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Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995

Idaho National Engineering and Environmental Laboratory

U.S. Nuclear Regulatory Commission
Office for Analysis and Evaluation of Operational Data
Washington, DC 20555-0001

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Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995

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Prepared by
J. P. Poloski, D. G. Marksberry*, C. L. Atwood, W. J. Galyean

Idaho National Engineering and Environmental Laboratory
Lockheed Martin Idaho Technologies Company
Idaho Falls, ID 83415-3129

*Nuclear Regulatory Commission

Prepared for
Safety Programs Division
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
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ABSTRACT

This report was produced at the Idaho National Engineering and Environmental Laboratory for the U.S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data. Data for all unexpected reactor trips during power operations at commercial nuclear power plants from 1987 through 1995 were reviewed. Each event was reviewed and categorized according to the initial event and, additionally, was marked if certain other risk-significant events occurred, regardless of their position in the event sequence. The collected data were analyzed for time dependence, reactor-type dependence, and between-plant variance. Dependencies and trends are reported, along with the raw counts and the best estimate for 1995 initiating event frequencies. For some initiators whose frequencies are low enough that no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used to estimate frequencies. These included operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as loss-of-coolant accidents (LOCAs). Results of engineering analyses of the operating experience are compared with probabilistic risk assessment/individual plant examinations (PRA/IPEs) and other regulatory issues.

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EXECUTIVE SUMMARY

This report presents an analysis of initiating event frequencies at United States (U.S.) nuclear power plants. The evaluation is based primarily on the operating experience from 1987 through 1995 as reported in Licensee Event Reports (LERs). The objectives of the study are: (1) provide revised, historical frequencies for the occurrence of initiating events in U.S. nuclear power plants, (2) compare these estimates based on operating experience to estimates used in probabilistic risk assessments (PRAs), individual plant examinations (IPEs), and other regulatory issues; and (3) review the operating data from an engineering perspective to determine trends and patterns of plant performance on a plant-type [i.e., pressurized water reactor (PWR) or boiling water reactor (BWR)], plant-specific, and industry-wide basis.

This study used as one of its sources of data the operating experience from 1987 through 1995 as reported in LERs. The Sequence Coding and Search System (SCSS) database was used to identify LERs for review and classification for this study. Each LER was reviewed from a risk and reliability perspective by an engineer with nuclear power plant experience. Based on the LER review, approximately 2,000 reactor trip events were analyzed with regard to their effect on plant performance.

For some initiators whose frequency is low enough that no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used to estimate their frequencies. These included operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as loss-of-coolant accidents (LOCAs).

Major Findings

This report provides information on frequencies, trends, and between-plant variation for initiating events. An evaluation of the results indicates that:

- Combined initiating event frequencies for all initiators calculated from the 1987–1995 experience are lower than the frequencies used in NUREG-1150, *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*, and IPEs by a factor of five and four, respectively.

- General transients constitute 77% of all initiating events. Events that pose a more severe challenge to the plant's mitigation systems (nongeneral transients) constitute the remaining 23%.

- Over the nine-year span considered by this report, either a decreasing or constant time trend was observed for all categories of events. A decreasing trend was identified in approximately two-thirds of the more risk-significant categories that had sufficient data for trending analysis. The overall initiating event frequency decreased by a factor of two to three during the nine-year span. Most risk-significant initiator frequencies (such as total loss of feedwater flow, loss of instrument or control air, inadvertent closure of all main steam isolation valves (MSIVs), and total loss of condenser heat sink for BWRs) decreased at a faster rate than the overall initiating event frequency.

- Loss-of-coolant accident frequencies are lower than those used in NUREG-1150 and industry-wide IPEs.

- The frequencies (per critical year) estimated from the 1987–1995 experience for the risk-significant categories and general transients are the following. All but the first show a decreasing trend, and the values presented here apply to 1995.

–	Loss of Offsite Power (PWR and BWR)	4.6E-2
–	Total Loss of Condenser Heat Sink: PWR	1.2E-1
–	Total Loss of Condenser Heat Sink: BWR	2.9E-1
–	Total Loss of Feedwater Flow (PWR and BWR)	8.5E-2
–	General transients: PWR	1.2
–	General transients: BWR	1.5

For LOCA categories, the frequencies were evaluated using data and information prior to 1987 due to their relatively low frequency and the corresponding sparseness of data. No pipe break LOCA events were found in the U.S. operating experience. For the small pipe break LOCA frequency, the estimate from WASH-1400, *Reactor Safety Study,* was updated using U.S. reactor experience. For medium and large pipe break LOCAs, frequency estimates were calculated by using the frequency of leaks or through-wall cracks that have occurred which challenge the piping integrity. Further, conservative estimates were used for the probability of break given a leak (based on a technical review of information on fracture mechanics, data on high energy pipe failures and cracks, and assessment of pipe break frequencies estimated by others since WASH-1400). The pipe-break LOCA frequencies (per critical year) estimated from the experience are:

	Small LOCA	Medium LOCA	Large LOCA
PWR:	5E-4	4E-5	5E-6
BWR:	5E-4	4E-5	3E-5

No interfacing system loss-of-coolant accident (ISLOCA) events were identified in the U.S. operating experience.

Between-plant variation in initiating event frequencies was identified in the following categories: Total Loss of Condenser Heat Sink for PWRs, Loss of Condenser Vacuum for PWRs, Inadvertent Closure of All MSIVs for BWRs, Total Loss of Feedwater Flow, and General Transients for BWRs and for PWRs. Several plants whose uncertainty interval of the mean are statistically significantly higher than the industry average (i.e., the uncertainty interval is located completely to the right of the industry average mean) for several risk-significant categories have been identified. A listing of these plants is provided in Table 4-4 in the main report.

A comparison was made between initiating event frequencies based on the 1987–1995 operating experience for non-LOCA categories and the corresponding values from PRA/IPEs. Based on the cumulative mean frequency of the initiating events, the IPE-wide frequency is higher (approximately a factor of four) than the frequency estimated from operating experience. Table ES-1 provides a comparison of the operating experience to the average of the IPE population.

The mean frequencies calculated from the 1987–1995 operating experience for non-LOCA events have generally decreased by a factor of two as compared with the mean frequencies from NUREG/CR-3862, *Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments*, and NUREG-1150, which were based on experience at the time of the studies.

Table ES-1. Initiating event frequencies (per critical year) based on operating experience compared to the average of the IPE population.

	PWR Frequency—Mean (per critical year)[b]		BWR Frequency—Mean (per critical year)[b]	
Description	Operating Experience[a]	IPE[c]	Operating Experience[a]	IPE[c]
Small Pipe Break LOCA (G3)	5E-4[d]	9.2E-3	5E-4[d]	1.0E-2
Steam Generator Tube Rupture (F)	7.0E-3	2.0E-2	—	—
Loss of Offsite Power (B)	4.6E-2[d]	1.0E-1	4.6E-2[d]	1.3E-1
Total Loss of Condenser Heat Sink (L)	1.2E-1	3.0E-1	2.9E-1	4.3E-1
Total Loss of Feedwater Flow (P)	8.5E-2[d]	1.0E+0	8.5E-2[d]	5.7E-1
General Transients (Q)	1.2E+0	4.0E+0	1.5E+0	6.0E+0

a 1987–1995 experience except for Small Pipe Break LOCA category which included total U.S. operating experience (1969–1997)

b. Units are in per critical year. One critical year equals 8,760 hours of reactor criticality.

c. The values are the mean of the IPE population for the plant type (PWR or BWR) The units stated in the IPE are per calendar year For comparison purposes, the per calendar year was converted to critical year One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study. Therefore the rate per critical year equals the rate per calendar year divided by 0.75

d The estimate did not differentiate with respect to plant type (i e , PWR and BWR); therefore the value is same for either plant type.

The mean frequencies based on 1987–1995 experience are lower than the means from NUREG/CR-3862 and NUREG-1150 by a factor of four or more for the following categories: Loss of Offsite Power for BWRs and PWRs, and General Transients for BWRs and PWRs. (Note: NUREG-1150 used frequencies for non-LOCA categories from NUREG/CR-3862.)

The total initiating event frequency for BWRs and PWRs has decreased by about a factor of five and eight, respectively, since the NUREG/CR-3862 study was published in 1985.

Table ES-2 provides a comparison of the operating experience to the values reported in NUREG/CR-3862 and NUREG-1150.

A comparison was made with the frequencies used in the Anticipated Transients Without Scram (ATWS) Events Rulemaking (SECY-83-293). The frequency of ATWS transient initiators calculated from the 1987–1995 operating experience has decreased since the ATWS Rulemaking analysis was completed in 1983. This decrease indicates that the frequency of challenges that could result in a severe ATWS event has declined. The SECY-83-293 ATWS initiating frequencies would be reduced approximately by a factor of three for the PWR vendors while the BWR vendor is reduced by about a factor of four when updated with initiating event frequencies from this study. Assuming the average failure to scram probability used in SECY-83-293, the probability of ATWS per calendar year for PWRs and BWRs based on 1987–1995 experience and SECY-83-293 are as follows:

	PWR	BWR
• 1987–1995 experience	8.4E-6	3.3E-6
• SECY-83-293	2.4E-5	1.2E-5

Table ES-2. Initiating event frequencies (per critical year) based on operating experience compared to NUREG/CR-3862 and NUREG-1150.

Description	Operating Experience[a]	NUREG/CR-3862[c]	NUREG-1150[c]
	Mean Frequency (per critical year)[b]		
Small Pipe Break LOCA (G3)	5E-4[d]	—	1.3E-3[d]
Steam Generator Tube Rupture (F)	7.0E-3	—	1.0E-2
Loss of Offsite Power (B)—PWR	4.6E-2[d]	1.9E-1	1.9E-1
Loss of Offsite Power (B)—BWR	4.6E-2[d]	1.1E-1	1 1E-1
Total Loss of Condenser Heat Sink (L)—PWR	1.2E-1	2.4E-1	2.4E-1
Total Loss of Condenser Heat Sink (L)—BWR	2.9E-1	9.1E-1	9.1E-1
Total Loss of Feedwater Flow (P) —PWR	8.5E-2[d]	2.2E-1	2.2E-1
Total Loss of Feedwater Flow (P) —BWR	8.5E-2[d]	9.3E-2	9.3E-2
General Transient—PWR (Q)	1.2E+0	1.0E+1	1.0E+1
General Transients—BWR (Q)	1.5E+0	8.6E+0	8.6E+0
Total of all events—PWR	1.4E+0	1.1E+1[e]	1.1E+1[e]
Total of all events—BWR	1.8E+0	9.7E+0[e]	9.9E+0[e]

a 1987–1995 experience except for Small Pipe Break LOCA category which included total U S operating experience (1969–1997)

b Units are in per critical year One critical year equals 8,760 hours of reactor criticality

c The units stated in the report are per reactor year (i e , numbers of years from start of commercial operation) For comparison purposes, the per reactor year was converted to critical year One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study Therefore the rate per critical year equals the rate per calendar year divided by 0.75

d The estimate did not differentiate with respect to plant type (i e , PWR and BWR); therefore the value is same for either plant type

e. This total represents the sum of all frequencies presented in the referenced report.

FOREWORD

This report provides information relevant to initiating events of unplanned, automatic and manual reactor trips. The results, findings, conclusions, and information contained in this and related reliability studies conducted by the Office for Analysis and Evaluation of Operational Data are intended to support several risk-informed regulatory activities. These reports can provide information on relevant operating experience that can be used to enhance plant inspections of risk-important systems. In addition, this information can be used to support staff technical reviews of proposed license amendments, including risk-informed applications. This work also will be used in the development of risk-based performance indicators.

Findings and conclusions from the analyses of the rates of initiating events during the 1987-1995 time period at domestic nuclear power plants are presented in the Executive Summary. The analysis of certain rare or infrequent initiating event categories, such as loss-of-coolant accidents (LOCAs), are based on U.S. and world-wide experience and cover periods before 1987 and after 1995 as well. The quantitative analysis and engineering analysis are presented in Sections 3 and 4, respectively. This report provides an indication of how performance varies among plants. The information to support risk-informed regulatory activities involving unplanned, automatic and manual reactor trips is summarized in Table P-1. This table provides a condensed index of risk-important data and results presented in discussions, tables and figures.

Based on knowledge gained from the operating experience and the need to provide updated frequencies for NRC PRA programs, the task to update pipe break LOCA frequency estimates was included as an objective of this report. The goal of this effort is to refine the original estimates based on operating experience and current knowledge of pipe break mechanisms. It is recognized that the approach in this report will result in reduction of unnecessary conservatism in LOCA frequency estimates. However, the result is still conservative. Further probabilistic evaluations of the results from fracture mechanics research is required to develop best estimates of pipe break LOCA frequencies that factors in the evaluation current operating, surveillance, and maintenance practices at U.S. nuclear power plants.

For a perspective on the implications of these initiating event frequencies on overall plant risk, it is necessary to also consider other factors such as system and component reliabilities and common-cause failure probabilities. The paper, *Indications of U.S. Nuclear Industry Trends from the Risk-based Analysis of Operating Experience*,[#] provides some perspective on the implications of the findings of this report with respect to overall risk.

Additional insights may be gained about plant-specific performance by examining the specific events in light of the overall industry group performance. In addition, a review of recent experience in the licensee event reports (LERs) will determine whether performance has undergone any significant change since the last year of this study. The

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[#] Patrick W. Baranowsky, "Indications of U.S. Nuclear Industry Trends from the Risk-Based Analysis of Reactor Operating Experience" (A. Mosleh and R.A. Bari, eds.), *Probabilistic Safety Assessment and Management (PSAM4): Proceedings of the Fourth International Conference on Probabilistic Safety Assessment and Management, 13-18 September 1998*, Springer-Verlag, London, 1998.

LERs used in the analyses are listed in Appendix D in the report. A search of the LER database can be conducted through the NRC's Sequence Coding and Search System (SCSS) to identify the initiating events that occurred after the period covered by this report. SCSS contains the full text LERs and is accessible by NRC staff from the SCSS home page (http://scss.ornl.gov/). Nuclear industry organizations and the general public can obtain information from the SCSS on a cost recovery basis by contacting the Oak Ridge National Laboratory.

The NRC plans to periodically update the information in this report.

Charles E. Rossi, Director
Safety Programs Division
Office for Analysis and Evaluation
of Operational Data

Table P-1. Summary of risk-important information specific to initiating events.

Lists of LERs used to estimate initiating event frequencies	Appendix D: Tables D-5 through D-9
Frequency estimates of risk-significant events	Table 3-1; Section 3.2.1
Time trends for risk-significant event categories	Section 4.2; Figures 4-1 through 4-7
List of plants having mean frequencies greater than industry average for risk-significant event categories	Tables 4-2, 4-3, 4-4; Appendix G: Figures G-1 through G-6
Plant-specific frequencies of event categories with plant-to-plant variations	Appendix G: Tables G-6 through G-11
Summary of experience from rare events, such as: o pipe break LOCAs o interfacing system LOCA o steam generator tube ruptures o reactor coolant pump seal LOCA o stuck open safety/relief valves o anticipated transient without scram o loss of safety-related cooling water system	Sections: 4.4.1 4.4.2 4.4.3 4.4.4 4.4.5 4.4.6 4.4.7
Dominant contributors to risk-significant events, such as: o total loss of condenser heat sink o loss of condenser vacuum o inadvertent closure of all main steam isolation valves o total loss of main feedwater flow Insights from manual reactor trips and dual unit trips	Section 4.5.2
Insights from the conditional occurrence of risk-significant events that follow various reactor trip initiators	Section 4.5.3; Appendix D: Table D-13
Data of through-wall cracks in primary pressure boundary piping used to estimate pipe break LOCA frequencies	Appendix J: Tables J-11, J-12
Note: Plant name and docket numbers are provided in Tables K-1 and K-2 (Appendix K).	

ACKNOWLEDGMENTS

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The LOCA analysis (Appendix J) benefited greatly from the efforts of Vikram N. Shah and A. G. (Jack) Ware.

Technical reviews by Timothy J. Leahy of the INEEL, Dennis C. Bley of Buttonwood Consulting, Alan M. Kolaczkowski of SAIC, and Robert Bertucio of Scientech contributed substantially to this report.

ACRONYMS

ac	alternating current
ADS	automatic depressurization system
AEOD	Office for Analysis and Evaluation of Operational Data
ATWS	Anticipated Transient Without Scram
B&W	Babcock & Wilcox
BWR	boiling water reactor
CE	Combustion Engineering
dc	direct current
DEGB	double-ended guillotine break
EPRI	Electric Power Research Institute
ESF	engineered safety feature
FI	functional impact
GSI	generic safety issue
HAZ	heat affected zone
HPSI	high pressure safety injection system
I&C	instrumentation and control
IGSCC	intergranular stress corrosion cracking
INEEL	Idaho National Engineering and Environmental Laboratory
IPE	individual plant examination
IPF	initial plant fault
ISLOCA	interfacing system loss-of-coolant accident
LBLOCA	large (pipe) break loss-of-coolant accident
LER	licensee event report
LLNL	Lawrence Livermore National Laboratory

LOCA	loss-of-coolant accident
LOFW	loss of (main) feedwater
LOHS	loss of (condenser) heat sink
LOSP	loss of offsite power
LWR	light water reactor
MBLOCA	medium (pipe) break loss-of-coolant accident
MLE	maximum likelihood estimate
MSIV	main stream isolation valve
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PNNL	Pacific Northwest National Laboratory
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
Rx	reactor
SCSS	Sequence Coding and Search System
SGTR	steam generator tube rupture
SI	special interest
SBLOCA	small (pipe) break loss-of-coolant accident
SLOCA	small loss-of-coolant accident
SRV	safety relief valve
SWS	service water system

TBV	turbine bypass valve
VSLOCA	very small loss-of-coolant accident

Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995

1. INTRODUCTION

This report is the product of a study conducted by the Technical Assistance in Reliability and Risk Analysis Program (Job Code Number: E8246). It was sponsored by the U.S. Nuclear Regulatory Commission's (NRC's) Office for Analysis and Evaluation of Operational Data (AEOD) and written at the Idaho National Engineering and Environmental Laboratory (INEEL).

1.1 Purpose

This report presents an analysis of initiating event frequencies at United States (U.S.) nuclear power plants. The evaluation is based primarily on the operating experience from 1987 through 1995, as reported in Licensee Event Reports (LERs). The objectives of the study are: (1) provide revised, historical frequencies for the occurrence of initiating events in U.S. nuclear power plants; (2) compare these estimates based on operating experience to estimates used in probabilistic risk assessments (PRAs), individual plant examinations (IPEs), and other regulatory issues; and (3) review the operating data from an engineering perspective to determine trends and patterns of plant performance on a plant-type [i.e., pressurized water reactor (PWR) or boiling water reactor (BWR)], plant-specific, and industry-wide basis.

One of the sources of data used in this study was the operating experience from 1987 through 1995 as reported in LERs. The Sequence Coding and Search System (SCSS) database was used to identify LERs for review and classification for this study. Each LER was reviewed from a risk and reliability perspective by an engineer with nuclear power plant experience. Based on the LER review, approximately 2,000 reactor trip events were analyzed with regard to their effect on plant performance.

For some initiators whose frequency is low enough that no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used to estimate their frequencies. These included operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as loss-of-coolant accidents (LOCAs).

1.2 Report Organization

Section 1 provides the purpose of the study. Section 2 describes the criteria used to determine which events were included in the study, how the categories were organized and defined, and how the events were classified. Section 3 presents the results of the frequency estimation for initiating events and the comparisons to the PRA/IPE information. Section 4 provides the results of the engineering analysis of the operational data. Section 5 contains the references.

There are eleven appendices:

Appendix A. Initial Plant Fault and Functional Impact Category Definitions

Appendix B. Category Cross-Reference Tables to Previous Studies

Appendix C. Licensee Event Report Selection, Categorization, and Quality Management

Appendix D. Detailed Sorting Results and Estimates of Initial Plant Fault Frequencies

Appendix E. Statistical Methods

Appendix F. Results of Testing for Time Trend and Plant Effect

Appendix G. Results Based on Data after the Learning Period, Including Plant-Specific Results and Time Trends

Appendix H. Calendar Hours, Operating Hours, and Criticality Factors

Appendix I. Summary of Infrequent Events Associated with a Reactor Trip

Appendix J. LOCA Frequency Estimates

Appendix K. Plant Name and Docket Number Tables.

2. EVENT CLASSIFICATION

This section describes the criteria used to determine which events were included in the study, how the data were organized and defined, and how the events were sorted.

2.1 Included Events

To be included in this study, an event had to meet all of the following criteria:

- Include an unplanned reactor trip (not a scheduled reactor trip on the daily operations schedule)
- Sequence of events starts when reactor is critical and at or above the point of adding heat
- Occur during the calendar years 1987 through 1995 inclusive
- Occur at a U.S. commercial nuclear power plant (excluding Fort St. Vrain and LaCrosse)
- Be reported by a Licensee Event Report (LER).

In addition to the above criteria, certain rare events, such as loss-of-coolant accidents (LOCAs), were supplemented with additional experience prior to 1987 and after 1995 to provide a better estimate (i.e., lower the uncertainty) of their frequencies than estimates based solely on the 1987–1995 experience. Rare and infrequent events are discussed further in Section 4.4.

2.2 Data Organization

Each reactor trip event was reviewed for the following information:

- All occurrences of risk-significant events in the reactor trip sequence that could impact the ability to remove reactor decay heat
- The first event in the sequence of events that causes or leads to the unplanned, automatic or manual reactor trip
- An occurrence of a manual reactor trip.

A database was created to collect and store this information into three groups or data sets:

- The functional impact group—contains one or more risk-significant events that occur during each reactor trip
- The initial plant fault group - contains the reactor trip event initiator for each reactor trip
- The special interest group - contains occurrences of events not included in the above, such as diesel starts and loads and manual reactor trips that occur after the event initiator.

Two lists of event categories that describe the transients typically used in PRAs were developed for review of the reactor trip events. Examples of risk-significant event categories from the functional impact group include loss of offsite power, loss-of-coolant accidents, and total loss of feedwater flow. Examples of reactor trip initiator categories from the initial plant fault group include an identical list of event categories used in the functional impact group and a list of general transient categories. These two data sets and the special interest group are discussed in detail in the following sections.

2.2.1 Functional Impact Group

Definition. The event categories in the functional impact group includes risk-significant events that could impact the ability to remove decay heat. The functional impact group contains 26 categories under 12 headings. The event categories used in previous reports on initiating event frequencies (EPRI 1982, Mackowiak 1985) were reviewed and sometimes regrouped into categories that matched event groupings typically used in recent PRAs. Only those events that could impact the ability to remove decay heat were included in functional impact group. Event categories classified as general transients were excluded in the functional impact group. General transients are a compilation of all reactor trip events that had no direct impact on mitigating systems' ability to remove decay heat.

The headings and categories associated with the functional impact group are listed in Table 2-1 and defined in Appendix A. Table B-1 in Appendix B provides a cross-reference of the categories used in this report with the categories used in previous reports.

Purpose. The purpose of the functional impact group is to determine the frequency at which risk-significant events are likely to occur in association with the reactor trip, regardless of their order in the reactor trip sequence. The results presented in the main body of the report are focused on the analysis and evaluation of risk-important event categories from the functional impact group. This focus was chosen by the NRC to support several risk-informed regulatory activities. Frequency estimates of functional impact categories are best suited for PRA analyses where the occurrence of a risk-significant event category (e.g., total loss of the main feedwater system or steam generator tube rupture) is not specifically modeled in the accident sequence event tree as a conditional failure. For this case, the frequency of a functional impact category (or groups of similar categories) is used as the initiating event frequency for quantification of the appropriate event tree.

Event classification. For each reactor trip, the analysts examined the sequence of events occurring any time before and shortly after the reactor trip. Each occurrence of an event from the table of functional impact categories was noted in the database for each reactor trip event. One or more functional impact events may be identified in a single reactor trip event sequence. However, a reactor trip sequence may have no functional impact events as would be expected for most reactor trips.

For example, consider the case where a total loss of feedwater flow causes a plant transient resulting in a reactor trip and turbine trip, and then the loss of offsite power (due to, for example, the failure to transfer the plant electrical power source from the main generator to the preferred offsite power source). The functional impact categories applicable for this reactor trip sequence are Loss of Offsite Power (category B1) and Total Loss of Feedwater Flow (category P1). The order in which the functional impact events occur is not considered in this study. The turbine trip event was not selected, since the event was not a functional impact category.

As discussed in the example above, a reactor trip sequence may have multiple occurrences of functional impact events. About 9% of all functional impact events are multiple occurrences. This will result in a slight increase in frequency estimates for selected categories. However, the increase in values are well within the uncertainty intervals estimated in the analysis. Nevertheless, the data for each functional impact category is

Table 2-1. Initial plant fault and functional impact headings and categories.

A (Reserved)

B Loss of Offsite Power
- B1 Loss of Offsite Power

C Loss of Safety-Related Bus
- C1 Loss of Vital Medium Voltage ac Bus
- C2 Loss of Vital Low Voltage ac Bus
- C3 Loss of Vital dc Bus

D Loss of Instrument or Control Air
- D1 Loss of Instrument or Control Air System

E Loss of Safety-Related Cooling Water
- E1 Total Loss of Service Water
- E2 Partial Loss of Service Water

F Steam Generator Tube Rupture
- F1 Steam Generator Tube Rupture

G Loss-of-Coolant Accident (LOCA)/Leak
- G1 Very Small LOCA/Leak
- G2 Stuck Open: 1 Safety/Relief Valve
- G3 Small Pipe Break LOCA
- G4 Stuck Open: Pressurizer PORV
- G5 Stuck Open: 2 or More Safety/Relief Valves
- G6 Medium Pipe Break LOCA
- G7 Large Pipe Break LOCA
- G8 Reactor Coolant Pump Seal LOCA: PWR

H Fire
- H1 Fire

J Flood
- J1 Flood

K High Energy Line Break
- K1 Steam Line Break Outside Containment
- K2 Feedwater Line Break
- K3 Steam Line Break Inside Containment: PWR

L Total Loss of Condenser Heat Sink
- L1 Inadvertent Closure of All MSIVs
- L2 Loss of Condenser Vacuum
- L3 Turbine Bypass Unavailable

M (Reserved)

N Interfacing System LOCA
- N1 Interfacing System LOCA

P Total Loss of Feedwater Flow
- P1 Total Loss of Feedwater Flow

Q General Transients (Other Initial Plant Fault[a])
- QC4 Loss of ac Instrumentation and Control Bus[a]
- QC5 Loss of Nonsafety-Related Bus[a]
- QG9 Primary System Leak[a]
- QG10 Inadvertent Open/Close: 1 Safety/Relief Valve[a]
- QK4 Steam or Feed Leakage[a]
- QL4 Loss of Nonsafety-Related Cooling Water[a]
- QL5 Partial Closure of MSIVs[a]
- QL6 Condenser Leakage[a]
- QP2 Partial Loss of Feedwater Flow[a]
- QP3 Total Loss of Condensate Flow[a]
- QP4 Partial Loss of Condensate Flow[a]
- QP5 Excessive Feedwater Flow[a]
- QR0 RCS High Pressure (RPS Trip)[a]
- QR1 RCS Low Pressure (RPS Trip)[a]: PWR
- QR2 Loss of Primary Flow (RPS Trip)[a]: PWR
- QR3 Reactivity Control Imbalance[a]
- QR4 Core Power Excursion (RPS Trip)[a]
- QR5 Turbine Trip[a]
- QR6 Manual Reactor Trip[a]
- QR7 Other Reactor Trip (Valid RPS Trip)[a]
- QR8 Spurious Reactor Trip[a]
- QR9 Spurious Engineered Safety Feature Actuation[a]

a. Initial plant fault only.

provided in Appendix D in this report to allow adaptation of industry average frequencies provided in this report for PRA-specific applications.

2.2.2 Initial Plant Fault Group

Definition. The initial plant fault is the first event in a sequence of events causing or leading to an unplanned, automatic, or manual reactor trip. The initial plant fault group contains 48 mutually exclusive categories under 13 headings. Twelve headings include risk-significant categories that could impact the ability to remove decay heat (e.g., loss of offsite power, loss-of-coolant accident, and total loss of condenser heat sink). These 12 headings and associated categories are identical to all of the risk-significant headings and categories used in the functional impact group. The initial plant fault group also includes an additional heading with 22 categories typically classified as general transients in PRAs. As described above, general transients are a compilation of all reactor trip initiators that had no direct impact on mitigating systems ability to remove decay heat. General transient-type categories used in previous reports on initiating event frequencies (EPRI 1982, Mackowiak et al. 1985) were modified in this study in order to develop a list of categories that better supports current PRAs.

The headings and categories associated with the initial plant fault group are listed in Table 2-1 and defined in Appendix A. Table B-1 in Appendix B provides a cross-reference of the categories used in this report with the categories used in previous reports.

Purpose. The events in the initial plant fault group are used in the engineering analysis section of the main report to develop insights from the conditional occurrences of risk-significant events. The events from the initial plant fault and functional impact groups were merged to compare the number of risk-significant events occurring after the reactor trip initiator (i.e., initial plant fault event).

Frequency estimates of initial plant fault categories are best suited for PRA analyses where the occurrence of one or more risk-significant event categories (e.g., total loss of the main feedwater system or steam generator tube rupture) are specifically included in the accident sequence event tree model as a conditional failure. The combination of these conditional functional and/or system successes and failures are depicted along the top heading across the event tree. For this type of event tree model, the frequency of a initial plant fault category (or a group of similar categories) is used as the initiating event frequency for quantification of the event tree. The conditional probability of a risk-significant event category subsequent to the initial plant fault event can be estimated from the data in the appropriate initial plant fault and functional impact categories. However, if a particular event category is not modeled in the event tree as a conditional event, then the frequency estimate of the functional impact category, which includes all occurrences of the event in the frequency estimate, may be more appropriate as the initiating event frequency for event tree quantification. The LER listing and frequency estimates of initial plant fault event categories are provided in the Appendix D of this report.

Event classification. For each reactor trip event, the analysts examined the sequence of events leading to the reactor trip and selected the event that occurred first from the list of 48 initial plant fault categories. Only one initial plant fault category was selected for each reactor trip. For example, consider the previous case where a total loss of feedwater flow causes a plant transient that results in a reactor trip and turbine trip, and then the loss of offsite power. The initial plant fault category appropriate for this reactor trip sequence would be Total Loss of Feedwater Flow (category P1), since it happened first. In this example, the total loss of feedwater may be the result of the failure or misoperation of components in the main feedwater system or associated with another system. However, if the root cause could not be matched to a category from the initial plant fault group, then the next event in the reactor trip sequence that could be matched was selected.

2.2.3 Special Interest Group

A third group, designated as special interest, includes additional events that are often of interest but are not associated with an initial plant fault or functional impact category, such as diesel starts and loads, and a manual reactor trip that occurs after the event initiator (i.e., initial plant fault). Station blackout and Anticipated Transient Without Scram (ATWS) events fall into this category because these events involve an initiating event and a failure of a mitigating system. All applicable special interest events are flagged regardless of their place in the sequence of events. This information was collected for future studies. One special interest category was analyzed in this report: the occurrence of a manual reactor trip after each initial plant fault was evaluated in Section 4, Engineering Analysis of Results.

2.3 Results

Initial plant fault and functional impact categories are always associated with their respective group heading. Classifying events at the category level maximizes the database programming flexibility. By altering the heading/category associations, the database developed in this study can be adapted for plants with individual plant examination (IPE) assumptions and definitions that may differ materially from the associations used in this report.

A summary count of the initial plant fault and functional impact categories for each heading is shown in Table 2-2. Detailed results for all categories are provided in Appendix D. The cumulative totals for each initial plant fault and functional impact category are shown in Table D-3 of Appendix D. Table D-4 provides a breakdown of the initial plant fault and functional impact counts by category and by plant type (i.e., BWR and PWR). The counts reported in these tables reflect the number of events from the 1987–1995 operating experience. Events from prior experience used to supplement certain rare event categories (i.e., reactor coolant pump seal LOCA, total loss of service water) are not included in these tables.

One must remember when reviewing these tables that each reactor trip event has only one initial plant fault. Therefore, the total of all initial plant fault counts is the same as the total number of reactor trips. However, a reactor trip may have one or more functional impact events, but, in most cases, a reactor trip sequence will have no functional impact event.

Table 2-2. Summary count of the events by initial plant fault (IPF) and functional impact (FI) heading.

IPF Total	FI Total	Heading	IPF Total	FI Total	Heading
17	33	Loss of Offsite Power (B)	1	2	Flood (J)
11	17	Loss of Safety-Related Bus (C)	9	9	High Energy Line Break (K)
26	36	Loss of Instrument or Control Air (D)	64	200	Total Loss of Condenser Heat Sink (L)
0	6	Loss of Safety-Related Cooling Water (E)	0	0	Interfacing System Loss-of-Coolant Accident (N)
3	3	Steam Generator Tube Rupture (F)	86	159	Total Loss of Feedwater Flow (P)
12	16	Loss-of-Coolant Accident/Leak (G)	1,725	—[a]	General Transient —Other initial plant fault[a] (Q)
31	39	Fire (H)	1,985	520	Totals

a Initial plant fault heading only

3. RISK-BASED ANALYSIS OF THE PLANT OPERATING DATA

3.1 Introduction

This section presents the results of the initiating event frequencies based on the functional impact categories. The frequencies are analyzed to uncover trends and patterns on plant performance at a plant-specific and industry-wide basis. The results and data from plant-specific probabilistic risk assessments (PRAs) and individual plant examinations (IPEs) are compared with the results calculated from the 1987–1995 experience. Results of frequency estimation for initial plant fault categories are presented in Appendix D.

3.2 Frequencies and Trends of Initiating Events

3.2.1 Frequencies of Initiating Events

Table 3-1 provides industry-wide summaries for event headings and categories. As explained in Appendix E, small data sets yield only simple generic estimates. If the data set is larger (includes more observed events), it may be possible to detect differences among plants, a time trend, or both. Therefore, Table 3-1 identifies the categories with between-plant variation and a time trend. When no time trend was modeled, the frequencies provided in the table referred to all the years of the study. When a time trend was modeled, the frequencies refer to the end point of the trend line (i.e., 1995, the last year of the study). In a few cases when differences between boiling water reactors (BWRs) and pressurized water reactors (PWRs) existed, separate estimates are presented for the two plant types.

In the case where a category had no or very few event occurrences, the single constant rate model was used to calculate the mean frequency. This model used a Jeffreys noninformative prior in a Bayes updated distribution. As explained in Appendix E, the mean of the distribution for this model is $(n + 0.5)/t$, where n is the observed number of events and t is the total time period of the operating experience in critical years. For example, the mean frequency for an event category with no observed events and applicable to both BWRs and PWRs would be 0.5/729 or 6.9E-4 events per critical year.

The results in this report represent the average industry frequencies of initiating events and plant-specific frequencies for those categories that displayed large between-plant variations (see Section 4.3.2). Some event categories, such as electrical bus failure, loss of instrument air, fire, flood and loss of service water, may not lead to reactor trips in all plants due to plant-specific design features and therefore may result in significant variations in the frequency of these events. The estimates in Table 3-1 reflect the expected frequency based on current operating experience of these types of events from the total population of plants. The estimates provided in this report for these types of events give an indication of the general expectation for how often these events occur in the regulatory population of plants and their relative frequency compared to other events, such as general transients, total loss of feedwater events and loss of offsite power events.

The investigation of possible trends is discussed in the Section 3.2.2. A discussion of the models used in the analysis is provided in Appendix E. The reason for choosing each model is summarized in Appendix F. Detailed results are given in Appendix G, including tables of plant-specific estimates and figures showing plant-specific estimates and modeled time trends. The estimation of frequencies of rare events, such as LOCAs, are discussed in Section 4, Engineering Analysis of Results.

Table Format and Content. The format for the entries in Table 3-1 is as follows. Each row of text refers to an event description of a heading or category. The next two columns in the row correspond to the heading/category code and the associated number of events. Columns 4 through 6 are the mean frequency, the

5th percentile and the 95th percentile of the frequency distribution. The values in columns 4 through 6 are in units of events per critical year. The last two columns identify if a time trend or between-plant variation, respectively, were found. As explained in Section 3.2.2, the value in column four represents the mean frequency based on the endpoint of the trend line (i.e., 1995, the last year of the study) for those categories with a decreasing trend (see column seven).

Units of frequency estimates. The unit of measure used in this to present results of initiating event frequencies is (events) *per critical year*. There are two exceptions where the results are reported in (events) *per calendar year*: anticipated transient without scram (ATWS) probabilities reported in Section 3.4, Comparison to the ATWS Rule, and the pipe break loss-of-coolant accident (LOCA) frequencies reported in Appendix J, LOCA Frequency Estimates. The results in these two cases were converted to (events) *per calendar year* for the ease of comparison to historical results. Pipe break LOCA frequencies reported in the main report and Executive Summary were converted to (events) per critical year, as discussed below.

Definition of critical year and calendar year. Table 3-1 presents the means in units of events per critical year, where one critical year equals 8,760 hours of reactor criticality. A critical year is not the same as a calendar year unless the reactor is critical throughout the entire calendar year. To estimate the expected number of events in a calendar year, multiply the value in Table 3-1 by the fraction of time when the reactor is critical. This fraction is called the criticality factor in this report. The criticality factors (by plant and year) are provided in Appendix H. The industry average criticality factor is about 75%.

Operating experience used to estimate frequencies. Frequencies of initiating event categories except for several rare event categories are based on U.S. operating experience from 1987 through 1995. Frequency estimates for pipe break LOCA-related events are based on total U.S and world-wide operating experience which included experience prior to 1987 and after 1995 (See Appendix J). Frequency estimates of reactor coolant pump seal LOCA, stuck open two or more safety/relief valves, and total loss of service water categories are based on total U.S. operating experience (1969 through 1997). The U.S. commercial operating experience used in this report are:

	Critical Years
U.S. 1987–1995	499-PWR; 230-BWR
U.S. 1969–1997	1019-PWR; 525-BWR

Except where noted, results (frequencies) in this report were reported in units of per critical year. Critical years are based on commercial start date prior to1984 and low-power-license date for 1984 and beyond.

Table 3-1. Frequency estimates of functional impact categories: mean, percentiles, and trends. (See text for detailed explanation.)

Event	Functional Impact Event Category	Number of Functional Impact Occurrences[a]	Mean Frequency (per critical year)[b,c,k]	Percentiles		Model Used	
				5th %ile	95th %ile	Trend	Plant Difference[j]
Loss-of-Coolant Accident (LOCA)	G						
Large Pipe Break LOCA: PWR	G7	0	5E-6[d]	1E-7	1E-5	Constant[e]	No
Large Pipe Break LOCA. BWR	G7	0	3E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA. PWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: BWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Small Pipe Break LOCA	G3	0	5E-4[d]	1E-4	1E-3	Constant[e]	No
Very Small LOCA/Leak	G1	4	6.2E-3	2.3E-3	1.2E-2	Constant[e]	No
Stuck Open: Pressurizer PORV	G4	0	1.0E-3	3.9E-6	3.9E-3	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: PWR	G2	2	5.0E-3	1.2E-3	1.1E-2	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: BWR	G2	10	4.6E-2	2.5E-2	7 1E-2	Constant[e]	No
Stuck Open: 2 or More Safety/Relief Valves	G5	0	3.2E-4[d]	1.3E-6	1.2E-3	Constant[e]	No
Reactor Coolant Pump Seal LOCA. PWR	G8	2[d]	2.5E-3[d]	5.6E-4	5.4E-3	Constant[e]	No
Steam Generator Tube Rupture: PWR	F1	3	7.0E-3	2.2E-3	1.4E-2	Constant[e]	No
Loss of Offsite Power	B1	33	4.6E-2	8.2E-3	1 1E-1	Constant[e]	No
Total Loss of Condenser Heat Sink (combined):[f] PWR	L	75[f]	1.2E-1[c,f]	2.3E-2[i]	3.2E-1[i]	Decrease[f]	Yes[j]
Total Loss of Condenser Heat Sink (combined):[f] BWR	L	122[f]	2.9E-1[c,f]	2.0E-1	3.9E-1	Decrease[f]	No
Inadvertent Closure of All MSIVs: PWR	L1	35	3.8E-2[c]	1.9E-2	6.5E-2	Decrease	No
Inadvertent Closure of All MSIVs: BWR	L1	74	1.7E-1[c]	6.0E-2[i]	3.6E-1[i]	Decrease	Yes[j]
Loss of Condenser Vacuum: PWR	L2	35	6.9E-2	2.9E-5	3.0E-1	Constant[e]	Yes[j]
Loss of Condenser Vacuum: BWR	L2	46	2.0E-1	4.3E-2	4.6E-1	Constant[e]	No
Turbine Bypass Unavailable	L3	10	4.1E-3[c]	6.1E-4	1.2E-2	Decrease	No
Total Loss of Feedwater Flow	P1	159	8.5E-2[c]	1.3E-2[i]	2.5E-1[i]	Decrease	Yes[j]
General Transients (combined):[f] PWR	Q	1184[f,g]	1.2E+0[c,f]	6.1E-1[i]	2.1E+0[i]	Decrease[f]	Yes[j]
General Transients (combined):[f] BWR	Q	541[f,g]	1.5E+0[c,f]	8.5E-1[i]	2.5E+0[i]	Decrease[f]	Yes[j]
High Energy Line Steam Breaks/Leaks (combined)[h]	K	9[h]	1.3E-2	7.0E-3	2.1E-2	Constant[e]	No
Steam Line Break/Leak Outside Containment	K1	7	1.0E-2	5.0E-3	1 7E-2	Constant[e]	No
Steam Line Break/Leak Inside Containment: PWR	K3	0	1.0E-3	3.9E-6	3.9E-3	Constant[e]	No
Feedwater Line Break/Leak	K2	2	3.4E-3	7.9E-4	7.6E-3	Constant[e]	No

Event	Functional Impact Event Category	Number of Functional Impact Occurrences[a]	Mean Frequency (per critical year)[b,c,k]	Percentiles		Model Used	
				5th %ile	95th %ile	Trend	Plant Difference[j]
Loss of Safety-Related Bus	C						
Loss of Vital Medium Voltage ac Bus	C1	13	1.9E-2	1.1E-2	2.8E-2	Constant[e]	No
Loss of Vital Low Voltage ac Bus	C2	3	4.8E-3	1.5E-3	9.7E-3	Constant[e]	No
Loss of Vital dc Bus	C3	1	2.1E-3	2.4E-4	5.4E-3	Constant[e]	No
Loss of Safety-Related Cooling Water	E						
Total Loss of Service Water	E1	1[d]	9.7E-4[d]	1.1E-4	2.5E-3	Constant[e]	No
Partial Loss of Service Water	E2	6	8.9E-3	4.0E-3	1.5E-2	Constant[e]	No
Loss of Instrument or Control Air: PWR	D1	15[c]	9.6E-3[c]	3.9E-3	1.9E-2	Decrease	No
Loss of Instrument or Control Air: BWR	D1	21[c]	2.9E-2[c]	1.3E-2	5.5E-2	Decrease	No
Fire	H1	39	3.2E-2[c]	1.7E-2	5.2E-2	Decrease	No
Flood	J1	2	3.4E-3	7.9E-4	7.6E-3	Constant[e]	No
		Total — PWR	1.4E+0[c]	6.9E-1[i]	2.4E+0[i]	Decrease[f]	Yes[j]
		Total — BWR	1.8E+0[c]	9.5E-1[i]	2.9E+0[i]	Decrease[f]	Yes[j]

a. Reactor trip events from 1987 through 1995, inclusive, except when noted for certain rare events.

b. Frequencies are presented in per critical year (8,760 critical hours per critical year).

c. For categories with a decreasing trend, the frequencies reported are based on the endpoint of the trend line (i.e., 1995, the last year of the study).

d. No failures were identified in the 1987–1995 operating experience. The Medium and Large Pipe Break LOCA estimates were based on review of current literature and fracture mechanic analyses and using world-wide experience. (Appendix J contains the results of the LOCA analysis.) Frequency estimates for Small Pipe Break LOCA, Reactor Coolant Pump Seal LOCA, Stuck Open: 2 or More Safety/Relief Valves, and Total Loss of Service Water categories were based on total U.S. operating experience (1969–1997).

e. Any evidence for a trend was weak, not statistically significant. The trend, if any, is too small to be seen in the data. Therefore, no trend is modeled.

f. Combined number of occurrences of all categories for each plant type (BWR, PWR) under this heading was used to calculate this frequency and trend.

g. Total number of initial plant-fault occurrences for this plant type.

h. The frequency was based on the combined number of occurrences in the categories under this heading.

i. The interval includes variability from plants with events early in life (for example, learning periods) and are wider than the plants' current performance. See Appendix G for results with the early-in-life events excluded.

j. Due to modeling assumptions with regard to independent random events, the between-plant variation was evaluated with the first four months from date of commercial operation (early-in-life events) excluded for the affected plants.

k. For categories modeled with no trend and no between-plant variation, the estimates were calculated using a Jeffreys noninformative prior (one-half of an event added to the total number of events) in a Bayes updated distribution.

3.2.2 Investigation of Possible Trends

Four models were used in the trends and pattern analyses of the event frequencies. The choice of the model depended on the complexity of the data set. Data sets composed of only a few event occurrences used a simpler approach, whereas large data sets required more complicated modeling. The assumption implicit in the four models is that the events occur following a Poisson process. The four models used in increasing order of complexity are: (1) single constant rate; (2) constant rate, differences among plants; (3) trend in calendar time, with no differences among plants; and (4) both trend in calendar time and differences among plants.

To interpret the time trend models, see the subsection of Appendix E entitled "Answering the Question, 'Is There a Trend?'" The statement "λ is modeled as constant" means that any trend was too slight to be clearly visible in the data. A small trend may in fact be present, and a larger data set might reveal that trend. Appendices E, F, and G describe the methods and the results of the trending analysis.

When no time trend is modeled, the frequencies given refer to all the years of the study. When a time trend is modeled, the frequencies are based on the endpoint of the trend line (i.e., 1995, the last year of the study). The last year was selected since it reflects the most recent industry experience during the time period of this study (i.e., 1995). As an example, from Table 3-1 consider category L1, Inadvertent Closure of All MSIVs, and category L2, Loss of Condenser Vacuum, for PWRs. The reported mean frequency for the Inadvertent Closure of All MSIVs category is about a factor of two lower than the other category, in spite of the fact that both categories have the same number of events during the same PWR operating time period. The explanation of this difference is that the frequency was decreasing for the Inadvertent Closure of All MSIVs category, but not for the Loss of Condenser Vacuum category.

Results. As seen in Table 3-1, no increasing trends were found for any heading and category. A decreasing trend was found in approximately two-thirds of the headings and categories that had sufficient data for trending analyses (i.e., ten or more events). Section 4.2 of the report provides additional information on the analyses of the categories with yearly trends. Section 3.5 provides a sensitivity analysis that evaluates the learning period effect of new plants on the initiating event frequencies.

3.3 Comparison to PRAs

3.3.1 Comparison to IPE/PRAs

Tables 3-2 and 3-3 provide a comparison between the initiating event frequencies based on operating experience (1987–1995 experience for non-LOCA categories) and values extracted from individual plant examinations (IPEs) using the IPE database (Su et al. 1997). The database contains information on 28 BWR IPEs and 51 PWR IPEs. The IPE estimates were obtained by extracting the complete set of initiating-events information contained in the IPE database and pooling the values appropriate for each category and heading used in the comparison. The individual IPE totals were then collected into an IPE-wide (with PWRs and BWRs grouped separately) data set, for which the statistics appearing in the tables were calculated. Since the IPE database only contains point estimates of initiating event frequencies, the frequencies recorded in Tables 3-2 and 3-3 represent the arithmetic average of point estimates in the event category/heading population. The lower and upper range values represent the minimum and maximum value of the point estimates from the IPE population. The IPE frequencies were converted to units of per critical year for comparison to the estimates calculated from the operating experience.

Table 3-2. Comparison between functional impact (FI) frequencies and individual plant examination (IPE) values for BWR plants.

Description	Mean FI Frequency (per critical year)[a]	Mean of the IPE Population Frequency (per critical year)[a,b]	IPE Range of Values: Lower	IPE Range of Values: Median	IPE Range of Values: Upper
LOCAs (G)					
Large Pipe Break LOCA (G7)	3E-5	5.5E-4	1.0E-5	4.1E-4	2.8E-3
Medium Pipe Break LOCA (G6)	4E-5	2.0E-3	8.4E-5	1.0E-3	4.1E-3
Small Pipe Break LOCA (G3)	5E-4[c]	1.0E-2	1.3E-3	1.1E-2	4.1E-2
Very Small LOCA/Leak (G1)	6.2E-3[c]	5.9E-2	2.3E-3	3.0E-2	3.2E-1
Stuck Open: 1 Safety/Relief Valve (G2)	4.6E-2	1.1E-1	8.5E-5	8.5E-2	4.1E-1
Loss of Offsite Power (B)	4.6E-2[c]	1.3E-1	3.0E-2	8.5E-2	8.4E-1
Transients					
High Energy Line Break Outside Containment (K1)	1.0E-2[c]	8.4E-3	1.3E-8	2.5E-4	6.2E-2
Total Loss of Condenser Heat Sink (L)	2.9E-1	4.3E-1	6.0E-2	3.0E-1	2.1E+0
Total Loss of Feedwater Flow (P)	8.5E-2[c]	5.7E-1	6.0E-2	6.0E-1	1.3E+0
General Transients (Q)	1.5E+0	6.0E+0	2.5E+0	5.7E+0	1.0E+1
Loss of Support Systems/Other					
Loss of Vital Medium or Low Voltage ac Bus (C1 + C2)	2.3E-2[c]	1.9E-2	3.5E-3	1.3E-2	7.2E-2
Loss of Vital dc Bus (C3)	2.1E-3[c]	1.9E-2	1.2E-3	1.1E-2	7.1E-2
Loss of Instrument or Control Air (D)	2.9E-2	4.8E-2	1.6E-4	4.4E-2	1.3E-1
Total Loss of Service Water (E1)	9.7E-4[c]	1.2E-2	2.4E-4	5.9E-3	5.2E-2
Flood (J)	3.4E-3[c]	9.6E-2	1.3E-7	5.5E-3	1.0E+0
Total	1.8E+0[d]	7.4E+0			

a One critical year equals 8,760 hours of reactor criticality

b The IPE frequencies were converted from units of calendar year to critical year for comparison to the estimates calculated from the operating experience One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study Therefore the rate per critical year equals the rate per calendar year divided by 0 75

c The estimate did not differentiate with respect to plant type (i e , PWR and BWR); therefore the value is same for either plant type

d. Total mean frequency includes additional categories not shown in the table (See Table 3-1).

Table 3-3. Comparison between functional impact (FI) frequencies and individual plant examination (IPE) values for PWR plants.

Description	Mean FI Frequency (per critical year)[a]	Mean of the IPE Population Frequency (per critical year)[a,b]	IPE Range of Values		
			Lower	Median	Upper
LOCAs (G)					
Large Pipe Break LOCA (G7)	5E-6	4.1E-4	1.3E-5	4.1E-4	9.4E-4
Medium Pipe Break LOCA (G6)	4E-5	1.0E-3	1.3E-4	9.4E-4	3.6E-3
Small Pipe Break LOCA (G3)	5E-4[c]	9.2E-3	5.0E-4	6.0E-3	3.9E-2
Very Small LOCA/Leak (G1)	6.2E-3[c]	1.1E-2	8.0E-7	8.0E-3	2.7E-2
Stuck Open: 1 Safety/Relief Valve (G2)	5.0E-3	7.8E-2	4.8E-6	5.7E-3	4.1E-1
Steam Generator Tube Rupture (F)	7.0E-3	2.0E-2	4.4E-3	1.3E-2	5.2E-2
Loss of Offsite Power (B)	4.6E-2[c]	1.0E-1	4.6E-3	8.0E-2	5.9E-1
Transients					
High Energy Line Break Outside Containment (K1)	1.0E-2[c]	7.1E-3	4.6E-5	4.1E-3	5.2E-2
Total Loss of Condenser Heat Sink (L)	1.2E-1	3.0E-1	5.9E-2	2.5E-1	1.0E+0
Total Loss of Feedwater Flow (P)	8.5E-2[c]	1.0E+0	1.6E-2	9.2E-1	3.7E+0
General Transients (Q)	1.2E+0	4.0E+0	2.0E-1	3.7E+0	9.9E+0
Loss of Support Systems/Other					
Loss of Vital Medium or Low Voltage ac Bus (C1 + C2)	2.3E-2[c]	1.5E-1	8.7E-5	4.3E-2	7.7E-1
Loss of Vital dc Bus (C3)	2.1E-3[c]	1.9E-2	4.5E-4	1.1E-2	1.5E-1
Loss of Instrument or Control Air (D)	9.6E-3	6.6E-2	7.3E-5	3.0E-2	4.1E-1
Total Loss of Service Water (E1)	9.7E-4[c]	5.3E-2	1.5E-7	6.7E-3	8.8E-1
Flood (J)	3.4E-3[c]	1.3E-2	7.1E-6	4.1E-3	6.9E-2
Total	1.4E+0[d]	5.8E+0			

a One critical year equals 8,760 hours of reactor criticality

b The IPE frequencies were converted from units of calendar year to critical year for comparison to the estimates calculated from the operating experience One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study Therefore the rate per critical year equals the rate per calendar year divided by 0 75

c The estimate did not differentiate with respect to plant type (i e , PWR and BWR); therefore, the value is same for either plant type

d. Total mean frequency includes additional categories not shown in the table (See Table 3-1).

Results. The values provided in Tables 3-2 and 3-3 are intended to illustrate the range of values used in the IPEs and how well, in an approximate manner, they compare with the values calculated in this report. Figures 3-1 and 3-2 are plots of the data contained in Tables 3-2 and 3-3. Based on the cumulative mean frequency of the initiating events, the IPE-wide frequency is higher (approximately a factor of four) than the frequency estimated from the operating experience. Several of the IPE initiating event frequencies (based on the arithmetic average of the point estimates from the IPE population) are higher by more than a factor of five than the corresponding mean frequency estimated from the operating experience. The categories for these are: Very Small LOCA/Leak (BWR), Stuck Open: 1 Safety/Relief Valve (PWR), Total Loss of Feedwater Flow (PWR and BWR), Flood (BWR), Loss of Instrument or Control Air (PWR), Loss of Vital Medium or Low Voltage ac Bus (PWR), Loss of Vital dc Bus (PWR and BWR), Total Loss of Service Water (PWR and BWR), and Flood (BWR). Furthermore, the frequencies of these categories, based on operating experience, fall within the lower and upper range of the IPE value, where about half are close to the median IPE value.

A comparison between pipe break LOCA frequencies developed in this study (see Appendix J) and those from IPEs (based on the arithmetic average of the point estimates from the population) show the frequencies for small, medium, and large pipe break LOCAs in IPEs are generally higher by a factor of 20 to 80. In PWRs the IPE values are higher by factors of 17, 25, and 80 for small, medium, and large pipe break LOCAs, respectively. In BWRs the IPE values are higher by factors of 20, 50, and 20 for small, medium, and large pipe break LOCAs, respectively.

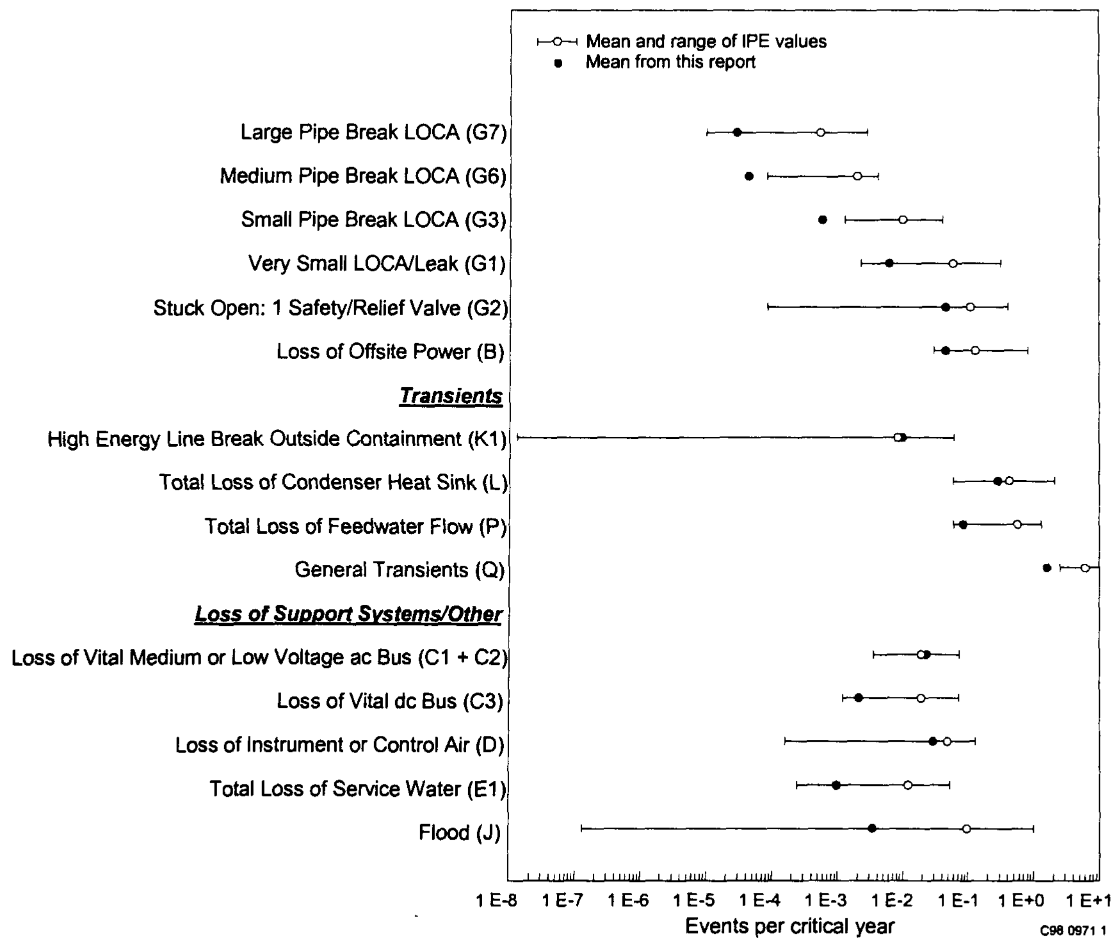

Figure 3-1. Comparison between functional impact frequencies and individual plant examination (IPE) values for BWR plants, from Table 3-2.

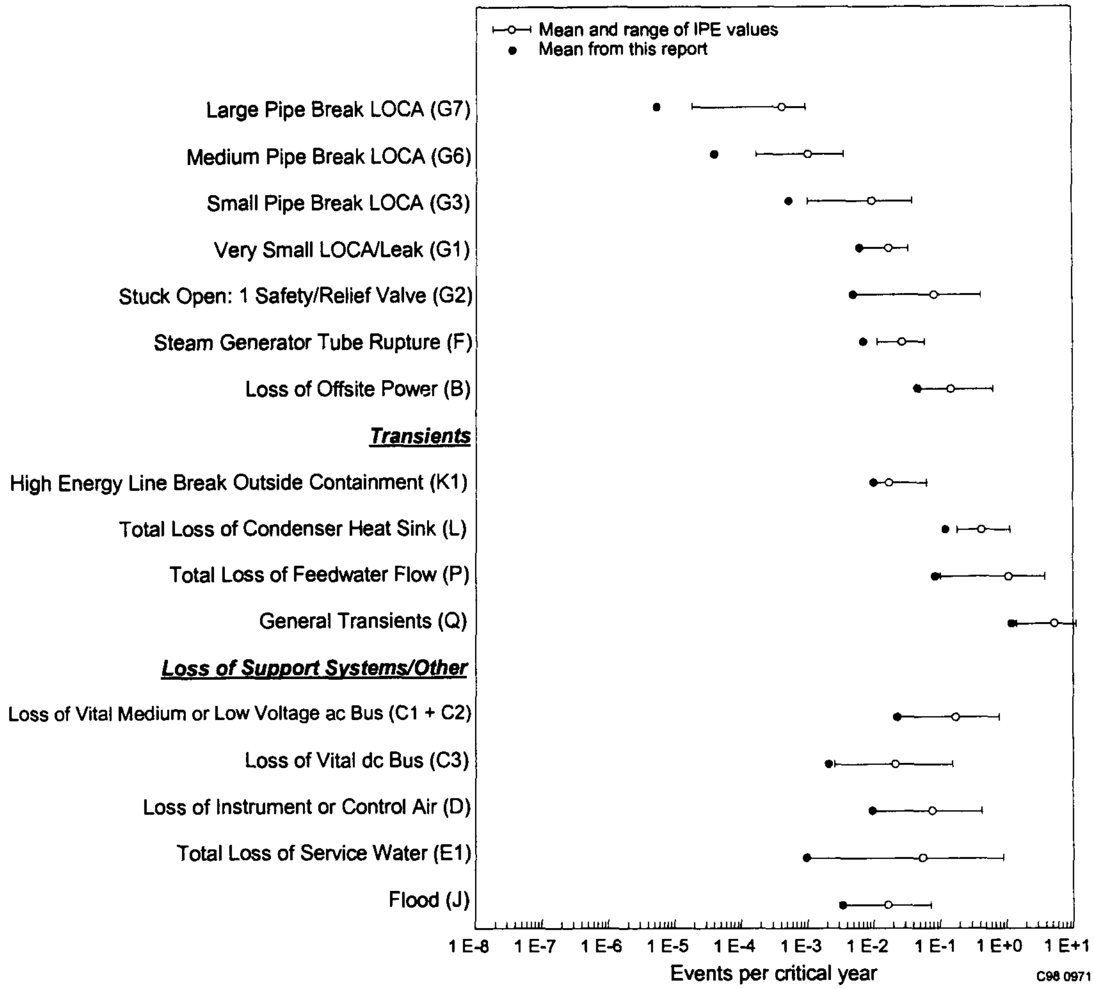

Figure 3-2. Comparison between functional impact frequencies and individual plant examination (IPE) values for PWR plants, from Table 3-3.

3.3.2 Comparison to NUREG-1150 and NUREG/CR-3862

This report follows several reports that have been produced independently over the last two decades by the Electric Power Research Institute (EPRI), the NRC, and the INEEL. In this report, both the data and the classification scheme are updated to reflect current PRA practices.

EPRI collected data for U.S. commercial power plant initiating events as a part of the study of the Anticipated Transient Without Scram (ATWS) topic. EPRI issued a report in 1978, with initiating event categories and associated frequencies (EPRI 1978) based on data submitted by the utilities. EPRI published a revision to this initial study in 1982 (EPRI 1982) and in 1985 the INEEL published NUREG/CR-3862, *Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments*, (Mackowiak et al. 1985). The latter report used Monthly Operating Reports and updated the EPRI data set to cover all plants from their commercial operation date through the end of 1983. The average operating time per plant from the operating experience used in NUREG/CR-3862 and the analysis of the1987–1995 data are approximately the same—9.5 calendar years per BWR plant and 8.4 calendar years per PWR plant.

Results. Tables 3-4 and 3-5 provide a comparison between the initiating event frequencies based on operating experience (1987–1995 experience for non-LOCA categories), and values from NUREG/CR-3862

Table 3-4. Comparison between functional impact (FI) frequencies and NUREG/CR-3862 and NUREG-1150 values for BWR plants.

Description	Mean FI Frequency (per critical year)[a]	NUREG/CR-3862 Mean Frequency (per critical year)[a,b]	NUREG-1150 Mean Frequency (per critical year)[a,b]
LOCAs (G)			
Large Pipe Break LOCA (G7)	3E-5	—	1 3E-4
Medium Pipe Break LOCA (G6)	4E-5	—	4 0E-4
Small Pipe Break LOCA (G3)	5E-4[c]	—	1 3E-3
Very Small LOCA/Leak (G1)	6.2E-3[c]	—	2.7E-2
Stuck Open: 1 Safety/Relief Valve (G2)	4.6E-2	—	1 9E-1
Loss of Offsite Power (B)	4.6E-2[c]	1.1E-1	1.1E-1
Transients			
Total Loss of Condenser Heat Sink (L)	2.9E-1	9.1E-1	9.1E-1
Total Loss of Feedwater Flow (P)	8.5E-2[c]	9.3E-2	9.3E-2
General Transients (Q)	1.5E+0	8.6E+0[f]	8.6E+0[f]
Loss of Safety-Related Bus (C)			
Loss of Vital Medium or Low Voltage ac Bus (C1 + C2)	2.3E-2[c]	—	5.0E-3
Loss of Vital dc Bus (C3)	2.1E-3[c]	—	6.0E-3
Fire (H1)	3.2E-2[c]	—	1.3E-2
Total of all events	1.8E+0[d]	9.7E+0[e]	9.9E+0[e]

a One critical year equals 8,760 hours of reactor criticality

b The units stated in the report are per reactor year (i e , numbers of years from start of commercial operation) For comparison purposes, the per reactor year was converted to critical year One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study Therefore the rate per critical year equals the rate per calendar year divided by 0 75

c The FI estimate did not differentiate with respect to plant type (i e , PWR and BWR); therefore, the value is the same for either plant type

d Total mean frequency includes additional categories not shown in the table (See Table 3-1).

e This total represents the sum of all frequencies presented in the referenced report

f. "Total of all events" entry minus the sum of all remaining entries (column) identified in this table.

Table 3-5. Comparison between functional impact (FI) frequencies and NUREG/CR-3862 and NUREG-1150 values for PWR plants.

Description	Mean FI Frequency (per critical year)[a]	NUREG/CR-3862 Mean Frequency (per critical year)[a,b]	NUREG-1150 Mean Frequency (per critical year)[a,b]
LOCAs (G)			
Large Pipe Break LOCA (G7)	5E-6	—	6 7E-4
Medium Pipe Break LOCA (G6)	4E-5	—	1 3E-3
Small Pipe Break LOCA (G3)	5E-4[c]	—	1.3E-3
Very Small LOCA/Leak (G1)	6.2E-3[c]	—	2.7E-2
Steam Generator Tube Rupture (F)	7.0E-3	—	1.0E-2
Loss of Offsite Power (B)	4.6E-2[c]	1.9E-1	1.9E-1
Transients			
Total Loss of Condenser Heat Sink (L)	1.2E-1	2.4E-1	2 4E-1
Total Loss of Feedwater Flow (P)	8.5E-2[c]	2.2E-1	2.2E-1
General Transients (Other) (Q)	1.2E+0	1.0E+1[f]	1.0E+1[f]
Loss of Safety-Related Bus (C)			
Loss of Vital Medium or Low Voltage ac Bus (C1 + C2)	2.3E-2[c]	—	5.0E-3
Loss of Vital dc Bus (C3)	2.1E-3[c]	—	6.0E-3
Fire (H1)	3.2E-2[c]	2.3E-2	1.3E-1
Total of all events	1.4E+0[d]	1.1E+1[e]	1.1E+1[e]

a One critical year equals 8,760 hours of reactor criticality

b The units stated in the report are per reactor year (i e , numbers of years from start of commercial operation) For comparison purposes, the per reactor year was converted to critical year One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study Therefore the rate per critical year equals the rate per calendar year divided by 0 75

c The FI estimate did not differentiate with respect to plant type (i e , PWR and BWR); therefore, the value is the same for either plant type

d Total mean frequency includes additional categories not shown in the table (See Table 3-1)

e This total represents the sum of all frequencies presented in the referenced report

f. "Total of all events" entry minus the remaining entries (column) identified in this table.

and NUREG-1150, *Severe Accident Risks: An Assessment For Five U.S. Nuclear Power Plants* (USNRC 1990). The frequencies from these two reports were matched to corresponding categories and headings used in the comparison. The frequencies were converted to units of per critical year for comparison to the estimates calculated from the operating experience.

The cumulative mean frequency of the initiating events for BWRs and PWRs has decreased by approximately a factor of 5 and 8 respectively, since the NUREG/CR-3862 report was published in 1985. This reduction is similar for the initiating event frequencies in NUREG-1150, since NUREG/CR-3862 was the source used in the study. The reduction of events in the General Transient categories is the major cause of the decrease in the total initial event frequencies for BWRs and PWRs. Several of the NUREG-1150 and

NUREG/CR-3862 frequencies are higher by a factor of three or more than the corresponding mean frequency estimated from operating experience. These categories are: Very Small LOCA/Leak (BWR), Stuck Open: 1 Safety/Relief Valve (BWR), Loss of Offsite Power (PWR), Total Loss of Condenser Heat Sink (BWR), Loss of Vital Medium or Low Voltage ac Bus (BWR), Loss of Vital dc Bus (PWR and BWR), Fire (PWR), and General Transients (PWR and BWR).

A comparison between pipe break LOCA frequencies developed in this study (see Appendix J) and those from NUREG-1150 show frequencies for small, medium, and large pipe break LOCAs in NUREG-1150 are generally higher by a factor of 2 to 140. For PWRs, the NUREG-1150 values are higher by factors of 2, 25, and 140 for small, medium, and large pipe break LOCAs, respectively. For BWRs, the NUREG-1150 values are higher by factors of 2, 10, and 3 for small, medium, and large pipe break LOCAs, respectively.

Appendix B provides a cross-reference of event categories to the NUREG/CR-3862 and EPRI NP-2230 studies to the categories defined for this study.

3.3.3 Loss of Offsite Power (LOSP)—Comparison to NUREG-1032

The NRC published NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants* (Baranowsky 1988), to report an evaluation of the risk from actual loss of offsite power (LOSP) events occurring at U. S. nuclear power plants up through 1985. A recent report, NUREG/CR-5496, *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1996* (Atwood et al. 1998), documents a similar study whose primary objective was to update the LOSP model parameters, frequency and recovery times, using plant events data from 1980–1996. The LOSP events defined in NUREG/CR-5496 were further grouped into three categories: plant-centered, grid-related, and caused by severe weather. NUREG/CR-5496 considered LOSP events when the plant was operating and shutdown, whereas this study evaluated only LOSP events that were associated with a reactor trip. NUREG/CR-5496 provides a more thorough evaluation as well as plant-specific data on LOSP frequencies that support PRA evaluations.

The combined LOSP frequency based on the 1987–1995 experience is lower by about a factor of two when compared to the results in NUREG-1032. The LOSP frequency based on 1987–1995 experience is about the same when compared with the combined frequency in NUREG/CR-5496, which includes plant-centered LOSP events during power operations, grid-related and severe weather-related LOSP events.

3.4 Comparison to the ATWS Rule

In 1980, after the evaluation of information gathered over the preceding ten years, the NRC concluded that the frequency of a severe Anticipated Accident Without Scram (ATWS) event may be unacceptably high. Following this evaluation, SECY-83-293 (Rulemaking Issue, Affirmation) (USNRC 1983) was issued to seek approval for publication of a final rule on the ATWS issue. The frequency of ATWS used in SECY 83-293 was based on EPRI data (EPRI 1978, 1982). Table 3-6 provides a summary comparison of the SECY-83-293 initiating event frequency estimates to the estimates based on 1987–1995 experience. The frequency estimates in Table 3-6 are based on the limiting set of transients for ATWS as defined in EPRI NP-2230, Table 4-2 (EPRI 1982). Further, the estimates based on 1987–1995 experience were converted from per critical year to per calendar year using the 75% criticality factor average calculated in this report. The frequency of ATWS transient initiators calculated from the 1987–1995 operating experience has decreased since the ATWS Rulemaking analysis was completed in 1983. This decrease indicates that the frequency of challenges that could result in a severe ATWS event has declined.

SECY-83-293, Enclosure D, "Recommendations of the ATWS Task Force," (USNRC 1983) states calculations of ATWS probabilities were performed by using simplified event trees for each generic reactor

Table 3-6. Comparison between initiating event frequencies (mean) calculated for this study and frequencies used in SECY-83-293 for estimating the probability of ATWS.

Plant Type	SECY-83-293 Initiating Event Frequency[a] (per calendar year)	Initiating Event Frequency[a] Using 1987–1995 Experience (per calendar year)
BWR	4.3E+0	1.2E+0
PWR	4.0E+0	1.4E+0

a. Frequency estimates are based on a limiting set of transients for ATWS as defined in EPRI 1982.

design. The event trees were evaluated for each prescribed ATWS preventive or mitigative option and for combinations of options recommended by the ATWS Task Force. Table 3-7 provides the probability of ATWS for the option required by 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," for each NSSS vendor. These ATWS probabilities were based on the initiating event frequency of ATWS initiators, and the failure probability of the reactor protection system (RPS) and ATWS mitigation systems (e.g., auxiliary feedwater system and high pressure injection systems). [Prior to the Salem RPS failure in 1983, NUREG-0460 (USNRC 1980) was issued and reported an ATWS probability as 2E-4 per year. This calculation was based on a RPS failure probability of 3E-5/demand and the initiating event frequency of six per year.]

Table 3-8 provides an updated ATWS probability for the various reactor vendors based on the results of this study. The same failure probability of the RPS and ATWS mitigation systems from Enclosure D to SECY-83-293 were used to update the ATWS probabilities in Table 3-8. The SECY-83-293 ATWS frequencies would be reduced approximately by a factor of three for the PWR vendors while the BWR vendor would be reduced by about a factor of four when updated with initiating event frequencies based on 1987–1995 experience. (SECY-83-293 used a generic RPS failure probability of 1.2E-5 per demand when calculating ATWS probabilities for the ATWS risk reduction options required by 10 CFR 50.62. At the time of this writing, RPS system reliability studies sponsored by AEOD are underway at INEEL that will provide revised failure probabilities based on operational data for each reactor vendor.)

Table 3-7. Comparison of the initiating event transient frequencies and ATWS probabilities between reactor vendors used in SECY-83-293.

Vendor	Initiating Event Frequency (per calendar year)	ATWS Probability (per calendar year)
Westinghouse	4	1.9E-6
Combustion Engineering/Babcock & Wilcox	4	2.2E-5
General Electric	4.3	1.2E-5

Table 3-8. Revised SECY-83-293 ATWS probabilities of reactor vendors using initiating event transient frequencies based on 1987–1995 experience.

Vendor	Initiating Event Frequency (per calendar year)	ATWS Probability (per calendar year)
Westinghouse	1.4	6.7E-7
Combustion Engineering/Babcock & Wilcox	1.4	7.7E-6
General Electric	1.2	3.3E-6

3.5 Effect of Learning Period at New Plants

A review of the data showed that some new plants experience a high frequency of initiating events, which drops sharply after the plant has been operating for a short time. The most dramatic such case was Vogtle 1. The cumulative count of all initiating events for Vogtle 1 is shown in Figure 3-3. Times are measured in days from the commercial start date, with negative times corresponding to events after the low power license date and before the commercial start date. The slope of the cumulative plot corresponds to event frequency (events per time).

This plot and similar plots for other plants are examined in Appendix E. Based on this examination, it was decided to count the early-in-life period, or learning period, as extending four months after the commercial start date. The vertical dashed line in Figure 3-1 marks this cutoff. If this early-in-life period is excluded, the data set is restricted to events and critical hours occurring after this date.

Twenty of the 112 plants considered in this study had their learning periods at least partly in 1987–1995, with the majority in 1987 and 1988. This suggests two effects of including or excluding the learning period in the analysis. (1) Inclusion of the learning period may amplify, or cause entirely, a decreasing trend in the event rate. (2) Inclusion of the learning period compares new, inexperienced plants with more mature plants. Therefore, inclusion of the learning period may amplify, or cause entirely, perceived differences between the plants. Of course, if only a few events occur in the learning period, the apparent between-plant variance can increase or decrease, depending on which random plants the events occur at. Furthermore, if between-plant variance is not modeled, the extra events will reduce the calculated relative uncertainty.

Results. A sensitivity analysis was performed to evaluate the effects of the learning period data on the endpoint of trend lines for those categories with a trend. All trends in this report are assumed to be exponentially decreasing, but a contributor to some of the trends that used all data may be influenced as plants completed their learning periods. As stated in Section 3.2.2, when a time trend is modeled, the frequencies presented in the report are based on the endpoint of the trend line (i.e., 1995, the last year of the study).

Table 3-9 gives a summary of results of selected functional impact categories based on using all data and using data only after the learning period (after the fourth month of commercial operation). Figure 3-4 provides a plot of both sets of frequencies for each functional impact category. As seen from the comparison of both sets of results, the frequency of categories using all data compares fairly well with the frequencies using only data after the learning period. The differences between the results using the two data sets range by a factor of 0.9 to 1.2, well within the uncertainly interval estimated in both analyses.

The summary of results based on all data are provided in Tables 3-1 and D-12 for functional impact and initial plant fault categories, respectively. The summary of results and plant-specific results based on only the data after the learning period are provided in Section 4 and in Appendix G.

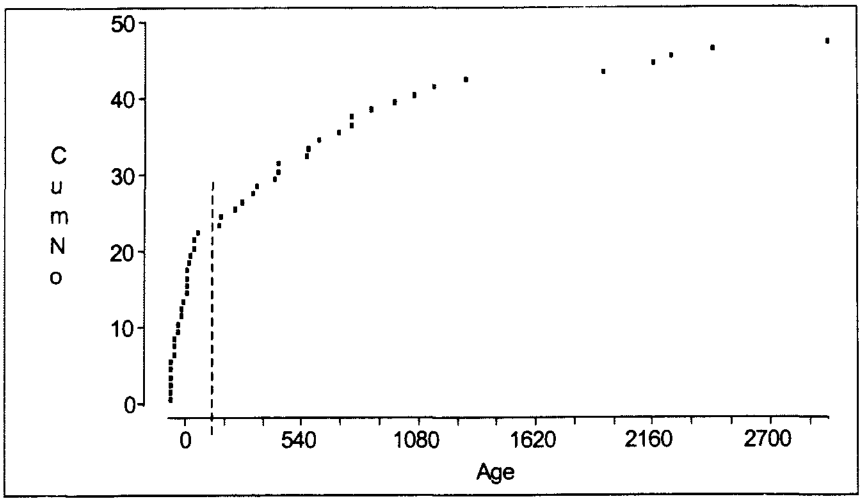

Figure 3-3. Vogtle 1, cumulative number of initiating events, by age (days) from commercial start date. All events after the low power license date are shown. The slope changes sharply near the dashed line.

Table 3-9. Effect of using all data, compared to results based on using only the data after the learning period.

Category	Frequency Using All Data (mean $_a$)	Frequency Using Data After Learning Period (mean $_b$)	Factor Increase In Mean (mean $_b \div$ mean $_a$)	Events In Learning Period and % of Total Experience
Loss of Offsite Power (B1)	4.6E-2	4.2E-2	0 91	3 (10%)
Loss of Instrument or Control Air: PWR (D1)	9.6E-3	9.8E-3	1 02	4 (36%)
Loss of Instrument or Control Air: BWR (D1)	2 9E-2	3.6E-2	1.24	2 (11%)
Fire (H1)	3.2E-2	3.2E-2	1.00	1 (3%)
High Energy Line Break (K)	1.3E-2	1.2E-2	0.92	1 (13%)
Total Loss of Condenser Heat Sink: PWR (L)	1.2E-1	1 4E-1	1.17	4 (6%)
Total Loss of Condenser Heat Sink: BWR (L)	2.9E-1	3.1E-1	1.07	9 (8%)
Total Loss of Feedwater Flow (P1)	8 5E-2	1.0E-1	1 18	27 (20%)
General Transient: PWR (Q)	1.2E+0	1 3E+0	1.08	114 (11%)
General Transient: BWR (Q)	1.5E+0	1.6E+0	1.07	34 (7%)

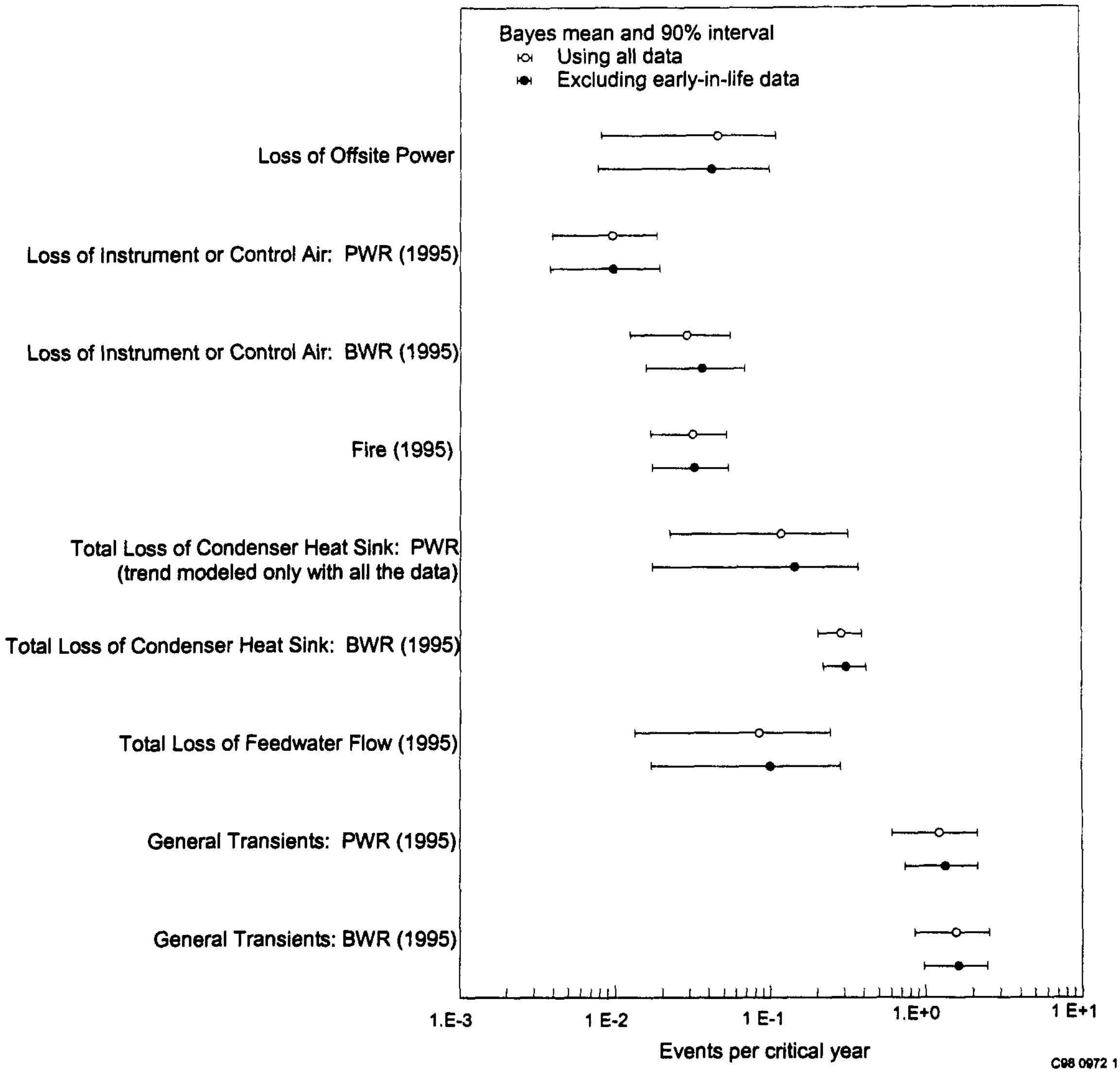

Figure 3-4. Comparison between functional impact frequencies using all the data or excluding the early period through the first four months after the commercial start date. The frequency refers to 1995 for those headings with a trend. The values are in Tables 3-1 and G-1.

4. ENGINEERING ANALYSIS OF RESULTS

4.1 Introduction

This section presents engineering insights concerning initiating event frequencies of functional impact categories. Frequencies for each category are analyzed to uncover trends and patterns in plant performance at a plant-type (PWR or BWR), plant-specific, and industry-wide basis. Best estimates for infrequent and rare events are provided. The leading contributors for risk-significant categories are summarized. In addition, conditional occurrences (percent of functional impacts occurring after the initial event) of the more risk-significant categories are presented. As discussed in Section 2.2, an initial plant fault is not analogous to the root cause of the reactor trip sequence. The initial plant fault is the very first event from the list of event categories (see Table 2-1) that causes or leads to a reactor trip. In some cases, the initial plant fault may be associated with the root cause, while in other cases it may not be the root cause event because the root cause was not associated with any initial plant fault category. For the latter case, the next chronological event in the reactor trip sequence is selected as the initial plant fault. This section provides a relational evaluation of initial plant fault events (i.e., reactor trip initiators) and the occurrence of risk-significant events (i.e., subsequent functional impact events) that occur after the initial plant fault event. Root cause and specific component analysis of reactor trips are not included in the scope of this study, but are a subject of further analyses in NRC/AEOD reactor trip review studies under development.

The following is a summary of the major findings:

- Over the nine-year span considered by this report, either a decreasing or constant time trend was observed for all categories of events. Overall, the frequency of reactor trips from all causes decreased over the period by about a factor of two to three. For BWRs, the 1987 reactor trip frequency was 4.4 events per critical year compared to 4.8 for PWRs. For 1995, the reactor trip frequency for BWRs and PWRs decreased to 1.8 and 1.4 events per critical year, respectively. A decreasing trend was identified in approximately two-thirds of the more risk-significant categories and headings that had sufficient data for trending analysis (i.e., ten or more events). The trends for Total Loss of Feedwater Flow, Total Loss of Condenser Heat Sink for BWRs, Inadvertent Closure of All MSIVs, and Loss of Instrument or Control Air were decreasing faster than the time trends for General Transient events.

- General Transients contributed 77% of all reactor trips. For the General Transient group, Turbine Trip was the major contributor (in terms of frequency) in both plant types (PWR and BWR). Of the more risk-significant categories (the remaining 23%), the more frequent events were Total Loss of Condenser Heat Sink and Total Loss of Feedwater Flow.

- The leading contributors to the Total Loss of Condenser Heat Sink (LOHS) in BWRs were transients resulting in an Inadvertent Closure of All MSIVs (60% of all LOHS events in BWRs) and Loss of Condenser Vacuum (37%). In PWRs, the contribution from each of these transients was about equal. The contribution of the Turbine Bypass Unavailable category to the Total Loss of Condenser Heat Sink frequency was almost negligible for both BWRs and PWRs.

- The major contributors to Total Loss of Feedwater Flow were directly related to problems within the feedwater system (54% of all Total Loss of Feedwater Flow events). Condensate system-related problems accounted for approximately 20% and support system (instrument air, electrical power, and cooling water) related problems resulted in 12% of the Total Loss of Feedwater Flow events.

- Plant-type variations between BWRs and PWRs were identified for three headings. The headings are Total Loss of Condenser Heat Sink, Loss of Instrument or Control Air, and General Transient. For the first two headings, the BWR frequency was slightly higher than the PWR frequency. The frequencies for the General Transient heading were about the same for both plant types.

- Between-plant variation was identified for five categories/headings. They are: Total Loss of Condenser Heat Sink (for PWRs), Loss of Condenser Vacuum (for PWRs), Inadvertent Closure of All MSIVs (for BWRs), Total Loss of Feedwater Flow, and General Transient (for PWRs and also for BWRs). Within these categories/headings, plant means were identified that are higher by a statistically significant amount than the industry average (the uncertainty interval for the plant mean is entirely to the right of the industry mean).

- The frequencies of loss-of-coolant (LOCA) events were evaluated. A summary of the these events are as follows:

  - ***Small pipe break LOCA***. No small pipe break LOCA events were found in the operating experience since WASH-1400, the Reactor Safety Study. For the small pipe break LOCA frequency, the estimate from WASH-1400 was updated using total U.S. reactor experience (i.e., no events in 1,019 PWR critical years and no events in 525 BWR critical years.) The updated frequencies for small pipe break LOCA for PWRs and BWRs is 5E-4 per critical year.

  - ***Medium and large pipe break LOCA.*** For medium and large break LOCAs, where no events have occurred world-wide, frequencies were estimated by calculating the frequency of leaks or through-wall cracks that have occurred which challenge the piping integrity. Further, conservative estimates were used for the conditional probability of a pipe break given a leak. The frequencies for medium pipe break LOCA for PWRs and BWRs is 4E-5 while the large pipe break LOCA frequencies are 5E-6 and 3E-5 per critical year for PWRs and BWRs, respectively.

  - ***Steam generator tube rupture***. This study identified three steam generator tube rupture (SGTR) events. The SGTR frequency estimate based on the three SGTR events is 7.0E-3 per critical year. Based on the current PWR population, this frequency correlates to about one event every two calendar years. The last SGTR identified in the 1987–1995 experience occurred at Palo Verde 2 in 1993.

  - ***Reactor coolant pump seal LOCA: PWR.*** Since 1981, no reactor coolant pump seal failures with a leak rate greater than technical specifications limit for identified leakage (usually 10 gpm) have been found in the review of the literature. Two catastrophic seal failures in PWRs were found in the total U.S. operating experience prior to 1981. No events were identified for BWRs. This study identified two reactor coolant pump seal leaks less than 6 gpm associated with a reactor trip. The reactor coolant pump seal LOCA frequency of 2.5E-3 per critical year was calculated in this study, based on 2 catastrophic seal failures with leak rates greater than 300 gpm in the total U.S. operating experience (1969–1997).

  - ***Interfacing systems LOCA.*** No interfacing system LOCA (ISLOCA) events were found in U.S. operating experience. ISLOCA frequencies of 2E-6 per calendar year for PWRs and less than 1E-8 per calendar year for BWRs were obtained from the ISLOCA Research Program (Galyean et al. 1993).

- No total loss of safety-related cooling water system events associated with a reactor trip have been identified during the time frame of this study. Only one total loss of safety-related service water system associated with a reactor trip was identified in the total U.S. operating experience (1969–1997). Six partial losses were identified in the 1987–1995 experience (associated with reactor trip events); however, none of these losses initiated the reactor trip sequence. The low frequency of loss of safety-related cooling water system indicates the normal plant line-ups during power operation provide a level of redundancy to these systems such that events having sufficient impact on plant operations to contribute to an initiating event are rare. A total loss of service water frequency of 9.7E-4 per critical year was calculated in this study, based on the one total failure in the total U.S. operating experience (1969–1997).

- Manual reactor trips occurred in 20% of all reactor trip events. Approximately one-fourth of all the manual reactor trip events were the result of a manual reactor trip as the initial plant fault. The remaining three-fourths of the manual reactor trips occurred after the initial plant fault.

- An evaluation of the more risk-significant events that occurred after the initiator in each reactor trip sequence reveal that:

  - One half of the more risk-significant events (under headings B through P) were transient induced, meaning they occurred after the reactor trip initiator (i.e., initial plant fault).

  - For Loss of Offsite Power (heading B) and Total Loss of Feedwater Flow (heading P), about half of the events occurred after the initial plant fault. Typically, the Loss of Offsite Power events occurred immediately after the turbine trip/reactor trip.

  - For Loss of Instrument or Control Air (heading D), only one-fourth of the events occurred after the initial plant fault.

  - For Total Loss of Condenser Heat Sink (heading L), about two-thirds of the events occurred after the initial plant fault.

  - Only 3 out of 103 Manual Reactor Trip (category QR6) events that occurred as the initial plant fault resulted in an additional event (functional impact) from a more risk-significant category (under heading B through P) after the manual reactor trip. This indicates that practically all manual reactor trips were associated with faults that were general transient in nature.

- Twelve cases were identified in the 1987–1995 experience where two reactors at a common site tripped simultaneously due to a related cause. These occurrences equate to an expectation across the industry of about one dual-unit trip per year. All but one dual-unit reactor trip were related to an electrical disturbance or loss of offsite power. The other dual-unit trip event was caused by manual reactor trips of both reactors due to the loss of the common station air system.

4.2 Industry-Wide Trends

The event classes with statistically significant trends for functional impact and initial plant fault categories and headings are shown in Table 4-1. The order of rows is roughly by the decreasing trend parameter *b*, although this order was modified slightly to keep most of the (Total Loss of Condenser Heat Sink) categories together. Further, trends were analyzed for those categories with a sufficient number of events.

Time trends were modeled by the formula $\lambda = \exp(a + by)$, where λ is the occurrence rate, y is the calendar year, and a and b are parameters. If b is zero, there is no trend. If b is negative, the trend is decreasing, and a plot of λ against y is an exponentially decreasing curve. Like λ, the parameters a and b are unknown parameters that apply to a hypothetical infinite population of plants. They are estimated from the limited observed data. A time trend was modeled whenever the evidence for a trend was statistically significant. In all of these cases, b was negative and the trend was decreasing. Usually, a case with $|b| < 0.1$ did not have a statistically significant trend. For the following discussion, the slope is considered *very gradual* if $|b| < 0.1$, *gradual* if $0.1 \le |b| < 0.2$, *moderate* if $0.2 \le |b| < 0.3$, and *steep* if $0.3 \le |b| < 0.4$. Over the nine-year span considered by this report, these ranges of $|b|$ correspond approximately to reductions of λ by factors of less than 2.2, from 2.2 to 5, from 5 to 11, and from 11 to 25. Appendix E provides a more detailed discussion on the trending method used in this report.

Table 4-1. Event categories and headings with modeled trends in frequencies.

Event	Steepness of Trend	Details
Loss of Instrument or Control Air (initial plant fault D1)	Moderate	$b = -0.26$, modeled as the same for BWRs and for PWRs
Loss of Instrument or Control Air (functional impact D1)	Moderate	$b = -0.21$, modeled as the same for BWRs and for PWRs
Total Loss of Feedwater Flow (functional impact P and initial plant fault P)	Gradual	$b = -0.18$ for functional impact and $b = -0.16$ for initial plant fault
Inadvertent Closure of All MSIVs (functional impact L1 and initial plant fault L1 for BWRs)	Gradual	$b = -0.15$ for functional impact and $b = -0.20$ for initial plant fault
Total Loss of Condenser Heat Sink (functional impact L and initial plant fault L for BWRs)	Gradual	$b = -0.12$ for functional impact and $b = -0.11$ for initial plant fault
Inadvertent Closure of All MSIVs (functional impact L1 for PWRs)	Gradual	$b = -0.14$
Fire (functional impact H1 and initial plant fault H1)	Gradual	$b = -0.13$ for functional impact and $b = -0.15$ for initial plant fault
General Transient (initial plant fault Q for PWRs)	Gradual	$b = -0.13$. These are 89% of all PWR initiating events
General Transient (initial plant fault Q for BWRs)	Very Gradual	$b = -0.09$. These are 81% of all BWR initiating events.

Results. Over the nine-year span considered by this report, either a decreasing or constant time trend was observed for all categories of events. Overall, the frequency of reactor trips from all causes had decreasedover the period by about a factor of two to three. For BWRs, the 1987 reactor trip frequency was 4.4 events per critical year compared to 4.8 for PWRs. For 1995, the reactor trip frequency for BWRs and PWRs decreased to 1.8 and 1.4 events per critical year, respectively. A decreasing trend was identified in approximately two-thirds of the more risk-significant categories and headings that had sufficient data for trending analysis (i.e., ten or more events). The trends for Total Loss of Feedwater Flow, Total Loss of Condenser Heat Sink for BWRs, Inadvertent Closure of All MSIVs, and Loss of Instrument or Control Air were decreasing faster than the time trends for General Transient events.

Because most initiating events are in the categories under the General Transient heading, it is not surprising that the trend for all initiating events is similar to the trend for the General Transient heading. In general, the trends seen for the more risk-significant categories are at least as steep as the trends for the General Transient heading. This generalization holds except for the more risk-significant categories with either a constant trend or so few events that no statistically significant trend could be detected.

