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Accident Sequence Precursor Program

Results, Trends, and Insights (SECY-04-0210)

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Note: The results presented in this paper were reported in the annual Commission Paper, SECY-04-0210, “Status of the Accident Sequence Precursor (ASP) Program and the Development of Standardized Plant Analysis Risk (SPAR) Models,” November 8, 2004. (See ADAMS Accession No. ML0425803665.)

Results, Trends, and Insights from the Accident Sequence Precursor (ASP) Program

This attachment discusses the results of accident sequence precursor (ASP) analyses conducted by the U.S. Nuclear Regulatory Commission (NRC), as they relate to events that occurred during Fiscal Years (FYs) 2001–2004. These results were presented in SECY-04-0210 (Ref. 1). Based on those results, this document also discusses the NRC's analysis of historical ASP trends, and the evaluation of the related insights. The 11 tables and 14 figures that augment this discussion appear at the end of this attachment.

1.0 ASP Event Analyses

Table 1 summarizes the status of the NRC's ASP analyses, as of September 15, 2004. Specifically, the table identifies the number of preliminary and final analyses that the NRC staff has completed for events that occurred during each fiscal year (2001–2004), as well as the number of preliminary analyses that are still underway, which include events *that will be rejected as precursors*. (Note that, as of August 31, 2004, the staff had not yet screened all of the FY 2004 events and unavailabilities.) The following subsections summarize the results of these analyses, which are further detailed in the associated Tables 2–9.

FY 2001 analyses. The ASP analyses for FY 2001 identified 23 precursors. Of those 23 precursors, 18 were identified on the basis of final analyses, and 5 are expected to be precursors because they relate to events that involved cracking of the control rod drive mechanism (CRDM).[1] All 23 of these precursors occurred at power.

Table 2 presents the results of the staff's ASP analyses for FY 2001 precursors that involved initiating events, while Table 3 presents the analysis results for precursors that involved precipitating conditions. In addition, Table 4 lists the CRDM cracking events that occurred during FYs 2001–2003.

FY 2002 analyses. The ASP analyses for FY 2002 identified 14 precursors. Of those 14 precursors, 8 were identified on the basis of final analyses, 1 is a potential precursor based on preliminary analysis, and 5 are potential precursors (expected to be precursors) because they relate to CRDM cracking events. All 14 of these precursors occurred at power.

The staff has completed its preliminary analysis of the multiple conditions that occurred at the Davis-Besse Nuclear Power Station coincident with degradation of the reactor pressure vessel (RPV) head; the document has been issued for peer review. This event is a potential *significant* precursor.[2]

Table 5 presents the results of the staff's ASP analyses for FY 2002 precursors that involved initiating events, while Table 6 presents the analysis results for precursors that involved precipitating conditions. In addition, as previously noted, Table 4 includes CRDM cracking events that occurred during FY 2002.

FY 2003 analyses. In February 2004, the NRC staff completed its screening and review of licensee event reports (LERs) concerning events that occurred during FY 2003. On the basis of that review, the ASP analyses have (thus far) identified 10 precursors, including 2 based on final analyses and 8 based on preliminary analyses. All 10 of these precursors occurred at power. An additional 22 analyses are ongoing, but the results of some of these analyses will not exceed the precursor threshold.

Table 7 presents the results of the staff's ASP analyses for FY 2003 precursors that involved initiating events, while Table 8 presents the analysis results for precursors that involved precipitating conditions. The staff may identify additional precursors after completing the ongoing analyses of FY 2003 events.

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1 As of September 15, 2004, the staff has not completed its preliminary ASP analyses of CRDM cracking events that occurred during FYs 2001–2003. However, based on sensitivity analyses completed to date, the staff anticipates that these events will yield an increase in core damage probability (ΔCDP) that is greater than or equal to 1×10^{-6}.

2 A significant precursor has a conditional core damage probability (CCDP) or change in core damage probability (ΔCDP) that is greater than or equal to 1×10^{-3}.

FY 2004 analyses. The staff has completed all screening and reviews for potential *significant* precursors through September 15, 2004. In particular, the staff had reviewed a combination of LERs and daily event notification reports (as required by Title 10, Section 50.72, of the *Code of Federal Regulations*, 10 CFR 50.72) to identify potential *significant* precursors. The staff is still screening and reviewing LERs concerning other potential precursor events that occurred during FY 2004.

The staff has also completed three preliminary analyses for losses of offsite power (LOOPs) that occurred during FY 2004 at Palo Verde Nuclear Generating Station, Units 1, 2, and 3. Table 9 addresses the Unit 2 event within the context of important precursors for FYs 2001–2004.

2.0 Industry Trends

This section discusses the results of trending analyses for all precursors and for precursors grouped by the order of magnitude of their CCDPs or ΔCDPs (called CCDP bins).

Statistically significant trend. The trending method used in this analysis is consistent with those methods used in the staff's risk studies (See Appendix E of Reference 5.) The trending method uses *p-value* for determining the probability of observing a trend as a result of chance alone. A trend is considered *statistically significant* if the *p-value* is smaller than 0.05. The *p-value* is shown for each trend in the figure provided at the end of this attachment.

Data coverage. Most of the data used in the trending analyses span the period from FY 1993 through FY 2002. In addition, the trends include the results of both final and preliminary analyses of potential precursors. However, the following exceptions apply to the data coverage of the trending analyses:

- *Significant precursors (10^{-3} bin).* The trend of significant precursors (i.e., CCDP and ΔCDP $\geq 1 \times 10^{-3}$) includes events that occurred during FYs 2003 and 2004. The results for FY 2004 are based on the staff's screening and review of a combination of LERs and daily event notification reports (10 CFR 50.72).[3] The staff analyzes all potential significant precursors immediately.
- *CRDM cracking events.* The staff is still conducting its preliminary analyses of cracking that occurred in CRDM housings during FYs 2001 and 2002. Sensitivity analyses conducted to date show that these cracking events are most likely potential precursors but not significant presursors. Therefore, the staff has included these events in the total count and trending of all precursors (i.e., CCDP and ΔCDP $\geq 1 \times 10^{-6}$). However, the staff has not included these events in the CCDP bin trends because their exact ΔCDP values are not yet known.

2.1 Occurrence Rate of All Precursors

The NRC's Industry Trends Program (ITP) provides the basis for addressing the agency's performance goal measure of "no statistically significant adverse industry trends in safety performance" (one measure associated with the Safety goal established in the NRC's Strategic Plan). Although the principle measures come from the Reactor Oversight Program, NRC's Office of Nuclear Reactor Regulation (NRR) also uses precursors identified by the ASP Program to assess industry performance. Thus, the method used to trend precursors is consistent with the analysis used to trend the other indicators in the ITP.

Results. No statistically significant trend has been observed in the occurrence rate for all precursors that occurred during the period from 1993 through 2002. Figure 1 depicts the occurrence rate per reactor-year for all precursors by fiscal year.

Figure 1 also shows the ASP results for events that occurred before FY 1993, which were derived using a less-rigorous methodology but are shown to provide historical perspective.

Data coverage. The trend of all precursors includes the ongoing analyses of events that involved cracking in CRDM housings.

2.2 Occurrence Rate of Precursors by CCDP Bin

In addition to the rate of occurrence of all precursors, the staff analyzed the data to determine whether trends exist in the rate of occurrence of precursors with CCDPs of different orders of magnitude. The method used in this analysis is based on a staff technical paper presented at the International Topical Meeting on

[3] The staff has completed all screening and reviews through September 15, 2004.

Probabilistic Safety Assessment (See Reference 2.)

Figure 2a is a histogram displaying the number of precursors per fiscal year for the CCDP $\geq 10^{-3}$ bin. (Note that Figure 2a shows the number of precursors instead of the occurrence rate.) This figure does not show a trend line because the staff did not detect a statistically significant trend.

By contrast, Figures 2b–d are histograms of the occurrence rate as a function of fiscal year for the other three CCDP bins (10^{-4}, 10^{-5}, and 10^{-6}). Because these figures represent statistically significant trends, each figure shows the trend line of the mean occurrence rate, with the 90-percent confidence band indicated by error bars.

Results. The trending analysis of the four CCDP bins ($\geq 10^{-3}$, 10^{-4}, 10^{-5}, and 10^{-6}) yielded the following results for the period from FY 1993 through FY 2002:

CCDP Bin	Trend
$\geq 10^{-3}$	No statistically significant trend
10^{-4}	Decreasing trend - almost statistically significant
10^{-5}	Decreasing trend - statistically significant
10^{-6}	Increasing trend - statistically significant

While no trend was detected in the highest CCDP bin ($\geq 10^{-3}$) and trends in bins 10^{-4} and 10^{-5} are decreasing, an increasing trend was detected in the lowest CCDP bin (10^{-6}). The cause of this increasing trend will be investigated in an upcoming study. In FY 2005, the staff will initiate a detailed evaluation of the ASP data to investigate the nature of trends and identify insights that can be applied in the NRC's regulatory programs.

Data coverage. The trends of precursor bins do *not* include the ongoing analyses of events that involved cracking in CRDM housings. The trend of the CCDP $\geq 10^{-3}$ bin includes events for FYs 2003 and 2004. All other bins cover the period from FY 1993 through FY 2002.

3.0 Insights and Other Trends

The discussion of *significant* precursors in Section 3.1 covers the period from FY 1993 through FY 2004, although the FY 2004 results are based on the staff's screening and review of a combination of LERs and daily event notification reports (10 CFR 50.72).[4] Section 3.4, which addresses the LOOP initiating events, also covers the period from FY 1993 through FY 2004.[5] The insights presented in the remaining sections cover the period from FY 1993 through FY 2002.

