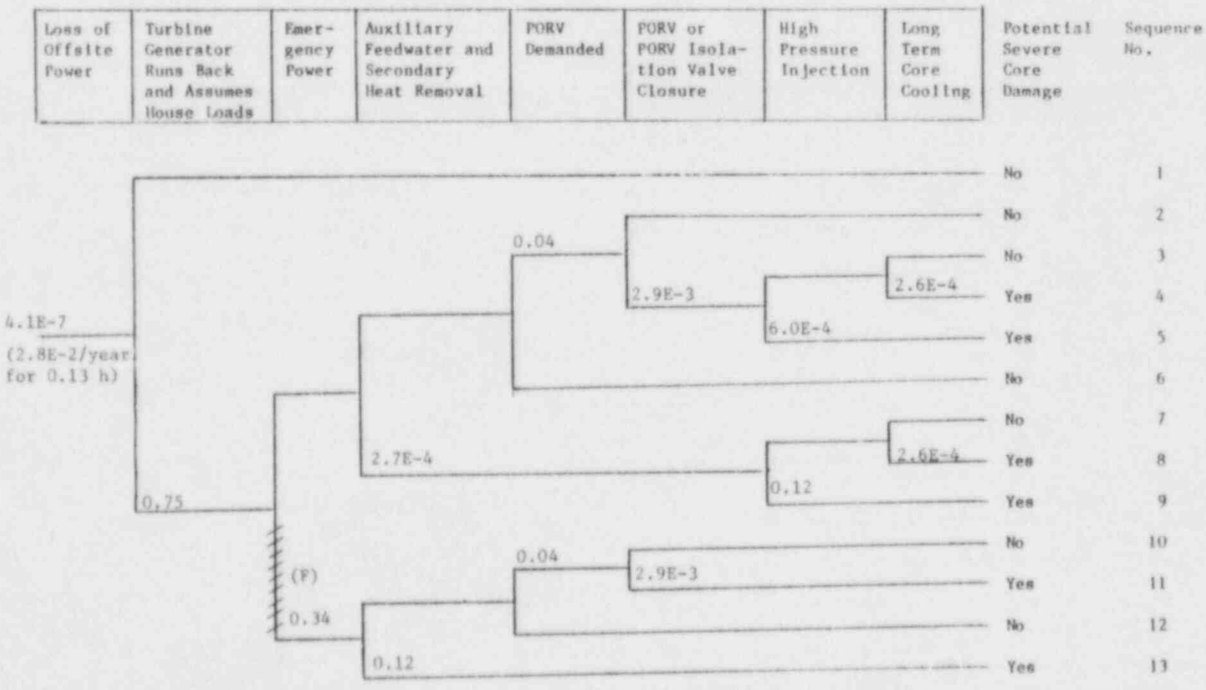
NSIC 159134 - Sequence of Interest for Loss of Offsite Power at Arkansas Nuclear - Unit 1



NSIC 159136 - Sequence of Interest for Loss of Offsite Power at Arkansas Nuclear 2

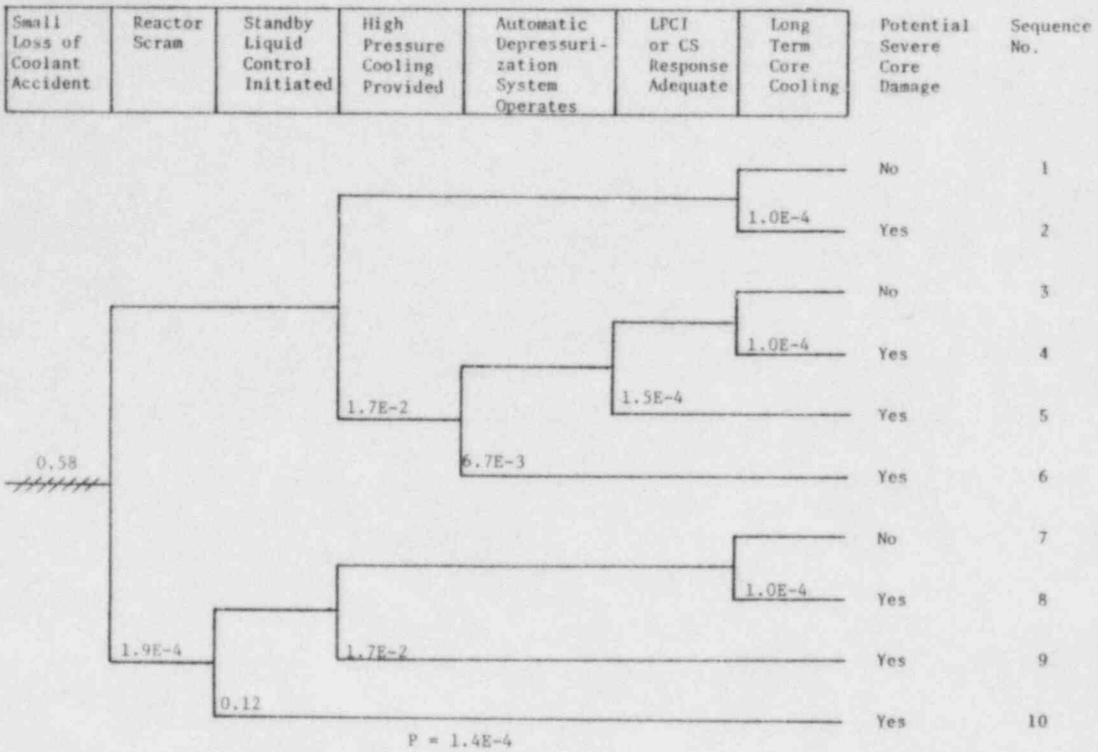


NSIC 159347 - Sequence of Interest for Unavailability of All High Steam Flow Signals at Surry 2

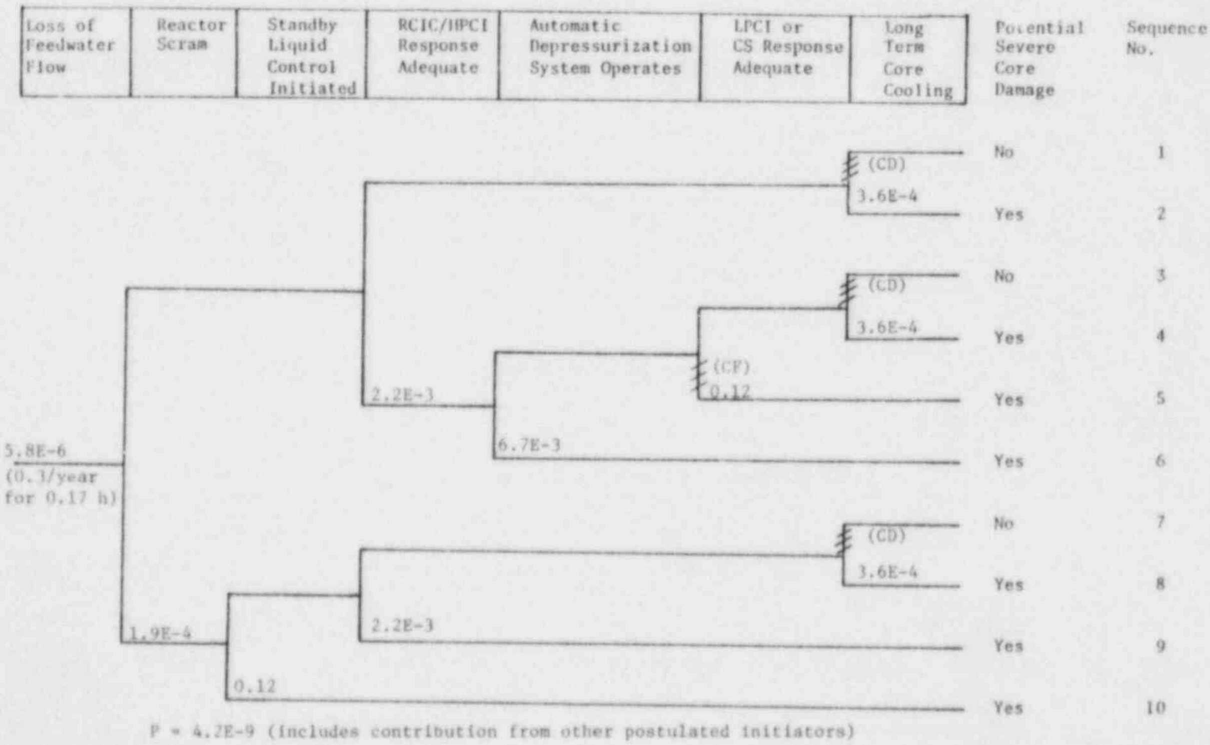


P = 9.4E-9 (includes factor of 0.75 for likelihood of occurring at power)

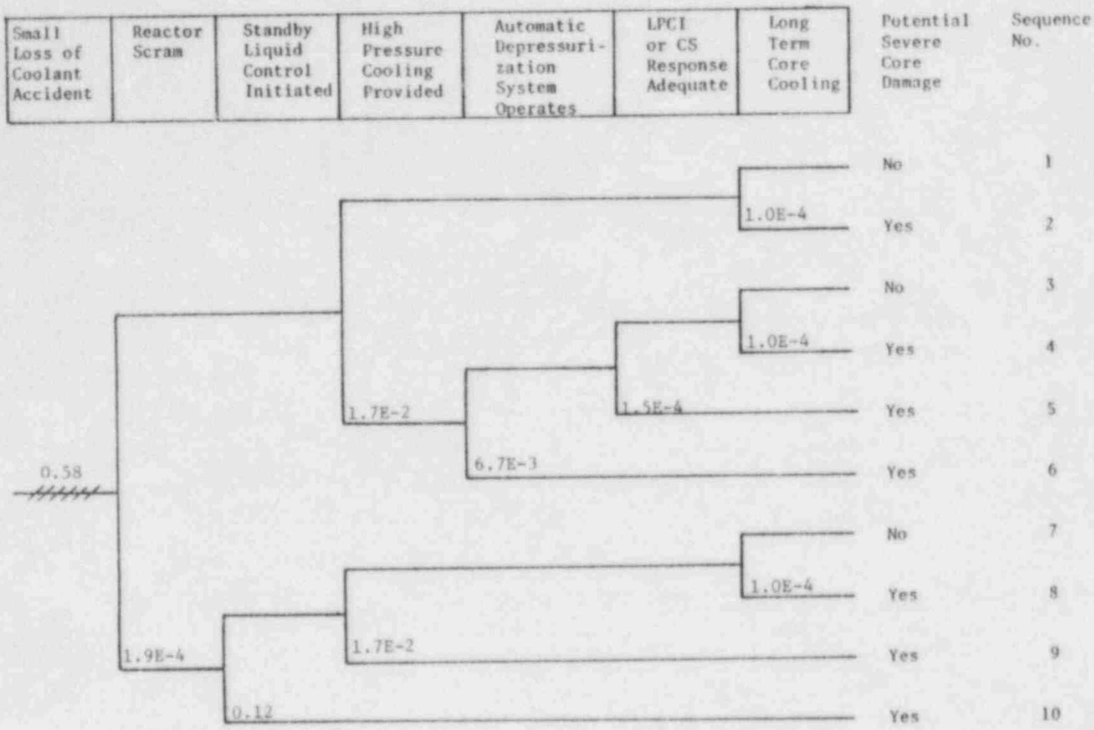
NSIC 160453 - Sequence of Interest for Loss of Emergency Power at Davis-Besse 1



NSIC 160497 - Sequence of Interest for Inadvertent Opening of Relief Valve at Pilgrim 1

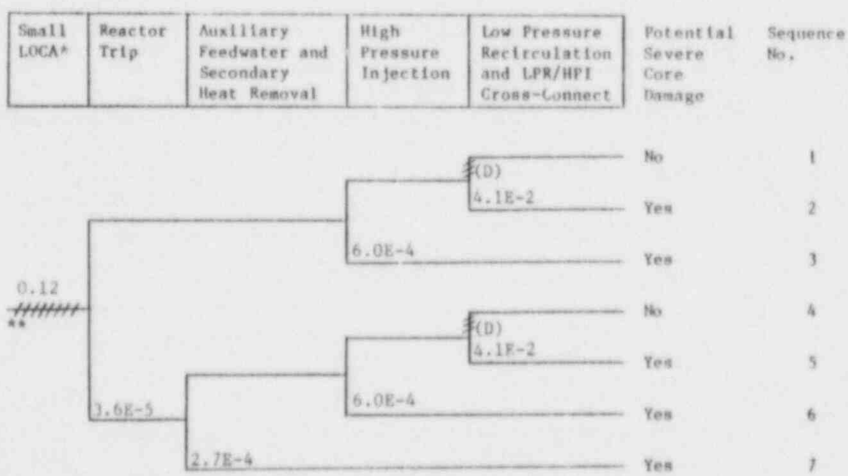


NSIC 160532 - Sequence of Interest for Component Cooling Water Inoperable at Pilgrim 1



P = 1.4E-4

NSIC 160559 - Sequence of Interest for Inadvertent Opening of Relief Valve at Pilgrim 1

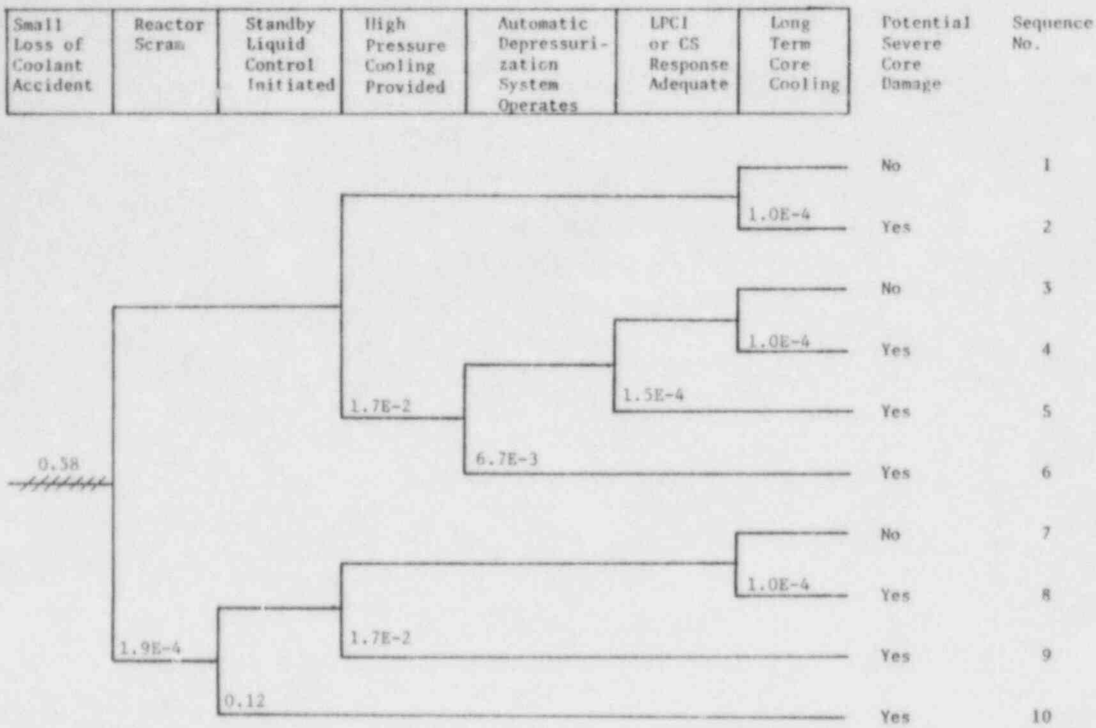


P = 5.0E-3

NSIC 160846 - Sequence of Interest for Loss of 24-V dc Nonnuclear Instrumentation Power Supply at Crystal River 3

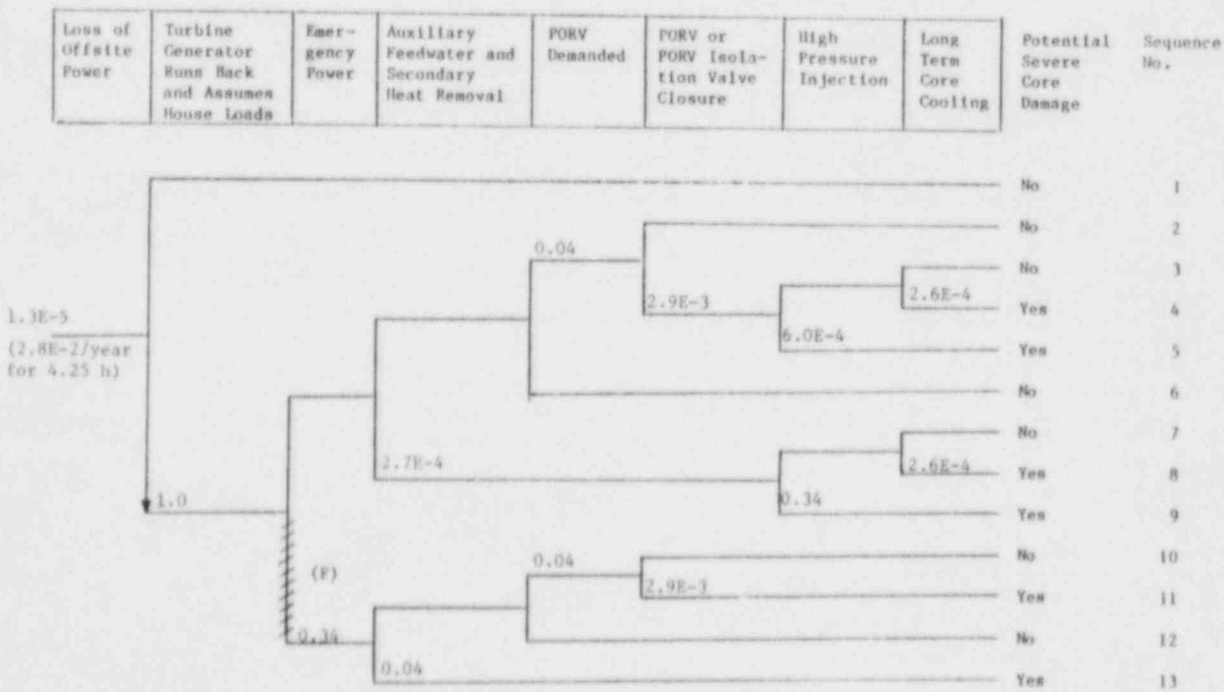
*due to loss of NNI-X power supply

**continued small LOCA requires operator error in not closing PORV block valve



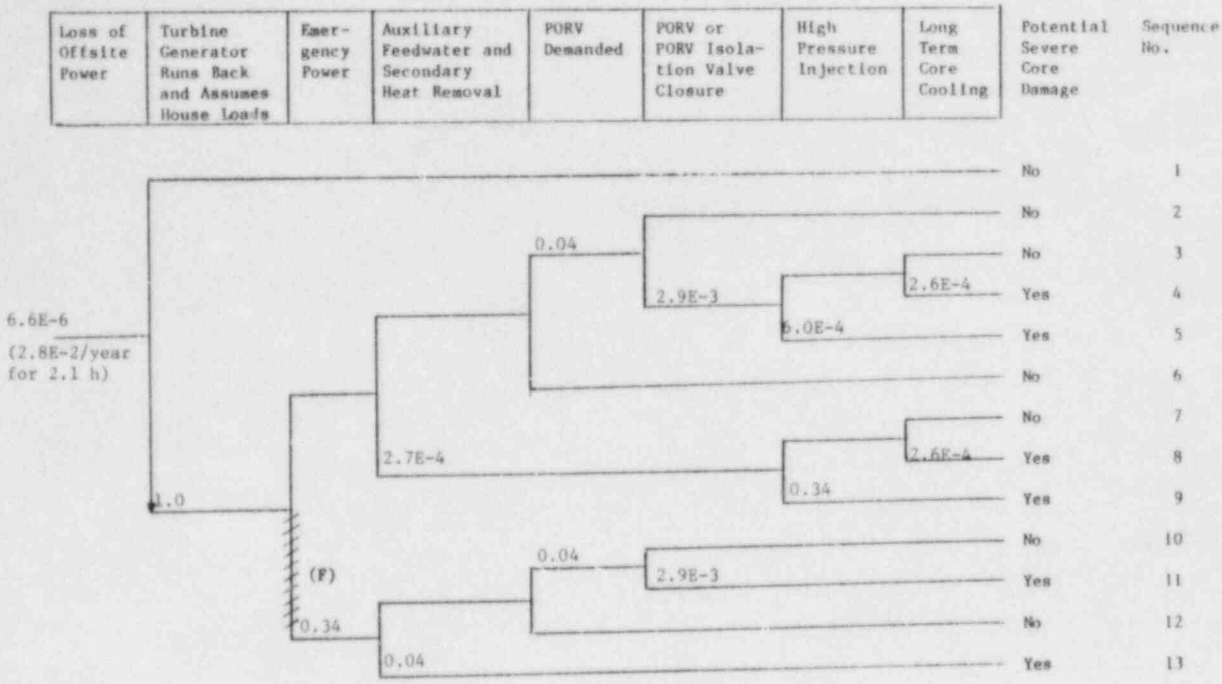
P = 1.4E-4

NSIC 160926 - Sequence of Interest for Stuck Open Relief Valve at Pilgrim 1



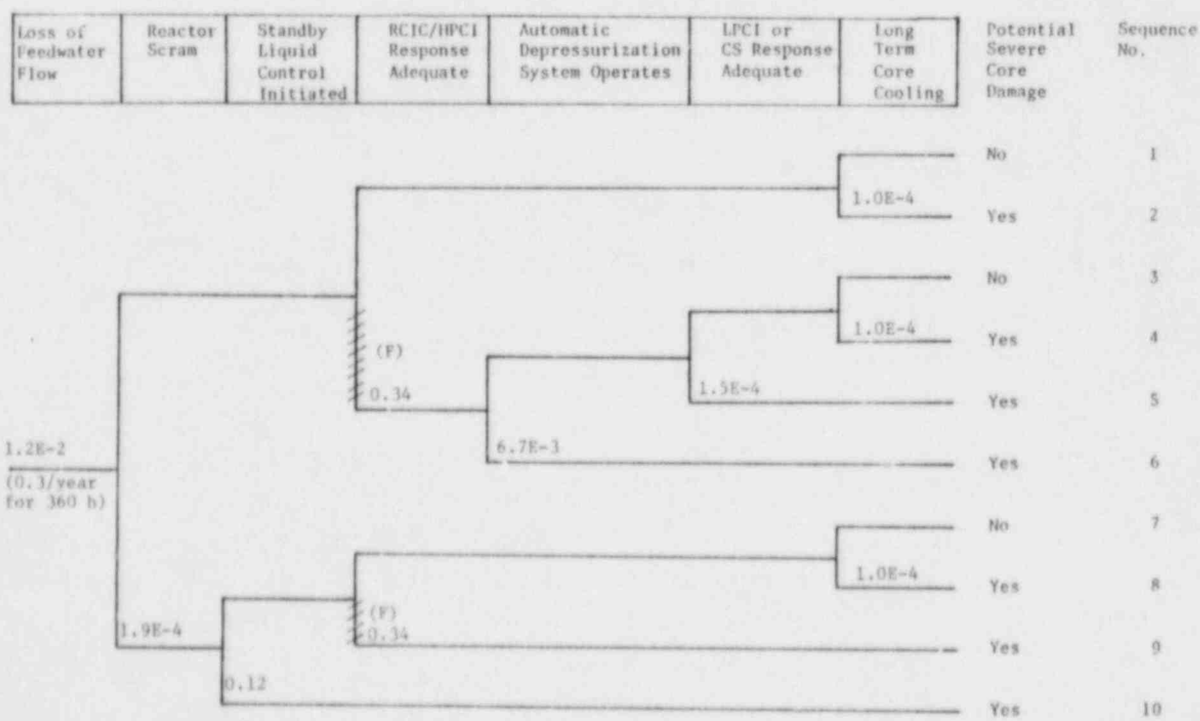
P = 1.4E-7 (includes factor of 0.75 for likelihood of occurring at power)

NSIC 161601 - Sequence of Interest for Loss of Service Water System to Diesel Generators at Salem 1



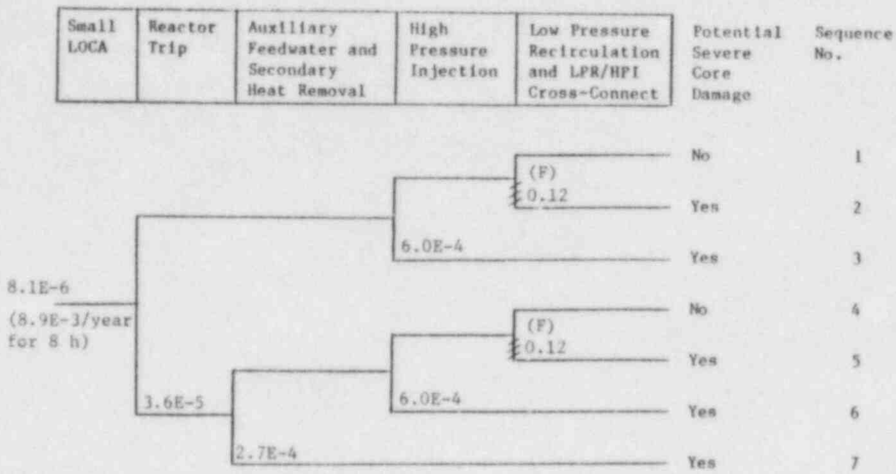
P = 6.7E-8 (includes factor of 0.75 for likelihood of occurring at power)

NSIC 161649 - Sequence of Interest for Loss of Emergency Power System at Sequoyah 1



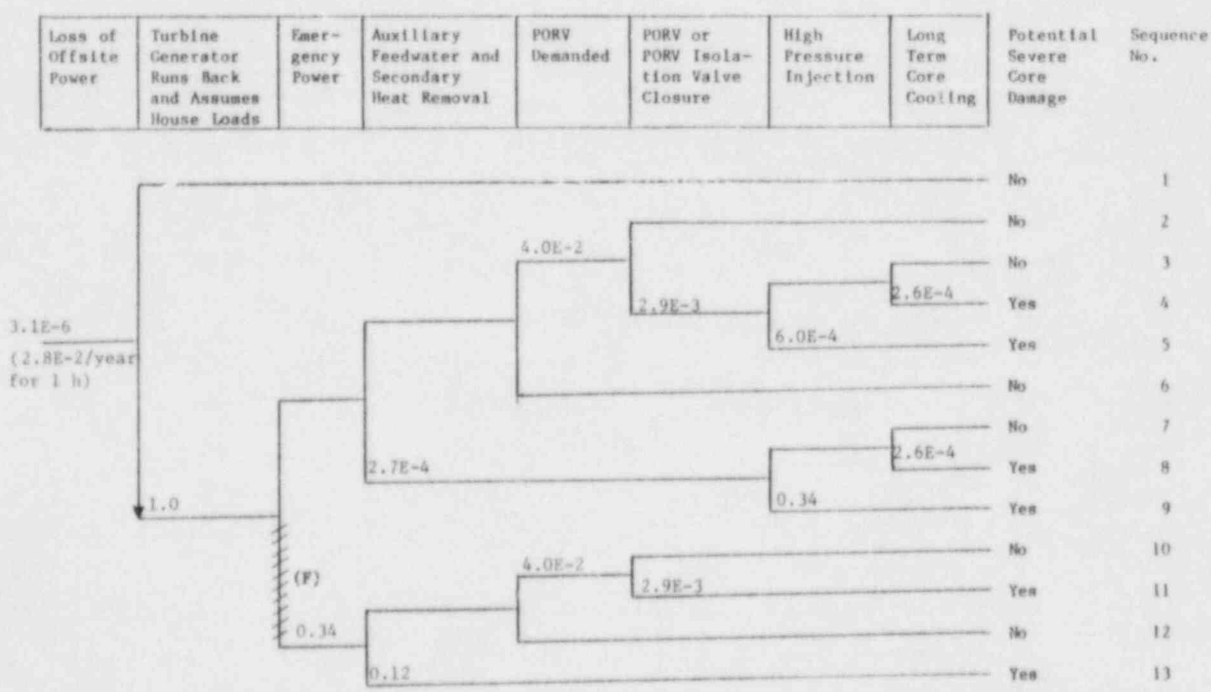
P = 3.2E-5 (includes contribution from other postulated initiators)

NSIC 161906 - Sequence of Interest for RCIC and HPCI Inoperable at Quad Cities 2



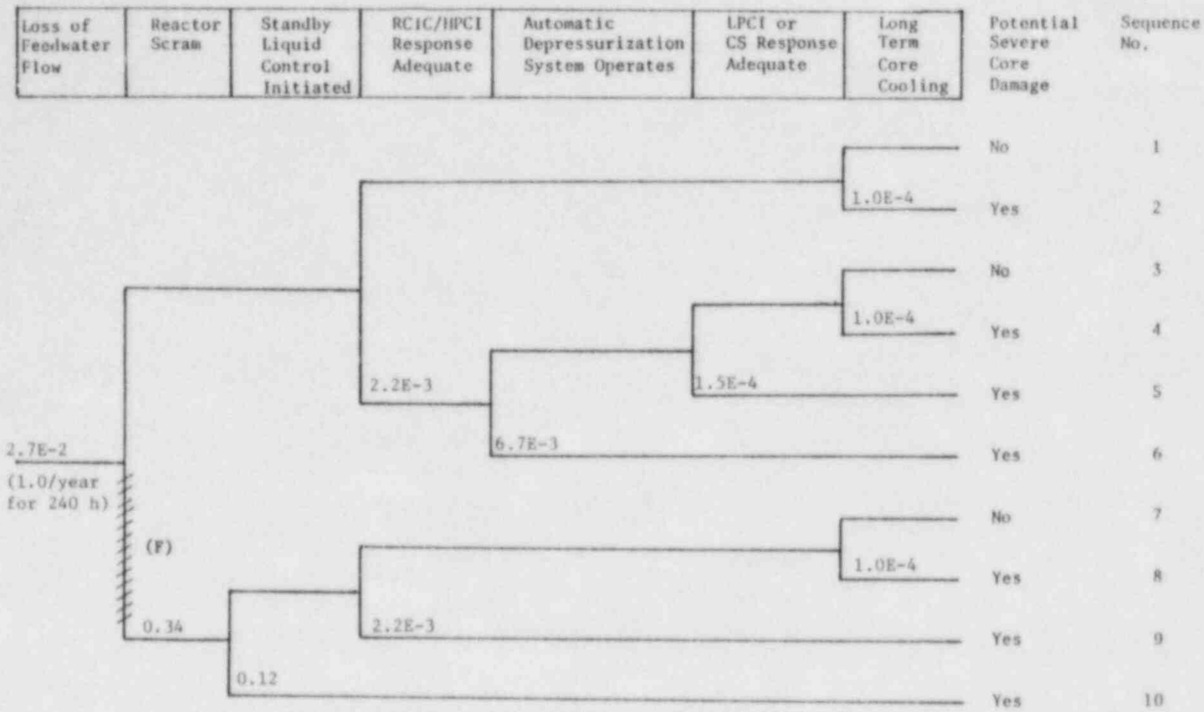
P = 9.8E-7 (includes contribution from other postulated initiators)

NSIC 162083 - Sequence of Interest for Loss of RWT Level Instrumentation at Arkansas Unit 2



