United States Nuclear Regulatory Commission

Accident Sequence Precursor (ASP) Program

Summary Description

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TABLE OF CONTENTS

ACRONYMS

ΔCDP	Increase in core damage probability (ΔCDP = CCDP – CDP)
ASP	accident sequence precursor
AFW	auxiliary feedwater
CCDP	conditional core damage probability
CDP	core damage probability
EPIX	Equipment Performance and Information Exchange (database)
FSAR	final safety analysis report
INPO	Institute of Nuclear Power Operations
IPE	individual plant examination
ITP	Industry Trends Program
LER	licensee event reports
LOCA	loss-of-coolant accident
MD 8.3	Management Directive 8.3, “NRC Incident Investigation Program”
PRA	probabilistic risk assessment
NRC	Nuclear Regulatory Commission
RES	Office of Nuclear Regulatory Research
ROP	Reactor Oversight Process
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (PRA software)
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk (models)

1. INTRODUCTION

The Accident Sequence Precursor (ASP) Program involves the systematic review and evaluation of operational events that have occurred at licensed U.S. commercial nuclear power plants. The ASP Program identifies and categorizes precursors to potential severe core damage accident sequences.

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) established the ASP Program in 1979 in response to the Risk Assessment Review Group report (see NUREG/CR-0400, September 1978). Evaluations done for the 1969–1979 period were the first efforts in this type of analysis.

1.2 Program Objectives

The ASP Program has the following objectives:

- Provide a comprehensive, risk-informed view of nuclear power plant operational experience and a measure for trending nuclear power plant core damage risk.
- Provide a partial check on dominant core damage scenarios predicted by probabilistic risk assessments (PRAs).
- Provide feedback to regulatory activities.

The NRC also uses the ASP Program to monitor performance against the safety goal established in the agency's Strategic Plan (Ref. 1). Specifically, the program provides input to the following performance measures:

- Zero events per year identified as a *significant* precursor of a nuclear reactor accident, i.e., conditional core damage probability (CCDP) or change in core damage probability (ΔCDP) greater than or equal to 1×10^{-3}.
- No more than one significant adverse trend in industry safety performance (determination principally made from the Industry Trends Program but supported by ASP results).

1.3 Relationships to Other NRC Programs

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."

Significance Determination Process. The main purpose of the SDP is to determine the safety significance of inspection findings. The SDP is part of the Reactor Oversight Process (ROP) and evaluates inspection findings in all seven cornerstones of safe operation—initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, worker radiation safety, physical protection. The SDP uses a three-phased approach to determine the significance of inspection findings in the initiating events, mitigating systems, and barrier integrity cornerstones. Reference 2 provides additional information on the ROP and SDP.

NRC Incident Investigation Program—event response evaluation. The main purpose of the event response evaluation element of the NRC Incident Investigation Program is to determine the appropriate level of reactive inspection in response to a significant event. The event response evaluation process is part of the Reactor Oversight Process and provides a prompt evaluation of significant operational events (as defined in Management Directive (MD) 8.3, "NRC Incident Investigation Program") involving reactor and fuel cycle facilities and NRC or Agreement State licensed materials. Reference 2 provides additional information on the ROP and MD 8.3.

Similarities between ASP, SDP, and event response processes. The risk models and technical methods used in ASP, SDP Phase 3, and event response assessments are generally similar. The Standardized Plant Analysis Risk (SPAR) models are typically used in all three processes. Most of the methods applied in SDP Phase 3 and event response assessments are derived from the ASP Program.

The SDP Phase 1 is a screening procedure that identifies the inspection findings to be evaluated under SDP Phase 2 or 3. The ASP and event response processes also employ screening procedures. Risk significance estimation under the SDP Phase 2 process is quite different from ASP, SDP Phase 3, and event response processes.

Differences between processes. Since ASP, SDP, and event response programs serve different functions, there are some inherent differences in the processes. For example, the SDP, which is used to determine the safety significance of inspection findings, analyzes each finding individually and screens out events where there are no licensee performance deficiencies. In contrast, the ASP program evaluates all potentially significant plant events and degraded conditions, and analyzes concurrent multiple degraded conditions together.

