

## **POLICY ISSUE (Information)**

September 29, 2010

SECY-10-0125

FOR: The Commissioners

FROM: Brian W. Sheron, Director  
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SUBJECT: STATUS OF THE ACCIDENT SEQUENCE PRECURSOR PROGRAM  
AND THE STANDARDIZED PLANT ANALYSIS RISK MODELS

### PURPOSE

To inform the Commission of the status of the Accident Sequence Precursor (ASP) Program, provide the annual quantitative ASP results, and communicate the status of the standardized plant analysis risk (SPAR) models. This paper does not address any new commitments or resource implications.

### BACKGROUND

In a memorandum to the Chairman dated April 24, 1992, the staff of the U.S. Nuclear Regulatory Commission (NRC) committed to report periodically to the Commission on the status of the ASP Program including development of associated risk models (e.g., SPAR models). The ASP Program systematically evaluates U.S. nuclear power plant (NPP) operating experience to identify, document, and rank the operating events most likely to lead to inadequate core cooling and severe core damage (precursors). The ASP Program provides insights to NRC's risk-informed and performance-based regulatory programs and monitors performance against safety measures established in the agency's Congressional Budget Justification (see NUREG-1100, Volume 26, "Congressional Budget Justification: Fiscal Year 2011," issued February 2010). The SPAR Model Program develops and improves independent risk-analysis tools and capabilities to support the use of probabilistic risk assessment (PRA) in the agency's risk-informed regulatory activities. The staff uses SPAR models to support the Significance Determination Process (SDP), the ASP Program, the Incident Investigation

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Program event assessment process, and the Generic Issue Program resolution process and to inform licensing and inspection activities.

## DISCUSSION

This section summarizes the status, accomplishments, and results of the ASP Program and SPAR Model Program since the previous status report, SECY-09-0143, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," dated September 29, 2009.

### ASP Program

The staff has completed the analyses of all precursor events that were identified in fiscal year (FY) 2009 (19 precursors). Precursors are events with a conditional core damage probability (CCDP) or increase in core damage probability ( $\Delta$ CDP) that is greater than or equal to  $1 \times 10^{-6}$ . In addition, the staff has completed the screening of FY 2010 events for *significant* precursors. *Significant* precursors have a CCDP or  $\Delta$ CDP greater than or equal to  $1 \times 10^{-3}$ . No *significant* precursors were identified in either FY 2009 or FY 2010, and the staff continues to complete the review, analysis, and documentation of all potential precursors. The last *significant* precursor identified was the Davis-Besse event in FY 2002. The ASP Program provides input to the agency's safety-performance measure of zero events per year identified as a *significant* precursor of a nuclear reactor accident.

The staff evaluated precursor data during the period of FY 2001 through FY 2009 to identify statistically significant adverse trends for the Industry Trends Program (ITP). No statistically significant trend was detected for all precursors during this 9-year period. The ASP Program results are trended in the ITP to provide an input to the agency's safety-performance measure of no significant adverse trend in industry safety performance.

In addition to the trend analysis of all precursors, the staff performs trend analyses on other precursor subgroups. These subgroups include precursors with a high safety significance (i.e., CCDP or  $\Delta$ CDP greater than or equal to  $1 \times 10^{-4}$ ), initiating events, degraded conditions, loss of offsite power initiating events, precursors at boiling-water reactors (BWRs), and precursors at pressurized-water reactors (PWRs). Statistically significant decreasing trends were detected for two subgroups of precursors—precursors corresponding to high safety significance (i.e., CCDP or  $\Delta$ CDP greater than or equal to  $1 \times 10^{-4}$ ) and precursors that occurred at PWRs. No trends were observed in other precursor subgroups. Enclosure 1 provides additional details on results and trends of the ASP Program.

### SPAR Model Program

During FY 2010, the staff completed enhancements to Systems Analysis Program for Hands-On Integrated Reliability Evaluations (SAPHIRE) software. The SAPHIRE computer code allows users to develop PRA models and is used to analyze the SPAR models. SAPHIRE Version 8 was released in April 2010. This new version includes features and capabilities that are new or improved over Version 7 to address new requirements for risk-informed programs. User interfaces were developed for performing:

- SDP Phase 2 analyses using the SPAR models.
- Degraded condition assessments for SDP Phase 3 and ASP analyses, and Management Directive (MD) 8.3 evaluations.
- Initiating event assessments for ASP analyses and MD 8.3 evaluations.
- PRA analyses requiring more significant modeling or data revisions.

The 77 SPAR models representing the 104 operating commercial NPPs were revised and augmented to take advantage of the new features and capabilities of SAPHIRE Version 8. In addition to the above model enhancements, the staff completed an evaluation of the potential core damage risk reduction associated with the extensive damage mitigation strategies and guidance required by 10 CFR 50.54(hh) for about two-thirds of the SPAR models. The evaluations of the remaining SPAR models are scheduled to be completed by October 2010.

The staff, with the cooperation of industry experts, completed a peer review of a representative BWR SPAR model and PWR SPAR model in accordance with ASME/ANS RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," and Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The peer review teams concluded that, within the constraints of the program, the SPAR models are an appropriate tool to provide an independent check on utility PRAs. The peer review teams also provided additional findings and observations to enhance the SPAR models.