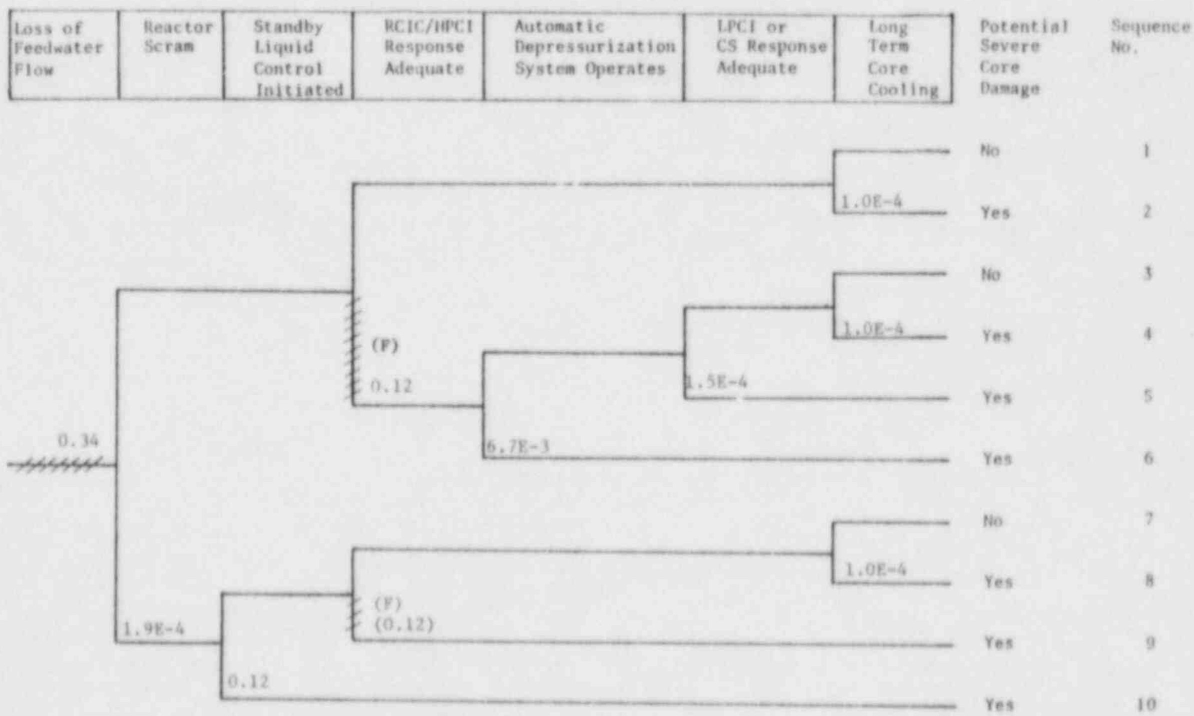
P = 1.3E-7

NSIC 163356 - Sequence of Interest for Output Breakers for Both Station Batteries Opened at Palisades



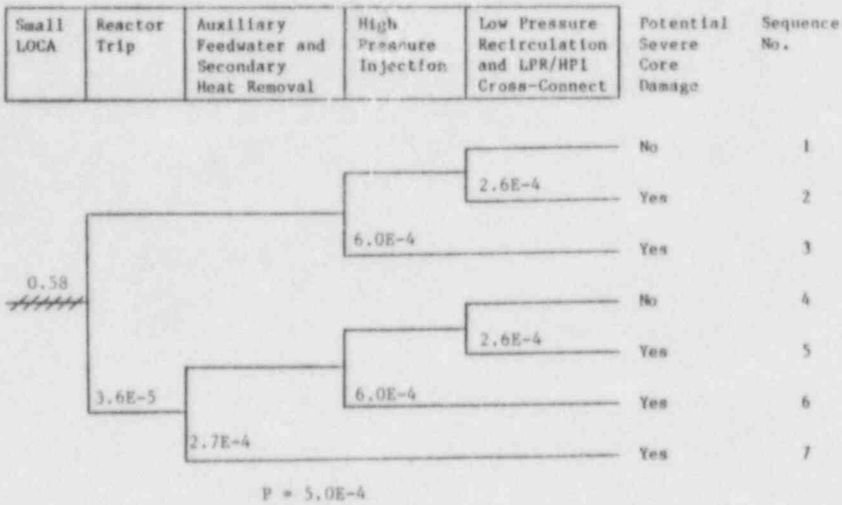
P = 9.8E-4 (includes contribution from other postulated initiators)

NSIC 163405 - Sequence of Interest for Failure of 76 Control Rods to Insert at Browns Ferry 3

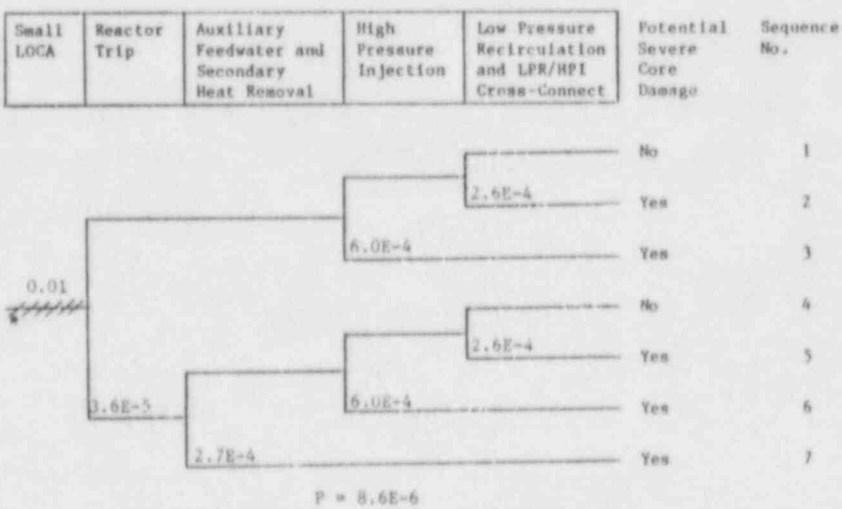


P = 3.3E-4

NSIC 163478 - Sequence of Interest for RCIC and HPCI Failure to Inject Following Scram at Hatch 1

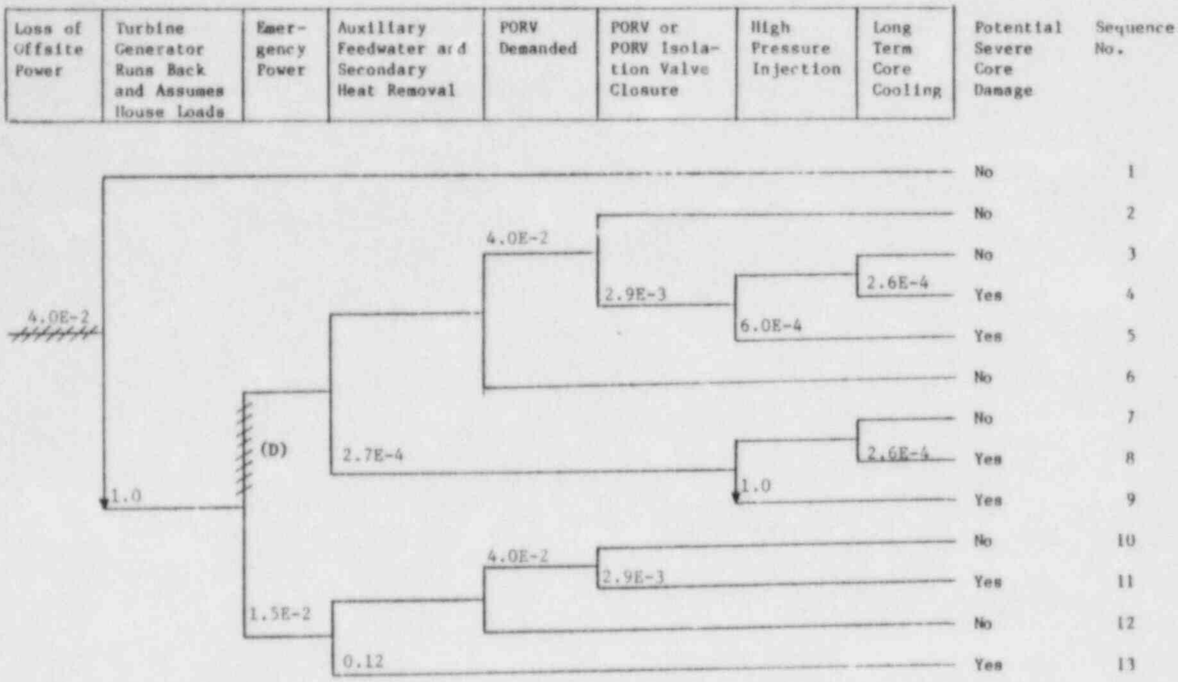


NSIC 163499 - Sequence of Interest for Reactor Coolant Pump Seal Failure at Arkansas Nuclear 1



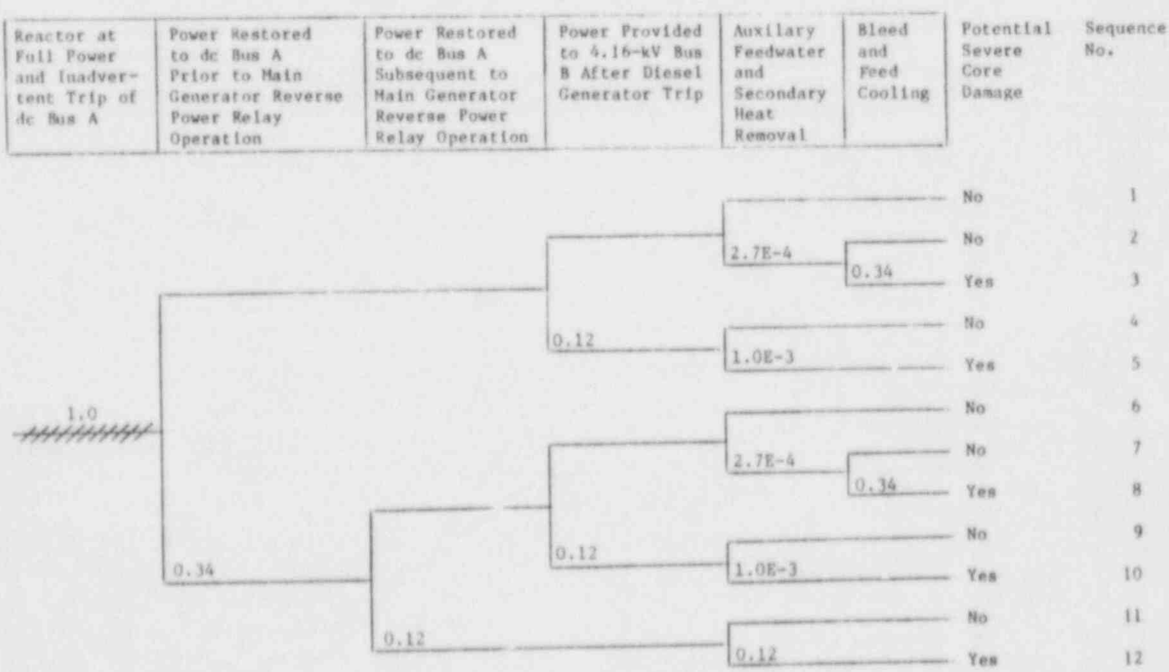
NSIC 164149 - Sequence of Interest for Isolable Small Break LOCA at Robinson 2

*auto isolation on safety injection closure of upstream valves



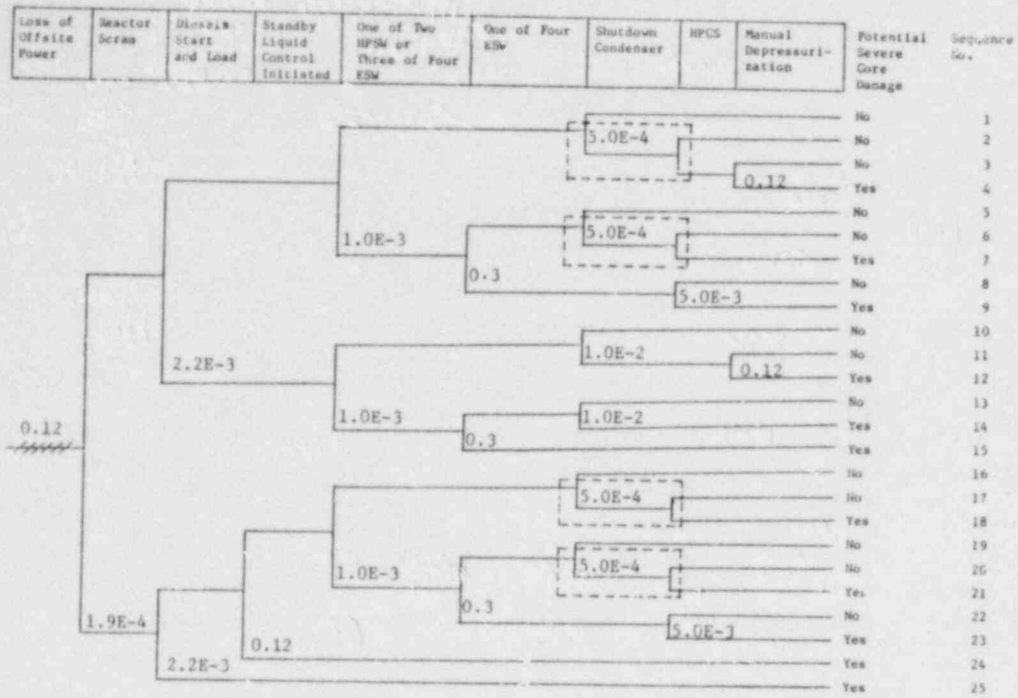
P = 6.1E-5 (includes factor of 0.75 for likelihood of occurring at power)

NSIC 164453 - Sequence of Interest for Brief Loss of Offsite Power and Degraded Load Shed at San Onofre 1



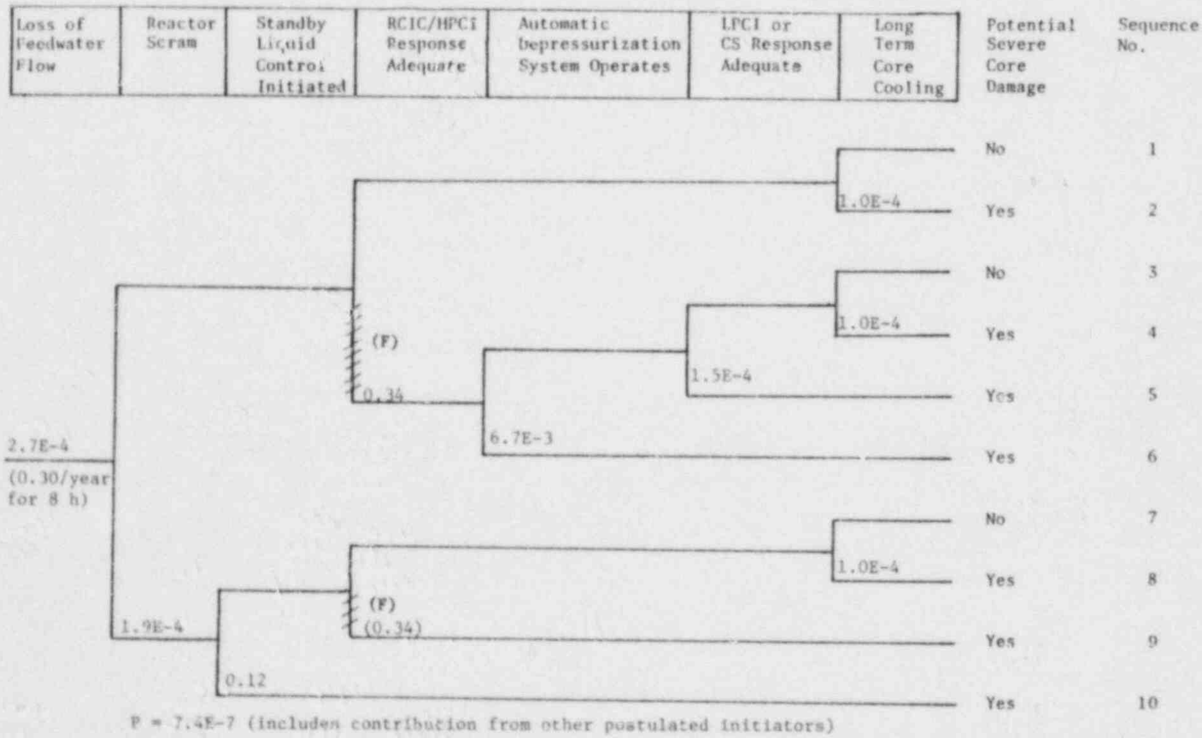
P = 5.1E-3

NSIC 164617 - Sequence of Interest for Loss of dc Bus and Diesel Generator Trip at Millstone 2



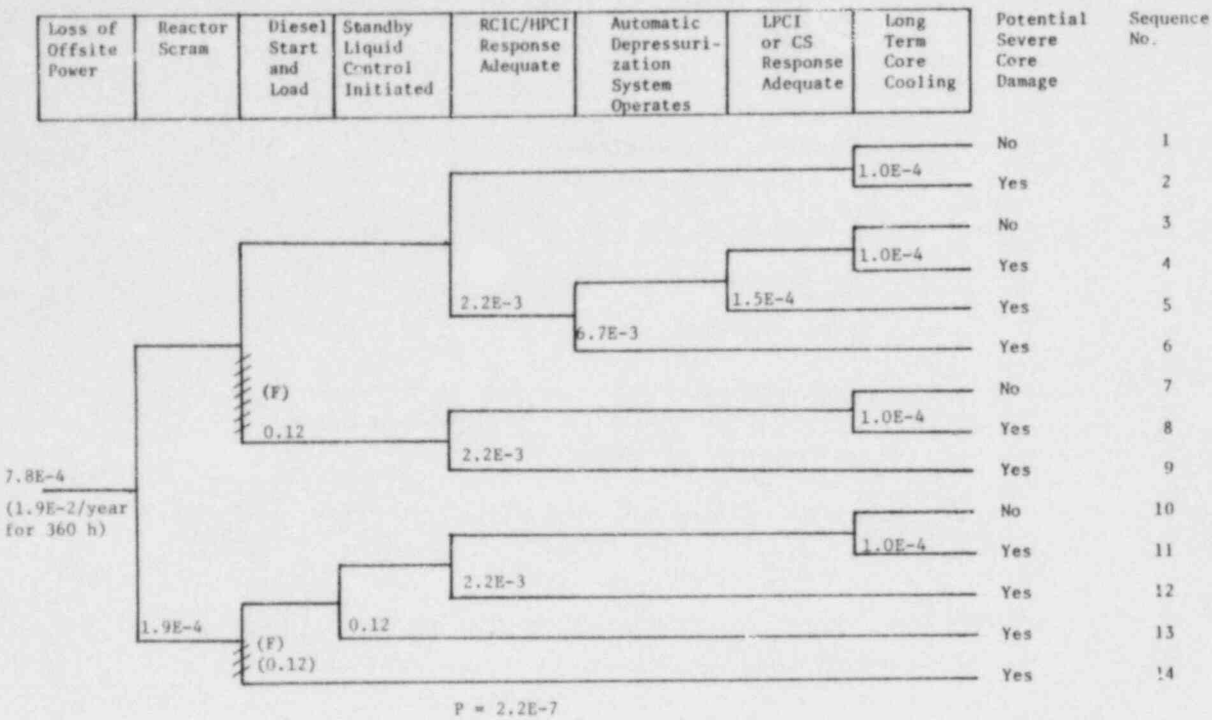
P = 1.0E-5

NSIC 164703 - Sequence of Interest for LOOP at La Crosse

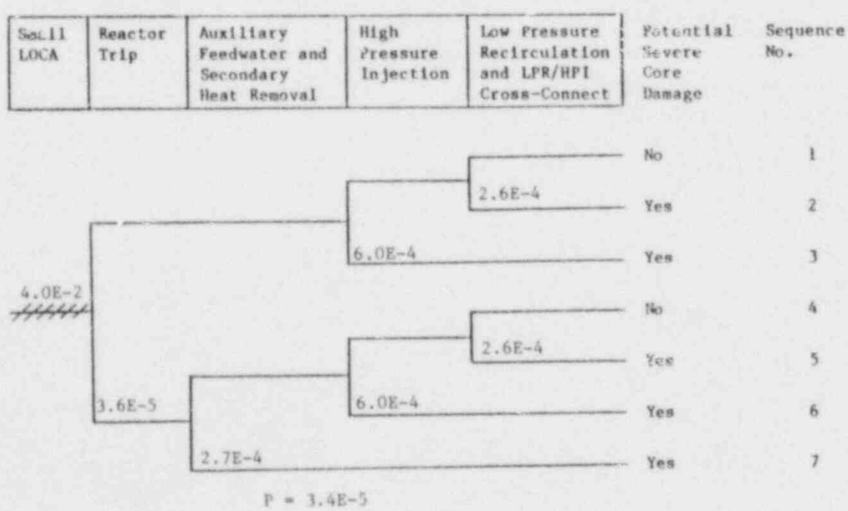


P = 7.4E-7 (includes contribution from other postulated initiators)

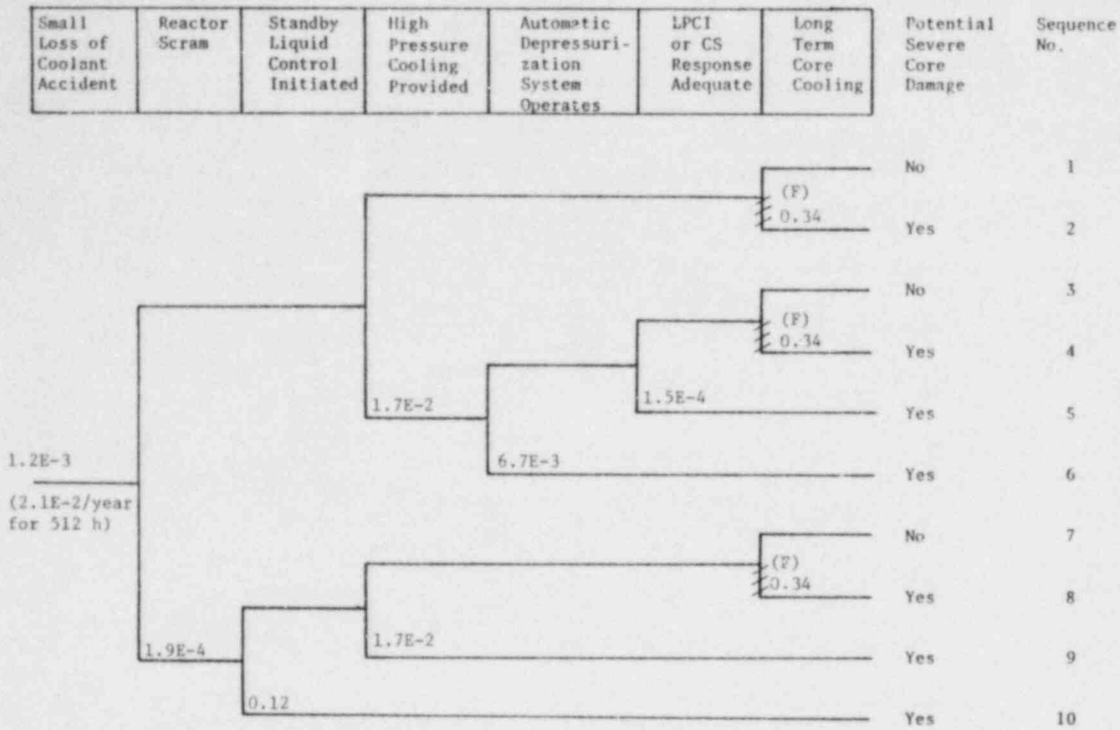
NSIC 164955 - Sequence of Interest for Loss of RCIC at HPCI at Hatch 1



NSIC 165438 - Sequence of Interest for Unavailability of Emergency Power at Hatch 1

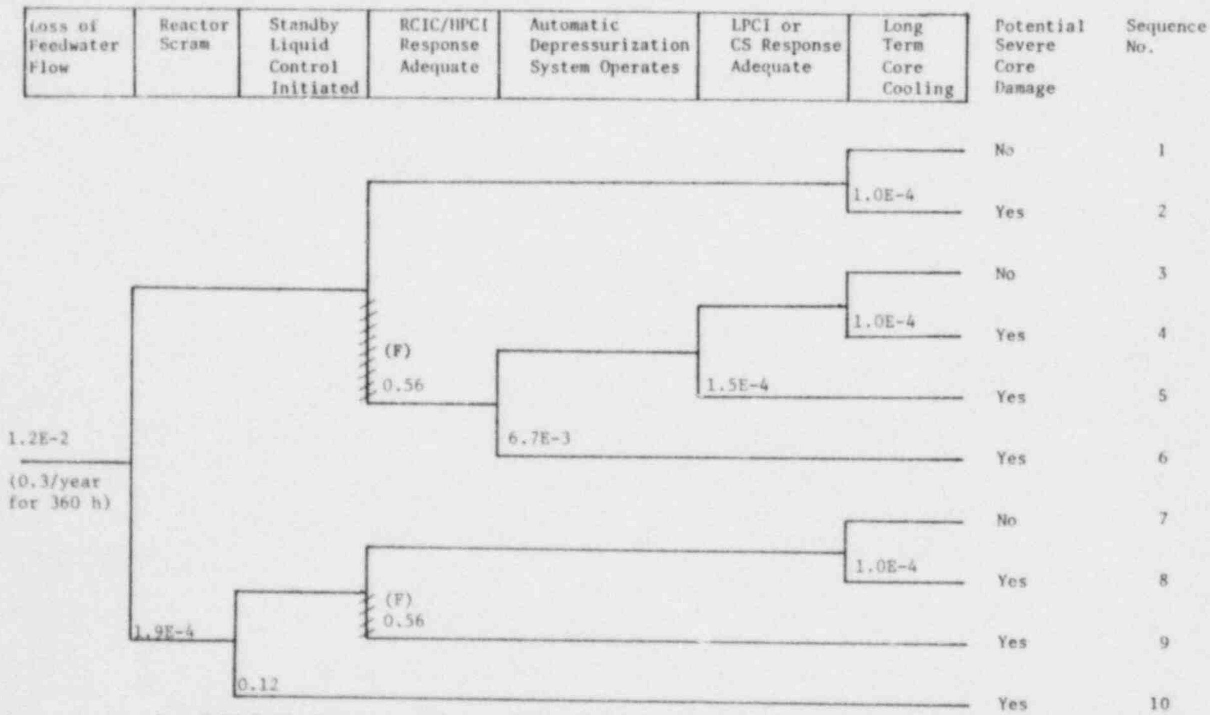


NSIC 165900 - Sequence of Interest for Inadvertent Opening of Pressurizer Relief Valve and Block Valve at Haddam Neck



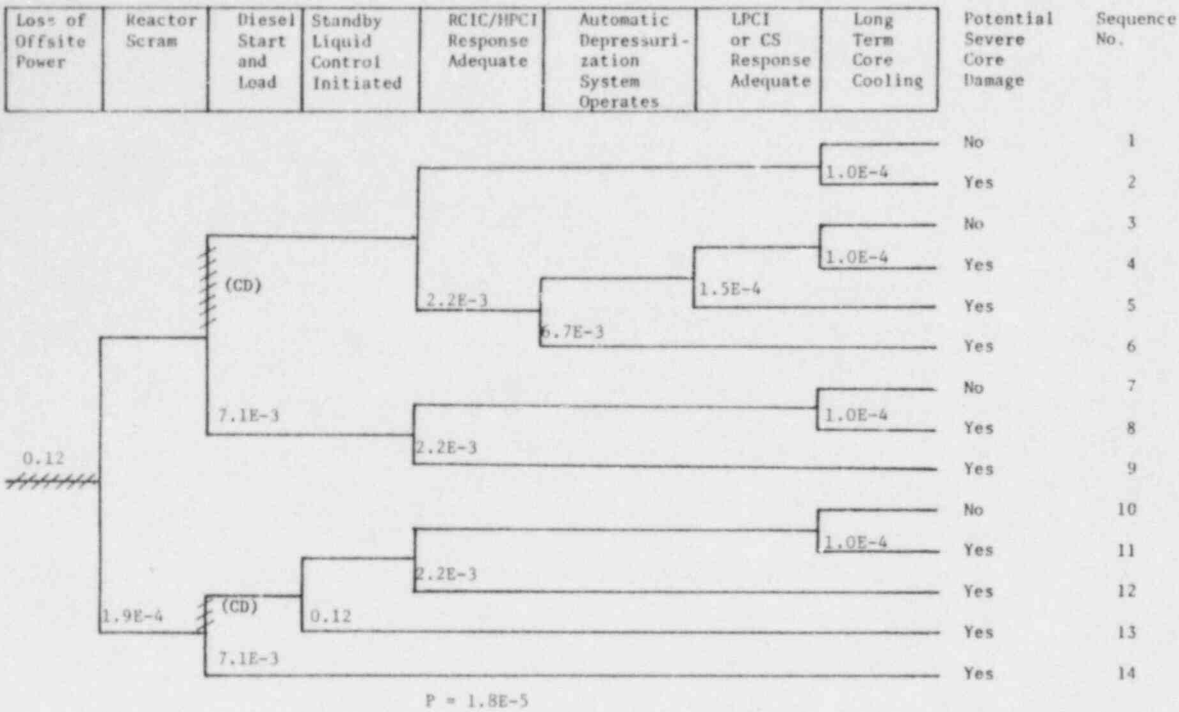
P = 6.7E-3 (includes contribution from other postulated initiators)

NSIC 166072 - Sequence of Interest for Loss of RHR at Brunswick 1

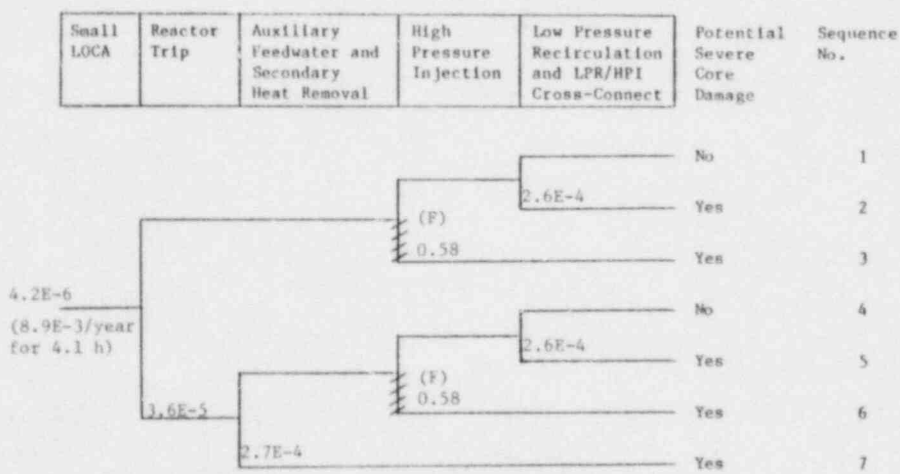


P = 5.6E-5 (includes contribution from other postulated initiators)