1.4 Precursor Definitions and Threshold

Definition of an operational event. An *operational event* can be

- An actual initiating event (e.g., loss of offsite power, loss-of-coolant accident), or
- A condition found during a test, inspection, or engineering evaluation involving a reduction in safety system reliability or function for a specific duration.

The ASP Program uses the term *operational event* interchangeably with the terms "initiating event" or "condition."

Definition of a precursor. An *accident sequence precursor* is an initiating event or degraded condition that, when coupled with one or more postulated events, could result in a plant condition involving inadequate core cooling and severe reactor core damage. The ASP Program uses nominal initiating event frequencies and/or nominal failure probabilities for estimating the conditional probability of the postulated event portion of the analysis.

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

At-power precursor. An at-power precursor is an operational event that usually meets <u>one</u> of the following criteria:

- The total failure of a system required to mitigate effects of a core damage initiator,
- The degradation of two or more safety system trains required to mitigate effects of a core damage initiator,

- The degradation of one safety system train for an extended period of time,
- A core damage initiator such as a loss of offsite power or small-break loss-of-coolant accident, or
- A reactor trip or loss-of-feedwater with a degraded safety system.

Shutdown precursor. A shutdown precursor is an operational event that meets both of the following criteria:

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

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

Thresholds. Two thresholds are noted in the ASP Program in the order of increasing severity.

- An initiating event with a CCDP or a condition with a ΔCDP (importance) greater than or equal to 1×10^{-6} is called a *precursor*.
- An initiating event with a CCDP or a condition with a ΔCDP (importance) greater than or equal to 1×10^{-3} is called a *significant precursor*.

2. EVALUATION PROCESS

2.1 Introduction

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

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

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

These four initiators are primarily associated with loss of reactor core cooling. ASP Program analysts examine licensee event reports (LERs) and other event documentation to determine the impact that operational events have on potential core damage sequences associated with these initiators. Operational events are occasionally identified that impact other initiators, such as an in-plant fire, in-plant flooding, seismic, and tornado. Unique models are developed to address these events.

Details of various elements of the evaluation process are discussed below.

2.2 Selection of Potential Precursors for Analysis

2.2.1 Event Information Sources

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

- *LER database.* LERs are mandatory reports of operational events from licensees of nuclear power plants submitted according to the requirements of 10 CFR 50.73. An LER contains information concerning equipment performance, loss of function (system/train failures), personnel errors, and the effect on the plant. This information is used to identify potential precursors in the screening process discussed in the next section.
- NRC inspection findings from routine inspections, special inspection team inspections, and augmented inspection team inspections.

In addition to these sources, special requests from NRC staff are used to identify potential precursors.

2.2.2 Initial Review Process

The ASP Program employs the following two-step review process in the selection of operational events for potential precursors:

Screening. LERs are screened against a set of screening criteria to identify those which should be reviewed as candidate precursors. Findings from inspections and special requests from NRC staff are also included as candidates. Screening usually involves the removal of LERs that are not related to a reactor trip or degraded structure, system, or component.

Engineering reviews. Those events selected through the screening process will undergo one- or two-engineer review(s) to determine if the reported event should be examined in greater detail. Operational events that, in the judgment of the initial reviewing engineer, clearly meet the ASP rejection criteria for analysis as a potential precursor are not subjected to another review. All other operational events are reviewed by two engineers to determine if they met the ASP criteria for detailed analysis before the decision is made to reject or to analyze the event.

An engineering review is a bounding review, meant to capture events that in any way appeared to deserve detailed review and to eliminate events that are clearly unimportant. This process involves eliminating events that satisfy predefined criteria for rejection and accepting all others as potentially significant and requiring analysis.

Rejection criteria. LERs are eliminated from further consideration as precursors if they involved only one of the following:

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

Acceptance criteria. All operational events that can not be eliminated using the rejection criteria undergo detailed analysis. Operational events identified for further consideration typically include the following:

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

2.3 Detailed Analysis of Potential Precursors

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

2.3.1 Process Overview

The overall event analysis process involves the modification of a PRA model to reflect attributes of an operational incident, solution of the modified model to estimate the risk significance of the incident and documentation of the analysis and its results. The process is structured to ensure the analysis is comprehensive and traceable. A detailed review by the analyst and a subsequent independent review(s) minimize the likelihood of errors, and enhance the quality of the risk analysis.

