Recent Accomplishments and Near-Term Anticipated Accomplishments

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.

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 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, which 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 has chosen two pilot sites (the Oconee and Shearon Harris nuclear power plants) and developed a frequently asked question (FAQ) process for resolving implementation issues. To date, 42 operating nuclear power units have submitted letters of intent to adopt 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 participated in an industry fire protection information forum and conducted a regional inspector workshop, two pilot plant observation visits, six public FAQ meetings with the NEI-805 task force, and a review of the fire probabilistic risk assessment (FPRA) for the Shearon Harris plant. In addition, the staff continues to evaluate NEI's request for additional enforcement discretion. The request is based upon the results of the pilot plants, which would propose to extend the existing discretion for plants that are transitioning to NFPA 805.

Over the next 6 months, the staff expects to conduct another workshop for the regional inspectors, an additional NFPA 805 pilot plant observation visit, six more public FAQ meetings, and a review of the Oconee FPRA. The staff also plans to begin the review of the pilot plant license amendment requests to transition to the NFPA 805 licensing basis and to reach a resolution regarding the NEI request for additional enforcement discretion.

2. 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, is addressing issues related to the risk assessment of digital systems. In this effort, the TWG is placing particular emphasis on risk-informing digital system reviews for operating plants and new reactors. The TWG efforts will be consistent with the NRC's Policy Statement on Probabilistic Risk Assessment (PRA), 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 the first Digital System Project Plan on July 12, 2007 and an updated project plan was issued in February 2008. The TWG has held eight 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 systems for design certification and combined operating license (COL) application PRAs. The staff discussed the draft ISG with stakeholders in a public meeting held in February 2008. Based upon further discussions with the ACRS, final issuance of the ISG is expected to occur in the early summer of 2008.

In addition, over the next 6 months, the TWG plans to continue work to finalize potential review areas where risk insights from PRA modeling of digital systems may be applied to staff reviews for operating plants and new reactors. The staff expects to receive an industry topical report to use with a pilot plant application. The staff plans to publish two NUREG-series reports: one on approaches for using traditional PRA methods for digital systems and another on the benchmark implementation of two dynamic methodologies for reliability modeling of digital systems. These two reports are part of the agency's overall effort to advance the state of the art in digital systems risk and reliability modeling to the point where it will be possible to risk-inform licensing reviews for digital systems and incorporate related models into nuclear power plant PRAs.

3. 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 stabilize and enhance PRA quality, the staff is also following up on a Commission staff requirements memorandum (SRM), "Staff Requirements—Meeting with Advisory Committee on Reactor Safeguards, 2:30 p.m., Friday, October 20, 2006, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (open to public attendance)" (M061020), dated November 8, 2006, to evaluate different HRA models in an effort either to identify a single one as acceptable for use or to provide guidance as to when each should be used. 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 completed and documented in the draft NUREG/IA-0215/HWR-844. The staff expects that the pilot study will be completed by December 2008 and a final NUREG/IA will be submitted for publication in December 2009. The staff has also established a memorandum of understanding with the Electric Power Research Institute (EPRI) to work together to identify areas where HRA has a significant impact on regulatory decisionmaking. The staff will present its findings to the Advisory Committee on Reactor Safeguards (ACRS) for their review.

4. 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, with approval anticipated in fiscal year (FY) 2008, while revisions to the B&W STS are expected to be issued in spring 2008.

- 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 staff expects the submittal for the second pilot plant, Fort Calhoun Station, in FY 2008.

- 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. In addition, the staff is currently reviewing the associated Technical Specification Task Force guidance (TSTF-425) to revise the STS, which the staff expects to approve and make available via the consolidated line item improvement process in FY 2008.

- Initiative 6, "Modification of Limiting Condition for Operation (LCO) 3.0.3, 'Actions and Completion Times,'" revises the surveillance requirement LCO by requiring that risk be considered in determining the correct course of action. A revised CE topical report was submitted for staff review in December 2007. That topical report supports a future revision of the CE STS to incorporate this initiative. In addition, the staff expects a topical report for Westinghouse plants to be submitted in FY 2008.

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

The staff prepared a proposed rule containing emergency core cooling system evaluation requirements, which 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 (LOCA) requirements to provide a risk-informed alternative maximum break size. In October 2006, the staff produced a draft final rule and briefed the 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 May 16, 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 the rulemaking should be assigned a medium priority. Nonetheless, the SRM also directed that the staff continue to make progress on the 10 CFR 50.46 rulemaking and apply resources to the effort in FY 2008. In March 2008, the staff provided the Executive Director for Operations its proposed schedule for completing the final rule.

- In a meeting with the ACRS on November 27, 2007, the staff discussed draft NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” and draft NUREG-1903, “Seismic Considerations for the Transition Break Size.” The ACRS recommended the publication of these NUREGs. The agency published NUREG-1903 in February 2008 and NUREG-1829 in March 2008. These documents provide part of the technical basis for the selection of the risk-informed maximum break size under 10 CFR 50.46a rulemaking.

