Results, Trends, and Insights of the Accident Sequence Precursor Program

1.0 Introduction

This enclosure discusses the results of accident sequence precursor (ASP) analyses conducted by the staff as they relate to events that occurred during fiscal years (FYs) 2009–2010. Based on those results, this document also discusses the staff's analysis of historical ASP trends and the evaluation of the related insights.

2.0 Background

The U.S. Nuclear Regulatory Commission (NRC) established the ASP Program in 1979 in response to recommendations made in NUREG/CR-0400, "Risk Assessment Review Group Report," issued September 1978. The ASP Program systematically evaluates U.S. nuclear power plant (NPP) operating experience to identify, document, and rank the operating events that are most likely to lead to inadequate core cooling and severe core damage (precursors).

To identify potential precursors, the staff reviews plant events from licensee event reports (LERs) and inspection reports (IRs). The staff then analyzes any identified potential precursors by calculating a probability of an event leading to a core damage state. A plant event can be one of two types, either (1) an occurrence of an initiating event, such as a reactor trip or a loss of offsite power (LOOP), with or without any subsequent equipment unavailability or degradation or (2) a degraded plant condition depicted by unavailability or degradation of equipment without the occurrence of an initiating event.

For the first type, the staff calculates a conditional core damage probability (CCDP). This metric represents a conditional probability that a core damage state is reached given an occurrence of an initiating event (and any subsequent equipment failure or degradation).

For the second type, the staff calculates an increase in core damage probability (ΔCDP). This metric represents the increase in core damage probability for a time period that a piece or multiple pieces of equipment are deemed unavailable or degraded.

The ASP Program considers an event with a CCDP or a ΔCDP greater than or equal to 1×10^{-6} to be a precursor.[1] The ASP Program defines a *significant* precursor as an event with a CCDP or ΔCDP greater than or equal to 1×10^{-3}.

Program Objectives. The ASP Program has the following objectives:

- Provide a comprehensive, risk-based view of NPP operating experience and a measure for trending core damage risk.

- Provide a partial check on dominant core damage scenarios predicted by probabilistic risk assessments (PRAs).

[1] For initiating event analyses, the precursor threshold is a CCDP greater than or equal to 1×10^{-6} or the plant-specific CCDP for a non-recoverable loss of balance-of-plant systems, whichever is greater. This initiating event precursor threshold prevents reactor trips with no losses of safety system equipment from being precursors.

- Provide feedback to regulatory activities.

The NRC also uses the ASP Program to monitor performance against the safety measures established in the agency's Congressional Budget Justification (Reference 1), which was formulated to support the agency's safety and security strategic goals and objectives.[2] Specifically, the program provides input to the following safety measures:

- Zero events per year identified as a *significant* precursor of a nuclear reactor accident.

- Less than one significant adverse trend in industry safety performance (determination principally made from the Industry Trends Program (ITP) but partially supported by ASP results).

Program Scope. The ASP Program is one of three agency programs that assess the risk significance of issues and events. The other two programs are the Significance Determination Process (SDP) and the event response evaluation process as defined in Management Directive (MD) 8.3, "NRC Incident Investigation Program." The SDP evaluates the risk significance of licensee performance deficiencies while assessments performed under MD 8.3 are used in the determination of the appropriate level of reactive inspection in response to a significant event. Compared to the other two programs, the ASP Program assesses additional scope of operating experience at U.S. NPPs. For example, the ASP Program analyzes initiating events as well as degraded conditions where no identified deficiency occurred in the licensee's performance. The ASP Program scope also includes events with concurrent, multiple degraded conditions.

3.0 ASP Program Status

The following subsections summarize the status and results of the ASP Program as of September 30, 2010.

FY 2009 Analyses. The ASP analyses for FY 2009 identified 19 precursors. Eighteen of the 19 precursors occurred while the plants were at power. The staff used the SDP to identify and assess 13 of the 19 precursors without performing duplicative analyses. In these cases, only the SDP significance category (i.e., the "color" of the finding) is reported in the ASP Program.

The CCDP or ΔCDP for no FY 2009 analysis exceeded the probability $1{\times}10^{-4}$; therefore, in accordance with the revised review process (see Reference 2); the staff issued these ASP analyses as final after completion of internal reviews (i.e., no formal external reviews were performed).

Table 1 presents the results of the staff's ASP analyses for FY 2009 precursors that involved initiating events. Table 2 presents the analysis results for FY 2009 precursors that involved degraded conditions.

[2] The performance measures involving precursor data (i.e., number of *significant* precursors and trend of all precursors) are the same for FYs 2009–2011.

Table 1. FY 2009 Precursors Involving Initiating Events.

Event Date	Plant	Description	CCDP/ SDP Color
11/03/08	Dresden 3	Failure to prevent inadvertent, uncontrolled control rod withdrawal by non-licensed operators. ***Enforcement Action (EA)-09-172***	WHITE[3]
03/26/09	Sequoyah 1	Partial loss of offsite power results in dual-unit reactor trips and extended loss of offsite power to a safety bus in each unit. ***LER 327/09-003***	4×10^{-6}
03/26/09	Sequoyah 2	Partial loss of offsite power results in dual-unit reactor trips and extended loss of offsite power to a safety bus in each unit. ***LER 327/09-003***	4×10^{-6}
07/12/09	Oyster Creek	Loss of offsite power with unavailability of isolation condenser due to foreign material. ***IR 50-219/09-09***	5×10^{-5}
07/30/09	Braidwood 2	Loss of offsite power coincident with a reactor trip due to loss of reactor coolant pumps. ***LER 457/09-002***	4×10^{-5}
08/19/09	Wolf Creek	Loss of offsite power due to lightning strike. ***IR 50-482/09-07***	9×10^{-6}

Table 2. FY 2009 Precursors Involving Degraded Conditions.