NSIC 166082 - Sequence of Interest for Loss of RCIC and HPCI at Brunswick 2

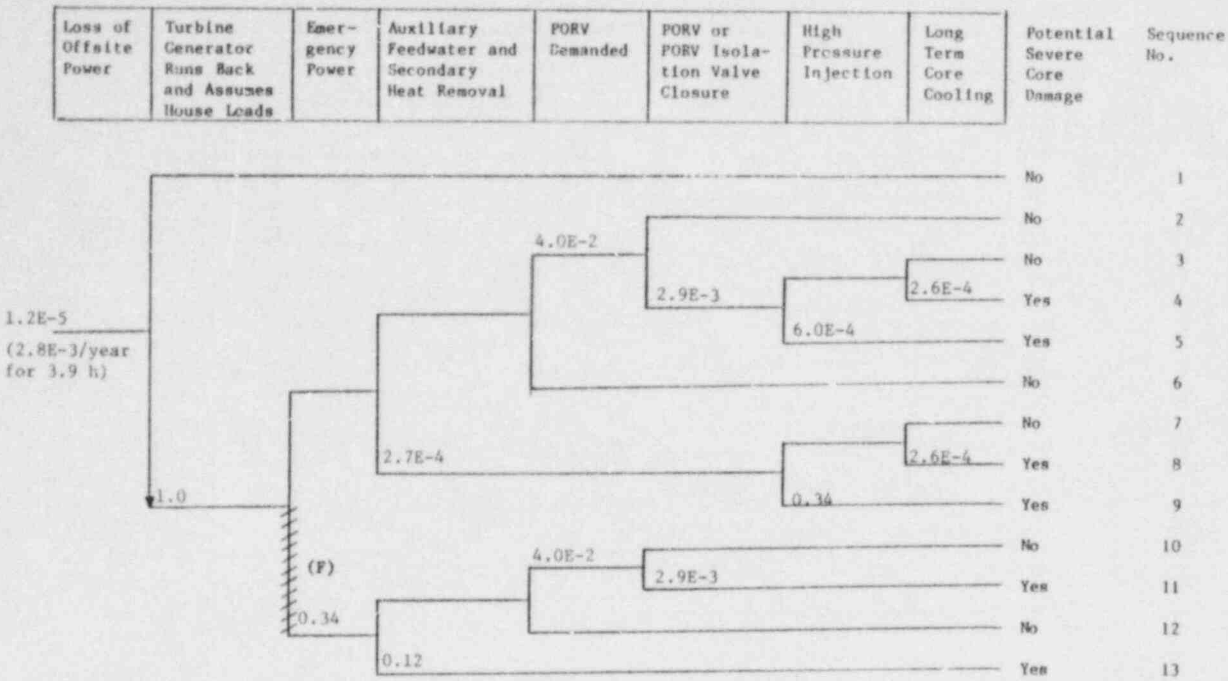


NSIC 166384 - Sequence of Interest for LOOP at Monticello



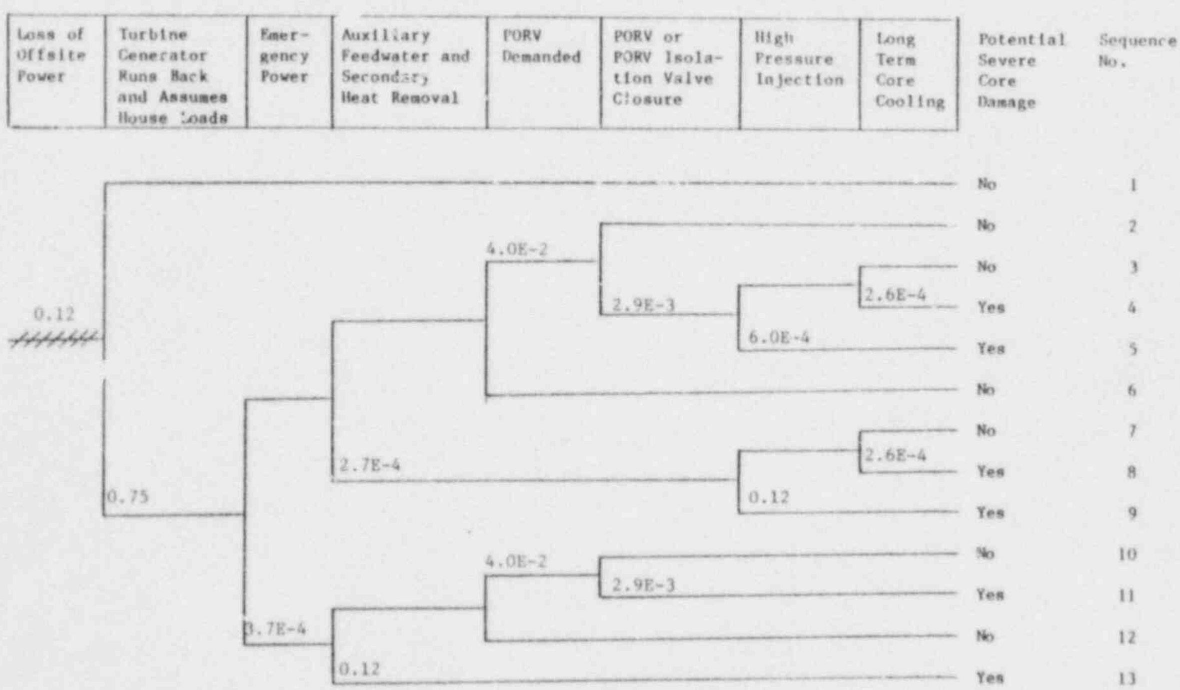
P = 2.4E-6 (includes contribution from other postulated initiators)

NSIC 166650 - Sequence of Interest for Safety Injection Supply Valve Found Unlocked and Closed at Beaver Valley



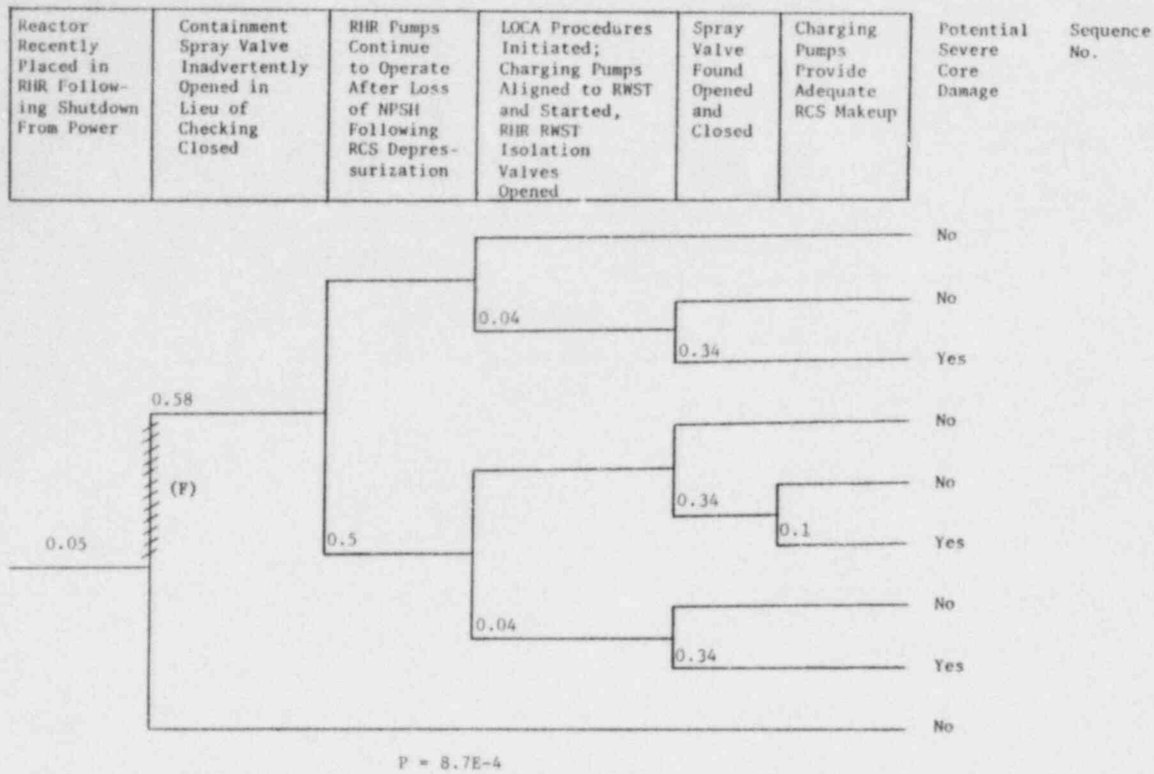
P = 5.0E-7

NSIC 166745 - Sequence of Interest for Two Shutdown Sequencers and Diesel Generator Unavailable at Palisades

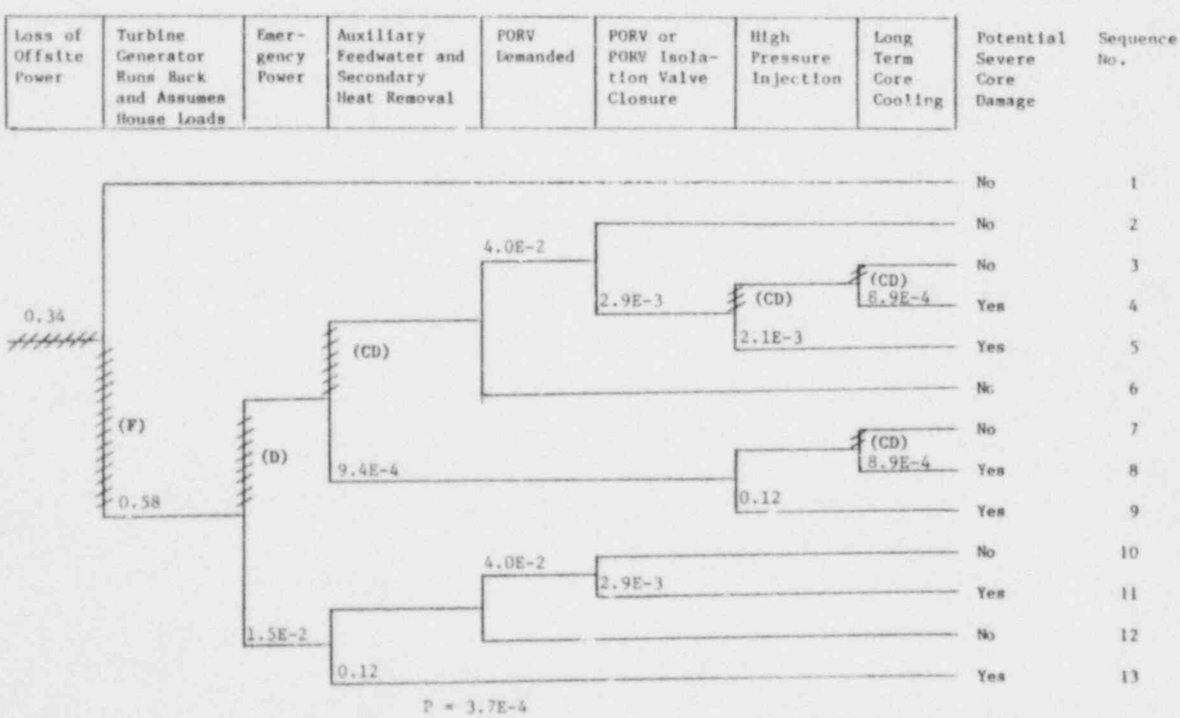


P = 5.2E-6 (includes factor of 0.75 for likelihood of occurring at power)

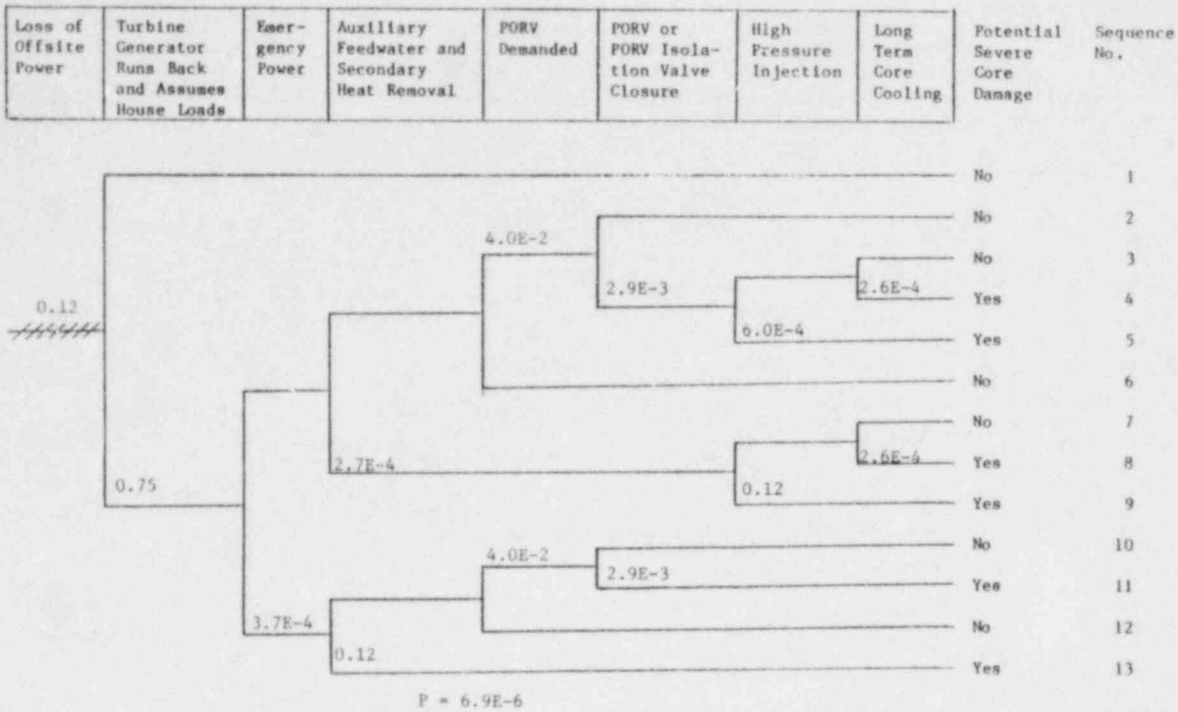
NSIC 167117 - Sequence of Interest for Effective LOOP at Rancho Seco



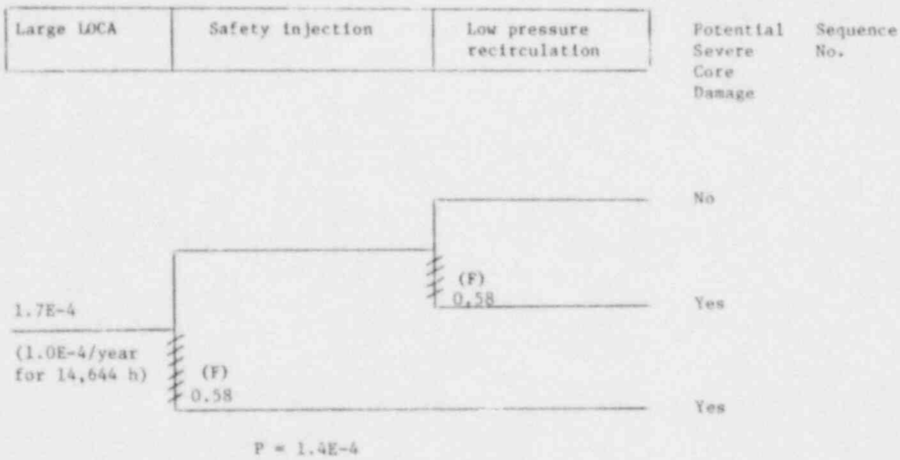
NSIC 167611 - Sequence of Interest for Inadvertent Spray Initiation and Draining of Reactor Coolant System at Sequoyah 1



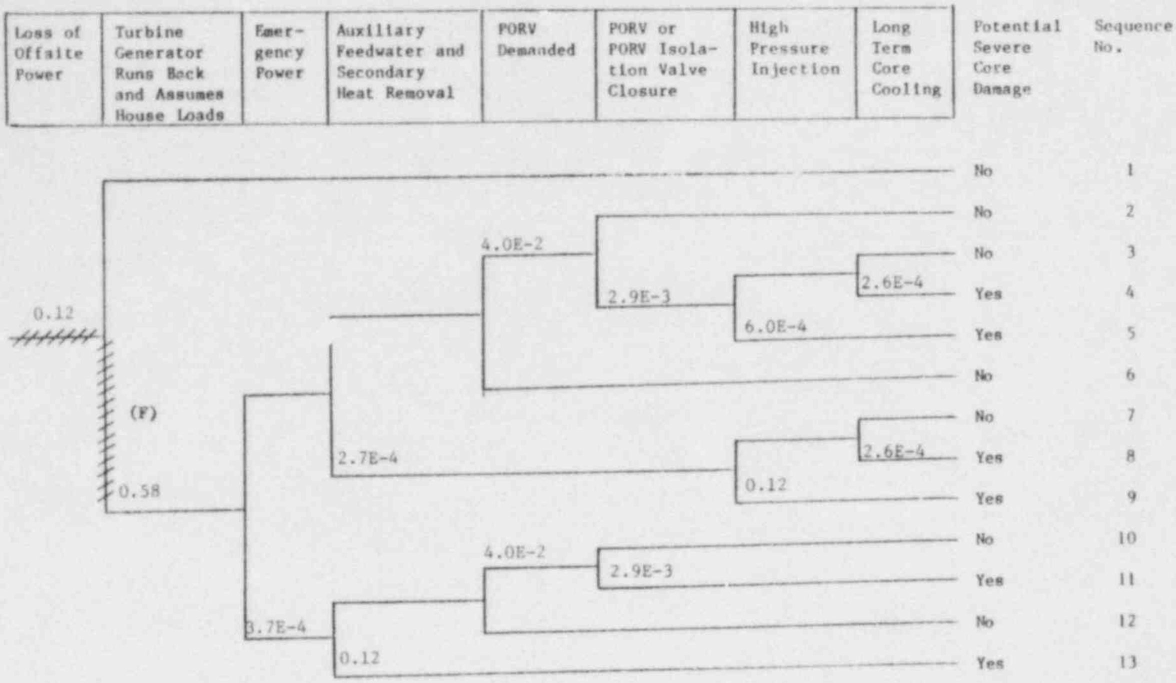
NSIC 167624 - Sequence of Interest for Loss of Offsite Power and Diesel Generator Failure to Start at Crystal River 3



NSIC 168548 - Sequence of Interest for Effective LOOP at Rancho Seco

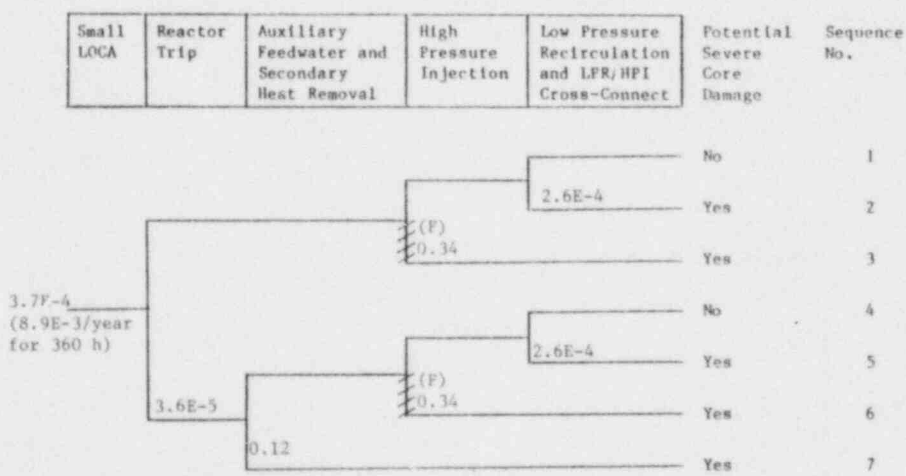


NSIC 168829 - Sequence of Interest for Both Safety Injection Trains Inoperable Due to Valve Problems at San Onofre 1



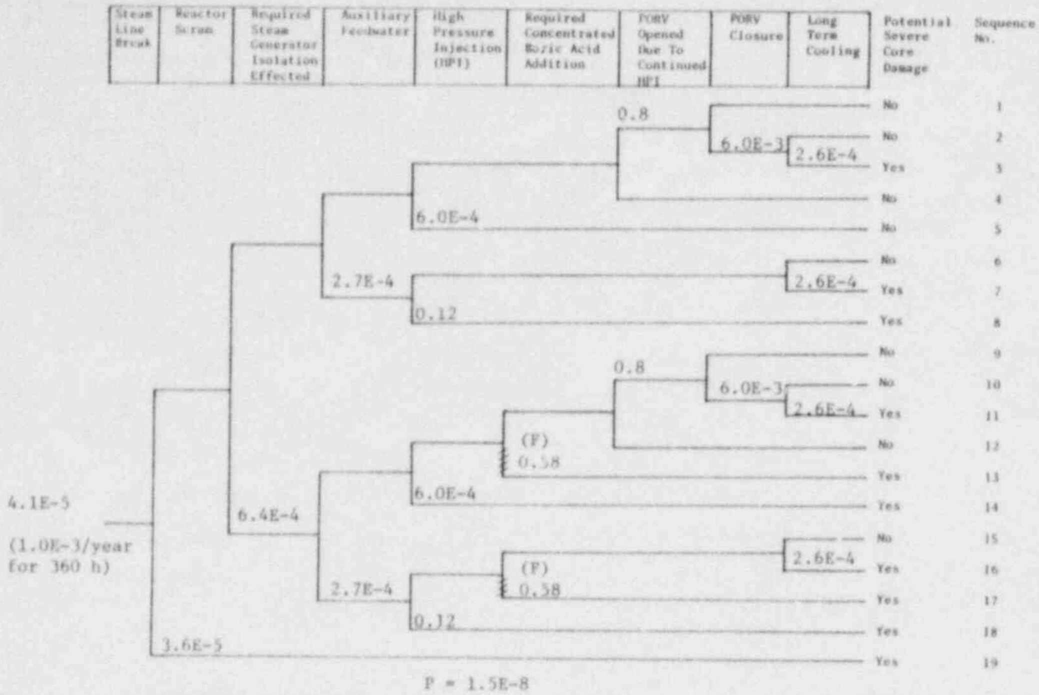
P = 5.4E-6

NSIC 169042 - Sequence of Interest for Degraded HPI System During LOOP at Arkansas Unit 1

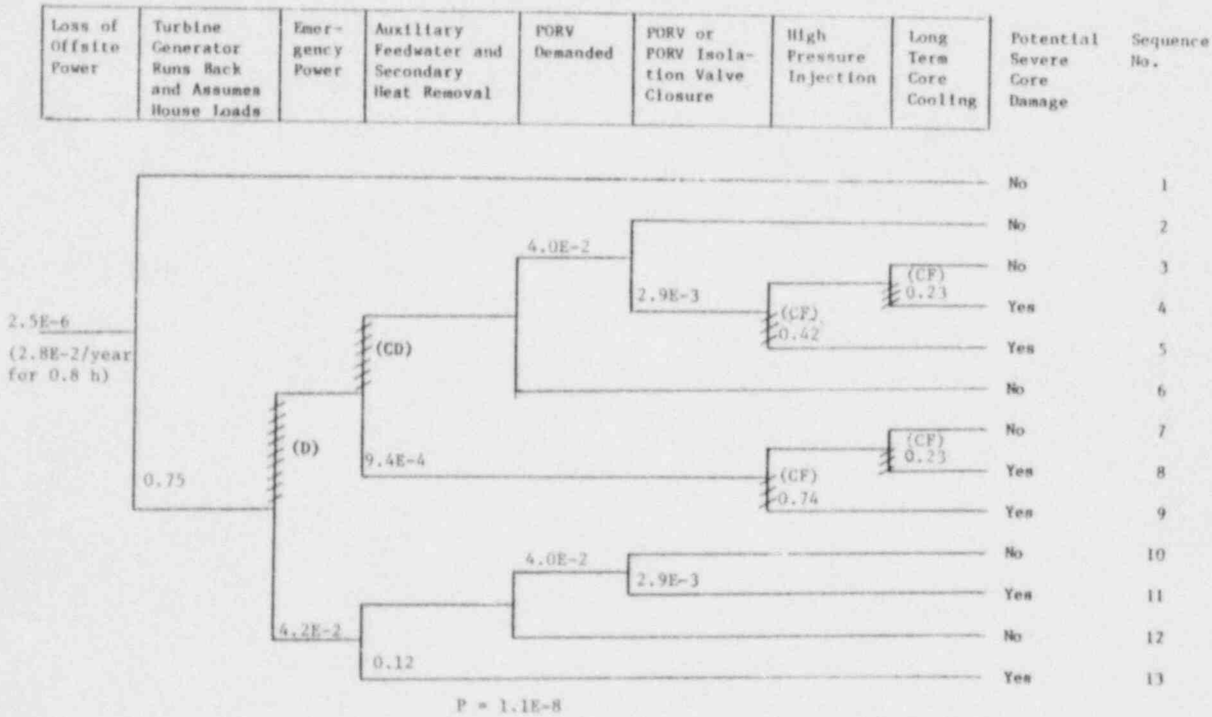


P = 9.3E-5 (includes contribution from other postulated initiators and a factor of 0.75 for the likelihood of occurring at power)

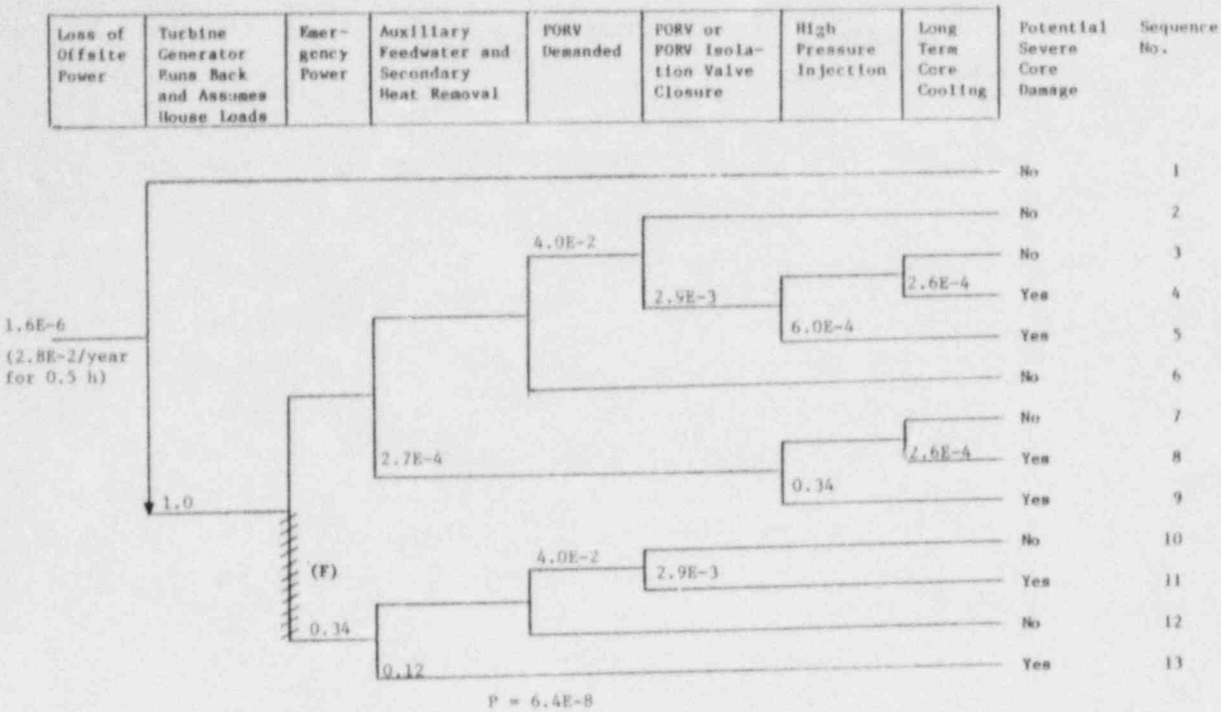
NSIC 169587 - Sequence of Interest for Safety Injection Path to RCS Obstructed at Turkey Point 4



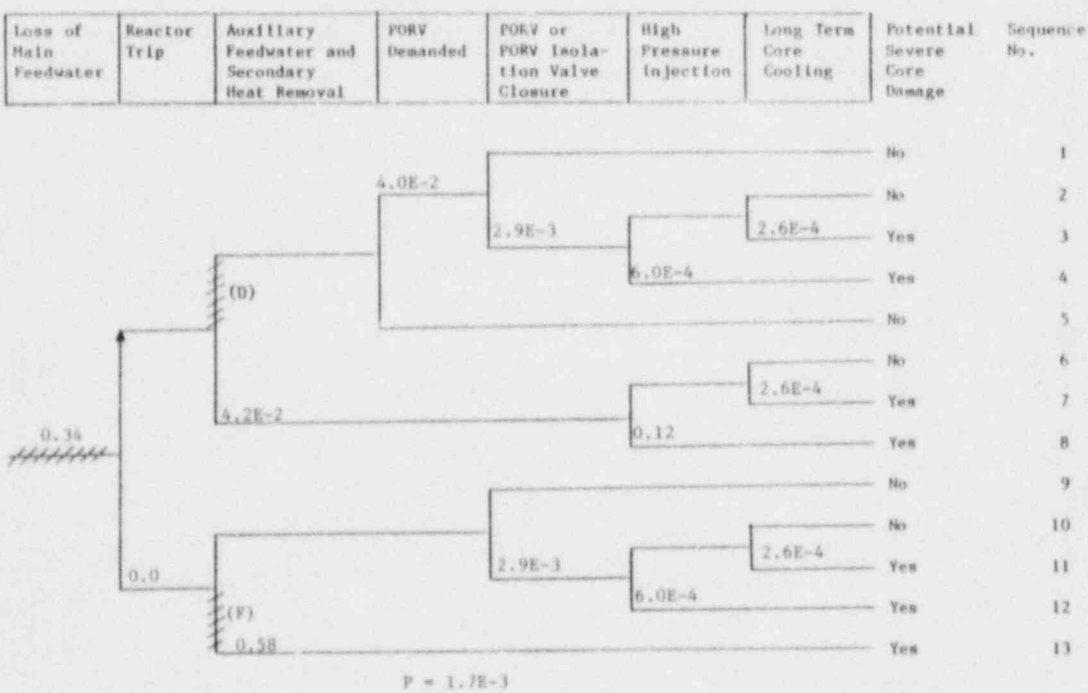
NSIC 170098 - Sequence of Interest for Two Boron Injection Tank Inlet Valves Fail to Open at Salem 1



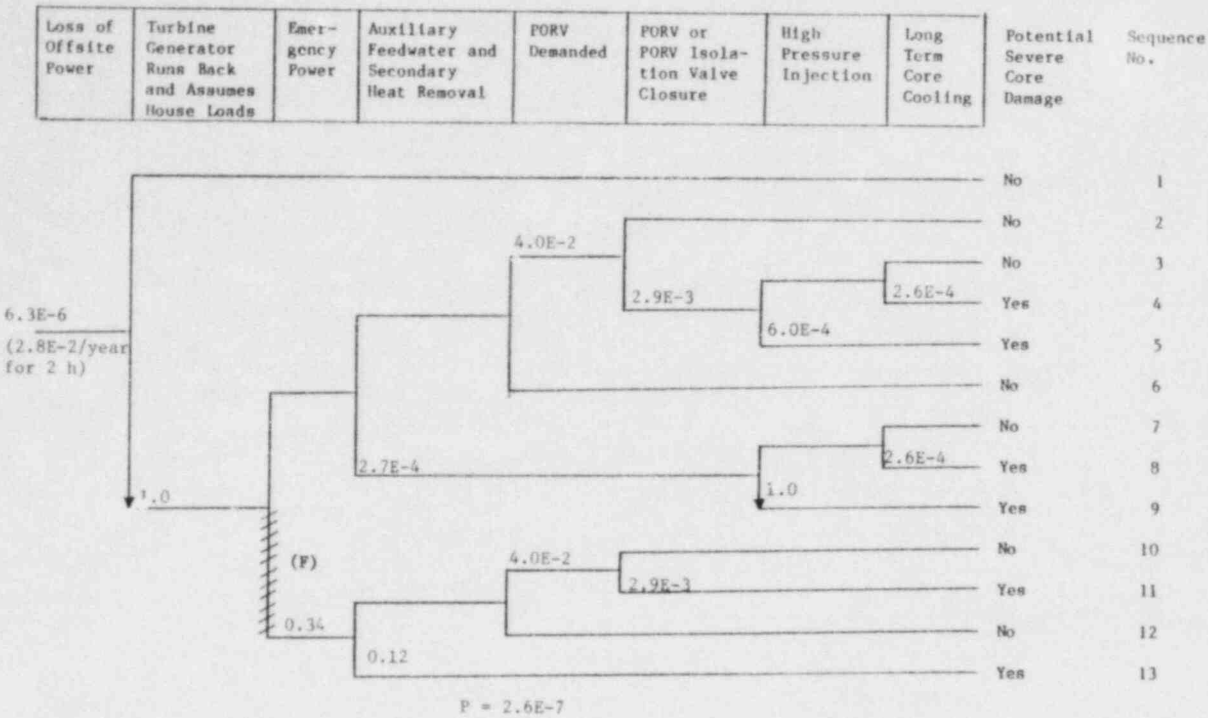
NSIC 170199 - Sequence of Interest for Unavailability of Diesel Generator and Component Cooling Water Train at Kewaunee



