POLICY ISSUE
(Information)

October 6, 2014 SECY-14-0107

FOR: The Commissioners

FROM: Brian W. Sheron, Director
Office of Nuclear Regulatory Research

SUBJECT: STATUS OF THE ACCIDENT SEQUENCE PRECURSOR PROGRAM AND THE STANDARDIZED PLANT ANALYSIS RISK MODELS

PURPOSE:

To inform the Commission of the status, accomplishments, and results of the Accident Sequence Precursor (ASP) Program, including quantitative ASP results, and to communicate the status of the development and maintenance 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. Subsequently in SECY-02-0041, “Status of Accident Sequence Precursor and SPAR Model Development Programs,” the staff expanded the annual ASP status report to include: (1) the evaluation of precursor data trends and (2) the 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 (i.e., precursors[1]). The ASP Program provides insights into the NRC’s risk-informed and performance-based regulatory programs and

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[1] Enclosure 1 provides background on the process used by the staff to identify precursors.

monitors performance against safety measures in the agency's Congressional Budget Justification (see NUREG-1100, Volume 30, "FY15 Congressional Budget Justification," issued March 2014).

Under the SPAR Model Program, the staff develops and maintains independent risk-analysis tools and capabilities to support NPP-related risk-informed regulatory activities. The staff uses

SPAR models for the Reactor Oversight Process (ROP) Significance Determination Process (SDP); the ASP Program; Management Directive (MD) 8.3, "NRC Incident Investigation Program," event assessment process; and MD 6.4, "Generic Issues Program," resolution process. In addition, the staff uses SPAR models to risk-inform 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-13-0107, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," dated October 4, 2013.

ASP Program

The staff continues to review operational events from licensee event reports and inspection reports to identify potential precursors to a core damage event. Operational events that exceed the threshold mentioned previously are considered precursors in the ASP Program. *Significant* precursors have a conditional core damage probability (CCDP[2]) or a change in core damage probability (ΔCDP[3]) greater than or equal to 1×10^{-3}. The staff has identified twelve precursor events for fiscal year (FY) 2013. The staff did not identify any *significant* precursors for FY 2013, and has not identified any potentially *significant* precursors for FY 2014 to date, although detailed evaluations of some FY 2014 events are still in progress.

The ASP Program evaluates the trend for all precursors as an input to the agency's Industry Trends Program (ITP). The ITP provides an input to the agency's safety performance measure that is part of the Congressional Budget Justification of no significant adverse trend in industry safety performance. For the period of FY 2004 through FY 2013, the staff found no statistically significant trend for all precursors.

In addition to the trend analysis of all precursors provided for the ITP, the staff performs trend analyses on other precursor subgroups for additional insights. These subgroups include:

– Precursors with a CCDP or ΔCDP greater than or equal to 1×10^{-4}
– Precursors involving an initiating event
– Precursors involving degraded conditions
– Precursors involving a complete loss of offsite power

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2 The term CCDP is the probability of the occurrence of core damage given that an initiating event has occurred.

3 The term ΔCDP is the increase in probability of core damage (from the baseline core damage probability) due to a failure of plant equipment or an identified deficiency during the time the failure or deficiency existed.

- Precursors that occurred at boiling-water reactors (BWRs)
- Precursors that occurred at pressurized-water reactors (PWRs)

For the period of FY 2004 through FY 2013, the staff found a statistically significant increasing trend in precursors with a CCDP or ΔCDP greater than or equal to 1×10^{-4}; no statistically significant trends were identified for the other subgroups during this same period. This increasing trend results from the occurrence of seven precursors in this subgroup in the past four years after zero events were identified in the previous six years.[4] The staff reviewed these events for risk-informed insights, looking at the systems causing the events, the dominant risk sequences, and the plant types affected by the events. The most common similarity was that seven of the eight events were caused by multiple electrical failures. These electrical failures varied from failures of electrical equipment (such as circuit breakers) to losses of offsite power. Regulatory actions taken as a result of these events included the issuance of several enforcement actions, five information notices, and a bulletin (see Table 3 in Enclosure 1). However, no changes to the NRC's regulations were deemed necessary.

Enclosure 1, "Results, Trends, and Insights of the Accident Sequence Precursor Program," provides additional details on results and trends of the ASP Program.

SPAR Model Program

The staff continued to maintain and update the 79 SPAR models representing 104 commercial nuclear power reactors[5]. The scope of every SPAR model includes internal events, at power, through core damage (i.e., Level 1 model). In addition, the staff continued to expand SPAR model capability beyond internal events at full-power operation. For example, 20 of these 79 SPAR models, representing 24 nuclear power reactors, include other hazard groups and are referred to as SPAR All Hazard (SPAR-AHZ[6]) models. Currently, 17 of the SPAR-AHZ models include hazards such as fires, internal floods, and seismic events based on the results of the assessments conducted for Supplement 5, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities," to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," and other readily available information. The staff has also recently completed incorporation of internal fire scenarios from the fire probabilistic risk assessments (PRAs) done in compliance with National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," for the Shearon Harris Nuclear Power Plant and the Donald C. Cook Nuclear Power Plant. In addition to more detailed fire PRA modeling, the SPAR models for Harris and Cook include improved external hazard modeling and model validation. The staff has also leveraged the ongoing Level 3 PRA project for the Vogtle Electric Generating Plant, Units 1 and 2, to develop improved external hazard and fire modeling for the

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[4] No precursors with a CCDP or ΔCDP greater than or equal to 1×10^{-4} were identified in FY 2013. However, the seven precursors identified in the previous three years (FYs 2010-2012) combined with no precursors being identified for the preceding six years (FYs 2004-2009) still cause a statistically significant trend over the 10-year period for this subgroup.

[5] Three of the 79 SPAR models are associated with nuclear power plants that have permanently shut down (Kewaunee, San Onofre, and Crystal River). While these SPAR models are no longer being updated, they remain available for agency use.

[6] These models were formerly named SPAR external event (SPAR-EE) models, but have been renamed SPAR-AHZ to reflect recent improvements in external hazard modeling efforts and for consistency with the ASME PRA Standard model scope.

Vogtle SPAR model. In the new reactor area, the staff has developed SPAR models for the AP1000, Advanced Boiling Water Reactor (ABWR) (for both the Toshiba and General Electric-Hitachi designs), U.S. Advanced Pressurized Water Reactor (US-APWR), and the U.S. Evolutionary Power Reactor (U.S. EPR). The staff is expanding the capability of the AP1000 SPAR model to include hazards such as seismic, fire, flooding, and low-power shutdown events. The Office of Nuclear Regulatory Research staff continues to work with the Office of Nuclear Reactor Regulation and the Office of New Reactors to identify future enhancements to the SPAR models, including continuing the development of new all-hazard SPAR models.

In FY 2010, the staff completed PRA standard-based peer reviews of a representative BWR SPAR model and a representative PWR SPAR model. These peer reviews were performed in accordance with American Society of Mechanical Engineers (ASME)/ American Nuclear Society (ANS) RA-S-2008, "Standard for Level 1/Large Early Release Frequency 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 provide an appropriate tool to conduct an independent check on the technical adequacy of utility PRAs. The teams also identified a number of facts and observations related to areas where enhancements could be implemented on the SPAR models and supporting documentation. The staff prioritized these enhancements and is addressing high-priority issues as available resources permit. Major activities undertaken to address these peer-review items in FY 2014 include the following:

- Structuring the SPAR model documentation to more closely align with the structure of ASME/ANS PRA standard.

- Incorporating improved loss of offsite power modeling and support system initiating events modeling into the SPAR models (e.g., loss of service water or component cooling water).

The pace of these activities was significantly reduced during FY 2013 because of sequestration-related budget cuts. With funding restored in FY 2014, the staff continued the resolution of peer-review items, including documentation enhancements and model upgrades. The staff plans to complete the PWR and BWR SPAR Model peer-review enhancements in August 2015.

The staff continues to maintain and improve the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software to support the SPAR Model Program. SAPHIRE is a personal-computer-based software application used to develop PRA models and to perform analyses with SPAR Models. During FY 2014, significant SAPHIRE activities included the following:

- Oversight of the SAPHIRE software quality-assurance program, including performance of an annual audit of software quality-assurance activities, tools, and documents in accordance with NUREG/BR-0167, "Software Quality Assurance Program and Guidelines."

- Implementation of new SAPHIRE features, including: a truncation convergence function, a results editor feature to assist users in reviewing and analyzing model results, the ability to use an external PRA-solving engine that is widely used by U.S. utilities, and improved Level 2 PRA modeling capabilities.

- Continued research on advanced quantification methods to improve accuracy and calculation speeds.

Enclosure 2, “Status of the Standardized Plant Analysis Risk Models,” provides a detailed status of SPAR models and related activities.

Planned ASP and SPAR Model Activities

- The staff will continue the screening, review, and analysis (preliminary and final) of potential precursors for FY 2014 and FY 2015 events to support the monitoring of the agency’s safety measures.

- The staff will continue to assess the ASP Program screening criteria for enhancement considering lessons learned from the performance of initiating event analyses.

- The staff will continue to implement enhancements to the internal event SPAR models for full-power operations. Planned enhancements include incorporating new models for support-system initiators, revised success criteria based on insights from ongoing thermal-hydraulic analyses, and a periodic parameter data update.

- The staff will continue quality-assurance activities for both the agency SPAR models and the SAPHIRE code. This will ensure that agency risk tools continue to be of sufficient quality for performing SDP, ASP, and MD 8.3 event assessments in support of the staff’s risk-informed regulatory activities.

- The staff will continue to evaluate the need for additional SPAR model capability (beyond full-power internal events) based on experience gained from SDP, ASP, and MD 8.3 event assessments and feedback from user offices.

- The staff will continue development of new SPAR-AHZ models, including incorporation of modeling derived from the NFPA 805 application process. The staff will continue to develop new all-hazard SPAR model capabilities for operating reactors.

<u>SUMMARY</u>:

Under the ASP Program, the staff continues to evaluate the safety significance of operating events at NPPs and to provide insights into the NRC’s risk-informed and performance-based regulatory programs. The staff identified no *significant* precursors in FY 2014 for events evaluated to date. A statistically significant increasing trend in precursors with a CCDP or ΔCDP greater than or equal to 1×10^{-4} was observed. There is an increase of precursors in this subgroup with seven events in the past four years after zero events were identified in the previous six years. Six of the seven events were caused by various types of electrical failures (ranging from failures of electrical equipment such as circuit breakers to losses of offsite power). The SPAR Model 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.

/RA/

Brian W. Sheron, Director
Office of Nuclear Regulatory Research

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 U.S. Nuclear Regulatory Commission (NRC) staff as they relate to events that occurred during fiscal years (FYs) 2013 and 2014. 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 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 operational events that have a conditional core damage probability (CCDP[1]) or an increase in core damage probability (ΔCDP[2]) greater than or equal to 1×10^{-6}. That is, for any given operational event analyzed the likelihood of inadequate core cooling and severe core damage was greater than or equal to one in one million.

To identify potential precursors, the staff reviews operational events, including the impact of external events (e.g., fires, floods, and seismic events) from licensee event reports (LERs) and inspection reports (IRs) on a unit basis (i.e., a single event that affects a multiunit site is counted as a precursor for each unit). The staff then analyzes any identified potential precursors by calculating the probability of an event leading to a core damage state. An operational event can be one of two types—(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 characterized by the unavailability or degradation of equipment without the occurrence of an initiating event.