Figures 4-1 through 4-7 provide a plot of the frequency (events per critical year) of the more risk-significant categories and headings, and the General Transient headings. Plots of all trends, including the initial plant fault categories, are provided in Appendix G.

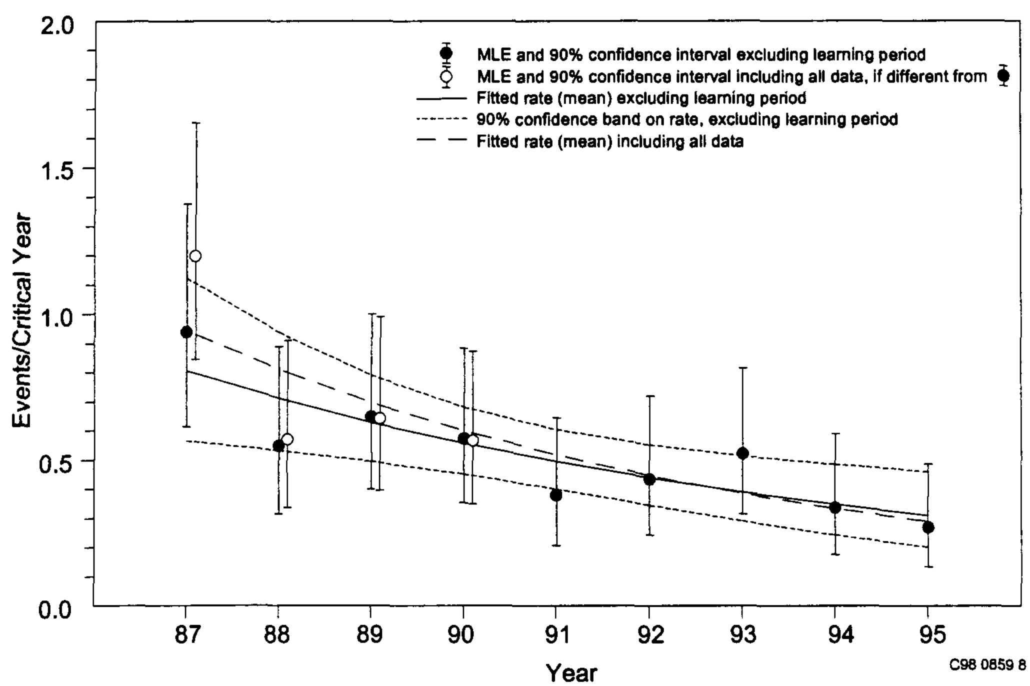

Figure 4-1. Time-dependent frequency for functional impact heading L, Total Loss of Condenser Heat Sink, for BWRs (events per critical year). The points and vertical lines are based on data from individual years while the dotted lines are a 90% confidence band on the frequency excluding the learning period at new plants. The p-value for each trend is 0.001. Between-plant differences were not seen.

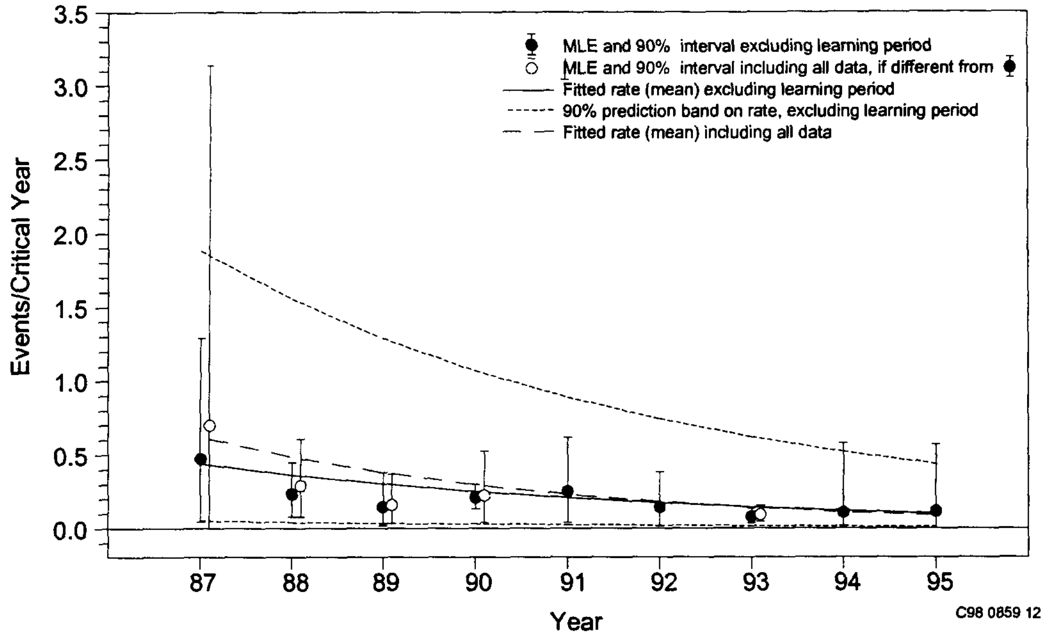

Figure 4-2. Time-dependent frequency for functional impact heading P, Total Loss of Feedwater Flow, for PWRs and BWRs (events per critical year). The points and vertical lines are based on data from individual years and reflect between-plant variation where observed. The dotted lines are a 90% prediction band on the frequency at a random plant, excluding the learning period at new plants. The p-value for each trend is 0.0001.

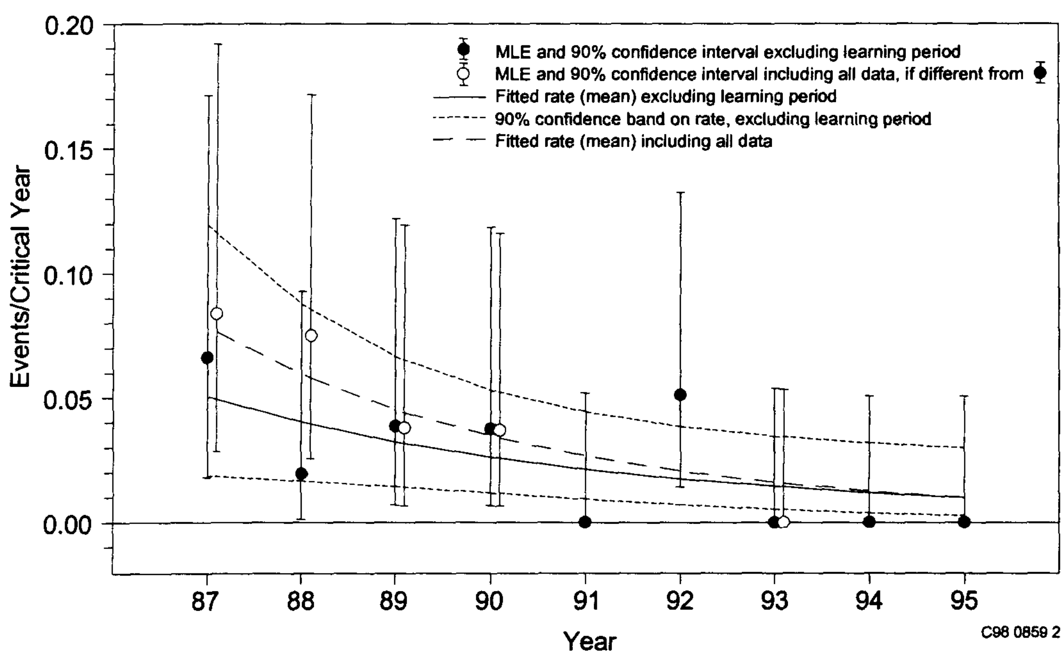

Figure 4-3. Time-dependent frequency for functional impact heading D, Loss of Instrument or Control Air System, for PWRs (events per critical year). The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the frequency excluding the learning period at new plants. The p-values for the trends with and without the learning period data are 0.0002 and 0.005, respectively. Between-plant differences were not seen.

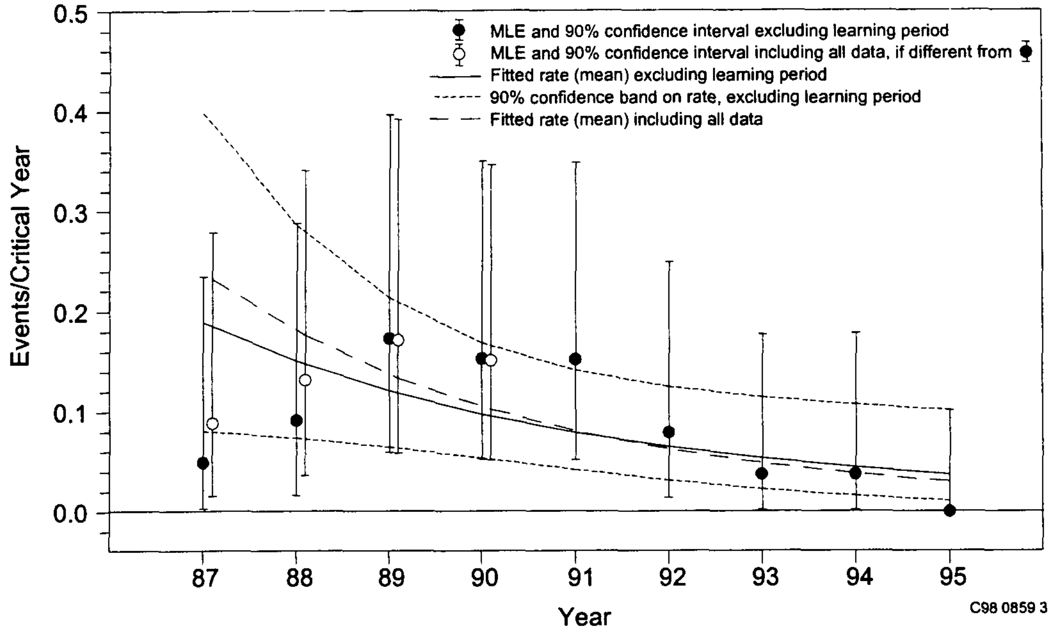

Figure 4-4. Time-dependent frequency for functional impact heading D, Loss of Instrument or Control Air System, for BWRs (events per critical year). The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the frequency excluding the learning period at new plants. The p-values for the trends with and without the learning period data are 0.0002 and 0.005, respectively. Between-plant differences were not seen.

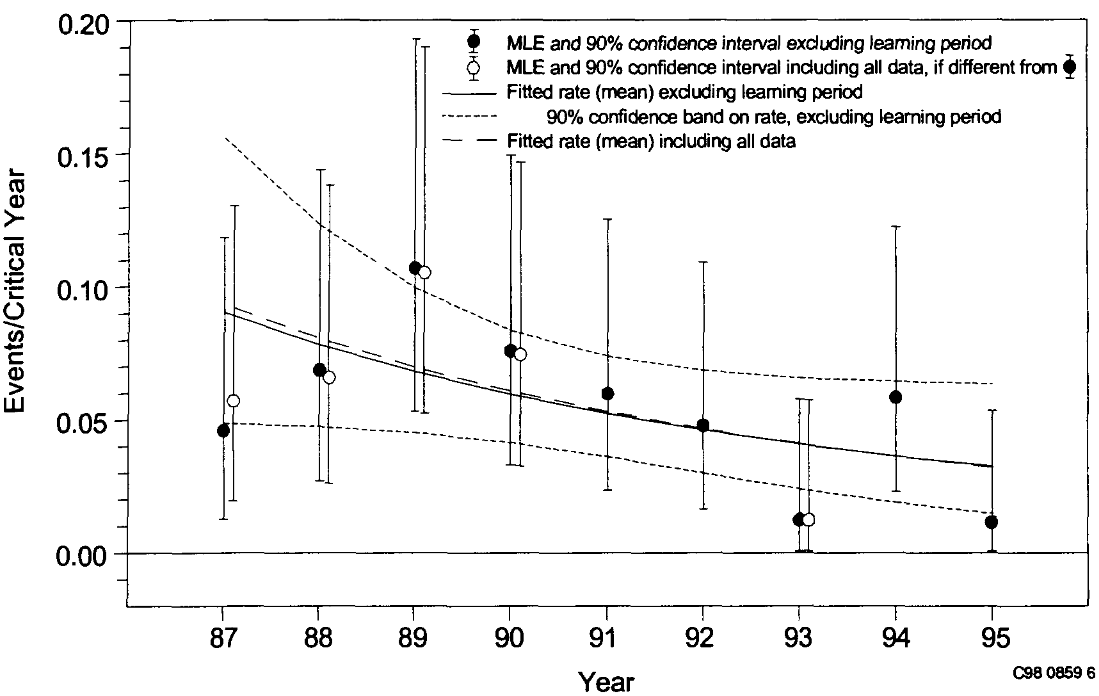

Figure 4-5. Time-dependent frequency for functional impact category H, Fire, for PWRs and BWRs (events per critical year). The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the frequency excluding the learning period at new plants. The p-values for the trends with and without the learning period data are 0.033 and 0.044, respectively. Between-plant differences were not seen.

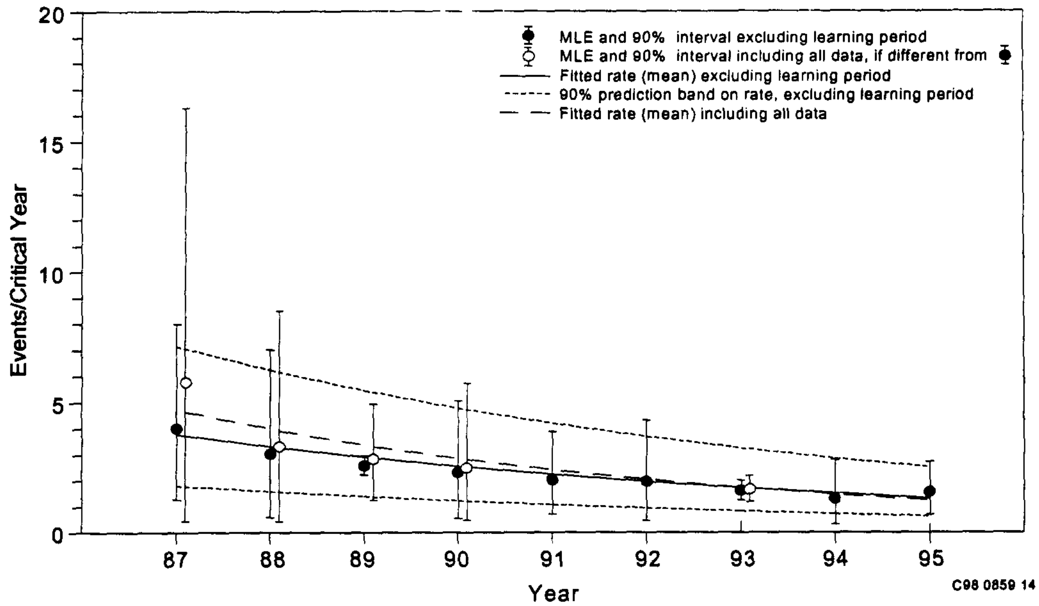

Figure 4-6. Time-dependent frequency for initial plant fault heading Q, General Transients, for PWRs (events per critical year). The points and vertical lines are based on data from individual years and reflect between-plant variation where observed. The dotted lines are a 90% prediction band on the frequency at a random plant excluding the learning period at new plants. The p-value for each trend is 0.0001.

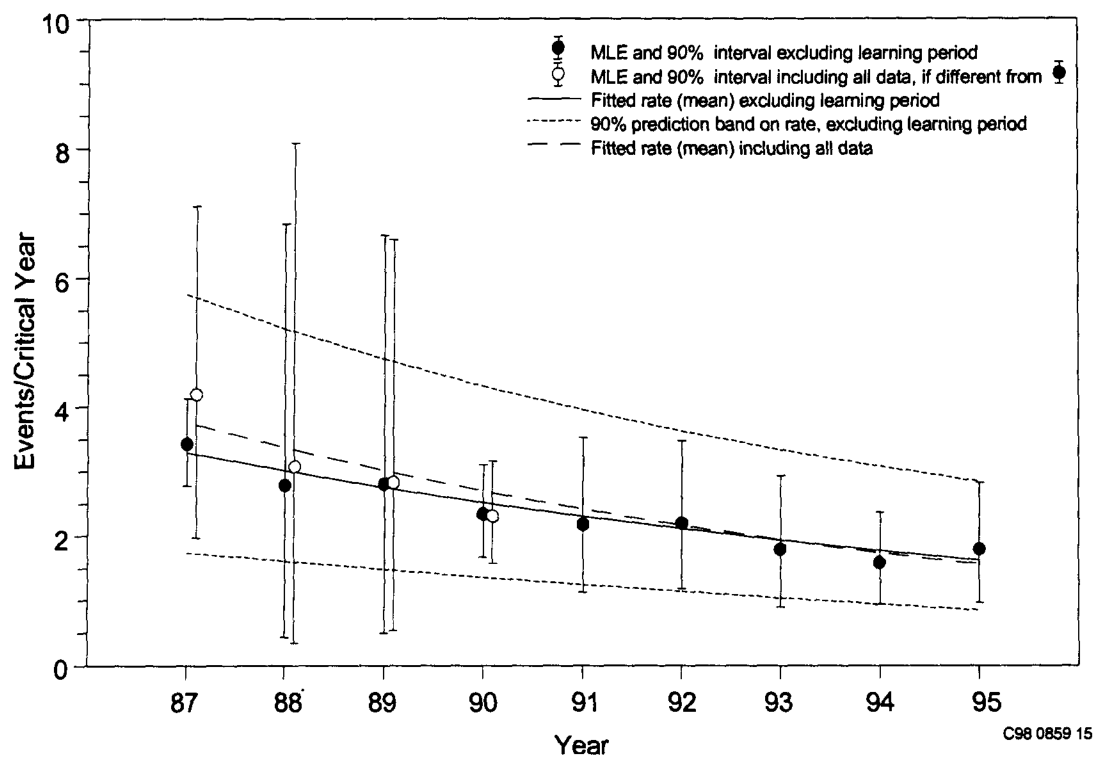

Figure 4-7. Time-dependent frequency for initial plant fault heading Q, General Transients, for BWRs (events per critical year). The points and vertical lines are based on data from individual years and reflect between-plant variation where observed. The dashed lines are a 90% prediction band on the fitted frequency at a random plant excluding the learning period at new plants. The p-value for each trend is 0.0001.

4.3 Plant-Type and Plant-Specific Evaluation

The Pearson chi-squared test was performed to detect differences between plants and between plant types. To ensure these types of effects were not interrelated, the data sets were analyzed for the presence of simultaneous effects (i.e., the two effects that appeared most nearly significant). Further, when between-plant variation was detected, actual numerical differences were examined in addition to statistical significance. Actual numerical differences were measured by considering the ratio of the largest plant frequency (Bayes mean) divided by the smallest plant frequency. If the ratio was larger than 6, the plant-specific frequencies were presented in this report, in tabular and graphical form. (The number 6 was chosen after examination of the data; in no cases was the ratio between 4 and 6.) Otherwise, the plant-specific frequencies were not presented individually. Instead, only the industry distribution, which included between-plant variation, was presented. For example, the General Transient category Q for PWRs had a time trend and between-plant differences that were both statistically significant. However, the ratio of the highest plant-specific mean to the lowest was only 2.6 in any one year. Therefore, the industry distribution is used, but plant-specific results are not presented. Appendices E, F, and G describe the methods and the results of the patterns analyses.

4.3.1 Plant-Type Variations

Plant-type effects (i.e., PWR versus BWR) were identified for Loss of Instrument or Control Air (D), Total Loss of Condenser Heat Sink (L), and General Transient (Q) headings. For the first two headings, the BWR frequency was slightly higher than the PWR frequency. The frequencies for the General Transient heading were about the same for both plant types.

Figure 4-8 provides a bar-chart comparison of frequencies of General Transient categories for PWRs and BWRs based on 1987–1995 experience. PWR frequencies are higher by about a factor of two for Loss of Nonsafety-Related Bus (QC5), Partial Loss of Feedwater Flow (QP2), and Other Reactor Trip (Valid RPS Trip) (QR7). BWR frequencies are higher by about a factor of two than PWR frequencies for Condenser Leakage (QL6), RCS High Pressure (RPS Trip) (QR0), and Manual Reactor Trip (QR6). The PWR frequency for Reactivity Control Imbalance (QR3) is a factor of six higher than BWR, while Core Power Excursion RPS Trip (QR4) category is a factor of eight higher for BWR.

4.3.2 Between-Plant Variations

The models used to estimate plant-specific results cannot account for sharp changes in the event frequency at a plant. Therefore, the analysis of between-plant variations used only the data after the learning period of new plants (i.e., after the first four months of commercial operation).

Figures 4-9 through 4-11 are plots of the plant-specific results (mean and uncertainty) for functional impact categories where the plant mean is greater than the industry average. The first line labeled "Industry" or "All PWRs" in each figure is the entire population of plants in its pooled grouping. When both between-plant differences and a time trend are modeled, the frequencies given refer to the end point of the trend line (1995, the last year of the study).

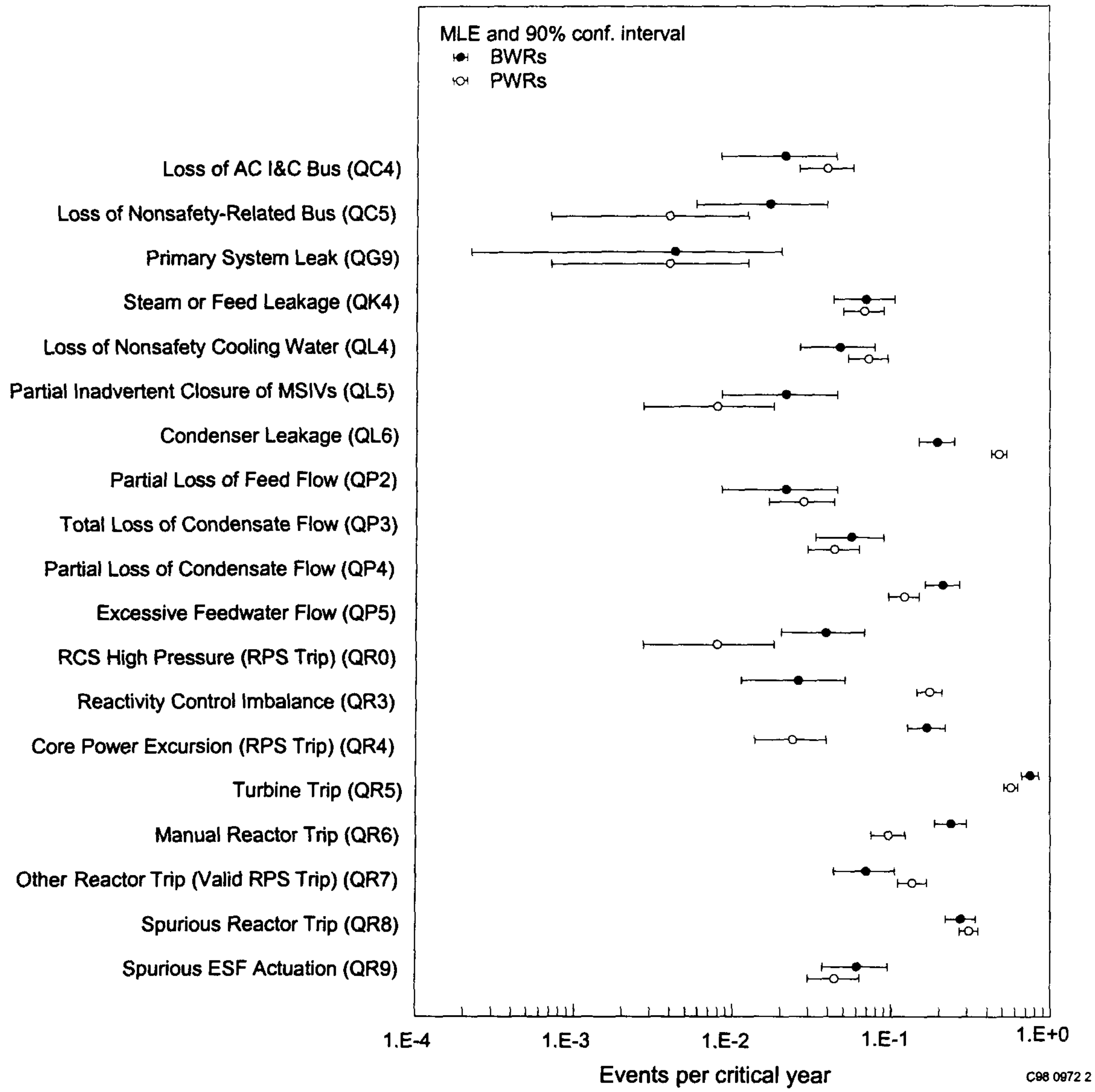

Figure 4-8. A chart comparison of frequencies of General Transient categories for PWRs and BWRs based on all data in the1987–1995 experience.

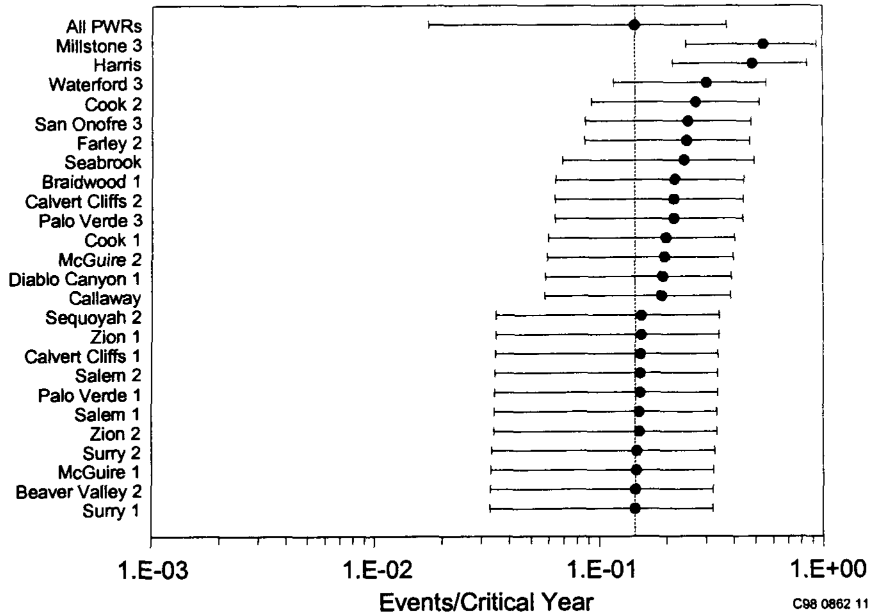

Figure 4-9. Plant-specific frequencies for functional impact heading L, Total Loss of Condenser Heat Sink for PWRs in 1995. Only plants with estimated (mean) values higher than the industry mean are shown. Much of the variation between plants is from functional impact category L2, Loss of Condenser Vacuum. The ratio of the highest mean to the industry mean is 3.8.

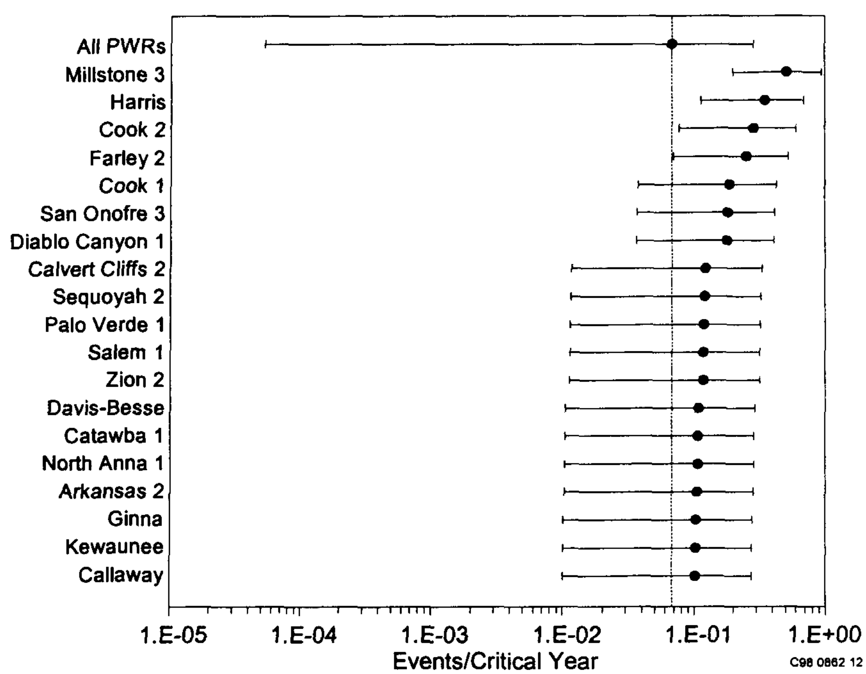

Figure 4-10. Plant-specific frequencies for functional impact category L2, Loss of Condenser Vacuum, for PWRs. Only plants with estimated (mean) values higher than the industry mean are shown. The ratio of the highest mean to the industry mean is 7.6.

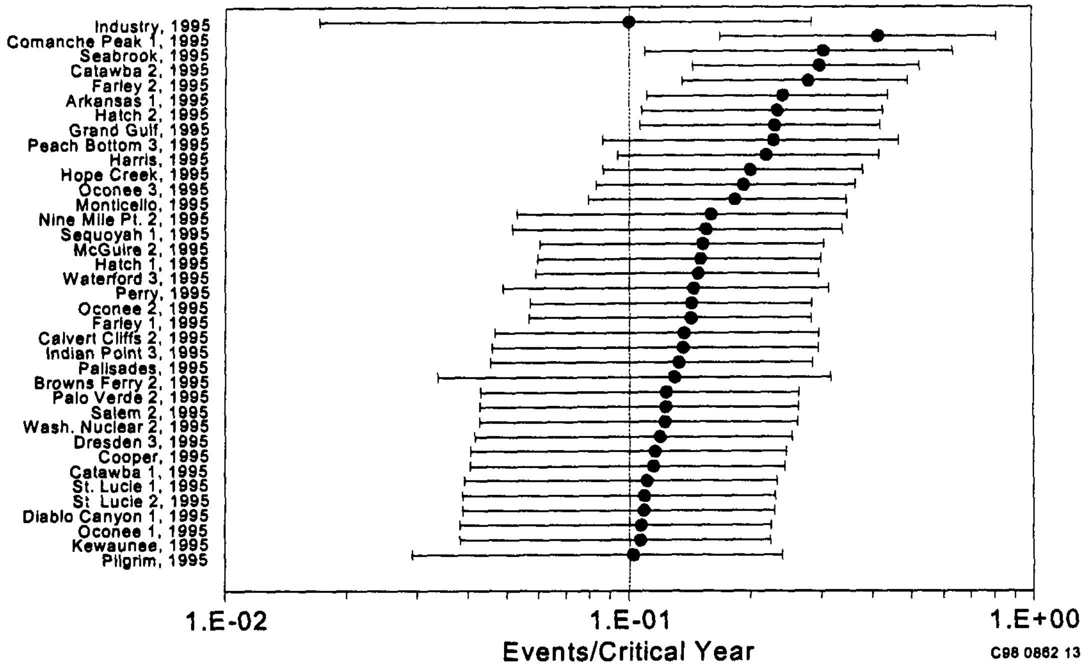

Figure 4-11. Plant-specific frequencies for functional impact category P1, Total Loss of Feedwater Flow, for PWRs and BWRs in 1995. Only plants with estimated (mean) values higher than the industry mean are shown. The ratio of the highest mean to the industry mean is 4.1.

Tables 4-2 and 4-3 provide a summary of the categories and headings identified as having between-plant variation and the plants having frequencies higher than the industry average. (The plant-specific frequencies were presented in this report if the ratio of the largest plant frequency divided by the smallest plant frequency was larger than 6.) The entry "X" in Tables 4-2 and 4-3 denotes a plant mean that is higher than the industry average, but within the industry uncertainty interval for the functional impact category. The entry "XX" identifies a plant mean that is higher by a statistically significant amount (the uncertainty interval for the plant mean is entirely to the right of the industry mean). If the industry is nearly symmetrical about the industry mean, as with Total Loss of Feedwater Flow, then nearly half the plants have an "X". The particular marked plants could change as more data accumulate. However, the plants marked by "XX" have a statistically significant difference in their behavior, which may be candidates for further investigation.

Further evaluation of Table 4-2 shows two of the four plants with an "XX" in the Loss of Condenser Vacuum category (L2) do not have a "XX" in the associated Total Loss of Condenser Heat Sink heading (L). This can be attributed to a lower occurrence of events in the Inadvertent Closure of All MSIVs category (L1), which also falls under the Total Loss of Condenser Heat Sink heading (L). The lower frequency of inadvertent MSIV isolations for these two plants have a greater contribution to the overall frequency of loss of condenser heat sink frequency than the higher loss of condenser vacuum frequency.

Table 4-4 provides a list of plants with mean frequencies higher by a statistically significant amount (the uncertainty interval for the plant means are entirely to the right of the industry mean). The functional impact category Loss of Condenser Vacuum is not listed in Table 4-4 since this category is a subset under the heading Total Loss of Condenser Heat Sink.

Table 4-2. Plants having mean frequencies greater than industry average for the plant-specific variations identified for PWR functional impact categories/headings.

Plant	Total Loss of Condenser Heat Sink (L)	Loss of Condenser Vacuum (L2)[a]	Total Loss of Feedwater Flow (P1)
Arkansas 1			**XX**
Arkansas 2		X	
Beaver Valley 2	X		
Braidwood 1	X		
Callaway	X	X	
Calvert Cliffs 1	X		
Calvert Cliffs 2	X	X	X
Catawba 1		X	X
Catawba 2			**XX**
Comanche Peak 1			**XX**
Cook 1	X	X	
Cook 2	X	**XX**	
Davis-Besse		X	
Diablo Canyon 1	X	X	X
Farley 1			X
Farley 2	X	**XX**	**XX**
Ginna		X	
Harris	**XX**	**XX**	X
Indian Point 3			X
Kewaunee		X	X
McGuire 1	X		
McGuire 2	X		X
Millstone 3	**XX**	**XX**	
North Anna 1		X	
Oconee 1			X
Oconee 2			X
Oconee 3			X
Palisades			X
Palo Verde 1	X	X	
Palo Verde 2			X
Palo Verde 3	X		
Salem 1	X	X	
Salem 2	X		X
San Onofre 2		X	
San Onofre 3	X		
Seabrook	X		**XX**
Sequoyah 1			X
Sequoyah 2	X	X	
St. Lucie 1			X
St. Lucie 2			X

Table 4-2. (continued).

Plant	Total Loss of Condenser Heat Sink (L)	Loss of Condenser Vacuum (L2)[a]	Total Loss of Feedwater Flow (P1)
Surry 1	X		
Surry 2	X		
Waterford 3	X		X
Zion 1	X		
Zion 2	X	X	

a. Loss of Condenser Vacuum is a category under the Total Loss of Heat Sink heading.

Table 4-3. Plants having mean frequencies greater than industry average for the plant-specific variations identified for BWR functional impact categories.

Plant	Total Loss of Feedwater Flow (P1)
Browns Ferry 2	X
Cooper	X
Dresden 3	X
Grand Gulf	**XX**
Hatch 1	X
Hatch 2	**XX**
Hope Creek	X
Monticello	X
Nine Mile Pt. 2	X
Peach Bottom 3	X
Perry	X
Pilgrim	X
Wash. Nuclear 2	X

Table 4-4. Plants with mean frequencies that are higher by a statistically significant amount (the uncertainty interval for the plant mean is entirely to the right of the industry mean).

Total Loss of Condenser Heat Sink: PWR	Total Loss of Feedwater Flow
Harris	Arkansas 1
Millstone 3	Catawba 2
	Comanche Peak 1
	Farley 2
	Grand Gulf
	Hatch 2
	Seabrook

4.4 Infrequent and Rare Events

For some rare event categories, such as loss-of-coolant accidents (LOCAs) whose frequencies are low enough that either few or no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used. These include operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as large and medium break LOCAs. Tables 4-5 and 4-6 list the frequency estimates for rare events obtained from available literature. (Note that the values reported in these two tables are in units of per calendar year.)

4.4.1 Loss-of-Coolant Accidents (LOCAs)—Pipe Breaks

The LOCA frequency estimates provided in this study span the break sizes in the primary system, boundary piping used in NUREG-1150 (USNRC 1990) analyses. Total U.S. nuclear power plant operating experience was used to update WASH-1400 (USNRC 1975) estimates for small pipe break LOCA frequencies. No small break LOCA events were found since WASH-1400 was published. For medium and large pipe-break LOCA, frequencies were estimated by calculating the frequency of leaks or through-wall cracks that have challenged the piping integrity. Further, conservative estimates were used for the probability of a break given a leak. This probability is based on a technical review of information on fracture mechanics, data on high-energy pipe failures and cracks, and assessments of pipe-break frequencies estimated by others since WASH-1400. Due to differences observed in both operating experience and engineering characteristics, separate frequency estimates are given for PWRs and BWRs.

The estimates presented in this report represent a reasonable but conservative adjustment to the previous understanding of the probability of pipe ruptures and LOCA frequencies. Up to the time of this study no definitive LOCA frequency estimates have been made since NUREG-1150, which used WASH-1400 values in many cases. Experience data and engineering understanding of piping failures are much improved since then. In light of this experience, a more complete analysis using data, fracture mechanics analyses, and an expert elicitation process could likely produce more definitive estimates. In the meantime, the available data and current operating experience are sufficient to support updating the best estimates of LOCA frequencies. Since the purpose of probabilistic risk assessments (PRAs) is to reflect best estimates and the associated uncertainties, the results presented here are a reasonable appraoch at producing more accurate PRAs.

Based on this knowledge from the operating experience and the need to provide updated frequencies for NRC PRA programs, the task to update pipe break LOCA frequency estimates was included as an objective of this report. The goal of this effort is to refine the original estimates based on operating experience and current knowledge of pipe break mechanisms. It is recognized that the approach in this report will result in reduction of unnecessary conservatism in LOCA frequency estimates. However, the result is still conservative. Further probabilistic evaluations of the results from fracture mechanics research is required to develop best estimates of pipe break LOCA frequencies that factors in the evaluation current operating, surveillance, and maintenance practices at U.S. nuclear power plants.

Table 3-1 provides the results of the frequency estimates for small, medium, and large pipe break LOCAs.

Appendix J describes the various primary pressure boundary pipe degradation mechanisms at work, intergranular stress corrosion cracking in BWRs, details of the LOCA frequency calculations, and comparisons between the various parameters used in the LOCA frequency results that was extracted from the available information on fracture mechanics analyses and computer code simulations. The last three sections in the appendix are the tables of events used in the analyses, list of references, and a bibliography of information reviewed during the conduct of the effort.

Table 4-5. Frequency estimates of LOCA-related events for BWR plants from available literature.

BWR	Frequency (per calendar year) Mean	Median	Error Factor	Notes
Very small LOCA				
NUREG-1150[a]	2.0E-2		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
IPE population	4.4E-2	2.3E-2	12[b]	Distribution of IPE point estimates
Small LOCA				
WASH-1400[d]		1.0E-3	10	Appendix III, Table 6-9
NUREG-1150[a]	1.0E-3		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
EPRI TR-100380[e]	1.8E-3		7.8	Section 5
IPE population	7.8E-3	8.0E-3	5[b]	Distribution of IPE point estimates
Medium LOCA				
WASH-1400[d]		3.0E-4	10	Appendix III, Table 6-9
NUREG-1150[a]	3.0E-4		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
EPRI TR-100380[e]	2.8E-4		7.8	Section 5
IPE population	1.4E-3	7.6E-4	7[b]	Distribution of IPE point estimates
Large LOCA				
WASH-1400[d]		1.0E-4	10	Appendix III, Table 6-9
NUREG-1150[a]	1.0E-4		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
EPRI TR-100380[e]	3.0E-4		7.8	Section 5
IPE population	4.1E-4	3.0E-4	16[b]	Distribution of IPE point estimates
ISLOCA[f]				
WASH-1400[d]	4.0E-6[f]		10	Appendix V, Section 4.4
NUREG/CR-5928[g]	<1E-8[f]			Negligible, Section 4
NUREG/CR-5124[h]	4.0E-6[f]			Average of 3 plants, Table 4.6
IPE population	9.6E-4[f]	5.0E-6	1000[b]	Distribution of IPE point estimates

a (USNRC 1990)

b Range factor of values for the 28 BWR IPEs, estimated by taking the square-root of the ratio of the maximum value to the minimum value

c (Ericson et al 1990)

d (USNRC 1975)

e (Jamali 1992)

f Values represent the core damage frequency.

g (Galyean et al. 1993)

h. (Chu et al. 1988)

Table 4-6. Frequency estimates of LOCA-related events for PWR plants from available literature.

PWR	Frequency (per calendar year) Mean	Median	Error Factor	Notes
Very small LOCA				
NUREG-1150[a]	2.0E-2		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
IPE population	8.0E-3	6.0E-3	183[b]	Distribution of IPE point estimates
Small LOCA				
WASH-1400[d]		1.0E-3	10	Appendix III, Table 6-9
NUREG-1150[a]	1.0E-3		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
EPRI TR-100380[e]	1.0E-3		7.8	Section 5
IPE population	6.9E-3	4.5E-3	8.9[b]	Distribution of IPE point estimates
Medium LOCA				
WASH-1400[d]		3.0E-4	10	Appendix III, Table 6-9
NUREG-1150[a]	1.0E-3		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
EPRI TR-100380[e]	3.2E-4		7.6	Section 5
IPE population	7.45E-4	7.1E-4	5.1[b]	Distribution of IPE point estimates
Large LOCA				
WASH-1400[d]		1.0E-4	10	Appendix III, Table 6-9
NUREG-1150[a]	5.0E-4		3	NUREG/CR-4550[c] Vol. 1, Table 8.2-4
EPRI TR-100380[e]	1.4E-4		7.6	Section 5
IPE population	3.0E-4	3.0E-4	8.4[b]	Distribution of IPE point estimates
ISLOCA[f]				
WASH-1400[d]	4.0E-6[f]		10	Appendix V, Section 4.4
NUREG-1150[a]	4.0E-7[f]	2.0E-8	100	NUREG/CR-4550[c] Vol. 2, Sec. 5.2
NUREG/CR-5928[g]	2.0E-6[f]			Section 6
IPE population	5.1E-5[f]	5.6E-7	2055[b]	Distribution of IPE point estimates
Steam Generator Tube Rupture				
NUREG-1150[a]	1.0E-2		7.9	NUREG/CR-4550[c] Vol. 2, Sec. 5.2
NUREG/CR-6365[h]	6.3E-3	6.1E-3	1.77	These values were recalculated based on the 8 single tube failures from U. S. nuclear power plants.
IPE population	1.4E-2	1.0E-2	3.4[b]	Distribution of IPE point estimates
Reactor Coolant Pump Seal LOCA				
NUREG-1150[a]	2.0E-2	—	3	NUREG/CR-4550[c] Vol. 2, Table 8.2-4

a. (USNRC 1990)

b. Range factor of values for the 51 PWR IPEs, estimated by taking the square-root of the ratio of the maximum value to the minimum value

c. (Ericson et al 1990)

d (USNRC 1975)

e. (Jamali 1992)

f Values represent the core damage frequency.

g (Galyean et al 1993)

h. (MacDonald et al. 1996)

4.4.2 Interfacing System LOCA

Interfacing system loss-of-coolant accidents (ISLOCAs) are a class of accidents that can result in the over-pressurization and rupture of systems that interface with the reactor coolant system outside containment. ISLOCAs have been a concern with regard to public health risk due to the potential for fission product release directly to the environment, bypassing the containment structure. No ISLOCA events have been identified in the total U.S. operating experience (1969–1997). However, during the course of this study, one ISLOCA precursor was identified. The identification of only one ISLOCA precursor in the nearly 2,000 LERs over the nine-year study period is not unexpected, given that only LERs containing documented reactor trips or manual trips from power were included in the study set. The types of activities that would normally lead to the identification of an ISLOCA precursor—maintenance and testing on systems that interface with the reactor coolant system—are usually performed on interfacing systems while the plant is shutdown.

The ISLOCA precursor event identified in this study did not result in a release of reactor coolant to the environment. The Arkansas Nuclear One Unit 1 event occurred as a result of a High Pressure Safety Injection System (HPSI) check valve failing to reseat along with the presence of a differential pressure (d/p) condition between two primary loops due to a tripped reactor coolant pump. This d/p and the failed open check valve allowed reactor coolant system water to backflow outside of containment via the HPSI system. The backflow of the high-temperature reactor coolant heated the HPSI piping enough to cause some combustible material in contact with the piping to start smoldering. The smoke activated the fire alarm which alerted operators to the condition. For additional information, see LER 313/89-002 and LER 313/89-004.

Concerns over frequency and other issues associated with ISLOCA precursor events have prompted risk assessments of ISLOCAs for Babcock & Wilcox (B&W), Combustion Engineering, Inc. (CE), and Westinghouse four-loop Ice Condenser plants [NUREG/CR-5604 (Gaylean and Gertman 1992), NUREG/CR-5744 (Kelly et al. 1992), and NUREG/CR-5745 (Kelly et al. 1992a), respectively]. These assessments provide qualitative and quantitative information on hardware, human factors, and accident consequence issues that dominate the ISLOCA risk to these three reactor vendor types. To accomplish this, a methodology based on PRA, human factors, and human reliability techniques was developed. Refer to any of the referenced reports for more details.

The frequency of ISLOCA was not estimated as a single initiating event probability but evaluated in the context of a sequence of events that considered such factors as likelihood of high pressure, the probability of rupture, the likelihood of operator recovery. The ISLOCA frequencies selected for comparison to PRA/IPEs are 2.0E-6 per critical year for PWRs and less than 1.0E-8 per critical year (negligible) for BWRs. These estimates, which were calculated from the ISLOCA Research Program [NUREG/CR-5928 (Gaylean et al. 1993)], represent average total core damage frequencies from all ISLOCA events. Caution is advised when comparing ISLOCA initiating event frequencies. Wide variation exists in modeling ISLOCA sequences. In the early models (e.g. Event-V in WASH-1400), it was assumed that any LOCA bypassing the containment sump would result in core damage, and that any pressurization of non-primary system piping up to primary system pressure would result in a rupture. Hence the "initiating event" frequency (e.g., failure of two check valves in series) was synonymous with core damage frequency. Results from the ISLOCA Research Program indicate the results from the early models in WASH-1400 were overly conservative. Piping found in commercial nuclear power plants has a significant safety margin to failure beyond its design specification, such that even most low-pressure piping has a high likelihood of maintaining integrity even at reactor coolant system pressures. Also there are often options available to the control room operators for either isolating any ruptures outside containment or maintaining inventory long enough to shutdown the reactor and depressurizing the reactor coolant system thereby avoiding core damage. Consequently, there is no consistent definition for an ISLOCA initiating event. Some definitions include the events leading to the leakage, while others include the entire sequence of events. This lack of consistency is evident in the range of values found in the IPEs. For PWRs, the range of values used in IPEs for ISLOCA "initiating events" comprises a low of 6.0E-10 per critical year to a high of 2.5E-3 per critical year. For BWRs, the range spans a low 1.3E-8 per critical year to a high of 1.3E-2 per critical year.

4.4.3 Steam Generator Tube Rupture

This study identified three steam generator tube rupture (SGTR) events. The SGTR frequency estimate based on the three SGTR events is 7.0E-3 per critical year. Based on the current PWR population, this frequency is equivalent to about one event every two calendar years. The last SGTR identified in the 1987–1995 experience occurred at Palo Verde 2 in 1993. Since no SGTR events were identified in the last two years of this study, the 1996 through 1997 operating experience was screened for SGTR events to determine if a trend existed. No SGTR events were found in the additional two years. Further trend analysis of SGTR frequency, using the operating experience prior to 1987 (five events) and after 1995 (no events), showed no statistical evidence of a decreasing trend in the SGTR frequency. This result is driven by the small size of the data population. A sensitivity calculation showed a trend would become statistically significant in the year 2001 if no other SGTR events occur up to that year. Although the limited data provided no statistical basis for a decreasing trend in SGTR frequency, there may be engineering reasons (e.g., better inspection techniques, increased sleeving or plugging of tubes, and improved secondary system chemistry control) for observing no SGTR events from 1993 to the present.

Table D-10 of Appendix D provides a summary listing of SGTR events identified in this study.

4.4.4 Reactor Coolant Pump Seal LOCA

No catastrophic seal failures in a reactor coolant pump have occurred in the U.S. since 1980. Instances of reactor coolant pump seal failures in PWRs in the 1970s resulted in addressing seal degradation as a potential mechanism for LOCAs in PWRs. Reactor coolant pump seal failures in BWRs during the 1970s occurred at a frequency similar to that experienced in PWRs; however, operating experience indicated that BWRs exhibited a lower leak rate for a majority of the seal failures. The low leak rate, larger reactor vessel water injection capabilities, and isolation valves on the reactor coolant pump loops mitigated the potential problem in BWRs.

NUREG/CR-4400 (Azarm and Mitra 1985) identified eight reactor coolant pump seal failures in PWRs that occurred between 1971 and 1985 with leak rates greater than 10 gpm. Two of these events had leak rates in excess of 100 gpm. NUREG/CR-4400 also identified numerous smaller leak rate events caused by seal degradations. A recent AEOD sponsored report, NUREG/CR-6582 (Shah et al. 1997), *Assessment of Pressurized Water Reactor Primary System Leaks*, provides an evaluation of reactor coolant pump seal performance based on 1985–1996 operating experience. The NUREG/CR-6582 report identified eight reactor coolant pump seal leak events in PWRs: one event had a leak rate of 40 gpm and the remaining seven events had leak rates less than the technical specifications minimum for identified leakage (10 gpm). Two reactor coolant pump seal failure events associated with reactor trips were identified in the 1987–1995 operating experience. Both events had leak rates less than 40 gpm and involved seal degradations that were caused by events external to the seals. The event descriptions of these two events are summarized in Appendix I.

NUREG/CR-6582 reported a decline in reactor coolant pump seal leaks during the 1991–1996 time frame. Furthermore, NUREG/CR-6582 suggested that the improvement in reactor coolant pump seal performance in PWRs could be due to modifications in the seal designs and the replacement of pump seals in PWRs with the improved seals. The lack of a catastrophic seal failure in the last eighteen years may be the result of the improved seal designs increasing the time for the plant to respond to a seal failure, thereby preventing the occurrence of a catastrophic seal failure that results in small LOCA. A detailed analysis of reactor coolant pump seal performance was outside the scope of this study.

Frequency estimates. Due to the rare occurrence of reactor coolant pump seal LOCA, total U. S. operating experience (1969–1997) for PWRs was used to calculate the frequency. The reactor coolant pump seal LOCA frequency estimate based on two events in the total U.S. PWR operating experience (1969–1997) or 1019 critical years is 2.5E-3 per critical year. This estimate was calculated using a Jeffreys noninformative prior in a Bayes updated distribution.

The two reactor coolant pump seal failure events used in the calculation are: a 500-gpm leak rate at Robinson in May 1975; and a 300-gpm leak rate at Arkansas Nuclear One Unit 1 in May 1980. Both events resulted in the manual actuation of safety injection. The plant was manually tripped during the Arkansas event, whereas Robinson event resulted in an automatic reactor trip due to a turbine trip on high steam generator level during the rapid load reduction. The event descriptions of these two events are summarized in Appendix I.

A sensitivity calculation was performed to compare the frequency estimates based on the two events during the total PWR operating experience and no events during the 1987–1995 experience. Using a Jeffreys noninformative prior in a Bayes updated distribution, the seal LOCA frequency based on the total U.S. experience was only about a factor of two higher than the estimate calculated from 1987–1995 experience.

4.4.5 Inadvertent and Stuck Open Safety/Relief Valves

This study identified twelve reactor trip events in the 1987–1995 operating experience associated with primary system safety/relief valves (SRVs) failure to close. Safety relief valves included in this study are PWR pressurizer power-operated relief valves (PORV), BWR main steam line code safety valves, and BWR Automatic Depressurization System relief valves. The mechanisms that caused the safety relief valves to initially open can be divided into three groups: SRV openings induced by a primary system pressure transient (two events); spurious SRV openings while at power (three events); and surveillance testing of SRVs in BWRs while at power (seven events). A transient that resulted in an SRV prematurely opening due to a lower than normal setpoint was not classified as a stuck-open-SRV event.

Two PWR events were identified where an SRV inadvertently opened during power operations and immediately closed after the manual reactor trip. A review of the stuck-open SRV events revealed the following:

- All stuck-open-SRV events involved one valve.
- Only one event (Fort Calhoun stuck open pressurizer code safety valve on 07/03/92) resulted in high-pressure safety injection actuation. The normal reactor coolant makeup system was able to maintain reactor coolant system inventory.
- No additional risk-significant events (i.e., no other events from a functional impact category) were identified during the reactor trip sequence associated with the SRV events.
- All but one event where the SRV failed to close during operational testing occurred at low power levels (less than 20%).
- All events that were spurious and transient induced resulted in a manual reactor trip. All but one of these events occurred at high power levels (greater than 90%).
- No inadvertently open or stuck open pressurizer PORV events were found in the 1987–1995 operating experience.

- All three inadvertent open SRV events in the BWRs resulted in the valve remaining in the stuck open position throughout plant shutdown, whereas, the SRVs immediately closed after the manual reactor trip in both inadvertent open SRV events in the PWRs.

Appendix I provides the basis for the classification of inadvertent and stuck open SRV events.

Frequency calculations. The frequency estimates for the three stuck open safety/relief valve categories (G2, G4, and G5) for BWRs and PWRs were calculated using a Jeffreys noninformative prior in a Bayes updated distribution. The frequency estimates were based on the 1987–1995 operating experience, except for the Stuck Open: 2 or More Safety/Relief Valves category, which was based on total U.S. operating experience (1969–1997). Table 3-1 provides the results of the calculations.

4.4.6 Very Small LOCA/Leak

This category is defined as a pipe break or component failure resulting in a reactor coolant system leak rate between 10 to 100 gpm. Four very small LOCA/leak events occurred in PWRs during the 1987–1995 time period. No events were found for BWRs. These events are: a 74-gpm steam generator tube leak from a tube plug at North Anna Unit 1 (LER 338/89-005); a 40-gpm leak from a failed reactor coolant pump seal, cavity pressure sensing instrument line at Arkansas Nuclear One Unit 2 (LER 368/88-011); a 40-gpm leak from a crack in the letdown drain line inside containment at McGuire Unit 1 (LER 369/87-017); and one 87-gpm leak resulting from a failure of an instrument line compression fitting at Oconee 3 (LER 287/91-008). The reactor was manually tripped in the Arkansas event. In the other three events, the leaks occurred after the automatic reactor trip. No other risk-significant events (i.e., no other events from a functional impact category) occurred during any of these events.

Frequency estimates. The frequency of a very small LOCA/leaks is 6.2E-3 per critical year, based on four events in the 1987–1995 operating experience or 729 critical years. The estimate was calculated by using a Jeffreys noninformative prior in a Bayes updated distribution.

4.4.7 ATWS

An anticipated transient without scram (ATWS) event is an operational transient (for example, total loss of feedwater flow, loss of condenser vacuum, or loss of offsite power) followed by failure of the reactor protection system to shut down the reactor. In 1980, during a routine shutdown at Browns Ferry Unit 3, seventy-six of the 185 control rods failed to fully insert when the reactor was manually tripped. The cause of the control rod malfunction was the retention of a significant amount of water in the scram discharge volume. In February 1983, Salem Unit 1 experienced failure of the reactor trip system breakers due to a common cause failure. Although the reactor was not shut down by the automatic trip function of the reactor protection system, the reactor was manually tripped 30 seconds later. Both events prompted the issuance of NRC bulletins to correct these generic problems.

This study identified several reactor trip events that had ATWS implications. In December 1995, three control rods failed to fully insert (2.5 inches from the bottom) during a turbine trip/reactor trip at South Texas Unit 1 (LER 498/95-013). In January 1996, a similar event at Wolf Creek occurred when five control rods failed to fully insert (2 to 8 inches from the bottom) during a manual reactor trip (LER 482/96-001). In both cases, the partially stuck control rods were located in high burn-up fuel assemblies. NRC Bulletin 96-01 (USNRC 1996) was issued to assess the operability of control rods, particularly in high burn-up fuel assemblies.

In January 1985, Sequoyah Unit 2 experienced a reactor trip as a result of a valid trip signal. However, the "A" reactor trip breaker failed to open automatically due to a failed solid-state protection system under-voltage, output board (LER 328/85-002). A similar event occurred at Shearon Harris in June 1991. In this event the plant experienced an automatic reactor trip due to a spurious low-reactor coolant system, loop-flow signal. The "A" reactor trip breaker failed to open automatically due to a failed solid-state protection system under-voltage, output board (LER 400/91-010).

Since an ATWS is a conditional event, meaning an additional system failure is required, the ATWS event is not included as an initiating event category.

4.4.8 Loss of Safety-Related Cooling Water System

No total loss of safety-related cooling water system events that were associated with a reactor trip have been identified in the 1987–1995 operating experience. Only one total loss of safety-related service water system that was associated with a reactor trip was identified by the ASP Program (Forester et al. 1997) and NUREG-1275 (Lam and Leeds 1988) in the total U.S. operating experience (1969–1997). In January 1982, Brunswick 2 experienced a reactor trip due to low condenser vacuum and a main steam line isolation. When attempting to align suppression pool cooling to remove decay heat, the pumps in both Residual Heat Removal Service Water (RHRSW) loops failed to start due to low suction header pressure lockout signals. However, decay heat removal was restored through the condenser shortly after the reactor trip. Flow from one RHRSW loop was established four hours after the reactor trip. The suction header pressure switches were found inoperable or degraded because the sensing lines were partially plugged with sediment, a switch was damaged in loop A, and the power supply to the loop B pressure switch was turned off (LER 324/82-005). The conditional core damage probability estimated from the ASP program (Forester et al. 1997) for this event was 2.4E-4.

Six partial losses associated with a reactor trip were identified in the 1987–1995 experience. None of these losses resulted in the initiation of a reactor trip. Two of the six partial losses of safety-related service water events had a direct effect on the performance of risk-important systems. At Vermont Yankee in 1991, following the expected start of both Emergency Diesel Generators (EDG) during a loss of offsite power (LOSP) event, the EDG heat exchangers were operating at reduced flow and the station air compressor coolers were operating with reduced and reversed flow. Even with the reduced flow, the EDGs were able to provide power throughout the event (LER 271/91-009 and 012). The conditional core damage probability estimated from the ASP program (Minarick et al. 1992) for this event was 2.9E-4. At Calvert Cliffs Unit 1 in 1987, an isolation of the safety-related component cooling water system to the reactor coolant pump seals resulted in the securing of the reactor coolant pumps 15 minutes after the reactor trip. The pump seals were not damaged (LER 317/87-00).

One event was found in the 1987–1995 operating experience in which a loss of a *nonsafety-related* cooling water system affected the operation of a risk-important component. In 1989, Palo Verde Unit 3 experienced a six-gpm reactor coolant pump seal leak caused by the loss of the component cooling water system. The component cooling water system was lost due to a failure of its nonsafety-related electrical bus to fast transfer following a reactor trip. The charging system was secured approximately thirty minutes after the reactor trip to prevent pressurizer level from exceeding the maximum limit (LER 530/89-001).

Summaries of the total and partial loss of safety-related service water events are provided in Appendix I.

An NRC report on service water system (SWS) failures and degradations (Houghton 1998) identified no failures during the 1986–1996 time period that resulted in an actual complete loss of cooling capability. The few short term losses of SWS that could impact core cooling capability were identified and recovered promptly. These involved four events while the plant was at power and five events during shutdown operations. For the at-power events, the recovery time were usually less than 30 minutes. Most of these events involved failure of one train while the other redundant train of SWS or emergency power was out-of-service for maintenance or testing. However, none of these events were associated with a reactor trip.

Frequency estimate. Due to the rare occurrence of total losses of safety-related service water systems, the total U. S. operating experience (1969–1997) was used to estimate the frequency. A total loss of safety-related service water frequency of 9.7E-4 per critical year was calculated in this study. This estimate is based on the complete system failure at Brunswick Unit 2 in the 1544 critical years of total U.S. operating (1969–1997) experience. The estimate was calculated using a Jeffreys noninformative prior in a Bayes updated distribution.

The low frequency of total loss of safety-related cooling water systems indicates that during power operation, plant designs provide a level of redundancy of these systems sufficient to mitigate the effects of disturbances of safety-related cooling water systems on plant operations. Further, the low frequency associated with a loss of safety-related cooling water systems implies that events totally disrupting the service water systems are rare.

4.5 Insights

4.5.1 Dominant Transients

Table 4-7 provides a listing of the dominant transients for two groups of event categories: General Transients (under heading Q) and the more risk-significant categories (under functional impact headings B through P). The values listed in the Table 4-7 for each group represent percentages of the number of events in each category as compared to the total number of events for each plant type (PWR and BWR) and to the overall total number of events in both plant types (PWR and BWR combined).

General transients contribute 77% of the events. For the General Transient group, Turbine Trips was the major contributor in both plant types (PWR and BWR). Of the more risk-significant categories (the remaining 23%), the more frequent events in PWRs and BWRs were Total Loss of Condenser Heat Sink and Total Loss of Feedwater Flow. The frequency of Total Loss of Feedwater Flow was higher in PWRs, whereas Total Loss of Condenser Heat Sink frequency was higher in BWRs. Both events posed a challenge for the plant's mitigation systems to remove decay heat after a reactor trip. However, the Total Loss of Condenser Heat Sink in BWRs imposes a more severe challenge to the primary containment. Steam relief to the suppression pool results in the pressurization of the primary containment and the reduction of heat removal capacity of the suppression pool. The initiating transients leading to the Total Loss of Condenser Heat Sink and Total Loss of Feedwater Flow are discussed in the next section.

4.5.2 Dominant Contributors to Risk-Significant Events

The contributions to the Total Loss of Condenser Heat Sink, Loss of Condenser Vacuum, Inadvertent Closure of All Main Steam Isolation Valves (MSIVs), and Total Loss of Feedwater Flow categories discussed below are based on the summary counts of initial plant faults correlated to subsequent functional impacts provided in Table D-13 of Appendix D. (An explanation of how to use the data provided in Table D-13 is included in the notes at the bottom of the table.)

Total Loss of Condenser Heat Sink. Table 4-8 provides a summary of the contributors to the Total Loss of Condenser Heat Sink. The top contributors to the Total Loss of Condenser Heat Sink (LOHS) in BWRs are transients that result in an Inadvertent Closure of All MSIVs (60% of all LOHS events in BWRs) and the Loss of Condenser Vacuum (37%). In PWRs, the contribution of each of these transients was about

Table 4-7. Summary of dominant transient categories contribution to the functional impact and general transient headings.

	% Contribution of Functional Impacts (PWR and BWR)	% Contribution Within a Plant Type	
		PWR	BWR
Functional Impacts (23% of all events)			
Total Loss of Feedwater Flow (P1)	31	39	21
Total Loss of Condenser Heat Sink (L)	38	28	51
Loss of Instrument or Control Air (D1)	7	5	9
Loss of Offsite Power (B1)	6	9	4
Fire (H1)	8	10	5
Leaks/Stuck Open Safety/Relief Valves (F1, G1,G2)	3	3	4
Others (each less than 1%)	7	6	6
Totals	100	100	100

	% Contribution of General Transients (PWR and BWR)	% Contribution Within a Plant Type	
		PWR	BWR
General Transients (77% of all events)			
Turbine Trip (QR5)	26	23	32
Partial Loss of Feedwater Flow (QP2)	16	20	8
Spurious Reactor Trip (QR8)	12	13	12
Manual Reactor Trip (QR6)	6	4	10
Excessive Feedwater (QP5)	6	5	9
Reactivity Control Imbalance (QR3)	5	7	1
Other Reactor Trip (QR7)	5	6	3
Others (each 3 % or less)	24	22	25
Totals	100	100	100

Table 4-8. Summary of the contributors to Total Loss of Condenser Heat Sink.

Category	% Contribution (PWR and BWR)	% Contribution Within a Plant Type PWR	BWR
Inadvertent Closure of all MSIVs (L1)	55	46	60
Loss of Condenser Vacuum (L2)	40	46	37
Turbine Bypass Unavailable (L3)	5	8	3
Totals	100	100	100

equal. The Turbine Bypass Unavailable category was a negligible contributor to the Total Loss of Condenser Heat Sink frequency for both BWRs and PWRs.

Loss of Condenser Vacuum. Table 4-9 provides a summary of the contributors to the Loss of Condenser Vacuum. The two major causes of Total Loss of Condenser Heat Sink were problems associated directly with the condenser (e.g., vacuum pump/steam air injector problem, condenser leakage) and problems associated with the circulating water system (i.e., pump trip, traveling screen blockage).

Inadvertent Closure of All Main Steam Isolation Valves (MSIVs). Table 4-10 provides a summary of the contributors to the Inadvertent Closure of All Main Steam Isolation Valves (MSIVs) events. Most MSIV isolations were caused by General Transients (heading Q). The top contributors to the Inadvertent Closure of All MSIVs were problems associated directly with the MSIVs: spurious actuations of the engineering safety features system, turbine trips, and problems associated with the feedwater and condensate system. The contribution of these four items was about the same.

Total Loss of Feedwater Flow. Table 4-11 provides a summary of the contributors to the Total Loss of Feedwater Flow. The major contributor to Total Loss of Feedwater Flow is directly related to problems within the feedwater system. These problems include: the trip of the only operating feedwater pump while operating at reduced power; the loss of a startup or an auxiliary feedwater pump normally used during plant startup; the loss of all operating feed pumps due to trips caused by low suction pressure, loss of seal water, or high water level (BWR reactor level or PWR steam generator level); anticipatory reactor trip due to loss of all operating feed pumps; and manual reactor trip in response to feed problems (characteristic of a total loss of feedwater flow), but prior to automatic reactor trip.

Table 4-9. Summary of the contributors to Loss of Condenser Vacuum.

Category	% Contribution (PWR and BWR)
Problems directly related to the condenser (Initial plant fault categories L2 & QL6)	58
Problems related to the circulating water system: Loss of Nonsafety-Related Cooling Water (QL4)	36
Others	6

Table 4-10. Summary of the contributors to Inadvertent Closure of All Main Steam Isolation Valves (MSIVs).

Category	% Contribution (PWR and BWR)
Problems related to MSIVs (Initial plant fault categories L1 & QL5)	22
Spurious ESF Actuation (QR9)	21
Transients related to feedwater flow problems (Initial plant fault categories P1, QP3, & QP5)	15
Turbine Trip (QR5)	14
Others	28

Table 4-11. Summary of the contributors to the Total Loss of Feedwater Flow.

Category	% Contribution (PWR and BWR)
Total Loss of Feedwater Flow (Initial plant fault category P1)	54
Loss of Condensate System Flow (QP3 & QP4)	20
Loss of Support System (Initial plant fault categories D1, C1, QC4, QC5, & QL4)	13
Others	13

Manual Reactor Trip. Manual reactor trip events occurred in 20% (406 events) of all reactor trip events (manual and automatic). Of the events containing a manual reactor trip, approximately one-fourth of these were the result of a manual reactor trip as the initial plant fault. The remaining three-fourths occurred subsequent to the initial plant fault from categories other than Manual Reactor Trip. (These subsequent manual trip events were collected in the database under the special interest group.) Manual reactor trips that are the initial plant fault were usually the result of other plant or component conditions that could not be characterized by a transient category in the initial plant fault group. Examples of the underlying cause of manual trips include: reactor coolant pump oil leak, high reactor water conductivity, feedwater flow oscillations, sudden reduction in electrical load, and erratic turbine control valve positioning. A review of the manual reactor trip data resulted in the following insights:

For manual reactor trips that occurred as the initial plant fault;

- The frequency of manual reactor trip as the initial plant fault is two times higher for BWRs than for PWRs (Refer to Figure 4-9).

- Only three out of 103 Manual Reactor Trip (category QR6) events that occurred as the initial plant fault were associated with an additional event from a more risk-significant category (under heading B through P) after the manual reactor trip. This indicates that nearly all manual reactor trips were associated with faults that were general transient in nature.

For manual reactor trips occurring subsequent to the initial plant fault;

- The frequency of Manual Reactor Trip events subsequent to Total Loss of Condenser Heat Sink events were higher in BWRs than PWRs by a factor of 30, whereas, the frequency of Manual Reactor Trip events subsequent to feedwater-related events (i.e., total and partial loss of feedwater flow, excessive feedwater flow) were higher in PWRs than BWRs by a factor of 25.

- Manual Reactor Trip events in PWRs were three times more likely to occur as the result of a General Transient event (under heading Q) than the combination of the other categories of the more risk-significant events (under headings B through P). In BWRs, the division was about even.

- The top contributors to Manual Reactor Trip events in BWRs were Total Loss of Condenser Heat Sink (31%) and Stuck Open Safety/Relief Valves (12%) events. The leading contributor to Manual Reactor Trip events in PWRs was the Total Loss of Feedwater Flow (20%).

Dual-unit reactor trips. This study identified twelve cases in the 1987–1995 experience where two reactors at a common site tripped simultaneously due to a related cause. This frequency equates to an expectation across the industry of about one dual-unit trip per year. All but one dual-unit reactor trip were related to an electrical disturbance or loss of offsite power. The electrical problems were caused by an electrical fault in the plant switchyard or site transmission line that affected both units simultaneously, or by an electrical fault in one unit that propagates to the neighboring unit through a common or connected switchyard. Three of these dual-unit trip events were related to voltage surges caused by lightning strikes to the plant that caused multiple control rods to drop into the core. One other dual-unit trip event was caused by manual reactor trips of both reactors due to the loss of the common station air system.

Table D-15 of Appendix D provides a listing of the dual reactor trip events found in the 1987–1995 experience.

4.5.3 Conditional Occurrences of Risk-Significant Events

The data under each risk-significant event category was evaluated for insights regarding the conditional occurrence of risk-significant events following the reactor trip initiator. In order to perform this review, the data from the initial plant fault and functional impact groups (see Section 2.2) were compared for each risk-significant event heading (initial plant fault and functional impact headings B through P). For the purpose of this discussion, an event that occurred as the reactor trip initiator was called an initial plant fault and a risk-significant event that occurred after the reactor trip initiator was called a *subsequent functional impact*. The evaluation of the conditional occurrences of risk-significant events in the 1987–1995 operating experience reveal that:

- In general, one half of the more risk-significant events (under headings B through P) were transient induced, meaning they occurred after the initial plant fault.

- *Loss of Offsite Power (heading B).* About one-half of the loss of offsite power events occurred after the reactor trip initiator, meaning they were subsequent functional impact events. As would be expected following the loss of offsite power, condenser heat sink and main feedwater are lost. The only risk-important event that was associated with, but not caused by, the LOSP event was the degradation of service water flow to both emergency diesel generators at Vermont Yankee in 1991. (See Section 4.4.8 for further details of this event.)

- *Loss of Instrument or Control Air (heading D).* Approximately 60% of all loss of instrument or control air events had no impact on plant mitigation systems to remove reactor decay heat. The remaining 40% of the events were associated with the total loss of feedwater flow and loss of condenser heat sink events. Approximately three-fourths of all loss of instrument and control air events occurred as the reactor trip initiator.

- *Partial Loss of Service Water (category E2).* Two of the six partial losses of safety-related cooling water events had an effect on the performance of risk-important systems. At Vermont Yankee in 1991, the degradation of the service water system resulted in reduced flow to the emergency diesel generator heat exchangers and station air compressor coolers during a loss of offsite power event. At Calvert Cliffs in 1990, a loss of control air to containment isolation valves on the component cooling water system isolated cooling to the reactor coolant pump seals. (Section 4.4.8 provides further details of these events.)

- *Loss-of-Coolant Accident (categories under heading G)*. Of the LOCA-related categories with events found in the 1987–1995 operating experience (i.e., Very Small LOCA/Leak and Stuck Open: 1 Safety/Relief Valve), no other subsequent functional impact events or risk-significant events occurred during the reactor trip sequence.

- *Fire (heading H)*. Most fire events (80%) that were associated with a reactor trip sequence occurred as the reactor trip initiator. None of the fire events had an adverse impact on safety-related systems or structures. In three events, offsite brush or forest fires which affected transmission lines resulted in the subsequent loss of offsite power. In another reactor trip event, a reactor coolant pump lube oil fire was extinguished inside containment. A review of the other fire events indicate that the most fires occurred in the Turbine Building (51%) and in the plant switchyard (28%). A manual reactor trip was initiated for 25% of the fire events. Only 15% of fire events were related to hydrogen gas (e.g., main generator, turbine generator bearing seals, isolated phase bus ducts, standby gas treatment system). This is consistent with the results of the NRC/AEOD special study on fire events (Shuaibi and Houghton 1997).

- *Total Loss of Condenser Heat Sink (heading L)*. Approximately 80% of the total loss of condenser heat sink events were not associated with another functional impact event in the reactor trip sequence, meaning they had no impact on plant mitigation systems to remove reactor decay heat. Of the remaining 20%, two-thirds of these events were associated with total loss of feedwater flow (44%) and loss of instrument or control air (22%). About two-thirds of all total loss of condenser heat sink events occurred after the reactor trip initiator.

- *Total Loss of Feedwater Flow (heading P)*. Approximately 80% of the total loss of feedwater events were not associated with another functional impact event in the reactor trip sequence. Of the remaining 20%, two-thirds of these events were associated with the loss of condenser heat sink (mostly caused by the inadvertent isolations of all main steam line isolation valves under functional impact category L1) and the remaining one-third were mostly associated with fire and loss of instrument or control air events. About one-half of all total loss of feedwater flow events occurred after the reactor trip initiator. (Main feedwater isolations, caused by a valid automatic system response immediately after the reactor trip, were not included under the Total Loss of Feedwater Flow category.)

Two tables were used to derive insights that were discussed above. Table 4-12 provides a comparison of the initial plant fault and the subsequent functional impact events for various headings. To illustrate the use of this table, consider the 33 events under the Loss of Offsite Power heading (B). Table 4-12 shows that of the 33 reactor trip sequences that contained a LOSP event, 52% of these have the initial event (i.e., initial plant fault event) as LOSP. There were 16 LOSP events (48% of all LOSP events) that occurred later in the event sequence (i.e., subsequent functional impact). The initial event in these 16 LOSP event sequences are related to events from other initial plant fault categories (exclusive of the LOSP category) and from the General Transient categories. One should be cautioned not to relate the total number of subsequent functional impact events to the number of reactor trips, since an event sequence may have more than one subsequent functional impact event.

Table 4-12. Summary of the events identified in Table D-13 of Appendix D. Each percentage refers to the total number for the row.

Heading or Category	Total Number	Initial Plant Faults	Subsequent Functional Impact
Very Small LOCA/Leak (G1)	4	2(50%)	2 (50%)
Stuck Open: 1 Safety/Relief Valve (G2)	12	10(83%)	2 (17%)
Steam Generator Tube Rupture (F)	3	3 (100%)	0 (0%)
Loss of Offsite Power (B)	33	17 (52%)	16 (48%)
High Energy Line Break (K)	9	9 (100%)	0 (0%)
Total Loss of Feedwater Flow (P)	159	86 (54%)	73 (46%)
Total Loss of Condenser Heat Sink (L)	200	64 (32%)	136 (68%)
Loss of Safety-Related Bus (C)	17	11 (65%)	6 (35%)
Loss of Instrument or Control Air (D)	36	26 (72%)	10 (28%)
Loss of Safety-Related Water (E)	6	0 (0%)	6 (100%)
Fire (H)	39	31 (79%)	8 (21%)
Flood (J)	2	1 (50%)	1 (50%)
Subtotal	520	260 (50%)	260 (50%)
General Transient (Q)	1725	1725	Not applicable

Table D-13 in Appendix D provides a matrix that maps the subsequent functional impact events to the initial plant fault categories. (Instructions on the use of Table D-13 are provided in the Notes section of the table.)

Figure 4-12 provides a graphical display of the information presented in Table D-13 excluding the general transient categories.

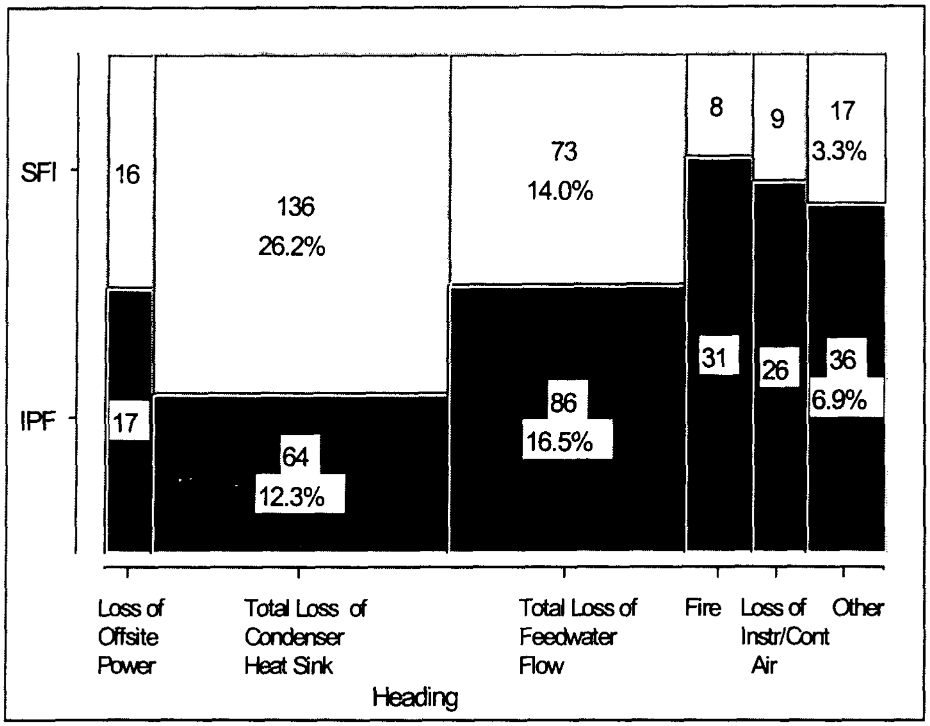

Figure 4-12. Graphical summary of Table D-13. Areas are proportional to counts, and percentages refer to the total count for the graph. The General Transient (Q) events are excluded because they dominate the data and are all initial plant faults. "Other" means all other functional impact categories combined.

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Appendix A

Initial Plant Fault and Functional Impact Category Definitions

Appendix A

Initial Plant Fault and Functional Impact Category Definitions

This appendix presents the definitions and rules used by the analysts to sort the LERs into the initial plant fault and functional impact categories. These definitions are applicable for both initial plant fault and functional impact categories. Examples are included in a given category to further explain the category use.

Unless stated otherwise, the categories listed below are applicable to both boiling water reactors (BWRs) and pressurized water reactors (PWRs).

Events defined by these categories must be associated with a manual or automatic reactor trip. The reactor trip must occur with the plant critical at or above the point of adding heat. The event must occur shortly before or shortly after the reactor trip. The event may contribute to the reactor trip (as a functional impact and/or initial plant fault) or may occur subsequent to the reactor trip (as a functional impact). Engineering judgment was applied to determine whether a failure event that occurs tens of minutes after the reactor trip should be classified under a functional impact category.

A. (Reserved)

B. Loss of Offsite Power

(B1) Loss of Offsite Power(LOSP)

A simultaneous loss of electrical power to all safety-related buses that causes emergency power generators to start and supply power to the safety-related buses.

The offsite power boundary extends from the offsite electrical power grid to the output breaker (inclusive) of the stepdown transformer that feeds the first safety-related bus with an emergency power generator. The plant switchyard and service-type transformers are included within the offsite power boundary.

This category includes the momentary or prolonged degradation of grid voltage that causes all emergency power generators to start (if operable) and load onto its associated safety-related buses (if available).

This category does not include an LOSP event that occurs while the plant is shutdown. Also, it does not include any momentary undervoltage event that results in the automatic start of all emergency power generators, but in which the generators do not tie on to their respective buses due to the short duration of the undervoltage.

C. Loss of Safety-Related Bus

(C1) Loss of Vital Medium Voltage ac Bus (≥600 V, <10 kV)

(C2) Loss of Vital Low Voltage ac Bus (<600 V)

(C3) Loss of Vital dc Bus

Loss of a safety-related electrical bus is any sustained de-energization of a safety-related bus due to the inability to connect to any of the normal or alternative electrical power supplies. The bus must be damaged or its power source unavailable for reasons beyond an open, remotely-operated feeder-breaker from a live power source.

Examples include: supply cable grounds; failed insulators; damaged disconnects; transformer deluge actuations; or improper uses of grounding devices.

This category does not include a momentary de-energization of a bus caused by a slow automatic transfer to an available power source. Losses of all lower voltage buses caused by the loss of the medium voltage feeder bus are not classified under this category unless a lower voltage bus was damaged beyond use. A loss of power to a single component in another system because of a failed or mis-positioned breaker that does not affect the entire bus is not included in this category, but is instead classified as a failure of the system that the single breaker serves. For example: a circuit breaker failure that causes a loss of power to a condensate pump that results in inadequate condensate flow, would be classified as a Partial Loss of Condensate (QP4).

D. Loss of Instrument or Control Air System

(D1) Loss of Instrument or Control Air System

A total or partial loss of an instrument or control air system that leads to a reactor trip or occurs shortly after the reactor trip.

Examples include: ruptured air headers; damaged air compressors with insufficient backup capability; losses of power to air compressors; line fitting failures; improper system line-ups; and undesired operations of pneumatic devices in other systems caused by low air header pressure.

This category does not include a loss of air to a single component in another system because of a blockage or incorrect line-up that does not affect the header pressure, but is instead classified as a failure of the system that the single component serves. For example: a solenoid valve malfunction that causes a loss of plant air to a single feedwater valve and causes the feedwater valve to shut, would be classified as a Partial Loss of Feedwater (QP2). A loss of a redundant component in the air system is not classified as a partial loss of the air system as long as the remaining, similar components provide the required level of performance.

E. Loss of Safety-Related Cooling Water

(E1) Total Loss of Service Water

(E2) Partial Loss of Service Water

A service water system (SWS) can be an open-cycle or a closed-cycle cooling water system. An open-cycle SWS takes suction from the plant's ultimate heat sink (e.g., the ocean, bay, lake, pond or cooling towers), removes heat from safety-related systems and components, and discharges the water back to the ultimate heat sink. A closed-cycle or intermediate SWS removes heat from safety-related equipment and discharges the heat through a heat exchanger to an open-cycle service water system.

These categories include the total or partial loss of a safety-related SWS, or a nonsafety-related SWS that provides cooling to safety-related components during normal plant operations. For the latter case, a standby safety-related service water system automatically starts upon the loss of the nonsafety-related system or during an accident sequence initiation.

Partial Loss of Service Water is a loss of one train of a multiple train system or partial loss of a single train system that impairs the ability of the system to perform its function. Examples include pump cavitation; strainer fouling; and piping rupture.

These categories do not include a loss of a redundant component in a SWS as long as the remaining, similar components provide the required level of performance. For example, a loss of a single SWS pump is not classified as a partial loss of a SWS as long as the remaining operating or standby pumps can provide the required level of performance. A loss of service water to a single component in another system because of a blockage or incorrect line-up that does not affect the cooling to other components serviced by the train is not included under this category, but is instead classified as a failure of the system that the single component serves.

F. Steam Generator Tube Rupture: PWR (SGTR)

A rupture of one or more steam generator tubes that result in a loss of primary coolant to the secondary side of the steam generator at a rate greater than or equal to 100 gpm.

A SGTR can occur as the initial plant fault, such as a tube rupture caused by high cycle fatigue or loose parts, or as a consequence of another initiating event. The latter case would be classified as a functional impact.

This category applies to PWRs only. This category includes excessive leakage caused by the failure of a previous SGTR repair (i.e., leakage past a plug).

G. Loss-of-Coolant Accident (LOCA)/Leak

(G1) Very Small LOCA/Leak

A pipe break or component failure that results in a loss of primary coolant between 10 to 100 gpm, but does not require the automatic or manual actuation of high pressure injection systems.

Examples include: reactor coolant pump (PWR) or recirculating pump (BWR) seal failures; valve packing failures; steam generator tube leaks; and instrument line fitting failures.

Note: Leakage from a pressurizer code safety valve (PWR), main steam line safety valve (BWR), or Automatic Depressurization System relief valve (BWR) are classified under category G2 or G5. Leakage from a pressurizer power operated relief valve is classified under category G4. A steam generator tube rupture (PWR) is classified under category F1. A small primary system leak (less than 10 gpm) that results in a manual reactor trip is classified under category Primary System Leak (QG9). Category QG9 only applies to the initial plant fault group, however.

(G2) Stuck Open: 1 Safety/Relief Valve

A failure of one primary system safety and/or relief valve (SRV) to fully close that results in the loss of primary coolant.

The valves included in this category are pressurizer code safety valves (PWR), main steam line safety valves (BWR), and Automatic Depressurization System relief valves (BWR). The stuck open SRV may or may not cause the automatic or manual actuation of high pressure injection systems.

This category includes a stuck open valve that cannot be subsequently closed upon manual demand (BWRs) or does not subsequently close on its own immediately after the reactor trip (BWRs). The mechanism that opens the valve is not a defining factor. The different mechanisms than can open an SRV are transient-induced opening, manual opening during valve testing (BWRs), and spurious opening.

In BWRs, only a stuck open SRV event initiated by routine surveillance testing of the valve during power operations would be classified as an initial plant fault under this category since no other initial plant fault category applies. All stuck open single SRV events in BWRs and PWRs are classified as a functional impact under this category. An inadvertent open SRV event during power operations which closes on its own after the reactor trip and before the manual or automatic actuation of a high pressure injection system are classified under category QG10, Inadvertent Open/Close: 1 Safety/Relief Valve. Category QG10 only applies to the initial plant fault group, however.

This category does not include a weeping safety valve.

Note: A stuck open pressurizer power-operated relief valve (PWR) is classified under category G4.

(G3) Small Pipe Break LOCA

For a BWR, a pipe in the primary system boundary with a break size less than 0.004 ft^2 (or a 1 inch inside diameter pipe equivalent) for liquid and less than 0.05 ft^2 (or an approximately 4 inch inside diameter pipe equivalent) for steam. For a PWR, a pipe break in the primary system boundary with an inside diameter between ½ to 2 inches.

The above break size ranges were used in the NUREG-1150 analyses of a selected group of plants. The plant-specific range of LOCA sizes should be divided into groups for which plant response, in terms of required system operability, is the same or very similar. The following generic definition was used in NUREG-1150: a small break LOCA is a break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject and provide sufficient core cooling to prevent core damage. However, low capability systems (i.e., 100 to 1500 gpm) are sufficient to make up the inventory completion.

Note: A steam generator tube rupture is classified under category F1. A steam generator tube leak is classified under category G1 or QG9. A stuck open safety or relief valve is classified under category G2, G4, or G5.

(G4) Stuck Open: Pressurizer PORV

A pressurizer power-operated relief valve (PORV) that fails to close.

This category applies to PWRs only.

(G5) Stuck Open: 2 or More Safety/Relief Valves

Two or more primary system safety and/or relief valves that fails to close.

The valves included in this category are pressurizer code safety valves (PWR), main steam line safety valves (BWR), and Automatic Depressurization System relief valves (BWR).

This category does not include a weeping safety valve.

Note: A stuck open pressurizer power-operated relief valve (PWR) is classified under category G4, Stuck Open PORV.

(G6) Medium Pipe Break LOCA

For a BWR, a pipe in the primary system boundary with a break size between 0.004 to 0.1 ft^2 (or an approximately 1 to 5 inches inside diameter pipe equivalent) for liquid and between 0.05 to 0.1 ft^2 (or an approximately 4 to 5 inches inside diameter pipe equivalent) for steam. For a PWR, a pipe break in the primary system boundary with an inside diameter between 2 to 6 inches.

The above break size ranges were used in the NUREG-1150 analyses of a selected group of plants. The plant-specific range of LOCA sizes should be divided into groups for which plant response, in terms of required system operability, is the same or very similar. The following generic definition was used in NUREG-1150: a medium break LOCA is a break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject and provide sufficient core cooling to prevent core damage. However, the loss from the break is such that high capacity systems (i.e., 1500 to 5000 gpm) are needed to makeup the inventory depletion.

(G7) Large Pipe Break LOCA

For a BWR, a pipe in the primary system boundary with a break size greater than 0.1 ft^2 (or an approximately 5 inches inside diameter pipe equivalent) for liquid and steam. For a PWR, a pipe break in the primary system boundary with an inside diameter greater than 6 inches.

The above break size ranges were used in the NUREG-1150 analyses of a selected group of plants. The plant-specific range of LOCA sizes should be divided into groups for which plant response, in terms of required system operability, is the same or very similar. The following generic definition was used in NUREG-1150: a large break LOCA is a break that depressurizes the reactor to the point where the low pressure systems can injection automatically providing sufficient core cooling to prevent core damage.

(G8) Reactor Coolant Pump Seal LOCA: PWR

A catastrophic failure the reactor coolant pump seal assembly that results in a primary coolant leak into the primary containment at a rate greater than 100 gpm.

This category applies to PWRs only.

A reactor coolant pump seal leak with a leak rate less than 100 gpm is classified under category G1 or QG9.

H. Fire

(H1) Fire

Smoke or flames inside the plant or site boundary that results in damage to safety- or nonsafety-related equipment.

Examples include: fires located in the plant switchyard (e.g., transformers, switchgear); burning thermal or electrical insulation; transformer, circuit breaker, and power supply fires; rags ignited by hot relief valve tailpipes; burning lube oil; and offsite brush fires that caused a loss of an electrical power transmission line. Fire-related events classified under this category typically require a response by plant personnel, however, damage to plant equipment determined in the post event evaluation to be caused by a fire that went undetected is also included in this category.

This category does not include a smoldering lightning arrestor caused by a lightning strike; the "smoking" of a set of breaker auxiliary contacts or a small relay coil; a simple fire in a trash can or ash tray; or a fire to an administrative support building (e.g., trailer) that does not effect plant structures, equipment or components required to maintain the plant in a safe condition.

J. Flood

(J1) Flood

A major on-site pipe break other than a high energy line break (as defined by heading K) that causes damage to structures, equipment, or components.

An example of this is leakage from condensate or feedwater lines (as defined under category QK4) as long as the leakage resulted in damage to structures, equipment or components.

This category does not include an activation of a transformer deluge system or natural flooding (e.g., river overflowing, heavy rains, etc.).

K. High Energy Line Break

(K1) Steam Line Break Outside Containment

A break of one inch equivalent diameter or more in a steam line located outside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

Examples include: operation of rupture disks; and breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

(K2) Feedwater Line Break

A break of one inch equivalent diameter or more in a feedwater or condensate line that contains main turbine working fluid at or above atmospheric saturation conditions.

Examples include: breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

(K3) Steam Line Break Inside Containment (PWR)

A break of one inch equivalent diameter or more in a steam line located inside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

This category applies to PWRs only. Examples include: breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

L. Total Loss of Condenser Heat Sink

(L1) Inadvertent Closure of All MSIVs

A complete closure of at least one MSIV in each main steam line.

An example includes the automatic closure of all MSIVs as part of an engineered safety feature actuation.

This category does not include a manual closure of all MSIVs to limit cooldown rate after a reactor trip, as long as the MSIVs are capable of being reopened by operator demand.

(L2) Loss of Condenser Vacuum

A decrease in condenser vacuum that leads to an automatic or manual reactor trip, or manual turbine trip; or a complete loss of condenser vacuum that prevents the condenser from removing decay heat after a reactor trip.

The main condenser boundary includes the condenser air ejectors and condenser vacuum pumps.

Initial plant faults that contribute to a loss of condenser vacuum include: circulating water pump trips (category QL4); traveling screen blockage (category QL4); and condenser leakage (category QL6).

This category does not include the loss of condenser vacuum caused by the loss of offsite power.

In addition, reactor trips that are the indirect result of a low condenser vacuum, such as a loss of feedwater caused by condensate pumps tripping on high condensate temperature because of loss of vacuum, are counted as Loss of Condenser Vacuum.

A loss of condenser vacuum resulting from a manual trip in response to a plant event that had no direct effect on the main condenser vacuum was not included in this category.

Note: In BWRs, a low condenser vacuum signal will generate a reactor trip. In PWRs, a low vacuum signal will cause turbine-driven main feedwater pumps to trip, which will result in a reactor trip on low steam generator level.

(L3) Turbine Bypass Unavailable

The failure of one or more turbine bypass valves (TBVs) to maintain the reactor pressure and temperature at the desired operating condition.

Turbine bypass failures included in this category may result in an automatic or manual reactor trip during an unsuccessful turbine run back; and the sustained use of one or more atmospheric dump valves (PWR) or safety relief valves to the suppression pool (BWR) after the reactor trip.

This category does not include turbine bypass valve closures caused by the loss of offsite power.

M. (Reserved)

N. Interfacing System LOCA

(N1) Interfacing System LOCA

A backflow of high pressure coolant from the primary system through low pressure system piping which results in the breach of the pipe or component.

P. Total Loss of Feedwater Flow

(P1) Total Loss of Feedwater Flow

A complete loss of all main feedwater flow.

Examples include: the trip of the only operating feedwater pump while operating at reduced power; the loss of a startup or an auxiliary feedwater pump normally used during plant startup; the loss of all operating feed pumps due to trips caused by low suction pressure, loss of seal water, or high water level (BWR reactor level or PWR steam generator level); anticipatory reactor trip due to loss of all operating feed pumps; and manual reactor trip in response to feed problems characteristic of a total loss of feedwater flow, but prior to automatic Reactor Protection System signals.

This category also includes the inadvertent isolation or closure of all feedwater control valves prior to the reactor trip, however, a main feedwater isolation caused by valid automatic system response after a reactor trip is not included.

This category does not include the total loss of feedwater caused by the loss of offsite power.

Q. General Transients

Categories under this heading are only used for initial plant faults, not for functional impact classification. The general transient categories result in automatic or manual reactor trips but do not degrade safety system response. Because these categories are only applicable as an initial plant fault, they will only be used when the event they describe is the first event from this entire list of categories to occur.

(QC4) Loss of ac Instrumentation and Control Bus

A sustained de-energization of an ac instrumentation and control bus due to the inability to connect to any of the normal or alternative electrical power supplies.

An event classified in this category is normally associated with damage to the bus itself, or damage to its uninterruptable power supply or supply breaker. The bus had to be damaged or its power source unavailable for reasons other than a remotely-operated feeder-breaker being open. This category includes only those failures of safety- and nonsafety-related ac instrumentation and control buses that lead to an automatic or manual reactor trip.

This category does not include a momentary undervoltage of a bus caused by a slow automatic transfer, or a loss of one output from the bus (e.g., failure of one output breaker), but is instead classified as a loss of the affected system.

(QC5) Loss of Nonsafety-Related Bus

A sustained deenergization of a nonsafety-related bus other than an ac instrumentation and control bus due to the inability to connect to any of the normal or alternative electrical power supplies.

This category is normally associated with damage to the bus itself, or damage to its feeder transformer or supply breaker. The bus had to be damaged or its power source unavailable for reasons other than a remotely-operated feeder-breaker being open. This category includes faults to high (>10kV), medium (>600V, <10KV), and low (>120V, <600V) nonsafety-related ac buses that lead to an automatic or manual reactor trip.

This category does not include a momentary undervoltage of a bus caused by a slow automatic transfer, or a loss of one output from the bus (e.g., failure of one output breaker), but is instead classified as a failure of the affected system.

There were no events classified as failures of a nonsafety-related dc bus.

