Precursors to Potential Severe Core Damage Accidents: Fiscal Year 1999

A Status Report: October 1, 1998–September 30, 1999

Manuscript Completed: September 2001

Prepared by
D.G. Marksberry

Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This report describes the seven operational events and conditions in Fiscal Year 1999 that were precursors to potential severe core damage accidents. All these events had conditional probabilities of subsequent severe core damage greater than or equal to 1.0×10^{-6}. These events were identified by first computer-screening licensee event reports from licensees of commercial nuclear power plants to identify candidate precursors. Candidate precursors were selected and evaluated in a process similar to that used in previous assessments. Other events designated by the Nuclear Regulatory Commission (NRC), such as inspection reports, also underwent a similar evaluation. Selected events underwent an engineering evaluation to identify, analyze, and document the precursors. Preliminary precursors were submitted for peer review by licensee and NRC staff to ensure that the plant design and its response to the precursors were correctly characterized. This study is a continuation of earlier work that evaluated 1969–1998 events. The report discusses the general rationale for this study, the selection and documentation of events as precursors, and the estimation of conditional probabilities of subsequent severe core damage for the events.

CONTENTS

Appendices

Figures

Tables

ACKNOWLEDGMENTS

The Accident Sequence Precursor Program and this report benefitted from the following individuals for their technical review and analysis of operational events:

Erul Chelliah
Marc Harper
Pat O'Reilly
Sunil Weerakkody

from the Operating Experience Risk Analysis Branch (OERAB) in the Office of Nuclear Regulatory Research (RES);

Angela Andrews
Randy Belles
Joe Cletcher
Don Copinger
Mike Muhlheim
Mike Poore
Debbie Queener

from the Oak Ridge National Laboratory; and

Joe Minerick

from Science Applications International Corporation, Oak Ridge TN.

Contribution from the following NRC staff had a significant contribution to individual analysis of precursor events for the 1999 fiscal year:

Bruce Bartlett (Region III)
Sonia Burgess (Region III)
Tsun-Yung Chang (RES)
Joseph Maynen (Region III)
John Ridgely (RES)
Walt Rogers (Region II)
Mel Shannon (Region II)
Jim Trapp (Region I)

Comments from peer reviews provided by licensees and NRC staff contributed substantially to accuracy of the results.

Trending analysis of precursor data was provided by Dale Rasmuson from RES/OERAB.

Technical reviews from the following RES/OERAB staff contributed substantially to the final report:

Patrick Baranowsky
Steve Mays
Pat O'Reilly

ABBREVIATIONS

AFW	auxiliary/emergency feedwater system
ASP	accident sequence precursor
BWR	boiling-water reactor
CCDP	conditional core damage probability
CCW	component cooling water system
CDP	core damage probability
ECCS	emergency core cooling system
EOP	emergency operating procedure
EP	emergency power system
EQ	earthquake (initiator)
FSAR	final safety analysis report
FY	fiscal year
IPE	individual plant examination
LER	licensee event report
LOFW	loss of feedwater system
LOOP	loss of offsite power
NRC	U.S. Nuclear Regulatory Commission
PWR	pressurized-water reactor
PRA	probabilistic risk assessment
RES	NRC Office of Nuclear Regulatory Research
SCSS	Sequence Coding and Search System
SGTR	steam generator tube rupture
SLB	steam line break
SLC	standby liquid control system
SLOCA	small loss-of-coolant accident
SPAR	Standardized Plant Analysis Risk (model)
SW	service water system
TRIP	reactor trip (initiator)

1. INTRODUCTION

The Accident Sequence Precursor (ASP) Program involves the systematic review and evaluation of operational events or conditions that have occurred at licensed U.S. nuclear power plants. The ASP Program identifies and categorizes precursors to potential severe core damage accident sequences. This report details the review and evaluation of operational events and conditions that occurred in Fiscal Year (FY) 1999 and were reported in licensee event reports and/or Nuclear Regulatory Commission (NRC) inspection reports.

1.1 Background

The ASP Program owes its genesis to the Risk Assessment Review Group (Ref. 1), which concluded that "unidentified event sequences significant to risk might contribute... a small increment...[to the overall risk]." The report continues, "It is important, in our view, that potentially significant [accident] sequences, and precursors, as they occur, be subjected to the kind of analysis contained in WASH-1400" [Ref. 2]. Evaluations done for the 1969–1981 period were the first efforts in this type of analysis.

The present report is a continuation of the work published in NUREG/CR-2497, *Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report*, as well as in later status reports. Since its inception, the ASP Program has published 18 reports documenting the results of its review of operational experience for precursors covering the years 1969–1998. These reports have been issued yearly since 1986.

1.2 Program Objectives

The primary objective of the ASP Program is to systematically evaluate U.S. nuclear plant operating experience to identify, document, and rank operating events most likely to lead to inadequate core cooling and core damage (precursors).

In addition, the secondary objectives of the ASP Program are

- To categorize the precursors by their plant-specific and generic implications,
- To provide a measure for trending nuclear plant core damage risk, and
- To provide a partial check on probabilistic risk assessment-predicted dominant core damage scenarios.

The program is also used to monitor the agency's performance against the following Strategic Plan goals for maintaining safety (Ref. 3):

> No more than one event per year which is a significant precursor (i.e., conditional core damage probability or importance $\geq 1 \times 10^{-3}$) of a nuclear reactor accident.
>
> No statistically significant adverse industry trends in safety performance.[1]

1.3 Precursor Definitions and Threshold

Definition of a precursor. An *accident sequence precursor* is an operational event that is an important element of a postulated core-damage accident sequence. An *operational event* can be an actual initiating event (e.g., loss of offsite power) and/or a condition found during a test, inspection, or engineering evaluation involving a reduction in safety system reliability or function for a specific duration.

Accident sequences of interest to the ASP Program are those that would have resulted in inadequate core cooling and severe core damage if additional failures had occurred. Precursors are initiating events or conditions that, when coupled with one or more postulated events, could result in a plant condition involving inadequate core cooling. The ASP Program uses nominal failure probabilities and initiating event frequencies for estimating the conditional probability of the postulated event portion of the analysis.

---

[1] The industry trend used in this performance measure was provided by the ASP Program for FY 1999 (see Section 3.0). Future trending anlaysis will be provided by the Office of Nuclear Reactor Regulation.

The ASP program currently performs detailed analyses for operational events and conditions affecting at-power and shutdown conditions.

At-power precursor. An at-power precursor is an operational event or condition that meets one of the following criteria:

- The total failure of a system required to mitigate effects of a core damage initiator.
- The degradation of two or more trains required to mitigate effects of a core damage initiator.
- A core damage initiator such as a loss of offsite power (LOOP) or small-break loss-of-coolant accident (SLOCA).
- A reactor trip or loss-of-feedwater with a degraded safety system.

Shutdown Precursor. A shutdown precursor is an operational event or condition that meets all of the following criteria:

- The event involved a core damage initiator such as a loss of shutdown cooling, loss of reactor vessel inventory, loss of offsite power, unavailability of emergency power, or a loss-of-coolant accident, and
- the initiator could only have occurred with the plant in a shutdown condition.

CCDP vs. Importance. The figure of merit for ASP analyses is the conditional core damage probability (CCDP) for initiating events and increase in core damage probability (ΔCDP) or *importance* for conditions. The *importance* is the measure of the incremental increase between the CCDP for the period in which the condition existed and the nominal CDP for the same period.

For example, Figure 1.1 shows the hypothetical risk results from two conditional assessments for different events at two plant sites. In one case, the absolute CCDP probability is higher than the second case. However, when considering the CCDP relative to the baseline CDP for both cases, the importances (i.e., the risk significance) are equal. The nominal CDP is usually very much smaller than the CCDP for significant events.

Threshold. An initiating event with a CCDP or a condition with an importance greater than or equal to 1.0 x 10^{-6} is considered a precursor in the ASP Program.

1.4 Approach

The process for identifying, analyzing, and documenting precursors is summarized in Section 2 and described in detail in Appendix A. Preliminary precursor analyses are transmitted for peer review by the responsible licensees and NRC staff. All comments are evaluated, and the analyses are revised as appropriate.

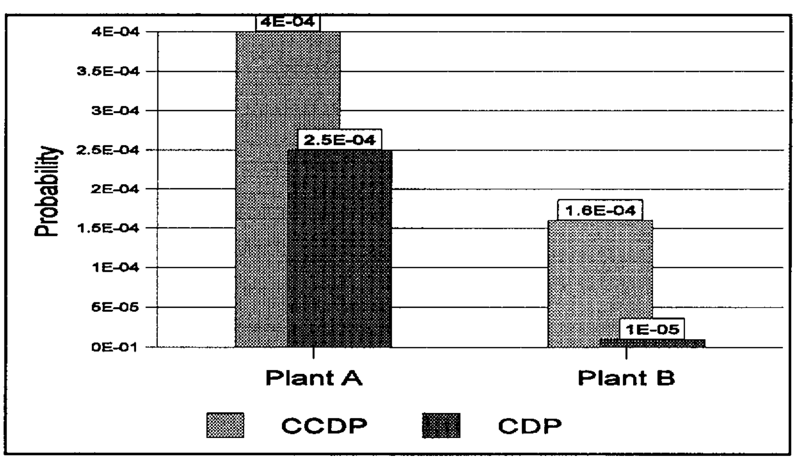

Figure 1.1. Example showing the importance measure relative to the CCDP and CDP for two cases. The calculated importance for both plants are 1.5 x10^{-4} (= CCDP - CDP).

2. EVALUATION PROCESS

The process for selecting, analyzing, and reviewing precursors is summarized below. Figure 2.1 illustrates this process. Details of various elements of the evaluation process are provided in Appendix A.

2.1 Selection of Potential Precursors for Analysis

Data sources. In the evaluation of Fiscal Year 1999 events, two primary sources were used to identify potential precursors:

- Licensee event reports (LERs) in the Sequence Coding and Search System (SCSS) database.
- Inspection reports from Nuclear Regulatory Commission (NRC) special inspections and augmented inspection team inspections.

Results from a special Accident Sequence Program (ASP) Program assessment of 141 issues identified at the Donald C. Cook Nuclear Power Plant, Units 1 and 2 during the FY 1999 period (Ref. 4) were included in this report.

Initial review process. The ASP Program employs the following three-phased process in its screening, review, and analysis of operational experience for precursors:

- *Screening* of LERs using a computerized search of the SCSS data base against a set of screening criteria to identify those which should be reviewed as candidate precursors. Findings from inspections and special requests from NRC staff are also included as candidates.
- *Engineering review* of the available documentation of the operational event or condition identified in the previous phase to determine whether it qualifies for detailed analysis as a potential precursor. This review may include simple bounding analysis.
- *Detailed analysis* of the event or condition, including quantification using ASP probabilistic risk assessment models.

2.2 Detailed Analysis of Potential Precursors

Quantification of the significance of an operational event or condition involves the determination of a conditional probability of subsequent severe core damage given the failures observed during an operational event or condition. This conditional probability is estimated by mapping observed failures or conditions onto the ASP accident sequence models, and calculating a conditional core damage probability. These plant-specific models, called Standardized Plant Analysis Risk (SPAR) models, contain event trees and linked fault trees. The models depict potential paths to severe core damage.

The effect of a precursor on event tree branches is assessed by evaluating the operational event or condition specifics against system design information and modeling assumptions. The evaluation of the event or condition includes all actual or potential concurrent failures, degradations, or outages of safety- and non-safety-related mitigation systems. The evaluation also includes estimates of the likelihood of equipment failures, human errors, and recovery actions. This information is used to modify the SPAR model. Random failures are assumed for other branches of the event tree models not related to the specific operational event or condition being analyzed. The quantification of the revised model results in a conditional probability of core damage given the operational event or condition.