3.1 Significant Precursors

The ASP Program provides the basis for the FY 2004 performance goal measure of "no more than one event per year identified as a *significant* precursor of a nuclear accident" (one measure associated with the Safety Goal established in the NRC's Strategic Plan). Specifically, the Strategic Plan defines a *significant* precursor as an event that has a probability of at least 1 in 1000 (10^{-3}) of leading to a reactor accident. (See Reference 3.) It should be noted that this performance goal was changed to zero events per year beginning in FY 2005.

Table 11 summarizes the top 20 *significant* precursors that occurred during the period from FY 1974 through FY 2004.

Results. Figure 2a depicts the number of *significant* precursors that occurred during FY 1993–2004. A review of the data for that period reveals the following insights:

- The staff did not identify any *significant* precursors during FYs 2001, 2003, and 2004.

- The multiple conditions coincident with degradation of the RPV head at Davis-Besse represent a potential *significant* precursor for FY 2002. The specific conditions included cracking of CRDM nozzles, degradation of the RPV head, potential clogging of the emergency sump, and potential degradation of the high-pressure injection (HPI) pumps.

- The performance goal measure of "no more

4 The staff has completed all screening and reviews through September 15, 2004.

5 FY 2004 includes LOOP events counted through September 15, 2004.

than one event per year identified as a *significant* precursor of a nuclear accident” has not been exceeded during the period from FY 1993 through FY 2004.

- The staff did not observe any statistically significant trend in the occurrence of *significant* precursors during FYs 1993–2004.
- *Significant* precursors have occurred, on average, about once every 4 years. The events in this group involve differing failure modes, causes, and systems.
- Two additional precursors with a CCDP $\geq 1 \times 10^{-3}$ have occurred in the past 12 years. Specifically, the event at Wolf Creek Generating Station (1994) involved a reactor coolant system (RCS) draindown to the refueling water storage tank during hot shutdown, while the event at Unit 2 of the Catawba Nuclear Station (1996) involved an extended, plant-centered LOOP with an emergency diesel generator (EDG) out of service for maintenance.

3.2 Important Precursors

Precursors with a CCDP or ΔCDP $\geq 1 \times 10^{-4}$ are considered *important* in the ASP Program. An *important* precursor generally has a CCDP higher than the core damage probability (CDP) estimated by most plant-specific probabilistic risk assessments (PRAs).

The staff identified three *important* precursors that occurred during FYs 2001 and 2002. By contrast, the staff’s preliminary analyses of plants affected by the power blackout in the Northeast United States in August 2003 identified five potential *important* precursors. In addition, the staff has preliminarily identified one potential *important* precursor for FY 2004.

The staff is continuing to analyze events that occurred during FYs 2003 and 2004, and these ongoing analyses may identify additional *important* precursors. Table 9 summarizes the *important* precursors identified so far.

Data coverage. Results summarized below do *not* include events that occurred during FYs 2003 and 2004.

Results. A review of the data for FYs 1993–2002 reveals the following insights:

- The mean occurrence rate of *important* precursors exhibited a *decreasing* trend that is *almost statistically significant* during the period from FY 1993 through FY 2002, as shown in Figure 3.
- *Important* precursors occur infrequently (about two per year on average).
- Twenty-one *important* precursors occurred during the period from FY 1993 through FY 2002 period. Of these, 33 percent involved a LOOP initiating event.

3.3 Initiating Events vs. Conditions

A precursor can be the result of either (1) an operational event involving an actual initiating event such as a LOOP, or (2) a condition found during a test, inspection, or engineering evaluation. A condition involves a reduction in safety system reliability or function for a specific duration (although no reactor trip initiator actually occurred during this time).

Results. A review of the data for FYs 1993–2002 reveals the following insights:

- Over the past 10 years, conditions outnumbered initiating events (73 percent compared to 27 percent, respectively). This predominance was most notable in FYs 2001 and 2002, when conditions contributed to 91 percent and 100 percent of the identified precursors, respectively.
- The mean occurrence rate of precursors involving initiating events has exhibited a *decreasing* trend that is statistically significant for the period from FY 1993 through FY 2002, as shown in Figure 4. Specifically, the occurrence rate of such precursors decreased over this period by a factor of seven.
- The mean occurrence rate of precursors involving conditions has exhibited an *increasing* trend that is statistically significant for the period from FY 1993 through FY 2002, as shown in Figure 5. Specifically, the occurrence rate of such precursors increased over this period by a factor of two. As discussed in Section 2.2, above, the nature of increasing trends will be investigated in an up coming study.

3.4 Precursors Involving Loss of Offsite Power Initiating Events

The LOOP event at Quad-Cities Station Unit 2, which was attributable to a failure of

the main power transformer, was the only precursor involving an initiating event during FY 2001. No LOOP events occurred during FY 2002.

In FY 2003, the power blackout in the Northeast United States in August 2003 caused nine plants to lose offsite power, and the staff's preliminary analyses identified eight of those events as potential precursors.[6] Three additional LOOP events occurred during FY 2003. The staff is continuing its preliminary analyses of those events, which occurred at Palisades Nuclear Power Plant and Units 2 and 3 of the Peach Bottom Atomic Power Station.

As of September 15, 2004, six LOOP events have occurred during FY 2004. The staff has completed its preliminary analyses of the LOOP events at Palo Verde Units 1, 2, and 3, but is still conducting the remaining analyses of the events at Vermont Yankee Generating Station, Unit 2 of the Brunswick Steam Electric Plant, and Unit 3 of the Dresden Nuclear Power Station.

Data coverage. Results summarized below include LOOP events for FYs 2003 and 2004. Although the staff is still analyzing six LOOP events that occurred during FY 2003 and 2004, experience has shown that these LOOP events are most likely precursors.

Results. A review of the data for FYs 1993–2004 reveals the following insights:

- The mean occurrence rate of precursors resulting from a LOOP has exhibited an *increasing* trend that is statistically significant for the period from FY 1993 through FY 2004, as shown in Figure 6. Specifically, the occurrence rate of such precursors increased over this period by a factor of three.
- Without the LOOP events that occurred as a result of the blackout in the Northeast United States on August 14, 2003, the identified precursors did not exhibit any statistically significant trend (either increasing or decreasing) for the period from FY 1993 through FY 2004. The review of the LOOP events associated with the Northeast blackout is the focus of the staff's action plan for resolving U.S. nuclear power plant issues relating to the electric power grid concerns.
- Approximately one-half (48 percent) of the LOOP precursor events that occurred during FYs 1993–2004 were evaluated to be *important* precursors (CCDP$\geq$1.0x10^{-4}).
- A simultaneous unavailability of an emergency power system train was involved in 4 of the 34 LOOP precursor events during FYs 1993–2004. Specifically, those four events involved a safety bus at Palo Verde Unit 2, which failed to sequence loads (2004); an EDG at Peach Bottom Unit 3, which tripped about 2.5 hours into the LOOP event as a result of low jacket coolant pressure (2003); an output breaker to an EDG at Indian Point Station Unit 2, which tripped open after closing (1999); and an EDG that was out of service for maintenance at Catawba Unit 2 (1996). Three of these four precursors had a CCDP >1x10^{-4}.

3.5 Precursors at Boiling- vs. Pressurized-Water Reactors

Five precursors in FY 2001 and one in FY 2002 occurred at a boiling-water reactor (BWR). The precursor counts for pressurized-water reactors (PWRs) include the ongoing analyses of events involving cracking in CRDM housings.

A review of the data for FYs 1993–2002 reveals the following results for BWRs and PWRs:

BWRs

- The mean occurrence rate of precursors at BWRs does not exhibit a trend that is statistically significant for the period from FY 1993 through FY 2002, as shown in Figure 7.
- No precursors occurred at BWRs during the 4-year period from FY 1997 through FY 2000.

PWRs

- The mean occurrence rate of precursors at PWRs does not exhibit a trend that is

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6 The ASP analysis of the LOOP event at Davis-Besse on August 14, 2003, showed that this event did not meet the threshold of a precursor in the ASP Program. (The CCDP was less than 1 x 10^{-6}.) The plant had been shut down for more than two years before this event occurred.

statistically significant for the period from FY 1993 through FY 2002, as shown in Figure 8.

- Historically, an average of 11 precursors per year occurred at PWRs during FYs 1993–2002.

3.6 Precursors Caused by Unavailability of Safety-Related Equipment[7]

Most precursors involve the unavailability of safety-related equipment. Such events typically occur during periods of extended unavailability of equipment without a reactor trip, or in combination with a reactor trip in which a risk-important component is unable to perform its safety function as a result of an unavailability condition.

A review of the data for FYs 1993–2002 reveals the following insights concerning the unavailability of safety-related equipment:

Equipment unavailabilities at BWRs

- Nine precursors involving the unavailability of safety-related equipment occurred at BWRs during FYs 1993–2002. The events in this group involved various failure modes, causes, and systems.

Emergency core cooling systems

- An unavailability of safety-related high- and/or low-pressure injection trains contributed to 58 percent of all identified precursors that occurred at PWRs during FYs 1993–2002. Most of these unavailabilities were caused by failures in either the emergency core cooling system (ECCS) (29 percent) or emergency power sources (24 percent), or resulted from design-basis issues involving other structures or systems that impact either the ECCS or one of its support systems (29 percent).

- The 19 precursors that involved a failure in an ECCS train yield the following insights:

  – Eighteen precursors involved a conditional unavailability that was identified during testing, inspection, or engineering reviews.

  – Fourteen precursors involved a condition that affected sump recirculation during postulated loss-of-coolant accidents of varying break sizes.

