Recent Accomplishments and Near-Term Anticipated Accomplishments-2009

This summary highlights the major risk-informed and performance-based initiatives that the staff of the U.S. Nuclear Regulatory Commission (NRC) is currently working on or has recently completed in 2009.

1. Fire Protection for Nuclear Power Plants

In 2004, the Commission approved a voluntary risk-informed and performance-based fire protection rule for existing NPPs. The rule endorsed a National Fire Protection Association (NFPA) consensus standard, NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants.” In addition, the Nuclear Energy Institute (NEI) developed NEI 04-02, “Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c),” dated September 30, 2005, that the staff endorsed in Regulatory Guide (RG) 1.205, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,” issued May 2006. The staff is working with two pilot sites (the Oconee and Shearon Harris NPPs) and has developed a frequently asked question (FAQ) process for resolving implementation issues. To date, 50 operating nuclear power units, including the pilots, have committed to transition to NFPA 805 as their licensing basis.

The staff continues its effort to implement the risk-informed fire protection rule. During the past 6 months, the staff conducted a regional inspector workshop, several pilot plant regulatory audits and supplementary clarification visits, six public FAQ meetings with the NEI-805 task force, and received supplements to the NFPA 805 License Amendment Requests (LARs) for Shearon Harris and Oconee.

The staff continues to work on the infrastructure to support the risk-informed fire protection rule. The staff is working to update RG 1.205, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants” originally issued in May 2006, to include lessons learned during the implementation of the transition to NFPA 805 by the pilot plants. In addition, the staff is developing a new Standard Review Plan section 9.5.1.2, “Risk-Informed, Performance-Based Fire Protection” to provide staff guidance for the review of licensee applications to transition to NFPA 805. The staff is also preparing a new triennial fire protection inspection procedure for licensees who have transitioned to NFPA 805. These staff activities and schedules are discussed in more detail in a plan addressing fire protection issues that was originally issued in SECY-08-0171 on November 5, 2009, and since updated every 6 months.

The staff requested and obtained Commission approval to continue to extend enforcement discretion such that nonpilot plants exhibiting sufficient progress in their NFPA 805 transition efforts will have a 6-month window to implement lessons learned from the NFPA 805 pilot plant LARs.

Over the next several months, the staff expects to continue the review of the Shearon Harris and Oconee NFPA 805 LARs and to conduct public meetings to share insights gained from the plant LARs with the nonpilot plants.

2. Risk-Informed Technical Specifications

The staff continues to work on the risk-informed technical specifications initiatives to add a risk-informed component to the standard technical specifications (STS). The following summaries highlight the major accomplishments in this area:

- Initiative 1, “Modified End States,” would allow licensees to repair equipment during hot shutdown rather than cold shutdown. The topical reports supporting this initiative for boiling-water reactor (BWR), Combustion Engineering (CE), and Babcock & Wilcox (B&W) plants have been approved, and revisions to the BWR and CE STS have been made available. The Westinghouse topical report submitted in September 2005 is currently under review, and the staff anticipates that the review will be completed in summer 2009 while revisions to the B&W STS are expected to be made available in fall 2009.

- Initiative 4b, “Risk-Informed Completion Times,” modifies technical specification completion times to reflect a configuration risk management approach that is more consistent with the approach described in the Maintenance Rule, as specified in Title 10, Section 50.65(a)(4), of the *Code of Federal Regulations*. As reported previously in SECY-07-0191, “Implementation and Update of the Risk-Informed and Performance-Based Plan,” dated October 31, 2007, the staff issued the license amendment for the first pilot plant, South Texas Project, in July 2007. The associated changes to the STS are to be submitted to the staff by the end of 2009. The industry has expressed significant interest in implementing this change over the next 5 years, with more than 40 submittals identified as being planned.

- Initiative 5b, “Risk-Informed Surveillance Frequencies,” relocates surveillance test intervals to a licensee-controlled document and provides a risk-informed method to change the intervals. The staff approved the industry’s guidance document (Revision 0 of NEI 04-10, “Risk-Informed Technical Specifications Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies”) in September 2006 along with the license amendment for the pilot plant, Limerick Generating Station. Revision 1 of NEI 04-10, which relocates staggered testing requirements and makes other administrative changes, was approved in September 2007. The associated Technical Specification Task Force guidance (TSTF-425) to revise the STS was made available in July 2009. The industry has expressed significant interest in implementing this change over the next 5 years, with 50 submittals identified as being planned.