The ASP Program currently performs detailed analyses for operational events and conditions affecting at-power and shutdown conditions.

2.3 Sources of Input Information for Detailed Analysis

Various sources of plant- and event-specific information are used in performing the detailed analysis. Information describing the event or condition in the LER or inspection report can be supplemented with additional information obtained from inspectors knowledgeable of the specific event or condition, NRC staff experts in relevant technical areas, and the licensee through follow-up event assessment.

The adaption of the SPAR model to the event or condition may need design-related information from plant-specific sources, such as the Updated Final Safety Analysis Report, Technical Specifications, individual plant examination for internal and external events, and plant operating procedures. In addition, the component failure probabilities and initiating event frequencies may be updated in the SPAR model using results from system reliability and initiating event studies, based on recent operating experience.

Lastly, formal comments from the licencee's review of the preliminary analysis (as described below) will be used to complete the final analysis.

2.4 Potentially Significant Events Considered Impractical to Analyze

In some cases, events are impractical to analyze because of the inability to reasonably model a condition within a probabilistic risk assessment (PRA) framework, considering the level of detail typically available in PRA models and the resources available to the ASP Program. These events usually involve component degradations in which the extent of the degradation cannot be readily determined or the impact of the degradation on plant response cannot be readily ascertained.

2.5 Containment-Related Events

Events involving loss of containment functions—containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control—are classified as "containment-related" events. Containment-related events are not currently considered precursor events under the ASP Program. The ASP Program is currently developing containment-related event models. The potential for increased risk to the public due to containment-related events justifies their inclusion in the report. Only a qualitative discussion is provided for containment-related events.

2.6 "Interesting" Events

Events or conditions that provide insight into unusual failure modes with the potential to compromise continued core cooling but are not precursors (i.e., conditional core damage probability or importance < 1×10^{-6}) are documented as "interesting" events. Only a qualitative discussion is provided for "interesting" events.

2.7 Independent Review Process

Preliminary precursor analyses undergo two independent reviews. Figure 2.2 illustrates this process. In the first review, the preliminary analysis is reviewed in-house by a second analyst. After completion of the first review and any corresponding revision, the final preliminary analysis is transmitted to the pertinent nuclear plant licensee and to NRC staff for peer review. The licensee is requested to review and comment on the technical adequacy of the analyses, including the depiction of their plant equipment and equipment capabilities. The review guidance provided to the licensee is provided in Appendix B. Review comments are evaluated for applicability to the ASP analysis.

After the preliminary analysis is revised based on licensee and NRC staff comments, the modified analysis is reviewed by the second analyst for final review and revised again, if necessary. Technical audits by branch management are performed for preliminary and final analyses prior to issuance. Technical differences are discussed with the reviewer. The response to comments and differences are documented in the final ASP analysis report.

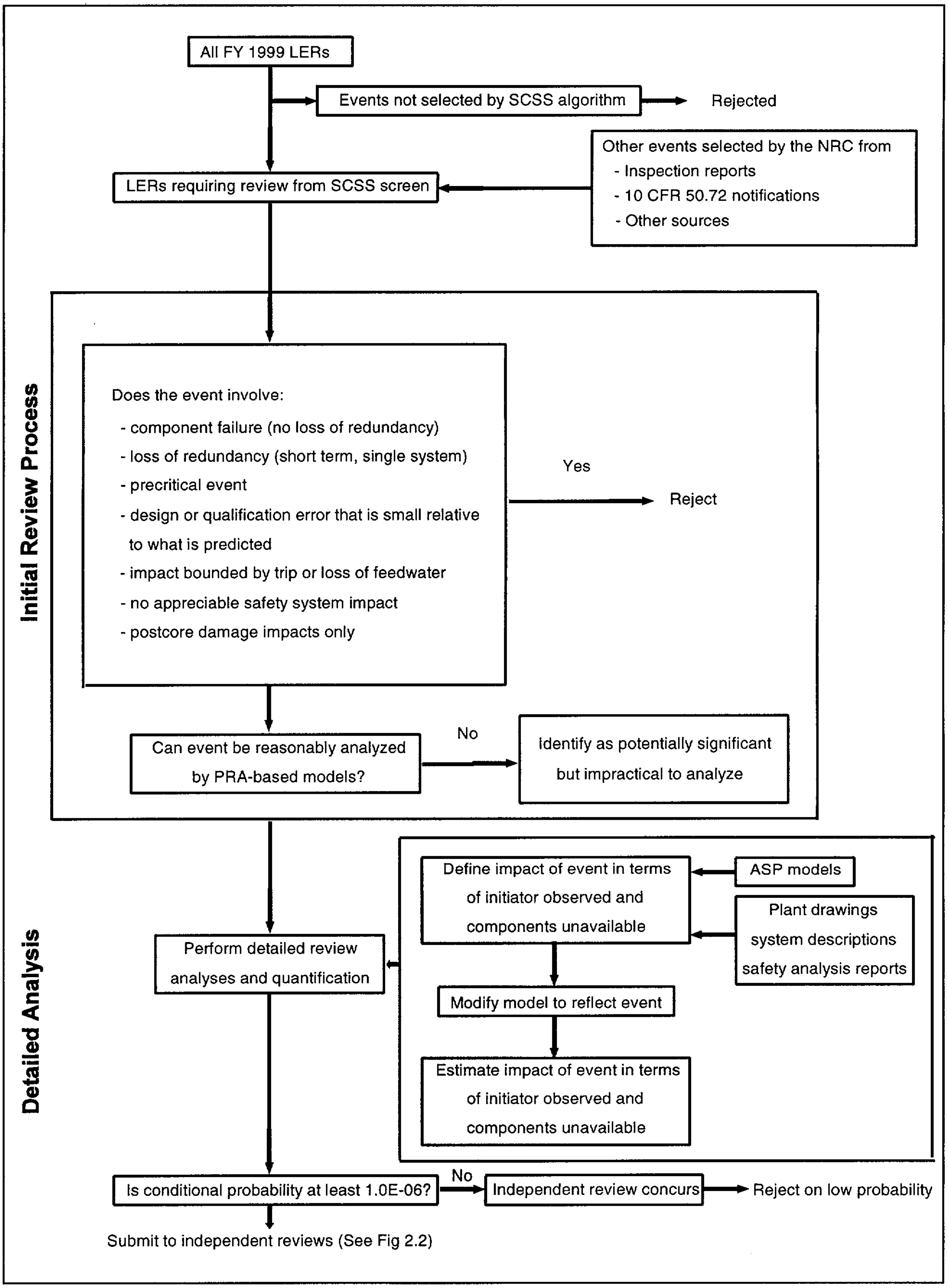

Figure 2.1. Accident Sequence Precursor Program analysis process.

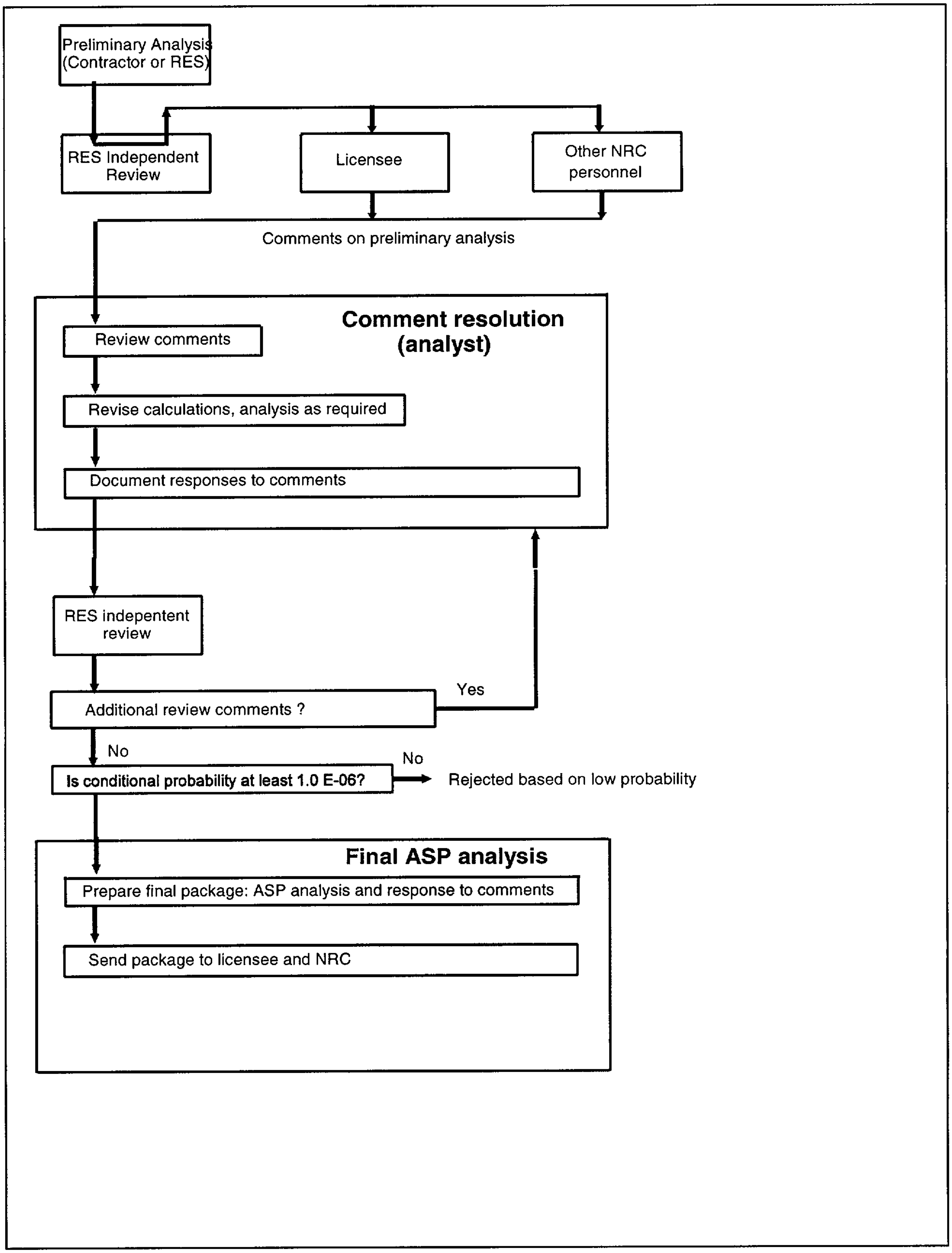

Figure 2.2. Accident Sequence Precursor Program review process.

3. RESULTS, TRENDS, AND INSIGHTS

This section describes the Fiscal Year (FY) 1999 precursors, the analysis of historical Accident Sequence Precursor (ASP) trends, and the evaluation of insights.

3.1 FY 1999 ASP Event Analysis

Seven of the initiating events and conditions that occurred during FY 1999 had a conditional core damage probability (CCDP) or a change in core damage probability (ΔCDP or importance) $\geq 1 \times 10^{-6}$. Of these seven precursors, all were at-power precursors.

Each precursor analysis was transmitted to the respective licensee and NRC staff for peer review and comment. The responses to comments are documented in the final analysis report.

3.1.1 At-Power Precursors

Seven events and conditions that occurred with the plant at-power had a CCDP or an importance (ΔCDP) $\geq 1 \times 10^{-6}$. The results of final ASP analyses for FY 1999 are presented in Table 3.1 for at-power precursors involving initiating events and Table 3.2 for at-power precursors involving conditions. Section 1.3 provides the criteria for an at-power precursor.

3.1.2 Shutdown Precursors

No shutdown precursors were identified in the evaluation of FY 1999 events; Section 1.3 discusses the criteria for a shutdown-related precursor.