As a minimum, a risk analysis consists of the following:

- Development of a risk-focused understanding of the event that occurred, relevant plant design and operational features as well as the status of the plant.
- Comparison of the event with the existing risk model to identify any changes that are necessary to support the analysis.
- Risk model elaboration, if necessary, to allow the risk-related features of the observed event to be properly represented in the model.
- Model modification to reflect event specifics.
- Initial model solution to estimate the risk significance of the event without consideration of crew activities to recover risk-significant failures.
- Recovery analysis to address potential crew actions to recover any failed components associated with risk-significant sequences.
- Analyst review of the results to ensure that the logic model and incident mapping process is correct. The focus of this review is to identify inconsistencies, errors, and incompleteness in the risk model. Then the risk model is modified and re-solved.
- Final documentation of the inputs (facts), assumptions, results, and uncertainties.
- Independent review(s) of the completed analysis.

In addition, a supplemental effort that can improve analysis accuracy and confidence in the results should be performed for higher risk-significance or controversial events:

- Sensitivity and uncertainty analysis to gain additional understanding of the impact of analysis assumptions and data variability on analysis results.

The event analysis process is iterative. Review of the model for applicability may highlight the need for additional detail related to the event. Review of the initial analysis results (significant sequences and cut sets) frequently identifies the need for additional detail concerning the event, plant design, operational information, or the need for greater model fidelity.

2.3.2 Analysis Overview

ASP analyses, as well as Significance Determination Process (SDP) Phase 3 and MD 8.3 analyses, are retrospective analyses of operational experience. In these analyses, a ‘failure memory’ approach is used to estimate the risk significance of degraded conditions and initiating events. In a failure memory approach, risk model elements (basic events) associated with observed failures and other off-normal situations are configured to be failed, while those associated with observed successes and unchallenged components are assumed capable of failing, typically with nominal probability.

A *failure* is defined in terms of the inability of a component (or operator action) to function in the context of a particular risk sequence and mission time.[1] An event analysis is performed on the failures and off-normal situations *observed* during the initiating event or degraded condition(s) discovered during surveillance test, engineering evaluation, or inspection. A degraded condition may represent a failed or unavailable structure, system, or component (SSC) that was unable to perform its mission upon demand or a degraded SSC with a higher probability of failure to complete its mission.

All other components in the risk model that were not impacted or challenged by the operational event are modeled with nominal (i.e., random) failure probabilities. An event involving a reactor trip is analyzed as an initiating event, although the non-initiator parts of an initiating event can be addressed in a supplemental degraded condition analysis. Postulated failures, such as the postulated failure of pump B instead of the observed failure of pump A because pump B's failure is of higher risk significance, are not assumed in the event analysis (except as a sensitivity analysis).

2.3.3 Analysis Types

The detailed analysis of a potential precursor considers the immediate impact of an initiating event and/or the potential impact of the equipment failure(s) or operator error(s) on the readiness of systems in the plant for mitigation of off-normal and accident conditions. The ASP Program considers three types of analysis: condition analysis, initiating event analysis, and shutdown analysis.

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

In this analysis, nominal initiating event frequencies are used in the analysis. The failure probability of the component that was determined to be not functional during the required mission time will be adjusted to reflect the degree in which the component will fail during the required mission time (usually 1.0). In some cases, extensive engineering analysis or expert judgment is required to determine the degree of degradation of the component. Nominal failure probabilities of all other components are used in the analysis. The ASP analysis uses a maximum period of unavailability of one year.

A condition analysis can include the unavailability of multiple components that were discovered at different times. In this special case, the time period in which the components were unavailable must overlap to some degree. The conditional probability of the increase in risk caused by the unavailable components is integrated over the worst case one-year time period.

[1] A component can be considered failed for some sequences and not failed for others in which the requirements for successful mitigation are more relaxed. Component functionality is often unrelated to inoperability as defined in a plant's Technical Specifications. A component that has been declared inoperable based on Technical Specifications may be functional (and therefore not failed) from a risk standpoint.