The staff continued to expand the SPAR model capability beyond internal events at full-power operation. The staff previously completed a total of 15 SPAR external event (EE) models (e.g., fires, floods, and seismic events). Three SPAR models which include both internal and external hazards were used in identifying and evaluating severe accident sequences for the consequential steam generator tube rupture (SGTR) project in support of the Steam Generator Action Plan. Consequential SGTR events are potentially risk significant due to the possibility of a severe core damage event (caused by either internal or external initiating events) in leading to failure of the steam generator tubes and bypass of the containment building. The SPAR-EE models were also recently used to provide background information to the Office of Nuclear Reactor Regulation (NRR) on the impact of assessing external hazard risk in 10 CFR 50.65 maintenance risk assessments. In addition, the staff plans to incorporate internal fire scenarios from the National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants," pilot applications into two SPAR models. The Office of Nuclear Regulatory Research staff continues to work with NRR and NRO to identify future enhancements to the SPAR-EE models.

The staff has continued the development and enhancement of the shutdown template models, resulting in a total of seven shutdown SPAR models that have become available to support the Reactor Oversight Process evaluations of shutdown events and degraded conditions.

The staff also developed new SPAR models for the Advanced Boiling Water Reactor (ABWR) and the AP1000. These models will allow confirmation of PRA results presented in licensing submittals and evaluation of risk-informed applications prior to new plant operation, and

assessment of operational findings and events once operation commences. The SPAR model for the AP1000 design was completed in February 2010 and quality assurance activities are being completed for the ABWR model. Enclosure 2 provides a detailed status of SPAR models.

### UPCOMING ACTIVITIES

The following upcoming activities continue to be responsive to agency programs:

- The staff will continue the screening, review, and analysis (preliminary and final) of potential precursors for FY 2010 and FY 2011 events to support the agency's safety measures.
- In accordance with existing user need requests, the staff will continue to implement enhancements to the SPAR models for full-power operations. Anticipated enhancements include incorporating new models for support-system initiators and revised success criteria based on insights from thermal-hydraulic analyses. Furthermore, the staff is working with industry representatives to resolve other PRA technical issues common to both licensee PRAs and NRC SPAR models. In support of this effort, the memorandum of understanding addendum on PRA with the Electric Power Research Institute has been extended through 2016.
- The staff will continue to evaluate the need for additional SPAR model capability (beyond full-power, internal initiators) based on experience gained from SDP, ASP, and MD 8.3 event assessments and respond to any new user need requests.
- The staff has reviewed the SPAR model peer review comments. A project plan is being developed to address peer review comments, where appropriate, and is planned to be completed in 2013. The main objective of this effort is to ensure the SPAR models continue to be of sufficient quality for performing SDP Phase 3, ASP, and MD 8.3 event assessments in support of the staff's risk-informed activities.
- The staff will use information obtained as part of the National Fire Protection Association Standards 805 pilot application process to create two new SPAR fire models with updated fire scenarios.
- The staff will continue the development of SPAR models for new reactors to allow confirmation of PRA results presented in licensing submittals, evaluation of risk-informed applications prior to new plant operation, and assessment of operational findings and events once operation commences. The next new reactor SPAR model planned to be developed is for the U.S. Advanced Pressurized-Water Reactor.

### SUMMARY

The ASP Program continues to evaluate the safety significance of operating events at NPPs and to provide insights to NRC's risk-informed and performance-based regulatory programs. The staff identified no *significant* precursors in FY 2010. No statistically significant trend was detected for all precursors during the FY 2001 through FY 2009 period. The SPAR Model

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Program is continuing to develop and improve independent risk analysis tools and capabilities to support the use of PRA in the agency's risk-informed regulatory activities.

COORDINATION

The Office of the General Counsel reviewed this Commission paper and has no legal objection.

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Enclosures:

1. Results, Trends, and Insights  
of the ASP Program
2. Status of the SPAR Models

# 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 ( $\Delta$ CCDP). 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  $\Delta$ CCDP greater than or equal to  $1 \times 10^{-6}$  to be a precursor.<sup>1</sup> The ASP Program defines a *significant* precursor as an event with a CCDP or  $\Delta$ CCDP greater than or equal to  $1 \times 10^{-3}$ .

**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).

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<sup>1</sup> For initiating event analyses, the precursor threshold is a CCDP greater than or equal to  $1 \times 10^{-6}$  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.<sup>2</sup> 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  $\Delta$ CDP 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.

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<sup>2</sup> 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. <b>Enforcement Action (EA)-09-172</b>	WHITE <sup>3</sup>
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. <b>LER 327/09-003</b>	4×10 <sup>-6</sup>
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. <b>LER 327/09-003</b>	4×10 <sup>-6</sup>
07/12/09	Oyster Creek	Loss of offsite power with unavailability of isolation condenser due to foreign material. <b>IR 50-219/09-09</b>	5×10 <sup>-5</sup>
07/30/09	Braidwood 2	Loss of offsite power coincident with a reactor trip due to loss of reactor coolant pumps. <b>LER 457/09-002</b>	4×10 <sup>-5</sup>
08/19/09	Wolf Creek	Loss of offsite power due to lightning strike. <b>IR 50-482/09-07</b>	9×10 <sup>-6</sup>

**Table 2. FY 2009 Precursors Involving Degraded Conditions.**

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

<sup>3</sup> 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<sup>-6</sup> to 10<sup>-5</sup>.

<sup>4</sup> 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<sup>-5</sup> to 10<sup>-4</sup>.



