Recent Accomplishments and Near-Term Anticipated Accomplishments-2011

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 2011.

1. Fire Protection for Nuclear Power Plants

In 2004, the Commission approved a voluntary risk-informed and performance-based fire protection rule for existing nuclear power plants. The rule endorsed 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. To date, nearly half of the U.S. operating nuclear power units, including the pilots, have committed to transition to NFPA 805 as their licensing basis.

The Oconee and Shearon Harris plants were the pilot plants for 50.48(c). The Shearon Harris NFPA 805 pilot application was approved via a safety evaluation in June 2010. The Oconee NFPA 805 pilot application was approved via a safety evaluation in December 2010.

NEI 04-02 was revised (Revision 2) in April 2008 and the staff revised RG 1.205 (Revision 1) in December 2009 to reflect lessons learned from the pilot reviews. The staff developed NUREG-800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 9, "Auxiliary Systems," Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program Review Responsibilities," issued December 2009, to provide staff guidance for the review of licensee applications to transition to NFPA 805. Additionally, a Frequently Asked Question (FAQ) process has been developed to review and establish a preliminary staff position on application, review, and implementation issues.

Lessons learned from the pilot applications indicated that the staff and the industry underestimated the complexity and resources necessary to complete the reviews. In a staff requirements memorandum (SRM) to SECY-11-0033, "Proposed NRC Staff Approach to Address Resource Challenges Associated with Review of a Large Number of NFPA 805 License Amendment Requests," dated April 20, 2011, the Commission approved the staff's recommendation to increase resources for NFPA 805 applications, develop a staggered review process, and to modify the current enforcement policy. The revised enforcement policy was sent to the Commission in SECY-11-0061, "A Request to Revise the Interim Enforcement Policy for Fire Protection Issues on 10 CFR 50.48(c) to Allow Licensees to Submit License Amendment Requests in a Staggered Approach," dated April 29, 2011 and approved in SRM-SECY-11-0061, dated June 10, 2011. To enhance the efficiency and effectiveness of the NFPA 805 application reviews, the industry developed an application template and the staff developed a safety evaluation template. The staff has received five applications; one has been accepted and four are undergoing acceptance reviews.

2. Risk-Informed Technical Specifications

The staff continues to work on the risk-informed technical specifications (RITS) 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 was approved in March 2010 while revisions to the B&W STS were approved and made available in December 2010.

- 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 industry has expressed significant interest in implementing this change over the next 5 years, with more than 40 submittals identified as planned. In July 2010, Southern Nuclear Company (SNC) submitted a letter of intent for Vogtle (for Units 1 and 2) to implement RITS Initiative 4b. The NRC granted the associated fee waiver request and the agency expects a pilot application in March 2012. By letter dated September 19, 2008, Luminant Power submitted a new reactor combined license application for Comanche Peak Nuclear Power Plant Units 3 and 4 that included Initiative 4b. The application is currently being reviewed by staff.

- 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 to 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. Numerous plants (over 30) have received approval via safety evaluations under Initiative 5b. By letter dated September 19, 2008, Luminant Power submitted a new reactor combined license application for Comanche Peak Nuclear Power Plant Units 3 and 4 that included Initiative 5b. This application is currently being reviewed by staff.

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

In 1998, the Commission decided to consider promulgating new regulations that would provide an alternative risk-informed approach for special treatment requirements in the current regulations for power reactors. The final rule (10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors"), 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 staff completed its review of Westinghouse topical report WCAP-16308-NP (Revision 0, July 2006), "Pressurized Water Reactor Owners Groups 10 CFR 50.69 Pilot Program – Categorization Process – Wolf Creek Generating Station," and issued its final safety evaluation on March 26, 2009 (ADAMS Accession No. ML090260674). By letter dated December 6, 2010, SNC informed the NRC of its intent to submit a license amendment request for implementation of 10 CFR 50.69 for Vogtle Units 1 and 2 and requested pilot plant status and a waiver of review fees. By letter dated June 17, 2011, the staff informed SNC that the NRC has granted the fee waiver request for the proposed licensing action in accordance with 10 CFR 170.11(b). SNC has indicated that it plans to submit a licensing action request in early 2012. Following the initial pilot application, lessons learned from the application review will be used to revise the associated industry guidance and RG 1.201.

