Part 2: Risk-Informed Regulation Implementation Activities

Chapter 1. Reactor Safety Arena

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1. INTRODUCTION

The NRC has generally regulated nuclear reactors based on deterministic approaches. Deterministic approaches to regulation consider a set of challenges to safety and determine how those challenges should be mitigated. As discussed in Part 1 and in the Commission's PRA Policy Statement, a probabilistic approach to regulation enhances and extends this traditional, deterministic approach by (1) allowing consideration of a broader set of potential challenges to safety, (2) providing a logical means for prioritizing these challenges based on risk significance, and (3) allowing consideration of a broader set of resources to defend against these challenges.

Until the accident at Three Mile Island (TMI) in 1979, the Atomic Energy Commission (now the NRC) only used probabilistic criteria in certain specialized areas of licensing reviews. For example, human-made hazards (e.g., nearby hazardous materials and aircraft) and natural hazards (e.g., tornadoes, floods, and earthquakes) were typically addressed in terms of probabilistic arguments and initiating frequencies to assess site suitability. The Standard Review Plan (NUREG-0800) for licensing reactors and some of the Regulatory Guides supporting NUREG-0800 provided review and evaluation guidance with respect to these probabilistic considerations.

The TMI accident substantially changed the character of the analysis of severe accidents worldwide. It led to a substantial research program on severe accident phenomenology. In addition, both major investigations of the accident (the Kemeny and Rogovin studies) recommended that PRA techniques be used more widely to augment the traditional nonprobabilistic methods of analyzing nuclear plant safety. In 1984, the NRC completed a study (NUREG-1050) that addressed the state-of-the-art in risk analysis techniques.

In early 1991, the NRC published NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." In NUREG-1150, the NRC used improved PRA techniques to assess the risk associated with five nuclear power plants. This study was a significant turning point in the use of risk-based concepts in the regulatory process and enabled the Commission to greatly improve its methods for assessing containment performance after core damage and accident progression. The methods developed for and results from these studies provided a valuable foundation in quantitative risk techniques.

For the last several years, NRC's work to expand the use of PRA in regulatory processes has been documented in the PRA Implementation Plan (See SECY-99-211). Many of the early actions focused upon the development of skills, tools, and infrastructure for the application of risk information.

To understand the role to be played by the activities discussed in this risk-informed regulation implementation plan, it is necessary to recognize the significant accomplishments already completed. The staff's future actions build upon these activities and would not be possible without them. Thus, this chapter includes some information about recent activities, with the level of detail commensurate with their relationship to ongoing activities. For example, certain activities are currently being implemented, and thus the only action described in the plan is to monitor results and to seek out opportunities for improvement.

In considering what areas in the reactor safety arena to target for greater use of risk information, the NRC staff examined the sources of risk, the existing regulatory processes, and where there were the best opportunities for improvements. This led to a focus on reactors operating at power, but also gave consideration to (1) low power and shutdown conditions, (2) reactors undergoing decommissioning with fuel stored in pools (discussed under the nuclear waste arena), and (3) advanced reactor designs.

As discussed further in Part 1 and in the NRC's Strategic Plan (NUREG-1614), the NRC has developed goals, then planned strategies and activities to accomplish those goals. Within these strategies and activities, there are opportunities for greater use of risk information; however, much of the staff's activities in the reactor safety arena are not "risk-informed" as this is generally meant. For example, most licensing actions (i.e., many technical specification changes, license transfers, license renewals) do not require the use of risk information.

The evolution of the staff's application of risk information to the regulation of nuclear reactors is briefly discussed below. Detailed information on specific staff activities being undertaken to transition to risk-informed regulation, organized along the lines of the Commission's Strategic Plan, is provided in section 3 of this chapter.

One of the first examples are the Appendices in 10 CFR Part 52 certifying the evolutionary standardized reactor designs. Part 52 requires that a PRA be performed for any future design and also that the design meet certain technical requirements to prevent and mitigate severe accidents. A rulemaking in the planning stage would further require that operators of standard design plants maintain a "living" PRA.

SECY-97-171 (Consideration of Severe Accident Risk in NRC Regulatory Decisions) discussed how severe accident risk had been considered in the past as well as areas where it might be for the future. For instance, the NRC promulgated new rules requiring plants to deal with accidents that were beyond the normal design basis (station blackout and anticipated transients without scram) on the basis of risk information. The regulatory analysis guidelines by which NRC makes decisions about whether requirements are cost-beneficial backfits also consider risk of severe accidents. As discussed in Part 1, the development of the Safety Goal Policy was also a major step. Beginning in 1988, the staff also undertook a plan to consider severe accident risks for existing plants. This plan included several activities, including issuance of a Generic Letter (GL 88-20) asking licensees to conduct Individual Plant Examinations (IPEs) to look for plant-specific vulnerabilities to severe accidents. Other activities considered containment performance and utility severe accident management programs.

With the enhanced capabilities to assess risk, the staff also recognized that there were opportunities to reduce unnecessary regulatory burden. Stakeholder input was sought to identify areas that presented burden and in which risk information indicated that the burden may not be commensurate with the risks. Initial efforts focused on discrete areas to gain experience with use of the tools and guidance. As noted, the staff first developed the basic guiding principles (safety goal, PRA policy, and general guidance for licensing action decisions) and then proceeded with pilot applications. Over the last several years, the staff has reviewed individual licensing actions in such areas as graded quality assurance, in-service inspection, in-service testing, or changes to allowed outage times in the technical specifications. Having completed several pilots, the staff has concluded that greater use of risk information in the regulatory process could be accomplished in a manner that maintained safety, improved safety focus, and reduced unnecessary burden. Thus, the staff is now focusing upon other activities, such as rulemaking, to offer voluntary options for licensees.   These activities include both specific technical areas (e.g., fire protection) as well as broader changes such as the adjustment of special treatment requirements.