Initiating event analysis. If the event or failure resulted in an automatic or manual reactor trip and occurred while the plant was at power, then the event is evaluated according to the likelihood that it and the ensuing plant response could lead to core damage.
In this analysis, the frequency of the initiating event will be set to 1.0 (because it happened). If any component of a mitigating system failed during the event, including operator errors, then the failure probability of the failed equipment will be set to 1.0.

If a future surveillance test determined that a component was unable to function during the required mission time, then the failure probability of the component will be adjusted to reflect the degree in which the component will fail during the required mission time—similar to a *condition analysis*. Nominal failure probabilities of all other components are used in the analysis.

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

- If the event could have impacted at-power operation, its impact is assessed.
- If the event could only occur at cold shutdown or refueling shutdown, then its impact on continued decay heat removal during shutdown is assessed.

2.3.4 Quantification Process

Quantification overview. The significance of an operational event involves the determination of a conditional probability of subsequent core damage given the observed failures. This conditional probability is estimated by mapping observed failures onto the risk model, and calculating a conditional core damage probability. These plant-specific models, called SPAR models, contain event trees and linked fault trees. The event trees and fault trees in a SPAR model depict potential paths to core damage.

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

Figure of merit. For an *initiating event* involving a reactor trip, risk significance is estimated based on the probability of proceeding to core damage given the observed failures. The overall result is expressed in a conditional core damage probability. For a *degraded condition*, risk significance is estimated based on the increase in core damage probability over the duration that the condition existed. For conditions, the overall result is the difference between the conditional core damage probability (given the observed failure) and the risk model's nominal (or baseline) core damage probability. This difference is known as the 'importance' or ΔCDP.

An operational event is selected and documented as a precursor to a potential core damage accident (accident sequence precursor) if the conditional probability of subsequent core damage is at least 1×10^{-6}. A detailed description of the ASP analysis process can be found in Appendix A of the RASP handbook (Ref. 3).

2.3.5 ASP Evaluation Process

For the most part, the ASP program has followed the same process since the program's inception in the early 1980's. Since the implementation of the Reactor Oversight Process (ROP) in April 2000, specifically the Significance Determination Process (SDP) and event response evaluation under Management Directive 8.3 (MD 8.3), similarities between processes provided an opportunity to achieve better efficiencies in the ASP Program. The new ASP process has been revised to utilize the results of the SDP or MD 8.3 programs, when appropriate, to eliminate duplication of analysis of the same operational event.

Selection of precursors with an SDP or documented MD 8.3 evaluation. For degraded conditions or significant operational occurrences for which there is an SDP or documented MD 8.3 quantitative risk evaluation, the ASP program will utilize the results of these evaluations, where applicable, without performing a separate ASP analysis.

Selection of precursors with no SDP or documented MD 8.3 evaluation. The ASP program will continue to perform analyses for those events for which there is no SDP or MD 8.3 evaluation. Examples of these types of events include most initiating events, and plant conditions for which there are no performance deficiencies or where there are concurrent multiple degraded conditions.

Potentially significant precursors. For all events (including those being evaluated by the SDP or MD 8.3 processes) that, based on preliminary evaluations, could be *significant* precursors, the ASP program will perform an expedited analysis, in order to support the reporting requirements in the annual NRC Performance and Accountability Report to Congress, NUREG-1542 (Ref. 1), and to support the new Abnormal Occurrence criteria described in Appendix D of NUREG-0090, Volume 29 (Ref. 4).

2.4 Sources of Input Information for Detailed Analysis

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

The adaptation of the SPAR model to the operational event may need design-related information from plant-specific sources, such as the emergency operating procedures, Updated Final Safety Analysis Report, and Individual Plant Examinations for internal events (e.g., loss of offsite power) and external events (e.g., seismic). In addition, the component failure probabilities and initiating event frequencies may be updated in the SPAR model using results from system reliability and initiating event studies, based on recent operational experience.

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

2.4.1 Event-Related Information

Information describing the operational event in the LER can be supplemented with additional information from the following sources:

- NRC inspection reports, which can be found on the NRC web page via the external (public) server.

- Inspectors, including senior reactor analysts, knowledgeable of the specific operational event.
- NRC staff experts in the relevant technical areas.
- The licensee through follow-up event assessment.
- Assessment reports issued by the licensee to support public regulatory conferences.