In addition, the NRC staff issued draft Inspection Procedure 37060, "10 CFR 50.69 Risk-Informed Categorization and Treatment of Structures, Systems, and Components Inspection," on February 16, 2011. NEI and one licensee provided comments on the procedure. The NRC staff addressed the comments and plans to issue final guidance in 2011. The NRC will focus its inspection efforts 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 SSCs of lower safety significance (e.g., 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. The NRC views inspection guidance developed with industry stakeholder input 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. During the pilot application review, the staff expects to continue to work with the industry and pilot licensee to modify the inspection procedure to reflect lessons learned and information gleaned from the pilot's proposed treatment program.

4. NRC Risk Network

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 associated training. The Risk Network (previously referred to as Risk Tool Enhancement) project represents a structured assessment involving internal stakeholders in the Office of Nuclear Reactor Regulation (NRR), Office of Nuclear Regulatory Research (RES), and each Region to define, prioritize, and implement enhancements to risk tools used by risk analysts, inspectors and their management. In February 2010, the staff issued a Risk Network project plan that organizes input received from internal stakeholders on enhancements for maintaining the quality, improving the efficient use, and advancing the state of the art of the NRC's risk tools.

There are currently 73 tasks managed under the Risk Network Project that address the enhancement or maintenance of NRC risk tools, procedures, or training. Technical leads for each task were identified and the tasks were prioritized in terms of their benefit to the agency and resources needed. In addition, the agency established the Risk Network oversight team, consisting of managers from NRR, RES, and each Region. The purpose of this team is to oversee the Risk Network project schedule and work products. The desired outcome of the Risk Network project is to ensure the availability of high quality NRC risk analysis tools that are technically sound and to ensure adequate training for the staff to use the risk tools. Approximately 20 of the 73 tasks have been completed.

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

The staff continues to work on several risk-informed rulemaking initiatives. The summary below highlights the 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, 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 May 16, 2007, to provide a plan (including resource and schedule estimates) for responding to the ACRS recommendation and related comments. Then, in SRM-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 the staff to continue to make progress on the 10 CFR 50.46a rulemaking and to apply resources to the effort in fiscal year (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 13, 2009 (Performance-Based Emergency Core Cooling System Acceptance Criteria) for public comment. The public comment period ended in January 2010. After reviewing public comments, and after making any changes to address these comments and ACRS comments, the staff submitted a final rulemaking package to the Commission for approval in December 2010.

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

The staff continues to address the issue of risk-informed regulatory guidance for new light-water reactors (LWRs). A Commission briefing was held on the topic on October 14, 2010. Subsequently, On March 2, 2011, the Commission issued SRM to SECY-10-0121, “Modifying the Risk-Informed Regulatory Guidance for New Reactors” (ADAMS Accession No. ML110610166), to direct the staff to continue to use the existing risk-informed framework, including current regulatory guidance, for licensing and oversight activities for new plants, pending additional analysis. The Commission also directed the staff to undertake a number of tasks, including the following:

- The staff should articulate, in a single document, a coherent overview of the Commission’s policies and decisions regarding new reactor safety performance for the purposes of public communication and NRC staff knowledge management.

- The staff should engage external stakeholders in a series of tabletop exercises to test various realistic performance deficiencies, events, modifications, and licensing bases changes against current NRC policy, regulations, guidance and all other requirements (e.g., technical specifications, license conditions, and code requirements) that are or will be relevant to the licensing bases of new reactors.

- The staff will submit a notation vote paper with options and recommendations to the Commission by June 4, 2012.