It should be noted that, where necessary, the staff has also added requirements as a result of risk information. For example, the maintenance rule (10 CFR 50.65) was recently modified to require licensees to assess and manage the increase in risk that may result from maintenance activities.

Risk information is being used to focus staff activities with respect to inspection and enforcement and to adjust specific requirements on licensees. This was one of the drivers for the revised reactor oversight process and is discussed in detail in section 3.

The staff has also been using risk information for several years for event assessment. One example is the accident sequence precursor program that determines conditional core damage probability for particular events.

Finally, the staff is continuing activities to enhance its capabilities to conduct or review risk analyses through various research programs. These include activities to improve tools, enhance data, and to identify areas where requirements can be adjusted in a risk-informed manner.

2. RECENTLY-COMPLETED RISK-INFORMED INITIATIVES

Maintenance Risk Assessments and Risk Management

In 1991, the NRC issued a new rule, 10 CFR 50.65, that required licensees to monitor the effectiveness of maintenance activities. The rule arose out of recognition that balance of plant equipment failures can initiate transients that are potentially significant events and therefore that risk could be reduced by limiting failures of such equipment. Further, risk will be reduced if mitigating systems have high reliability and availability. The scope of equipment included within the monitoring requirements of this rule was intentionally broad to include many SSCs that are not classified as safety-related. In monitoring SSCs against performance goals, the guidance allows variances in the level of monitoring (e.g., at plant level, system level, or train level) based upon the importance of the system to safety. Thus, two significance categories (high safety-significant and low safety-significant) were developed, along with ranking methodologies.

Part of the 1991 rule stated that licensees *should* perform an assessment of the total plant equipment that is out of service to determine the overall effect on performance of safety systems. NRC later determined that this provision should be strengthened. On June 18, 1999, the NRC issued a final rule that requires licensees to assess and manage risk that may result from maintenance activities. As part of this rule, the Commission also determined that the scope of the assessment may be limited to structures, systems and components that a risk-informed evaluation process has shown to be significant to public health and safety. The NRC has endorsed an industry guidance document, through a Regulatory Guide, for implementation of this revised rule. The effective date of the revised requirements is November 28, 2000.

Alternative Source Term

Extensive research has led to an enhanced understanding of the timing, magnitude, and chemical form of fission product releases following nuclear accidents. The results of this work are summarized in NUREG-1465 which was published in February 1995. Application of this new knowledge to operating reactors can result in cost savings and simplification of operations without adverse impacts on public health and safety. In order to allow licensee use of the new source term as it is applied to design basis accident calculations (and resulting plant features and operations), the NRC has issued a rule (10 CFR 50.67) and associated guidance. In addition, the staff conducted pilot reviews at several plants. The final rule was published on December 23, 1999, and the final regulatory guide was published on July 28, 2000.

Licensee Reporting Requirements

The NRC conducted a rulemaking to revise requirements for licensee reporting of events and conditions with several objectives, including elimination of reports of no risk significance. Thus, reports of conditions “outside the design basis of the plant” have been eliminated, and instead the reporting criteria for such conditions focus upon whether there is a loss of function. Other risk-informed aspects of this rulemaking include the revised requirements for reporting of system actuations, by inclusion of a list of risk-significant systems. Such data supports NRC needs for risk studies about reliability and availability of systems. The final rule was approved by the Commission on July 11, 2000, and the rule will be published soon.

3. CURRENT INITIATIVES

This section presents the risk-informed implementation activities underway in the reactor safety arena. As discussed in Part 1, this section is organized along the lines of the Strategic Plan. That is, the activities are listed under the performance goal and strategy that they support. The table below lists the performance goals and the strategies that are supported by an activity. Note that many activities support more that one strategy. In such a case, the activity is discussed under the strategy that it primarily supports; however, the other strategies that it supports are also listed.

Maintain Safety

Strategy 1: *We will sharpen our focus on safety to include a transition to a revised NRC reactor oversight program for our inspection, assessment, and enforcement activities.*

RS-MS1-1	Oversight framework
RS-MS1-2	Inspection program
RS-MS1-3	Performance indicators
RS-MS1-4	Assessment process
RS-MS1-5	Enforcement
RS-MS1-6	Process improvements

Strategy 3: *We will evaluate operating experience and the results of risk assessments for safety implications*

RS-MS3-1	Risk-based performance indicators
RS-MS3-2	Performance indicator analysis
RS-MS3-3	Plant reliability studies (Sequence Coding and Search System)
RS-MS3-4	Accident sequence precursor program
RS-MS3-5	System reliability studies
RS-MS3-6	Individual plant examinations for external events (IPEEE)

Strategy 5: *We will ensure that changes to operating licenses and exemptions to regulations maintain safety and meet regulatory requirements*

RS-MS5-1	Risk-informed licensing guidance (1) RG 1.174 (2) Application-specific guidance .1 Graded quality assurance .2 In-service inspection .3 In-service testing .4 Technical specifications (3) Guidance for non-risk-informed licensing applications (4) Guidance for reviewing risk-important human actions

Strategy 8: *We will continue to develop and incrementally use risk-informed and, where appropriate, less-prescriptive regulatory approaches to maintain safety*

RS-MS8-1	RIP50 (Option 2)
RS-MS8-2	RIP50 (Option 3)
RS-MS8-3	Standard technical specifications
RS-MS8-4	Fire protection
RS-MS8-5	Safeguards
RS-MS8-6	Reactor pressure vessel integrity

Efficiency, Effectiveness and Realism

Strategy 1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions*

RS-EE1-1	Advanced reactors
RS-EE1-2	Standards
RS-EE1-3	Methods
RS-EE1-4	Analytical tools
RS-EE1-5	International cooperation
RS-EE1-6	Regulatory effectiveness
RS-EE1-7	See standard technical specifications (MS8-3)