- Initiative 6, “Modification of Selected TS for Conditions Leading to Exigent Plant Shutdown,” revises the completion times for loss-of-function conditions to allow up to 24 hours for corrective actions. A revised CE topical report was submitted for staff review in December 2007, and staff approval is anticipated in fall 2009. Other vendor topical reports are anticipated after approval of the CE report along with STS changes to implement the approved CE topical report.

3. Develop an Alternative Risk-Informed Approach to Special Treatment Requirements

The Commission decided in 1998 to consider promulgating new regulations that would provide an alternative risk-informed approach for special treatment requirements in the current regulations for power reactors. Special treatment requirements for structures, systems, and components go beyond industry-established requirements for equipment classified as "commercial grade." Special treatment requirements provide additional confidence that the equipment is capable of meeting its functional requirements under design basis conditions. These special treatment requirements include additional design considerations, qualification, change control, documentation, reporting, maintenance, testing, surveillance, and quality assurance requirements.

The Commission approved the final rule, with some modifications, in an affirmation session on October 7, 2004. The final rule was published in the *Federal Register* on November 22, 2004 (69 FR 68008). The NRC staff issued Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Revision 1, on April 28, 2006.

The topical report proposed a categorization process used by Wolf Creek Nuclear Operating Corporation in support of a future licensee submittal requesting approval to implement 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors," at the Wolf Creek Generating Station. The staff completed its review of the topical report and issued its final safety evaluation on March 26, 2009 (ADAMS Accession No. ML090260674). The staff found the categorization process described in the topical report to be acceptable, but did not approve nor endorse any specific treatment process. Treatment programs being implemented under 10 CFR 50.69 do not require prior approval from the NRC as part of the license amendment review process.

The staff plans to develop guidance for sample inspections to be conducted at plants voluntarily choosing to implement 10 CFR 50.69. The performance of sample inspections is consistent with the statement of considerations accompanying the final 10 CFR 50.69 rule. The staff plans to issue draft guidance to obtain stakeholder input and issue final guidance by the summer 2011. Inspection efforts will be focused on the most risk significant aspects related to implementation of 10 CFR 50.69 (i.e., proper categorization of SSCs and treatment of Risk-Informed Safety Class (RISC) 1 and RISC-2 SSCs). Additionally, the inspections are expected to be performance based, with lower safety significant function SSCs, such as those classified RISC-3, not receiving a major portion of inspection focus unless adverse performance trends are observed.

The staff recognizes the need for an effective, stable and predictable regulatory climate for the implementation of 10 CFR 50.69. Inspection guidance developed with industry stakeholder input is viewed as an efficient vehicle for reaching a common understanding of what constitutes an acceptable treatment program for SSCs since specific treatment plans are not reviewed as part of a licensee's application to implement 10 CFR 50.69.

4. Initiative to Enhance Risks Tools for Oversight

The NRC staff uses a suite of risk tools to support oversight of nuclear reactors such as risk assessment software, Standardized Plant Analysis Risk (SPAR) models, databases, guidance for the Significance Determination Process (SDP) and other risk methodologies, and associated training. In May 2009, the staff initiated a structured assessment involving internal stakeholders in NRR, RES and each region to define, prioritize, and implement enhancements to those risk tools used by risk analysts, inspectors and their management in the agency's oversight of nuclear reactors. This evaluation helps to identify needed modifications to be considered for maintaining the quality of the risk tools, and making enhancements to improve their efficient use and advance the state-of-the-art quality of risk tools.

This summer, the staff obtained input from internal stakeholders in a series of meetings. Over 300 suggestions were received. The suggestions were grouped and prioritized in terms of benefit to the agency and resources needed. Currently, the staff is developing a 5-year program plan identifying those enhancements which we intend to implement.
The desired outcome of this initiative is to assure availability of a suite of high quality NRC risk analysis tools that are technically sound for consistent risk assessments of inspection findings, operational events and issues in other regulatory applications. In addition, the initiative should also provide appropriate training information associated with each risk tool to assure NRC staff is adequately trained in the proper use of the risk analysis tools.

5. Risk-Informed Rulemaking and Related Activities Currently in Progress

The staff continues to work on several risk-informed rulemaking initiatives. The following summary highlights major accomplishments.