Since the issuance of the SRM, the staff has held six public meetings with stakeholders. The kickoff meeting on the response to the SRM took place on March 24, 2011 (ADAMS Accession No. ML110840607). The first tabletop exercise on risk-informed in-service inspection of piping took place on May 4, 2011 (ADAMS Accession No. ML111330381). Workshops held on May 26, 2011 and June 1, 2011 related to Risk-Informed Technical Specifications Initiative (RITS) 4b (completion times), and Maintenance Rule 50.65(a)(4) (ADAMS Accession Nos. ML111650176 and ML111650341). The tabletop exercise on RITS 5b (Surveillance Frequency Control Program) took place on June 29, 2011 (ML11182A976). On August 9, 2011, a tabletop exercise on 50.69 (Risk-Informed Categorization and Treatment of Structures, Systems, and Components) was conducted (ADAMS Accession No. ML112290891). In addition, participants at the August 9, 2011, workshop discussed proposed changes to NEI 96-07 (10 CFR 50.59 guidance) on the new reactor change process under Section VIII.B.5.c of each design certification rule. This regulation relates to changes to ex-vessel severe accident design features during construction and operation. The proposed changes to the NEI guidance resulted from a public workshop held on December 2, 2010 on the subject matter (ADAMS Accession No. ML110130408).

The staff has drafted a summary-level public communication brochure regarding new reactor safety performance. The brochure is currently undergoing internal review before issuance.

An informational briefing of the ACRS subcommittee on Reliability and PRA took place on September 20, 2011.

7. Human Reliability Analysis

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

The Commission directed ACRS in 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 interact frequently with ACRS to incorporate its input on all facets of the work, including the technical approach and its development, implementation and deployment process. 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 approximately a dozen regulatory and industry organizations and members of the Halden Reactor Project (HRP). 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 NRC published the results of the study in NUREG/IA-0216, "International HRA Empirical Study-Phase 1 Report," Volume 1, issued November 2009 and Volume 2, issued August 2011. Volume 1 documents the pilot study, and Volume 2 documents the results of SGTR scenarios. Volume 3, to be published by December of 2011, will document the results of loss of feedwater scenarios. The overall lessons learned from the study will be published as a separate NUREG report expected to be published by February 2012.

Utilizing the results from the international HRA study and previous HRA method evaluations, the staff is performing technical work to address SRM-M061020. The approach aims to address (1) the issue of variability in HRA through the adoption of a formalization process that guides the identification of potential human failures, (2) the use of an explicit human performance framework for establishing causal relationships of human failures to underlying failure mechanisms, and (3) the use of the current understanding of cognitive psychology as a technical basis for postulating failure events, failure mechanisms, and underlying performance drivers. It also intends to use a mathematical formulation consistent with the overall PRA framework to estimate failure probabilities. The staff believes that this approach will result in a single architecture for HRA that ensures consistency and adequacy for all HRA applications. This work is being performed collaboratively with the Electric Power Research Institute (EPRI) under a Memorandum of Understanding (MOU) to address the issue of variability in HRA. The staff expects to complete the work in September 2012.

As part of the direction in SRM-M090204B to collect data and test HRA methods using U.S. nuclear plant operating crews, the staff has established an MOU with a U.S. utility and has initiated a new study to evaluate a specific set of HRA methods used in regulatory applications through a comparison of HRA predictions to crew performance in simulator experiments performed in a U.S. nuclear power plant. The results will be used to determine the potential limitations of data collected in non-U.S. simulators when used to evaluate U.S. applications and to improve the insights developed from the International HRA Empirical Study. The staff expects to complete the work in September 2013. In addition to these data collection activities, the staff is working to develop the capability to use an in-house research simulator to improve the human factors basis for HRA.

In regard to HRA data, RES signed an agreement with a U.S. utility in March 2011 to collaborate on the collection of human performance information on the utility's training programs for HRA. The information sources include the licensed operator simulator training, job performance measures, and emergency drills.