Reduce Unnecessary Regulatory Burden (UB)

Strategy 1: *We will utilize risk information and performance-based approaches to reduce unnecessary regulatory burden*

RS-UB1-1	See Option 2 (MS8-1)
RS-UB1-2	See Option 3 (MS8-2)
RS-UB1-3	See licensing guidance (MS5-1)
RS-UB1-4	See Standard TS (MS8-3)

Strategy 3: *We will improve our reactor oversight process by redirecting resources from those areas less important to safety*

RS-UB3-1 (to 3-6)	See oversight process (MS1-1)

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 1: *We will sharpen our focus on safety to include a transition to a revised NRC reactor oversight program for our inspection, assessment, and enforcement activities.*

Performance Goal: Reduce unnecessary regulatory burden on stakeholders

Strategy 3: *We will improve our reactor oversight program by redirecting resources from those areas less important to safety.*

The NRC decided to revamp its inspection, assessment, and enforcement processes in response to a number of considerations, including improvements in the performance of the nuclear industry, the desire by NRC to apply more objective, timely, safety-significant criteria in assessing performance, and for efficiencies in implementation. This involved a number of related individual activities.

Implementation Activity MS1-1 (UB3-1): Establishment of a framework for deciding on inspection, assessment and enforcement actions for nuclear power reactors that focuses on activities and systems that are risk-significant. **(NRR)**

Implementation Activity MS1-2 (UB3-2): Establishment of a revised baseline inspection program for all nuclear power plants with additional inspections that may be performed in response to a specific event or problem at a plant. **(NRR)**

Implementation Activity MS1-3 (UB3-3): Establishment of performance indicators, to be submitted by licensees and used to monitor performance **(NRR)**

Implementation Activity MS1-4 (UB3-4): Establishment of an assessment process for determining NRC actions based upon indicator and inspection information. **(NRR)**

Implementation Activity MS1-5 (UB3-5): Revisions to the enforcement policy to integrate with the overall assessment process. **(OE and NRR)**

Implementation Activity MS 1-6 (UB3-6): Providing technical support to enhancements to the risk-informed reactor oversight process. **(RES and NRR)**

The basic approach under the new oversight process is to monitor performance with respect to reactor safety cornerstones (initiating events, mitigation system performance, barrier integrity, and emergency preparedness), radiation safety (worker exposure and general public protection during routine operations), and security. Indicators that can be used to monitor performance against these cornerstones have been developed. NRC has also identified “inspectable areas” which relate to these cornerstones and for which performance indicators alone are not sufficient to monitor performance (NRC is also inspecting the performance indicator reporting process).

The risk-informed oversight effort was developed using the results of research work and previous risk studies to identify the most significant systems, structures and components (risk matrices) and to develop processes by which the risk significance of inspection findings could be determined (significance determination process). For instance, in judging the areas and the amount of inspection effort to apply, the risk significance of the activities or systems involved was considered. Further, risk information was used where possible in setting the thresholds for the performance indicators. When judging the importance of inspection findings, the significance determination process uses risk information to assess the significance of the issue. These assessments are then input to an assessment process (action matrix) to define the agency response, depending upon both the significance of individual findings as well as overall cornerstone performance. The notebooks used for the SDP will be updated as needed to support implementation of the program.

Performance is then assessed by categorizing the indicators and inspection findings using significance thresholds to decide upon agency response. Depending upon the results in the various cornerstone areas, NRC will continue its baseline inspection, will inspect licensee corrective actions to deal with problem areas, will undertake additional inspections to focus upon the cause of the degraded performance, or if performance is unacceptable, the plant will not be permitted to operate until the problems are corrected.

The enforcement program changes have being integrated with the assessment process. For violations of low significance, the matter will be discussed in the inspection report, with no formal enforcement action. The licensee will place the issue in its corrective action program. For more significant issues, a Notice of Violation may be issued, with the assessment process being used to establish the appropriate response. Certain types of violations will continue to use the previous enforcement policy with use of severity levels and possible civil penalties. (See May 1, 2000 FR notice for details of the revised enforcement policy.)

Performance indicator information, inspection findings, and the results of the NRC assessment process are made publicly available through the NRC web site. The uniform use of this process allows for enhanced communication with licensees and the public.

The revised process reflected in the above noted activities were developed with input from a wide range of stakeholders. The new processes were piloted with a subset of the reactors, and the new program was implemented nationwide in April 2000.

The NRC has convened a task group to assess inspector training and qualifications in light of the new reactor oversight program and other risk-informed initiatives. The task group consists of representatives from NRR, HR and the regions. The task group began meeting in July and August to plan its review.

The activities now underway include full implementation of the oversight program at all power reactors, a lessons-learned review, and study of possible enhancements particularly for certain cross-cutting issues, like corrective action.

Major Milestones

Generate analytic tool or method to support periodic inspection of licensee's corrective action program 12/2000

Complete report on assessment of the extent to which human performance is reflected in the reactor oversight process 12/2000

Report on lessons-learned from full implementation 6/2001

Maintain and update significant determination process (SDP) notebooks, rev.2 . . . . 9/2001
of the revised oversight program

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 3: *We will evaluate operating experience and the results of risk assessments for safety implications.*

Implementation Activity MS3-1: Develop and implement risk-based performance indicators (RBPIs) **(RES and NRR)**

The reactor oversight process uses performance indicators and findings from risk-informed inspections to assess plant performance relative to the "cornerstones of safety." Risk-based performance indicators (RBPIs) will assess performance in three cornerstones of safety -- initiating events, mitigating systems, and containment barrier integrity. The RBPIs will address quantitative measures of performance in areas whose influence on core damage frequency and on containment performance is explicit. RBPIs will reflect significant changes in these performance parameters for a broad set of systems and operational aspects associated with licensee performance under the cornerstones of safety. As discussed in SECY-99-007 and SECY-99-007A, the RBPIs will enhance the reactor oversight process as summarized below:

- Reliability indicators will be developed at the component/train/system level;
- Indicators for shutdown modes and fire events will be developed consistent with the current state-of-the-art models, data, and methods for these areas;
- RBPI threshold values will be more plant-specific to reflect risk-significant differences in plant designs;
- An indicator will be developed to consistently assess the integrated risk significance of performance indicators and inspection findings on overall plant performance and will provide an additional input to the Action Matrix;
- Trending of risk-significant performance at the industry-wide level, including insights and identification of key contributors to any observed trends, will be provided where there are insufficient data or models to develop a plant-specific indicator.