3.1.3 Containment-Related Events

No containment-related events were identified in the evaluation of FY 1999 events; Section 2.5 discusses the criteria for a containment-related event.

3.1.4 "Interesting" Events

No "interesting" events were identified in the evaluation of FY 1999 events; Section 2.6 discusses the criteria for an interesting event.

3.2 Overall Industry Trends

This section provides the results of trending analyses for all precursors and for CCDP bins. The data used in the trend analyses and an explanation of the trend plot are provided in Appendix C.

3.2.1 Occurrence Rate of Precursors

The 1993–1999 ASP results are trended by fiscal year. The trend plot is shown in Figure 3.1.

The mean occurrence rate of precursors has exhibited a decreasing trend that is statistically significant (p-value = 0.005) (Figure 3.1). The occurrence rate of precursors has decreased over the period by a factor of 2 to 3.

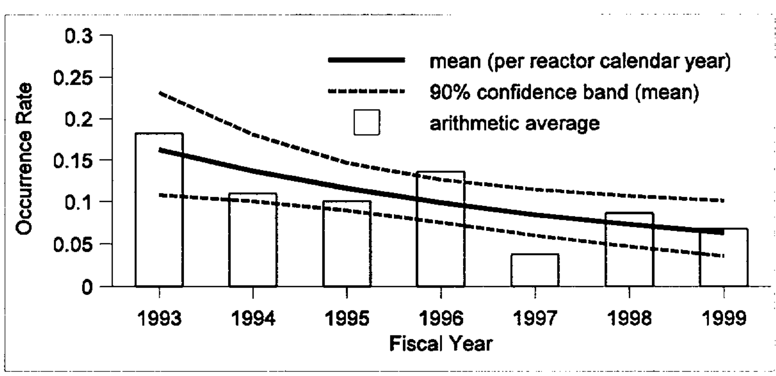

Figure 3.1. ***All precursors***—occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.0053).

Table 3.1. FY 1999 at-power precursors involving initiating events.

Plant	Description/Event Identifier	Plant type	Event date	CCDP	Event type
Davis - Besse	Manual reactor trip while recovering from a component cooling system leak and de-energization of safety-related bus D1 and non-safety bus D2 (LER 346/98-011)	PWR	10/14/98	1.4×10^{-5}	Transient
Indian Point 2	Loss of offsite power to safety-related buses following a reactor trip and a emergency diesel output breaker trip (LER 247/99-015)	PWR	08/31/99	2.8×10^{-6} (See note)	Loss of offsite power

Note: The Indian Point 2 precursor analysis is being re-analyzed in light of additional information received after the final analysis was issued. However, any change in results that increases or decreases the CCDP into a different CCDP bin (e.g., 10^{-4}, 10^{-5}, 10^{-7}) will not change the significance of trends or insights presented in this report.

Table 3.2. FY 1999 at-power precursors involving conditional unavailabilities.

Plant	Description/Event identifier	Plant type	Event date	CCDP	Importance (CCDP – CDP)	Event type
Oconee 1	Postulated high-energy line leaks or breaks leading to failure of safety-related 4 kV switchgear (LER 269-99-001)	PWR	2/24/99	3.4×10^{-5}	8.2×10^{-6}	Unavail-ability
Oconee 2	Postulated high-energy line leaks or breaks leading to failure of safety-related 4 kV switchgear (LER 269-99-001)	PWR	2/24/99	3.2×10^{-5}	5.6×10^{-6}	Unavail-ability
Oconee 3	Postulated high-energy line leaks or breaks leading to failure of safety-related 4 kV switchgear (LER 269-99-001)	PWR	2/24/99	3.1×10^{-5}	5.2×10^{-6}	Unavail-ability
Cook 1 and 2	Lack of capability to operate emergency service water following a seismic event (Inspection reports 50-315/316/97-024 and 50-315/316/99-010)	PWR	6/11/99	5.2×10^{-5}	3.2×10^{-5}	Unavail-ability

3.2.2 Occurrence Rate of Precursors by CCDP Bins

The occurrence rate of all precursors has exhibited a decreasing trend that is statistically significant during the FY 1993–1999 period (Figure 3.1). The data were analyzed to determine whether trends exist in the occurrence rate of precursors with CCDP of different orders of magnitude. The analysis method is based on a staff technical paper reported in Reference 5.

The objective of this analysis was to determine whether a trend exists in the ASP precursor bin data. The results of the trending analysis of the four probability bins (10^{-3}, 10^{-4}, 10^{-5}, 10^{-6}) for the FY 1993–1999 period are as follows:

CCDP Bin	Trend (FY 1993–1999)
10^{-6}	No statistically significant trend
10^{-5}	Decreasing–statistically significant
10^{-4}	Decreasing–statistically significant
$\geq 10^{-3}$	No statistically significant trend

A histogram of the occurrence rate as a function of fiscal year for each probability bins is provided in Figures 3.2a-d. The trend line of the mean occurrence rate (with the 90% confidence band) is shown in the two figures with statistically significant trends. No trend line is shown when a statistically significant trend was not detected.

3.3 Insights

3.3.1 Significant Precursors

The ASP Program is used to monitor the agency's performance against the following Strategic Plan goal: "No more than one event per year identified as a significant precursor of a nuclear accident." A *significant precursor* is defined in the Strategic Plan as an event that has a 1/1000 (10^{-3}) or greater probability of leading to a reactor accident (Ref. 3).

No precursors were identified during FY 1999 with a CCDP or importance $\geq 1\times10^{-3}$. Precursors with an CCDP or importance $\geq 1\times10^{-3}$ have occurred, on the average, about once every 3 to 4 years. The events in this group appear to involve no common attributes, such as failure modes, causes, or systems, among the events.

Two precursors with a CCDP $\geq 1\times10^{-3}$ have occurred since 1991—the Wolf Creek reactor coolant system draindown to the refueling water storage tank during hot shutdown (1994) and the Catawba 2 extended plant-centered loss of offsite power with an emergency diesel generator out of service for maintenance (1996).

3.3.2 Important Precursors

Precursors with a CCDP or importance $\geq 1\times10^{-4}$ are considered *important* in the ASP Program. There were no important precursors in FY 1999. The review of the ASP data reveals the following:

- The mean occurrence rate of *important* precursors has exhibited a decreasing trend that is statistically significant (p-value = 0.002) during the FY 1993–1999 period (Figure 3.3). The mean occurrence rate of precursors decreased over this period by a factor of 10 to 11.

- During the FY 1993–1999 period, 14 *important* precursors have occurred. Of these, almost half (43%) involved a loss of offsite power initiating event.

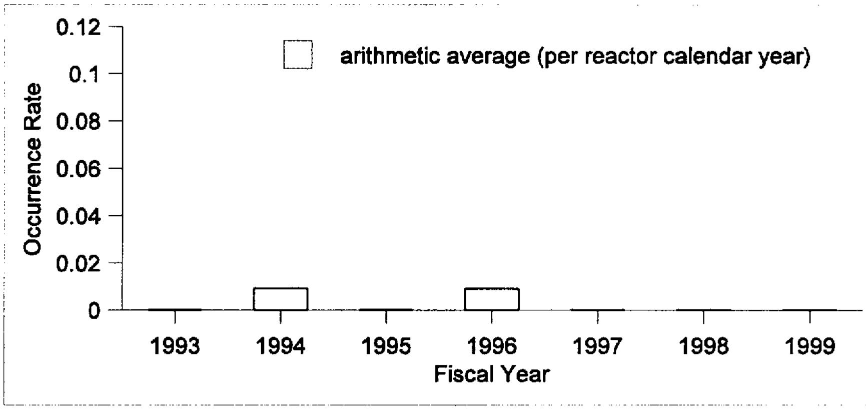

Figure 3.2a. ***Precursors in CCDP bin 10^{-3}***—occurrence rate, by fiscal year. No trend detected that is statically significant (p-value = 0.49).

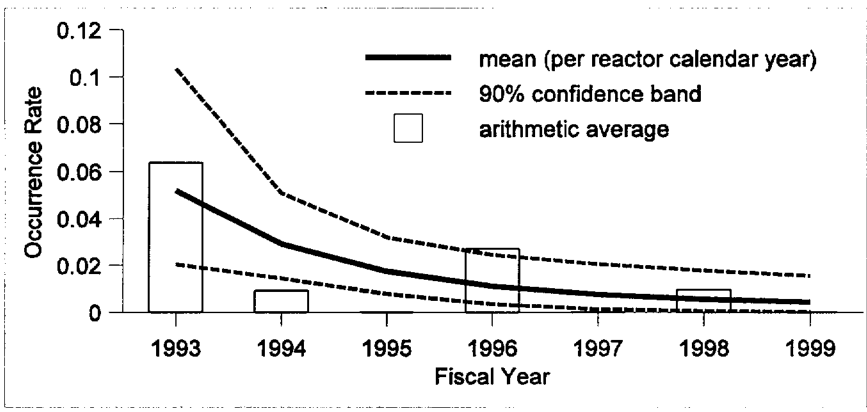

Figure 3.2b. ***Precursors in CCDP bin 10^{-4}***—occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.0018).

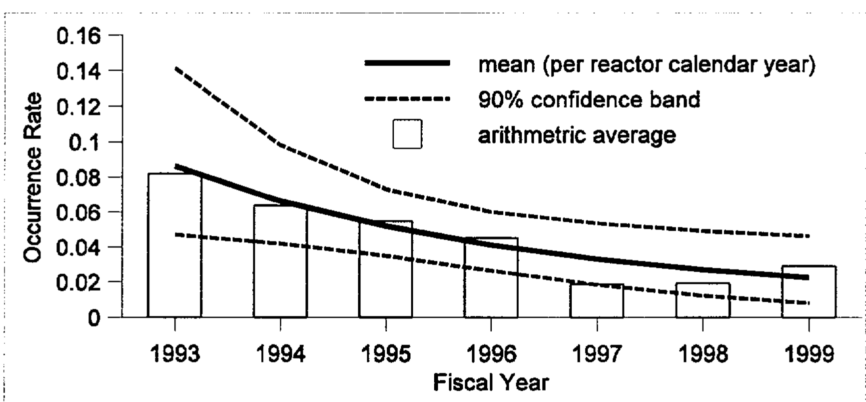

Figure 3.2c. ***Precursors in CCDP bin 10^{-5}***—occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.0076).

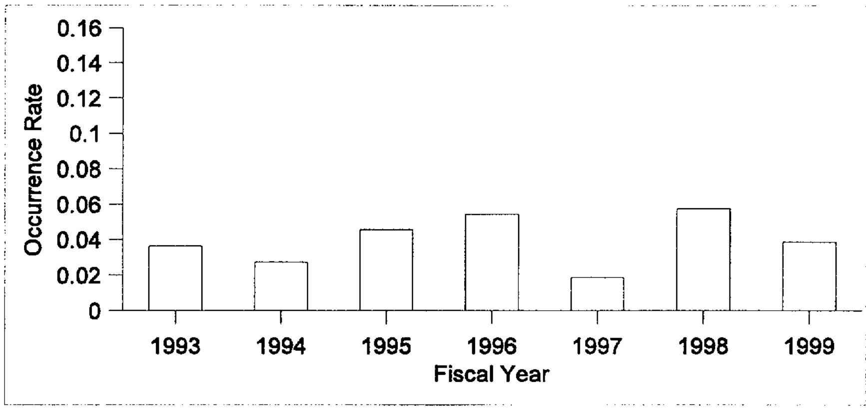

Figure 3.2d. ***Precursors in CCDP bin 10^{-6}***—occurrence rate, by fiscal year. No trend detected that is statistically significant (p-value = 0.70).