Event Date	Condition Duration	Plant	Description	ΔCDP/ SDP Color
10/16/08	31 years	St. Lucie 1	Air intrusion into component cooling water system causes pump cavitation. ***EA-09-321***	YELLOW[4]
11/02/08	180 days	Duane Arnold	Breaker failure results in emergency diesel generator failure during surveillance test. ***EA-09-083***	WHITE
12/02/08	47 days	Ginna	Inadequate preventative maintenance on the turbine-driven auxiliary feedwater pump governor results in pump failure. ***EA-09-045***	WHITE
02/25/09	26 days	Seabrook	Inadequate design on the emergency diesel generator cooling water flange leads cooling water leak and subsequent failure. ***EA- 09-145***	WHITE
02/27/09	66 days	Browns Ferry 1	Inadequate procedure revision could lead to failure in operator response to a fire. ***EA-09-307***	WHITE

3 A WHITE finding corresponds to a licensee performance deficiency of low to moderate safety significance and has increase in core damage frequency in the range of 10^{-6} to 10^{-5}.

4 A YELLOW finding corresponds to a licensee performance deficiency of substantial safety significance and has an increase in core damage frequency in the range of 10^{-5} to 10^{-4}.

02/27/09	66 days	Browns Ferry 2	Inadequate procedure revision could lead to failure in operator response to a fire. ***EA-09-307***	WHITE
02/27/09	66 days	Browns Ferry 3	Inadequate procedure revision could lead to failure in operator response to a fire. ***EA-09-307***	WHITE
03/06/09	19 months	Browns Ferry 1	Failure to protect cables of redundant safety systems from fire damage. ***EA-09-307***	YELLOW
03/06/09	18 years	Browns Ferry 2	Failure to protect cables of redundant safety systems from fire damage. ***EA-09-307***	YELLOW
03/06/09	13 years	Browns Ferry 3	Failure to protect cables of redundant safety systems from fire damage. ***EA-09-307***	YELLOW
06/24/09	322 days	Braidwood 1	Failure of containment sump suction valve to open. ***EA-09-259***	WHITE
07/02/09	51 days	Ginna	Corrosion binding of the governor control valve results in the turbine-driven auxiliary feedwater pump failure. ***EA-09-249***	WHITE
08/15/09	23 days	Farley 2	Standby service water pump unavailable for 23 days. ***IR 50-364/10-07***	9×10^{-6}

FY 2010 Analyses. The staff immediately performs an initial review of events to determine if they have the potential to be *significant* precursors. Specifically, the staff reviews a combination of LERs (as required by Title 10, Section 50.73, "Licensee Event Report System," of the *Code of Federal Regulations* [10 CFR 50.73]) and daily event notification reports (as required by 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors") to identify potential *significant* precursors. The staff has completed the review of FY 2010 events and no *significant* precursor was identified.

4.0 Industry Trends

This section discusses the results of trending analyses for all precursors and *significant* precursors.

Statistically Significant Trend. Statistically significant is defined in terms of the "p-value." A p-value is a probability indicating whether to accept or reject the null hypothesis that no trend exists in the data. P-values of less than or equal to 0.05 indicate that there is 95 percent confidence that a trend exists in the data (i.e., reject the null hypothesis of no trend).

Data Coverage. Based on insights gained in SECY-06-028, "Status of the Accident Sequence Precursor Program and the Development of Standardized Plant Analysis Risk Models," dated October 5, 2006, the staff chose FY 2001 as the trend analyses' starting point to provide a data period with a consistent ASP Program scope and to align it with the first full year of the Reactor Oversight Process (ROP). ASP Program changes that occurred in FY 2001 (e.g., inclusion of SDP findings and external initiated events) resulted in a step increase in the number of precursors identified compared to those identified in previous years. The data period for trending analyses ends in FY 2009 (the last full year of completed ASP analyses) but will become a shifting 10-year period in the future.

The following exception applies to the data coverage of the trending analyses:

- ***Significant Precursors.*** The data for *significant* precursors includes events that occurred during FY 2010. The results for FY 2010 are based on the staff's screening and review of a combination of LERs and daily event notification reports (as of September 30, 2010). The staff analyzes all potential *significant* precursors immediately.

4.1 Occurrence Rate of All Precursors

NRC's ITP provides the basis for addressing the agency's safety-performance measure on the "number of statistically significant adverse trends in industry safety performance" (one measure associated with the safety goal established in NRC's Strategic Plan). The mean occurrence rate[5] of all precursors identified by the ASP Program is one indicator used by the ITP to assess industry performance.

Results. A review of the data for that period reveals the following insights:

- The mean occurrence rate of all precursors does not exhibit a trend that is statistically significant (p-value = 0.22) for the period from FY 2001–2009 (see Figure 1).