The staff prepared a proposed rule containing emergency core cooling system evaluation requirements that could be used as an alternative to the current requirements in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water Nuclear Power Reactors." That proposed rulemaking is designed to redefine the large-break loss-of-coolant accident requirements to provide a risk-informed alternative maximum break size. In October 2006, the staff produced a draft final rule and briefed the Advisory Committee on Reactor Safeguards (ACRS). In response, the ACRS recommended that the Commission should not issue the proposed rule in its present form. As a result, the staff prepared SECY-07-0082, "Rulemaking To Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements: 10 CFR 50.46a, 'Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,'" dated May16, 2007, which provided a plan (including resource and schedule estimates) for responding to the ACRS recommendation and related comments. Then, in an SRM related to SECY-07-0082 dated August 10, 2007, the Commission agreed with the staff's recommendation that completing the rulemaking should be assigned a medium priority. Nonetheless, the SRM also directed that the staff continue to make progress on the 10 CFR 50.46a rulemaking and to apply resources to the effort in FY 2008.

On April 1, 2008, the Executive Director for Operations provided the staff's schedule for completing the final rule to the Commission. Following Commission approval, the NRC published a supplemental proposed rule, 74 FR 40765, August 10, 2009 (Performance-Based

Emergency Core Cooling System Acceptance Criteria) for public comment. The public comment period ends on October 27, 2009. After reviewing public comments, and making any changes based on those comments, a final rulemaking package will be provided to the Commission.

On October 3, 2007, the staff published a proposed rulemaking on “Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.” The proposed rule contained a new 10 CFR 50.61a that will provide new requirements that a pressurized-water reactor licensee could voluntarily use as an alternative to complying with the existing requirements. NRC received over 40 comments during the public comment period that ended on December 17, 2007. Some comments recommend major changes to the rule such as deleting the requirements that licensees identify and document the distribution of flaws in their reactor vessel and use a data-based trend curve contained in the rule. The use of the trend curve in the rule was changed and a supplemental proposed rule was published (73-FR-46557 August 11, 2009). Three comments were received during the comment period that closed on September 10, 2008.

The staff completed the final rulemaking package that will amend the regulations in Title 10 CFR 50.61 that describe the fracture toughness requirements for protection against pressurized thermal shock (PTS) events for pressurized-water reactors. The Commission approved the final PTS rule in its September 22, 2009, SRM on SECY-09-0059, “Final Rule Related to Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a).” The rule will be published in the Federal Register after Office of Management and Budget review and clearance.

6. Advanced Reactor Regulatory Structure

The staff issued NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing,” Volumes 1 and 2, in December 2007. This NUREG documents a framework that provides an approach, scope, and criteria that could be used to develop an alternative set of risk-informed and performance-based requirements to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” for future NPPs. Part of the framework establishes a probabilistic approach for identifying design basis events (i.e., Anticipated Operational Occurrences and Design Basis Accidents). The staff is developing a draft regulatory guide using the approach from the framework. This draft guide was scheduled to be complete for internal NRC review at the end of 2008; however, due to higher priority work, the schedule was revised for draft at the end of 2009.

In SECY-09-0056, “Staff Approach Regarding a Risk-Informed and Performance-Based Revision to Part 50 of Title 10 of the *Code of Federal Regulations a*nd Developing a Policy Statement on Defense-In-Depth for Future Reactors,” dated April 7, 2008 (ADAMS Accession No. ML090360197), the staff stated that it plans to defer rulemaking activities for risk-informed and performance-based reactor requirements for future reactors until it conducts a test review of a license application for the NGNP prototype design or other non-light-water reactor (LWR) design. Moreover, the staff indicated that it plans to defer activities to finalize a defense-in-depth policy statement until additional experience and related insights are gained from the Next Generation Nuclear Plant or other non-LWR reviews. However, it further stated that it plans to continue to develop a position on defense-in-depth that is integrated with other related policy and key technical positions, and test these proposed positions during the review of an actual design.

7. Infrastructure for Risk-Informed and Performance-Based Environment for New Light Water Reactors

During FY 2009, the staff developed a white paper and held two public meeting regarding the implementation of risk-informed applications for new LWRs. The discussions in these forums address the regulatory framework as applied to currently operating reactors and highlight potential implementation issues if and when applied to new reactor designs.

In FY 2010, the Agency will continue to develop the infrastructure and programs to foster a risk-informed and performance-based environment. These activities will include:

- Continued development of requirements specific to new and advanced LWRs in consensus probabilistic risk assessment (PRA) standards.