The staff developed a prototype event timeline tool to assist NRC inspectors in the conduct of event inspections. Additionally, the staff is coordinating with NRR and the Region II office to visit H.B. Robinson to collect the human performance information on its March 28, 2010 fire event. This visit is planned for the week of November 8, 2011.

8. Human Reliability Analysis Development for Fire PRA

Under a joint MOU, 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, NUREG/CR-6850/EPRI 1011989, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," issued September 2005 (ADAMS Accession Nos. ML052580075 and 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 supports the NFPA 805 transition initiative and the 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 selected plants was completed. In May 2009, feedback from both of these efforts was reviewed and addressed, resulting in a revised NUREG-1921/EPRI 1019196, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," in November 2009. The NRC internally reviewed the revised draft, and an overview was presented to the ACRS HRA subcommittee in June 2009. Following some additional revisions, the report was issued as a draft for public comment in December 2009. This work is one input to the work being done under SRM-M061020 and related research.

The public comment period for the draft report closed in March 2010. Comments were received from four reviewers. In addition, feedback was provided by the PWR Owners Group in a pilot application of the fire HRA guidelines. The joint EPRI/NRC-RES team is currently completing the final report. In addition, the joint team is presenting the fire HRA module for the second time (August 1-5, 2011 and November 14-18, 2011) in the Joint EPRI/NRC-RES Fire PRA training course. Publication of the final report is expected in 2011.

9. Analytical Tools for Risk Applications

SAPHIRE Version 8, initially released in April 2010, includes features and capabilities to address requirements for risk-informed programs. SAPHIRE Version 8 includes user interfaces developed for performing the following tasks:

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

The NRC staff continues to maintain and improve the SAPHIRE Version 8 software to support risk-informed programs. The software is repeatedly reviewed and tested to make it more efficient, reliable, and maintainable. Many of the software error fixes and modifications are developed in response to user requests, and user feedback will continue to be addressed. All SAPHIRE Version 8 maintenance activities, modifications and improvements are performed in accordance with the documented SAPHIRE software quality assurance practices. The NRC periodically audits the developers' implementation of these practices against the requirements in NUREG/BR-0167, "Software Quality Assurance Program and Guidelines," issued February 1993.

In FY 2011 new features and capabilities have been implemented in SAPHIRE Version 8 to better support NRC regulatory activities. SAPHIRE Version 8 has been modified to run on multi-core (multiple processors internal to a single computer) computers. The effective use of multi-core computers has decreased the overall analysis time needed to quantify SPAR model results. SAPHIRE Version 8 has also been modified to better support its use in SDP Phase 2 analyses and inspection planning activities. SAPHIRE Version 8 now includes user-friendly links to SPAR model documentation and produces new risk insights reports, which summarize plants' risk information for NRC resident inspectors. The staff has also provided training on SAPHIRE Version 8 to those resident inspectors who are participating in the piloting of a proposed new process for SDP Phase 2 analyses using SPAR models.

Companion documentation for the SAPHIRE Version 8 software has been published as NUREG/CR-7039, "System Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE Version 8)," Volumes 1-7, issued June 2011. The documentation includes an overview of SAPHIRE Version 8 features, a tutorial, a users' guide, and a technical reference. The completion of the SAPHIRE Version 8 documentation provides a valuable resource for users of the software.

10. SPAR Model Development Program

Standardized Plant Analysis Risk (SPAR) models are plant-specific PRA models that treat accident sequence progression, plant systems and components, and plant operator actions. The standardized models represent the as-built and as-operated plant. As such, they permit the staff to perform risk-informed regulatory activities by independently assessing the risk of events or degraded conditions at operating nuclear power plants. During fiscal year (FY) 2011, the staff accomplished the following tasks:

- In FY 2010, the staff, in cooperation with industry experts, completed a peer review of a representative boiling-water reactor (BWR) SPAR model and a pressurized-water reactor

(PWR) SPAR model in accordance with American National Standard, ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated April 1, 2002, and RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." In FY 2011 the staff reviewed the peer review comments and initiated projects to address these comments, where appropriate. This effort is planned to be completed in late calendar year 2013.