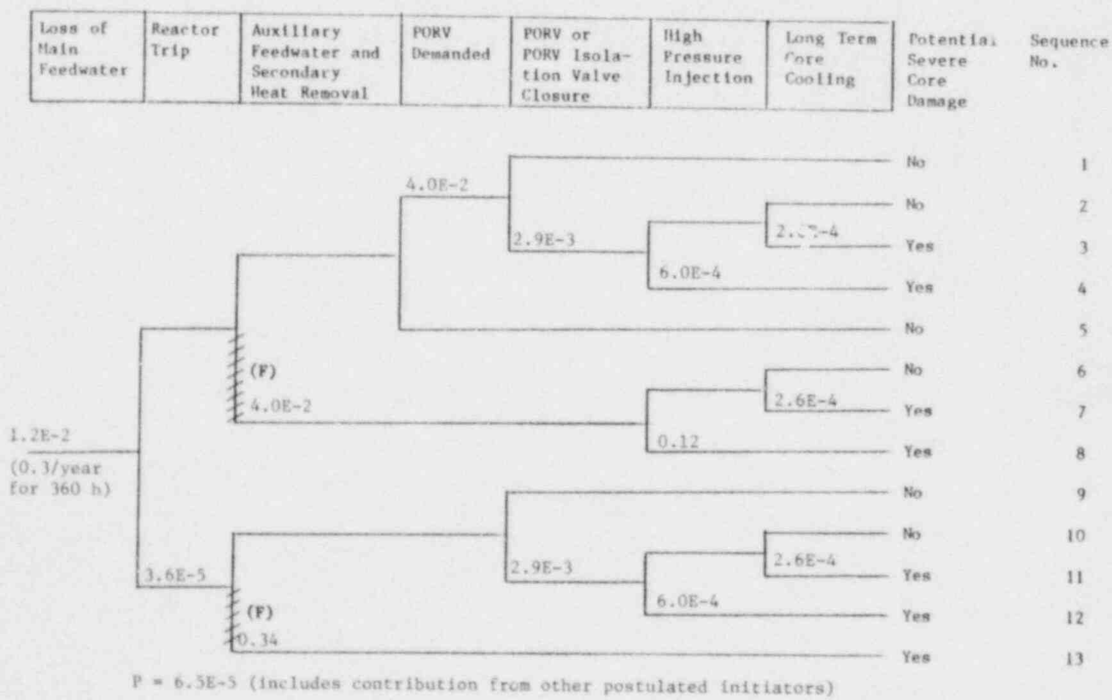
NSIC 171202 - Sequence of Interest for Unavailability of Emergency Power System at Turkey Point 3



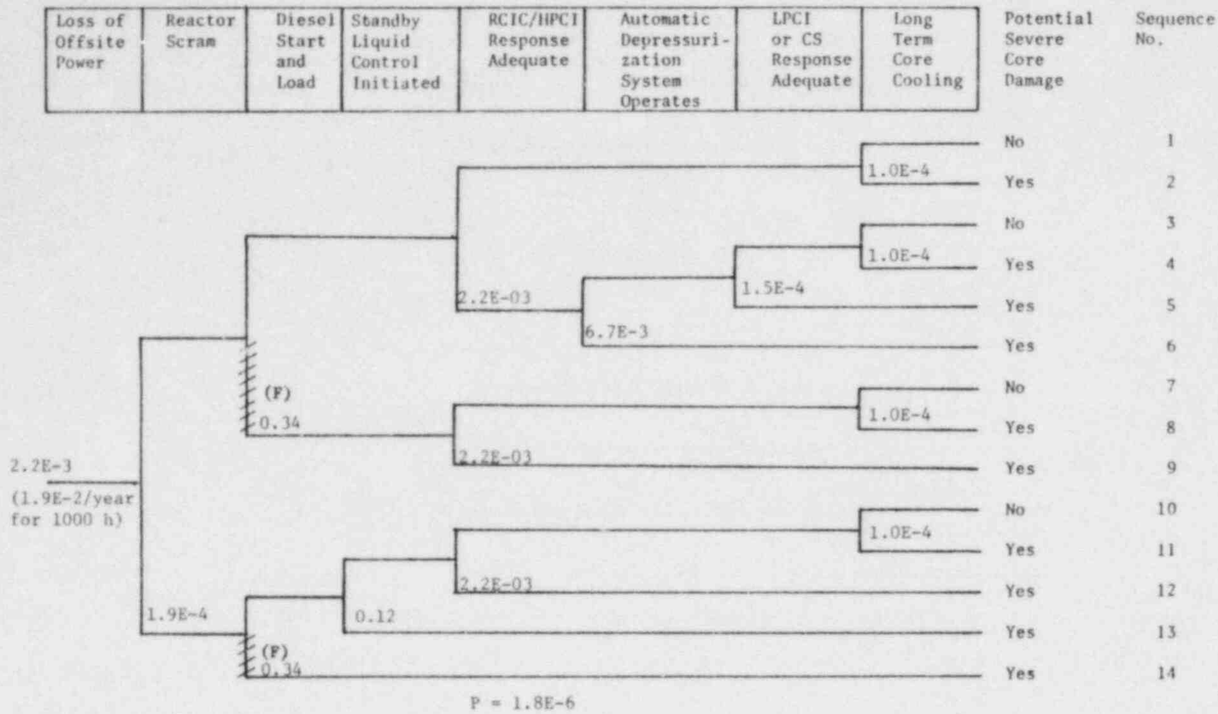
NSIC 171667 - Sequence of Interest for Loss of Two Instrument Buses at Davis-Besse 1



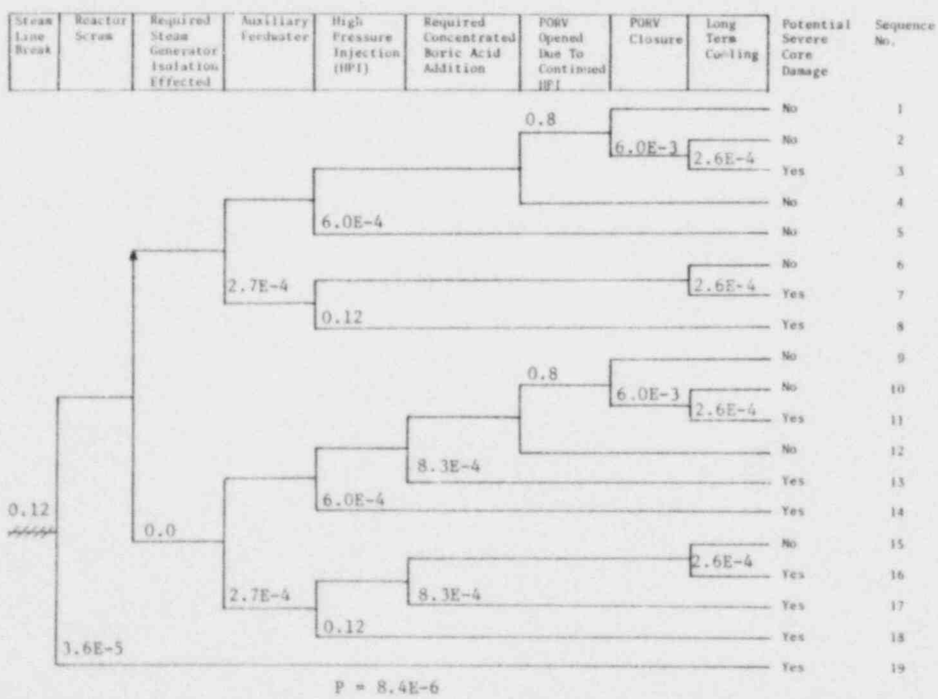
NSIC 171700 - Sequence of Interest for Unavailability of Emergency Power System at San Onofre 1



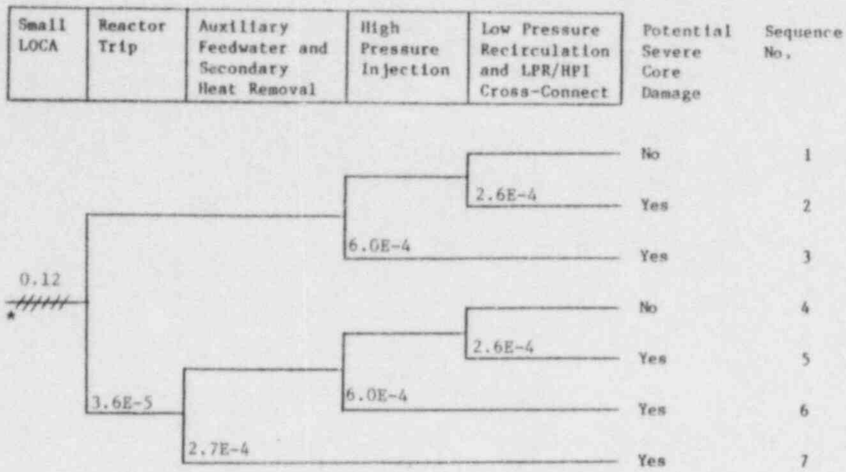
NSIC 171733 - Sequence of Interest for Unavailability of Three Auxiliary Feedwater Pumps at Zion 2



NSIC 171842 - Sequence of Interest for Failure of Diesel Generators at Dresden 3



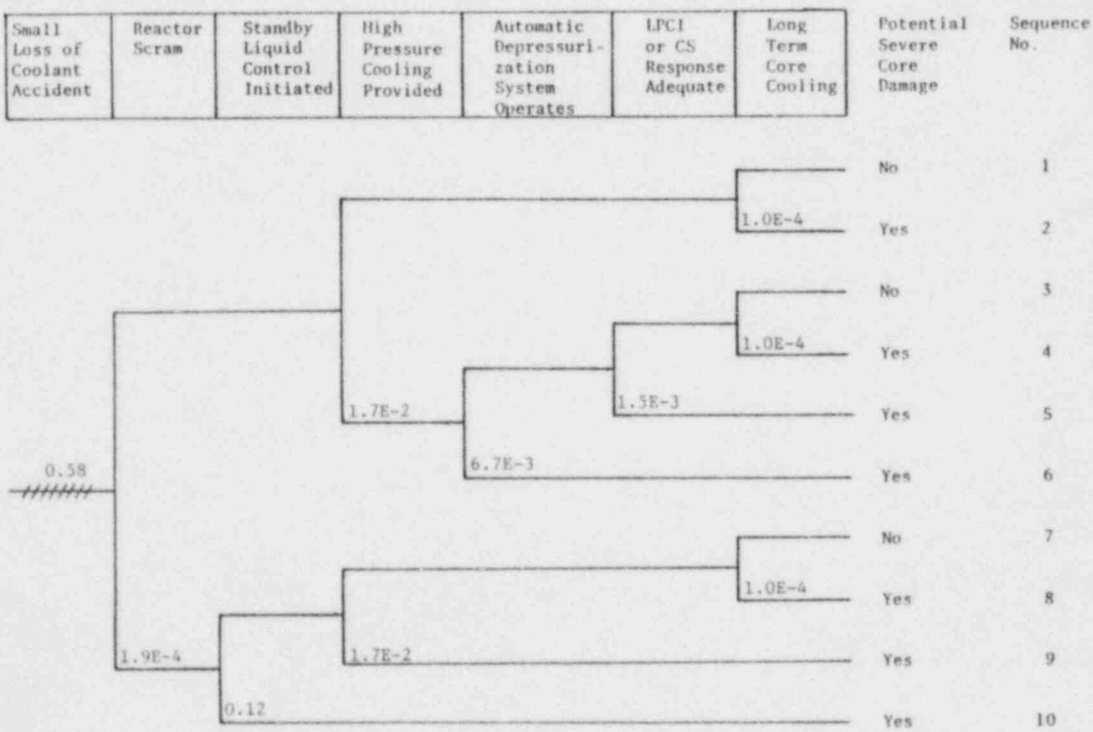
NSIC 171939 - Sequence of Interest for Inadvertent Steam Dump Valve Opening and Safety Injection at North Anna 2



P = 7.7E-5 (includes factor of 0.75 for likelihood of occurring at power)

NSIC 172198 - Sequence of Interest for Inadvertent Pressurizer Code Safety Valve Opening at St. Lucie 1

***reseating of safety valve V1200 would provide mitigation**



P = 1.4E-5

NSIC 174073 - Sequence of Interest Tree for Stuck Open Relief Valve at Duane Arnold

Appendix E

SENSITIVITY ANALYSIS

This appendix provides supporting event rankings and conditional probability distributions for the sensitivity analysis described in Sect. 6.3.

Each of the following figures contains the reference conditional event probability ranking and distribution plots based on the mean value—conditional event probabilities (denoted by the solid line) as well as a ranking and distribution for the minimum and maximum value of the parameter of interest. Table E.1 provides a listing of the variables; the minimum, maximum, and mean values; and the applicable figure numbers, in order of presentation.

Table E.1. ASP variables, values, and applicable Appendix E figures

Variable under consideration[a]	Value			Figure No.
	Min	Max	Mean	
Recovery classes				
R1	0.3	1.0	0.58	E.1
R2	0.1	0.8	0.34	E.2
R3	0.03	0.3	0.12	E.3
R4	0.01	0.1	0.04	E.4
Plant-specific tailoring factors				
P2	0.1	0.8	0.34	E.5
P3	0.03	0.3	0.12	E.6
P4	0.01	0.1	0.04	E.7
β factor	0.03	0.3	0.12	E.8
BWR functions				
HPCI/RCIC	1.04E-4	1.04E-2	2.24E-3	E.9
High-pressure cooling provided following LOCA	3.0E-3	5.0E-2	1.70E-2	E.10
Emergency power	9.38E-5	1.06E-2	2.22E-3	E.11
Automatic depressurization	1.94E-7	8.68E-2	6.67E-3	E.12
SBLC only	0.03	0.3	0.12	E.13
Scram only	4.45E-7	1.52E-3	1.87E-4	E.14
LPCI/core spray	6.79E-8	1.50E-3	1.50E-4	E.15
Reactor isolation (large SLB)	2.78E-7	2.67E-2	2.33E-3	E.16
Long-term core cooling	1.71E-6	5.98E-4	1.02E-4	E.17
PWR functions				
Reactor trip	~0.0	1.50E-3	3.60E-5	E.18
AFW given reactor trip success	5.84E-5	7.57E-4	2.73E-4	E.19
AFW given reactor trip failure	1.24E-6	2.73E-2	2.73E-3	E.20
PORV demanded	0.01	0.1	4.00E-2	E.21
PORV closure given SLB	3.59E-4	0.01	2.90E-3	E.22
HPI given AFW success	2.01E-5	3.04E-3	6.03E-4	E.23
Long-term core cooling	1.20E-5	1.20E-3	2.57E-4	E.24
Emergency power	9.34E-6	1.98E-3	3.68E-4	E.25
SG isolation (large SLB)	2.23E-5	3.18E-3	6.37E-4	E.26
Concentrated boric acid addition given HPI success (large SLB)	2.30E-5	4.35E-3	8.27E-4	E.27
PORV opened due to HPI (large SLB)	0.64	1.0	8.00E-1	E.28
PORV closure given SLB	8.0E-4	0.02	6.00E-3	E.29
BWR initiators				
LOFW	0.041	1.0	0.3	E.30
LOOP	5.16E-3	4.72E-2	1.9E-2	E.31
LOCA	7.46E-3	4.60E-2	2.12E-2	E.32
Large SLB	4.53E-7	1.0E-2	1.0E-3	E.33
PWR initiators				
LOFW	0.041	1.0	0.30	E.34
LOOP	9.02E-3	6.19E-2	2.75E-2	E.35
LOCA	3.90E-3	1.70E-2	8.90E-3	E.36

[a]Failures of PWR feed and bleed, turbine generator runback, and AFW given emergency power are not considered here individually because they are uniformly plant tailored. See applicable figures for effects of variation in plant-specific tailoring factors.

ORNL–DWG 84–4217 ETD

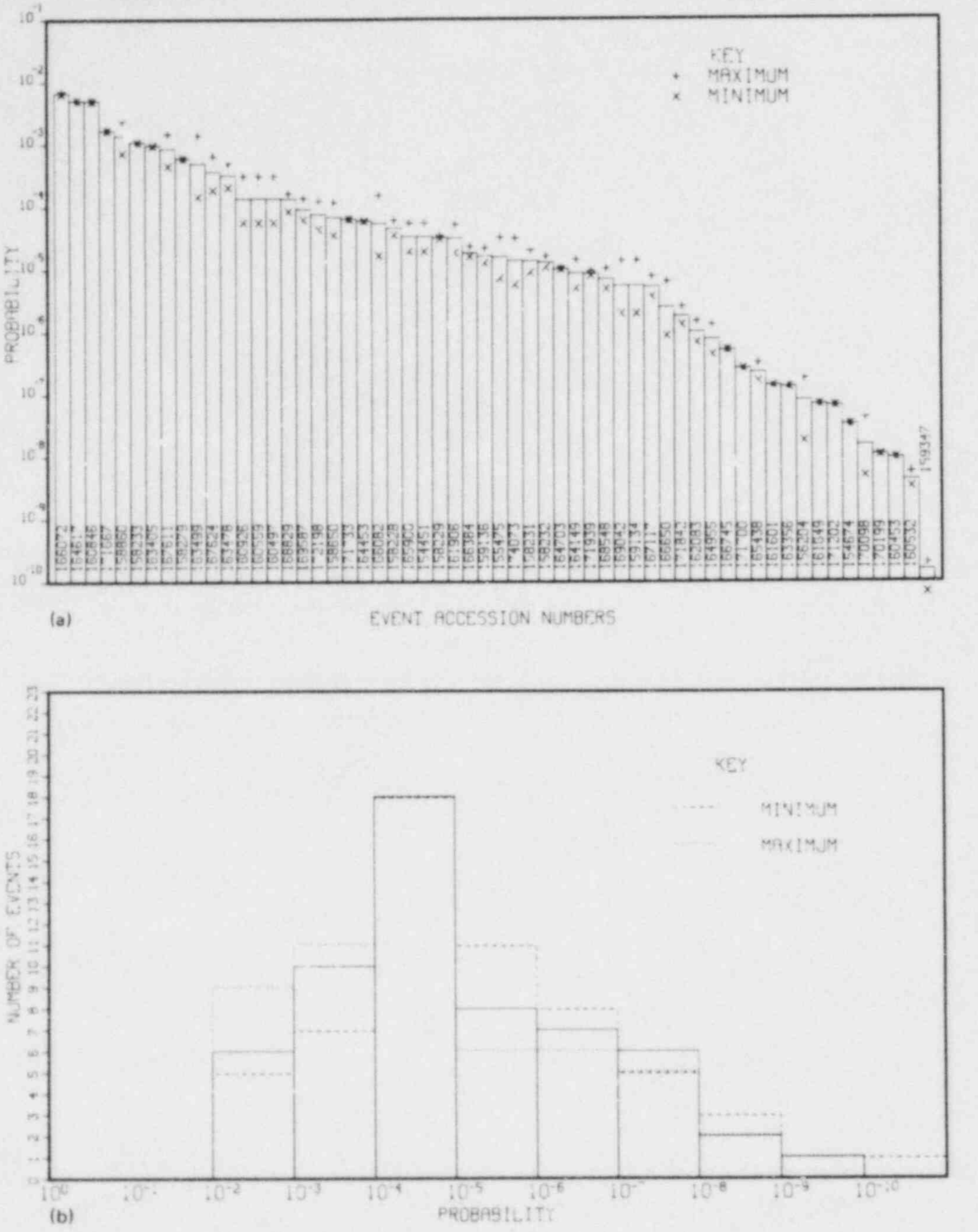


Fig. E.1. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in recovery class R1 numeric values.

ORNL-DWG 84-4182 ETD

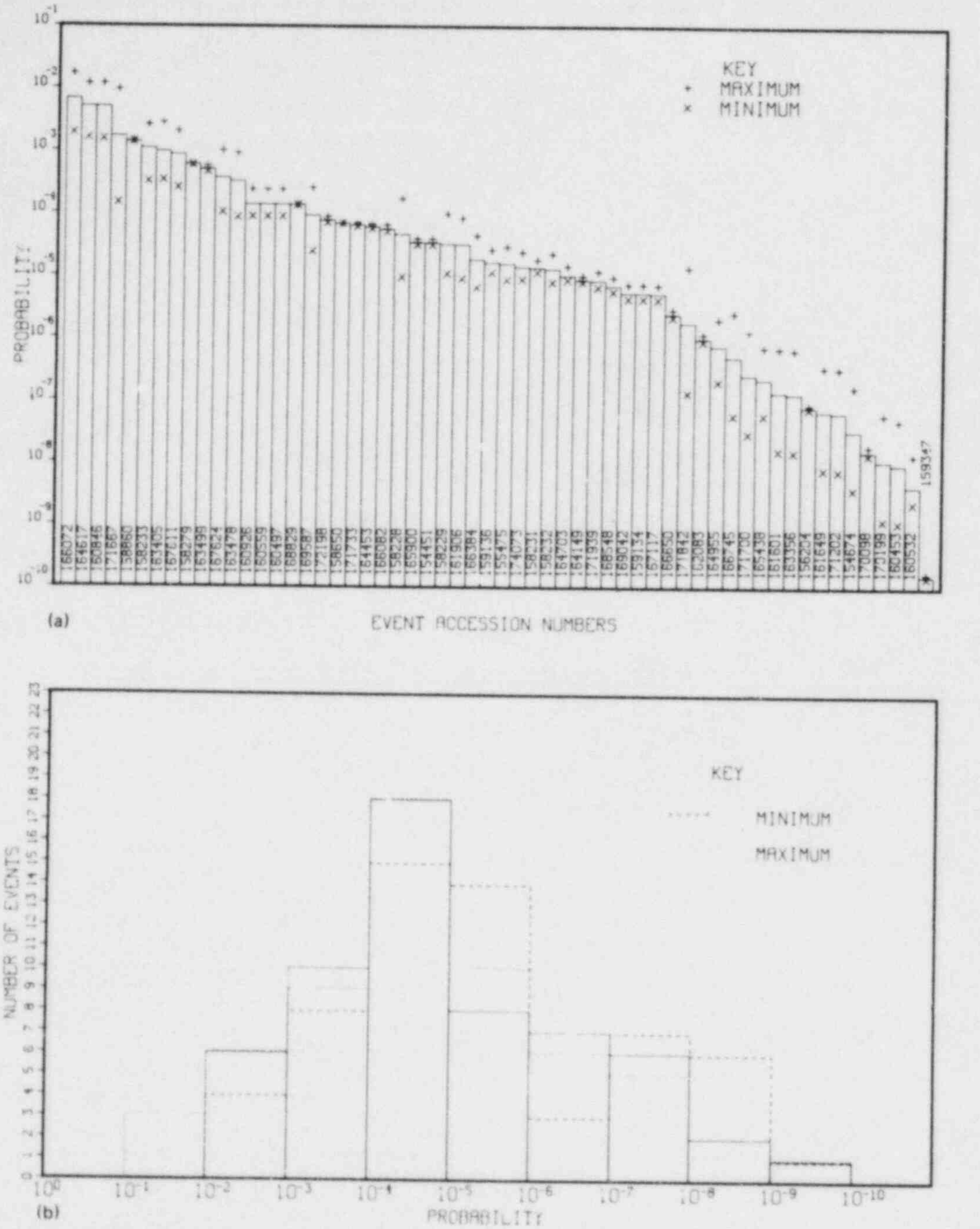


Fig. E.2. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in recovery class R2 numeric values.

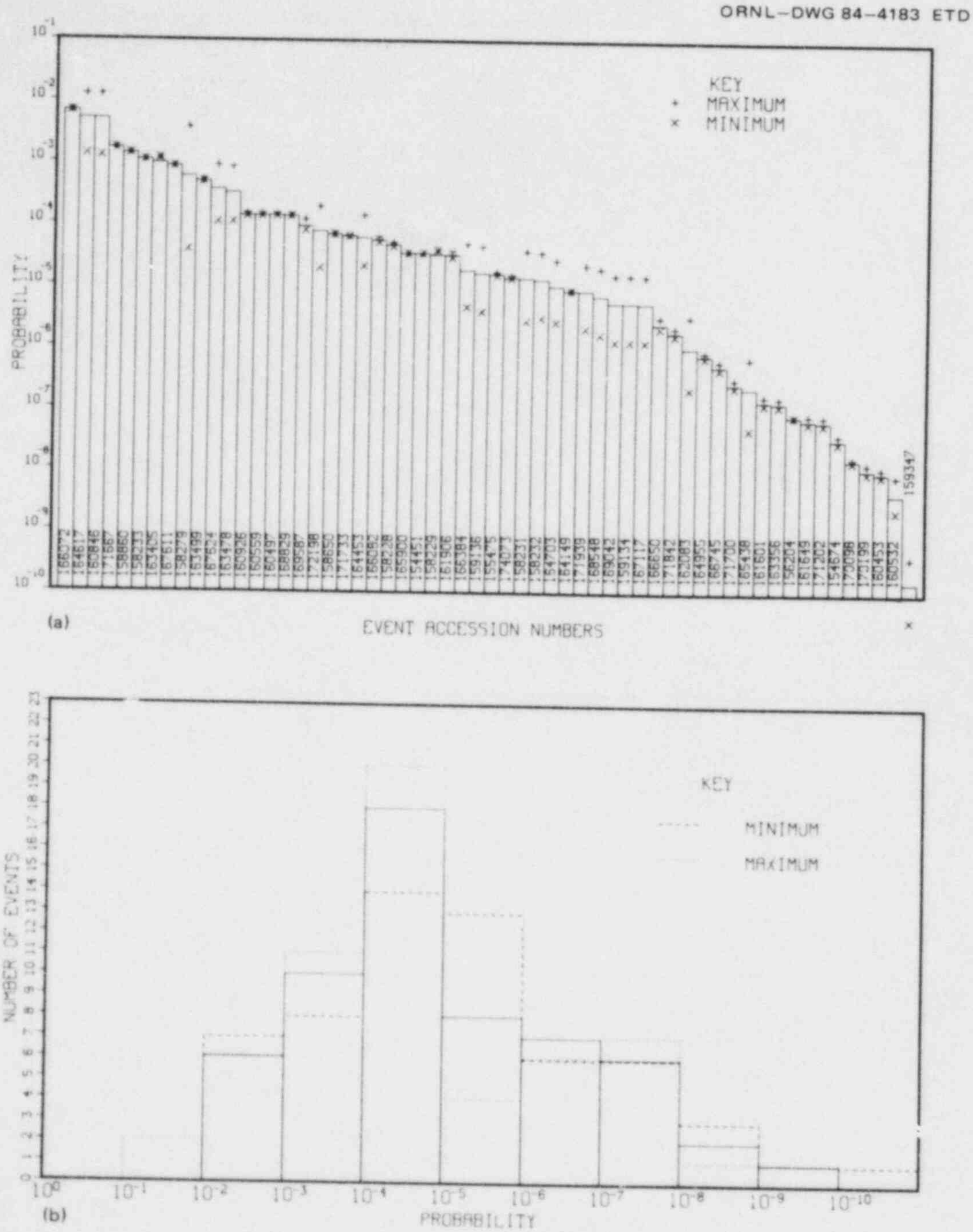


Fig. E.3. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in recovery class R3 numeric values.

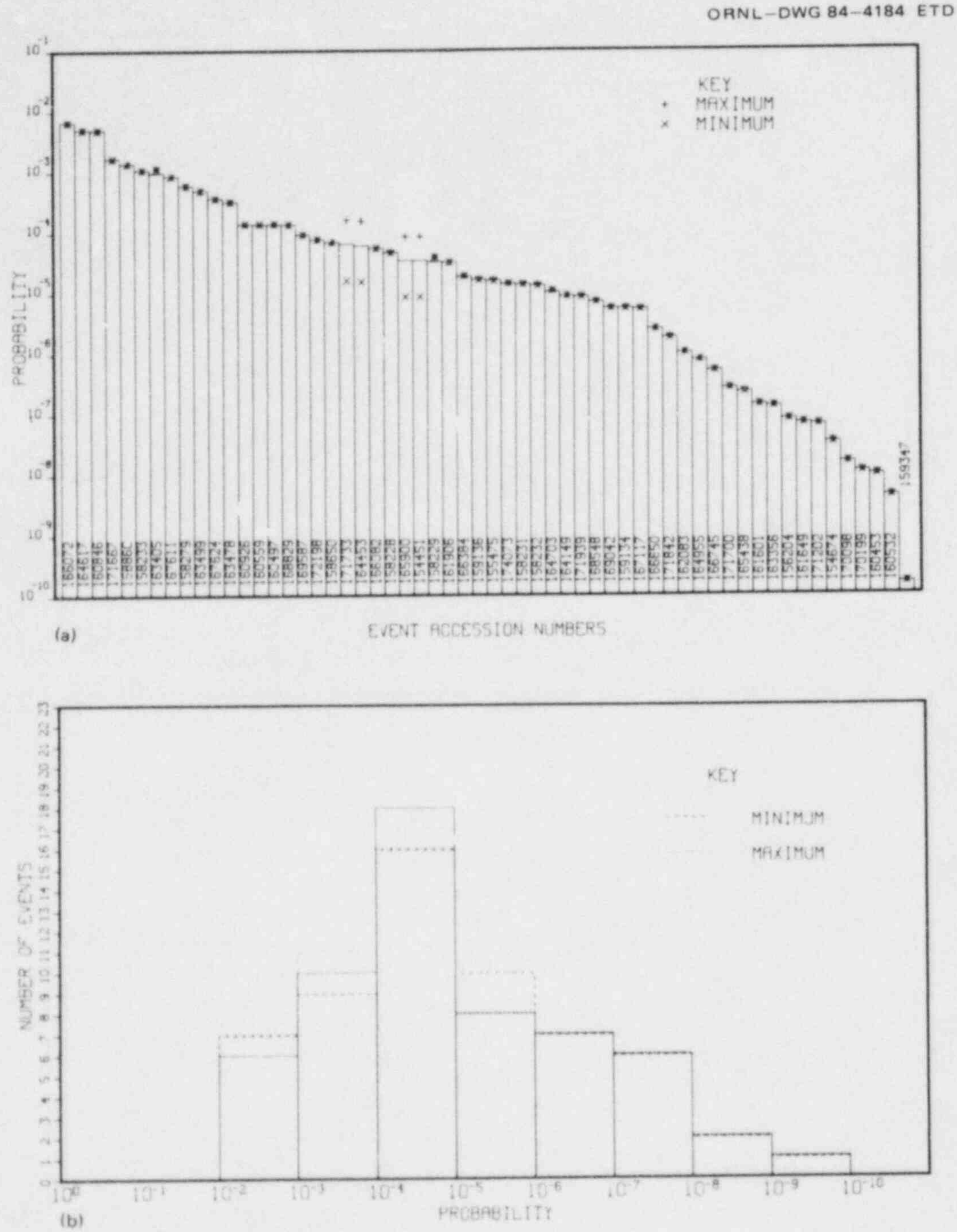


Fig. E.4. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in recovery class R4 numeric values.

ORNL-DWG 84-4185 ETD

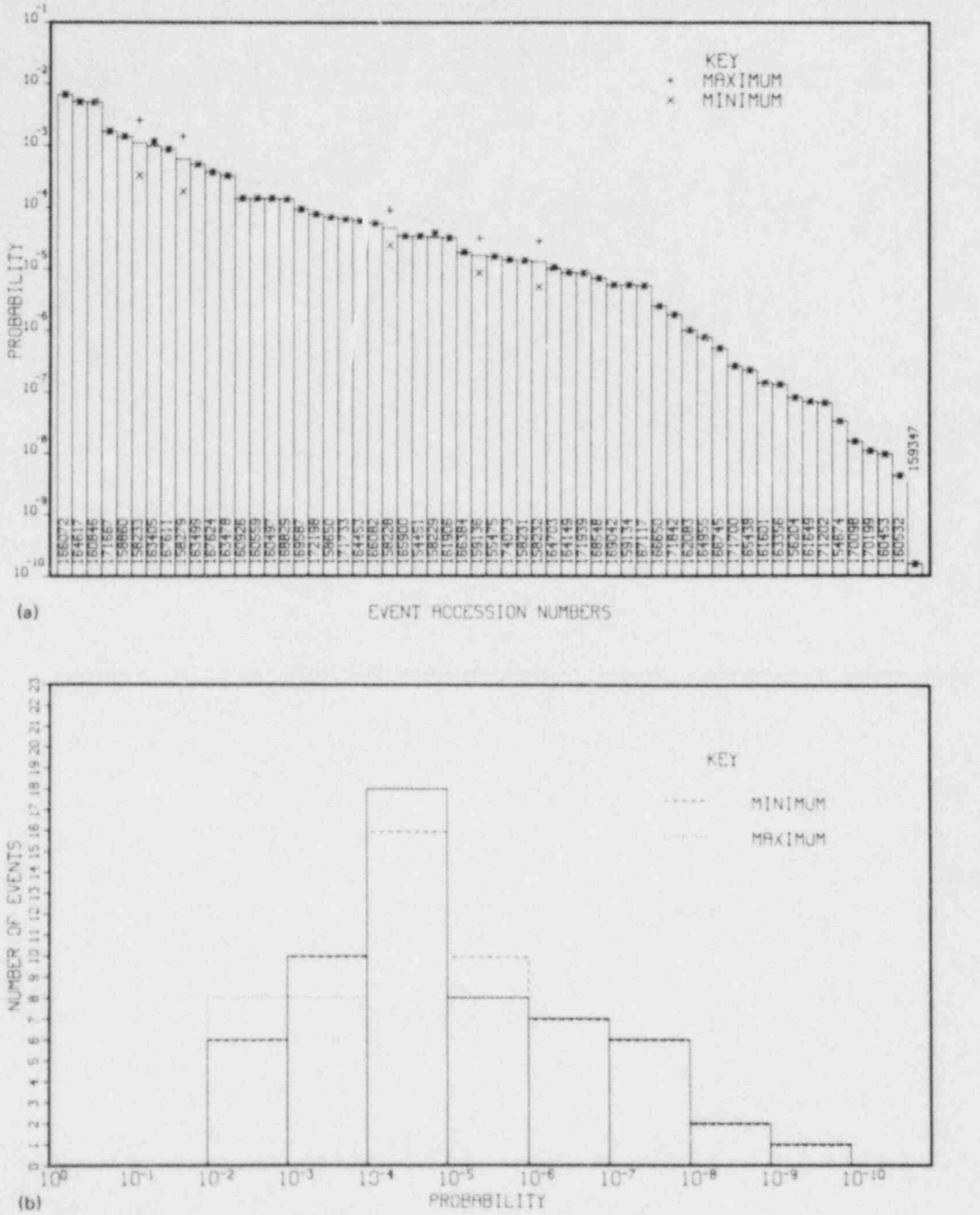


Fig. E.5. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in plant-specific tailoring factor P2 numeric values.