Auxiliary/emergency feedwater systems

- The unavailability of one or more trains of the auxiliary and emergency feedwater (AFW/EFW) systems contributed to 44 percent of all precursors that occurred at PWRs. Most of these unavailabilities were caused by failures in the AFW/EFW systems (24 percent) or emergency power sources (44 percent), or resulted from design-basis issues involving other structures or systems that impact either the AFW/EFW systems or one of their support systems (32 percent).

- The 12 precursors that involved a failure in an AFW/EFW train yield the following insights:

  – Five of the train failures occurred following a reactor trip.

  – Ten of the precursors involved the unavailability of the turbine-driven AFW/EFW pump train.

Emergency power sources in PWRs

- The unavailability of emergency power sources,[8] such as EDGs and hydroelectric generators (at Oconee), contributed to 26 percent of all precursors that occurred at PWRs. Most of these unavailabilities were caused by random hardware failures in the emergency power system (57 percent). The other unavailabilities were attributable to design-basis issues (23 percent) and losses of service water (23 percent).

- All LOOP events at PWRs had operable turbine-driven AFW pumps.

Section 3.4 (above), discusses insights related to precursors that involved a LOOP with a simultaneous EDG unavailability.

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[7] The sum of percentages presented in this section does not always equal 100-percent because some precursors involve multiple equipment unavailabilities.

[8] Not all EDG unavailabilities are precursors. An EDG unavailability for a period of less than one surveillance test cycle (1 month) is screened out in the ASP Program (assuming no other complications). In addition, the risk contributions of EDG unavailabilities vary plant-to-plant and may result in a ΔCDP less than the threshold of a precursor (1×10^{-6}). Reference 4 provides a detailed engineering analysis of EDG unavailabilities.

3.7 Causes of Precursors Involving Conditions

Precursors involving conditional unavailability of safety-related systems and components

are attributable to numerous causes. For the purposes of this review, the staff classified ASP data into five causal categories, including (1) design-basis issues, (2) hardware/material failures, (3) procedure errors, (4) maintenance deficiencies, and (5) other personnel/human errors that were not related to categories 1–4.

Design-basis issues arise when the design of plant structures, systems, and/or components deviates from the regulatory requirements and assumptions used in safety analyses.

Results. Figure 9a depicts the distribution of precursor causes within the five categories, while Figure 9b plots precursors related to *design-basis issues*. A review of the data for FYs 1993–2002 reveals the following insights:

- More than half (55 percent) of all identified precursors related to *design-basis issues,* and almost half of those had a ΔCDP $\geq 1\text{x}10^{-5}$.
- Human-related deficiencies (i.e., procedures, maintenance, and other human errors) accounted for 30 percent of the identified condition-related precursors. Random hardware failures account for the remaining 70 percent of condition-related precursors.

3.8 Annual ASP Index

The staff derives the annual ASP index for order-of-magnitude comparisons with industry-average core damage frequency (CDF) estimates derived from PRAs and individual plant examinations (IPEs). The index for a given fiscal year is the sum of the CCDPs divided by the number of reactor-years (RYs).

Results. Figure 10 depicts the annual ASP indices for FYs 1993–2002, with indices prior to FY 1993 provided for historical context. A review of the ASP indices reveals the following insights:

- Based on order of magnitude, the average ASP index for the period from FY 1993 through FY 2002 is consistent with the CDF estimates from the IPEs.[9]
- The increase in the ASP index for FY 2002 is attributable to the multiple conditions coincident with degradation of the RPV head at Davis-Besse. Both the preliminary ASP analysis results and the associated index are subject to change based on peer review comments.[10]

Limitations. Using CCDPs from ASP results to estimate CDF is difficult because (1) the mathematical relationship requires a significant level of detail, (2) statistics for frequency of occurrence of specific precursor events are sparse, and (3) the assessment must also account for events and conditions that did not meet the ASP precursor criteria.

The ASP models and process do not explicitly address all CDF scenarios, such as fires, flooding, and external events. Thus, they are incomplete for use in estimating total CDF. In addition, using CCDP can overestimate the CDF because of double counting.

Because of these and other limitations, the staff has primarily used the CCDPs as a relative trending indication. Nonetheless, ASP results can be linked to CDF by using an annual ASP index. The IPEs also give incomplete estimates of total CDF, although the IPEs are reasonably similar in scope to the current ASP Program.

3.9 Consistency with PRAs and IPEs

A secondary objective of the ASP Program is to provide a partial validation of the dominant core damage scenarios predicted by PRAs and IPEs. Most of the identified precursor events are consistent with failure combinations identified in PRAs and IPEs. However, a review of the precursor events for FYs 1994–2002 reveals that approximately 20 percent of the identified precursors involved event initiators or failure modes that were not explicitly modeled in the PRA or IPE concerning the specific plant at which the precursor event occurred. Table 10 lists

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9 The CDF estimates in the IPEs range from 1 $\text{x}10^{-6}$/RY to 3 $\text{x}10^{-4}$/RY, with an average value of 6 $\text{x}10^{-5}$/RY.

10 All preliminary results and subsequent indices are subject to change. The indices for FYs 2001 and 2002 also include the CRDM cracking events, for which the staff used a ΔCDP value of 5 $\text{x}10^{-5}$ for each event.

these precursors. The occurrence of these precursors do not imply that explicit modeling is needed; however, there could be insights that could be fed-back to future revisions of the PRA.

4.0 References

1. U.S. Nuclear Regulatory Commission. "Status of the Accident Sequence Precursor (ASP) Program and the Development of Standardized Plant Analysis Risk (SPAR) Models," Commission paper SECY-04-0210, November 8, 2004. (ADAMS Accession No. ML0425803665.)

2. Rasmuson, D.M., and P.D. O'Reilly. "Analysis of Annual Accident Sequence Precursor Occurrence Rates for 1984–94," in *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.

3. U.S. Nuclear Regulatory Commission. NUREG-1100, Vol. 20, "Performance Budget, Fiscal Year 2005." NRC: Washington, DC. February 2004.

4. U.S. Nuclear Regulatory Commission. NUREG/CR-5500, Vol. 5, "Reliability Study: Emergency Diesel Generator Power System, 1987–1993." NRC: Washington, DC. September 1999.

5. U.S. Nuclear Regulatory Commission. NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995." NRC: Washington, DC. February 1999.

Table 1. Status of ASP analyses as of September 15, 2004.

Status	FY 2001	FY 2002	FY 2003	FY 2004[a]
Final analysis completed	18	8	2	0
Preliminary analysis completed	0	1	8	3
Preliminary analysis underway *(includes events that may be precursors)*	5	5	22	7[a]

a. As of September 15, 2004, the staff has not yet screened all of the FY 2004 events and unavailabilities.

Table 2. FY 2001 at-power precursors involving initiating events (as of September 15, 2004).

Plant	Description/Event Identifier	Plant Type	Event Date	CCDP
Quad-Cities 2	Reactor scram attributable to failure of main power transformer. ***LER # 265/01-001***	BWR	8/2/01	$5x10^{-6}$
LaSalle 2	Reactor scram attributable to actuation of the under-voltage protective circuit on the Division 1 Engineered Safety Feature (ESF) Bus. ***LER # 374/01-003***	BWR	9/3/01	$1x10^{-5}$

Table 3. FY 2001 at-power precursors involving conditional unavailabilities (as of September 15, 2004).

Plant	Description/Event Identifier	Plant Type	Event Date	Importance (CCDP-CDP)
Oconee 1, 2, & 3	Non-seismic 16-inch fire system piping header transited through the auxiliary building and posed a potential flooding problem if the piping ruptured during a seismic event. ***IR # 269/01-08***	PWR	11/1/00	$5x10^{-6}$ (Unit 1) $4x10^{-6}$ (Units 2 & 3)
Prairie Island 1 & 2	Bearing lubrication for the cooling water pumps degraded following a LOOP. ***LER # 282/00-004***	PWR	11/1/00	$1x10^{-6}$
Limerick 2	A manual scram occurred because a main steam relief valve (MSRV) failed open as a result of erosion and oxidation of the first stage pilot valve disk seating area. ***LER # 353/01-001***	BWR	2/23/01	$3x10^{-6}$
Fermi 2	EDG 14 was inoperable for more than the time allowed by the Technical Specifications(7 days). ***LER # 341/01-001***	BWR	3/28/01	$3x10^{-6}$
Kewaunee	The licensee failed to provide a fixed fire suppression system for Fire Area TU-95B, and this could result in a postulated lube oil fire involving the AFW pump. ***IR # 305/02-06***	PWR	3/28/01	$1x10^{-5}$
Surry 1 and 2	EDG failed as a result of insufficient lubrication. ***LER # 280/01-001***	PWR	4/15/01	$3x10^{-6}$ (Unit 1) $6x10^{-6}$ (Unit 2)
Calvert Cliffs 1	AFW pump turbine bearing failed as a result of steam intrusion. ***LER # 317/01-001***	PWR	5/16/01	$1x10^{-5}$
Dresden 3	The high-pressure coolant injection (HPCI) system was inoperable following a water hammer event. ***LER # 249/02-005***	BWR	7/5/01	$3x10^{-6}$
Palisades	Smoke detectors in the cable room were not installed in accordance with code. ***LER # 255/01-008***	PWR	7/27/01	$1x10^{-6}$
ANO 1	Fire protection and procedures were inadequate for the north switchgear room, Fire Zone 99-M. ***IR # 313/01-06***	PWR	8/3/01	$4x10^{-6}$
Cook 1 & 2	Degraded ESW flow rendered both Unit 2 EDGs inoperable. ***LER # 316/01-003***	PWR	8/29/01	$1x10^{-5}$ (Unit 1) $7x10^{-6}$ (Unit 2)