Major Milestones

Issue Phase 1 RBPI development progress report for external stakeholder comment 1/2001

Brief ACRS on Phase 1 RBPI development progress 5/2001

Brief Commission on Phase 1 RBPI development progress 8/2001

Brief Commission on Phase 2 RBPI development progress 4/2003

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 3: *We will evaluate operating experience and the results of risk assessments for safety implications.*

Implementation Activity MS3-2: Maintain and disseminate performance indicator information. **(RES)**

The staff will continue to evaluate operational data to identify adverse safety trends and individual plants with marginal performance. This evaluation provides a scrutable process for identifying and predicting changes in safety performance and for conducting effective plant safety assessments.

Major Milestones

Issue FY2000 performance indicators in support of performance plan 1/2001

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 3: *We will evaluate operating experience and the results of risk assessments for safety implications.*

Implementation Activity MS3-3: Assess nuclear power plant equipment reliability and availability. **(RES)**

The NRC operates and maintains the Sequence Coding and Search System (SCSS) which contains information about events at nuclear power plants in a computer-searchable framework. This effort entails (1) coding an estimated 1100 licensee event reports (LERs) into SCSS each year; (2) maintaining the availability to the NRC staff and public of more than 45,000 LERs that are searchable from more than 150 fields on a user-friendly internet web site, which permits the NRC staff to perform more than 5,000 searches of this resource each year; (3) responding to approximately 200 requests for special searches, source documents, consultation, and inquiries involving SCSS data from the NRC; and (4) responding to approximately 1 high-priority, quick-turnaround technical assistance task involving SCSS data each year. The SCSS provides important access to information about events at nuclear power plants which is used to support agency evaluations and decisions.

The NRC also operates and maintains the Common-Cause Failure (CCF) database. Common-cause failures have the potential to adversely impact the safety of nuclear power plants and are significant contributors to plant risk. It is important, therefore, that operating events at power reactors be reviewed to disclose risk-significant interactions, phenomena, and behavior in the design and operating of power reactors that were not previously recognized or analyzed. Operation and maintenance of the CCF database involves (1) updating the CCF component data and parameter estimations for approximately 40 safety-system components using licensee event report (LER) data and nuclear power industry's Equipment Performance Information Exchange (EPIX) data; (2) issuing reports on CCF insights; and (3) supporting the international exchange of common-cause failure data.

These operational data will provide information on relevant operating experience that will be used to enhance plant inspections of risk-important systems. In addition, the data will be used by the NRC staff to perform technical reviews of proposed license amendments, including risk-informed applications. The data will also be used in the development of risk-based performance indicators to support improvements in the reactor oversight process.

Major Milestones

Continue to operate and maintain the Sequence Coding and Search System (SCSS) during FY2001 . . . . 9/2001

Continue to operate and maintain the Common-Cause Failure (CCF) database during FY2001 . . . . 9/2001

Continue to operate and update the agency's Reliability and Availability Database System (RADS) with new EPIX data as it is received from the Institute for Nuclear Power Operations (INPO) 9/2001

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 3: *We will evaluate operating experience and the results of risk assessments for safety implications.*

Implementation Activity MS3-4: Produce accident sequence precursor analyses. **(RES)**

The Accident Sequence Precursor (ASP) program provides analyses of the risk significance of operating experience on a plant-specific basis. It also provides trending information on industry and group-specific risk performance. Thus, the ASP program provides a risk perspective of nuclear power plant events to inform agency response and regulatory decisions. The staff will continue to review LERs, inspection reports and other sources to identify events that should be analyzed and will continue to produce an annual report of ASP results.

The staff plans to develop an enhanced capability to use expert opinion, human performance analysis, and state-of-the-art engineering analysis (such as thermal-hydraulic calculations) to improve the realism and consistency of ASP analyses. The staff also plans to improve the efficiency of the selection process for event/condition analyses using the results of the reactor oversight program's "significance determination process" results, inspection findings, and initial event assessment analyses. The planned improvements are expected to reduce the resources needed for screening events and improve the timeliness of final analyses. The staff also plans to improve its capability to derive ASP insights to make a more direct impact on the regulatory process. An eventual goal is to expand the number of staff capable of conducting ASP event assessments using state-of-the-art SPAR models.

Major Milestones

Provide 1999 annual report 3/2001

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 3: *We will evaluate operating experience and the results of risk assessments for safety implications.*

Implementation Activity MS3-5: Complete system reliability and related studies. **(RES)**

The staff will continue to conduct assessments of the risk significance of operational events and data trends from reactor operations. Trends in a) initiating events, b) system reliability and availability, c) non-compliance with license requirements, d) occurrences of common-cause failure modes, and e) events occurring at low power and shutdown conditions are assessed in conjunction with current PRAs so that safety vulnerabilities can be identified on a timely basis. These assessments also provide the information needed to update PRA assumptions to improve the accuracy and credibility of risk assessments.