- The staff places a priority on creating methods and guidance for the risk assessment of shutdown events, with emphasis on SDP Phase 3 analyses. For this purpose, eight SPAR models that contain selected shutdown event scenarios, as well as internal event scenarios, have been developed. These models are supported by a handbook for the analysts, a model maker's guideline for the construction of other models and scenarios, an event tree template library, and a human error probability library. At this time the staff has no plans to make further operating reactor SPAR shutdown models. Currently available models, together with the supporting documents, can be used to support SDP Phase 3 evaluations of shutdown events and degraded conditions for other plants by generating further models from the existing templates.

- The staff has performed MELCOR analyses to investigate success criteria associated with specific Level 1 PRA sequences. In some cases, these analyses confirm the existing technical basis. In other cases, they support modifications that have been made to increase the realism of the agency's SPAR models. To date, calculations have been performed for a number of sequences for the Peach Bottom and Surry plants utilizing the MELCOR models developed for these two plants under the State-of-the-Art Reactor Consequence Analyses project. These results have been incorporated in the technical bases supporting the Surry and Peach Bottom SPAR models and have been extended to include an additional 19 BWR SPAR models and 8 PWR SPAR models. Ongoing work in this area includes the development of additional MELCOR input models and the investigation of Level-1 PRA end-state characterization (e.g., realism of core damage surrogates).

- In FY 2010 the staff completed an evaluation of the potential core damage risk reduction associated with the extensive damage mitigation strategies and guidance required by 10 CFR 50.54(hh) for approximately two-thirds of the 78 SPAR models. An evaluation of the remaining SPAR models was completed in the first quarter of FY 2011.

- During FY 2011 the staff developed two design-specific internal events SPAR models for the Advanced Boiling Water Reactor (ABWR), one for the ABWR/Toshiba reactor design and one for the ABWR/General Electric (GE) design. As part of the SPAR model development, the requisite supporting documentation was also completed. The first draft of the ABWR/Toshiba model has been provided to the Office of New Reactors (NRO) for review. The ABWR/GE SPAR model has been completed and will be transitioned to a routine maintenance status. A preliminary design-specific internal events SPAR model for the U.S. Advanced Pressurized-Water Reactor (U.S. APWR) has also been provided to NRO for review and comment. Although the AP1000 model was completed in February 2010, a modification was made to the SPAR model to include a seismic model. This modification has been completed and submitted to NRO for review. The staff plans to continue developing new reactor SPAR models as necessary to support NRO needs. Because design standardization is a key aspect of the new plants, it should only be necessary to develop one internal events SPAR model for each of the new designs.

- The staff has executed an addendum to the MOU with EPRI to conduct cooperative nuclear safety research for PRA. Several of the initiatives included in the addendum are intended to help resolve technical issues that account for the key differences between NRC SPAR models and licensee PRA models. In support of this effort, the memorandum of understanding addendum on PRA with EPRI has been extended through 2016. The staff also continues to work with the National Aeronautical and Space Administration (NASA) to address PRA issues of mutual interest. In addition, the NRC has utilized the cooperative agreement and grant program to establish collaborative PRA research projects with the University of Maryland and the Massachusetts Institute of Technology.

- In accordance with existing user need requests, the staff will continue to implement enhancements to the SPAR models for full-power operations. Anticipated enhancements include incorporating new models for support-system initiators and revised success criteria based on insights from thermal-hydraulic analyses.

- Based on a user need from NRR, further work is in progress to add additional capability to the existing SPAR models. One significant ongoing activity is the incorporation of internal fire scenarios from NFPA 805 pilot applications into the SPAR models.

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

11. Risk-Related Generic Issues

The Generic Issues Program (GIP) is an agency-wide program to address difficult safety or security issues that do not clearly fit elsewhere at the NRC. Several active generic issues involve risk.