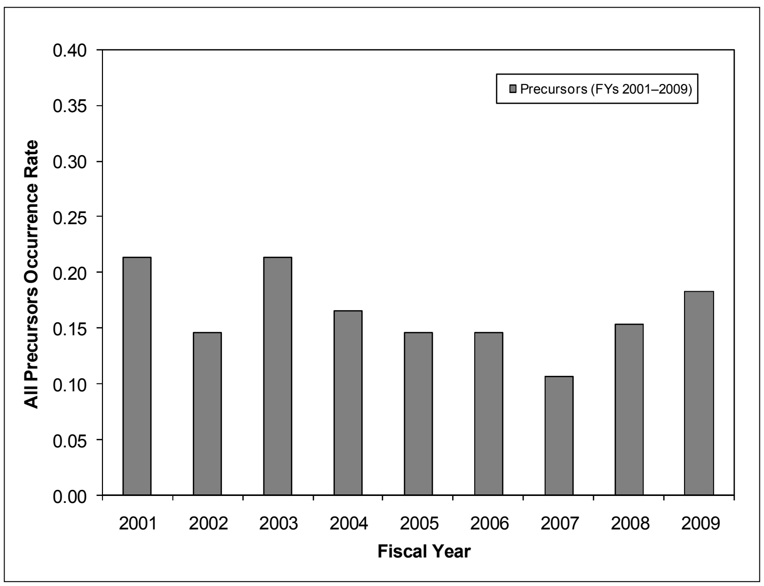

Figure 1. Total Precursors.

- The analysis detected a statistically significant decreasing trend (p-value = <0.0001) for precursors with a high safety significance (i.e., CCDP or ΔCDP greater than or equal to 1×10^{-4}) during this same period (see Figure 2).

5 The occurrence rate is calculated by dividing the number of precursors by the number of reactor years.

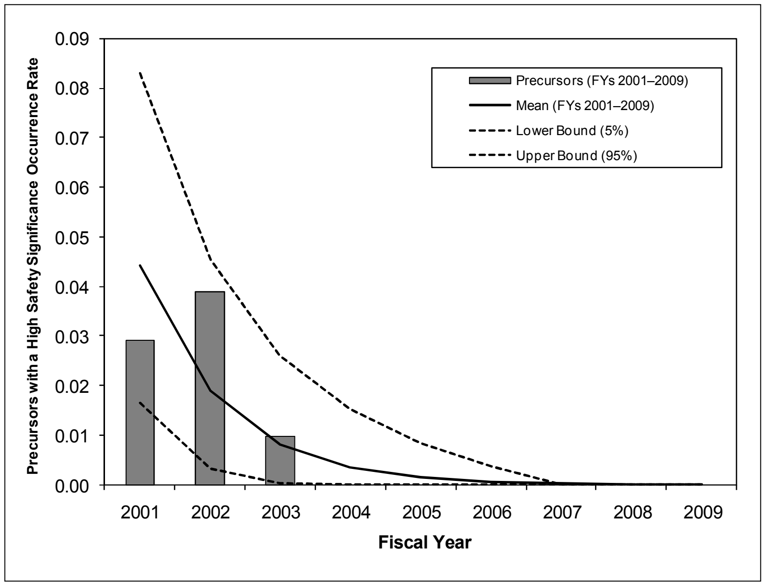

Figure 2. Precursors with High Safety Significance.

4.2 *Significant* Precursors

The ASP Program provides the basis for the safety measure of zero “number of significant accident sequence precursors of a nuclear reactor accident” (one measure associated with the safety goal established in NRC’s Congressional Budget Justification [Reference 1]). Specifically, a *significant* precursor is an event that has a probability of at least 1 in 1,000 (greater than or equal to 1×10^{-3}) of leading to a reactor accident.

Results. A review of the data for that period reveals the following insights:

- No *significant* precursors were identified in FY 2010.

- The staff has identified only one *significant* precursor since FY 2001. In FY 2002, the staff identified a *significant* precursor involving concurrent, multiple degraded conditions at Davis-Besse. Table 3 provides a complete list of all *significant* precursors from 1969–2010, including event descriptions. This table is provided for historical perspective and contains no new information or new insights.

- Over the past 20 years, *significant* precursors have occurred, on average, about once every 5 years and involve differing failure modes, causes, and systems.

Table 3. All Significant Precursors that Have Occurred Since 1969.[6]

Plant	CCDP/ ΔCDP	Date	Description
Davis-Besse	6×10^{-3}	2/27/02	The analysis included concurrent, multiple degraded conditions. These conditions included cracking of control rod drive mechanism (CRDM) nozzles and reactor pressure vessel (RPV) head degradation; potential clogging of the emergency sump; and potential degradation of the high-pressure injection (HPI) pumps. ***LER 346/02-002***
Catawba 2	2×10^{-3}	2/6/96	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 loss offsite power (LOOP) event. Although both emergency diesel generators (EDGs) started, the output breaker of EDG '1B' to vital Bus '1B' failed to close on demand, leaving Bus '1B' without 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	When the plant was in cold shutdown, operators implemented two unpermitted simultaneous evolutions, which resulted in the transfer of 9,200 gallons of reactor coolant system (RCS) inventory to the reactor water storage tank (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 as a result of this event. ***LER 482/94-013***
Harris 1	6×10^{-3}	4/3/91	A degraded condition resulted from relief valve and drain line failures in the alternative minimum flow systems for the safety injection (SI) 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***

[6] ASP analyses have been performed since 1969, and the associated methodologies and PRA models have evolved over the past 41 years. Consequently, the results obtained in the earlier years may be conservative when compared to those obtained using the current methodology and PRA models.