2.4.2 Plant Design and Operation Information

Once an event is understood, important sequences identified, and the SPAR model selected, the appropriateness of the existing risk model in describing the potential risk impact is reviewed by the analyst. This review ensures that the SPAR model reflects the as-built, as-operated plant for the sequences impacted by the operational event. Areas where additional modeling detail is required to adequately reflect the observed event are identified. The adaptation of the plant-specific SPAR model to the event may require design-related information from plant-specific sources such as:

- Updated Final Safety Analysis Report.
- Technical Specifications.
- Individual Plant Examinations (probabilistic risk assessments) for internal events (e.g., loss of offsite power) and external events (e.g., seismic).
- Plant system descriptions and drawings.
- Plant emergency and normal operating procedures.
- NRC resident inspector.
- NRC regional senior reactor analyst, who is knowledgeable in risk assessments.

2.4.3 Updates of Model Parameters

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

- System reliability studies for high-pressure injection systems in boiling-water and pressurized-water reactors, auxiliary feedwater systems (if applicable), emergency diesel generators, safety-related service water systems, and reactor protection systems.
- Component reliability studies for motor- and air-operated valves, and motor- and turbine-driven pumps.
- Common-cause failure parameter estimations and database for 42 components in boiling-water and pressurized-water reactors.
- Initiating event studies for reactor trips, loss-of-coolant accidents, fires, and loss of offsite power events.

These risk studies are updated annually and available from the NRC public web page, “Reactor Operational Experience Results and Databases” (Ref. 5). Data used to update model parameters are based on LERs and records from the Institute of Nuclear Power Operations’

(INPO), Equipment Performance and Information Exchange (EPIX) database (proprietary database available to NRC staff only).

3. INDEPENDENT REVIEW PROCESS

3.1 New ASP Review Process

The new ASP review process has been revised to be more-risk informed and to reduce NRC and licensee staff burdens (Ref. 6). The level of the review an analysis receives is based on CCDP or ΔCDP of the operating event, as described below.

Lower risk events. If an ASP analysis results in a CCDP or ΔCDP of less than 1×10^{-4}, formal review comments by the licensee will no longer be requested. A summary of the analysis and results will be issued to the pertinent licensee and other stakeholders for information. However, if the licensee or other stakeholders choose to comment on these analyses, the NRC will continue to address their comments.

Higher risk events. If an ASP analysis results in a CCDP or ΔCDP greater than 1×10^{-4}, formal review comments by the licensee will continue to be requested. These ASP analyses will be issued as final to all stakeholders after resolution of the review comments.

3.2 Review Process for Lower Risk Events

Analyses for lower risk events undergo an internal RES technical review using a checklist. A technical audit is performed by a senior analyst prior to issuance. The analyses may also be sent to applicable NRR and Region staff for an informal review. After issuing the analysis as final, the analysis will be transmitted to the licensee and any comments received from the licensee will be addressed. If necessary, appropriate modifications to the analysis will be made and the analysis re-issued.

3.3 Review Process for Higher-Risk Events

For higher risk events described above, a potential precursor undergoes a three-step independent review process: Preliminary analysis reviews, peer reviews, and final analysis reviews.

3.3.1 Preliminary Reviews

In the first review, the draft preliminary analysis is reviewed in-house by a second analyst. The typical two-day review is performed using a checklist. Technical audit by a senior analyst is performed for preliminary analyses prior to issuance. After completion of the first review and any corresponding revision, the preliminary analysis is transmitted to the pertinent nuclear plant licensee and to the NRC staff for peer review.

3.3.2 Peer Reviews

The licensee is requested to review and comment on the technical adequacy of select analyses, including the depiction of their plant equipment and equipment capabilities. The reviews by licensees are optional.

Review guidance (see Appendix) is sent to the licensee along with the preliminary ASP analysis. The information in the guidance include

- Provides specific guidance for performing the requested review,

- Identifies the criteria used in the analysis to determine whether any credit should be given for the use of licensee-identified additional equipment or specific actions in recovering from the event, and
- Describes the specific information that the licensee should provide to support such a claim.