Major Milestones

Complete study on CCF insights 1/2001

Complete system reliability and related studies
(air-operated valves, motor-operated valves, B&W reactor
pump seal, CE reactor pump seal) 2/2001

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 3: *We will evaluate operating experience and the results of risk assessments for safety implications.*

Implementation Activity MS3-6: Review IPEEE submittals and issue insights report. **(RES)**

The results of the licensees' individual plant evaluation (external events), IPEEEs, provide valuable information on the strengths and weaknesses of existing plants. The staff will complete its review of these evaluations and develop risk insights. The results of this effort will be used to support development of guidance and standards on the use of risk assessment and plant-specific risk information to support the risk-informed reactor oversight program. The outcomes of this work are potential safety enhancements and licensee burden reductions through greater use of risk analysis methods and results.

Major Milestones

Complete review of IPEEE submittals (discussed in Attachment 2) 1/2001

Issue draft report for public comments on general perspectives from IPEEE program 4/2001

Issue final IPEEE insights report 10/2001

Complete draft NUREG report on fracture testing of IAEA RPV material from cooperative program 9/2001

Finalize draft of RG 1.99 revision and publish for comment 12/2001

Publish a NUREG/CR report describing a generic flaw density and size distribution . . . 12/2001

Complete analyses and propose technical basis for revision of PTS rule 9/2002

Performance Goal: Maintain safety, protection of the environment, and of the common defense and security

Strategy 5: *We will ensure that changes to operating licenses and exemptions to regulations maintain safety and meet regulatory requirements.*

Performance Goal: Reduce unnecessary regulatory burden on Stakeholders

Strategy 1: *We will utilize risk information and performance-based approaches to reduce unnecessary regulatory burden.*

Implementation Activity MS5-1 (UB1-3) Establish guidance for risk-informed licensing basis changes. **(NRR&RES)**

Subactivity 1: Regulatory Guide 1.174 (and update) and SRP Chapter 19

The PRA policy statement encourages greater use of PRA in all regulatory activities. One major activity is using PRA to support decisions to modify an individual plant's licensing basis. The staff prepared guidance documents to guide such risk-informed changes to a plant's licensing basis, as in requests for technical specification changes. The guidance describes acceptable means for assessing the nature and impact of licensing basis changes when the change request is supported by risk information. In being risk-informed, rather than solely based upon risk information, the NRC is retaining certain principles such as consistency with the defense-in-depth philosophy and maintenance of sufficient safety margins. The RG (and the SRP) were issued for public comment before being issued.

NRC is conducting periodic reviews of these documents to identify any desired improvements. The first review, documented in a memo dated June 30, 1999, identified four topics for revision. The topics were (1) discussion of an ASME standard on PRA quality, (2) shutdown risk, (3) seismic margins method, and (4) clarification on fuel burnup and composition.

Subactivity 2: Application-specific guidance

In addition to the general guidance documents for risk-informed licensing basis changes, NRC also developed documents specific to particular topics for which there was interest in pursuing risk-informed changes. These include graded quality assurance, inservice inspection, inservice testing, and technical specifications.

2.1 Graded Quality Assurance (RG 1.176)
2.2 Inservice Inspection (RG 1.178 and SRP 3.9.8)
2.3 Inservice Testing (RG 1.175 and SRP 3.9.7)
2.4 Technical Specifications (RG 1.177 and SRP 16.1)

Implementation Activity 2.1 : Updating the Graded QA RG

The graded QA RG 1.176 was issued in August of 1998. A GQA program was approved for the South Texas Project in November of 1997. Because the RG was issued after the pilot application was approved, it reflects the lessons learned from the pilot review.

While implementing the GQA pilot application, STP determined that they will derive much less benefit than they had anticipated from application of GQA. In particular, according to the licensee, special requirements in other regulations require continued complex and costly controls on many SSCs regardless of the reduced QA requirements. Consequently, there have been no further applications to apply GQA. In July of 1999, STP submitted a request for exemptions from many of the special treatment requirements, including greater relief from quality assurance requirements than allowed by the GQA program. The staff is utilizing this exemption request within the ongoing effort to risk-inform the special treatment requirements of 10 CFR Part 50 (see item MS8-1).

Furthermore, a 10 CFR 50.54(a) rule change became effective April 26, 1999, that allows licensees to apply any QA program alternatives or exceptions that have been approved by the NRC staff for any other nuclear power plant (if the bases can be shown to be applicable at the plant making the change) without requesting prior staff review and approval.

Therefore, changes to RG 1.176 will be not be performed until further experience is gained with the effects of the 10 CFR 50.54(a) rule change on licensees' QA programs, and the staff's review of the STP exemption request in completed. Based upon the current projected review schedule for the STP exemption request, the staff anticipates that the effort to revise RG 1.176 would commence no sooner than the spring of 2001 and take approximately one year.

Implementation Activity 2.2: Review of Inservice Inspection Topical Reports

On June 11, 1998, the staff sent to the Commission SECY-98-139, which transmitted Regulatory Guide 1.178 and Standard Review Plan (SRP) Section 3.9.8 for trial use. These documents provide guidance to licensees and staff regarding risk-informed inservice inspection (RI-ISI) programs for piping systems.

The industry submitted topical reports with two different methods, one developed by Westinghouse Owners Group and the other by Electric Power Research Institute (EPRI), for incorporating risk insights into their RI-ISI programs.

The staff was reviewing three ASME Code Cases (N-560, N-577, and N-578) in parallel with the pilot plant submittals.

Since the issuance of SECY-98-139, the staff has completed reviews of RI-ISI programs from four pilot plants and approved the methodologies presented in the two industry topical reports. However, industry schedules for RI-ISI submittals slipped such that the pilot plant reviews could not be completed until August 1999. The industry submittal for one of the methodologies (EPRI Report) was done in October 1999. The three ASME code cases have recently been revised to incorporate lessons learned in the review and approval of the pilot plant submittals and industry methodologies.