- 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 will contain a new paragraph 10 CFR 50.61a which will provide new requirements that a pressurized-water reactor licensee could voluntarily use as an alternative to complying with the existing requirements. The NRC received over 40 comments during the public comment period, which 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. Some proposed changes, if accepted, may require re-noticing the proposed rule. The staff is currently developing responses to the comments and revising the schedule for the rulemaking.

- By letter dated January 26, 2006, the Westinghouse Owners Group, later the Pressurized-Water Reactor Owners Group (PWROG), submitted topical report WCAP-16168-NP, Revision 1, “Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval.” The PWROG topical report provides the technical and regulatory basis for decreasing the frequency of inspections by extending the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI inservice inspection interval for reactor vessel welds from the current 10 years to 20 years for ASME Code Section XI, Category B-A and B-D, reactor vessel welds. The staff completed its review of the topical report and issued its draft safety evaluation on March 6, 2008 to solicit comments on factual errors and to clarify concerns raised in the safety evaluation.

6. Byproduct Materials Rulemaking

In SECY-07-0113, “Final Rule: 10 CFR Parts 30, 31, 32, and 150—Exemptions from Licensing, General Licenses, and Distribution of Byproduct Material: Licensing and Reporting Requirements (RIN 3150-AH41),” dated July 6, 2007, the Commission approved the final rule. This rulemaking was risk-informed, in part, by the use of NUREG-1717, “Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials,” issued June 2001. The Commission published the final rule in the *Federal Register* on October 16, 2007 (72 FR 58473). The effective date of the rule was December 17, 2007. This initiative has been implemented and will not be included in future RPP updates.

7. 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 system's response to initiating events and to quantify associated consequential outcome frequencies. Over the past 6 months, the staff released SAPHIRE Version 7.27 which includes the capability to perform uncertainty analysis on importance measures and other improvements. Over the next 6 months, completion of the first beta version of SAPHIRE Version 8 is expected. SAPHIRE Version 8 features and capabilities address new code requirements in support of risk-informed programs, including the development of a user interface for SDP Phase 2 assessments. Also, the common-cause failure module contained in Version 7, originally scheduled for last year, is expected to be completed during this time frame.

8. Industry Trends Program Support

The Industry Trends Program Support program uses data collected from Licensee Event Reports, the Institute of Nuclear Power Operations' Equipment Performance and Information Exchange (EPIX) System and Monthly Operating Reports to develop current estimates of industry and plant-specific system and component reliabilities, initiating event frequencies, common-cause failure parameters, and fire event frequencies. The staff updated the NRC's public Internet site with the 2006 estimates, trends, charts, graphs and summary tables for the industry-wide data. This year the staff made the transition to using component information and data from EPIX in plant-specific fault tree models. Formerly, the NRC used data from LERs in generic fault trees. The new data source and models are expected to provide more meaningful results in two ways. First, by using plant-specific models, we are able to make plant-specific system performance estimates. Also, our "industry-wide" estimates will now include plant-to-plant (design) variation. Second, since EPIX failure records provide a large population of data, component availability and reliability can be estimated with a higher degree of assurance of accuracy than in the past and confidence in the statistics of analysis will be proportionately higher.

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. In the coming months, the staff plans to issue the following NUREG series reports to provide guidance for RASP:

- "Estimating Pipe Break Loss-of-Coolant Accident Frequencies Using NUREG-1829 Information,"
- "Common-Cause Failure Analysis in Event Assessment," and
- "Data Guidance for the Risk Assessment Standardization Project."

9. Reactor Performance Data Collection Program

The staff has been collecting data and information for twenty years to support studies and risk analyses of nuclear power plant operational experience. The information comes from Licensee Event Reports (LERs), the Institute of Nuclear Power Operations' Equipment Performance and Information System (and its processor database, the Nuclear Plant Reliability Data System), and utilities' Monthly Operating Reports. In FY 2001, these data collection activities were consolidated in a single system, the Integrated Data Collection and Coding (IDCCS). Over the past 6 months, the staff updated the IDCCS with FY 2007 data and incorporated the latest LERs into the LERSearch database so that it now reflects LERs from 1981 through January 2008. The staff plans to further enhance that database in FY 2008 to provide additional search options and provide more risk-related operational data.

10. 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 nuclear power plants. Over the past 6 months, the staff accomplished the following:

- Completed initial cutset-level review of all SPAR models (except four models held back because of licensee delays in updating their models).

- Initiated revised Level 2/large early release frequency (LERF) model development for selected plant types.

- Commenced cooperative research activities under the Office of Nuclear Regulatory Research/EPRI memorandum of understanding addendum to address resolution of key technical issues with the industry.

- Completed data update of 39 SPAR models based on NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” issued February 2007.

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

- Completed proof-of-concept to extend Level 1 SPAR models to incorporate containment systems in support of Level 2/LERF modeling.