For the first type of event, the staff calculates a CCDP. This metric represents a conditional probability that a core damage state is reached given the occurrence of an initiating event (and any subsequent equipment failure or degradation). For the second type of event, the staff calculates a ΔCDP. This metric represents the increase in core-damage probability for a time period during which a component or multiple components are deemed unavailable or degraded.

The ASP Program defines an event with a CCDP or a ΔCDP greater than or equal to 1×10^{-6} to be a precursor. For initiating event analyses, and to focus analyses on the more safety-significant events, the ASP Program excludes as precursors reactor transients whose results would be similar to or less significant than the loss of balance-of-plant systems (e.g., feedwater and condenser heat sink) with no degradation of safety-related equipment. Therefore, the ASP Program uses an initiating-event precursor threshold of a CCDP of 1×10^{-6} or the plant-specific

[1] The term CCDP is the probability of the occurrence of core damage given that an initiating event has occurred.

[2] The term ΔCDP is the increase in probability of core damage (from the baseline core damage probability) due to a failure of plant equipment or an identified deficiency during the time the failure or deficiency existed.

CCDP[3] for the non-recoverable loss of balance-of-plant systems, whichever is greater. Since 1988, this initiating-event precursor threshold screens out reactor trips with no losses of safety-system equipment from being precursors because of their relatively low risk significance. The ASP Program defines a *significant* precursor as an event with a CCDP or ΔCDP greater than or equal to 1×10^{-3}.

Figure 1 illustrates the complete ASP analysis process.

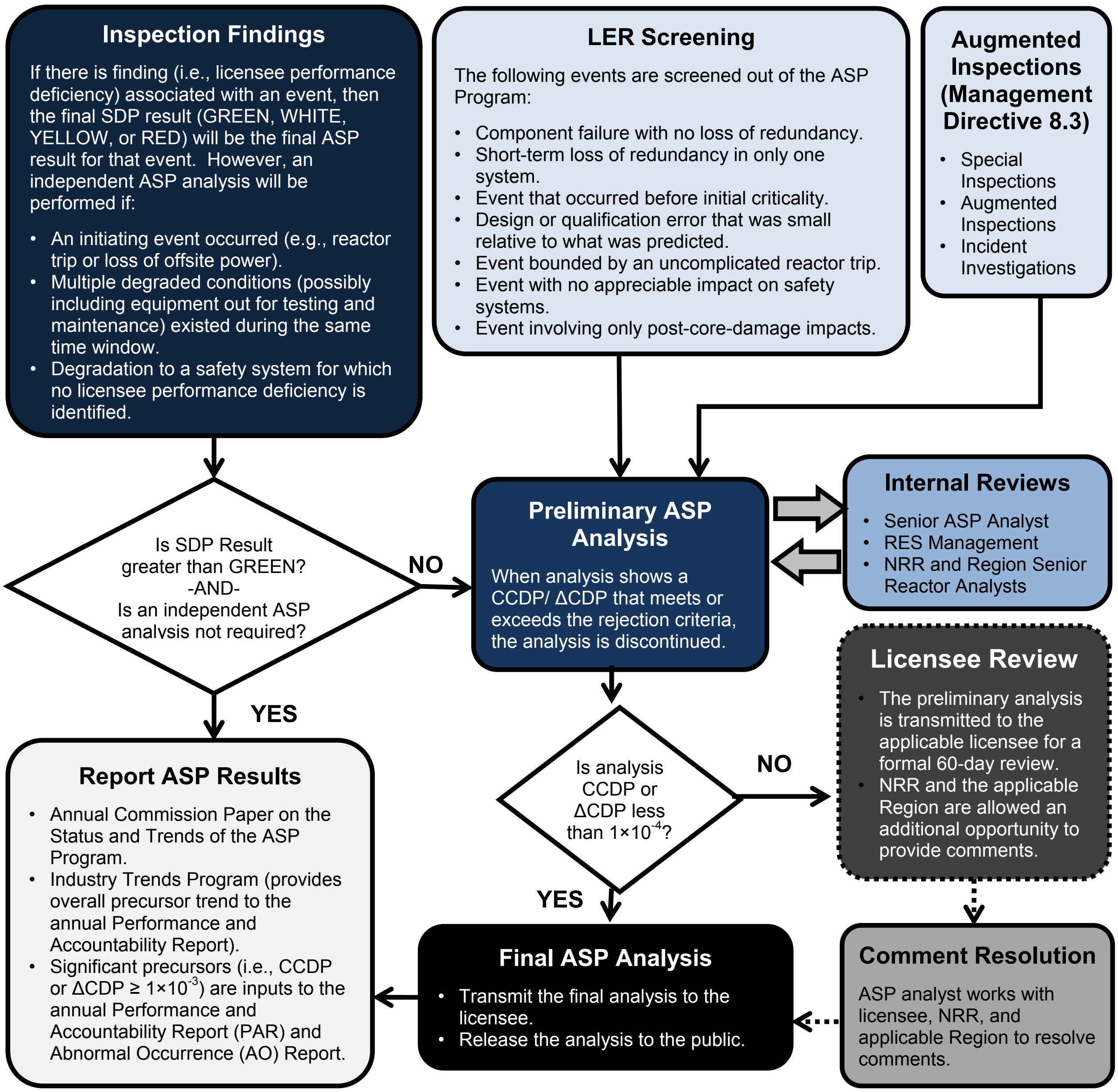

Figure 1. ASP process flowchart.

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3 The plant-specific CCDP is determined using NRC's Standardized Plant Analysis Risk (SPAR) models to analyze the non-recoverable loss of the condenser heat sink and the non-recoverable loss of main feedwater initiating events for each plant. If the results from either of these analyses are greater than 1×10^{-6}, the highest value is used as the precursor threshold for the subject plant.

Program Objectives. The ASP Program has the following objectives:

- Provide a comprehensive, risk-informed view of NPP operating experience and a measure for trending core-damage risk.
- Provide a partial validation of the current state of practice in risk assessment.
- Provide feedback to regulatory activities.

The NRC also uses the ASP Program results as an input to the NRC's Abnormal Occurrence Report and to monitor performance against the safety measures in the agency's Congressional Budget Justification (Ref. 1), which was formulated to support the agency's safety and security strategic goals and objectives. Specifically, the program provides input to the following safety measures:

- Zero events per year identified as a *significant* precursor of a nuclear reactor accident.
- No more than one significant adverse trend in industry safety performance (determination principally made from the Industry Trends Program but partially supported by ASP results).

Program Scope. The ASP Program is one of three agency programs that assess the risk significance of 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," and in Inspection Manual Chapter (IMC) 309, "Reactive Inspection Decision Basis for Reactors." The SDP evaluates the risk significance of licensee performance deficiencies, while assessments performed under MD 8.3 or IMC 309 are used to determine the appropriate level of reactive inspection in response to an event. Compared to the other two programs, the ASP Program assesses an additional scope of operating experience at U.S. NPPs. For example, the ASP Program analyzes initiating events as well as degraded conditions for which no identified deficiency occurred in the licensee's performance. The ASP Program also evaluates 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, 2014.

FY 2013 Analyses. The ASP analyses for FY 2013 identified 17 precursors (6 initiating events and 11 degraded conditions). An additional WHITE finding, identified under the SDP, was bounded by a non-recoverable loss of condenser heat sink and thus was screened out as an ASP precursor. One precursor occurred while the plant was shutdown. For 14 of the 17 precursors, the performance deficiency identified under the Reactor Oversight Process (ROP) documented the risk-significant aspects of the event completely. In these cases, the SDP significance category (i.e., the "color" of the finding) is reported in the ASP Program. For the remaining events, an independent ASP analysis was performed to determine the risk significance of three loss of offsite power (LOOP) initiating events.

Preliminary ASP analyses for loss of offsite power that occurred at LaSalle, Units 1 and 2 precursor events on April 17, 2014 will be issued as final after completion of internal reviews in accordance with the ASP review process (see Ref. 2 and Figure 1).

Table 1 presents the results of the staff's ASP analyses for FY 2013 precursors that involved initiating events. Table 2 presents the analysis results for FY 2013 precursors that involved degraded conditions.

Table 1. FY 2013 Precursors Involving Initiating Events

Event Date	Plant	Description	CCDP
12/22/12	Browns Ferry 2	Unplanned automatic reactor scram due to loss of power to the reactor protection system. ***LER 260/12-006***	WHITE[4]
2/8/13	Pilgrim	LOOP events due to Winter Storm Nemo. ***LER 293/13-003***	8×10^{-5}
3/31/13	Arkansas Nuclear One 1	Generator Stator drop causing Unit 1 LOOP while shutdown and Unit 2 trip with loss of Switchgear 2A1. ***LER 313/13-001***	YELLOW[5]
3/31/13	Arkansas Nuclear One 2	Generator Stator drop causing Unit 1 LOOP while shutdown and Unit 2 trip with loss of Switchgear 2A1. ***LER 313/13-001***	YELLOW
4/17/13	LaSalle 1	Dual Unit Loss of Offsite Power Due to Lightning Strike. ***LER 373/13-002***	1×10^{-5}
4/17/13	LaSalle 2	Dual Unit Loss of Offsite Power Due to Lightning Strike. ***LER 373/13-002***	1×10^{-5}

Table 2. FY 2013 Precursors Involving Degraded Conditions

Condition Duration	Plant	Description	ΔCDP/ SDP Color
34 days	Robinson	Failure of dedicated shutdown diesel generator. ***Enforcement Action (EA)-13-129***	WHITE
21 years[6]	Dresden 2[7]	Failure to establish procedure to address the effect of external flooding on the plant. ***EA-13-079***	WHITE
21 years[4]	Dresden 3[5]	Failure to establish procedure to address the effect of external flooding on the plant. ***EA-13-079***	WHITE
31 years[4]	Sequoyah 1[5]	Inadequate electrical conduit seals for the Essential Raw Cooling Water Pumping Station could result in loss of diesel generators during a flooding event. ***EA-13-045***	WHITE
30 years[4]	Sequoyah 2[5]	Inadequate electrical conduit seals for the Essential Raw Cooling Water Pumping Station could result in loss of diesel generators during a flooding event. ***EA-13-045***	WHITE
1 year	Monticello[5]	Failure to maintain flood plan to protect the site against external flooding events. ***EA-13-096***	YELLOW
22 days	Duane Arnold	Emergency diesel generator inoperability results in safety system's functional failure. ***EA-13-182***	WHITE

4 A WHITE finding corresponds to a licensee performance deficiency of low-to-moderate safety significance and has an increase in core-damage frequency in the range of greater than 10^{-6} to 10^{-5} per reactor year.

5 A YELLOW finding corresponds to a licensee performance deficiency of moderate-to-high safety significance and has an increase in core-damage frequency in the range of greater than 10^{-5} to 10^{-4} per reactor year.

6 Note that although these degraded conditions lasted for many years, ASP and SDP analyses limit the exposure period to 1 year.

7 These seven events resulted from the efforts undertaken by licensees and inspectors as part of the Fukushima Near-Term Task Force Recommendation 2.3 walkdowns (Ref. 8).

Condition Duration	Plant	Description	ΔCDP/ SDP Color
17 years[4]	Point Beach 1[5]	Flooding procedure failed to protect safety-related equipment. ***EA-13-125***	WHITE
17 years[4]	Point Beach 2[5]	Flooding procedure failed to protect safety-related equipment. ***EA-13-125***	WHITE
25 days	Waterford 3	Emergency diesel generator inoperable due to room exhaust-fan fire. ***EA-13-233***	WHITE
64 days	Duane Arnold	Reactor core isolation cooling turbine trip. ***EA-13-223***	WHITE

FY 2014 Analyses. The staff performs an initial review of all events to determine if they have the potential to be *significant* precursors. Specifically, the staff reviews LERs (in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.73, "Licensee Event Report System") and daily event-notification reports (in accordance with 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors") to identify potential *significant* precursors. The staff has completed the initial review of FY 2014 events and identified no potentially *significant* precursors (as of September 30, 2014). The staff will inform the Commission if a *significant* precursor is identified during the more detailed evaluations of events.