- Continued discussions at public forums regarding risk-informed initiatives 4b and 5b on technical specification completion times and surveillance frequency control program, respectively, for new reactors.

- Continued discussions regarding the implementation of risk-informed applications for new LWRs, including public meetings, briefings before the ACRS, and identification of any policy for Commission consideration.

- Continued support of risk-informed elements of inservice inspection of piping as it pertains to new reactors.

- Update to the risk insights documents prepared to assist the staff in its review of the five new reactor design centers with combined license applications (ABWR, AP1000, ESBWR, U.S. EPR, US-APWR) and support of inspection activities of new reactors by Region II staff.

8. Phased Approach to Probabilistic Risk Assessment Quality

The increased use of PRAs in NRC's regulatory decisionmaking process requires consistency in the quality, scope, methodology, and data used in such analyses. A key aspect of implementing a phased approach to PRA quality is the development of PRA standards and related guidance documents. To achieve that objective, professional societies, the nuclear industry, and the staff have undertaken initiatives to develop national consensus standards and guidance on the use of PRA in regulatory decisionmaking.

Revision 2 to Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," was issued in March 2009. This revision addressed concerns regarding model uncertainties and related assumptions and included guidance for new and advanced LWRs. This revision also endorsed Addenda A to the joint PRA ASME/ANS standard, "Level 1 and Large Early Release Frequency (LERF) PRA Standard" (ASME/ANS RA-Sa-2009), that was published in February 2009. This standard applies to at-power internal and external hazards for operating reactors. Additional related documents endorsed in RG 1.200 include revisions to NEI documents on PRA and fire PRA peer review (i.e., NEI 05-04 and NEI 07-12, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard" and "Fire Probabilistic Risk Assessment Peer Review Guidelines," respectively).

The staff is supporting other PRA standards efforts and will consider endorsing these standards, once issued, in future revisions to RG 1.200. These other standard efforts include:

- Level 1/LERF standard for internal events at low-power and shutdown conditions for operating reactors.

- Level 1/LERF standard for at-power internal and external hazards for new and advanced LWRs.

- Level 2 and Level 3 for at-power internal hazards for LWRs and non-LWRs.

- Levels 1, 2, and 3 for internal and external hazards for all operating modes for advanced non-LWRs.

The staff is also working with the American Society of Mechanical Engineers (ASME) in development of training on the ASME/American Nuclear Society (ANS) probabilistic risk assessment (PRA) standard. This training comprises two modules. The first module is a 1-hour Web-based course designed for both managers and PRA practitioners and is scheduled to be available by December 2009. The second module is classroom style designed for PRA practitioners. It is divided into four separate elements covering internal events and internal floods over a 3 ½-day period. It is scheduled to be available in 2010.

With the issuance of Revision 2 to RG 1.200, risk-informed application-specific regulatory guides were updated to reference RG 1.200 to address the issue of the technical acceptability of the base PRA. Draft Regulatory Guides 1226 and 1227, proposed revisions to RGs 1.174 and 1.177 ("An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" and "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," respectively) were issued for public review and comment in August 2009.

In FY 2009, the agency also supported the review of the draft PRA standard for advanced non-LWRs, and the standard on risk-informed approaches to establishing nuclear safety design criteria for modular helium-cooled reactor plants.

The staff issued NUREG-1855, "Treatment of Uncertainties from PRAs in Risk-Informed Decision Making," in March 2009. The NRC report and a complementary Electric Power Research Institute (EPRI) report provide guidance on meeting the requirements in the ASME/ANS PRA standard on uncertainties and provide guidance on how to treat the results from the uncertainty analyses in decisionmaking for risk-informed activities. NRC and EPRI cosponsored a 2-day public workshop in May 2009. The meeting had over 70 participants including representatives from industry (owners groups, vendors, utilities, consultants, etc.) and the National Aeronautics and Space Administration. The workshop was held to explain how to use the NRC and EPRI guidance to satisfy the requirement in the PRA standard on uncertainties and how to address those uncertainties in decisionmaking. Participants found the workshop useful but requested additional examples on implementation of the guidance. The staff is developing additional examples and expanding the scope of the NUREG. A revision is planned for 2010.

9. Human Reliability Analysis

The staff is addressing issues associated with the differences in the many HRA methods available for quantifying human failure events in a PRA. In addition to supporting the agency's plan to stabilize and enhance PRA quality, the staff also is following up on a Commission staff requirements memorandum (M061020).