(QG9) Primary System Leak

A small leak of primary coolant, inside the primary containment, at a rate less than 10 gpm and results in an automatic or manual reactor trip. A crack in one or more steam generator tubes that result in a loss of primary coolant to the secondary side of the steam generator at a rate less than 10 gpm.

A plant shutdown is required for primary leak rates that exceeds technical specification limits. Most shutdown events of this nature do not result in a reactor trip. Manual reactor trips are sometimes initiated to expedite the controlled shutdown to avoid violating technical specification requirements. Automatic reactor trips may occur during the controlled shutdown caused by problems not related to the leak itself. Examples include feedwater flow problems at low power or operator errors.

This category includes those primary leak events that prompt a controlled reactor shutdown and somehow result in an automatic or manual reactor trip.

(QG10) Inadvertent Open/Close: 1 Safety/Relief Valve

One or more primary system safety and/or relief valves that inadvertently opens during normal power operations and then closes on its own prior to the manual or automatic actuation of a high pressure injection system.

The valves included in this category are pressurizer code safety valves (PWR), pressurizer power-operated relief valves (PWR), main steam line safety valves (BWR), and Automatic Depressurization System relief valve (BWR).

This category only applies to the initial plant fault group.

(QK4) Steam or Feed Leakage

A loss of the main turbine working fluid at or above atmospheric saturation conditions from the steam or main feedwater system up to and including a pipe break less than one inch equivalent diameter.

This category includes a small steam or feedwater leak that leads to an automatic or manual reactor trip. Examples include: flange leaks, packing leaks, blown fittings and leaks through other system connections.

Note: Pipe breaks one inch equivalent diameter or more are classified under heading K, High Energy Line Breaks.

(QL4) Loss of Nonsafety-Related Cooling Water

A total or partial loss of a nonsafety-related cooling water system that leads to an automatic or manual reactor trip.

This category includes the loss of nonsafety-related cooling water systems that provide cooling to nonsafety-related balance-of-plant components. Examples of cooling water systems include turbine building service water systems, nonsafety-related mechanical draft cooling towers, and condenser circulating water systems.

This category does not include partial or total loss of a safety-related cooling water system (e.g., service water, component cooling water) that results in a reactor trip due to the loss of nonsafety-related balance-of-plant equipment in which it serves during normal plant operations. This event is classified under the appropriate category under heading E, Loss of Safety-Related Cooling Water.

(QL5) Partial Closure of MSIVs

Any combination of partial or full closure of one or more main steam isolation valves (MSIVs) with at least one main steam line open to pass steam to the main condenser.

This category includes partial MSIV closures that leads to an automatic or manual reactor trip. Examples include: full closure of one MSIV and partial closure of one MSIV.

Note: In BWRs, a reactor trip signal will be generated by the closure of any single MSIV.

(QL6) Condenser Leakage

Faults in the condenser shell, tubing, or connective components that result in leakage (fluid or gas) to or from the condenser.

Example include condenser expansion joint ruptures or leaks, tube leaks/ruptures that require shutdown for conductivity/chemistry although condenser vacuum is normal, and breaks in piping attached to the condenser.

(QP2) Partial Loss of Feedwater Flow

A reduction in main feedwater flow that leads to an automatic or manual reactor trip.

All main feedwater system component malfunctions in conjunction with a steam generator low level alarm were considered to be at least a partial loss of feedwater (if not a total loss of feedwater as defined by category P1). Examples include the partial or full closure of a feedwater regulation valve, and a trip of one feedwater pump.

This category does not include steam generator level shrinkage events due to the injection of colder water (usually during low power operations). In addition, protective trip of a single main feedwater pump due to inadequate suction pressure caused by a partial loss of condensate flow is classified under QP4, Partial Loss of Condensate Flow.

(QP3) Loss of Condensate Flow

A complete loss of condensate flow that leads to an automatic or manual reactor trip.

Examples include: the failure of all condensate pumps or booster pumps; and a malfunction that causes a loss of all condensate flow to the main feedwater system.

Note: An event that results in a total loss of condensate flow as the initial plant fault will result in the total loss of feedwater flow, therefore, this event will also be classified under function impact category P1, Total Loss of Feedwater Flow.

(QP4) Partial Loss of Condensate Flow

A reduction in condensate flow that leads to an automatic or manual reactor trip.

Examples include: the failure of less than all condensate pumps; and a fault in the feed heater or condensate path that causes a reduction of condensate flow.

(QP5) Excessive Feedwater Flow

An inadvertent increase in feedwater flow that leads to an automatic or manual reactor trip.

Excessive feedwater transients as the initiating event can generate various reactor protection system (RPS) trip signals. Examples of events caused by excessive feedwater as the transient initiator include: an automatic reactor trip on high rate of power change caused by moderator temperature effects (BWR); a turbine trip/reactor trip due to high reactor water (BWR) or high steam generator (PWR) level; a reactor water (BWR) or steam generator (PWR) low-level reactor trip that follow the tripping of feedwater pumps caused by high levels due to excessive feedwater flow; and a manual reactor trip in response to improper feedwater regulation valve operation.

This category does not include a transient that results in a swell (increased level) in the reactor vessel or steam generator caused by other than excessive feedwater flow (usually depressurization of steam).

(QR0) RCS High Pressure (RPS Trip)

A transient not classified under any other category that causes reactor pressure to increase to the high pressure RPS trip setpoint.

(QR1) RCS Low Pressure (RPS Trip): PWR

A transient not classified under any other category that causes primary pressure to decrease to the low pressure RPS trip setpoint.

This category only applies to PWRs.

(QR2) Loss of Primary Flow (RPS Trip): PWR

A total loss or reduction in reactor coolant system flow that results in a RPS trip.

This category only applies to PWRs. Examples of events that may cause a reactor coolant pump (RCP) trip include momentary undervoltage transients and RCP faults.

This category does include RCP trips caused by a damaged RCP electrical bus as a result of Fire (H1).

(QR3) Reactivity Control Imbalance

A reactivity anomaly that leads to an automatic or manual reactor trip.

Examples include: high negative or positive neutron flux rate RPS trip (PWR); and automatic and manual reactor trips caused by a dropped control rod, an inadvertent rod withdrawal, a rod control system malfunction, a neutron flux imbalance, or an indication of core instability.

This category does not include a reactivity anomaly that results in a high reactor power RPS trip (classified under category QR4, Core Power Excursion.)

(QR4) Core Power Excursion (RPS Trip)

A reactivity anomaly that causes reactor power exceeding the high reactor power RPS trip setpoint.

Examples of events that typically cause a high reactor power RPS trip include: an inadvertent rod withdrawal that do not cause a high neutron flux rate RPS trip; improper operation of the mode selector switch during startup or shutdown that enables a power level trip lower than the present power level (BWR); a neutron flux spike due to pressure changes or recirculation flow changes (BWR); steam pressure oscillations caused by a turbine control system malfunction (BWR); neutron flux exceeding the flow-biased average power range monitor (APRM) scram setpoint (BWR); and a power increase caused by overfeeding cold feedwater (BWR).

(QR5) Turbine Trip

An inadvertent trip of the main turbine that results in a cessation of steam flow to the turbine, and leads to an automatic or manual reactor trip.

Manual turbine trips performed to cause a reactor trip were never an initiating event. Events of this type were considered as special interest group.

The main turbine as defined in this category includes the main turbine and its auxiliaries; the electrohydraulic control system; turbine throttle valves; main generator and its auxiliaries; and the main generator output breakers.

This category includes: inadvertent closure of all turbine throttle valves; EHC fault; main generator trip due to a switchyard equipment fault (e.g., output breaker, main transformer, switchyard breaker, offsite transmission line); response to electrical grid undervoltage voltage or frequency anomaly; inadequate plant response to an electric load rejection; unplanned manual turbine trip; and a spurious turbine trip.

(QR6) Manual Reactor Trip

A manual initiation of a reactor trip, either purposely or by human error.

This category does not include: the improper operation of the mode selector switch during startup or shutdown that enables a power level trip lower than the present power level (classified under category QR4, Core Power Excursion).

(QR7) Other Reactor Trip (Valid RPS Trip)

All other reactor trips (other than those listed above) that result when an actual plant condition reaches the RPS trip setpoint for that condition.

(QR8) Spurious Reactor Trip

An automatic reactor trip caused by hardware failure or human error in a RPS instrumentation or logic channel, or a reactor trip breaker.

Examples include: incorrect venting of an instrument line during maintenance that causes false signal being sent to the RPS; and any other RPS system fault or human error that generates a reactor trip signal that does not reflect actual plant conditions.

(QR9) Spurious Engineered Safety Feature Actuation

A spurious actuation of the Engineering Safety Features (ESF) system caused by hardware failure or human error in an ESF instrumentation or logic channel that results in a reactor trip.

Appendix B

Category Cross-Reference Tables to Previous Studies

Appendix B

Category Cross-Reference Tables to Previous Studies

Past Reports

This report follows several reports that have been produced independently over the last two decades by the Electric Power Research Institute (EPRI), the NRC, and the INEEL. In this report, both the data and the classification scheme are updated to reflect current probabilistic risk assessment (PRA) practices.

The EPRI collected data for U.S. commercial power plant initiating events as a part of the study of the Anticipated Transient Without Scram (ATWS) topic. EPRI issued a report in 1978, with initiating event categories and associated rates (EPRI NP-801) based on data submitted by the utilities. EPRI NP-2230 (EPRI 1982) was an update to this initial study, and the INEEL published NUREG/ CR-3862 (Mackowiak et al. 1985) report for the NRC in 1985. The latter report used Monthly Operating Reports and updated the EPRI data set to cover all plants from their commercial operation date through the end of 1983.

Previous reports made assumptions about the significance of the event categories as a function of various vendor designs and plant types. This method led to the use of categories that were specific to the plant type and that differentiated between similar events that occurred for different reasons or with slightly different plant parameters or effects, for example, there are six different categories that describe a turbine-generator trip in the EPRI studies.

In the years since these studies began, the methodology of risk assessments has changed. Current PRA usage tends towards more general initiating event categories than those used in the late 1970s, with consequences modeled in the event tree rather than as separate initiators. The methodology changes illustrated by these reports make developing a broad-based list of initiating event categories desirable. Therefore, the categories were themselves modified in this study to develop a list that better supports nuclear power plant risk assessment methods in use in the mid-1990s.

Category Cross Reference

Table B-1 shows the mapping between the categories used in this report and those used in the EPRI report (NP-2230) and NUREG/CR-3862 report. Categories that could not be mapped between studies were combined with the total general transient frequency.

Table B-1. Cross-reference with categories from previous studies.

This NUREG/CR Category	NP-2230 EPRI and NUREG/CR-3862 PWR Category	BWR Category
B1-Loss of Offsite Power	35-Loss of station power	31-Loss of offsite power 32-Loss of auxiliary power
C1-Loss of Vital Medium Voltage ac Bus C2-Loss of Vital Low Voltage ac Bus C3-Loss of Vital dc Bus		
D1-Loss of Instrument and Control Air		
E1-Total Loss of Service Water E2-Partial Loss of Service Water		
F1-Steam Generator Tube Rupture G7-Large Pipe Break LOCA G6-Medium Pipe Break LOCA G3-Small Pipe Break LOCA G8-Reactor Coolant Pump Seal LOCA N1-Interfacing System LOCA		
G1-Very Small LOCA/Leak	4-Leakage from control rods 5-Leakage from primary system 7-Pressurizer leakage 26-Steam generator leakage	
G4-Stuck Open: Pressurizer PORV G2-Stuck Open: 1 Safety/Relief Valve G5-Stuck Open: 2 or More Safety/Relief Valves	29-Sudden opening of steam relief valve(s)	11-Inadvertent opening of a safety/relief valve (Stuck)
H1-Fire		
J1-Flood		
K1-Steam Line Break Outside Containment K3-Steam Line Break Inside Containment K2-Feedwater Line Break	28-Miscellaneous leakage in secondary system	
L1-Inadvertent Closure of All MSIVs	18-Closure of all MSIVs	5-Main steam isolation valve closure
L2-Loss of Condenser Vacuum	25-Loss of condenser vacuum	8-Loss of normal condenser vacuum
L3-Turbine Bypass Unavailable		
P1-Total Loss of Feedwater Flow	16-Total loss of feedwater flow (all loops)	22-Loss of all feedwater flow
General transients.	*General transients*	*General transients*
QC4-Loss of ac Instrumentation and Control Bus QC5-Loss of Safety-Related Bus QG9-Primary System Leak QG10-Inadvertent Open/Close Safety/Relief Valve QK4-Steam of Feed Leakage QL4-Loss of Nonsafety-Related Cooling Water QL5-Partial Closure of MSIVs QL6-Condenser Leakage QP2-Partial Loss of Feedwater Flow QP3-Total Loss of Condensate Flow QP4-Partial Loss of Condensate Flow QP5-Excessive Feedwater Flow QR0-RCS High Pressure (RPS Trip) QR1-RCS Low Pressure (RPS Trip): PWR QR2-Loss of Primary Flow (RPS Trip): PWR QR3-Reactivity Control Imbalance	1-Loss of RCS flow (1 loop) 2-Uncontrolled rod withdrawal 3-CRDM problems and/or rod drop 6-Low pressurizer pressure 8-High pressurizer pressure 9-Inadvertent safety injection signal 10-Containment pressure problems 11-CVCS malfunction-boron dilution 12-Pressure/temp/power imbalance 13-Startup of inactive coolant pump 14-Total loss of RCS flow 15-Loss of reduction in feedwater (1 loop) 17-Full or partial closure of MSIV (1 loop)	1-Electric load rejection 2-Electric load rejection with turbine bypass valve failure 3-Turbine trip 4-Turbine trip with turbine bypass valve failure 6-Inadvertent closure of one MSIV 7-Partial MSIV closure 9-Pressure regulator fails open 10-Pressure regulator fails closed 12-Turbine bypass fails open 13-Turbine bypass or control valve causes increased pressure (closed) 14-Recirculation control failure-increasing flow

Table B-1. (continued).

	NP-2230 EPRI and NUREG/CR-3862	
This NUREG/CR Category	PWR Category	BWR Category
General transients (continued)	*General transients: (continued)*	*General transients. (continued)*
QR4-Core Power Excursion (RPS trip) QR5-Turbine Trip QR6-Manual Reactor Trip QR7-Other Reactor Trip (Valid RPS Trip) QR8-Spurious Reactor Trip QR9-Spurious ESF Actuation	19-Increase in feedwater flow (1 loop) 20-Increase in feedwater flow (all loops) 21-Feedwater flow instability-operator error 22-Feedwater flow instability-misc. mechanical causes 23-Loss of condensate pump (1 loop) 24-Loss of condensate pumps (all loops) 27-Condenser leakage 29-Sudden opening of steam relief valve(s) 30-Loss of circulating water 31-Loss of component cooling 32-Loss of service water system 33-Turbine trip, throttle valve closure, EHC problems 34-Generator trip or generator caused faults 36-Pressurizer spray failure 37-Loss of power to necessary plant systems 38-Spurious trips-cause unknown 39-Automatic trip-no transient condition 40-Manual trip-no transient condition 41-Fire within plant	15-Recirculating control failure-decreasing flow 16-Trip of one recirculation pump 17-Trip of all recirculating pumps 18-Adnormal startup of idle recirculating pump 19-Recirculating pump seizure 20-Feedwater-increasing flow at power 21-Loss of feedwater flow heater 23-Trip of one feedwater pump (or condensate pump) 24-Feedwater flow-low 25-Low feedwater flow during startup or shutdown 26-High feedwater flow during shutdown or startup 29-Inadvertent insertion of rod(s) 27-Rod withdrawal at power 28-High flux due to rod withdrawal at startup 30-Detected fault in reactor protection system 33-Inadvertent startup of HPCI/HPCS 34-Scram due to plant occurrences 35-Spurious trip via instrumentation, RPS fault 36-Manual scram-no out-of-tolerance condition 37-Cause unknown

REFERENCES

EPRI (Electric Power Research Institute), 1982, *ATWS: A Reappraisal, Part 3: Frequency of Anticipated Transients (Interim Report)* EPRI NP-2230,

Mackowiak, D. P. et al., 1985, *Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments*, NUREG/CR-3862,

Appendix C

Licensee Event Report Selection, Categorization, and Quality Management

Appendix C

Licensee Event Report Selection, Categorization, and Quality Management

Licensee Event Report Selection

All Licensee Event Reports (LERs) from 1987 through 1995 that documented unplanned reactor trips from criticality were needed for review. A search of the Sequence Coding and Search System (SCSS) database at Oak Ridge National Laboratory produced 2,024 events for consideration.

SCSS sequence information was appended to each of the selected LER records to supplement the review process.

Each LER abstract, SCSS information, and, when necessary, full text was electronically and manually reviewed to categorize the appropriate events discussed in the LER.

The criteria used to determine which events were included in the study are summarized below.

An event had to meet all of the following criteria:

- Include an unplanned reactor trip (one not on the daily operations schedule)
- Sequence of events starts when reactor is critical and at or above the point of adding heat
- Occur during the calendar years 1987 through 1995 inclusive
- Occur at a U.S. commercial nuclear power plant (excluding Fort St. Vrain and LaCrosse)
- Be reported by Licensee Event Report (LER).

Selection Quality Checks

Several actions were taken to ensure the LER set included all appropriate events without including inappropriate events. Figure C-1 provides a flowchart of the overall LER selection and quality control process.

The set was compared to the Performance Indicator Database and the Plant Operational Events Database at the Idaho National Engineering and Environmental Laboratory (INEEL) for the same time period. Two additional LERs were found and added to the 2,024 retrieved from SCSS.

Following the database comparisons, the records that indicated an initial power level of less than or equal to 5% were manually reviewed to determine if the reactor was critical when the event began. This review found nearly 50 records that did not meet all requirements for inclusion in this study. Those records were removed. Also, during the sorting and review of the LERs, any LERs that indicated a planned reactor shutdown were removed from the study. These checks for improper inclusion identified

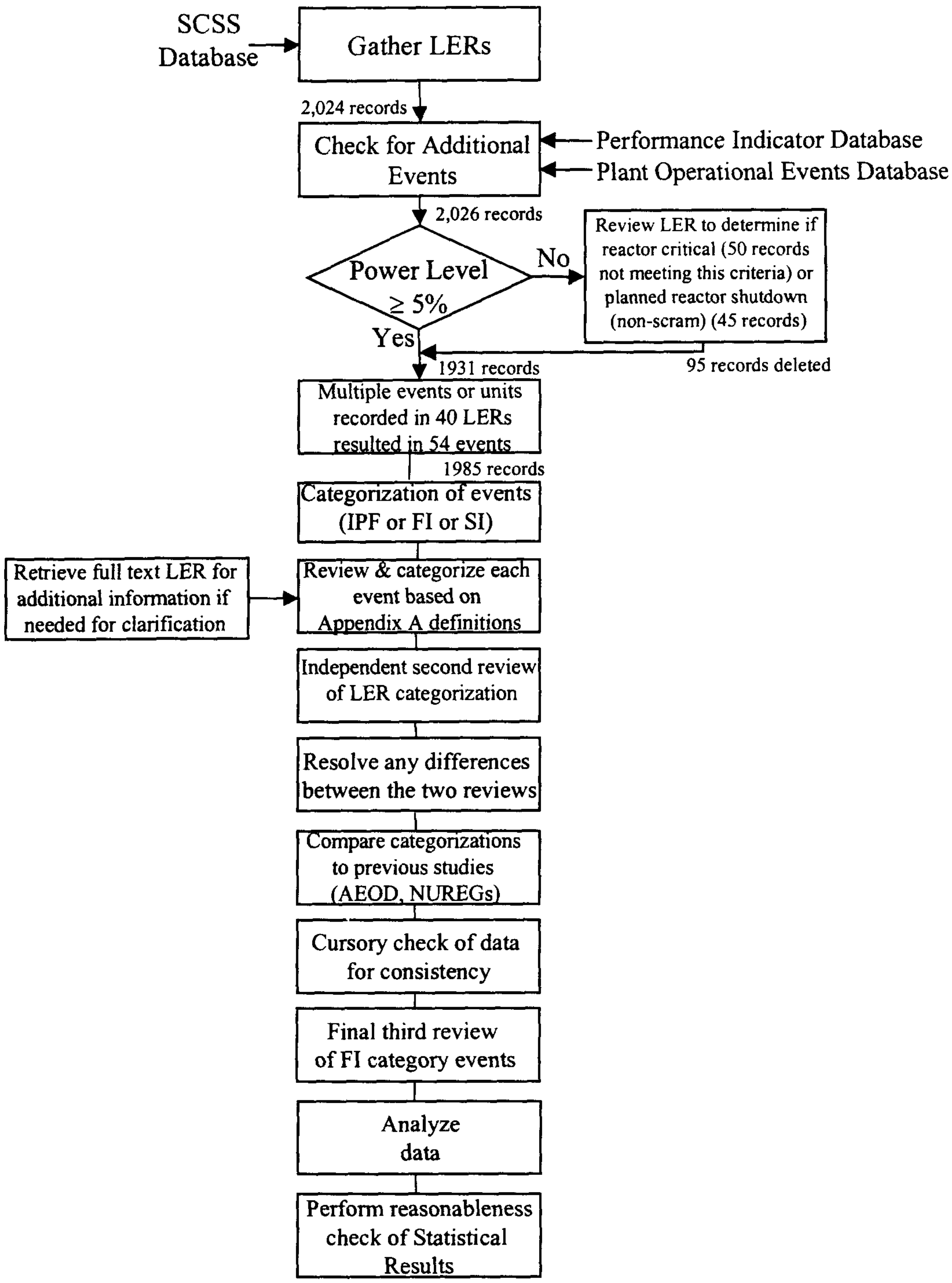

Figure C-1. Flowchart of the LER selection and quality control process.

about 45 events that were either non-reactor trips or were planned. These records were deactivated. Thus, approximately 1,931 records were in the database when categorization was started.

During manual categorization, about 40 LERs were found that included multiple events. These events were split out and given their own records so they could be sorted separately. The final number of events reviewed was 1,985.

Categorization

Analysts with backgrounds in nuclear power plant operation and who are familiar with LER reviews read the LERs and categorized the events into the following groups: initial plant fault (IPF), functional impact (FI), and special interest (SI).

A review form was made for each record that contained the record number, date, abstract, LER number, plant type, and list of SCSS codes; plus a list of the categories for each of the three groups (i.e., the IPF, FI, and SI groups). This coding sheet served as a draft record during the manual review process.

The initial reviewer reviewed the information provided on the review sheet and categorized the event according to the rules and definitions stated in Appendix A. Specifically, the reviewer selected a single initial plant failure from the initial plant fault list, marked all functional impact categories that occurred, and mark all appropriate special interest categories.

If the abstract and SCSS codes were inconclusive, the reviewer retrieved and reviewed a full text version of the LER from the INEEL files. Borderline issues, such as the division between the *Core Power Excursion* and *Reactivity Control/Imbalance* categories, were addressed at scheduled roundtable discussions with all the project reviewers. These discussions promoted consistency in the logic and interpretation used by the analysts to code each LER and more detailed definitions. As expected, a few cases arose that required adding a special rule, e.g., should a smoking lightning arrestor be counted as a fire? (No).

Multiples

Early in the manual review process, two anomalies were noted. First, some LERs described more than one event for the same plant or unit, each on different dates. The coding sheets for those LERs were copied, and the events of each date were counted separately (thus creating additional database entries as mentioned above).

Appropriate Unit

The second anomaly noted was that some LERs were written for a particular unit at a station, but described simultaneous events for another unit as well. Again, the record was duplicated and each event individually coded with the docket number of the plant described. The database was modified to account for the actual unit(s) involved.

Second Check

Upon completion of the first review for each LER, a different analyst conducted a second review. In the second review, the analyst examined the coding sheet completed by the first analyst and either agreed with the classification or proposed a different classification. Differences were resolved through discussions between the two analysts or were brought to a scheduled roundtable meeting.

Categorization Quality Checks

After completing the first and second manual reviews, several quality checks were conducted.

Report Comparisons

A number of reports that covered at least part of the period covered by this report were available for comparison against the categorization made for this study. Table C-1 lists the reports examined and the categories against which they were verified. Some of the reports were not in final form when they were made available for this check and might undergo slight name changes before publication.

Logic Checks

Once all categorization information was entered in the report database, (cursory) logic checks were run to search for possible inconsistencies. Checks included were as follows:

- If the IPF category was also an FI category, then the FI category is marked
- If an FI category is marked, then its appropriate FI heading is also marked
- If an FI heading is marked, then at least one of the categories under it is marked
- The SI electrical disturbance fields are marked only if the loss of offsite power FI category is marked
- No more than one SI electrical disturbance field is marked

Table C-1. Study/Category comparisons.

Study	Categories
Grant, G. M., et al., 1996, *Emergency Diesel Generator Power System Reliability 1987–1993*, INEL-95-0035, February.	Special Interest: EDG start & load
Shah, V., et al., 1998, *Assessment of PWR Primary System Leaks*, Final Draft NUREG/CR-6582, November.	VSLOCA, SGTR, ISLOCA, Inadvertant opening or stuck open SRV or PORV
Shuaibi, M. and J. R. Houghton, 1997, *Special Study: Fire Events—Feedback of US Operating Experience,* U.S. Nuclear Regulatory Commission, AEOD S97-03, June.	Fire
Houghton, J. R., et al., 1998, *Special Study: Operating Experience Feedback from Service Water System Failures and Degradations (1986-1995),* U.S. Nuclear Regulatory Commission, AEOD S98-01, February.	Total Loss of Service Water, Partial Loss of Service Water
Atwood, C. L., et al., 1998, *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980-1996*, NUREG/CR-5496, INEEL/EXT-97-00887, November.	Loss of Offsite Power

- If the manual reactor trip or loss of nonsafety-related bus IPF categories are chosen, then the SI categories by the same name are checked

- The IPF category is not left blank

- If the loss of offsite power category is chosen, then total loss of feedwater flow is not chosen except when loss of feedwater is an IPF.

Third Review for FIs

All records with an FI category marked were given a third review to verify the FI definitions were met and consistently applied.

Main Steam Isolation Valves/ESF

A computer search was performed to find all records with an SCSS code that indicated actuation of some level of engineered safety feature (ESF). Those records were reviewed to determine if the ESF actuation included a closure of all main steam isolation valves and were marked appropriately.

Statistical Analysis Quality Checks

After calculating and formatting the statistical results into tables, the results were transferred from the electronic versions of the tables into a spread sheet and checked against several consistency algorithms listed below:

- 5% < median < 95%

- Medians < means

- Gamma 5% = $\Gamma^{-1}(5\%, \alpha, 1/\beta)$
  where α = shape parameter
  β = scale parameter

- Gamma mean = α / β

- Gamma 95% = $\Gamma^{-1}(95\%, \alpha, 1/\beta)$

- Lognormal 5% = median / (error factor)

- Lognormal mean = $\exp(\mu + \sigma^2/2)$,
  where μ = ln(median)
  σ = ln(error factor)/1.645

- Lognormal 95% = median × (error factor)

Appendix D

Detailed Sorting Results and Estimates of Initial Plant Fault Frequencies

Appendix D

Detailed Sorting Results and Estimates of Initial Plant Fault Frequencies

Appendix D contains detailed tables of the LER selection and categorization. The tables and results presented in this appendix are based on the *all* of the operating experience from 1987 through 1995. No data have been excluded. Appendix G provides detailed results of the operating experience from 1987 through 1995 with the first four months of experience (from the start of commercial operation) removed from the 1987–1995 operating experience for the affected plants. The tables are listed below with a brief description of their contents:

Table D-1. Heading codes and titles. This table cross references the text names of the initial plant fault and functional impact headings with their numeric codes. These are used in later tables.

Table D-2. Category codes and titles. This table cross-references the text names of the initial plant fault and functional impact categories with their numeric codes. These codes are used in later tables.

Table D-3. Summary of initial plant fault and functional impact category counts based on all the operating experience from 1987 through 1995. This table provides a count of the initial plant faults and functional impacts by category.

Table D-4. Summary of initial plant fault and functional impact category counts by reactor type based on all the operating experience from 1987 through 1995. This table provides a count of the initial plant faults and functional impacts by category and by reactor type (PWR and BWR).

Table D-5. Initial plant fault categories with assigned LERs based on all the operating experience from 1987 through 1995. This table lists each initial plant fault category, as ordered by the headings, with the number of events that were assigned to each of them. Following each category is the list of the LERs assigned to that category. LERs with multiple events are listed multiple times and identified with a footnote.

Table D-6. LERs with assigned initial plant fault code based on all the operating experience from 1987 through 1995. This table lists each LER for which a initial plant fault assignment was made, followed by the numeric codes for the initial plant fault categories assigned. LERs with multiple events are listed multiple times and identified with a footnote.

Table D-7. Functional impact categories with assigned LERs based on all the operating experience from 1987 through 1995. This table lists each functional impact category, as ordered by the headings, with the number of events that were assigned to each of them. Following each category is the list of the LERs assigned to that category. LERs with multiple events are listed multiple times and identified with a footnote.

Table D-8. LERs with assigned functional impact code based on all the operating experience from 1987 through 1995. This table lists each LER for which a functional impact assignment was made, followed by the numeric codes for the functional impact categories assigned. LERs with multiple events are listed multiple times and identified with a footnote.

Table D-9. LERs from Table D-8 with multiple functional impact codes (P heading not included). This table lists each functional impact combination from Table D-8 and gives the LERs that were coded with that combination. It does not include Total Loss of Feedwater Flow (P) entries.

Table D-10. Steam generator tube rupture and very small LOCA leak rates based on all the operating experience from 1987 through 1995. This table lists the LERs that were coded steam generator tube rupture (F1) and very small LOCAs/leak (G1), and gives the leak rate and source for each event.

Table D-11. Initial plant fault and functional impact mean frequencies and associated uncertainty distributions based on all the operating experience from 1987 through 1995. Tabulation of the initial plant fault and functional impacts mean frequencies in units of per critical year.

Table D-12. Frequency estimates of initial plant fault categories: mean, percentiles, and trends based on all the operating experience from 1987 through 1995. This table provides summary count of initial plant faults and mean frequencies and associated uncertainties.

Table D-13. Summary count of initial plant fault events correlated to the subsequent functional impact events based on all the operating experience from 1987 through 1995. This table provides an accounting of the initial plant faults and the subsequent functional impacts that occurred after the reactor trip initiator.

Table D-14. Summary of manual reactor trips that occurred subsequent to the initial plant fault based on all the operating experience from 1987 through 1995. This table provides an accounting of the manual reactor trips identified in the operating experience from 1987 through 1995 correlated to the initial plant faults. A total of 406 manual reactor trips were identified of which 103 were the initial plant fault. The remaining 303 manual reactor trips are classified according to initial plant fault that led to the manual reactor trip and the reactor type (i.e., BWR and PWR).

Table D-15. Summary of dual reactor trips based on all the operating experience from 1987 through 1995. This table provides an accounting of the dual unit reactor trips that occurred. One LER reports the reactor trips of two units.

Table D-1. Heading codes and titles.

Code	Name	Code	Name
A	(Reserved)	H	Fire
B	Loss of Offsite Power	J	Flood
C	Loss of Safety-Related Bus	K	High Energy Line Break
D	Loss of Instrument or Control Air	L	Total Loss of Condenser Heat Sink
E	Loss of Safety-Related Cooling Water	M	(Reserved)
F	Steam Generator Tube Rupture	N	Interfacing System LOCA
G	Loss of Coolant Accident (LOCA)/Leak	P	Total Loss of Feedwater Flow
		Q	General Transients

Table D-2. Category codes and titles.

Code	Name	Code	Name
A1	(Reserved)	J1	Flood
B1	Loss of Offsite Power	K1	Steam Line Break Outside Containment
C1	Loss of Vital Medium Voltage ac Bus	K2	Feedwater Line Break
C2	Loss of Vital Low Voltage ac Bus	K3	Steam Line Break Inside Containment
C3	Loss of Vital dc Bus	L1	Inadvertent Closure of All MSIVs
D1	Loss of Instrument or Control Air System	L2	Loss of Condenser Vacuum
E1	Total Loss of Service Water	L3	Turbine Bypass Unavailable
E2	Partial Loss of Service Water	M1	(Reserved)
F1	Steam Generator Tube Rupture	N1	Interfacing System LOCA
G1	Very Small LOCA/Leak	P1	Total Loss of Feedwater Flow
G2	Stuck Open: 1 Safety/Relief Valve	QC4	Loss of ac Instrumentation and Control Bus
G3	Small Pipe Break LOCA	QC5	Loss of Nonsafety-Related Bus
G4	Stuck Open: Pressurizer PORV	QG9	Primary System Leak
G5	Stuck Open: 2 or more Safety/Relief Valves	QG10	Inadvertent Open/Close: 1 Safety /Relief Valve
G6	Medium Pipe Break LOCA	QK4	Steam or Feed Leakage
G7	Large Pipe Break LOCA	QL4	Loss of Nonsafety-Related Cooling Water
G8	Reactor Coolant Pump Seal LOCA: PWR	QL5	Partial Closure of MSIVs
H1	Fire	QL6	Condenser Leakage

Table D-2. (continued).

Code	Name	Code	Name
QP2	Partial Loss of Feedwater Flow	QR3	Reactivity Control Imbalance
QP3	Total Loss of Condensate Flow	QR4	Core Power Excursion (RPS Trip)
QP4	Partial Loss of Condensate Flow	QR5	Turbine Trip
QP5	Excessive Feedwater Flow	QR6	Manual Reactor Trip
QR0	RCS High Pressure (RPS Trip)	QR7	Other Reactor Trip (Valid RPS Trip)
QR1	RCS Low Pressure (RPS Trip): PWR	QR8	Spurious Reactor Trip
QR2	Loss of Primary Flow (RPS Trip): PWR	QR9	Spurious Engineered Safety Feature Actuation

Table D-3. Summary of initial plant fault (IPF) and functional impact (FI) category counts based on all the operating experience from 1987 through 1995.

IPF Total	FI Total	Category	IPF Total	FI Total	Category
17	33	B1—Loss of Offsite Power	7	7	K1—Steam Line Break Outside Containment
10	13	C1—Loss of Vital Medium Voltage ac Bus	2	2	K2—Feedwater Line Break
1	3	C2—Loss of Vital Low Voltage ac Bus	0	0	K3—Steam Line Break Inside Containment
0	1	C3—Loss of Vital dc Bus	21	109	L1—Inadvertent Closure of All MSIVs
26	36	D1—Loss of Instrument or Control Air System	40	81	L2—Loss Of Condenser Vacuum
0	0	E1—Total Loss of Service Water	3	10	L3—Turbine Bypass Unavailable
0	6	E2—Partial Loss of Service Water	0	0	N1—Interfacing System LOCA
3	3	F1—Steam Generator Tube Rupture	86	159	P1—Total Loss of Feedwater Flow
2	4	G1—Very Small LOCA/Leak	31	—[a]	QC4—Loss of ac Instrumentation and Control Bus
10	12	G2— Stuck Open: 1 Safety/Relief Valve	25	—[a]	QC5—Loss of Nonsafety-Related Bus
0	0	G3—Small Pipe Break LOCA	6	—[a]	QG9—Primary System Leak
0	0	G4—Stuck Open: Pressurizer PORV	2	—[a]	QG10—Inadvertent Open/Close: 1 Safety/Relief Valve
0	0	G5—Stuck Open: 2 or more Safety/Relief Valves	3	—[a]	QK4—Steam or Feed Leakage
0	0	G6—Medium Pipe Break LOCA	50	—[a]	QL4—Loss of Nonsafety-Related Cooling Water
0	0	G7—Large Pipe Break LOCA	47	—[a]	QL5—Partial Closure of MSIVs
0	0	G8—Reactor Coolant Pump Seal LOCA: PWR	9	—[a]	QL6—Condenser Leakage
31	39	H1—Fire	285	—[a]	QP2—Partial Loss of Feedwater Flow
1	2	J1—Flood	19	—[a]	QP3—Total Loss of Condensate Flow
			35	—[a]	QP4—Partial Loss of Condensate Flow

Table D-3. (continued).

IPF Total	FI Total	Category	IPF Total	FI Total	Category
110	—[a]	QP5—Excessive Feedwater Flow	457	—[a]	QR5—Turbine Trip
13	—[a]	QR0—RCS High Pressure (RPS Trip): PWR	103	—[a]	QR6—Manual Reactor Trip
8	—[a]	QR1—RCS Low Pressure (RPS Trip): PWR	84	—[a]	QR7—Other Reactor Trip (Valid RPS Trip)
40	—[a]	QR2—Loss of Primary Flow (RPS Trip)	217	—[a]	QR8—Spurious Reactor Trip
94	—[a]	QR3—Reactivity Control Imbalance	36	—[a]	QR9—Spurious Engineered Safety Feature Actuation
51	—[a]	QR4—Core Power Excursion (RPS Trip)	1,985	520	

a. Initial plant fault only.

Table D-4. Summary of initial plant fault (IPF) and functional impact (FI) category counts by plant type based on all the operating experience from 1987 through 1995.

BWR IPF	BWR FI	Category	PWR IPF	PWR FI
4	9	B1—Loss of Offsite Power	13	24
7	7	C1—Loss of Vital Medium Voltage ac Bus	3	6
1	2	C2—Loss of Vital Low Voltage ac Bus	0	1
0	1	C3—Loss of Vital dc Bus	0	0
13	21	D1— Loss of Instrument or Control Air System	13	15
0	0	E1—Total Loss of Service Water	0	0
0	3	E2—Partial Loss of Service Water	0	3
—[b]	—[b]	F1—Steam Generator Tube Rupture	3	3
0	0	G1—Very Small LOCA/Leak	2	4
10	10	G2— Stuck Open: 1 Safety/Relief Valve	0	2
0	0	G3—Small Pipe Break LOCA	0	0
—[b]	—[b]	G4— Stuck Open: Pressurizer PORV	0	0
0	0	G5— Stuck Open: 2 or more Safety/Relief Valves	0	0
0	0	G6—Medium Pipe Break LOCA	0	0
0	0	G7—Large Pipe Break LOCA	0	0
—[b]	—[b]	G8—Reactor Coolant Pump Seal LOCA: PWR	0	0
10	11	H1—Fire	21	28
1	2	J1—Flood	0	0
2	2	K1—Steam Line Break Outside Containment	5	5
0	0	K2—Feedwater Line Break	2	2
0	0	K3—Steam Line Break Inside Containment	0	0
16	74	L1—Inadvertent Closure of All MSIVs	5	35
27	46	L2—Loss of Condenser Vacuum	13	35
2	4	L3—Turbine Bypass Unavailable	1	6
0	0	N1—Interfacing System LOCA	0	0
24	52	P1—Total Loss of Feedwater Flow	62	107
12	—[a]	QC4—Loss of ac Instrumentation and Control Bus	19	—[a]
5	—[a]	QC5—Loss of Nonsafety-Related Bus	20	—[a]

Table D-4. (continued).

BWR IPF	BWR FI	Category	PWR IPF	PWR FI
4	—[a]	QG9—Primary System Leak	2	—[a]
		QG10—Inadvertent Open/Close: 1 Safety/Relief Valve	2	—[a]
1	—[a]	QK4—Steam or Feed Leakage	2	—[a]
16	—[a]	QL4—Loss of Nonsafety-Related Cooling Water	34	—[a]
11	—[a]	QL5—Partial Closure of MSIVs	36	—[a]
5	—[a]	QL6—Condenser Leakage	4	—[a]
45	—[a]	QP2—Partial Loss of Feedwater Flow	240	—[a]
5	—[a]	QP3—Total Loss of Condensate Flow	14	—[a]
13	—[a]	QP4—Partial Loss of Condensate Flow	22	—[a]
49	—[a]	QP5—Excessive Feedwater	61	—[a]
9	—[a]	QR0—RCS High Pressure (RPS Trip)	4	—[a]
—[b]	—[b]	QR1—RCS Low Pressure (RPS Trip): PWR	8	—[a]
—[b]	—[b]	QR2—Loss of Primary Flow (RPS Trip): PWR	40	—[a]
6	—[a]	QR3—Reactivity Control Imbalance	88	—[a]
39	—[a]	QR4—Core Power Excursion (RPS Trip)	12	—[a]
173	—[a]	QR5—Turbine Trip	284	—[a]
55	—[a]	QR6—Manual Reactor Trip	48	—[a]
16	—[a]	QR7—Other Reactor Trip (Valid RPS Trip)	68	—[a]
63	—[a]	QR8—Spurious Reactor Trip	154	—[a]
14	—[a]	QR9—Spurious Engineered Safety Feature Actuation	22	—[a]
658	244	Totals	1,327	276

a Initial plant fault only

b. Applicabale only to PWRs

Table D-5. Initial plant fault categories with assigned LERs based on all the operating experience from 1987 through 1995.

Loss of Offsite Power—B1
17
029/91-002-0
249/89-001-1
255/87-024-0
261/92-017-0
270/92-004-0
271/91-009-1
293/93-022-0
302/89-023-0
302/92-001-0
309/88-006-0
317/87-012-1[c]
317/87-012-1[c]
324/89-009-1
327/92-027-0[c]
327/92-027-0[c]
369/91-001-0
456/88-022-0

Loss of Vital Medium Voltage ac Bus —C1
10
219/90-005-0
263/91-019-0
265/87-013-0
277/92-010-0
280/89-044-0
293/93-004-0
318/94-001-1
336/88-011-1
353/94-010-1
388/92-001-0

Loss of Vital Low Voltage ac Bus—C2
1
374/94-004-0

Loss of Vital dc Bus—C3
None

Loss of Instrument or Control Air System —D1
26
237/94-005-2
245/87-038-0
247/89-002-0
249/93-004-0
265/88-026-0
280/90-006-0
285/90-026-1
317/87-003-0
327/92-018-0
331/90-015-0
346/87-015-0
354/89-017-0
369/87-021-0
374/92-016-1
387/89-001-0
400/87-041-0
410/88-001-0
410/90-009-0
416/88-013-0
416/90-028-0
424/88-043-0
456/88-025-0[c]
456/88-025-0[c]
457/88-019-0
461/87-017-0
530/92-001-0
369/89-004-0
529/93-001-2

Total Loss of Service Water—E1
None

Partial Loss of Service Water—E2
None

Steam Generator Tube Rupture—F1
3
338/87-017-1

Very Small LOCA/Leak—G1
2
368/88-011-0
287/91-008-0

Stuck Open: 1 Safety/Relief Valve—G2
10
237/90-006-1
254/89-004-0
265/91-012-0
265/93-006-0
324/90-004-3
352/95-008-0
354/87-047-0
373/93-002-0
397/92-033-0[b]
397/92-033-0[b]

Small Pipe Break LOCA—G3
None

Stuck Open: Pressurizer PORV—G4
None

Stuck Open: 2 or more Safety/Relief Valves—G5
None

Medium Pipe Break LOCA—G6
None

Large Pipe Break LOCA—G7
None

Reactor Coolant Pump Seal LOCA—G8
None

Fire—H1
31
219/92-005-0
237/90-002-2
269/89-002-0
275/90-005-0
295/94-005-0
295/94-010-0
298/89-026-0
304/90-011-1
305/87-009-0
305/88-001-0
305/92-017-0
311/91-017-0
316/91-006-0
317/92-008-0
321/90-012-0
321/91-001-0
323/88-008-0
323/89-010-0
334/94-005-0[c]
335/94-007-0
341/89-038-1
341/91-015-0
354/90-003-0
373/87-014-0
382/90-012-0
389/92-006-0
400/89-017-1
412/87-030-2
461/88-028-0
498/89-005-0
528/88-010-1

Flood—J1
1
440/91-027-0

Steam Line Break Outside Containment—K1
7
255/87-016-0
328/93-001-0
331/91-001-0
336/95-032-0
368/89-006-0
440/87-027-1
455/90-010-1

Feedwater Line Break—K2
2
336/91-012-1
423/90-030-2

Steam Line Break Inside Containment—K3
None

Inadvertent Closure of All MSIVs—L1
21
220/90-026-0
245/87-007-0
249/87-016-0
249/89-006-0
260/94-005-0
293/89-011-0
293/92-018-0
298/87-005-0
313/94-002-0
324/90-009-0
362/90-002-1
366/90-001-1
373/94-015-0
388/87-006-0
397/88-003-0
397/93-027-0
423/87-027-0
440/87-042-0
454/87-019-2
458/93-017-0
483/90-007-0

Loss of Condenser Vacuum—L2
40
155/88-008-0
219/89-011-0
219/90-008-0
245/89-015-0
249/87-010-0
260/95-007-0
263/94-004-0
275/92-004-0
277/91-022-1
278/90-008-0
278/92-005-0
278/93-004-0
278/95-001-0
293/89-023-0
315/89-001-0
315/95-003-0
316/87-004-0
316/92-007-0
316/94-005-0
321/93-001-0
324/88-001-7
346/89-005-0
353/90-012-0
354/87-037-0
362/93-004-0
364/91-004-0
364/94-001-0
366/95-003-0[b]
373/92-003-0
400/89-001-2
410/87-064-0
410/87-081-0
410/89-035-0
410/94-007-0
413/94-001-0
416/87-009-2
461/87-050-0
461/91-006-0
461/93-007-0

Table D-5. (continued).

528/95-012-0

Turbine Bypass Unavailable—L3

3

325/90-017-0
341/87-008-0
455/87-011-1

Interfacing System LOCA—N1

None

Total Loss of Feedwater Flow—P1

86

029/90-011-0
155/94-010-1
237/89-012-0
255/90-001-1
255/95-003-0
263/88-007-0
269/88-009-0
269/94-002-0
270/94-005-0
272/93-002-0
275/90-002-0
275/95-015-0
278/87-002-0
278/93-002-0
278/94-005-0
281/93-006-0
286/88-001-0
287/94-003-0
298/87-003-0
298/87-009-0
302/88-024-0
309/91-006-0
311/90-029-1
311/93-002-0
312/88-019-0
313/87-004-0
313/89-048-0
313/91-005-0
313/95-004-0
318/91-005-0
318/92-005-0
321/88-013-0
327/90-012-0
327/94-008-0
333/87-008-0
335/89-003-0
336/91-004-0
341/87-017-0
341/88-004-0
346/87-001-0
348/87-003-0
348/87-010-0
354/88-027-0
361/87-031-1
364/89-007-0
364/89-010-0
364/90-001-0
364/91-002-0
364/95-005-0[b]
364/95-005-0[b]
366/89-005-0
366/92-009-0
369/90-001-0
382/88-016-0
389/87-003-0
397/87-002-0
400/87-008-0
400/87-013-0
400/87-017-0
400/87-018-0
400/87-037-0
410/88-014-0
410/91-023-0
413/91-019-0
414/87-007-1
414/87-025-0
414/88-031-0
414/89-002-0
414/95-005-0
416/90-029-0
416/91-004-0
423/87-021-0
440/87-012-0
440/87-037-0
440/87-072-0
440/90-001-0
443/93-001-0
445/90-013-0
445/90-030-0
445/92-014-0
445/92-019-0
445/95-004-1
454/90-014-0
455/88-008-0
499/89-020-0
529/87-008-0

Loss of ac Instrument and Control Bus—QC4

31

255/92-038-1
263/87-006-0
266/91-005-0
266/91-008-0
281/88-004-0
285/92-023-0
287/92-003-0
287/94-002-0
295/91-016-0
321/87-011-1
327/89-005-1
327/90-021-2
327/95-008-0
335/87-010-0
335/87-017-0
341/90-003-2
352/87-046-0
354/87-034-0
366/87-006-1
366/87-009-1
373/95-014-0
387/91-008-0
395/87-027-0
397/87-020-0
412/87-018-1
440/88-012-0
440/95-008-0
455/87-007-1
456/95-004-0
482/92-002-0
483/91-006-0

Loss of Nonsafety-Related Bus—QC5

25

272/90-029-0
272/94-011-0
286/87-002-0
311/95-004-1
317/93-003-0[c]
318/88-002-2
338/95-001-0
352/93-011-0
362/87-011-2
362/89-001-3
362/91-001-0
397/87-022-0
400/92-009-0
410/91-017-1
412/88-002-1
416/89-019-0
424/90-016-0
424/90-023-0
443/91-002-0
443/95-002-0
445/95-003-1
458/87-012-1
528/95-014-0
529/92-002-1
529/93-004-0

Primary System Leak—QG9

6

333/95-010
352/95-006
354/87-014
414/87-010
458/87-002
528/87-018

Inadvertent Open/Close: 1 Safety/Relief Valve—QG10

2

395/89-011-1
395/89-015-2

Steam or Feed Leakage—QK4

3

272/90-030-0
318/92-001-0
341/93-013-0

Loss of Non-Safety-Related Cooling Water—QL4

50

244/95-008-0
245/90-016-1
249/93-014-0
249/95-019-0
263/87-014-0
263/94-003-0
272/93-011-0
275/95-017-0
286/91-003-0
289/92-002-0
302/91-003-1
305/92-020-1
311/89-013-1
318/95-005-0
323/94-012-0
323/95-002-0
325/95-011-0
328/95-007-0
331/92-018-1
333/90-023-0
333/93-004-0
335/93-007-0[b]
335/93-007-0[b]
335/93-007-0[b]
336/93-012-1
338/88-002-0
341/93-004-0
352/94-001-0
354/88-012-1
354/94-012-0
362/87-017-0
364/92-010-0
366/95-003-0[b]
368/90-020-0
382/87-020-0
389/93-008-0
400/87-021-0
400/89-004-0
423/87-001-0
423/88-014-0
423/88-024-0
423/89-008-0
423/90-011-0
423/90-013-1
423/90-014-0
423/92-011-0
440/93-010-0
443/92-025-0
461/88-019-0
483/95-005-0

Partial Closure of MSIVs—QL5

47

029/89-005-0
219/87-029-0
245/92-028-0

a Reserved.

b. One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units.

Table D-5. (continued).

249/90-005-0	Condenser	260/92-004-1	311/89-005-0	335/88-008-0	370/88-008-0
250/89-020-1	Leakage—QL6	261/87-020-0 [b]	311/90-036-0	335/91-003-0	370/89-002-0
261/89-005-0	9	261/90-007-0	311/92-009-0	335/91-005-0	370/89-003-1
261/95-004-0	304/90-010-0	263/87-004-0	311/93-005-0	335/94-001-0	370/91-010-1
265/93-001-0	318/92-003-0	263/89-009-0	311/94-008-0	335/95-010-0	370/92-007-0
266/92-008-0	397/91-035-0	269/92-004-1	312/89-004-0	336/87-002-0	370/92-009-0
269/93-010-1	400/92-007-0	269/92-015-0	313/89-037-0	336/87-009-2	370/93-001-0
272/89-027-0	400/92-010-0	270/87-002-0	313/89-038-0	336/87-011-0	370/93-002-0
277/89-023-0	416/89-012-0	270/94-002-0	315/87-021-0	336/87-012-0	373/87-022-0
311/89-008-0	416/95-008-0	272/89-007-0	316/90-012-0	336/90-006-0	373/87-038-0
311/94-011-0	440/87-035-0	272/90-012-0	316/93-008-0	336/93-004-2 [b]	373/91-006-0
316/94-001-0	461/89-029-0	272/93-013-0	316/95-002-0	336/93-004-2 [b]	373/93-015-0
318/92-006-0		272/94-003-0	317/91-003-0	338/88-020-0	373/94-010-1
321/88-009-0	Partial Loss of	275/87-023-1	317/95-002-0	338/89-005-0	373/95-016-0
323/87-003-1	Feedwater	275/88-025-1	317/95-006-0	338/90-001-0	382/87-016-0
328/91-006-0	Flow—QP2	275/91-002-1	318/87-002-1	339/90-010-0	382/89-013-0
331/89-008-0	285	275/92-002-0	318/88-004-0	339/91-009-0	382/89-024-1
331/90-016-0	029/88-003-0	280/94-006-0	321/87-013-0	339/92-001-0	389/87-002-0
331/91-005-1	206/89-019-0	280/95-001-1	321/90-013-0	339/94-003-1	389/92-004-0
338/91-017-1	206/90-011-0	281/89-010-0	321/92-009-0	344/87-001-0	389/95-002-0
339/92-007-0	213/95-016-0	281/90-003-0	321/93-016-0	344/90-034-0	395/87-015-0
341/89-036-0	219/92-009-0	281/90-004-0	324/87-004-0	344/92-020-1	395/92-004-1
348/87-002-0	220/87-028-1	281/93-002-0	325/92-003-0	346/87-006-0	397/89-031-0
368/90-019-0	220/91-014-0	281/93-003-0	327/95-017-0	346/89-003-1	397/93-002-1
369/95-005-0	237/87-023-1	286/87-001-0	328/88-027-1	346/93-003-0	397/93-007-1
382/87-028-0	237/87-024-0	286/87-004-0	328/89-005-0 [b]	346/93-005-0	400/87-042-0
395/88-006-0	244/90-007-0	286/88-002-0	328/89-005-0 [b]	348/90-005-0	400/88-007-0
413/89-008-1	244/90-010-1	286/91-005-0	328/89-005-0 [b]	348/92-008-0	400/89-003-0
414/88-025-0	244/92-003-0	286/92-015-1	331/90-019-0	354/91-005-0	400/89-005-0
414/93-003-1	244/93-006-0	286/95-012-0	331/95-005-0	354/91-008-0	400/89-006-0
414/94-006-0	244/94-007-0	287/90-002-0	333/87-017-0	354/94-007-0	410/88-025-0
414/95-001-0	245/88-003-0	289/87-004-1	333/90-027-0	361/87-001-0	412/87-014-0
423/88-023-0	245/90-015-0	289/87-006-0	333/93-009-3	361/87-004-1	412/87-034-0
423/94-011-0	247/88-006-0	295/88-013-0	333/95-013-1	361/92-008-0	413/87-013-0
424/87-027-0	247/88-019-0	298/93-038-0	334/88-009-0	364/89-013-0	413/87-015-0
424/89-018-0	247/92-002-0	302/88-006-2	334/89-001-0	364/92-007-1	413/87-026-0
424/90-001-0	250/94-006-0	302/91-014-0	334/89-002-0	366/88-008-0	413/89-017-0
425/90-007-0	251/88-010-0	302/91-017-0	334/90-007-0	369/88-007-1	413/89-022-0
425/90-008-0	255/89-020-0	302/92-027-0	334/91-023-1	369/92-008-0	414/87-002-1
425/92-002-0	255/90-002-0	304/91-002-1	335/87-002-0	369/95-001-1	414/87-019-0
443/93-009-1	255/91-015-0	309/94-008-0	335/87-013-1	370/87-019-0	414/87-027-1
443/94-001-1	260/91-017-0	311/89-003-0	335/88-003-0	370/88-001-0	414/88-019-1

a. Reserved

b One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units

Table D-5. (continued).

414/88-020-1	455/87-018-0	529/92-001-1	277/93-004-0	275/87-002-0	364/93-004-0
414/88-021-1	455/88-001-1	529/95-005-0	309/88-001-0	275/88-021-0	366/91-005-0
414/88-023-0	455/88-004-1	530/93-001-0	309/91-010-0	275/91-007-0	366/95-001-0
414/89-001-0	455/91-005-0	530/94-005-0	309/91-012-0	277/89-012-1	368/89-024-0
414/90-013-0	455/92-003-1		327/92-012-0	278/95-003-0	368/95-002-0
414/92-006-0	456/90-021-0	Total Loss of	328/88-023-1	280/89-026-0	370/92-004-0
414/94-003-0	456/91-012-0	Condensate	335/87-016-0	281/91-011-0	373/87-032-0
416/95-011-0	457/88-013-0	Flow—QP3	335/88-004-0	287/92-001-0	373/94-011-2
423/87-008-0	457/88-016-0	19	341/92-012-0	293/89-015-0	382/87-008-0
423/87-020-0	457/90-010-0	270/89-004-0	344/92-027-0	293/90-013-0	382/91-013-1
423/87-025-0	457/92-002-0	287/91-007-0	344/92-028-0	295/88-005-0	382/93-002-0
423/87-034-0	457/92-006-0	305/88-004-0	354/88-013-1	295/90-004-0	387/87-013-0
423/90-005-0	457/93-007-0	321/93-013-0	366/87-008-0	298/87-002-0	387/89-002-1
424/87-012-0	457/94-005-0	325/95-018-0	366/88-020-0	304/91-004-0	388/90-005-0
424/87-013-0	461/88-017-1	366/88-017-0	387/89-005-0	309/87-006-1	389/89-005-0
424/87-029-0	461/89-022-0[b]	370/87-003-0	400/87-024-0	311/87-011-1	397/91-032-0
424/87-034-0	461/92-002-1	370/92-006-0	400/87-025-0	311/88-017-0	410/87-031-1
424/87-035-0	461/92-010-0	389/90-001-0	400/87-031-0	311/88-024-0	410/87-058-0
424/88-013-0	482/87-027-0	400/87-005-0	400/88-032-0	311/92-007-0	410/88-017-0
424/88-044-0	482/87-030-0	400/87-019-0	410/92-017-0	315/87-008-0	410/88-019-0
424/89-005-0	483/88-001-0	400/87-063-0	412/87-035-0	316/87-008-0	412/87-023-0
424/89-012-0	483/88-006-0	400/88-028-0	413/91-015-0	317/87-011-0	412/87-029-0
424/89-016-1[b]	483/88-010-0	414/87-021-2	424/87-011-0	317/88-009-0	412/89-003-0
424/89-016-1[b]	483/90-017-0	416/88-006-0	456/87-060-0	317/95-005-1	413/93-008-0
424/90-011-0	483/92-007-0	440/87-030-0		318/95-003-0	414/88-012-0
424/92-008-0	498/90-005-0	445/95-004-1	Excessive	324/88-018-0	416/89-006-0
425/89-021-1	498/90-006-0	457/88-020-0	Feedwater	324/90-008-2	416/90-011-0
425/89-029-0	498/90-023-0	457/88-029-1	Flow—QP5	324/90-016-0	423/88-009-0
425/91-005-0	498/94-009-1		110	324/91-001-1	424/87-014-0
425/93-004-0	498/94-015-1	Partial Loss of	155/95-007-0	325/88-023-0	424/87-033-0
440/92-017-0	498/95-001-0	Condensate	213/90-018-0	327/88-047-1	425/90-009-0
440/95-007-0	499/89-019-0	Flow—QP4	219/87-011-1	327/89-035-0	440/87-064-0
443/90-025-0	499/89-021-0	35	219/94-003-0	328/88-028-0[b]	440/88-001-1
445/90-017-0	499/90-004-0	213/90-020-0	237/87-016-0	331/90-002-0	443/92-017-0
445/90-023-0	499/91-001-0	219/91-005-0	245/89-021-0	333/90-009-0	445/90-025-0
445/92-022-0	499/92-003-0	244/90-019-0	247/92-007-0	334/88-008-0	445/90-027-0
445/95-007-0	499/92-010-0	249/87-011-0	249/87-012-0	334/91-022-0	454/89-002-0
446/93-003-0	499/93-004-0	249/92-021-1	250/90-011-1	341/87-056-0	455/87-019-1
446/93-011-0	528/87-014-0	249/94-018-0	251/93-003-0	341/93-007-0	457/88-014-1
446/95-004-0	528/88-024-0	251/90-008-0	254/91-025-0	344/87-024-0	457/88-028-0
454/87-018-1	528/95-008-0	261/91-011-0	270/87-004-0	344/88-043-0	458/87-003-0
454/88-004-0	529/87-010-0	263/87-009-0	271/87-017-0	344/92-014-0	458/89-007-0
454/90-006-0	529/88-014-0	263/93-008-0	271/88-007-0	346/87-011-0	461/87-025-0
455/87-009-1	529/89-003-1	269/90-013-0	271/95-021-0	353/95-008-0	461/87-029-0

a Reserved

b One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units

Table D-5. (continued).

461/87-055-0	40	Reactivity	344/89-006-1	456/89-006-0[c]	325/95-015-1
461/87-060-0	213/94-009-0	Control	346/90-016-1	456/90-008-0	331/93-010-0
461/89-022-0[b]	275/87-004-0	Imbalance—	364/91-001-0	457/88-031-0	333/87-018-0
461/89-032-0	275/94-020-0[c]	QR3	364/91-005-0	457/89-004-0	333/89-020-1
482/90-011-0	275/94-020-0[c]	94	364/92-008-0	457/91-003-0	333/89-023-0
528/91-009-0	280/87-011-1	029/87-003-1	369/88-013-1	482/87-017-0[b]	341/87-035-0
530/94-007-0	280/87-024-1	029/89-007-0	370/89-001-0	482/87-017-0[b]	341/88-020-0
	281/95-007-0	155/92-009-1	370/90-008-0	482/87-041-0	341/95-005-0
RCS High	282/89-010-1	206/89-023-0	370/91-007-0	483/89-008-0	346/88-028-0
Pressure (RPS	295/93-007-0	206/91-010-0	370/91-012-1	483/95-001-0	352/88-012-1
Trip)—QR0	302/93-009-0	213/93-002-0	370/93-008-0	499/89-026-0	374/88-003-0
13	312/88-015-0	250/95-007-0	382/87-012-1[b]	499/92-001-0	374/94-006-0
029/87-012-0	317/87-013-0	251/94-004-0	382/89-017-1	530/87-004-0	395/93-001-0
249/93-007-0	317/93-003-0[c]	260/93-006-0	382/90-002-0	530/90-004-0	410/88-026-0
263/93-006-1	334/88-007-0	266/95-005-0	389/89-007-0	530/93-004-0	410/89-009-0
271/87-005-0	334/92-009-0	269/87-010-0	389/92-001-1		410/89-036-0
271/90-004-0	334/94-005-0[c]	275/91-008-0	389/93-007-1	Core Power	410/89-040-0
271/90-009-0	348/91-009-0	280/92-001-0	397/92-037-3	Excursion (RPS	413/87-006-1
289/89-003-0	361/92-012-0	280/95-003-0	412/87-012-0[b]	Trip)—QR4	416/92-013-0
293/89-026-1	362/92-003-0	281/95-004-0	412/87-012-0[b]	51	424/87-032-0
312/88-018-0	368/91-005-0	281/95-005-0	412/88-009-0	155/92-010-0	440/88-020-0
333/93-020-0	369/87-004-0	282/91-011-0	414/88-022-0	219/87-005-0	440/88-024-0
336/93-013-0	369/95-006-0	286/89-015-0	416/88-010-0	220/87-014-0	456/87-027-0
341/88-021-1	382/90-003-1	287/90-001-2	416/90-026-0	220/91-002-0	457/88-022-0
458/88-003-0	389/87-001-0	287/90-003-0	423/88-028-0	220/92-003-0	458/92-026-0
	400/87-035-0	287/91-006-1	423/89-009-1	220/92-008-0	461/87-036-0
RCS Low	412/88-007-1	289/91-002-0	423/90-019-1	220/94-005-0	461/87-042-0
Pressure (RPS	413/91-013-1	306/89-004-1[b]	424/87-008-0	245/87-034-0	528/91-010-0[c]
Trip)—QR1	414/91-008-1	306/89-004-1[b]	425/89-027-0	247/88-002-0	528/91-010-0[c]
8	424/88-001-0	306/90-003-1	445/90-028-0	261/87-022-0	
250/87-003-0	425/93-006-0	306/90-012-0	445/92-025-0	263/91-003-0	Turbine Trip—
266/87-005-0	443/91-009-0	311/88-009-0	445/95-002-0[c]	263/91-015-0	QR5
275/90-017-1	445/91-004-0	313/88-003-0	445/95-002-0[c]	265/90-011-0	457
285/92-028-0	482/90-001-0	316/90-004-0	454/87-017-1[b]	271/88-009-0	029/88-008-1
302/91-018-0	483/95-004-0	318/87-008-0	454/87-017-1[b]	277/94-003-0	029/91-004-0
328/90-017-0	498/90-014-0	327/93-002-0	454/88-002-0	278/87-001-1	155/88-009-0
382/94-007-0	498/91-021-0	328/89-008-0	454/90-011-1	289/90-004-0	155/89-008-0
499/91-010-1	498/95-009-0	334/87-013-0	455/88-006-0	295/88-017-0	155/92-014-0
	499/89-009-0	334/89-018-0	455/94-002-0	298/88-002-0	206/87-003-1
Loss of Primary	529/89-009-1	336/94-009-1	456/87-032-0	298/89-001-0	213/92-009-0
Flow (RPS	529/94-002-0	338/87-004-0	456/88-016-0	298/94-004-0	219/89-015-0
Trip)—QR2		341/88-019-1	456/89-006-0[c]	324/91-021-0	219/89-016-0
					219/89-017-1

a Reserved

b One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units

Table D-5. (continued).

219/89-021-1	261/89-004-1	278/90-003-1	304/88-009-0	321/91-004-0	338/88-013-0
219/94-007-0	261/89-006-0	278/91-001-0	304/90-013-0	321/91-007-0	338/89-014-0
219/95-008-0	265/87-005-0	278/91-010-0	305/87-008-0	321/91-013-1	339/93-002-1
220/90-017-0	265/87-009-1	278/95-007-0	305/89-016-0	321/92-014-0	341/87-002-0
220/90-020-0	265/87-020-0	280/92-007-0	305/95-003-0	321/94-003-0	341/87-031-1
220/91-012-0	265/88-001-0	281/87-003-0	306/90-001-0	321/94-014-0	341/88-030-0
220/92-004-0	265/88-005-0	281/88-010-0	306/94-002-0	323/87-004-1	341/89-006-0
220/94-002-0	265/89-001-0	281/88-022-0	309/87-007-0	323/89-005-0	341/93-010-0
220/95-002-0	265/89-005-0	281/89-009-0	309/88-010-0	323/93-001-1	341/93-014-1
237/91-011-0	265/90-010-1	281/93-004-0	309/89-001-0	324/87-001-2	341/95-006-0
237/91-024-0	265/92-001-0	281/95-006-0	309/89-003-0	324/90-015-0	344/91-004-0
244/89-004-0	265/93-013-0	282/90-017-0	309/91-005-1	325/87-019-0	346/87-010-1
244/90-013-0	265/95-005-0	285/92-014-0	309/92-001-0	325/88-024-2	348/88-021-0
245/89-005-0	266/95-006-0	285/92-023-0	311/87-004-1	325/91-007-0	348/91-007-1
247/92-018-0	269/91-011-1	285/93-011-0	311/87-005-0	327/88-045-1	348/91-010-0
247/95-001-0[b]	269/92-003-0	285/93-018-0	311/88-007-0	327/90-022-0	348/95-001-0
247/95-016-0	269/93-008-0	286/87-003-0	313/87-005-0	327/92-010-0	352/87-048-2
249/87-006-0	270/87-001-0	286/88-005-0	313/89-002-0	327/93-003-0	352/91-009-0
249/88-017-0	270/88-003-0	286/88-006-0	313/89-018-0	327/94-014-0	352/95-002-1[c]
249/91-006-0	270/89-002-0	286/90-004-0	313/91-001-1	327/95-010-0	352/95-002-1[c]
249/92-025-1	270/89-003-0	286/91-004-0	313/93-001-0	328/92-001-0	353/89-013-0
249/95-001-0	270/93-005-0	287/88-006-0[b]	313/95-005-0	328/93-006-0	353/93-001-0
249/95-008-1	270/95-002-0	287/88-006-0[b]	313/95-009-0	328/95-001-0	353/93-005-0
249/95-017-0	271/88-008-0	287/89-002-0	315/92-012-0	328/95-002-0	353/95-010-0
250/90-013-0	271/90-015-0	287/92-002-0	316/90-013-0	328/95-003-0	354/87-039-0
251/89-011-0	271/91-005-0	287/93-001-0	316/91-004-0	331/88-008-1	354/88-022-0
251/93-002-0	271/91-014-0	287/94-001-0	316/91-010-0	331/89-011-1	354/88-029-0
251/94-006-0	271/94-004-1	289/87-008-2	316/93-007-0	331/90-014-0	354/89-025-0
254/87-005-0	272/87-007-0	289/88-006-0	316/94-008-0	333/87-012-0	354/90-001-0
254/88-016-0	272/88-015-0	289/91-003-0	316/95-004-0	334/87-001-1	354/90-028-1
254/89-010-1	272/91-024-0	293/90-008-0	317/87-015-0	334/87-012-0	354/93-004-0
254/90-004-0	272/94-005-0	293/92-016-0	317/88-006-0	334/93-013-0	354/94-014-0
254/93-023-0	272/94-007-1	293/93-014-0	317/93-004-0	334/94-008-0	354/94-015-0
255/87-027-0	272/94-009-0	293/94-005-0	317/94-006-1	335/87-011-0	361/90-016-1
255/92-001-0	275/87-001-0	295/88-011-0	317/94-007-1	335/89-005-0	362/89-006-1
255/92-034-1	275/87-006-1	295/89-002-0	318/87-009-1	335/92-006-0	362/93-001-0
255/92-035-0	275/88-026-1	295/90-017-0	318/94-004-0	335/94-003-0	364/89-008-0
255/92-039-0	275/90-014-0	298/87-011-0	318/94-006-0	335/94-005-0	364/89-012-0
260/91-019-0	275/93-011-0	298/89-025-0	321/87-001-0	335/95-003-1	364/89-015-0
260/94-013-1	275/95-009-0	298/90-011-0	321/87-002-0	336/87-007-1	364/92-005-0
260/95-002-0	277/91-028-1	301/87-002-0	321/88-003-0	336/91-001-1	364/94-003-0
261/88-001-0	277/92-009-0	301/89-002-0	321/88-005-0	338/87-015-1	364/94-004-0
261/88-010-0	277/92-012-0	301/89-004-0	321/88-018-0	338/87-020-0	364/95-008-0
261/88-011-1	277/92-015-0	301/93-002-0	321/90-020-0	338/88-005-0	366/91-004-0

a. Reserved

b. One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units.

Table D-5. (continued).

366/92-026-0	397/87-018-0[b]	424/87-030-0	456/90-001-0	499/90-005-0	278/90-002-1
368/87-007-0	397/89-002-0	424/87-041-0	456/90-023-0	499/91-003-0	278/94-002-0
368/87-008-0	397/89-028-0	424/87-047-0[b]	457/88-012-1	499/91-004-0	280/87-019-0
369/88-001-1	397/90-031-0	424/87-047-0[b]	457/89-002-0	499/91-007-1	280/90-004-0[c]
369/90-032-0	397/95-002-0	424/87-063-0	457/91-006-0	499/93-001-1	280/90-004-0[c]
369/93-009-0	397/95-004-0	424/88-006-0	457/92-001-0	499/94-007-0	285/89-019-0
369/94-004-0	397/95-006-1	424/88-008-0	457/92-007-0	499/95-008-0	285/95-003-0[b]
370/87-016-1	400/87-012-0	424/88-022-1	457/94-003-0	528/87-003-0	285/95-003-0[b]
370/87-021-0	400/87-038-0	424/88-024-0	458/88-007-0	528/88-021-0	286/95-018-0
370/92-010-0	400/95-010-0	424/93-008-0	458/88-018-4	528/90-006-0	293/95-003-0
373/87-003-0	410/87-043-0	425/89-019-0	458/88-021-1	528/92-012-0	298/87-014-0
373/87-005-1	410/88-039-1	425/89-031-0	458/89-008-0	528/92-016-0	298/88-019-0
373/89-009-1	410/89-014-0	425/90-002-0	458/89-042-0	528/93-001-0	304/90-001-0
373/90-006-0	410/90-013-1	425/91-007-0	458/90-008-0	529/91-004-1	309/94-011-0
373/90-010-0	410/91-022-0	425/92-010-0	458/90-014-0	530/89-001-3	309/95-001-0
374/87-014-0	410/93-012-0	425/94-001-0	458/90-047-0	530/91-008-0	315/95-012-0
374/90-010-0	410/94-001-1	425/94-002-0	458/92-005-0		317/87-004-0
374/91-012-0	410/95-005-1	440/87-045-0	458/93-024-2	Manual Reactor	317/88-012-1
374/91-014-0	412/87-019-0	440/88-026-0	461/87-043-0	Trip—QR6	318/87-006-0
374/92-012-0	412/87-028-0	440/95-005-0	461/89-028-0	103	318/92-007-0
374/94-001-0	412/87-036-0	443/90-015-1	461/90-013-0	155/88-002-0	321/90-021-0
374/94-008-1	412/90-008-0	443/90-022-0	461/92-001-0	155/94-007-0	321/92-024-0
382/87-007-1	412/91-005-0	443/91-001-0	482/87-004-0	213/94-018-1	323/87-024-1
382/91-011-1	412/95-006-0	443/91-006-0	482/87-037-0	237/95-009-0	323/89-007-0
382/95-002-0	413/91-021-0	443/91-008-0	482/89-002-0	245/91-007-0	323/89-008-0
387/88-006-0	414/87-029-0	443/93-003-0	482/90-013-0	249/87-013-1	327/94-011-0
387/88-010-0	414/88-028-0	445/90-029-0	482/92-016-0	250/87-010-0	327/95-019-0
387/89-027-0	414/92-001-0	445/91-002-0	483/88-004-1	250/87-034-0	331/ 89-001-0
387/92-017-0	416/87-012-0	445/91-020-0	483/88-007-0	250/91-008-0	331/94-010-0
387/93-008-1	416/88-002-0	445/91-023-0	483/90-005-0	251/90-004-0	333/91-006-1
388/90-002-0	416/90-017-1	445/92-001-0	483/90-016-0	254/89-003-0	335/90-007-0
388/91-012-0	416/91-002-1	445/93-011-0	483/92-010-0	254/94-008-0	339/93-003-0
388/94-002-0	416/91-005-1	445/94-001-0	498/88-048-0	254/95-001-0	341/89-007-0
388/95-005-0	416/91-007-0	445/94-006-0	498/88-049-0	255/87-011-0	341/91-004-0
389/87-004-0	416/92-017-2	446/93-005-0	498/89-001-0	255/87-021-0	341/92-002-0
389/87-007-1	416/95-010-0	446/94-010-0	498/89-015-1	255/87-025-0	344/87-037-0
389/92-005-0	423/87-031-1	454/88-005-1	498/90-015-0	260/91-014-1	353/92-012-0
395/87-021-0	423/90-009-0	454/92-001-0	498/90-025-1	260/91-018-0	354/92-013-0
395/89-006-0	423/91-014-1	455/87-005-0	498/91-022-0	261/94-006-0	354/93-012-0
395/89-012-0	423/92-027-0	456/87-050-0	498/95-013-0	261/94-016-1	354/95-005-0
395/89-020-0	423/92-029-0	456/87-052-0	499/89-017-0	265/94-005-0	361/89-019-0
397/87-018-0[b]	423/93-004-1	456/87-057-1	499/89-023-0	272/88-009-0	361/91-007-1
				277/92-006-0	

a Reserved.

b. One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-5. (continued).

364/92-001-0	219/92-010-0	348/89-006-0	029/88-002-1	269/89-013-0	321/93-009-0
366/88-024-0	220/87-024-0	364/87-009-0	029/89-013-0	269/91-006-1	321/93-012-0
366/92-015-0	237/91-004-1	368/95-003-0	206/89-021-1	270/93-007-0	323/88-002-1
366/93-005-0	244/88-003-0	382/88-001-0	206/90-007-1	272/88-003-0	323/88-010-0
368/90-005-0	244/88-005-0	382/88-033-0	206/91-017-0	272/89-012-0	325/91-018-0
373/93-011-0	244/90-012-0	400/87-062-0	213/88-008-0	272/93-004-0	328/88-024-0
374/88-012-0	244/92-002-0	410/92-022-0	213/88-009-0	275/88-002-0	328/90-008-0
374/91-010-0	247/95-001-0 [b]	412/87-015-0	213/88-012-0	275/88-020-0	328/92-008-1
374/92-004-0	250/87-009-1	412/87-020-1	213/93-013-1	275/91-009-0	328/92-011-0
382/88-002-0	255/92-037-0	412/87-024-0	219/90-004-0	277/89-033-0	331/89-009-0
397/90-021-0	260/94-004-0	412/87-026-0	219/92-007-0	278/92-003-0	331/90-004-0
397/93-006-0	260/95-004-0	412/87-032-1	220/90-019-0	280/88-003-0	331/92-013-1
397/94-008-0	261/87-020-0[b]	413/87-028-0	220/91-008-0	280/88-029-0	333/90-001-0
410/88-028-0	271/87-015-1	414/95-004-0	220/92-009-0	280/93-002-0	333/90-026-1
410/88-051-0	282/93-005-0	416/93-008-0	220/93-002-1	281/93-005-0	333/93-013-0
410/89-024-0	301/95-003-0	424/87-009-0[b]	220/93-006-0	282/87-004-0	335/91-006-0
410/95-003-0	304/88-007-1	424/87-009-0[b]	220/94-004-0	282/87-013-0	336/90-012-0
410/95-007-0	305/93-001-0[b]	424/87-010-0[b]	220/94-007-0	285/95-005-0	339/95-004-0
410/95-008-0	305/93-001-0[b]	424/87-010-0[b]	237/91-037-0	286/87-012-0	341/87-011-0
414/87-018-0	306/89-002-0	424/87-025-1	244/90-016-0	287/89-004-0	341/90-011-0
440/87-073-1	306/90-009-0	440/88-015-0	245/87-036-0	287/91-005-0	344/88-001-1
440/93-015-0	315/88-001-0	443/93-018-0	245/93-018-0	287/92-004-0	344/88-026-1
440/94-002-0	315/91-004-0	445/90-009-0	247/87-004-0	287/95-002-0	344/88-028-0
443/89-008-0	316/87-005-0	445/91-008-0	247/87-009-0	289/93-003-0	344/89-017-1
445/93-007-0	316/87-007-0	455/87-002-1[b]	247/88-018-0	293/89-038-0	346/90-002-1
446/94-003-0	316/95-005-0	455/87-002-1[b]	247/89-013-0	301/88-001-0	346/92-002-1
446/94-012-0	318/87-005-1	455/88-009-0	247/91-001-1	301/91-006-0	348/87-004-0
446/94-014-0	318/93-003-0	455/93-003-0	247/91-013-0	302/87-009-2	348/91-006-0
455/87-010-0	318/94-005-0	456/89-004-0	249/89-002-0	302/87-011-0	348/91-008-0
458/94-028-0	321/92-021-0	482/87-022-1	250/87-023-0	305/88-006-0	354/87-017-0
458/95-012-0	323/87-001-1	482/90-012-0	250/89-004-0	305/95-005-0	354/87-051-0
461/90-012-0	323/87-016-0	483/87-032-0	251/87-001-0	306/90-002-0	354/94-011-0
461/91-008-0	324/90-012-1	498/88-026-0	251/90-003-0	311/87-002-0	361/91-003-0
461/93-006-0	324/92-001-1	498/90-016-0	251/92-004-0	311/88-006-0	364/87-001-0
461/95-001-0	325/92-005-0	498/91-012-1	255/91-012-0	311/88-014-0	364/92-002-0
461/95-005-0	328/88-028-0[b]	499/89-022-0	260/92-006-0	311/88-016-0	364/92-006-0
529/90-001-0	331/89-003-0	528/87-018-1	261/87-025-0	311/92-014-0	366/88-006-0
529/91-003-0	331/91-003-0	528/88-011-0	261/90-002-0	315/88-013-0	366/88-011-0
	333/87-020-0	528/88-015-0	263/89-038-0	315/89-003-0	366/88-018-0
Other Reactor	334/87-002-0	529/87-019-0	263/90-017-0	316/87-013-0	366/90-003-0
Trip (Valid RPS	334/89-007-0		263/91-014-0	316/89-014-0	366/94-007-0
Trip)—QR7	335/94-004-0	Spurious Reactor	265/93-005-0	316/95-006-0	368/90-014-1
84	336/93-019-0	Trip—QR8	265/93-024-0	317/94-001-0	369/87-009-0
213/87-005-1	338/89-017-0	217	269/89-001-0	321/91-017-0	369/87-036-0

a Reserved

b. One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units.

Table D-5. (continued).

369/88-005-1	414/94-007-0	425/91-006-0	458/92-001-2	Spurious	362/88-002-1
369/89-022-1	416/88-012-2	440/87-007-0	458/94-023-1	Engineered	366/87-003-0
369/90-027-0	416/89-010-0	440/88-017-1	482/95-001-0	Safety Feature	368/88-020-0
369/91-004-0	416/89-016-0	440/88-023-0	483/89-006-0	Actuation—QR9	369/87-017-1
369/92-009-0	416/91-010-0	443/90-018-0	483/92-002-0	36	382/91-019-0
374/89-011-1	416/91-012-0	443/92-024-0	498/88-045-0	237/87-032-0	382/91-022-0
374/90-001-1	416/92-010-1	443/93-012-0	498/90-020-0	237/89-019-1	400/95-011-1
382/87-012-1[b]	416/94-011-0	445/92-009-0	498/92-003-1	237/90-001-0	412/93-002-1
382/91-010-0	416/95-004-1	445/93-001-1	499/89-013-0	254/92-004-0	414/89-003-1
389/94-003-0	416/95-007-0	445/93-002-0	499/89-016-0	265/87-011-0	416/88-019-1
395/87-024-0	423/87-002-0	454/90-002-0	499/90-002-0	265/94-006-0	424/94-001-0
395/88-002-0	423/87-026-0	454/94-009-1	499/90-013-0	275/89-009-1	455/93-008-1
395/88-007-1	424/87-018-0[b]	455/87-001-1	499/95-003-0	277/89-015-1	456/90-018-0
395/88-009-1	424/87-018-0[b]	455/87-006-1	528/89-004-0	278/92-008-0	456/94-012-0
395/92-003-0	424/87-050-0	455/88-012-0	529/87-004-1	280/93-001-0	457/88-026-0
397/87-019-0	424/87-066-0	455/90-001-0	529/89-010-0	281/91-007-1	458/94-030-0
397/89-035-0	424/88-025-2	456/87-035-0	529/92-006-0	285/94-001-0	482/87-002-0
400/87-004-0	424/93-009-0	456/88-023-0	529/94-006-0	298/88-021-0	482/89-004-0
400/91-010-0	424/95-002-0[c]	456/93-001-0	530/90-007-0	315/88-011-0	530/91-003-1
410/87-033-0	424/95-002-0[c]	457/88-018-0		318/95-002-1	
412/88-013-0	425/89-020-0	458/88-002-0		325/87-017-1	
414/94-005-0	425/89-024-0	458/89-035-0		353/90-015-0	

a Reserved

b One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units

Table D-6. LERs with assigned initial plant fault (IPF) code based on all the operating experience from 1987 through 1995.