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

**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:

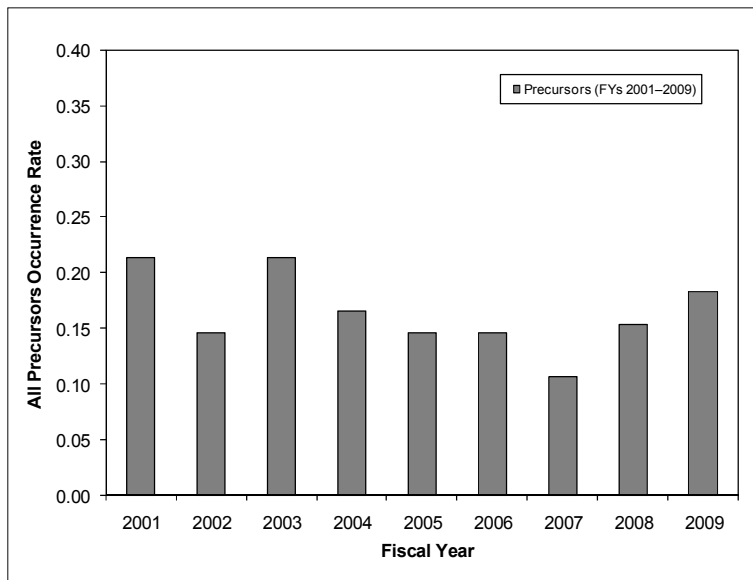
- **Significant Precursors.** The data for *significant* precursors includes events that occurred during FY 2010. The results for FY 2010 are based on the staff's screening and review of a combination of LERs and daily event notification reports (as of September 30, 2010). The staff analyzes all potential *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<sup>5</sup> 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:

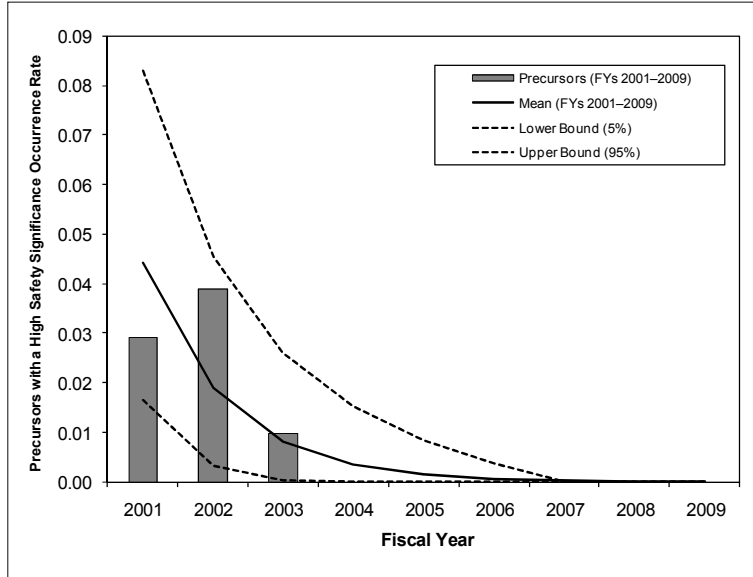
- The mean occurrence rate of all precursors does not exhibit a trend that is statistically significant (p-value = 0.22) for the period from FY 2001–2009 (see Figure 1).



**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., CDDP or  $\Delta$ CDDP greater than or equal to  $1 \times 10^{-4}$ ) during this same period (see Figure 2).

<sup>5</sup> 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  $1 \times 10^{-3}$ ) 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.<sup>6</sup>**

Plant	CCDP/ ΔCDP	Date	Description
Davis-Besse	$6 \times 10^{-3}$	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. <b>LER 346/02-002</b>
Catawba 2	$2 \times 10^{-3}$	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. <b>LER 414/96-001</b>
Wolf Creek 1	$3 \times 10^{-3}$	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. <b>LER 482/94-013</b>
Harris 1	$6 \times 10^{-3}$	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. <b>LER 400/91-008</b>

<sup>6</sup> 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	$1 \times 10^{-3}$	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. <b>LER 250/86-039</b>
Catawba 1	$3 \times 10^{-3}$	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). <b>LER 413/86-031</b>
Davis-Besse 1	$1 \times 10^{-2}$	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 reseal 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. <b>LER 346/85-013</b>
Hatch 1	$2 \times 10^{-3}$	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. <b>LER 321/85-018</b>

Plant	CCDP/ ΔCDP	Date	Description
LaSalle 1	$2 \times 10^{-3}$	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. <b>LER 373/84-054</b>
Salem 1	$5 \times 10^{-3}$	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. <b>LER 272/83-011</b>
Davis Besse 1	$2 \times 10^{-3}$	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 reset. <b>LER 346/81-037</b>
Brunswick 1	$7 \times 10^{-3}$	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. <b>LER 325/81-032</b>

Plant	CCDP/ ΔCDP	Date	Description
Millstone 2	$5 \times 10^{-3}$	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. <b>LER 336/81-005</b>
St. Lucie 1	$1 \times 10^{-3}$	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). <b>LER 335/80-029</b>
Davis Besse 1	$1 \times 10^{-3}$	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. <b>LER 346/80-029</b>
Crystal River 3	$5 \times 10^{-3}$	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. <b>LER 302/80-010</b>