- GI-191, Assessment of Debris Accumulation on PWR Sump Performance: This generic issue concerns the possibility that, following a loss of coolant accident in a PWR, debris accumulating on the emergency core cooling system sump screen may result in clogging and restrict water flow to the pumps. As a result of this generic issue and the related generic letter (GL 2004-02), all PWR licensees increased the size of their containment sump strainers, significantly reducing the risk of strainer clogging. An associated issue, which needs to be resolved to close GI-191, regards the potential for debris to bypass the sump strainers and enter the reactor core. In 2008, the NRC staff determined that additional industry-sponsored testing was necessary to support resolution of this issue. Some testing was performed, but testing and NRC evaluation are continuing because of NRC staff concerns about the testing results and related assumptions. In SRM-SECY-10-0113, “Closure Options for Generic Safety Issue – 191, Assessment of Debris Accumulation of Pressurized Water Reactor Sump Performance,” dated December 23, 2010, the Commission determined that it was prudent to allow the nuclear industry to complete testing on in-vessel effects and zone of influence in 2011, and to develop a path forward by mid

2012. The SRM directed the staff to evaluate alternative approaches, including risk-informed approaches, for resolving GSI-191 and to present them to the Commission by mid 2012. The Commission further agreed that modifications should be completed within two operating cycles for smaller LOCAs and three operating cycles for larger LOCAs after development of the path forward. NRC staff will determine a closure date for this Generic Issue after meeting with the Commission in mid-2012.

- GI-199, Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: In support of early site permits (ESPs) and combined license applications (COLs) for new reactors, the NRC staff reviewed updates to the seismic source and ground motion models provided by applicants. These seismic updates included new Electric Power Research Institute models to estimate earthquake ground motion and updated models for earthquake sources in the Central and Eastern US (CEUS). These reviews identified higher seismic hazard estimates than previously assumed at some sites. As a result, the staff concluded on May 26, 2005 (ADAMS Accession No. ML051450456), that the issue of increased seismic hazard estimates in the CEUS be examined under the GIP as GI-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" (ADAMS Accession No. ML051600272). After the initial screening analysis for GI-199 suggested that estimates of the seismic hazard for some currently operating plants in the CEUS has increased, the issue proceeded to the safety/risk assessment stage of the GIP. Subsequently, during the safety/risk assessment stage of the GIP, the NRC staff reviewed and evaluated the new information received with the ESP/COL submittals, along with 2008 U.S. Geological Survey seismic hazard estimates and recent geological research literature. The staff compared the new seismic hazard data with the earlier evaluations conducted as part of the IPEEE program. From this evaluation, the staff concluded that the likelihood of exceeding the seismic hazard used in the IPEEE program could be higher than previously understood for some currently operating CEUS sites. The NRC staff completed the safety/risk assessment stage of GI-199 on September 2, 2010 (ADAMS Accession No. ML100270582), concluding that GI-199 should transition to the regulatory assessment stage of the GIP. Information Notice 2010-018, "Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,' " dated September 2, 2010 (ADAMS Accession No. ML101970221) summarizes the results of the GI-199 safety/risk assessment. The issue was transmitted from RES to NRR for action and the NRR staff developed a draft generic letter that has been issued for public comment, which concludes in November, 2011. After addressing public comments, the NRC staff plans to issue the GL in early 2012.

- GI-204, Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure: In July 2010, NRR submitted a proposed issue to the GIP on flooding impacts at operating reactors due to potential dam failures. The submittal was motivated by recent findings under the Reactor Oversight Process by NRR and Regional staff with respect to flooding protection which could be challenged by potential upstream dam failure. The screening assessment of the generic issue concluded that further evaluation of external flooding of nuclear power plants due to an upstream dam failure is warranted, which will require a risk-informed evaluation of the impact of potential flooding scenarios, such as the likelihood of potential dam failures, flooding analysis, and consequential impacts at nuclear power plants. No immediate safety concerns were identified during the conduct of the screening assessment. In addition to the submittal to the GI Program, NRR is planning to release (in coordination with GI-204) information on these items as two Information Notices on (1) dam failure frequencies, and (2) impacts on severe flood considerations resulting from the U.S. Army Corps of Engineers' 2004 study.