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
029/87-003-1	QR3	213/88-012-0	QR8	220/90-019-0	QR8	237/95-009-0	QR6
029/87-012-0	QR0	213/90-018-0	QP5	220/90-020-0	QR5	244/88-003-0	QR7
029/88-002-1	QR8	213/90-020-0	QP4	220/90-026-0	L1	244/88-005-0	QR7
029/88-003-0	QP2	213/92-009-0	QR5	220/91-002-0	QR4	244/89-004-0	QR5
029/88-008-1	QR5	213/93-002-0	QR3	220/91-008-0	QR8	244/90-007-0	QP2
029/89-005-0	QL5	213/93-013-1	QR8	220/91-012-0	QR5	244/90-010-1	QP2
029/89-007-0	QR3	213/94-009-0	QR2	220/91-014-0	QP2	244/90-012-0	QR7
029/89-013-0	QR8	213/94-018-1	QR6	220/92-003-0	QR4	244/90-013-0	QR5
029/90-011-0	P1	213/95-016-0	QP2	220/92-004-0	QR5	244/90-016-0	QR8
029/91-002-0	B1	219/87-005-0	QR4	220/92-008-0	QR4	244/90-019-0	QP4
029/91-004-0	QR5	219/87-011-1	QP5	220/92-009-0	QR8	244/92-002-0	QR7
155/88-002-0	QR6	219/87-029-0	QL5	220/93-002-1	QR8	244/92-003-0	QP2
155/88-008-0	L2	219/89-011-0	L2	220/93-006-0	QR8	244/93-006-0	QP2
155/88-009-0	QR5	219/89-015-0	QR5	220/94-002-0	QR5	244/94-007-0	QP2
155/89-008-0	QR5	219/89-016-0	QR5	220/94-004-0	QR8	244/95-008-0	QL4
155/92-009-1	QR3	219/89-017-1	QR5	220/94-005-0	QR4	245/87-007-0	L1
155/92-010-0	QR4	219/89-021-1	QR5	220/94-007-0	QR8	245/87-034-0	QR4
155/92-014-0	QR5	219/90-004-0	QR8	220/95-002-0	QR5	245/87-036-0	QR8
155/94-007-0	QR6	219/90-005-0	C1	237/87-016-0	QP5	245/87-038-0	D1
155/94-010-1	P1	219/90-008-0	L2	237/87-023-1	QP2	245/88-003-0	QP2
155/95-007-0	QP5	219/91-005-0	QP4	237/87-024-0	QP2	245/89-005-0	QR5
206/87-003-1	QR5	219/92-005-0	H1	237/87-032-0	QR9	245/89-015-0	L2
206/89-019-0	QP2	219/92-007-0	QR8	237/89-012-0	P1	245/89-021-0	QP5
206/89-021-1	QR8	219/92-009-0	QP2	237/89-019-1	QR9	245/90-015-0	QP2
206/89-023-0	QR3	219/92-010-0	QR7	237/90-001-0	QR9	245/90-016-1	QL4
206/90-007-1	QR8	219/94-003-0	QP5	237/90-002-2	H1	245/91-007-0	QR6
206/90-011-0	QP2	219/94-007-0	QR5	237/90-006-1	G2	245/92-028-0	QL5
206/91-010-0	QR3	219/95-008-0	QR5	237/91-004-1	QR7	245/93-018-0	QR8
206/91-017-0	QR8	220/87-014-0	QR4	237/91-011-0	QR5	247/87-004-0	QR8
213/87-005-1	QR7	220/87-024-0	QR7	237/91-024-0	QR5	247/87-009-0	QR8
213/88-008-0	QR8	220/87-028-1	QP2	237/91-037-0	QR8	247/88-002-0	QR4
213/88-009-0	QR8	220/90-017-0	QR5	237/94-005-2	D1	247/88-006-0	QP2

a. Reserved.

b. One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
247/88-018-0	QR8	250/87-003-0	QR1	255/87-011-0	QR6	261/87-025-0	QR8
247/88-019-0	QP2	250/87-009-1	QR7	255/87-016-0	K1	261/88-001-0	QR5
247/89-002-0	D1	250/87-010-0	QR6	255/87-021-0	QR6	261/88-010-0	QR5
247/89-013-0	QR8	250/87-023-0	QR8	255/87-024-0	B1	261/88-011-1	QR5
247/91-001-1	QR8	250/87-034-0	QR6	255/87-025-0	QR6	261/89-004-1	QR5
247/91-013-0	QR8	250/89-004-0	QR8	255/87-027-0	QR5	261/89-005-0	QL5
247/92-002-0	QP2	250/89-020-1	QL5	255/89-020-0	QP2	261/89-006-0	QR5
247/92-007-0	QP5	250/90-011-1	QP5	255/90-001-1	P1	261/90-002-0	QR8
247/92-018-0	QR5	250/90-013-0	QR5	255/90-002-0	QP2	261/90-007-0	QP2
247/95-001-0[b]	QR5	250/91-008-0	QR6	255/91-012-0	QR8	261/91-011-0	QP4
247/95-001-0[b]	QR7	250/94-006-0	QP2	255/91-015-0	QP2	261/92-017-0	B1
247/95-016-0	QR5	250/95-007-0	QR3	255/92-001-0	QR5	261/94-006-0	QR6
249/87-006-0	QR5	251/87-001-0	QR8	255/92-034-1	QR5	261/94-016-1	QR6
249/87-010-0	L2	251/88-010-0	QP2	255/92-035-0	QR5	261/95-004-0	QL5
249/87-011-0	QP4	251/89-011-0	QR5	255/92-037-0	QR7	263/87-004-0	QP2
249/87-012-0	QP5	251/90-003-0	QR8	255/92-038-1	QC4	263/87-006-0	QC4
249/87-013-1	QR6	251/90-004-0	QR6	255/92-039-0	QR5	263/87-009-0	QP4
249/87-016-0	L1	251/90-008-0	QP4	255/95-003-0	P1	263/87-014-0	QL4
249/88-017-0	QR5	251/92-004-0	QR8	260/91-014-1	QR6	263/88-007-0	P1
249/89-001-1	B1	251/93-002-0	QR5	260/91-017-0	QP2	263/89-009-0	QP2
249/89-002-0	QR8	251/93-003-0	QP5	260/91-018-0	QR6	263/89-038-0	QR8
249/89-006-0	L1	251/94-004-0	QR3	260/91-019-0	QR5	263/90-017-0	QR8
249/90-005-0	QL5	251/94-006-0	QR5	260/92-004-1	QP2	263/91-003-0	QR4
249/91-006-0	QR5	254/87-005-0	QR5	260/92-006-0	QR8	263/91-014-0	QR8
249/92-021-1	QP4	254/88-016-0	QR5	260/93-006-0	QR3	263/91-015-0	QR4
249/92-025-1	QR5	254/89-003-0	QR6	260/94-004-0	QR7	263/91-019-0	C1
249/93-004-0	D1	254/89-004-0	G2	260/94-005-0	L1	263/93-006-1	QR0
249/93-007-0	QR0	254/89-010-1	QR5	260/94-013-1	QR5	263/93-008-0	QP4
249/93-014-0	QL4	254/90-004-0	QR5	260/95-002-0	QR5	263/94-003-0	QL4
249/94-018-0	QP4	254/91-025-0	QP5	260/95-004-0	QR7	263/94-004-0	L2
249/95-001-0	QR5	254/92-004-0	QR9	260/95-007-0	L2	265/87-005-0	QR5
249/95-008-1	QR5	254/93-023-0	QR5	261/87-020-0[b]	QP2	265/87-009-1	QR5
249/95-017-0	QR5	254/94-008-0	QR6	261/87-020-0[b]	QR7	265/87-011-0	QR9
249/95-019-0	QL4	254/95-001-0	QR6	261/87-022-0	QR4	265/87-013-0	C1

a. Reserved.

b One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
265/87-020-0	QR5	269/92-004-1	QP2	272/88-003-0	QR8	275/90-017-1	QR1
265/88-001-0	QR5	269/92-015-0	QP2	272/88-009-0	QR6	275/91-002-1	QP2
265/88-005-0	QR5	269/93-008-0	QR5	272/88-015-0	QR5	275/91-007-0	QP5
265/88-026-0	D1	269/93-010-1	QL5	272/89-007-0	QP2	275/91-008-0	QR3
265/89-001-0	QR5	269/94-002-0	P1	272/89-012-0	QR8	275/91-009-0	QR8
265/89-005-0	QR5	270/87-001-0	QR5	272/89-027-0	QL5	275/92-002-0	QP2
265/90-010-1	QR5	270/87-002-0	QP2	272/90-012-0	QP2	275/92-004-0	L2
265/90-011-0	QR4	270/87-004-0	QP5	272/90-029-0	QC5	275/93-011-0	QR5
265/91-012-0	G2	270/88-003-0	QR5	272/90-030-0	QK4	275/94-020-0[c]	QR2
265/92-001-0	QR5	270/89-002-0	QR5	272/91-024-0	QR5	275/94-020-0[c]	QR2
265/93-001-0	QL5	270/89-003-0	QR5	272/93-002-0	P1	275/95-009-0	QR5
265/93-005-0	QR8	270/89-004-0	QP3	272/93-004-0	QR8	275/95-015-0	P1
265/93-006-0	G2	270/92-004-0	B1	272/93-011-0	QL4	275/95-017-0	QL4
265/93-013-0	QR5	270/93-005-0	QR5	272/93-013-0	QP2	277/89-012-1	QP5
265/93-024-0	QR8	270/93-007-0	QR8	272/94-003-0	QP2	277/89-015-1	QR9
265/94-005-0	QR6	270/94-002-0	QP2	272/94-005-0	QR5	277/89-023-0	QL5
265/94-006-0	QR9	270/94-005-0	P1	272/94-007-1	QR5	277/89-033-0	QR8
265/95-005-0	QR5	270/95-002-0	QR5	272/94-009-0	QR5	277/91-022-1	L2
266/87-005-0	QR1	271/87-005-0	QR0	272/94-011-0	QC5	277/91-028-1	QR5
266/91-005-0	QC4	271/87-015-1	QR7	275/87-001-0	QR5	277/92-006-0	QR6
266/91-008-0	QC4	271/87-017-0	QP5	275/87-002-0	QP5	277/92-009-0	QR5
266/92-008-0	QL5	271/88-007-0	QP5	275/87-004-0	QR2	277/92-010-0	C1
266/95-005-0	QR3	271/88-008-0	QR5	275/87-006-1	QR5	277/92-012-0	QR5
266/95-006-0	QR5	271/88-009-0	QR4	275/87-023-1	QP2	277/92-015-0	QR5
269/87-010-0	QR3	271/90-004-0	QR0	275/88-002-0	QR8	277/93-004-0	QP4
269/88-009-0	P1	271/90-009-0	QR0	275/88-020-0	QR8	277/94-003-0	QR4
269/89-001-0	QR8	271/90-015-0	QR5	275/88-021-0	QP5	278/87-001-1	QR4
269/89-002-0	H1	271/91-005-0	QR5	275/88-025-1	QP2	278/87-002-0	P1
269/89-013-0	QR8	271/91-009-1	B1	275/88-026-1	QR5	278/90-002-1	QR6
269/90-013-0	QP4	271/91-014-0	QR5	275/89-009-1	QR9	278/90-003-1	QR5
269/91-006-1	QR8	271/94-004-1	QR5	275/90-002-0	P1	278/90-008-0	L2
269/91-011-1	QR5	271/95-021-0	QP5	275/90-005-0	H1	278/91-001-0	QR5
269/92-003-0	QR5	272/87-007-0	QR5	275/90-014-0	QR5	278/91-010-0	QR5

a. Reserved.

b. One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
278/92-003-0	QR8	281/90-004-0	QP2	286/88-001-0	P1	289/87-008-2	QR5
278/92-005-0	L2	281/91-007-1	QR9	286/88-002-0	QP2	289/88-006-0	QR5
278/92-008-0	QR9	281/91-011-0	QP5	286/88-005-0	QR5	289/89-003-0	QR0
278/93-002-0	P1	281/93-002-0	QP2	286/88-006-0	QR5	289/90-004-0	QR4
278/93-004-0	L2	281/93-003-0	QP2	286/89-015-0	QR3	289/91-002-0	QR3
278/94-002-0	QR6	281/93-004-0	QR5	286/90-004-0	QR5	289/91-003-0	QR5
278/94-005-0	P1	281/93-005-0	QR8	286/91-003-0	QL4	289/92-002-0	QL4
278/95-001-0	L2	281/93-006-0	P1	286/91-004-0	QR5	289/93-003-0	QR8
278/95-003-0	QP5	281/95-004-0	QR3	286/91-005-0	QP2	293/89-011-0	L1
278/95-007-0	QR5	281/95-005-0	QR3	286/92-015-1	QP2	293/89-015-0	QP5
280/87-011-1	QR2	281/95-006-0	QR5	286/95-012-0	QP2	293/89-023-0	L2
280/87-019-0	QR6	281/95-007-0	QR2	286/95-018-0	QR6	293/89-026-1	QR0
280/87-024-1	QR2	282/87-004-0	QR8	287/88-006-0[b]	QR5	293/89-038-0	QR8
280/88-003-0	QR8	282/87-013-0	QR8	287/88-006-0[b]	QR5	293/90-008-0	QR5
280/88-029-0	QR8	282/89-010-1	QR2	287/89-002-0	QR5	293/90-013-0	QP5
280/89-026-0	QP5	282/90-017-0	QR5	287/89-004-0	QR8	293/92-016-0	QR5
280/89-044-0	C1	282/91-011-0	QR3	287/90-001-2	QR3	293/92-018-0	L1
280/90-004-0[c]	QR6	282/93-005-0	QR7	287/90-002-0	QP2	293/93-004-0	C1
280/90-004-0[c]	QR6	285/89-019-0	QR6	287/90-003-0	QR3	293/93-014-0	QR5
280/90-006-0	D1	285/90-026-1	D1	287/91-005-0	QR8	293/93-022-0	B1
280/92-001-0	QR3	285/92-014-0	QR5	287/91-006-1	QR3	293/94-005-0	QR5
280/92-007-0	QR5	285/92-023-0	QC4	287/91-007-0	QP3	293/95-003-0	QR6
280/93-001-0	QR9	285/92-028-0	QR1	287/91-008-0	G1	295/88-005-0	QP5
280/93-002-0	QR8	285/93-011-0	QR5	287/92-001-0	QP5	295/88-011-0	QR5
280/94-006-0	QP2	285/93-018-0	QR5	287/92-002-0	QR5	295/88-013-0	QP2
280/95-001-1	QP2	285/94-001-0	QR9	287/92-003-0	QC4	295/88-017-0	QR4
280/95-003-0	QR3	285/95-003-0[b]	QR6	287/92-004-0	QR8	295/89-002-0	QR5
281/87-003-0	QR5	285/95-003-0[b]	QR6	287/93-001-0	QR5	295/90-004-0	QP5
281/88-004-0	QC4	285/95-005-0	QR8	287/94-001-0	QR5	295/90-017-0	QR5
281/88-010-0	QR5	286/87-001-0	QP2	287/94-002-0	QC4	295/91-016-0	QC4
281/88-022-0	QR5	286/87-002-0	QC5	287/94-003-0	P1	295/93-007-0	QR2
281/89-009-0	QR5	286/87-003-0	QR5	287/95-002-0	QR8	295/94-005-0	H1
281/89-010-0	QP2	286/87-004-0	QP2	289/87-004-1	QP2	295/94-010-0	H1
281/90-003-0	QP2	286/87-012-0	QR8	289/87-006-0	QP2	298/87-002-0	QP5

a. Reserved

b One LER that describes multiple reactor trip events from one plant unit

c. One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
298/87-003-0	P1	304/88-007-1	QR7	309/88-006-0	B1	311/93-002-0	P1
298/87-005-0	L1	304/88-009-0	QR5	309/88-010-0	QR5	311/93-005-0	QP2
298/87-009-0	P1	304/90-001-0	QR6	309/89-001-0	QR5	311/94-008-0	QP2
298/87-011-0	QR5	304/90-010-0	QL6	309/89-003-0	QR5	311/94-011-0	QL5
298/87-014-0	QR6	304/90-011-1	H1	309/91-005-1	QR5	311/95-004-1	QC5
298/88-002-0	QR4	304/90-013-0	QR5	309/91-006-0	P1	312/88-015-0	QR2
298/88-019-0	QR6	304/91-002-1	QP2	309/91-010-0	QP4	312/88-018-0	QR0
298/88-021-0	QR9	304/91-004-0	QP5	309/91-012-0	QP4	312/88-019-0	P1
298/89-001-0	QR4	305/87-008-0	QR5	309/92-001-0	QR5	312/89-004-0	QP2
298/89-025-0	QR5	305/87-009-0	H1	309/94-008-0	QP2	313/87-004-0	P1
298/89-026-0	H1	305/88-001-0	H1	309/94-011-0	QR6	313/87-005-0	QR5
298/90-011-0	QR5	305/88-004-0	QP3	309/95-001-0	QR6	313/88-003-0	QR3
298/93-038-0	QP2	305/88-006-0	QR8	311/87-002-0	QR8	313/89-002-0	QR5
298/94-004-0	QR4	305/89-016-0	QR5	311/87-004-1	QR5	313/89-018-0	QR5
301/87-002-0	QR5	305/91-010-0	P1	311/87-005-0	QR5	313/89-037-0	QP2
301/88-001-0	QR8	305/92-017-0	H1	311/87-011-1	QP5	313/89-038-0	QP2
301/89-002-0	QR5	305/92-020-1	QL4	311/88-006-0	QR8	313/89-048-0	P1
301/89-004-0	QR5	305/93-001-0[b]	QR7	311/88-007-0	QR5	313/91-001-1	QR5
301/91-006-0	QR8	305/93-001-0[b]	QR7	311/88-009-0	QR3	313/91-005-0	P1
301/93-002-0	QR5	305/95-003-0	QR5	311/88-014-0	QR8	313/93-001-0	QR5
301/95-003-0	QR7	305/95-005-0	QR8	311/88-016-0	QR8	313/94-002-0	L1
302/87-009-2	QR8	306/89-002-0	QR7	311/88-017-0	QP5	313/95-004-0	P1
302/87-011-0	QR8	306/89-004-1[b]	QR3	311/88-024-0	QP5	313/95-005-0	QR5
302/88-006-2	QP2	306/89-004-1[b]	QR3	311/89-003-0	QP2	313/95-009-0	QR5
302/88-024-0	P1	306/90-001-0	QR5	311/89-005-0	QP2	315/87-008-0	QP5
302/89-023-0	B1	306/90-002-0	QR8	311/89-008-0	QL5	315/87-021-0	QP2
302/91-003-1	QL4	306/90-003-1	QR3	311/89-013-1	QL4	315/88-001-0	QR7
302/91-014-0	QP2	306/90-009-0	QR7	311/90-029-1	P1	315/88-011-0	QR9
302/91-017-0	QP2	306/90-012-0	QR3	311/90-036-0	QP2	315/88-013-0	QR8
302/91-018-0	QR1	306/94-002-0	QR5	311/91-017-0	H1	315/89-001-0	L2
302/92-001-0	B1	309/87-006-1	QP5	311/92-007-0	QP5	315/89-003-0	QR8
302/92-027-0	QP2	309/87-007-0	QR5	311/92-009-0	QP2	315/91-004-0	QR7
302/93-009-0	QR2	309/88-001-0	QP4	311/92-014-0	QR8	315/92-012-0	QR5

a. Reserved.

b. One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
315/95-003-0	L2	317/91-003-0	QP2	321/87-011-1	QC4	323/88-010-0	QR8
315/95-012-0	QR6	317/92-008-0	H1	321/87-013-0	QP2	323/89-005-0	QR5
316/87-004-0	L2	317/93-003-0[c]	QC5	321/88-003-0	QR5	323/89-007-0	QR6
316/87-005-0	QR7	317/93-003-0[c]	QR2	321/88-005-0	QR5	323/89-008-0	QR6
316/87-007-0	QR7	317/93-004-0	QR5	321/88-009-0	QL5	323/89-010-0	H1
316/87-008-0	QP5	317/94-001-0	QR8	321/88-013-0	P1	323/93-001-1	QR5
316/87-013-0	QR8	317/94-006-1	QR5	321/88-018-0	QR5	323/94-012-0	QL4
316/89-014-0	QR8	317/94-007-1	QR5	321/90-012-0	H1	323/95-002-0	QL4
316/90-004-0	QR3	317/95-002-0	QP2	321/90-013-0	QP2	324/87-001-2	QR5
316/90-012-0	QP2	317/95-005-1	QP5	321/90-020-0	QR5	324/87-004-0	QP2
316/90-013-0	QR5	317/95-006-0	QP2	321/90-021-0	QR6	324/88-001-7	L2
316/91-004-0	QR5	318/87-002-1	QP2	321/91-001-0	H1	324/88-018-0	QP5
316/91-006-0	H1	318/87-005-1	QR7	321/91-004-0	QR5	324/89-009-1	B1
316/91-010-0	QR5	318/87-006-0	QR6	321/91-007-0	QR5	324/90-004-3	G2
316/92-007-0	L2	318/87-008-0	QR3	321/91-013-1	QR5	324/90-008-2	QP5
316/93-007-0	QR5	318/87-009-1	QR5	321/91-017-0	QR8	324/90-009-0	L1
316/93-008-0	QP2	318/88-002-2	QC5	321/92-009-0	QP2	324/90-012-1	QR7
316/94-001-0	QL5	318/88-004-0	QP2	321/92-014-0	QR5	324/90-015-0	QR5
316/94-005-0	L2	318/91-005-0	P1	321/92-021-0	QR7	324/90-016-0	QP5
316/94-008-0	QR5	318/92-001-0	QK4	321/92-024-0	QR6	324/91-001-1	QP5
316/95-002-0	QP2	318/92-003-0	QL6	321/93-001-0	L2	324/91-021-0	QR4
316/95-004-0	QR5	318/92-005-0	P1	321/93-009-0	QR8	324/92-001-1	QR7
316/95-005-0	QR7	318/92-006-0	QL5	321/93-012-0	QR8	325/87-017-1	QR9
316/95-006-0	QR8	318/92-007-0	QR6	321/93-013-0	QP3	325/87-019-0	QR5
317/87-003-0	D1	318/93-003-0	QR7	321/93-016-0	QP2	325/88-023-0	QP5
317/87-004-0	QR6	318/94-001-1	C1	321/94-003-0	QR5	325/88-024-2	QR5
317/87-011-0	QP5	318/94-004-0	QR5	321/94-014-0	QR5	325/90-017-0	L3
317/87-012-1[c]	B1	318/94-005-0	QR7	323/87-001-1	QR7	325/91-007-0	QR5
317/87-012-1[c]	B1	318/94-006-0	QR5	323/87-003-1	QL5	325/91-018-0	QR8
317/87-013-0	QR2	318/95-002-1	QR9	323/87-004-1	QR5	325/92-003-0	QP2
317/87-015-0	QR5	318/95-003-0	QP5	323/87-016-0	QR7	325/92-005-0	QR7
317/88-006-0	QR5	318/95-005-0	QL4	323/87-024-1	QR6	325/95-011-0	QL4
317/88-009-0	QP5	321/87-001-0	QR5	323/88-002-1	QR8	325/95-015-1	QR4
317/88-012-1	QR6	321/87-002-0	QR5	323/88-008-0	H1	325/95-018-0	QP3

a. Reserved

b One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
327/88-045-1	QR5	328/92-001-0	QR5	333/87-020-0	QR7	334/94-008-0	QR5
327/88-047-1	QP5	328/92-008-1	QR8	333/89-020-1	QR4	335/87-002-0	QP2
327/89-005-1	QC4	328/92-011-0	QR8	333/89-023-0	QR4	335/87-010-0	QC4
327/89-035-0	QP5	328/93-001-0	K1	333/90-001-0	QR8	335/87-011-0	QR5
327/90-012-0	P1	328/93-006-0	QR5	333/90-009-0	QP5	335/87-013-1	QP2
327/90-021-2	QC4	328/95-001-0	QR5	333/90-023-0	QL4	335/87-016-0	QP4
327/90-022-0	QR5	328/95-002-0	QR5	333/90-026-1	QR8	335/87-017-0	QC4
327/92-010-0	QR5	328/95-003-0	QR5	333/90-027-0	QP2	335/88-003-0	QP2
327/92-012-0	QP4	328/95-007-0	QL4	333/91-006-1	QR6	335/88-004-0	QP4
327/92-018-0	D1	331/ 89-001-0	QR6	333/93-004-0	QL4	335/88-008-0	QP2
327/92-027-0[c]	B1	331/88-008-1	QR5	333/93-009-3	QP2	335/89-003-0	P1
327/92-027-0[c]	B1	331/89-003-0	QR7	333/93-013-0	QR8	335/89-005-0	QR5
327/93-002-0	QR3	331/89-008-0	QL5	333/93-020-0	QR0	335/90-007-0	QR6
327/93-003-0	QR5	331/89-009-0	QR8	333/95-010-0	QG9	335/91-003-0	QP2
327/94-008-0	P1	331/89-011-1	QR5	333/95-013-1	QP2	335/91-005-0	QP2
327/94-011-0	QR6	331/90-002-0	QP5	334/87-001-1	QR5	335/91-006-0	QR8
327/94-014-0	QR5	331/90-004-0	QR8	334/87-002-0	QR7	335/92-006-0	QR5
327/95-008-0	QC4	331/90-014-0	QR5	334/87-012-0	QR5	335/93-007-0[b]	QL4
327/95-010-0	QR5	331/90-015-0	D1	334/87-013-0	QR3	335/93-007-0[b]	QL4
327/95-017-0	QP2	331/90-016-0	QL5	334/88-007-0	QR2	335/93-007-0[b]	QL4
327/95-019-0	QR6	331/90-019-0	QP2	334/88-008-0	QP5	335/94-001-0	QP2
328/88-023-1	QP4	331/91-001-0	K1	334/88-009-0	QP2	335/94-003-0	QR5
328/88-024-0	QR8	331/91-003-0	QR7	334/89-001-0	QP2	335/94-004-0	QR7
328/88-027-1	QP2	331/91-005-1	QL5	334/89-002-0	QP2	335/94-005-0	QR5
328/88-028-0[b]	QP5	331/92-013-1	QR8	334/89-007-0	QR7	335/94-007-0	H1
328/88-028-0[b]	QR7	331/92-018-1	QL4	334/89-018-0	QR3	335/95-003-1	QR5
328/89-005-0[b]	QP2	331/93-010-0	QR4	334/90-007-0	QP2	335/95-010-0	QP2
328/89-005-0[b]	QP2	331/94-010-0	QR6	334/91-022-0	QP5	336/87-002-0	QP2
328/89-005-0[b]	QP2	331/95-005-0	QP2	334/91-023-1	QP2	336/87-007-1	QR5
328/89-008-0	QR3	333/87-008-0	P1	334/92-009-0	QR2	336/87-009-2	QP2
328/90-008-0	QR8	333/87-012-0	QR5	334/93-013-0	QR5	336/87-011-0	QP2
328/90-017-0	QR1	333/87-017-0	QP2	334/94-005-0[c]	H1	336/87-012-0	QP2
328/91-006-0	QL5	333/87-018-0	QR4	334/94-005-0[c]	QR2	336/88-011-1	C1

a Reserved

b One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
336/90-006-0	QP2	341/87-002-0	QR5	344/88-028-0	QR8	348/91-010-0	QR5
336/90-012-0	QR8	341/87-008-0	L3	344/88-043-0	QP5	348/92-008-0	QP2
336/91-001-1	QR5	341/87-011-0	QR8	344/89-006-1	QR3	348/95-001-0	QR5
336/91-004-0	P1	341/87-017-0	P1	344/89-017-1	QR8	348/95-005-0	QL5
336/91-012-1	K2	341/87-031-1	QR5	344/90-034-0	QP2	352/87-046-0	QC4
336/93-004-2[b]	QP2	341/87-035-0	QR4	344/91-004-0	QR5	352/87-048-2	QR5
336/93-004-2[b]	QP2	341/87-056-0	QP5	344/92-014-0	QP5	352/88-012-1	QR4
336/93-012-1	QL4	341/88-004-0	P1	344/92-020-1	QP2	352/91-009-0	QR5
336/93-013-0	QR0	341/88-019-1	QR3	344/92-027-0	QP4	352/93-011-0	QC5
336/93-019-0	QR7	341/88-020-0	QR4	344/92-028-0	QP4	352/94-001-0	QL4
336/94-009-1	QR3	341/88-021-1	QR0	346/87-001-0	P1	352/95-002-1[c]	QR5
336/95-032-0	K1	341/88-030-0	QR5	346/87-006-0	QP2	352/95-002-1[c]	QR5
338/87-004-0	QR3	341/89-006-0	QR5	346/87-010-1	QR5	352/95-006-0	QG9
338/87-015-1	QR5	341/89-007-0	QR6	346/87-011-0	QP5	352/95-008-0	G2
338/87-017-1	F1	341/89-036-0	QL5	346/87-015-0	D1	353/89-013-0	QR5
338/87-020-0	QR5	341/89-038-1	H1	346/88-028-0	QR4	353/90-012-0	L2
338/88-002-0	QL4	341/90-003-2	QC4	346/89-003-1	QP2	353/90-015-0	QR9
338/88-005-0	QR5	341/90-011-0	QR8	346/89-005-0	L2	353/92-012-0	QR6
338/88-013-0	QR5	341/91-004-0	QR6	346/90-002-1	QR8	353/93-001-0	QR5
338/88-020-0	QP2	341/91-015-0	H1	346/90-016-1	QR3	353/93-005-0	QR5
338/89-005-0	QP2	341/92-002-0	QR6	346/92-002-1	QR8	353/94-010-1	C1
338/89-014-0	QR5	341/92-012-0	QP4	346/93-003-0	QP2	353/95-008-0	QP5
338/89-017-0	QR7	341/93-004-0	QL4	346/93-005-0	QP2	353/95-010-0	QR5
338/90-001-0	QP2	341/93-007-0	QP5	348/87-002-0	QL5	354/87-014-0	QG9
338/91-017-1	QL5	341/93-010-0	QR5	348/87-003-0	P1	354/87-017-0	QR8
338/95-001-0	QC5	341/93-013-0	QK4	348/87-004-0	QR8	354/87-034-0	QC4
339/90-010-0	QP2	341/93-014-1	QR5	348/87-010-0	P1	354/87-037-0	L2
339/91-009-0	QP2	341/95-005-0	QR4	348/88-021-0	QR5	354/87-039-0	QR5
339/92-001-0	QP2	341/95-006-0	QR5	348/89-006-0	QR7	354/87-047-0	G2
339/92-007-0	QL5	344/87-001-0	QP2	348/90-005-0	QP2	354/87-051-0	QR8
339/93-002-1	QR5	344/87-024-0	QP5	348/91-006-0	QR8	354/88-012-1	QL4
339/93-003-0	QR6	344/87-037-0	QR6	348/91-007-1	QR5	354/88-013-1	QP4
339/94-003-1	QP2	344/88-001-1	QR8	348/91-008-0	QR8	354/88-022-0	QR5
339/95-004-0	QR8	344/88-026-1	QR8	348/91-009-0	QR2	354/88-027-0	P1

a Reserved

b One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
354/88-029-0	QR5	362/91-001-0	QC5	366/87-008-0	QP4	368/95-002-0	QP5
354/89-017-0	D1	362/92-003-0	QR2	366/87-009-1	QC4	368/95-003-0	QR7
354/89-025-0	QR5	362/93-001-0	QR5	366/88-006-0	QR8	369/87-004-0	QR2
354/90-001-0	QR5	362/93-004-0	L2	366/88-008-0	QP2	369/87-009-0	QR8
354/90-003-0	H1	364/87-001-0	QR8	366/88-011-0	QR8	369/87-017-1	QR9
354/90-024-0	QL5	364/87-009-0	QR7	366/88-017-0	QP3	369/87-021-0	D1
354/90-028-1	QR5	364/89-007-0	P1	366/88-018-0	QR8	369/87-036-0	QR8
354/91-005-0	QP2	364/89-008-0	QR5	366/88-020-0	QP4	369/88-001-1	QR5
354/91-008-0	QP2	364/89-010-0	P1	366/88-024-0	QR6	369/88-005-1	QR8
354/92-013-0	QR6	364/89-012-0	QR5	366/89-005-0	P1	369/88-007-1	QP2
354/93-004-0	QR5	364/89-013-0	QP2	366/90-001-1	L1	369/88-013-1	QR3
354/93-012-0	QR6	364/89-015-0	QR5	366/90-003-0	QR8	369/89-004-0	F1
354/94-007-0	QP2	364/90-001-0	P1	366/91-004-0	QR5	369/89-022-1	QR8
354/94-011-0	QR8	364/91-001-0	QR3	366/91-005-0	QP5	369/90-001-0	P1
354/94-012-0	QL4	364/91-002-0	P1	366/92-009-0	P1	369/90-027-0	QR8
354/94-014-0	QR5	364/91-004-0	L2	366/92-015-0	QR6	369/90-032-0	QR5
354/94-015-0	QR5	364/91-005-0	QR3	366/92-026-0	QR5	369/91-001-0	B1
354/95-005-0	QR6	364/92-001-0	QR6	366/93-005-0	QR6	369/91-004-0	QR8
361/87-001-0	QP2	364/92-002-0	QR8	366/94-007-0	QR8	369/92-008-0	QP2
361/87-004-1	QP2	364/92-005-0	QR5	366/95-001-0	QP5	369/92-009-0	QR8
361/87-031-1	P1	364/92-006-0	QR8	366/95-003-0 [b]	QL4	369/93-009-0	QR5
361/89-019-0	QR6	364/92-007-1	QP2	366/95-003-0 [b]	L2	369/94-004-0	QR5
361/90-016-1	QR5	364/92-008-0	QR3	368/87-007-0	QR5	369/95-001-1	QP2
361/91-003-0	QR8	364/92-010-0	QL4	368/87-008-0	QR5	369/95-005-0	QL5
361/91-007-1	QR6	364/93-004-0	QP5	368/88-011-0	G1	369/95-006-0	QR2
361/92-008-0	QP2	364/94-001-0	L2	368/88-020-0	QR9	370/87-003-0	QP3
361/92-012-0	QR2	364/94-003-0	QR5	368/89-006-0	K1	370/87-016-1	QR5
362/87-011-2	QC5	364/94-004-0	QR5	368/89-024-0	QP5	370/87-019-0	QP2
362/87-017-0	QL4	364/95-005-0[b]	P1	368/90-005-0	QR6	370/87-021-0	QR5
362/88-002-1	QR9	364/95-005-0[b]	P1	368/90-014-1	QR8	370/88-001-0	QP2
362/89-001-3	QC5	364/95-008-0	QR5	368/90-019-0	QL5	370/88-008-0	QP2
362/89-006-1	QR5	366/87-003-0	QR9	368/90-020-0	QL4	370/89-001-0	QR3
362/90-002-1	L1	366/87-006-1	QC4	368/91-005-0	QR2	370/89-002-0	QP2

a. Reserved.

b One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
370/89-003-1	QP2	374/88-012-0	QR6	382/91-019-0	QR9	389/93-007-1	QR3
370/90-008-0	QR3	374/89-011-1	QR8	382/91-022-0	QR9	389/93-008-0	QL4
370/91-007-0	QR3	374/90-001-1	QR8	382/93-002-0	QP5	389/94-003-0	QR8
370/91-010-1	QP2	374/90-010-0	QR5	382/94-007-0	QR1	389/95-002-0	QP2
370/91-012-1	QR3	374/91-010-0	QR6	382/95-002-0	QR5	395/87-015-0	QP2
370/92-004-0	QP5	374/91-012-0	QR5	387/87-013-0	QP5	395/87-021-0	QR5
370/92-006-0	QP3	374/91-014-0	QR5	387/88-006-0	QR5	395/87-024-0	QR8
370/92-007-0	QP2	374/92-004-0	QR6	387/88-010-0	QR5	395/87-027-0	QC4
370/92-009-0	QP2	374/92-012-0	QR5	387/89-001-0	D1	395/88-002-0	QR8
370/92-010-0	QR5	374/92-016-1	D1	387/89-002-1	QP5	395/88-006-0	QL5
370/93-001-0	QP2	374/94-001-0	QR5	387/89-005-0	QP4	395/88-007-1	QR8
370/93-002-0	QP2	374/94-004-0	C2	387/89-027-0	QR5	395/88-009-1	QR8
370/93-008-0	QR3	374/94-006-0	QR4	387/91-008-0	QC4	395/89-006-0	QR5
373/87-003-0	QR5	374/94-008-1	QR5	387/92-017-0	QR5	395/89-011-1	QG10
373/87-005-1	QR5	382/87-007-1	QR5	387/93-008-1	QR5	395/89-012-0	QR5
373/87-014-0	H1	382/87-008-0	QP5	388/87-006-0	L1	395/89-015-2	QG10
373/87-022-0	QP2	382/87-012-1[b]	QR8	388/90-002-0	QR5	395/89-020-0	QR5
373/87-032-0	QP5	382/87-012-1[b]	QR3	388/90-005-0	QP5	395/92-003-0	QR8
373/87-038-0	QP2	382/87-016-0	QP2	388/91-012-0	QR5	395/92-004-1	QP2
373/89-009-1	QR5	382/87-020-0	QL4	388/92-001-0	C1	395/93-001-0	QR4
373/90-006-0	QR5	382/87-028-0	QL5	388/94-002-0	QR5	397/87-002-0	P1
373/90-010-0	QR5	382/88-001-0	QR7	388/95-005-0	QR5	397/87-018-0[b]	QR5
373/91-006-0	QP2	382/88-002-0	QR6	389/87-001-0	QR2	397/87-018-0[b]	QR5
373/92-003-0	L2	382/88-016-0	P1	389/87-002-0	QP2	397/87-019-0	QR8
373/93-002-0	G2	382/88-033-0	QR7	389/87-003-0	P1	397/87-020-0	QC4
373/93-011-0	QR6	382/89-013-0	QP2	389/87-004-0	QR5	397/87-022-0	QC5
373/93-015-0	QP2	382/89-017-1	QR3	389/87-007-1	QR5	397/88-003-0	L1
373/94-010-1	QP2	382/89-024-1	QP2	389/89-005-0	QP5	397/89-002-0	QR5
373/94-011-2	QP5	382/90-002-0	QR3	389/89-007-0	QR3	397/89-028-0	QR5
373/94-015-0	L1	382/90-003-1	QR2	389/90-001-0	QP3	397/89-031-0	QP2
373/95-014-0	QC4	382/90-012-0	H1	389/92-001-1	QR3	397/89-035-0	QR8
373/95-016-0	QP2	382/91-010-0	QR8	389/92-004-0	QP2	397/90-021-0	QR6
374/87-014-0	QR5	382/91-011-1	QR5	389/92-005-0	QR5	397/90-031-0	QR5
374/88-003-0	QR4	382/91-013-1	QP5	389/92-006-0	H1	397/91-032-0	QP5

a Reserved

b One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
397/91-035-0	QL6	400/88-032-0	QP4	410/89-040-0	QR4	412/88-007-1	QR2
397/92-033-0[b]	G2	400/89-001-2	L2	410/90-009-0	D1	412/88-009-0	QR3
397/92-033-0[b]	G2	400/89-003-0	QP2	410/90-013-1	QR5	412/88-013-0	QR8
397/92-037-3	QR3	400/89-004-0	QL4	410/91-017-1	QC5	412/89-003-0	QP5
397/93-002-1	QP2	400/89-005-0	QP2	410/91-022-0	QR5	412/90-008-0	QR5
397/93-006-0	QR6	400/89-006-0	QP2	410/91-023-0	P1	412/91-005-0	QR5
397/93-007-1	QP2	400/89-017-1	H1	410/92-017-0	QP4	412/93-002-1	QR9
397/93-027-0	L1	400/91-010-0	QR8	410/92-022-0	QR7	412/95-006-0	QR5
397/94-008-0	QR6	400/92-007-0	QL6	410/93-012-0	QR5	413/87-006-1	QR4
397/95-002-0	QR5	400/92-009-0	QC5	410/94-001-1	QR5	413/87-013-0	QP2
397/95-004-0	QR5	400/92-010-0	QL6	410/94-007-0	L2	413/87-015-0	QP2
397/95-006-1	QR5	400/95-010-0	QR5	410/95-003-0	QR6	413/87-026-0	QP2
400/87-004-0	QR8	400/95-011-1	QR9	410/95-005-1	QR5	413/87-028-0	QR7
400/87-005-0	QP3	410/87-031-1	QP5	410/95-007-0	QR6	413/89-008-1	QL5
400/87-008-0	P1	410/87-033-0	QR8	410/95-008-0	QR6	413/89-017-0	QP2
400/87-012-0	QR5	410/87-043-0	QR5	412/87-012-0[b]	QR3	413/89-022-0	QP2
400/87-013-0	P1	410/87-058-0	QP5	412/87-012-0[b]	QR3	413/91-013-1	QR2
400/87-017-0	P1	410/87-064-0	L2	412/87-014-0	QP2	413/91-015-0	QP4
400/87-018-0	P1	410/87-081-0	L2	412/87-015-0	QR7	413/91-019-0	P1
400/87-019-0	QP3	410/88-001-0	D1	412/87-018-1	QC4	413/91-021-0	QR5
400/87-021-0	QL4	410/88-014-0	P1	412/87-019-0	QR5	413/93-008-0	QP5
400/87-024-0	QP4	410/88-017-0	QP5	412/87-020-1	QR7	413/94-001-0	L2
400/87-025-0	QP4	410/88-019-0	QP5	412/87-023-0	QP5	414/87-002-1	QP2
400/87-031-0	QP4	410/88-025-0	QP2	412/87-024-0	QR7	414/87-007-1	P1
400/87-035-0	QR2	410/88-026-0	QR4	412/87-026-0	QR7	414/87-010-0	QG9
400/87-037-0	P1	410/88-028-0	QR6	412/87-028-0	QR5	414/87-018-0	QR6
400/87-038-0	QR5	410/88-039-1	QR5	412/87-029-0	QP5	414/87-019-0	QP2
400/87-041-0	D1	410/88-051-0	QR6	412/87-030-2	H1	414/87-021-2	QP3
400/87-042-0	QP2	410/89-009-0	QR4	412/87-032-1	QR7	414/87-025-0	P1
400/87-062-0	QR7	410/89-014-0	QR5	412/87-034-0	QP2	414/87-027-1	QP2
400/87-063-0	QP3	410/89-024-0	QR6	412/87-035-0	QP4	414/87-029-0	QR5
400/88-007-0	QP2	410/89-035-0	L2	412/87-036-0	QR5	414/88-012-0	QP5
400/88-028-0	QP3	410/89-036-0	QR4	412/88-002-1	QC5	414/88-019-1	QP2

a Reserved

b One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
414/88-020-1	QP2	416/89-019-0	QC5	423/88-023-0	QL5	424/87-033-0	QP5
414/88-021-1	QP2	416/90-011-0	QP5	423/88-024-0	QL4	424/87-034-0	QP2
414/88-022-0	QR3	416/90-017-1	QR5	423/88-028-0	QR3	424/87-035-0	QP2
414/88-023-0	QP2	416/90-026-0	QR3	423/89-008-0	QL4	424/87-041-0	QR5
414/88-025-0	QL5	416/90-028-0	D1	423/89-009-1	QR3	424/87-047-0[b]	QR5
414/88-028-0	QR5	416/90-029-0	P1	423/90-005-0	QP2	424/87-047-0[b]	QR5
414/88-031-0	P1	416/91-002-1	QR5	423/90-009-0	QR5	424/87-050-0	QR8
414/89-001-0	QP2	416/91-004-0	P1	423/90-011-0	QL4	424/87-063-0	QR5
414/89-002-0	P1	416/91-005-1	QR5	423/90-013-1	QL4	424/87-066-0	QR8
414/89-003-1	QR9	416/91-007-0	QR5	423/90-014-0	QL4	424/88-001-0	QR2
414/90-013-0	QP2	416/91-010-0	QR8	423/90-019-1	QR3	424/88-006-0	QR5
414/91-008-1	QR2	416/91-012-0	QR8	423/90-030-2	K2	424/88-008-0	QR5
414/92-001-0	QR5	416/92-010-1	QR8	423/91-014-1	QR5	424/88-013-0	QP2
414/92-006-0	QP2	416/92-013-0	QR4	423/92-011-0	QL4	424/88-022-1	QR5
414/93-003-1	QL5	416/92-017-2	QR5	423/92-027-0	QR5	424/88-024-0	QR5
414/94-003-0	QP2	416/93-008-0	QR7	423/92-029-0	QR5	424/88-025-2	QR8
414/94-005-0	QR8	416/94-011-0	QR8	423/93-004-1	QR5	424/88-043-0	D1
414/94-006-0	QL5	416/95-004-1	QR8	423/94-011-0	QL5	424/88-044-0	QP2
414/94-007-0	QR8	416/95-007-0	QR8	424/87-008-0	QR3	424/89-005-0	QP2
414/95-001-0	QL5	416/95-008-0	QL6	424/87-009-0[b]	QR7	424/89-012-0	QP2
414/95-004-0	QR7	416/95-010-0	QR5	424/87-009-0[b]	QR7	424/89-016-1[b]	QP2
414/95-005-0	P1	416/95-011-0	QP2	424/87-010-0[b]	QR7	424/89-016-1[b]	QP2
416/87-009-2	L2	423/87-001-0	QL4	424/87-010-0[b]	QR7	424/89-018-0	QL5
416/87-012-0	QR5	423/87-002-0	QR8	424/87-011-0	QP4	424/90-001-0	QL5
416/88-002-0	QR5	423/87-008-0	QP2	424/87-012-0	QP2	424/90-011-0	QP2
416/88-006-0	QP3	423/87-020-0	QP2	424/87-013-0	QP2	424/90-016-0	QC5
416/88-010-0	QR3	423/87-021-0	P1	424/87-014-0	QP5	424/90-023-0	QC5
416/88-012-2	QR8	423/87-025-0	QP2	424/87-018-0[b]	QR8	424/92-008-0	QP2
416/88-013-0	D1	423/87-026-0	QR8	424/87-018-0[b]	QR8	424/93-008-0	QR5
416/88-019-1	QR9	423/87-027-0	L1	424/87-025-1	QR7	424/93-009-0	QR8
416/89-006-0	QP5	423/87-031-1	QR5	424/87-027-0	QL5	424/94-001-0	QR9
416/89-010-0	QR8	423/87-034-0	QP2	424/87-029-0	QP2	424/95-002-0[c]	QR8
416/89-012-0	QL6	423/88-009-0	QP5	424/87-030-0	QR5	424/95-002-0[c]	QR8
416/89-016-0	QR8	423/88-014-0	QL4	424/87-032-0	QR4	425/89-019-0	QR5

a Reserved

b One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
425/89-020-0	QR8	440/88-020-0	QR4	445/90-017-0	QP2	446/94-014-0	QR6
425/89-021-1	QP2	440/88-023-0	QR8	445/90-023-0	QP2	446/95-004-0	QP2
425/89-024-0	QR8	440/88-024-0	QR4	445/90-025-0	QP5	454/87-017-1[b]	QR3
425/89-027-0	QR3	440/88-026-0	QR5	445/90-027-0	QP5	454/87-017-1[b]	QR3
425/89-029-0	QP2	440/90-001-0	P1	445/90-028-0	QR3	454/87-018-1	QP2
425/89-031-0	QR5	440/91-027-0	J1	445/90-029-0	QR5	454/87-019-2	L1
425/90-002-0	QR5	440/92-017-0	QP2	445/90-030-0	P1	454/88-002-0	QR3
425/90-007-0	QL5	440/93-010-0	QL4	445/91-002-0	QR5	454/88-004-0	QP2
425/90-008-0	QL5	440/93-015-0	QR6	445/91-004-0	QR2	454/88-005-1	QR5
425/90-009-0	QP5	440/94-002-0	QR6	445/91-008-0	QR7	454/89-002-0	QP5
425/91-005-0	QP2	440/95-005-0	QR5	445/91-020-0	QR5	454/90-002-0	QR8
425/91-006-0	QR8	440/95-007-0	QP2	445/91-023-0	QR5	454/90-006-0	QP2
425/91-007-0	QR5	440/95-008-0	QC4	445/92-001-0	QR5	454/90-011-1	QR3
425/92-002-0	QL5	443/89-008-0	QR6	445/92-009-0	QR8	454/90-014-0	P1
425/92-010-0	QR5	443/90-015-1	QR5	445/92-014-0	P1	454/92-001-0	QR5
425/93-004-0	QP2	443/90-018-0	QR8	445/92-019-0	P1	454/94-009-1	QR8
425/93-006-0	QR2	443/90-022-0	QR5	445/92-022-0	QP2	455/87-001-1	QR8
425/94-001-0	QR5	443/90-025-0	QP2	445/92-025-0	QR3	455/87-002-1[b]	QR7
425/94-002-0	QR5	443/91-001-0	QR5	445/93-001-1	QR8	455/87-002-1[b]	QR7
440/87-007-0	QR8	443/91-002-0	QC5	445/93-002-0	QR8	455/87-005-0	QR5
440/87-012-0	P1	443/91-006-0	QR5	445/93-007-0	QR6	455/87-006-1	QR8
440/87-027-1	K1	443/91-008-0	QR5	445/93-011-0	QR5	455/87-007-1	QC4
440/87-030-0	QP3	443/91-009-0	QR2	445/94-001-0	QR5	455/87-009-1	QP2
440/87-035-0	QL6	443/92-017-0	QP5	445/94-006-0	QR5	455/87-010-0	QR6
440/87-037-0	P1	443/92-024-0	QR8	445/95-002-0[c]	QR3	455/87-011-1	L3
440/87-042-0	L1	443/92-025-0	QL4	445/95-002-0[c]	QR3	455/87-018-0	QP2
440/87-045-0	QR5	443/93-001-0	P1	445/95-003-1	QC5	455/87-019-1	QP5
440/87-064-0	QP5	443/93-003-0	QR5	445/95-004-1	QP3	455/88-001-1	QP2
440/87-072-0	P1	443/93-009-1	QL5	445/95-007-0	QP2	455/88-004-1	QP2
440/87-073-1	QR6	443/93-012-0	QR8	446/93-003-0	QP2	455/88-006-0	QR3
440/88-001-1	QP5	443/93-018-0	QR7	446/93-005-0	QR5	455/88-008-0	P1
440/88-012-0	QC4	443/94-001-1	QL5	446/93-011-0	QP2	455/88-009-0	QR7
440/88-015-0	QR7	443/95-002-0	QC5	446/94-003-0	QR6	455/88-012-0	QR8
440/88-017-1	QR8	445/90-009-0	QR7	446/94-010-0	QR5	455/90-001-0	QR8
		445/90-013-0	P1	446/94-012-0	QR6	455/90-010-1	K1

a Reserved.

b One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
455/91-005-0	QP2	457/88-022-0	QR4	458/93-024-2	QR5	482/87-017-0[b]	QR3
455/92-003-1	QP2	457/88-026-0	QR9	458/94-023-1	QR8	482/87-022-1	QR7
455/93-003-0	QR7	457/88-028-0	QP5	458/94-028-0	QR6	482/87-027-0	QP2
455/93-008-1	QR9	457/88-029-1	QP3	458/94-030-0	QR9	482/87-030-0	QP2
455/94-002-0	QR3	457/88-031-0	QR3	458/95-012-0	QR6	482/87-037-0	QR5
456/87-027-0	QR4	457/89-002-0	QR5	461/87-017-0	D1	482/87-041-0	QR3
456/87-032-0	QR3	457/89-004-0	QR3	461/87-025-0	QP5	482/89-002-0	QR5
456/87-035-0	QR8	457/90-010-0	QP2	461/87-029-0	QP5	482/89-004-0	QR9
456/87-050-0	QR5	457/91-003-0	QR3	461/87-036-0	QR4	482/90-001-0	QR2
456/87-052-0	QR5	457/91-006-0	QR5	461/87-042-0	QR4	482/90-011-0	QP5
456/87-057-1	QR5	457/92-001-0	QR5	461/87-043-0	QR5	482/90-012-0	QR7
456/87-060-0	QP4	457/92-002-0	QP2	461/87-050-0	L2	482/90-013-0	QR5
456/88-016-0	QR3	457/92-006-0	QP2	461/87-055-0	QP5	482/92-002-0	QC4
456/88-022-0	B1	457/92-007-0	QR5	461/87-060-0	QP5	482/92-016-0	QR5
456/88-023-0	QR8	457/93-007-0	QP2	461/88-017-1	QP2	482/95-001-0	QR8
456/88-025-0[c]	D1	457/94-003-0	QR5	461/88-019-0	QL4	483/87-032-0	QR7
456/88-025-0[c]	D1	457/94-005-0	QP2	461/88-028-0	H1	483/88-001-0	QP2
456/89-004-0	QR7	458/87-002-0	QG9	461/89-022-0[b]	QP2	483/88-004-1	QR5
456/89-006-0[c]	QR3	458/87-003-0	QP5	461/89-022-0[b]	QP5	483/88-006-0	QP2
456/89-006-0[c]	QR3	458/87-012-1	QC5	461/89-028-0	QR5	483/88-007-0	QR5
456/90-001-0	QR5	458/88-002-0	QR8	461/89-029-0	QL6	483/88-010-0	QP2
456/90-008-0	QR3	458/88-003-0	QR0	461/89-032-0	QP5	483/89-006-0	QR8
456/90-018-0	QR9	458/88-007-0	QR5	461/90-012-0	QR6	483/89-008-0	QR3
456/90-021-0	QP2	458/88-018-4	QR5	461/90-013-0	QR5	483/90-005-0	QR5
456/90-023-0	QR5	458/88-021-1	QR5	461/91-006-0	L2	483/90-007-0	L1
456/91-012-0	QP2	458/89-007-0	QP5	461/91-008-0	QR6	483/90-016-0	QR5
456/93-001-0	QR8	458/89-008-0	QR5	461/92-001-0	QR5	483/90-017-0	QP2
456/94-012-0	QR9	458/89-035-0	QR8	461/92-002-1	QP2	483/91-006-0	QC4
456/95-004-0	QC4	458/89-042-0	QR5	461/92-010-0	QP2	483/92-002-0	QR8
457/88-012-1	QR5	458/90-008-0	QR5	461/93-006-0	QR6	483/92-007-0	QP2
457/88-013-0	QP2	458/90-014-0	QR5	461/93-007-0	L2	483/92-010-0	QR5
457/88-014-1	QP5	458/90-047-0	QR5	461/95-001-0	QR6	483/95-001-0	QR3
457/88-016-0	QP2	458/92-001-2	QR8	461/95-005-0	QR6	483/95-004-0	QR2
457/88-018-0	QR8	458/92-005-0	QR5	482/87-002-0	QR9	483/95-005-0	QL4
457/88-019-0	D1	458/92-026-0	QR4	482/87-004-0	QR5	498/88-026-0	QR7
457/88-020-0	QP3	458/93-017-0	L1	482/87-017-0[b]	QR3	498/88-045-0	QR8

a Reserved.

b. One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units.

Table D-6. (continued).

LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code	LER Number	IPF Code
498/88-048-0	QR5	499/89-017-0	QR5	528/87-014-0	QP2	529/89-010-0	QR8
498/88-049-0	QR5	499/89-019-0	QP2	528/87-018-1	QG9	529/90-001-0	QR6
498/89-001-0	QR5	499/89-020-0	P1	528/88-010-1	H1	529/91-003-0	QR6
498/89-005-0	H1	499/89-021-0	QP2	528/88-011-0	QR7	529/91-004-1	QR5
498/89-015-1	QR5	499/89-022-0	QR7	528/88-015-0	QR7	529/92-001-1	QP2
498/90-005-0	QP2	499/89-023-0	QR5	528/88-021-0	QR5	529/92-002-1	QC5
498/90-006-0	QP2	499/89-026-0	QR3	528/88-024-0	QP2	529/92-006-0	QR8
498/90-014-0	QR2	499/90-002-0	QR8	528/89-004-0	QR8	529/93-001-2	F1
498/90-015-0	QR5	499/90-004-0	QP2	528/90-006-0	QR5	529/93-004-0	QC5
498/90-016-0	QR7	499/90-005-0	QR5	528/91-009-0	QP5	529/94-002-0	QR2
498/90-020-0	QR8	499/90-013-0	QR8	528/91-010-0[c]	QR4	529/94-006-0	QR8
498/90-023-0	QP2	499/91-001-0	QP2	528/91-010-0[c]	QR4	529/95-005-0	QP2
498/90-025-1	QR5	499/91-003-0	QR5	528/92-012-0	QR5	530/87-004-0	QR3
498/91-012-1	QR7	499/91-004-0	QR5	528/92-016-0	QR5	530/89-001-3	QR5
498/91-021-0	QR2	499/91-007-1	QR5	528/93-001-0	QR5	530/90-004-0	QR3
498/91-022-0	QR5	499/91-010-1	QR1	528/95-008-0	QP2	530/90-007-0	QR8
498/92-003-1	QR8	499/92-001-0	QR3	528/95-012-0	L2	530/91-003-1	QR9
498/94-009-1	QP2	499/92-003-0	QP2	528/95-014-0	QC5	530/91-008-0	QR5
498/94-015-1	QP2	499/92-010-0	QP2	529/87-004-1	QR8	530/92-001-0	D1
498/95-001-0	QP2	499/93-001-1	QR5	529/87-008-0	P1	530/93-001-0	QP2
498/95-009-0	QR2	499/93-004-0	QP2	529/87-010-0	QP2	530/93-004-0	QR3
498/95-013-0	QR5	499/94-007-0	QR5	529/87-019-0	QR7	530/94-005-0	QP2
499/89-009-0	QR2	499/95-003-0	QR8	529/88-014-0	QP2	530/94-007-0	QP5
499/89-013-0	QR8	499/95-008-0	QR5	529/89-003-1	QP2		
499/89-016-0	QR8	528/87-003-0	QR5	529/89-009-1	QR2		

a. Reserved

b One LER that describes multiple reactor trip events from one plant unit

c. One LER that describes reactor trip events from multiple plant units

Table D-7. Functional impact categories with assigned LERs based on all the operating experience from 1987 through 1995.

Loss of Offsite Power—B1
33
029/91-002-0
219/89-015-0
219/92-005-0[a]
237/90-002-2[a]
249/89-001-1
255/87-024-0
261/92-017-0
270/92-004-0[a]
271/91-009-1[a]
293/93-004-0[a]
293/93-022-0
301/89-002-0[a]
302/89-023-0
302/92-001-0
304/91-002-1[a]
309/88-006-0
315/91-004-0[a]
317/87-012-1[c]
317/87-012-1[c]
323/88-008-0[a]
324/89-009-1[a]
327/92-027-0[c]
327/92-027-0[c]
334/93-013-0
336/88-011-1[a]
369/91-001-0
370/93-008-0[a]
373/93-015-0
395/89-012-0
412/87-036-0
443/91-008-0
455/87-019-1
456/88-022-0

Loss of Vital Medium Voltage ac Bus —C1
13
219/90-005-0
263/91-019-0[a]
265/87-013-0[a]
277/92-010-0[a]
280/89-044-0
293/93-004-0[a]
304/91-002-1[a]
318/94-001-1
336/88-011-1[a]
353/94-010-1
361/90-016-1
388/92-001-0
483/89-008-0

Loss of Vital Low Voltage ac Bus —C2
3
293/93-004-0[a]
374/94-004-0
425/90-002-0

Loss of Vital dc Bus—C3
1
321/91-017-0[a]

Loss of Instrument or Control Air System —D1
36
219/92-005-0[a]
237/94-005-2[a]
245/87-038-0
247/89-002-0
249/93-004-0
265/88-026-0
270/92-004-0[a]
271/91-009-0[a]
271/91-014-0
280/90-006-0[a]
285/90-026-1
298/89-026-0[a]
301/89-002-0[a]
317/87-003-0[a]
327/92-018-0
331/90-015-0
341/90-003-2[a]
346/87-015-0
354/89-017-0
369/87-021-0[a]
373/89-009-1
374/92-016-1
387/89-001-0[a]
400/87-041-0[a]
410/88-001-0
410/90-009-0[a]
416/88-013-0
416/90-028-0[a]
416/91-005-1[a]
416/91-007-0[a]
424/88-043-0
456/88-025-0[a,c]
456/88-025-0[a,c]
457/88-019-0
461/87-017-0
530/92-001-0

Total Loss of Service Water—E1
None

Partial Loss of Service Water —E2
6
245/90-016-1
271/91-009-1[a]
317/87-003-0[a]
346/87-011-0
416/89-019-0
423/90-011-0[a]

Steam Generator Tube Rupture—F1
3
338/87-017-1
369/89-004-0
529/93-001-2

Very Small LOCA/Leak—G1
4
287/91-008-0
338/89-005-0
368/88-011-0
369/87-017-1[a]

Stuck Open: 1 Safety/Relief Valve—G2
12
237/90-006-1
254/89-004-0
265/91-012-0
265/93-006-0
285/92-023-0
317/94-007-0
324/90-004-3
352/95-008-0
354/87-047-0
373/93-002-0
397/92-033-0[b]
397/92-033-0[b]

Small Pipe Break LOCA—G3
None

Stuck Open: Pressurizer PORV—G4
None

a. One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes

b One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-7. (continued).

Stuck Open: 2 or more Safety/Relief Valves—G5

None

Medium Pipe Break LOCA—G6

None

Large Pipe Break LOCA—G7

None

Reactor Coolant Pump Seal LOCA (PWR) —G8

None

Fire—H1

39

213/94-018-1
219/92-005-0[a]
237/90-002-2[a]
269/89-002-0
275/90-005-0
295/94-005-0
295/94-010-0
298/89-026-0[a]
304/90-011-1
305/87-009-0
305/88-001-0
305/92-017-0
309/91-005-1
311/91-017-0
316/91-006-0
317/92-008-0
321/90-012-0
321/91-001-0
323/88-008-0[a]
323/89-010-0
334/94-005-0[c]
335/94-007-0
341/89-038-1
341/91-015-0
341/93-014-1
346/87-001-0[a]
354/90-003-0[a]
373/87-014-0
382/90-012-0
382/95-002-0
389/92-006-0
395/88-002-0
400/89-003-0
400/89-017-1
412/87-030-2
461/88-028-0
498/89-005-0
528/88-010-1
528/89-004-0

Flood—J1

2

440/91-027-0[a]
440/93-010-0[a]

Steam Line Break Outside Containment—K1

7

255/87-016-0
328/93-001-0
331/91-001-0
336/95-032-0
368/89-006-0[a]
440/87-027-1[a]
455/90-010-1

Feedwater Line Break—K2

2

336/91-012-1
423/90-030-2[a]

Steam Line Break Inside Containment—K3

None

Inadvertent Closure of All MSIVs—L1

109

220/87-014-0
220/90-026-0
237/87-032-0
237/89-019-1
237/90-001-0
237/91-004-1
237/91-024-0
237/94-005-2[a]
245/87-007-0
245/92-028-0
249/87-012-0
249/87-016-0
249/88-017-0
249/89-006-0
254/92-004-0
254/93-023-0
260/94-005-0
263/91-019-0[a]
265/87-011-0
265/88-001-0
265/88-005-0
265/92-001-0
265/93-013-0
265/94-006-0
269/94-002-0[a]
271/87-017-0
271/90-004-0
277/89-015-1
278/92-008-0
278/95-003-0
280/90-006-0[a]
281/91-007-1
293/89-011-0
293/92-018-0
293/93-022-0[a]
295/91-016-0
298/87-005-0
298/89-026-0[a]
298/90-011-0
302/88-024-0[a]
311/88-014-0
313/94-002-0
318/95-002-1
321/92-021-0
321/93-009-0
321/93-012-0
323/87-016-0
324/87-001-2
324/88-018-0
324/89-009-1[a]
324/90-009-0
325/87-017-1
325/91-018-0
325/95-015-1
331/89-008-0[a]
334/89-007-0
341/90-003-2[a]
348/89-006-0
353/90-015-0
354/88-027-0[a]
362/90-002-1
366/87-003-0
366/88-006-0
366/88-017-0[a]
366/89-005-0[a]
366/90-001-1
366/94-007-0
369/87-017-1[a]
370/93-008-0[a]
373/94-015-0
373/95-014-0
374/92-012-0
382/91-011-1
382/91-019-0
382/91-022-0
387/91-008-0
388/87-006-0
389/87-003-0[a]
397/87-002-0[a]
397/87-020-0
397/87-022-0[a]

a. One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes.

b. One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units.

Table D-7. (continued).

397/88-003-0
397/93-027-0
400/87-062-0
400/88-028-0[a]
400/95-011-1
410/88-014-0[a]
412/87-024-0
412/93-002-1
414/89-003-1
416/90-017-1
416/91-005-1[a]
416/95-008-0[a]
423/87-027-0
423/90-030-2[a]
440/87-012-0[a]
440/87-042-0
440/87-072-0[a]
443/94-001-1
454/87-019-2
456/90-018-0
456/94-012-0
458/93-017-0
458/94-030-0
482/87-002-0
483/90-007-0
498/88-049-0
530/89-001-3
530/94-007-0

Loss of Condenser Vacuum—L2

81

155/88-008-0
219/89-011-0
219/90-008-0
244/95-008-0
245/89-015-0
245/90-016-1
249/87-010-0
249/93-014-0
260/95-007-0
263/87-014-0
263/94-003-0
263/94-004-0
272/93-011-0
275/92-004-0
275/95-017-0
277/91-022-1
277/92-010-0[a]
278/90-008-0
278/92-005-0
278/93-004-0
278/95-001-0
293/89-023-0
304/90-010-0
305/92-020-1
315/89-001-0
315/95-003-0
316/87-004-0
316/92-007-0
316/94-005-0
318/92-003-0
321/93-001-0
324/88-001-7
325/95-011-0
328/95-007-0
331/92-018-1
338/88-002-0
341/91-004-0
341/93-004-0
346/89-005-0
353/90-012-0
354/87-037-0[a]
354/88-012-1[a]
362/87-017-0
362/93-004-0
364/91-004-0
364/92-010-0
364/94-001-0
366/95-003-0[b]
368/89-006-0[a]
373/92-003-0
387/89-001-0[a]
400/87-021-0[a]
400/89-001-2
400/89-004-0[a]
400/92-007-0
400/92-010-0
410/87-064-0
410/87-081-0
410/89-035-0
410/90-009-0[a]
410/94-007-0
413/94-001-0
416/87-009-2
416/89-012-0
416/95-008-0[a]
423/87-001-0
423/88-014-0
423/88-024-0
423/90-011-0[a]
423/90-013-1
423/90-014-0
440/87-027-1[a]
440/91-027-0[a]
440/93-010-0[a]
461/87-050-0
461/88-019-0
461/89-029-0
461/91-006-0
461/93-007-0
483/95-005-0
528/95-012-0

Turbine Bypass Unavailable—L3

10

293/93-004-0[a]
317/87-003-0[a]
325/90-017-0
331/89-008-0[a]
341/87-008-0
368/89-006-0[a]
370/89-002-0
382/90-003-1[a]
443/91-002-0
455/87-011-1

Interfacing System LOCA—N1

None

Total Loss of Feedwater Flow—P1

159

029/90-011-0
155/94-010-1
237/89-012-0
244/90-019-0
249/87-011-0
249/92-021-1
255/90-001-1
255/95-003-0
260/91-018-0
261/91-011-0
263/87-006-0
263/87-009-0
263/88-007-0
263/91-019-0[a]
265/87-013-0[a]
269/88-009-0
269/94-002-0[a]
270/89-004-0
270/94-002-0
270/94-005-0
272/93-002-0
275/90-002-0
275/95-015-0
277/89-033-0
278/87-002-0
278/93-002-0
278/94-005-0
281/93-006-0
286/87-002-0
286/88-001-0
287/91-007-0
287/92-003-0
287/94-002-0
287/94-003-0
293/94-005-0

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a. One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes.

b. One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-7. (continued).

298/87-003-0	344/92-028-0	395/87-027-0	416/90-028-0[a]
298/87-009-0	346/87-001-0[a]	397/87-002-0[a]	416/90-029-0
302/88-024-0[a]	348/87-003-0	397/87-022-0[a]	416/91-004-0
305/88-004-0	348/87-010-0	400/87-008-0	416/91-007-0[a]
305/91-010-0	348/91-010-0	400/87-013-0	423/87-021-0
309/91-006-0	352/91-009-0	400/87-017-0	424/90-016-0
311/90-029-1	354/87-037-0[a]	400/87-018-0	440/87-012-0[a]
311/93-002-0	354/88-012-1[a]	400/87-019-0	440/87-030-0
312/88-019-0	354/88-027-0[a]	400/87-021-0[a]	440/87-037-0
313/87-004-0	354/90-003-0[a]	400/87-024-0	440/87-072-0[a]
313/87-005-0	361/87-031-1	400/87-025-0	440/88-012-0
313/89-048-0	364/89-007-0	400/87-031-0	440/90-001-0
313/91-005-0	364/89-010-0	400/87-037-0	443/91-001-0
313/95-004-0	364/90-001-0	400/87-041-0[a]	443/92-017-0
315/91-004-0[a]	364/91-002-0	400/87-063-0	443/93-001-0
318/91-005-0	364/95-005-0[b]	400/88-028-0[a]	445/90-013-0
318/92-005-0	364/95-005-0[b]	400/88-032-0	445/90-030-0
321/88-013-0	366/87-008-0	400/89-004-0[a]	445/92-014-0
321/91-007-0	366/88-017-0[a]	410/88-014-0[a]	445/92-019-0
321/91-017-0[a]	366/89-005-0[a]	410/91-023-0	445/95-003-1
325/95-018-0	366/92-009-0	410/92-017-0	445/95-004-1
327/90-012-0	366/95-001-0	412/87-035-0	446/93-011-0
327/94-008-0	369/87-021-0[a]	413/91-015-0	454/90-014-0
331/89-008-0[a]	369/90-001-0	413/91-019-0	455/88-008-0
333/87-008-0	370/87-003-0	414/87-007-1	456/88-025-0[a,c]
335/87-016-0	370/92-006-0	414/87-021-2	456/88-025-0[a,c]
335/89-003-0	382/87-028-0	414/87-025-0	457/88-029-1
336/91-004-0	382/88-016-0	414/88-031-0	482/92-002-0
341/87-017-0	382/90-003-1[a]	414/89-002-0	499/89-020-0
341/88-004-0	389/87-003-0[a]	414/95-005-0	529/87-008-0
341/92-012-0	389/90-001-0	416/88-006-0	529/93-004-0

a. One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes

b. One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-8. LERs with assigned functional impact (FI) code based on all the operating experience from 1987 through 1995.

LER Number	FI Code	LER Number	FI Code	LER Number	FI Code	LER Number	FI Code
029/90-011-0	P1	245/92-028-0	L1	263/94-004-0	L2	275/95-015-0	P1
029/91-002-0	B1	247/89-002-0	D1	265/87-011-0	L1	275/95-017-0	L2
155/88-008-0	L2	249/87-010-0	L2	265/87-013-0[a]	C1	277/89-015-1	L1
155/94-010-1	P1	249/87-011-0	P1	265/87-013-0[a]	P1	277/89-033-0	P1
213/94-018-1	H1	249/87-012-0	L1	265/88-001-0	L1	277/91-022-1	L2
219/89-011-0	L2	249/87-016-0	L1	265/88-005-0	L1	277/92-010-0[a]	C1
219/89-015-0	P1	249/88-017-0	L1	265/88-026-0	D1	277/92-010-0[a]	L2
219/90-005-0	C1	249/89-001-1	B1	265/91-012-0	G2	278/87-002-0	P1
219/90-008-0	L2	249/89-006-0	L1	265/92-001-0	L1	278/90-008-0	L2
219/92-005-0[a]	B1	249/92-021-1	P1	265/93-006-0	G2	278/92-005-0	L2
219/92-005-0[a]	D1	249/93-004-0	D1	265/93-013-0	L1	278/92-008-0	L1
219/92-005-0[a]	H1	249/93-014-0	L2	265/94-006-0	L1	278/93-002-0	P1
220/87-014-0	L1	254/89-004-0	G2	269/88-009-0	P1	278/93-004-0	L2
220/90-026-0	L1	254/92-004-0	L1	269/89-002-0	H1	278/94-005-0	P1
237/87-032-0	L1	254/93-023-0	L1	269/94-002-0[a]	L1	278/95-001-0	L2
237/89-012-0	P1	255/87-016-0	K1	269/94-002-0[a]	P1	278/95-003-0	L1
237/89-019-1	L1	255/87-024-0	B1	270/89-004-0	P1	280/89-044-0	C1
237/90-001-0	L1	255/90-001-1	P1	270/92-004-0[a]	B1	280/90-006-0[a]	D1
237/90-002-2[a]	B1	255/95-003-0	P1	270/92-004-0[a]	D1	280/90-006-0[a]	L1
237/90-002-2[a]	H1	260/91-018-0	P1	270/94-002-0	P1	281/91-007-1	L1
237/90-006-1	G2	260/94-005-0	L1	270/94-005-0	P1	281/93-006-0	P1
237/91-004-1	L1	260/95-007-0	L2	271/87-017-0	L1	285/90-026-1	D1
237/91-024-0	L1	261/91-011-0	P1	271/90-004-0	L1	285/92-023-0	G2
237/94-005-2[a]	D1	261/92-017-0	B1	271/91-009-1[a]	D1	286/87-002-0	P1
237/94-005-2[a]	L1	263/87-006-0	P1	271/91-009-1[a]	B1	286/88-001-0	P1
244/90-019-0	P1	263/87-009-0	P1	271/91-009-1[a]	E2	287/91-007-0	P1
244/95-008-0	L2	263/87-014-0	L2	271/91-014-0	D1	287/91-008-0	G1
245/87-007-0	L1	263/88-007-0	P1	272/93-002-0	P1	287/92-003-0	P1
245/87-038-0	D1	263/91-019-0[a]	C1	272/93-011-0	L2	287/94-002-0	P1
245/89-015-0	L2	263/91-019-0[a]	L1	275/90-002-0	P1	287/94-003-0	P1
245/90-016-1[a]	E2	263/91-019-0[a]	P1	275/90-005-0	H1	293/89-011-0	L1
245/90-016-1[a]	L2	263/94-003-0	L2	275/92-004-0	L2	293/89-023-0	L2

a One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes

b One LER that describes multiple reactor trip events from one plant unit

c. One LER that describes reactor trip events from multiple plant units

Table D-8. (continued).

LER Number	FI Code	LER Number	FI Code	LER Number	FI Code	LER Number	FI Code
293/92-018-0	L1	309/88-006-0	B1	318/95-002-1	L1	328/95-007-0	L2
293/93-004-0[a]	B1	309/91-005-1	H1	321/88-013-0	P1	331/89-008-0[a]	L1
293/93-004-0[a]	C1	309/91-006-0	P1	321/90-012-0	H1	331/89-008-0[a]	L3
293/93-004-0[a]	C2	311/88-014-0	L1	321/91-001-0	H1	331/89-008-0[a]	P1
293/93-004-0[a]	L3	311/90-029-1	P1	321/91-007-0	P1	331/90-015-0	D1
293/93-022-0[a]	B1	311/91-017-0	H1	321/91-017-0[a]	C3	331/91-001-0	K1
293/93-022-0[a]	L1	311/93-002-0	P1	321/91-017-0[a]	P1	331/92-018-1	L2
293/94-005-0	P1	312/88-019-0	P1	321/92-021-0	L1	333/87-008-0	P1
295/91-016-0	L1	313/87-004-0	P1	321/93-001-0	L2	334/89-007-0	L1
295/94-005-0	H1	313/87-005-0	P1	321/93-009-0	L1	334/94-005-0[c]	H1
295/94-010-0	H1	313/89-002-0	N1	321/93-012-0	L1	335/87-016-0	P1
298/87-003-0	P1	313/89-048-0	P1	323/87-016-0	L1	335/89-003-0	P1
298/87-005-0	L1	313/91-005-0	P1	323/88-008-0[a]	B1	335/94-007-0	H1
298/87-009-0	P1	313/94-002-0	L1	323/88-008-0[a]	H1	336/88-011-1[a]	B1
298/89-026-0[a]	D1	313/95-004-0	P1	323/89-010-0	H1	336/88-011-1[a]	C1
298/89-026-0[a]	H1	315/89-001-0	L2	324/87-001-2	L1	336/91-004-0	P1
298/89-026-0[a]	L1	315/91-004-0[a]	B1	324/88-001-7	L2	336/91-012-1	K2
298/90-011-0	L1	315/91-004-0[a]	P1	324/88-018-0	L1	336/95-032-0	K1
301/89-002-0[a]	B1	315/95-003-0	L2	324/89-009-1[a]	B1	338/87-017-1	F1
301/89-002-0[a]	D1	316/87-004-0	L2	324/89-009-1[a]	L1	338/88-002-0	L2
302/88-024-0[a]	L1	316/91-006-0	H1	324/90-004-3	G2	338/89-005-0	G1
302/88-024-0[a]	P1	316/92-007-0	L2	324/90-009-0	L1	341/87-008-0	L3
302/89-023-0	B1	316/94-005-0	L2	325/87-017-1	L1	341/87-017-0	P1
302/92-001-0	B1	317/87-003-0[a]	E2	325/90-017-0	L3	341/88-004-0·	P1
304/90-010-0	L2	317/87-003-0[a]	D1	325/91-018-0	L1	341/89-038-1	H1
304/90-011-1	H1	317/87-003-0[a]	L3	325/95-011-0	L2	341/90-003-2[a]	D1
304/91-002-1[a]	B1	317/87-012-1[c]	B1	325/95-015-1	L1	341/90-003-2[a]	L1
304/91-002-1[a]	C1	317/87-012-1[c]	B1	325/95-018-0	P1	341/91-004-0	L2
305/87-009-0	H1	317/92-008-0	H1	327/90-012-0	P1	341/91-015-0	H1
305/88-001-0	H1	317/94-007-0	G2	327/92-018-0	D1	341/92-012-0	P1
305/88-004-0	P1	318/91-005-0	P1	327/92-027-0[c]	B1	341/93-004-0	L2
305/91-010-0	P1	318/92-003-0	L2	327/92-027-0[c]	B1	341/93-014-1	H1
305/92-017-0	H1	318/92-005-0	P1	327/94-008-0	P1	344/92-028-0	P1
305/92-020-1	L2	318/94-001-1	C1	328/93-001-0	K1	346/87-001-0[a]	H1

a. One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes

b. One LER that describes multiple reactor trip events from one plant unit.

c. One LER that describes reactor trip events from multiple plant units.

Table D-8. (continued).

LER Number	FI Code	LER Number	FI Code	LER Number	FI Code	LER Number	FI Code
346/87-001-0[a]	P1	364/94-001-0	L2	373/93-002-0	G2	397/92-033-0[b]	G2
346/87-011-0	E2	364/95-005-0[b]	P1	373/93-015-0	B1	397/92-033-0[b]	G2
346/87-015-0	D1	364/95-005-0[b]	P1	373/94-015-0	L1	397/93-027-0	L1
346/89-005-0	L2	366/87-003-0	L1	373/95-014-0	L1	400/87-008-0	P1
348/87-003-0	P1	366/87-008-0	P1	374/92-012-0	L1	400/87-013-0	P1
348/87-010-0	P1	366/88-006-0	L1	374/92-016-1	D1	400/87-017-0	P1
348/89-006-0	L1	366/88-017-0[a]	L1	374/94-004-0	C2	400/87-018-0	P1
348/91-010-0	P1	366/88-017-0[a]	P1	382/87-028-0	P1	400/87-019-0	P1
352/91-009-0	P1	366/89-005-0[a]	L1	382/88-016-0	P1	400/87-021-0[a]	L2
352/95-008-0	G2	366/89-005-0[a]	P1	382/90-003-1[a]	L3	400/87-021-0[a]	P1
353/90-012-0	L2	366/90-001-1	L1	382/90-003-1[a]	P1	400/87-024-0	P1
353/90-015-0	L1	366/92-009-0	P1	382/90-012-0	H1	400/87-025-0	P1
353/94-010-1	C1	366/94-007-0	L1	382/91-011-1	L1	400/87-031-0	P1
354/87-037-0[a]	L2	366/95-001-0	P1	382/91-019-0	L1	400/87-037-0	P1
354/87-037-0[a]	P1	366/95-003-0[b]	L2	382/91-022-0	L1	400/87-041-0[a]	D1
354/87-047-0	G2	368/88-011-0	G1	382/95-002-0	H1	400/87-041-0[a]	P1
354/88-012-1[a]	L2	368/89-006-0[a]	K1	387/89-001-0[a]	D1	400/87-062-0	L1
354/88-012-1[a]	P1	368/89-006-0[a]	L2	387/89-001-0[a]	L2	400/87-063-0	P1
354/88-027-0[a]	L1	368/89-006-0[a]	L3	387/91-008-0	L1	400/88-028-0[a]	L1
354/88-027-0[a]	P1	369/87-017-1[a]	G1	388/87-006-0	L1	400/88-028-0[a]	P1
354/89-017-0	D1	369/87-017-1[a]	L1	388/92-001-0	C1	400/88-032-0	P1
354/90-003-0[a]	H1	369/87-021-0[a]	D1	389/87-003-0[a]	L1	400/89-001-2	L2
354/90-003-0[a]	P1	369/87-021-0[a]	P1	389/87-003-0[a]	P1	400/89-003-0	H1
361/87-031-1	P1	369/89-004-0	F1	389/90-001-0	P1	400/89-004-0[a]	L2
361/90-016-1	C1	369/90-001-0	P1	389/92-006-0	H1	400/89-004-0[a]	P1
362/87-017-0	L2	369/91-001-0	B1	395/87-027-0	P1	400/89-017-1	H1
362/90-002-1	L1	370/87-003-0	P1	395/88-002-0	H1	400/91-010-0	M1
362/93-004-0	L2	370/89-002-0	L3	395/89-012-0	B1	400/92-007-0	L2
364/89-007-0	P1	370/92-006-0	P1	397/87-002-0[a]	L1	400/92-010-0	L2
364/89-010-0	P1	370/93-008-0[a]	B1	397/87-002-0[a]	P1	400/95-011-1	L1
364/90-001-0	P1	370/93-008-0[a]	L1	397/87-020-0	L1	410/87-064-0	L2
364/91-002-0	P1	373/87-014-0	H1	397/87-022-0[a]	L1	410/87-081-0	L2
364/91-004-0	L2	373/89-009-1	D1	397/87-022-0[a]	P1	410/88-001-0	D1
364/92-010-0	L2	373/92-003-0	L2	397/88-003-0	L1	410/88-014-0[a]	L1

a. One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes.

b One LER that describes multiple reactor trip events from one plant unit.

c One LER that describes reactor trip events from multiple plant units.

Table D-8. (continued).

LER Number	FI Code	LER Number	FI Code	LER Number	FI Code	LER Number	FI Code
410/88-014-0[a]	P1	416/91-004-0	P1	440/90-001-0	P1	457/88-019-0	D1
410/89-035-0	L2	416/91-005-1[a]	D1	440/91-027-0[a]	J1	457/88-029-1	P1
410/90-009-0[a]	D1	416/91-005-1[a]	L1	440/91-027-0[a]	L2	458/93-017-0	L1
410/90-009-0[a]	L2	416/91-007-0[a]	D1	440/93-010-0[a]	L2	458/94-030-0	L1
410/91-023-0	P1	416/91-007-0[a]	P1	440/93-010-0[a]	J1	461/87-017-0	D1
410/92-017-0	P1	416/95-008-0[a]	L1	443/91-001-0	P1	461/87-050-0	L2
410/94-007-0	L2	416/95-008-0[a]	L2	443/91-002-0	L3	461/88-019-0	L2
412/87-024-0	L1	423/87-001-0	L2	443/91-008-0	B1	461/88-028-0	H1
412/87-030-2	H1	423/87-021-0	P1	443/92-017-0	P1	461/89-029-0	L2
412/87-035-0	P1	423/87-027-0	L1	443/93-001-0	P1	461/91-006-0	L2
412/87-036-0	B1	423/88-014-0	L2	443/94-001-1	L1	461/93-007-0	L2
412/93-002-1	L1	423/88-024-0	L2	445/90-013-0	P1	482/87-002-0	L1
413/91-015-0	P1	423/90-011-0[a]	E2	445/90-030-0	P1	482/92-002-0	P1
413/91-019-0	P1	423/90-011-0[a]	L2	445/92-014-0	P1	483/89-008-0	C1
413/94-001-0	L2	423/90-013-1	L2	445/92-019-0	P1	483/90-007-0	L1
414/87-007-1	P1	423/90-014-0	L2	445/95-003-1	P1	483/95-005-0	L2
414/87-021-2	P1	423/90-030-2[a]	K2	445/95-004-1	P1	498/88-049-0	L1
414/87-025-0	P1	423/90-030-2[a]	L1	446/93-011-0	P1	498/89-005-0	H1
414/88-031-0	P1	424/88-043-0	D1	454/87-019-2	L1	499/89-020-0	P1
414/89-002-0	P1	424/90-016-0	P1	454/90-014-0	P1	528/88-010-1	H1
414/89-003-1	L1	425/90-002-0	C2	455/87-011-1	L3	528/89-004-0	H1
414/95-005-0	P1	440/87-012-0[a]	L1	455/87-019-1	B1	528/95-012-0	L2
416/87-009-2	L2	440/87-012-0[a]	P1	455/88-008-0	P1	529/87-008-0	P1
416/88-006-0	P1	440/87-027-1[a]	K1	455/90-010-1	K1	529/93-001-2	F1
416/88-013-0	D1	440/87-027-1[a]	L2	456/88-022-0	B1	529/93-004-0	P1
416/89-012-0	L2	440/87-030-0	P1	456/88-025-0[a,c]	D1	530/89-001-3	L1
416/89-019-0	E2	440/87-037-0	P1	456/88-025-0[a,c]	D1	530/92-001-0	D1
416/90-017-1	L1	440/87-042-0	L1	456/88-025-0[a,c]	P1	530/94-007-0	L1
416/90-028-0[a]	D1	440/87-072-0[a]	L1	456/88-025-0[a,c]	P1		
416/90-028-0[a]	P1	440/87-072-0[a]	P1	456/90-018-0	L1		
416/90-029-0	P1	440/88-012-0	P1	456/94-012-0	L1		

a. One LER that describes one reactor trip event from one plant unit and with multiple assigned functional impact codes.

b. One LER that describes multiple reactor trip events from one plant unit

c One LER that describes reactor trip events from multiple plant units

Table D-9. LERs from Table D-8 with multiple functional impact (FI) codes (P heading not included).

FI Heading Combination	LER Number	FI Heading Combination	LER Number	FI Heading Combination	LER Number
B, C, L	293/93-004-0	B, H	323/88-008-0	D, L	410/90-009-0
B, D, E	271/91-009-1	B, L	293/93-022-0	D, L	416/91-005-1
B, D, H	219/92-005-0	B, L	324/89-009-1	E, L	245/90-016-1
D, H, L	298/89-026-0	B, L	370/93-008-0	E, L	423/90-011-0
D, E, L	317/87-003-0	C, L	263/91-019-0	G, L	369/87-017-1
B, C	336/88-011-1	C, L	277/92-010-0	J, L	440/91-027-0
B, C	304/91-002-1	D, L	237/94-005-2	J, L	440/93-010-0
B, D	270/92-004-0	D, L	280/90-006-0	K, L	368/89-006-0
B, D	301/89-002-0	D, L	341/90-003-2	K, L	423/90-030-2
B, H	237/90-002-2	D, L	387/89-001-0	K, L	440/87-027-1

Table D-10. Steam generator tube rupture (F1) and very small LOCA (G1) leak rates based on all the operating experience from 1987 through 1995.

LER (Plant)	Leak Rate	Category	Comment	LER (Plant)	Leak Rate	Category	Comment
287/91-008-0 (Oconee 3)	87 gpm	G1	Failure of instrument line compression fitting	368/88-011-0 (Arkansas 2)	40 gpm	G1	Sensing line reducing fitting and RCP shaft seal
338/87-017-1 (North Anna 1)	637 gpm[a]	F1	Steam generator tube rupture, leak rate not reported	369/87-017-1 (McGuire 1)	40 gpm	G1	Letdown line drain line crack (inside containment
338/89-005-0 (North Anna 1)	74 gpm	G1	Steam generator tube leak	369/89-004-0 (McGuire 1)	540 gpm	F1	Steam generator tube rupture
				529/93-001-2 (Palo Verde 2)	240 gpm	F1	Steam generator tube rupture

a. Value taken from NUREG/CR-6365.

Table D-11. Initial plant fault (IPF) and functional impact (FI) mean frequencies and associated uncertainty distributions based on all the operating experience from 1987 through 1995 (except for certain rare events).

Category	FI (per critical year)	Distribution[a]	IPF (per critical year)	Distribution[a]
B—Loss of Offsite Power	4.61E-2	gamma(1.99, 43.38)	2.37E-2	gamma(1.97, 83.35)
C—Loss of Safety-Related Bus				
C1—Loss of Vital Medium Voltage AC Bus	1.85E-2	gamma(13.5, 728.29)	1 44E-2	gamma(10 5, 728.29)
C2—Loss of Vital Low Voltage ac Bus	4.81E-3	gamma(3.5, 728.29)	2.06E-3	gamma(1.5, 728.29)
C3—Loss of Vital dc Bus	2.06E-3	gamma(1.5, 728.29)	6.87E-4	gamma(0.5, 728.29)
D, BWRs, 1995—Loss of Instrument or Control Air System	2.91E-2	lognormal(2.63E-2, 2.10)	1.27E-2	lognormal(1.06E-3, 2.69)
D, PWRs, 1995—Loss of Instrument or Control Air System	9.60E-3	lognormal(8.58E-3, 2.18)	5 82E-3	lognormal(4.85E-3, 2.70)
E1—Total Loss of Service Water	9.72E-4	gamma(1.5, 1543.30)	9.72E-4	gamma(1.5, 1543.30)
E2—Partial Loss of Service Water	8.92E-3	gamma(6.5, 728.29)	6.87E-4	gamma(0.5, 728.29)
F, PWRs—Steam Generator Tube Rupture	7.02E-3	gamma(3.5, 498.55)	7.02E-3	gamma(3.5, 498.55)
G—Loss of Coolant Accident/Leak				
G1—Very Small LOCA/Leak	6.18E-3	gamma(4.5, 728.29)	3.43E-3	gamma(2.5, 728.29)
G2—Stuck Open: 1 Safety/Relief Valve: BWR	4.57E-2	gamma(10.5, 229.74)	4.57E-2	gamma(10.5, 229.74)
G2—Stuck Open: 1 Safety/Relief Valve: PWR	5.01E-3	gamma(2.5, 498.55)	5.01E-3	gamma(2.5, 498.55)
G3—Small Pipe Break LOCA	5.0E-4	lognormal(4.0E4, 3)	5.0E-4	lognormal(4.0E-4, 3)
G4— Stuck Open: Pressurizer PORV	1.00E-3	gamma(0.5, 498.55)	1.00E-3	gamma(0.5, 498.55)
G5— Stuck Open: 2 or more Safety/Relief Valves	3.24E-4	gamma(0.5, 1543.30)	3.24E-4	gamma(0.5, 1543.30)
G6—Medium Pipe Break LOCA: PWR	4.0E-5	lognormal(1.0E-5, 10)	4.0E-5	lognormal(1.0E-5, 10)
G6—Medium Pipe Break LOCA: BWR	4.0E-5	lognormal(1.0E-5, 10)	4.0E-5	lognormal(1.0E-5, 10)
G7—Large Pipe Break LOCA: PWR	5.0E-6	lognormal(1.0E-6, 10)	5.0E-6	lognormal(1.0E-6, 10)
G7—Large Pipe Break LOCA: BWR	3.0E-5	lognormal(1.0E-5, 10)	3.0E-5	lognormal(1.0E-5,10)
G8—Reactor Coolant Pump Seal LOCA : PWR	2.45E-3	gamma(2.5, 1018.77)	2.45E-3	gamma(2.5, 1018 77)
H, 1995—Fire	3.16E-2	lognormal(2.99E-2, 1.75)	2.34E-2	lognormal(2.17E-2, 1.91)
J—Flood	3.43E-3	gamma(2.5, 728.29)	2.06E-3	gamma(1.5, 728 29)
K—High Energy Line Break	1.30E-2	gamma(9 5, 728.29)	1.30E-2	gamma(9.5, 728.29)
K1—Steam Line Break Outside Containment	1.03E-2	gamma(7.5, 728.29)	1 03E-2	gamma(7.5, 728.29)
K2—Feedwater Line Break	3.43E-3	gamma(2.5, 728 29)	3.43E-3	gamma(2.5, 728.29)

Table D-11. (continued).

Category	FI (per critical year)	Distribution[a]	IPF (per critical year)	Distribution[a]
K3, PWRs—Steam Line Break Inside Containment	1.00E-3	gamma(0.5, 498.55)	1.00E-3	gamma(0.5, 498.55)
L, BWRs, 1995 —Loss of Condenser Heat Sink	2.86E-1	lognormal(2.81E-1, 1.38)	1.02E-1	lognormal(9.62E-2, 1.71)
L1, BWRs, 1995 —Inadvertent Closure of All MSIVs	1.71E-1	lognormal(1.48E-1, 2.45)	3.12E-1	lognormal(2 61E-2, 2.66)
L2, BWRs —Loss of Condenser Vacuum	2.02E-1	gamma(2.344, 11.60)	1.19E-1	gamma(1.83, 15.33)
L, PWRs —Loss of Condenser Heat Sink	—	—	3 76E-2	gamma(0.662, 17.62)
L, PWRs , 1995—Loss of Condenser Heat Sink	1.17E-1	lognormal(8.46E-2, 3.76)	—	—
L1, PWRs—Inadvertent Closure of MSIVs	—	—	1.10E-2	gamma(5.5, 498.55)
L1, PWRs, 1995—Inadvertent Closure of MSIVs	3.80E-2	lognormal(3.54E-2, 1.85)	—	—
L2, PWRs—Loss of Condenser Vacuum	6.87E-2	gamma(0.354, 5.14)	2.58E-2	gamma(0.246, 9.56)
L3—Turbine Bypass Unavailable	—	—	2.06E-3	gamma(1.5, 728.29)
L3, 1995—Turbine Bypass Unavailable	4.10E-3	lognormal(2.72E-3, 4.44)	—	—
P, 1995—Total Loss of Feedwater Flow	8.45E-2	lognormal(5.70E-2, 4.30)	5.44E-2	lognormal(3.03E-2, 5.94)
Q-B, BWRs, 1995—Other Initial Plant Fault	1.55E+0	lognormal(1.46E-0, 1.73)	—	—
Q-P, PWRs, 1995—Other Initial Plant Fault	1.22E+0	lognormal(1.14E+0, 1.87)	—	—

Note: Refer to Tables 3-1 and D-12 for special notes concerning the specifics on the values in this table.

a. For the gamma(param1, param2), param2 is critical years and the mean is param1 divided by param2. For the logmomal(param1, param2), param1 is the fitted median while param2 is the error factor.

Table D-12. Frequency estimates of initial plant fault categories: mean, percentiles, and trends based on all the operating experience from 1987 through 1995 (except for certain rare events).

Event	Initial Plant Fault Category	Number of Initial Plant Fault Occurrences[a]	Mean Frequency (per critical year)[b,c,k]	Percentiles: 5th %ile	Percentiles: 95th %ile	Model Used: Trend	Model Used: Plant Difference[j]
Loss of Coolant Accident (LOCA)	G						
Large Pipe Break LOCA: PWR	G7	0	5E-6[d]	1E-7	1E-5	Constant[e]	No
Large Pipe Break LOCA. BWR	G7	0	3E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: PWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: BWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Small Pipe Break LOCA	G3	0	5E-4[d]	1E-4	1E-3	Constant[e]	No
Very Small/Leak	G1	2	3.4E-3	7.9E-4	7.6E-3	Constant[e]	No
Stuck Open: Pressurizer PORV	G4	0	1.0E-3	3.9E-6	3.9E-3	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: PWR	G2	2	5.0E-3	1.2E-3	1.1E-2	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: BWR	G2	10	4.6E-2	2.5E-2	7.1E-2	Constant[e]	No
Stuck Open: 2 or More Safety/Relief Valves	G5	0	3.2E-4[d]	1.3E-6	1.2E-3	Constant[e]	No
Reactor Coolant Pump Seal LOCA. PWR	G8	2[d]	2.5E-3[d]	5.6E-4	5.4E-3	Constant[e]	No
Steam Generator Tube Rupture: PWR	F1	3	7.0E-3	2.2E-3	1.4E-2	Constant[e]	No
Loss of Offsite Power	B1	17	2.4E-2	4.1E-3	5.6E-2	Constant[e]	No
Total Loss of Condenser Heat Sink (combined)[f]. PWR	L	19[f]	3.8E-2[f]	5.3E-4	1.3E-1	Constant[e]	Yes[j]
Total Loss of Condenser Heat Sink (combined)[f]. BWR	L	45[f]	1.0E-1[c,f]	5.6E-2	1.7E-1	Decrease	No
Inadvertent Closure of All MSIVs: PWR	L1	5	1.1E-2	4.6E-3	2.0E-2	Constant[e]	No
Inadvertent Closure of All MSIVs: BWR	L1	16	3.1E-2[c]	9.8E-3	6.9E-2	Decrease	No
Loss of Condenser Vacuum: PWR	L2	13	2.6E-2	<1.0E-6	1.3E-1	Constant[e]	Yes[j]
Loss of Condenser Vacuum: BWR	L2	27	1.2E-1	1.9E-2	2.9E-1	Constant[e]	No
Turbine Bypass Unavailable	L3	1	2.1E-3	2.4E-4	5.4E-3	Constant[e]	No
Total Loss of Feedwater Flow	P1	86	5.4E-2[c]	5.1E-3[i]	1.8E-1[i]	Decrease	Yes[j]
General Transients (combined)[f]. PWR	Q	1,184[f,g]	1.2E+0[c,f]	6.1E-1[i]	2.1E+0[i]	Decrease	Yes[j]
General Transients (combined)[f]. BWR	Q	541[f,g]	1.5E+0[c,f]	8.5E-1[i]	2.5E+0[i]	Decrease	Yes[j]
High Energy Line Steam Breaks/Leaks (combined)[h]	K	9[h]	1.3E-2	7.0E-3	2.1E-2	Constant[e]	No
Steam Line Break/Leak Outside Containment	K1	7	1.0E-2	5.0E-3	1.7E-2	Constant[e]	No
Steam Line Break/Leak Inside Containment: PWR	K3	0	1.0E-3	3.9E-6	3.9E-3	Constant[e]	No
Feedwater Line Break/Leak	K2	2	3.4E-3	7.9E-4	7.6E-3	Constant[e]	No

Table D-12. (continued).

Event	Initial Plant Fault Category	Number of Initial Plant Fault Occurrences[a]	Mean Frequency (per critical year)[b,c,k]	Percentiles		Model Used	
				5th %ile	95th %ile	Trend	Plant Difference[j]
Loss of Safety-Related Bus	C						
Loss of Vital Medium Voltage ac Bus	C1	10	1.4E-2	8.0E-3	2.2E-2	Constant[e]	No
Loss of Vital Low Voltage ac Bus	C2	1	2.1E-3	2.4E-4	5.4E-3	Constant[e]	No
Loss of Vital dc Bus	C3	0	6.9E-4	2.7E-6	2.6E-3	Constant[e]	No
Loss of Safety-Related Cooling Water	E						
Total Loss of Service Water	E1	0[d]	3.2E-4[d]	1.3E-6	1.2E-3	Constant[e]	No
Partial Loss of Service Water	E2	0	6.9E-4	2.7E-6	2.6E-3	Constant[e]	No
Loss of Instrument or Control Air: PWR	D1	13	5.8E-3[c]	1.8E-3	1.3E-2	Decrease	No
Loss of Instrument or Control Air: BWR	D1	13	1.3E-2[c]	3.9E-3	2.9E-2	Decrease	No
Fire	H1	31	2.3E-2[c]	1.1E-2	4.1E-2	Decrease	No
Flood	J1	1	2.1E-3	2.4E-4	5.4E-3	Constant[e]	No
		Total —PWR	1.4E+0[c]	6.9E-1[i]	2.4E+0[i]	Decrease[f]	Yes[j]
		Total —BWR	1.8E+0[c]	9.5E-1[i]	2.9E+0[i]	Decrease[f]	Yes[j]

a. Reactor trip events from 1987 through 1995, inclusive, except when noted for certain rare events.

b. Frequencies are presented in per critical year (8,760 critical hours per critical year).

c. For categories with a decreasing trend, the frequencies reported are based on the endpoint of the trend line (i.e., 1995, the last year of the study).

d. No failures were identified in the 1987–1995 operating experience. The Medium and Large Pipe Break LOCA estimates were based on review of current literature and fracture mechanic analyses and using world-wide experience. (Appendix J contains the results of the LOCA analysis.) Frequency estimates for Small Pipe Break LOCA, Reactor Coolant Pump Seal LOCA, Stuck Open: 2 or More Safety/Relief Valves, and Total Loss of Service Water categories were based on total U.S. operating experience (1969–1997).

e. Any evidence for a trend was weak, not statistically significant. The trend, if any, is too small to be seen in the data. Therefore, no trend is modeled.

f. Combined number of occurrences of all categories for each plant type (BWR, PWR) under this heading was used to calculate this frequency and trend.

g. Total number of initial plant-fault occurrences for this plant type.

h. The frequency was based on the combined number of occurrences in the categories under this heading.

i. The interval includes variability from plants with events early in life (for example, learning periods) and are wider than the plants' current performance. See Appendix G for results with the early-in-life events excluded.

j. Due to modeling assumptions with regard to independent random events, the between-plant variation was modeled with the first four months from date of commercial operation (early-in-life events) excluded for the affected plants. See Appendix G for these results.

k. For categories modeled with no trend and no between-plant variation, the estimates were calculated using a Jeffreys noninformative prior (one-half of an event added to the total number of events) in a Bayesian updated distribution.

Table D-13. Summary count of the initial plant fault (IPF) events correlated to the subsequent functional impact (FI) events based on all the operating experience from 1987 through 1995.

IPF/FI Combination	IPF Count	Loss of Offsite Power	Loss of Vital Medium Voltage ac Bus	Loss of Vital Low Voltage ac Bus	Loss of Vital dc Bus	Loss of Inst./ Control Air System	Partial Loss of Service Water (SW)	Steam Generator Tube Rupture	Very Small LOCA	Stuck Open: 1 Safety/ Relief Valve	Fire	Flood	Steam Line Break Outside Contain-ment	Feedwater Line Break	Inadvertent Closure of All MSIVs	Loss of Condenser Vacuum	Turbine Bypass Unavail-able	Total Loss of Feed-water Flow
		B1	C1	C2	C3	D1	E2	F1	G1	G2	H1	J1	K1	K2	L1	L2	L3	P1
B1/B1D1	2	2	—	—	—	2	—	—	—	—	—	—	—	—	—	—	—	—
B1/B1E2	1	1	—	—	—	—	1	—	—	—	—	—	—	—	—	—	—	—
B1/B1L1	2	2	—	—	—	—	—	—	—	—	—	—	—	—	2	—	—	—
B1/B1	12	12	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
B1 Total	17	17	—	—	—	2	1	—	—	—	—	—	—	—	2	—	—	—
C1/B1C1C2L3	1	1	1	1	—	—	—	—	—	—	—	—	—	—	—	—	1	—
C1/C1L1P1	1	—	1	—	—	—	—	—	—	—	—	—	—	—	1	—	—	1
C1/C1B1	1	1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
C1/C1L2	1	—	1	—	—	—	—	—	—	—	—	—	—	—	—	1	—	—
C1/C1P1	1	—	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1
C1/C1	5	—	5	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
C1 Total	10	2	10	1	—	—	—	—	—	—	—	—	—	—	1	1	1	2
C2/C2	1	—	—	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—
C2 Total	1	—	—	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—
C3 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
D1/D1E2L3	1	—	—	—	—	1	1	—	—	—	—	—	—	—	—	—	1	—
D1/D1L1	2	—	—	—	—	2	—	—	—	—	—	—	—	—	2	—	—	—
D1/D1L2	2	—	—	—	—	2	—	—	—	—	—	—	—	—	—	2	—	—
D1/D1P1	5	—	—	—	—	5	—	—	—	—	—	—	—	—	—	—	—	5
D1/D1	16	—	—	—	—	16	—	—	—	—	—	—	—	—	—	—	—	—
D1 Total	26	—	—	—	—	26	1	—	—	—	—	—	—	—	2	2	1	5
E1 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
E2 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
F1/F1	3	—	—	—	—	—	—	3	—	—	—	—	—	—	—	—	—	—
F1 Total	3	—	—	—	—	—	—	3	—	—	—	—	—	—	—	—	—	—
G1/G1	2	—	—	—	—	—	—	—	2	—	—	—	—	—	—	—	—	—

Table D-13. (continued).

IPF/FI Combination	IPF Count	Loss of Offsite Power	Loss of Vital Medium Voltage ac Bus	Loss of Vital Low Voltage ac Bus	Loss of Vital dc Bus	Loss of Inst./ Control Air System	Partial Loss of Service Water (SW)	Steam Generator Tube Rupture	Very Small LOCA	Stuck Open: 1 Safety/ Relief Valve	Fire	Flood	Steam Line Break Outside Contain-ment	Feedwater Line Break	Inadvertent Closure of All MSIVs	Loss of Condenser Vacuum	Turbine Bypass Unavail-able	Total Loss of Feed-water Flow
		B1	C1	C2	C3	D1	E2	F1	G1	G2	H1	J1	K1	K2	L1	L2	L3	P1
G1 Total	2	—	—	—	—	—	—	—	2	—	—	—	—	—	—	—	—	—
G2/G2	10	—	—	—	—	—	—	—	—	10	—	—	—	—	—	—	—	—
G2 Total	10	—	—	—	—	—	—	—	—	10	—	—	—	—	—	—	—	—
G3 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
G4 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
G5 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
G6 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
G7 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
G8 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
H1/B1D1H1	1	1	—	—	—	1	—	—	—	—	1	—	—	—	—	—	—	—
H1/D1H1L1	1	—	—	—	—	1	—	—	—	—	1	—	—	—	1	—	—	—
H1/B1H1	2	2	—	—	—	—	—	—	—	—	2	—	—	—	—	—	—	—
H1/H1P1	1	—	—	—	—	—	—	—	—	—	1	—	—	—	—	—	—	1
H1/H1	26	—	—	—	—	—	—	—	—	—	26	—	—	—	—	—	—	—
H1 Total	31	3	—	—	—	2	—	—	—	—	31	—	—	—	1	—	—	1
J1/J1L2	1	—	—	—	—	—	—	—	—	—	1	—	—	—	—	1	—	—
J1 Total	1	—	—	—	—	—	—	—	—	—	1	—	—	—	—	1	—	—
K1/K1L2L3	1	—	—	—	—	—	—	—	—	—	—	—	1	—	—	1	1	—
K1/K1L2	1	—	—	—	—	—	—	—	—	—	—	—	1	—	—	1	—	—
K1/K1	5	—	—	—	—	—	—	—	—	—	—	—	5	—	—	—	—	—
K1 Total	7	—	—	—	—	—	—	—	—	—	—	—	7	—	—	2	1	—
K2/K2	1	—	—	—	—	—	—	—	—	—	—	—	—	1	—	—	—	—
K2/L1	1	—	—	—	—	—	—	—	—	—	—	—	—	1	1	—	—	—
K2 Total	2	—	—	—	—	—	—	—	—	—	—	—	—	2	1	—	—	—
K3 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
L1/L1	21	—	—	—	—	—	—	—	—	—	—	—	—	—	21	—	—	—

Table D-13. (continued).