The preliminary analysis is also sent to various technical branches within the NRC as well as the cognizant regional office for review and comment. Review comments are evaluated for applicability and pertinence to the ASP analysis.

3.3.3 Final Analysis Reviews

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

4. FACTORS AFFECTING RESULTS (UNCERTAINTIES)

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

4.1 Technical Adequacy of SPAR Models

The Office of Nuclear Regulatory Research (RES) has implemented an updated SPAR model quality assurance plan covering the Revision 3 SPAR models. Processes in place to verify, validate, and benchmark these models according to the guidelines and standard established by the SPAR Model Development Program. As part of this process, reviews of the Revision 3 SPAR models and results are preformed against the licensee probabilistic risk assessment (PRA) models. In addition, processes are 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 a handbook for the risk analysis of operational events (as known as the RASP handbook, Ref. 3). This handbook provides analysts with guidance for maintaining SPAR models that are sufficiently representative of the as-built, as-operated plant to support model uses.

4.2 Cooperative Research for PRA

RES has executed an addendum to the memorandum of understanding with the Electric Power Research Institute 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 results.

The objective of this effort is to work with the broader PRA community to resolve PRA issues and develop PRA methods, tools, data, and technical information useful to both the NRC and industry. The agency has established working groups that include support from Office of Nuclear Reactor Regulation, the Office of New Reactors, and the regional offices. Initial cooperative efforts include the following:

- Support system initiating event analysis
- Treatment of loss of offsite in PRAs
- Initiating event guideline development
- Treatment of uncertainty in risk analyses
- Aggregation of risk metrics
- Standard approach for injection following containment failure (boiling-water reactors)
- 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

4.3 Recovery of Failed Equipment

Crediting recovery of failed or unavailable equipment can have a significant impact on the analysis results. The actual likelihood of the failure to recover from an event at a particular plant is difficult to assess and may vary substantially from the nominal values currently used in the ASP analyses. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, and others concerning the likelihood of recovering from specific failures (typically observed during testing) within a time period that would prevent core damage following an postulated initiating event.

To improve consistency in human reliability estimates of recovery actions in ASP analyses, the ASP Program has adopted the SPAR-H method for performing human reliability analysis (Ref. 7). The simplified human reliability analysis approach makes use of a three-page worksheet to rate a series of performance shaping factors and dependency factors to arrive at a screening level human error probability for a given task.

4.4 Uncertainty and Sensitivity Analyses

Uncertainty analyses. The results of any ASP analysis are conditional on modeling assumptions and on the data used to support the risk estimation. An ASP analysis considers two kinds of uncertainties.

- *Parameter uncertainty.* The uncertainties regarding the numerical values of the parameters used in the model (parameter uncertainty) are estimated using generic industry data adjusted for plant-specific operational experience and design features. These uncertainty distributions are then propagated through the SPAR model to produce a mean value of the conditional core damage probability (CCDP) or importance (ΔCDP) as well as the 5^{th} and 95^{th} percentile values.

  The SAPHIRE code has recently been updated to perform ΔCDP uncertainty calculations (one of the few PRA codes able to do this). The uncertainty associated with ΔCDP calculations behaves significantly different than the uncertainty associated with CCDP.

- *Model uncertainty.* The issue of alterative model assumptions (often referred to as model uncertainty) is handled by performing sensitivity studies. Sensitivity studies show the affects of key assumptions on the mean CCDP or ΔCDP. These include initiating event frequencies, recovery actions, and operator actions.

Sensitivity analyses. Sensitivity analyses are also frequently performed in an ASP analysis to indicate analysis inputs or elements whose changes in value cause the greatest changes in partial or final risk outputs. They are performed to identify which components are sensitive to results. A PRA model, such as a SPAR model, is conditional on the validity of its assumptions and on the availability of data for its parameters. An epistemic model[2] represents the state of knowledge regarding the numerical values of the parameters and the validity of the model assumptions. The validity of numerical values is handled through the uncertainty analysis. The validity of the model, and its associated uncertainties, can be handled through sensitivity analysis.