The Commission directed the Advisory Committee on Reactor Safeguards (ACRS) in staff requirements memorandum (SRM) (M061020) to "work with the staff and external stakeholders to evaluate the different human reliability models in an effort to propose a single model for the agency to use or guidance on which model(s) should be used in specific circumstances." Consequently, the staff will present its findings to the ACRS for its review. Moreover, the staff has initiated efforts to address SRM-M090204B to collect data and test HRA methods using U.S. nuclear plant operating crews.

The staff supports and participates in the International HRA Empirical Study, an experimental study performed collaboratively by about a dozen regulatory and industry organizations and members of the Halden Reactor Project. This study involves the collection of reactor operator crew performance observations and comparison with the results of different HRA methods used to evaluate the actions involved in simulated scenarios. The pilot phase of this study was documented in the draft NUREG/IA-0216/HWR-844. The staff expects the study will be completed by December 2010. The staff plans to document the methodology and results of the study in a final NUREG/IA to be submitted for publication in December 2011.

The staff also has established a Memorandum of Understanding (MOU) with the EPRI to work together to identify areas where HRA has a significant impact on regulatory decisionmaking. The main tasks of this work include:

- Identification of current and anticipated regulatory applications in which HRA results could have an impact on the decision.

- Examination of the suitability and adequacy of these methods in the areas in which they are applied.

- Determination of whether a single model could support all regulatory needs or whether an improved small set of methods along with appropriate guidance and training material is more suitable.

- Submission of the results to public comment.
- Production of a final NUREG report.

The staff expects to complete the work in September 2011.

10. HRA Development for Fire PRA

Under a joint MOU, NRC's Office of Nuclear Regulatory Research (RES) and EPRI have embarked on a cooperative program to improve the state-of-the-art in fire risk studies. This program produced a joint document, EPRI 1011989 & NUREG/CR-6850, entitled "Fire PRA Methodology for Nuclear Power Facilities" (ML052580075, ML052580118) that addresses fire risk for at-power operations. Because this joint NRC/EPRI report does not describe a methodology for developing best-estimate human failure probabilities, a new effort is underway to develop such a methodology and associated guidance, including peer review and testing. The results of this HRA methodology development effort is expected to support the NFPA 805 transition initiative and possible resolution of other regulatory issues, such as multiple spurious operation and operator manual actions.

In 2008, a peer review was performed and testing on the selected plants was completed. In May 2009, feedback from both of these efforts was reviewed and addressed, resulting in a revised draft of the NUREG-1921 "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines." This revised draft was internally reviewed, and an overview was presented to the ACRS HRA subcommittee in June 2009. In the next 6 months, the draft report is expected to be released for public comment, and the staff, along with its EPRI counterparts, will respond to these comments. Publication of the final report is expected in 2010.

11. Analytical Tools for Risk Applications

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) is a software application developed for performing PRA using a personal computer running the Windows operating system. SAPHIRE is used to model a plant's response to initiating events or conditions and to quantify associated consequential outcome frequencies. Over the past 6 months, SAPHIRE Version 8 has been in beta testing. It runs the Agency's Standardized Plant Risk (SPAR) models that are used in many regulatory inspections and research programs. SAPHIRE Version 8 features and capabilities address new code requirements in support of risk-informed programs, including the development of a user interface for significance determination process (SDP) Phase 2 assessments. SAPHIRE 8 is scheduled for release in April 2010, and SAPHIRE 7 will be phased out.

12. SPAR Model Development and Risk Assessment Standardization Project

SPAR models are plant-specific PRA models that model accident sequence progression, plant systems and components, and plant operator actions. The standardized models represent the as-built, as-operated plant and, as such, permit the staff to perform risk-informed regulatory activities by independently assessing the risk of events or degraded conditions at operating NPPs. Over the past 6 months, the staff accomplished the following:

- Completed initial detailed cut-set level review of all SPAR models and updated 12 SPAR models.

- Continued the cooperative research activities under the RES/EPRI MOU addendum to address resolution of key technical issues with the industry. The staff, working with industry, jointly issued the first draft guidance document on Support System Initiating Events (SSIE) in January 2009. The staff continues working with industry to resolve the remaining SSIE issues and also is working with industry to issue a guidance document on treatment of loss of offsite power (LOOP) in PRAs in late 2009. The staff plans to continue this cooperative effort with EPRI to address the remaining SPAR/PRA model issues over the next 2 years.