IPF/FI Combination	IPF Count	Loss of Offsite Power	Loss of Vital Medium Voltage ac Bus	Loss of Vital Low Voltage ac Bus	Loss of Vital dc Bus	Loss of Inst./ Control Air System	Partial Loss of Service Water (SW)	Steam Generator Tube Rupture	Very Small LOCA	Stuck Open: 1 Safety/ Relief Valve	Fire	Flood	Steam Line Break Outside Contain-ment	Feedwater Line Break	Inadvertent Closure of All MSIVs	Loss of Condenser Vacuum	Turbine Bypass Unavail-able	Total Loss of Feed-water Flow
		B1	C1	C2	C3	D1	E2	F1	G1	G2	H1	J1	K1	K2	L1	L2	L3	P1
L1 Total	21	—	—	—	—	—	—	—	—	—	—	—	—	—	21	—	—	—
L2/L2P1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—	1
L2/L2	39	—	—	—	—	—	—	—	—	—	—	—	—	—	—	39	—	—
L2 Total	40	—	—	—	—	—	—	—	—	—	—	—	—	—	—	40	—	1
L3/L3	3	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	3	—
L3 Total	3	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	3	—
N1 Total	0	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
P1/H1P1	1	—	—	—	—	—	—	—	—	—	1	—	—	—	—	—	—	1
P1/L1P1	9	—	—	—	—	—	—	—	—	—	—	—	—	—	9	—	—	9
P1/P1	76	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	76
P1 Total	86	—	—	—	—	—	—	—	—	—	1	—	—	—	9	—	—	86
QC4/D1L1	1	—	—	—	—	1	—	—	—	—	—	—	—	—	1	—	—	—
QC4/G2	1	—	—	—	—	—	—	—	—	1	—	—	—	—	4	—	—	—
QC4/L1	4	—	—	—	—	—	—	—	—	—	—	—	—	—	4	—	—	—
QC4/P1	6	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	6
QC4 Total	31	—	—	—	—	1	—	—	—	1	—	—	—	—	5	—	—	6
QC5/L1P1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—	—	1
QC5/E2	1	—	—	—	—	—	1	—	—	—	—	—	—	—	—	—	—	—
QC5/L3	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—
QC5/P1	4	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	4
QC5 Total	25	—	—	—	—	—	1	—	—	—	—	—	—	—	1	—	1	5
QG9 Total	6	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QG10 Total	2	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QK4 Total	3	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QL4/E2L2	2	—	—	—	—	—	2	—	—	—	—	—	—	—	—	2	—	—
QL4/J1L2	1	—	—	—	—	—	—	—	—	—	—	1	—	—	—	1	—	—

Table D-13. (continued).

IPF/FI Combination	IPF Count	Loss of Offsite Power	Loss of Vital Medium Voltage ac Bus	Loss of Vital Low Voltage ac Bus	Loss of Vital dc Bus	Loss of Inst./ Control Air System	Partial Loss of Service Water (SW)	Steam Generator Tube Rupture	Very Small LOCA	Stuck Open: 1 Safety/ Relief Valve	Fire	Flood	Steam Line Break Outside Containment	Feedwater Line Break	Inadvertent Closure of All MSIVs	Loss of Condenser Vacuum	Turbine Bypass Unavailable	Total Loss of Feedwater Flow
		B1	C1	C2	C3	D1	E2	F1	G1	G2	H1	J1	K1	K2	L1	L2	L3	P1
QL4/L2P1	3	—	—	—	—	—	—	—	—	—	—	—	—	—	—	3	—	3
QL4/L2	23	—	—	—	—	—	—	—	—	—	—	—	—	—	—	23	—	—
QL4 Total	50	—	—	—	—	—	2	—	—	—	—	1	—	—	—	29	—	3
QL5/L1L3P1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—	1	1
QL5/L1	2	—	—	—	—	—	—	—	—	—	—	—	—	—	2	—	—	—
QL5/P1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1
QL5 Total	47	—	—	—	—	—	—	—	—	—	—	—	—	—	3	—	1	2
QL6/L1L2	1	—	—	—	—	—	—	—	—	—	—	—	—	—	1	1	—	—
QL6/L2	6	—	—	—	—	—	—	—	—	—	—	—	—	—	—	6	—	—
QL6 Total	9	—	—	—	—	—	—	—	—	—	—	—	—	—	1	7	—	—
QP2/B1C1	1	1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QP2/B1	1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QP2/G1	1	—	—	—	—	—	—	—	1	—	—	—	—	—	—	—	—	—
QP2/H1	1	—	—	—	—	—	—	—	—	—	1	—	—	—	—	—	—	—
QP2/L3	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—
QP2/P1	2	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	2
QP2 Total	285	2	1	—	—	—	—	—	1	—	1	—	—	—	—	—	1	2
QP3/L1P1	2	—	—	—	—	—	—	—	—	—	—	—	—	—	2	—	—	2
QP3/P1	14	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	14
QP3 Total	19	—	—	—	—	—	—	—	—	—	—	—	—	—	2	—	—	16
QP4/P1	16	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	16
QP4 Total	35	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	16
QP5/B1	1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QP5/E2	1	—	—	—	—	—	1	—	—	—	—	—	—	—	—	—	—	—
QP5/L1	5	—	—	—	—	—	—	—	—	—	—	—	—	—	5	—	—	—
QP5/P1	2	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	2
QP5 Total	110	1	—	—	—	—	1	—	—	—	—	—	—	—	5	—	—	2

Table D-13. (continued).

IPF/FI Combination	IPF Count	Loss of Offsite Power	Loss of Vital Medium Voltage ac Bus	Loss of Vital Low Voltage ac Bus	Loss of Vital dc Bus	Loss of Inst./ Control Air System	Partial Loss of Service Water (SW)	Steam Generator Tube Rupture	Very Small LOCA	Stuck Open: 1 Safety/ Relief Valve	Fire	Flood	Steam Line Break Outside Contain-ment	Feedwater Line Break	Inadvertent Closure of All MSIVs	Loss of Condenser Vacuum	Turbine Bypass Unavail-able	Total Loss of Feed-water Flow
		B1	C1	C2	C3	D1	E2	F1	G1	G2	H1	J1	K1	K2	L1	L2	L3	P1
QR0/L1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—	—	—
QR0 Total	13	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—	—	—
QR1 Total	8	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QR2/L3P1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1	1
QR2 Total	40	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1	1
QR3/B1L1	1	1	—	—	—	—	—	—	—	—	—	—	—	—	1	—	—	—
QR3/C1	1	—	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QR3 Total	94	1	1	—	—	—	—	—	—	—	—	—	—	—	1	—	—	—
QR4/L1	2	—	—	—	—	—	—	—	—	—	—	—	—	—	2	—	—	—
QR4 Total	51	—	—	—	—	—	—	—	—	—	—	—	—	—	2	—	—	—
QR5/B1D1	1	1	—	—	—	1	—	—	—	—	—	—	—	—	—	—	—	—
QR5/D1L1	1	—	—	—	—	1	—	—	—	—	—	—	—	—	1	—	—	—
QR5/D1P1	1	—	—	—	—	1	—	—	—	—	—	—	—	—	—	—	—	1
QR5/B1	5	5	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QR5/C1	1	—	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QR5/C2	1	—	—	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—
QR5/D1	2	—	—	—	—	2	—	—	—	—	—	—	—	—	—	—	—	—
QR5/G2	1	—	—	—	—	—	—	—	—	1	—	—	—	—	—	—	—	—
QR5/H1	3	—	—	—	—	—	—	—	—	—	3	—	—	—	—	—	—	—
QR5/L1	14	—	—	—	—	—	—	—	—	—	—	—	—	—	14	—	—	—
QR5/P1	6	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	6
QR5 Total	457	6	1	1	—	5	—	—	—	1	3	—	—	—	15	—	—	7
QR6/H1	1	—	—	—	—	—	—	—	—	—	1	—	—	—	—	—	—	—
QR6/L2	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1	—	—
QR6/P1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1
QR6 Total	103	—	—	—	—	—	—	—	—	—	1	—	—	—	—	1	—	1
QR7/B1P1	1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1

Table D-13. (continued).

IPF/FI Combination	IPF Count	Loss of Offsite Power	Loss of Vital Medium Voltage ac Bus	Loss of Vital Low Voltage ac Bus	Loss of Vital dc Bus	Loss of Inst./ Control Air System	Partial Loss of Service Water (SW)	Steam Generator Tube Rupture	Very Small LOCA	Stuck Open: 1 Safety/ Relief Valve	Fire	Flood	Steam Line Break Outside Contain-ment	Feedwater Line Break	Inadvertent Closure of All MSIVs	Loss of Condenser Vacuum	Turbine Bypass Unavail-able	Total Loss of Feed-water Flow
		B1	C1	C2	C3	D1	E2	F1	G1	G2	H1	J1	K1	K2	L1	L2	L3	P1
QR7/L1	7	—	—	—	—	—	—	—	—	—	—	—	—	—	7	—	—	—
QR7 Total	84	1	—	—	—	—	—	—	—	—	—	—	—	—	7	—	—	1
QR8/C3P1	1	—	—	—	1	—	—	—	—	—	—	—	—	—	—	—	—	1
QR8/H1	2	—	—	—	—	—	—	—	—	—	2	—	—	—	—	—	—	—
QR8/L1	6	—	—	—	—	—	—	—	—	—	—	—	—	—	6	—	—	—
QR8/P1	1	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	1
QR8 Total	217	—	—	—	1	—	—	—	—	—	2	—	—	—	6	—	—	2
QR9/G1L1	1	—	—	—	—	—	—	—	1	—	—	—	—	—	1	—	—	—
QR9/L1	22	—	—	—	—	—	—	—	—	—	—	—	—	—	22	—	—	—
QR9 Total	36	—	—	—	—	—	—	—	1	—	—	—	—	—	23	—	—	—
Q Total	1,725	11	3	1	1	6	4	—	2	2	7	1	—	—	72	37	4	64
Total	1,985	33	13	3	1	36	6	3	4	12	39	2	7	2	109	81	10	159

Notes. To illustrate the use of this table, consider the initial plant fault category B1--Loss of Offsite Power (LOSP). The first column in Table D-13 lists the combinations of functional impact events that occurred after each initial plant fault category. In the first reactor trip sequence represented as B1/B1D1, two LOSP events occurred as the initial plant fault event (or reactor trip initiator), followed by the subsequent loss of instrument or control air (functional impact category D1). The loss of instrument or control air is called the subsequent functional impact event. The initial plant fault category identifier for the reactor trip initiator is located on the left side of the slash (**B1**/B1D1). The subsequent functional impact category identifier is located on the right side of the slash (B1/B1**D1**). Since the LOSP event has a category in the initial plant fault and functional impact groups, the sequence will have a B1 identifier on both sides of the slash. The number of functional impact events associated with each unique sequence is included in the column under the functional impact categories across the top of the table. In the second reactor trip event sequence (B1/B1E2) for the LOSP initial plant fault category, one reactor trip sequence involved partial loss of service water that occurred after the LOSP event. The third sequence combination (B1/B1L1) involved two event where the inadvertent closure of all main steam isolation valves occurred after the LOSP event. In the last event sequence (B1/B1), no other subsequent functional impact occurred after the LOSP event in 11 reactor trip sequences. To find the 14 LOSP events that occurred after reactor trip initiators from other initial plant fault categories, follow the LOSP column B1 down through the table.

Table D-14. Summary of manual reactor trips that occurred subsequent to the initial plant fault based on all the operating experience from 1987 through 1995.

Manual Reactor Trips	Category	BWR	PWR
2	B1—Loss of Offsite Power	1	1
3	C1—Loss of Vital Medium Voltage ac Bus	2	1
13	D1— Loss of Instrument or Control Air System	6	7
3	F1—Steam Generator Tube Rupture	0	3
1	G1—Very Small LOCA/Leak	0	1
10	G2— Stuck Open: 1 Safety/Relief Valve	10	0
7	H1—Fire	2	5
1	J1—Flood	1	0
6	K1—Steam Line Break Outside Containment	2	4
2	K2—Feedwater Line Break	0	2
15	L2—Loss of Condenser Vacuum	14	1
30	P1—Total Loss of Feedwater Flow	2	28
1	QC4—Loss of ac Instrument and Control Bus	0	1
9	QC5—Loss of Nonsafety-Related Bus	1	8
4	QG9—Primary System Leak	4	0
	QG10—Inadvertent Open/Close: 1 Safety/Relief Valve	0	2
2			
2	QK4—Steam or Feed Leakage	1	1
35	QL4—Loss of Nonsafety-Related Cooling Water	10	25
7	QL5—Partial Closure of MSIVs	0	7
4	QL6—Condenser Leakage	3	1
65	QP2—Partial Loss of Feedwater Flow	5	60
7	QP3—Total Loss of Condensate Flow	2	5
10	QP4—Partial Loss of Condensate Flow	2	8
12	QP5—Excessive Feedwater	5	7
1	QR2—Loss of Primary Flow (RPS Trip)	0	1
34	QR3—Reactivity Control Imbalance	4	30
13	QR5—Turbine Trip	5	8
2	QR8—Spurious Reactor Trip	1	1
2	QR9—Spurious Engineered Safety Feature Actuation	0	2
303	Totals	83	220

a QR6-Manual Reactor Trip initial plant fault category had 103 events of which 55 were BWR and 48 were PWR

Table D-15. Summary of dual unit reactor trips based on all the operating experience from 1987 through 1995.

Plants	LER #	IPF
Diablo Canyon 1, 2	275/94-020	QR2
Surry 1, 2	280/90-004	QR6
Calvert Cliffs 1, 2	317/87-012	B1
Calvert Cliffs 1, 2	317/93-003	QC5/QR2
Sequoyah 1, 2	327/92-027	B1
Beaver Valley 1, 2	334/94-005	QR2/H1
Limerick 1, 2	352/95-002	QR5
Vogtle 1, 2	424/95-002	QR8
Comanche Peak 1, 2	445/95-002	QR3
Braidwood 1, 2	456/88-025	D1
Braidwood 1, 2	456/89-006	QR3
Palo Verde 1, 3	528/91-010	QR4

Appendix E

Statistical Methods

Appendix E

Statistical Methods

To characterize event occurrence frequencies, operational data on reactor trips from U.S. commercial nuclear power plants from 1987 through 1995 were collected and reviewed. For new plants, data started at the low power license date.

This appendix describes the methods for the detection of trends and estimation of occurrence frequencies. The descriptions give details of the models and discussion of some of the reasoning behind the choice of models.

DATA COLLECTION AND CHARACTERIZATION

Event Occurrences

Collection and categorization of event occurrences is described in Appendix C. Quality assurance measures are also detailed there.

Critical Hours

The critical hours for each plant (1987-1995) were taken from the database CRITHRS (INEEL 1997), maintained at the INEEL. These hours come directly from the plants' monthly operating reports.

Operating Experience Used to Estimate Frequencies

Frequencies in this report are reported in units of events per critical year, where a critical year is defined as 8760 critical hours. Frequencies for initiating event categories except for several rare event categories are based on U.S. operating experience from 1987 through 1995.

Critical Hours Used for Certain Rare Events

Frequency estimates for pipe break LOCA-related events are based on total U.S and world-wide operating experience which included experience prior to 1987 and after 1995 (See Appendix J).

Estimates for reactor coolant pump seal LOCA, stuck open two or more safety/relief valves, and total loss of service water categories are based on total U.S. operating experience (1969 through 1997). The critical hours from 1984-1997 come from databases (INEEL 1997; INEEL 1998) maintained at the INEEL and based on licensee monthly operating reports. These critical hours included all experience after the low power license date. The critical hours from 1969-1983 come from Mackowiak et al. (1985), and ultimately from a review of the NRC "gray books." Those hours included all experience after the commercial start date. The critical hours for Big Rock Point and Dresden 1 are not given in Mackowiak et al. (1985), and therefore were estimated as 68% of the calendar hours. The U.S. values used in this report are:

<u>Critical Years (U.S., 1969-1997)</u>

1018.77 for PWRs 524.53 for BWRs

Hours to Be Counted for New Plants

For all the remaining categories of initiating event, the U. S. operating experience in 1987-1995 was used. A reviewer of an earlier draft pointed out that new plants often experience a high frequency of initiating events, which drops sharply after the plant has been operating for a short time. To describe the *current* performance of plants, the relevant data set consists of the time period after the initial learning experience. Inclusion of the learning period could give misleading results.

On the other hand, some readers may wish to see results based on all the data. Therefore, it was decided to analyze the data both ways, including and excluding the learning period. Readers must recognize the following facts.

- The analytical models used in this report assume that each plant has either a constant event frequency or a gradually changing event frequency. A sudden change in the frequency is modeled only by separately considering two time periods, before and after the sudden change.

- Some (broad) categories of event had high frequencies at certain new plants during a learning period, and markedly lower frequencies immediately after the learning period.

- For plants with such a learning period, the assumption of a constant rate or gentle trend gives erroneous or misleading results. Any trend seen is real, but cannot be assumed to continue forever, because part of the mechanism for improvement no longer exists. Also, for a plant with many events during the learning period, the modeling assumptions result in a relatively high rate for the plant, which is calculated as persisting over the entire time period analyzed. This is incorrect.

- For these reasons, this report analyzes both sets of data, but never displays plant-specific results based on data including the learning period. Plant-specific results are shown only for data excluding the learning period.

Determination of the Learning Period

When excluding a learning period, it is desirable not to exclude more of the operating history than necessary. Therefore, we examined the data to see how long any learning period lasted.

Plants with commercial start date after January 1, 1987 were considered for examination, and those with 24 or more initiating events were examined. The results for those ten plants are displayed below, in descending order of number of events.

The plots all measure time in days from the commercial start date. Negative times correspond to times after the low power license date but before the commercial start. The cumulative counts of events are plotted against elapsed time, and the slope of the cumulative plot represents frequency, events per time. These plots suggest a learning experience at many plants, when many initiating events occur in quick succession. The learning period is followed by a constant or gradually decreasing event frequency.

Nine of the ten plants examined (all but South Texas 1, the plant with the fewest total failures) showed a clear learning period. Eight plants had a clear "last" event in the learning period, with a gap between that event and the next event. The ages at these final learning events were –50, 0, 31, 33, 57, 64, 116, and 218 days. The end of the learning period for Nine Mile Point 2 is harder to discern, but it is somewhere from 97 to 170 days. To exclude the entire learning period for most plants, but not to exclude more than necessary, we decided to count the end of the learning period as four months (approximately 120 days) after the commercial start date. Excluding the so-defined learning period excluded 177 events, and reduced the database from 1985 to 1808 events. It reduced the total critical years from 728.29 to 717.26. This reduced database was used for the frequency calculations that exclude the learning period.

The dashed line in each plot shows approximately the four-month cut-off.

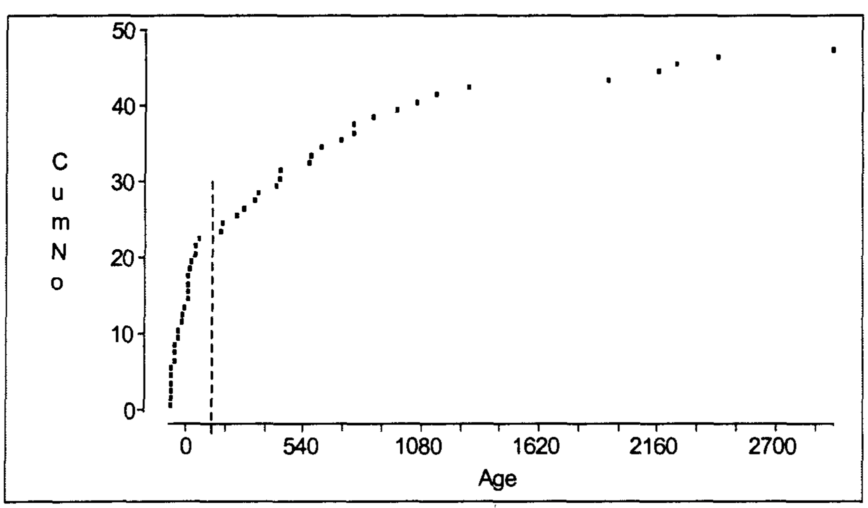

Figure E-1. Vogtle 1, cumulative number of initiating events, by age (days) from commercial start date. All events after the low power license date are shown.

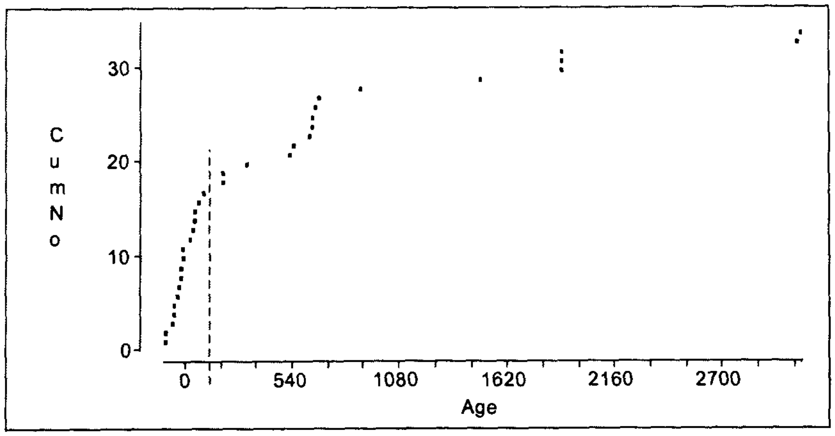

Figure E-2. Harris, cumulative number of initiating events, by age (days) from commercial start date. At the left, the plot does not show events between the low power license date (10/24/86) and the start of data collection (1/1/87).

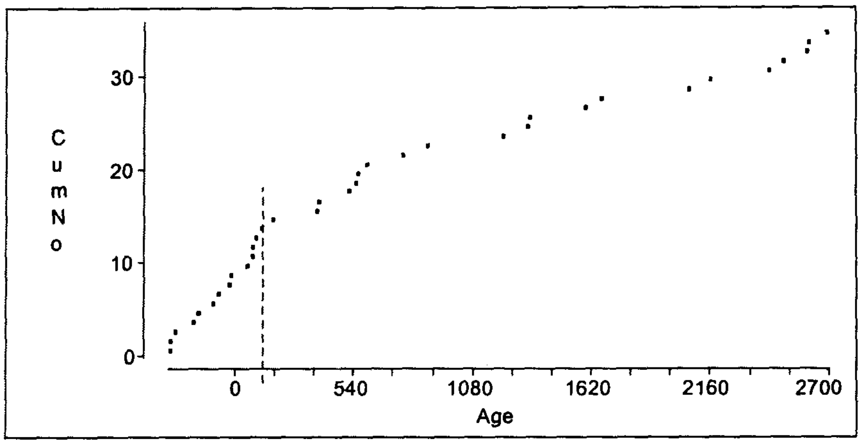

Figure E-3. Nine Mile Point 2, cumulative number of initiating events, by age (days) from commercial start date. At the left, the plot does not show events between the low power license date (10/31/86) and the start of data collection (1/1/87).

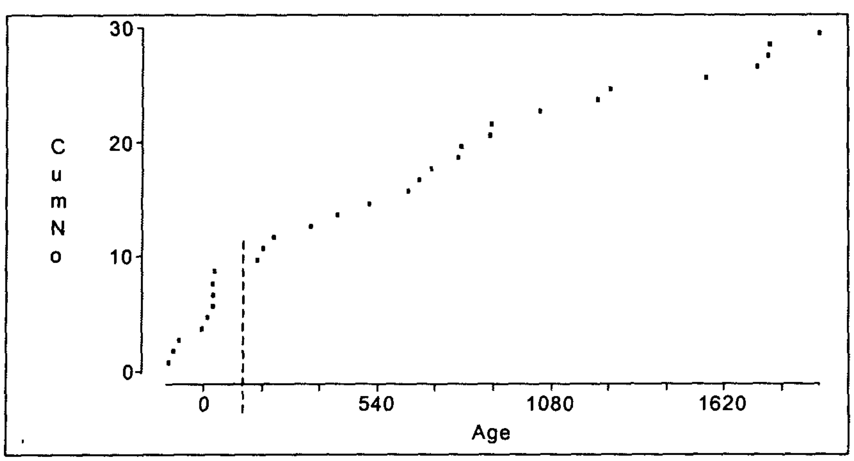

Figure E-4. Comanche Peak 1, cumulative number of initiating events, by age (days) from commercial start date. All events after the low power license date are shown.

Figure E-5. Clinton, cumulative number of initiating events, by age (days) from commercial start date. At the left, the plot does not show events between the low power license date (9/29/86) and the start of data collection (1/1/87).

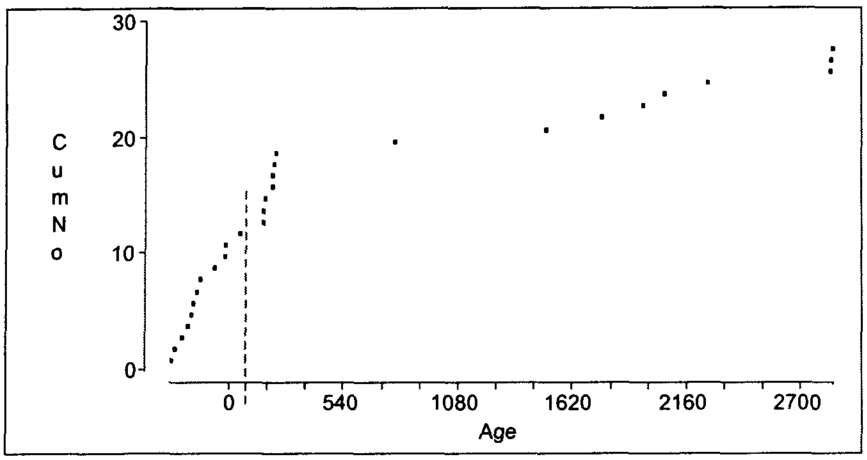

Figure E-6. Perry, cumulative number of initiating events, by age (days) from commercial start date. This plant went 20 months between the low power license and the commercial start. At the left, the plot does not show events between the low power license date (3/18/86) and the start of data collection (1/1/87).

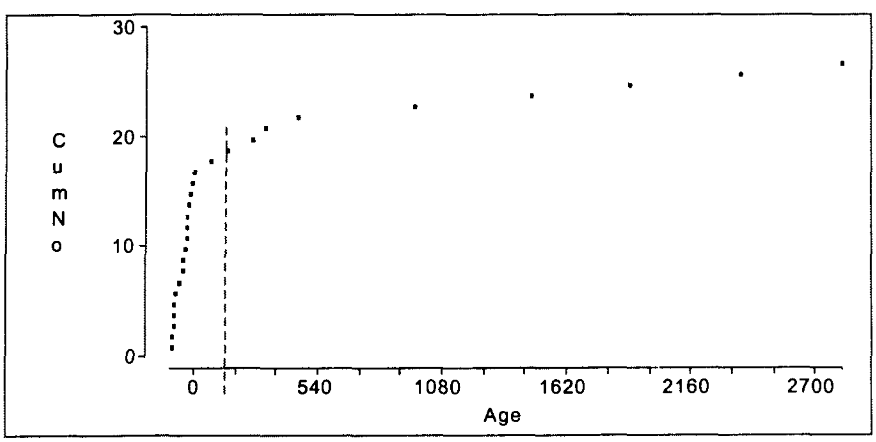

Figure E-7. Beaver Valley 2, cumulative number of initiating events, by age (days) from commercial start date. All events after the low power license date are shown.

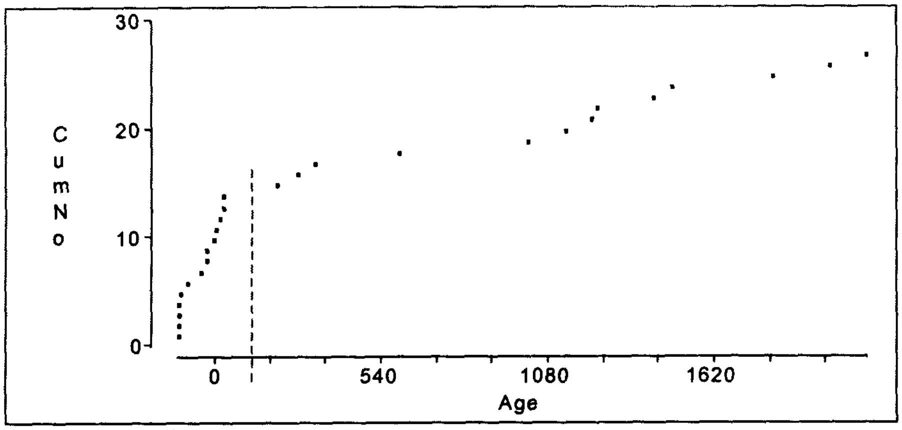

Figure E-8. Braidwood 2, cumulative number of initiating events, by age (days) from commercial start date. All events after the low power license date are shown.

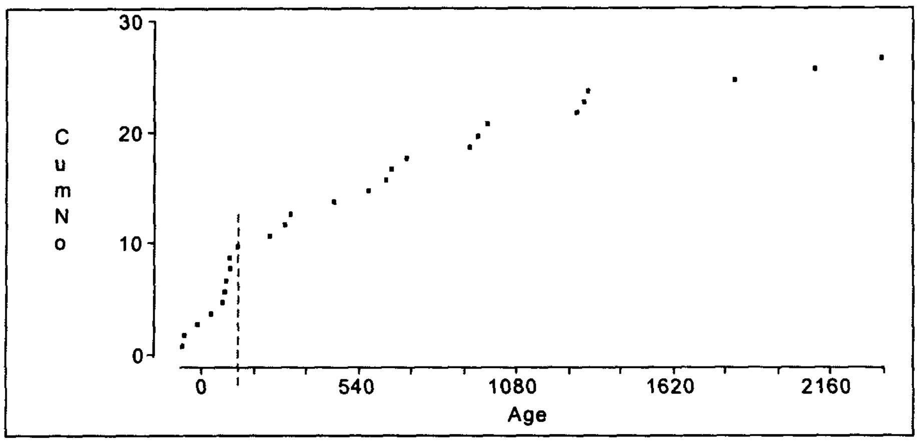

Figure E-9. South Texas 2. Cumulative number of initiating events, by age (days) from commercial start date. All events after the low power license date are shown.

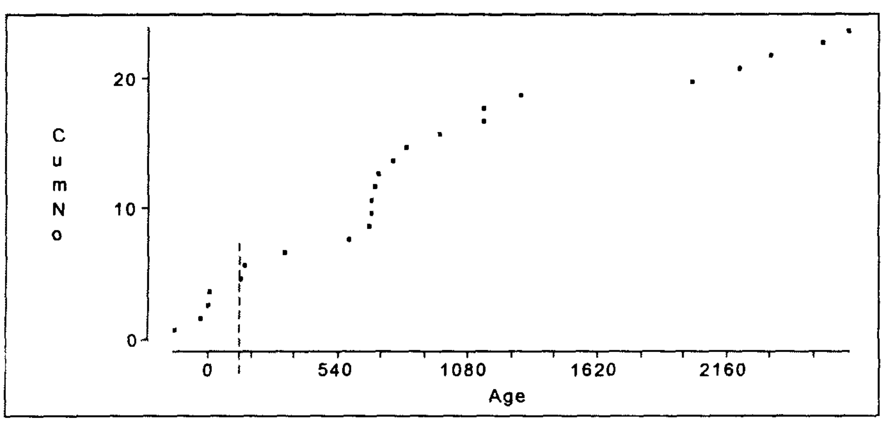

Figure E-10. South Texas 1, cumulative number of initiating events, by age (days) from commercial start date. All events after the low power license date are shown.

The total numbers of critical years in U. S. commercial reactors, 1987-1995, are shown in Table E-1

Table E-1. Critical years in U.S. reactors, 1987-1995.

	PWRs	BWRs
From Low Power License Date	498.55	229.74
From 4 Months after Commercial Start	491.25	226.01

ANSWERING THE QUESTION "IS THERE A TREND?"

The occurrence frequency, λ, of any kind of event is the average number of such events per plant time. For example, the loss of offsite power (LOSP) occurrence frequency for 1990 is the average number of LOSP events that would have occurred in 1990 per plant critical year. This is a theoretical quantity, the frequency that would have been observed if an infinitely large number of plants could have been observed, all in the condition of the actual plants in 1990. It is a large-population average. The *actual* number of events per plant critical year in 1990 differs from this average somewhat, because of the random nature of initiating events. The actual number of events reported for 1990, divided by the actual number of plant critical years in 1990, is an *estimate* of the underlying process parameter λ.

To assess how far the estimate might be from the underlying parameter, we must assume a model. Any model, such as constant occurrence frequency or exponentially decreasing frequency, is a simplification of reality. In particular, no frequency is really constant. Nevertheless, models are indispensable, both for presenting conclusions of a data analysis and for use as inputs to a probabilistic risk assessment (PRA).

In this study, a frequency with a trend was modeled as

$$\lambda = \exp(a + by)$$

where y is the calendar year. If b is zero, there is no trend. If b is negative, the trend is decreasing, and a plot of λ against y is an exponentially decreasing curve. Like λ, the parameters a and b are unknown parameters that apply to a hypothetical infinite population of plants. They are estimated from the limited observed data.

If there is really no trend ($b = 0$), the *estimate* of b may still differ substantially from 0, because of the random nature of the events. The *p-value* is defined as the probability of observing such an extreme b as a result of chance alone. It measures the strength of the evidence that b is nonzero.

To illustrate these ideas, consider Inadvertent Closure of All MSIVs, functional impact (FI) events in BWRs. Based on 70 events after the learning period, the estimate of b is -0.151. The p-value is defined as p-value = Prob($|$ estimated b from 70 events $| \geq 0.151$, if there is really no trend) .

This probability is calculated to be 0.0005, a very small number. A result is called *statistically significant* if the p-value is smaller than 0.05. Thus, based on a data set with 70 FI events, the trend is statistically very significant. It is customary to use 0.05 as a cutoff, but in principle a different cutoff could be used.

Figure E-11 plots the estimate of b and a 95% confidence interval for b, based on the 70 FI events. The fact that the p-value is much less than 0.05 corresponds to the fact that the 95% confidence interval for b is well to the left of zero.

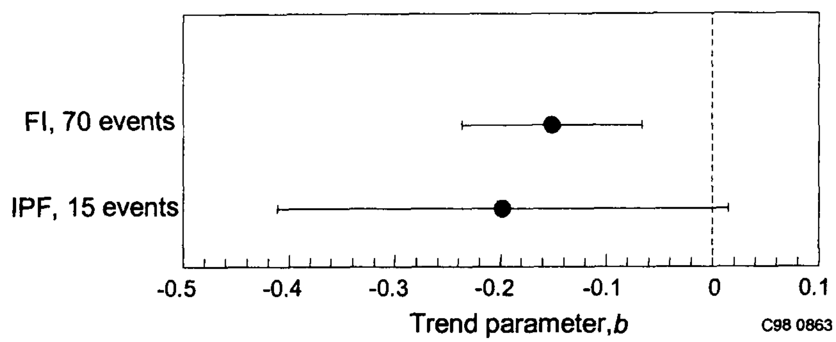

Figure E-11. Point estimates and 95% confidence intervals for trend parameter, b, for Inadvertent Closure of All MSIVs, based on 70 functional impacts and 15 initial plant faults.

For comparison, consider the 15 initial plant faults (IPFs) for Inadvertent Closure of All MSIVs. These events are a small subset of the 70 FI events. The estimated slope is –0.198. However, this estimate is based on only 15 events, so the estimate has considerable uncertainty. In fact, the p-value is

p-value = Prob($|$ estimated b from 15 events $| \geq 0.198$, if there is really no trend) = 0.062.

Note the paradox, that the estimated b is larger in absolute value for IPFs than for FIs, but the trend is not statistically significant, because the p-value is greater than 0.05. This illustrates the fact that a p-value does not directly measure the size of b; it measures the strength of the evidence that b is nonzero. The strength of the evidence depends on both the apparent size of the parameter and the number of events in the data set. In this example, because of the conflict between the p-value and the estimated magnitude of the trend, the decision of whether to model a trend for IPFs is not easy. It is discussed in the final section of this appendix, and in Tables F-1 and F-2 of Appendix F.

One might ask, "If some trend is always present, why not always model it?" The answer to this reasonable question is the following. A data set provides limited information. With enough ingenuity we could postulate a very complicated model, with many unknown parameters. In fact, five increasingly complex models are discussed in the subsection below. However, using an unnecessarily complex model wastes our limited resources, by estimating quantities that are negligible. The most efficient procedure is to focus on the important quantities, and to ignore the others. Therefore, if a trend is weak, too small to be clearly evident from the data, we normally do not model a trend at all.

In summary, one must interpret data models by remembering the following facts.

- Any model, such as no trend or an exponentially decreasing trend, is a simplification of reality, useful but not absolutely correct.

- It is impossible to prove that no trend is present. Even if no trend is seen, the possibility of a very slight trend can never be ruled out.

- A steep trend can be seen clearly with even a small data set. A very gradual trend can be seen only when the data set is large. This report models a trend if the trend is strong enough to be seen clearly (that is, to be statistically significant) in the observed data. The trend is statistically significant if a process with no trend would produce such a large *estimated* trend with probability < 0.05. The arbitrariness of 0.05 is acknowledged, but some cutoff is needed, and this one is customary. (The one exception to the rigid use of 0.05 is discussed in the final subsection of this appendix, on methods for choosing an appropriate model.)

- The statement "λ is modeled as constant" means that any trend was too slight to be clearly visible in the data. A small trend may in fact be present, and a larger data set might reveal that trend.

All the above discussion can be rephrased to deal with the question "Is there between-plant variation?" Just as with a trend, some between-plant variation always exists. However, this report models such variation only if it is large enough to be clearly evident in the data.

MODELS OF THE EVENT FREQUENCIES

The statistical method used to estimate the event occurrence frequency depended on the complexity of the data set. A data set with only a few event occurrences must be analyzed in a simple way. A data set with a large number of events occurrences requires more complicated modeling, so that the estimates can reflect the trends or patterns that are evident in the data. The five models that were used are described here, beginning with the simplest.

The assumption underlying all the models is that the events occur following a Poisson process, so that in any small time interval Δt, the probability of an event occurring is $\lambda\Delta t$. The basic properties of this model are described by Engelhardt (1994) and in many statistics books. The different models are determined by the form of λ, specifically, whether λ is constant, or dependent on the specific plant, or dependent on the calendar year, or dependent on both.

In every case, the desired result is a Bayesian distribution for the event occurrence frequency or frequencies. Such distributions can be used in PRAs. In some models, this Bayesian distribution is obtained directly, by using the data to update some prior distribution. The prior distribution either is chosen to be noninformative (not reflecting any strong prior information or belief), or is inferred from the data themselves. In other models, classical (non-Bayesian) methods are used, and the Bayesian distribution is constructed afterwards so that the Bayesian uncertainty intervals match the classical intervals. The result is a Bayesian distribution that depends on the data but not on prior information or belief.

The models are described here. A separate section explains the data-analysis methods used to decide which model is most appropriate.

Single Constant Frequency

Here λ is assumed to be the same for all plants and all time. This simple model is appropriate when very few events have occurred. Let n be the observed number of events in t critical hours. The Jeffreys noninformative prior distribution is updated by the data to produce a posterior distribution, which has a gamma form. The two parameters are the shape parameter, equal to $n + ½$, and the scale parameter, equal to t hours. This distribution can be used in PRAs. The mean of the distribution is $(n + ½)/t$. For further explanation, see Engelhardt (1994).

Although this possibility did not arise in the data, it could happen that boiling water reactors (BWRs) and pressurized water reactors (PWRs) each have a constant event occurrence frequency, which is different for the two types of reactor. In that case, the two data sets would be analyzed separately.

Constant Frequencies, Differing among Plants

This model says that the ith plant has an event frequency λ_i, which is constant over time but possibly different from the frequencies of the other plants. The model used was a parametric empirical Bayes model. We modeled the plants as belonging to a family, and treated any one plant as drawn randomly from the family. The distribution of λ_i within this family was modeled parametrically, and for mathematical convenience, the distribution was assumed to be a gamma(a, b) distribution. (During any data analysis, this assumption was checked to make sure that it was consistent with the data.) Therefore, the model was that λ_i for the ith plant is generated randomly from a gamma(a, b) distribution, and that the random number of failures in the observed t_i critical hours is Poisson with mean $\lambda_i t_i$.

The empirical Bayes method estimates a and b from the data. That is, the likelihood function for the data is based on the observed number of event occurrences and critical hours at each plant and the assumed gamma-Poisson model. This function of a and b was maximized through an iterative search of the parameter space, using a SAS routine given in Englehardt (1994). In order to avoid fitting a degenerate, spike-like distribution whose variance is less than the variance of the observed failure counts, the parameter space in this search was restricted to cases where b was less than the total number of observed critical hours. The a and b corresponding to the maximum likelihood were taken as estimates of the beta distribution parameters representing the observed data for the failure mode.

The resulting distribution was then updated by the data for each plant, to produce a plant-specific distribution for λ_i. A refinement, due to Kass and Steffey (1989) was also used, which adjusted these plant-specific distributions to account for the fact that a and b were only estimated, not known exactly. The form of each adjusted plant-specific distribution was approximated by a gamma distribution, which is printed in the report for possible use in PRA work. For further discussion, see Englehardt (1984).

Trend In Calendar Time, With No Differences Among Plants

When a trend in time was apparent, but no strong differences between plants were evident, the form of the occurrence frequency was modeled as

$$\lambda = \exp(a + by) \tag{E-1}$$

or equivalently, $\log(\lambda) = a + by$, where y denotes the calendar year. This model is a *loglinear* model, and methods for analyzing data from such a model are explained by Atwood (1995) and by certain advanced texts. If b is negative, as was the case for every data set analyzed with this model in this report, the trend is decreasing.

The SAS procedure GENMOD (SAS 1993) was used to analyze data using this model. This procedure uses a classical approach, not a Bayesian one. Denote $\exp(a + by)$ by $\lambda(y)$. GENMOD finds the maximum likelihood estimate (MLE) of $\log\lambda(y)$, denoted here by $\log\hat{\lambda}(y)$ and the standard error of the estimate, denoted here by $se(y)$. Using the approximate normality of the MLE, valid for large data counts, it produces approximate 90% confidence intervals, of the form

$$\log\hat{\lambda}(y) \pm 1.645 se(y) \ ,$$

because 1.645 is the 95th percentile of the normal distribution.

For use in PRA, this report uses the Bayesian distribution that gives the same uncertainty intervals as produced by GENMOD. This Bayesian distribution models $\log\lambda(y)$ as having a normal distribution with mean $\log\hat{\lambda}(y)$ and standard deviation equal to $se(y)$. Then a 90% interval (containing 90% of the probability determined by this Bayesian distribution) is exactly the same as the confidence interval just given. This Bayesian distribution can be interpreted as quantifying the uncertainty in $\log\lambda(y)$, based on the data and not on any prior information or belief.

Finally, the normal distribution for $\log\lambda(y)$ can be re-expressed as a lognormal distribution for $\lambda(y)$. This is the distribution presented in this report.

A modification of this model is the model

$$\log \lambda(y) = a + by + cI_{BWR} ,$$

where I_{BWR} is an indicator variable for the plant-type, 1 if the plant is a BWR and 0 if it is a PWR. This model has a single slope parameter, b, but different intercepts for the two plant types. If c is positive, in any year y the frequency $\lambda(y)$ is modeled as being greater at BWRs than at PWRs by a factor $\exp(c)$. This 3-parameter model is intermediate between the 2-parameter model (E-1) and the model that uses equation (E-1) with one pair of parameters for BWRs and a different pair of parameters for PWRs, for a total of 4 parameters.

To display the results of model (E-1) graphically, one could plot the fitted equation (E-1) with a solid line, and plot the confidence intervals (with ends above and below the fitted value) for each year, and then connect the ends of the confidence intervals with dotted lines. In this report, two modifications of such a plot are made. First, a confidence interval is constructed to be valid at a single year. Therefore, the band just described would not contain the entire true curve with 90% confidence—it would only contain the curve at any one year of interest with 90% confidence. Therefore, a slightly wider band is plotted, one that contains the entire curve with 90% confidence. (See page 34 of Atwood [1995] for details.) Second, in deference to the ultimate Bayesian use of the results, the Bayes mean is plotted, not the fitted value. The fitted value corresponds to the Bayes median, and can be somewhat smaller than the Bayes mean.

Trend In Calendar Time With Extra-Poisson Scatter

When fitting the above trend model, the goodness of fit was always examined. Lack of fit is seen if the estimated frequencies for the individual years are scattered around the trend line more than would be expected under the assumption of Poisson counts.

To model a trend with lack of fit, we assumed that the count during any year was not Poisson distributed, but instead had a negative binomial distribution. The negative binomial distribution was chosen because it is commonly used when extra-Poisson variance must be modeled. The mean count was assumed to change exponentially over time, and the coefficient of variation was assumed to be constant. This led to a three parameter model. The three parameters were estimated by maximum likelihood, and the asymptotic distribution of the maximum likelihood estimators was used to quantify the uncertainty in the estimates. Mathematically, this is identical to an empirical Bayes analysis with a trend in the mean; however the interpretation is different. The SAS program for performing the analysis was written and validated, as described in the LOSP report (Atwood et al. 1998). In the present study, the model was needed only for category B, Loss of Offsite Power.

If the trend term was not statistically significant in this model, the data set was fitted to a model with negative binomial counts and no trend. Mathematically, this is identical to an empirical Bayes model of between-year variation, but the interpretation is different.

Both Trend In Calendar Time And Differences Among Plants

In this model, the counts are again assumed to be Poisson distributed, and the event occurrence frequency satisfies

$$\log\lambda = a + by + \upsilon \ , \qquad \text{(E-2)}$$

where y is the calendar year and υ is an additive effect that depends on the particular plant. Assume that υ follows a normal distribution with mean 0 and standard deviation σ_υ, which is estimated by s_υ. That is, at a random plant, υ takes a value from this normal distribution. The mean of υ is assumed to be exactly zero because any nonzero value is absorbed into a.

The SAS macro GLIMMIX (Wolfinger 1997) was used to analyze data using this model. The method is documented to some extent (Wolfinger and O'Connell 1993) and is based on repeated calls to the SAS procedure MIXED (SAS 1992). In particular, GLIMMIX treats the expression

$$\frac{N-\lambda t}{\lambda t} + \log\lambda$$

as the response term in calls to MIXED, where N is the number of events for any particular plant and year, t is the corresponding number of critical hours, and λ is the event frequency for that plant and year. This response term has mean $\log\lambda$, and it is fitted to a line of form $a + by$. The fitting coefficients, a and b, are used to construct fitted values of λ, and the response expression is redefined using these fitted values of λ. MIXED is called again, and the process is repeated until it stabilizes.

Once the fixed effects have been accounted for in this way, the random effects must be evaluated. This involves estimating the between-plant variation, that is, the variance of υ, and the values of the individual plant effects. Several estimation methods are offered. The default, "restricted maximum likelihood," is explained by Searle et al. (1992). In simple examples it produces the usual unbiased estimates of the variance terms. This default method was used in all the analyses of this report.

To test the GLIMMIX approach, an example was considered with no time trend, only random plant effects. Both the empirical Bayes approach and GLIMMIX were then applied to the example. In both cases, the estimated event frequencies for extreme plants were pulled in somewhat toward the industry mean. In this example, GLIMMIX estimated the variance of the counts as smaller than calculated from the Poisson model.

Apparently as a result, GLIMMIX pulled the individual plant frequencies toward the industry mean less strongly than did the empirical Bayes method—GLIMMIX attributed more of the observed variation in the counts to true variation between plants and less to randomness of the counts.

When the variance of the counts is much more than the expected variance, this is evidence of underfit, or lack of fit. When the variance of the counts is much less than the expected variance, this is evidence of overfit. When we saw evidence of underfit or overfit, we calculated the square root of the ratio of the observed to the expected variance. This quantity estimates the factor by which a confidence band on the trend line is to narrow or too wide. In all cases, the factor was 25% or less, so no adjustment was made to the model.

The output from GLIMMIX can include estimates of user-specified quantities, such as estimates of $a + by$ for various values of y, and estimates of $\log \lambda_i = a + by + \upsilon_i$, the logarithm of the event-occurrence frequency at plant i. Any such estimate is accompanied by its standard error.

All the estimates are based on normal distributions. The normality of the estimators follows from the assumed large sample size, and the normality of υ is a model assumption. Therefore, each 90% interval is of the form (estimate ± 1.645 standard error), because 1.645 is the 95th percentile of the normal distribution. As in the other sections of this appendix, these non-Bayesian results were re-expressed as Bayes distributions, by using the distributions whose intervals matched the confidence intervals. These Bayes distributions are normal for the mean log occurrence frequency for some particular year or for $\log \lambda_i$, the log-frequency at plant i. In terms of the original plants, the Bayes distribution for any occurrence frequency is lognormal.

For example, let y correspond to the year 1995. If $\log(a + by)$ is estimated by $\log(\hat{a} + \hat{b}y)$, with standard error $se(y)$, a 90% confidence interval for $\log(a + by)$ is $\log(\hat{a} + \hat{b}y) \pm 1.645 \mathrm{se}(y)$. A 90% prediction interval for the log-frequency at a random plant in 1995 is

$$\log(\hat{a} + \hat{b}y) \pm 1.645 \sqrt{se^2(y) + s_\upsilon^2} \ .$$

The Bayes distribution that quantifies the uncertainty on the log-frequency at a random plant in 1995 is normal with mean $\log(\hat{a} + \hat{b}y)$ and variance $[se^2(y) + s_\upsilon^2]$. The Bayes distribution on the frequency itself, not the logarithm, is the corresponding lognormal distribution.

As a second example, denote the log-frequency at plant i in 1995 by $\log(\lambda_i)$. Suppose that it is estimated by $\log(\hat{\lambda}_i)$, with standard error se_i. The Bayes distribution for $\log(\lambda_i)$ is normal with mean $\log(\hat{\lambda}_i)$ and standard deviation se_i. The Bayes distribution for λ_i itself is the corresponding lognormal distribution.

For graphical display, a *prediction band* is used, of the form

$$\hat{a} + \hat{b}y \pm [se^2(y) + s_\upsilon^2]^{1/2} [\chi^2_{0.90}(2)]^{1/2} \ .$$

Denote the quantity in Equation (E-2) by $\log \lambda(y, \upsilon)$. The interpretation of the prediction band is

P[a random data set yields a band containing $\log\lambda(y,\upsilon)$ for all y and random υ] ≥ 0.90.

The rest of this subsection sketches a proof that the prediction band has the asserted property.

The band is derived as a modification of the band for fixed effects. The method was originally developed by Working, Hotelling, and Sheffé, and is modified here for the present application. Consider

$$\begin{pmatrix} \hat{a} - a + \upsilon \\ \hat{b} - b \end{pmatrix}.$$

Define the covariance matrix of this two-element vector to be **W**. Define **V** as the covariance matrix of $(\hat{a}, \hat{b})^{\mathrm{T}}$. Because $(\hat{a}, \hat{b})$ is based on past data and υ is based on a plant to be randomly chosen, it follows that $(\hat{a}, \hat{b})$ and υ are statistically independent. Therefore, **W** and **V** are related as follows:

$$\mathbf{V} = \begin{pmatrix} v_{11} & v_{12} \\ v_{21} & v_{22} \end{pmatrix} \qquad \mathbf{W} = \begin{pmatrix} v_{11} + \sigma_\upsilon^2 & v_{12} \\ v_{21} & v_{22} \end{pmatrix}.$$

Define $\mathbf{c}^{\mathrm{T}} = (1, y)$. Define **U** to be an invertible square matrix such that $\mathbf{U}^{\mathrm{T}}\mathbf{U} = \mathbf{W}$. We have

$$\left|\hat{a} + \hat{b}y - (a + by) + \upsilon\right| = \left|\mathbf{c}^T \begin{pmatrix} \hat{a} - a + \upsilon \\ \hat{b} - b \end{pmatrix}\right|$$

$$= \left|(\mathbf{Uc})^T (\mathbf{U}^T)^{-1} \begin{pmatrix} \hat{a} - a + \upsilon \\ \hat{b} - b \end{pmatrix}\right|$$

$$\leq \left|\mathbf{Uc}\right| \left|(\mathbf{U}^T)^{-1} \begin{pmatrix} \hat{a} - a + \upsilon \\ \hat{b} - b \end{pmatrix}\right|$$

by the Cauchy – Schwarz inequality

$$= \left|\mathbf{c}^T \mathbf{W} \mathbf{c}\right|^{1/2} \left|(\hat{a} - a + \upsilon, \hat{b} - b)\mathbf{W}^{-1} \begin{pmatrix} \hat{a} - a + \upsilon \\ \hat{b} - b \end{pmatrix}\right|^{1/2}$$

The second term in the product is the square root of a chi-squared random variable with two degrees of freedom, because of the definition of **W**. The first term in the product is $\left(\mathbf{c}^{\mathrm{T}}\mathbf{V}\mathbf{c} + \sigma_\upsilon^2\right)^{1/2}$. Here $\mathbf{c}^{\mathrm{T}}\mathbf{V}\mathbf{c}$ is the variance of $\hat{a} + \hat{b}y$. Denote the estimated standard deviation of $\hat{a} + \hat{b}y$ by *se*(*y*), the same notation used below Equation (E-1). Then $\left(\mathbf{c}^{\mathrm{T}}\mathbf{V}\mathbf{c} + \sigma_\upsilon^2\right)^{1/2}$ is estimated by $\left[se^2(y) + s_\upsilon^2\right]^{1/2}$. Therefore, the desired band is of the form

$$\hat{a} + \hat{b}y \pm \left[se^2(y) + s_\upsilon^2\right]^{1/2} \left[\chi^2_{0.90}(2)\right]^{1/2}.$$

As mentioned before, the interpretation of this prediction band is that

$$P\left\{\left|a + by + \upsilon - \hat{a} - \hat{b}y\right| \leq \left[se^2(y) + s_\upsilon^2\right]^{1/2} \left[\chi^2_{0.90}(2)\right]^{1/2}\right.$$

for a random data set, a random υ, and all $y\} \geq 0.90$.

The probabilities are approximate, because they rely on the asymptotic normal distributions with estimated variances.

METHODS FOR CHOOSING APPROPRIATE MODEL

The Pearson chi-squared test was performed to try to detect a statistically significant difference between years, with the data from different plants pooled. The test was also performed to try to detect a statistically significant difference between plants, with the data from different years pooled. Similarly, the test was performed to try to detect a statistically significant difference between plant types. In general, testing one effect at a time can be misleading, because the effects can be interrelated. In this case, however, nearly all the plants were observed for about the same time period, 1987–1995, so the confounding of effects is almost certainly small. To be safe, however, the data sets were also analyzed for the presence of simultaneous fixed effects (the two effects that appeared most nearly significant). This simultaneous analysis was performed except for two kinds of event categories: categories with fewer than 10 events, when no statistically significant effects had been seen in the one-at-a-time analyses; and detailed subcategories, when no simultaneous effects had been seen in the larger summary category.

To analyze a data set for two fixed effects, such as time trend and reactor type (BWR or PWR), the procedure GENMOD was used to analyze both in a single model. The statistical significance of adding a parameter is shown by GENMOD, and the usual cut-off of 0.05 was used to determine whether or not to call each parameter statistically significant. This cut-off was not applied mechanically, and judgment was used in the few borderline cases, as discussed in Appendix F.

When using GENMOD, we tried to use the Pearson chi-squared statistic to decide if a model fitted well enough. This statistic has an asymptotic chi-squared distribution, if the sample size is large. If the Pearson chi-squared statistic was close to the likelihood-ratio chi-squared statistic, the sample size was deemed acceptable for the asymptotic distribution. If not, the Pearson statistic was noted, but the results were regarded as inconclusive. When examining the adequacy of the model we calculated the test statistics in two ways, with the plants pooled and with each plant contributing a separate datum for each year.

The same method was used with GLIMMIX, except the likelihood-ratio chi-squared statistic is called the deviance in the GLIMMIX output. In these analyses, the plants were never pooled.

When analyzing a data set for between-plant variation only, two approaches were used. First, we attempted to compute the empirical Bayes distribution. If no non-degenerate empirical Bayes distribution could be found, we did not (could not) proceed with empirical Bayes modeling. If, however, an empirical Bayes distribution could be calculated, we used the Pearson chi-squared test of equality of the frequencies at the various plants, with 0.05 as the cut-off for deciding whether to model the differences between plants.

One exception was made to this use of 0.05. For any single category, the data sets for functional impacts (FIs) and for initial plant faults (IPFs) were typically similar, but the FI data set included the IPF data set as a subset. If a trend or between-plant variation was statistically significant in the FI data, but not quite statistically significant in the smaller IPF data, we modeled the pattern as being present in both data sets, even though the smaller IPF data set was not quite large enough to give statistical significance. The individual decisions, and their bases, are given in Appendix F.

Further, when between-plant variation was detected, actual numerical differences were examined in addition to statistical significance. Actual numerical differences were measured by considering the ratio of the largest plant frequency (Bayes mean) divided by the smallest plant frequency. If the ratio was larger than 6, the plant-specific frequencies were presented in this report, in tabular and graphical form. (The number 6 was chosen after examination of the data; in no cases was the ratio between 4 and 6.) Otherwise, the plant-specific frequencies were not presented individually. Instead, only the industry distribution, which included between-plant variation, was presented.

When an apparent difference between reactor types (BWRs and PWRs) was seen, we sought an engineering basis for the difference. However, we did not seek an engineering explanation for time trends or for differences between plants.

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Appendix F

Results of Testing for Time Trend and Plant Effect

Appendix F

Results of Testing for Time Trend and Plant Effect

This section summarizes the findings when the methods of Appendix E were used to choose the appropriate model for each data set. The results in each case are summarized in Tables F-1 and F-2. Table F-1 considers the data after the plants' learning periods. Table F-2 considers the full data set, but only those categories and headings where the event count differs from the count in Table F-1. The functional impact (FI) categories were analyzed first, because they were larger than the corresponding initial plant fault (IPF) categories. Therefore, the tables list the functional impact categories before the initial plant fault categories.

Table F-1. Bases for choices of models when using only events after the learning period.

FI B, Loss of Offsite Power —30 events

Difference between years is statistically significant (p-value = 0.045). When a trend is modeled, there is statistically significant lack of fit (p-value = 0.03). This is interpreted below as extra random scatter in the counts, beyond what is expected if counts have Poisson distribution. Therefore, a model was fit allowing for trend, and assuming negative binomial counts instead of Poisson counts, hence larger variance.

Trend in year not statistically significant (p-value = 0.16 when extra-Poisson scatter accounted for). No plant effect evident.

Conclusion: Model no effects, but account for extra-Poisson scatter by assuming negative binomial counts, not Poisson counts. This is the same approach used in the LOSP report (Atwood et al. 1998).

IPF B, Loss of Offsite Power—16 events

Difference between years is nearly statistically significant (p-value = 0.06). When trend is modeled, lack of fit is statistically significant (p-value = 0.04). Although not quite statistically significant, we follow the model for FI B, and interpreted this below as extra random scatter in the counts, beyond what is expected if counts have Poisson distribution.

Trend in year not statistically significant (p-value = 0.15 when extra-Poisson scatter accounted for). No plant effect evident.

Conclusion: Just as for FI B, model no effects, but account for extra-Poisson scatter by assuming negative binomial counts, not Poisson counts.

FI C, Loss of Safety-Related Bus—16 events

No trend in time, no differences between plants.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI C1 or C2, Loss of Vital ac Bus—16 events

No trend in time, no differences between plants.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI C3, Loss of Vital dc Bus—1 event

Too few events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

Table F-1. (continued).

IPF C, Loss of Safety-Related Bus—11 events

No trend in time, no differences between plants.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF C1 or C2, Loss of Vital ac Bus—11 events (same events as IPF C)

No trend in time, no differences between plants.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF C3, Loss of Vital dc Bus—0 events

Too few events to show a pattern.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI D, Loss of Instrument or Control Air System—30 events

Decreasing trend in year (p-value = 0.005).

Effect of plant type (BWR, 19 events, higher than PWR, 11 events in more time; p-value = 0.0004).

The interaction between plant-type and year-trend is not statistically significant.

Pearson chi-square shows good fit to model with common trend and effect of plant type.

Conclusion:

- Model common trend, multiplicative effect for plant type.
- Give confidence bands on decreasing rates, use lognormal distributions for 1995 rates.

IPF D, Loss of Instrument or Control Air System —20 events

Decreasing trend in year (p-value = 0.006).

There is an effect of plant type (BWR, 11 events, higher than PWR, 9 events; p-value = 0.03) just as seen for FI D.

Pearson chi-square shows good fit to this model.

Conclusion:

- Model common trend, multiplicative effect for plant type.
- Give confidence bands on decreasing rates, use lognormal distributions for 1995 rates.

FI E1, Total Loss of Service Water—no events in the 1987–1995 experience

Too few events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use the combined total U.S. operating experience (1969–1997) for BWRs and PWRs (1 event).

FI E2, Partial Loss of Service Water—6 events

No differences between plants evident.

In spite of the small number of events, the trend is almost statistically significant (p-value is calculated as 0.052, but this is an asymptotic approximation). It is difficult to decide whether trend is present. Because of the sparseness of the data, and to be conservative, we do not model a trend.

Conclusion: model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

Table F-1. (continued).

IPF E1, Total Loss of Service Water—no events
Too few events to show patterns.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use the combined total U.S. operating experience (1969–1997) for BWRs and PWRs (0 event).

IPF E2, Partial Loss of Service Water—0 events
Too few events to show any pattern.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI F, Steam Generator Tube Rupture (PWR)—3 events
Too few events to show any pattern.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF F, Steam Generator Tube Rupture (PWR) —3 events
Same three events as for functional impact.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI G1, Very Small LOCA/Leak—4 events
Too few events to show patterns.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI G2, Stuck Open: 1 Safety/Relief Valve—2 events PWR
Effect of plant type (p-value=0.0002) is confirmed by engineering considerations.
Too few PWR events to show additional patterns.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI G2, Stuck Open: 1 Safety/Relief Valve—10 events BWR
Effect of plant type (p-value=0.0002) is confirmed by engineering considerations.
No trend in time, no differences between plants evident.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI G3, FI G6, FI G7—no events
Too few events to show patterns.
Conclusion: Operating experience from U.S. and foreign reactors, and evaluations of engineering aspects of pipe break LOCAs were used to generate frequencies. (Refer to Appendix J.)

FI G4, Stuck Open: Pressurizer PORV—no events
Too few events to show patterns.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use 1987–1995 operating experience for PWRs (no events).

Table F-1. (continued).

FI G5, Stuck Open: 2 or More Safety/Relief Valves—no events

Too few events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use the combined total U.S. operating experience (1969–1997) for BWRs and PWRs (no events).

FI G8, Reactor Coolant Seal LOCA—no events in the 1987–1995 experience

Too few events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use the total U.S. operating experience (1969–1997) for PWRs (2 events).

IPF G2, Stuck Open: 1 Safety/Relief Valve—0 events PWR

Effect of plant type (p-value=0.0000) is confirmed by engineering considerations.

Too few PWR events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF G2, Stuck Open: 1 Safety/Relief Valve—10 events BWR

Effect of plant type (p-value=0.0000) is confirmed by engineering considerations.

These are the same events as for the FI G2 (BWR).

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF G3, FI G6, FI G7—no events

Too few events to show patterns.

Conclusion: Operating experience from U.S. and foreign reactors, and evaluations of engineering aspects of pipe break LOCAs were used to generate frequencies. (Refer to Appendix J.)

IPF G4, Stuck Open: Pressurizer PORV—no events

Too few events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use 1987–1995 operating experience for PWRs (no events).

IPF G5, Stuck Open: 2 or More Safety/Relief Valves—no events

Too few events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use the combined total U.S. operating experience (1969–1997) for BWRs and PWRs (no events).

IPF G8, Reactor Coolant Pump Seal LOCA--no events in the 1987–1995 experience

Too few events to show patterns.

Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate. Use the total U.S. operating experience (1969–1997) for PWRs (2 events). Same two events as for functional impact.

FI H, Fire—38 events

If constant rate assumed, some between-plant variation may be present (p-value = 0.036, but this is an approximate calculation, based on only 38 events among 112 plants). Empirical Bayes analysis gives ratio of highest plant-specific rate to lowest plant-specific rate = 4.2, relatively small.

Table F-1. (continued).

If between-plant differences ignored, trend is present (p-value = 0.044). Fit is good, with scatter about what would be expected from Poisson counts; therefore, we have no evidence here of variance unaccounted for, such as between-plant variation. Furthermore, pooling data in 1987-90 and in 1991-95 results in statistically significant difference between the two time periods (p-value = 0.032).
The data set is too small to allow modeling of both trend and between-plant differences.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

IPF H, Fire—30 events
Similar results as for FI H, except no evidence of differences between plants.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

FI J, Flood—2 events
Too few events to show a pattern
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF J, Flood—1 event
Too few events to show a pattern
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI K, High Energy Line Break —8 events
No trend in time, no differences between plants evident.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant rate.

FI K1 through K3—6 or fewer events
Too few events to show patterns.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF K, High Energy Line Break—8 events
These are the same events as for FI K.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF K1 through K3—6 or fewer events
These are the same events as for the FI categories.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI L, Total Loss of Condenser Heat Sink—184 events
There is a plant-type effect (BWRs higher than PWRs).
Rather than using a single model with a factor for plant-type, we get a better fit by doing separate analyses on BWRs and PWRs as follows.

PWRs—71 events
Between-plant variance is present (p-value = 0.0000). The ratio of highest plant-specific rate/lowest plant-specific rate is 6.7.
No overall statistically significant trend (p-value = 0.11 when between-plant differences ignored).

Table F-1. (continued).

Modeling both trend in time and between-plant variation produces barely significant trend (p-value = 0.049) and evidence of overfit.
Conclusion: Model effect of all plants by empirical Bayes. Present plant-specific rates.

BWRs—113 events
Decreasing trend in year (p-value = 0.001).
No lack of fit from individual plants.
No between plant differences seen when trend is ignored.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

FI L1, Inadvertent Closure of All MSIVs—103 events

PWRs—33 events
Decreasing trend in year marginally statistically significant (p-value = 0.048), and good fit.
No statistically significant differences between plants.
Conclusion: Give confidence bands on decreasing rate, use lognormal distribution for 1995 rate.

BWRs—70 events
Decreasing trend in year (p-value = 0.0009). Very good fit of the data to this model.
Between-plant differences: statistically significant (p-value = 0.02), but ratio of largest plant-specific mean/lowest plant specific mean is only 3.0.
Modeling both trend in time and between-plant variation produces some evidence of overfit, and band around the trend line that may be too wide by 10%. Ratio of largest plant-specific mean/lowest plant-specific mean is 3.0.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific rates, but use between-plant distribution to quantify uncertainty interval for 1995.

FI L2, Loss of Condenser Vacuum — 76 events

PWRs—34 events
No evidence at all of trend.
Between-plant variation is present (p-value = 0.0000), and ratio of highest plant-specific mean/lowest plant-specific mean = 18.4.
Conclusion: Model effect of all plants by empirical Bayes.

BWRs — 42 events
No statistical significance trend (p-value = 0.11)
No between-plant variation (p-value = 0.13)
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI L3, Turbine Bypass Unavailable—8 events in all reactor types
No significant difference between reactor types.
No statistically significant trend in time (p-value = 0.12)
No differences between plants.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate for industry.

Table F-1. (continued).

IPF L, Total Loss of Condenser Heat Sink—58 events

PWRs—18 events
No trend in time.
Significant between-plant variation (p-value = 0.026), but ratio of highest plant-specific rate/lowest plant-specific rate is 7.2.
Conclusion: Model effect of all plants by empirical Bayes. Present plant-specific rates. This is the same presentation as used for FI L for PWRs.

BWRs—40 events
Trend in time not quite statistically significant (p-value = 0.07). But magnitude of the trend is about the same as for FI L for BWRs. Furthermore, pooling data in 1987-90 and in 1991-95 results in statistically significant difference between the two time periods (p-value = 0.028).
No between-plant variation seen.
Conclusion: By analogy with FI L, give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

IPF L1, Inadvertent Closure of All MSIVs—20 events

PWRs—5 events
Too few events to show any pattern.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

BWRs—15 events
Trend in year not quite statistically significant (p-value = 0.06). However, estimated magnitude of trend is 30% larger than for FI L1 for BWRs. Furthermore, pooling data in 1987-90 and in 1991-95 results in statistically significant difference between the two time periods (p-value = 0.038).
No between-plant differences.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

IPF L2, Loss of Condenser Vacuum—37 events

PWRs—13 events
No evidence of trend.
Between-plant variation is present (p-value = 0.002), and ratio of highest plant-specific mean/lowest plant-specific mean = 14.6.
Conclusion: Model between-plant differences by empirical Bayes.

BWRs—24 events
No trend in time.
No statistically significant between-plant variation (p-value = 0.13).
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF L3 Turbine Bypass Unavailable—1 event in all reactor types
Too few events to show patterns.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate for industry.

Table F-1. (continued).

FI P, Total Loss of Feedwater Flow—132 events

Between-plant variation is present (p-value = 0.0001 when time trend ignored), but ratio of highest plant-specific mean/lowest plant-specific mean is only 4.8.

Decreasing trend is present (p-value = 0.0001 when differences between plants ignored). Fit is adequate (approximate p-value for lack of fit = 0.12).

When both trend and between-plant differences are simultaneously modeled, the resulting ranking of worst plants is similar to that from empirical Bayes, though not exactly the same. The ratio between the highest plant-specific mean/lowest plant-specific mean is 9.3. The model appears to overfit the data, although the data set is too small to permit an accurate assessment of the goodness of fit. The estimated effect of the overfit is to make the uncertainty interval for 1995 too wide by about 10%. Rather than making such an adjustment, we instead check to see that the model gives reasonable numbers.

To check the reasonableness of the most complex model (the model that includes both a trend and between-plant differences), consider the slope of the trend and the dispersion around the trend:

- The slope of the trend (slope of $\log\lambda$) is about the same whether or not between-plant differences are modeled. The slopes are −0.185 and −0.184, respectively.
- Suppose we do not model trend, and consider two possibilities, with and without between-plant differences. Modeling between-plant differences increases the "error factor" (= upper limit/best est.) by about a factor of 2. Suppose instead that we model trend, and again consider two models, with and without between-plant differences. Modeling between-plant differences increases the "error factor" by about a factor of 2. This is similar to the effect when no trend was modeled.
- These two considerations indicate that the most complex model appears to give reasonable answers in spite of the overfit.

Conclusion: Model both trend and between-plant differences. Give prediction band on decreasing rate at a random plant. Use lognormal distribution for 1995 rate at a random plant. Present plant-specific results for 1995.

IPF P, Total Loss of Feedwater Flow—72 events

Between-plant variation is present (p-value = 0.0001 when time trend ignored), but ratio of highest plant-specific mean/lowest plant-specific mean is only 5.4.

Decreasing trend is present (p-value = 0.0009 when differences between plants ignored). Fit is very good.

As with FI P, modeling both trend and between-plant differences appears to overfit the data (with the effect of making the interval for 1995 too wide by an estimated 25%.) The ratio of highest plant-specific mean/lowest plant-specific mean is 12.4. Rather than shrinking the interval by an estimated amount, we performed the same checks as for FI P, and conclude that the most complex model appears to give reasonable answers.

Conclusion: Model both trend and between-plant differences. Give prediction band on decreasing rate at a random plant. Use lognormal distribution for 1995 rate at a random plant. Present plant-specific results for 1995.

IPF Q-P, General Transient (PWR)—1,070 events

Between-plant variation is present (p-value = 0.0000, when trend is ignored).

Trend is present (p-value = 0.0001, when between-plant differences are ignored).

When both trend and between-plant differences are modeled, there is some evidence of underfit, additional variation beyond that of the assumed Poisson distribution. It may be a result of pooling somewhat diverse categories. The estimated effect is to make the error bands too narrow by about 20%.

Table F-1. (continued).

The same checks made for FP indicate that the most complex model appears to give reasonable answers. Using this model, the ratio of the highest plant-specific rate in 1995 to the lowest such mean is 2.6.
Conclusion: Model both trend and between-plant differences. Give prediction band on decreasing rate at a random plant. Use lognormal distribution for 1995 rate at a random plant. Do not present plant-specific results. Mention that uncertainty bands may be somewhat too narrow because of extra variance that is not accounted for.

IPF Q-B, General Transient (BWR)—507 events

Between-plant variation is present (p-value = 0.0005, when trend is ignored), but ratio of highest plant-specific mean/lowest plant-specific mean is only 2.3.
Trend is present (p-value = 0.0001, when between-plant differences are ignored).
When both trend and between-plant differences are modeled, there is some evidence of underfit, additional variation beyond that of the assumed Poisson distribution. It may be a result of pooling somewhat diverse categories. The estimated effect is to make the error bands too narrow by about 10%. Using this model, the ratio of highest plant-specific mean in 1995 to the lowest such mean is only 2.0.
The same checks made for FP indicate that the most complex model appears to give reasonable answers.
Conclusion: Model both trend and between-plant differences. Give prediction band on decreasing rate at a random plant. Use lognormal distribution for 1995 rate at a random plant. Do not present plant-specific results. Mention that uncertainty bands may be slightly too narrow because of extra variance that is not accounted for.

All PWR Transients—1,198 events

Because almost 90% of these events are the general transient category, these events are to be modeled the same way, modeling both trend and between-plant variation. The ratio of the highest plant-specific mean to the lowest is only 2.9. There is evidence of underfit, which has the effect of making the error bands too narrow by an estimated 20%.
Conclusion: Model both trend and between-plant differences. Give prediction band on decreasing rate at a random plant. Use lognormal distribution for 1995 rate at a random plant. Do not present plant-specific results. Mention that uncertainty bands may be somewhat too narrow because of extra variance that is not accounted for.

All BWR Transients—610 events

Because over 80% of these events are the general transient category, these events are to be modeled the same way, modeling both trend and between-plant variation. The ratio of the highest plant-specific mean to the lowest is only 2.0. There is evidence of underfit, which has the effect of making the error bands too narrow by an estimated 13%.
Conclusion: Model both trend and between-plant differences. Give prediction band on decreasing rate at a random plant. Use lognormal distribution for 1995 rate at a random plant. Do not present plant-specific results. Mention that uncertainty bands may be slightly too narrow because of extra variance that is not accounted for.

Table F-2. Bases for choices of models when using all data after Low Power License Date. Only cases with different counts from Table F-1 are shown.

FI B, Loss of Offsite Power —33 events

Difference between years is statistically significant (p-value = 0.045). When a trend is modeled, there is statistically barely significant lack of fit (p-value = 0.047). This is interpreted below as extra random scatter in the counts, beyond what is expected if counts have Poisson distribution. Therefore, a model was fit allowing for trend, and assuming negative binomial counts instead of Poisson counts, hence a larger variance.

Trend in year not quite statistically significant (p-value = 0.052 when extra-Poisson scatter accounted for). Not modeling a trend is conservative.

No plant effect evident.

Conclusion: Model no effects, but account for extra-Poisson scatter by assuming negative binomial counts, not Poisson counts. This is the same approach used in the LOSP report (Atwood et al. 1998).

IPF B, Loss of Offsite Power—17 events

Difference between years is not statistically significant (p-value = 0.090). When trend is modeled, lack of fit is not quite statistically significant (p-value = 0.063). Although not quite statistically significant, we follow the model for FI B, and interpret the between-year variance as extra random scatter in the counts, beyond what is expected if counts have Poisson distribution.

Trend in year not statistically significant (p-value = 0.066 when lack of fit is ignored, p-value = 0.095 when extra-Poisson scatter accounted for).

No plant effect evident.

Conclusion: Model no effects, but account for extra-Poisson scatter by assuming negative binomial counts, not Poisson counts.

FI D, Loss of Instrument or Control Air System—36 events

Decreasing trend in year (p-value = 0.0002).

Effect of plant type (BWR, 21 events, higher than PWR, 15 events in more time; p-value = 0.009).

The interaction between plant-type and year-trend is not statistically significant.

Pearson chi-square shows good fit to model with common trend and effect of plant type.

Conclusion:

- Model common trend, multiplicative effect for plant type.
- Give confidence bands on decreasing rates, use lognormal distributions for 1995 rates.

IPF D, Loss of Instrument or Control Air System —26 events

Decreasing trend in year (p-value = 0.0002).

There is an effect of plant type (BWR, 13 events, higher than PWR, 13 events; p-value = 0.045) just as seen for FI D.

Pearson chi-square shows good fit to this model.

Conclusion:

- Model common trend, multiplicative effect for plant type.
- Give confidence bands on decreasing rates, use lognormal distributions for 1995 rates.

FI H, Fire—39 events

If constant rate is assumed, between-plant variation is not quite statistically significant (p-value calculated as 0.073, although this is an approximate calculation, based on only 39 events among 112 plants).

Table F-2. (continued).

If between-plant differences are ignored, trend is present (p-value = 0.033). Fit is good, with scatter about what would be expected from Poisson counts. Furthermore, pooling data in 1987-90 and in 1991-95 results in statistically significant difference between the two time periods (p-value = 0.028).
The data set is too small to allow modeling of both trend and between-plant differences.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

IPF H, Fire—31 events
Similar results as for FI H, except no evidence of differences between plants.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

FI K, High Energy Line Break —9 events
No trend in time, no differences between plants evident.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant rate.

FI K1 through K3—7 or fewer events
Too few events to show patterns.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF K, High Energy Line Break—9 events
These are the same events as for FI K.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

IPF K1 through K3—7 or fewer events
These are the same events as for the FI categories.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

FI L, Loss of Condenser Heat Sink—197 events
There is a plant-type effect (BWRs higher than PWRs).
Rather than using a single model with a factor for plant-type, we get a better fit by doing separate analyses on BWRs and PWRs as follows.

PWRs—75 events
Between-plant variance is present (p-value = 0.0000). The ratio of the highest plant-specific mean to the lowest is not meaningful, because the data collection period contains the learning periods of only some of the plants.
Statistically significant trend (p-value = 0.034 when between-plant differences ignored). This differs from the result in Table F-1, with the difference resulting from the learning-period data.
Modeling both trend in time and between-plant variation produces some evidence of overfit, though the data set is too small to allow an accurate assessment of goodness of fit.
Conclusion: Model both between-plant differences and time trend. Do not present plant-specific rates, but use between-plant distribution to quantify uncertainty interval for 1995.

BWRs—122 events
Decreasing trend in year (p-value = 0.001).
No lack of fit from individual plants.
No between plant differences seen when trend is ignored.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

Table F-2. (continued).

FI L1, Inadvertent Closure of All MSIVs—109 events

PWRs—35 events
Decreasing trend in year statistically significant (p-value = 0.022), and good fit.
No statistically significant differences between plants.
Conclusion: Give confidence bands on decreasing rate, use lognormal distribution for 1995 rate.

BWRs—74 events
Decreasing trend in year (p-value = 0.0002). Very good fit of the data to this model.
Between-plant differences is barely statistically significant (p-value = 0.048).
Modeling both trend in time and between-plant variation produces good fit.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific rates, but use between-plant distribution to quantify uncertainty interval for 1995.

FI L2, Loss of Condenser Vacuum — 81 events

PWRs—35 events
No evidence at all of trend.
Between-plant variation is present (p-value = 0.0000). The ratio of the highest plant-specific mean to the lowest is not meaningful, because the data collection period contains the learning periods of only some of the plants.
Conclusion: Model effect of all plants by empirical Bayes.

BWRs—46 events
No statistical significance trend (p-value = 0.12)
Statistically significant between-plant variation (p-value = 0.018). This differs from the result in Table F-1, with the difference resulting from the learning-period data.
Conclusion: Model effect of all plants by empirical Bayes. Do not present plant-specific rates.

FI L3, Turbine Bypass Unavailable—10 events in all reactor types
No significant difference between reactor types.
Statistically significant trend in time (p-value = 0.023). This differs from the result in Table F-1, with the difference resulting from the learning-period data.
No differences between plants.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

IPF L, Loss of Condenser Heat Sink—64 events

PWRs—19 events
No trend in time.
Significant between-plant variation (p-value = 0.038). The ratio of the highest plant-specific mean to the lowest is not meaningful, because the data collection period contains the learning periods of only some of the plants.
Conclusion: Model effect of all plants by empirical Bayes. Do not present plant-specific rates.

BWRs—45 events
Trend in time is statistically significant (p-value = 0.006). This differs from the result in Table F-1, with the difference resulting from the learning-period data.
No between-plant variation seen.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

Table F-2. (continued).

IPF L1, Inadvertent Closure of All MSIVs—21 events

PWRs—5 events
Too few events to show any pattern.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate.

BWRs—16 events
Trend in year statistically significant (p-value = 0.037).
No between-plant differences.
Conclusion: Give confidence band on decreasing rate, use lognormal distribution for 1995 rate.

IPF L2, Loss of Condenser Vacuum—37 events

PWRs—13 events
No evidence of trend.
Between-plant variation is present (p-value = 0.001). The ratio of the highest plant-specific mean to the lowest is not meaningful, because the data collection period contains the learning periods of only some of the plants.
Conclusion: Model between-plant differences by empirical Bayes. Do not present plant-specific results, but use between-plant distribution to quantify uncertainty interval for 1995.

BWRs—27 events
Trend in time is not statistically significant (p-value = 0.13).
Statistically significant between-plant variation (p-value = 0.012). This differs from the result in Table F-1, with the difference resulting from the learning-period data. The ratio of the highest plant-specific mean to the lowest is not meaningful, because the data collection period contains the learning periods of only some of the plants.
Conclusion: Model between-plant differences by empirical Bayes. Do not present plant-specific results.

IPF L3 Turbine Bypass Unavailable—3 event in all reactor types
Too few events to show patterns, though a trend is suggested (p-value = 0.09) because two of the three events occurred during plants' learning periods.
Conclusion: Model no effects, use Bayes updated distribution with a Jeffreys noninformative prior for constant generic rate for industry.

FI P, Total Loss of Feedwater Flow—159 events
Between-plant variation is present (p-value = 0.0000 when time trend ignored)
Decreasing trend is present (p-value = 0.0001 when differences between plants ignored). Fit is poor (approximate p-value for lack of fit = 0.017). Plants with bad learning periods contribute to this lack of fit.
When both trend and between-plant differences are simultaneously modeled, the model appears to overfit the data slightly, although the data set is too small to permit an accurate assessment of the goodness of fit.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific results, but use between-plant distribution to quantify uncertainty interval for 1995.

IPF P, Total Loss of Feedwater Flow—86 events
Between-plant variation is present (p-value = 0.0000 when time trend ignored).
Decreasing trend is present (p-value = 0.0001 when differences between plants ignored). Fit is good.

Table F-2. (continued).

Modeling both trend and between-plant differences appears to overfit the data (with the effect of making the interval for 1995 too wide by an estimated 20%.) However, the data set is too small to permit an accurate assessment of the goodness of fit.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific results, but use between-plant distribution to quantify uncertainty interval for 1995.

IPF Q-P, General Transient (PWR)—1,184 events

Between-plant variation is present (p-value = 0.0000, when trend is ignored).
Trend is present (p-value = 0.0001, when between-plant differences are ignored). The fit is bad, with p-value = 0.002; this small p-value indicates strong evidence against the assumed model. A contributor to the lack of fit is the presence of some plants with bad years, resulting from their learning periods. The model does not allow for such sudden changes of some plants.
When both trend and between-plant differences are modeled, there is very strong evidence of underfit, additional variation beyond that of the assumed Poisson distribution. Plants with high counts during their learning periods contribute to this lack of fit. The estimated effect is to make the error bands too narrow by about 40%.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific results, but use between-plant distribution to quantify uncertainty interval for 1995.

IPF Q-B, General Transient (BWR)—541 events

Between-plant variation is present (p-value = 0.0000, when trend is ignored).
Trend is present (p-value = 0.0001, when between-plant differences are ignored).
When both trend and between-plant differences are modeled, there is some evidence of underfit, additional variation beyond that of the assumed Poisson distribution. The estimated effect is to make the error bands too narrow by about 12%.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific results, but use between-plant distribution to quantify uncertainty interval for 1995.

All PWR Transients—1,327 events

Because almost 90% of these events are the general transient category, these events are to be modeled the same way, modeling both trend and between-plant variation. There is very strong evidence of underfit, which has the effect of making the error bands too narrow by an estimated 40%.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific results, but use between-plant distribution to quantify uncertainty interval for 1995.

All BWR Transients—658 events

Because over 80% of these events are the general transient category, these events are to be modeled the same way, modeling both trend and between-plant variation. There is evidence of underfit, which has the effect of making the error bands too narrow by an estimated 16%.
Conclusion: Model both trend and between-plant differences. Do not present plant-specific results, but use between-plant distribution to quantify uncertainty interval for 1995.

REFERENCE

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Appendix G

Results Based on Data after Learning Period, Including Plant-Specific Results and Time Trends

Appendix G

Results Based on Data after Learning Period, Including Plant-Specific Results and Time Trends

This appendix gives results based on the data with the early-in-life data excluded, the period up to four months after the commercial start date. Tables G-1 and G-2 show the industry rates for the functional impact and initial plant fault categories, respectively. They are similar in form to Table 3-1, but based on the restricted data set. This appendix provides results of both initial plant fault and functional impact categories.

The appendix also gives plant-specific event rates, in both tabular and graphical form, and plots of modeled time trends. It includes event categories for which differences were seen between plants or years. For some categories, the between-plant differences were statistically significant, but still small in absolute terms. A p-value only measures the strength of the evidence, and the evidence of between-plant differences can be strong when many events have occurred, even if the magnitude of the differences is small. For example, the category General Transients for PWRs had a time trend and between-plant differences that were both statistically significant. However, the ratio of the highest plant-specific mean to the lowest was only 2.6 in any one year.

Table G-3 lists the categories and headings that show statistically significant evidence of between-plant differences. The categories in Table G-3 are ordered according to ratio of highest plant-specific mean to lowest plant-specific mean. Also shown is the ratio of highest plant-specific mean to the industry mean. In each case, the between-plant variation is modeled and included in the reported uncertainty intervals. However, plant-specific results are presented only when the ratio highest/lowest > 6. Thus, plant-specific results are not presented for the three cases at the bottom of Table G-3. The relatively small differences between the plants do not justify presentation of plant-specific values. (Appendix E provides a detailed description of the trends and patterns analyses.)

Table G-4 provides a listing of the new plants that began operation during the 1987-1995 time frame. Also included is the critical year information for the learning period adjustments. The critical years information is based on total monthly critical hours. When a cutoff date fell in the middle of the month we did not know how many of the total critical hours fell before and after the cutoff date, so we assigned the maximum possible number after the cutoff. For example, at Palo Verde 2, the end of the learning period was set at Jan. 19, 1987, four months after the commercial start date. That plant had 217.5 critical hours (= 0.025 critical years) in January 1987. Those hours were assigned as coming after January 19, but this may be incorrect.

Table G-5 is a listing of the LERs that occurred during the learning period. Two LERs have numbers that do not match the docket number of the unit. (One LER reported events at two sister units.)

Tables G-6 through G-11 give the plant-specific rates. Figures G-1 through G-6 also give these rates, but present the plants in a different order. The tables list the plants in alphabetical order, but the figures give the plants in descending order by mean event rate. The plants with no observed events are identified by name in the tables but not in the figures. The first line, labeled "All PWRs" or "Industry" in each table and in each figure, describes the whole population of plants considered. The interval for this population includes most of the plant-specific estimates given in the table or figure.

Table Format and Content. The format for the entries in Tables G-4 through G-9 is as follows: Each line refers to a Bayesian distribution for the event rate. The first three numbers in the line (columns 2 through 4) are the mean, the 5th percentile, and the 95th percentile of the rate, in units of events per critical year.

Then the distribution is given, either a gamma or a lognormal distribution. If a gamma distribution is specified, the form is gamma(shape parameter, scale parameter), where the shape parameter is unitless and the scale parameter is in critical years. The mean of the distribution is (shape parameter)/(scale parameter), and the percentiles must be found by a computer calculation. If, instead, a lognormal distribution is specified, the form is lognormal (median, error factor), where the median has units events-per-critical-year and the error factor is unitless. *Both the median and mean are given; do not confuse the two columns.* The percentiles are related to the other parameters by: 5th percentile = median/(error factor), 95th percentile = median×(error factor). The mean is related by mean = $\exp(\mu + \sigma^2/2)$, with $\mu = \ln(\text{median})$ and $\sigma = \ln(\text{error factor})/1.645$.

When only between-plant differences are modeled, with no time trend, the rates given refer to all the years of the study. When both between-plant differences and a time trend are modeled, the rates given refer to 1995, the last year of the study.

Plants that were in the study but decommissioned before 1995 were used in the analysis to determine the combined industry-wide frequency of each event category. However, these plants are not shown in the tables and figures with between-plant differences. These plants are San Onofre 1, Trojan, Yankee Rowe, and Rancho Seco.

Time Dependent Trends. Figures G-7 through G-20 show the trends that were modeled. Consider first the case when no between-plant differences were modeled. The annual results are plotted, a mean and confidence interval based on each year's data. New plants began commercial operation in 1987-1990, and one PWR began operation in 1993. Therefore, for those years, two point estimates and two confidence intervals are shown next to each other, one based on excluding the learning period (up to four months after commercial start) and one based on using all the data. Each data set — including or excluding the learning-period data — can be used to fit an exponentially decreasing frequency. A 90% confidence band on the decreasing frequency is constructed; to reduce clutter, only one band is constructed, based on excluding the learning period. This band is simultaneously valid at all times, as explained in Appendix E. Ultimately, a Bayes distribution is derived for the fitted curve, and the mean of this distribution is plotted. The two data sets result in two such means, which are both plotted. In summary, the plot shows the annual point estimates and confidence intervals (or a pair of them in years with learning periods), two exponentially decreasing Bayes means (based on the two data sets), and a 90% confidence band corresponding to the data that excludes the learning period. The maximum likelihood estimate (MLE) of the event frequency, based on any one year's data, equals the number of events divided by the number of reactor critical years. It is the usual simple point estimate of the frequency.

Now consider the case when both a time trend and between-plant variation are modeled. In such a case, the between-plant variation can sometimes be seen even with only one year's data. Then the plotted vertical line is not a confidence interval on the mean rate, but instead is the empirical Bayes distribution that models the industry variability for that year. Thus, the vertical lines show whatever variation could be seen based on one year's data: they are 90% confidence intervals for years when between-plant variation could not be modeled, and empirical Bayes 90% intervals for the industry for years when between-plant variation could be modeled. Similarly, the 90% band on the fitted curve is a prediction band, with 90% confidence of containing the true rate for a random plant in all years. For details, see Appendix E.

Table G-1. Frequency estimates of functional impact categories: mean, percentiles, and trends using only data after the first four months from date of commercial operation.

Event	Functional Impact Event Category	Number of Functional Impact Occurrences[a]	Mean Frequency (per critical year)[b,c,i]	Percentiles		Model Used	
				5th %ile	95th %ile	Trend	Plant Difference
Loss-of-Coolant Accident (LOCA)	G						
Large Pipe Break LOCA. PWR	G7	0	5E-6[d]	1E-7	1E-5	Constant[e]	No
Large Pipe Break LOCA. BWR	G7	0	3E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: PWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: BWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Small Pipe Break LOCA	G3	0	5E-4[d]	1E-4	1E-3	Constant[e]	No
Very Small/Leak	G1	4	6.3E-3	2.3E-3	1.2E-2	Constant[e]	No
Stuck Open: Pressurizer PORV	G4	0	1.0E-3	4.0E-6	3.9E-3	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: PWR	G2	2	5.1E-3	1.2E-3	1.1E-2	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: BWR	G2	10	4.7E-2	2.6E-2	7.2E-2	Constant[e]	No
Stuck Open: 2 or More Safety/Relief Valves	G5	0	3.2E-4[d]	1.3E-6	1.2E-3	Constant[e]	No
Reactor Coolant Pump Seal LOCA: PWR	G8	2[d]	2.5E-3[d]	5.6E-4	5.4E-3	Constant[e]	No
Steam Generator Tube Rupture: PWR	F1	3	7.1E-3	2.2E-3	1.4E-2	Constant[e]	No
Loss of Offsite Power	B1	30	4.2E-2	7.8E-3[j]	1.0E-1[j]	Constant[e,j]	No
Total Loss of Condenser Heat Sink (combined):[f] PWR	L	71[f]	1.4E-1[f]	1.7E-2	3.7E-1	Constant[e]	Yes
Total Loss of Condenser Heat Sink (combined):[f] BWR	L	113[f]	3.1E-1[c,f]	2.2E-1	4.2E-1	Decrease	No
Inadvertent Closure of All MSIVs: PWR	L1	33	4.0E-2[c]	2.0E-2	6.9E-2	Decrease	No
Inadvertent Closure of All MSIVs: BWR	L1	70	1.8E-1[c]	5.5E-2	4.2E-1	Decrease	Yes
Loss of Condenser Vacuum: PWR	L2	34	6.8E-2	5.3E-5	2.9E-1	Constant[e]	Yes
Loss of Condenser Vacuum: BWR	L2	42	1.9E-1	1.4E-1	2.4E-1	Constant[e]	No
Turbine Bypass Unavailable	L3	8	1.2E-2	6.1E-3	1.9E-2	Constant[e]	No
Total Loss of Feedwater Flow	P1	132	1.0E-1[c]	1.7E-2	2.8E-1	Decrease	Yes
General Transients (combined):[f] PWR	Q	1070[f,g]	1.3E+0[c,f]	7.4E-1	2.1E+0	Decrease[f]	Yes
General Transients (combined):[f] BWR	Q	507[f,g]	1.6E+0[c,f]	9.8E-1	2.5E+0	Decrease[f]	Yes
High Energy Line Steam Breaks/Leaks (combined)[i]	K	8[i]	1.2E-2	6.1E-3	1.9E-2	Constant[e]	No
Steam Line Break/Leak Outside Containment	K1	6	9.1E-3	4.1E-3	1.6E-2	Constant[e]	No
Steam Line Break/Leak Inside Containment: PWR	K3	0	1.0E-3	4.0E-6	3.9E-3	Constant[e]	No
Feedwater Line Break/Leak	K2	2	3.5E-3	8.0E-4	7.7E-3	Constant[e]	No

Table G-1. (continued).

Event	Functional Impact Event Category	Number of Functional Impact Occurrences[a]	Mean Frequency (per critical year)[b,c,i]	Percentiles 5th %ile	Percentiles 95th %ile	Model Used Trend	Model Used Plant Difference
Loss of Safety-Related Bus	C						
Loss of Vital Medium Voltage ac Bus	C1	13	1.9E-2	1.1E-2	2.8E-2	Constant[e]	No
Loss of Vital Low Voltage ac Bus	C2	3	4.9E-3	1.5E-3	9.8E-3	Constant[e]	No
Loss of Vital dc Bus	C3	1	2.1E-3	2.5E-4	5.5E-3	Constant[e]	No
Loss of Safety-Related Cooling Water	E						
Total Loss of Service Water	E1	1[d]	9.7E-4[d]	1.1E-4	2.5E-3	Constant[e]	No
Partial Loss of Service Water	E2	6	9.1E-3	4.1E-3	1.6E-2	Constant[e]	No
Loss of Instrument or Control Air: PWR	D1	11[c]	9.8E-3[c]	3.9E-3	2.0E-2	Decrease	No
Loss of Instrument or Control Air: BWR	D1	19[c]	3.6E-2[c]	1.6E-2	6.9E-2	Decrease	No
Fire	H1	38	3.2E-2[c]	1.7E-2	5.4E-2	Decrease	No
Flood	J1	2	3.5E-3	8.0E-4	7.7E-3	Constant[e]	No
		Total — PWR[f]	1.5E+0[c]	8.2E-1	2.4E+0	Decrease	Yes
		Total — BWR[f]	1.9E+0[c]	1.1E+0	2.8E+0	Decrease	Yes

a. Reactor trip events from 1987 through 1995, inclusive, except when noted for certain rare events.

b. Frequencies are presented in per critical year (8,760 hours per critical year).

c. For categories with a decreasing trend, the frequencies reported are based on the endpoint of the trend line (i.e., 1995, the last year of the study).

d. No failures were identified in the 1987–1995 operating experience. The Medium and Large Pipe Break LOCA estimates were based on review of current literature and fracture mechanic analyses and using world-wide experience. (Appendix J contains the results of the LOCA analysis.) Frequency estimates for Small Pipe Break LOCA, Reactor Coolant Pump Seal LOCA, Stuck Open: 2 or More Safety/Relief Valves, and Total Loss of Service Water categories were based on total U.S. operating experience (1969–1997).

e. Any evidence for a trend was weak, not statistically significant. The trend, if any, is too small to be seen in the data. Therefore, no trend is modeled.

f. Combined number of occurrences of all categories for each plant type (BWR, PWR) under this heading was used to calculate this frequency and trend.

g. Total number of initial plant fault occurrences for this plant-type.

h. The frequency was based on the combined number of occurrences in the categories under this heading.

i. For categories modeled with no trend and no between-plant variation, the estimates were calculated using a Jeffreys non-informative prior (one half of an event added to the total number of events) in a Bayesian updated distribution.

j. The scatter seen from year to year was more than would be expected from Poisson counts. The uncertainty interval reflects this extra scatter.