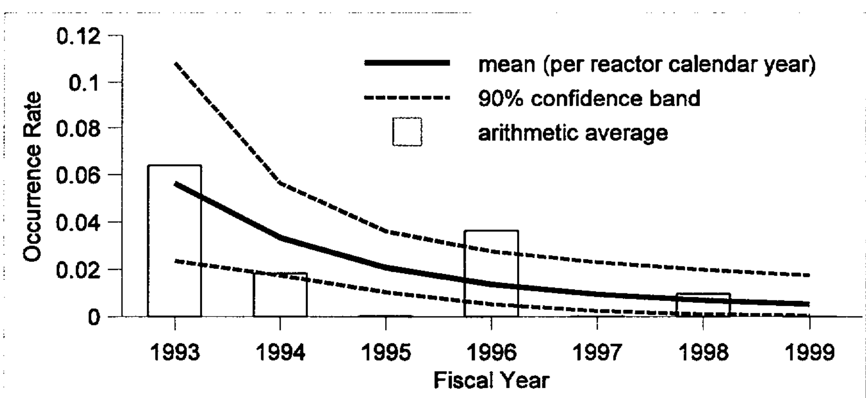

Figure 3.3. ***"Important" precursors (CCDP ≥ 10^{-4})***—occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.0017).

3.3.3 Initiating Events vs. Conditions

A precursor can be the result of an operational event involving an actual initiating event (e.g., loss of offsite power) and/or a condition found during a test, inspection, or engineering evaluation involving a reduction in safety system reliability or function for a specific duration (and no initiator actually occurred during this time).

Five of the seven precursors in FY 1999 involved the unavailability of equipment; two involved initiating events. A review of the data reveals the following:

- The results for FY 1999 are consistent with the FY 1993–1998 results. Historically, conditional unavailability of equipment (60%) outnumbered initiating events (40%). During the FY 1993–1999 period, 71% of the precursors involved conditional unavailability of equipment. This indicates that risk-significant conditions are being identified prior to unplanned demands of the degraded safety systems.

- The mean occurrence rate of precursors involving initiating events has exhibited a decreasing trend that is statistically significant (p-value = 0.005) during the FY 1993–1999 period (Figure 3.4). The occurrence rate of precursors decreased over this period by a factor of 4 to 5.

- No trend that was statistically significant was detected for precursors involving conditional unavailability of equipment (p-value = 0.2). See Figure 3.5.

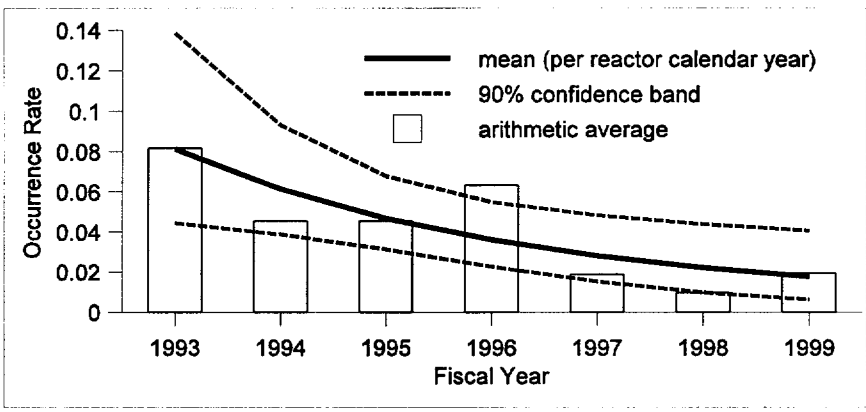

Figure 3.4. ***Precursors involving initiating events***—occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.0049).

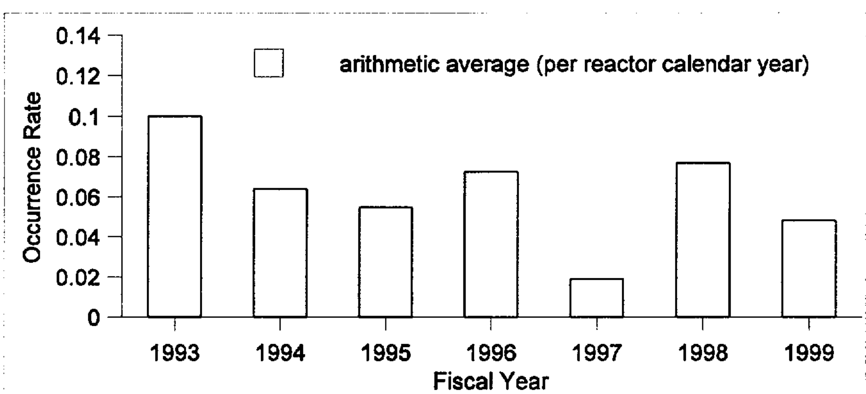

Figure 3.5. ***Precursors involving conditional unavailability of equipment***—occurrence rate, by fiscal year. No trend detected that is significantly significant (p-value = 0.18).

3.3.4 Precursors Involving Loss of Offsite Power Initiating Events

One precursor involving a loss of offsite power (LOOP) initiating event occurred during FY 1999—a plant-centered LOOP to safety-related buses following a reactor trip at Indian Point 2. The review of the ASP data reveals the following:

- The results for FY 1999 are consistent with the period FY 1993–1998, when the average number of precursors involving a LOOP initiator was about two a year.
- The mean occurrence rate of LOOP-related precursors exhibited a decreasing trend during the FY 1993–1999 period. The trend is almost statistically significant [2] (p-value 0.09) (Figure 3.6).
- Almost half (43%) of the LOOP events that occurred during FYs 1993-1999 are considered *important* precursors (CCDP $\geq 1 \times 10^{-4}$) in the ASP Program.
- A simultaneous unavailability of emergency power system train and a LOOP were also involved in three of the LOOP-related precursors during the FY 1993–1999 period. Two of the precursors involving a LOOP event and emergency power train unavailability had a CCDP $\geq 1\times10^{-4}$.
- None of the precursors since 1989 have involved a grid-related LOOP event.

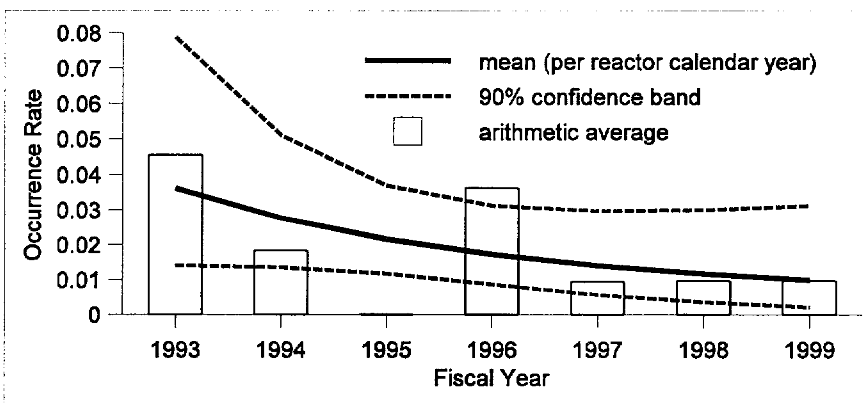

Figure 3.6. ***Precursors involving loss of offsite power initiating events***— occurrence rate, by fiscal year. The trend detected is almost statistically significant (p-value = 0.091).

[2] The trend will become statistically significant if only one LOOP initiating event occurs during the next two fiscal years.

3.3.5 Precursors at BWRs vs PWRs

None of the precursors in FY 1999 occurred at a BWR; there has been only one such precursor since 1996. A review of the data for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) during the FY 1993–1999 period reveals the following:

- The mean occurrence rates of precursors at BWRs and at PWRs have both exhibited a decreasing trend that are statistically significant (p-values of 0.009 and 0.05, respectively) (Figures 3.7 and 3.8).
- The mean occurrence rate of precursors has decreased during the FY 1993–1999 period by a factor of 10 at BWRs compared to a factor of 2 at PWRs.
- A precursor is 7 to 8 times as likely to occur at a PWR, than at a BWR. This is based on the occurrence rate at the end of the trend line (i.e., FY 1999) in Figures 3.7 and 3.8.

According to the staff's review of individual plant examinations (NUREG-1560, Ref. 6), the core damage frequencies estimated in the individual plant examinations were generally lower for BWRs than for PWRs. NUREG-1560 attributed the difference to the larger number of injection systems in the BWR design along with the ability to rapidly depressurize to allow the use of low-pressure injection systems. This may explain, in part, the lower number of precursors at BWRs.

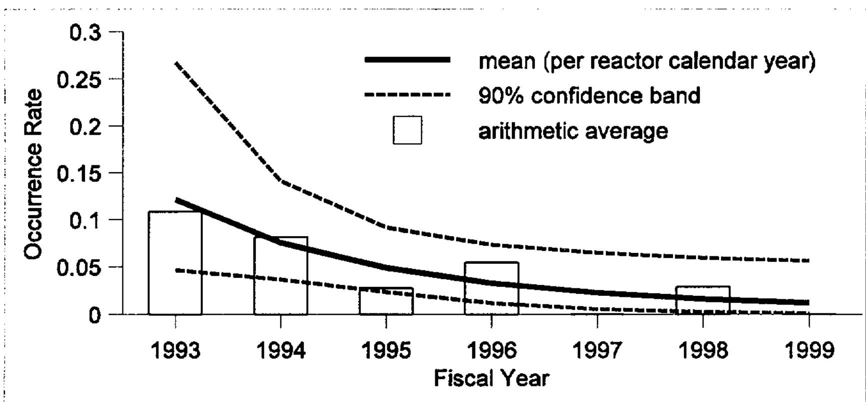

Figure 3.7. ***Precursors involving BWRs***—occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.0094).

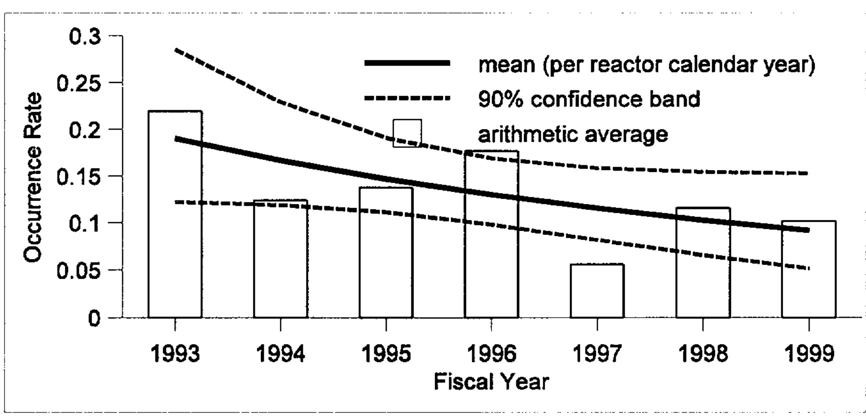

Figure 3.8. ***Precursors involving PWRs***—occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.047).

3.3.6 Consistency with PRAs/IPEs

Most of the precursor events are consistent with failure combinations identified in probabilistic risk assessments (PRAs) and individual plant examinations (IPEs). A review of the precursor events for the period 1994–1999 shows that several precursors involved event initiators or conditional availability of equipment not typically modeled in PRAs or IPEs. These events make up 21% of the precursors for this period.

Precursors not typically modeled in PRAs and IPEs are listed in Table 3.3.

Table 3.3. Precursors not typically modeled in PRAs or IPEs.