- Completed new next-generation low-power/shutdown (LP/SD) models for two plants.

- Completed the development of an AP1000 model for the Office of New Reactors (NRO) in response to a recent user need from NRO, "Development of Standardized Plant Analysis Risk Models for New Reactors," dated March 25, 2008. Prior to new plant operation, the NRC staff may need to perform risk assessments to evaluate risk-informed applications after combined license issuance. The staff will begin development of an Advanced Boiling Water Reactor (ABWR) SPAR model in FY 2010.

- In addition to the above model enhancements, the staff completed an evaluation of the strategies implemented in support of B.5.b to mitigate severe accidents related to core damage for a out two-thirds of the licensees. The remaining licensees are scheduled to be evaluated by October 2010.

- In August 2009, the staff, working with industry Owners Groups, completed the peer review of a boiling-water reactor (BWR) SPAR model in accordance with the latest American National Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications. The peer review teams are made up of industry experts, senior reactor analysts from the Regions, and other knowledgeable staff. Industry participation ensured that the SPAR models were reviewed consistent with the industry peer reviews. Because the SPAR models are standardized, it was decided that a peer review of a typical BWR SPAR model and a typical pressurized-water reactor (PWR) SPAR model would be sufficient for the 77 SPAR models representing the 104 operating reactors. The PWR SPAR model peer review is scheduled for October 2009.

In FY 2010, the staff plans to continue implementing enhancements to the Revision 3 SPAR models and to complete additional external events and LP/SD models to support the Accident Sequence Precursor (ASP) Program and the Significance Determination Process. Major enhancements planned for FY 2010 include improvements to applicable SPAR model success criteria based on the results of staff-developed and reviewed thermal-hydraulic analysis and transition of the SPAR models to SAPHIRE version 8. The changes to the SPAR models are required to take advantage of the many new features in SAPHIRE 8. The staff also continues to provide technical support to SPAR model users and risk-informed programs.

13. Reactor Performance Data Collection/Industry Trends

The staff has been collecting data and information for over 20 years to support studies and risk analyses of nuclear power plant operational experience. The information comes from diverse sources including Licensee Event Reports (LERs), the Institute of Nuclear Power Operations Equipment Performance and Information System (EPIX) and its processor database, the Nuclear Plant Reliability Data System, and Monthly Operating Reports. These data collection efforts have been consolidated into a single system, the Integrated Data Collection and Coding (IDCCS). Over the past 6 months, the staff updated the IDCCS with calendar year 2008 data including the latest LERs into the LERSearch database so that it now reflects LERs from 1981 through March 2009. LERSearch, the LER search system on the NRC internal Web site, has been enhanced to provide additional search options and more risk-related operational data. Over the next year, the staff plans to make LERSearch publicly available on the external NRC Web site.

Beginning in the summer 2009, the staff began an effort to make a comprehensive update to the existing fire database. The updated database will be based on proprietary industry-reported data provided to NRC as a result of a cooperative agreement with EPRI. It is expected that this database will replace the existing one. The staff plans to develop and track enhanced fire metrics that will be made possible by use of the new database.

The Industry Trends Program Support program uses data collected from LERs, EPIX, EPRI-sponsored fire activities, and Monthly Operating Reports to regularly update estimates of industrywide and plant-specific system and component reliabilities, initiating event frequencies, common-cause failure parameters, and fire event frequencies. These data are important for implementing a risk-informed and performance-based approach to regulation.

This program also produces guidance and data for the Risk Assessment Standardization Project (RASP). The RASP is developing standard procedures and methods for risk assessment of inspection findings and reactor incidents. Such procedures and methods can be used to implement the performance-based aspects of NRC's regulatory practice in activities such as the reactor oversight process. Over the coming year, the staff plans to issue the following draft NUREG series reports for comment to provide guidance for RASP:

- "Estimating Pipe Break Loss-of-Coolant Accident Frequencies Using NUREG-1829 Information."
- "Common-Cause Failure Analysis in Event Assessment."
- "Data Guidance for the Risk Assessment Standardization Project."
- "Industry Performance of Relief Valves at U.S. Nuclear Power Plants through 2007."