Table G-2. Frequency estimates of initial plant fault categories: mean, percentiles, and trends using only data after the first four months from date of commercial operation.

Event	Initial Plant Fault Category	Number of Initial Plant Fault Occurences[a]	Mean Frequency (per critical year)[b,c,i]	Percentiles		Model Used	
				5th %ile	95th %ile	Trend	Plant Specific
Loss of Coolant Accident (LOCA)	G		—				
Large Pipe Break LOCA. PWR	G7	0	5E-6[d]	1E-7	1E-5	Constant[e]	No
Large Pipe Break LOCA: BWR	G7	0	3E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: PWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Medium Pipe Break LOCA: BWR	G6	0	4E-5[d]	1E-6	1E-4	Constant[e]	No
Small Pipe Break LOCA	G3	0	5E-4[d]	1E-4	1E-3	Constant[e]	No
Very Small/Leak	G1	2	3.5E-3	8.0E-4	7.7E-3	Constant[e]	No
Stuck Open: Pressurizer PORV	G4	0	1.0E-3	4.0E-6	3.9E-3	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve · PWR	G2	0	1.0E-3	4.0E-6	3.9E-3	Constant[e]	No
Stuck Open: 1 Safety/Relief Valve: BWR	G2	10	4.7E-2	2.6E-2	7.2E-2	Constant[e]	No
Stuck Open : 2 or More Safety/Relief Valves	G5	0	3.2E-4[d]	1.3E-6	1.2E-3	Constant[e]	No
Reactor Coolant Pump Seal LOCA: PWR	G8	2[d]	2.5E-3[d]	5.6E-4	5.4E-3	Constant[e]	No
Steam Generator Tube Rupture: PWR	F1	3	7.1E-3	2.2E-3	1.4E-2	Constant[e]	No
Loss of Offsite Power	B1	16	2.3E-2	3.0E-3	5.8E-2	Constant[e]	No
Total Loss of Condenser Heat Sink (combined)[f]· PWR	L	18[f]	3.6E-2[f]	2.0E-4	1.4E-1	Constant[e]	Yes
Total Loss of Condenser Heat Sink (combined)[f]· BWR	L	40[f]	1.2E-1[c,f]	6.6E-2	1.9E-1	Decrease	No
Inadvertent Closure of All MSIVs: PWR	L1	5	1.1E-2	4.7E-3	2.0E-2	Constant[e]	No
Inadvertent Closure of All MSIVs: BWR	L1	15	3.3E-2[c]	1.1E-2	7.4E-2	Decrease	No
Loss of Condenser Vacuum: PWR	L2	13	2.6E-2	<1.0E-6	1.3E-1	Constant[e]	Yes
Loss of Condenser Vacuum: BWR	L2	24	1.1E-1	7.5E-2	1.5E-1	Constant[e]	No
Turbine Bypass Unavailable	L3	1	2.1E-3	2.5E-4	5.5E-3	Constant[e]	No
Total Loss of Feedwater Flow	P1	72	6.6E-2[c]	6.5E-3	2.2E-1	Decrease	Yes
General Transients (combined)[g]: PWR	Q	1,070[f,g]	1.3E+0[c,f]	7.4E-1	2.1E+0	Decrease[f]	Yes
General Transients (combined)[g]: BWR	Q	507[f,g]	1.6E+0[c,f]	9.8E-1	2.5E+0	Decrease[f]	Yes
High Energy Line Steam Breaks/Leaks (combined)[h]	K	8[h]	1.2E-2	6.1E-3	1.9E-2	Constant[e]	No
Steam Line Break/Leak Outside Containment	K1	6	9.1E-3	4.1E-3	1.6E-2	Constant[e]	No
Steam Line Break/Leak Inside Containment: PWR	K3	0	1.0E-3	4.0E-6	3.9E-3	Constant[e]	No
Feedwater Line Break/Leak	K2	2	3.5E-3	8.0E-4	7.7E-3	Constant[e]	No

Table G-2. (continued).

Event	Initial Plant Fault Category	Number of Initial Plant Fault Occurences[a]	Mean Frequency (per critical year)[b,c,i]	Percentiles		Model Used	
				5th %ile	95th %ile	Trend	Plant Specific
Loss of Safety-Related Bus	C						
Loss of Vital Medium Voltage ac Bus	C1	10	1.5E-2	8.1E-3	2.3E-2	Constant[e]	No
Loss of Vital Low Voltage ac Bus	C2	1	2.1E-3	2.5E-4	5.5E-3	Constant[e]	No
Loss of Vital dc Bus	C3	0	7.0E-4	2.7E-6	2.7E-3	Constant[e]	No
Loss of Safety-Related Cooling Water	E						
Total Loss of Service Water	E1	0	3.2E-4[d]	1.3E-6	1.3E-3	Constant[e]	No
Partial Loss of Service Water	E2	0	7.0E-4	2.7E-6	2.7E-3	Constant[e]	No
Loss of Instrument or Control Air: PWR	D1	9	6.5E-3[c]	1.9E-3	1.5E-2	Decrease	No
Loss of Instrument or Control Air: BWR	D1	11	1.7E-2[c]	5.4E-3	3.9E-2	Decrease	No
Fire	H1	30	2.4E-2[c]	1.2E-2	4.3E-2	Decrease	No
Flood	J1	1	2.1E-3	2.5E-4	5.5E-3	Constant[e]	No
		Total — PWR[f]	1.5E+0[c]	8.2E-1	2.4E+0	Decrease	Yes
		Total — BWR[f]	1.9E+0[c]	1.1E+0	2.8E+0	Decrease	Yes

a. Reactor trip events from 1987 through 1995, inclusive, except when noted for certain rare events.

b. Rates are presented in per critical year (8,760 critical hours per critical year).

c. For categories with a decreasing trend, the rates reported are based on the endpoint of the trend line (i.e., 1995, the last year of the study).

d. No failures were identified in the 1987-1995 operating experience. The Medium and Large Pipe Break LOCA estimates are based on the best estimates calculated from literature and fracture mechanics analyses and using world-wide experience (Appendix J contains the results of the LOCA analysis). Frequency estimates for Small Pipe Break LOCA, Reactor Coolant Pump Seal LOCA, Stuck Open: 2 or More Safety/Relief Valves, and Total Loss of Service Water categories were based on total U.S. operating experience (1969-1997).

e. Any evidence for a trend was weak, not statistically significant. The trend, if any, is too small to be seen in the data. Therefore, no trend is modeled.

f. Combined number of occurrences of all categories for each plant-type (BWR, PWR) under this heading was used to calculate this rate and trend.

g. Total number of initial plant fault occurrences for this plant-type.

h. The frequency was based on the combined number of occurrences in the categories under this heading.

i. For categories with no trend and no between-plant variation modeled, the estimates were calculated using a Jeffreys non-informative prior (one half of an event added to the total number of events) in a Bayesian updated distribution.

Table G-3. Cases with statistically significant between-plant differences.

Category or Heading	Events	Highest/ Lowest	Highest/ Industry	p-value for Between-Plant Difference
Loss of Condenser Vacuum: PWRs (functional impact L2)	34	18.4	7.6	0.0000
Loss of Condenser Vacuum: PWRs (initial plant fault L2)	13	14.6	7.9	0.002
Total Loss of Feedwater Flow (initial plant fault P)	72	12.4	5.0	0.0001
Total Loss of Feedwater Flow (functional impact P)	132	9.3	4.1	0.0001
Total Loss of Condenser Heat Sink: PWRs (initial plant fault L)	18	7.2	4.7	0.03
Total Loss of Condenser Heat Sink: PWRs (functional impact L)	71	6.7	3.8	0.0000
Inadvertent Closure of All MSIVs: BWRs (functional impact L1)	70	3.6	2.0	0.02
General Transient: PWRs (initial plant fault Q)	1070	2.6	1.7	0.0000
General Transient: BWRs (initial plant fault Q)	507	2.0	1.3	0.0000

Table G-4. Information about new plants, with plants listed in alphabetical order. Only time in 1987-1995 is counted in critical year totals.

Name	Docket	Low Power License Date	Commercial Start Date	Critical Years After Low Power License Date	Critical Years After Learning Period	Difference
Beaver Valley 2	412	05/28/87	11/17/87	7.172	6.746	0.427
Braidwood 1	456	05/21/87	07/29/88	6.466	5.509	0.957
Braidwood 2	457	12/18/87	10/17/88	6.473	5.840	0.632
Byron 2	455	11/06/86	08/21/87	7.847	7.098	0.749
Clinton 1	461	09/29/86	11/24/87	6.561	5.758	0.803
Comanche Peak 1	445	02/08/90	08/13/90	4.695	4.142	0.553
Comanche Peak 2	446	02/02/93	08/03/93	2.220	1.707	0.513
Fermi 2	341	03/20/85	01/23/88	6.134	5.381	0.753
Harris	400	10/24/86	05/02/87	7.380	6.890	0.490
Hope Creek	354	04/11/86	12/20/86	7.607	7.328	0.279
Limerick 2	353	07/10/89	01/08/90	5.650	5.115	0.535
Nine Mile Pt. 2	410	10/31/86	04/05/88	6.010	5.303	0.708
Palo Verde 2	529	12/09/85	09/19/86	6.375	6.375	0.000
Palo Verde 3	530	03/25/87	01/08/88	6.142	5.684	0.459
Perry	440	03/18/86	11/18/87	6.165	5.515	0.650
Seabrook	443	05/26/89	08/19/90	4.672	4.055	0.617
South Texas 1	498	08/21/87	08/25/88	5.057	4.486	0.571
South Texas 2	499	12/16/88	06/19/89	4.525	4.054	0.471
Vogtle 1	424	01/16/87	06/01/87	7.740	7.312	0.427
Vogtle 2	425	02/09/89	05/20/89	6.055	5.620	0.435
Total				120.945	109.916	11.030

Table G-5. The 177 events before the end of the learning period, sorted by LER number.

OBS	LER	Name	Docket	Date	Commercial Start Date
1	341/87-002-0	Fermi 2	341	02/26/87	01/23/88
2	241/87-008-0	Fermi 2	341	03/01/87	01/23/88
3	341/87-011-0	Fermi 2	341	04/06/87	01/23/88
4	341/87-017-0	Fermi 2	341	05/13/87	01/23/88
5	341/87-031-1	Fermi 2	341	07/20/87	01/23/88
6	341/87-035-0	Fermi 2	341	07/31/87	01/23/88
7	341/87-056-0	Fermi 2	341	12/31/87	01/23/88
8	341/88-004-0	Fermi 2	341	01/10/88	01/23/88
9	341/88-019-1	Fermi 2	341	05/07/88	01/23/88
10	341/88-020-0	Fermi 2	341	05/08/88	01/23/88
11	341/88-021-1	Fermi 2	341	05/10/88	01/23/88
12	353/89-013-0	Limerick 2	353	11/10/89	01/08/90
13	354/87-014-0	Hope Creek	354	02/11/87	12/20/86
14	354/87-017-0	Hope Creek	354	02/24/87	12/20/86
15	400/87-004-0	Harris	400	01/21/87	05/02/87
16	400/87-005-0	Harris	400	01/22/87	05/02/87
17	400/87-008-0	Harris	400	02/27/87	05/02/87
18	400/87-012-0	Harris	400	03/11/87	05/02/87
19	400/87-013-0	Harris	400	03/13/87	05/02/87
20	400/87-017-0	Harris	400	03/31/87	05/02/87
21	400/87-018-0	Harris	400	04/03/87	05/02/87
22	400/87-019-0	Harris	400	04/12/87	05/02/87
23	400/87-021-0	Harris	400	04/14/87	05/02/87
24	400/87-024-0	Harris	400	04/21/87	05/02/87
25	400/87-025-0	Harris	400	04/22/87	05/02/87
26	400/87-031-0	Harris	400	05/24/87	05/02/87
27	400/87-035-0	Harris	400	06/17/87	05/02/87
28	400/87-037-0	Harris	400	06/21/87	05/02/87
29	400/87-038-0	Harris	400	06/22/87	05/02/87
30	400/87-041-0	Harris	400	08/04/87	05/02/87
31	400/87-042-0	Harris	400	07/09/87	05/02/87
32	410/87-031-1	Nine Mile Pt. 2	410	06/12/87	04/05/88
33	410/87-033-0	Nine Mile Pt. 2	410	06/15/87	04/05/88
34	410/87-043-0	Nine Mile Pt. 2	410	07/11/87	04/05/88
35	410/87-058-0	Nine Mile Pt. 2	410	10/01/87	04/05/88
36	410/87-064-0	Nine Mile Pt 2	410	10/23/87	04/05/88
37	410/87-081-0	Nine Mile Pt. 2	410	12/26/87	04/05/88
38	410/88-001-0	Nine Mile Pt. 2	410	01/20/88	04/05/88
39	410/88-014-0	Nine Mile Pt. 2	410	03/13/88	04/05/88
40	410/88-017-0	Nine Mile Pt. 2	410	03/21/88	04/05/88
41	410/88-019-0	Nine Mile Pt. 2	410	06/02/88	04/05/88
42	410/88-025-0	Nine Mile Pt. 2	410	06/22/88	04/05/88
43	410/88-025-0	Nine Mile Pt. 2	410	06/28/88	04/05/88
44	410/88-028-0	Nine Mile Pt 2	410	07/11/88	04/05/88
45	412/87-012-0	Beaver Valley 2	412	08/07/87	11/17/87

Table G-5. (continued).

OBS	LER	Name	Docket	Date	Commercial Start Date
46	412/87-012-0	Beaver Valley 2	412	08/07/87	11/17/87
47	412/87-014-0	Beaver Valley 2	412	08/15/87	11/17/87
48	412/87-015-0	Beaver Valley 2	412	08/15/87	11/17/87
49	412/78-018-1	Beaver Valley 2	412	08/18/87	11/17/87
50	412/87-019-0	Beaver Valley 2	412	08/25/87	11/17/87
51	412/87-020-1	Beaver Valley 2	412	09/09/87	11/17/87
52	412/87-012-0	Beaver Valley 2	412	09/28/87	11/17/87
53	412/87-024-0	Beaver Valley 2	412	09/29/87	11/17/87
54	412/87-026-0	Beaver Valley 2	412	10/08/87	11/17/87
55	412/87-028-0	Beaver Valley 2	412	10/14/87	11/17/87
56	412/87-029-0	Beaver Valley 2	412	10/15/87	11/17/87
57	412/87-030-2	Beaver Valley 2	412	10/16/87	11/17/87
58	412/87-032-1	Beaver Valley 2	412	10/24/87	11/17/87
59	412/87-034-0	Beaver Valley 2	412	10/29/87	11/17/87
60	412/87-035-0	Beaver Valley 2	412	11/10/87	11/17/87
61	412/87-036-0	Beaver Valley 2	412	11/17/87	11/17/87
62	412/88-002-1	Beaver Valley 2	412	01/27/88	11/17/87
63	424/87-008-0	Vogtle 1	424	03/19/87	06/01/87
64	424/87/009-0	Vogtle 1	424	03/20/87	06/01/87
65	424/87-009-0	Vogtle 1	424	03/20/87	06/01/87
.66	424/87-010-0	Vogtle 1	424	03/21/87	06/01/87
67	424/87-010-0	Vogtle 1	424	03/21/87	06/01/87
68	424/87-011-0	Vogtle 1	424	03/26/87	06/01/87
69	424/87-012-0	Vogtle 1	424	04/05/87	06/01/87
70	424/87-013-0	Vogtle 1	424	04/10/87	06/01/87
71	424/87-014-0	Vogtle 1	424	04/11/87	06/01/87
72	424/87-018-0	Vogtle 1	424	04/29/87	06/01/87
73	424/87-018-0	Vogtle 1	424	04/29/87	06/01/87
74	424/87-025-1	Vogtle 1	424	05/09/87	06/01/87
75	424/87-027-0	Vogtle 1	424	05/13/87	06/01/87
76	424/87-029-0	Vogtle 1	424	05/24/87	06/01/87
77	424/87-030-0	Vogtle 1	424	06/03/87	06/01/87
78	424/87-032-0	Vogtle 1	424	06/06/87	06/01/87
79	424/87-033-0	Vogtle 1	424	06/07/87	06/01/87
80	424/87-034-0	Vogtle 1	424	06/07/87	06/01/87
81	424/87-035-0	Vogtle 1	424	06/14/87	06/01/87
82	424/87-041-0	Vogtle 1	424	06/23/87	06/01/87
83	424/87-047-0	Vogtle 1	424	07/08/87	06/01/87
84	424/87-047-0	Vogtle 1	424	07/08/87	06/01/87
85	424/87-050-0	Vogtle 1	424	07/28/87	06/01/87
86	425/89-019-0	Vogtle 2	425	05/02/89	05/20/89
87	425/89-020-0	Vogtle 2	425	05/12/89	05/20/89
88	425/89-021-1	Vogtle 2	425	05/22/89	05/20/89
89	425/89-024-0	Vogtle 2	425	07/26/89	05/20/89
90	440/87-007-0	Perry	440	02/13/87	11/18/87

Table G-5. (continued).

OBS	LER	Name	Docket	Date	Commercial Start Date
91	440/87-012-0	Perry	440	03/02/87	11/18/87
92	440/87-027-1	Perry	440	04/13/87	11/18/87
93	440/87-030-0	Perry	440	05/01/87	11/18/87
94	440/87-035-0	Perry	440	05/24/87	11/18/87
95	440/87-037-0	Perry	440	05/27/87	11/18/87
96	440/87-042-0	Perry	440	06/17/87	11/18/87
97	440/87-045-0	Perry	440	06/30/87	11/18/87
98	440/87-064-0	Perry	440	09/09/87	11/18/87
99	440/87-072-0	Perry	440	10/27/87	11/18/87
100	440/87-073-1	Perry	440	10/29/87	11/18/87
101	440/88-001-1	Perry	440	01/03/88	11/18/87
102	443/89-008-0	Seabrook	443	06/22/89	08/19/90
103	443/90/015-1	Seabrook	443	06/20/90	08/19/90
104	443/90-018-0	Seabrook	443	07/05/90	08/19/90
105	443/90-022-0	Seabrook	443	08/22/90	08/19/90
106	443/90-025-0	Seabrook	443	11/09/90	08/19/90
107	445/90-009-0	Comanche Peak 1	445	04/21/90	08/13/90
108	445/90-013-0	Comanche Peak 1	445	05/09/90	08/13/90
109	445/90-017-0	Comanche Peak 1	445	05/27/90	08/13/90
110	445/90-023-0	Comanche Peak 1	445	08/08/90	08/13/90
111	445/90-025-0	Comanche Peak 1	445	08/25/90	08/13/90
112	445/90-027-0	Comanche Peak 1	445	09/07/90	08/13/90
113	445/90-028-0	Comanche Peak 1	445	09/08/90	08/13/90
114	445/90-029-0	Comanche Peak 1	445	09/10/90	08/13/90
115	445/90-030-0	Comanche Peak 1	445	09/15/90	08/13/90
116	446/93-003-0	Comanche Peak 2	446	05/04/93	08/03/93
117	446/93-005-0	Comanche Peak 2	446	05/20/93	08/03/93
118	446/93-011-0	Comanche Peak 2	446	11/17/93	08/03/93
119	445/87-001-1	Byron 2	455	01/15/87	08/21/87
120	445/87-002-1	Bryon 2	455	02/05/87	08/21/87
121	455/87-002-1	Byron 2	455	02/05/87	08/21/87
122	455/87-005-0	Byron 2	455	03/31/87	08/21/87
123	455/87-006-1	Byron 2	455	04/27/87	08/21/87
124	455/87-007-1	Byron 2	455	05/04/87	08/21/87
125	455/87-009-1	Byron 2	455	06/29/87	08/21/87
126	455/87-010-0	Byron 2	455	07/01/87	08/21/87
127	455/87-011-1	Byron 2	455	07/25/87	08/21/87
128	455/87-018-0	Byron 2	455	10/01/87	08/21/87
129	455/87-019-1	Byron 2	455	10/02/87	08/21/87
130	456/87-027-0	Braidwood 1	456	06/06/87	07/29/88
131	456/87-032-0	Braidwood 1	456	07/01/87	07/29/88
132	456/87-035-0	Braidwood 1	456	07/05/87	07/29/88
133	456/87-050-0	Braidwood 1	456	09/23/87	07/29/88
134	456/87-052-0	Braidwood 1	456	09/24/87	07/29/88
135	456/87-057-1	Braidwood 1	456	10/09/87	07/29/88

Table G-5. (continued).

OBS	LER	Name	Docket	Date	Commercial Start Date
136	456/87-060-0	Braidwood 1	456	12/06/87	07/29/88
137	456/88-016-0	Braidwood 1	456	08/11/88	07/29/88
138	456/88-022-0	Braidwood 1	456	10/16/88	07/29/88
139	456/88-023-0	Braidwood 2	457	10/17/88	10/17/88
140	456/88-025-0	Braidwood 1	456	11/15/88	07/29/88
141	456/88-025-0	Braidwood 2	457	11/15/88	10/17/88
142	457/88-012-1	Braidwood 2	457	06/20/88	10/17/88
143	457/88-013-0	Braidwood 2	457	06/21/88	10/17/88
144	457/88-014-1	Braidwood 2	457	06/22/88	10/17/88
145	457/88-016-0	Braidwood 2	457	06/24/88	10/17/88
146	457/88-018-0	Braidwood 2	457	07/02/88	10/17/88
147	457/88-019-0	Braidwood 2	457	07/24/88	10/17/88
148	457/88-020-0	Braidwood 2	457	09/04/88	10/17/88
149	457/88-022-0	Braidwood 2	457	09/19/88	10/17/88
150	457/88-026-0	Braidwood 2	457	09/23/88	10/17/88
151	457/88-028-0	Braidwood 2	457	11/17/88	10/17/88
152	457/88-029-1	Braidwood 2	457	10/25/88	10/17/88
153	457/88-031-0	Braidwood 2	457	11/05/88	10/17/88
154	461/87-017-0	Clinton 1	461	03/22/87	11/24/87
155	461/87-025-0	Clinton 1	461	05/06/87	11/24/87
156	461/87-029-0	Clinton 1	461	05/24/87	11/24/87
157	461/87-036-0	Clinton 1	461	07/13/87	11/24/87
158	461/87-042-0	Clinton 1	461	08/12/87	11/24/87
159	461/87-043-0	Clinton 1	461	07/24/87	11/24/87
160	461/87-050-0	Clinton 1	461	08/25/87	11/24/87
161	461/87-055-0	Clinton 1	461	09/21/87	11/24/87
162	461/87-060-0	Clinton 1	461	10/02/87	11/24/87
163	498/88-026-0	South Texas 1	498	03/30/88	08/25/88
164	498/88-045-0	South Texas 1	498	07/19/88	08/25/88
165	498/88-048-0	South Texas 1	498	08/16/88	08/25/88
166	498/88-049-0	South Texas 1	498	08/26/88	08/25/88
167	499/89-009-0	South Texas 2	499	04/05/89	06/19/89
168	499/89-013-0	South Texas 2	499	04/15/89	06/19/89
169	499/89-016-0	South Texas 2	499	06/02/89	06/19/89
170	499/89-017-0	South Texas 2	499	07/13/89	06/19/89
171	499/89-019-0	South Texas 2	499	08/23/89	06/19/89
172	499/89-020-0	South Texas 2	499	08/29/89	06/19/89
173	499/89-021-0	South Texas 2	499	09/05/89	06/19/89
174	499/89-022-0	South Texas 2	499	09/19/89	06/19/89
175	499/89-023-0	South Texas 2	499	09/22/89	06/19/89
176	499/89-026-0	South Texas 2	499	10/13/89	06/19/89
177	530/87-004-0	Palo Verde 3	530	12/17/87	01/08/88

Table G-6. Plant-specific rates (events per critical year) for functional impact heading L, Total Loss of Condenser Heat Sink for PWRs.

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
All PWRs	1.44E-1	1.74E-2	3.72E-1	gamma[a] (1.527, 10.63)
Arkansas 1	1.39E-1	3.16E-2	3.10E-1	gamma (2.477, 17.76)
Arkansas 2	1.40E-1	3.17E-2	3.10E-1	gamma (2.477, 17.74)
Beaver Valley 1	1.43E-1	3.24E-2	3.18E-1	gamma (2.475, 17.30)
Beaver Valley 2	1.45E-1	3.29E-2	3.23E-1	gamma (2.473, 17.00)
Braidwood 1	2.19E-1	6.43E-2	4.47E-1	gamma (3.285, 15.03)
Braidwood 2	9.27E-2	1.02E-2	2.45E-1	gamma (1.446, 15.60)
Byron 1	1.39E-1	3.15E-2	3.09E-1	gamma (2.477, 17.82)
Byron 2	8.61E-2	9.36E-3	2.28E-1	gamma (1.434, 16.64)
Callaway	1.91E-1	5.75E-2	3.86E-1	gamma (3.385, 17.77)
Calvert Cliffs 1	1.53E-1	3.46E-2	3.40E-1	gammaa (2.465, 16.11)
Calvert Cliffs 2	2.17E-1	6.40E-2	4.43E-1	gamma (3.292, 15.18)
Catawba 1	1.42E-1	3.21E-2	3.14E-1	gamma (2.476, 17.49)
Catawba 2	1.42E-1	3.22E-2	3.15E-1	gamma (2.475, 17.43)
Comanche Peak 1	1.03E-1	1.16E-2	2.72E-1	gamma (1.463, 14.15)
Comanche Peak 2	1.24E-1	1.42E-2	3.24E-1	gamma (1.479, 11.95)
Cook 1	1.99E-1	5.96E-2	4.04E-1	gamma (3.358, 16.88)
Cook 2	2.70E-1	9.25E-2	5.23E-1	gamma (4.016, 14.87)
Crystal River 3	1.44E-1	3.27E-2	3.21E-1	gamma (2.474, 17.12)
Davis-Besse	1.42E-1	3.22E-2	3.16E-1	gamma (2.475, 17.42)
Diablo Canyon 1	1.93E-1	5.80E-2	3.91E-1	gamma (3.379, 17.54)
Diablo Canyon 2	1.38E-1	3.13E-2	3.06E-1	gamma (2.478, 17.97)
Farley 1	1.36E-1	3.09E-2	3.02E-1	gamma (2.478, 18.24)
Farley 2	2.47E-1	8.64E-2	4.74E-1	gamma (4.146, 16.80)
Fort Calhoun	8.54E-2	9.27E-3	2.26E-1	gamma (1.432, 16.77)
Ginna	1.38E-1	3.13E-2	3.06E-1	gamma (2.478, 17.99)
Haddam Neck	9.09E-2	9.98E-3	2.40E-1	gamma (1.443, 15.88)
Harris	4.87E-1	2.13E-1	8.51E-1	gamma (6.066, 12.46)
Indian Point 2	8.74E-2	9.52E-3	2.31E-1	gamma (1.436, 16 44)
Indian Point 3	1.01E-1	1.13E-2	2.64E-1	gamma (1.459, 14.51)
Kewaunee	1.37E-1	3.10E-2	3.03E-1	gamma (2.478, 18.13)
Maine Yankee	8.92E-2	9.76E-3	2.35E-1	gamma (1.440, 16.15)
McGuire 1	1.46E-1	3.31E-2	3.25E-1	gamma (2.472, 16.92)
McGuire 2	1.96E-1	5.89E-2	3.99E-1	gamma (3.367, 17.16)
Millstone 2	9.25E-2	1.02E-2	2.44E-1	gamma (1 446, 15 64)
Millstone 3	5.47E-1	2.46E-1	9.45E-1	gamma (6.377, 11.66)
North Anna 1	1 41E-1	3.20E-2	3.14E-1	gamma (2.476, 17.52)
North Anna 2	8.28E-2	8.92E-3	2.19E-1	gamma (1 427, 17.23)
Oconee 1	1 37E-1	3.11E-2	3.04E-1	gamma (2.478, 18 10)
Oconee 2	8 22E-2	8.85E-3	2.18E-1	gamma (1.425, 17 33)
Oconee 3	8.38E-2	9.06E-3	2.22E-1	gamma (1.429, 17 05)

Table G-6. (continued)

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Palisades	9.27E-2	1 02E-2	2.45E-1	gamma (1.446, 15.60)
Palo Verde 1	1.52E-1	3 43E-2	3.38E-1	gamma (2.467, 16.23)
Palo Verde 2	8.98E-2	9.85E-3	2 37E-1	gamma (1.441, 16.05)
Palo Verde 3	2.16E-1	6.38E-2	4 42E-1	gamma (3.294, 15.24)
Point Beach 1	8.24E-2	8.87E-3	2.18E-1	gamma (1 426, 17.31)
Point Beach 2	8.28E-2	8.92E-3	2.19E-1	gamma (1.427, 17.24)
Prairie Island 1	8.15E-2	8.75E-3	2.16E-1	gamma (1.424, 17 48)
Prairie Island 2	8.13E-2	8.73E-3	2.16E-1	gamma (1.424, 17.50)
Rancho Seco	1.32E-1	1.52E-2	3.47E-1	gamma (1.480, 11.17)
Robinson 2	8.99E-2	9.86E-3	2.37E-1	gamma (1.441, 16.03)
Salem 1	1.51E-1	3 42E-2	3.37E-1	gamma (2.467, 16.29)
Salem 2	1.52E-1	3.44E-2	3.39E-1	gamma (2.466, 16 18)
San Onofre 1	1.06E-1	1.20E-2	2.79E-1	gamma (1.466, 13.80)
San Onofre 2	8.51E-2	9.23E-3	2.25E-1	gamma (1.432, 16.82)
San Onofre 3	2.50E-1	8.74E-2	4.81E-1	gamma (4.128, 16 49)
Seabrook	2.40E-1	6.90E-2	4.95E-1	gamma (3.188, 13.27)
Sequoyah 1	9.72E-2	1.08E-2	2.56E-1	gamma (1.454, 14.95)
Sequoyah 2	1 54E-1	3.48E-2	3.43E-1	gamma (2.464, 16.00)
South Texas 1	1.01E-1	1.13E-2	2.66E-1	gamma (1.460, 14.45)
South Texas 2	1.04E-1	1.17E-2	2.73E-1	gamma (1.464, 14 07)
St Lucie 1	8.44E-2	9 13E-3	2.23E-1	gamma (1.430, 16 95)
St Lucie 2	1.40E-1	3.17E-2	3.10E-1	gamma (2 477, 17.75)
Summer	8 48E-2	9.18E-3	2.24E-1	gamma (1.431, 16.88)
Surry 1	1.45E-1	3.29E-2	3.22E-1	gamma (2.473, 17.05)
Surry 2	1.47E-1	3.33E-2	3.27E-1	gamma (2.472, 16 80)
Three Mile Isl 1	8.23E-2	8.85E-3	2.18E-1	gamma (1.426, 17.32)
Trojan	1.10E-1	1.25E-2	2.89E-1	gamma (1.471, 13 32)
Turkey Point 3	9.29E-2	1.03E-2	2.45E-1	gamma (1.447, 15 57)
Turkey Point 4	9.17E-2	1.01E-2	2.42E-1	gamma (1.445, 15.76)
Vogtle 1	8.51E-2	9.23E-3	2.25E-1	gamma (1.432, 16 82)
Vogtle 2	9 40E-2	1.04E-2	2.48E-1	gamma (1 449, 15 41)
Waterford 3	3.02E-1	1.16E-1	5.59E-1	gamma (4.795, 15.87)
Wolf Creek	1 40E-1	3 17E-2	3.10E-1	gamma (2.477, 17.72)
Yankee-Rowe	1.05E-1	1 18E-2	2.75E-1	gamma (1.465, 13 98)
Zion 1	1.54E-1	3.47E-2	3.42E-1	gamma (2.464, 16.03)
Zion 2	1 51E-1	3.42E-2	3.36E-1	gamma (2 467, 16 32)

a As explained in the text, the parameters shown for the gamma distribution are the shape parameter and the scale parameter Means and percentiles are given in columns 2 through 4 of this table For more details, see the text preceding these tables Units of means and percentiles are events per critical year Units of the gamma scale parameter are critical year.

Table G-7. Plant-specific rates (events per critical year) for initial plant fault heading L, Total Loss of Condenser Heat Sink for PWRs.

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
All PWRs	3.61E-2	2.02E-4	1 35E-1	gamma[a] (0.536, 14 87)
Arkansas 1	6.87E-2	6.36E-3	1 87E-1	gamma (1 312, 19 10)
Arkansas 2	2.40E-2	9 81E-5	9.20E-2	gamma (0.504, 20 99)
Beaver Valley 1	2 45E-2	1 01E-4	9.37E-2	gamma (0.505, 20.63)
Beaver Valley 2	2.48E-2	1.04E-4	9 49E-2	gamma (0 506, 20.39)
Braidwood 1	2 63E-2	1 14E-4	1 00E-1	gamma (0.509, 19 36)
Braidwood 2	2.59E-2	1.11E-4	9 89E-2	gamma (0 508, 19.64)
Byron 1	6.85E-2	6.36E-3	1.87E-1	gamma (1 314, 19 17)
Byron 2	2.44E-2	1 01E-4	9 34E-2	gamma (0.505, 20 68)
Callaway	6 75E-2	6 35E-3	1 83E-1	gamma (1 324, 19 60)
Calvert Cliffs 1	2.58E-2	1.11E-4	9.87E-2	gamma (0.508, 19 68)
Calvert Cliffs 2	2 62E-2	1.13E-4	9.98E-2	gamma (0.509, 19.46)
Catawba 1	6.95E-2	6.36E-3	1.90E-1	gamma (1 304, 18.75)
Catawba 2	2.43E-2	1.00E-4	9 32E-2	gamma (0.505, 20 74)
Comanche Peak 1	2.82E-2	1 26E-4	1 07E-1	gamma (0.513, 18.18)
Comanche Peak 2	3.23E-2	1.49E-4	1.23E-1	gamma (0.516, 15 94)
Cook 1	1.15E-1	1 62E-2	2 90E-1	gamma (1.682, 14 57)
Cook 2	1.68E-1	2 60E-2	4.13E-1	gamma (1.799, 10 68)
Crystal River 3	2 47E-2	1 03E-4	9 44E-2	gamma (0.506, 20 49)
Davis-Besse	6.98E-2	6 36E-3	1.91E-1	gamma (1 302, 18.66)
Diablo Canyon 1	6 81E-2	6.36E-3	1 85E-1	gamma (1.318, 19 35)
Diablo Canyon 2	2 38E-2	9.65E-5	9 11E-2	gamma (0 503, 21.17)
Farley 1	2.35E-2	9.45E-5	9.00E-2	gamma (0 502, 21 39)
Farley 2	1.12E-1	1 63E-2	2 79E-1	gamma (1.725, 15 36)
Fort Calhoun	2.42E-2	9.98E-5	9 28E-2	gamma (0.504, 20 80)
Ginna	2.37E-2	9.63E-5	9 10E-2	gamma (0.503, 21 19)
Haddam Neck	2.55E-2	1 08E-4	9.73E-2	gamma (0.507, 19 92)
Harris	7 06E-2	6.36E-3	1.93E-1	gamma (1 294, 18.32)
Indian Point 2	2.47E-2	1.03E-4	9 45E-2	gamma (0 506, 20 48)
Indian Point 3	2 76E-2	1.22E-4	1.05E-1	gamma (0.512, 18.54)
Kewaunee	2.36E-2	9.53E-5	9.05E-2	gamma (0 503, 21.31)
Maine Yankee	2.51E-2	1 06E-4	9 60E-2	gamma (0.507, 20.19)
McGuire 1	2 49E-2	1.04E-4	9 52E-2	gamma (0 506, 20.33)
McGuire 2	2 41E-2	9.91E-5	9.24E-2	gamma (0.504, 20 89)
Millstone 2	2 58E-2	1 11E-4	9.87E-2	gamma (0.508, 19.67)
Millstone 3	7.09E-2	6.36E-3	1 94E-1	gamma (1.290, 18 20)
North Anna 1	2.42E-2	9.97E-5	9 28E-2	gamma (0 504, 20 81)
North Anna 2	2.36E-2	9 57E-5	9.06E-2	gamma (0 503, 21 27)
Oconee 1	2 36E-2	9 55E-5	0 06E-2	gamma (0 503, 21.28)
Oconee 2	2.35E-2	9 47E-5	9 01E-2	gamma (0 502, 21 37)
Oconee 3	2.39E-2	9 73E-5	9 15E-2	gamma (0 503, 21 08)

Table G-7. (continued)

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Palisades	2 59E-2	1.11E-4	9 89E-2	gamma (0 508, 19.64)
Palo Verde 1	7 36E-2	6.34E-3	2.03E-1	gamma (1 263, 17.16)
Palo Verde 2	2 52E-2	1 07E-4	9.65E-2	gamma (0 507, 20.09)
Palo Verde 3	2 61E-2	1.12E-4	9 96E-2	gamma (0 509, 19 50)
Point Beach 1	2 35E-2	9.49E-5	9.03E-2	gamma (0 502, 21.34)
Point Beach 2	2.36E-2	9 56E-5	9.06E-2	gamma (0.503, 21 27)
Prairie Island 1	2 33E-2	9.35E-5	8 95E-2	gamma (0.502, 21 51)
Prairie Island 2	2.33E-2	9.33E-5	8.94E-2	gamma (0.502, 21 53)
Rancho Seco	3 40E-2	1.56E-4	1.29E-1	gamma (0.515, 15 16)
Robinson 2	2.53E-2	1.07E-4	9.66E-2	gamma (0.507, 20.07)
Salem 1	2 56E-2	1.09E-4	9.79E-2	gamma (0.508, 19 82)
Salem 2	2 58E-2	1.10E-4	9.84E-2	gamma (0 508, 19.73)
San Onofre 1	2.88E-2	1.30E-4	1.10E-1	gamma (0 513, 17.82)
San Onofre 2	2 42E-2	9.94E-5	9.26E-2	gamma (0.504, 20 86)
San Onofre 3	1.14E-1	1.63E-2	2.83E-1	gamma (1 708, 15 03)
Seabrook	2.83E-2	1.27E-4	1.08E-1	gamma (0.513, 18.10)
Sequoyah 1	2 69E-2	1 18E-4	1.03E-1	gamma (0.510, 18.99)
Sequoyah 2	2.60E-2	1 12E-4	9.92E-2	gamma (0 509, 19 58)
South Texas 1	2.77E-2	1.23E-4	1.06E-1	gamma (0.512, 18 48)
South Texas 2	2.83E-2	1.27E-4	1.08E-1	gamma (0.513, 18 10)
St. Lucie 1	2.40E-2	9.81E-5	9.19E-2	gamma (0 504, 20 99)
St. Lucie 2	2.40E-2	9.81E-5	9.19E-2	gamma (0.504, 20.99)
Summer	2.41E-2	9.88E-5	9.23E-2	gamma (0.504, 20 92)
Surry 1	2.48E-2	1.03E-4	9.47E-2	gamma (0.506, 20 43)
Surry 2	2.50E-2	1.05E-4	9.57E-2	gamma (0.506, 20.23)
Three Mile Isl 1	2.35E-2	9 48E-5	9.02E-2	gamma (0.502, 21.36)
Trojan	2.97E-2	1.35E-4	1.13E-1	gamma (0 514, 17.34)
Turkey Point 3	2.59E-2	1.11E-4	9.90E-2	gamma (0.509, 19 61)
Turkey Point 4	2 57E-2	1.10E-4	9.80E-2	gamma (0 508, 19 79)
Vogtle 1	2 42E-2	9.93E-5	9.26E-2	gamma (0 504, 20 86)
Vogtle 2	2.62E-2	1.13E-4	9.99E-2	gamma (0.509, 19 45)
Waterford 3	2 38E-2	9 67E-5	9.12E-2	gamma (0.503, 21 15)
Wolf Creek	2.40E-2	9.83E-5	9.20E-2	gamma (0 504, 20 97)
Yankee-Rowe	2 85E-2	1.28E-4	1 08E-1	gamma (0.513, 18.01)
Zion 1	2.59E-2	1.11E-4	9 90E-2	gamma (0.509, 19 61)
Zion 2	2 56E-2	1 09E-4	9.77E-2	gamma (0.508, 19 85)

a As explained in the text, the parameters shown for the gamma distribution are the shape parameter and the scale parameter For more details, see the text preceding these tables. Units of means and percentiles are events per critical year. Units of the gamma scale parameter are critical year.

Table G-8. Plant-specific rates (events per critical year) for functional impact heading L, Loss of Condenser Vacuum for PWRs.

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
All PWRs	6.78E-2	5.27E-5	2.86E-1	gamma[a] (0.384, 5.66)
Arkansas 1	2.92E-2	1.80E-5	1.24E-1	gamma (0.371, 12.73)
Arkansas 2	1.05E-1	1.03E-2	2.84E-1	gamma (1.351, 12.82)
Beaver Valley 1	3.02E-2	1.88E-5	1.29E-1	gamma (0.372, 12.30)
Beaver Valley 2	3.09E-2	1.93E-5	1.32E-1	gamma (0.372, 12.03)
Braidwood 1	3.44E-2	2.20E-5	1.46E-1	gamma (0.373, 10.87)
Braidwood 2	3.34E-2	2 12E-5	1.42E-1	gamma (0.373, 11.18)
Byron 1	2.91E-2	1 79E-5	1.24E-1	gamma (0.371, 12.78)
Byron 2	3.01E-2	1.87E-5	1.28E-1	gamma (0.372, 12.36)
Callaway	1.02E-1	1.00E-2	2.76E-1	gamma (1.354, 13.25)
Calvert Cliffs 1	3.32E-2	2.11E-5	1.41E-1	gamma (0.373, 11.22)
Calvert Cliffs 2	1.23E-1	1 17E-2	3.32E-1	gamma (1.332, 10.87)
Catawba 1	1.07E-1	1 05E-2	2.90E-1	gamma (1.349, 12.56)
Catawba 2	2.99E-2	1.85E-5	1.27E-1	gamma (0.372, 12.42)
Comanche Peak 1	3.91E-2	2.57E-5	1.66E-1	gamma (0.375, 9.57)
Comanche Peak 2	5.21E-2	3.53E-5	2.21E-1	gamma (0.376, 7.22)
Cook 1	1.87E-1	3.76E-2	4.29E-1	gamma (2.219, 11 87)
Cook 2	2.87E-1	7.66E-2	6.06E-1	gamma (2.931, 10.21)
Crystal River 3	3.06E-2	1.91E-5	1.31E-1	gamma (0.372, 12 14)
Davis-Besse	1.08E-1	1.05E-2	2.92E-1	gamma (1.348, 12.48)
Diablo Canyon 1	1.79E-1	3.63E-2	4.09E-1	gamma (2.237, 12.52)
Diablo Canyon 2	2.87E-2	1.76E-5	1.22E-1	gamma (0.371, 12.92)
Farley 1	2.82E-2	1 72E-5	1.20E-1	gamma (0.371, 13.18)
Farley 2	2.53E-1	6.98E-2	5.29E-1	gamma (3.038, 12.00)
Fort Calhoun	2.97E-2	1.84E-5	1.27E-1	gamma (0.372, 12.50)
Ginna	1.03E-1	1.01E-2	2.79E-1	gamma (1 353, 13.08)
Haddam Neck	3.24E-2	2.05E-5	1.38E-1	gamma (0.373, 11.50)
Harris	3 49E-1	1.12E-1	6.92E-1	gamma (3.683, 10.54)
Indian Point 2	3.07E-2	1.91E-5	1.31E-1	gamma (0.372, 12.12)
Indian Point 3	3 76E-2	2.45E-5	1.60E-1	gamma (0.374, 9.96)
Kewaunee	1 02E-1	1.00E-2	2.76E-1	gamma (1.354, 13.23)
Maine Yankee	3.16E-2	1 98E-5	1.34E-1	gamma (0 372, 11.80)
McGuire 1	3.11E-2	1.95E-5	1.33E-1	gamma (0.372, 11.96)
McGuire 2	2 95E-2	1.82E-5	1.26E-1	gamma (0.371, 12.59)
Millstone 2	3.32E-2	2.11E-5	1.41E-1	gamma (0.373, 11.22)
Millstone 3	5 13E-1	1 99E-1	9.45E-1	gamma (4.864, 9 49)
North Anna 1	1.07E-1	1 04E-2	2.89E-1	gamma (1.349, 12.59)
North Anna 2	2.85E-2	1 74E-5	1.21E-1	gamma (0 371, 13 03)
Oconee 1	2.84E-2	1 74E-5	1.21E-1	gamma (0.371, 13 05)
Oconee 2	2 82E-2	1.72E-5	1.20E-1	gamma (0.371, 13.15)
Oconee 3	2 90E-2	1.78E-5	1.23E-1	gamma (0.371, 12 82)

Table G-8. (continued)

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Palisades	3.34E-2	2.12E-5	1.42E-1	gamma (0.373, 11.18)
Palo Verde 1	1.19E-1	1.14E-2	3.21E-1	gamma (1.337, 11.27)
Palo Verde 2	3.19E-2	2.00E-5	1.36E-1	gamma (0.372, 11.68)
Palo Verde 3	3.38E-2	2.16E-5	1.44E-1	gamma (0.373, 11.03)
Point Beach 1	2.83E-2	1.73E-5	1.21E-1	gamma (0.371, 13.12)
Point Beach 2	2.85E-2	1.74E-5	1.21E-1	gamma (0.371, 13.04)
Prairie Island 1	2.78E-2	1.70E-5	1.19E-1	gamma (0.371, 13.31)
Prairie Island 2	2.78E-2	1.69E-5	1.19E-1	gamma (0.371, 13.34)
Rancho Seco	5.85E-2	3.94E-5	2.48E-1	gamma (0.376, 6.43)
Robinson 2	3.19E-2	2.01E-5	1.36E-1	gamma (0.372, 11.66)
Salem 1	1.18E-1	1.13E-2	3.20E-1	gamma (1.338, 11.33)
Salem 2	3.30E-2	2.09E-5	1.41E-1	gamma (0.373, 11.28)
San Onofre 1	4.08E-2	2.70E-5	1.73E-1	gamma (0.375, 9.19)
San Onofre 2	2.96E-2	1.83E-5	1.26E-1	gamma (0.372, 12.56)
San Onofre 3	1.82E-1	3.68E-2	4.17E-1	gamma (2.230, 12.27)
Seabrook	3.95E-2	2.60E-5	1.68E-1	gamma (0.375, 9.49)
Sequoyah 1	3.57E-2	2.30E-5	1.52E-1	gamma (0.374, 10.45)
Sequoyah 2	1.21E-1	1.16E-2	3.28E-1	gamma (1.334, 11.03)
South Texas 1	3.78E-2	2.47E-5	1.61E-1	gamma (0.374, 9.90)
South Texas 2	3.95E-2	2.60E-5	1.68E-1	gamma (0.375, 9.49)
St. Lucie 1	2.92E-2	1.80E-5	1.25E-1	gamma (0.371, 12.71)
St. Lucie 2	2.92E-2	1.80E-5	1.24E-1	gamma (0.371, 12.71)
Summer	2.94E-2	1.81E-5	1.25E-1	gamma (0.371, 12.63)
Surry 1	3.08E-2	1.92E-5	1.31E-1	gamma (0.372, 12.07)
Surry 2	3.14E-2	1.97E-5	1.34E-1	gamma (0.372, 11.85)
Three Mile Isl 1	2.82E-2	1.73E-5	1.20E-1	gamma (0.371, 13.14)
Trojan	4.33E-2	2.89E-5	1.84E-1	gamma (0.375, 8.68)
Turkey Point 3	3.35E-2	2.13E-5	1.42E-1	gamma (0.373, 11.14)
Turkey Point 4	3.28E-2	2.08E-5	1.40E-1	gamma (0.373, 11.35)
Vogtle 1	2.96E-2	1.83E-5	1.26E-1	gamma (0.372, 12.56)
Vogtle 2	3.40E-2	2.17E-5	1.45E-1	gamma (0.373, 10.97)
Waterford 3	2.88E-2	1.77E-5	1.23E-1	gamma (0.371, 12.89)
Wolf Creek	2.93E-2	1.80E-5	1.25E-1	gamma (0.371, 12.69)
Yankee-Rowe	3.99E-2	2.63E-5	1.70E-1	gamma (0.375, 9.39)
Zion 1	3.35E-2	2.13E-5	1.42E-1	gamma (0.373, 11.14)
Zion 2	1.18E-1	1.13E-2	3.19E-1	gamma (1.338, 11.36)

a As explained in the text, the parameters shown for the gamma distribution are the shape parameter and the scale parameter. Units of means and percentiles are events per critical year. Units of the gamma scale parameter are critical year.

Table G-9. Plant-specific rates (events per critical year) for initial plant fault category L2, Loss of Condenser Vacuum at PWRs.

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
All PWRs	2.61E-2	4 46E-7	1.26E-1	gamma[a] (0.250, 9 59)
Arkansas 1	1 46E-2	1.59E-7	7.17E-2	gamma (0.240, 16 41)
Arkansas 2	1 47E-2	1.59E-7	7.18E-2	gamma (0.240, 16 39)
Beaver Valley 1	1.51E-2	1.66E-7	7 37E-2	gamma (0.241, 16.00)
Beaver Valley 2	1.53E-2	1.71E-7	7.50E-2	gamma (0.241, 15.73)
Braidwood 1	1.66E-2	1.93E-7	8.10E-2	gamma (0.242, 14 60)
Braidwood 2	1.62E-2	1.87E-7	7.93E-2	gamma (0.242, 14.90)
Byron 1	1.46E-2	1.58E-7	7.15E-2	gamma (0.240, 16 46)
Byron 2	1.50E-2	1.65E-7	7.34E-2	gamma (0.241, 16 06)
Callaway	1.43E-2	1.53E-7	7.02E-2	gamma (0.240, 16 77)
Calvert Cliffs 1	1 62E-2	1.86E-7	7.91E-2	gamma (0.242, 14 95)
Calvert Cliffs 2	1.64E-2	1.91E-7	8.03E-2	gamma (0.242, 14.71)
Catawba 1	7.44E-2	4.55E-3	2.17E-1	gamma (1.076, 14.47)
Catawba 2	1.49E-2	1.64E-7	7.31E-2	gamma (0.241, 16.12)
Comanche Peak 1	1.82E-2	2 21E-7	8.90E-2	gamma (0.243, 13.33)
Comanche Peak 2	2.22E-2	2.78E-7	1.08E-1	gamma (0.244, 11.00)
Cook 1	1.35E-1	1 76E-2	3.44E-1	gamma (1.603, 11.89)
Cook 2	2.07E-1	3.35E-2	5.01E-1	gamma (1.866, 9 03)
Crystal River 3	1 52E-2	1.69E-7	7.44E-2	gamma (0.241, 15.84)
Davis-Besse	7 47E-2	4.55E-3	2.18E-1	gamma (1.074, 14.39)
Diablo Canyon 1	7.24E-2	4.53E-3	2.11E-1	gamma (1.087, 15 01)
Diablo Canyon 2	1.45E-2	1.56E-7	7.09E-2	gamma (0.240, 16 60)
Farley 1	1 43E-2	1.52E-7	6.98E-2	gamma (0.240, 16.85)
Farley 2	1.30E-1	1.76E-2	3.29E-1	gamma (1.642, 12 63)
Fort Calhoun	1.49E-2	1 63E-7	7.28E-2	gamma (0.241, 16 19)
Ginna	1 45E-2	1 55E-7	7.08E-2	gamma (0.240, 16.62)
Haddam Neck	1.59E-2	1.80E-7	7.76E-2	gamma (0.241, 15.21)
Harris	7.59E-2	4.56E-3	2.22E-1	gamma (1.068, 14.07)
Indian Point 2	1.52E-2	1.69E-7	7.45E-2	gamma (0.241, 15.83)
Indian Point 3	1.77E-2	2.12E-7	8.64E-2	gamma (0 243, 13.71)
Kewaunee	1 43E-2	1.53E-7	7.02E-2	gamma (0.240, 16.75)
Maine Yankee	1 56E-2	1.75E-7	7.61E-2	gamma (0.241, 15.51)
McGuire 1	1.54E-2	1 72E-7	7.53E-2	gamma (0.241, 15.66)
McGuire 2	1 48E-2	1.61E-7	7.23E-2	gamma (0.241, 16.28)
Millstone 2	1.62E-2	1.86E-7	7.91E-2	gamma (0.242, 14.95)
Millstone 3	1.53E-2	1.70E-7	7.47E-2	gamma (0.241, 15.78)
North Anna 1	1 49E-2	1.62E-7	7.27E-2	gamma (0.241, 16 20)
North Anna 2	1 44E-2	1.54E-7	7.04E-2	gamma (0.240, 16 70)
Oconee 1	1 44E-2	1.54E-7	7.04E-2	gamma (0.240, 16 72)
Oconee 2	1 43E-2	1.52E-7	6.99E-2	gamma (0 240, 16 82)
Oconee 3	1.46E-2	1 57E-7	7.13E-2	gamma (0 240, 16 50)

Table G-9. (continued)

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Palisades	1.62E-2	1 87E-7	7 93E-2	gamma (0.242, 14.90)
Palo Verde 1	8 02E-2	4 56E-3	2 37E-1	gamma (1 043, 13.01)
Palo Verde 2	1 57E-2	1.77E-7	7 67E-2	gamma (0.241, 15.40)
Palo Verde 3	1 64E-2	1 90E-7	8 01E-2	gamma (0.242, 14.76)
Point Beach 1	1 43E-2	1 53E-7	7 00E-2	gamma (0.240, 16.79)
Point Beach 2	1.44E-2	1.54E-7	7.04E-2	gamma (0.240, 16 71)
Prairie Island 1	1 41E-2	1.50E-7	6.92E-2	gamma (0.240, 16.98)
Prairie Island 2	1.41E-2	1.49E-7	6 91E-2	gamma (0.240, 17.00)
Rancho Seco	2.38E-2	2.95E-7	1 16E-1	gamma (0.243, 10.20)
Robinson 2	1 57E-2	1.77E-7	7 68E-2	gamma (0.241, 15.38)
Salem 1	1 60E-2	1.83E-7	7 82E-2	gamma (0.242, 15.11)
Salem 2	1 61E-2	1.85E-7	7 87E-2	gamma (0.242, 15.01)
San Onofre 1	1.88E-2	2.31E-7	9 16E-2	gamma (0.243, 12.95)
San Onofre 2	1 48E-2	1.62E-7	7 25E-2	gamma (0.241, 16.25)
San Onofre 3	7.33E-2	4 54E-3	2.14E-1	gamma (1.082, 14 75)
Seabrook	1 83E-2	2 23E-7	8 95E-2	gamma (0.243, 13.25)
Sequoyah 1	1.71E-2	2.02E-7	8.34E-2	gamma (0.242, 14.20)
Sequoyah 2	1 63E-2	1.88E-7	7.96E-2	gamma (0.242, 14 85)
South Texas 1	1 78E-2	2 14E-7	8.68E-2	gamma (0.243, 13.65)
South Texas 2	1.83E-2	2 23E-7	8 95E-2	gamma (0.243, 13.24)
St. Lucie 1	1 47E-2	1.59E-7	7.18E-2	gamma (0.240, 16.40)
St. Lucie 2	1.47E-2	1.59E-7	7 18E-2	gamma (0.240, 16.40)
Summer	1.47E-2	1 60E-7	7.22E-2	gamma (0.241, 16 32)
Surry 1	1.53E-2	1 70E-7	7.47E-2	gamma (0.241, 15.78)
Surry 2	1 55E-2	1 74E-7	7 58E-2	gamma (0.241, 15.56)
Three Mile Isl 1	1 43E-2	1.52E-7	7 00E-2	gamma (0.240, 16.81)
Trojan	1 96E-2	2.43E-7	9 54E-2	gamma (0.243, 12.44)
Turkey Point 3	1 63E-2	1 87E-7	7 95E-2	gamma (0 242, 14.87)
Turkey Point 4	1 60E-2	1 83E-7	7 83E-2	gamma (0.242, 15 08)
Vogtle 1	1 48E-2	1 62E-7	7.25E-2	gamma (0.241, 16.25)
Vogtle 2	1 65E-2	1.91E-7	8 04E-2	gamma (0.242, 14 70)
Waterford 3	1 45E-2	1 56E-7	7 10E-2	gamma (0.240, 16.57)
Wolf Creek	1.47E-2	1 59E-7	7 19E-2	gamma (0.240, 16 38)
Yankee-Rowe	1 85E-2	2 26E-7	9 02E-2	gamma (0.243, 13 15)
Zion 1	1.63E-2	1.87E-7	7 95E-2	gamma (0.242, 14.87)
Zion 2	1 60E-2	1.82E-7	7.81E-2	gamma (0.242, 15 13)

a As explained in the text, the parameters shown for the gamma distribution are the shape parameter and the scale parameter For more details, see the text preceding these tables. Units of means and percentiles are events per critical year. Units of the gamma scale parameter are critical year.

Table G-10. Plant-specific rates (events per critical year) for functional impact category P1, Total Loss of Feedwater Flow in 1995 for PWRs and BWRs. Time trend and between-plant variation are modeled; four plants that were decommissioned before 1995 were used in the analysis, but are not shown in the listing below.

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Industry, 1995	1.00E-1	1.71E-2	2.84E-1	lognormal[a] (6.97E-2, 4.075)
Arkansas 1, 1995	2.42E-1	1.11E-1	4.40E-1	lognormal (2.21E-1, 1.988)
Arkansas 2, 1995	4.71E-2	1.23E-2	1.14E-1	lognormal (3.75E-2, 3.039)
Beaver Valley 1, 1995	4.80E-2	1.25E-2	1.16E-1	lognormal (3.81E-2, 3.056)
Beaver Valley 2, 1995	5.18E-2	1.31E-2	1.27E-1	lognormal (4.08E-2, 3.118)
Big Rock Point, 1995	7.99E-2	2.43E-2	1.81E-1	lognormal (6.63E-2, 2.729)
Braidwood 1, 1995	5.78E-2	1.39E-2	1.45E-1	lognormal (4.49E-2, 3.227)
Braidwood 2, 1995	5.70E-2	1.38E-2	1.42E-1	lognormal (4.43E-2, 3.210)
Browns Ferry 2, 1995	1.30E-1	3.36E-2	3.15E-1	lognormal (1.03E-1, 3.060)
Browns Ferry 3, 1995	9.94E-2	1.71E-2	2.81E-1	lognormal (6.92E-2, 4.053)
Brunswick 1, 1995	8.79E-2	2.59E-2	2.02E-1	lognormal (7.24E-2, 2.793)
Brunswick 2, 1995	5.03E-2	1.28E-2	1.23E-1	lognormal (3.97E-2, 3.101)
Byron 1, 1995	7.61E-2	2.35E-2	1.71E-1	lognormal (6.34E-2, 2.700)
Byron 2, 1995	8.27E-2	2.50E-2	1.88E-1	lognormal (6.85E-2, 2.740)
Callaway, 1995	4.62E-2	1.22E-2	1.11E-1	lognormal (3.69E-2, 3.021)
Calvert Cliffs 1, 1995	5.31E-2	1.32E-2	1.31E-1	lognormal (4.16E-2, 3.149)
Calvert Cliffs 2, 1995	1.37E-1	4.65E-2	2.95E-1	lognormal (1.17E-1, 2.519)
Catawba 1, 1995	1.15E-1	4.05E-2	2.43E-1	lognormal (9.92E-2, 2.449)
Catawba 2, 1995	2.98E-1	1.45E-1	5.26E-1	lognormal (2.76E-1, 1.908)
Clinton 1, 1995	5.59E-2	1.36E-2	1.39E-1	lognormal (4.36E-2, 3.195)
Comanche Peak 1, 1995	4.17E-1	1.69E-1	8.17E-1	lognormal (3.71E-1, 2.202)
Comanche Peak 2, 1995	8.64E-2	1.65E-2	2.36E-1	lognormal (6.23E-2, 3.779)
Cook 1, 1995	7.78E-2	2.38E-2	1.76E-1	lognormal (6.47E-2, 2.715)
Cook 2, 1995	5.21E-2	1.31E-2	1.28E-1	lognormal (4.09E-2, 3.129)
Cooper, 1995	1.16E-1	4.06E-2	2.45E-1	lognormal (9.98E-2, 2.460)
Crystal River 3, 1995	8.19E-2	2.48E-2	1.86E-1	lognormal (6 79E-2, 2 741)
Davis-Besse, 1995	8.04E-2	2.44E-2	1.82E-1	lognormal (6.67E-2, 2.729)
Diablo Canyon 1, 1995	1.09E-1	3.87E-2	2.29E-1	lognormal (9.40E-2, 2.431)
Diablo Canyon 2, 1995	4.71E-2	1.23E-2	1.14E-1	lognormal (3.75E-2, 3.036)
Dresden 2, 1995	8.32E-2	2.49E-2	1.90E-1	lognormal (6.88E-2, 2.758)
Dresden 3, 1995	1.19E-1	4.15E-2	2.53E-1	lognormal (1.02E-1, 2.470)
Duane Arnold, 1995	7.84E-2	2.40E-2	1.77E-1	lognormal (6.52E-2, 2 716)
Farley 1, 1995	1.42E-1	5.67E-2	2.82E-1	lognormal (1.26E-1, 2.231)
Farley 2, 1995	2.79E-1	1.36E-1	4.92E-1	lognormal (2.58E-1, 1.902)
Fermi 2, 1995	9.62E-2	2 77E-2	2.23E-1	lognormal (7.87E-2, 2 838)
Fitzpatrick, 1995	8 46E-2	2.52E-2	1.93E-1	lognormal (6 99E-2, 2 768)
Fort Calhoun, 1995	4.79E-2	1.25E-2	1.16E-1	lognormal (3 81E-2, 3 054)

Table G-10. (continued).

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Ginna, 1995	7.43E-2	2.30E-2	1.67E-1	lognormal (6.20E-2, 2.689)
Grand Gulf, 1995	2.31E-1	1.07E-1	4.20E-1	lognormal (2.12E-1, 1.986)
Haddam Neck, 1995	5.19E-2	1.31E-2	1.28E-1	lognormal (4.08E-2, 3.127)
Harris, 1995	2.20E-1	9 42E-2	4.18E-1	lognormal (1.98E-1, 2.106)
Hatch 1, 1995	1.51E-1	5.96E-2	2.99E-1	lognormal (1.33E-1, 2 240)
Hatch 2, 1995	2.34E-1	1.08E-1	4.26E-1	lognormal (2.14E-1, 1.988)
Hope Creek, 1995	2.01E-1	8.64E-2	3.81E-1	lognormal (1.81E-1, 2.099)
Indian Point 2, 1995	4.90E-2	1.26E-2	1.19E-1	lognormal (3.88E-2, 3.075)
Indian Point 3, 1995	1.37E-1	4.60E-2	2.95E-1	lognormal (1.16E-1, 2.533)
Kewaunee, 1995	1.07E-1	3.81E-2	2.24E-1	lognormal (9.25E-2, 2.425)
LaSalle 1, 1995	4.91E-2	1.27E-2	1.20E-1	lognormal (3.89E-2, 3.076)
LaSalle 2, 1995	4.96E-2	1.27E-2	1.21E-1	lognormal (3.92E-2, 3.083)
Limerick 1, 1995	7.70E-2	2.37E-2	1.73E-1	lognormal (6.41E-2, 2.707)
Limerick 2, 1995	6.25E-2	1.45E-2	1.59E-1	lognormal (4.80E-2, 3.310)
Maine Yankee, 1995	7.99E-2	2.43E-2	1.81E-1	lognormal (6.63E-2, 2.733)
McGuire 1, 1995	8.05E-2	2 44E-2	1.83E-1	lognormal (6.68E-2, 2.736)
McGuire 2, 1995	1.53E-1	6.03E-2	3.04E-1	lognormal (1.35E-1, 2.246)
Millstone 1, 1995	4.77E-2	1.24E-2	1.16E-1	lognormal (3.79E-2, 3.052)
Millstone 2, 1995	8.21E-2	2.47E-2	1.87E-1	lognormal (6.79E-2, 2 755)
Millstone 3, 1995	8.01E-2	2.43E-2	1.82E-1	lognormal (6.65E-2, 2.732)
Monticello, 1995	1.83E-1	7.94E-2	3.46E-1	lognormal (1 66E-1, 2 088)
Nine Mile Pt. 1, 1995	5.65E-2	1.37E-2	1.41E-1	lognormal (4 39E-2, 3.209)
Nine Mile Pt. 2, 1995	1.60E-1	5.30E-2	3 49E-1	lognormal (1.36E-1, 2.565)
North Anna 1, 1995	4.86E-2	1.26E-2	1.18E-1	lognormal (3.85E-2, 3.063)
North Anna 2, 1995	4.61E-2	1.22E-2	1.11E-1	lognormal (3.68E-2, 3.020)
Oconee 1, 1995	1.07E-1	3 82E-2	2.25E-1	lognormal (9.28E-2, 2.426)
Oconee 2, 1995	1 43E-1	5.70E-2	2.84E-1	lognormal (1.27E-1, 2.232)
Oconee 3, 1995	1.93E-1	8.32E-2	3.65E-1	lognormal (1.74E-1, 2.095)
Oyster Creek, 1995	4.99E-2	1.28E-2	1.22E-1	lognormal (3.95E-2, 3 089)
Palisades, 1995	1.33E-1	4 55E-2	2.85E-1	lognormal (1 14E-1, 2.504)
Palo Verde 1, 1995	5.36E-2	1.33E-2	1.33E-1	lognormal (4.20E-2, 3 156)
Palo Verde 2, 1995	1.24E-1	4.29E-2	2.63E-1	lognormal (1.06E-1, 2 479)
Palo Verde 3, 1995	5.65E-2	1.37E-2	1.41E-1	lognormal (4.40E-2, 3.205)
Peach Bottom 2, 1995	9.81E-2	2 82E-2	2.28E-1	lognormal (8.01E-2, 2.845)
Peach Bottom 3, 1995	2.29E-1	8.63E-2	4.67E-1	lognormal (2 01E-1, 2.326)
Perry, 1995	1 45E-1	4.89E-2	3.13E-1	lognormal (1.24E-1, 2.531)
Pilgrim, 1995	1.03E-1	2 90E-2	2.40E-1	lognormal (8 35E-2, 2.876)
Point Beach 1, 1995	4.58E-2	1 21E-2	1.10E-1	lognormal (3.66E-2, 3 016)
Point Beach 2, 1995	4 60E-2	1.22E-2	1 11E-1	lognormal (3.67E-2, 3 018)

Table G-10. (continued).

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Prairie Island 1, 1995	4 53E-2	1.21E-2	1 09E-1	lognormal (3 62E-2, 3.006)
Prairie Island 2, 1995	4.49E-2	1.20E-2	1.08E-1	lognormal (3.59E-2, 2.999)
Quad Cities 1, 1995	4.88E-2	1.26E-2	1.19E-1	lognormal (3 87E-2, 3 073)
Quad Cities 2, 1995	7 99E-2	2.42E-2	1 81E-1	lognormal (6.62E-2, 2.734)
River Bend, 1995	4.89E-2	1 26E-2	1 19E-1	lognormal (3.87E-2, 3.074)
Robinson 2, 1995	8.42E-2	2.52E-2	1.92E-1	lognormal (6.96E-2, 2 760)
Salem 1, 1995	8.27E-2	2.48E-2	1.88E-1	lognormal (6.84E-2, 2.755)
Salem 2, 1995	1.24E-1	4.27E-2	2.63E-1	lognormal (1 06E-1, 2.484)
San Onofre 2, 1995	7.76E-2	2.38E-2	1.75E-1	lognormal (6.45E-2, 2.712)
San Onofre 3, 1995	4.73E-2	1.24E-2	1.14E-1	lognormal (3 76E-2, 3 041)
Seabrook, 1995	3 05E-1	1 10E-1	6.36E-1	lognormal (2.64E-1, 2.410)
Sequoyah 1, 1995	1.56E-1	5.17E-2	3.38E-1	lognormal (1.32E-1, 2.559)
Sequoyah 2, 1995	5.47E-2	1.35E-2	1.36E-1	lognormal (4.27E-2, 3 173)
South Texas 1, 1995	6.21E-2	1 44E-2	1.58E-1	lognormal (4 77E-2, 3.310)
South Texas 2, 1995	6.59E-2	1.48E-2	1.69E-1	lognormal (5.01E-2, 3 379)
St. Lucie 1, 1995	1.11E-1	3 92E-2	2.33E-1	lognormal (9.55E-2, 2.437)
St. Lucie 2, 1995	1.09E-1	3.87E-2	2.30E-1	lognormal (9.43E-2, 2 434)
Summer, 1995	7.76E-2	2 38E-2	1.75E-1	lognormal (6.46E-2, 2.710)
Surry 1, 1995	5.03E-2	1.28E-2	1.23E-1	lognormal (3.98E-2, 3.096)
Surry 2, 1995	8.44E-2	2.53E-2	1.92E-1	lognormal (6.98E-2, 2 759)
Susquehanna 1, 1995	4.73E-2	1 24E-2	1 14E-1	lognormal (3.76E-2, 3.043)
Susquehanna 2, 1995	4.64E-2	1.22E-2	1.12E-1	lognormal (3.70E-2, 3.027)
Three Mile Isl 1, 1995	4.63E-2	1.22E-2	1.12E-1	lognormal (3.69E-2, 3.022)
Turkey Point 3, 1995	5.49E-2	1 35E-2	1.36E-1	lognormal (4.29E-2, 3.178)
Turkey Point 4, 1995	5.33E-2	1.33E-2	1.32E-1	lognormal (4.18E-2, 3 151)
Vermont Yankee, 1995	4.56E-2	1.21E-2	1.10E-1	lognormal (3.64E-2, 3.011)
Vogtle 1, 1995	8 11E-2	2.47E-2	1 84E-1	lognormal (6.73E-2, 2.729)
Vogtle 2, 1995	5 90E-2	1 41E-2	1 48E-1	lognormal (4.57E-2, 3.246)
Wash. Nuclear 2, 1995	1.23E-1	4.26E-2	2.61E-1	lognormal (1.06E-1, 2.477)
Waterford 3, 1995	1.49E-1	5 90E-2	2.96E-1	lognormal (1 32E-1, 2.238)
Wolf Creek, 1995	7.72E-2	2 37E-2	1 74E-1	lognormal (6.42E-2, 2.708)
Zion 1, 1995	5.14E-2	1.30E-2	1.26E-1	lognormal (4 05E-2, 3.123)
Zion 2, 1995	5.08E-2	1.29E-2	1 25E-1	lognormal (4 01E-2, 3.110)

a As explained in the text, the parameters shown for the lognormal distribution are the *median* and the error factor Means and percentiles are given in columns 2 through 4 of this table. For more details, see the text preceding these tables. Units of means and percentiles are events per critical year.

Table G-11. Plant-specific rates (events per critical year) for initial plant fault category P1, Total Loss of Feedwater Flow in 1995 for PWRs and BWRs. Time trend and between-plant variation are modeled; four plants that were decommissioned before 1995 were used in the analysis, but are not shown in the listing below.

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Industry, 1995	6.59E-2	6.47E-3	2.16E-1	lognormal (3 74E-2, 5.774)
Arkansas 1, 1995	2.28E-1	1 05E-1	4.15E-1	lognormal (2.09E-1, 1.987)
Arkansas 2, 1995	2.83E-2	5.00E-3	7.91E-2	lognormal (1 99E-2, 3.976)
Beaver Valley 1, 1995	2.89E-2	5 05E-3	8.11E-2	lognormal (2.02E-2, 4.006)
Beaver Valley 2, 1995	3.11E-2	5 25E-3	8.82E-2	lognormal (2.15E-2, 4.098)
Big Rock Point, 1995	7 09E-2	1.90E-2	1.70E-1	lognormal (5.68E-2, 2.989)
Braidwood 1, 1995	3 49E-2	5.53E-3	1.01E-1	lognormal (2.36E-2, 4.273)
Braidwood 2, 1995	3.42E-2	5.49E-3	9.88E-2	lognormal (2.33E-2, 4.242)
Browns Ferry 2, 1995	4.22E-2	5 94E-3	1.26E-1	lognormal (2.74E-2, 4 611)
Browns Ferry 3, 1995	6.50E-2	6 46E-3	2.12E-1	lognormal (3.70E-2, 5.728)
Brunswick 1, 1995	3.20E-2	5.29E-3	9.12E-2	lognormal (2.20E-2, 4.154)
Brunswick 2, 1995	3.04E-2	5.17E-3	8 62E-2	lognormal (2.11E-2, 4.082)
Byron 1, 1995	6.66E-2	1.81E-2	1.59E-1	lognormal (5 36E-2, 2.956)
Byron 2, 1995	7.31E-2	1.96E-2	1.75E-1	lognormal (5.86E-2, 2.992)
Callaway, 1995	2 77E-2	4.95E-3	7.71E-2	lognormal (1.95E-2, 3.946)
Calvert Cliffs 1, 1995	3.21E-2	5.31E-3	9.17E-2	lognormal (2.21E-2, 4.155)
Calvert Cliffs 2, 1995	1 49E-1	5.11E-2	3.17E-1	lognormal (1.27E-1, 2.493)
Catawba 1, 1995	6.91E-2	1 86E-2	1.65E-1	lognormal (5.55E-2, 2 974)
Catawba 2, 1995	2.96E-1	1.47E-1	5.18E-1	lognormal (2.75E-1, 1.879)
Clinton 1, 1995	3.37E-2	5.45E-3	9.71E-2	lognormal (2.30E-2, 4.223)
Comanche Peak 1, 1995	2.41E-1	7.97E-2	5.26E-1	lognormal (2.05E-1, 2.570)
Comanche Peak 2, 1995	5 44E-2	6 33E-3	1.71E-1	lognormal (3.29E-2, 5.200)
Cook 1, 1995	2.88E-2	5.05E-3	8.08E-2	lognormal (2.02E-2, 4.002)
Cook 2, 1995	3 14E-2	5 26E-3	8.95E-2	lognormal (2.17E-2, 4.123)
Cooper, 1995	1.22E-1	4.26E-2	2.58E-1	lognormal (1.05E-1, 2.461)
Crystal River 3, 1995	7.28E-2	1.94E-2	1 75E-1	lognormal (5.82E-2, 2.998)
Davis-Besse, 1995	7.10E-2	1.91E-2	1.70E-1	lognormal (5.70E-2, 2.984)
Diablo Canyon 1, 1995	1.12E-1	3 99E-2	2.36E-1	lognormal (9 70E-2, 2 432)
Diablo Canyon 2, 1995	2 82E-2	5 00E-3	7.88E-2	lognormal (1.99E-2, 3.970)
Dresden 2, 1995	7.48E-2	1.97E-2	1.80E-1	lognormal (5.96E-2, 3.026)
Dresden 3, 1995	2 98E-2	5 12E-3	8.41E-2	lognormal (2.07E-2, 4.052)
Duane Arnold, 1995	2.89E-2	5.06E-3	8.11E-2	lognormal (2.03E-2, 4.004)
Farley 1, 1995	1.08E-1	3.86E-2	2.27E-1	lognormal (9.37E-2, 2.426)
Farley 2, 1995	3.34E-1	1 74E-1	5 64E-1	lognormal (3.14E-1, 1 798)
Fermi 2, 1995	3.40E-2	5 46E-3	9 80E-2	lognormal (2.31E-2, 4.238)
Fitzpatrick, 1995	7.63E-2	2.00E-2	1 85E-1	lognormal (6.08E-2, 3.036)
Fort Calhoun, 1995	2 88E-2	5.05E-3	8.08E-2	lognormal (2.02E-2, 4.000)

Table G-11. (continued).

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Ginna, 1995	2.77E-2	4.95E-3	7.72E-2	lognormal (1.96E-2, 3.950)
Grand Gulf, 1995	1.11E-1	3.96E-2	2.34E-1	lognormal (9.62E-2, 2.430)
Haddam Neck, 1995	3 14E-2	5.26E-3	8.92E-2	lognormal (2.17E-2, 4.119)
Harris, 1995	3.03E-2	5.18E-3	8.56E-2	lognormal (2.11E-2, 4.064)
Hatch 1, 1995	6.65E-2	1.81E-2	1.58E-1	lognormal (5.36E-2, 2.955)
Hatch 2, 1995	1.13E-1	4.00E-2	2.37E-1	lognormal (9.75E-2, 2.435)
Hope Creek, 1995	6.87E-2	1.86E-2	1.64E-1	lognormal (5.52E-2, 2.969)
Indian Point 2, 1995	2.95E-2	5.11E-3	8.31E-2	lognormal (2.06E-2, 4.035)
Indian Point 3, 1995	8.39E-2	2.13E-2	2.06E-1	lognormal (6.62E-2, 3.106)
Kewaunee, 1995	6.41E-2	1.76E-2	1.52E-1	lognormal (5.17E-2, 2.940)
LaSalle 1, 1995	2.96E-2	5.11E-3	8.33E-2	lognormal (2.06E-2, 4.035)
LaSalle 2, 1995	2.98E-2	5.14E-3	8.42E-2	lognormal (2.08E-2, 4.048)
Limerick 1, 1995	2.85E-2	5.02E-3	7.98E-2	lognormal (2.00E-2, 3.985)
Limerick 2, 1995	3.77E-2	5.71E-3	1.11E-1	lognormal (2.51E-2, 4.399)
Maine Yankee, 1995	7.11E-2	1.90E-2	1.71E-1	lognormal (5.69E-2, 2.997)
McGuire 1, 1995	7.17E-2	1.91E-2	1.72E-1	lognormal (5.74E-2, 2.998)
McGuire 2, 1995	2.85E-2	5.02E-3	7.98E-2	lognormal (2.00E-2, 3.987)
Millstone 1, 1995	2.87E-2	5.03E-3	8.05E-2	lognormal (2.01E-2, 4.000)
Millstone 2, 1995	7.40E-2	1.95E-2	1.79E-1	lognormal (5.90E-2, 3.027)
Millstone 3, 1995	7.12E-2	1.90E-2	1.71E-1	lognormal (5.70E-2, 2.993)
Monticello, 1995	6.35E-2	1.75E-2	1.50E-1	lognormal (5.13E-2, 2.933)
Nine Mile Pt. 1, 1995	3.42E-2	5.47E-3	9.88E-2	lognormal (2.32E-2, 4.252)
Nine Mile Pt. 2, 1995	9.58E-2	2.40E-2	2.36E-1	lognormal (7.52E-2, 3.137)
North Anna 1, 1995	2.91E-2	5.08E-3	8.19E-2	lognormal (2.04E-2, 4.014)
North Anna 2, 1995	2.76E-2	4.95E-3	7.70E-2	lognormal (1.95E-2, 3.946)
Oconee 1, 1995	1.10E-1	3.93E-2	2.32E-1	lognormal (9.54E-2, 2.428)
Oconee 2, 1995	6.36E-2	1.75E-2	1.51E-1	lognormal (5.13E-2, 2.936)
Oconee 3, 1995	6.64E-2	1.81E-2	1.58E-1	lognormal (5.34E-2, 2.954)
Oyster Creek, 1995	3.00E-2	5.16E-3	8.48E-2	lognormal (2.09E-2, 4.057)
Palisades, 1995	1.43E-1	4.94E-2	3.04E-1	lognormal (1.23E-1, 2.481)
Palo Verde 1, 1995	3.24E-2	5.34E-3	9.25E-2	lognormal (2.22E-2, 4.163)
Palo Verde 2, 1995	7.47E-2	1.97E-2	1.80E-1	lognormal (5.96E-2, 3.018)
Palo Verde 3, 1995	3.41E-2	5.47E-3	9.83E-2	lognormal (2.32E-2, 4.238)
Peach Bottom 2, 1995	3.44E-2	5.49E-3	9.92E-2	lognormal (2.33E-2, 4.250)
Peach Bottom 3, 1995	2.70E-1	1.11E-1	5.23E-1	lognormal (2.41E-1, 2.168)
Perry, 1995	8 73E-2	2.23E-2	2.13E-1	lognormal (6 90E-2, 3.093)
Pilgrim, 1995	3.56E-2	5.57E-3	1.03E-1	lognormal (2.40E-2, 4.306)
Point Beach 1, 1995	2.75E-2	4.93E-3	7.65E-2	lognormal (1.94E-2, 3.938)
Point Beach 2, 1995	2.76E-2	4.94E-3	7.68E-2	lognormal (1.95E-2, 3 943)
Prairie Island 1, 1995	2.72E-2	4 90E-3	7.54E-2	lognormal (1 92E-2, 3 922)

Table G-11. (continued)

Plant	Mean	5th %ile	95th %ile	Distribution and Parameters
Prairie Island 2, 1995	2.69E-2	4.88E-3	7 46E-2	lognormal (1.91E-2, 3.912)
Quad Cities 1, 1995	2.95E-2	5.10E-3	8.30E-2	lognormal (2.06E-2, 4.035)
Quad Cities 2, 1995	2.95E-2	5.10E-3	8.32E-2	lognormal (2 06E-2, 4 039)
River Bend, 1995	2.95E-2	5.10E-3	8 31E-2	lognormal (2 06E-2, 4 035)
Robinson 2, 1995	3.07E-2	5.20E-3	8.71E-2	lognormal (2.13E-2, 4.090)
Salem 1, 1995	7.43E-2	1.96E-2	1.79E-1	lognormal (5.93E-2, 3.023)
Salem 2, 1995	1.32E-1	4.57E-2	2.80E-1	lognormal (1.13E-1, 2.478)
San Onofre 2, 1995	6.83E-2	1.85E-2	1.63E-1	lognormal (5.49E-2, 2 969)
San Onofre 3, 1995	2.84E-2	5.01E-3	7.94E-2	lognormal (1 99E-2, 3 980)
Seabrook, 1995	1.24E-1	2.88E-2	3.15E-1	lognormal (9.52E-2, 3.310)
Sequoyah 1, 1995	1.71E-1	5.85E-2	3.67E-1	lognormal (1.47E-1, 2 506)
Sequoyah 2, 1995	3.30E-2	5.39E-3	9 46E-2	lognormal (2.26E-2, 4.191)
South Texas 1, 1995	3.78E-2	5.70E-3	1.11E-1	lognormal (2.51E-2, 4.411)
South Texas 2, 1995	4.02E-2	5.83E-3	1.19E-1	lognormal (2.64E-2, 4.521)
St. Lucie 1, 1995	6.64E-2	1.81E-2	1.58E-1	lognormal (5.35E-2, 2.957)
St. Lucie 2, 1995	6.56E-2	1.79E-2	1.56E-1	Lognormal (5.28E-2, 2.954)
Summer, 1995	2.87E-2	5.04E-3	8 04E-2	lognormal (2.01E-2, 3.993)
Surry 1, 1995	3.03E-2	5.18E-3	8.56E-2	lognormal (2.10E-2, 4.066)
Surry 2, 1995	7.55E-2	2.00E-2	1.82E-1	lognormal (6.02E-2, 3 017)
Susquehanna 1, 1995	2.84E-2	5.01E-3	7.95E-2	lognormal (2.00E-2, 3.983)
Susquehanna 2, 1995	2.79E-2	4.97E-3	7.78E-2	lognormal (1.97E-2, 3.958)
Three Mile Isl 1, 1995	2.77E-2	4.96E-3	7.72E-2	lognormal (1.96E-2, 3.947)
Turkey Point 3, 1995	3.32E-2	5.40E-3	9.52E-2	lognormal (2.27E-2, 4.198)
Turkey Point 4, 1995	3.22E-2	5.33E-3	9.20E-2	lognormal (2.21E-2, 4.157)
Vermont Yankee, 1995	2.73E-2	4.92E-3	7.59E-2	lognormal (1.93E-2, 3.930)
Vogtle 1, 1995	2.96E-2	5.13E-3	8.34E-2	lognormal (2 07E-2, 4 031)
Vogtle 2, 1995	3.55E-2	5.57E-3	1.03E-1	lognormal (2.40E-2, 4.297)
Wash. Nuclear 2, 1995	7.41E-2	1.96E-2	1.79E-1	lognormal (5.92E-2, 3.016)
Waterford 3, 1995	6.59E-2	1.80E-2	1.57E-1	lognormal (5.31E-2, 2.951)
Wolf Creek, 1995	2.86E-2	5.03E-3	7.99E-2	lognormal (2 01E-2, 3 987)
Zion 1, 1995	3.12E-2	5.23E-3	8.87E-2	lognormal (2.15E-2, 4.118)
Zion 2, 1995	3.08E-2	5.20E-3	8.73E-2	lognormal (2 13E-2, 4.096)

a. As explained in the text, the parameters shown for the lognormal distribution are the *median* and the error factor. Means and percentiles are given in columns 2 through 4 of this table. For more details, see the text preceding these tables Units of means and percentiles are events per critical year.

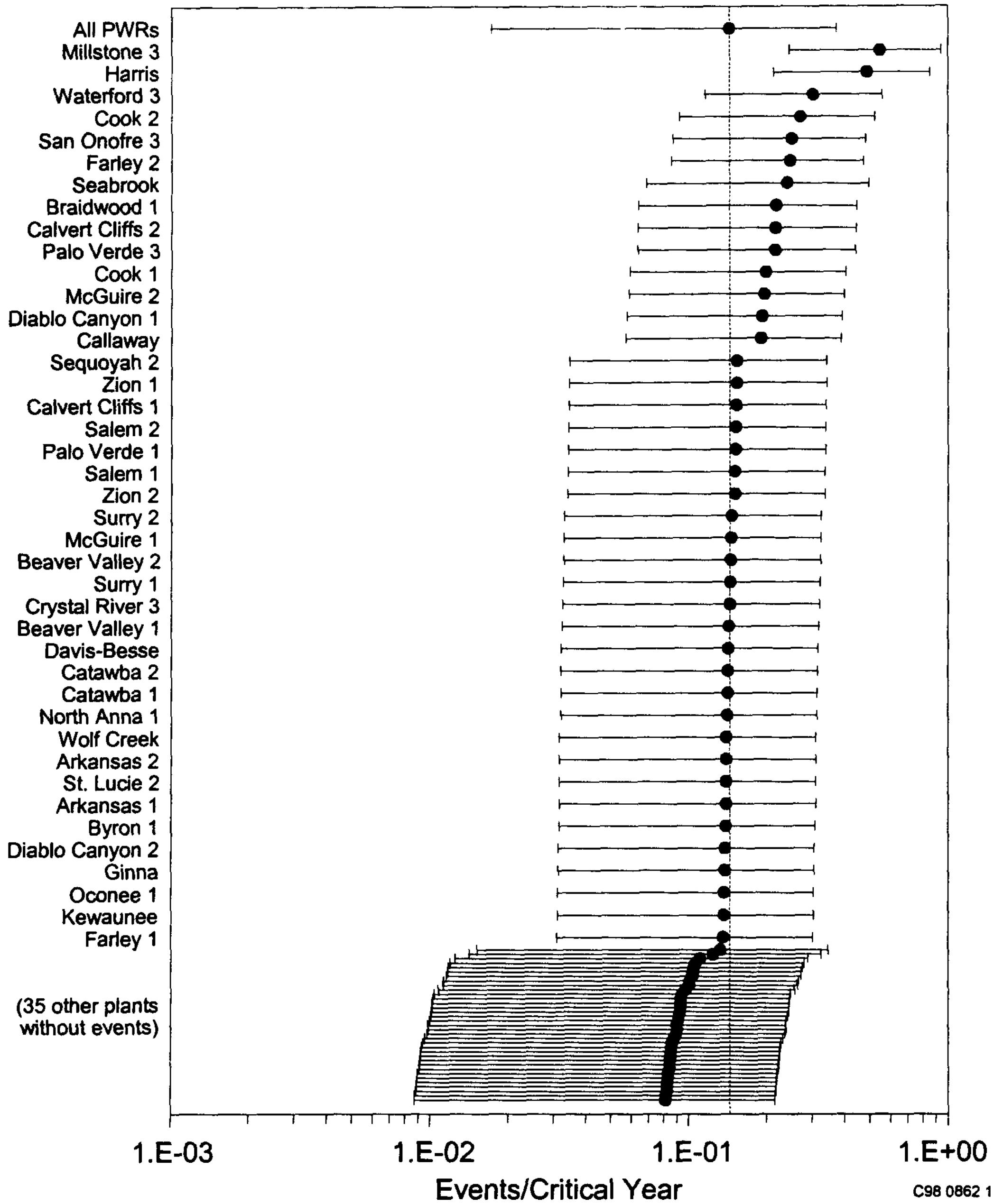

Figure G-1. Plant-specific rates (means and 90% intervals) for functional impact heading L, Total Loss of Condenser Heat Sink for PWRs. The ratio of the highest mean to the lowest is 6.7.

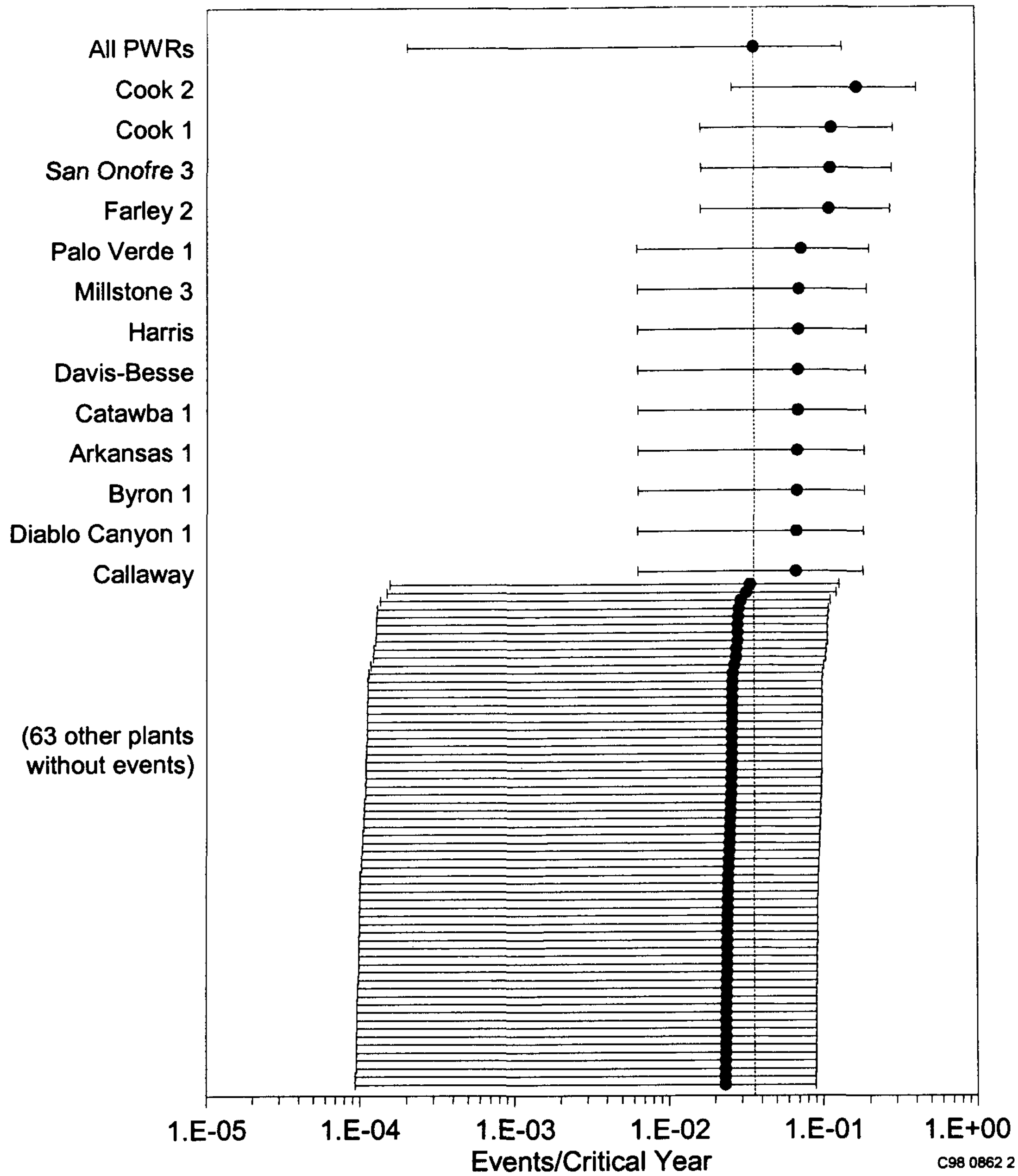

Figure G-2. Plant-specific rates (means and 90% intervals) for initial plant fault heading L, Total Loss of Condenser Heat Sink for PWRs. The ratio of the highest mean to the lowest is 7.2.

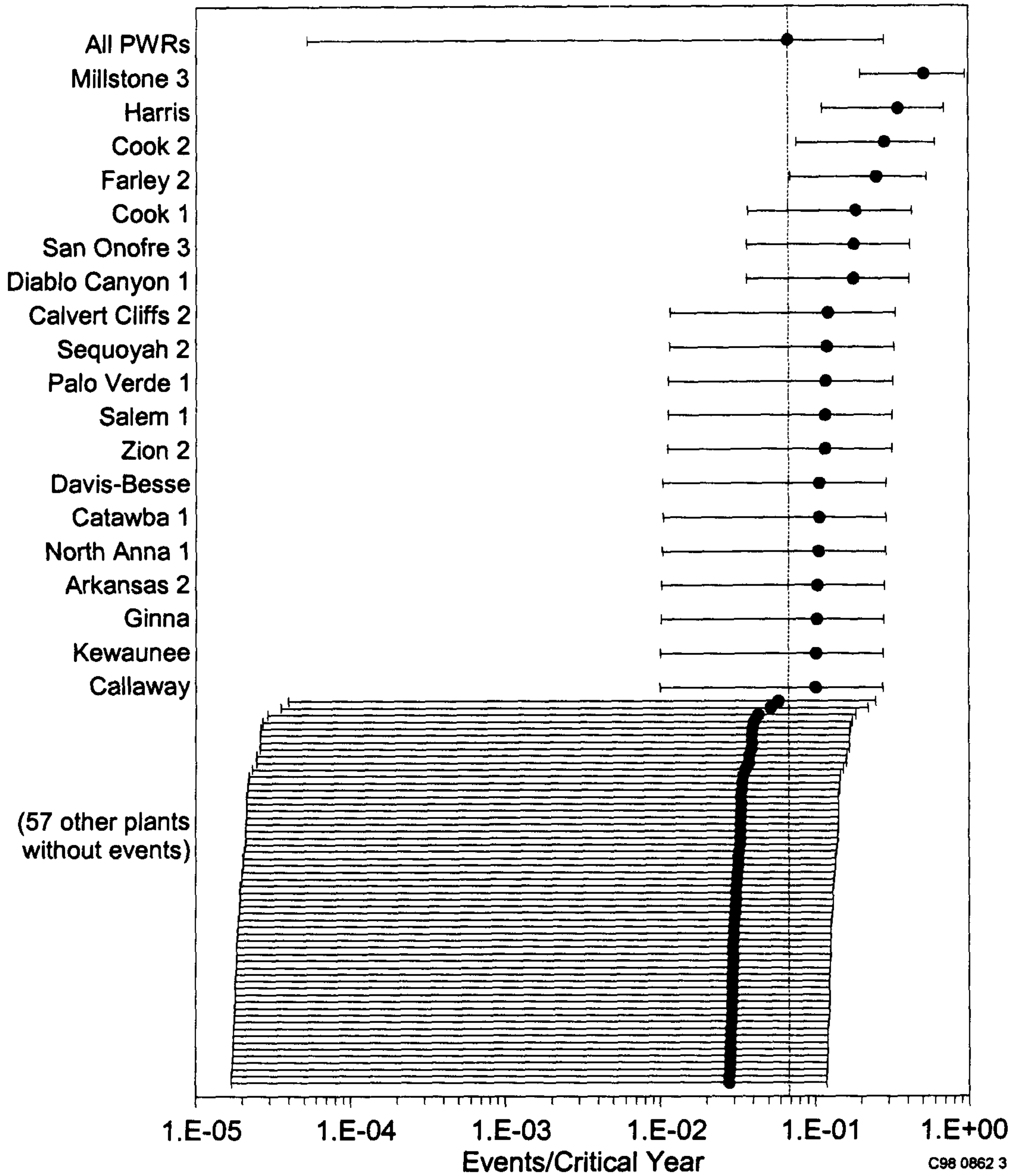

Figure G-3. Plant-specific rates (means and 90% intervals) for functional impact category L2, Loss of Condenser Vacuum for PWRs. The ratio of the highest mean to the lowest is 18.4.