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 less than or equal to 0.05 indicate that there is 95 percent confidence that a trend exists in the data (i.e., leading to a rejection of the null hypothesis of no trend).

Data Coverage. The data period for the ASP trending analyses is a rolling 10-year period aligned with a rolling 10-year period used in the Industry Trends Program.

4.1 Occurrence Rate of All Precursors

The NRC's Industry Trends Program 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 in the NRC's Strategic Plan). The mean occurrence rate of all precursors identified by the ASP Program is one indicator used by the Industry Trends Program to assess industry performance.[8]

Results. The mean occurrence rate of all precursors does not exhibit a statistically significant trend (p-value = 0.956) for the 10-year period from FY 2004–2013 (see Figure 2).

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

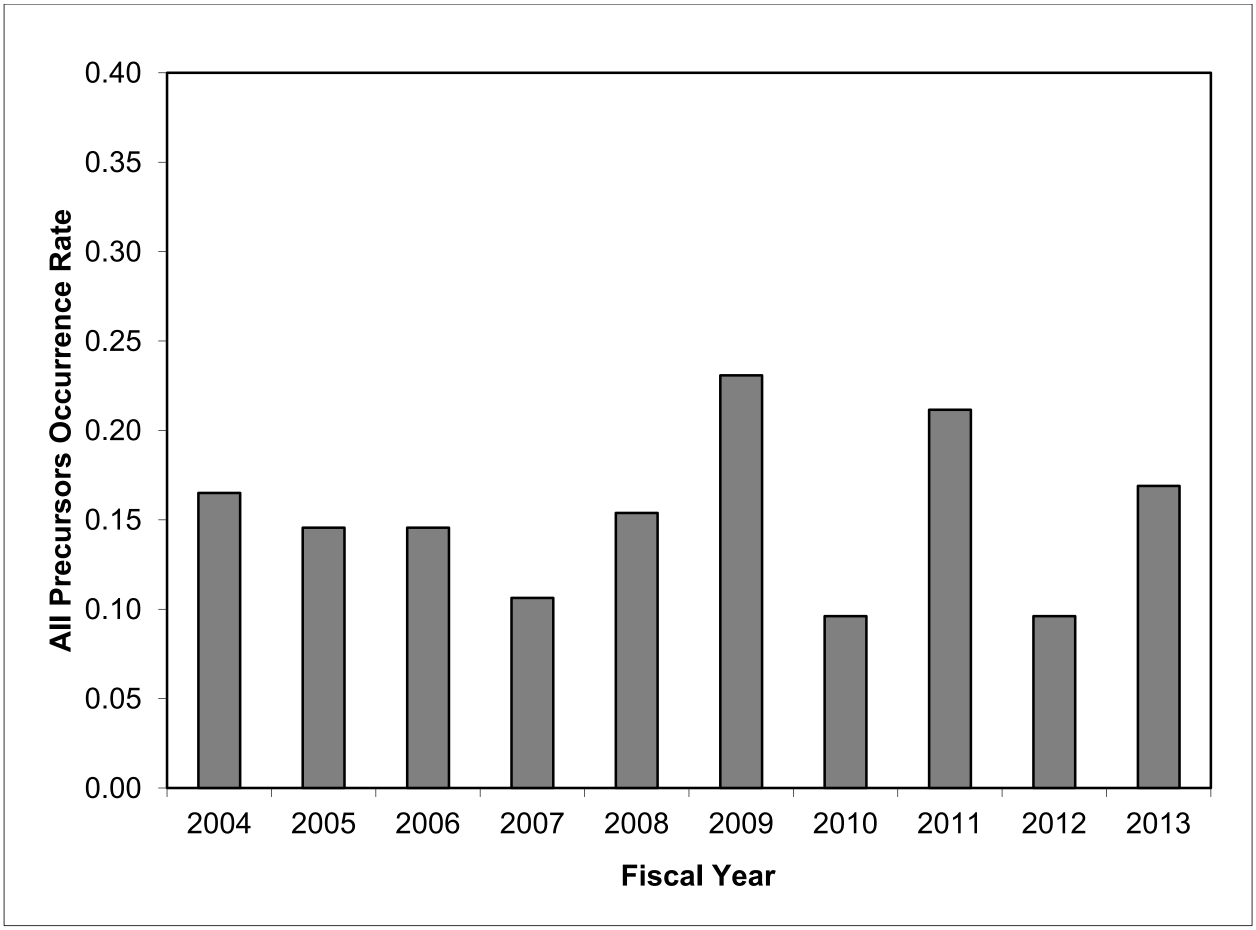

Figure 2. Occurrence rate of all precursors shows no statistically significant trend for the period from FY 2004 through FY 2013 (p-value = 0.956).

4.2 *Significant* Precursors

The ASP Program provides the input for determining if the safety measure regarding the "number of significant accident sequence precursors of a nuclear reactor accident" is zero. This is a safety measure associated with the safety goal in the NRC's Congressional Budget Justification (Ref. 1).

Results. A review of the data for the 10-year period from FY 2004 through FY 2013 reveals the following insights:

- No *significant* precursors have been identified during FYs 2004 through FY 2013. The staff has completed the initial review of FY 2014 events and identified no potentially significant precursors (as of September 30, 2014).

- The last *significant* precursor was identified in FY 2002 and involved concurrent, multiple degraded conditions at the Davis-Besse nuclear power plant.[9]

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[9] Commission Paper SECY-10-0125, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models" (Ref. 7), provides a complete list of all *significant* precursors from 1969 through 2010.

5.0 Insights and Other Trends

The following sections provide additional ASP trends and insights for the 10-year period from FY 2004 through FY 2013.

5.1 Occurrence Rate of Precursors with a CCDP or ΔCDP ≥ 1×10^{-4}

Precursors with a CCDP or ΔCDP ≥ 1×10^{-4} are considered important in the ASP Program because they generally have a CCDP higher than the annual core-damage probability (CDP) estimated by most plant-specific probabilistic risk assessments (PRAs).

The staff did not identify any precursors with a CCDP or ΔCDP ≥ 1×10^{-4} for FY 2013. Over the past 10-year period (FY 2004 through FY 2013), a total of seven precursors with CCDP or ΔCDP ≥ 1×10^{-4} were identified. These seven precursors were identified between FY 2010 and FY 2012. As summarized in Table 3, the staff issued a total of six generic communications involving five information notices (INs) and one bulletin relating to four of these events. In addition, the staff issued two RED findings, one YELLOW finding, and three WHITE findings based on identified performance deficiencies associated with these precursor events.

Table 3. FY 2010–2013 Precursors with a CCDP or ΔCDP ≥ 1×10^{-4})

Date	Plant (Risk Measures)	Event or Condition	Risk Insights (Generic Communications)
3/28/10	H. B. Robinson **(CCDP = 4×10^{-4})**	Fire causes loss of non-vital buses along with a partial loss of offsite power with reactor coolant pump seal cooling challenges. ***LER 261/10-002***	Neither the fire nor the minor equipment failures individually should have led to a high risk event. However, poor operator performance created a much higher risk scenario. Risk was dominated by transient-induced reactor coolant pump seal loss of coolant accidents (LOCAs). The SDP assessment resulted in **two WHITE findings** (one performance deficiency was for failure to adequately implement the requirements contained in OPS-NGGC-1000, "Fleet Conduct of Operations," and the other performance deficiency was for improper implementation of the Commission-approved requalification program). (NRC IN 2010-09, "Importance of Understanding Circuit Breaker Control Power Indications.")
10/23/10	Browns Ferry, Unit 1 **(RED Finding[10])**	Failure to establish adequate design control and perform adequate maintenance causes valve failure that led to a residual heat removal loop being unavailable. ***EA-11-018***	A valve failure coupled with a hypothetical fire that required execution of self-induced station blackout (SBO) procedures would have led to an unrecoverable situation. The self-induced SBO procedures added one to two orders of magnitude to the risk of this event. Risk was dominated by fire-initiated scenarios. (NRC IN 2012-14, "Motor-Operated Valve Inoperable due to Stem-Disc Separation.")

10 A RED finding corresponds to a licensee performance deficiency of high safety significance and has an increase in core-damage frequency greater than 10^{-4}.

Date	Plant (Risk Measures)	Event or Condition	Risk Insights (Generic Communications)
6/7/11	Fort Calhoun **(RED Finding)**	Fire in safety-related 480-volt electrical breaker because of deficient design controls during breaker modifications. Eight other breakers were susceptible to similar fires. ***EA-12-023***	The plant operated with a poorly designed modification to nine breakers, all of which had a potential for a fire, especially in a relatively minor seismic event. Risk comes from a very wide variety of sequences.
8/23/11	North Anna, Unit 1 **(CCDP = 3×10^{-4})**	Dual unit loss of offsite power caused by earthquake that coincided with the Unit 1 turbine-driven auxiliary feedwater (AFW) pump being out of service because of testing and the subsequent failure of a Unit 2 emergency diesel generator (EDG). ***LER 338/11-003***	Earthquake coupled with routine maintenance on the AFW pump and an unrelated failure of an EDG. Risk was dominated by SBO sequences. The SDP assessment resulted in **a WHITE finding** (one performance deficiency was for failure to establish and maintain maintenance procedures appropriate to the circumstances for the safety-related EDGs). (NRC IN 2012-01, "Seismic Considerations – Principally Issues Involving Tanks," and NRC IN 2012-25, "Performance Issues with Seismic Instrumentation and Associated Systems for Operating Reactors.")
1/13/12	Wolf Creek **(CCDP = 5×10^{-4})**	Multiple switchyard faults cause reactor trip and subsequent loss of offsite power. ***LER 482/12-001***	A LOOP of moderate length (two to three hours) caused by equipment failures in the switchyard. Risk was dominated by SBO sequences. ASP evaluated the LOOP initiating event while the SDP analysis performed a condition assessment on the loss of the startup transformer resulting in **a YELLOW finding** (one performance deficiency was for failure to identify that electrical maintenance contractors had not installed insulating sleeves on wires that affected the differential current protection circuit, contrary to work-order instructions).
1/30/12	Byron, Unit 2 **(CCDP = 1×10^{-4})**	Transformer and breaker failures cause loss of offsite power, reactor trip, and de-energized safety buses. ***LER 454/12-001***	The key issue for this event is the potential for operators to fail to recognize this scenario. Operator errors could lead to SBO-like sequences. (NRC IN 2012-03, "Design Vulnerability in Electric Power System," and NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System.")
5/24/12	River Bend **(CCDP = 3×10^{-4})**	Loss of normal service water, circulating water, and feedwater due to electrical fault. ***LER 458/12-003***	Initiating event coupled with postulated loss of safety-related service water would lead to complete loss of heat sink.

Results. A review of the data for the 10-year period from FY 2004 through FY 2013 reveals the following insights:

- The mean occurrence rate of precursors with a CCDP or ΔCDP ≥ 1×10^{-4} exhibits a statistically significant trend (p-value = 0.007; see Figure 3).