Table 4. FYs 2001–2003 CRDM cracking events.[a]

Plant	Event Date	Description/Event Identifier
Oconee 1	12/4/00	RPV head leakage resulted from primary water stress corrosion cracking (PWSCC) of one CRDM nozzle. ***LER # 269/00-006***
Oconee 3	2/18/01	RPV head leakage resulted from PWSCC of nine CRDM nozzles. ***LER # 287/01-001***
ANO 1	3/24/01	RPV head leakage resulted from PWSCC of one CRDM nozzle. ***LER # 313/01-002***
Oconee 2	4/28/01	RPV head leakage resulted from PWSCC of four CRDM nozzles. ***LER # 270/01-002***
Palisades	6/21/01	RPV head leakage resulted from PWSCC of one CRDM nozzle. ***LER # 255/01-004***
Crystal River	10/1/01	RPV head leakage resulted from PWSCC of one CRDM nozzle. ***LER # 302/01-004***
TMI 1	10/12/01	RPV head leakage resulted from PWSCC of five CRDM nozzles. ***LER # 289/01-002***
Surry 1	10/28/01	RPV head leakage resulted from PWSCC of two CRDM nozzles. ***LER # 280/01-003***
North Anna 2	11/13/01	RPV head leakage resulted from PWSCC of one CRDM nozzle. ***LER # 339/01-003***
Davis-Besse	2/27/02	Cracking of CRDM nozzles, RPV head degradation, potential clogging of the emergency sump, and potential degradation of the HPI pumps. ***LER # 346/02-002***
St. Lucie 2	4/3/02	RPV head leakage resulted from PWSCC of one CRDM nozzle. ***LER # 389/03-002***

a. The staff issued the preliminary analysis of Davis-Besse for peer review in September 2004. The analyses of cracking events at the remaining plants are ongoing. The risk associated with multiple cracks at a given plant will be considered collectively in one analysis for each plant (i.e., only one precursor for each plant)

Table 5. FY 2002 at-power precursors involving initiating events (as of September 15, 2004).

Plant	Description/Event Identifier	Plant Type	Event Date	CCDP
	None			

Table 6. FY 2002 at-power precursors involving conditional unavailabilities (as of September 15, 2004).

Plant	Description/Event Identifier	Plant Type	Event Date	Importance (CCDP-CDP)
Harris	Debris accumulated in suction lines to the “A” residual heat removal (RHR) pump and “A” containment spray pump. ***LER # 400/01-003***	PWR	10/8/01	6×10^{-6}
Point Beach 1 & 2	Potential common-mode failure of all AFW pumps. ***LER # 266/01-005***	PWR	11/29/01	7×10^{-4} (Units 1 & 2)
Callaway	Foreign object rendered the “B” emergency service water (ESW) pump inoperable, and foreign material in the condensate storage tank (CST) caused a failure of the “A” AFW pumps. ***LER # 483/01-002***	PWR	12/3/01	2×10^{-5}
Davis-Besse[a]	Cracking of CRDM nozzles, RPV head degradation, potential clogging of the emergency sump, and potential degradation of the HPI pumps. ***LER # 346/02-002***	PWR	2/27/02	6×10^{-3}
Braidwood 1	The bleed path for the power-operated relief valve (PORV) was inoperable because of leaking accumulator check valves. ***LER # 456/02-002***	PWR	4/16/02	4×10^{-6}
Columbia 2	Four safety-related systems had unreliable breakers. ***IR # 397/02-05***	BWR	4/25/02	6×10^{-6}
Oconee 3	The emergency power supply cable from the auxiliary service water switchgear to the HPI pump was inadequately installed. ***IR # 247/02-15***	PWR	5/30/02	9×10^{-6}
Indian Point 2	Moderate degradation of the control room west wall could allow smoke and gases to penetrate the control room in the event of a turbine building fire. ***IR # 247/02-10***	PWR	7/19/02	7×10^{-6}

a. Preliminary analysis results are subject to change.

Table 7. FY 2003 at-power precursors involving initiating events (as of September 15, 2004).[a]

Plant	Description/Event Identifier	Plant Type	Event Date	CCDP
Fermi 2[b]	Reactor trip and loss of offsite power occurred as a result of the power blackout in the Northeast United States on August 14, 2003. ***LER # 314/03-002***	BWR	8/14/03	$2x10^{-4}$
Fitzpatrick[b]	Reactor trip and loss of offsite power occurred as a result of the power blackout in the Northeast United States on August 14, 2003. ***LER # 333/03-001***	BWR	8/14/03	$9x10^{-5}$
Ginna[b]	Reactor trip and loss of offsite power occurred as a result of the power blackout in the Northeast United States on August 14, 2003. ***LER # 244/03-002***	PWR	8/14/03	$2x10^{-4}$
Indian Point 2 & 3[b]	Reactor trip and loss of offsite power occurred as a result of the power blackout in the Northeast United States on August 14, 2003. ***LER # 247/03-005***	PWR	8/14/03	$1x10^{-4}$ (Unit 2)
				$7x10^{-5}$ (Unit 3)
Nine Mile Point 1 & 2[b]	Reactor trip and loss of offsite power occurred as a result of the power blackout in the Northeast United States on August 14, 2003. ***LER # 220/03-002***	BWR	8/14/03	$3x10^{-5}$ (Unit 1)
				$5x10^{-4}$ (Unit 2)
Perry[b]	Reactor trip and loss of offsite power occurred as a result of the power blackout in the Northeast United States on August 14, 2003. ***LER # 440/03-002***	BWR	8/14/03	$5x10^{-4}$

a. Other analyses are ongoing.
b. Preliminary analysis results are subject to change.

Table 8. FY 2003 at-power precursors involving conditional unavailabilities (as of September 15, 2004).[a]

Plant	Description/Event Identifier	Plant Type	Event Date	Importance (CCDP-CDP)
Point Beach 1 & 2	A design deficiency in the AFW orifices could cause debris plugging of the pumps' recirculation line, and subsequent common-cause failure of all AFW pumps. ***LER # 266/02-003***	PWR	10/29/02	$6x10^{-5}$ (Unit 1)
				$4x10^{-4}$ (Unit 2)

a. Other analyses are ongoing.

Table 9. FYs 2001–2004 important precursors (as of September 15, 2004).

Plant	Description/Event Identifier	Event Date	CCDP
Point Beach 1 & 2	This condition involved a design deficiency in the air-operated minimum-flow recirculation valves of the AFW pumps. The valves fail closed on loss of instrument air, and this could potentially lead to pump deadhead conditions and a common-mode, non-recoverable failure of the AFW pumps. Because the pressurizer PORVs also depend on instrument air, an event involving a loss of instrument air may also result in the loss of feed-and-bleed cooling capability. ***LER # 266/01-005***	11/29/01	$7x10^{-4}$ (Both Units)
Davis-Besse[a]	Cracking of CRDM nozzles, RPV head degradation, potential clogging of the emergency sump, and potential degradation of the HPI pumps. ***LER # 346/02-002***	2/27/02	$6x10^{-3}$
Point Beach 2	This condition involved a design deficiency in the flow-restricting orifices in the recirculation lines of the AFW pumps. Because of this design deficiency, the orifices are vulnerable to debris plugging when the suction supply for the AFW pumps is switched to its safety-related water supply (the service water system). Blocked flow in the recirculation lines of the AFW pumps, combined with inadequacies in plant emergency operating procedures, could potentially lead to pump deadhead conditions and a common-mode, non-recoverable failure of the pumps. The mean ΔCDP was $6x10^{-5}$ for Unit 1. ***LER # 266/02-003***	10/29/02	$4x10^{-4}$
Northeast Blackout Fermi 2[a] Ginna[a] Indian Point 2[a] Nine Mile Point 2[a] Perry[a]	Reactor trip and loss of offsite power resulted from the power blackout in the Northeast United States on August 14, 2003. The plant-to-plant variations in CCDP are primarily attributable to the varying durations of the LOOP at each site, minor problems with mitigating systems in several plants, and design differences among the plants. The offsite power recovery times used in the ASP analyses are based on the times at which the grid control centers gave permission to use the power, as reported in the LERs and information compiled by the NRC's regional offices. The related ASP analyses also considered any additional time to get power from the switchyard to a safety bus, and the probability of failing to successfully restore the power. An important plant design feature with respect to the risk of station blackout is the time to battery depletion. Other important design features include the configuration of EDGs and alternative power sources, and the availability of turbine-powered mitigating systems. ***See Table 7 for LER numbers.***	8/14/03	$2x10^{-4}$ $2x10^{-4}$ $1x10^{-4}$ $5x10^{-4}$ $5x10^{-4}$
Palo Verde 2[a]	A ground fault in the electrical grid resulted in losses of offsite power to all three units at Palo Verde. With the exception of one EDG in Unit 2, all EDGs started and loaded onto engineered safeguard buses. (The Unit 2 Train "A" buses had been de-energized as a result of the loss of the EDG.) The CCDPs for Units 1 and 3 are $4x10^{-5}$. ***IR # 528/04-12***	6/14/04	$7x10^{-4}$

a. Preliminary analysis results are subject to change.

Table 10. Precursors involving failure modes and event initiators that were not explicitly modeled in the PRA or IPE concerning the specific plant at which the precursor event occurred.