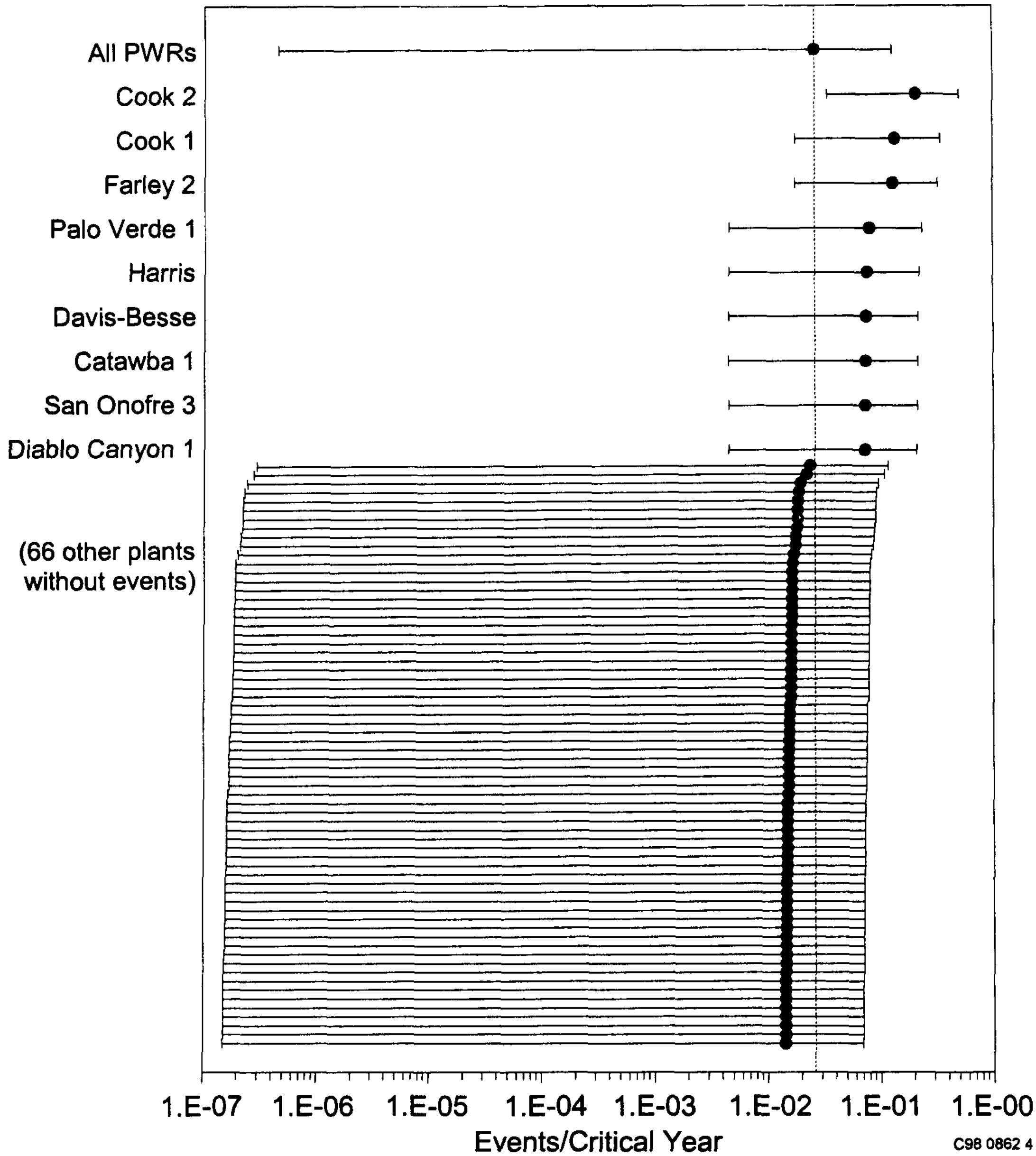

Figure G-4. Plant-specific rates (means and 90% intervals) for initial plant fault category L2, Loss of Condenser Vacuum for PWRs. The ratio of the highest mean to the lowest is 14.6.

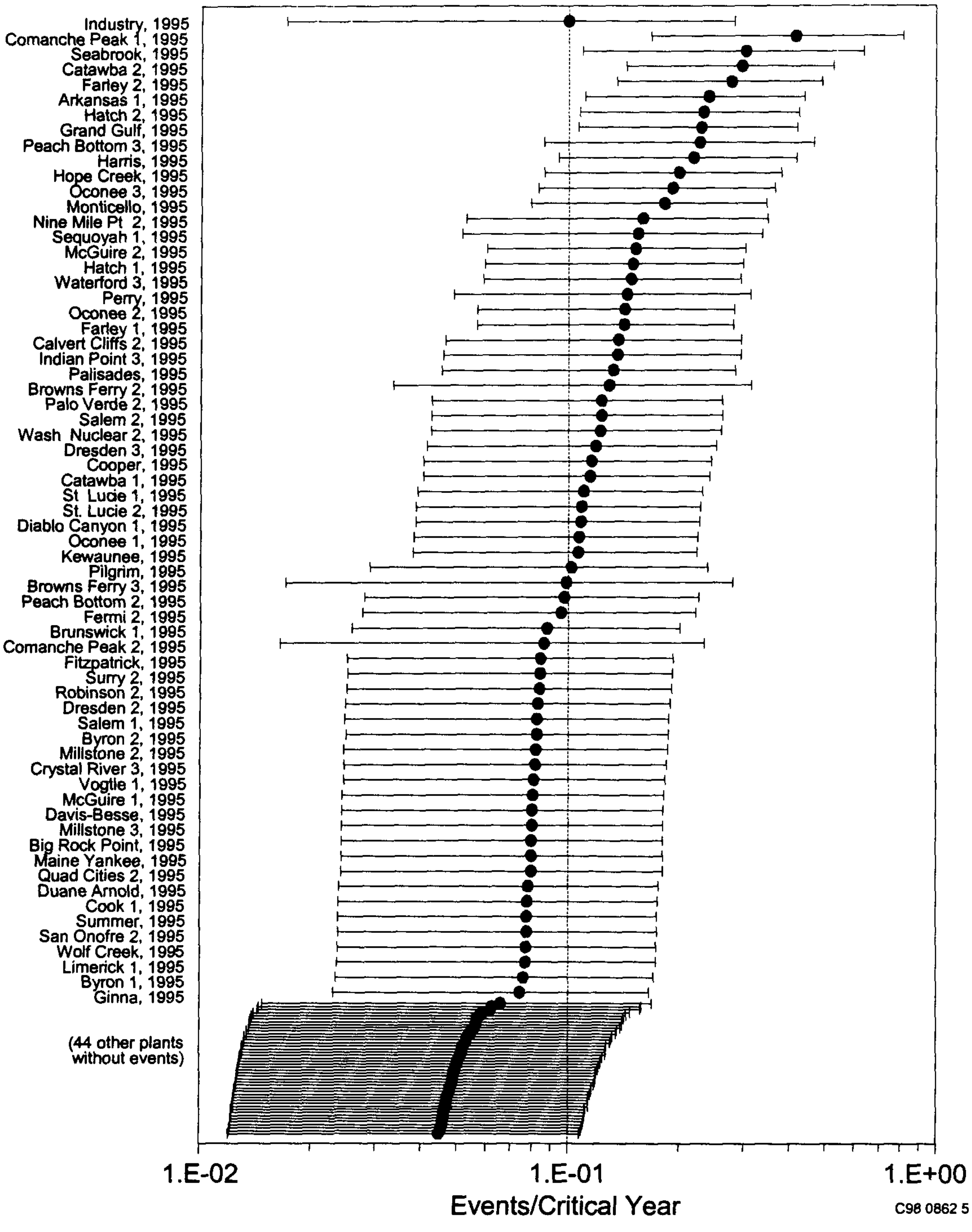

Figure G-5. Plant-specific rates (means and 90% intervals) for functional impact category P1, Total Loss of Feedwater Flow in 1995 for PWRs and BWRs. The ratio of the highest mean to the lowest is 9.3.

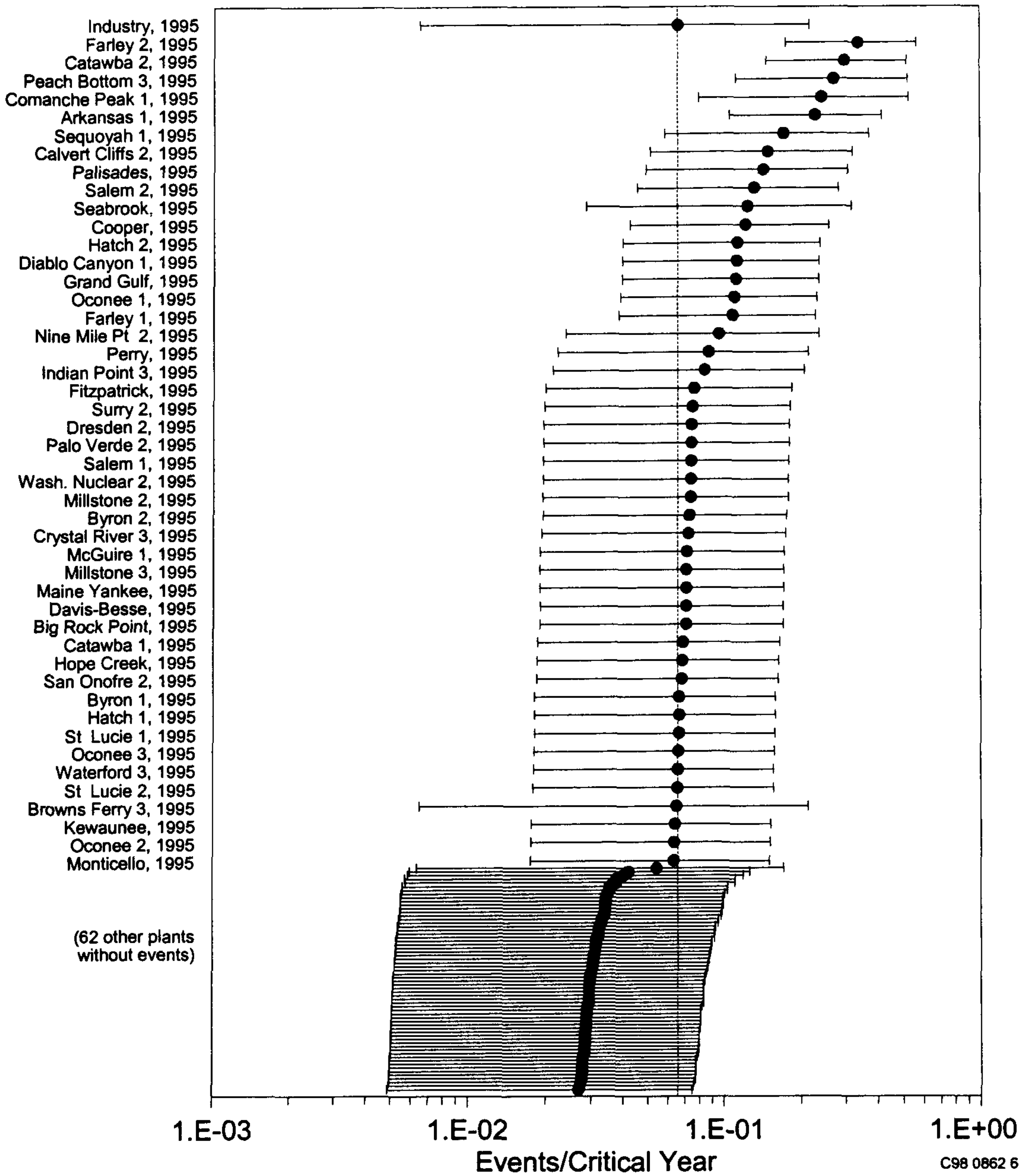

Figure G-6. Plant-specific rates (means and 90% intervals) for initial plant fault category P1, Total Loss of Feedwater Flow in 1995 for PWRs and BWRs. The ratio of the highest mean to the lowest is 12.4.

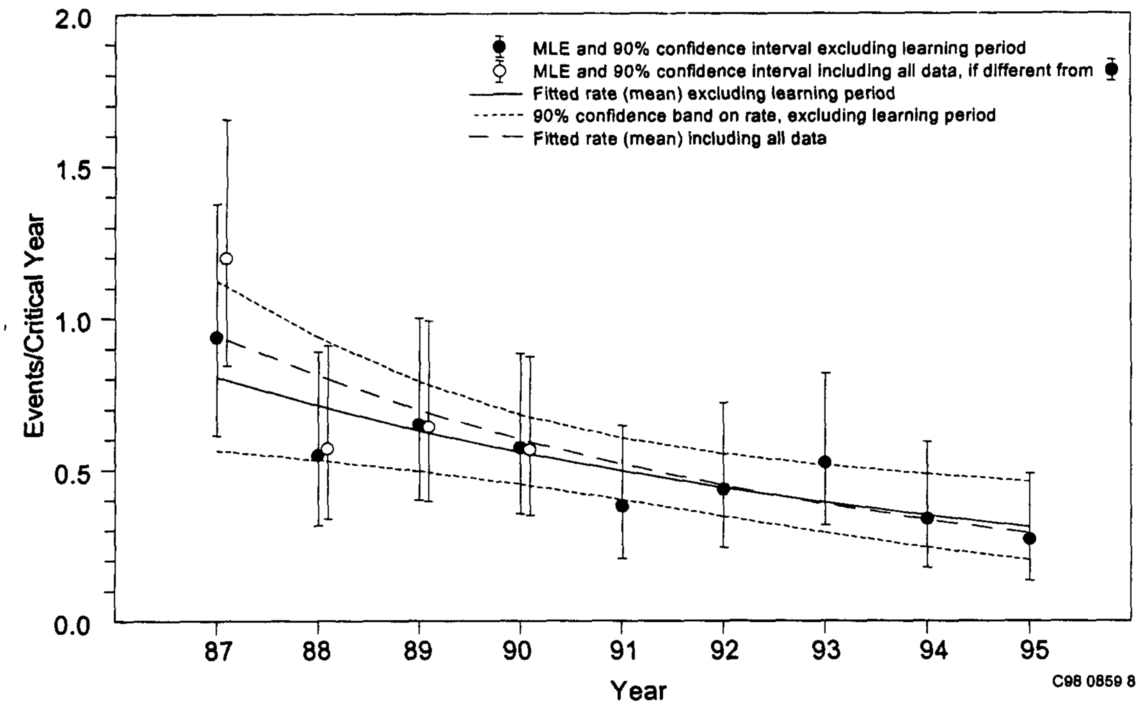

Figure G-7. Time-dependent rate for functional impact heading L, Total Loss of Condenser Heat Sink for BWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants. The 1987-88 events during the learning period make the fitted trend steeper.

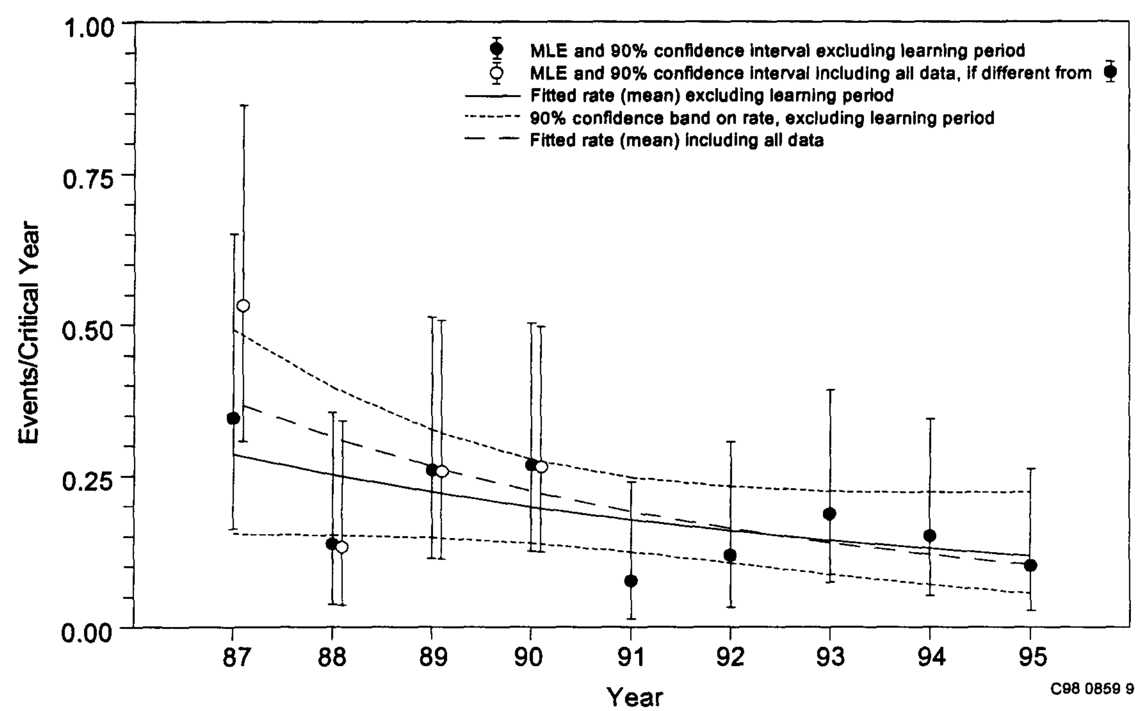

Figure G-8. Time-dependent rate for initial plant fault heading L, Total Loss of Condenser Heat Sink for BWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants.

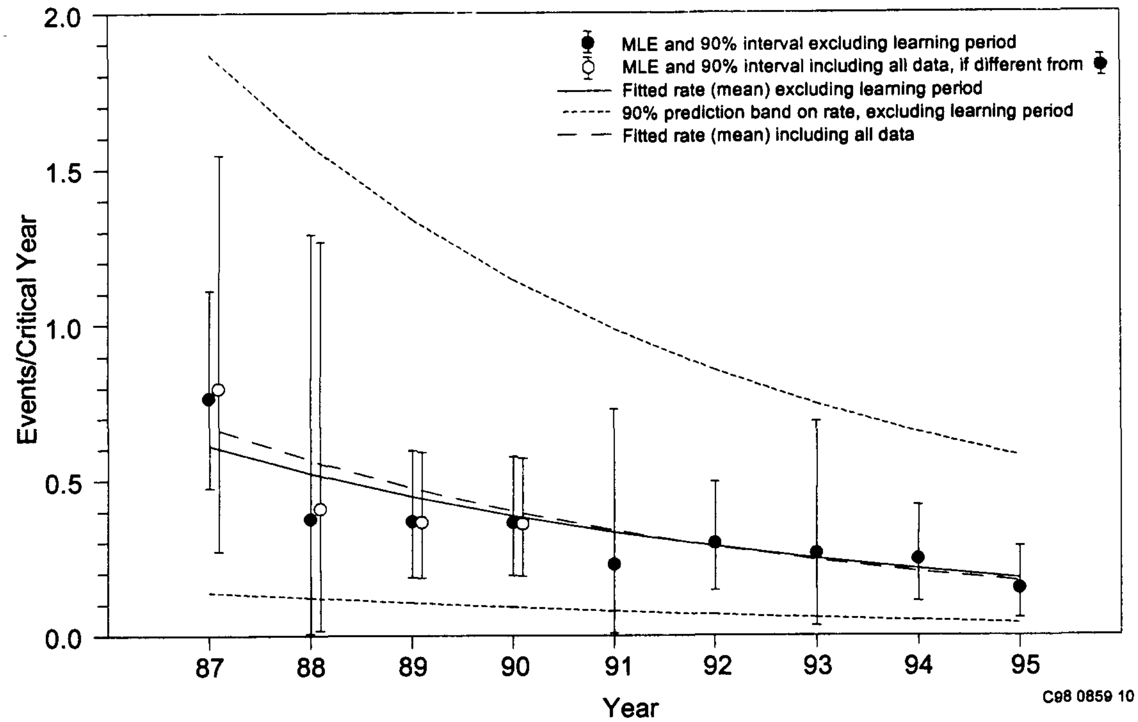

Figure G-9. Time-dependent rate for functional impact category L1, Inadvertent Closure of All MSIVs for BWRs. The points and vertical lines are based on data from individual years, including between-plant variation when possible, and the dotted lines are a 90% prediction band on the rate at a random plant, based on excluding the learning period at new plants. Including or excluding the learning period makes little difference.

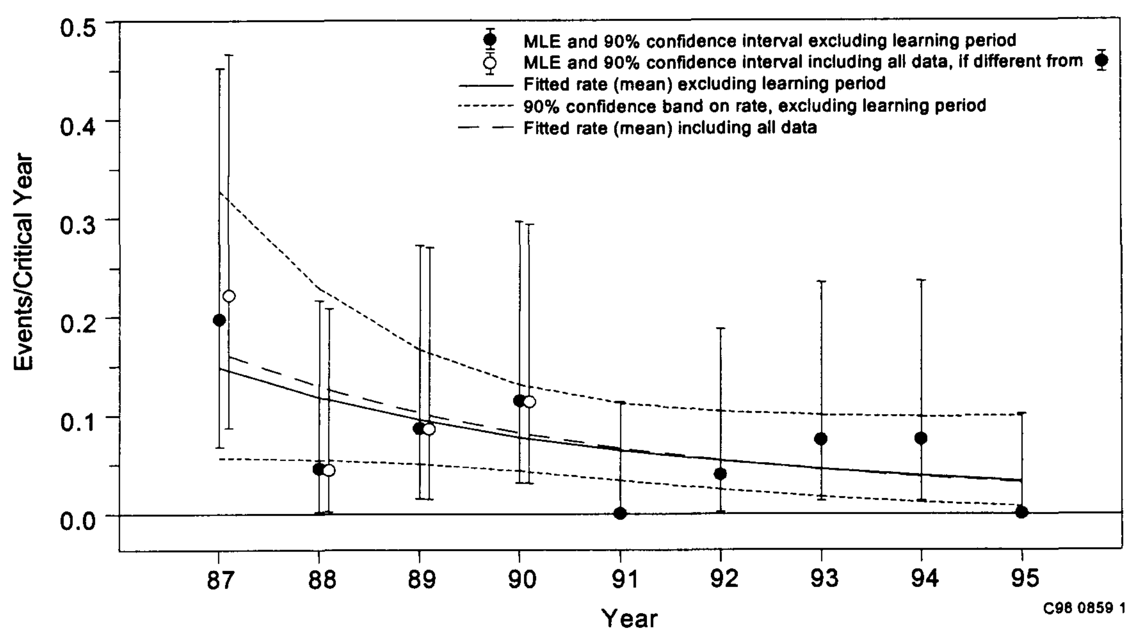

Figure G-10. Time-dependent rate for initial plant fault category L1, Inadvertent Closure of All MSIVs for BWRs. The points and vertical lines are based on data from individual years, including between-plant variation when possible, and the dotted lines are a 90% prediction band on the rate at a random plant, based on excluding the learning period at new plants. Including or excluding the learning period makes little difference.

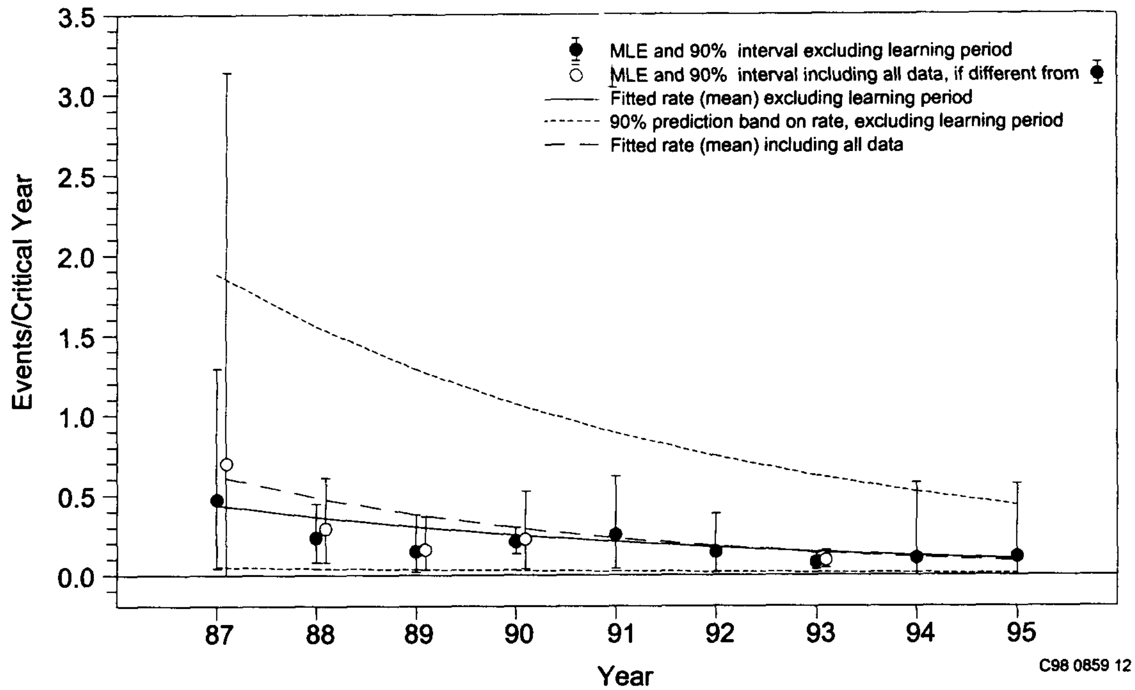

Figure G-11. Time-dependent rate for functional impact category P1, Total Loss of Feedwater Flow for PWRs and BWRs. The points and vertical lines are based on data from individual years, including between-plant variation when possible, and the dotted lines are a 90% prediction band on the rate at a random plant, based on excluding the learning period at new plants. The 1987-90 events during the learning period make the fitted trend steeper.

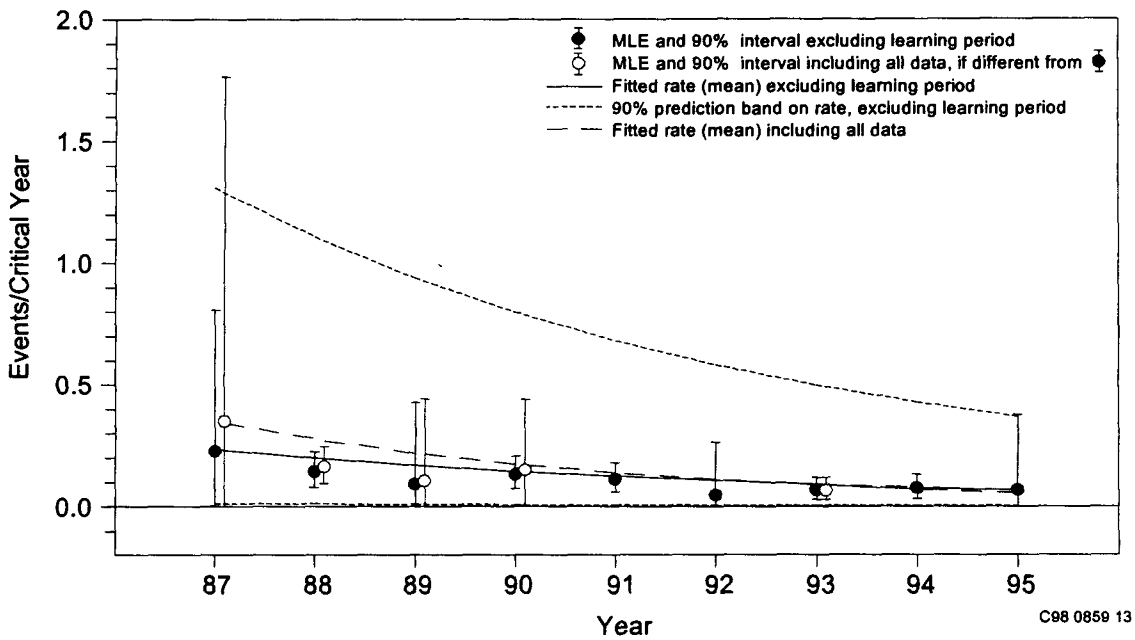

Figure G-12. Time-dependent rate for initial plant fault category P1, Total Loss of Feedwater Flow for PWRs and BWRs. The points and vertical lines are based on data from individual years, including between-plant variation when possible, and the dotted lines are a 90% prediction band on the rate at a random plant, based on excluding the learning period at new plants. The 1987-90 events during the learning period make the fitted trend steeper.

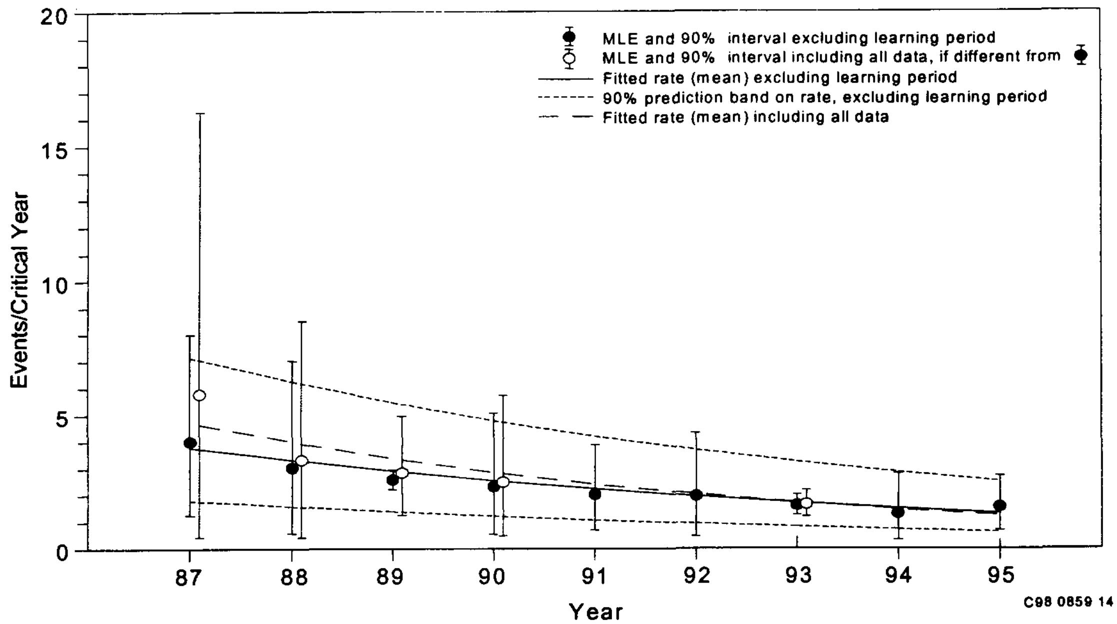

Figure G-13. Time-dependent rate for initial plant fault category Q, General Transients for PWRs. The points and vertical lines are based on data from individual years, including between-plant variation when possible, and the dotted lines are a 90% prediction band on the rate at a random plant, based on excluding the learning period at new plants. The 1987-90 events during the learning period make the fitted trend steeper.

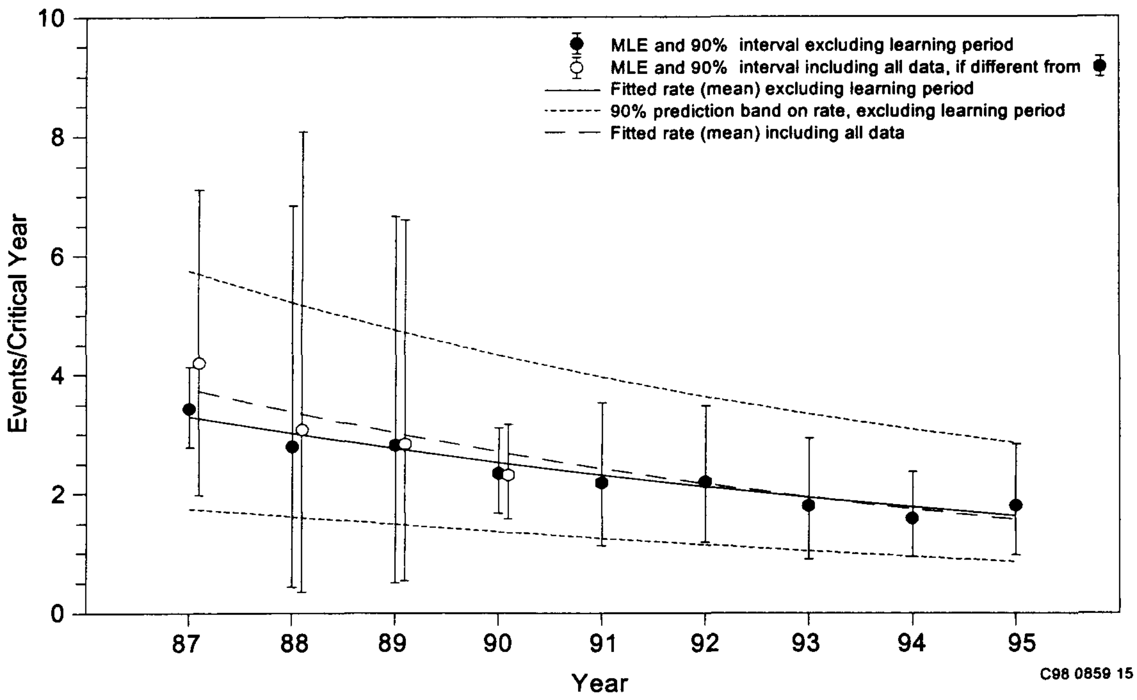

Figure G-14. Time-dependent rate for initial plant fault category Q, General Transients for BWRs. The points and vertical lines are based on data from individual years, including between-plant variation when possible, and the dotted lines are a 90% prediction band on the rate at a random plant, based on excluding the learning period at new plants. The 1987-90 events during the learning period make the fitted trend steeper.

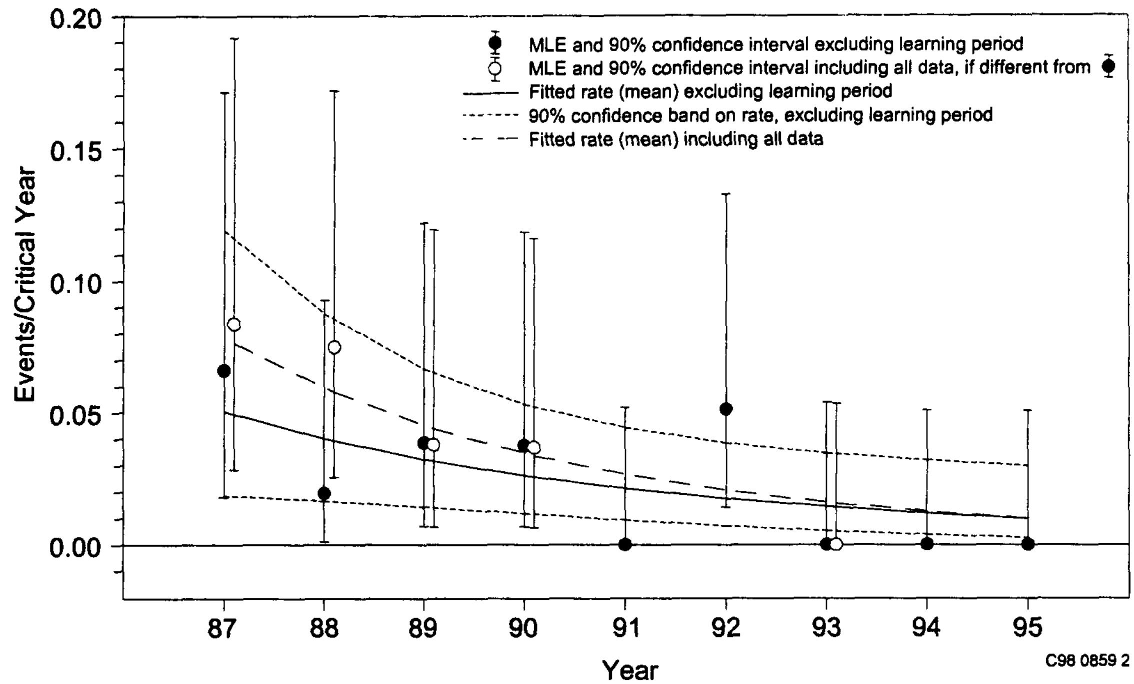

Figure G-15. Time-dependent rate for functional impact category D1, Loss of Instrument or Control Air for PWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants. The 1987-88 events during the learning period make the fitted trend steeper.

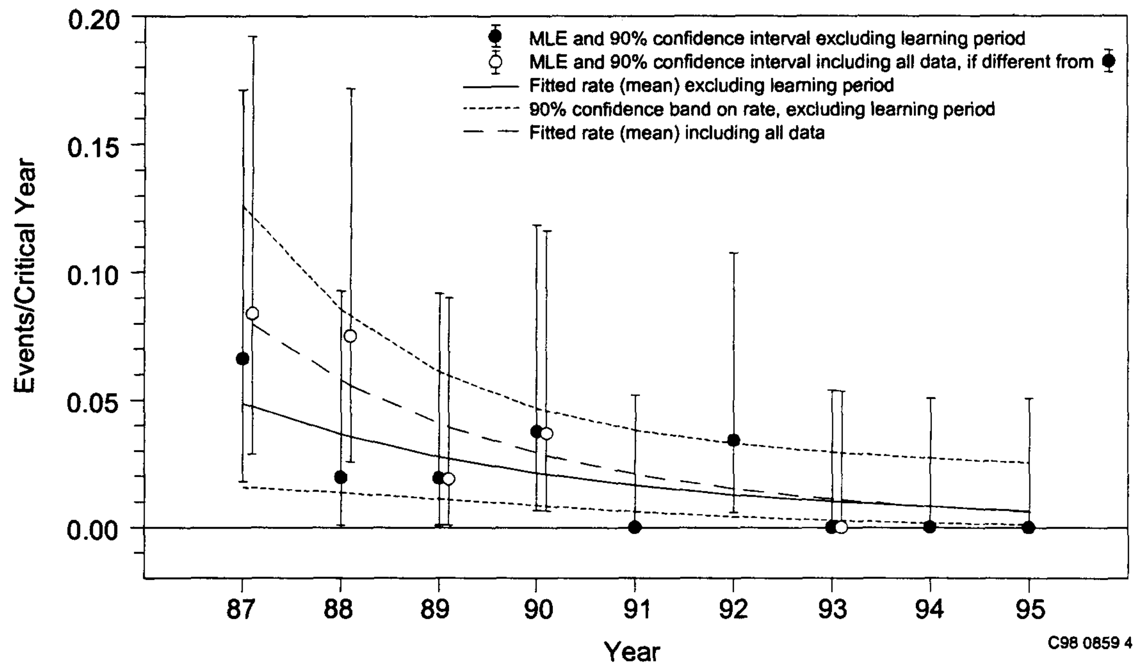

Figure G-16. Time-dependent rate for initial plant fault category D1, Loss of Instrument or Control Air for PWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants. The 1987-88 events during the learning period make the fitted trend steeper.

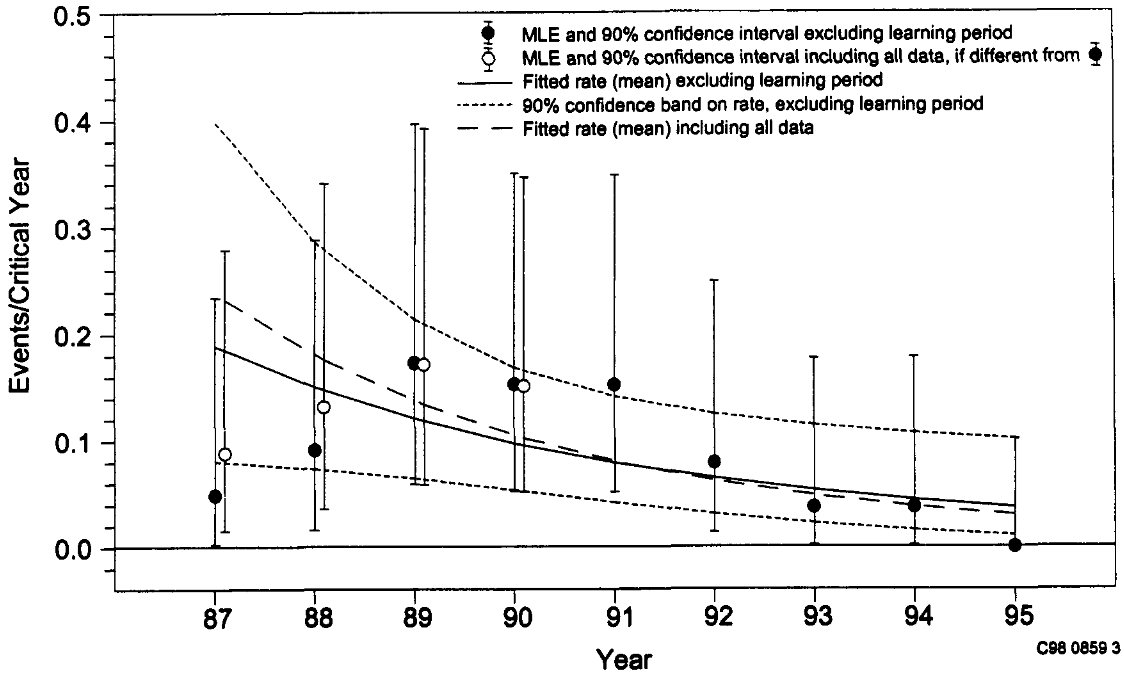

Figure G-17. Time-dependent rate for functional impact category D1, Loss of Instrument or Control Air for BWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants. The 1987-88 events during the learning period make the fitted trend steeper.

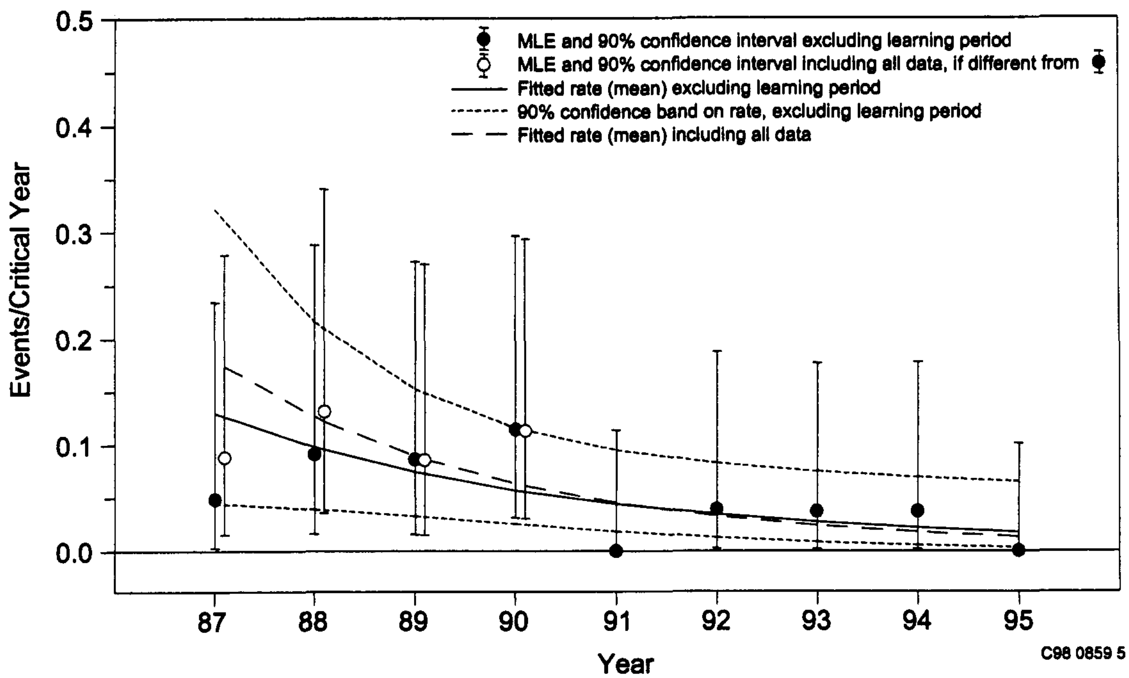

Figure G-18. Time-dependent rate for initial plant fault category D1, Loss of Instrument or Control Air for BWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants. The 1987-88 events during the learning period make the fitted trend steeper.

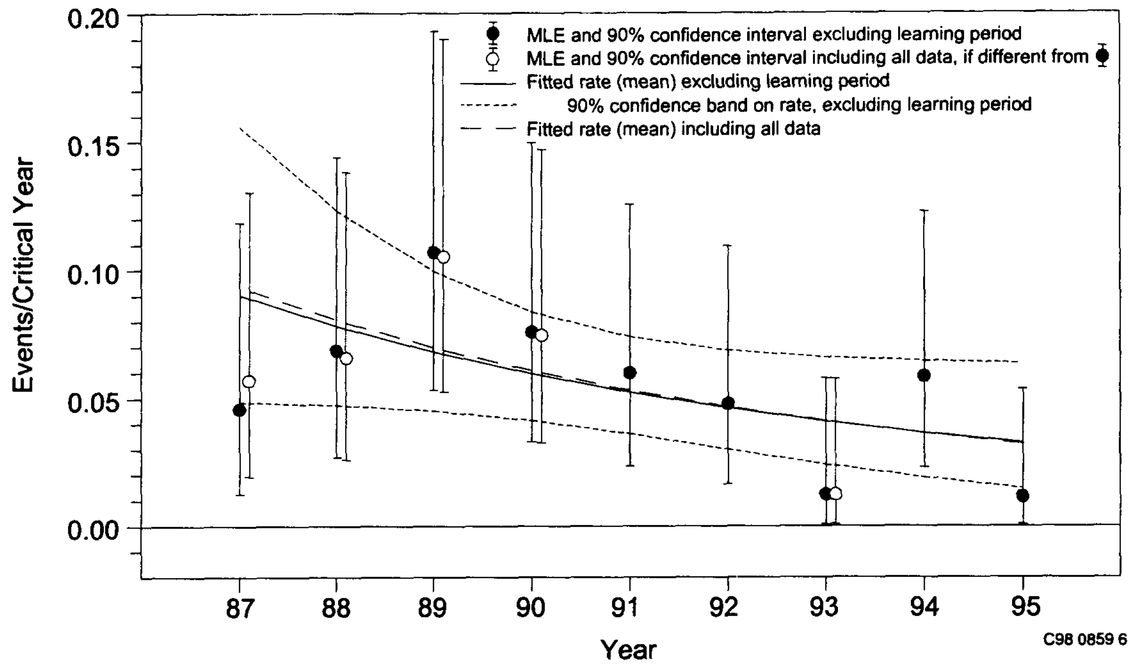

Figure G-19. Time-dependent rate for functional impact category H1, Fire for PWRs and BWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants. Including or excluding the learning period makes little difference.

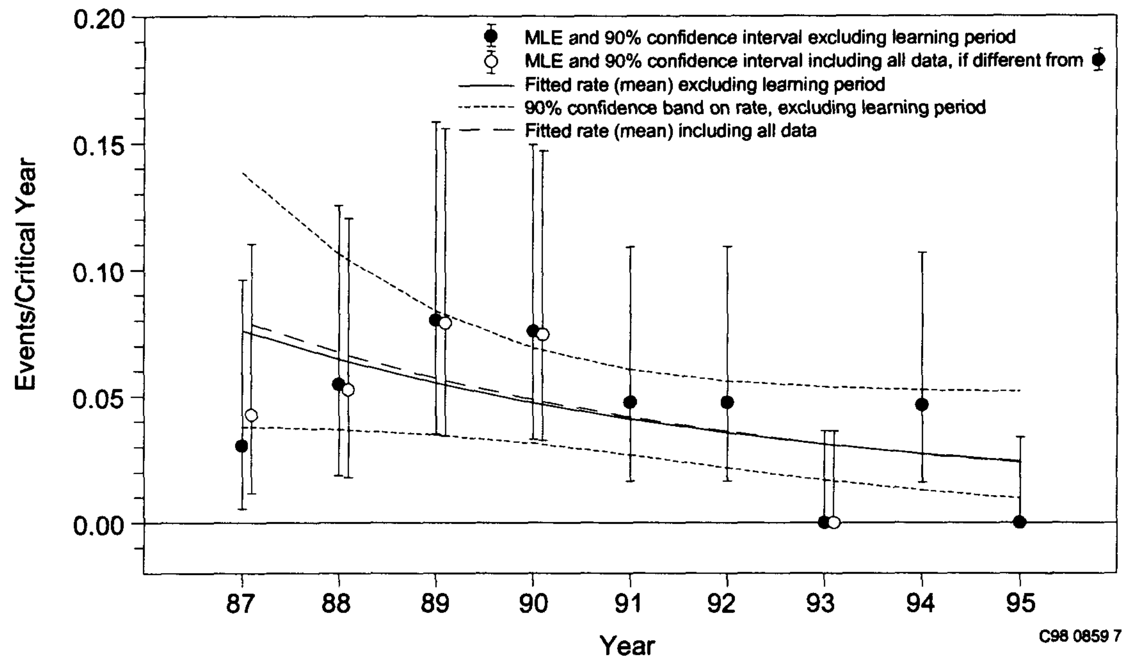

Figure G-20. Time-dependent rate for initial plant fault category H1, Fire for PWRs and BWRs. The points and vertical lines are based on data from individual years, and the dotted lines are a 90% confidence band on the rate, based on excluding the learning period at new plants. Including or excluding the learning period makes little difference.

Appendix H

Calendar Hours, Critical Hours, and Criticality Factors

Appendix H

Calendar Hours, Critical Hours, and Criticality Factors

In the Appendices of this report, rates are given in terms of events per critical year, where a critical year consists of 8,760 (= 365 days × 24 hours/day) critical hours of reactor operation. The total critical hours specified in Table H-1 was multiplied by (1 year/8760 hours) to obtain an equivalent critical year. A critical year is not necessarily the same as a calendar year unless the reactor is critical for the entire year. To convert critical years into events per *calendar* year, the criticality factors given here can be useful. The criticality factor for a plant is the fraction of time the reactor was critical in a given calendar year. Therefore the events per calendar year = events per critical year × criticality factor.

For example, suppose that an event is expected to occur about 0.5 times every critical year, on average and that Table H-3 shows the criticality factor for a plant of interest is 0.8 (reactor has been critical about 80% of the time). Then the same event correlated to units of calendar year is 0.4 events per calendar year [(0.5 events/critical year) × (0.8 critical year/calendar year)] or about two events every five calendar years.

Note that Browns Ferry 1 does not appear in the tables or this report since the last recorded hours of reactor critical operation were in 1985.

Table H-1. Critical hours, by plant.

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
Arkansas1	7856	6157	5999	6500	8150	7138	7599	8658	7576	65632
Arkansas2	7715	6032	6610	8247	7341	6454	8390	7740	6910	65439
Beaver Valley1	7339	7067	5888	8156	5029	8227	5981	7026	6895	61607
Beaver Valley 2	2313	8284	6308	6791	8733	7421	6829	8494	7657	62829
Big Rock Point	6216	6394	6921	6759	7461	4791	6959	6599	8319	60417
Braidwood 1	3426	5746	5587	7830	5353	7237	8081	7001	6379	56640
Braidwood 2	0	4796	7618	6904	6727	8396	7152	6518	8589	56700
Browns Ferry 2	0	0	0	0	4646	8496	5854	7310	8652	34958
Browns Ferry 3	0	0	0	0	0	0	0	0	989	989
Brunswick 1	5789	6661	5749	5948	6061	2518	0	7990	7521	48237
Brunswick 2	8329	5646	5780	5927	5236	2378	5915	6549	8760	54520
Byron 1	6210	6485	8743	7144	7243	8731	7152	7175	7234	66117
Byron 2	6813	8676	7060	6667	8502	7102	7470	8710	7740	68739
Callaway	6228	8202	7482	7365	8734	7289	7569	8760	7419	69048
Calvert Cliffs 1	6616	6399	1807	1925	6687	5050	8619	5912	8545	51559
Calvert Cliffs 2	5958	7827	1718	0	4651	7924	6072	8000	7206	49357
Catawba 1	6076	7070	7485	6349	6373	6396	6991	8734	7782	63257
Catawba 2	7213	6497	6448	6048	6700	8349	7295	7069	7157	62774
Clinton 1	5350	7399	4244	4827	7080	6025	6970	8308	7274	57477
Comanche Peak 1	0	0	0	5303	5489	7103	7021	8674	7539	41128
Comanche Peak 2	0	0	0	0	0	0	5189	5828	8427	19444
Cook 1	6012	8434	6170	6945	7754	5752	8760	6257	6081	62165
Cook 2	6290	2716	6581	4959	8053	3169	8492	5168	8308	53735
Cooper	8424	5968	6673	6953	6899	8467	5147	3076	5851	57458
Crystal River 3	5334	7457	4274	5591	7187	6684	7446	7382	8760	60116

Table H-1. (continued).

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
Davis-Besse	7426	2127	8547	4967	7055	8759	7305	7705	8760	62650
Diablo Canyon 1	8476	5682	7189	8504	7197	7298	8631	7041	7267	67286
Diablo Canyon 2	6059	6191	8137	7433	7486	8673	7385	7560	8492	67415
Dresden 2	5764	6975	7253	5959	5280	7553	4887	5981	3012	52662
Dresden 3	7209	6346	7312	7453	5356	5689	7117	3085	5708	55275
Duane Arnold	5668	6610	6921	6641	8278	7193	6963	8236	7345	63856
Farley 1	8307	7428	7613	8696	6987	7210	8543	7593	7433	69810
Farley 2	6538	8784	7205	6501	8480	7158	6932	8704	7246	67547
Fermi 2	5148	5326	6002	7421	6746	7140	8142	191	7616	53730
Fitzpatrick	6161	6061	8087	6356	4675	0	7158	7292	6529	52318
Fort Calhoun	6608	6510	7817	5622	8030	5792	7081	8726	7290	63477
Ginna	8015	7679	6649	7393	7592	7634	7562	7289	7851	67663
Grand Gulf	7203	8498	7006	6911	8230	7349	7141	8465	7040	67842
Haddam Neck	4729	6177	5883	2825	6693	7040	7146	6810	6809	54112
Harris	6214	6585	6963	7849	7142	6581	8733	7248	7337	64651
Hatch 1	7192	6009	8760	5940	6790	8566	7099	7638	8760	66754
Hatch 2	8520	6359	6496	8685	6779	7005	7874	7620	7122	66458
Hope Creek	7570	7090	6814	8020	7380	7094	8567	7113	6988	66636
Indian Point 2	6347	7492	5644	5837	4763	8625	6631	8760	5885	59984
Indian Point 3	5497	7313	5352	5511	7669	5397	1304	0	1873	39915
Kewaunee	7861	7756	7436	7701	7306	7726	7608	7781	7691	68864
La Salle 1	5609	5931	6115	8475	6747	6568	7402	5313	8302	60463
La Salle 2	4781	6648	6693	6343	8446	6078	5912	8282	6082	59265
Limerick 1	6151	8476	5785	6003	8177	6240	8650	7909	8115	65507
Limerick 2	0	0	1962	7559	7029	8653	7402	8720	8170	49495
Maine Yankee	5724	6950	8210	6216	7585	6951	6992	7960	321	56909
McGuire 1	6836	6784	7211	4808	6328	6863	5164	6339	8080	58412
McGuire 2	7047	7314	6943	5937	8561	6215	6426	7711	8203	64357
Millstone 1	6971	8662	7377	8021	3100	5984	8481	5575	7004	61175
Millstone 2	8242	6953	6028	6552	5141	3204	7690	4349	3392	51550
Millstone 3	6351	7196	6716	7909	2962	6491	6276	8455	5288	57644
Monticello	7174	8769	6679	8487	7076	8566	7391	7624	8760	70526
Nine Mile Pt. 1	8171	0	0	3366	6988	5206	7442	8428	7412	47014
Nine Mile Pt. 2	2703	4525	5206	4800	6972	5648	7377	8374	4834	50440
North Anna 1	4585	8020	5023	8748	6698	7242	6475	8042	8739	63572
North Anna 2	6842	8735	6919	7012	8602	7308	7329	8560	7124	68431
Oconee 1	6914	8769	7371	7775	7288	7586	7928	7372	7595	68596
Oconee 2	8605	6989	7386	7506	8760	7229	7423	7387	8276	69561
Oconee 3	6142	7230	7683	8731	6741	6803	8655	6836	7650	66471
Oyster Creek	5620	5789	5015	7805	5298	7546	7691	6202	8532	59497
Palisades	4227	4990	6051	5143	6846	6686	4707	5872	6639	51161
Palo Verde 1	4589	5763	1522	4198	7599	6117	6782	8675	5218	50463
Palo Verde 2	6985	5750	4226	5376	6719	8480	4723	6103	5275	53637
Palo Verde 3	946	8370	1210	8169	6418	7010	8008	5998	6552	52680
Peach Bottom 2	1730	0	5331	7173	5553	6130	7728	7851	8632	50129

Table H-1. (continued).

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
Peach Bottom 3	1823	0	801	7844	5359	7696	6613	8588	8028	46753
Perry	4505	6939	4997	5880	8055	6630	4219	4399	8378	54002
Pilgrim	0	0	5614	7196	5760	7498	7083	6259	7066	46475
Point Beach 1	7389	7848	7728	7424	7623	7493	7836	8135	7815	69290
Point Beach 2	7583	7708	7244	7739	7645	7546	7925	7851	7276	68516
Prairie Island 1	7288	7836	8741	7840	7988	6851	8508	7292	8760	71104
Prairie Island 2	8760	7814	7852	7786	8760	6538	7381	8743	7699	71334
Quad Cities 1	6252	8478	6621	7318	5030	6250	7020	2651	8031	57652
Quad Cities 2	6941	6293	8435	6305	7795	5693	4726	5874	4295	56355
Rancho Seco	0	5544	2355	0	0	0	0	0	0	7898
River Bend	5995	8280	6052	6835	7642	3487	6272	5684	8725	58973
Robinson 2	6354	5792	4262	5675	7131	5867	6191	6964	7421	55657
Salem 1	6413	6937	6276	6055	6637	5582	5950	6588	2661	53098
Salem 2	6423	5993	7650	5351	7260	5149	5514	6336	2468	52144
San Onofre 1	7383	3818	3583	4163	5790	8022	0	0	0	32759
San Onofre 2	6193	8286	5227	7693	5733	8242	7280	8760	6614	64027
San Onofre 3	7135	5931	8252	6298	8270	6702	6727	8760	7250	65324
Seabrook	0	0	194	5525	6646	7138	8204	5560	7663	40930
Sequoyah 1	0	380	8671	6577	6882	7794	1281	6021	6842	44449
Sequoyah 2	0	5202	6344	6941	8537	7205	2546	5598	8238	50609
South Texas 1	0	5172	5751	5534	6239	6122	720	7080	7684	44302
South Texas 2	0	0	4514	6005	6441	8594	740	5281	8064	39638
St. Lucie 1	6972	7554	8290	5570	7151	8561	6860	7794	6716	65467
St. Lucie 2	7382	8784	6627	6691	8760	6784	6759	7104	6603	65495
Summer	6222	6068	7276	7346	7266	8553	7358	6091	8517	64696
Surry 1	6178	3755	4272	6723	8760	7141	8432	6663	7581	59506
Surry 2	6555	5028	1504	7974	6036	8479	6389	8261	7165	57392
Susquehanna 1	6465	8290	6593	6769	8623	6747	5275	8292	7176	64230
Susquehanna 2	8484	6157	6916	8198	7119	7256	8276	6674	7777	66856
Three Mile Isl 1	6435	6761	8717	7166	7567	8746	7750	8363	7954	69458
Trojan	4731	5925	5423	5811	1409	4797	0	0	0	28096
Turkey Point 3	1910	5408	5807	5284	2252	6034	8501	7718	7928	50841
Turkey Point 4	4503	5050	4147	6803	1426	7226	7442	7568	8638	52803
Vermont Yankee	7375	8404	7416	7523	8265	7743	7021	8646	7618	70011
Vogtle 1	5386	6822	8413	7171	7180	8563	7673	7890	8702	67799
Vogtle 2	0	0	6135	7326	8455	7254	7795	8107	7969	53040
Wash Nuclear 2	6199	6311	6858	5909	4407	5758	6962	6590	6935	55929
Waterford 3	7224	6625	7233	8131	6994	7307	8707	7623	7310	67152
Wolf Creek	6153	6118	8715	7096	6295	7612	7060	7606	8649	65303
Yankee-Rowe	7248	7487	8137	5391	6332	0	0	0	0	34595
Zion 1	6877	6748	5268	5097	4653	4605	6988	4274	6345	50855
Zion 2	5570	7005	8334	3123	5544	5759	5427	6219	6348	53329
Total	615265	666069	666134	706552	735347	734405	726224	751613	778247	6379857

Table H-2. Calendar hours, by plant.[a]

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
Arkansas 1	+	+	+	+	+	+	+	+	+	78888
Arkansas 2	+	+	+	+	+	+	+	+	+	78888
Beaver Valley 1	+	+	+	+	+	+	+	+	+	78888
Beaver Valley 2	3588	+	+	+	+	+	+	+	+	75348
Big Rock Point	+	+	+	+	+	+	+	+	+	78888
Braidwood 1	5196	+	+	+	+	+	+	+	+	75516
Braidwood 2	0	7164	+	+	+	+	+	+	+	68484
Browns Ferry 2	+	+	+	+	+	+	+	+	+	78888
Browns Ferry 3	+	+	+	+	+	+	+	+	+	78888
Brunswick 1	+	+	+	+	+	+	+	+	+	78888
Brunswick 2	+	+	+	+	+	+	+	+	+	78888
Byron 1	+	+	+	+	+	+	+	+	+	78888
Byron 2	8556	+	+	+	+	+	+	+	+	78888
Callaway	+	+	+	+	+	+	+	+	+	78888
Calvert Cliffs 1	+	+	+	+	+	+	+	+	+	78888
Calvert Cliffs 2	+	+	+	+	+	+	+	+	+	78888
Catawba 1	+	+	+	+	+	+	+	+	+	78888
Catawba 2	+	+	+	+	+	+	+	+	+	78888
Clinton 1	7380	+	+	+	+	+	+	+	+	77888
Comanche Peak 1	0	0	0	6540	+	+	+	+	+	51660
Comanche Peak 2	0	0	0	0	0	0	6780	+	+	25500
Cook 1	+	+	+	+	+	+	+	+	+	78888
Cook 2	+	+	+	+	+	+	+	+	+	78888
Cooper	+	+	+	+	+	+	+	+	+	78888
Crystal River 3	+	+	+	+	+	+	+	+	+	78888
Davis-Besse	+	+	+	+	+	+	+	+	+	78888
Diablo Canyon 1	+	+	+	+	+	+	+	+	+	78888
Diablo Canyon 2	+	+	+	+	+	+	+	+	+	78888
Dresden 2	+	+	+	+	+	+	+	+	+	78888
Dresden 3	+	+	+	+	+	+	+	+	+	78888
Duane Arnold	+	+	+	+	+	+	+	+	+	78888
Farley 1	+	+	+	+	+	+	+	+	+	78888
Farley 2	+	+	+	+	+	+	+	+	+	78888
Fermi 2	+	+	+	+	+	+	+	+	+	78888
Fitzpatrick	+	+	+	+	+	+	+	+	+	78888
Fort Calhoun	+	+	+	+	+	+	+	+	+	78888
Ginna	+	+	+	+	+	+	+	+	+	78888
Grand Gulf	+	+	+	+	+	+	+	+	+	78888

Table H-2. (continued).

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
Haddam Neck	+	+	+	+	+	+	+	+	+	78888
Harris	8700	+	+	+	+	+	+	+	+	78788
Hatch 1	+	+	+	+	+	+	+	+	+	78888
Hatch 2	+	+	+	+	+	+	+	+	+	78888
Hope Creek	+	+	+	+	+	+	+	+	+	78888
Indian Point 2	+	+	+	+	+	+	+	+	+	78888
Indian Point 3	+	+	+	+	+	+	+	+	+	78888
Kewaunee	+	+	+	+	+	+	+	+	+	78888
La Salle 1	+	+	+	+	+	+	+	+	+	78888
La Salle 2	+	+	+	+	+	+	+	+	+	78888
Limerick 1	+	+	+	+	+	+	+	+	+	78888
Limerick 2	0	0	3396	+	+	+	+	+	+	56772
Maine Yankee	+	+	+	+	+	+	+	+	+	78888
McGuire 1	+	+	+	+	+	+	+	+	+	78888
McGuire 2	+	+	+	+	+	+	+	+	+	78888
Millstone 1	+	+	+	+	+	+	+	+	+	78888
Millstone 2	+	+	+	+	+	+	+	+	+	78888
Millstone 3	+	+	+	+	+	+	+	+	+	78888
Monticello	+	+	+	+	+	+	+	+	+	78888
Nine Mile Pt. 1	+	+	+	+	+	+	+	+	+	78888
Nine Mile Pt. 2	5340	+	+	+	+	+	+	+	+	78888
North Anna 1	+	+	+	+	+	+	+	+	+	78888
North Anna 2	+	+	+	+	+	+	+	+	+	78888
Oconee 1	+	+	+	+	+	+	+	+	+	78888
Oconee 2	+	+	+	+	+	+	+	+	+	78888
Oconee 3	+	+	+	+	+	+	+	+	+	78888
Oyster Creek	+	+	+	+	+	+	+	+	+	78888
Palisades	+	+	+	+	+	+	+	+	+	78888
Palo Verde 1	+	+	+	+	+	+	+	+	+	78888
Palo Verde 2	+	+	+	+	+	+	+	+	+	78888
Palo Verde 3	1620	+	+	+	+	+	+	+	+	76884
Peach Bottom 2	+	+	+	+	+	+	+	+	+	78888
Peach Bottom 3	+	+	+	+	+	+	+	+	+	78888
Perry	+	+	+	+	+	+	+	+	+	78888
Pilgrim	+	+	+	+	+	+	+	+	+	78888
Point Beach 1	+	+	+	+	+	+	+	+	+	78888
Point Beach 2	+	+	+	+	+	+	+	+	+	78888
Prairie Island 1	+	+	+	+	+	+	+	+	+	78888
Prairie Island 2	+	+	+	+	+	+	+	+	+	78888

Table H-2. (continued).

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
Quad Cities 1	+	+	+	+	+	+	+	+	+	78888
Quad Cities 2	+	+	+	+	+	+	+	+	+	78888
Rancho Seco	+	+	3780	0	0	0	0	0	0	21324
River Bend	+	+	+	+	+	+	+	+	+	78888
Robinson 2	+	+	+	+	+	+	+	+	+	78888
Salem 1	+	+	+	+	+	+	+	+	+	78888
Salem 2	+	+	+	+	+	+	+	+	+	78888
San Onofre 1	+	+	+	+	+	8028	0	0	0	51852
San Onofre 2	+	+	+	+	+	+	+	+	+	78888
San Onofre 3	+	+	+	+	+	+	+	+	+	78888
Seabrook	0	0	4836	+	+	+	+	+	+	57852
Sequoyah 1	+	+	+	+	+	+	+	+	+	78888
Sequoyah 2	+	+	+	+	+	+	+	+	+	78888
South Texas 1	0	7164	+	+	+	+	+	+	+	73308
South Texas 2	0	0	7068	+	+	+	+	+	+	61716
St. Lucie 1	+	+	+	+	+	+	+	+	+	78888
St. Lucie 2	+	+	+	+	+	+	+	+	+	78888
Summer	+	+	+	+	+	+	+	+	+	78888
Surry 1	+	+	+	+	+	+	+	+	+	78888
Surry 2	+	+	+	+	+	+	+	+	+	78888
Susquehanna 1	+	+	+	+	+	+	+	+	+	78888
Susquehanna 2	+	+	+	+	+	+	+	+	+	78888
Three Mile Isl 1	+	+	+	+	+	+	+	+	+	78888
Trojan	+	+	+	+	+	+	0	0	0	52608
Turkey Point 3	+	+	+	+	+	+	+	+	+	78888
Turkey Point 4	+	+	+	+	+	+	+	+	+	78888
Vermont Yankee	+	+	+	+	+	+	+	+	+	78888
Vogtle 1	7140	+	+	+	+	+	+	+	+	78516
Vogtle 2	0	0	6684	+	+	+	+	+	+	60396
Wash. Nuclear 2	+	+	+	+	+	+	+	+	+	78888
Waterford 3	+	+	+	+	+	+	+	+	+	78888
Wolf Creek	+	+	+	+	+	+	+	+	+	78888
Yankee-Rowe	+	+	+	+	+	1356	0	0	0	45180
Zion 1	+	+	+	+	+	+	+	+	+	78888
Zion 2	+	+	+	+	+	+	+	+	+	78888
Total	922776	949044	967128	980196	981120	975624	962820	963600	963600	8665908

a Plus sign (+) indicates a full year, taken as 8760 calendar hours. Zero (0) means that initial criticality had not yet occurred or plant had been decommissioned.

Table H-3. Criticality factor = (critical hours)/(calendar hours), by plant.[a]

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Average
Arkansas 1	0.897	0.703	0.685	0.742	0.930	0.815	0.868	0.988	0.865	0.832
Arkansas 2	0.881	0.689	0.755	0.941	0.838	0.737	0.958	0.884	0.789	0.830
Beaver Valley 1	0.838	0.807	0.672	0.931	0.574	0.939	0.683	0.802	0.787	0.781
Beaver Valley 2	0.645	0.946	0.720	0.775	0.997	0.847	0.780	0.970	0.874	0.834
Big Rock Point	0.710	0.730	0.790	0.772	0.852	0.547	0.794	0.753	0.950	0.766
Braidwood 1	0.659	0.656	0.638	0.894	0.611	0.826	0.922	0.799	0.728	0.750
Braidwood 2	—	0.669	0.870	0.788	0.768	0.958	0.816	0.744	0.980	0.805
Browns Ferry 2	0.000	0.000	0.000	0.000	0.530	0.970	0.668	0.834	0.988	0.443
Browns Ferry 3	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.113	0.013
Brunswick 1	0.661	0.760	0.656	0.679	0.692	0.287	0.000	0.912	0.859	0.611
Brunswick 2	0.951	0.644	0.660	0.677	0.598	0.271	0.675	0.748	1.000	0.691
Byron 1	0.709	0.740	0.998	0.816	0.827	0.997	0.816	0.819	0.826	0.838
Byron 2	0.796	0.990	0.806	0.761	0.971	0.811	0.853	0.994	0.884	0.871
Callaway	0.711	0.936	0.854	0.841	0.997	0.832	0.864	1.000	0.847	0.875
Calvert Cliffs 1	0.755	0.730	0.206	0.220	0.763	0.577	0.984	0.675	0.976	0.654
Calvert Cliffs 2	0.680	0.894	0.196	0.000	0.531	0.905	0.693	0.913	0.823	0.626
Catawba 1	0.694	0.807	0.854	0.725	0.727	0.730	0.798	0.997	0.888	0.802
Catawba 2	0.823	0.742	0.736	0.690	0.765	0.953	0.833	0.807	0.817	0.796
Clinton 1	0.725	0.845	0.485	0.551	0.808	0.688	0.796	0.948	0.830	0.729
Comanche Peak 1	—	—	—	0.811	0.627	0.811	0.801	0.990	0.861	0.796
Comanche Peak 2	—	—	—	—	—	—	0.765	0.665	0.962	0.763
Cook 1	0.686	0.963	0.704	0.793	0.885	0.657	1.000	0.714	0.694	0.788
Cook 2	0.718	0.310	0.751	0.566	0.919	0.362	0.969	0.590	0.948	0.681
Cooper	0.962	0.681	0.762	0.794	0.788	0.967	0.588	0.351	0.668	0.728
Crystal River 3	0.609	0.851	0.488	0.638	0.820	0.763	0.850	0.843	1.000	0.762
Davis-Besse	0.848	0.243	0.976	0.567	0.805	1.000	0.834	0.880	1.000	0.794
Diablo Canyon 1	0.968	0.649	0.821	0.971	0.822	0.833	0.985	0.804	0.830	0.853
Diablo Canyon 2	0.692	0.707	0.929	0.849	0.855	0.990	0.843	0.863	0.969	0.855
Dresden 2	0.658	0.796	0.828	0.680	0.603	0.862	0.558	0.683	0.344	0.668
Dresden 3	0.823	0.724	0.835	0.851	0.611	0.649	0.812	0.352	0.652	0.701
Duane Arnold	0.647	0.755	0.790	0.758	0.945	0.821	0.795	0.940	0.838	0.809
Farley 1	0.948	0.848	0.869	0.993	0.798	0.823	0.975	0.867	0.848	0.885
Farley 2	0.746	1.003	0.823	0.742	0.968	0.817	0.791	0.994	0.827	0.856
Fermi 2	0.588	0.608	0.685	0.847	0.770	0.815	0.929	0.022	0.869	0.681
Fitzpatrick	0.703	0.692	0.923	0.726	0.534	0.000	0.817	0.832	0.745	0.663
Fort Calhoun	0.754	0.743	0.892	0.642	0.917	0.661	0.808	0.996	0.832	0.805

Table H-3. (continued).

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Average
Ginna	0.915	0.877	0.759	0.844	0.867	0.871	0.863	0.832	0.896	0.858
Grand Gulf	0.822	0.970	0.800	0.789	0.940	0.839	0.815	0.966	0.804	0.860
Haddam Neck	0.540	0.705	0.672	0.322	0.764	0.804	0.816	0.777	0.777	0.686
Harris	0.714	0.752	0.795	0.896	0.815	0.751	0.997	0.827	0.838	0.820
Hatch 1	0.821	0.686	1.000	0.678	0.775	0.978	0.810	0.872	1.000	0.846
Hatch 2	0.973	0.726	0.742	0.991	0.774	0.800	0.899	0.870	0.813	0.842
Hope Creek	0.864	0.809	0.778	0.916	0.842	0.810	0.978	0.812	0.798	0.845
Indian Point 2	0.725	0.855	0.644	0.666	0.544	0.985	0.757	1.000	0.672	0.760
Indian Point 3	0.627	0.835	0.611	0.629	0.875	0.616	0.149	0.000	0.214	0.506
Kewaunee	0.897	0.885	0.849	0.879	0.834	0.882	0.868	0.888	0.878	0.873
La Salle 1	0.640	0.677	0.698	0.968	0.770	0.750	0.845	0.607	0.948	0.766
La Salle 2	0.546	0.759	0.764	0.724	0.964	0.694	0.675	0.945	0.694	0.751
Limerick 1	0.702	0.968	0.660	0.685	0.933	0.712	0.987	0.903	0.926	0.830
Limerick 2	—	—	0.578	0.863	0.802	0.988	0.845	0.995	0.933	0.872
Maine Yankee	0.653	0.793	0.937	0.710	0.866	0.793	0.798	0.909	0.037	0.721
McGuire 1	0.780	0.774	0.823	0.549	0.722	0.783	0.590	0.724	0.922	0.740
McGuire 2	0.804	0.835	0.793	0.678	0.977	0.709	0.734	0.880	0.936	0.816
Millstone 1	0.796	0.989	0.842	0.916	0.354	0.683	0.968	0.636	0.800	0.775
Millstone 2	0.941	0.794	0.688	0.748	0.587	0.366	0.878	0.496	0.387	0.653
Millstone 3	0.725	0.821	0.767	0.903	0.338	0.741	0.716	0.965	0.817	0.754
Monticello	0.819	1.001	0.762	0.969	0.808	0.978	0.844	0.870	1.000	0.894
Nine Mile Pt. 1	0.933	0.000	0.000	0.384	0.798	0.594	0.850	0.962	0.846	0.596
Nine Mile Pt. 2	0.506	0.517	0.594	0.548	0.796	0.645	0.842	0.956	0.804	0.667
North Anna 1	0.523	0.915	0.573	0.999	0.765	0.827	0.739	0.918	0.998	0.806
North Anna 2	0.781	0.997	0.790	0.800	0.982	0.834	0.837	0.977	0.813	0.867
Oconee 1	0.789	1.001	0.841	0.888	0.832	0.866	0.905	0.841	0.867	0.870
Oconee 2	0.982	0.798	0.843	0.857	1.000	0.825	0.847	0.843	0.945	0.882
Oconee 3	0.701	0.825	0.877	0.997	0.769	0.777	0.988	0.780	0.873	0.843
Oyster Creek	0.642	0.661	0.573	0.891	0.605	0.861	0.878	0.708	0.974	0.754
Palisades	0.482	0.570	0.691	0.5877	0.781	0.763	0.537	0.670	0.758	0.649
Palo Verde 1	0.524	0.658	0.174	0.479	0.867	0.698	0.774	0.990	0.837	0.666
Palo Verde 2	0.797	0.656	0.482	0.614	0.767	0.968	0.539	0.697	0.854	0.708
Palo Verde 3	0.584	0.955	0.138	0.932	0.733	0.800	0.914	0.685	0.877	0.700
Peach Bottom 2	0.197	0.000	0.609	0.819	0.634	0.700	0.882	0.896	0.985	0.635
Peach Bottom 3	0.208	0.000	0.091	0.895	0.612	0.879	0.755	0.980	0.916	0.593
Perry	0.514	0.792	0.570	0.671	0.919	0.757	0.482	0.502	0.956	0.685

Table H-3. (continued).

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Average
Pilgrim	0.000	0.000	0.641	0.821	0.658	0.856	0.809	0.714	0.807	0.589
Point Beach 1	0.844	0.896	0.882	0.847	0.870	0.855	0.894	0.929	0.892	0.878
Point Beach 2	0.866	0.880	0.827	0.883	0.873	0.861	0.905	0.896	0.831	0.869
Prairie Island 1	0.832	0.894	0.998	0.895	0.912	0.782	0.971	0.832	1.000	0.901
Prairie Island 2	1.000	0.892	0.896	0.889	1.000	0.746	0.843	0.998	0.879	0.904
Quad Cities 1	0.714	0.968	0.756	0.835	0.574	0.713	0.801	0.303	0.917	0.731
Quad Cities 2	0.792	0.718	0.963	0.720	0.890	0.650	0.539	0.671	0.490	0.714
Rancho Seco	0.000	0.633	0.623	—	—	—	—	—	—	0.370
River Bend	0.684	0.945	0.691	0.780	0.872	0.398	0.716	0.649	0.996	0.748
Robinson 2	0.725	0.661	0.487	0.648	0.814	0.670	0.707	0.795	0.847	0.706
Salem 1	0.732	0.792	0.716	0.691	0.758	0.637	0.679	0.752	0.304	0.673
Salem 2	0.733	0.684	0.873	0.611	0.829	0.588	0.629	0.723	0.282	0.661
San Onofre 1	0.843	0.436	0.409	0.475	0.661	0.999	—	—	—	0.632
San Onofre 2	0.707	0.946	0.597	0.878	0.654	0.941	0.831	1.000	0.755	0.812
San Onofre 3	0.815	0.677	0.942	0.719	0.944	0.765	0.768	1.000	0.828	0.828
Seabrook	—	—	0.040	0.631	0.759	0.815	0.936	0.635	0.875	0.707
Sequoyah 1	0.000	0.043	0.990	0.751	0.786	0.890	0.146	0.687	0.781	0.563
Sequoyah 2	0.000	0.594	0.724	0.792	0.975	0.822	0.291	0.639	0.940	0.642
South Texas 1	—	0.722	0.656	0.632	0.712	0.699	0.082	0.808	0.877	0.604
South Texas 2	—	—	0.639	0.685	0.735	0.981	0.084	0.603	0.921	0.642
St. Lucie 1	0.796	0.862	0.946	0.636	0.816	0.977	0.783	0.890	0.767	0.830
St. Lucie 2	0.843	1.003	0.756	0.764	1.000	0.774	0.772	0.811	0.754	0.830
Summer	0.710	0.693	0.831	0.839	0.829	0.976	0.840	0.695	0.972	0.820
Surry 1	0.705	0.429	0.488	0.768	1.000	0.815	0.963	0.761	0.865	0.754
Surry 2	0.748	0.574	0.172	0.910	0.689	0.968	0.729	0.943	0.818	0.728
Susquehanna 1	0.738	0.946	0.753	0.773	0.984	0.770	0.602	0.947	0.819	0.814
Susquehanna 2	0.968	0.703	0.790	0.936	0.813	0.828	0.945	0.762	0.888	0.847
Three Mile Isl 1	0.735	0.772	0.995	0.818	0.864	0.998	0.885	0.955	0.908	0.880
Trojan	0.540	0.676	0.619	0.663	0.161	0.548	—	—	—	0.534
Turkey Point 3	0.218	0.617	0.663	0.603	0.257	0.689	0.970	0.881	0.905	0.644
Turkey Point 4	0.514	0.576	0.473	0.777	0.163	0.825	0.850	0.864	0.986	0.669
Vermont Yankee	0.842	0.959	0.847	0.859	0.943	0.884	0.801	0.987	0.870	0.887
Vogtle 1	0.754	0.779	0.960	0.819	0.820	0.978	0.876	0.901	0.993	0.864
Vogtle 2	—	—	0.918	0.836	0.965	0.828	0.890	0.926	0.910	0.878
Wash. Nuclear 2	0.708	0.720	0.783	0.675	0.503	0.657	0.795	0.752	0.792	0.709
Waterford 3	0.825	0.756	0.826	0.928	0.798	0.834	0.994	0.870	0.834	0.851

Table H-3. (continued).

Plant	1987	1988	1989	1990	1991	1992	1993	1994	1995	Average
Wolf Creek	0.702	0.698	0.995	0.810	0.719	0.869	0.806	0.868	0.987	0.88
Yankee-Rowe	0.827	0.855	0.929	0.615	0.723	—	—	—	—	0.766
Zion 1	0.785	0.770	0.601	0.582	0.531	0.526	0.798	0.488	0.724	0.645
Zion 2	0.636	0.800	0.951	0.356	0.633	0.657	0.620	0.710	0.725	0.676
Average	0.667	0.702	0.689	0.721	0.749	0.753	0.754	0.780	0.808	0.736
PWRs	0.688	0.738	0.713	0.724	0.769	0.788	0.770	0.810	0.811	0.757
BWRs	0.626	0.630	0.640	0.715	0.711	0.682	0.724	0.720	0.801	0.695

a. The criticality factor is undefined when there are zero calendar hours. These values are denoted by dashes.

Appendix I

Summary of Infrequent Events Associated with a Reactor Trip

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Summary of Infrequent Events Associated with a Reactor Trip

I.1 STUCK OPEN AND INADVERTENT OPEN SAFETY/RELIEF VALVE EVENTS

This study identified 14 reactor trip events in the 1987-1995 operating experience associated with primary system safety/relief valves (SRVs) that failed to close. Safety/relief valves included in this study are PWR pressurizer power-operated relief valves (PORVs), BWR main steam line code safety valves, and BWR Automatic Depressurization System relief valves. The mechanisms that caused the valves to open can be divided into three groups: SRV openings induced by a primary system pressure transient (2 events); spurious SRV openings during routine power operations (5 events); and surveillance testing of SRVs in BWRs while at power (7 events). Table I-1 lists the SRV-related reactor trip events found in the 1987-1995 operating experiences.

Each event was reviewed and an engineering judgement was made to determine whether the event should be included in a functional impact category based on the risk significance of the event. Not withstanding, if the SRV-related occurrence was the very first event in the reactor trip sequence that causes or leads to an unplanned, automatic or manual reactor trip, then the LER was included in an initial plant fault category. However, if the same event SRV-related occurrence was judged not to be risk significant, then the event was not classified as a function impact. The bases for the classification of stuck open SRV events found in the 1987-1995 operating experience are discussed below.

- Two spurious SRV opening events in a PWR resulted in a manual reactor trip. The valves closed shortly after the reactor trip, but prior to the pressure reaching the safety injection setpoint. These two events were classified as a general transient under the initial plant fault category QG10, Inadvertent Open/Close: 1 Safety/Relief Valve, and not judged to be functional impacts because the events did not have a risk-related impact on post trip recovery. (LERs: 395/89-011, 395/89-015)

- Three spurious SRV opening events in BWRs occurred during routine power operations and prompted manual reactor trips. The SRV being tested failed to close which resulted in a challenge to the suppression pool during plant cooldown in all three events. These events were classified under the initial plant fault category G2 because the stuck open SRV occurrences were the very first event from the initial plant fault list to occur. These events were also classified under the functional impact category G2 because of the challenges to the suppression pool. (LERs: 265/91-012, 265/91-012, 352/95-008)

- Two events involved a stuck open pressurizer code safety valve following a pressure transient and automatic reactor trip. The safety valves failed to fully close in both events. The Fort Calhoun event (LER 285/92-028) resulted in a safety injection actuation and a 200 gpm leak rate during cooldown. The Calvert Cliffs event (LER 317/94-007) resulted in a maximum leak rate of 25 gpm during cooldown. Both events were classified under the functional impact category G2. The initiating transients were the initial plant faults in both cases.

- Seven events involved failures of SRVs to close during routine SRV testing in BWRs while at reduced power levels (most less than 20% power). Each event prompted a manual reactor trip as required by technical specification. Since no other conditions occurred prior to the SRV failing to close, all seven events were classified under the initial plant fault category G2. In three events, the SRVs closed promptly after manual reactor trip. In two events, the SRV closed on its own 15 and 77 minutes after the reactor trip. In the last three events, the SRVs remained stuck open throughout cooldown. No events resulted in the automatic actuation of a high pressure injection system. All seven events were classified under the functional impact category G2 due to the inability of the control room operator to close the SRV within the time period specified in the technical specifications (usually two minutes) and the demand on the suppression pool during blowdown. (LERs: 254/89-004, 324/93-004, 354/87-047, 373/93-002, 397/92-033, 397/92-033)

- Four events that were related to premature opening of SRVs during a pressure transient were not classified as either an initial plant fault or function impact. In each event, the SRV prematurely opened during the pressure transient due to an out of tolerance lift setpoint and closed on its own shortly after opening. The premature SRV openings did not have an adverse impact on post trip recovery.

Table I-1. Safety/relief valve (SRV) related reactor trip events found in the 1987-1995 operating experience.

	BWR		PWR	
Spurious openings				
Closed promptly after trip			Summer	(395/89-011)
			Summer	(395/89-015)
Stuck open	Dresden 2	(237/90-006)		
	Quad Cities 2	(265/91-012)		
	Limerick 1	(352/95-008)		
Transient induced openings				
Closed promptly after trip[a] (Prematurely opened)	Hope Creek	(354/88-022)	Ft. Calhoun	(285/92-028)
	WNP-2	(397/95-002)	San Onofre 3	(362/90-002)
Stuck open			Ft. Calhoun	(285/92-023)
			Calvert Cliffs	(317/94-007)
Testing induced openings				
Closed promptly after trip	WNP-2	(397/92-033)[b]	N/A	
	WNP-2	(397/92-033)[b]		
Closed after time delay	Quad Cities 2	(265/93-006)	N/A	
	LaSalle 1	(373/93-002)		
Stuck open	Quad Cities 1	(254/89-004)	N/A	
	Brunswick 2	(324/90-004)		
	Hope Creek	(354/87-047)		

a. Events in shaded area were not classified as stuck open SRV events

b. One LER described two separate events (07/06/92, 07/11/92).

I.2 REACTOR COOLANT PUMP SEAL LOCA/LEAK EVENTS

Two events were used to estimate the Reactor Coolant Pump Seal LOCA frequency from catastrophic reactor coolant pump seal failures. The maximum leak rates from both events were 300 and 500 gpm. The description of these events from the Accident Sequence Precursors (ASP) Program status reports (ASP Series) are provided below.

Robinson Unit 2 (No LER). On May 12, 1975, during routine power operations, Robinson Unit 2 experienced gradual flow variations to the number 1 seal for the "C" reactor coolant pump (RCP). The seal leakoff spiked several times, oscillated full range several times, then stabilized with a seal flow greater than six gpm. Plant load was reduced to 36% and the "C" RCP was secured. A reactor trip occurred due to a turbine trip on high steam generator level, resulting from the rapid load reduction and the use of steam dumps for cooldown. The flow control valve in the combined return line from the three RCP thermal barrier cooling lines closed due to high flow caused by cooling water flashing in the "C" RCP thermal barrier. The flashing was caused by hot primary coolant flowing upward through the "C" RCP thermal barrier. Closure of the flow control valve resulted in loss of thermal barrier cooling in all three RCPs. RCPs "A" and "B" were stopped because flashing in the seal return line threatened to cause loss of seal flow due to pressure surges. The flashing was caused by the high primary flow rate through the No. 1 seal of RCP "C". The RCP "C" No. 1 seal return flow isolation valve was closed to decrease pressure surges in the letdown line. Seal flow was lost on RCPs "A" and "B". Leakage through RCP "C" No. 2 seal resulted in high Reactor Cooldown Drain Tank (RCDT) pressures. The RCDT was drained to the containment sump. The flow control valve in the combined return line from the three RCP thermal barriers was blocked open, restoring thermal barrier cooling on all three RCPs. Reactor coolant pump "C" was started with increased seal flow and RCS cooldown was started using condenser dump. A high standpipe alarm was received for RCP "C" and the pump was stopped. Rapidly falling pressurizer level indicated failure of RCP "C" No. 2 and No. 3 seals.

Safety injection pumps "A", "B", and "C" were started to makeup for rapidly decreasing pressurizer level. Pressurizer level stabilized and safety injection pump "C" was stopped. Auxiliary pressurizer spray was used to reduce plant pressure to the operating pressure of the RHR system. During this pressure reduction, the safety injection accumulators partially discharged into the RCS before their isolation valves were closed. Based on system response to the use of auxiliary spray, the utility concluded that a second steam bubble existed in the system, probably in the steam generator tubes, since little gas or steam escaped when the vessel head was later vented.

A total of 132,500 gallons of water leaked into containment. The maximum lead rate of 500 gpm was reported in NUREG/CR-4400 (Azarm and Boccio 1985)The conditional core damage probability for this event that was estimated from the ASP program (Minirack et al. 1982) was 2.5E-3.

Arkansas Nuclear One Unit 1 (LER 313/80-015). On July 17, 1980, Arkansas Nuclear One Unit 1 experienced a reactor coolant pump (RCP) "C" seal failure, resulting in excessive reactor coolant system (RCS) leakage to the containment. A controlled power reduction was begun, and approximately one-half hour later letdown was secured to reduce RCS inventory loss. RCS leak was estimated to be 10-20 gpm. RCS leak rate increased during the power reduction and the plant was subsequently rapidly taken off line. RCP "C" was tripped after the turbine was taken off line but with the reactor critical. RCS leak rate increased substantially when RCP "C" was tripped, and the RCP "C" lift pumps were started and stopped four times in succession in an attempt to reduce the leak rate. On the fourth attempt a reduction in leak rate was noticed. RCS leak rate had increased to a maximum of approximately 350 gpm. The reactor was manually tripped and high pressure injection (HPI) pumps B and C started and all HPI valves opened to provide RCS makeup. The RCP "C" seal return line was isolated to prevent inventory loss through that line and RCP seal flow increased to quench the steam/water leaking by the failed seal. A one-half psi increase in containment pressure occurred and the reactor building emergency coolers were put in service to minimize the pressure increase. One HPI pump was secured and the HPI valves closed 1.3 hours after the seal failure. Two HPI pumps were used to provide continued RCS makeup from the borated water storage tank. Individual SLBIC trains were inadvertently initiated twice during the cooldown, resulting in start of the turbine-driven emergency feedwater pump. This pump was subsequently stopped and the auxiliary feedwater pump lined up to feed the steam generators. During the RCS cooldown, containment entry was required to isolate the two core flood tanks to prevent their discharging into the RCS below 600 psig. A decrease in core flood tank level of 18 in. and 12 in. occurred prior to effecting isolation. Throughout the incident a greater than 100°F margin to saturation existed.

Approximately 60,000 gallons of water collected in containment. The maximum lead rate of 300 gpm was reported in NUREG/CR-4400 (Azarm and Boccio 1985). The conditional core damage probability for this event that was estimated from the ASP program (Cottrell et al. 1984) was 5.0E-4.

This study identified two reactor coolant pump seal failure events in the 1987-1995 operating experience that were associated with a reactor trip. Since the leak rates in both events did not exceed 40 gpm, they were not used to estimate the Reactor Coolant Pump Seal LOCA frequency.

Arkansas Nuclear One Unit 2 (LER 368/88-011). On August 1, 1988, Arkansas Nuclear One Unit 2 experienced a complete severance of a reactor coolant pump (RCP) seal sensing line due to vibratory fatigue, initiating reactor coolant pump seal degradation. The reactor was then manually tripped and the affected RCP was stopped. The maximum leak rate was 40 gpm, however, most of the leakage was coming from the sensing line, not the seal. Later investigations revealed that the carbon faces in the second and fourth stages were broken and those in the first and third stage were cracked (Shah 1998). This event was included in the Very Small LOCA/Leak category (G1).

Palo Verde Unit 3 (LER 530/89-001). On March 3, 1989, Palo Verde Unit 3 experienced a six gpm reactor coolant pump (RCP) seal leak caused by the loss of component cooling water to the pump seal. A fast bus transfer of in-plant nonsafety-related electrical loads did not occur immediately after the reactor trip. This resulted in the loss of component cooling water to the RCP seal cooler. In responding to post trip plant response, the charging system was secured approximately 30 minutes after the reactor trip to prevent pressurizer level from exceeding the maximum limit. The loss of seal injection provided by the charging system allowed hot reactor coolant to circulate up through the RCP seals. One RCP seal became degraded and began leaking at a rate of six gpm prior to the restoration of seal injection. This event was included in the Very Small LOCA/Leak category (G1). In addition, this event was not classified as a total loss of a safety-related cooling water system because the component cooling water system at this plant is powered by a nonsafety-related electrical bus.

I.3 TOTAL AND PARTIAL LOSS OF SERVICE WATER SYSTEMS

One total loss of a safety-related service water system event that was associated with a reactor trip was used to estimate the frequency of the Total Loss of Service Water category (E1). This was the only event in the 1969-1997 operating experienced found in the Accident Sequence Precursors (ASP) database. The description of this event from the ASP series report is reproduced below.