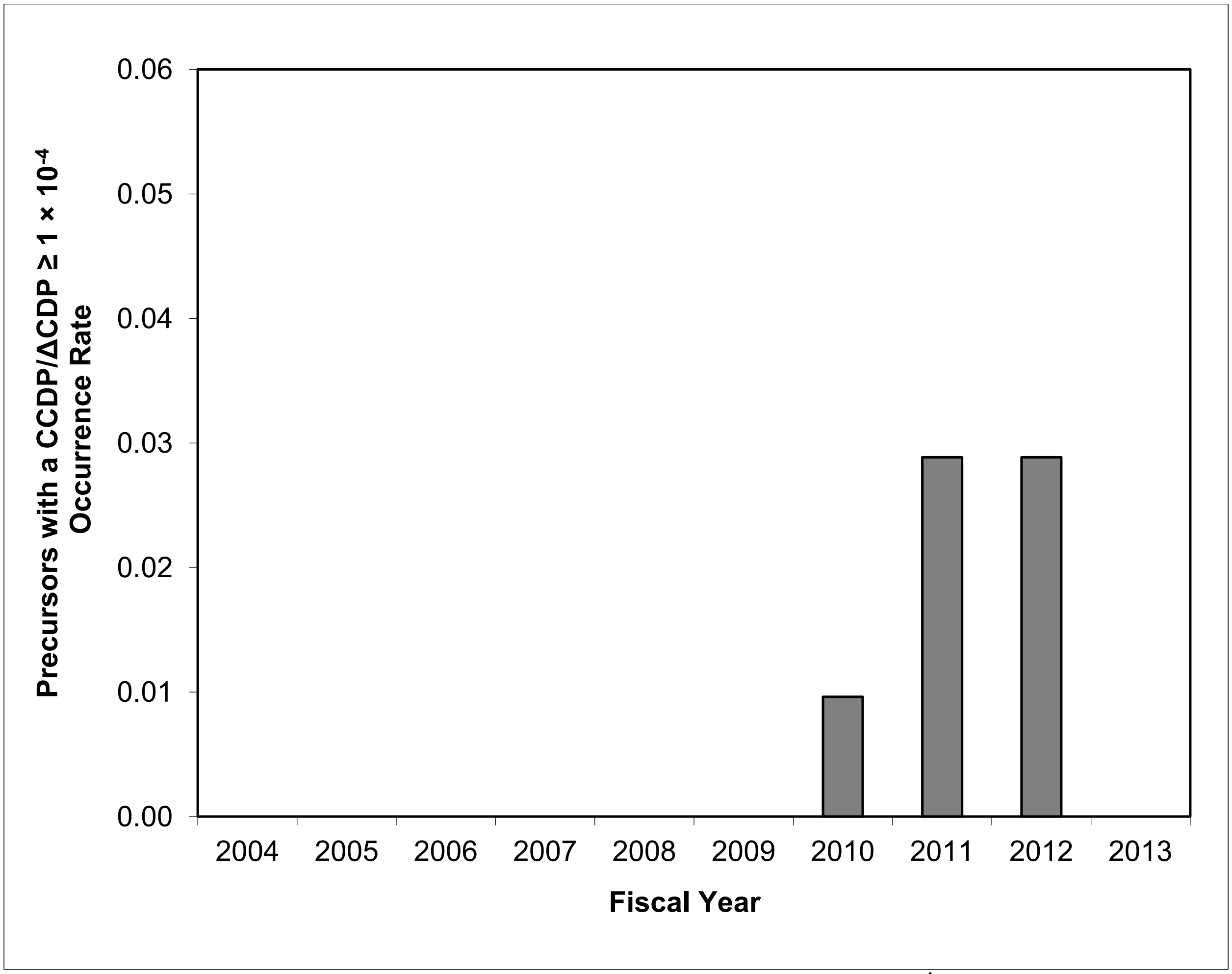

Figure 3. Occurrence rate of precursors with a CCDP or ΔCDP ≥ 1 × 10^{-4} shows a statistically significant trend for the period from FY 2004 through FY 2013 (p-value = 0.007).

Figure 3 shows that no precursor with a CCDP or ΔCDP ≥ 1×10^{-4} occurred between 2004 and 2009 and seven such precursors occurred during FYs 2010-2012. The events related to these precursors over this period involved differing reactor types, causes, systems, and components.

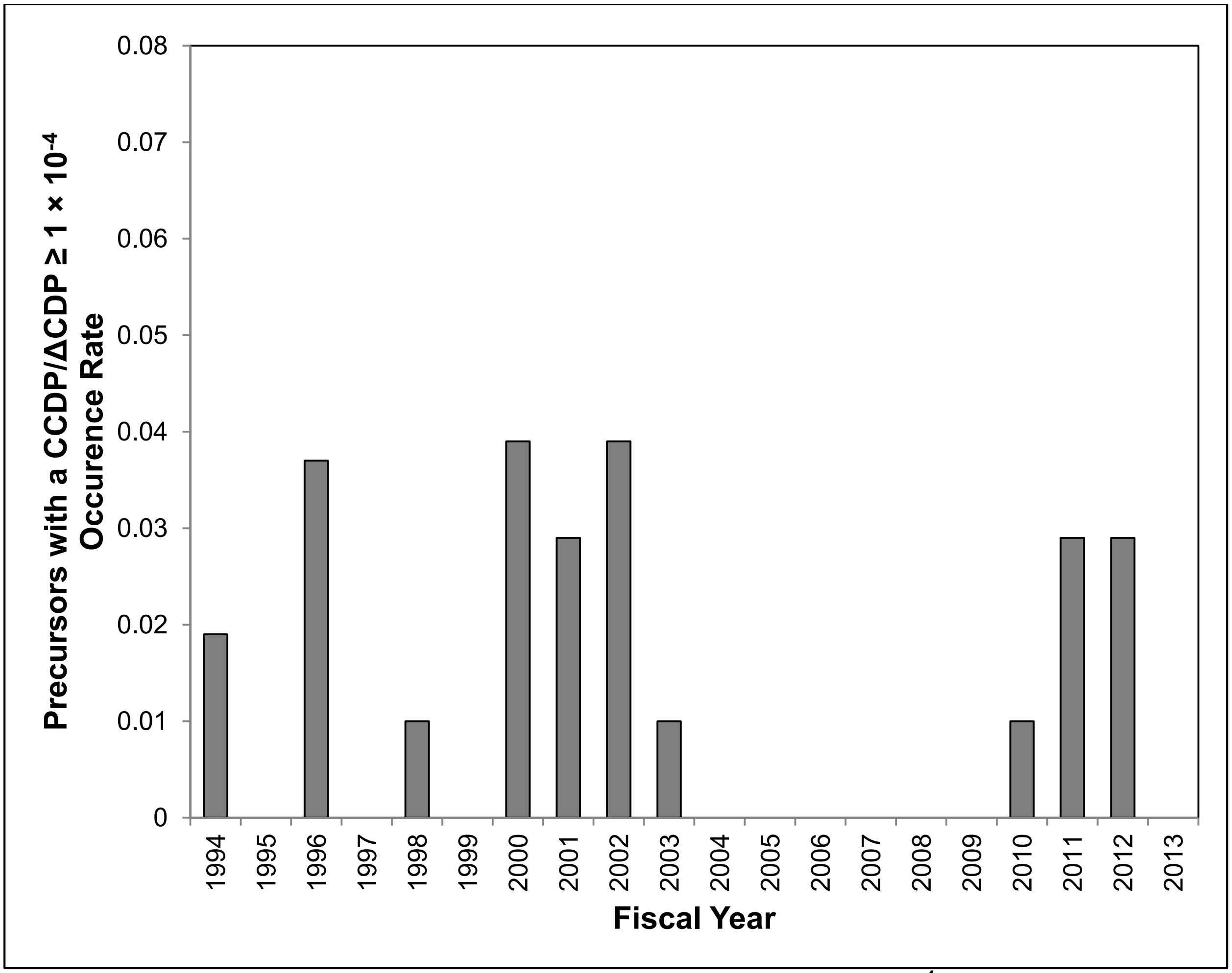

Figure 3A. Occurrence rate of precursors with a CCDP or ΔCDP ≥ 1 × 10^{-4} for the 20-year period from FY 1994 through FY 2013

Figure 3A shows the 20-year historical occurrence rate of precursors with a CCDP or ΔCDP ≥ 1×10^{-4}. Over the last 20 years, 27 precursors with a CCDP or ΔCDP ≥ 1×10^{-4} have occurred. Of these 27 precursors, 26 percent involved a LOOP initiating event. This is generally consistent with recent operating experience.

A review of the precursors in Table 3 reveals the following:

- Six of the seven precursors involved electrical events in electrical distribution systems. Six of the electrical events resulted in reactor trips, of which four were associated with LOOP initiating events. Fort Calhoun was in cold shutdown during the seventh electrical non-trip event.

- LOOP initiating events with no complications typically do not have a CCDP ≥ 1×10^{-4}. However, the three LOOP events reviewed featured complications that involved one or more additional failures or test/maintenance unavailabilities of standby safety equipment that resulted in higher CCDPs (North Anna, Byron, and Wolf Creek). The LOOP at Byron was unique in that operator action was required to establish emergency power to the safety buses because of a design vulnerability associated with a single-phase open-circuit condition.[11]

- Two precursors involved fires of electrical components caused by electrical faults (Robinson and Fort Calhoun). In the case of Robinson, multiple electrical fires occurred during the initial fault and a second fire was caused during plant restoration (i.e., the operating crew attempted to reset an electrical distribution system control relay before isolating the fault, which re-initiated the electrical fault and caused a second fire). The fires at Robinson were extinguished by plant personnel using dry chemical fire extinguishers. The electrical fire in a switchgear room at Fort Calhoun was extinguished by the automatic fire-suppression system.

- Four of the six precursors involving reactor trips had failures that were recoverable. In fact, the recovery actions were successfully implemented by the operators during each of these actual events.[12] These recovery actions were credited in the ASP analysis and contributed to risk reductions in these four events.

- Two of the seven precursors did not result in a reactor trip, but involved conditions resulting in the unavailability of safety components for some period of time (Browns Ferry 1, Fort Calhoun). These components were not recoverable in the time necessary to mitigate a hypothetical initiating event.

- Three precursors involved failures and initiators that contributed to rarely seen accident sequences.
  - The Robinson electrical fault with subsequent reactor trip resulted in a complete loss of reactor coolant pump (RCP) seal cooling and a partial loss of seal injection for 39 minutes. In PRA models, including the standardized plant analysis risk (SPAR) models, loss of RCP seal injection and cooling significantly increases the likelihood of a RCP seal loss-of-coolant accident (LOCA) within 13 minutes of the loss of seal injection and cooling. The operators restarted the charging pumps within one minute; however, an open valve in the charging system diverted flow away from the RCP seals. The operators recovered seal cooling at 13 minutes. Recovery of seal injection was not credited in the ASP analysis and recovery of seal cooling within 13 minutes was assigned a very high failure probability (0.8), which contributed to the high risk result.
  - The Bryon Unit 2 LOOP and design vulnerability resulted in the complete loss of electrical power to the safety buses. The operators were able to diagnose the problem and restore power from the emergency diesel generators (EDGs) to the safety buses in eight minutes. Offsite power was restored to both safety buses approximately 34 hours after the LOOP occurred. Recovery of emergency power to the safety bus before station battery depletion was modeled in the ASP analysis.
  - A beyond-design-basis earthquake at North Anna induced a LOOP event and subsequent reactor trips in both units. During the LOOP event, one of four EDGs onsite

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11 See NRC Bulletin 2012-01, “Design Vulnerabilities in Electric Power System” (Ref. 6).