14. Digital Systems Probabilistic Risk Assessment

The Risk-Informing Digital Instrumentation and Control Task Working Group (TWG), in support of the Digital Instrumentation and Control Steering Committee, has been addressing issues related to the risk assessment of digital instrumentation and control (I&C) systems. In this effort, the TWG has been placing particular emphasis on risk-informing digital I&C system reviews for operating plants and new reactors. The TWG efforts have been consistent with NRC's Policy Statement on Probabilistic Risk Assessment, which states in part that the agency supports the use of PRA in regulatory matters "to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." Toward that end, the TWG issued an updated project plan on March 14, 2008. The TWG has held several public meetings with industry stakeholders since April 2007. On December 3, 2007, the staff issued the draft interim staff guidance (ISG) for new reactors for public comment. This ISG is intended for use in reviewing current methods in modeling digital I&C systems for design certification and combined license (COL) application PRAs. The TWG discussed the draft ISG with stakeholders in public meetings held in February, March, and May 2008 and with the Advisory Committee on Reactor Safeguards (ACRS) on March 20, 2008, and May 11, 2008. The TWG also supported a Commission brief on April 7, 2008. After addressing ACRS and industry comments, the staff issued the TWG ISG on August 11, 2008.

The ACRS also provided comments during two briefings by the staff on the application of traditional PRA methods to digital I&C systems (April 17, 2008, and May 8, 2008). The ACRS emphasized the importance of failure mode identification, the limitations of sensitivity studies that dealt with probabilities, the usefulness of available failure rate data sources, and the current limitations of "traditional" PRAs in identifying failure modes. Given the ACRS comments and the staff's concerns, the staff reassessed the problem statement and associated project plan on the application of current PRA methods to risk-inform specific digital I&C system issues for operating reactors. The concern is that given the stated limitations in PRA technology, the development and implementation of a risk-informed methodology per the current project plan using traditional PRA methods may be premature.

However, the staff continues its research into PRA methodologies for assessment of digital instrumentation and control (I&C) system risk. Previous and current research projects have identified a set of desirable characteristics for reliability models of digital I&C systems and have applied various probabilistic reliability modeling methods to an example digital system. This work is documented in several NUREG/CR reports that have received extensive internal and external stakeholder review. Reports published in the past year include NUREG/CR-6962, "Traditional Probabilistic Risk Assessment Methods for Digital Systems" (October 2008); NUREG/CR-6985, "A Benchmark Implementation of Two Dynamic Methodologies for the Reliability Modeling of Digital Instrumentation and Control Systems" (February 2009); and NUREG/CR-6997, "Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods" (September 2009). The results of these "benchmark" studies have been compared to the set of desirable characteristics to identify areas where additional research might improve the capabilities of the methods.

One specific area that is currently being pursued is the quantification of software reliability. Given the lack of a consensus on how, or even if, to model software failure in an NPP PRA, a workshop involving experts with knowledge of software reliability and/or NPP PRA was convened in May 2009. At the workshop, the experts established a philosophical basis for modeling software failures in a reliability model. The staff is now performing a review of quantitative software reliability methods, and plans are underway to develop one or two technically sound approaches to modeling and quantifying software failures in terms of failure rates and probabilities. Assuming such approaches can be developed, they will then be applied to an example software-based protection system in a proof-of-concept study.

In October 2008, the staff also led an Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA)/Committee on the Safety of Nuclear Installations (CSNI)/Working Group on Risk (WGRisk) technical meeting on digital I&C risk modeling. The objectives of this meeting were to make recommendations regarding current methods and information sources used for quantitative evaluation of the reliability of digital I&C systems for PRAs of NPPs, and to identify, where appropriate, the near- and long-term developments that would be needed to improve modeling and evaluating the reliability of these systems. During the meeting, it was recognized that although many studies have been performed in various countries, the models of digital I&C systems developed so far have a wide variation in terms of scope and level of detail, and no consensus has been reached on what is an acceptable method for modeling digital systems. The participants agreed that probabilistic data are scarce, so an urgent need exists to address this shortcoming. Although the meeting did not result in identification of specific recommendations regarding what methods or information sources should be used for quantitative evaluation of the reliability of digital I&C systems for PRAs of NPPs, it did provide a useful forum for the participants to share and discuss their respective experiences with modeling these systems.