12. Use of Risk Insights to Enhance Safety Focus of Small Modular Reactor Reviews

SRM-SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated May 11, 2011, directs the staff to provide the Commission with a paper that explores the feasibility (e.g., regulatory infrastructure changes, resource requirements, and timing for implementation) of including risk information in categorizing SSCs as safety-related and non safety-related for the design-specific small modular reactor (SMR) review plans in both the short and long terms. SRM-SECY-11-0024 directs the staff to consider stakeholder input, as appropriate and to consult with the Office of the General Counsel on the Commission paper to determine whether legal obstacles to this approach would require a rule change. Consequently, a Feasibility Study Team (FST) has been established to respond to the SRM.

As requested as part of this exploration, the FST has included a review of previous Commission policies on the spectrum of new/advanced reactor policy issues that may have used "safety-related" or "non-safety related" SSC classification as part of the policy resolution.

The FST also explored the potential application of risk insights to the overall regulatory framework rather than limit it to SMRs. The FST is preparing a paper that explores the feasibility of including risk information in categorizing SSCs, which will be provided to the Commission.

13. Revised Fuel Cycle Oversight Process

The staff submitted SECY 10-0031, "Revising the Fuel Cycle Oversight Process," dated March 19, 2010, to the Commission for its consideration and approval of the plan to revise the fuel cycle oversight process. The Commission was briefed on SECY-10-0031 on April 29, 2010. Following the April 29 briefing, the staff received SRM-M100429, dated May 12, 2010 and SRM-SECY-10-0031, dated August 4, 2010. In response to these memoranda, the staff developed and discussed with the ACRS on December 15, 2010, a paper comparing integrated safety analyses (ISAs) for fuel cycle facilities and probabilistic risk assessments (PRAs) for reactors. ACRS provided a letter to the Commission with comments on this paper and recommendations on the fuel cycle oversight program. SRM-SECY-10-0031 also directs the staff to work on specific elements of the oversight program. The staff submitted SECY-11-0140, "Enhancements to the Fuel Cycle Oversight Process," dated October 7, 2011, to the Commission to address these tasks and to provide the Commission with recommendations for next steps to enhance the fuel cycle oversight process.

14. Part 61: Site-Specific Analyses Rulemaking

Depleted uranium is considered source material, in accordance with 10 CFR Part 40, "Domestic Licensing of Source Material," and if depleted uranium were treated as a waste, it would fall under the definition of low-level radioactive waste in accordance with 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, “Response to Commission Order CLI-05-20 regarding Depleted Uranium,” dated October 7, 2008, whether quantities of depleted uranium at issue in the waste stream from commercial uranium enrichment facilities warrant the amendment of 10 CFR 61.55(a)(6) or 10 CFR 61.55(a) waste classification tables.

The Commission directed the staff in (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, the 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.

Subsequently, in SRM-SECY-10-0043, “Blending of Low-Level Radioactive Waste,” dated October 13, 2010, the Commission directed the staff to include blended waste in the limited scope rulemaking for depleted uranium. The Commission also indicated that the “staff should also consider additional opportunities for stakeholder involvement and education in development of the rule, such as additional public meetings or extension of the public comment period on the rule.”

The staff began work on the proposed rule in October 2010. On May 3, 2011, the NRC published draft proposed rule text and supporting technical bases in the Federal Register for public comment. On May 18, 2011, staff held a public meeting on the draft proposed rule text and technical bases. The staff briefed the ACRS full committee and a subcommittee on the proposed rule and draft guidance four times between June and September, 2011. The staff will consider comments that it receives from ACRS and the public as it finalizes the proposed rule before submitting it to the Commission.