12 Even though recovery actions were successfully accomplished during the actual events, the ASP Program does not take complete credit for these successful human actions. Human Reliability Analysis (HRA) is performed for each recovery action to calculate the probability of failure to recover. HRA considers complications in human performance that were observed during the actual event and impacts on human performance, both negative and positive, that could be experienced during each postulated accident sequence.

failed and the Unit 1 turbine-driven auxiliary feedwater (AFW) pump was out of service for surveillance testing. The station blackout diesel generator was manually aligned to the safety bus in 49 minutes. The turbine-driven AFW pump was placed back into service in 33 minutes. Offsite power was restored to all four safety buses approximately nine hours after the LOOP occurred. These recovery actions were modeled in the ASP analysis.

5.2 Precursors Involving Initiating Events and Degraded Conditions

A review of the data for the 10-year period from FY 2004 through FY 2013 reveals the following insights for precursors involving initiating events and degraded conditions.

Initiating Events

- The mean occurrence rate of precursors involving initiating events does not exhibit a trend that is statistically significant (p-value = 0.782) for the period from FY 2004 through FY 2013 (see Figure 4).

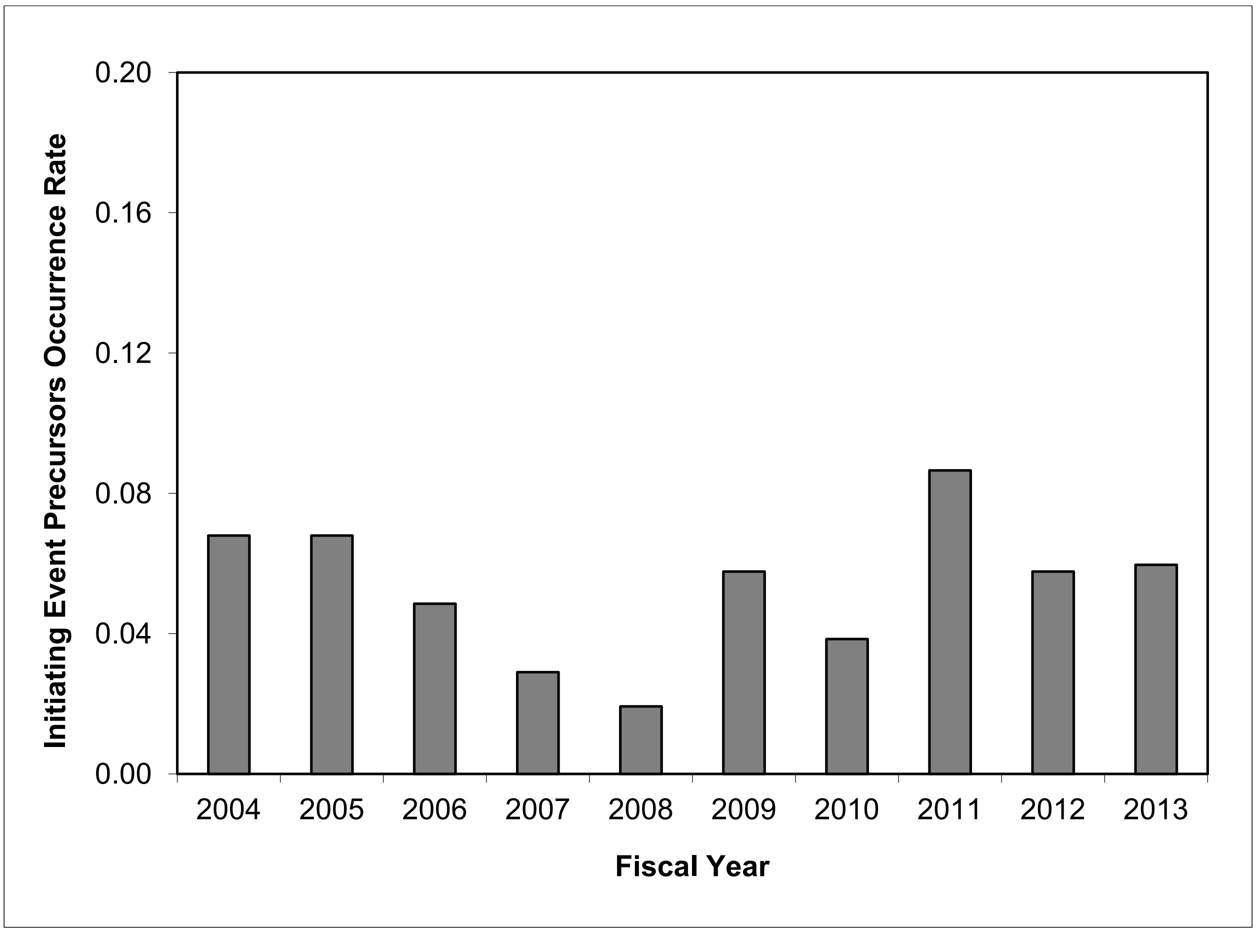

Figure 4. Occurrence rate of precursors involving initiating events shows no statistically significant trend for the period from FY 2004 through FY 2013 (p-value = 0.782)

- Of the 55 precursors involving initiating events, 55 percent were LOOP events. This is expected because uncomplicated transients typically do not exceed the ASP threshold (10^{-6}), while essentially all LOOPs do exceed the threshold. While the frequency of complicated transients is about the same as the frequency of LOOPs, the risk estimates for LOOPs are somewhat higher.

Degraded Conditions

- The mean occurrence rate of precursors involving degraded conditions does not exhibit a trend that is statistically significant (p-value = 0.939) during FY 2004 through FY 2013 (see Figure 5).

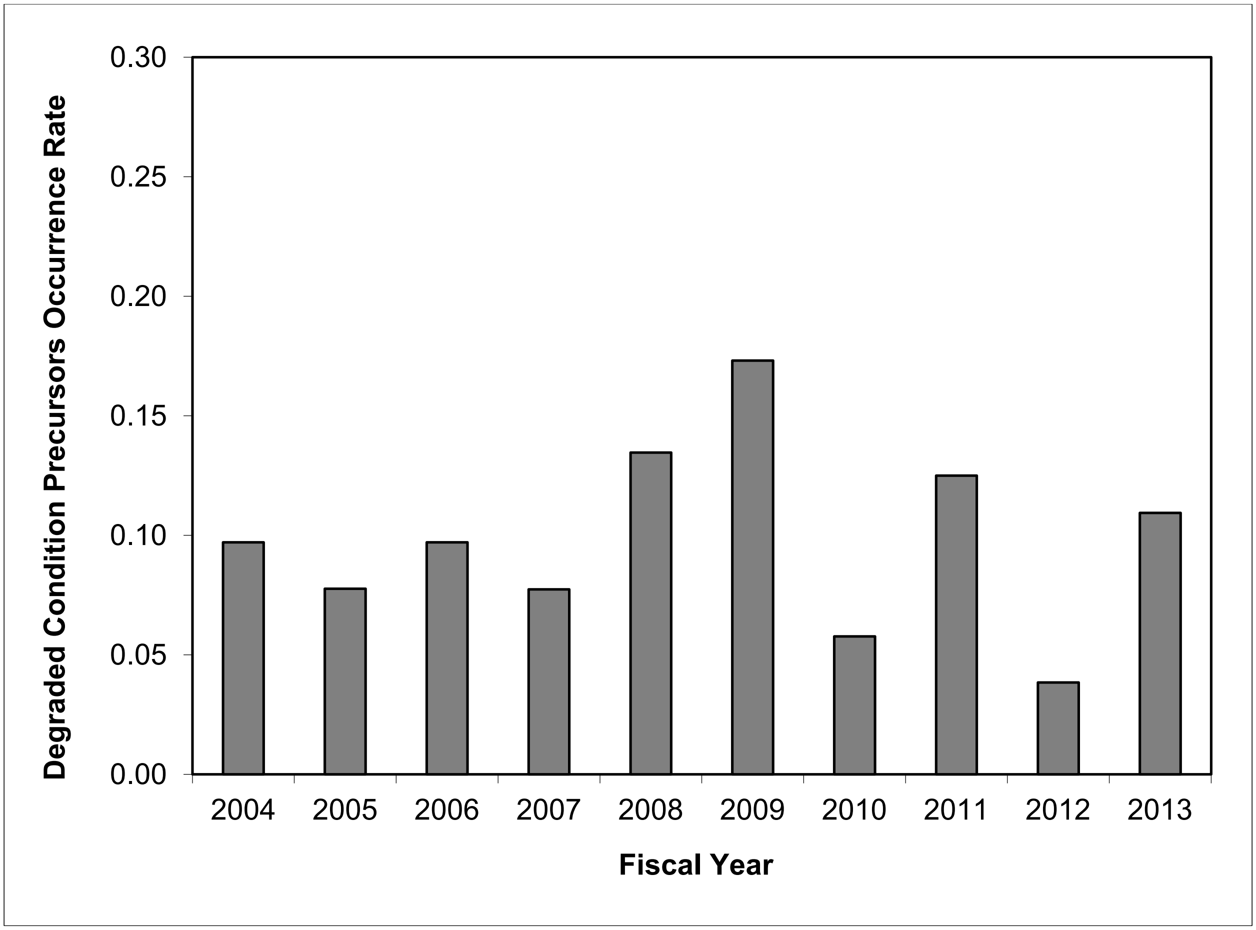

Figure 5. Occurrence rate of precursors involving degraded conditions shows no statistically significant trend for the period from FY 2004 through FY 2013 (p-value = 0.939)

- Over the past 10 years, precursors involving degraded conditions outnumbered initiating events by 85 percent.

- From FY 2004 through FY 2013, 33 percent of precursors involved degraded conditions existing for a decade or longer.[13] Of these precursors, 44 percent involved degraded conditions dating back to initial plant construction.

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13 Note that although these degraded conditions lasted for many years, ASP analyses limit the exposure period to 1 year.

5.3 Precursors Involving a Complete Loss of Offsite Power Initiating Event

In FY 2013, five precursors resulted from a complete LOOP initiating event. Typically, all complete LOOP events meet the precursor threshold.

Results. A review of the data for the 10-year period from FY 2004 through FY 2013 reveals the following insights:

- The mean occurrence rate of precursors resulting from a LOOP does not exhibit a statistically significant trend (p-value = 0.371; see Figure 6).