In addition to the four pilot plant, the staff has approved three other risk-informed ISI submittals. Five more are currently under review. However, as expected, the pilot program and other reviews have identified areas where the industry wants to modify the original generic framework. The Westinghouse Owners Group has submitted an addendum to its RI-ISI topical report that proposes to incorporate augmented programs not covered by the methodology approved by the staff. Furthermore, both the Westinghouse Owners Group and EPRI are meeting with the staff to discuss the possibility of expanding the RI-ISI methodology to include augmented inspection programs such as high energy line break exclusion zone piping and components other than piping.

The staff believes that work to revise Regulatory Guide 1.178 and the SRP should not commence until further experience is acquired with the revised industry methodologies (expected to be evaluated by the end of December 2000) and the finalization of the three ASME Code Cases. Contingent upon completion ot these documents, the staff expects to finalize Regulatory Guide 1.178 and the SRP by the end of December 2001.

Implementation Activity 2.3: Inservice Testing

The guidance for inservice testing has been issued and a pilot application was completed in 1998. Several other applications, generally of limited scope, have been approved or are in the process of being reviewed. As with the other application guides, the staff plans to review its experience for possible revision of the guidance. Candidate areas, as discussed in August 30, 1999, memo include potential relaxation of performance monitoring and corrective actions for low safety-significant SSCs if common cause failure is not of risk significance.

This review is planned to begin after March 2001.

It should also be noted that the ASME has been developing a series of Code Cases aimed at incorporating risk insights into the ASME Code test requirements for pumps and valves. Once approved, it may be appropriate to adjust the guidance in the RG to reflect their issuance.

Implementation Activity 2.4: Technical Specifications

Plant-specific licensing actions using the risk-informed guidance on TS have been processed largely in the area of relaxations of allowed outage times for particular SSC.

The August 30, 1999, review of the guidance document identified one area for possible revision relating to the nexus of configuration risk management with the maintenance rule (50.65(a)(4)). The staff's activities related to risk-informing TS include several other initiatives discussed under another activity(see item MS8-3).

Subactivity 3: Guidance for use when reviewing non-risk-informed submittals

As Policy Issue 4 in SECY-98-300 (Options for Risk-Informed Revisions to 10CFR Part 50), the staff recommended developing additional guidance with respect to the use of risk-informed approaches in regulatory activities. This guidance would be used in deciding if undue risk may exist when all other regulatory requirements appear to be met. In the related Staff Requirements Memorandum (SRM), the Commission approved the recommendation and requested the staff to submit clarifying guidance for Commission approval.

SECY-99-246 (Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews) proposed interim guidance for applying risk-informed decisionmaking in the review of non-risk-informed license amendment requests. Central to the process is a determination as to whether the license amendment request, if granted, could create "special circumstances" under which plant operation may pose an undue risk to public health and safety even though all other regulatory requirements appear to be satisfied. In the related SRM, the Commission approved the use of this guidance on an interim basis while the staff proceeds to engage stakeholders and finalize the guidance, and directed the staff to inform the Commission if it determines (during the interim period) that a license amendment application meets the special circumstances standard. The interim guidance has been disseminated to industry via Regulatory Issue Summary 2000-7 (issued 3/28/00).

The NRC plans to formally issue the guidance as a new appendix to Chapter 19 of the Standard Review Plan. The appendix provides guidance to the NRC staff on the use of risk information in those rare instances where license amendment requests meet regulatory requirements but raise significant risk concerns due to some special circumstances associated with the request. The proposed appendix was published in the Federal Register for public comment on April 10, 2000. The NRC held a public meeting in Rockville, Maryland to discuss the appendix on May 16, 2000, prior to close of the comment period.

The proposed guidance has also been discussed with the Advisory Committee for Reactor Safeguards (ACRS) and the Committee to Review Generic Requirements (CRGR). The staff forwarded final guidance documents to the Commission on September 26, 2000, following resolution of public comments and final review by ACRS and CRGR. Should the Commission approve, the staff intends to issue a Regulatory Issue Summary announcing the final guidance.

Subactivity 4: Provide guidance for reviewing risk-important human actions in plant-specific licensing actions.

The staff has prepared a draft report that contains technical bases to develop risk-informed guidance and acceptance criteria for the review of licensee proposals affecting operator actions credited in safety analyses. The technical bases uses a graded approach, with actions of higher risk importance receiving more detailed NRC review than those of lesser significance.

<u>Major Milestones</u>

Issue final guidance for special circumstances for
use of risk-information in non-risk-informed licensing actions 11/2000

Publish NUREG/CR on risk-important human actions 10/2000

Publish NUREG Guidance for reviewing risk-important
human actions . . . . 3 months after ACRS/CRGR review

Complete revisions to RG 1.174 . . . . 8/2001

Complete update of application-specific guides . . . . following RG 1.174 update

Complete RG 1.178 and SRP on risk-informed ISI 12/2001

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 8: *We will continue to develop and incrementally use risk-informed and, where appropriate, less-prescriptive regulatory approaches to maintain safety.*

Performance Goal: Reduce Unnecessary Regulatory Burden on stakeholders

Strategy 1; *We will utilize risk information and performance-based approaches to reduce unnecessary regulatory burden.*

Implementation Activity MS8-1 (UB1-1): Develop an alternative risk-informed approach to special treatment requirements in Part 50, that would vary the treatment applied to structures, systems and components (SSC) on the basis of their safety significance using a risk-informed categorization method. **(NRR)**

The 1995 policy statement on the NRC's expanded use of PRA states "the use of PRA technology should be increased in all 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."