Plant	CCDP/ ΔCDP	Date	Description
Hatch 2	$1 \times 10^{-3}$	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. <b>LER 366/79-045</b>
Oyster Creek	$2 \times 10^{-3}$	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. <b>LER 219/79-014</b>
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. <b>LER 320/79-012</b>
Salem 1	$1 \times 10^{-2}$	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. <b>LER 272/78-073</b>
Calvert Cliffs 1	$3 \times 10^{-3}$	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. <b>LER 317/78-020</b>
Farley 1	$5 \times 10^{-3}$	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. <b>LER 348/78-021</b>



Plant	CCDP/ ΔCDP	Date	Description
Rancho Seco	$1 \times 10^{-1}$	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. <b>LER 312/78-001</b>
Davis-Besse 1	$5 \times 10^{-3}$	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. <b>LER 346/77-110</b>
Davis-Besse 1	$7 \times 10^{-2}$	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. <b>LER 346/77-016</b>
Cooper	$1 \times 10^{-3}$	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). <b>LER 298/77-040</b>
Zion 2	$2 \times 10^{-3}$	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. <b>LER 304/77-044</b>
Millstone 2	$1 \times 10^{-2}$	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. <b>LER 336/76-042</b>
Kewaunee	$5 \times 10^{-3}$	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. <b>LER 305/75-020</b>

Plant	CCDP/ ΔCDP	Date	Description
Brunswick 2	9×10 <sup>-3</sup>	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. <b>LER 324/75-013</b>
Browns Ferry 1	2×10 <sup>-1</sup>	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. <b>LER 259/75-006</b>
Turkey Point 3	2×10 <sup>-2</sup>	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. <b>LER 250/74-LTR</b>
Point Beach 1	5×10 <sup>-3</sup>	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. <b>LER 266/74-LTR</b>
Point Beach 1	1×10 <sup>-3</sup>	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. <b>LER 266/71-LTR</b>

## 5.0 Insights and Other Trends

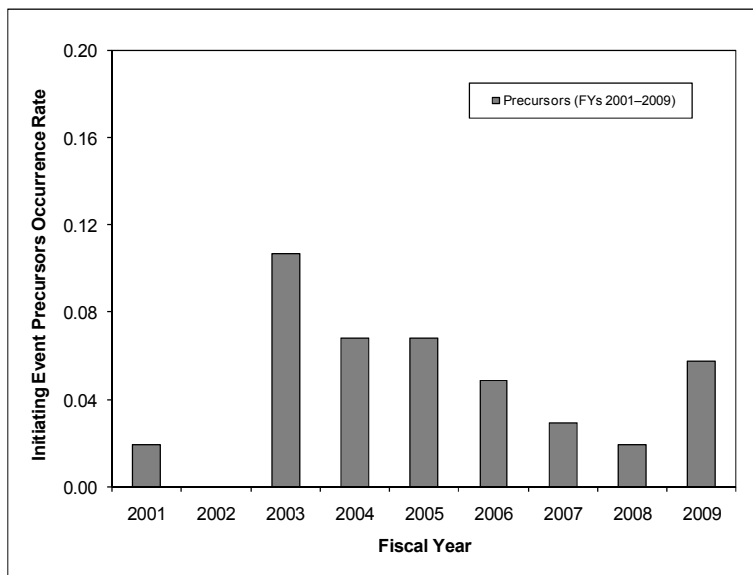
The following sections provide additional ASP trends and insights from the period FY 2001–2009.

## 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.

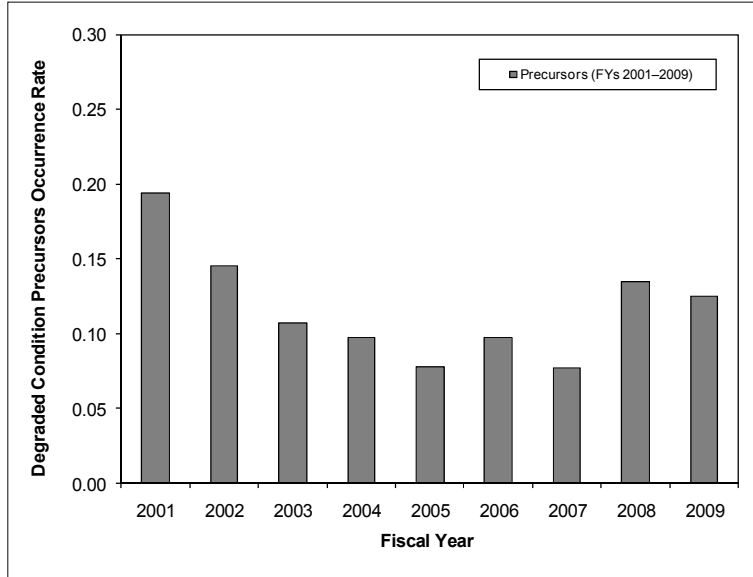


Figure 4. Precursors Involving Degraded Conditions.