Plant	CCDP/ ΔCDP	Date	Description
Turkey Point 3	1×10^{-3}	12/27/86	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 power-operated relief valve (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***
Catawba 1	3×10^{-3}	6/13/86	A weld break on the letdown piping, near the component cooling water (CCW) 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 outlet flange on the variable letdown orifice failed as a result of excessive cavitation-induced vibration. This event was a small-break loss of coolant accident (LOCA). ***LER 413/86-031***
Davis-Besse 1	1×10^{-2}	6/9/85	While at 90% 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 auxiliary feedwater (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 High-Pressure Injection (HPI) Pump ‘1’ to reduce RCS pressure. ***LER 346/85-013***
Hatch 1	2×10^{-3}	5/15/85	Water from a heating, ventilation, and air conditioning (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 that had been inadvertently activated as a result of unrelated maintenance activities). This resulted in the lifting of the safety relief valve (SRV) four times. The SRV stuck open on the fourth cycle initiating a transient. Moisture also energized the high-pressure coolant injection (HPCI) trip solenoid making HPCI inoperable. Reactor core isolation cooling (RCIC) was unavailable due to maintenance. ***LER 321/85-018***

Plant	CCDP/ ΔCDP	Date	Description
LaSalle 1	2×10^{-3}	9/21/84	While at 23% power, an operator error caused a reactor scram and main steam isolation valve (MSIV) closure. RCIC was found to be unavailable during testing (one RCIC pump was isolated and the other pump tripped during the test). Residual heat removal (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	5×10^{-3}	2/25/83	When the reactor was at 25% power, both reactor trip breakers failed to open on demand of a low-low SG level trip signal. A manual trip was initiated about seconds after the automatic trip breaker failed to open, and was successful. The same event occurred 3 days later, at 12% power. Mechanical binding of the latch mechanism in the breaker under-voltage trip attachment failed both breakers in both events. ***LER 272/83-011***
Davis Besse 1	2×10^{-3}	6/24/81	With the plant at 74% 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. ***LER 346/81-037***
Brunswick 1	7×10^{-3}	4/19/81	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 water box 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***

Plant	CCDP/ ΔCDP	Date	Description
Millstone 2	5×10^{-3}	1/2/81	When the reactor was at full power, a 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 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 4.16 kV Bus 'B' to be de-energized. An operator recognition error caused the PORV to be opened at 2380 psi. ***LER 336/81-005***
St. Lucie 1	1×10^{-3}	6/11/80	At 100% 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	1×10^{-3}	4/19/80	When the reactor was in cold shutdown, two essential buses 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***
Crystal River 3	5×10^{-3}	2/26/80	The 24 V power supply to the nonnuclear instrumentation (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 about 70% of NNI inoperable or inaccurate, the operator correctly decided that insufficient information was 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 about 43,000 gallons of primary water into the containment. ***LER 302/80-010***

Plant	CCDP/ ΔCDP	Date	Description
Hatch 2	1×10^{-3}	6/3/79	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	2×10^{-3}	5/2/79	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	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	1×10^{-2}	11/27/78	While at full 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 maladjustment of the over-speed trip mechanism). An inadvertent SI occurred as a result of decreasing average coolant temperature and SI signals. ***LER 272/78-073***
Calvert Cliffs 1	3×10^{-3}	4/13/78	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 buses. ***LER 317/78-020***
Farley 1	5×10^{-3}	3/25/78	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***

Plant	CCDP/ ΔCDP	Date	Description
Rancho Seco	1×10^{-1}	3/20/78	When the reactor was at power, a failure of the NNI power supply resulted in a loss of MFW, which caused a reactor trip. Because instrumentation drift falsely indicated that the SG 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 SG dry out. ***LER 312/78-001***
Davis-Besse 1	5×10^{-3}	12/11/77	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	7×10^{-2}	9/24/77	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	1×10^{-3}	8/31/77	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	2×10^{-3}	7/12/77	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	1×10^{-2}	7/20/76	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***
Kewaunee	5×10^{-3}	11/5/75	Mixed bed resin beads were leaking from the demineralizer in the makeup water system and migrated to the condensate storage tank (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***

Plant	CCDP/ ΔCDP	Date	Description
Brunswick 2	$9×10^{-3}$	4/29/75	At 10% 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 procedures. HPCI system failed to run and was manually shut down as a result of high torus level. Train B of RHR failed as a result of a failed SW supply valve to the heat exchanger. The reactor experienced an automatic scram on manual closure of the MSIV. ***LER 324/75-013***
Browns Ferry 1	$2×10^{-1}$	3/22/75	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	$2×10^{-2}$	5/8/74	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	$5×10^{-3}$	4/7/74	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% plugged (both motor-driven pumps). 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	$1×10^{-3}$	1/12/71	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***

5.0 Insights and Other Trends

The following sections provide additional ASP trends and insights from the period FY 2001–2009.

5.1 Initiating Events vs. Degraded Conditions

A review of the data for FY 2001–2009 yields insights described below.

Initiating Events

- The mean occurrence rate of precursors involving initiating events does not exhibit a trend that is statistically significant (p-value = 0.89) for the period from FY 2001–2009, as shown in Figure 3.