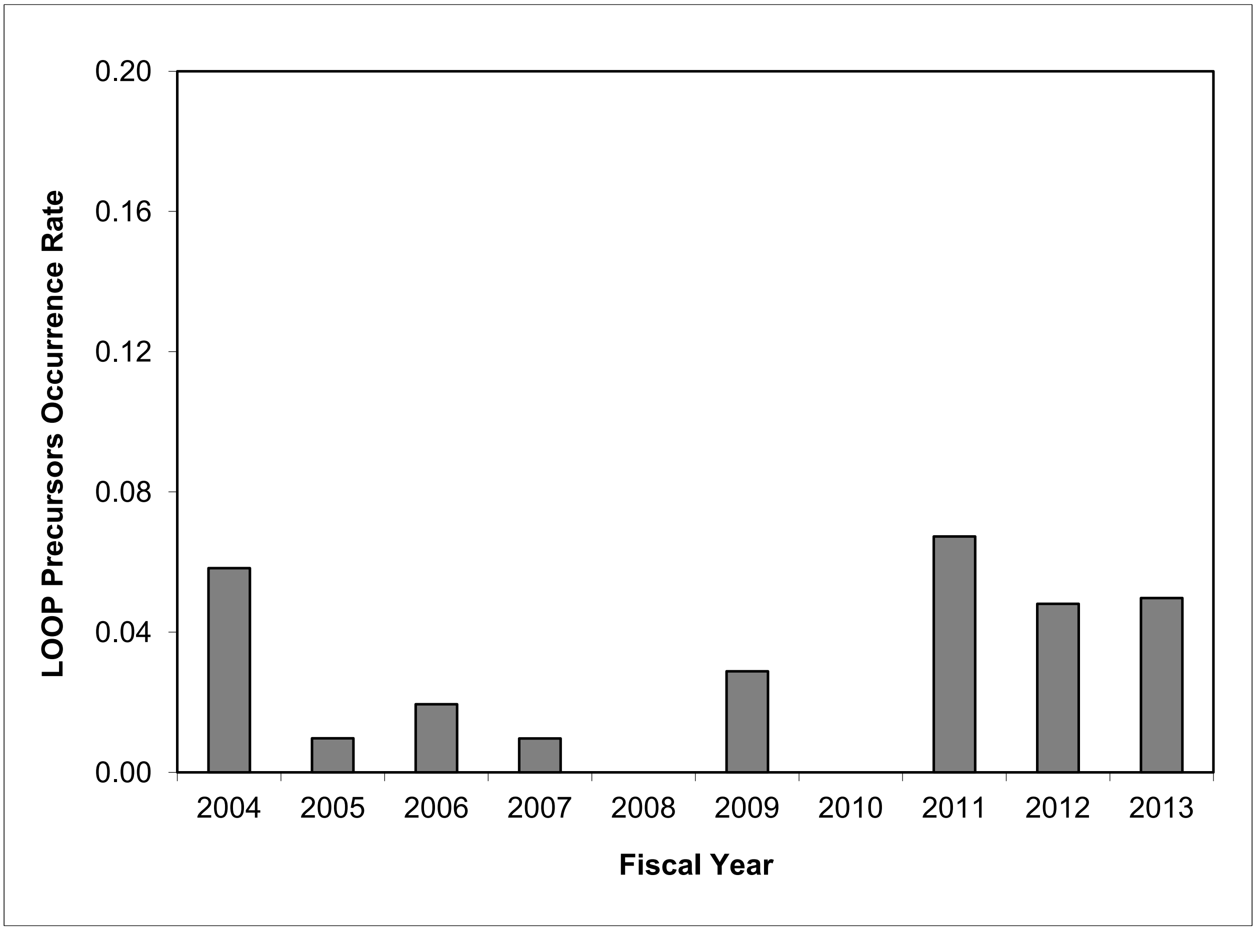

Figure 6. Occurrence rate of precursors involving LOOP events shows no statistically significant trend for the period from FY 2004 through FY 2013 (p-value = 0.371)

- Of the 30 LOOP precursors, 43 percent resulted from external events and 13 percent resulted from a degraded electrical grid outside of the NPP boundary. Seven of the 13 LOOP precursors that were caused by external events occurred in FY 2011[14]. This is unusual and unprecedented, but there is no indication of a trend from these events.

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[14] These FY 2011 events were Surry Units 1 and 2 tornado precursor events that occurred on April 16, 2011, Browns Ferry Units 1, 2, and 3 tornado precursor events that occurred on April 27, 2011, and North Anna Units 1 and 2 earthquake precursor events that occurred on August 23, 2011.

- Three of the 30 LOOP precursor events involved the simultaneous unavailability of an emergency power system train.

5.4 Precursors at BWRs and PWRs

A review of the data for the 10-year period from FY 2004 through FY 2013 reveals the following insights for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs).

BWRs

- The mean occurrence rate of precursors that occurred at BWRs does not exhibit a statistically significant trend (p-value = 0.216; see Figure 7).

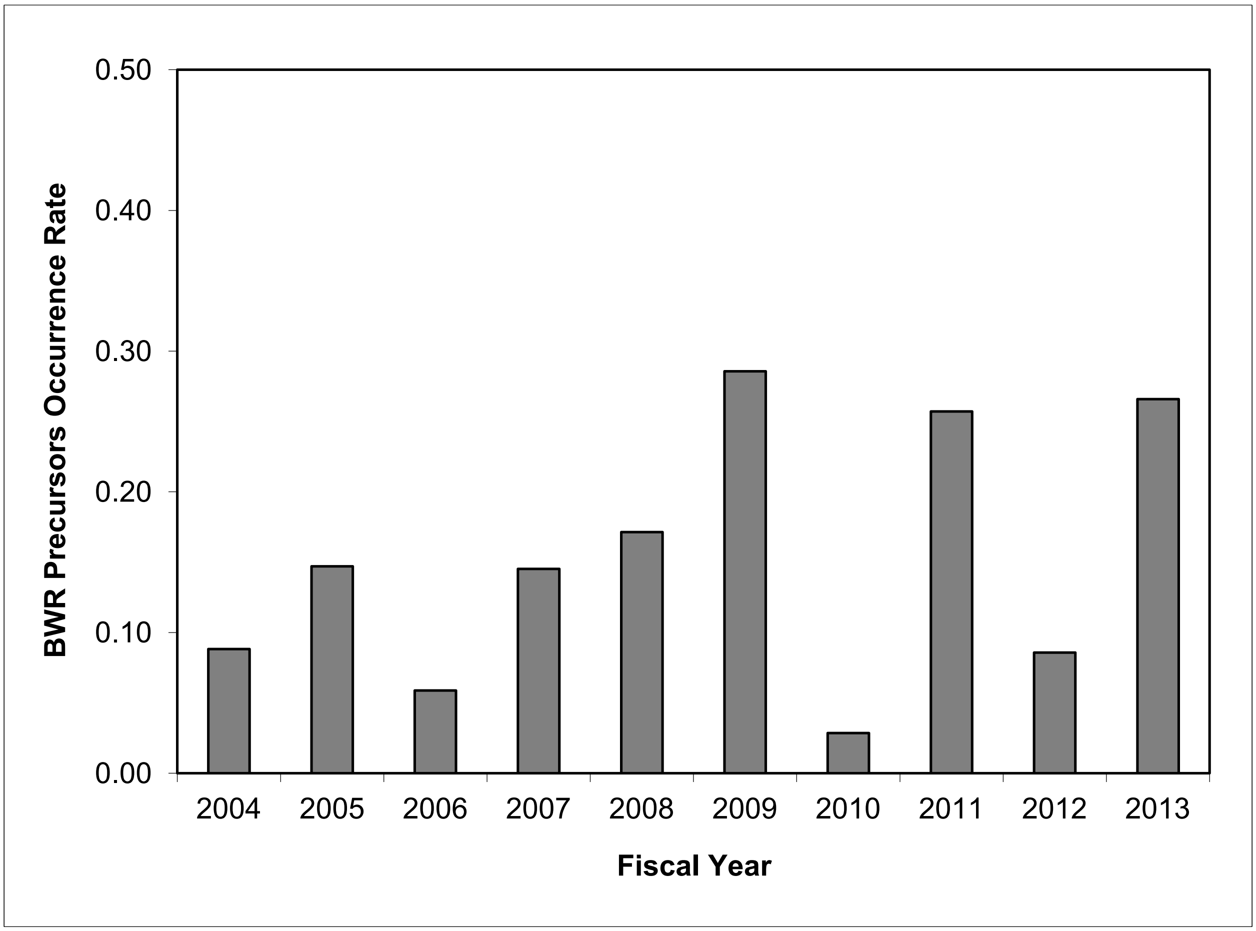

Figure 7. Occurrence rate of precursors involving events at BWRs shows no statistically significant trend for the period from FY 2004 through FY 2013 (p-value = 0.216)

- LOOP events contributed to 52 percent of precursors involving initiating events at BWRs.

- Of the 32 precursors involving the unavailability of safety-related equipment that occurred at BWRs, most were caused by failures in the emergency power system (34 percent), emergency core cooling systems (22 percent), safety-related cooling water systems (3 percent), or electrical distribution system (6 percent).

PWRs

- The mean occurrence rate of precursors that occurred at PWRs does not exhibit a statistically significant trend (p-value = 0.238; see Figure 8).

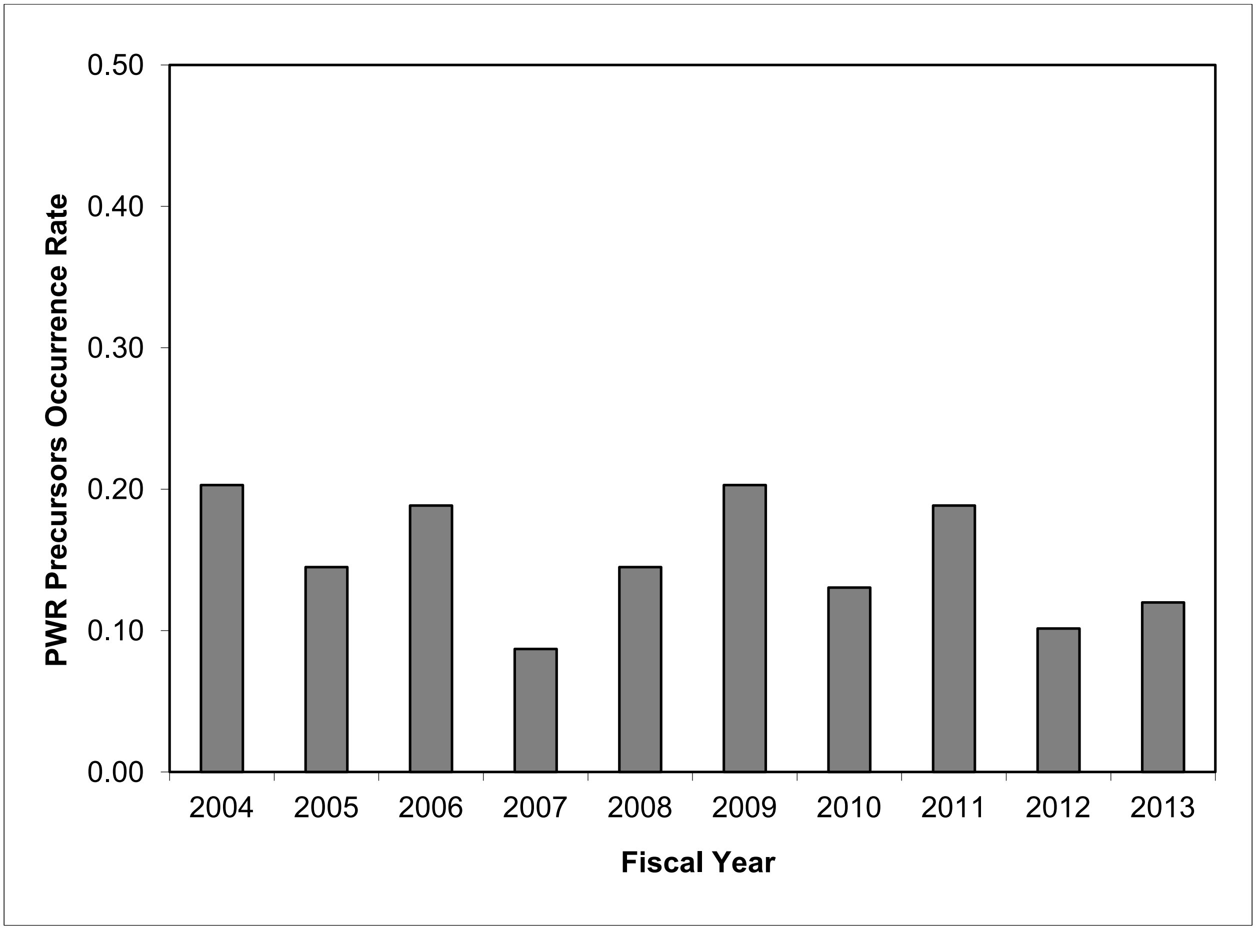

Figure 8. Occurrence rate of precursors involving events at PWRs shows no statistically significant trend for the period from FY 2004 through FY 2013 (p-value = 0.238)

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

- Of the 70 precursors involving the unavailability of safety-related equipment that occurred at PWRs, most were caused by failures in the emergency power system (26 percent), emergency core cooling systems (11 percent), auxiliary feedwater system (13 percent), safety-related cooling water systems (13 percent), or electrical distribution system (11 percent).
  - Of the 8 precursors involving failures in the emergency core-cooling systems, 6 precursors (75 percent) were because of conditions affecting sump recirculation during postulated LOCAs of varying break sizes. Design errors caused most of these precursors (67 percent).
  - Of the 9 precursors involving failures of the auxiliary feedwater system, random hardware failures (78 percent) and design errors (22 percent) were the largest failure contributors. Eight of the 9 precursors (89 percent) involved the unavailability of the turbine-driven auxiliary feedwater pump train.