Plant	Year	Event Description
Columbia 2	2002	Common-cause failure (CCF) of breakers used in four safety-related systems. ***IR # 397/02-05***
Davis-Besse	2002	Cracking of control rod drive mechanism nozzles and reactor pressure vessel head degradation, potential clogging of the emergency sump, and potential degradation of the high-pressure injection pumps. ***LER # 346/02-002***
Callaway	2002	Potential common mode failure of all auxiliary feedwater pumps due to foreign material in the condensate storage tank caused by degradation of the floating bladder. ***LER # 483/01-002***
Point Beach 1 & 2	2002	Potential common mode failure of all auxiliary feedwater (AFW) pumps due to a design deficiency in the AFW pumps' air-operated minimum flow recirculation valves. The valves fail closed on loss of instrument air and this could potentially lead to pump deadhead conditions and a common mode, non-recoverable failure of the AFW pumps. ***LER # 266/01-005***
Harris	2002	Potential failure of residual heat removal pump 'A' and containment spray pump "A" due to debris in the pumps' suction lines. ***LER # 400/01-003***
Kewaunee	2001	Failure to provide a fixed fire suppression system could result in a postulated fire that propagates and causes the loss of control cables in both safe shutdown trains. ***IR # 305/02-06***
Prairie Island 1 & 2	2000	A 1988 change in the backwash system for the cooling water pump drive shaft bearing lubrication water supply system could result in loss of plant cooling water during postulated loss-of-offsite-power conditions. ***LER # 282/00-004***
Oconee 1, 2, & 3	2000	Non-seismic 16-inch fire system piping header transited through the auxiliary building and posed a potential flooding problem should the piping rupture during a seismic event. ***IR # 269/00-08***
Cook 1 & 2	1999	Postulated high-energy line leaks or breaks in turbine building leading to failure of multiple safety-related equipment. ***LER # 315/99-026***
Oconee 1, 2, & 3	1999	Postulated high-energy line leaks or breaks in turbine building leading to failure of safety-related 4 kV switchgear. ***LER # 269/99-001***
Cook 2	1998	Postulated high-energy line break in turbine building leading to failure of all component cooling water pumps. ***LER # 316/98-005***
Oconee 1, 2, & 3	1998	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. ***LER # 269/98-004***
Haddam Neck	1996	Potentially inadequate residual heat removal pump net positive suction head following a large- or medium-break loss-of-coolant accident due to design errors. ***LER # 213/96-016***
LaSalle 1 & 2	1996	Fouling of the cooling water systems due to concrete sealant injected into the service water tunnel. ***LER # 373/96-007***
Wolf Creek	1996	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 with concurrent unavailability of the turbine-driven auxiliary feedwater pump. ***LER # 482/96-001***
Wolf Creek	1994	Blowdown of the reactor coolant system to the refueling water storage tank during hot shutdown. ***LER # 482/94-013***

Table 11. Significant (CCDP $\geq 10^{-3}$) accident sequence precursors during the 1969–2004 period—ordered by event date. *(See note.)*

Plant	CCDP	Date	Description
Davis-Besse	6×10^{-3} Preliminary	2/27/02	***Multiple conditions coincident with reactor pressure vessel (RPV) head degradation*** The analysis included multiple degraded conditions discovered on various dates. These conditions included cracking of control rod drive mechanisms (CRDM) nozzles and reactor pressure vessel (RPV) head degradation on February 27, 2002; potential clogging of the emergency sump on September 4, 2002; and potential degradation of the high-pressure injection (HPI) pumps on October 22, 2002. ***LER # 346/02-002***
Catawba 2	2×10^{-3}	2/6/96	***Loss of offsite power (LOOP) with an emergency diesel generator (EDG) B unavailable*** When the reactor was at hot shutdown, a transformer in the switchyard shorted out during a storm, causing breakers to open and resulting in a LOOP event. Although both EDGs started, the output breaker of EDG “1B” to essential bus “1B” failed to close on demand, leaving bus “1B” without AC power. After 2 hours and 25 minutes, operators successfully closed the EDG “1B” output breaker. ***LER # 414/96-001***
Wolf Creek 1	3×10^{-3}	9/17/94	***Reactor coolant system (RCS) blowdown to refueling water storage tank (RWST)*** When the plant was in cold shutdown, operators implemented two unpermitted simultaneous evolutions, which resulted in the transfer of 9,200 gallons (34,825 liters) of RCS inventory to the RWST. Operators immediately diagnosed the problem and terminated the event by closing the residual heat removal (RHR) cross-connect motor-operated valve (MOV). The temperature of the RCS increased by 7 °F (4 °C) as a result of this event. ***LER # 482/94-013***
Harris 1	6×10^{-3}	4/3/91	***HPI unavailability for one refueling cycle*** A degraded condition resulted from relief valve and drain line failures in the alternative minimum flow systems for the charging/safety injection pumps, which would have diverted a significant amount of safety injection flow away from the reactor coolant system. The root cause of the degradation is believed to have been water hammer, as a result in air left in the alternative minimum flow system following system maintenance and test activities. ***LER # 400/91-008***
Turkey Point 3	1×10^{-3}	12/27/86	***Turbine load loss with trip; control rod drive (CRD) auto insert fails; manual reactor trip; power operated relief valve (PORV) sticks open*** The reactor was tripped manually following a loss of turbine governor oil system pressure and the subsequent rapid electrical load decrease. Control rods failed to insert automatically because of two cold solder joints in the power mismatch circuit. During the transient, a PORV opened but failed to close (the block valve had to be closed). The loss of governor oil pressure was due to a cleared orifice blockage and the auxiliary governor dumping control oil. ***LER # 250/86-039***

Plant	CCDP	Date	Description
Catawba 1	$3x10^{-3}$	6/13/86	***Chemical and volume control system (CVCS) leak (130 gpm) from the component cooling water (CCW)/CVCS heat exchanger joint (i.e., small-break loss-of-coolant accident (LOCA))*** A weld break on the letdown piping, near the CCW/CVCS heat exchanger caused excessive RCS leakage. A loss of motor control center (MCC) power caused the variable letdown orifice to fail open. The weld on the 1-inch (2.54-cm) outlet flange on the variable letdown orifice failed as a result of excessive cavitation-induced vibration. This event was a small-break LOCA. ***LER # 413/86-031***
Davis-Besse 1	$1x10^{-2}$	6/9/85	***Loss of feedwater; scram; operator error fails auxiliary feedwater (AFW); PORV fails open*** While at 90-percent power, the reactor tripped with main feedwater (MFW) pump "1" tripped and MFW pump "2" unavailable. Operators made an error in initiating the steam and feedwater rupture control system and isolated AFW to both steam generators (SGs). The PORV actuated three times and did not reseat at the proper RCS pressure. Operators closed the PORV block valves, recovered AFW locally, and used HPI pump "1" to reduce RCS pressure. ***LER # 346/85-013***
Hatch 1	$2x10^{-3}$	5/15/85	***Heating, ventilation, and air conditioning (HVAC) water shorts panel; safety relief valve (SRV) fails open; high-pressure coolant injection (HPCI) fails; reactor core isolation cooling (RCIC) unavailable*** Water from an HVAC vent fell onto an analog transmitter trip system panel in the control room (the water was from the control room HVAC filter deluge system which had been inadvertently activated as a result of unrelated maintenance activities). This resulted in the lifting of the SRV four times. The SRV stuck open on the fourth cycle initiating a transient. Moisture also energized the HPCI trip solenoid making HPCI inoperable. RCIC was unavailable due to maintenance. ***LER # 321/85-018***
Lasalle 1	$2x10^{-3}$	9/21/84	***Operator error causes scram; RCIC unavailable; RHR unavailable*** While at 23-percent power, an operator error caused a reactor scram and MSIV closure. RCIC was found to be unavailable during testing (one RCIC pump was isolated and the other pump tripped during the test). RHR was found to be unavailable during testing due to an inboard suction isolation valve failing to open on demand. Both RHR and RCIC may have been unavailable after the reactor scram. ***LER # 373/84-054***
Salem 1	$5x10^{-3}$	2/25/83	***Trip with automatic reactor trip capability failed*** When the reactor was at 25-percent power, both reactor trip breakers failed to open on demand of a low-low SG level trip signal. A manual trip was initiated approximately 3 seconds after the automatic trip breaker failed to open, and was successful. The same event occurred 3 days later, at 12-percent power. Mechanical binding of the latch mechanism in the breaker under-voltage trip attachment failed both breakers in both events. ***LER # 272/83-011***