## 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.

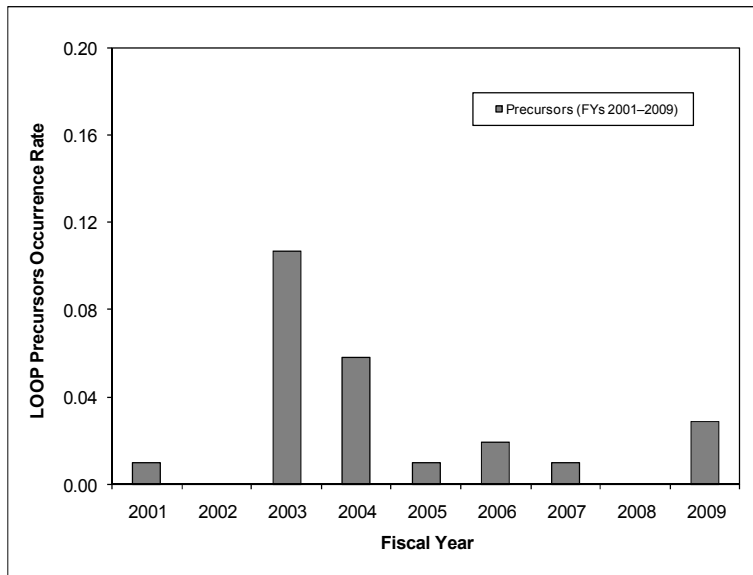


Figure 5. Precursors Involving LOOP Events.

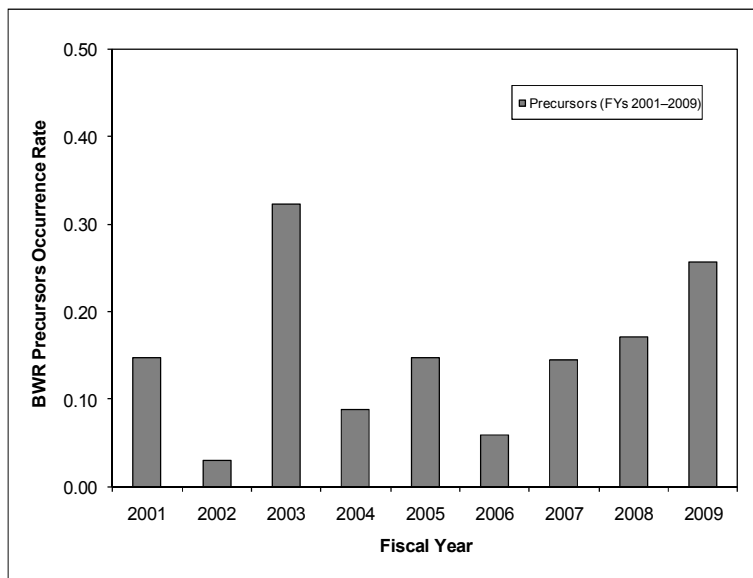
- Of the 25 LOOP events that occurred during the FY 2001–2009 period, 44 percent resulted from a degraded electrical grid outside of the NPP boundary. Eight of the 11 grid-related LOOP precursors were the result of the 2003 Northeast Blackout.
- A simultaneous unavailability of an emergency power system 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.<sup>7</sup>

#### **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.



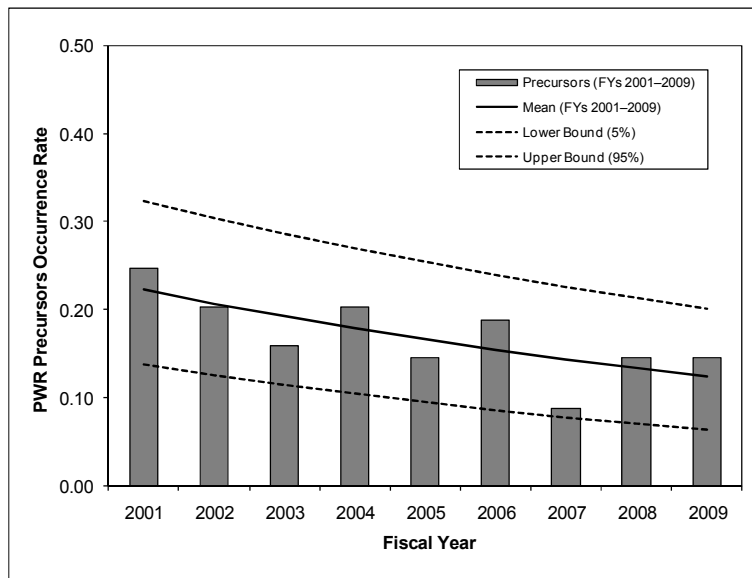
**Figure 6. Precursors involving BWRs.**

- 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).