15. Developing SDP Tools for the Fuel Cycle Oversight Process (FCOP) Revision

In an SRM dated April 3, 2008, the Commission directed NRC staff to "…continue to make the fuel cycle performance review process more transparent and risk-informed." To develop an oversight process that has an improved degree of objectivity, predictability, transparency, and consistency that incorporates risk-informed and performance-based tools, the Executive Director for Operations (EDO) directed the staff to undertake a comprehensive effort to develop a new oversight process for fuel cycle facilities. The Commission provided additional guidance on the desired revisions in the SRM dated February 17, 2009, and in a memorandum dated March 13, 2009, from the Office of Nuclear Materials Safety and Safeguards (NMSS) and Region 2 to the EDO, the staff described its plan for revising the fuel cycle oversight process.

Currently NMSS, NSIR, OE, and Region 2 are revising and the developing the framework of the FCOP. RES is supporting this effort by developing SDP tools that will be used to evaluate the significance of inspection findings in fuel cycle facilities. Beginning in July 2009, NRC and Brookhaven National Laboratory (BNL) started to work on developing the SDP tools. Over the next 4 months, NRC is expected to have an internal peer review and the Nuclear Energy Institute (NEI) peer review of the first draft of the tools. The staff is working with licensees, NEI, and other stakeholders through a series of public workshops to inform the development of the revised oversight process, including the proposed SDP tools.

16. Disposal of Significant Quantities of Depleted Uranium

Depleted uranium is considered source material, in accordance with 10 CFR Part 40, "Domestic Licensing of Source Material," and if treated as a waste would fall under the definition of low-level radioactive waste per 10 CFR 61.55(a). The Commission reaffirmed this waste classification in Memorandum and Order CLI-05-20 dated October 19, 2005. Consistent with Commission policy to increase the use of risk assessment technology in all regulatory matters, the NRC staff considered in a risk-informed screening analysis (SECY-08-0147), dated October 7, 2008, whether quantities of depleted uranium at issue in the waste stream from commercial uranium enrichment facilities warrant amending 10 CFR 61.55(a)(6) or 10 CFR 61.55(a) waste classification tables.

The Commission directed the staff in a Staff Requirements Memorandum (SRM-SECY-08-0147), dated March 18, 2009, to pursue a limited rulemaking to specify a requirement for a site-specific analysis and associated technical requirements for unique waste streams including, but not limited to, the disposal of significant quantities of depleted uranium. In pursuing this limited rulemaking, NRC is not proposing to alter the waste classification scheme. However, for unique waste streams including, but not limited to, significant quantities of depleted uranium, a need may exist to place additional criteria on its disposal at a specific facility or to deny such disposal based on unique site characteristics. Those restrictions would be determined via a site-specific performance assessment analysis, which satisfies the requirements, developed through the rulemaking process.

On June 24, 2009, NRC announced in the *Federal Register* , 74 FR 30175, that it is seeking early public input on major issues associated with potential rulemaking for land disposal of unique waste streams including, but not limited to, significant quantities of depleted uranium in near-surface, low-level radioactive waste facilities. NRC staff conducted public workshops in Rockville, Maryland, on September 2-3, 2009, and in Salt Lake City, Utah, on September 23-24, 2009, to discuss issues associated with rulemaking.

After the public workshops, the staff will begin development of the technical basis for the draft rulemaking. The technical basis will consider the input from the stakeholders at the workshops and the input provided to the docket.

17. Risk Assessment of Red Oil Excursions

Brookhaven National Laboratory performed an independent quantitative risk assessment of red oil excursion events in the MOX Fuel Fabrication Facility based on the design in the application for an operating license. This study provided risk insights and an independent perspective on risk to the NRC staff concerning this phenomenon.

18. Risk-Informed Prioritization of Fuel Cycle Items to be Inspected

A method of prioritizing the operational readiness inspection of Items Relied on for Safety (IROFS) at new fuel cycle facilities had been previously developed by personnel at Region II. This method was applied to two centrifuge enrichment plants by a joint panel of technical experts familiar with these facilities. The method considers two major factors in evaluating the priority of an IROFS: 1) the increase in risk if the IROFS were ineffective, and 2) the rank of the item in terms of the probability of a deficiency rendering it ineffective. The first factor, the increase in risk, is evaluated based on information from the applicant's Integrated Safety

Analysis. The second factor is evaluated by the prioritizing panel to assign a numerical rank considering the following sub-factors:

a) complexity of the system;
b) knowledge and experience of the applicant with the type of process;
c) reliability of the type of IROFS;
d) level of detail provided concerning the process design.

IROFS were then prioritized by the panel in order of the product of the two major factors. This prioritized list was then provided to Region II to assist in inspection planning. Lessons learned from this exercise will be used to further refine the prioritization method for application to other facilities.