ORNL-DWG 84-4186 ETD

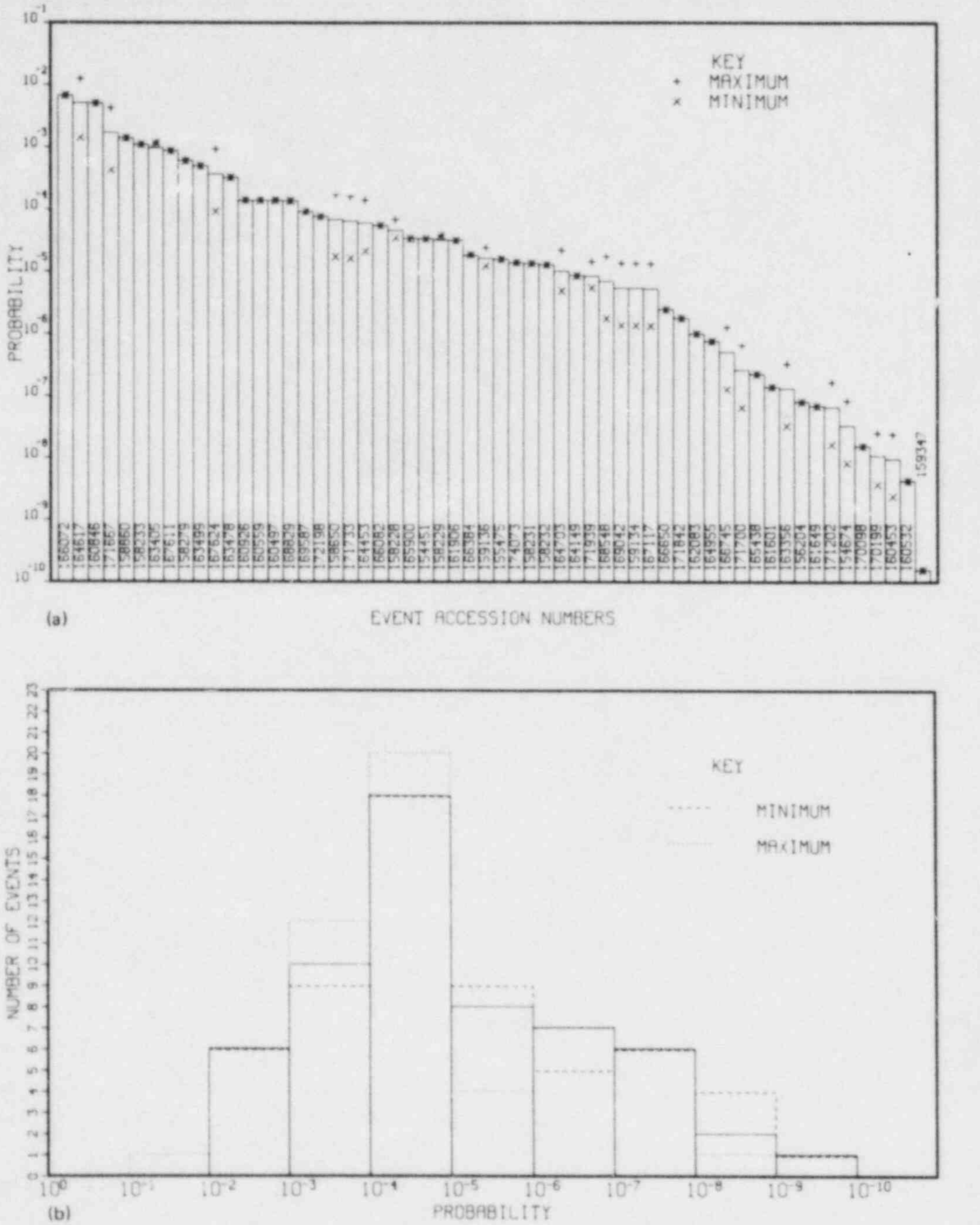


Fig. E.6. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in plant-specific tailoring factor P3 numeric values.

ORNL-DWG 84-4187 ETD

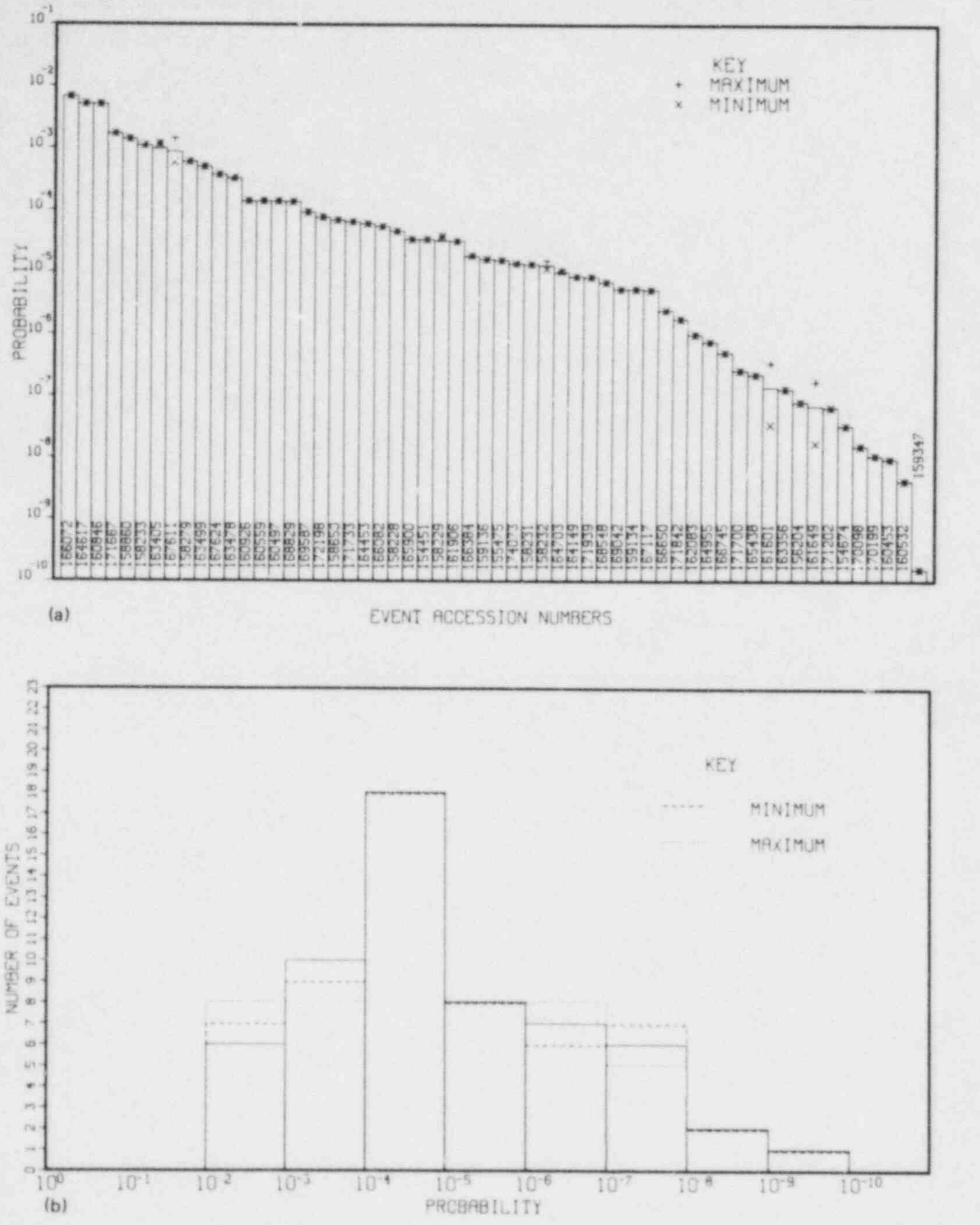


Fig. E.7. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in plant-specific tailoring factor P4 numeric values.

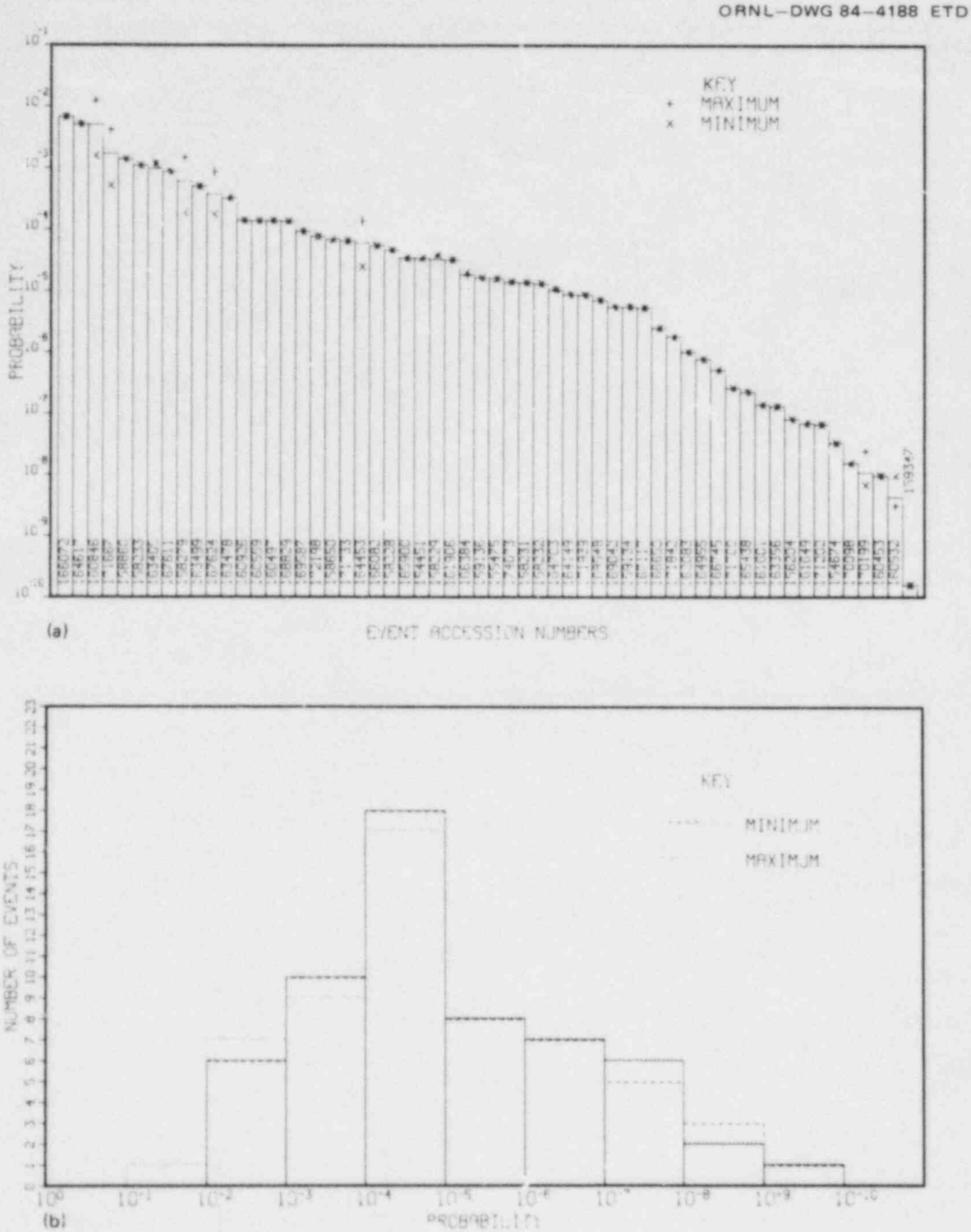


Fig. E.8. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in β factor numeric values.

ORNL-DWG 84-4189 ETD

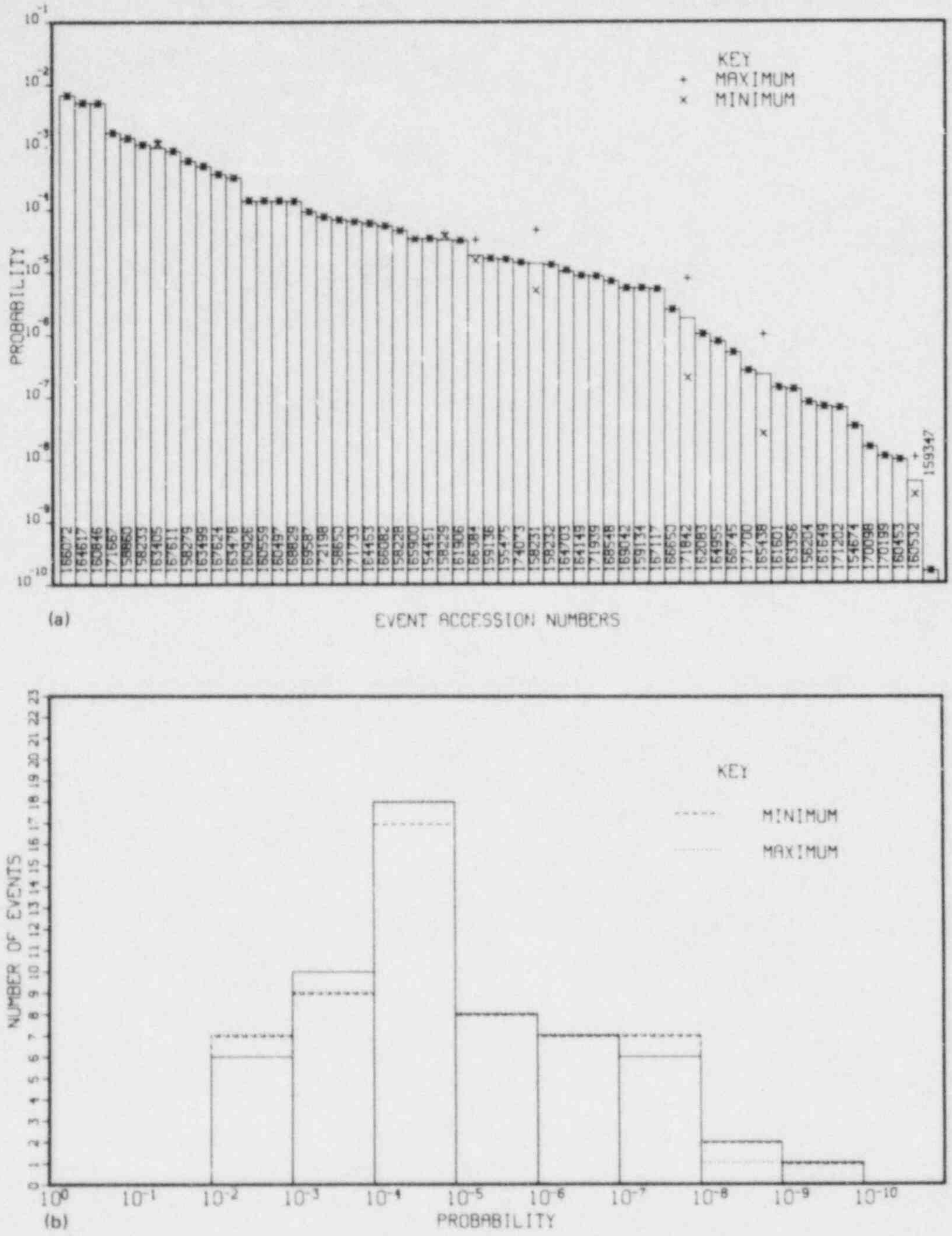


Fig. E.9. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR HPCI/RCIC numeric values.

ORNL-DWG 84-4190 ETD

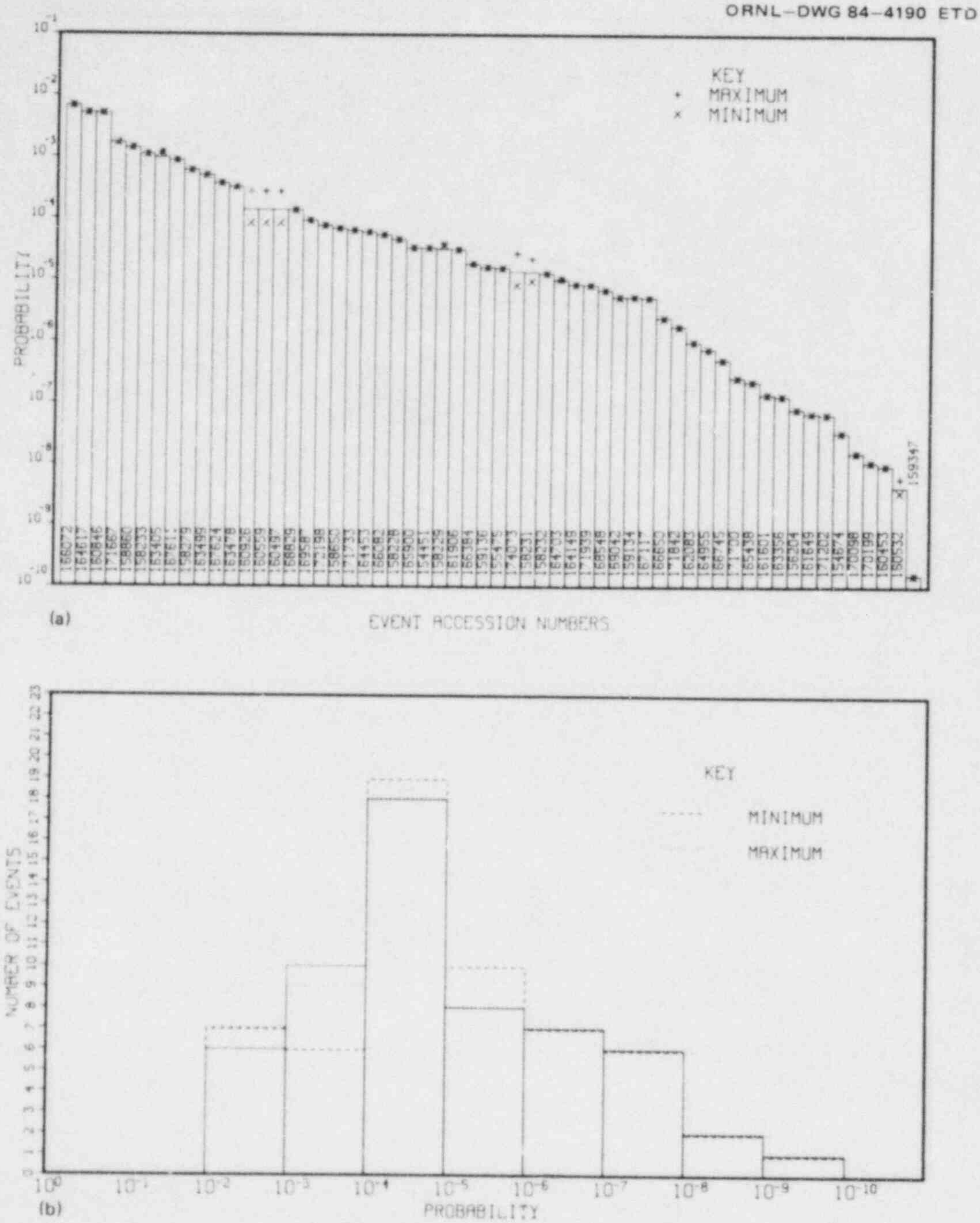


Fig. E.10. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR high-pressure cooling provided following LOCA numeric values.

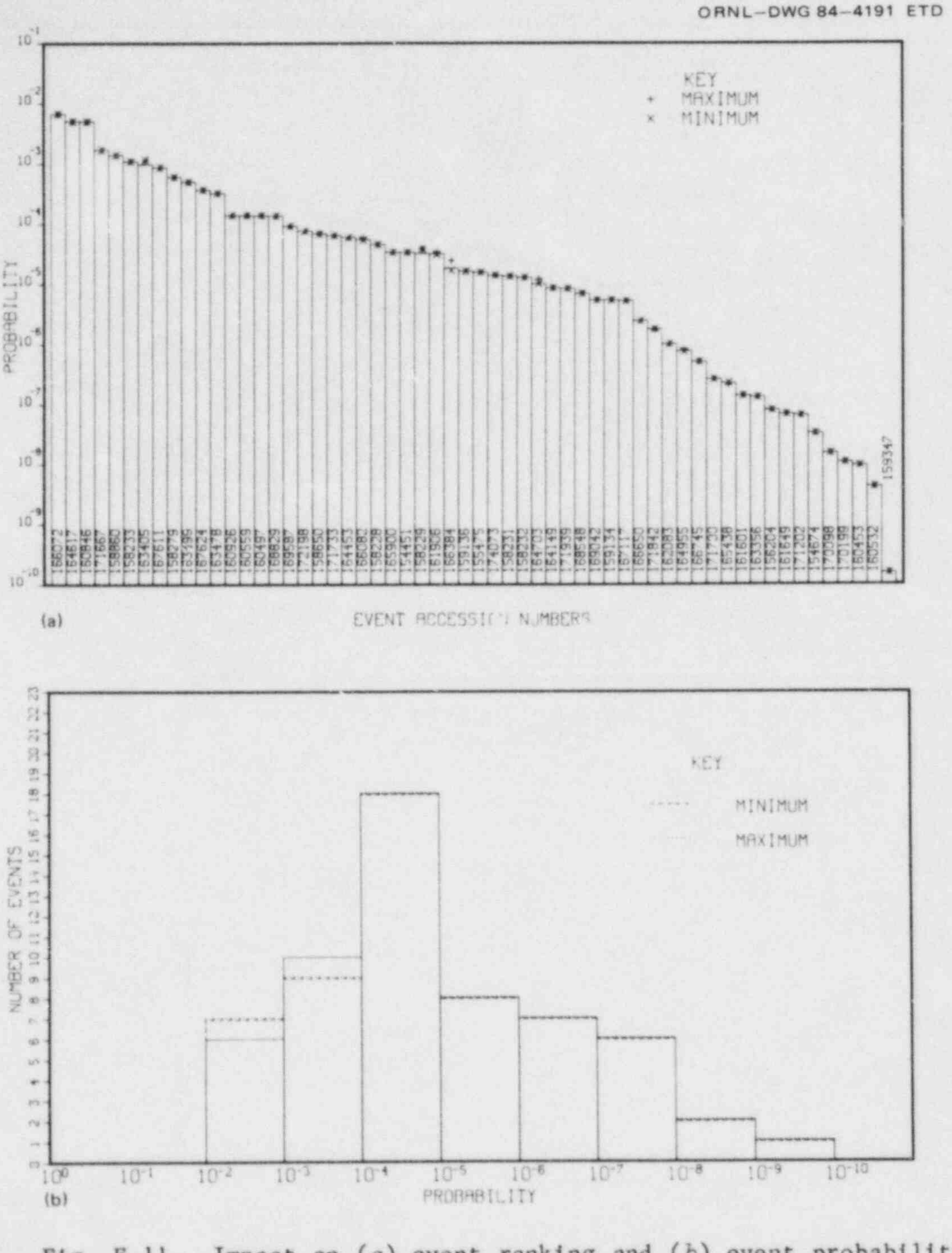


Fig. E.11. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR emergency power numeric values.

ORNL-DWG 84-4192 ETD

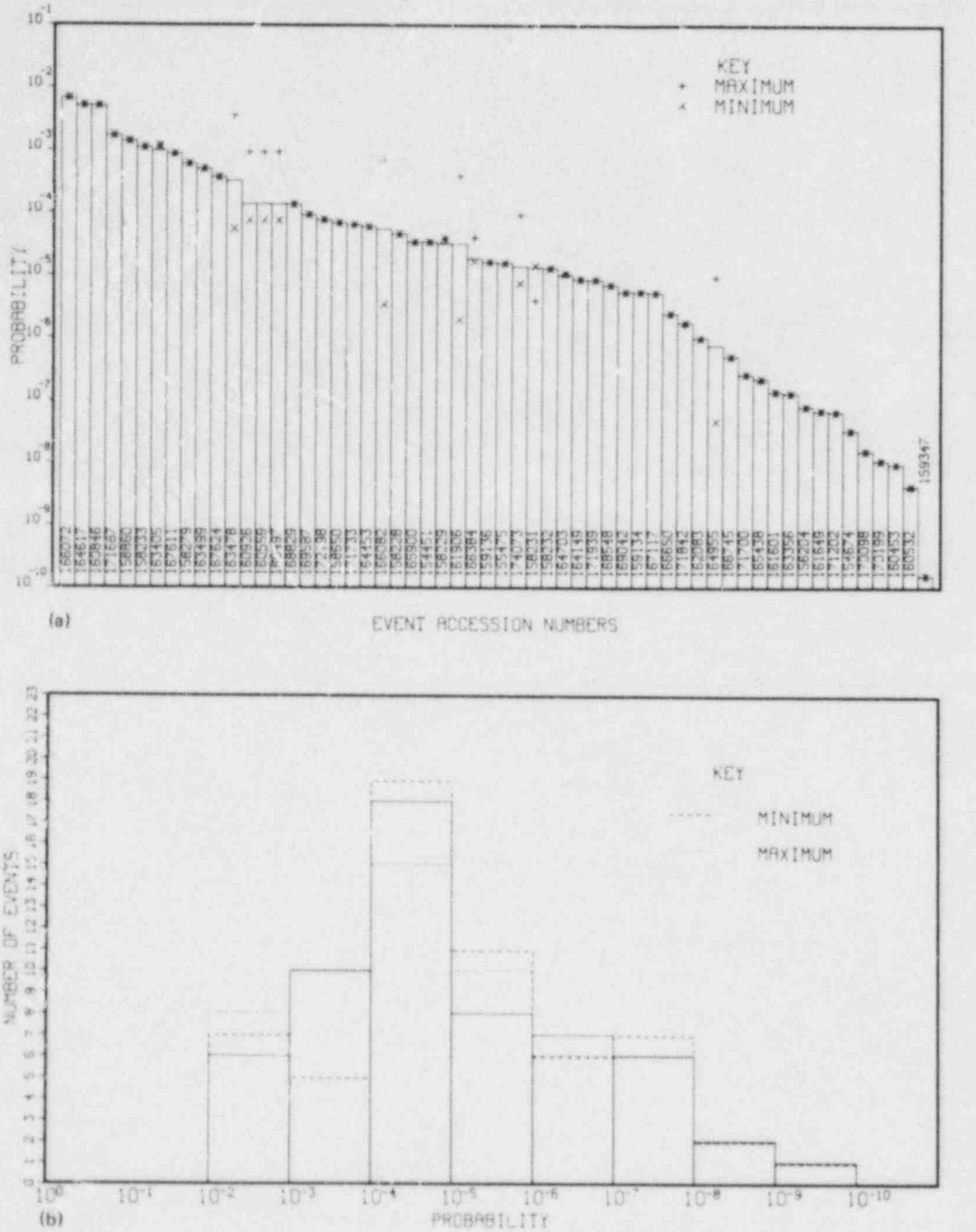


Fig. E.12. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR automatic depressurization numeric values.

ORNL-DWG 84-4193 ETD

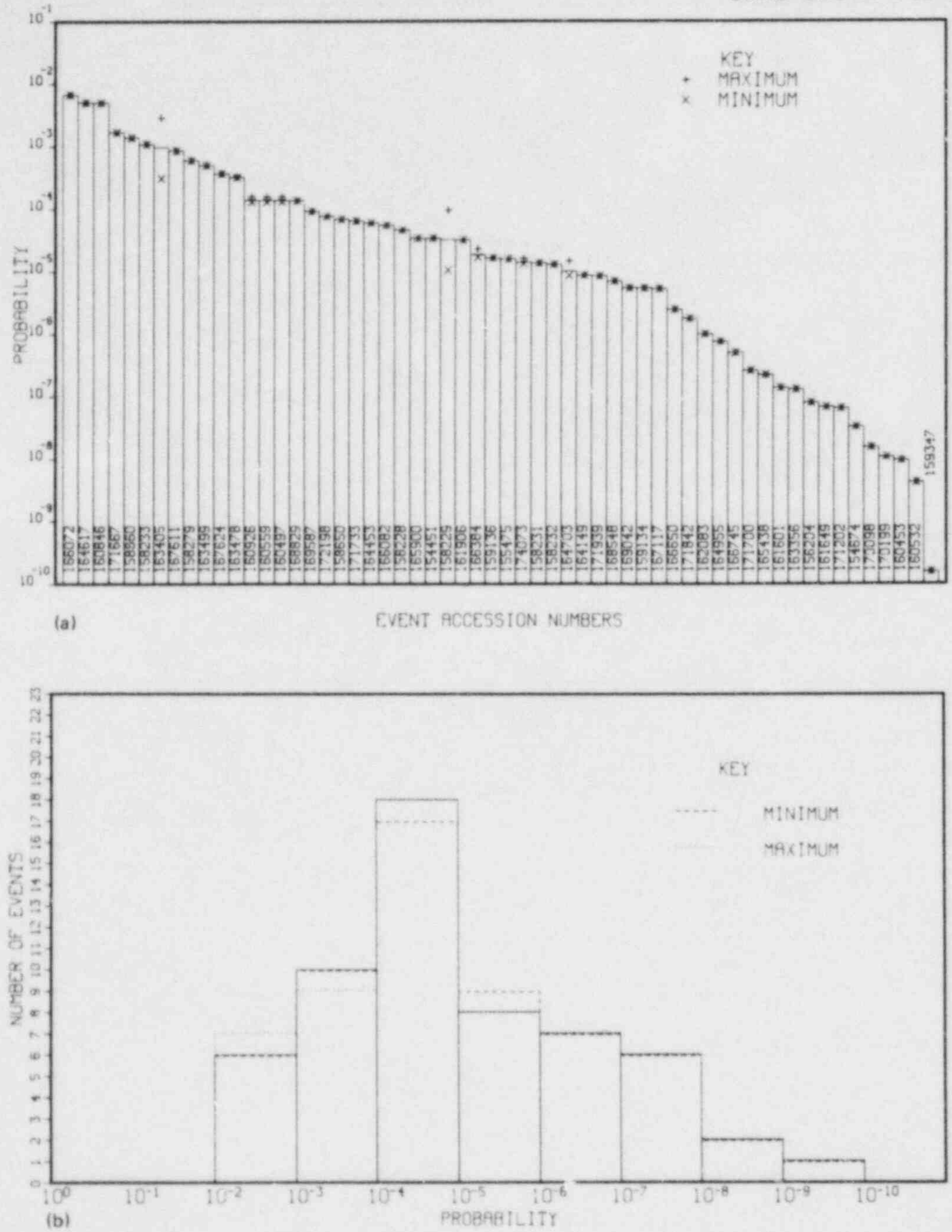


Fig. E.13. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR SBLC numeric values.