- Completed and made publicly available the RASP Handbook Volume 1, Part 1 “Internal Events Analysis”; Volume 2, “External Events Analysis”; and Volume 3, “SPAR Model Reviews” (checklists).

- Made operational the RASP Tool Box Web page which provides Web links to tools and access to references for senior reactor analysts and others.

In FY 2008, the staff plans to continue implementing enhancements to the Revision 3 SPAR models and complete additional external events and LP/SD models to support the Accident Sequence Precursor (ASP) Program and the Significance Determination Process. In addition, the staff plans to extend the Level 1 SPAR models to incorporate containment systems for additional plant classes and complete data update of all SPAR models.

11. Phased Approach to Probabilistic Risk Assessment Quality

The increased use of PRAs in the 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. ASME and ANS recently published a joint PRA standard, "Level 1 and Large Early Release Frequency (LERF) PRA Standard" (ASME/ANS RA-S-2008), which applies to at-power internal events, internal fire events, and external events for operating reactors.

The staff has initiated work on Revision 2 of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." This revision will endorse the joint ASME/ANS Level 1/LERF PRA standard. The staff recently held public meetings on this topic and plans to issue Revision 2 for public comment in mid-2008, and final regulatory guide In December 2008. ASME and ANS continue to work on other PRA standards:

- Level 1/LERF standard for internal events at low power and shutdown conditions for operating reactors
- Level 1 and Large Release Frequency for at-power internal and external events for new reactors
- Level 2 and Level 3 for at-power internal events for operating reactors
- Level 1,2 and 3 for internal and external events for all operating modes for advanced non-light water reactors

The staff is supporting these efforts and will consider endorsing these standards, once issued, in a future revision to Regulatory Guide 1.200.

The staff is also working with ASME (under a Cooperative Agreement) in development of training on the ASME/ANS PRA standard. This effort addresses Parts 1 and 2 of the standard which covers the technical requirements for a Level 1/LERF PRA for at-power internal events. This training will be a combination of web-based and class-room training. The web-based training will be available in mid-to-late 2008.

In November 2007, the staff issued draft NUREG-1855, "Treatment of Uncertainties from PRAs in Risk-Informed Decision-Making," for public review and comment. It is being developed in collaboration with the Electric Power Research Institute (EPRI) who has issued a draft report on uncertainties, as part of the NRC/EPRI Memorandum of Understanding. These two documents are meant to be complimentary. The NRC report along with the EPRI report provides information and guidance on uncertainties associated with PRA. They are meant to 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 decision making for risk-informed activities. The staff held two public meetings and plans to issue a final version in late 2008.

This regulatory guide and NUREG report (including the EPRI report) will assist the staff in establishing the technical acceptability of the PRA results to be used in regulatory decision making. When used in support of an application, these documents will obviate the need for an in-depth review of the base PRA by NRC reviewers, and provide for a more focused and consistent review process.

12. New Reactor License Application Reviews

In February 2008, the staff issued interim staff guidance (ISG) for public comment on PRA information necessary to support design certification and combined license applications. This ISG was the result of public meetings held during 2007 and is intended to clarify new PRA regulatory changes and the associated regulatory guidance. Public comments are currently being reviewed and the staff plans to hold an additional public meeting prior to issuing the final ISG in late spring or early summer 2008.

Over the past six months, the staff has also issued risk insights reports for the advanced boiling-water reactor, AP1000, the U.S. Evolutionary Power Reactor, the Economic Simplified Boiling-Water Reactor, and the U.S. Advanced Pressurized Water Reactor designs. These reports will assist the staff in performing risk-informed reviews of design certifications and COL applications for these designs. By applying risk insights, the staff will be able to focus their attention and resources upon aspects of the design that contribute most to safety.

13. 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 establishes the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future nuclear power plants. It documents a framework that provides an approach, scope, and criteria that could be used to develop an alternative set of requirements to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” for future nuclear power plants. The staff will, as resources permit, continue its pre-application discussions supporting a proposed risk-informed licensing approach for the pebble-bed modular reactor (PBMR). The licensing strategy for the next generation nuclear plant (NGNP) that is currently being prepared in cooperation with the Department of Energy will also include recommendations for the use of risk insights in the licensing of the NGNP prototype facility. These experiences will inform the staff's recommendation to the Commission in 2009 on the possible use of risk-informed and performance-based approaches within rulemakings to support the licensing of Generation IV reactors.

14. Staff Training on Risk-Informed Regulation

The staff has significantly revised and updated the course content for internal training courses P-101 and P-107, “Risk-Informed Regulation for Technical Staff” and “Risk-Informed Regulation for Technical Managers,” respectively. Pilot classes have been conducted and the courses are now available for all technical staff members. In addition, an abbreviated 90-minute version of the training is provided as part of the qualification program on reactor regulation awareness for technical reviewers in the office of Nuclear Reactor Regulation (NRR). These courses are now regularly scheduled and conducted. This initiative has been implemented and will not be included in future RPP updates.