The Commission's current regulatory framework is based largely upon prevention and mitigation of particular design-basis events (DBE) under specific assumptions. Those SSCs needed to withstand these DBE are then subject to regulatory requirements intended to provide a high degree of assurance that those SSCs will function under these design basis conditions. These requirements, now referred to as "special treatment requirements," include qualification, change control, documentation, reporting, maintenance, testing, surveillance and quality assurance requirements. While these requirements have been effective in maintaining safety, in some cases they also result in unnecessary regulatory burden, in that the requirements apply to all SSCs with this category regardless of their relative contribution to plant safety.

The agency is committed to risk informing 10 CFR Part 50. This effort consists of two parts: (1) making changes to the overall scope of systems, structures, and components (SSCs) covered by those section in Part 50 requiring special treatment (such as quality assurance, technical specifications, environmental qualification), and (2) making changes to specific requirements in the body of regulations. (These efforts are referred to as Option 2 and Option 3, respectively, in SECY-98-300). The staff has developed detailed plans for accomplishing these activities. These activities are described further in SECY-99-256 and SECY-00-194 for Option 2 SECY-99-264, and in SECY-00-086, and SECY-00-0198 for Option 3 .

Under Option 2, the NRC is developing alternative regulations in 10 CFR Part 50 (and other applicable parts) that would modify the requirements for special treatment to focus on those SSCs that have been identified as important to protect public health and safety by using a risk-informed categorization approach. The categorization process is to be performed as part of an integrated decision-making process which uses both risk insights and traditional engineering insights. SSCs that are significant contributors to plant safety would continue to be subjected to the full set of special treatment requirements. In some cases, SSCs that are significant contributors to plant safety (using risk insights), but which are not presently required to satisfy special treatment requirements will, under the rule, have to meet certain requirements to maintain availability and reliability. Other SSCs that are not significant contributors to plant safety, but which are currently subjected to special treatment requirements, would be subject to reduced or no requirements.

In SECY-98-300 and in SECY-99-256, the staff outlined that the proposed approach would enable licensees and NRC to focus their resources on SSCs that make a significant contribution to plant safety. Through revision of the special treatment requirements, the underlying design functional requirements are retained, but burden reduction is anticipated with reducing or removing special treatment requirements on a set of low-significant SSCs. This approach was viewed as being consistent with the PRA policy statement, available PRA technology, efficiency of NRC resources, and responsiveness to stakeholder concerns. Further, changes to operationally-oriented requirements were viewed as a better opportunity for using risk insights rather than changes to design requirements in that existing facilities are already designed and constructed.

An Advance Notice of Proposed Rulemaking (ANPR) was published in March 2000 to seek public input on this proposal. In SECY-00-194, dated September 7, 2000, the staff discussed the public comments on the ANPR, the staff's preliminary views on the comments, and a conceptual approach for implementing the rulemaking plan. The next step is to prepare a proposed rule for Commission review.

To support the rulemaking efforts, NEI has proposed to prepare a guideline document to describe the categorization and special treatment requirements to apply in conjunction with such a rule change. The staff is reviewing the categorization and treatment sections of the guideline document separately and provided initial comments on the categorization section to NEI in a letter dated September 26, 2000.

In addition, the categorization of SSCs needs to use a PRA of acceptable quality to appropriately categorize the SSCs as to their relative risk significance. The industry has developed a peer review process to apply to PRAs; NEI-00-02 presents the peer review process including the technical elements used to judge the PRAs. The NRC is also reviewing this document in the context of its use in support of this initiative and provided initial comments to NEI in a letter dated September 19, 2000.

The staff concluded that pilot activities would be useful to help refine the rule and guidance documents. The NSSS owners' groups are developing plans for pilot activities, focusing on a limited number of systems to test efficacy of the categorization and treatment guidelines, but have not yet provided details of their plans to the NRC.

The licensee for South Texas has submitted an exemption request that would allow them to apply the concepts underlying this rulemaking (categorization, removal of special treatment requirements) at their facility, by receiving exemptions to certain existing requirements that would prevent them from otherwise undertaking such a program. This review is considered to be a "proof-of-concept" prototype for the rulemaking. The experience from their efforts and the staff review are being coordinated with the rulemaking activities and guidance development.

Major Milestones

Rulemaking

Proposed Rule 8/2001
Final Rule 12/2002

South Texas Exemption

Draft Safety Evaluation 11/2000

Final Safety Evaluation 4/2001

Pilot reviews

Owners groups submittals/plans 1/2001

Staff review complete 6/2001

NEI Guidance review

Staff completes review of categorization/treatment/peer review guideline documents . . 6/2001

Performance Goal: Maintain safety, protection of the environment, and of the common defense and security

Strategy 8: *We will continue to develop and incrementally use risk-informed and, where appropriate, less-prescriptive regulatory approaches to maintain safety.*

Performance Goal: Reduce unnecessary regulatory burden on stakeholders

Strategy 1: *We will utilize risk information and performance-based approaches to reduce unnecessary regulatory burden*.

Implementation Activity MS8-2 (UB1-2): Make changes to specific technical requirements in Part 50 using risk information. **(RES and NRR))**

The staff's work on Option 3 will focus on providing a better balance to the Part 50 technical requirements among those needed to provide defense in depth, to maintain appropriate safety margins, and to limit risk. This improved balance will be achieved by systematic consideration of the Part 50 requirements. The staff's approach to risk-informing Part 50 will necessitate a broad assessment of Part 50 requirements, rather than a review of individual regulations. As such, the staff's work may involve changing regulations in sets, rather than individually. That is, risk-informing Part 50 may involve relaxing requirements in some areas in combination with increasing requirements in other areas to achieve a better balance.

The Option 3 effort entails two phases. In Phase 1, the staff will study the ensemble of technical requirements contained in 10 CFR Part 50 and associated implementing documents to: (1) identify candidate changes to requirements and design basis accidents, (2) prioritize candidate changes to requirements and design basis accidents, and (3) identify recommended changes to requirements. The focus of the staff's work will be on providing a better balance to the Part 50 technical requirements among defense-in-depth, safety margin, and risk considerations. This work will result in a set of recommended changes to the requirements and priorities for implementation. During Phase 2, the technical bases for rule changes will be developed and rulemaking will be conducted.