Year	Plant(s)	Event description
1994	Wolf Creek	Blowdown of the reactor coolant system to the refueling water storage tank during hot shutdown
1996	Wolf Creek	Reactor trip with the loss of one train of emergency service water due to the formation of frazil ice on the circulating water traveling screens and the unavailability of the turbine-driven auxiliary feedwater pump
1996	LaSalle 1 and 2	Fouling of the cooling water systems due to concrete sealant injected into the service water tunnel
1996	Haddam Neck	Potentially inadequate residual heat removal pump net positive suction head following a large- or medium-break loss-of-coolant accident due to design errors
1998	Oconee 1, 2, and 3	Incorrect calibration of the borated water storage tank (BWST) level instruments resulted in a situation where the emergency operating procedure (EOP) requirements for BWST-to-reactor building emergency sump transfer would never have been met; operators would be working outside the EOP
1998	Cook 2	Potential failure of all component cooling water pumps due to steam intrusion resulting from a postulated high-energy line break
1999	Oconee 1, 2, and 3	Postulated high-energy line leaks or breaks leading to failure of safety-related 4 kV switchgear

4. REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC). NUREG/CR–0400, "Risk Assessment Review Group Report." NRC: Washington, D.C. September 1978.

2. U.S. Nuclear Regulatory Commission. NUREG–75/014 (WASH–1400), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants." NRC: Washington, D.C. October 1975.

3. U.S. Nuclear Regulatory Commission. NUREG–1614, Vol. 2, Part 1, "Strategic Plan, Fiscal Year 2000 - Fiscal Year 2005." NRC: Washington, D.C. February 2000.

4. Weerakkody, S.D., et al. NUREG–1728, "Assessment of Risk Significance Associated With Issues Identified at D.C. Cook Nuclear Plant." NRC: Washington, D.C. October 2000.

5. Rasmuson, D.M. and P.D. O'Reilly. "Analysis of Annual Accident Sequence Precursor Occurrence Rates for 1984–94." *Proceeding of the International Topical Meeting on Probabilistic Safety Assessment.* American Nuclear Society (ANS), Park City, Utah. 29 September–3 October 1996. Vol. III, pp. 1645-1652. ANS: LaGrange Park, Illinois. 1994.

6. U.S. Nuclear Regulatory Commission. NUREG–1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Summary Report." NRC: Washington, D.C. October 1996.

5. BIBLIOGRAPHY

Component failure probabilities and initiating events frequencies are updated in the SPAR model using results from component and system reliability and initiating event studies. These studies, which are based on recent operating experience, include plant-specific values for parameters (failure probabilities, initiating event frequencies) with detectable plant-to-plant variations.

A list of studies and databases used in ASP analyses is provided below. All studies listed here have been peer reviewed by NRC staff and external stakeholders.

Initiating Events

1. J. P. Poloski, et al., *Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995*, NUREG/CR–5750, February 1999.

2. C. L. Atwood, et al., *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1996*, NUREG/CR–5496, November 1998.

3. J. R. Houghton, *Special Study: Fire Events–Feedback of U. S. Operating Experience*, AEOD/S97-03, June 1997.

Component Reliability

1. J. R. Houghton, H. G. Hamzehee, *Component Performance Study: Turbine-Driven Pumps, 1987–1998*, NUREG–1715, Vol. 1, April 2000.

2. J. R. Houghton, H. G. Hamzehee, *Component Performance Study: Motor-Driven Pumps, 1987–1998*, NUREG–1715, Vol. 2, June 2000.

System Reliability

1. J. P. Poloski, et al., *Reliability Study: Auxiliary/Emergency Feedwater System, 1987–1995*, NUREG/CR–5500, Vol. 1, August 1998.

2. S. A. Eide, et al., *Reliability Study: Westinghouse Reactor Protection System, 1984–1995*, NUREG/CR–5500, Vol. 2, April 1999.

3. S. A. Eide, et al., *Reliability Study: General Electric Reactor Protection System, 1984–1995*, NUREG/CR–5500, Vol. 3, May 1999.

4. G. M. Grant, et al., *Reliability Study: High-Pressure Coolant Injection (HPCI) System, 1987–1993*, NUREG/CR–5500, Vol. 4, September 1999.

5. G. M. Grant, et al., *Reliability Study: Emergency Diesel Generator Power System, 1987–1993*, NUREG/CR–5500, Vol. 5, September 1999.

6. G. M. Grant, et al., *Reliability Study: Isolation Condenser System, 1987–1993*, NUREG/CR–5500, Vol. 6, September 1999.

7. J. P. Poloski, et al., *Reliability Study: Reactor Core Isolation Cooling (RCIC) System, 1987–1993*, NUREG/CR–5500, Vol. 7, September 1999.

8. J. P. Poloski, et al., *Reliability Study: High-Pressure Core Spray (HPCS) System, 1987–1993*, NUREG/CR–5500, Vol. 8, September 1999.

Common-Cause Failure

1. F. M. Marshall, et al., *Common-Cause Failure Parameter Estimates*, NUREG/CR–5497, October 1998.

2. A. Mosleh, et al., *Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment*, NUREG/CR–5485, November 1998.

3. F. M. Marshall, et al., *Common-Cause Failure Database and Analysis System: Overview*, NUREG/CR–6268, Vol. 1, June 1998.

4. F. M. Marshall, et al., *Common-Cause Failure Database and Analysis System: Event Definition and Classification*, NUREG/CR–6268, Vol. 2, June 1998.

5. F. M. Marshall, et al., *Common-Cause Failure Database and Analysis System: Data Collection and Event Coding*, NUREG/CR–6268, Vol. 3, June 1998.

6. K. J. Kvarfordt, et al., *Common-Cause Failure Database and Analysis System: Software Reference Manual*, NUREG/CR–6268, Vol. 4, June 1998.

APPENDIX A
EVALUATION PROCESS DETAILS

The Accident Sequence Precursor (ASP) Program is concerned with the identification and documentation of operational events and conditions that have involved portions of core damage sequences and with the estimation of associated frequencies and probabilities.

Identification of precursors requires the review of operational events and conditions for instances in which plant functions that provide protection against core damage have been challenged or compromised. Based on previous experience with reactor plant operational events and conditions, it is known that most operational events and conditions can be directly or indirectly associated with four initiators:

- Reactor trip which includes loss of main feedwater within its sequences,
- Loss of offsite power,
- Small-break loss-of-coolant accident, and
- Steam generator tube rupture in pressurized-water reactors.

These four initiators are primarily associated with loss of core cooling. ASP Program analysts examine licensee event reports (LERs) and other event documentation to determine the impact that operational events and conditions have on potential core damage sequences associated with these initiators. (Operational events and conditions are occasionally identified that impact other initiators, such as a large-break loss-of-coolant accident. Unique models are developed to address these events.)

Details of various elements of the evaluation process are discussed below. Figure 2.1 of the main report illustrates this process.

A.1 Selection of Potential Precursors for Analysis

A.1.1 Event Data Sources

In the evaluation of events, two primary sources were used to identify potential precursors:

- LERs in the Sequence Coding and Search System (SCSS) database.

  The SCSS database contained 831 full-texted LERs for FY 1999. In addition, each LER is coded for equipment performance, loss of function (system/train failures), personnel errors, and the effect on the plant. These codes are used to identify potential precursors in the screening process discussed in the next section.

- NRC inspection reports from routine inspections, special inspections, and augmented inspection team inspections.

  Results from a special ASP Program assessment of 141 issues identified at the Donald C. Cook Nuclear Power Plant, Units 1 and 2 during the FY 1999 period were included in this report.

Sources used in the Cook study to identify potential precursors for analysis include all LERs, NRC inspection reports, licensee condition reports, and self-assessment reports performed by the licensee. Details of the Cook risk study can be found in NUREG-1728, Vols.1 and 2, "Assessment of Risk Significance Associated With Issues Identified at D.C. Cook Nuclear Power Plant" (Ref. A.1).

A.1.2 Initial Review Process

The ASP Program employs the following three-phased process in its screening, review, and analysis of operational events and conditions for precursors:

SCSS screening. A computerized SCSS screening algorithm was developed to address the intensive review activity to identify all potential precursors over a yearly period. The purpose of the algorithm is to reduce significantly the number of LERs subject to detailed review by ASP Project staff, yet still identify all potential precursors reported in LERs.

The algorithm capitalizes on the intensive LER review and encoding already used for SCSS and utilizes SCSS' search capabilities. The algorithm was constructed in a manner analogous to the ASP manual screening conducted in earlier years of the program.

The algorithm provides a reduction of about 75% in the number of LERs that are required to be reviewed.

Engineering reviews and review criteria. Those events selected underwent one- or two-engineer review(s) to determine if the reported event should be examined in greater detail. Events that, in the judgment of the initial reviewing engineer, clearly failed to satisfy the ASP criteria for analysis as a potential precursor were not subject to another evaluation. All other events were reviewed by two engineers to determine if they met the ASP criteria for detailed analysis before the decision was made to reject or to analyze the event. This initial review was a bounding review, meant to capture events that in any way appeared to deserve detailed review and to eliminate events that were clearly unimportant. This process involved eliminating events that satisfied predefined criteria for rejection and accepting all others as potentially significant and requiring analysis.

LERs were eliminated from further consideration as precursors if they involved only one of the following:

- Component failure with no loss of redundancy,
- Short-term loss of redundancy in only one system,
- An event that occurred prior to initial criticality,
- Design or qualification error that was small relative to what was predicted (e.g., an error of a few percent in an actuation setpoint),
- An event bounded by a general reactor trip or a loss of feedwater,
- An event with no appreciable impact on safety systems, or
- An event involving only post-core-damage impacts.

Events identified for further consideration typically included the following:

- Unexpected core damage initiators (i.e., loss of offsite power, steam generator tube rupture, and small-break loss-of-coolant accident);
- All events in which a reactor trip was demanded and a safety-related component failed;
- All support system failures, including failures in cooling water systems, instrument air, instrumentation and control, and electric power systems;
- Any event in which two or more failures occurred;
- Any event or condition that was not predicted or that proceeded differently from the plant design basis; and
- Any event that, based on the reviewers' experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Detailed analysis. Events determined to be potentially significant as a result of this initial review were then subjected to a thorough, detailed analysis. This extensive analysis was intended to identify those events considered to be precursors to potential severe core damage accidents, either because of an initiating event or because of failures that could have affected the course of postulated off-normal events or accidents.

The detailed analysis of each event considered the immediate impact of an initiating event or the potential impact of the equipment failures or operator errors on the readiness of systems in the plant for mitigation of off-normal and accident conditions. In the review of each selected event, three general scenarios (involving both the actual event and postulated additional failures) were considered.

- If the event or failure was immediately detectable and occurred while the plant was at power, then the event was evaluated according to the likelihood that it and the ensuing plant response could lead to severe core damage.

- If the event or failure had no immediate effect on plant operation (i.e., if no initiating event occurred), then the review considered whether the plant would require the failed items for mitigation of potential severe core damage sequences should a postulated initiating event occur during the failure period.

- If the event or failure was identified while the plant was not at power, then the event was first assessed to determine whether it could have impacted at-power operation.

  If the event could have impacted at-power operation, its impact was assessed.

  If the event could only occur at cold shutdown or refueling shutdown, then its impact on continued decay heat removal during shutdown was assessed.

For each actual or postulated initiating event associated with an operational event or condition, the sequence of operation of various mitigating systems required to prevent core damage was considered. Events were selected and documented as precursors to potential severe core damage accidents (accident sequence precursors) if the conditional probability of subsequent core damage was at least 1.0×10^{-6}. Events of low significance were thus excluded, allowing attention to be focused on the more important events. This approach is consistent with the approach used to define 1987–1998 precursors, but differs from that of earlier ASP reports which addressed all events meeting the precursor selection criteria regardless of conditional core damage probability.

A.2 Sources of Input Information for Detailed Analysis

Various sources of plant- and event-specific information are used to perform the detailed analysis.