Plant	CCDP	Date	Description
Davis Besse 1	$2x10^{-3}$	6/24/81	***Loss of vital bus; failure of an AFW pump; main steam safety valve lifted and failed to reseat*** With the plant at 74-percent power, the loss of bus "E2" occurred due to a maintenance error during CRDM breaker logic testing. A reactor trip occurred, due to loss of CRDM power (bus "E2"), and instrumentation power was also lost (bus "E2" and a defective logic card on the alternate source). During the recovery, AFW pump "2" failed to start due to a maladjusted governor slip clutch and bent low speed stop pin. A main steam safety valve lifted, and failed to reseat (valve was then gagged). ***LER # 346/81-037***
Brunswick 1	$7x10^{-3}$	4/19/81	***RHR heat exchanger damaged*** While the reactor was in cold shutdown during a maintenance outage, the normal decay heat removal system was lost because of a failure of the single RHR heat exchanger that was currently in service. The failure occurred when the starting of a second RHR service water pump caused the failure of a baffle in the waterbox of the RHR heat exchanger, thereby allowing cooling water to bypass the tube bundle. The redundant heat exchanger was inoperable because maintenance was in progress. ***LER # 325/81-032***
Millstone 2	$5x10^{-3}$	1/2/81	***Loss of DC power and one EDG as a result of operator error; partial LOOP*** When the reactor was at full power, the 125v DC emergency bus was lost as a result of operator error. The loss of the bus caused the reactor to trip, but the turbine failed to trip because of the unavailability of DC bus "A." Loads were not switched to the reserve transformer (following the manual turbine trip) because of the loss of DC bus "A." Two breakers (on the "B" 6.9kV and 4.16kV buses) remained open, thereby causing a LOOP. EDG "B" tripped as a result of leakage of the service water (SW) flange, which also caused the "B" 4.16 kV bus to be de-energized. An operator recognition error caused the PORV to be opened at 2380 psia. ***LER # 336/81-005***
St. Lucie 1	$1x10^{-3}$	6/11/80	***Reactor coolant pump seal LOCA due to loss of component cooling water (CCW); top vessel head bubble*** At 100-percent power, a moisture-induced short circuit in a solenoid valve caused a CCW containment isolation valve to shut causing loss of CCW to all reactor coolant pumps (RCPs). While reducing pressure to initiate the shutdown cooling system (SCS), the top head water flashed to steam, thus forming a bubble (initially undetected by the operators). During the cooldown, the SCS relief valves lifted and low-pressure safety injection (LPSI) initiated (i.e., the other LPSI pump started charging, while the other was used for cooldown). ***LER # 335/80-029***
Davis Besse 1	$1x10^{-3}$	4/19/80	***Loss of 2 essential busses*** When the reactor was in cold shutdown, two essential busses were lost due to breaker ground fault relay actuation during an electrical lineup. Decay heat drop line valve was shut, and air was drawn into the suction of the decay heat removal pumps, resulting in loss of a decay heat removal path. ***LER # 346/80-029***

Plant	CCDP	Date	Description
Crystal River 3	$5x10^{-3}$	2/26/80	***Loss of 24-volt DC power to non-nuclear instrumentation (NNI)*** The 24-volt power supply to the NNI was lost as a result of a short to ground. This initiated a sequence of events in which the PORV opened (and stayed open) as a direct result of the loss of the NNI power supply. HPI initiated as a result of depressurization through the open PORV, and with approximately 70 percent of NNI inoperable or inaccurate, the operator correctly decided that there was insufficient information available to justify terminating HPI. Therefore, the pressurizer was pumped solid, one safety valve lifted, and flow through the safety valve was sufficient to rupture the reactor coolant drain tank rupture disk, thereby spilling approximately 43,000 gallons (162,800 liters) of primary water into the containment. ***LER # 302/80-010***
Hatch 2	$1x10^{-3}$	6/3/79	***Loss of feedwater; HPCI fails to start; RCIC is unavailable*** During a power increase, the reactor tripped due to a condensate system trip. HPCI failed to initiate on low-low level due to a failed turbine stop valve. In addition, water from leaking mechanical seal lines and an unknown valve caused water to back up and contaminate the pump oil. RCIC was out of service for unspecified reasons. ***LER # 366/79-045***
Oyster Creek	$2x10^{-3}$	5/2/79	***Loss of feedwater flow*** While testing the isolation condenser, a reactor scram occurred. The feedwater pump tripped and failed to restart. The recirculation pump inlet valves were closed. The isolation condenser was used during cooldown. ***LER # 219/79-014***
Three Mile Island 2	1	3/28/79	***Loss of feedwater; PORV failed open; operator errors led to core damage*** Operators misinterpreted plant conditions, including the RCS inventory, during a transient that was triggered by a loss of feedwater and a stuck-open PORV. As a result, the operators prematurely shut off the high-pressure safety injection system, turned off the reactor coolant pumps, and failed to diagnose and isolate a stuck-open pressurizer relief valve. With the no RCS inventory makeup, the core became uncovered and fuel damage occurred. In addition, contaminated water was spilled into the containment and auxiliary buildings. ***LER # 320/79-012***
Salem 1	$1x10^{-2}$	11/27/78	***Loss of vital bus and scram; multiple components lost*** While the reactor was at 100-percent power, vital instrument bus “1B” was lost as a result of the failure of an output transformer and two regulating resistors. Loss of the vital bus caused a false low RCS loop flow signal, thereby causing a reactor trip. Two AFW pumps failed to start (one because of the loss of vital bus “1B”, and the other because of a maladjustment of the over-speed trip mechanism). Inadvertent safety injection occurred as a result of decreasing average coolant temperature and safety injection signals. ***LER # 272/78-073***
Calvert Cliffs 1	$3x10^{-3}$	4/13/78	***LOOP; one EDG failed to start*** With the plant shutdown, a protective relay automatically opened the switchyard breakers, resulting in a LOOP. EDG “11" failed to start. EDG “22" started and supplied the safety busses. ***LER # 317/78-020***

Plant	CCDP	Date	Description
Farley 1	5x10^{-3}	3/25/78	***Low-Low water level in one SG trip/scram; turbine-driven AFW pump fails*** A low level condition in a single SG resulted in a reactor trip. The turbine-driven AFW pump failed to start. Both motor-driven AFW pumps started, but were deemed ineffective because all recirculation bypass valves were open (thereby diverting flow). A recirculation valve was manually closed. ***LER # 348/78-021***
Rancho Seco	1x10^{-1}	3/20/78	***Failure of NNI and steam generator dryout*** When the reactor was at power, a failure of the NNI power supply resulted in a loss of main feedwater, which caused a reactor trip. Because instrumentation drift falsely indicated that the steam generator contained enough water, control room operators did not take prompt action to open the AFW flow control valves to establish secondary heat removal. This resulted in steam generator dryout. ***LER # 312/78-001***
Davis-Besse 1	5x10^{-3}	12/11/77	***AFW pumps inoperable during test*** During AFW pump testing, operators found that control over both pumps was lost because of mechanical binding in the governor of one pump and blown control power supply fuses for the speed changer motor on the other pump. ***LER # 346/77-110***
Davis-Besse 1	7x10^{-2}	9/24/77	***Stuck-open pressurizer PORV*** A spurious half-trip of the steam and feedwater rupture control system initiated closure of the startup feedwater valve. This resulted in reduced water level in SG "2." The pressurizer PORV lifted nine times and then stuck open because of rapid cycling. ***LER # 346/77-016***
Cooper	1x10^{-3}	8/31/77	***Partial loss of feedwater; reactor scram; RCIC and HPCI degraded*** A blown fuse caused the normal power supply to the feedwater and RCIC controllers to fail. The alternate power supply was unavailable due to an unrelated fault. A partial loss of feedwater occurred, and the reactor tripped on low water level. RCIC and HPCI operated, however, both pumps did not accelerate to full speed (RCIC due to the failed power supply and HPCI due a failed governor actuator). ***LER # 298/77-040***
Zion 2	2x10^{-3}	7/12/77	***Testing causes instrumentation errors*** With the reactor in hot shutdown, testing caused operators to lose indications of reactor and secondary system parameters. In addition, inaccurate inputs were provided to control and protection systems. ***LER # 304/77-044***
Millstone 2	1x10^{-2}	7/20/76	***Loop from grid disturbance; errors in EDG loading fail the emergency core cooling systems (ECCS)*** With the reactor at power, a main circulating water pump was started, and this resulted in an in-plant voltage reduction to below the revised trip set point. This isolated the safety-related buses and started the EDGs. Each time a major load was tied onto the diesel, the revised under-voltage trip set points tripped the load. As a result, at the end of the EDG loading sequence, all major loads were isolated even though the EDGs were tied to the safety-related buses. ***LER # 336/76-042***

Plant	CCDP	Date	Description
Kewaunee	$5x10^{-3}$	11/5/75	***Inoperable AFW pumps during startup as a result of leaks from the demineralizer into the condensate storage tank (CST)*** Mixed bed resin beads were leaking from the demineralizer in the makeup water system and migrated to the CST. As a result, during startup, both motor-driven AFW pump suction strainers became clogged, thereby resulting in low pump flow. The same condition occurred for the turbine-driven AFW pump suction strainer. ***LER # 305/75-020***
Brunswick 2	$9x10^{-3}$	4/29/75	***Multiple valve failures; RCIC inoperable as a result of stuck-open down/safety valve*** At 10-percent power, the RCIC system was determined to be inoperable, and SRV "B" was stuck open. The operator failed to scram the reactor according to the EOPs. HPCI system failed to run and was manually shut down as a result of high torus level. Loop "B" of RHR failed as a result of a failed service water supply valve to the heat exchanger. The reactor experienced an automatic scram on manual closure of the main steam isolation valve (MSIV). ***LER # 324/75-013***
Browns Ferry 1	$2x10^{-1}$	3/22/75	***Cable tray fire*** The fire was started by an engineer, who was using a candle to check for air leaks through a firewall penetration seal to the reactor building. The fire resulted in significant damage to cables related to the control of Units 1 and 2. All Unit 1 emergency core cooling systems were lost, as was the capability to monitor core power. Unit 1 was manually shut down and cooled using remote manual relief valve operation, the condensate booster pump, and control rod drive system pumps. Unit 2 was shut down and cooled for the first hour by the RCIC system. After depressurization, Unit 2 was placed in the RHR shutdown cooling mode with makeup water available from the condensate booster pump and control rod drive system pump. ***LER # 259/75-006***
Turkey Point 3	$2x10^{-2}$	5/8/74	***Failure of three AFW pumps to start during test*** Operators attempted to start all three AFW pumps while the reactor was at power for testing. Two of the pumps failed to start as a result of over-tightened packing. The third pump failed to start because of a malfunction in the turbine regulating valve pneumatic controller. ***LER # 250/74-LTR***
Point Beach 1	$5x10^{-3}$	4/7/74	***Inoperable AFW pumps during shutdown*** While the reactor was in cooldown mode, motor-driven AFW pump "A" did not provide adequate flow. The operators were unaware that the in-line suction strainers were 95 percent plugged (both motor-driven pumps "A" and "B"). A partially plugged strainer was found in each of the suction lines for both turbine-driven AFW pumps. ***LER # 266/74-LTR***
Point Beach 1	$1x10^{-3}$	1/12/71	***Failure of containment sump valves*** During a routine check of the containment tendon access gallery, air was observed leaking from the packing of one sump isolation valve. Operators attempted to open the valve, but the valve failed to open due to a shorted solenoid in the hydraulic positioner. The redundant sump isolation valve was also found inoperable due to a stuck solenoid in the hydraulic positioner. ***LER # 266/71-LTR***

*NOTE (to Table 10):

- Events are selected on the basis of CCDPs, as estimated by the ASP Program.

