APPENDIX A

TECHNICAL BASIS FOR THE AT-POWER SIGNIFICANCE DETERMINATION PROCESS (SDP)

1.0 OBJECTIVE

The objective of this appendix to Inspection Manual Chapter (IMC) 0308, Attachment 3, “Technical Basis for the Significance Determination Process” is to provide a technical basis for the risk categorization process used to estimate the risk significance of inspection findings at-power (within the safety cornerstones of initiating events, mitigating systems, and barrier integrity) as described in IMC 0609, Appendix A, “At Power Significance Determination Process.”

2.0 BACKGROUND

Since the initial implementation of the Reactor Oversight Process (ROP), the at-power SDP has involved a three phased approach. The initial phase (Phase 1) was designed to screen findings of low risk significance to green to allow the staff to focus more resources on risk significant findings. The second phase (Phase 2) was designed to estimate the risk significance of the finding, provide an engineering understanding of the finding, and serve as an additional screening tool to identify low risk significant findings that did not screen out in the initial phase. The at-power Phase 2 process consisted mainly of site-specific pre-solved tables and risk-informed notebooks which, from a high level, were a set of tables and guidance designed using risk insights from the licensee’s risk model. The third phase (Phase 3) was designed to add specificity to the Phase 2 risk evaluation if needed (i.e., provide more detailed analyses, reduce uncertainties, etc).

The majority of this IMC was dedicated to providing the technical bases for the pre-solved tables and risk-informed notebooks. However, over the years, maintaining the site-specific pre-solved tables and risk-informed notebooks located in IMC 0609, App A proved to be a challenging task. As plants implemented equipment modifications and associated revisions to the plant risk model, the accuracy of the pre-solved tables and risk-informed notebooks began to degrade. Instead of revising all of the plant specific pre-solved tables and risk-informed notebooks, the agency decided to transition from the pre-solved tables and risk-informed notebooks to Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) and the site-specific Standardized Plant Assessment Risk (SPAR) models. SAPHIRE and the site-specific SPAR models provide an efficient and effective infrastructure that facilitates risk model fidelity and updates.

In the transition from the pre-solved tables and risk-informed notebooks to SAPHIRE and the site-specific SPAR models it is important to note some key technical differences. The risk-informed notebooks are train-level models that had been benchmarked against the licensee PRA models. They were designed as a tool to provide an “order of magnitude” estimate of the delta CDF. The SPAR models are component level models that have also been benchmarked with licensee PRA models. The SPAR models use a combination of generic and plant specific data from NUREG/CR-6928. For legacy, reference, and knowledge transfer purposes, the pre-solved tables, risk-informed notebooks, and associated ROP guidance documents have been archived.

3.0 TECHNICAL BASIS FOR THE AT-POWER SDP

The technical basis for the at-power SDP is divided into two sections. The first section (3.1) provides a technical justification for the screening questions. The screening questions are categorized by safety cornerstone and provide a logical series of questions to determine if a finding can be characterized as having low safety significance. The second section (3.2) provides technical justification for the detailed risk evaluation. In contrast to the site-specific pre-solved tables and risk-informed notebooks which had a robust and detailed technical justification in this IMC, the detailed technical justification for SAPHIRE and the site-specific SPAR models can be found in a variety of staff documentation (e.g., NUREGs). As such, only an overview of the technical justification is provided in this IMC.

3.1 TECHNICAL BASIS FOR THE AT-POWER SCREENING QUESTIONS

The initial screening is intended to screen out those findings that have minimal impact on risk early in the process as an efficiency measure. The at-power screening questions apply to the reactor safety cornerstones of initiating events, mitigating systems and barrier integrity. To support the issuance of SECY 99-007A, the staff performed a simple sensitivity test of the at-power inspection finding screening process. The test was designed to ensure that findings with proven risk importance would not be screened out by the process. The staff reviewed the 1996 accident sequence precursors (ASP) to potential severe core damage events. In 1996, the NRC identified in NUREG/CR-4674, Vol. 25, fourteen precursors with a conditional core damage probability (CCDP) greater than 1E-6 affecting thirteen units. In all there were seven at power precursor events involving initiating events and six at power precursor events involving the unavailability of mitigating systems. All of the risk significant ASP events and degraded conditions successfully made it past the screening questions and would have required further evaluation using a more detailed risk evaluation. This sensitivity test that was used during the initial development stages of the Reactor Oversight Process (ROP) provides a level of confidence that potentially risk significant inspection findings will not be inadvertently screened out early in the process and will receive a more detailed level of evaluation.

Over the years since the initial implementation of the ROP, the at-power screening questions have been clarified and refined based on lessons learned and experience. In addition, some new screening questions have been added to improve the overall effectiveness of the screening process. However, the screening questions as a whole have not changed enough to warrant another sensitivity study similar to the effort to support SECY 99-007A. The staff recognizes that the at-power screening questions are not an all inclusive set. Therefore, as a conservative measure, if a finding screens to green in accordance with the applicable screening questions and the staff has reason to believe that there is still potential that the finding is risk significant, the staff reserves the opportunity to perform a more detailed risk evaluation.