ORNL-DWG 84-4194 ETD

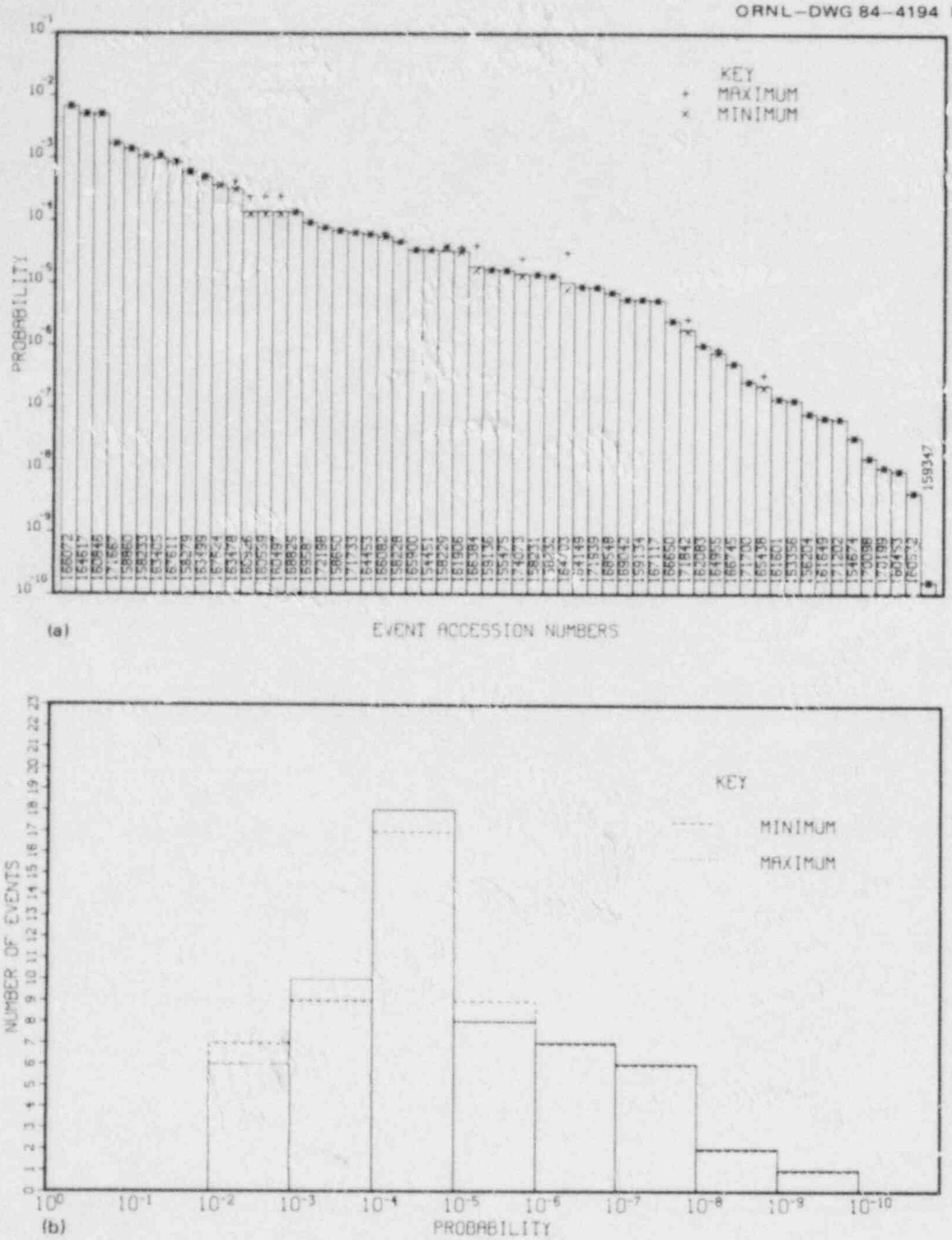


Fig. E.14. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR scram numeric values.

ORNL-DWG 84-4195 ETD

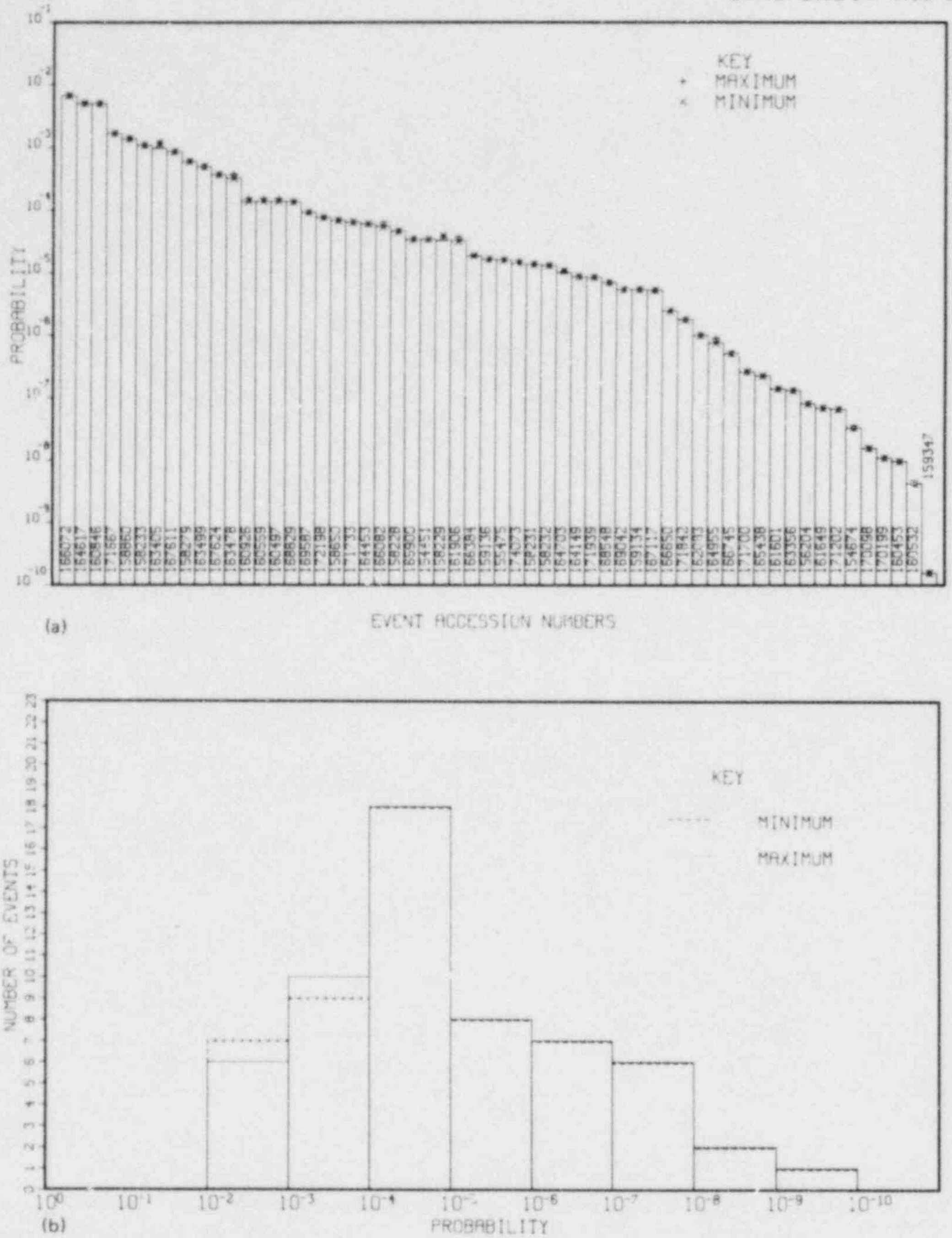


Fig. E.15. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR LPCI/core spray numeric values.

ORNL-DWG 84-4196 ETD

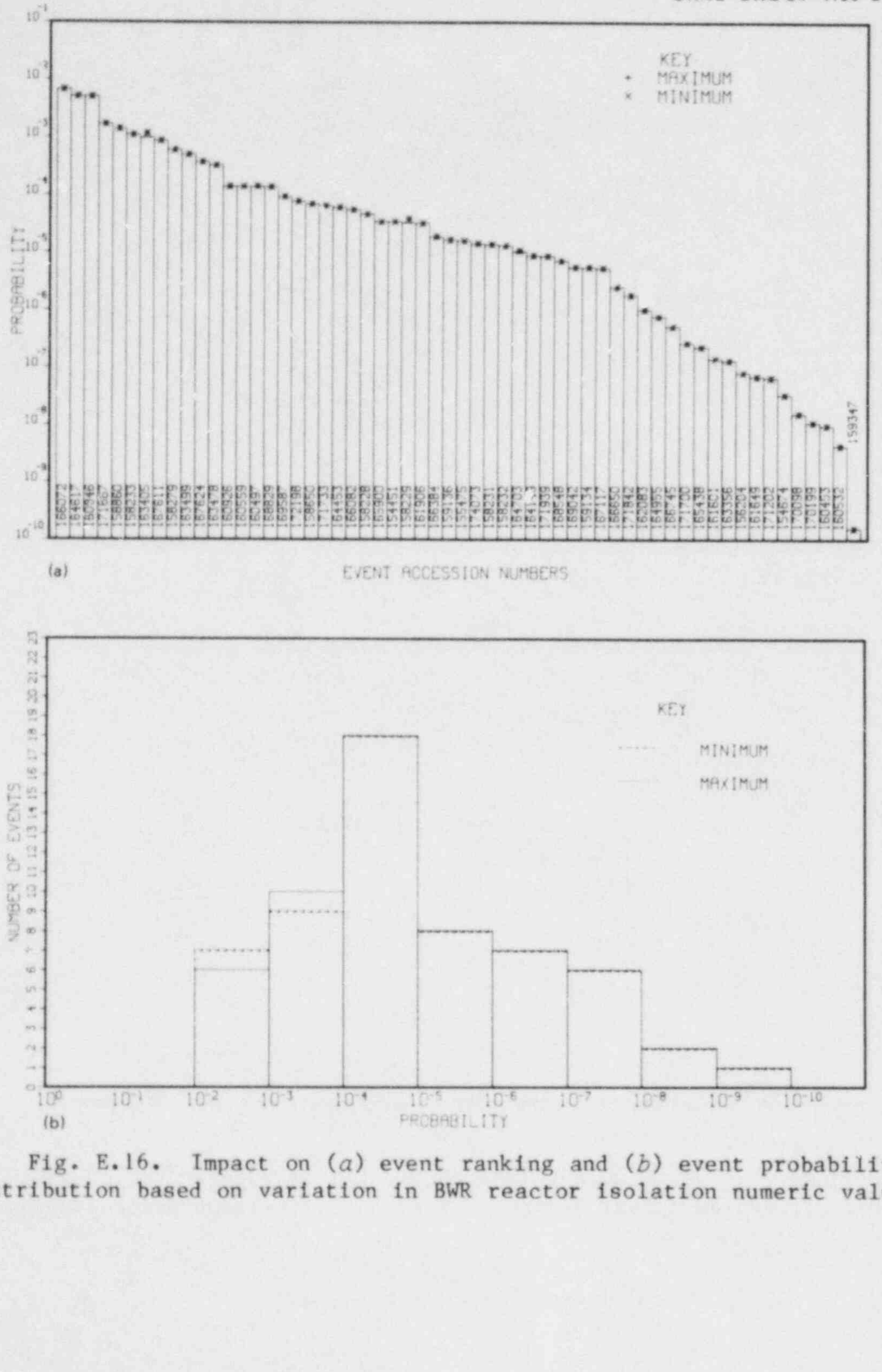
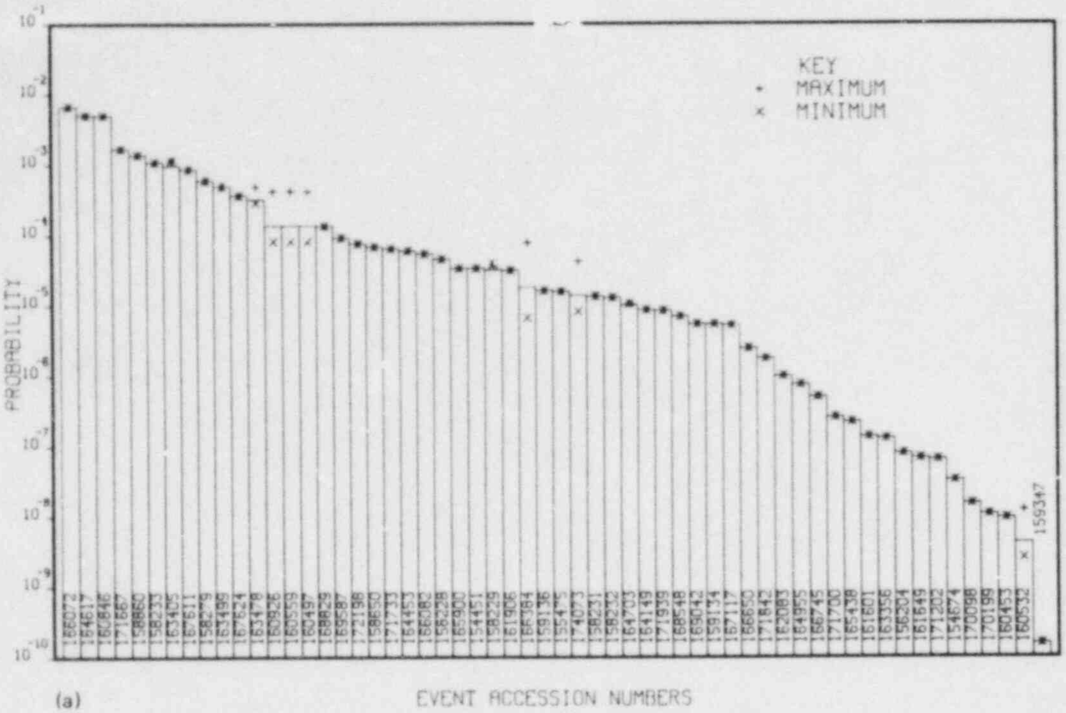


Fig. E.16. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR reactor isolation numeric values.

ORNL-DWG 84-4197 ETD



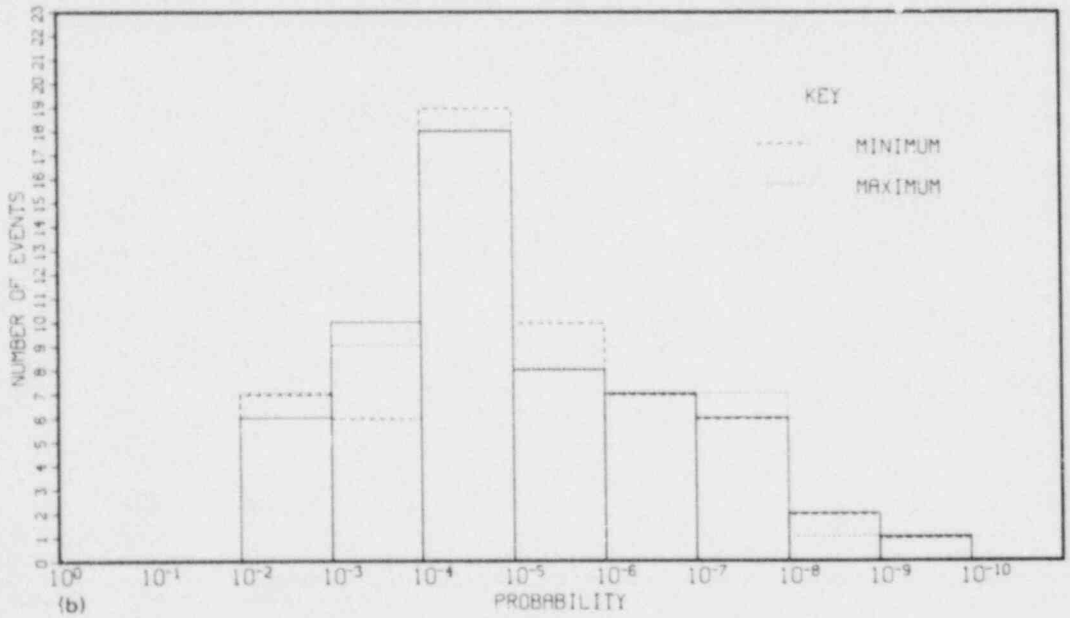


Fig. E.17. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR long-term core cooling numeric values.

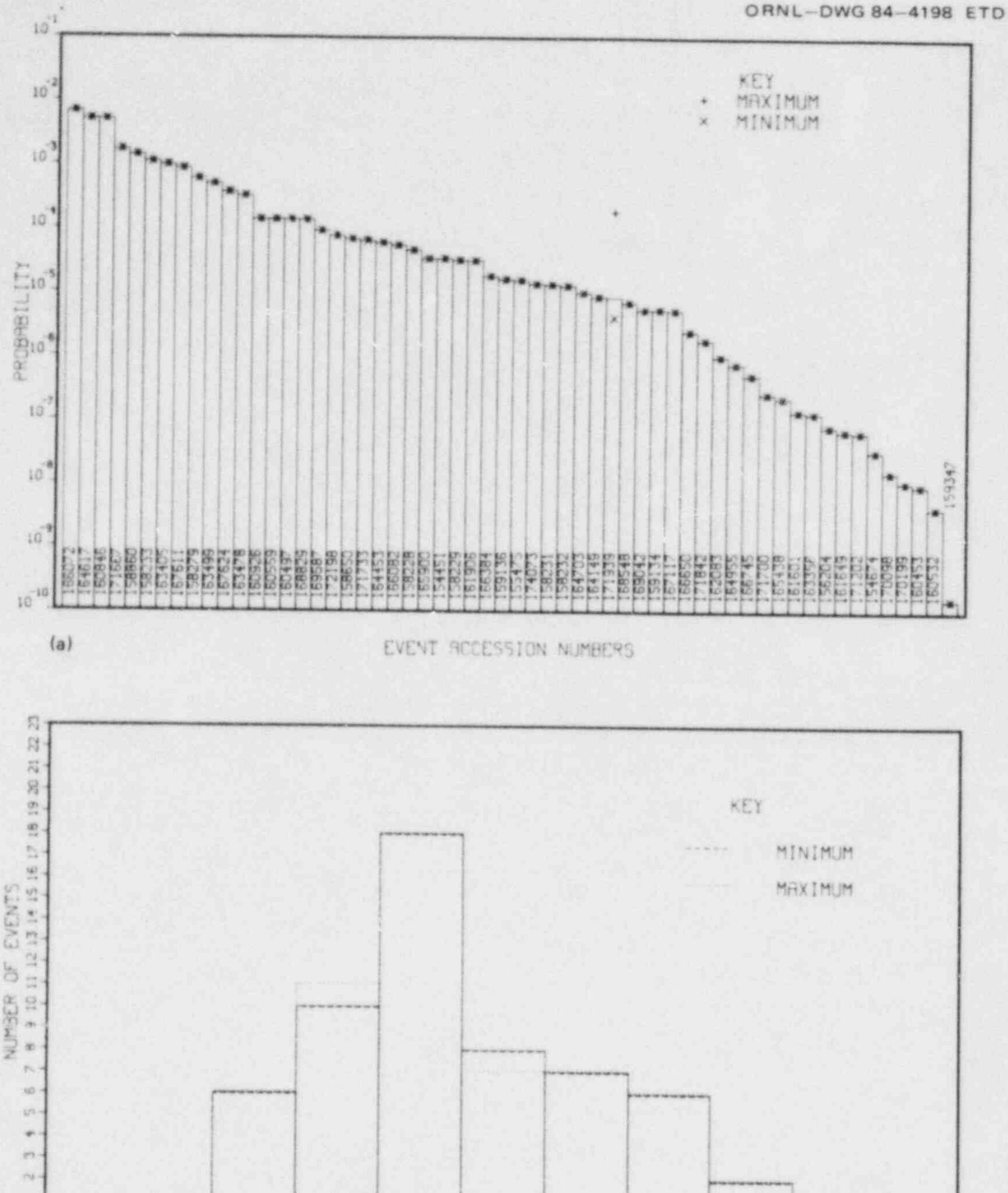


Fig. E.18. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR reactor trip numeric values.

ORNL-DWG 84-4199 ETD

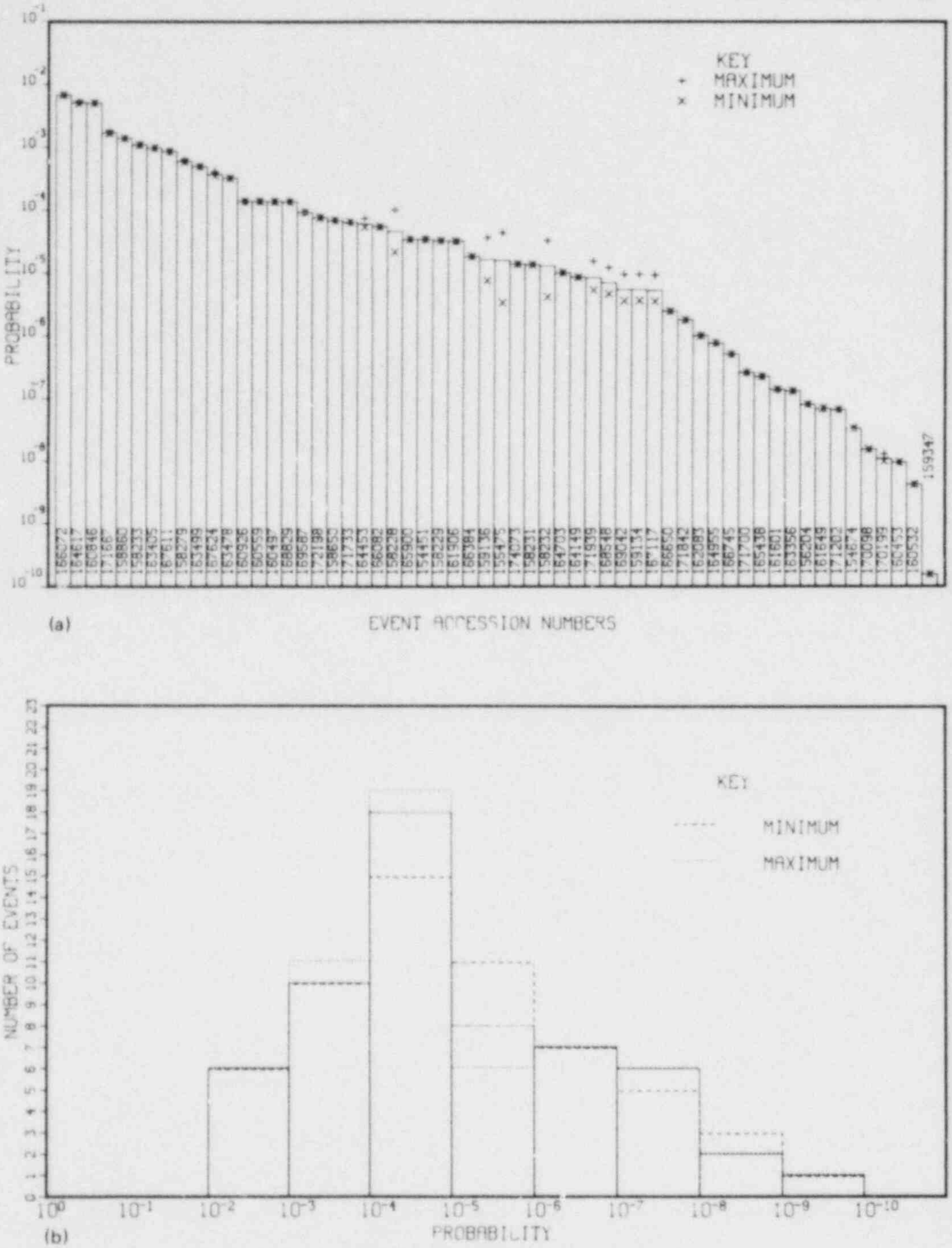


Fig. E.19. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR AFW given reactor trip success numeric values.

ORNL-DWG 84-4200 ETD

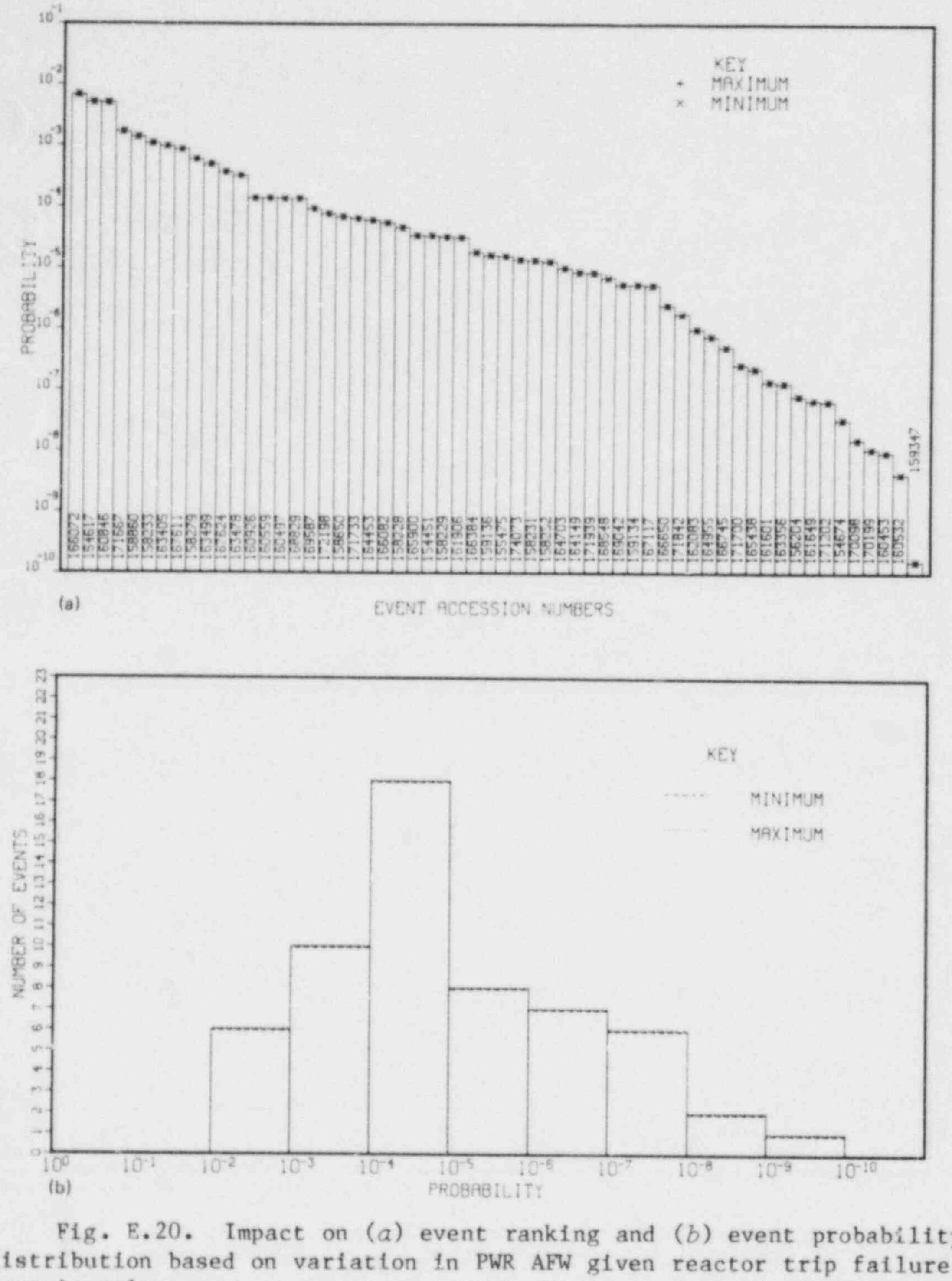


Fig. E.20. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR AFW given reactor trip failure numeric values.

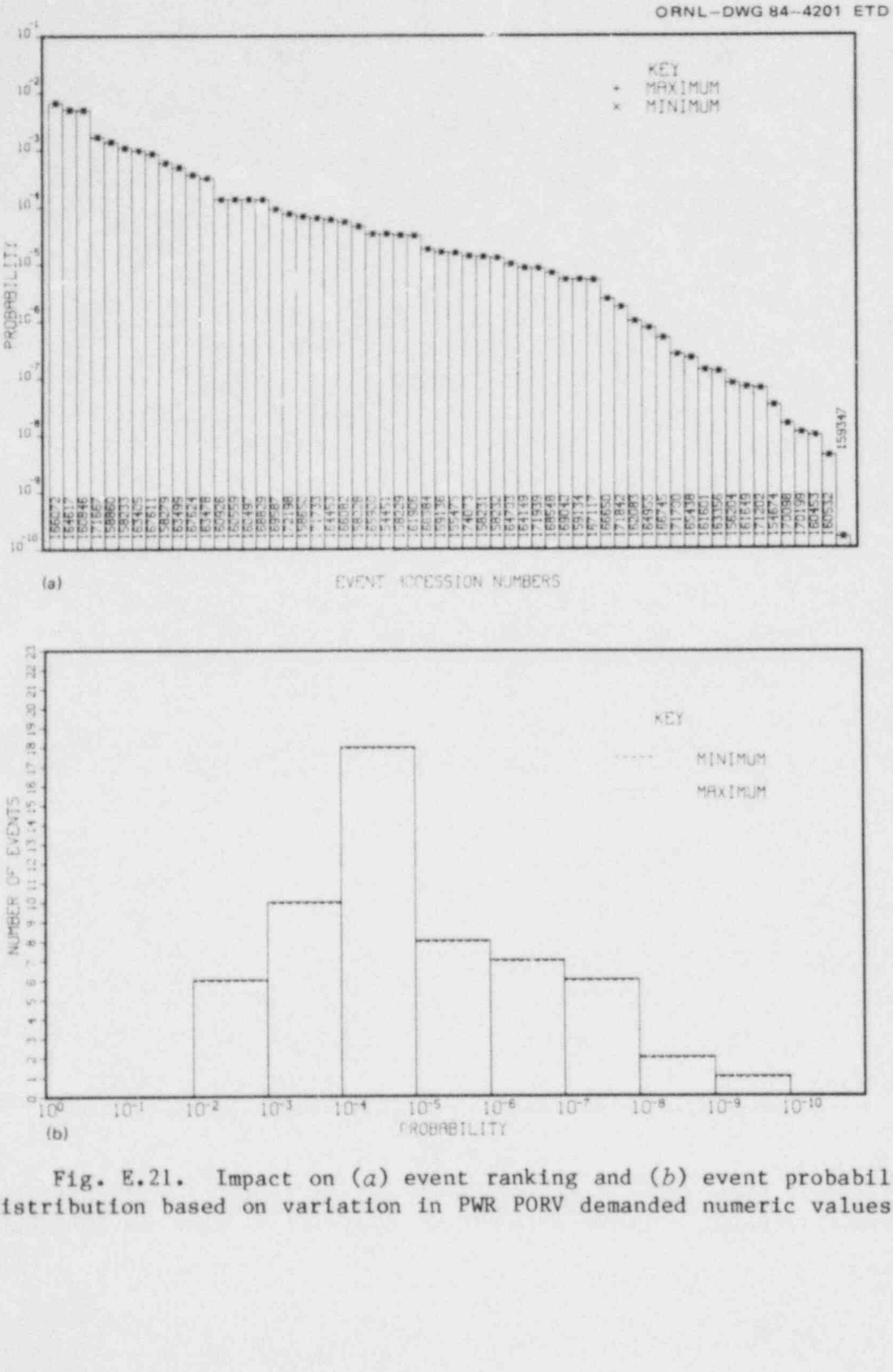


Fig. E.21. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR PORV demanded numeric values.