- Because of model and data uncertainties, it is difficult to differentiate between events with CCDPs that are within a factor of about 3.

- ASP analyses have been performed since 1969, and the associated methodologies and PRA models have evolved over the past 30 years. Consequently, the results obtained in the earlier years may be conservative when compared to those obtained using the current methodology and PRA models.

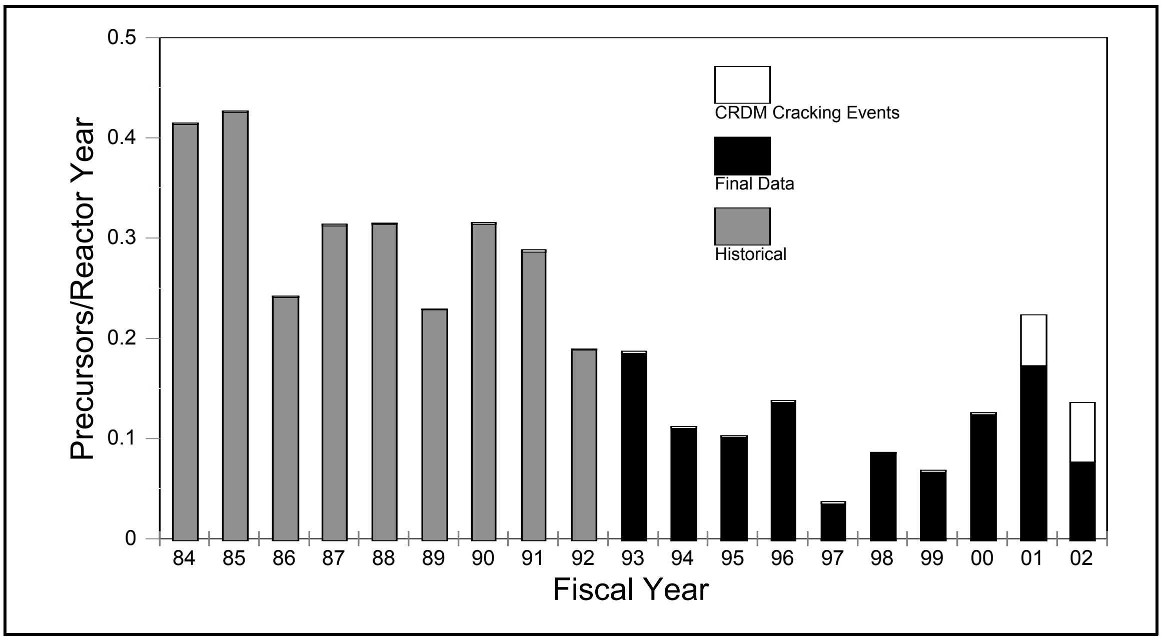

Figure 2: Total precursors–occurrence rate, by fiscal year. No trend line is shown because no trend was detected that was statistically significant (p-value= 0.34). The results prior to FY 1993 are shown to provide perspective. The ongoing analyses of events involving cracks in the CRDM housings are included FY 2001 and 2002 data.

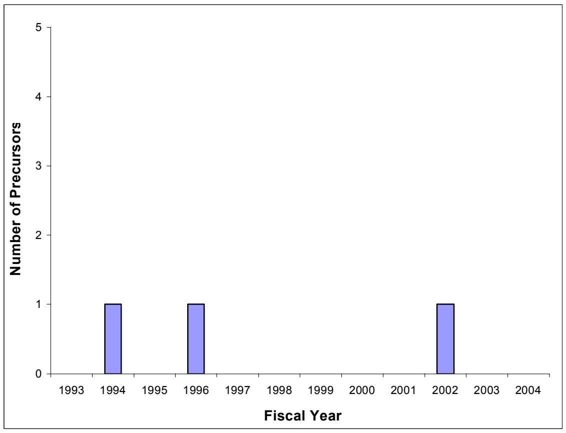

Figure 2a: Precursors in CCDP bin 10^{-3}-number of precursors, by fiscal year. No trend line is shown because no trend was detected that is statistically significant.

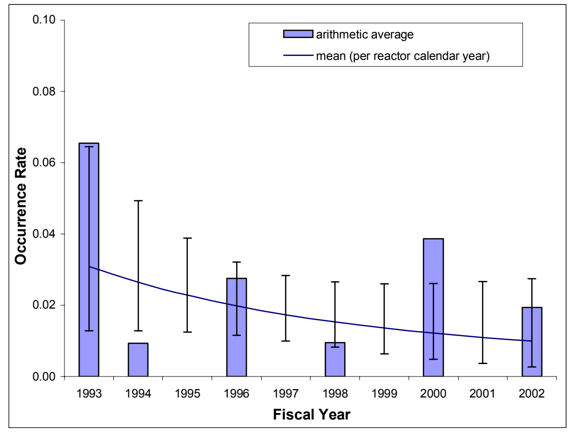

Figure 2b: Precursors in CCDP bin 10^{-4}-occurrence rate, by fiscal year. The decreasing trend is almost statistically significant (p-value = 0.11).

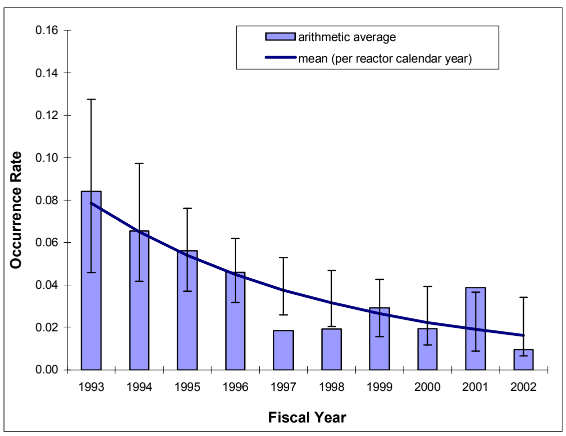

Figure 2c: Precursors in CCDP bin 10^{-5}-occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.002).

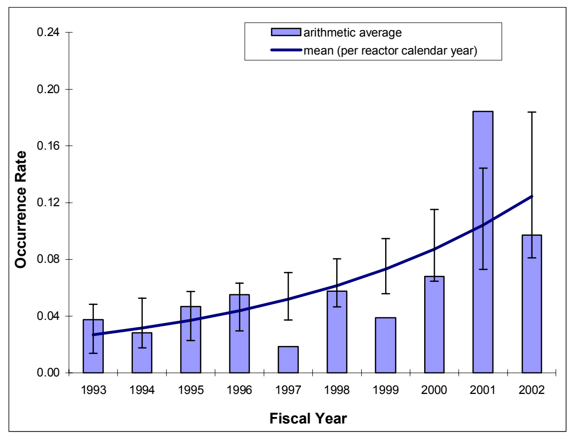

Figure 2d: Precursors in CCDP bin 10^{-6}-occurrence rate, by fiscal year. The increasing trend is statistically significant (p-value = 0.0001).

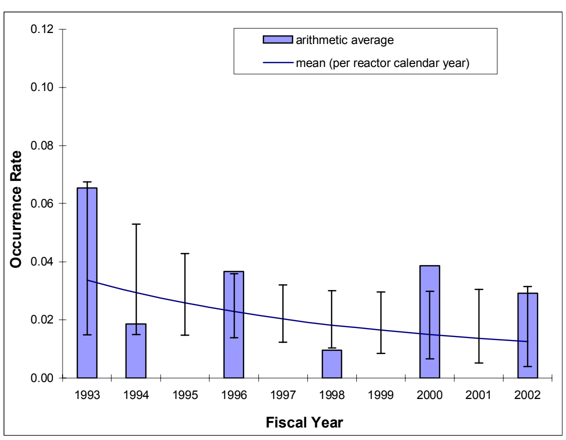

Figure 3: Important precursors (CCDP = 10^{-4})-occurrence rate, by fiscal year. The decreasing trend is almost statistically significant (p-value = 0.14).

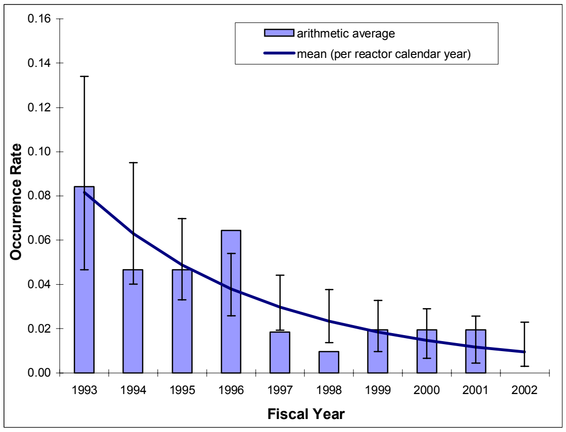

Figure 4: Precursors involving initiating events-occurrence rate, by fiscal year. The decreasing trend is statistically significant (p-value = 0.0001).