Event-related information. Information describing the event or condition in the LER can be supplemented with additional information from the following sources:

- NRC inspection reports, which can be found on the NRC web page via the external (public) server.
- Inspectors knowledgeable of the specific event or condition.
- NRC staff experts in the relevant technical areas.
- The licensee through follow-up event assessment.
- Assessment reports issued by the licensee to support public regulatory conferences.

Plant design and operation information. The adaption of the plant-specific SPAR model to the event or condition may require design-related information from plant-specific sources such as:

- Updated Final Safety Analysis Report.
- Technical Specifications.
- Individual plant examinations for internal and external events.
- Operating procedures obtained from the licensee.
- NRC resident inspector.
- Site visit by the ASP analyst.
- Other similar-related LERs previously issued by the licencee (may contain design and operational information).

Updates of model parameters. Component failure probabilities and initiating event frequencies may be updated in the SPAR model using results from component and system reliability and initiating event studies. These studies, which are based on the operating experience, include plant-specific values for parameters (failure probabilities, initiating event frequencies) with detectable plant-to-plant variations. Examples of studies include:

- System reliability studies for high-pressure injection systems in boiling-water and pressurized water reactors, auxiliary feedwater systems (if applicable), emergency diesel generators, safety-related service water systems and reactor protection systems.

- Component reliability studies for motor- and air-operated valves, and motor- and turbine-driven pumps.

- Common-cause failure parameter estimations and database for 42 components in boiling-water and pressurized water reactors.

- Initiating event studies for reactor trips, loss-of-coolant accidents, fires, and loss of offsite power events.

An all-inclusive list of studies and databases used in ASP analyses is provided in the Bibliography in this report. All studies listed in the Bibliography have been peer reviewed by NRC staff and industry organizations.

Peer reviews. Lastly, all preliminary analyses of precursors are sent to the NRC staff and to the licensee for peer review. This peer review provides the opportunity for reviewers to point out errors in the model and analysis assumptions, present additional information concerning the functionality and recoverability of failed equipment, and suggest updates to model parameters based on plant-specific data.

A.3 Detailed Analysis of Potential Precursors

Summary. Quantification of the significance of an operational event or condition involves the determination of a conditional probability of subsequent severe core damage given the failures observed during an operational event or condition. This conditional probability is estimated by mapping observed failures or conditions onto the ASP accident sequence models, and calculating a conditional core damage probability. These plant-specific models, called Standardized Plant Analysis Risk (SPAR) models, contain event trees and linked fault trees. The models depict potential paths to severe core damage.

The effect of a precursor on event tree branches is assessed by evaluating the operational event or condition specifics against system design information and modeling assumptions. The evaluation of the event or condition includes all actual or potential concurrent failures, degradations, or outages of safety- and non-safety-related mitigation systems. The evaluation also include estimates of the likelihood of equipment failures, human errors, and recovery actions. This information is used to modify the SPAR model. Random failures are assumed for other branches of the event tree models not related to the specific operational event or condition being analyzed. The quantification of the revised model results in a revised conditional probability of core damage given the operational event or condition.

A.4 Potentially Significant Events Considered Impractical to Analyze

In some cases, events are impractical to analyze because of the inability to reasonably model within a probabilistic risk assessment (PRA) framework, considering the level of detail typically available in PRA models and the resources available to the ASP Program. These events usually involve component degradations in which the extent of the degradation cannot be readily determined or the impact of the degradation on plant response cannot be readily ascertained.

A.5 Containment-Related Events

Events involving loss of containment functions---containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control—are classified as "containment-related" events. Containment-related events are not currently considered precursor events under the ASP Program; the ASP Program is currently developing containment-related event models. The potential for increased exposure to the public justifies their inclusion in the report. Only a qualitative discussion is provided for containment-related events.

A.6 "Interesting" Events

Events or conditions that provide insight into unusual failure modes with the potential to compromise continued core cooling but are not considered to be precursors (i.e., conditional core damage probability or importance $< 1 \times 10^{-6}$) are documented as "interesting" events. Only a qualitative discussion is provided for "interesting" events.

A.7 Independent Review Process

Preliminary precursor analyses undergo two independent reviews. The process is illustrated in Figure 2.2 of the main report. In the first review, the preliminary analysis is reviewed in-house by a second analyst. After completion of the first review and any corresponding revision, the final preliminary analysis is transmitted to the pertinent nuclear plant licensee and to the

NRC staff for peer review. The licensee is requested to review and comment on the technical adequacy of the analyses, including the depiction of their plant equipment and equipment capabilities. The review guidance provided to the licensee is provided in Appendix B. Review comments are evaluated for applicability and pertinence to the ASP analysis.

After the preliminary analysis is revised based on licensee and NRC staff comments, the modified analysis is reviewed by the second analyst for final review and revised again, if necessary. Technical audits by branch management are performed for preliminary and final analyses prior to issuance. Technical differences are discussed with the reviewer. The response to comments and differences are documented in the final ASP analysis report.

A.8 Uncertainties or Factors Affecting Results

Improvements have been applied in the analysis of FY 1999 precursors, such as updates of model parameters (initiating event frequencies and failure probabilities) based on the operating experience; a new human reliability analysis methodology developed for the SPAR Revision 3 models; and detailed event-related information and plant-specific design-related information from the regional senior reactor analysts (SRAs).

However, as with any analytic procedure, the availability of information and modeling assumptions can affect results. Several of these potential sources of uncertainties affecting the results and measures taken in the ASP Program to reduce these uncertainties are addressed below.

SCSS screening. As described in Section A.1.2, above, not all LERs are reviewed. A search algorithm is used to search through encoded LERs in the SCSS database. The algorithm was executed on the SCSS database and compared to results from past manual ASP reviews for eight years (Ref. A.2). The algorithm was successful in screening 15,000 LERs during the time period to approximately 3,000. The number of precursors in the remaining group is 299 out of a total of 305 identified during manual review. Excluding three of these missed precursors that would not be labeled as precursors under the present ASP Program, the success rate for this test was 99%. The other three events were not found by the algorithm due to incomplete coding in the SCSS database.

With the new reactor oversight process, specifically, the Significance Determination Process (SDP) used by the inspection staff in assessing plant performance, results from the SDP as documented in inspection reports can be used as an additional check on the adequacy of SCSS screening results.[1]

Modeling details. The event trees used in the analysis are plant-class specific and reflect differences between plants in the eight plant classes that have been defined. The fault trees are structured to reflect the plant-specific systems. While major differences between plants are represented in this way, the plant models may not reflect all of the important differences such as the impact of potential support system failures,[2] station blackout issues, and plant-specific, non-safety-related accident mitigation strategies (e.g., cross-ties, battery load shed procedures) not credited in the individual plant examination. However, every effort is made to obtain current design and operational information which may have an important contribution to the risk being analyzed.

Event trees in the SPAR Revision 2QA models do not include infrequent events, such as medium, large, and interfacing systems LOCAs, losses of support systems as initiating events, and external events (e.g., tornados, fires, earthquakes).

For those cases where the SPAR model lacks the modeling capability for analyzing an operational event or condition, the appropriate details will be developed and added to the SPAR model on an as needed basis.

---

[1] The Reactor Oversight Process was implemented for all plants midway through FY 2000.

[2] Limited support system modeling is provided in the current revision 2 SPAR models. Modeling of support systems will be provided in the revision 3 models.

Recovery of failed equipment. Assignment of recovery credit for an event can have a significant impact on the assessment of the event. The actual likelihood of failing to recover from an event at a particular plant is difficult to assess and may vary substantially from the values currently used in the ASP analyses. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, and others concerning the likelihood of recovering from specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event. Programmatic constraints have prevented substantial efforts in estimating actual recovery probabilities.

The values currently used are based on a review of recovery actions during historic events and also include consideration of human error during recovery. These values have been reviewed both within and outside the ASP Program. While it is acknowledged that substantial uncertainty exists, these values are considered reasonable for estimating the risk associated with an operational event or condition.

To improve consistency in human reliability estimates in ASP analyses, the ASP Program has adopted an improved method for performing human reliability analysis. The ASP human reliability analysis methodology makes use of a two-page worksheet to rate a series of performance shaping factors and dependency factors to arrive at a screening level human error probability for a given task. This method, which is being applied in the Revision 3 of the SPAR models, is an improvement over the previous method used in the Revision 2 of the SPAR models.

This improved method was applied in the analysis of FY 1999 precursors. The human error and non-recovery probabilities were revised in the SPAR, Rev. 2 models using the new methodology.

A.9 References

1. Weerakkody, S.D., et al. NUREG–1728, "Assessment of Risk Significance Associated With Issues Identified at D.C. Cook Nuclear Plant." Nuclear Regulatory Commission: Washington, D.C. October 2000.

2. Poore, W.P. "LER Screening Algorithm For Identification of Potential Accident Sequence Precursor Events." *Proceeding of the International Topical Meeting on Probabilistic Safety Assessment.* American Nuclear Society (ANS), Park City, Utah. 29 September–3 October 1996. Vol. III, pp. 1645-1652. ANS: LaGrange Park, Illinois. 1994.

APPENDIX B
GUIDANCE FOR LICENSEE REVIEW OF PRELIMINARY ASP ANALYSIS

The information in this appendix is sent to the licensee along with the preliminary Accident Sequence Precursor (ASP) analysis. The information provides specific guidance for performing the requested review, identifies the criteria used by the analyst to determine whether any credit should be given in the analysis for the use of licensee-identified additional equipment or specific actions in recovering from the event, and describes the specific information that the licensee should provide to support such a claim.

Background

The preliminary precursor analysis of an event or condition that occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP Program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage.

The types of events evaluated include actual initiating events, such as a loss of off-site power or loss-of-coolant accident, degradation of plant conditions, and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences.

This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and other pertinent reports, such as the licensee event report (LER) and/or NRC inspection reports.

Modeling Techniques

The models used for the analysis of events were developed by the Idaho National Engineering and Environmental Laboratory. The models were developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The developed models are called Standardized Plant Analysis Risk (SPAR) models. The SPAR models are based on linked fault trees. Fault trees were developed for each top event on the event trees to a super component level of detail.

SPAR Revision 2 models have four types of initiating events: (1) transients, (2) small loss-of-coolant accidents, (3) steam generator tube rupture (if applicable) and (4) loss of offsite power. The only support system modeled in Revision 2 is the electric power system. The SPAR models have transfer events tree for station blackout and anticipated transient without scram.

The models may be modified to include additional detail for the systems/components of interest for a particular event. This may include additional equipment or mitigation strategies as outlined in the FSAR or IPE. Probabilities are modified to reflect the particular circumstances of the event being analyzed.

Guidance for Peer Review

Comments regarding the analysis should address:

- Does the "Event Summary" section:
  - accurately describe the event as it occurred; and
  - provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?
- Does the "Modeling Assumptions" section:
  - accurately describe the modeling done for the event;
  - accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions; and
  - include assumptions regarding the likelihood of equipment recovery?

Appendix G of Reference 1 provides examples of comments and responses for previous ASP analyses.

Criteria for Evaluating Comments

Modifications to the event analysis may be made based on the comments that you provide. Specific documentation will be required to consider modifications to the event analysis. References should be made to portions of the LER or other event documentation concerning the sequence of events. System and component capabilities should be supported by references to the FSAR, IPE, plant procedures, or analyses. Comments related to operator response times and capabilities should reference plant procedures, the FSAR, the IPE, or applicable operator response models. Assumptions used in determining failure probabilities should be clearly stated.

Criteria for Evaluating Additional Recovery Measures

Additional systems, equipment, or specific recovery actions may be considered for incorporation into the analysis. However, to assess the viability and effectiveness of the equipment and methods, the appropriate documentation must be included in your response. This includes:

- normal or emergency operating procedures,
- piping and instrumentation diagrams (P&IDs),
- electrical one-line diagrams,
- results of thermal-hydraulic analyses, and
- operator training (both procedures and simulation).