15. Waste Confidence Rule and Extended Storage and Transportation of Spent Nuclear Fuel

In SECY-11-0029, “Plan for the Long-Term Update to the Waste Confidence Rule and Integration with the Extended Storage and Transportation Initiative,” dated February 28, 2011, the staff provided the Commission with a plan to update the waste confidence decision and rule and to enhance the technical and regulatory basis of the existing regulatory framework for the regulation of spent nuclear fuel for extended periods. This plan incorporates work initiated under SRM-COMSECY-10-0007, “Project Plan for Regulatory Program Review to Support Extended Storage and Transportation of Spent Nuclear Fuel,” dated December 6, 2010, that directs the staff (1) to continue efforts to enhance the process for licensing and inspection of spent fuel storage, (2) to continue current research activities that support long term storage, and (3) to complete the extended storage and transportation gap assessments identified as Phase 1 of the project. The Office of Nuclear Material Safety and Safeguards and RES are coordinating the gap assessment and technical research. These efforts will include the use of risk information and performance-based approaches in the regulatory bases for extended storage and transportation through the technical gap assessment, directed research on significant technical issues, and incorporation of this approach in future revisions to guidance and possible changes in regulations. The Gap Assessment Report will be finalized in FY 2012, along with a plan for completing the technical research in the following years.

16. Regulatory Basis to Support Rulemaking for Potential Reprocessing Facilities

In SRM-SECY-07-0081, "Regulatory Options for Licensing Facilities Associated with the Global Nuclear Energy Partnership," dated June 27, 2007, the Commission directed the NRC staff to proceed with a regulatory gap analysis and to identify changes in the regulatory requirements necessary to license a potential reprocessing facility. As part of a regulatory gap analysis, the staff identified the need to develop quantitative risk insights for the variety of chemical-radiological operations that might occur at potential spent nuclear fuel reprocessing facilities. Staff from RES and the Office of Nuclear Material Safety and Safeguards are collaborating to develop analytical tools that can account for potential hazards at reprocessing facilities and provide quantitative insights on the radiological risks associated with fission product and actinide separations. The staff plans to apply these risk insights to develop appropriate risk guidelines for reprocessing facilities.

17. Risk-Informed Security

As part of a user need request, RES worked with the Office of Nuclear Security and Incident Response (NSIR) to identify ways risk can be used to better inform NRC's approach to security regulations, licensing actions, and inspection activities. In response to this user need, RES and NSIR held a workshop exploring the potential use of risk-informed approaches for regulating security at nuclear power plants. The workshop was held September 14 and 15, 2010 at Sandia National Laboratories (SNL) in Albuquerque, New Mexico. SNL delivered a report summarizing the conclusions of the workshop to the NRC in December, 2010 and it was provided to the Commission. NSIR is continuing to have discussions with RES on how to better risk inform security. Any new major initiatives are awaiting the outcome of the agency-wide task force recommendations on risk-informing the NRC regulatory process.

18. Risk-Informed Emergency Action Levels

The NRC staff is exploring the feasibility of creating a technical approach to risk inform emergency response protective actions taken related to emergency action levels (EALs). The effort involves the use of a PRA approach to quantify risk for selected EAL scenarios by computing their conditional core damage probabilities (CCDP) using plant SPAR models. The CCDP results will serve as a means to compare and evaluate each EAL within a given emergency classification level (ECL) to determine if the current EALs are generally risk consistent. The initial analysis, based on Peach Bottom and Surry as representative BWR and PWR plants, has been completed. This work is being documented in a draft NUREG/CR, "Risk Informing Emergency Preparedness Oversight: Evaluation of Emergency Action Levels Using SAPHIRE, Volume 1-A Pilot Study Using Peach Bottom and Surry SPAR Level 1 Model." The final version of the NUREG/CR is expected to be published in spring 2012.