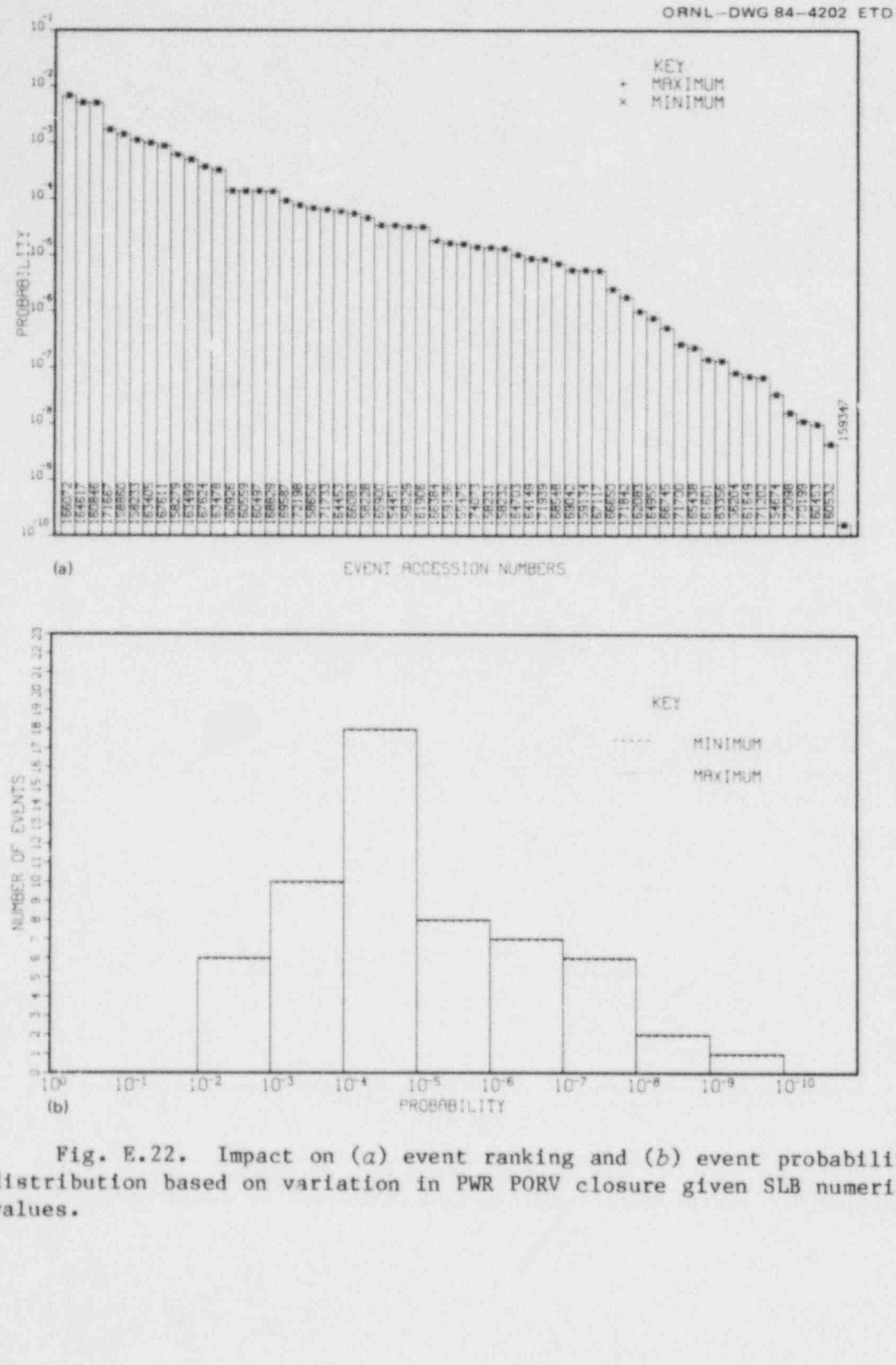


Fig. E.22. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR PORV closure given SLB numeric values.

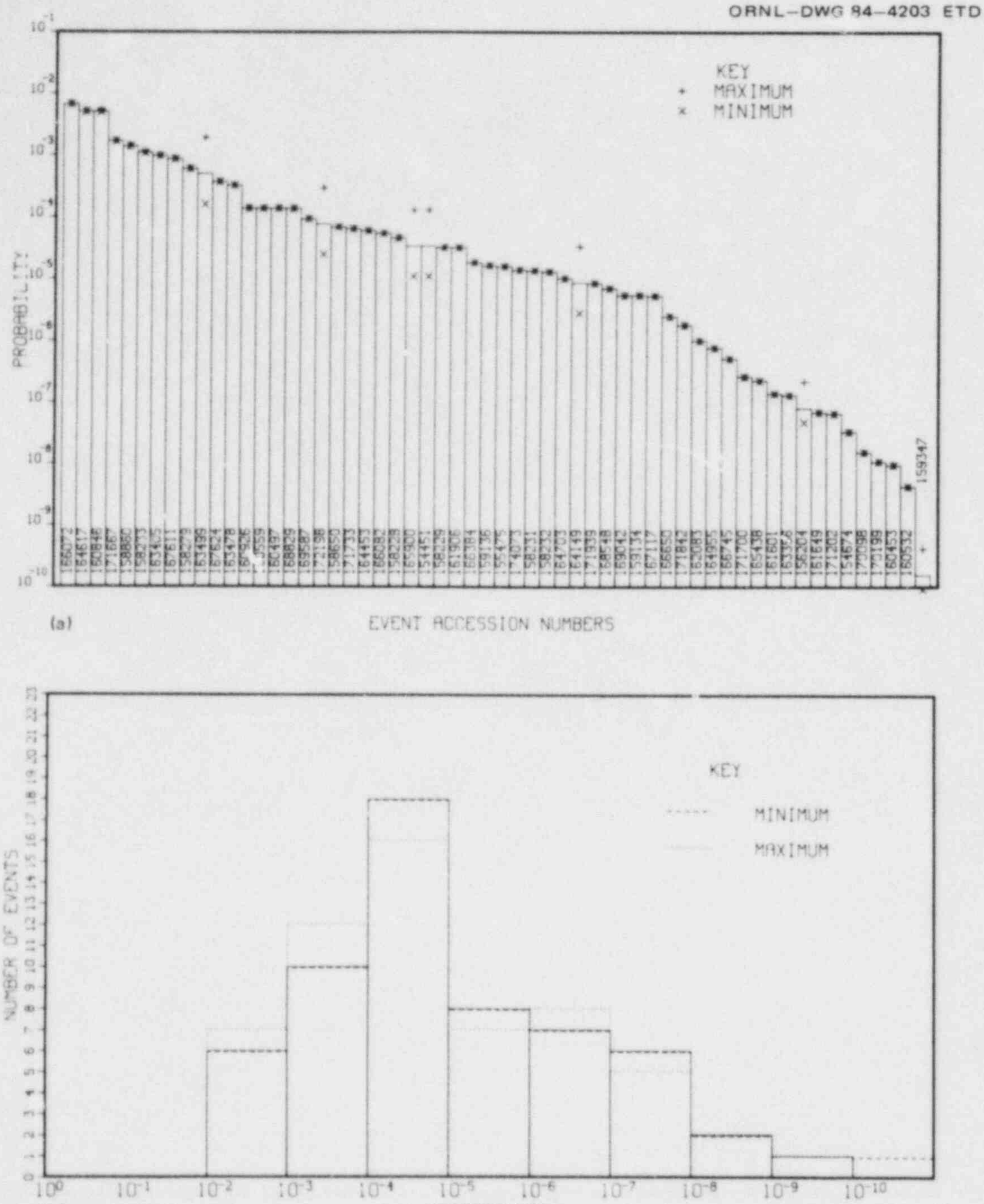


Fig. E.23. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR HPI given AFW success numeric values.

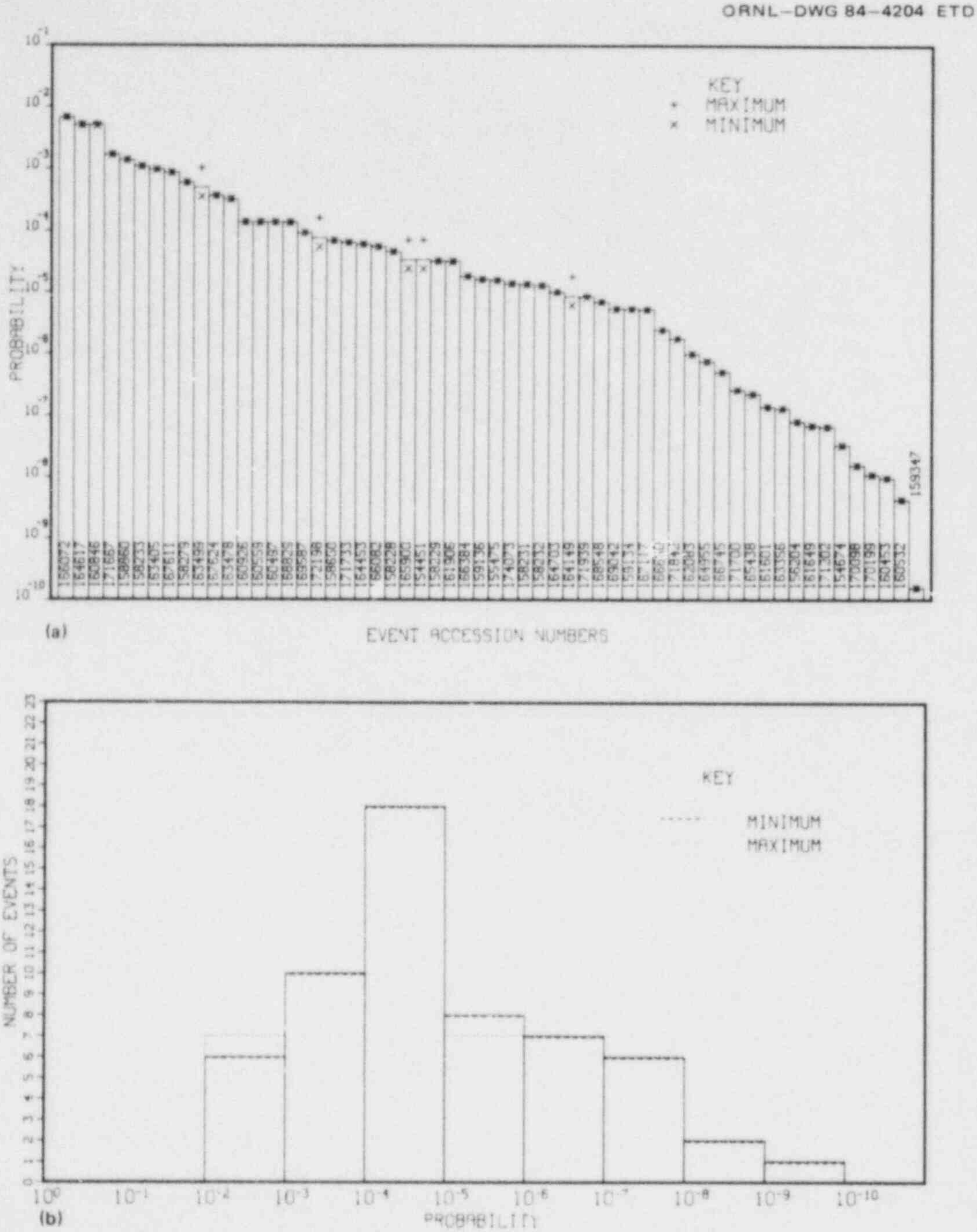


Fig. E.24. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR long-term core cooling numeric values.

ORNL-DWG 84-4205 ETD

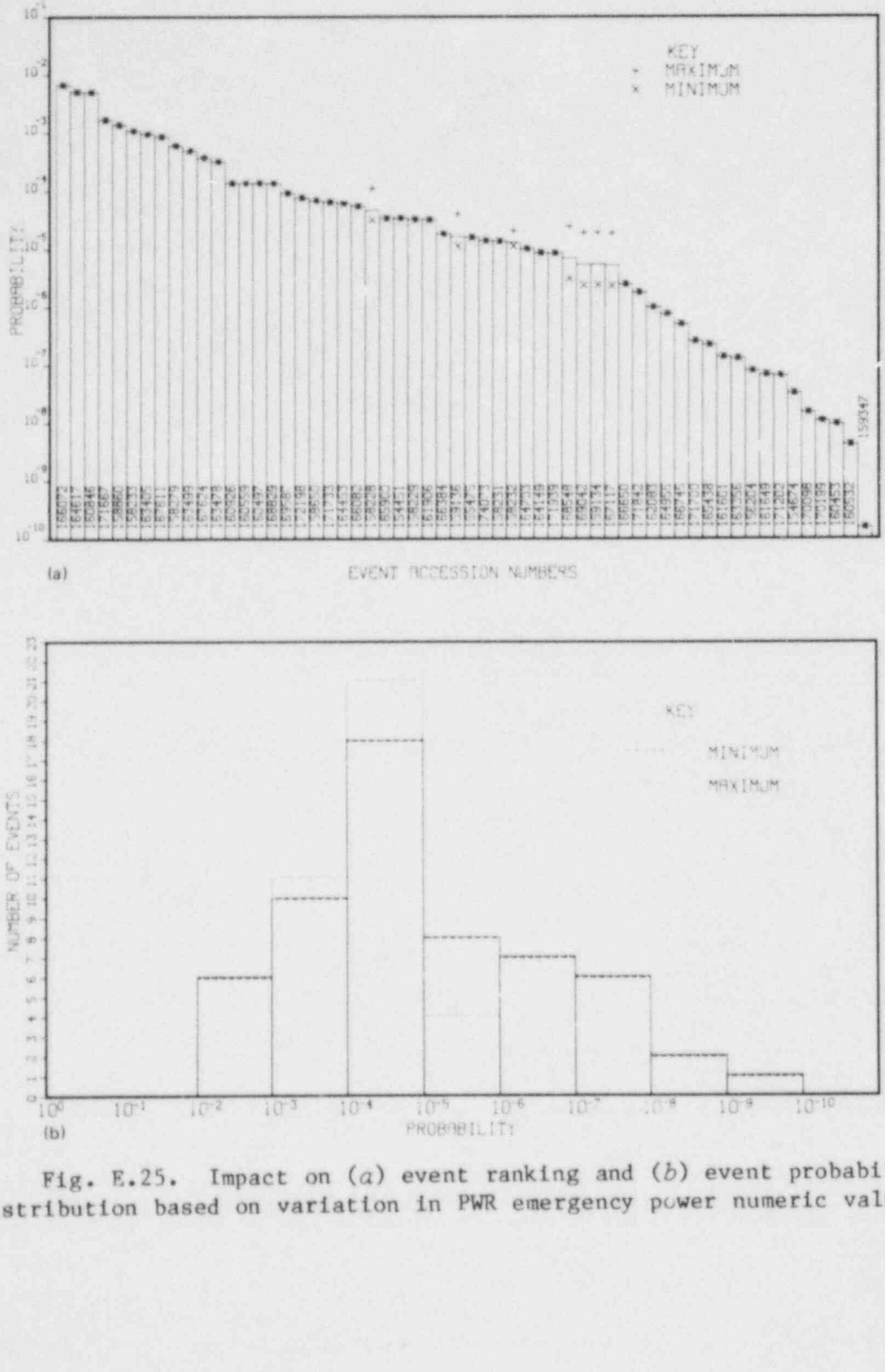


Fig. E.25. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR emergency power numeric values.

ORNL-DWG 84-4206 ETD

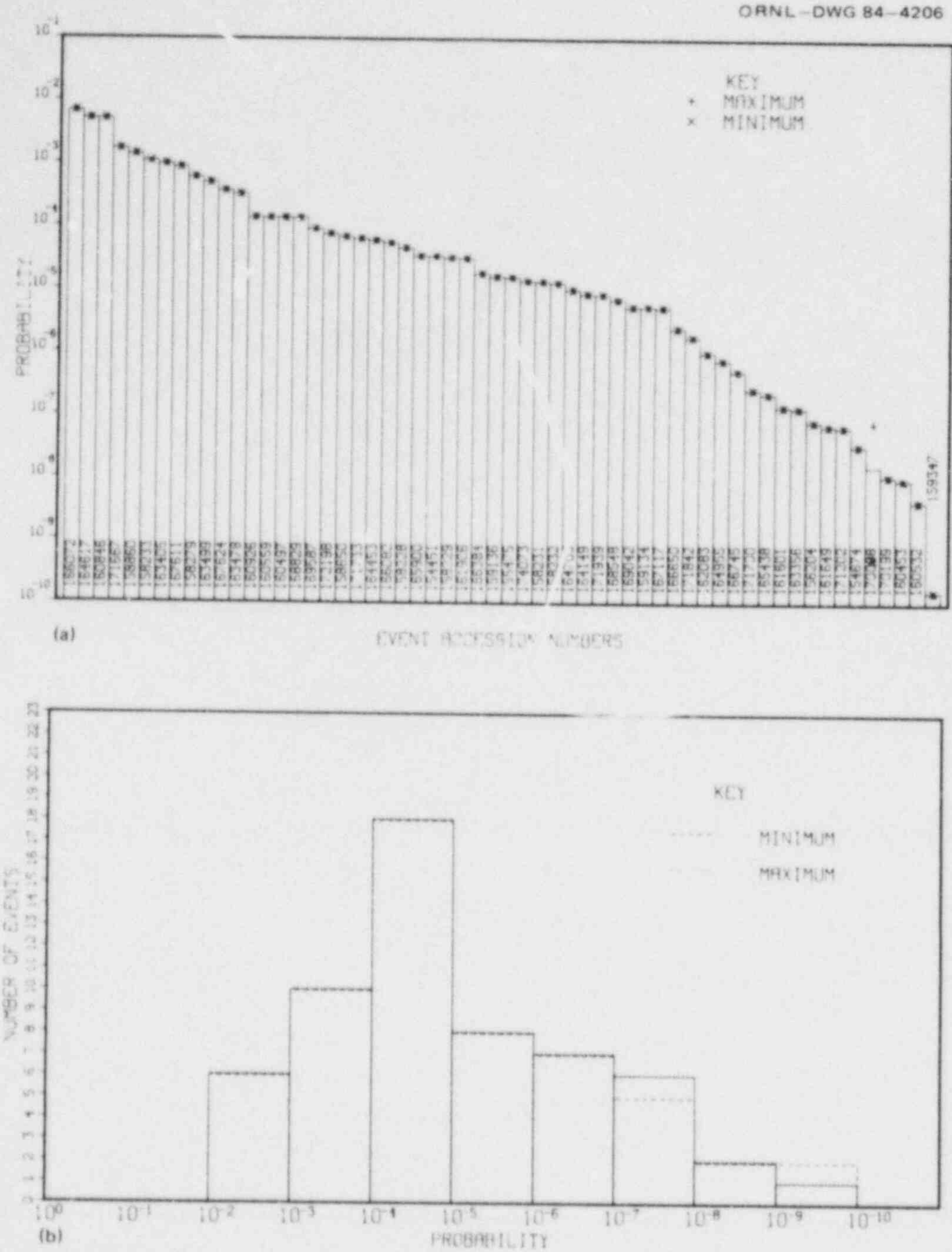


Fig. E.26. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR SG isolation numeric values.

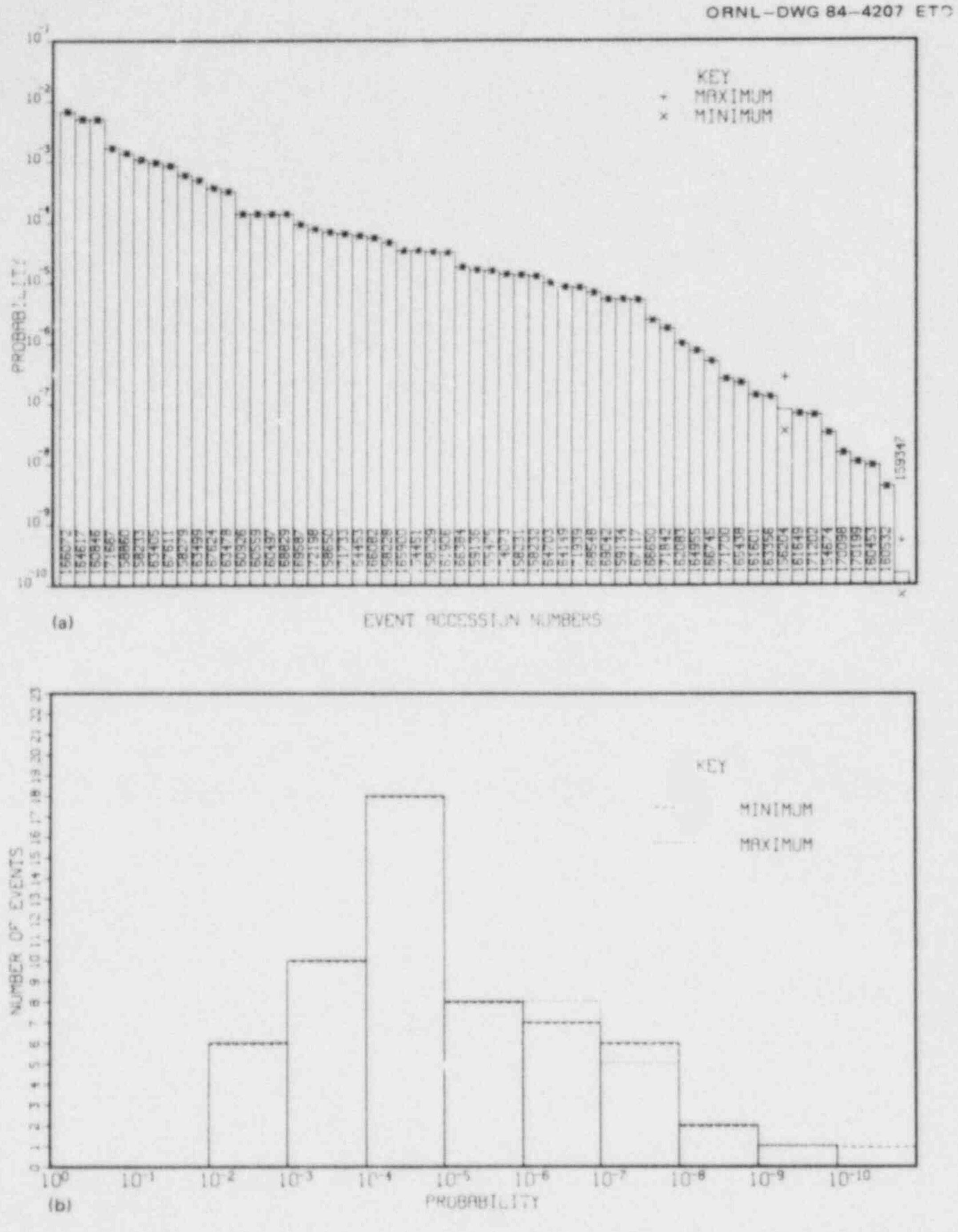


Fig. E.27. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR concentrated boric acid addition given HPI success numeric values.

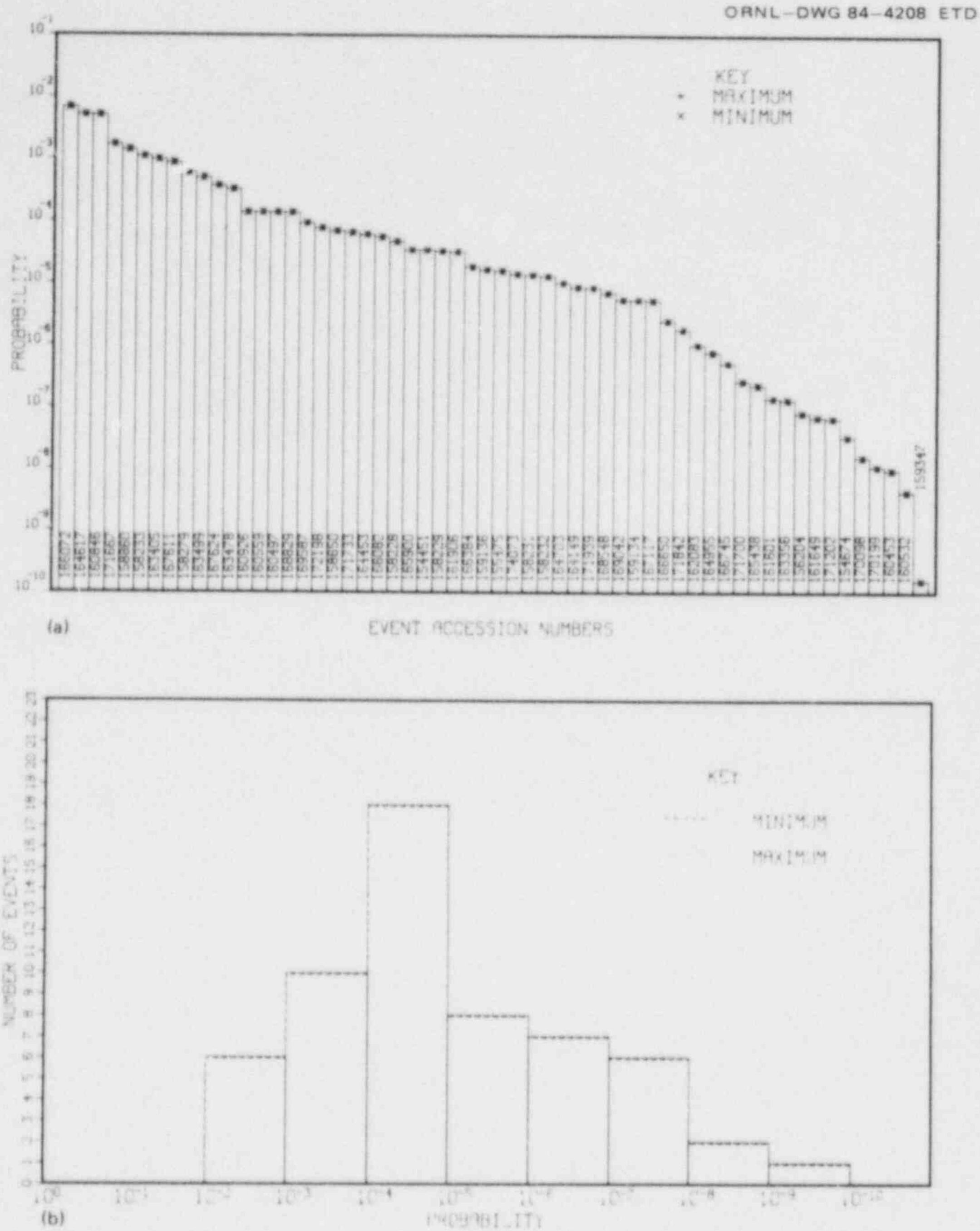


Fig. E.28. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR PORV opened due to HPI numeric values.

ORNL-DWG 84-4209 ETD

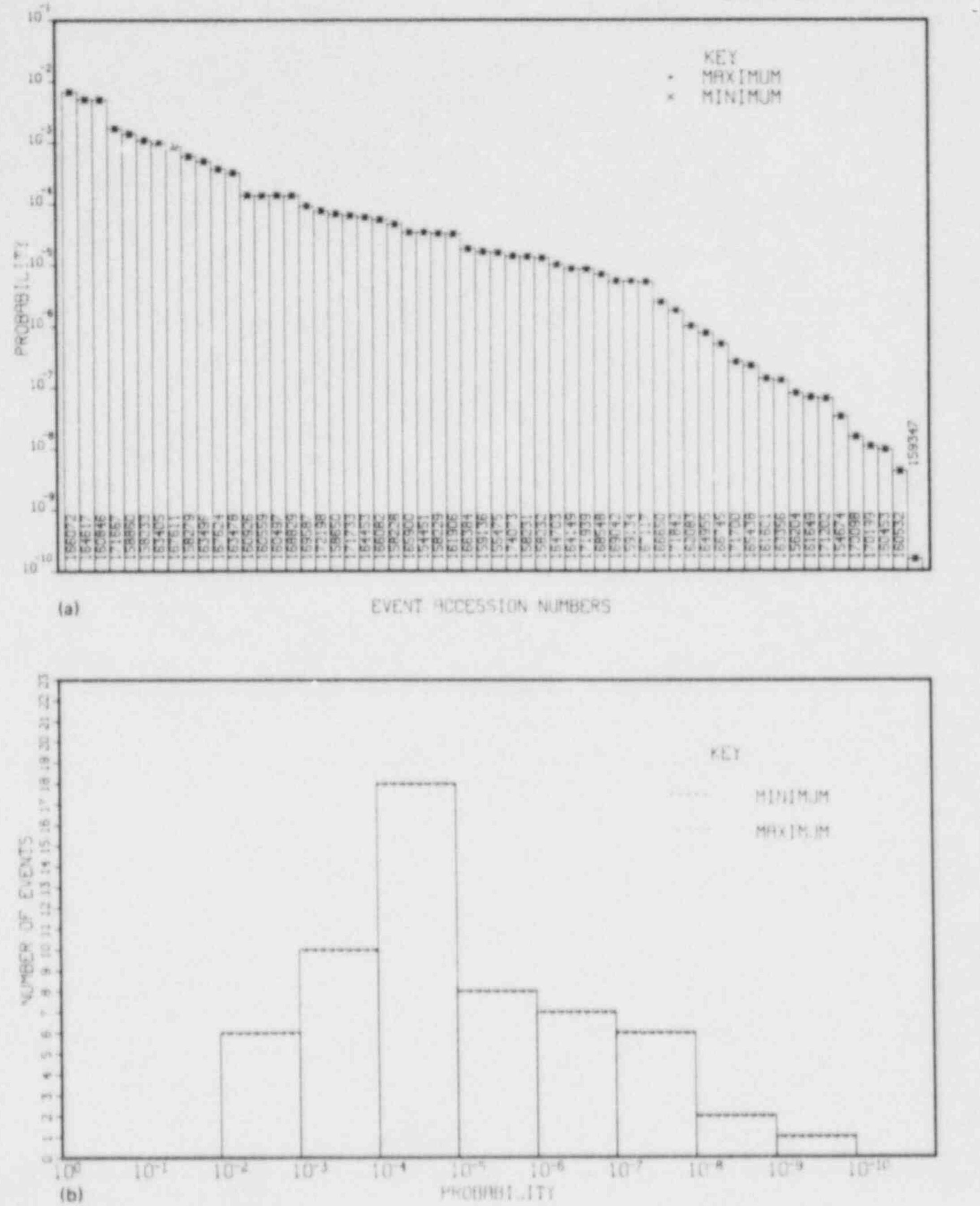


Fig. E.29. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR PORV closure given SLB numeric values.

ORNL-DWG 84-4210 ETD

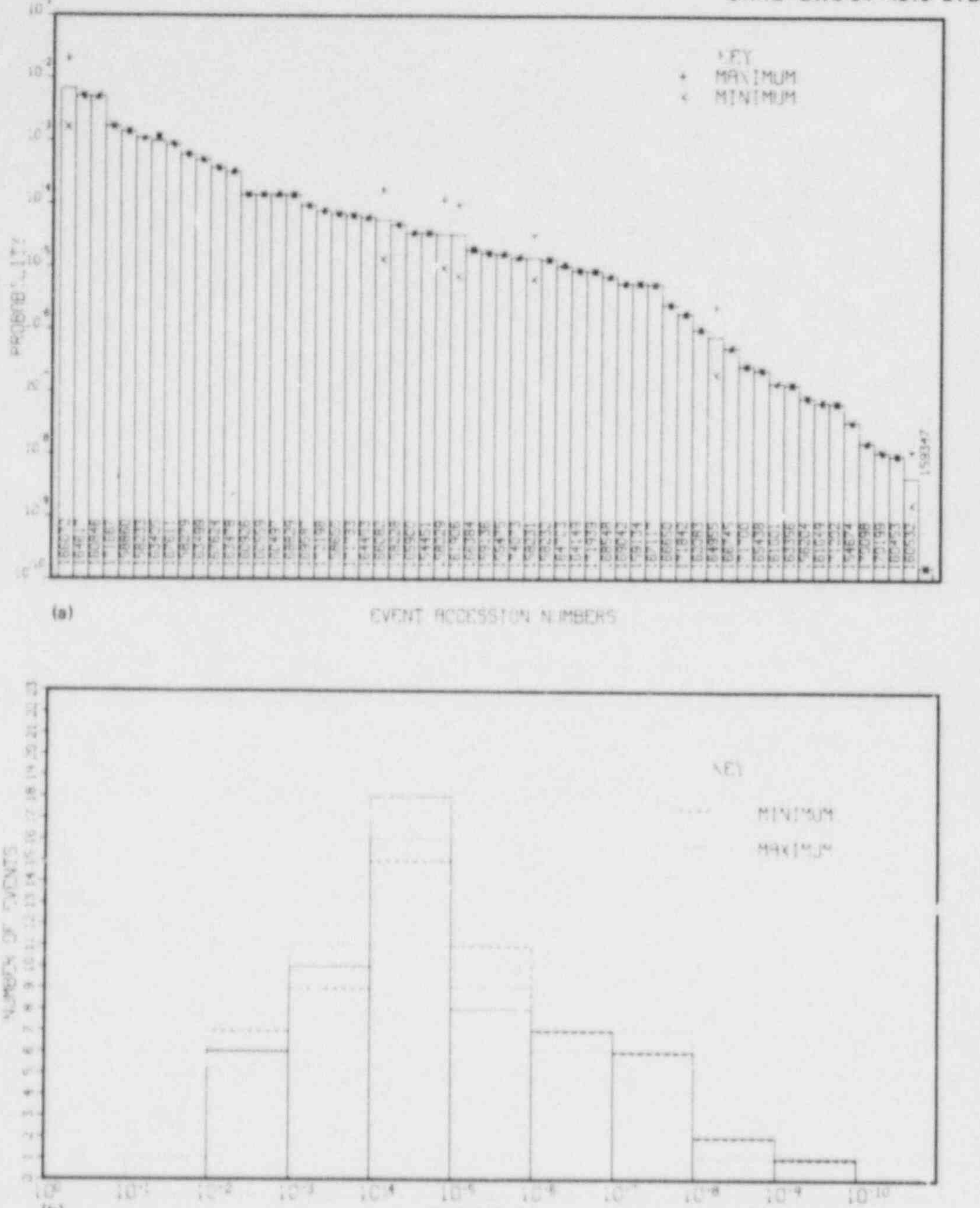


Fig. E.30. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR LOFW numeric values.