3.2 TECHNICAL BASIS FOR THE DETAILED RISK EVALUATION

IMC 0609, Appendix A briefly describes how SPAR models (e.g., SDP Workspace, Event Condition Assessment, General Analysis) can be used to develop a plant specific estimate of the risk significance of an inspection finding. The SPAR models consist of a set of plant-specific Probabilistic Risk Assessment (PRA) models that employ a standard approach for event-tree and fault-tree development as well as a standard approach for input data for initiating event frequencies, equipment performance, and human performance. These input data can be modified to be more plant- and event-specific when needed.

The NRC staff ensures the SPAR models continue to be of sufficient quality for performing event and condition assessments of operational events and degraded plant conditions in support of the staff's risk-informed activities. In 2006, the staff implemented an updated SPAR Model Quality Assurance Plan with the objective of maintaining sufficient technical adequacy of the SPAR models. There are 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 Idaho National Lab (INL) performed a one-time "cut-set level" review of all the SPAR models with the licensee's PRA model. INL also maintains a process to regularly update the SPAR models and typically 10-12 models are updated per year. Additionally, INL provides support for real time updates of the SPAR model if necessary to complete an SDP evaluation. The staff performs periodic reviews of the SPAR models, compares results against the licensee PRA models, and regularly updates the SPAR models based on operational findings and feedback to better represent the as-built, as-operated plant. The staff has processes in place for the proper use of these models in agency programs, which are documented in the Risk Assessment of Operational Events Handbook (Ref. 1) also known as the "RASP Handbook."

In addition, the staff (with the cooperation of industry experts) performed a peer review of a representative boiling-water reactor (BWR) SPAR model and pressurized-water reactor (PWR) SPAR model in accordance with American National Standard, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 2), and Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 3). This benchmarking effort is another avenue to verify the technical adequacy of the SPAR models.

The staff utilizes the SAPHIRE software to analyze the SPAR models. A number of quality assurance activities have been instituted to ensure the software continues to meet the needs of the SPAR Model Program. The staff contributed to the development of this revision of SAPHIRE by reviewing the software requirements, testing preliminary versions of the software, and providing feedback to be incorporated in the final software design. In addition, an Independent Verification and Validation (IV&V) evaluation of SAPHIRE was performed. The IV&V team assessed the SAPHIRE software for conformance with the NRC's documented technical requirements specified within the NURGE/BR-0167 "Software Quality Assurance Program and Guidelines" (Ref. 4), and where applicable, the IEEE Std 1012-2004 "Standard for Software Verification and Validation" (Ref. 5) as a secondary reference. The staff continues to maintain and improve the SAPHIRE software to support the SPAR Model Program. All SAPHIRE maintenance activities, modifications and improvements are performed in accordance with the established SAPHIRE Software Quality Assurance Plan. Companion documentation for the SAPHIRE software is published as NUREG/CR-7039, Volumes 1 through 7 (Ref. 6).

4.0 REFERENCES

1. Risk Assessment of Operational Events, "Volume 1 – Internal Events," Revision 1.03, U.S. Nuclear Regulatory Commission, Washington, DC, August, 2009.

   Risk Assessment of Operational Events, "Volume 2 – External Events," Revision 1.01, U.S. Nuclear Regulatory Commission, Washington, DC, January, 2008.

   Risk Assessment of Operational Events, "Volume 3 – SPAR Model Reviews," Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, September, 2007.

   Risk Assessment of Operational Events, "Volume 4 – Shutdown Events," Revision 1.0, U.S. Nuclear Regulatory Commission, Washington, DC, April, 2011.

2. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

3. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, March 2009.

4. NUREG/BR-0167, "Software Quality Assurance Program and Guidelines," U.S. Nuclear Regulatory Commission, Washington, DC, February 1993.

5. IEEE Std 1012-2004, "IEEE Standard for Software Verification and Validation," Revision of IEEE Std 1012-1998, The Institute of Electrical and Electronics Engineers, Inc., New York, NY, June 2005.

6. NUREG/CR-7039, "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 8," Volumes 1-7, U.S. Nuclear Regulatory Commission, Washington, DC, June 2011.

7. IMC 0609, Appendix A, "At-Power Significance Determination Process"

8. IMC 0308, Attachment 3, "Technical Basis for the Significance Determination Process"

END

Attachment 1 – Revision History for IMC 0308, Attachment 3, Appendix A

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Training Required and Completion Date	Comment and Feedback Resolution Accession Number
N/A	06/25/04 CN 04-020	Initial Issue		
N/A	ML071860533 11/08/07 CN 07-035	This IMC has been revised to reflect the changes made to IMC 0609 (03/23/2007)	None	ML072830167
N/A	ML11222A063 06/19/12 CN-12-010	Added a technical basis for the at-power screening logic questions. Removed the technical basis for the at-power Phase 2 process in support of the transition from the pre-solved tables and risk-informed notebooks to SAPHIRE and the plant specific SPAR models. Incorporated feedback from ROPFF 0308.3-1370.	Senior Reactor Analysts and headquarters staff provided detailed instructor-led training to resident inspectors, region based inspectors, and other regional staff. June 2012	ML12142A084 Closed FBF: 0308.03-1370 ML12171A218