– Of the 18 precursors involving failures of the emergency power system, 15 precursors (83 percent) were from hardware failures.

– Design errors contributed 31 percent of all precursors involving the unavailability of safety-related equipment that occurred at PWRs.

5.5 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 PRAs and the 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-operating years in the fiscal year; this shows the cumulative plant average of the precursors for a given 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 2011 and has a ΔCDP of 5×10^{-6}. A review of the LER reveals that the degraded condition has existed since a design modification that was performed in FY 2007. In the integrated ASP index, the ΔCDP of 5×10^{-6} is included in FY 2007, 2008, 2009, 2010, and 2011. In addition, the ΔCDP is not prorated for any portion of the year that this condition existed but rather implemented for the entire year, which conservatively estimates the risk contribution during the first and last year. For an initiating event occurring in FY 2011, only FY 2011 includes the CCDP from this precursor.

Results. Figure 9 depicts the integrated ASP indices for the 10-year period from FY 2004 through FY 2013. A review of the ASP indices reveals the following insights:

- Based on the order of magnitude (10^{-5}), the average integrated ASP index for the period from FY 2004 through FY 2013 is consistent with the CDF estimates from the SPAR models and industry PRAs.

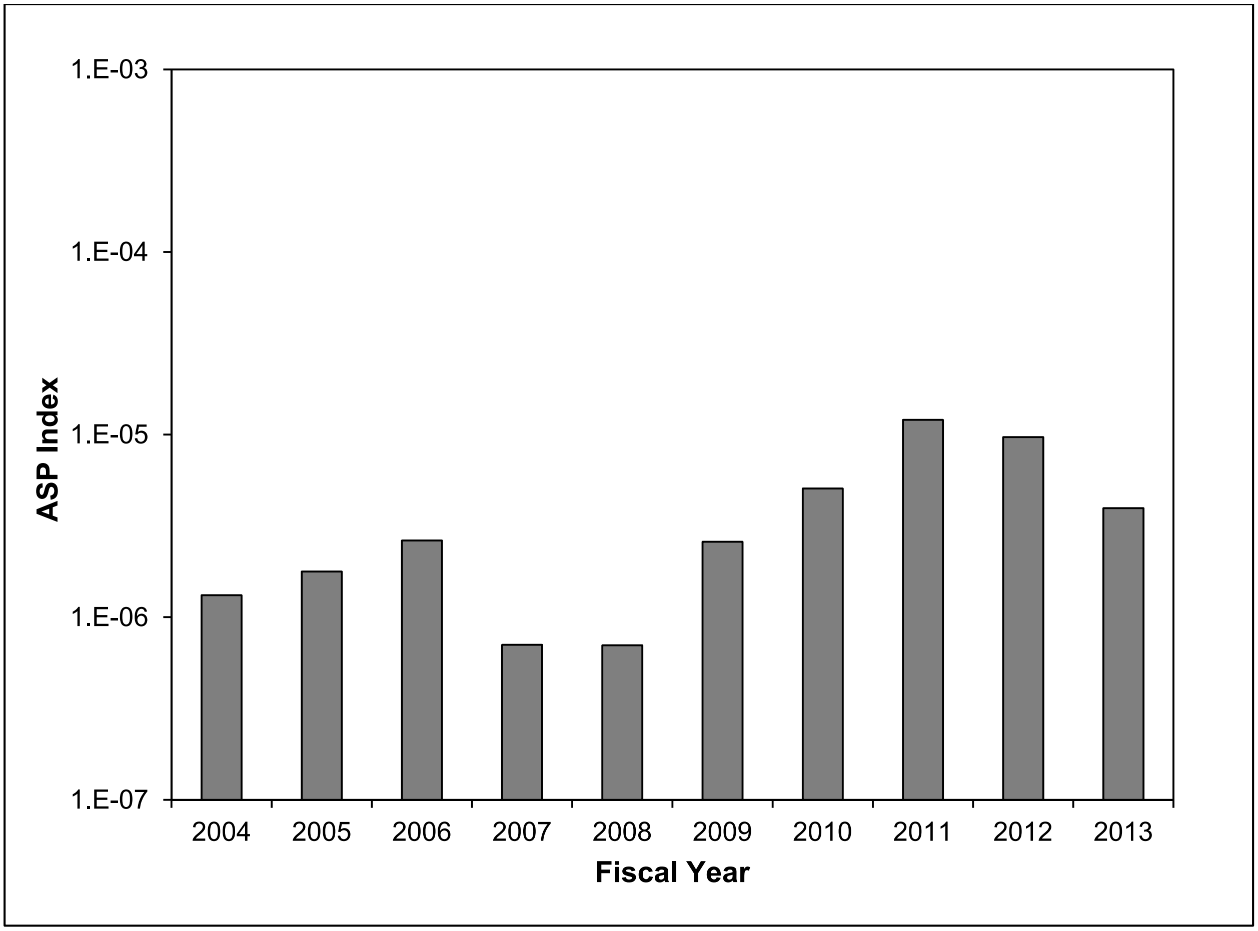

Figure 9. Integrated ASP index for the 10-year period from FY 2004 through FY 2013.

- Precursors over the period from FY 2004 through FY 2013 made the following contributions to the average integrated ASP index:
  - The average integrated ASP index was derived considering the contributions of the 157 precursors during this period.
  - The number of precursors was a little higher than typical in FY 2011 and a little lower than typical in FY 2012. However, the value of this index is relatively high in both FY 2011 and FY 2012 because of the increase in precursors with a CCDP or ΔCDP greater than or equal to 1×10^{-4}, which tends to drive the indicator to a much greater degree than the number of precursors. The staff considers that from a broad industry risk perspective, this increase is not significant.

Limitations. Using CCDPs and ΔCDPs from ASP results to estimate CDF is challenging because (1) the mathematical relationship between CCDPs, ΔCDPs, and CDF requires a significant level of computation, (2) data for the frequency of occurrence of specific precursor events are sparse, and (3) the assessment must also account for events and conditions that did not meet the criteria to be considered an ASP precursor (such as low-risk events including, but not limited to, balance-of-plant failure events).

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 previous years.

5.6 Operating Experience Insights Feedback for PRA Standards and Guidance

A secondary objective of the ASP Program is to provide insights into the current state of practice in risk assessment. ASP event analyses, both precursors and non-precursors, from FY 2013 were reviewed against the approaches to PRA described in the American Society of Mechanical Engineers (ASME)/ American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 4), as endorsed in Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 5). This review sought to identify aspects of the event analyses for which the risk-significant ASME/ANS PRA Standard did not provide guidance. None of the FY 2013 event analyses indicated an inadequacy in the state of PRA practice as described in ASME/ANS RA-Sa-2009. The staff continues to work with ASME/ANS on refinement to the standard to ensure that it provides sufficient guidance to assess the risk significance of external events, including external flooding.

6.0 Summary

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

- ***Significant Precursors.*** The staff identified no *significant* precursors (i.e., CCDP or ΔCDP greater than or equal to 1×10^{-3}) in FY 2013. The staff identified no potentially *significant* precursors in FY 2014. The ASP Program provides the input for determining if the safety measure regarding the "number of significant accident sequence precursors of a nuclear reactor accident" is zero. The final results will be provided in the FY 2014 NRC Performance and Accountability Report (NUREG-1542).

- ***Occurrence Rate of All Precursors.*** The occurrence rate of all precursors does not exhibit a trend that is statistically significant from FY 2004 through FY 2013. The trend of all precursors is one input to the Industry Trends Program to assess industry performance and is part of the input to the adverse trends safety measure. These results will be provided in the FY 2014 NRC Performance and Accountability Report.

- ***Additional Trend Results.*** During the same period, a statistically significant increasing trend was observed in precursors with a CCDP or ΔCDP greater than or equal to 1×10^{-4}. There is an increase of precursors in this subgroup over the past four years after no events were identified in the previous six years.

7.0 References

1. U.S. Nuclear Regulatory Commission, "FY 2015 Congressional Budget Justification," NUREG-1100, Vol. 30, March 2014, Agencywide Documents Access and Management System (ADAMS) Accession No. ML14064A167.

2. U.S. Nuclear Regulatory Commission, "Revised Review and Transmittal Process for Accident Sequence Precursor Analyses," Regulatory Issue Summary 2006-24, December 2006, ADAMS Accession No. ML060900007.

3. U.S. Nuclear Regulatory Commission, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," SECY-13-0107, October 4, 2013, ADAMS Accession No. ML13232A094.

4. American Society of Mechanical Engineers/American Nuclear Society, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009, March 2009.

5. U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 2, March 2009, ADAMS Accession No. ML090410014.

6. U.S. Nuclear Regulatory Commission, "Design Vulnerability in Electric Power System," NRC Bulletin 2012-01, July 27, 2012, ADAMS Accession No. ML12074A115.

7. U.S. Nuclear Regulatory Commission, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," SECY-10-0125, September 29, 2010, ADAMS Accession No. ML102100386.

8. U.S. Nuclear Regulatory Commission, "Enhancing Reactor Safety in the 21st Century – The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," July 12, 2011, ADAMS Accession No. ML112510271.

Status of the Standardized Plant Analysis Risk Models

1.0 Background

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) Standardized Plant Analysis Risk (SPAR) Model Program is to develop standardized risk analysis models and tools for staff analysts to support various 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 improved loss of offsite power (LOOP) and station blackout modules; an improved reactor coolant pump seal failure model; new support system initiating event models; and updated estimates of accident initiator frequencies and equipment reliability based on recent operating experience data.

The SPAR models consist of a standardized, plant-specific set of risk models that use the event-tree and fault-tree linking methodology. Although the SPAR models are plant-specific models, they rely on a set of standardized modeling conventions (e.g., standardized naming conventions, standard modeling approaches, and logic structure) to allow agency risk analysts to proficiently assess the risk significance of findings and operational events. 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. SPAR standardization is needed to allow agency risk analysts to efficiently use SPAR models for a wide variety of nuclear plants without having to relearn modeling conventions and basic assumptions. Although the system fault trees contained in the SPAR models generally are not as detailed as those in licensee probabilistic risk assessments (PRAs), in some cases SPAR models may contain more sophisticated modeling for common-cause failure, support systems, and loss of offsite power. To date, the staff has completed 79 SPAR models representing all 104 commercial nuclear power units. All SPAR models are developed under a comprehensive quality assurance program and have been benchmarked against licensee PRAs through either onsite quality assurance reviews or other information provided by the licensee.

The staff initiated the Risk Assessment Standardization Project (RASP) in 2004. A primary focus of RASP was to standardize risk analyses performed in SDP Phase 3, in ASP, and under Management Directive (MD) 8.3, "NRC Incident Investigation Program." Under this project, the staff initiated the following activities:

- Enhance SPAR models to be more plant-specific and improve the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) code used to manipulate the SPAR models.