This documentation must be current at the time of the event occurrence. Systems, equipment, or specific recovery actions that were not in place at the time of the event will not be considered. Also, the documentation should address the impact (both positive and negative) of the use of the specific recovery measure on:

- the sequence of events,
- the timing of events,
- the probability of operator error in using the system or equipment, and
- other systems/processes already modeled in the analysis (including operator actions).

An Example of a Recovery Measure Evaluation

A pressurized-water reactor plant experiences a reactor trip. During the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW modeling would be patterned after information gathered either from the plant FSAR or the IPE. However, if information is received about the use of an additional system (such as a standby steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be mitigated by the use of the standby feedwater system.

The mitigation effect for the standby feedwater system would be credited in the analysis provided that the following material was available:

- standby feedwater system characteristics are documented in the FSAR or accounted for in the IPE,

- procedures for using the system during recovery existed at the time of the event,

- the plant operators had been trained in the use of the system prior to the event,

- a clear diagram of the system is available (either in the FSAR, IPE, or supplied by the licensee),

- previous analyses have indicated that there would be sufficient time available to implement the procedure successfully under the circumstances of the event under analysis, and

- the effects of using the standby feedwater system on the operation and recovery of systems or procedures that are already included in the event modeling. In this case, use of the standby feedwater system may reduce the likelihood of recovering failed AFW equipment or initiating feed-and-bleed due to time and personnel constraints.

Materials Provided for Review

The following materials have been provided in the package to facilitate your review of the preliminary analysis of the event or condition:

- Preliminary ASP analysis.
- Specific LER, NRC inspection report, or other pertinent reports for each preliminary ASP analysis.

Schedule

Please refer to the transmittal letter for schedules and procedures for submitting your comments.

Reference

1. R. J. Belles, et al., "Precursors to Potential Severe Core Damage Accidents: 1997, A Status Report," USNRC Report NUREG/CR-4674 (ORNL/NOAC-232) Volume 26, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, and Science Applications International Corp., Oak Ridge, Tennessee, November 1998.

APPENDIX C
DATA TABLES AND TREND PLOTS

The data used to support the results in Section 3 of the main report are provided below.

E.1 Tables

Listing of precursors for FYs 1993--1999 (Table E.1)

This table lists all precursors during the FY 1993-1999 period. This data was obtained from the ASP Database maintained by the Office of Nuclear Regulatory Research. In addition, the data can be extracted from the NUREG/CR-4674-series listed at front of this report. The data fields used in the table are defined below:

Event Date—Usually the date that the event occurred or equipment unavailability discovered. This date is taken from the licensee event report (LER) or NRC inspection report.

Docket and LER Nos.—(XXX/YY-ZZZ) The first set of numbers (XXX) is the plant docket number. The set of numbers after the slash (YY-ZZZ) is the LER number. In some cases, a precursor in a unit 2 plant may be reported in an LER reported under the unit 1 docket number. When searching for the LER in this case, look for the LER no. YY-ZZZ under the docket associated with unit 1.

For cases where an LER was not issued, as indicated by "-S01", an inspection report was used in the analysis. The inspection report number is referenced in the precursor analysis report.

CCDP or ΔCDP—Conditional core damage probability (CCDP) of initiating event assessments and change in core damage probability (ΔCDP) of conditional assessments.

Event Type—For initiating events, such as loss of offsite power, the initiating event is the event type. For conditional assessments, the event type (assigned a "U") is the initiator of the dominating sequence. The event type designator is the same for both, except a "U" indicates a condition.

The event types are defined as follows:

EQ:	Earthquake
LOCA:	Loss-of-coolant accident
LOOP:	Loss of offsite power
SGTR:	Steam generator tube rupture (PWRs)
SLB:	High-energy steam line break
SLC:	Standby liquid control (BWRs)
TRIP:	Reactor trip (none of the above)

Data used in the trending analysis (Table E.2)

This table provides the counts of precursors by fiscal year and by category used in the trending analysis. In addition, the operating experience used to calculate occurrence rates is summarized in the table. Details are provided in Table E.3.

Data used to calculate reactor calender years for trending analysis (Table E.3)

The data used to calculate the reactor calender years is provided in Table E.3. The unit of measure used in the trending analysis is the reactor calendar year. All trend plots presents the means in units of precursor events per reactor calendar year.

A reactor calendar year includes the time that a plant unit was in operation and shutdown outages (e.g., maintenance, refueling). For most plants, this is 365 days. The exceptions are new or decommissioned plants. For new plants, the period starts at the date when the operating licensee was issued. For a plant undergoing decommissioning, the period ends at the date when the plant was defueled.

The use of a fiscal year is a departure from previous ASP annual reports, in which the calendar year was used in the reporting of trends and insights. A fiscal year is from October 1 to September 30 in the following calendar year.

E.2 Trend Plots

The data used in the trending analysis are provided in Tables E.2 and E.3. Each trend plot provides the following information:

- Fitted mean of the occurrence rate, as indicated by the solid line, where the rate is in units of per reactor calendar year.

- 90% confidence band (5% lower bound and 95% upper bound) on the fitted mean, as indicated by the dashed lines on both sides of the mean.

- The p-value of the trend. The p-value is the probability of observing a trend as a result of chance alone. A p-value is considered statistically significant in this report if the p-value is smaller than 0.05.

- Histogram of the occurrence rate, by fiscal year. The rate in the histogram is the arithmetic average (i.e., the number of precursors divided by the number of reactor calendar years). This is also known as the *maximum likelihood estimate.*

The precursor trends in the boiling-water reactor and pressurized water reactor groups were based on the operating experience of the reactor type. For all other precursor groups, the trends were based on total U.S. experience during the FYs 1993–1999.

Table E.1. Precursors (FY1993-1999) sorted by event date.

Event Date	Docket and LER Nos.	Plant	Precursor Title	CCDP or ΔCDP	Event Type
1992/10/17	483/92-011	Callaway	LOSS OF MAIN CONTROL ROOM ANNUNCIATORS	1.300e-05	ULOOP
1992/10/19	270/92-004	Oconee 2	LOOP WITH FAILURE OF BOTH KEOWEE UNITS	2.100e-04	LOOP
1992/12/02	269/92-018	Oconee 1	BOTH KEOWEE UNITS POTENTIALLY UNAVAILABLE	3.200e-05	ULOOP
1992/12/02	270/92-018	Oconee 2	BOTH KEOWEE UNITS POTENTIALLY UNAVAILABLE	3.200e-05	ULOOP
1992/12/02	287/92-018	Oconee 3	BOTH KEOWEE UNITS POTENTIALLY UNAVAILABLE	3.200e-05	ULOOP
1992/12/31	327/92-027	Sequoyah 1	LOSS OF OFFSITE POWER	1.800e-04	LOOP
1992/12/31	328/92-027	Sequoyah 2	LOSS OF OFFSITE POWER	1.800e-04	LOOP
1993/01/22	498/93-005	South Texas 1	UNAVAIL OF ONE EMERGENCY DIESEL GENERATOR AND THE TURBINE DRIVEN AUXILIARY FEEDWATER FOR 25 DAYS	1.200e-05	ULOOP
1993/01/29	289/93-002	Three Mile Isl 1	BOTH RHR HEAT EXCHANGERS UNAVAILABLE FOR 3 HOURS	3.100e-06	ULOCA
1993/02/25	413/93-002	Catawba 1	ESSENTIAL SERVICE WATER POTENTIALLY UNAVAIL	1.500e-04	ULOOP
1993/02/25	414/93-002	Catawba 2	ESSENTIAL SERVICE WATER POTENTIALLY UNAVAIL	1.500e-04	ULOOP
1993/03/13	293/93-004	Pilgrim 1	WEATHER-INDUCED LOOP, VESSEL PRESSURE/TEMPERATURE LIMITS VIOLATED	4.600e-06	LOOP
1993/03/14	529/93-001	Palo Verde 2	STEAM GENERATOR TUBE RUPTURE (240 GPM)	4.700e-05	SGTR
1993/03/26	440/93-011	Perry 1	CLOGGED SUPPRESSION POOL STRAINERS AND SERVICE WATER FLOOD	1.200e-04	TRIP
1993/04/16	339/93-002	North Anna 2	AFW DISABLED AFTER REACTOR TRIP	1.100e-06	TRIP
1993/04/22	265/93-010	Quad Cities 2	DEGRADATION OF BOTH EDGs	6.000e-05	ULOOP
1993/06/27	213/93-S01	Haddam Neck	DEGRADATION OF MCC-5, PRESSURIZER PORV, BOTH EDGS	6.500e-05	ULOOP
1993/08/02	316/93-007	Cook 2	REACTOR TRIP WITH DEGRADED AFW	2.400e-06	TRIP
1993/09/14	373/93-015	LaSalle 1	SCRAM AND LOOP	1.300e-04	LOOP
1993/09/30	313/93-003	Arkansas 1	BOTH TRAINS OF RECIRCULATION INOPERABLE FOR 14 H	5.100e-05	ULOCA
1993/10/06	412/93-012	Beaver Valley 2	FAILURE OF BOTH EDG LOAD SEQUENCERS	2.100e-06	ULOCA
1993/10/12	334/93-013	Beaver Valley 1	DUAL-UNIT LOOP	5.500e-05	LOOP
1993/12/27	370/93-008	McGuire 2	LOOP AND FAILURE OF AN MSIV TO CLOSE	9.300e-05	LOOP
1994/01/12	318/94-001	Calvert Cliff 2	TRIP, LOSS OF 13.8-KV BUS, AND SALT WATER COOLING SYSTEM UNAVAIL FOR 2 MIN	1.300e-05	TRIP
1994/02/08	266/94-002	Point Beach 1	BOTH EDGS INOPERABLE FOR 47 HRS	1.200e-05	ULOOP

Table E.1 (Continued)

Event Date	LER No.	Plant	Precursor Title	CCDP or ΔCDP	Event Type
1994/02/08	301/94-002	Point Beach 2	BOTH EDGS INOPERABLE FOR 47 HRS	1.200e-05	ULOOP
1994/02/16	213/94-004	Haddam Neck	BOTH PRESSURIZER PORVS AND VITAL 480 VAC BUS DEGRADED	1.400e-04	ULOOP
1994/03/07	304/94-002	Zion 2	UNAVAIL OF TURBINE-DRIVEN AFW PUMP AND AN EDG	2.300e-05	ULOOP
1994/06/08	237/94-018	Dresden 2	MCC TRIPS DUE TO IMPROPER BREAKER SETTINGS	6.100e-06	ULOOP
1994/08/04	237/94-021	Dresden 2	LONG-TERM UNAVAIL OF HIGH-PRESSURE CORE SPRAY (3.5 MONTHS)	3.100e-06	ULOCA
1994/09/08	458/94-023	River Bend 1	SCRAM, MAIN TURBINE-GENERATOR FAILS TO TRIP, RCIC AND CONTROL ROD DRIVE SYSTEM UNAVAILABLE	1.800e-05	TRIP
1994/09/17	482/94-013	Wolf Creek 1	RCS BLOWS DOWN TO RWST (9,200 GAL) DURING HOT SHUTDOWN	3.000e-03	LOCA
1994/11/03	250/94-005	Turkey Point 3	LOAD SEQUENCERS PERIODICALLY INOPERABLE OVER 1 YR PERIOD	1.800e-06	ULOCA
1994/11/03	251/94-005	Turkey Point 4	LOAD SEQUENCERS PERIODICALLY INOPERABLE OVER 1 YR PERIOD	1.800e-06	ULOCA
1995/01/09	335/95-004	St Lucie 1	FAILED PRESSURIZER PORVS, RCP SEAL FAILURE, RELIEF VALVE FAILURE, PLUS OTHER PROBLEMS	9.300e-05	UTRIP
1995/01/19	368/95-001	Arkansas 2	LOSS OF DC BUS COULD FAIL BOTH EFW TRAINS	1.100e-05	UTRIP
1995/01/25	336/95-002	Millstone Pt 2	CONTAINMENT SUMP ISOLATION VALVES POTENTIALLY UNAVAILABLE DUE TO PRESSURE LOCKING	3.100e-05	ULOCA
1995/03/09	213/95-010	Haddam Neck	MULTIPLE SAFETY INJECTION VALVES ARE SUSCEPTIBLE TO PRESSURE LOCKING	4.700e-06	ULOCA
1995/04/20	313/95-005	Arkansas 1	REACTOR TRIP WITH ONE EFW TRAIN UNAVAILABLE	2.000e-05	TRIP
1995/06/10	382/95-002	Waterford 3	REACTOR TRIP, BREAKER FAILURE AND FIRE, AND DEGRADED SHUTDOWN COOLING	1.700e-05	TRIP
1995/06/11	445/95-003	Comanche Pk 1	REACTOR TRIP, AFW PUMP TRIP, SECOND AFW PUMP UNAVAIL	2.900e-05	TRIP
1995/09/01	315/95-011	Cook 1	ONE SAFETY INJECTION PUMP UNAVAIL FOR SIX MONTHS	7.700e-06	TRIP
1995/09/11	352/95-008	Limerick 1	SAFETY RELIEF VALVE FAILS OPEN, SCRAM, SUPPRESSION POOL STRAINER FAILS	9.000e-06	TRIP
1995/11/20	389/95-005	St Lucie 2	FAILURE OF ONE EDG WITH CCF IMPLICATIONS	1.300e-05	ULOOP