<sup>7</sup> 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

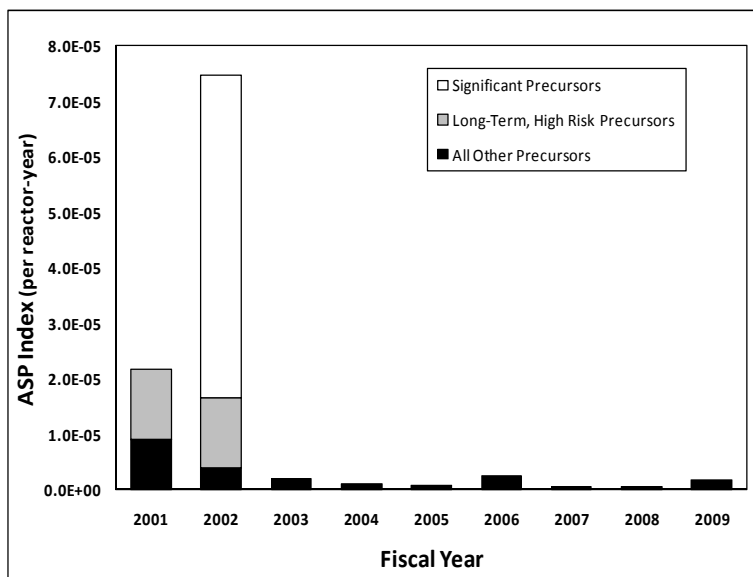
The staff derives the integrated ASP index for order-of-magnitude comparisons with industry-average core damage frequency (CDF) estimates derived from probabilistic risk assessments (PRAs) and NRC's standardized plant analysis risk (SPAR) models. The index or CDF from precursors for a given fiscal year is the sum of CCDPs and  $\Delta$ CDPs 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  $\Delta$ CDP of  $5 \times 10^{-6}$ . 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  $\Delta$ CDP of  $5 \times 10^{-6}$  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  $\Delta$ CDP greater than or equal to  $1 \times 10^{-3}$ ) 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  $\Delta$ CDPs of these two precursors were high ( $7 \times 10^{-4}$ ) 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  $\Delta$ CDPs 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.**

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

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. <b>LER 400/02-004</b>
2003	St. Lucie 2	Reactor pressure vessel head leakage because of cracking of control rod drive mechanism nozzles. <b>LER 389/03-002</b>
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. <b>LER 302/01-004, LER 289/01-002, LER 280/01-003, LER 339/02-001</b>
2002	Columbia	Common-cause failure of breakers used in four safety-related systems. <b>IR 50-397/02-05</b>
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. <b>LER 346/02-002</b>
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. <b>LER 483/01-002</b>
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. <b>LER 266/01-005</b>
2002	Shearon Harris	Potential failure of residual heat removal pump and containment spray pump because of debris in the pumps' suction lines. <b>LER 400/01-003</b>
2001	Oconee 1, 2, 3 Arkansas 1 Palisades	Reactor pressure vessel head leakage because of cracking of control rod drive mechanism nozzles. <b>LER 269/03-002, LER 270/02-002, LER 287/03-001, LER 313/02-003, LER 255/01-004</b>
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. <b>IR 50-305/02-06</b>

## 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  $1 \times 10^{-3}$ ) 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  $\Delta$ CDP greater than or equal to  $1 \times 10^{-4}$  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.

# Status of the Standardized Plant Analysis Risk Models

## 1.0 Background

The objective of the Standardized Plant Analysis Risk (SPAR) Model Program is to develop standardized risk analysis models and tools that staff analysts use in many regulatory activities, including the Accident Sequence Precursor (ASP) Program and Phase 3 of the Significance Determination Process (SDP). The SPAR models have evolved from two sets of simplified event trees initially used to perform precursor analyses in the early 1980s. Today's SPAR models for internal events are far more comprehensive than their predecessors. For example, the revised SPAR models include a new, improved loss of offsite power (LOOP)/station blackout module; an improved reactor coolant pump seal failure model; and updated estimates of accident initiator frequencies and equipment reliability based on more recent operating experience data.

The SPAR models consist of a standardized, plant-specific set of risk models that use the event-tree/fault-tree linking methodology. They employ a standard approach for event-tree development as well as a standard approach for input data for initiating event frequencies, equipment performance, and human performance. These input data can be modified to be more plant- and event-specific when needed. The system fault trees contained in the SPAR models generally are not as detailed as those contained in licensee probabilistic risk assessments (PRAs). To date, the U.S. Nuclear Regulatory Commission (NRC) staff has completed 77 SPAR models to represent all 104 commercial operating units and benchmarked them against licensee PRAs during the onsite quality assurance reviews of these models.

In August 2000, the staff initiated the SPAR Model Development Plan to address the following modeling issues:

- Internal initiating events during full-power operation.
- Internal initiating events during shutdown operations.
- External initiating events (including fires, floods, and seismic events).
- Calculation of large early release frequency (LERF).

The staff initiated the risk assessment standardization project (RASP) in February 2004. The primary focus of RASP is to standardize risk analyses in SDP Phase 3, ASP, and Management Directive (MD) 8.3, "NRC Incident Investigation Program." Under this project, the staff initiated the following activities:

- Enhancing SPAR models to be more plant specific and enhance the codes used to manipulate the SPAR models.
- Documenting consistent methods and guidelines for risk assessments of internal events during power operations, internal fires and floods, external events (e.g., seismic events and tornadoes), and internal events during shutdown operations.
- Providing on-call technical support for licensing and inspection issues.

## **2.0 SPAR Model Program Status**

The SPAR Model Program continues to play an integral role in the ASP analysis of operating events. Many other agency activities, such as the SDP analyses, MD 8.3 evaluations, and the Mitigating Systems Performance Index, involve the use of SPAR models. New SPAR models are under development in response to staff needs for assessing plant risk during shutdown operations and external events and for assessing accident progression to the plant damage state level.

In conformance with the SPAR Model Development Plan, the staff has completed the following activities in model and method development since the previous status report (SECY-09-143, "Status of the Accident Sequence Precursor Program and the Development of Standardized Plant Analysis Risk Models," dated September 29, 2009) as described below.