Brunswick Unit 2 (LER 324/82-005). On January 16, 1982, Brunswick Unit 2 experienced a scram due to low condenser vacuum. After the scram, a group 1 isolation occurred and the main steam isolations valves (MSIVs) closed. Operators aligned the reactor core isolation cooling system (RCIC) to supply makeup water to the reactor. Later, when operators attempted to align suppression pool cooling, they discovered that both residual heat removal service water (RHRSW) loops were inoperable. Low suction header pressure lockout signals prevent the start of pump in both loops. Operators reset the group 1 isolation, reopened the MSIVs, reestablished condenser vacuum, and realigned the main feedwater power conversion system for makeup and decay heat removal.

An inspection of the suction header pressure switches found that their sensing lines were partially plugged with sediment, which may have prevented the switches from sensing the actual header pressure, which was within acceptable limits. The suction header pressure switch for the RHRSW "A" loop was also found to be damaged. In addition, the power supply of the "B" loop suction header pressure switch was found to be switched off, apparently having been left that way after prior maintenance work. The pressure switch power feed breaker was reclosed, the RHRSW "B" loop interlock cleared, and the associated RHR train was started and aligned for suppression pool cooling. RHRSW "B" loop was tested and declared operable approximately four hours after the scram. The RHRSW "A" loop was made operable approximately eight hours after the scram. The conditional core damage probability for this event that was estimated from the ASP program (Forester et al. 1997) was 2.4E-4.

Six partial losses of safety-related services water events that were associated with a reactor trip were identified in the 1987-1995 operating experience. These events are summarized below.

Vermont Yankee (LER 271/91-009 and 012). On April 23, 1991, following the expected start of both emergency diesel generators (EDG) during a loss of offsite power (LOSP) event at Vermont Yankee, the EDG heat exchangers were operating at reduced flow and the station air compressor coolers were operating with reduced and reversed flow. The root cause of the event was a weak design modification resulting in an incorrect procedure. The incorrect procedure established an alternate cooling discharge path to the cooling towers and produced a high service water system back pressure of approximately 40 psid. System back pressure was further increased due to various system design and operating characteristics present during the LOSP event. The conditional core damage probability for this event that was estimated from the ASP program (Minarick et al. 1992) was 2.9E-4

Grand Gulf (LER 416/89-019). On December 30, 1989, Grand Gulf experienced a total loss of plant service water (PSW) due to a loss of power to the supply wells. The reactor was manually scrammed. Standby service water (SSW) was initiated and provided cooling for the component cooling water heat exchangers and the drywell chillers. The SSW basin level dropped below the technical specifications limit due to leakage of SSW into PSW. The PSW was restored 63 minutes after the loss and SSW basin inventory was recovered. The power loss to the supply wells was due to a malfunction of the microwave information and control systems. The conditional core damage probability for this event that was estimated from the ASP program (Minarick et al. 1990) was 1.2E-6.

Davis Besse (LER 346/87-011). On September 6, 1987, following a reactor trip at Davis Besse, service water pump no. 1 failed to auto start upon loading of the emergency diesel generator and had to be manually restarted. The reason for failure of the pump to auto start was a missing wire in its breaker cubicle. The conditional core damage probability for this event that was estimated from the ASP program (Minarick et al. 1989) was 6.1E-4.

Millstone Unit 1 (LER 245/90-016). On October 4, 1990, while reducing power during storm conditions, a manual reactor trip was initiated at Millstone Unit 1 because of degraded conditions in the service eater and circulating water supplies. Seaweed buildup on the intake structure traveling screens exceeded the screen wash system removal capability. Debris was carried over the traveling screen head shaft. Three of five traveling water screens incurred damage to the outer baskets because of high differential pressure. Service water pressure decreased due to pump cavitation and self cleaning strainer fouling. A manual scram was initiated when low service water pressures were observed combined with increasing containment temperature, pressure and decreasing condenser vacuum. The containment temperature and pressure increases were the consequence of degraded reactor building closed cooling water heat exchanger performance. The service water system recovered once the pumps regained adequate submergence. Cold shutdown was achieved with the remaining intact traveling screens, circulating water pumps, and service water pumps. The service water strainer bypass valve provided additional sea water cooling reliability. This event did not meet the threshold of an ASP event.

Millstone Unit 3 (LER 423/90-011). On March 30, 1990, Millstone Unit 3 initiated a manual reactor trip due to an anticipated turbine trip from a loss of condenser vacuum. Prior to the trip, the intake structure screen wash system was removed from service to install a repaired elbow. The effort to manually clear the screens from seaweed buildup was not enough to prevent two circulation water pumps from tripping. Operation of the service water system was not jeopardized due to the ratio of service water system flow (approximately 15,000 gallons per minute) to circulating water pump flow (approximately 150,000 gallons per minute) for one bay. When a circulating water pump trips, there is a reduction in flow resistance through the blocked screens. This allows differential level across the screens to return to an acceptable value. The conditional core damage probability for this event that was estimated from the ASP program (Minarick et al. 1991) was 1.1E-6.

Clavert Cliffs Unit 1 (LER 317/87-003). On January 27, 1987, Calvert Cliffs Unit 1 initiated a manual reactor trip due to the decrease in steam generator levels as the result of the control valves drifting shut. A loss of air pressure to the control valves was caused by an inadvertent isolation of the instrument air header from the instrument air compressors while performing a surveillance test. In addition, the containment isolation valves on the component cooling water system failed closed due to the loss of air pressure. The reactor coolant pumps were stopped 16 minutes after the loss of air, due to lack of cooling water. Instrument air and component cooling to the reactor coolant pumps were restored 25 minutes after the loss of air. This event did not meet the threshold of an ASP event.

I.4 INTERNAL FLOOD EVENTS

Two internal flood events that were associated with a reactor trip were identified in the 1987-1995 operating experience. Neither event affected safety-related equipment. Both events are summarized below.

Perry (LER 440/91-027). On December 22, 1991, Perry experienced a catastrophic failure of the 36 inch auxiliary circulating water supply line that occurred in a fiberglass elbow in the pipe just prior to the point where the pipe transitions from fiberglass to carbon steel. The reactor was manually tripped. The pipe was located in a yard area where the pipe exits the ground prior to entering the Heater Bay building. Several instruments and a power distribution component in the Emergency Service Water Pumphouse were damaged by water which entered the building through a series of conduits. This was the only known safety-related equipment affected as a result of flooding. The water which entered the conduit originated in an electrical manhole which became flooded during the pipe rupture event. Several instruments on the non-safety related control rod hydraulic skids became partially submerged from water which entered the Intermediate Building. This event did not meet the threshold of an ASP event.

Perry (LER 440/93-010). On March 26, 1993, Perry initiated a manual reactor trip due a rupture in a 30 inch section of underground non-safety-related service water piping. The catastrophic failure of the 30 inch service water pipe is believed to have resulted from axial pipe stress caused by pipe bending due to a localized loss of soil support. A majority of the water inside the plant entered through spare conduits near the ceiling of control complex. Other buildings affected by internal flooding include the auxiliary building, radwaste building, turbine building, intermediate building, turbine power complex and emergency service water pumphouse. Water in these buildings entered primarily through doors or electrical penetrations. Water levels in the buildings varied between one to eight inches, below levels which could compromise the operability of any safety-related equipment. This event did not meet the threshold of an ASP event.

I.5 FEEDWATER LINE BREAK EVENT

Two feedwater line break events that were associated with a reactor trip were identified in the 1987-1995 operating experience. Both events are summarized below.

Millstone Unit 2 (LER 336/91-012). On November 6, 1991, Millstone Unit 2 initiated a manual reactor tripped due to a rupture of an eight inch diameter pipe which contains pressurized, heated water and serves as a drain line from a first stage reheater drain tank high pressure feedwater heater. The cause of this rupture was severe wall thinning from two-phase erosion/corrosion, cavitation or a combination of both mechanisms. This event did not meet the threshold of an ASP event.

Millstone Unit 3 (LER 423/90-030). On December 31, 1990, Millstone Unit 3 initiated a manual reactor trip due to two six-inch moisture separator drain line piping breaks in the turbine building. The cause of the failure was severe wall thinning that was attributed to single phase erosion/corrosion. The piping failure resulted in the release of approximately 127,000 gallons of steam/water from the condensate piping and hotwell and 65,000 gallons of water from the condensate surge tank. The thermal energy of the fluid released from the ruptured piping activated the fire protection sprinkler system releasing an additional 25,000 gallons of water into the turbine building. In addition to mechanical and electrical damage in the turbine building, a power loss caused the isolation of instrument air to the containment, resulting in the loss of normal pressurizer spray flow and the isolation of normal letdown flow. This event did not meet the threshold of an ASP event.

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Appendix J

LOCA Frequency Estimates

CONTENTS

TABLES

Appendix J

LOCA Frequency Estimates

J-1. INTRODUCTION

This appendix documents an effort to estimate frequencies for loss-of-coolant accidents (LOCAs) in U.S. commercial nuclear power plants. Estimates are made for both pressurized water reactors (PWRs) and boiling water reactors (BWRs) using available operating experience data, information on corrosion mechanisms acting on primary pressure boundary piping, and information from fracture mechanics analysis on crack development and propagation mechanisms.

Most probabilistic risk assessments (PRAs) and individual plant examinations (IPEs) use pipe break-related LOCA initiating event frequencies that have their roots in WASH-1400 (USNRC 1975a). These frequencies are based on median values estimated from a range of point estimates derived from various sources (both nuclear and non-nuclear, both U.S. and foreign). Specifically, a set of pipe rupture frequencies was assembled using the different sources of information available at the time. From the set, median values and associated uncertainties were estimated using engineering judgment. The *mean* values from WASH-1400 for small, medium, and large LOCAs are 3E-3, 8E-4, and 3E-4 per reactor calendar year, respectively. These frequencies result in *mean* time between LOCAs of 375, 1250, and 3760 reactor calendar years respectively. With about 8000 worldwide reactor calendar years[a] of operation, approximately 32 LOCAs (24 small, 6 medium, and 2 large) would be expected. Clearly, these estimates are conservative.

No definitive LOCA frequency estimates have been made since NUREG-1150 (USNRC 1990), which used WASH-1400 values in many cases. Experience data and engineering understanding of pipe failures are much improved since then. The estimates presented in this report represent a reasonable but conservative adjustment to our understanding of the probability of pipe ruptures and LOCA frequencies. In light of this experience, a more complete analysis using data, fracture mechanics analyses, and results from pipe fracture experiments would likely produce more definitive estimates and uncertainties. In the meantime, the available data and current operating experience are sufficient to support an incremental adjustment to the conservative estimates of LOCA frequencies currently used in PRAs. Since the purpose of PRAs is to reflect best estimates and the associated uncertainties, the results presented in this appendix are a reasonable step at producing more accurate PRAs.

Based on this knowledge from the operating experience and the need to provide updated frequencies for NRC PRA programs, the task to update pipe break LOCA frequency estimates was included as an objective of this report. The goal of this effort is to refine the original estimates based on operating experience and current knowledge of pipe break mechanisms. The approach used in this report is intended to reduce unnecessary conservatism in LOCA frequency estimates. However, the results are still conservative. Further probabilistic evaluations utilizing fracture mechanics research are required to develop more realistic estimates of pipe break LOCA frequencies that factor in the effects of current operating, surveillance, and maintenance practices at U.S. nuclear power plants.

Summary of approach. The approaches used in the present analysis for estimating LOCA frequencies can be segregated into two basic types. First, the small (pipe) break LOCA (SBLOCA)

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a. Based on the worldwide IAEA annual report: *Operating Experience with Nuclear Power Stations in Member States in 1996*. Both operating and shutdown reactors were counted.

frequency is estimated from available U.S. operating experience data in a simple Bayes update of the SBLOCA frequency from WASH-1400. Because no difference could be discerned between the PWR and BWR operating experience data and the dominant failure mechanisms (compression fitting failure and failure of socket weld from vibratory fatigue affect both PWRs and BWRs), the data are combined into a single data set. This combined data are then used to update a prior distribution based on the WASH-1400 estimate to produce a single SBLOCA frequency estimate appropriate for both PWRs and BWRs.

To estimate frequencies for events even rarer than SBLOCA, a different process was needed. The frequency estimates for MBLOCAs and LBLOCAs rely on a precursor type of analysis of the data (i.e., throughwall crack and leak events), which are then combined with a conditional probability of a throughwall crack (i.e., the precursor event) transitioning into a rupture. This conditional probability of a break given a throughwall crack is based on a technical review of readily available information on fracture mechanics, data on high-energy pipe failures and cracks, and an assessment of pipe break frequencies estimated by others since WASH-1400. Due to differences observed in both operating experience and engineering characteristics, separate frequency estimates are given for PWRs and BWRs. Also, wherever possible, the LOCA frequencies and the parameters used to calculate them are compared to similar values derived from or presented in the available literature. This includes utilizing results from fracture mechanics computer codes such as PRAISE (Harris and Dedhia 1992).

LOCA sizes. The LOCA pipe break frequency estimates provided in this appendix span the break sizes (small, medium and large) in primary system boundary piping that were used in the NUREG-1150 analysis (as referenced in NUREG/CR-4550, Vol. 1 [Ericson et al, 1990]). The specific break sizes for BWR and PWR used in this report are provided in Table J-1.

The ranges in break sizes used in PRAs depend on the plant-specific design features. Differentiation of LOCA sizes is required since the plant-specific thermal-hydraulic response varies according to the size of the break and the design of the plant. NUREG/CR-4550, Vol. 1, defines the plant response, in terms of required system operability, for various break sizes. For example, a large LOCA is defined as a break that depressurizes the reactor to the point where the low pressure systems can inject automatically providing sufficient core cooling to prevent core damage. NUREG/CR-4550, Vol. 1, determined separate break sizes for use in the analyses as greater than 0.1 square feet (0.009 square meters) for BWRs and greater than 6 inches (150 mm) for PWRs. Because the LOCA size differentiation varies for different plant designs, the break sizes used in NUREG-1150 were adopted in this appendix. However, the data used to calculate LOCA pipe break frequencies are provided in this appendix to allow adaptation of LOCA frequencies for plant-specific applications. The definitions for small, medium and large breaks are provided in Appendix A in this report.

Appendix organization. A summary of results of the detailed analyses and comparison to LOCA frequencies from WASH-1400 and NUREG-1150 follows this introduction. The third major section of this appendix briefly describes some potential degradation mechanisms that might affect the reliability of the primary pressure boundary. The fourth section presents the details of the LOCA frequency calculations. This section includes three subsections, one for each LOCA size. Section 5 documents comparisons between the various parameters used in the LOCA frequency calculations and those that can be extracted from the available information on fracture mechanics analyses and computer code simulations. The last three sections are the tables of events used in the analyses, the list of references, and a bibliography of information reviewed during the conduct of the effort documented here.

Table J-1. Pipe break sizes used in the NUREG-1150 analyses and equivalent pipe diameter.

	Small		Medium		Large	
	Break Area (square feet)	Equivalent Inside Diameter (inches)	Break Area (square feet)	Equivalent Inside Diameter (inches)	Break Area (square feet)	Equivalent Inside Diameter (inches)
BWR						
Liquid piping	<0.004	<1	0.004-0.1	1–5	>0.1	>5
Steam piping	<0.05	<4	0.05-0.1	4–5	>0.1	>5
PWR	n/a	1/2–2	n/a	2–6	n/a	>6

J-2. SUMMARY OF RESULTS

J-2.1 Insights from the Operating Experience

Various sources of information and data were reviewed to identify events involving leaks and throughwall cracks in primary piping, and associated corrosion mechanisms. These sources include U.S. and worldwide operating experience. Specifically used were U.S. licensee event reports (LERs) found in the NRC Sequence Coding and Search System or SCSS (1980-1997, U.S. operating experience); NRC SECY papers; NRC generic communications, such as information notices, generic letters and bulletins (1970-1998, U.S. and worldwide experience); NRC generic safety issues documented in NUREG-0933 (through 1997); NRC technical NUREG-series reports (through 1997); Nuclear Power Experience database (1970-1997, U.S. experience); SKI pipe failure database (1970-1997, worldwide experience); and technical papers found in the literature.

A review of the total U.S. and worldwide nuclear power plant operating experience resulted in the following observations:

- The total world experience includes no reported large or medium pipe break LOCAs in about 8,000 worldwide reactor calendar years of operation.

- No small pipe break LOCAs were reported in the total U.S. operating experience (about 2,100 reactor calendar years).

- The two mechanisms responsible for throughwall cracks in primary pressure boundary piping greater than 2 inches (50 mm) in diameter are IGSCC in BWRs and thermal fatigue cracking in PWRs.

  Mechanisms responsible for degrading small diameter primary piping (<2 inches [<50 mm]) in BWRs and PWRs include IGSCC and other forms of stress corrosion cracking, thermal fatigue (PWR only), compression fitting failures in instrument lines, and vibration fatigue.

- All throughwall cracks in U.S. PWRs and those identified in worldwide experience were found in piping 10 inches (250 mm) in diameter and smaller. The last throughwall crack with medium or large LOCA implications in a U.S. PWR occurred in a 6-inch (150-mm) safety injection nozzle at Farley 2 in 1987.

Most throughwall cracks in U.S. BWRs that occurred up though 1986 were found in bypass line and riser pipe welds in the recirculation system (caused by IGSCC). The last and only throughwall crack since the IGSCC mitigation efforts implemented during the mid-1980s occurred in a 16-inch (410-mm) residual heat removal system suction line weld at Dresden 2 in 1990.

- A total of 58 throughwall cracks in medium and large break LOCAs sized primary pressure boundary piping were found in U.S. BWRs since 1965, most (about 70 percent) were found in large-sized piping. In PWR medium and large break LOCAs sized primary piping, one throughwall crack was identified in the total U.S. PWR operating experience and 4 additional events found in the worldwide experience. Only one throughwall crack in a PWR occurred in large-sized piping. No throughwall cracks in primary piping in BWRs and PWRs resulted in a catastrophic failure.

- Only three throughwall crack events in U.S. plants were detected by leak detection system (leak rates between 0.7 to 6 gpm) while operating at power. All others were found during inservice inspections.

J-2.2 Summary of LOCA Frequency Estimates

Results. Table J-2 summarizes the results of this analysis and compares the frequencies estimated here with those presented in WASH-1400 and NUREG-1150. (The *median* values presented in WASH-1400 have been converted to *means* for the comparisons made in this report. Also, all probability distributions on LOCA frequencies are assumed to be lognormal. Therefore, upper and lower bounds were determined by applying error factors to *median* values.)

Units of LOCA frequency estimates. The LOCA pipe break frequency estimates presented in this appendix were based on calendar years of operation (i.e., calendar year rather than critical year) in order to facilitate comparisons to WASH-1400 and NUREG-1150. The results presented in the executive summary and the main body of the report are given in critical years.

A critical year is not the same as a calendar year unless the reactor is critical throughout the entire calendar year. To convert the frequencies in this appendix to critical years, divide the frequency by an industry average criticality factor of 0.75. This average criticality factor was based on operating experiences that covers U.S. plants in operation during 1987-1995.

Operating experience used to estimate LOCA frequencies. Frequency estimates for pipe break LOCA-related events are based on a combination of total U.S. and worldwide operating experience that includes experience prior to 1987 and after 1995. The operating experience used to estimate the pipe break LOCA frequencies:

- Small pipe break LOCA: Pooled total U.S. PWR and BWR experience (1969-1997)

- Medium and large pipe break LOCA:

  BWR: Total U.S. BWR experience (1969-1997)
  PWR: Total worldwide "western-style" PWR experience (1969-1997).

Table J-3 lists the reactor calendar years of U.S. and worldwide operating experience used to calculate LOCA frequency estimates.

Table J-2. LOCA frequencies compared to WASH-1400 and NUREG-1150.

	Lower Bound (per reactor-calendar-year)[a]	Frequency (mean) (per reactor-calendar-year)[a]	Upper Bound (per reactor-calendar-year)[a]
	Small pipe break LOCA		
PWR			
This analysis[b]	1E-4	4E-4	1E-3
WASH-1400	1E-4	3E-3	1E-2
NUREG-1150	3E-4	1E-3	2E-3
BWR			
This analysis[b]	1E-4	4E-4	1E-3
WASH-1400	1E-4	3E-3	1E-2
NUREG-1150	3E-4	1E-3	2E-3
	Medium pipe break LOCA		
PWR			
This analysis[b]	1E-6	3E-5	1E-4
WASH-1400	3E-5	8E-4	3E-3
NUREG-1150	3E-4	1E-3	2E-3
BWR			
This analysis[b]	9E-7	3E-5	9E-5
WASH-1400	3E-5	8E-4	3E-3
NUREG-1150	8E-5	3E-4	7E-4
	Large pipe break LOCA		
PWR			
This analysis[b]	1E-7	4E-6	1E-5
WASH-1400	1E-5	3E-4	1E-3
NUREG-1150	1E-4	5E-4	1E-3
BWR			
This analysis[b]	9E-7	2E-5	9E-5
WASH-1400	1E-5	3E-4	1E-3
NUREG-1150	3E-5	1E-4	2E-4

a. The LOCA frequencies estimated in this analysis are based on calendar years of operation (i.e., reactor calendar years rather than reactor critical years) in order to facilitate comparisons to WASH-1400 and NUREG-1150. In the main body of this report, frequencies are given in reactor critical years. To convert the frequencies in this appendix to reactor critical years, divide the frequency by an industry average criticality factor of 75%.

b. LOCA frequency estimates calculated in this appendix.

Table J-3. Reactor-calendar-years experience for year 1969 through 1997, inclusive.

	BWR	PWR	Total LWR
U.S.[a]	710	1392	2102
non-U.S.[b]	1038	1970	3008
Total	1748	3362	5110

a. Does not include Big Rock Point, Dresden 1, Fermi 1, Fort St Vrain, Humbolt Bay, La Crosse, Peach Bottom 1.

b. Only includes "Western style" light water reactors (see Table J-10 for a list of included countries).

J-3. DEGRADATION MECHANISMS

J-3.1 Overview of Degradation Mechanisms

While pipe damage in general has been attributable to a number of degradation mechanisms, the only mechanisms that have caused leaks and throughwall cracks in medium and large diameter light water reactor primary piping are thermal fatigue and stress corrosion cracking mechanisms. Leaks and throughwall cracks in medium and large diameter PWR primary piping systems have resulted from thermal fatigue cracking. BWR recirculation system piping of varying diameters has experienced intergranular stress corrosion cracking (IGSCC). Other mechanism such as failure of closure bolts and studs on reactor vessels, reactor coolant pumps, and steam generators, and flow-accelerated corrosion and water hammer damage to unisolable primary piping have been identified in NRC generic communications and generic safety issues. But catastrophic failures due to these mechanisms seems relatively unlikely based on the available research and operating experience. Also, there are industry and NRC programs in place to limit the likelihood of problems from these mechanisms.

Improvements in reducing IGSCC in BWR piping are discussed in the next section. A recent NRC report, NUREG/CR-6582, *Assessment of Pressurized Water Reactor Primary System Leaks* (Shah et al, 1998, see section 3.3.2), provides a detailed discussion on thermal fatigue causing cracking of unisolable primary system branch lines (i.e., safety injection, residual heat removal, and charging lines) in PWRs. Therefore, thermal fatigue will not be addressed here. An overview of other degradation mechanisms and their effects on primary piping is provided below.

Bolting corrosion. The degradation of threaded fasteners in the primary coolant pressure boundary of PWR plants was the subject of an extensive research initiative during the 1980s under Generic Safety Issue 29 (GSI-29), *Bolting Degradation or Failure in Nuclear Power Plants* (Emrit et al 1996). Since 1974, the NRC has issued numerous generic communications on the topic. However, the NRC classified this issue as resolved in Generic Letter 91-17 based on the actions taken in response to NRC guidance and industry initiatives. Furthermore, the NRC conclusion of GSI-29 stated that leakage through bolted pressure joints was possible, but catastrophic joint failure of the reactor coolant pressure boundary was highly unlikely. Since the resolution of GSI-29, the NRC has issued no other generic communication concerning bolt failures.

Erosion/corrosion. Flow-accelerated corrosion has caused pipe damage, and even caused the rupture of secondary system piping made of carbon steel. However, primary coolant piping is fabricated from stainless steel or carbon steel clad with stainless steel. Thus, corrosion-resistant stainless steel is always in contact with primary coolant. The NRC has issued several generic communications and a generic safety issue (GSI-139, *Thinning of Carbon Steel Piping in LWRs* [Emrit et al, 1996]) on flow-accelerated corrosion problems in secondary system piping. However, no corrosion problems associated with primary coolant pressure boundary were referenced. NRC Information Notice 92-35 (USNRC 1992) was issued to describe an unexpectedly high rate of flow-accelerated corrosion in an unisolatable portion of the reactor feedwater piping inside containment at Susquehanna Unit 1. About a 30 percent reduction in thickness was measured in one location in a reducing tee riser in the feedwater distribution piping. However, no rupture or leak occurred in this event. This case appears to be isolated since all plants have developed and put into place a flow-accelerated corrosion monitoring program in response to regulatory and industry initiatives. Also, inspection programs required by Section XI of the ASME Boiler and Pressure Vessel Code help detect wall thinning caused by flow-accelerated corrosion in unisolable portions of BWR carbon steel piping. Since the Susquehanna event, no other generic communications concerning flow-accelerated corrosion in primary pressure boundary piping have been issued.

Thermal embrittlement. Another potential damage mechanism, which could occur in cast stainless steels with significant amounts of delta ferrite, is thermal embrittlement. Although the maximum effect takes place at temperatures much higher than reactor coolant temperatures, a very gradual loss of toughness may take place over long times at reactor coolant temperatures. However, no evidence of failure caused by this mechanism in primary coolant piping has been observed. In addition, the periodic inspection programs required by Section XI of the ASME Boiler and Pressure Vessel Code help verify that structural damage caused by thermal embrittlement is not in evidence in primary coolant piping.

Water hammer. Water hammer events have been reported at both BWRs and PWRs. In BWRs, the events occurred in the residual heat removal system (shutdown cooling mode), isolation condenser, and high pressure coolant injection systems. In PWRs, the events occurred in the feedwater, main steam, auxiliary feedwater, steam generator blowdown, and accumulator systems. However, none of the events resulted in damage to unisolable primary pressure boundary piping or components. Water hammer events in safety-related and balance-of-plant systems from 1985 to 1997 are summarized in NRC Information Notice 91-50 and supplement.

Failure mechanisms in small diameter piping. Based on the operating experience related to the occurrence of leaks in small diameter (<2 inches [<50 mm]) primary system piping, two failure mechanisms (vibratory fatigue failure of socket welds and failure of compression fittings) could potentially produce a catastrophic failure of a small pipe. In the 1985-1996 U.S. PWR experience examined in NUREG/CR-6582 (Shah et al, 1998) 29 primary system leaks were attributed to vibratory fatigue and 14 to compression fitting failures. (A similarly detailed review of the BWR operating experience on leaks was not performed. However, the same type of fitting is used in both PWRs and BWRs.) Leaks caused by vibratory fatigue and compression fitting failure appear to be the result of installation errors that are correctable. Specifically, vibratory fatigue is corrected by installing additional pipe restraints, and compression-fitting failures are the result of improper installation of the fitting.

There have been other leaks, but they were packing leaks or seal leaks, which are judged to not have the potential to grow to SBLOCA size. In addition, although theoretically both thermal fatigue and stress corrosion cracking could potentially cause a failure of small diameter pipe, these mechanisms are believed to be less likely to result in catastrophic failure (i.e., rupture) because they appear less frequently in field experience data (compared to vibratory fatigue and compression fitting failures).

Summary. Although a number of degradation mechanisms are possible in primary system piping, only a few have actually been observed in the operating experience and none resulted in a significant degradation of primary piping. The LOCA frequency estimates in this analysis are not predicated on the impossibility of other degradation mechanisms. The estimates are simply based on those mechanisms believed to dominate the LOCA frequencies.

In small diameter piping, vibratory fatigue and compression fitting failures appear to occur most frequently. For the medium and large size piping, IGSCC seems to be the mechanism of greatest historical concern for BWRs, although the mitigation strategies implemented in the 1980s appear to be having a noticeable positive effect. For PWR piping, thermal fatigue is the most frequent issue of concern, although for small diameter PWR pipe the vibratory fatigue and compression fitting failures are more frequent.

J-3.2 IGSCC Improvement

While pipe damage in general has been attributable to a number of degradation mechanisms, cracking mechanisms are the only ones that have caused reportable damage, including throughwall cracks and leaks, in light water reactor primary piping. While thermal fatigue cracking seems to be primarily associated with PWRs, BWR recirculation lines of varying diameters have experienced intergranular stress corrosion cracking (IGSCC). As described in the NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping* (USNRC 1988a), stress corrosion cracking near weldments in BWR piping has been occurring since the 1960s. Early cases were in relatively small diameter piping. In early 1982, cracking was identified in large diameter piping in a recirculation system of an operating U.S. BWR plant. Since then, BWR piping systems have been extensively inspected. These inspections have resulted in the detection of significant numbers of cracked weldments in most BWRs that began operating before the mid-1980s (i.e., before the IGSCC mitigation initiatives).

According to NUREG-0313, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, Rev. 2* (Haxelton and Koo 1988), piping weldments in BWRs are susceptible to IGSCC. The three elements that, in combination, cause IGSCC are a susceptible (sensitized) material, a significant tensile stress, and an aggressive (oxidizing) environment. A number of domestic and foreign BWR owners have replaced piping systems that have experienced IGSCC with more resistant material. Other owners are implementing countermeasures such as stress improvement or hydrogen water chemistry to reduce the susceptibility of the piping to IGSCC. In many cases, cracked weldments have been repaired by reinforcing them with weld overlays.

Nyman et al. (1997), Tables 4-7 and 4-9, estimate an improvement factor of greater than 20 on the crack occurrence frequency for mitigation efforts (such as using corrosion resistant clad, or using 316 or 304 nuclear grade stainless steel) aimed at eliminating IGSCC. In addition, improved inservice inspection surveillance practices increase the likelihood of early detection and repair of very small cracks before they grow throughwall. This factor of 20 improvement is supported by an analysis of the operating experience data in which they assessed the range of influence IGSCC has on pipe failure probability. In addition, the report references work done by EPRI (Danko 1983) that quantitatively estimates the improvement in pipe reliability gained through the implementation of mitigation strategies. U.S. operating experience shows most cracks in large diameter piping due to IGSCC occurred in the 1970s and early 1980s with the last event in 1990. This experience is consistent with the expected performance improvement following the implementation of mitigation strategies and inservice inspection efforts employed in U.S. BWRs.

The improvement factor of 20 can be applied to both medium and large sized BWR piping. This improvement factor is based on the ERPI sponsored study to demonstrate the benefits of pipe remedies for the mitigation of IGSCC. In the EPRI work, a program was established to test full-size welded pipes of a variety of heats of commercial grades of 304 stainless steels. A statistical test program was formulated that incorporated Type 304 stainless steels pipes of 4-inch (100-mm) diameter welded by standard field procedures and welded pipes with various pipe remedies.

The study concluded that IGSCC initiation would take 20 times longer with a sensitization-related pipe remedy in place. This implies cracking will not take place for more than 40 years after the improvement is implemented. Therefore, the reduction in BWR LOCA frequencies associated with IGSCC will be significantly greater than a factor of 20. Furthermore, the report concluded that a factor of 20 is ultraconservative for larger diameter piping since field failure times are much longer.

The sensitization-related remedies associated with the improvement factor are pipe replacement with Types 304 NG and 316 NG (nuclear grade) stainless steels, and solution heat treatment. These IGSCC mitigation actions were endorsed in Generic Letter 88-01 (USNRC 1988a) and NUREG-0313 (Hazelton and Koo 1988). Most U.S. utilities have implemented hydrogen water chemistry control programs, which will further mitigate IGSCC.

J-4. LOCA FREQUENCY CALCULATIONS

J-4.1 Large Break LOCA Frequency Estimates

The approach taken to estimate the frequency of a large (pipe) break LOCA (LBLOCA) uses the available experience to estimate the number of reactor calendar years and throughwall crack events in large-diameter piping to first estimate a leak frequency. For BWRs, a conservative IGSCC improvement factor of 20 was then applied to the leak frequency calculation. This accounts for experimental and engineering assessments relating to the improvements expected from replacing recirculation system piping with lines containing fewer welds and material less susceptible to IGSCC, improved inservice inspection and crack detection methods, and stress improvement and hydrogen water chemistry designed as a IGSCC countermeasure. A conservative conditional rupture probability (given a throughwall crack or leak) is also estimated and factored into the frequency calculation to produce a rupture (LOCA) frequency estimate. An error factor of 10 (assuming a lognormal distribution) was used to capture the uncertainties in the LBLOCA frequency estimates.

Conditional probability of a rupture given a throughwall crack. For the LBLOCA and MBLOCA estimates for both PWRs and BWRs, this analysis uses the conditional probability of a rupture given a leak that was proposed by Beliczey and Schulz (1990). A simplified correlation was derived from results and insights from structural mechanics models, experimental data, and operating experience with German PWRs. This probability is inversely dependent on pipe diameter and is defined as:

$$P_{R/TW} = 2.5/DN \qquad \text{(J-1)}$$

where

$P_{R/TW}$ = mean probability of rupture given a throughwall (TW) crack

DN = nominal pipe diameter in mm.

This correlation results in conditional probabilities of a rupture given a throughwall crack of 0.1 and 0.01 for a 1- and 10-inch (25- and 250-mm) diameter pipe, respectively. Although not part of the Beliczey and Schulz work, as an added measure of conservatism, a value of 0.01 for pipes larger than 10 nches in diameter has been assumed here.

This simple correlation for various piping diameters is supported by results presented in a recent report from the Swedish Nuclear Power Inspectorate or SKI (Nyman et al. 1997), which used Bayesian statistics and the worldwide SKI pipe failure database to estimate conditional break probabilities for stainless steel piping in nuclear power plants. Furthermore, results from probabilistic fracture mechanics analyses on PWR and BWR piping systems performed by Lawrence Livermore National Laboratory (Harris et al. 1989), Battelle (Rahman et al. 1995), and Pacific Northwest National Laboratory (Simonen, Harris and Dedhia 1998) also support the Beliczey and Schulz correlation. Comparisons to these studies are discussed in Section J-5.1.

IGSCC improvement factor. The available research indicates that IGSCC initiation will take 20 times longer than before the mitigation strategies were implemented. This implies cracking will not take place for more than 40 years after the improvement is implemented. Therefore, the reduction in BWR LOCA frequencies associated with IGSCC will be significantly greater than a factor of 20. Also, the last IGSCC related throughwall crack was found in 1990, which indicates that IGSCC mitigation efforts and improved inservice inspection requirements aimed at reducing the likelihood of IGSCC throughwall cracks are having an affect. Since all throughwall cracks in BWRs were due to IGSCC, the IGSCC improvement factor of 20 is included in the calculation to estimate leak frequencies that could result in a break in large- and medium-sized piping in BWRs. In other words, IGSCC will play a reduced role in estimating LOCA frequencies.

LOCA pipe break frequency calculation. The estimates for LBLOCA and MBLOCA frequencies were calculated using the following equation:

$$\text{LOCA Frequency} = (F_{TW})(P_{R/TW})(IGSCC_{BWR\text{-}only}) \qquad \text{(J-2)}$$

where

F_{TW} = Frequency of throughwall (TW) cracks in primary (unisolatable) piping

= (Number of throughwall cracks/number of reactor calendar year of operating experience)

$P_{R/TW}$ = Mean probability of rupture given a throughwall crack

= 2.5/(nominal pipe diameter in mm), for pipe diameters from 1 to 10 inches (from 25 to 250 mm)

= 0.01, for pipe diameters greater than 10 inches (250 mm)

IGSCC = IGSCC improvement factor, *for BWRs only* = 1/20 = 0.05.

Error factor. The error factor attached to the LBLOCA and MBLOCA estimates developed in this appendix was based on engineering judgment rather than any statistical process. The value of 10 (and a lognormal distribution) was selected since it represents a relatively wide uncertainty band and is consistent with the values used in WASH-1400. NUREG-1150 used error factors of 3 for all LOCA frequencies (all sizes and for both PWRs and BWRs). Given the number of factors and issues influencing LOCA frequencies and the uncertainty about each of them, an error factor of 10 seems more suitable.

J-4.1.1 PWR LBLOCA Frequency Estimation

The PWR LBLOCA frequency estimation is based on worldwide PWR experience through 1997 (i.e., 3362 calendar years of operation). Only experience from "western style" PWR designs is included in the estimate. The Russian-built VVERs (except for the two Finnish-built VVER reactors, which have been heavily modified to western design standards) and pressurized heavy water reactors (e.g., CANDU designs) are not included in this analysis because of limited data and the differences between these reactor designs and U.S. LWRs.

A search of the available literature identified only a single throughwall crack or leak event in large diameter piping. In this foreign reactor event, a 203-mm (8-inch) schedule 140, Type 316 stainless steel residual heat removal system line was found leaking (0.2 gpm) in 1989. The unisolable leak was in the weld joint between an elbow and a horizontal pipe section located between the hot leg and the first isolation valve. The crack extended 3.8 inches (96 mm) circumferentially around the pipe on the inside surface of the weld. About 0.06 inches (1.5 mm) of this crack extended completely through the wall. The crack was reported to be caused by thermal fatigue.

Calculation. Using this single leak event in 8-inch (203-mm) diameter piping, 3362 PWR-calendar years of worldwide PWR experience, and the conditional probability of a leak becoming a rupture (Equation. J-1), the corresponding probabilities of leakage and rupture are as follows.

For 8-inch (203-mm) piping: $\lambda_L = (1/3362) = 3.0E\text{-}4$
$\lambda_B = (2.5/203)\lambda_L = 3.6E\text{-}6$

Therefore, the PWR LBLOCA frequency is estimated as 3.6E-6/calendar year.

Table J-4 compares this estimate to the PWR LBLOCA frequencies presented in WASH-1400 and NUREG-1150.

J-4.1.2 BWR LBLOCA Frequency Estimation

The BWR LBLOCA frequency estimation is based on the total U.S. BWR operating experience from 1969 through 1997 (i.e., 710 calendar years of operation). Only U.S. BWR experience was used to estimate LBLOCA frequencies because data is not readily available from foreign BWRs. Three plants, Dresden 1, La Crosse, and Humboldt Bay, were excluded in the LOCA analysis since they are not representative of the design and operation of currently operating BWRs.

During this time, thousands of cracks have been detected in BWR recirculation system piping and feedwater nozzle regions, but no pipe ruptures. Most cracks were found in older BWR models (i.e., BWR2/3/4). About 34 of these cracks in large-sized primary pressure boundary piping have been

Table J-4. PWR LBLOCA estimated frequency from this analysis (based on worldwide PWR experience through 1997) and values from WASH-1400 and NUREG-1150.

Source	Lower Bound (per reactor calendar year)	Mean Frequency (per reactor calendar year)	Upper Bound (per reactor calendar year)
This analysis	1E-7	4E-6	1E-5
WASH-1400	1E-5	3E-4	1E-3
NUREG-1150	1E-4	5E-4	1E-3

Note: The upper and lower bounds are estimated using engineering judgment and attempt to capture the uncertainty in the various parameters used in the frequency calculation.

throughwall. All cracks were caused by IGSCC in piping weldments. About 2/3 of these throughwall cracks were found in the recirculation system riser pipe welds, in which 40 percent of the riser weld cracks were found during the same inspection outage at one plant. Only one crack was detected by identifiable leakage (3 gpm) while the plant was operating at power. Since the IGSCC issue during the mid-1980s, the last and only throughwall crack in BWR large- and medium-sized piping occurred in a 16-inch residual heat removal system suction line weld at Dresden 2 in 1990. Table J-11 lists the 34 BWR pipe crack events used in the LBLOCA frequency estimate

Calculation. The distribution of the 34 throughwall crack events is as follows: 2 cracks in 28-inch (711-mm) diameter piping, 1 crack in a 22-inch (559-mm) diameter piping, 2 cracks in 16-inch (406-mm) diameter piping, 23 cracks in 12-inch (305-mm) diameter piping, and 6 cracks in 10-inch (250-mm) diameter piping. Since all piping has diameters 10 inches or greater, the conditional probability of a rupture (given a throughwall crack) is 0.01. The corresponding probabilities of leakage (with the IGSCC improvement factor) and break in 710 calendar years of U.S. BWR experience are as follows.

For ≥10-inch (≥250-mm) piping: $\lambda_L = (34/710)(1/20) = 2.4E\text{-}3$

$\lambda_B = (0.01)\lambda_L = 2.4E\text{-}5$

Therefore, the BWR MBLOCA frequency is estimated as 2.4E-5/calendar year.

Table J-5 compares this estimate to the BWR LBLOCA frequencies presented in WASH-1400 and NUREG-1150.

J-4.2 Medium Break LOCA Frequency Estimates

The same approach taken to estimate the LBLOCA frequencies was used to estimate medium (pipe) break LOCA frequencies for PWRs and BWRs. It uses the available experience to estimate the number of reactor calendar years and throughwall crack events in medium-sized piping to estimate a leak frequency. For BWRs, a conservative IGSCC improvement factor of 20 was then applied to the leak frequency calculation. This accounts for experimental and engineering assessments relating to the improvements expected from replacing recirculation system piping with fewer number of welds and material less susceptible to IGSCC, improved inservice inspection and crack detection methods, and stress improvement and hydrogen water chemistry IGSCC countermeasures. A conservative conditional rupture probability (given a throughwall crack or leak) is factored into the frequency calculation

Table J-5. BWR LBLOCA estimated frequency from this analysis (based on total U.S. BWR experience through 1997) and values from WASH-1400 and NUREG-1150.

Source	Lower Bound (per reactor calendar year)	Mean Frequency (per reactor calendar year)	Upper Bound (per reactor calendar year)
This study	9E-7	2E-5	9E-5
WASH-1400	1E-5	3E-4	1E-3
NUREG-1150	3E-5	1E-4	2E-4

Note: The upper and lower bounds are estimated using engineering judgment and attempt to capture the uncertainty in the various parameters used in the frequency calculation.

producing a rupture (LOCA) frequency estimate (Equation. J-2). An error factor of 10 (and a lognormal distribution) was used to capture the uncertainties in the MBLOCA frequency estimates.

J-4.2.1 PWR MBLOCA Frequency Estimation

The PWR LBLOCA frequency estimation is based on worldwide PWR experience through 1997 (or 3362 calendar years of operation). Only experience from "western style" PWR designs are included in the estimate. The Russian-built VVERs (except for the two Finnish-built VVER reactors) and pressurized heavy water reactors (e.g., CANDU designs) are not included in this analysis due to limited available data for these reactor designs.

A search of the available literature identified five throughwall crack or leak events in medium-sized, unisolable primary pressure boundary piping. One throughwall crack was identified in the total U.S. PWR operating experience and four additional events found in the worldwide experience. No throughwall cracks in primary piping resulted in a catastrophic failure or a significant leak rate. All of these leaks were caused by thermal fatigue loadings that were not accounted for in the original design (Shah et al. 1998). The last throughwall crack in a U.S. PWR occurred in a 6-inch safety injection nozzle at Farley 2 in 1987. Table J-12 lists these events.

Four additional throughwall crack events were found in the worldwide experience; however, the pipes had inside diameters less than two inches. Two additional events (Crystal River 3, 1982 and Oconee 2, 1997) were found in the makeup/high pressure injection nozzles (with a 2.1-inch [53-mm] inside diameter) in U.S. Babcock and Wilcox designed reactors; however, the flow rate out of the primary coolant system would be limited by the 1.5-inch (38-mm) thermal sleeve within the nozzle connecting to the cold leg pipe. Therefore, these six events were not used in the MBLOCA frequency estimate due to the effective break area less than that defined for a MBLOCA in this analysis.

Calculation. The distribution of the five throughwall crack events is as follows: 1 crack in 2.5-inch (64-mm) diameter piping, and 4 cracks in 6-inch (150-mm) diameter piping. The corresponding probabilities of leakage and break in 3362 calendar years of worldwide PWR experience are as follows.

For 2.5-inch (64-mm) piping:

$\lambda_L = 1/3362 = 3.0E\text{-}4$

$\lambda_B = (2.5/64)\lambda_L = (3.9E\text{-}2)(3.0E\text{-}4) = 1.1E\text{-}5$

For 6-inch (150-mm) piping:

$\lambda_L = 4/3362 = 1.2E\text{-}3$

$\lambda_B = (2.5/150)\lambda_L = (1.6E\text{-}2)(1.2E\text{-}3) = 1.9E\text{-}5$

Therefore, the total PWR MBLOCA frequency is estimated as

$$\lambda_B = 1.1E\text{-}5 + 1.9E\text{-}5 = 3.0E\text{-}5/\text{calendar year}.$$

Table J-6 compares this estimate to the PWR MBLOCA frequencies presented in WASH-1400 and NUREG-1150.

Table J-6. PWR MBLOCA estimated frequency from this analysis (based on worldwide PWR experience through 1997) and values from WASH-1400 and NUREG-1150.

Source	Lower Bound (per reactor calendar year)	Mean Frequency (per reactor calendar year)	Upper Bound (per reactor calendar year)
This study	1E-6	3E-5	1E-4
WASH-1400	3E-5	8E-4	3E-3
NUREG-1150	3E-4	1E-3	2E-3

Note: The upper and lower bounds are estimated using engineering judgment and attempt to capture the uncertainty in the various parameters used in the frequency calculation.

J-4.2.2 BWR MBLOCA Frequency Estimation

The BWR LBLOCA frequency estimation is based on the total U.S. BWR operating experience from 1969 through 1997 (or 710 calendar years of operation). Only U.S. BWR experience was used to estimate LBLOCA frequencies due to limited readily available data from foreign BWRs. Three plants, Dresden 1, La Crosse, and Humboldt Bay, were excluded in the LOCA analysis since they are not representative of the design and operation of currently operating BWRs.

Fifteen throughwall cracks in medium-sized primary pressure boundary piping have been reported in U.S. BWRs. All cracks were caused by IGSCC in piping weldments. About 2/3 of these throughwall cracks were found in the recirculation system bypass pipe welds. One crack was detected by identifiable leakage (1.5 gpm) while the plant was operating at power. The last throughwall crack found in medium-sized piping was in 1984. The last throughwall crack in a bypass line was reported in 1975. Table J-13 lists the 15 BWR pipe crack events used in the MBLOCA frequency estimate.

Calculation. The distribution of throughwall crack events is as follows: 13 cracks in 4-inch (100-mm) diameter piping, and 2 cracks in 6-inch (150-mm) diameter piping. The corresponding probabilities of leakage and break in 710 calendar years of U.S. BWR experience are as follows:

For 4-inch (100-mm) piping:
$\lambda_L = (13/710)(1/20) = 9.2E\text{-}4$
$\lambda_B = (2.5/100)\lambda_L = 2.3E\text{-}5$

For 6-inch (150-mm) piping:
$\lambda_L = (2/710)(1/20) = 1.4E\text{-}4$, and
$\lambda_B = (2.5/150)\lambda_L = 2.3E\text{-}6$

Therefore, the total BWR MBLOCA frequency is estimated as

$$\lambda_B = 2.3E\text{-}5 + 2.3E\text{-}6 = 2.6E\text{-}5/\text{calendar year}.$$

Table J-7 compares this estimate to the BWR MBLOCA frequencies presented in WASH-1400 and NUREG-1150.

Table J-7. BWR MBLOCA estimated frequency from this analysis (based on total U.S. BWR experience through 1997) and values from WASH-1400 and NUREG-1150.

Source	Lower Bound (per reactor calendar year)	Mean Frequency (per reactor calendar year)	Upper Bound (per reactor calendar year)
This study	9E-7	3E-5	9E-5
WASH-1400	3E-5	8E-4	3E-3
NUREG-1150	8E-5	3E-4	7E-4

Note: The upper and lower bounds are estimated using engineering judgment and attempt to capture the uncertainty in the various parameters used in the frequency calculation.

J-4.3 Small Break LOCA Frequency Estimate

The small (pipe) break LOCA (SBLOCA) frequency estimates are based on U.S. commercial nuclear power plant operating experience, which is used to perform a Bayesian update of the WASH-1400 estimate. In WASH-1400, various sources of data were surveyed, reviewed, and used to produce order-of-magnitude estimates based on engineering judgment. These sources of data included both U.S. and foreign operating experience available at that time, U.S. naval experience, and non-nuclear experience (both U.S. and foreign). However, each data set was evaluated independently to produce a pipe rupture estimate. The entire set of individual estimates was then considered and a median value, and upper and lower bound values were selected.

Reviews of available data sources were conducted to identify any potential SBLOCA events in the U.S. operating experience. Data reviewed included licensee event reports (LERs), events identified in the Accident Sequence Precursor program, and data documented in NUREG/CR-6582 (Shah et al. 1998). Worldwide experience was not used in the SBLOCA frequency estimate due to limited readily available data.

This review yielded two leak events for potential consideration as SBLOCA events. At Oconee Unit 3 in 1987 (LER 287/91-008), a RVLIS instrumentation line that was located at the top of the reactor coolant system hot leg had pulled out of a compression fitting downstream of a root valve. The leak rate was calculated to average approximately 80 gpm at operating pressure. No safety injection actuations occurred, although one high pressure injection pump (which is also used for normal RCS makeup injection) was used to maintain RCS pressure and inventory at this leak rate. The break size was limited by the 3/8-inch (9.5-mm) upstream instrument connection to the RCS hot leg pipe[b]. At Catawba Unit 1 in 1986 (LER 413/86-031), a 360-degree circumferential throughwall crack in the weld on the outlet of the variable letdown orifice resulted in an average 87-gpm leak rate. No safety injection actuation occurred. Neither of these events qualified as a SBLOCA as defined in Appendix A, since the break size in the Oconee event was limited to 3/8-inch (less than the 1/2-inch [13-mm] lower limit of a SBLOCA) and the pipe break in the Catawba event was located outside the primary pressure boundary.

b. Letter from Duke Energy Corporation (M.S. Tuckman to NRC) dated September 14, 1998.

Calculation. The SBLOCA frequency was calculated using the WASH-1400 estimate as the prior distribution and updating it with zero failures in 2102 U.S. reactor calendar years of operation.

A sensitivity calculation was performed using a Bayes update with a Jefferys noninformative prior as follows:

Mean frequency of SBLOCA = 0.5/2102 = 2.3E-4/calendar year.

The lower and upper bounds in the uncertainty using a Jefferys noninformative prior are 9E-6 and 9E-4, respectively. As compared with the results using WASH-1400 and Jefferys as prior, the WASH-1400 prior yields an approximately similar mean value but a reduced uncertainty interval.

Table J-8 shows the estimated SBLOCA frequency, using the distribution from WASH-1400 as a prior. For comparison, the WASH-1400 and NUREG-1150 distributions are also shown.

J-5. COMPARISON OF RESULTS TO HISTORICAL EXPERIENCE AND RESEARCH

This section examines in more detail two of the more significant factors used in the medium and large break LOCA frequency calculations. Use of the conditional probability of rupture (given a throughwall crack or leak) and the IGSCC improvement factor are both motivated by reviews of the operating experience data. Throughwall cracks have been found in both PWRs and BWRs without degenerating into a rupture. Hence it is a simple fact that not all leaks lead to ruptures. The only questions are what is the conditional probability and what is the basis and support for whatever value is chosen. Similarly, the occurrence of IGSCC has decreased significantly over the last 10 years. The timing of this decrease coincides with the implementation of the IGSCC mitigation activities of the U.S. nuclear power industry. Again, an obvious correlation exists. The question is how to account for the effects of the IGSCC mitigation efforts when quantifying LOCA frequencies. The following sections address these issues.

J-5.1 Conditional Rupture Probability

For the MBLOCA and LBLOCA estimates for both PWRs and BWRs, this analysis uses the Beliczey and Schulz (1990) conditional probability of a break given a leak. This probability is inversely dependent on pipe diameter and is defined as:

$$P_{R/TW} = 2.5/DN \qquad \text{(J-1)}$$

where

$P_{R/TW}$ = mean probability of rupture given a throughwall (TW) crack

DN = nominal pipe diameter in mm.

Table J-8. LWR SBLOCA estimated frequency from this analysis (based on WASH-1400 as a prior and total U.S. BWR and PWR experience through 1997) and values from WASH-1400 and NUREG-1150.

Source	Lower Bound (per reactor calendar year)	Mean Frequency (per reactor calendar year)	Upper Bound (per reactor calendar year)
This analysis	1E-4	4E-4	1E-3
WASH-1400	1E-4	3E-3	1E-2
NUREG-1150	3E-4	1E-3	2E-3

This correlation was developed from results and insights from structural mechanics models, experimental data, and operating experience with German PWRs. Although not part of the Beliczey and Schulz work, as an added measure of conservatism a value of 0.01 for pipes larger than 10 inches (250 mm) in diameter has been assumed here.

J-5.1.1 Comparison to the SKI Pipe Failure Database

Although not defined identically, the results from the Beliczey and Schulz correlation compare reasonably well with results from studies conducted by the Swedish Nuclear Power Inspectorate (SKI). The results of conditional failure probabilities of nuclear piping presented in the SKI report by Nyman et al. (1997) used Bayesian statistics and the worldwide SKI pipe failure database for stainless steel piping. The SKI's LOCA Affected Piping (SLAP) Database contains data on reported pipe failures in light water reactors during the period 1970 to the present. As of October 1997, the database included about 2,360 failure reports from BWRs, PWRs, light water cooled and graphite moderated reactors, and Russian PWRs (VVER design). The data were collected from 274 plants and covered 4,741 reactor (calendar) years of operating experience. The pipe failure data were classified into three failure modes: crack, leak, and rupture.

The SKI data show that all complete failures (ruptures) in large diameter pipes occurred in balance-of-plant systems, support systems, or fire protection systems. Complete failures in LOCA-sensitive primary coolant pressure-boundary piping were restricted to small diameter piping of less than 1 inch (25 mm) (e.g., instrument lines, vent and drain lines, test and sample lines). In addition, the report provided a comparison of the Beliczey and Schulz correlation with results of conditional failure probabilities calculated by the SLAP database for stainless steel piping of various sizes (Nyman et al. 1997, Figure 4-2). This comparison plots the estimated conditional probabilities of rupture for pipes ranging in size from 1 inch to 10 inches (25 mm to 250 mm).

The expression for the conditional probability reported in the SKI report is derived from a Bayes update of the Jefferys noniformative prior as follows

$$P_{R/DP}=(2R + 1)/(2DP + 2) \tag{J-3}$$

where

$P_{R/DP}$ = mean probability of rupture given a degraded piping (DP) (i.e., cracks more than 20% throughwall)

R = number of rupture events, that is, complete failure

DP = number of occurrences of degraded piping with certain attributes (diameter, materials, etc.). Occurrences include consideration of flaw/crack indications (cracks more than 20% throughwall), leaks or ruptures.

The mean values of the conditional probabilities for stainless steel piping based on Bayesian statistics (Equation J-3) and the SKI worldwide pipe failure database, and the Beliczey and Schulz correlation (Equation J-1) are shown in Table J-9 (Nyman et al. 1997). The greatest difference appears for 1-inch (25-mm) pipe where the SLAP-based estimate predicts a value only one-half that of the Beliczey and Schulz correlation. For pipe 10 inches (250 mm) or larger in diameter, the conditional break probability of 0.01 compares reasonably well with the results presented in the SKI report. The SKI calculations produce a value of 0.0051 as the conditional probability of rupture for large diameter (≥ 10 inches [≥ 250 mm]) stainless steel piping. The results for conditional probabilities using Equation J-3 are

Table J-9. Conditional probability of rupture of stainless steel piping by diameter.

Diameter (mm)	$P_{R/DP}$ (mean) Bayesian Statistics, Equation. (J-3)	Correlation, Equation. (J-1)
DN[a] ≤ 25	5.8E-2	0.1
25 < DN ≤ 50	4.1E-2	5.0E-2
50 < DN ≤ 100	2.7E-2	2.9E-2[b]
100 < DN ≤ 250	1.5E-2	1.3E-2[c]
250 > DN	5.1E-3	5.7E-3[d]

a. DN = nominal pipe diameter in mm.

b. Calculated for piping diameter of 86 mm, which is the harmonic mean (inverse of the mean of the inverses) of DN75 and DN100

c. Calculated for piping diameter of 187 5 mm, which is the harmonic mean of DN150 and DN250

d. Calculated for piping diameter of 436.4 mm, which is the harmonic mean of DN300 and DN800.

within one standard deviation of the corresponding results using Equation J-1, except for the piping diameters less than or equal to 1 inch (25 mm).

The results presented in Table J-9 show that the conditional probability of rupture given a leak increases with decreasing pipe diameter. Beliczey and Schulz suggest the following reasons why this relation is plausible, at least qualitatively.

- Loadings due to vibrations, not taken into account at the construction and the design stages, are of decreasing influence with increasing diameter.
- Loadings from inertial forces originating from the liquid flow (for example, closing actions of valves) can be predicted more accurately during design as the diameter is increased.
- The number of layers of weld beads is larger, thus the influence of faults in weld beads is smaller as the pipe diameter is increased.
- Conditions during manufacturing can be better controlled and quality assurance is better with larger pipes.
- The number of recurring inspections is greater for larger pipes.
- The reliability of early leak detection is increased because of the larger amount of leakage with increasing pipe diameter.

This relationship is further supported by probabilistic fracture mechanics analyses on PWR and BWR piping systems performed by Lawrence Livermore National Laboratory (Holman and Chow 1989), Battelle (Rahman et al. 1995), and Pacific Northwest National Laboratory (Simonen, Harris and Dedhia 1998). These analyses show that the dominating contribution to LBLOCAs is not from very large piping (i.e., main recirculation loops in BWRs, hot/cold leg loops in PWRs), which have probabilities of a double-ended guillotine break of around 1E-10 to 1E-12 per reactor calendar year. Instead, ruptures of smaller diameter piping in the 6- to 12-inch (150- to 305-mm) diameter range, have a higher probability of occurrence.

J-5.1.2 Comparison to the Operating Experience

Results from the Beliczey and Schulz correlation for 4-inch (100-mm) piping compares reasonably well to a point estimate calculated from total U.S. BWR operating experience of 4-inch (100-mm) piping with no breaks. Treating leak events as demands, the conditional break probability can be estimated using a Jefferys noninformative prior in a Bayesian update calculation.

Using 18 throughwall crack events in 4-inch (100-mm) pipes found in the total U.S. BWR operating experience (including Dresden 1 events since this plant was judged atypical with respect to the initiation of cracks, not the progression from crack to rupture):

$P_{R/TW}$ in 4-inch (100-mm) pipe = 0.5/19 = 0.026

The conditional break probability calculated from the Beliczey and Schulz correlation:

$P_{R/TW}$ in 4-inch (100-mm) pipe = 2.5/(4 in. x 25.4 mm/in.) = 0.028.

The conditional break probability of 0.01 compares reasonably well to a point estimate calculated from total U.S. BWR operating experience of piping with diameters 10 inches (250 mm) or greater with no breaks. Treating leak events as demands, the conditional break probability can be estimated using a Jefferys noninformative prior in a Bayesian update calculation.

Using 36 throughwall crack events found in pipes 10 inches (250 mm) or greater found in the total U.S. BWR operating experience (including Dresden 1 and LaCrosse events):

$P_{R/TW}$ in pipe 10 inches (250 mm) or greater = 0.5/37 = 0.014

This value is approximately the same as the $P_{R/TW}$ of 0.01.

J-5.1.3 Comparison to NUREG/CR-4792

Conditional probabilities of a pipe break given a leak calculated from results of PRAISE code analyses reported in a Lawrence Livermore National Laboratory study compare reasonably well with results using the Beliczey and Schulz correlation for 10-inch (250-mm) diameter piping.

The objective of NUREG/CR-4792, Vol. 1 (Holman and Chow 1989) was to estimate the probability of leaks and double-ended guillotine break (DEGB) in the main steam, feedwater, and recirculation piping of an older, representative BWR-4 plant. The probabilistic fracture mechanics model, implemented in the PRAISE computer code, was modified to include an appropriate probabilistic model of the intergranular stress corrosion cracking (IGSCC) phenomenon. The model covers Type 304 stainless steel found in most BWR Mark I recirculation piping and the Type 316NG used as an IGSCC-resistant replacement for Type 304. (Replacement configurations have 40 percent fewer weld joints than the original system — 30 compared to 51 and no bypass lines.) Two sets of results with and without the influence of IGSCC were reported.

A model was developed that included the influence of IGSCC and replacement 316NG stainless steel piping in a fictitious recirculation system configuration that contained the same number of welds and included bypass lines as in original BWR-4 designs. The semiempirical model was based on laboratory and field data combined from several sources. The model assumed worst case residual stress conditions. The evaluation of the relative behavior of different material types did not include inservice inspection in the evaluations (although PRAISE has this capability), nor consider the influence of other IGSCC mitigating actions (e.g., weld overlay, inductive heat stress improvement, etc.) on the estimated failure probabilities.

The results (Holman and Chow 1989, Section 4.3) for 316NG stainless steel BWR piping showed that the 12-inch (305-mm) riser piping in the recirculation system dominated the probability of system failure (Holman and Chow 1989, Figure 4.11). The corresponding leak and break probabilities for the entire recirculation system (Holman and Chow 1989, Figure 4.9) are nominally zero during the first 10 years of operation. The cumulative end-of-life leak probability is about 5E-1 per loop after another 30 years of operation, or about 2E-2 per loop-year. The system probability of DEGB is zero for the first 30 years. Two breaks were predicted (out of 25,000 Monte Carlo replications) in the riser welds, the first of which occurred at about 30 years; all other welds groups experienced no DEGB events over the 40 years of plant life. The resultant end-of-life system break probability is about 2E-3 per loop. This implies a DEGB probability of 2E-4 per loop-year over the final 10 years of plant operations and zero during the first 30 years, even under worst case applied stresses and no inservice inspection. The report concluded that routine inservice inspection over the plant life could be expected to substantially lower the late-life probability of DEGB through early detection of cracks.

The conditional probability of a break given a leak is approximately 2E-4/2E-2 or 0.01, where the 12-inch (305-mm) riser piping dominated the failure probability. This conservative value compares well to the conditional probability calculated from the Beliczey and Schulz correlation for 10-inch (250-mm) piping:

$P_{R/TW}$ = 2.5/(10 in. x 25.4 in/mm) = 1.0E-2

J-5.1.4 Comparison to NUREG/CR-6004

Conditional probabilities of a pipe break given a leak, calculated from results of PROLBB code analyses and reported by Battelle, compare reasonably well with results using the Beliczey and Schulz correlation for 4-inch (100-mm) diameter bypass line piping.

The objective of NUREG/CR-6004 (Rahman et al. 1995) was to conduct probabilistic pipe fracture evaluations for application to leak-rate-detection requirements. The PROLBB computer code was developed to evaluate the conditional probability of failure of a circumferentially cracked pipe based on exceeding its maximum load-carrying capacity. The model included accurate deterministic models for estimating leak rates, area-of-crack openings, and maximum load-carrying capacity of pipes; it also included a complete statistical characterization of crack morphology parameters, material property variables, crack location, and standard methods of structural reliability theory.

The probabilistic model was applied to 16 nuclear piping systems in a BWR and a PWR. Several pipe sizes ranging in diameter from 4 inches (100 mm) to 32 inches (813 mm), and several pipe materials, including stainless steel, carbon steel, and cast stainless steel and welds, were considered for determining the conditional probability of failure. Two normal operating stress intensities of 50 and 100 percent the ASME Code Service Level A limit for Class 1 piping were used. Various types of cracking mechanisms, such as IGSCC, corrosion fatigue, and thermal fatigue, were also considered. In addition, both simple circumferential throughwall cracked pipes and complex-cracked pipes were analyzed. Crack locations were defined in both a deterministic sense (either base metal or weld metal) and a probabilistic sense (random location).

The results from NUREG/CR-6004 were used to estimate the conditional failure probability in a BWR for the worst case crack found in the operating experience for medium size piping. The event chosen was a throughwall crack in the heat affected zone of a weld joining the 4-inch (100-mm) bypass piping on the 28-inch (711-mm) main coolant recirculation piping that was reported at Dresden Unit 2 in 1974. A leak rate of 1.5 gpm (5.7 L/min) was observed while at power. From Figure 5.15 in the report, the conditional failure probability of a 4-inch (100-mm) bypass line with the Dresden Unit 2 crack has a range of 1E-1 to 5E-8 at 50 and 100 percent of Service Level A operating stresses, respectively. The conditional failure probability at the midpoint between the 50 and 100 percent curves at a 1.5 gpm (5.7 L/min) leak rate is about 1E-4, which is a conservative estimate since the stress intensity on the pipe at full power is much closer to 100 percent. The conditional probability of a break calculated from the Beliczey and Schulz correlation for 4-inch (100-mm) piping is 2.5/(4 in. x 25.4 in/mm) or 2E-2, which is about a factor of 50 higher. Although the result from the Beliczey and Schulz correlation is conservative in comparison to the NUREG/CR-6004 based estimate, this might not remain the case for significantly higher leak rates since the later estimate will increase with increasing leak rate, but the former remains a constant.

The conditional break probability of 0.01 for pipe larger than 10 inches (250 mm) in diameter is about two orders of magnitude greater than the conditional probability of a break given a leak estimated for an 18-inch (460-mm) stainless steel pipe from results of PROLBB code analyses.

The PROLBB code analyses results were used to estimate the conditional failure probability for the worst case crack event found in the operating experience for large diameter piping in BWRs. The event chosen was the throughwall crack in the heat affected zone of a weld joining the 10-inch (250-mm) recirculation inlet nozzle safe end to the thermal sleeve attachment that was reported at Duane Arnold in 1978. A leak rate of 3 gpm (11.4 L/min) was observed while at power. The crack was classified in NUREG/CR-6004 as a complex crack (a long circumferential surface crack that penetrates the pipe thickness for a short length). The conditional failure probability of a 18-inch (460-mm) riser pipe (a 10-inch [250-mm] pipe was not analyzed) with a complex crack in a weld is in the range of 5E-2 to 5E-7 at 50 and 100 percent of Service Level A operating stresses, respectively (from Figure 5.23 in the report). The report showed that a complex crack in a weld at 50 percent of Service Level A is the most restrictive. The conditional failure probability at the midpoint between the 50 and 100 percent curves at a 3 gpm (11.4 L/min) leak rate is about 1E-4, which is judged to be conservative since the stress intensity of the pipe at full power is much closer to 100 percent.

J-5.1.5 Comparison to PNNL Study Results

The conditional probabilities of a pipe break given a leak calculated from results using the Beliczey and Schulz correlation for 6-inch (150-mm) diameter piping compare reasonably well with results of a Pacific Northwest National Laboratory (PNNL) study documented by F. A. Simonen et al. (1998).

In this study, a probabilistic fracture mechanics model was used to simulate fatigue crack growth of fabrication flaws in stainless steel piping. The effects of leak detection thresholds on the calculated probabilities of large leaks and pipe breaks were evaluated for a range of pipe sizes used in PWR primary piping. The pc-PRAISE computer code was used to perform the probabilistic fracture mechanics analyses. Parameters that were analyzed included: two pipe sizes, 6-inch (150-mm) and 29-inch (740-mm) OD; disabling leaks of 30, 300, and 3000 gpm (114, 1140, and 11400 L/min); leak detection from 0.2 to several 1000s gpm (0.8 to several 3785s L/min); stainless steel piping; and primary pressures from 300 to 2235 psig (2 to 15.4 MPa).

The conditional probability of a break for a 6-inch (150-mm) diameter pipe with high internal pressure are about 1E-6 for a small leak (<30 gpm [<114 L/min]) and 1E-3 for a 300 gpm [1140 L/min] disabling leak. The conditional break probability calculated from the Beliczey and Schulz correlation for a 6-inch (150-mm) diameter pipe is 2.5/(6 in. x 25.4 in/mm) or 1.6E-2, which is noticeably more conservative than that produced in the Pacific Northwest National Laboratory work.

J-5.2 IGSCC Improvement Factor

The improvement factor of 20 attributed to IGSCC mitigation efforts can be applied to both medium and large size BWR piping. This improvement factor is based on ERPI sponsored work (Danko 1983) aimed at demonstrating the benefits of IGSCC mitigation strategies. A program was established to test full-size welded pipes of a variety of heats of commercial grade 304 stainless steels. The 4-inch (100-mm) pipes were welded by standard field procedures and subjected to various IGSCC remedies.

The study concluded that IGSCC initiation might take 20 times longer with a sensitization-related remedy in place. With the unmitigated field data identifying IGSCC occurring at approximately 2 years time, this implies cracking might not take place for more than 40 years after improvements are implemented. Therefore, the reduction in BWR LOCA frequencies associated with IGSCC will likely be significantly greater than a factor of 20. Furthermore, the report concluded that a factor of 20 is ultraconservative for larger diameter piping since failure times based on field data are much longer than 2 years.

The sensitization-related remedies credited with an improvement factor of at least 20 are pipe replacement with Types 304 NG (nuclear grade) and 316 NG stainless steels, and solution heat treatment. These IGSCC mitigation actions were endorsed in NRC Generic Letter 88-01 (USNRC 1988a) and NUREG-0313 (Hazelton and Koo 1988).

J-5.2.1 Comparison to the Operating Experience

The improvement factor of 20 compares reasonably well with the reduction in leak frequencies before and after most BWR plants implemented IGSCC mitigating strategies in the mid- to late-1980s. This data-driven improvement factor was calculated from total U.S. BWR operating experience by taking the ratio of throughwall crack/leak frequencies for the time before and after the date midway between the last two IGSCC leak events (November 1986 and November 1990).

The throughwall crack/leak frequencies based on the 48 throughwall crack and leak events caused by IGSCC during 762 calender years of BWR operation are:

1969-1988: 47 events / 458 calendar years = 0.103

1989-1997: 1 event / 304 calendar years = 0.003

The measure of the reduction in throughwall crack/leak frequency caused by IGSCC provides the IGSCC improvement factor:

IGSCC Improvement Factor = 0.103/0.0033 = 33

J-5.2.2 Comparison to NUREG/CR-4792

The IGSCC improvement factor of 20 compares reasonably well with the results of PRAISE code analyses that were performed by Lawrence Livermore National Laboratory and reported in NUREG/CR-

4792, Vol. 1, (Holman and Chou 1989). The Lawrence Livermore National Laboratory study showed that the leak probability of a weld in a recirculation system 12-inch (305-mm) riser pipe made of IGSCC resistant Type 316NG stainless steel improved significantly over the same piping system made of non-resistant Type 304 stainless steel. This improvement factor decreased over the life of the plant, from less than 100 at ten years, to less than five at end of life (see Figure 4.15 in Holman and Chou 1989).

The results also indicate the leak probability of a riser pipe weld is higher than those for all other weldments in the recirculation system, which is consistent with the operating experience that shows one-half of the throughwall crack events were in the riser weld or heat affected zone (HAZ). Furthermore, the leak probability in a 316NG stainless steel riser weldment is nominally zero during the first 15 years of plant operation and increases to about 1.5E-2 (cumulative probability) over the next 25 years (Figure 4.13 in Holman and Chou 1989). As discussed in the comparison of the Beliczey and Schulz correlation, the analysis did not consider inservice inspection. This assumption makes it more likely a crack will grow to throughwall, since in an actual situation the plant would have gone through one or more inservice inspection cycles.

J-5.2.3 Comparison to Harris (1993)

Although the factors associated with water chemistry-related strategies for IGSCC mitigation were not available in early 1980s, work on this issue was done at Lawrence Livermore National Laboratory and reported by Harris and Balkey (1993). In order to estimate the effectiveness of different remedies for BWR piping, the capability to consider changes in conditions at discrete times was incorporated into the PRAISE Code. To demonstrate this capability, the effect of water chemistry changes was evaluated on piping reliability after 20 years of operation with a nominal oxygen content (0.2 ppm). A girth weld in a 4-inch (100-mm) diameter, Type 304 stainless steel line subject to random residual stresses was analyzed. The stresses due to deadweight and thermal expansion constraint were 0.95 and 8.07 ksi, respectively, and a steady internal pressure of 1,330 psi (9.2 MPa) was considered. The PRAISE code calculated the cumulative failure probability (i.e., probability of a throughwall crack resulting in leakage) as a function of time. The analysis considered various levels of oxygen in the coolant during steady state operation, ranging from nominal oxygen level of 0.2 ppm to 0.002 ppm.

The results showed that the effect of oxygen in the coolant varied from minimal to substantial, depending on the change considered. There was an increasing beneficial effect with decreasing oxygen content. When the oxygen content was reduced to 0.002 ppm, the cumulative failure probability stopped increasing. In other words, the PRAISE code results indicate that reducing oxygen level to 0.002 ppm makes BWR piping no longer susceptible to IGSCC. It is very likely that most plants have implemented at least two IGSCC mitigation strategies, including hydrogen water chemistry.

J-6. DATA TABLES

Table J-10. PWRs from worldwide experience used in the PWR MBLOCA and LBLOCA frequency estimates.

Country	Number of Reactors[a]
Belgium	7
Brazil	1
China	2 French reactors
China-Taiwan	2
Finland	2 "westernized" VVERs
France	58
Germany	14 reactors, excluding reactors in former East Germany, which are VVER designs
Italy	1, now shut down
Japan	23
Korea	11
Netherlands	1
Slovenia	1 Westinghouse reactor
South Africa	2 French reactors
Spain	7
Sweden	3
Switzerland	3
United States	79

a. The world list of nuclear power plants, as of December 31, 1997, as given in the March 1998 issue of *Nuclear News*. PWRs were counted from this list for the countries shown. Both operating and shutdown reactors were counted, 217 PWRs in all.

Table J-11. BWR large diameter pipe leak and throughwall crack events from U. S. operating experience through 1997.

No.	Plant	Date	Description	Reference
n/a	LaCrosse	10/69	Feedwater nozzle safe end	USNRC 1975b, p.2-3
1	Nine Mile Point 1	03/70	10-inch (250-mm) spray nozzle	USNRC 1975b, p.2-3
n/a	Dresden 1	06/74	Steam supply line to emergency condenser	USNRC 1975b, p.2-2
2	Dresden 2	01/28/75	10-inch (250-mm) core spray line to safe end - A	USNRC 1975b, p.3-5
			Weeping; weld	NPE VII.C.25
3	Dresden 2	01/28/75	10-inch (250-mm) core spray line to safe end - B	USNRC 1975b, p.3-5
			Weeping; weld	NPE VII.C.25
4	Dresden 2	02/10/75	10-inch (250-mm) core spray line (8 feet [2.4 m] from safe end)	USNRC 1975b, p.E-1
			Weeping; weld HAZ	NPE VII.C.28
5	Duane Arnold	1978	10-inch (250-mm) riser to safe end	USNRC 1979, p.7.1
			3 gpm (11.4 L/min); safe end base metal	NPE V.B.29
6	Nine Mile Point 1	03/23/82	28-inch (711-mm) recirc loop discharge safe end	LER 220/82-009
			Weeping; weld HAZ	USNRC 1984, p.3-1 IN 82-39; NPE V.B.39
7	Monticello	11/02/82	12-inch (305-mm) riser to safe end - C	LER 263/82-013
			Weeping; weld HAZ	NPE V.B.44
8	Monticello	11/02/82	12-inch (305-mm) riser to safe end - E	LER 263/82-013
			Weeping; weld HAZ	NPE V.B.44
9	Monticello	11/02/82	12-inch (305-mm) riser to safe end - F	LER 263/82-013
			Weeping; weld HAZ	NPE V.B.44
10	Monticello	11/02/82	12-inch (305-mm) riser elbow to safe end - G	LER 263/82-013
			Weeping; weld HAZ	NPE V.B.44
11	Hatch 1	11/04/82	22-inch (559-mm) manifold end cap	LER 321/82-089
			Weeping, weld	NPE V.B.43
12	Brunswick 1	01/26/83	12-inch (305-mm) recirc loop discharge, Loop A	LER 325/83-001
			Weeping; weld HAZ	NPE V.B.42, 44
13	Brunswick 1	01/26/83	12-inch (305-mm) recirc loop discharge, Loop B	LER 325/83-001
			Weeping; weld HAZ	NPE V.B.42, 44
14	Monticello	05/05/84	12-inch (305-mm) riser to safe end	LER 263/84-011
			0 gpm (0 L/min); weld HAZ	NPE V.B.56
15	Quad Cities 1	04/14/84	12-inch (305-mm) riser pipe to elbow - J	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
16	Quad Cities 1	04/14/84	12-inch (305-mm) riser elbow to pipe - K	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
17	Quad Cities 1	04/14/84	12-inch (305-mm) riser pipe to elbow - K	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
18	Quad Cities 1	04/14/84	12-inch (305-mm) riser sweepolet to pipe - J	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57

Table J-11. (continued).

No.	Plant	Date	Description	Reference
19	Quad Cities 1	04/14/84	12-inch (305-mm) riser pipe to elbow - H	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
20	Quad Cities 1	04/14/84	12-inch (305-mm) riser elbow to pipe - H	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
21	Quad Cities 1	04/14/84	12-inch (305-mm) riser elbow to pipe - G	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
22	Quad Cities 1	04/14/84	12-inch (305-mm) riser elbow to pipe - J	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
23	Quad Cities 1	04/14/84	12-inch (305-mm) riser pipe to elbow - E	LER 254/84-005
			Weeping; weld HAZ	NPE V.B.57
24	Browns Ferry 2	02/21/85	28-inch (711-mm) header to 12-inch (305-mm) riser junction.	LER 260/85-001
			Weep, weld HAZ	NPE V.B.68
25	Duane Arnold	03/10/85	10-inch (250-mm) riser pipe	LER 331/85-010
			Weeping (after IHSI)	NPE V.B.80
26	Duane Arnold	03/10/85	16-inch (406-mm) RHR to recirc loop suction	LER 331/85-010
			Weeping (after IHSI)	NPE V.B.80
27	Quad Cities 2	04/01/85	12-inch (305-mm) riser sweephole to pipe	LER 265/85-008
			Weep, weld HAZ	NPE V.B.68
28	Brunswick 1	07/01/85	12-inch (305-mm) riser pipe - A	LER 325/85-026
			Weeping (after IHSI); weld HAZ	NPE V.B.69
29	Brunswick 1	07/01/85	12-inch (305-mm) riser pipe - D	LER 325/85-026
			Weeping (after IHSI); weld HAZ	NPE V.B.69
30	Brunswick 1	07/01/85	12-inch (305-mm) riser pipe - H	LER 325/85-026
			Weeping (after IHSI); weld HAZ	NPE V.B.69
31	Brunswick 1	07/01/85	12-inch (305-mm) riser pipe - J	LER 325/85-026
			Weeping (after IHSI); weld HAZ	NPE V.B.69
32	Brunswick 1	07/01/85	12-inch (305-mm) riser pipe - K	LER 325/85-026
			Weeping (after IHSI); weld HAZ	NPE V.B.69
33	Quad Cities 2	11/05/86	12-inch (305-mm) riser elbow to pipe	LER 265/86-017
			Weeping, weld	NPE V.B.77
34	Dresden 2	11/24/90	16-inch (406mm) RHR to recirc loop suction	LER 237/90-014
			Weeping, weld	

n/a – Event (and corresponding operating experience) not used in quantification because of the atypical design of the plant

HAZ – heat affected zone

NPE - Nuclear Power Experience (Stoller)

Table J-12. PWR large and medium sized pipe leak or throughwall crack events based on worldwide experience.

Plant	Event Date	NSSS Vendor	Piping System Pipe Dia (in.) Nominal (OD/ID)	Location	Size	Leak Rate (gpm)	Reference
Genkai Unit 1	06/06/88	PWR (Japan)	Residual Heat Removal to RCS hot leg pipe 8.0(8.6/7.0)	Heat affected zone of elbow-to-pipe weld	Crack extended 97 mm circumferentially at the inside surface, 1.5 mm long at the outside surface	0.2	USNRC 1989; Shirahama 1998
Farley Unit 2	12/09/87	West. 3-Loop (U.S.)	Safety Injection to RCS cold leg pipe 6.0(6.6/5.2)	Heat affected zone of elbow-to-pipe weld	Crack extended 120 degrees circumferentially at the inside surface, 22 mm long at the outside surface	0.7	USNRC 1988b; LER364/87-10; Strauch et al. 1990
Tihange Unit 1	06/18/88	PWR 3-Loop (Belgium)	Safety Injection to RCS hot leg 6.0(6.6/5.2)	Elbow base metal	Crack extended 89 mm circumferentially at the inside surface, 41 mm long at the outside surface	5.8	USNRC 1988c
Dampierre Unit 2	09/92	PWR 3-Loop (France)	Safety Injection to RCS hot leg 6.0(6.6/5.2)	Check valve-to-pipe weld and base metal of straight portion of pipe	Crack extended 110 mm circumferentially at the inside surface, 22 mm long at the outside surface	2.6	Jungelaus et al. 1998
Dampierre Unit 1	12/14/96	PWR 3-Loop (France)	Safety Injection to RCS hot leg 6.0(6.6/5.2)	Base metal of straight portion of pipe	Crack extended 80 mm circumferentially at the inside surface, 22 mm long at the outside surface	0.7	USNRC 1997; Jungelaus et al. 1998
Loviisa Unit 2	05/94	Modified VVER (Finland)	Hot leg drain line 2.4(2.5/2.1)	Weld between a T-joint piece and a reducer	65-degree circumferential crack	0.1	Hytonen 1998

Table J-13. BWR medium-size primary system leak events.

No.	Plant	Date	Description	Reference
n/a	Dresden 1	12/65	6-inch (150-mm) recirc. valve bypass line	USNRC 1975b, p.2-1
n/a	Dresden 1	02/66	4-inch (100-mm) suction line to recirc. pump	USNRC 1975b, p.2-1
n/a	Dresden 1	04/67	6-inch (150-mm) recirc. pump suction line	USNRC 1975b, p.2-2
n/a	Dresden 1	12/69	4- to 2-inch (100- to 50-mm) riser in vessel head vent	USNRC 1975b, p.2-1
n/a	Dresden 1	11/70	4-inch (100-mm) pipe connected to the recirc. loop	USNRC 1975b, p.2-1
n/a	Dresden 1	02/71	4-inch (100-mm) demineralizer supply line to recirc. loop	USNRC 1975b, p.2-1
n/a	Dresden 1	01/72	4- to 2-inch (100- to 50-mm) riser in vessel head vent	USNRC 1975b, p.2-1
1	Dresden 2	09/13/74	4-inch (100-mm) recirc. valve bypass line 1.5 gpm (5.7 L/min); weld HAZ	USNRC 1975b, p.3-1 NPE V.B.10
2	Millstone 1	09/18/74	4-inch (100-mm) recirc. valve bypass line 0 gpm (0 L/min); weld	USNRC 1975b, p.3-1 NPE V.B.10
3	Dresden 2	12/13/74	4-inch (100-mm) recirc. valve bypass line Weeping; weld	USNRC 1975b, p.3-1 NPE V.B.13
4	Quad Cities 2	12/23/74	4-inch (100-mm) recirc. valve bypass line, Loop A Weeping; weld	NPE V.B.14
5	Quad Cities 2	12/23/74	4-inch (100-mm) recirc. valve bypass line, Loop B 0 gpm (0 L/min); weld	NPE V.B.14
6	Hatch 1	12/74	6-inch (150-mm) vessel head spray line Leakage during startup testing Crack caused by water hammer	NPE VII.D.123
7	Millstone 1	11/76	6-inch (150-mm) vessel head spray to penetration 0 gpm (0 L/min); spool piece weld	NPE VII.D.105
8	Quad Cities 1	01/10/75	4-inch (100-mm) recirc. valve bypass line, Loop A Weeping; weld	USNRC 1975b, p.E-1 NPE V.B.14
9	Quad Cities 1	01/10/75	4-inch (100-mm) recirc. valve bypass line, Loop B 0 gpm (0 L/min); weld	USNRC 1975b, p.E-1 NPE V.B.14
10	Duane Arnold	03/78	4-inch (100-mm) RWCU system, inside isolation valve weld HAZ	NPE VIII.A.48
11	Vermont Yankee	10/80	4-inch (100-mm) RWCU system, inside isolation valve Weeping; weld HAZ	NPE VIII.A.71
12	Browns Ferry 2	05/17/84	4-inch (100-mm) jet pump instrument nozzle 0 gpm (0 L/min); weld	LER 296/84-006 NPE III.44
13	Peach Bottom 2	06/07/84	4-inch (100-mm) jet pump instrument nozzle	LER 277/84-010

Table J-13. *(continued).*

No.	Plant	Date	Description	Reference
			reducer to safe end	NPE III.42
			Weeping; weld HAZ	
14	Peach Bottom 3	06/10/84	4-inch (100-mm) jet pump instrument nozzle	LER 277/84-008
			reducer to safe end	NPE III.42
			Weeping; weld HAZ	
15	Brunswick 1	11/27/84	4-inch (100-mm) jet pump instrument nozzle	LER 325/84-017

n/a—Event (and corresponding operating experience) not used in quantification because of the atypical design of the plant.

HAZ—heat affected zone

NPE—Nuclear Power Experience (Stoller)

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Appendix K

Plant Name and Docket Number Tables

Appendix K

Plant Name and Docket Number Tables

Table K-1. List of plants by docket number.

Docket	Type	Name	Vendor	Docket	Type	Name	Vendor
029	PWR	Yankee-Rowe	WE	301	PWR	Point Beach 2	WE
155	BWR	Big Rock Point	GE	302	PWR	Crystal River 3	BW
206	PWR	San Onofre 1	WE	304	PWR	Zion 2	WE
213	PWR	Haddam Neck	WE	305	PWR	Kewaunee	WE
219	BWR	Oyster Creek	GE	306	PWR	Prairie Island 2	WE
220	BWR	Nine Mile Pt 1	GE	309	PWR	Maine Yankee	CE
237	BWR	Dresden 2	GE	311	PWR	Salem 2	WE
244	PWR	Ginna	WE	312	PWR	Rancho Seco	BW
245	BWR	Millstone 1	GE	313	PWR	Arkansas 1	BW
247	PWR	Indian Point 2	WE	315	PWR	Cook 1	WE
249	BWR	Dresden 3	GE	316	PWR	Cook 2	WE
250	PWR	Turkey Point 3	WE	317	PWR	Calvert Cliffs 1	CE
251	PWR	Turkey Point 4	WE	318	PWR	Calvert Cliffs 2	CE
254	BWR	Quad Cities 1	GE	321	BWR	Hatch 1	GE
255	PWR	Palisades	CE	323	PWR	Diablo Canyon 2	WE
260	BWR	Browns Ferry 2	GE	324	BWR	Brunswick 2	GE
261	PWR	Robinson 2	WE	325	BWR	Brunswick 1	GE
263	BWR	Monticello	GE	327	PWR	Sequoyah 1	WE
265	BWR	Quad Cities 2	GE	328	PWR	Sequoyah 2	WE
266	PWR	Point Beach 1	WE	331	BWR	Duane Arnold	GE
269	PWR	Oconee 1	BW	333	BWR	Fitzpatrick	GE
270	PWR	Oconee 2	BW	334	PWR	Beaver Valley 1	WE
271	BWR	Vermont Yankee	GE	335	PWR	St Lucie 1	CE
272	PWR	Salem 1	WE	336	PWR	Millstone 2	CE
275	PWR	Diablo Canyon 1	WE	338	PWR	North Anna 1	WE
277	BWR	Peach Bottom 2	GE	339	PWR	North Anna 2	WE
278	BWR	Peach Bottom 3	GE	341	BWR	Fermi 2	GE
280	PWR	Surry 1	WE	344	PWR	Trojan	WE
281	PWR	Surry 2	WE	346	PWR	Davis-Besse	BW
282	PWR	Prairie Island 1	WE	348	PWR	Farley 1	WE
285	PWR	Ft. Calhoun	CE	352	BWR	Limerick 1	GE
286	PWR	Indian Point 3	WE	353	BWR	Limerick 2	GE
287	PWR	Oconee 3	BW	354	BWR	Hope Creek	GE
289	PWR	Three Mile Isl 1	BW	361	PWR	San Onofre 2	CE
293	BWR	Pilgrim	GE	362	PWR	San Onofre 3	CE
295	PWR	Zion 1	WE	364	PWR	Farley 2	WE
296	BWR	Browns Ferry 3	GE	366	BWR	Hatch 2	GE
298	BWR	Cooper	GE	368	PWR	Arkansas 2	CE

Table K-1. (continued).

Docket	Type	Name	Vendor	Docket	Type	Name	Vendor
369	PWR	McGuire 1	WE	425	PWR	Vogtle 2	WE
370	PWR	McGuire 2	WE	440	BWR	Perry	GE
373	BWR	La Salle 1	GE	443	PWR	Seabrook	WE
374	BWR	La Salle 2	GE	445	PWR	Comanche Peak 1	WE
382	PWR	Waterford 3	CE	446	PWR	Comanche Peak 2	WE
387	BWR	Susquehanna 1	GE	454	PWR	Byron 1	WE
388	BWR	Susquehanna 2	GE	455	PWR	Byron 2	WE
389	PWR	St. Lucie 2	CE	456	PWR	Braidwood 1	WE
395	PWR	Summer	WE	457	PWR	Braidwood 2	WE
397	BWR	Wash. Nuclear 2	GE	458	BWR	River Bend	GE
400	PWR	Harris	WE	461	BWR	Clinton 1	GE
410	BWR	Nine Mile Pt 2	GE	482	PWR	Wolf Creek	WE
412	PWR	Beaver Valley 2	WE	483	PWR	Callaway	WE
413	PWR	Catawba 1	WE	498	PWR	South Texas 1	WE
414	PWR	Catawba 2	WE	499	PWR	South Texas 2	WE
416	BWR	Grand Gulf	GE	528	PWR	Palo Verde 1	CE
423	PWR	Millstone 3	WE	529	PWR	Palo Verde 2	CE
424	PWR	Vogtle 1	WE	530	PWR	Palo Verde 3	CE

Table K-2. List of plants by name.

Docket	Type	Name	Vendor	Docket	Type	Name	Vendor
313	PWR	Arkansas 1	BW	315	PWR	Cook 1	WE
368	PWR	Arkansas 2	CE	316	PWR	Cook 2	WE
334	PWR	Beaver Valley 1	WE	298	BWR	Cooper	GE
412	PWR	Beaver Valley 2	WE	302	PWR	Crystal River 3	BW
155	BWR	Big Rock Point	GE	346	PWR	Davis-Besse	BW
456	PWR	Braidwood 1	WE	275	PWR	Diablo Canyon 1	WE
457	PWR	Braidwood 2	WE	323	PWR	Diablo Canyon 2	WE
260	BWR	Browns Ferry 2	GE	237	BWR	Dresden 2	GE
296	BWR	Browns Ferry 3	GE	249	BWR	Dresden 3	GE
325	BWR	Brunswick 1	GE	331	BWR	Duane Arnold	GE
324	BWR	Brunswick 2	GE	348	PWR	Farley 1	WE
454	PWR	Byron 1	WE	364	PWR	Farley 2	WE
455	PWR	Byron 2	WE	341	BWR	Fermi 2	GE
483	PWR	Callaway	WE	333	BWR	Fitzpatrick	GE
317	PWR	Calvert Cliffs 1	CE	285	PWR	Ft. Calhoun	CE
318	PWR	Calvert Cliffs 2	CE	244	PWR	Ginna	WE
413	PWR	Catawba 1	WE	416	BWR	Grand Gulf	GE
414	PWR	Catawba 2	WE	213	PWR	Haddam Neck	WE
461	BWR	Clinton 1	GE	400	PWR	Harris	WE
445	PWR	Comanche Peak 1	WE	321	BWR	Hatch 1	GE
446	PWR	Comanche Peak 2	WE	366	BWR	Hatch 2	GE

Table K-2. (continued).

Docket	Type	Name	Vendor	Docket	Type	Name	Vendor
354	BWR	Hope Creek	GE	254	BWR	Quad Cities 1	GE
247	PWR	Indian Point 2	WE	265	BWR	Quad Cities 2	GE
286	PWR	Indian Point 3	WE	312	PWR	Rancho Seco	BW
305	PWR	Kewaunee	WE	458	BWR	River Bend	GE
373	BWR	La Salle 1	GE	261	PWR	Robinson 2	WE
374	BWR	La Salle 2	GE	272	PWR	Salem 1	WE
352	BWR	Limerick 1	GE	311	PWR	Salem 2	WE
353	BWR	Limerick 2	GE	206	PWR	San Onofre 1	WE
309	PWR	Maine Yankee	CE	361	PWR	San Onofre 2	CE
369	PWR	McGuire 1	WE	362	PWR	San Onofre 3	CE
370	PWR	McGuire 2	WE	443	PWR	Seabrook	WE
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423	PWR	Millstone 3	WE	498	PWR	South Texas 1	WE
263	BWR	Monticello	GE	499	PWR	South Texas 2	WE
220	BWR	Nine Mile Pt 1	GE	335	PWR	St. Lucie 1	CE
410	BWR	Nine Mile Pt 2	GE	389	PWR	St. Lucie 2	CE
338	PWR	North Anna 1	WE	395	PWR	Summer	WE
339	PWR	North Anna 2	WE	280	PWR	Surry 1	WE
269	PWR	Oconee 1	BW	281	PWR	Surry 2	WE
270	PWR	Oconee 2	BW	387	BWR	Susquehanna 1	GE
287	PWR	Oconee 3	BW	388	BWR	Susquehanna 2	GE
219	BWR	Oyster Creek	GE	289	PWR	Three Mile Isl 1	BW
255	PWR	Palisades	CE	344	PWR	Trojan	WE
528	PWR	Palo Verde 1	CE	250	PWR	Turkey Point 3	WE
529	PWR	Palo Verde 2	CE	251	PWR	Turkey Point 4	WE
530	PWR	Palo Verde 3	CE	271	BWR	Vermont Yankee	GE
277	BWR	Peach Bottom 2	GE	424	PWR	Vogtle 1	WE
278	BWR	Peach Bottom 3	GE	425	PWR	Vogtle 2	WE
440	BWR	Perry	GE	382	PWR	Waterford 3	CE
293	BWR	Pilgrim	GE	482	PWR	Wolf Creek	WE
266	PWR	Point Beach 1	WE	397	BWR	Wash. Nuclear 2	GE
301	PWR	Point Beach 2	WE	029	PWR	Yankee-Rowe	WE
282	PWR	Prairie Island 1	WE	295	PWR	Zion 1	WE
306	PWR	Prairie Island 2	WE	304	PWR	Zion 2	WE

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Lockheed Martin Idaho Technologies Co.
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Idaho Falls, ID 83415-3129

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11 ABSTRACT (200 words or less)

This report was produced at the Idaho National Engineering and Environmental Laboratory for the U.S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data. Data for all unexpected reactor trips during power operations at commercial nuclear power plants from 1987 through 1995 were reviewed. Each event was reviewed and categorized according to the initial event and, additionally, was marked if certain other risk-significant events occurred, regardless of their position in the event sequence. The collected data were analyzed for time dependence, reactor-type dependence, and between-plant variance. Dependencies and trends are reported, along with the raw counts and the best estimate for 1995 initiating event frequencies. For some initiators whose frequencies are low enough that no events would be expected in the 1987–1995 period, additional operating experience and information from other sources were used to estimate frequencies. These included operating experience from U.S. and foreign reactors, as well as evaluation of engineering aspects of certain rare events, such as loss-of-coolant accidents (LOCAs). Results of engineering analyses of the operating experience are compared with probabilistic risk assessment/individual plant examinations (PRA/IPEs) and other regulatory issues.

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