ORNL-DWG 84-4211 ETD

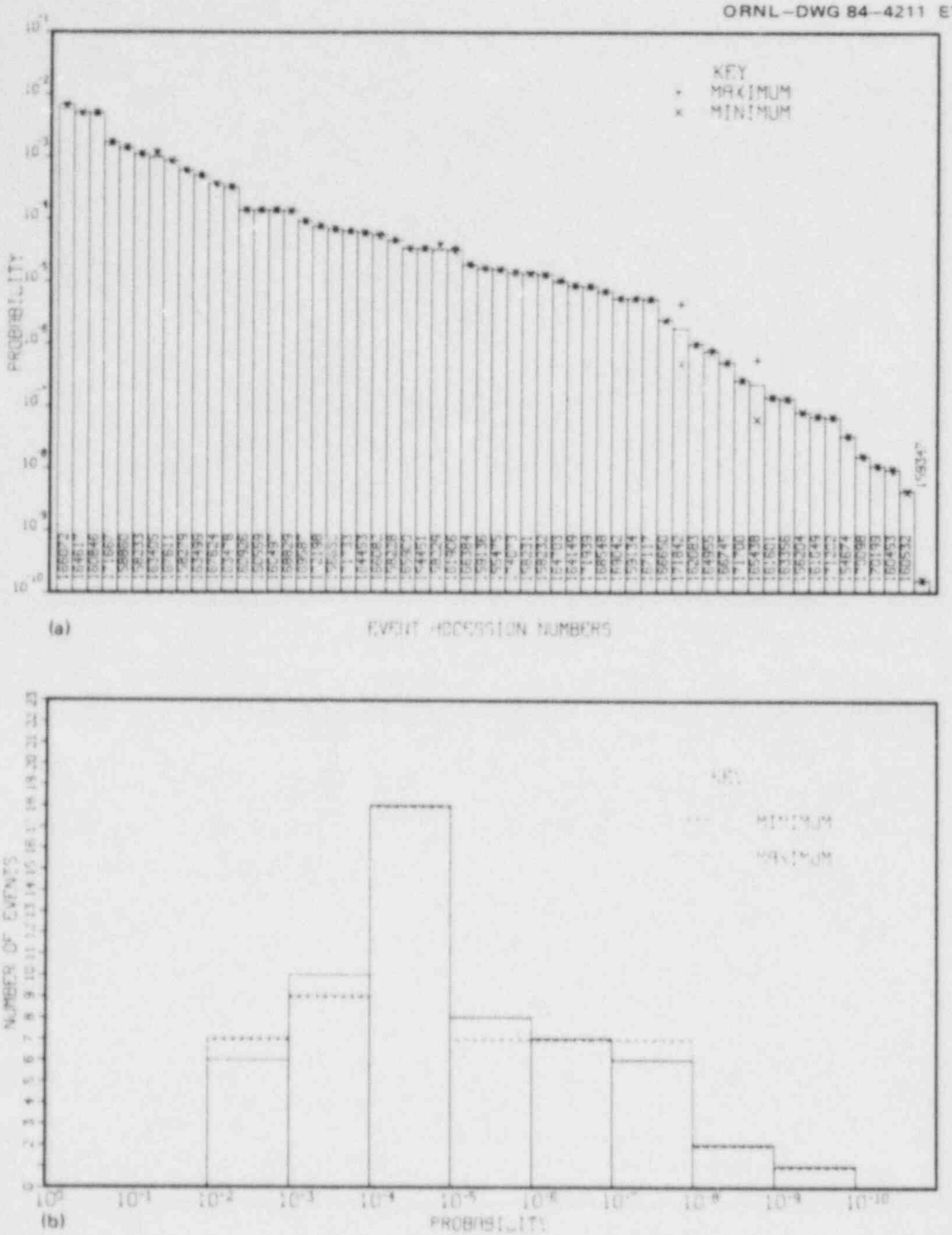


Fig. E.31. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR LOOP numeric values.

ORNL-DWG 84-4212 ETD

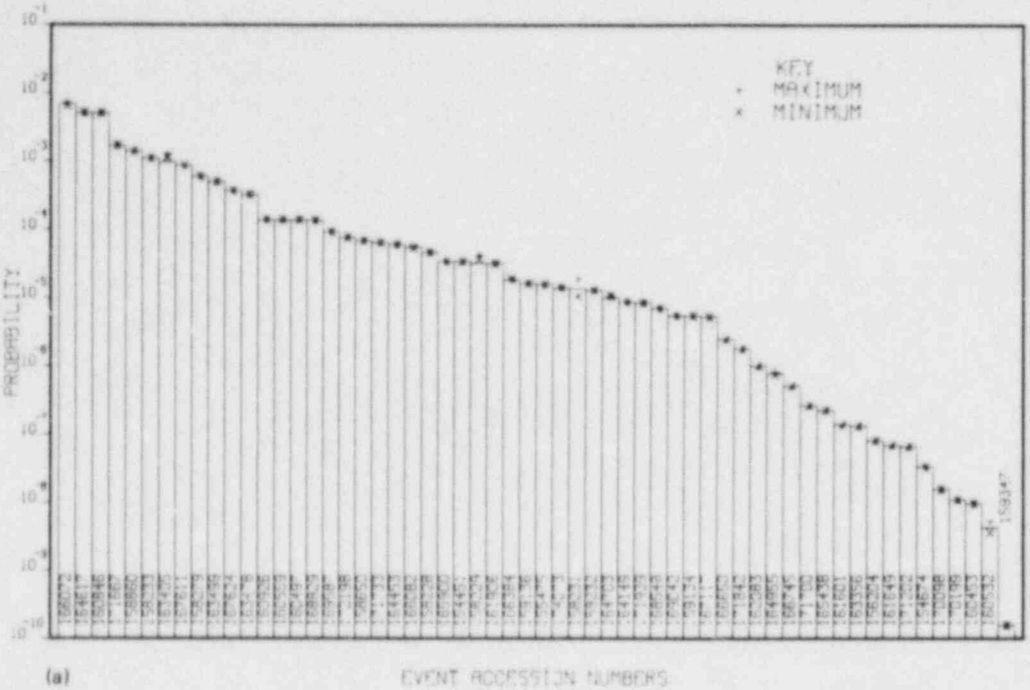
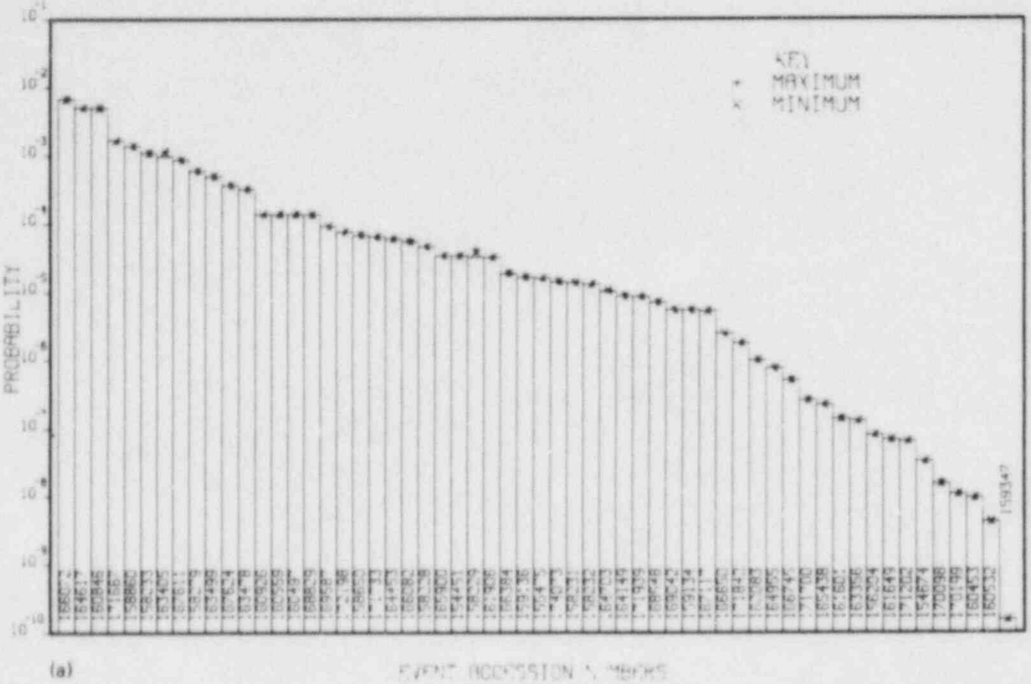


Fig. E.32. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR LOCA numeric values.

ORNL-DWG 84-4213 ETD



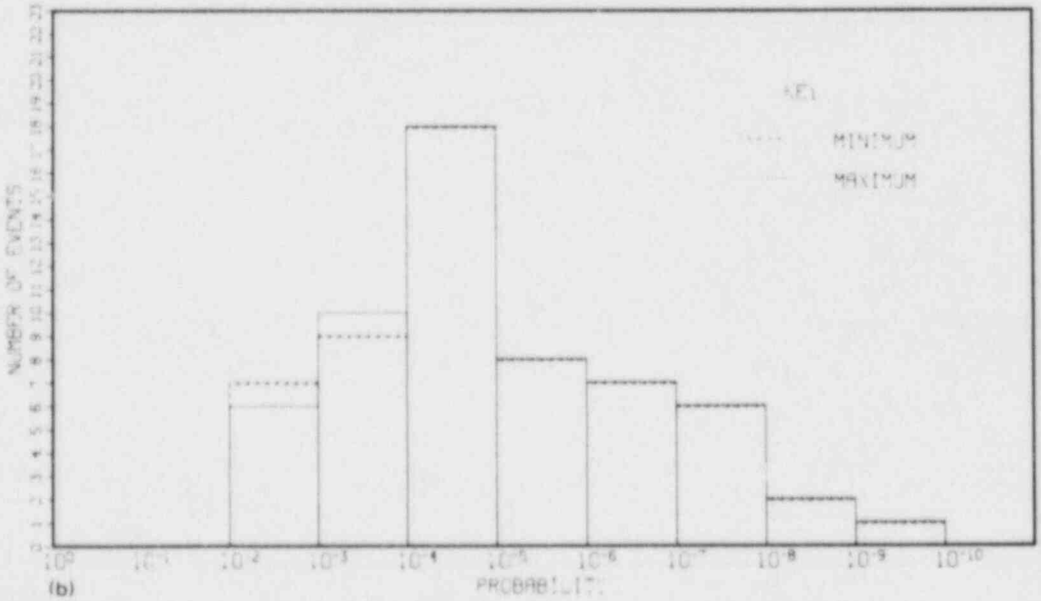
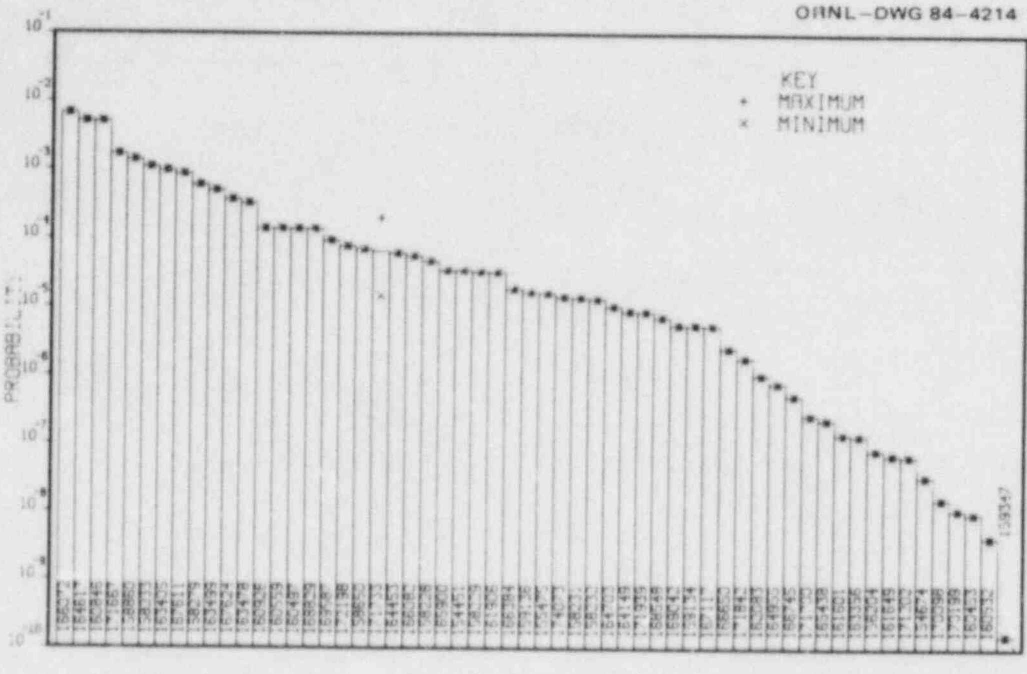
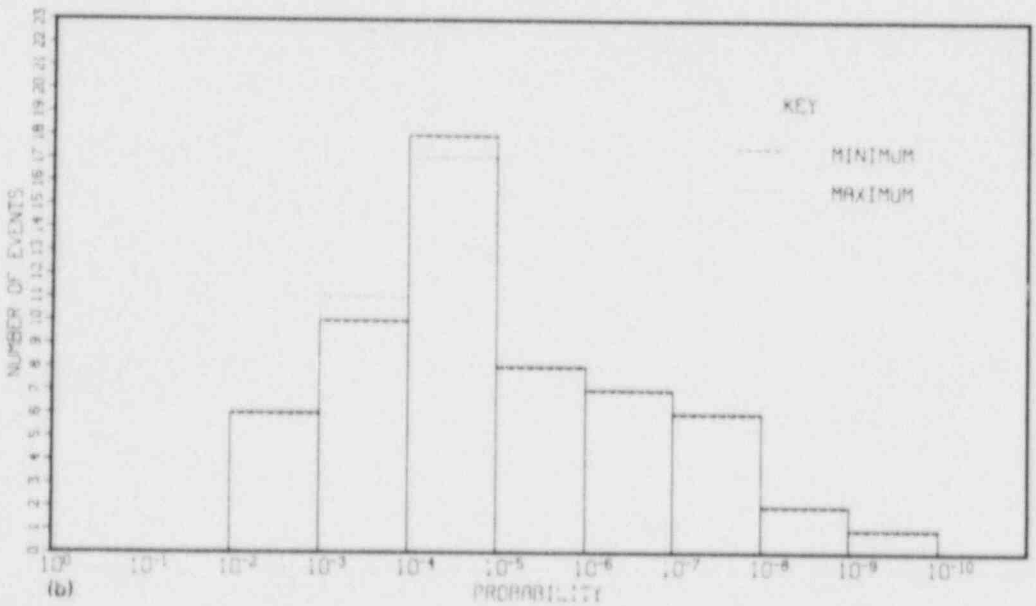


Fig. E.33. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in BWR large SLB numeric values.

ORNL-DWG 84-4214 ETD



(a)



(b)

Fig. E.34. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR LOFW numeric values.

ORNL-DWG 84-4215 ETD

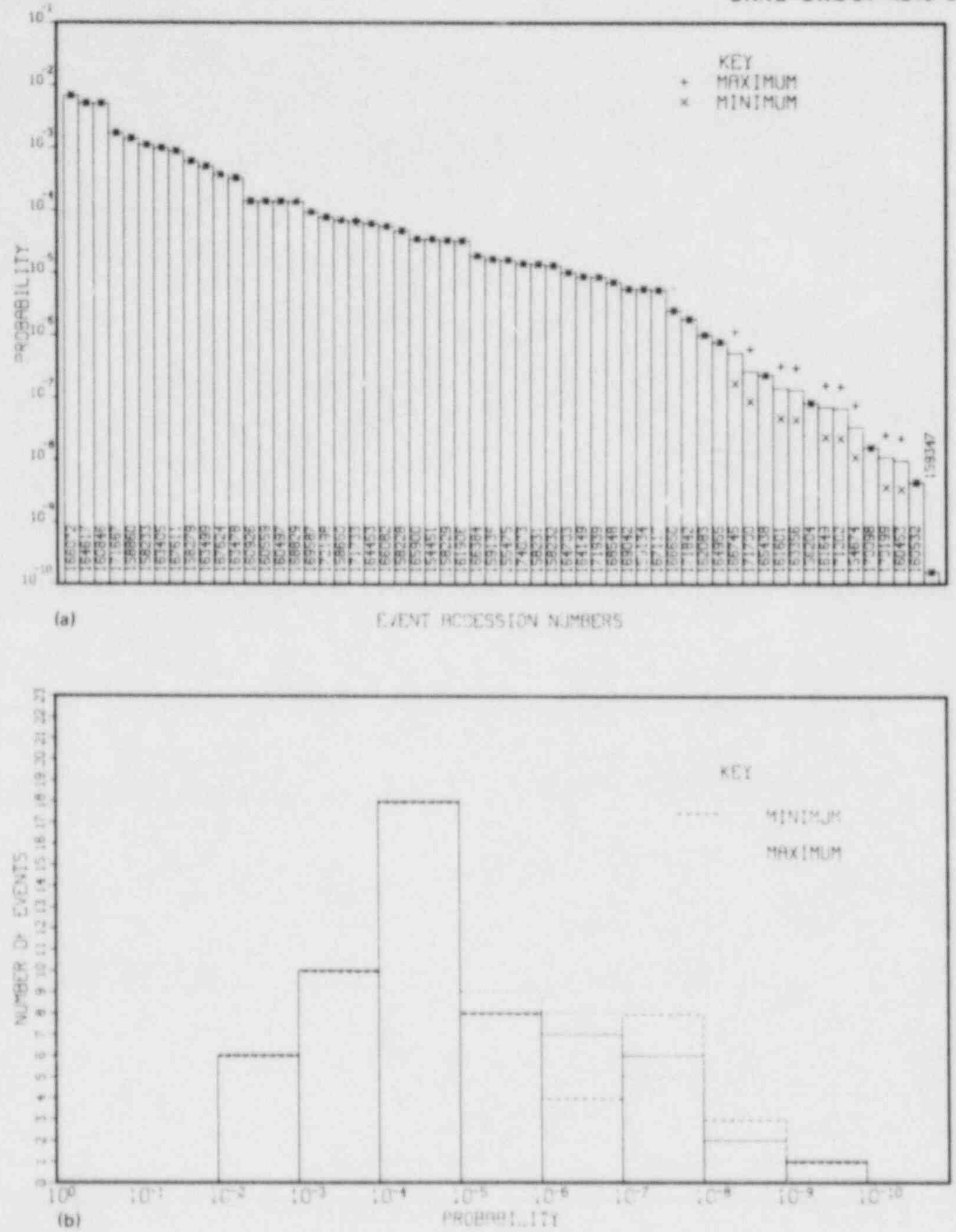


Fig. E.35. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR LOOP numeric values.

ORNL-DWG 84-4216 ETD

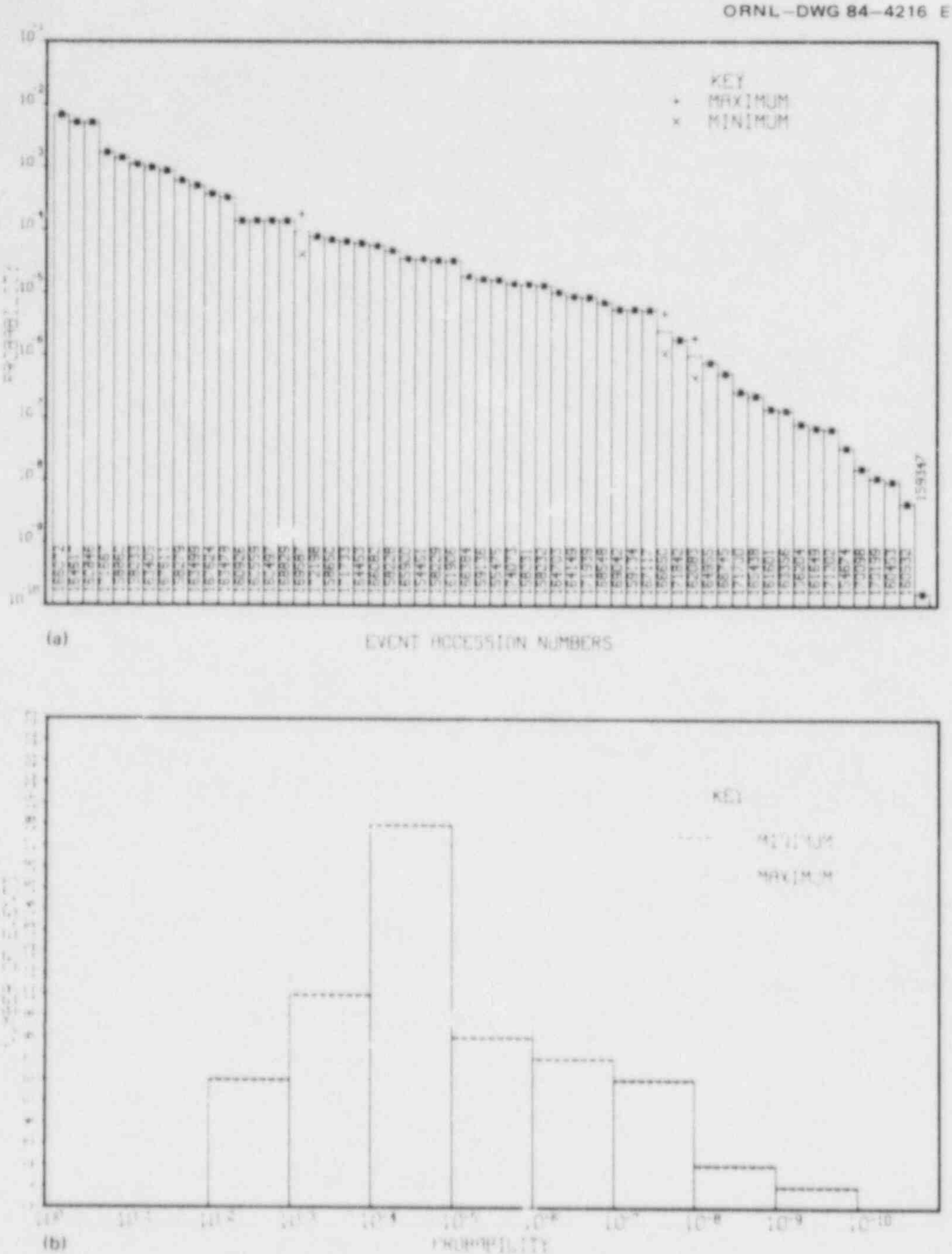


Fig. E.36. Impact on (*a*) event ranking and (*b*) event probability distribution based on variation in PWR LOCA numeric values.

Appendix F

DRAFTS, REVIEWS, AND REVIEWERS

There were two prior drafts of this document: a preliminary draft issued February 25, 1983, and a draft issued in July 1983. The preliminary draft identified only the events selected at that time (essentially the same as in this final report) together with the complete documentation of each. The preliminary draft was widely distributed to EPRI, INPO, the affected utilities, and the ASP Review Team, as well as within ORNL, SAI, and the Nuclear Regulatory Commission. The ASP Review Team is a group of probabilistic risk assessment experts assembled by the project to review project activities. Distribution of the July draft, which included details of the methods as well as of the sensitivity analysis, was somewhat less widespread but did include ORNL, SAI, and the NRC, as well as the ASP Review Team.

Lists of individual and corporate reviewers who submitted written comments on one or both drafts and dates reviews were received are included in this appendix. In addition to the persons listed and the authors' organizations, numerous other individuals and organizations were given copies of the draft and were invited to comment, including all participants in the Industry Workshop on the Accident Sequence Precursor Program held at EPRI, February 28–March 1, 1983 (these potential reviewers are not listed).

The listing of any individual or organization here is not meant to imply that that individual or organization concurs with all or any portion of the contents of this document. The methods, evaluations, and findings reported herein are entirely the responsibility of the authors. The authors, however, benefited immeasurably from the many comments of these reviewers; the cooperation of these reviewers in this endeavor is appreciated.

The ASP Review Team

In FY 1983 the ASP Project established a three-member review team of nationally recognized probabilistic risk assessment experts to review the activities of the project. The team met on two occasions (May 28 and August 31, 1983) to review the February 25, 1983, and July 1983 drafts, respectively. They also reviewed and commented on a preliminary copy of the final report. Team members are listed below with their current affiliations, although they participated on the Review Team as individuals.

Kenneth S. Canady, Duke Power Company

Norman C. Rasmussen, Massachusetts Institute of Technology

William E. Vesely, Battelle Columbus Laboratories

Utilities

Copies of applicable portions of the February 25, 1983, draft, which contained completed descriptions of all the selected 1980–81 events except for quantification of conditional probability of severe core damage, were distributed to the 25 affected nuclear utilities during the summer and early fall of 1983. Responses were received from 21 of these 25 utilities concerning 53 of the 61 different events. The information in these responses was then used by the ASP Project staff to supplement already existing information on the events. The utilities that responded are listed below.

Arkansas Power and Light Company, June 17, 1983

Baltimore Gas and Electric Company, June 13, 1983

Boston Edison Company, September 14, 1983

Carolina Power and Light Company, June 22, June 27, and December 1, 1983

Commonwealth Edison Company, June 14, 1983

Consolidated Edison Company of New York, Inc., June 15, 1983

Consumers Power Company, November 29, 1983

Dairyland Power Cooperative, June 8, 1983

Duke Power Company, April 14, 1983

Duquesne Light Company, June 2, 1983

Florida Power and Light Company, November 29, 1983

Florida Power Corporation, June 10, 1983

Georgia Power Company, June 20, 1983

Iowa Electric Light and Power Company, December 14, 1983

Northeast Nuclear Energy Company, June 10, 1983

Portland General Electric Company, June 15, 1983

Public Service Electric and Gas Company, June 17, 1983

Sacramento Municipal Utility District, June 10, 1983

Southern California Edison, June 27, 1983

Toledo Edison Company, June 17, 1983

Virginia Electric and Power Company, June 3, 1983

Nuclear Regulatory Commission

Copies of both the February 25, 1983, and the July 1983 drafts were distributed within the Nuclear Regulatory Commission for review and comment. Extensive comments were received from the NRC on both drafts.

Division of Accident Evaluation, September 20, 1983

Division of Engineering Technology, October 7, 1983

Division of Human Factors Safety, May 3, 1983

Division of Licensing, May 20, 1983

Division of Project and Resident Programs, October 4, 1983

Office of Resource Management, October 5, 1983

Division of Safety Technology, April 23 and September 14, 1983

Division of Systems Interaction, May 3, 1983

Office for Analysis and Evaluation of Operational Data, June 7, 1983

Other Organizations

Combustion Engineering, Inc., April 15, 1983

Institute of Nuclear Power Operations, May 13, 1983

National Centre of Systems Reliability, October 28, 1983

University Research Foundation (University of Maryland), July 27 and September 21, 1983

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 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.

Actual occurrence event tree. An event tree describing a historic sequence of events as it actually occurred at a particular plant.

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.

β-factor (method). The fraction of the total failure probability attributable to dependent failures when a two-train system is modeled using the β-factor method. (This model is described in more detail in Sect. 3.2.)

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, vessels).

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

Consequently degraded function. A function was considered consequently degraded if a component failure external to the function resulted in loss of the function's redundancy (e.g., if an auxiliary feedwater train was rendered unavailable during a potential loss of offsite power because of a diesel generator failure).

Consequently failed function. A function was considered consequently failed if (1) it failed because of the failure of another function or (2) it failed because of 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 high-pressure injection function during a postulated loss of offsite power due to the unavailability of one of two HPI pumps plus the unavailability of the diesel generator 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 there is insufficient core cooling to prevent the core from heating up to a temperature at which core materials melt.

Degraded function. A function with failed components which is operable but in which no additional redundancy exists.

Demand. A test or an operating condition that requires the availability of a component or a system. In this study, it includes actuations required because of the initiating events that were accounted for. One demand consisted of the actuation of all redundant components in a function, 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. That sequence in a set of sequences which 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 loss-of-coolant accident.

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-specific event tree. See *actual occurrence event tree*.

Event tree. A logic model which represents dependencies that exist 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).

Function. A system or combination of systems that performs a particular task. For example, in this study, the task of providing secondary-side cooling in the event of a loss of main feedwater was ascribed to the auxiliary feedwater and secondary heat removal function. This function includes the auxiliary feedwater system, the secondary relief valves, systems that provide required cooling water to the auxiliary feedwater system, and auxiliary feedwater control systems.

Functional event tree. A logic model in which dependencies that exist and combinations of actions required to achieve end states are described at the functional level.

Generic event tree. See *standardized event tree*.

Human error. Failure of a human to perform a given task within task constraints (note that this includes errors of omission as well as errors of commission).

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.

Importance. A ranking of components, systems, or events from a number of such entities based on the relative contribution of that entity toward some top event.

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 by utilities who operate nuclear plants to the NRC 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-0161 on Licensee Event Reports.

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.*

Probabilistic risk assessment. An analytical technique for integrating diverse aspects of design and operation to assess risk and develop an information base for analyzing plant-specific and/or generic issues.

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.

Risk achievement worth. The increase in risk of an event if some feature (function) were assumed to be absent or were assumed to be failed.

Risk reduction worth. The decrease in risk of an event if some feature (function) were assumed to be optimized or were assumed to be made perfectly reliable.

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.*

Standardized event trees. The set of functional event trees developed in this study to describe event sequences of interest for four initiating events: loss of main feedwater, loss of offsite power, small loss-of-coolant accident, and steam line break.

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. Analyses that provide 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 turbine-generator, auxiliaries, and engineered safety features.

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16. ABSTRACT (200 words or less)
Descriptions of fifty-eight operational events reported as Licensee Event Reports, which occurred at commercial light-water reactors during 1980-1981 and which are considered to be precursors to potential severe core damage, are presented, along with associated event trees, categorization, and subsequent analyses. This study is a continuation of the work presented in NUREG/CR-2497 which somewhat similarly evaluated the 1969-1979 events. The current study incorporates improvements which evolved from an assessment of the comments on the earlier report and applies these in the assessment of the LERs which occurred during 1980 and 1981. The report sequentially discusses (1) the general rationale for this study, (2) the program methods for LER review and documentation, (3) the calculation of function failure probabilities and initiating event frequencies based upon precursor data, (4) the use of the conditional probability of subsequent severe core damage estimates to rank precursor events and estimate an average industrywide risk of severe core damage, and (5) the conduct of sensitivity analyses on these results. There was some apparent decrease in most initiating event frequencies and function failure probabilities in the 1980 and 1981 period, as compared to the earlier report. Although it was not possible to conclude that these decreases were statistically significant, they did result in a reduction in the industry average estimated severe core damage frequency for 1980-1981 as compared to the 1969-1979 period.

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