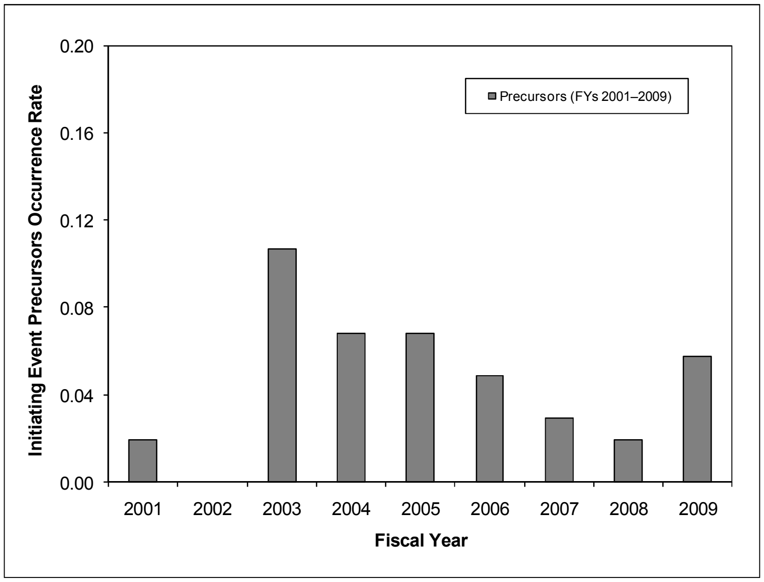

Figure 3. Precursors Involving Initiating Events.

- Of the 43 precursors involving initiating events during FY 2001–2009, 58 percent were LOOP events.

Degraded Conditions

- The mean occurrence rate of precursors involving degraded conditions does not exhibit a trend that is statistically significant (p-value = 0.19) during the FY 2001–2009 period, as shown in Figure 4.

- Over the past 9 years, precursors involving degraded conditions outnumbered initiating events (72 percent compared to 28 percent, respectively). This predominance was most notable in FY 2001 and FY 2002, when degraded conditions contributed to 91 percent and 100 percent of the identified precursors, respectively.

- From FY 2001–2009, 31 percent of precursors involving degraded conditions existing for a decade or longer. Of these precursors, 56 percent involved degraded conditions with condition start dating back to initial plant construction.

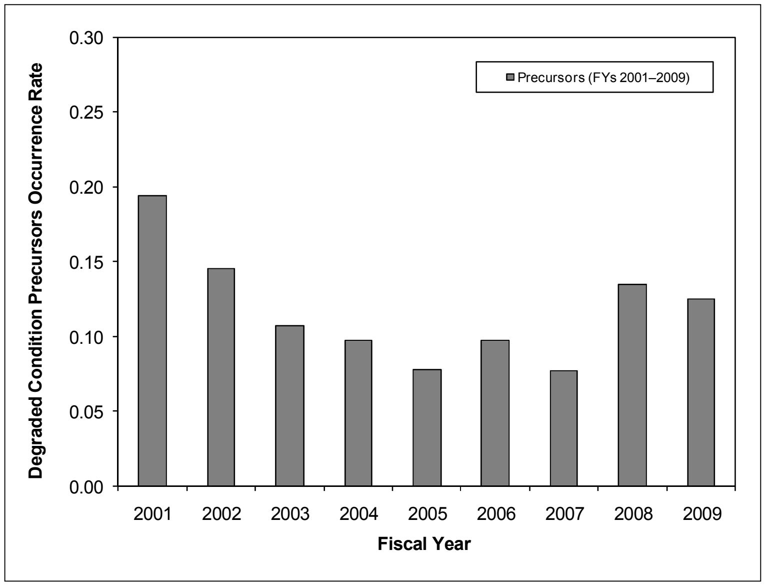

Figure 4. Precursors Involving Degraded Conditions.

5.2 Precursors Involving Loss of Offsite Power Initiating Events

Three of FY 2009 precursors resulted from a LOOP initiating event. Typically, all LOOP events meet the precursor threshold.

Results. A review of the data for FY 2001–2009 leads to the following insights:

- The mean occurrence rate of precursors resulting from a LOOP does not exhibit a trend that is statistically significant (p-value = 0.56) for the period from FY 2001–2009, as shown in Figure 5.

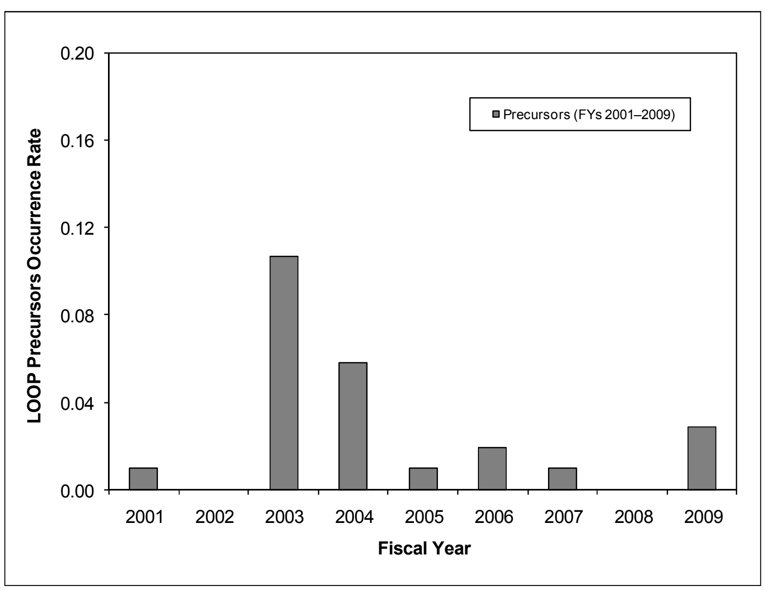

Figure 5. Precursors Involving LOOP Events.

- Of the 25 LOOP events that occurred during the FY 2001–2009 period, 44 percent resulted from a degraded electrical grid outside of the NPP boundary. Eight of the 11 grid-related LOOP precursors were the result of the 2003 Northeast Blackout.

- A simultaneous unavailability of an emergency power system train was involved in 2 of the 25 LOOP precursor events during FY 2001–2009.