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[2] The epistemic uncertainty is related to the variability in model parameters that associated with the analyst's confidence in the predictions of the PRA model (i.e., thermal-hydraulic calculation, the temperature and pressure ranges for a particular accident sequence) and also reflects how accurate the PRA model represents the actual system being modeled. The epistemic uncertainty is referred to as state-of-knowledge uncertainty.

Sensitivity analyses involve performing the risk calculations with changes to some assumption or facts that went into the 'best estimate' case. They can be as simple as changing the value of a model parameter to as complicated as making major model changes to reflect different success criteria. If possible, parametric changes should have an engineering basis (i.e., what if a certain human error probability was calculated in a high stress assumption?) rather than an arbitrary parametric change (i.e., what if the initiating event was 10 times more frequent?).

Documentation of results. Consistent with previous ASP Program practices, the published results represent the best estimate assumptions and model, and the significance of the precursor is determined using the mean value of the uncertainty distribution.

5. REFERENCES

1. U.S. Nuclear Regulatory Commission, "Performance and Accountability Report – Fiscal Year 2007," NUREG-1542, Volume 13, November 2007. http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1542/

2. U.S. Nuclear Regulatory Commission, "Reactor Oversight Process (ROP) Program Documents," http://www.nrc.gov/reactors/operating/oversight/program-documents.html, September 2007.

3. U.S. Nuclear Regulatory Commission, "Risk Assessment of Operational Events Handbook – Internal Events," Volume 1, ADAMS Accession Number ML080070303, January 2008.

4. U.S. Nuclear Regulatory Commission, "Report to Congress on Abnormal Occurrences – Fiscal Year 2006," NUREG-0090, Volume 29, April 2007. http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0090/

5. U.S. Nuclear Regulatory Commission, "Reactor Operational Experience Results and Databases," http://nrcoe.inel.gov/results/, August 2007.

6. U.S. Nuclear Regulatory Commission, "Revised Review and Transmittal Process for Accident Sequence Precursor Analyses," Regulatory Issue Summary 2006-24, December 6, 2006. http://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/2006/ri200624.pdf

7. U.S. Nuclear Regulatory Commission, "The SPAR-H Human Reliability Analysis Method," NUREG/CR-6883, August 2005. http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6883/

Appendix - Guidance for Licensee Review of Preliminary ASP Analysis

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

Background

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

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

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

Modeling Techniques

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

Revision 3 SPAR models generally have 11 types of initiating events:

- Transients
- Small loss-of-coolant accident (LOCA)
- Medium LOCA
- Large LOCA
- Interfacing system LOCA
- Steam generator tube rupture (PWR only)
- Loss of offsite power
- Loss of component cooling water (PWRs only)
- Loss of service water
- Loss of direct current power

They also have transfer event tree for station blackout and anticipated transient without scram.

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

Guidance for Peer Review

Comments regarding the analysis should address:

- Does the "Event Summary" section:
  - Accurately describe the event as it occurred; and
  - Provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?

- Does the "Modeling Assumptions" section:
  - Accurately describe the modeling done for the event;
  - Accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions; and
  - Include assumptions regarding the likelihood of equipment recovery?

Criteria for Evaluating Comments

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

Criteria for Evaluating Additional Recovery Measures

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

- Normal or emergency operating procedures,
- Piping and instrumentation diagrams,
- Electrical one-line diagrams,
- Results of thermal-hydraulic analyses, and
- Operator training (both procedures and simulation).

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

- Sequence of events
- Timing of events
- Probability of operator error in using the system or equipment.
- Other systems/processes already modeled in the analysis (including operator actions)

An Example of a Recovery Measure Evaluation

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

- The mitigation effect for the standby feedwater system would be credited in the analysis provided that the following material was available:
- Standby feedwater system characteristics are documented in the FSAR or accounted for in the updated IPE.
- Procedures for using the system during recovery existed at the time of the event.
- The plant operators had been trained in the use of the system prior to the event.
- A clear diagram of the system is available.
- Previous analyses have indicated that there would be sufficient time available to implement the procedure successfully under the circumstances of the event under analysis.
- The effects of using the standby feedwater system has been considered not have an adverse impact on recovery operations. In this case, use of the standby feedwater system may reduce the likelihood of recovering failed AFW equipment or initiating feed-and-bleed due to time and personnel constraints.

Schedule

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