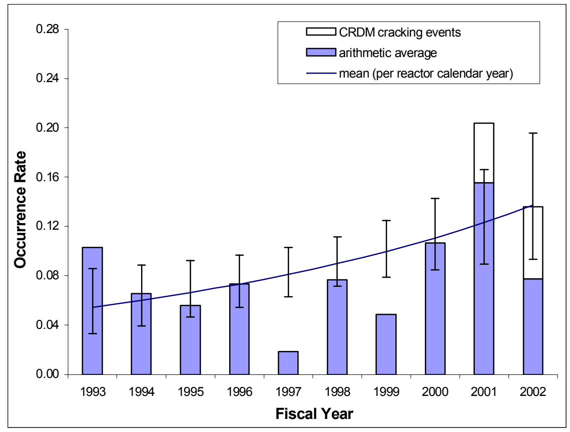

Figure 5: Precursors involving conditional unavailability of equipment-occurrence rate, by fiscal year. The increasing trend is statistically significant (p-value = 0.005).

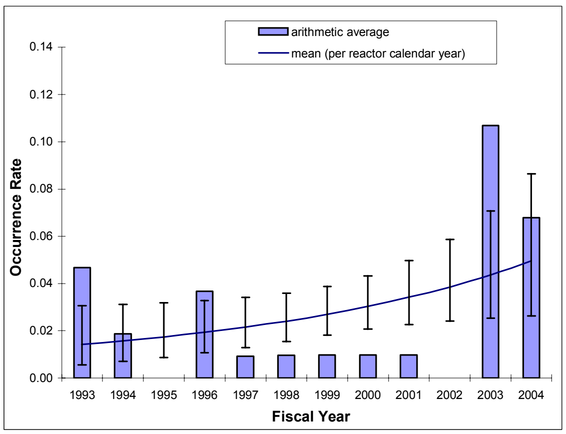

Figure 6: Precursors involving loss of offsite power initiating events-occurrence rate, by fiscal year. The increasing trend is statistically significant (p-value = 0.02).

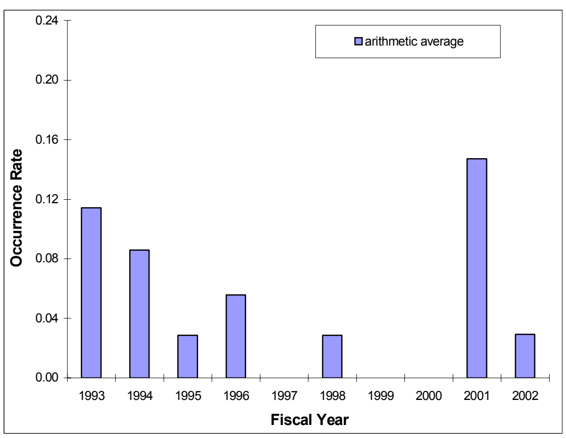

Figure 7: Precursors involving BWRs-occurrence rate, by fiscal year. No trend line is shown because no trend was detected that is statistically significant (p-value = 0.36).

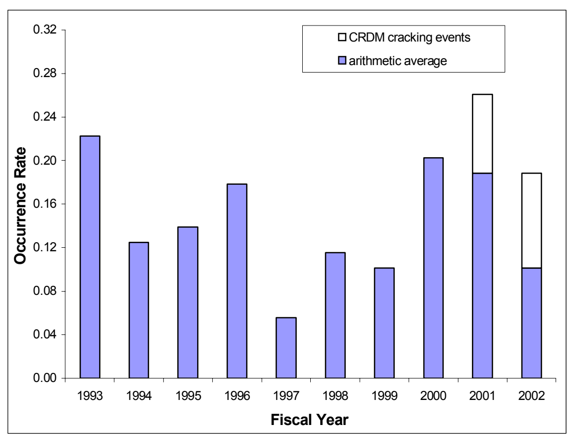

Figure 8: Precursors involving PWRs-occurrence rate, by fiscal year. No trend line is shown because no trend is detected that is statistically significant (p-value = 0.37).

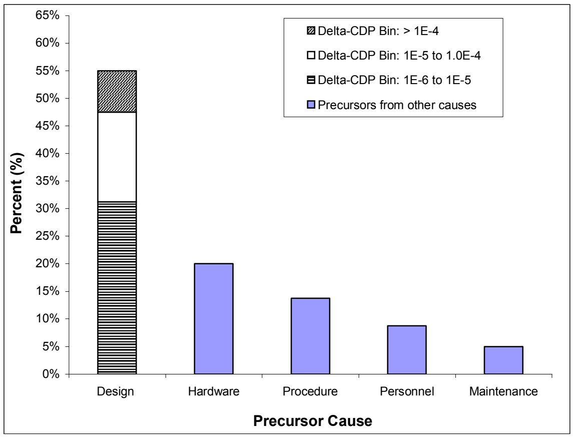

Figure 9a: Causes of precursors- percentage, by cause type.

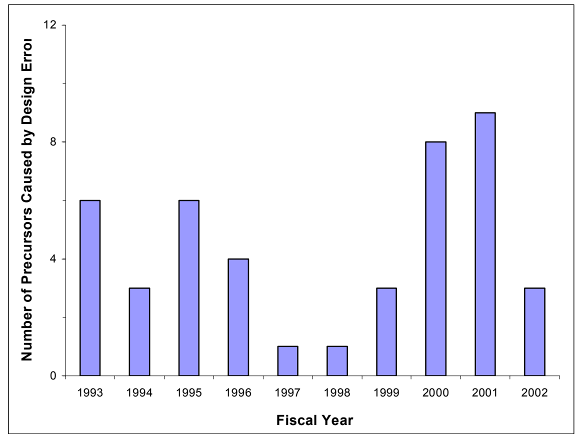

Figure 9b: Precursors caused by design-basis issues- number of precursors, by fiscal year.

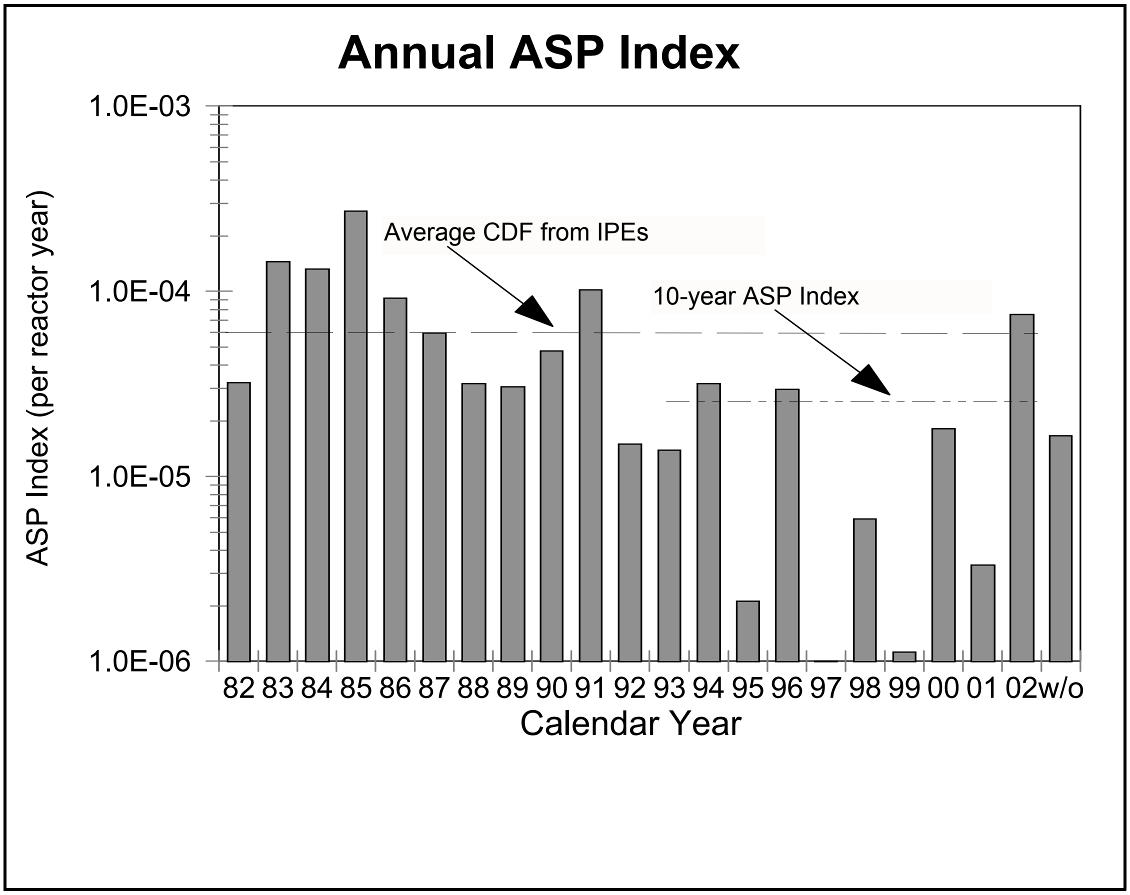

Figure 10: Annual ASP Index - ASP Index for a year is the total CCDP of all precursors divided by the total number of reactor years in a given year. Years with significant precursors (i.e., CCDP $\geq$ 1E-3): 1983 (2), 1984 (2), 1985 (3), 1986 (2), 1990 (1), 1991 (1), 1994 (1), 1996 (1), and 2002 (1). The ongoing analyses of events involving cracks in the CRDM housings are included FY 2001 and 2002 data. The CCDPs of these events are assumed to be 5×10^{-5}. The bar labeled "w/o" is FY 2002 index without the potential precursor at Davis-Besse.