- Document consistent methods and guidelines for risk assessments of internal events during power operations; internal fires and floods; external hazards (e.g., seismic events and tornadoes); and internal events during low-power and shutdown (LPSD) operations.

- Provide on-call technical support for staff involved with licensing and inspection issues.

This effort resulted in the development of the Risk Assessment of Operational Events Handbook (commonly referred to as the RASP Handbook) and better alignment between the SDP and ASP operational event assessment processes.

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 and MD 8.3 evaluations, involve the use of SPAR models. The NRC is developing new SPAR modules in response to staff needs for assessing plant risk for external hazards and for assessing accident progression to the plant damage state level.

The staff has completed the following activities in model and method development since the previous status report (SECY-13-0107, "Status of the Accident Sequence Precursor Program and the Standardized Plant Analysis Risk Models," dated October 4, 2013), as described below.

Technical Adequacy of SPAR Models

The staff implemented a Quality Assurance (QA) Plan covering the SPAR models in 2006. The SPAR QA plan was updated in fiscal year (FY) 2013. The main objective of this plan is to ensure that the SPAR models continue to represent the as-built, as-operated nuclear plants and continue to be of sufficient quality for performing event assessments of operational events in support of the staff's risk-informed activities. In addition to model development, the QA Plan provides mechanisms for internal and external peer review, validation and verification, and configuration control of the SPAR models. 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, when applicable. The QA Plan also provides a feedback process from the model users in conjunction with error reporting, tracking, and resolution. 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, which serves as a desktop guidance document for agency risk analysts.

In addition, in 2010 the staff (with the cooperation of industry experts) performed a peer review of SPAR models for a representative boiling-water reactor (BWR) and a representative pressurized-water reactor (PWR) in accordance with American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-S-2008, "Standard for Level 1/Large Early Release Frequency 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 noted a number of strengths for the SPAR models, including:

- The SPAR model structure is robust and well developed.
- The SPAR model fault trees are streamlined with an appropriate level of detail for its intended uses.
- The SPAR model structure and the SAPHIRE computer software are "state of the technology."

- The SPAR model is an efficient method to develop qualitative and quantitative insights for risk-informed applications, SDP evaluations, inspections, event assessments, and model evaluations.

The peer review teams also noted a number of enhancements that could be made to the SPAR models. The staff has reviewed the peer review comments and has initiated projects to address these comments, where appropriate. Activities in progress to address these peer review items include structuring the SPAR model documentation to more closely align with the structure of the PRA standard, incorporation of improved LOOP modeling, and addressing the high priority items for the BWR models. The pace of these activities was significantly reduced during FY 2013 because of sequestration-related budget cuts. With funding restored in FY 2014, the staff continued the resolution of peer review items, including documentation enhancements and model upgrades. The staff plans to complete the PWR peer review enhancements in August 2015, on schedule. The BWR peer review enhancements have been delayed, with an expected completion date of August 2015.

It should be noted that the SPAR models are generally used to categorize and prioritize operational events and conditions, including licensee non-compliance issues with existing regulations, while licensee PRA models developed to support licensing basis changes must meet the technical adequacy requirements of Regulatory Guide 1.200.

Routine SPAR Model Updates

Existing SPAR models need to be updated regularly as a result of any significant plant changes that may affect the risk profile of the plant. As the SPAR model is updated, its documentation (i.e., the model report and the plant risk information eBook summary reports) is also updated to represent the latest PRA information included in the SPAR model. Comparisons between the SPAR model baseline results and licensee model results (when voluntarily submitted by the licensee) are also performed. These comparisons include comparisons of baseline CDF, conditional core damage probability for each initiator type, top cut sets, and importance measures. These comparisons help ensure that SPAR models and associated risk assessments that support the SDP process are of high quality and reflect the as-built, as-operated plants. Although the level of effort was reduced to 6 updates per year because of budget constraints in FY 2013, the effort was increased again in FY 2014 to complete approximately 10 model updates per year.

In addition to these routine SPAR model updates, more limited SPAR models updates are performed to support specific operational event assessment activities when requested by agency risk analysts. These updates are normally required to better model specific features of an operational event that are not normally captured in a base PRA or to reflect an enhanced understanding of the as-built, as-operated plant as a result of event follow-up activities. In FY 2014 the staff updated 47 SPAR additional models to support specific SDP or ASP activities. These updates included 75 specific SPAR model modifications, 19 of which were considered significant upgrades to the SPAR models. As a result of these activities, well over half of the existing SPAR models were updated in FY 2014.

SPAR Models for the Analysis of All Hazards (External Events)

Development of SPAR All HaZard (SPAR-AHZ) models, which contain accident scenarios from all hazard categories applicable to a given site, has continued during FY 2014, although at a lower intensity because of budgetary constraints and balancing limited staff resources to work on other projects, such as the Level 3 PRA project for the Vogtle site. In FY 2014, one new SPAR-AHZ model, which includes internal fire models extracted from the National Fire Protection Association (NFPA) Standard 805-compliant fire model for the Vogtle plant, has been constructed and placed in the SPAR model library for use by NRC risk analysts. The NRC is currently working on the V.C. Summer and Peach Bottom SPAR-AHZ models. Development of these models includes licensee site visits to gather information and discuss modeling assumptions and results. Because the licensee-developed NFPA 805-compliant fire PRA models contain thousands of quantified sequences, a significant focus of the SPAR-AHZ effort was combining similar sequences to enhance model usability while maintaining the ability to retain the resolution contained in the licensee models. Currently, the NRC Office of Nuclear Regulatory Research (RES) and the NRC Office of Nuclear Reactor Regulation (NRR) are working together to identify ways to increase the pace of SPAR-AHZ model development, given expected resource constraints in FY 2015 and beyond.

New Reactor SPAR Models

Before new plant operation, the staff may perform risk assessments to inform potential risk-informed applications for Combined Licenses (COLs), focus construction inspection scope, or assess the significance of construction inspection findings. Once the plants begin operation, independent assessments using SPAR models will be used by the staff for the evaluation of operational findings and events similar to the assessments performed for current operating reactors.

There are currently six new reactor internal hazard SPAR models. These include one model for the AP1000, two Advanced Boiling-Water Reactor (ABWR) models (one for the Toshiba design and one for the General Electric-Hitachi design), one model for the U.S. Advanced Pressurized-Water Reactor (US-APWR), and one for the U.S Evolutionary Power Reactor (U.S. EPR). In addition to these internal events models, there is a seismic model for the AP1000 and a low power and shutdown model for the Toshiba ABWR. Since FY 2013, the staff has been developing a SPAR-AHZ model for the AP1000 reactor design. This AHZ model includes an internal flooding model (completed in FY 2013) and an internal fire model (completed in FY 2014). The staff is currently developing a low power and shutdown model for the AP1000 reactor design.

The staff plans to continue developing new reactor SPAR models, including external hazards and low power and shutdown models, as needed, to support licensing and oversight activities.

MELCOR Thermal Hydraulic Analysis for SPAR Model Success Criteria

The staff continues to perform MELCOR analyses 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. The latest round of activity is documented in two reports: (1) an upcoming NUREG report entitled "Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron," and (2) NUREG/CR-7177, entitled "Compendium of Analyses to Investigate Select Level 1 Probabilistic Risk Assessment End-State Definition and Success Criteria Modeling Issues," published in May 2014. The results of these studies will be used to confirm specific success criteria for a suite of four-loop Westinghouse plants, which are similar to Byron, with appropriate consideration of the design and operational differences of these plants. They also will be used to support application-specific consultation on the use of the SPAR models.

This effort directly supports the agency's goal of using state-of-the-art tools that promote effectiveness and realism. The NRC is communicating the project plans and results to internal and external stakeholders through mechanisms such as the Regulatory Information Conference and the industry's Modular Accident Analysis Program Users' Group.

3.0 Additional Activities

SAPHIRE Maintenance and Improvements

In FY 2014, new features and capabilities have been implemented in SAPHIRE to better support NRC regulatory activities. The new features include:

- A method to automatically adjust the model truncation level and produce a summary report of the convergence results.
- A cut set editor that allows users to efficiently review cut set results, quickly apply changes and sensitivity cases, and recalculate the results.
- The ability to use an external solving engine[1], which allows for comparisons of results using different solving methods.
- Level 2 PRA model quantification features (e.g., the ability to utilize decomposition event trees) and improved integration of Level 1 and Level 2 modeling.

Many of these advanced features were created to support specific NRC projects, and the features were advanced through different developmental versions of the software. In accordance with SAPHIRE configuration management practices, these developmental versions had restricted use and limited availability to users. At this time, all of the above stated features have been merged into a single SAPHIRE version, which is now available to the entire SAPHIRE user community.

All of these improvements to SAPHIRE have been performed in accordance with the SAPHIRE software QA program. A set of software QA documents has been developed for SAPHIRE. These documents cover topics such as the software development plan, configuration management, requirements tracking, and testing and acceptance. The NRC project manager performs an annual audit of the SAPHIRE software quality assurance program. The most recent audit was completed on January 16, 2014, and no significant issues were identified. The NRC Project Manager confirmed that the maintenance and implementation of the SAPHIRE software quality assurance program is consistent with the guidance contained in NUREG/BR-0167, "Software Quality Assurance Program and Guidelines."

The SAPHIRE developers continue to explore advanced features and enhancements that may be implemented in future SAPHIRE revisions. The SAPHIRE team is planning to demonstrate the feasibility of developing a web-based version of SAPHIRE. A web-based SAPHIRE application is envisioned to have several advantages that are not available with a desktop application, such as improved configuration management of models and analyses, enhanced

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[1] SAPHIRE now has the ability to use the FTREX solving engine that is typically used with the Computer Aided Fault Tree Analysis (CAFTA) system. CAFTA was developed by the Electric Power Research Institute (EPRI) and is used by the majority of utilities in the United States.

collaboration capabilities, and remote access to high-performance computing resources. The work to establish the feasibility of a web-based SAPHIRE version began in FY 2014 and is expected to be completed in calendar year (CY) 2015. In addition, the SAPHIRE team continues to research advanced PRA quantification techniques that can improve accuracy and solving speeds. The team has evaluated quantification approaches using Binary Decision Diagram based methods and has remained cognizant of ongoing academic research with Boolean satisfiability or “SAT” methods.

Cooperative Research for PRA

The staff has executed an addendum to the memorandum of understanding (MOU) with the Electric Power Research Institute (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.

During FY 2014, significant efforts have been made in implementing PRA methodologies for support system initiating event (SSIE) analysis and treatment of LOOP in PRAs. These methodologies are being implemented in the SPAR models as one of the activities associated with addressing the peer review comments. To date, 40 models have been enhanced with the improved SSIE modeling methodology and 66 models have been enhanced with the improved LOOP methodology. The staff plans to continue these cooperative efforts with EPRI and other stakeholders to address the remaining issues over the next several years.

Integrated Modeling

The Office of Nuclear Regulatory Research continues to enhance SAPHIRE and the SPAR models to support development of integrated models. To this end, RES recently completed an integrated model for Peach Bottom Unit 2 containing state-of-the-practice SPAR models for Level 1 internal events at power and during shutdown, other hazards, and Level 2 events. This effort included the incorporation of other ongoing modeling initiatives (e.g., modeling of SSIEs), use of modeling features new to SAPHIRE8 (e.g., decomposition event trees), and further validation of the Level 2 PRA model. This work directly benefits the RES Vogtle site Level 3 PRA project (SRM -SECY-11-0089) by guiding the approach to Level 2 and integrated hazard modeling.