5.3 Precursors at Boiling-Water Reactors versus Pressurized-Water Reactors

A review of the data for FY 2001–2009 reveals the results for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) described below.[7]

BWRs

- The mean occurrence rate of precursors that occurred at BWRs does not exhibit a trend that is statistically significant (p-value = 0.5) for the period from FY 2001–2009, as shown in Figure 6. The staff has reviewed the data to determine if any insights could be identified for the relative increase in precursors that have occurred at BWRs over the past three years. No clear insights were identified because the precursors involved different systems, components, and failure modes.

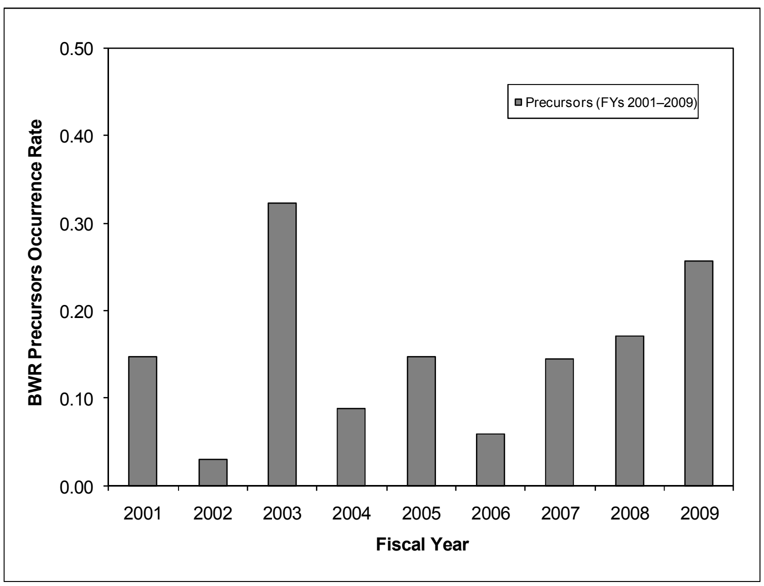

Figure 6. Precursors involving BWRs.

- LOOP events contributed to 61 percent of precursors involving initiating events at BWRs.

- Of the 29 precursors involving the unavailability of safety-related equipment that occurred at BWRs during FY 2001–2009, most were caused by failures in the emergency power system (41 percent), emergency core cooling systems (38 percent), electrical distribution system (14 percent), or safety-related cooling water systems (10 percent).

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7 The sum of percentages in this section does not always equal 100 percent because some precursors involve multiple equipment availabilities.

PWRs

- The mean occurrence rate of precursors that occurred at PWRs exhibits a statistically significant decreasing trend (p-value = 0.005) during the FY 2001–2009 period, as shown in Figure 7.

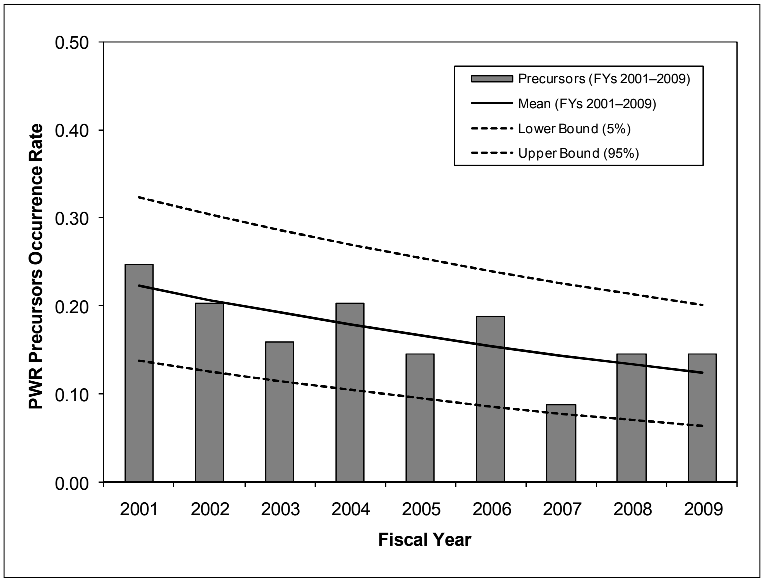

Figure 7. Precursors Involving PWRs.

- LOOP events contribute to 56 percent of precursors involving initiating events at PWRs.

- Of the 80 precursors involving the unavailability of safety-related equipment that occurred at PWRs during FY 2001–2009, most were caused by failures in the emergency core cooling systems (31 percent), auxiliary feedwater system (21 percent), emergency power system (19 percent), or safety-related cooling water systems (19 percent).

  - Of the 25 precursors involving failures in the emergency core cooling systems, 18 precursors (72 percent) were due to conditions affecting sump recirculation during postulated loss-of coolant accidents of varying break sizes. Design errors were the cause of most of these precursors (89 percent).

  - Of the 17 precursors involving failures of the auxiliary feedwater system, random hardware failures (47 percent) and design errors (35 percent) were the largest failure contributors. Fifteen of the 17 precursors (88 percent) involved the unavailability of the turbine-driven auxiliary feedwater pump train.

  - Of the 15 precursors involving failures of the emergency power system, 12 precursors (80 percent) were from hardware failures.

  - Design errors contributed 48 percent of all precursors involving the unavailability of safety-related equipment that occurred at PWRs during FY 2001–2009.