Table E.1 (Continued)

Event Date	LER No.	Plant	Precursor Title	CCDP or ΔCDP	Event Type
1996/01/10	272/96-002	Salem 1	CHARGING PUMP SUCTION VALVES FROM THE RWST POTENTIALLY UNAVAIL	5.800e-06	ULOCA
1996/01/10	311/96-002	Salem 2	CHARGING PUMP SUCTION VALVES FROM THE RWST POTENTIALLY UNAVAIL	5.800e-06	ULOCA
1996/01/30	482/96-001	Wolf Creek 1	REACTOR TRIP WITH A LOSS OF TRAIN A OF ESSENTIAL SERVICE WATER AND TURBINE-DRIVEN AFW PUMP	2.100e-04	TRIP
1996/02/06	414/96-001	Catawba 2	LOOP WITH EDG B UNAVAIL	2.100e-03	LOOP
1996/03/06	370/96-002	McGuire 2	2B EDG INOPERABLE FOR 2.2 MONTHS	1.800e-06	ULOOP
1996/05/19	313/96-005	Arkansas 1	REACTOR TRIP AND SUBSEQUENT STEAM GENERATOR DRYOUT	5.600e-04	USGTR
1996/05/21	443/96-003	Seabrook 1	TURBINE-DRIVEN EFW UNAVAIL	4.600e-05	ULOOP
1996/05/23	454/96-007	Byron 1	TRANSFORMER BUS FAULT CAUSES A LOOP DURING SHUTDOWN	1.700e-05	LOOP
1996/06/28	373/96-007	LaSalle 1	CONCRETE SEALANT FOULS COOLING WATER SYSTEMS	7.000e-06	TRIP
1996/06/28	374/96-007	LaSalle 2	CONCRETE SEALANT FOULS COOLING WATER SYSTEMS	7.000e-06	TRIP
1996/06/29	282/96-012	Prairie Island 1	LOOP TO SAFEGUARDS BUSES ON BOTH UNITS	5.300e-05	LOOP
1996/06/29	306/96-012	Prairie Island 2	LOOP TO SAFEGUARDS BUSES ON BOTH UNITS	5.300e-05	LOOP
1996/08/01	213/96-016	Haddam Neck	POTENTIALLY INADEQUATE RHR PUMP NPSH FOLLOWING A LARGE-OR MEDIUM BREAK LOCA	1.100e-04	ULOCA
1996/09/01	213/96-024	Haddam Neck	RHR PUMP UNAVAIL	2.900e-06	ULOCA
1997/01/22	309/97-004	Maine Yankee	RCS HOT-LEG RECIRC VLV SUBJECT TO PRESSURE LOCK	1.300e-05	ULOCA
1997/04/21	270/97-001	Oconee 2	UNISOLABLE RCS LEAK	2.200e-05	LOCA
1997/05/03	287/97-003	Oconee 3	TWO HIGH-PRESSURE INJECTION PUMPS DAMAGED FROM LOW WATER LEVEL IN LETDOWN STORAGE TANK	4.300e-06	ULOCA
1997/06/21	289/97-007	Three Mile Isl 1	FAILURE OF BOTH GENERATOR OUTPUT BREAKERS CAUSES LOOP	9.600e-06	LOOP
1997/11/02	335/97-011	St Lucie 1	NON-CONSERVATIVE RECIRCULATION ACTUATION SETPOINT	1.700e-05	ULOCA
1998/02/05	361/98-003	San Onofre 2	INOPERABLE SUMP RECIRCULATION VALVE	7.200e-06	ULOCA
1998/02/12	269/98-004	Oconee 1	CALIBRATION & CALCULATIONAL ERRORS COMPROMISE ECCS TRANSFER TO EMERGENCY SUMP	1.700e-06	ULOCA

Table E.1 (Continued)

Event Date	LER No.	Plant	Precursor Title	CCDP or ΔCDP	Event Type
1998/02/12	270/98-004	Oconee 2	CALIBRATION & CALCULATIONAL ERRORS COMPROMISE ECCS TRANSFER TO EMERGENCY SUMP	1.700e-06	ULOCA
1998/02/12	287/98-004	Oconee 3	CALIBRATION & CALCULATIONAL ERRORS COMPROMISE ECCS TRANSFER TO EMERGENCY SUMP	1.400e-06	ULOCA
1998/06/24	346/98-006	Davis-Besse 1	A TORNADO TOUCHDOWN CAUSES REACTOR TRIP AND LOOP	5.600e-04	LOOP
1998/07/14	155/98-001	Big Rock Point	LONG-TERM UNAVAILABILITY OF LIQUID POISON CONTROL SYSTEM	1.100e-05	UTRIP
1998/07/14	316/98-005	Cook 2	POTENTIAL FAILURE OF ALL COMPONENT COOLING WATER PUMPS DUE TO STEAM INTRUSION RESULTING FROM POSTULATED BREAK IN A UNIT 2 MAIN STEAM LINE	2.700e-06	USLB
1998/09/12	454/98-018	Byron 1	UNAVAILABILITY OF AN EDG FOR 18 DAYS	8.100e-06	ULOOP
1998/10/14	346/98-011	Davis-Besse 1	MANUAL REACTOR TRIP DUE TO COMPONENT COOLING SYSTEM LEAK AND DE-ENERGIZING OF BUSES	1.400e-05	TRIP
1999/02/24	269-99-001	Oconee 1	POSTULATED HIGH-ENERGY LINE LEAKS OR BREAKS LEADING TO FAILURE OF SAFETY-RELATED 4 KV SWITCHGEAR	8.200e-06	USLB
1999/02/24	269-99-001	Oconee 2	POSTULATED HIGH-ENERGY LINE LEAKS OR BREAKS LEADING TO FAILURE OF SAFETY-RELATED 4 KV SWITCHGEAR	5.600e-06	USLB
1999/02/24	269-99-001	Oconee 3	POSTULATED HIGH-ENERGY LINE LEAKS OR BREAKS LEADING TO FAILURE OF SAFETY-RELATED 4 KV SWITCHGEAR	5.200e-06	USLB
1999/06/11	315/99-S01[a]	Cook 1	LACK OF CAPABILITY TO OPERATE EMERGENCY SERVICE WATER FOLLOWING A SEISMIC EVENT	3.200e-05	UEQ
1999/06/11	316/99-S01[a]	Cook 2	LACK OF CAPABILITY TO OPERATE EMERGENCY SERVICE WATER FOLLOWING A SEISMIC EVENT	3.200e-05	UEQ
1999/08/31	247/99-015	Indian Point 2	LOOP TO SAFETY-RELATED BUSES FOLLOWING A REACTOR TRIP AND AN EDG OUTPUT BREAKER TRIP	2.800e-06[b]	LOOP

Note:

a. Inspection Report Nos. 50-315/316/97-024, and Nos. 50-315/316/99-010

b. See note to Table 3.1

Table H.2. Data used in the trending analysis.

	Total	Fiscal Year						
		FY-93	FY-94	FY-95	FY-96	FY-97	FY-98	FY-99
Reactor Calender Years		110.1	110.0	110.0	110.7	108.5	105.4	105.0
BWR		37.0	37.0	37.0	37.0	36.1	35.0	35.0
PWR		73.1	73.0	73.0	73.7	72.5	70.4	70.0
CCDP bins:								
10^{-3}	2	0	1	0	1	0	0	0
10^{-4}	13	7	1	0	3	0	1	0
10^{-5}	37	9	7	6	5	2	2	3
10^{-6}	26	4	3	5	6	2	6	4
Total Precursors ($\geq 1x10^{-6}$)	78	20	12	11	15	4	9	7
Significant Precursors ($\geq 1x10^{-3}$)	2	0	1	0	1	0	0	0
Important Precursors ($\geq 1x10^{-4}$)	15	7	2	0	4	0	1	0
Operational Event Type:								
Initiating Events	31	9	5	5	7	2	1	2
Conditional Unavailabilities	47	11	7	6	8	2	8	5
BWRs vs. PWRs:								
BWRs	11	4	3	1	2	0	1	0
PWRs	67	16	9	10	13	4	8	7
Loss of Offsite Power Initiator	14	5	2	0	4	1	1	1

Table E.3. Data used to calculate reactor calender years for trending analysis.

	Defuel/ Startup Date[a]	Reactor Calendar Days						
		FY-93	FY-94	FY-95	FY-96	FY-97	FY-98	FY-99
Decommissioned–PWR								
San Onofre 1	11/30/92	60	0	0	0	0	0	0
Trojan	11/09/92	39	0	0	0	0	0	0
Maine Yankee	6/23/97	365	365	365	365	265	0	0
Zion 1	4/28/97	365	365	365	365	209	0	0
Zion 2	2/26/98	365	365	365	365	365	148	0
Haddam Neck	12/5/96	365	365	365	365	65	0	0
Initial Startup–PWR								
Comanche Peak 2	8/3/93	306	365	365	365	365	365	365
Watts Barr	5/27/96	0	0	0	239	365	365	365
Operating–PWRs								
68 units x 365 days =		24820	24820	24820	24820	24820	24820	24820
Total PWR (reactor calendar years)		73.1	73.0	73.0	73.7	72.5	70.4	70.0
Decommissioned–BWR								
Big Rock	9/22/97	365	365	365	365	356	0	0
Millstone 1	10/31/96	365	365	365	365	30	0	0
Operating–BWRs								
35 units x 365 days =		12775	12775	12775	12775	12775	12775	12775
Total BWR (reactor calendar years)		37.0	37.0	37.0	37.0	36.1	35.0	35.0
TOTAL (PWR + BWR)		110.1	110.0	110.0	110.7	108.5	105.4	105.0

Notes:

a. Startup date from NUREG-1350, "Information Digest," (http://www.nrc.gov/NRC/NUREGS/SR1350/V12/index.html)
Defuel date from the NRC Status Reports (http://www.nrc.gov/NRR/DAILY/drlist.htm). Dates for San Onofre 1 and Trojan are shutdown dates from NUREG-1350.