### ***Transition of SPAR Models to SAPHIRE Version 8***

In fiscal year (FY) 2010, the 77 SPAR models representing the 104 operating commercial nuclear power plants (NPPs) were revised and augmented to take advantage of the new features and capabilities of SAPHIRE Version 8. Model enhancement included improved modeling of common-cause failure events and handling of recovery rule linking, documentation, and parameter data updates. In addition, the staff continues to provide technical support for SPAR model users and risk-informed programs. The staff also completes about a dozen routine SPAR model updates annually.

### ***Technical Adequacy of SPAR Models***

The staff implemented an updated SPAR Model Quality Assurance Plan covering the SPAR models in 2006. The main objective of this plan is to ensure the SPAR models continue to be of sufficient quality for performing event assessments of operational events in support of the staff's risk-informed activities. The staff has processes in place to verify, validate, and benchmark these models according to the guidelines and standards established by the SPAR Model Program. As part of this process, the staff performs reviews of the SPAR models and results against the licensee PRA models. The staff also has processes in place for the proper use of these models in agency programs such as the ASP Program, the SDP, and the MD 8.3 process. These processes are documented in the RASP handbook.

In addition, the staff, with the cooperation of industry experts, performed a peer review of a representative boiling-water reactor (BWR) SPAR model and pressurized-water reactor SPAR model in accordance with ASME/ANS RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," and Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The staff has reviewed the peer review comments and has initiated projects to address these comments, where appropriate. This effort is planned to be completed in 2013.

### ***SPAR Models for the Analysis of External Events***

The staff previously completed a total of 15 SPAR external event (SPAR-EE) models. The staff continues to work with the Office of Nuclear Reactor Regulations (NRR) and the Office of New

Reactors (NRO) to identify needed enhancements to the SPAR-EE models. One significant upcoming activity is the incorporation of internal fire scenarios from the National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," pilot applications into the SPAR models. Three SPAR models which include both internal and external hazards were used in identifying and evaluating severe accident sequences for the consequential steam generator tube rupture (SGTR) project in support of the Steam Generator Action Plan. Consequential SGTR events are potentially risk significant due to the possibility of a severe core damage event (caused by either internal or external initiating events) leading to failure of the steam generator tubes and bypass of the containment building. The SPAR-EE models also were recently used to provide background information to NRR on the impact of assessing external hazard risk in 10 CFR 50.65 maintenance risk assessments.

### ***SPAR Models for Analysis of Internal Initiating Events during Shutdown Operation***

The staff places a priority on creating methods and guidance for the risk assessment of shutdown events, with emphasis on SDP Phase 3 analyses. In FY 2009, the staff developed a detailed shutdown model maker guideline document to provide consistent guidance for the construction of shutdown SPAR (SPAR-SD) models. Two new SPAR-SD models were developed using the modeling guidelines, resulting in a total of seven shutdown SPAR models available to support SDP Phase 3 evaluations of shutdown events and degraded conditions. The work will continue in FY 2011 to create one new shutdown SPAR model, bringing the total number of SPAR-SD models to eight.

### ***MELCOR Thermal Hydraulic Analysis for SPAR Model Success Criteria***

The staff has performed MELCOR analyses, using input decks developed under the State-of-the-Art Reactor Consequence Analysis Project, to investigate success criteria associated with specific Level-1 PRA sequences. In some cases, these analyses confirm the existing technical basis and in other cases they support modifications that can be made to increase the realism of the agency's SPAR models.

To date, calculations have been performed for a number of sequences for the Peach Bottom and Surry plants. These results will be incorporated in the technical bases supporting the Surry and Peach Bottom SPAR models, and some results can be readily extended to other plants. The Office of Nuclear Regulatory Research (RES) is continuing to pursue opportunities for broadening the scope of this effort in terms of the types of sequences being investigated, as well as the applicability of the work to more plants. This includes the planned development of additional MELCOR input models, the investigation of Level-1 PRA end-state characterization (e.g., realism of core damage surrogates), and planned interactions with the Electric Power Research Institute (EPRI).

This effort directly supports the agency's goal of using state-of-the-art tools that promote effectiveness and realism. Project plans and results are being communicated to internal and external stakeholders via mechanisms such as the Regulatory Information Conference and the industry's Modular Accident Analysis Program Users' Group.

### **3.0 Additional SPAR Model Activities**

#### *SAPHIRE Version 8*

SAPHIRE Version 8, released in April 2010, includes features and capabilities that are new or improved over Version 7 to address new requirements for risk-informed programs. User interfaces were developed for performing:

- SDP Phase 2 analyses with the SPAR models.
- Condition assessments for SDP Phase 3 and ASP analyses, and MD 8.3 evaluations.
- Initiating event assessments for ASP analyses and MD 8.3 evaluations.
- PRA analyses requiring more significant modeling or data revisions.

Features and capabilities also have been improved for SPAR model development and use. Enhanced SPAR models for internal events during power operations have been developed to use the new SDP Phase 2 analysis interface. A new data input method and code improvements for the SPAR models were developed. New capabilities to model and analyze LERF PRA models have been incorporated. SAPHIRE Version 8 also includes the capability to perform phase mission time analysis that is also useful for modeling within the SPAR-SD models. In addition, SAPHIRE Version 8 has been designed with unique capabilities to use the SPAR models in an integrated manner (i.e., different model types such as internal and external events models combined into one model). Improved PRA methods also have been implemented for common cause failure modeling and for sequence solving. Finally, the software's general functionality has been enhanced, and the interface layout has been made more user-friendly.