5.4 Integrated ASP Index

The staff derives the integrated ASP index for order-of-magnitude comparisons with industry-average core damage frequency (CDF) estimates derived from probabilistic risk assessments (PRAs) and NRC's standardized plant analysis risk (SPAR) models. The index or CDF from precursors for a given fiscal year is the sum of CCDPs and ΔCDPs in the fiscal year divided by the number of reactor-calendar years in the fiscal year.

The integrated ASP index includes the risk contribution of a precursor for the entire duration of the degraded condition (i.e., the risk contribution is included in each fiscal year that the condition exists). The risk contributions from precursors involving initiating events are included in the fiscal year that the event occurred.

Examples. A precursor involving a degraded condition is identified in FY 2003 and has a ΔCDP of 5×10^{-6}. A review of the LER reveals that the degraded condition has existed since a design modification performed in FY 2001. In the integrated ASP index, the ΔCDP of 5×10^{-6} is included in FYs 2001, 2002, and 2003.

For an initiating event occurring in FY 2003, only FY 2003 includes the CCDP from this precursor.

Results. Figure 8 depicts the integrated ASP indices for FY 2001–2009. A review of the ASP indices leads to the following insights:

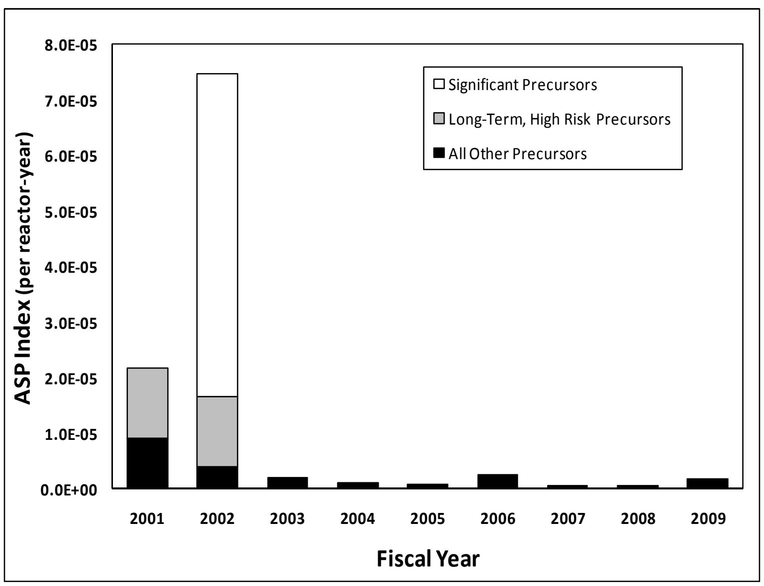

Figure 8. Integrated ASP Index.

- Based on order of magnitude (10^{-5}), the average integrated ASP index for the period from FY 2001–2009 is consistent with the CDF estimates from the SPAR models and industry PRAs.

- Precursors over the FY 2001–2009 period made the following contributions to the average integrated ASP index:

- The one *significant* precursor (i.e., CCDP or ΔCDP greater than or equal to 1×10^{-3}) contributed to 56 percent of the average integrated ASP index. The *significant* precursor (Davis-Besse, FY 2002) existed for one year.

- Two precursors involving long-term degraded conditions at Point Beach Units 1 and 2 contributed 24 percent of the average integrated ASP. The degraded conditions were discovered in FY 2002 and involved potential common-mode failure of all AFW pumps. The associated ΔCDPs of these two precursors were high (7×10^{-4}) and the degraded conditions had existed since plant construction.

- The remaining 20 percent of the average integrated ASP index resulted from contributions from the 149 precursors.

Limitations. Using CCDPs and ΔCDPs 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 also must account for events and conditions that did not meet the ASP precursor criteria.

The integrated ASP index provides the contribution of risk (per fiscal year) resulting from precursors and cannot be used for direct trending purposes because the discovery of precursors involving longer-term degraded conditions in future years may change the cumulative risk from the previous year(s).

5.5 Consistency between Precursors and Probabilistic Risk Assessments

A secondary objective of the ASP Program is to provide a partial check of the dominant core damage scenarios predicted by PRAs, including the agency SPAR models. Most identified precursors are consistent with accident sequences and system failures modes (i.e., at the event tree level) already identified in PRAs; however, some precursors involve event initiators or failure causes that are not explicitly modeled in the associated plant PRA. In addition, precursors can involve potential failures or degraded conditions that are often discovered through design or procedure reviews and are not explicitly considered in initiating event or component failure data. A recent example is the air intrusion into the component cooling water system that occurred at St. Lucie (Unit 1) in FY 2009. The plant did not lose component cooling water; however, the event resulted in an increased likelihood of losing the component cooling water system, which is not explicitly modeled in PRAs. In this case, the associated SPAR model did include the appropriate accident sequences and system-level failure modes to support the risk analysis.

In FY 2009, five precursors involved potential failures or failure causes that were not explicitly modeled and required some modifications or enhancements to the SPAR models. Although the event initiators and failure causes associated with some precursors are not explicitly included in the SPAR models, it has always been possible to incorporate them into the existing models to gain risk insights that can be important inputs to regulatory decision-making.

A review of the precursors that occurred during FYs 2001–2009 reveals that 30 percent of the identified precursors involved event initiators or failure causes that were not explicitly modeled in the associated SPAR model. These precursors did not reveal any risk-significant core damage scenarios that are not currently captured in the SPAR models. Because the SPAR quality assurance process ensures that the agency PRA models provide a reasonable representation of the as-built, as-operated nuclear plant, this also implies that licensee PRA models developed in accordance with approved standards should adequately capture these risk-significant core damage scenarios. Table 4 provides a complete list of ASP analyses (FYs 2001–2009) that required mapping the specific condition or event into the existing SPAR model.