SAPHIRE Version 8 was developed closely with the user community. For example, in developing the SDP user interface, pilot training classes were provided to NRC end users. The staff also widely participated in testing beta versions and provided feedback for consideration in its development. Moreover, an NRC internal peer review assessed the software requirements. In addition to these types of quality assurance activities, independent verification and validation and acceptance testing were performed for SAPHIRE Version 8.

Increased use of SAPHIRE Version 8 for NRC PRA activities is anticipated as the SPAR models are transitioned from the SAPHIRE Version 7 format, and as users become trained on its new features and capabilities. User feedback will continue to be addressed.

#### ***Audit by the NRC Office of Inspector General***

The NRC Office of the Inspector General (OIG) completed an audit report, OIG-06-A-24, "Evaluation of the NRC's Use of Probabilistic Risk Assessment in Regulating the Commercial Nuclear Power Industry," dated September 29, 2006, which made the following three recommendations:

- Develop and implement a formal, written process for maintaining PRA models that is sufficiently representative of the as-built, as-operated plant to support model uses.
- Develop and implement a fully documented process to conduct and maintain configuration control of PRA software.

- Conduct a full verification and validation of SAPHIRE Version 7.

Recommendations 1 and 2 were previously addressed by the staff and closed. OIG acknowledged that performing a full verification and validation of SAPHIRE Version 7 was not justified because of the development schedule of SAPHIRE Version 8. The staff has implemented four recommendations to the SAPHIRE Version 8 project software verification and validation. These recommendations were based on a comparison of the SAPHIRE testing, verification and validation to the Institute of Electrical and Electronics Engineers Standard for Software Verification and Validation 1012-1998. Subsequent discussions with the OIG staff indicated that the addition of these four recommendations, combined with code testing, would satisfy full verification and validation of SAPHIRE Version 8. Testing of SAPHIRE Version 8 has been completed. OIG considers this issue resolved, and the issue was closed with the release of SAPHIRE Version 8 in April 2010.

### ***Evaluation of Extensive Damage Mitigation Strategies and Guidance***

This project is in support of Staff Requirements Memorandum COMGBJ-06-0004, dated April 14, 2006. The objective of this project is to evaluate the change in risk of the 104 NRC licensed commercial NPPs based on the implementation of extensive damage mitigation strategies and guidance required by 10 CFR 50.54(hh), if those strategies and guidance are used by the licensee to mitigate reactor accidents typically modeled in the SPAR models. An evaluation of 52 SPAR models has been completed as part of the first two phases of this project. The third and final phase will result in the evaluation of the remaining 25 SPAR models. This project is on schedule to be completed in October 2010.

### ***New Reactor SPAR Models***

Prior to new plant operation, the staff may need to perform risk assessments to confirm PRA results provided in licensing submittals or to evaluate risk-informed applications. Once the plants begin operation, the results from licensee PRAs or independent assessments using SPAR models may be used by the NRC staff for the evaluation of operational findings and events similar to the assessments performed for current operating reactors.

During FY 2010 the staff developed a design-specific internal events SPAR model for the Advanced Boiling Water Reactor (ABWR) design reactor design. As part of the SPAR model development, the requisite supporting documentation was also completed. The first draft of the ABWR model has been provided to NRO for review. The staff also plans to initiate work on developing a design-specific internal events SPAR model for the U.S. Advanced Pressurized-Water Reactor. Because design standardization is a key aspect of the new plants, it should only be necessary to develop one SPAR model for each of the new designs.

The AP1000 model was completed in February 2010 and has been transitioned to a routine maintenance status. It has also been optimized for SAPHIRE Version 8.

### ***Cooperative Research for PRA***

The staff has executed an addendum to the memorandum of understanding with EPRI to conduct cooperative nuclear safety research for PRA. Several of the initiatives included in the addendum are intended to help resolve technical issues that account for the key differences



between NRC SPAR models and licensee PRA models. The staff also continues to work with the National Aeronautical and Space Administration to address PRA issues of mutual interest. In addition, the NRC has utilized the cooperative agreement and grant program to establish collaborative PRA research projects with the University of Maryland and the Massachusetts Institute of Technology.

The objective of this effort is to work with the broader PRA community to resolve PRA issues and to develop PRA methods, tools, data, and technical information useful to both NRC and industry. The agency has established working groups that include support from RES, NRR, NRO, and the Regions. Initial cooperative efforts include the following:

- Support system initiating event analysis.
- Treatment of LOOP in PRAs.
- Treatment of uncertainty in risk analyses.
- Standard approach for injection following BWR containment failure.
- Standard approach for containment sump recirculation during small and very small loss-of-coolant accident.
- Human reliability analysis.
- Digital instrumentation and control risk methods.
- Advanced PRA methods.
- Advanced reactor PRA methods.

Significant efforts have been made in the past year in the areas of support system initiating event analysis, treatment of LOOP in PRAs, and treatment of uncertainty in risk analysis. For example, in the area of support system initiating event analysis, the staff and industry have come to agreement on a common approach to modeling support system initiators and worked together to resolve common cause issues that significantly affect model quantification results. The staff plans to use the support system initiating event methodology and the improved treatment of LOOP events to further enhance the realism and accuracy of the SPAR models. These methodologies are planned to be implemented in the SPAR models as one of the activities associated with addressing the peer review comments. The staff plans to continue these cooperative efforts with EPRI and other stakeholders to address the remaining issues over the next several years.