Table 4. Precursors Involving Failure Modes or Initiators not explicitly Modeled in a PRA.

FY	Plant	Event Description
2009	St. Lucie 1	Air intrusion into component cooling water system causes pump cavitation. ***EA-09-321***
2009	Dresden	Failure to prevent inadvertent, uncontrolled control rod withdrawal by nonlicensed operators. ***EA-09-172***
2009	Browns Ferry 1, 2, 3	Failure to protect cables of redundant safety systems from fire damage. ***EA-09-307***
2008	Prairie Island 2	Potential unavailability of the component cooling water system during a postulated high-energy line break due to inadequate design. ***EA-09-167***
2008	Byron 1 & 2	Corrosion of equipment cooling water system piping. ***EA-08-046***
2008	San Onofre 2	Deficient electrical connections with potential to affect multiple safety systems. ***EA-08-296***
2008	Oconee 1	Procedure error leads to loss of reactor coolant system inventory while shutdown (Mode 6). ***EA-08-324***
2007	Cooper	Inadequate post-fire procedure could have prevented achieving safe shutdown. ***EA-07-204***
2007	McGuire 1 & 2	Potential inoperability of service water strainer backwash system during accident conditions. ***EA-08-220***
2006	Clinton	Potential air entrapment of high-pressure core spray because of incorrect suction source switchover set point. ***EA-06-291***
2006	Oconee 1, 2, 3	Failure to maintain design control for the standby shutdown facility flooding boundary. ***EA-06-199***
2005	Kewaunee	Design deficiency could cause unavailability of safety-related equipment during postulated internal flooding. ***EA-05-176***
2005	LaSalle 1 & 2 Crystal River 3	Single-failure vulnerability of safety bus protective relay schemes caused by common power metering circuits. ***EA-05-103, EA-05-114***
2005	Watts Bar	Component cooling backup line from essential raw cooling water was unavailable because silt blockage. ***IR 50-390/04-05***
2005	Watts Bar	Low-temperature, overpressure valve actuations while shut down. ***EA-05-169***
2004	Calvert Cliffs 2	Failed relay causes overcooling condition during reactor trip. ***LER 318/04-001***
2004	Palo Verde 1, 2, 3	Containment sump recirculation potentially inoperable because of pipe voids. ***LER 528/04-009***

FY	Plant	Event Description
2003	Shearon Harris	Postulated fire could cause the actuation of certain valves that could result in a loss of the charging pump, reactor coolant pump seal cooling, loss of reactor coolant system inventory, and other conditions. ***LER 400/02-004***
2003	St. Lucie 2	Reactor pressure vessel head leakage because of cracking of control rod drive mechanism nozzles. ***LER 389/03-002***
2002	Crystal River 3 Three Mile Island 1 Surry 1 North Anna 1 & 2	Reactor pressure vessel head leakage because of cracking of control rod drive mechanism nozzles. ***LER 302/01-004, LER 289/01-002, LER 280/01-003, LER 339/02-001***
2002	Columbia	Common-cause failure of breakers used in four safety-related systems. ***IR 50-397/02-05***
2002	Davis-Besse	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***
2002	Callaway	Potential common-mode failure of all auxiliary feedwater pumps because of foreign material in the condensate storage tank caused by degradation of the floating bladder. ***LER 483/01-002***
2002	Point Beach 1 & 2	Potential common-mode failure of all auxiliary feedwater pumps because of a design deficiency of the air-operated minimum flow recirculation valves. ***LER 266/01-005***
2002	Shearon Harris	Potential failure of residual heat removal pump and containment spray pump because of debris in the pumps' suction lines. ***LER 400/01-003***
2001	Oconee 1, 2, 3 Arkansas 1 Palisades	Reactor pressure vessel head leakage because of cracking of control rod drive mechanism nozzles. ***LER 269/03-002, LER 270/02-002, LER 287/03-001, LER 313/02-003, LER 255/01-004***
2001	Kewaunee	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 50-305/02-06***

6.0 Summary

This section summarizes the ASP results, trends, and insights:

- ***Significant Precursors.*** The staff did not identify any *significant* precursors (i.e., CCDP or ΔCDP greater than or equal to 1×10^{-3}) in FY 2010. The ASP Program provides the basis for the safety-performance measure of zero "number of significant accident sequence precursors of a nuclear reactor accident." These results will be provided in the FY 2010 Performance and Accountability Report.

- ***Occurrence Rate of All Precursors.*** The occurrence rate of all precursors does not exhibit a trend that is statistically significant during the FY 2001–2009 period. The trend of all precursors is one input into the ITP to assess industry performance and is part of the input into the adverse trends’ safety measure. These results will be provided in the FY 2010 Performance and Accountability Report.

- ***Additional Trend Results.*** During the same period, statistically significant decreasing trends were detected for two subgroups of precursors—precursors with a CCDP or ΔCDP greater than or equal to 1×10^{-4} and precursors that occurred at PWRs. No trends were observed in other precursor subgroups.

7.0 References

1. NUREG-1100, Vol. 26, “Performance Budget: Fiscal Year 2011,” U.S. Nuclear Regulatory Commission, Washington, DC, February 2010.

2. Regulatory Issue Summary 2006-24, “Revised Review and Transmittal Process for Accident Sequence Precursor Analyses,” U.S. Nuclear Regulatory Commission, Washington, DC, December 2006.

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