In SECY-00-0198, the staff provided (1) a status report on its study of possible risk-informed changes to the technical requirements of 10 CFR Part 50, (2) recommendations for risk-informed changes to 10 CFR 50.44 ("Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors") that will both enhance safety and reduce unnecessary burden, (3) policy issues for Commission decision, and (4) an update to the framework that describes the approach, process and guidelines the staff will apply in reviewing, formulating, and recommending risk-informed alternatives to 10 CFR Part 50 technical requirements.

Major Milestones

Provide recommendations and feasibility report to Commission on other Part 50 changes (such as 10 CFR 50.46) 6/2001

Provide recommendations and feasibility of changes to other rules (e.g., reactivity insertion accidents . . . . TBD

Performance Goal: Maintain safety, protection of the environment, and of the common defense and security.

Strategy 8: *We will continue to develop and incrementally use risk-informed, and, where appropriate, performance-based regulatory approaches to maintain safety.*

Performance Goal: Make NRC activities and decisions more effective, efficient, and realistic.

Strategy 1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions.*

Performance Goal: Reduce unnecessary regulatory burden on stakeholders

Strategy 1: *We will utilize risk information and performance-based approaches to reduce unnecessary regulatory burden.*

Implementation Activity MS8-3 (EE1-1 and UB1-4): Develop risk-informed improvements to the standard technical specifications (STS). **(NRR)**

Since the mid-1980's, the NRC has been reviewing and granting improvements to technical specifications that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on technical specification improvements of July 22, 1993, the Commission stated that it expects that licensees will utilize any plant-specific PRA or risk survey in preparing their technical specification related submittals. The Commission reiterated this point when it issued the revision to 10 CFR 50.36, “Technical Specifications,” in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision making and regulatory efficiency. Since that time, the industry and the NRC have been pursuing increased use of PRA in developing improvements to technical specifications. Guidance documents have been prepared to assist in this regard and a number of specific changes have been implemented (see implementation activity MS5-1).

Consistent with the Commission’s policy statements on technical specifications and the use of PRA, the NRC and the industry continue to develop risk-informed improvements to the current system of technical specifications. These improvements are intended to maintain or improve safety while reducing unnecessary burden and to bring technical specification requirements into congruence with the Commission’s other risk-informed regulatory activities.

Proposals for risk-informed improvements to the STS are identified by the industry and discussed with the NRC staff before a submittal is developed. The proposals are judged based on their ability to maintain or improve safety, the amount of unnecessary burden reduction they will likely produce, their ability to make NRC’s regulation of plant operations more efficient and effective, the amount of industry interest in the proposal, and the complexity of the proposed change.

The industry and the staff have identified eight initiatives to date for risk-informed improvements to the STS. They are:

1. Define the preferred end state for technical specification actions (usually Hot Shutdown for PWRs);

2. Increase the time allowed to delay entering required actions when a surveillance is missed;

3. Modify existing mode restraint logic to allow greater flexibility (i.e., use risk assessments for entry into higher mode limiting conditions for operation (LCOs) based on low risk);

4. Replace the current system of fixed completion times with reliance on a configuration risk management program (CRMP);

5. Optimize surveillance frequencies;

6. Modify LCO 3.0.3 actions to allow for a risk-informed evaluation to determine whether it is better to shut down or to continue to operate;

7. Define actions to be taken when equipment is not operable but is still functional.

8. Risk-inform the scope of the TS rule

<u>Major Milestones</u>

Staff completes review of Initiative 2 . . . . 12/2000

Industry submits proposal for Initiative 6 . . . . 1/2001

Staff completes review of Initiative 3 . . . . 3/2001

Industry submits proposal for Initiative 1 . . . . 3/2001

Staff completes review of Initiative 6 . . . . 7/2001

Staff completes review of Initiative 1 . . . . 9/2001

Industry submits proposals for Initiatives 4, 5, and 7 . . . . 12/2001

Staff completes review of Initiatives 4, 5, and 7 . . . . 12/2002

Industry submits proposal for Initiative 8 . . . . 12/2002

Staff completes review of Initiative 8 . . . . 12/2003

Performance Goal: Maintain Safety, protection of the environment, and of the common defense and security

Strategy 8 *We will continue to develop and incrementally use risk-informed and, where appropriate, less prescriptive performance-based regulatory approaches to maintain safety.*

Implementation Activity MS8-4: Develop alternative fire protection standards for nuclear power plants. **(NRR&RES)**

The staff has been working with the National Fire Protection Association (NFPA) to develop an alternative performance-based risk-informed fire protection standard for nuclear power plants. A draft of this standard, NFPA-805,has been issued for public comment and resolution of comments is expected in the spring of 2001. A rulemaking plan was submitted to the Commission in SECY-00-0009 that would, after final Commission endorsement, provide for incorporation of that standard into Commission regulations as a voluntary alternative to current fire protection requirements.

Another activity related to fire protection is the Circuit Analysis Resolution Program. In response to the need to resolve post-fire safe shutdown, fire-induced circuit failure analysis issues, the Boiling Water Reactor Owners Group (BWROG) and the Nuclear Energy Institute (NEI) have respectively developed deterministic and risk-informed post-fire safe shutdown methodology documents. These two documents have recently been combined into one document which provides a step-by-step means of deterministically conducting safe shutdown analyses, while is intended to provide optional risk-informed methods for selected analytical steps.

Methodology improvements related to fire protection are discussed under item EE1-3.

Major Milestones

Publish proposed rule 10/2001

Final Rule 4/2002

Performance Goal: Maintain safety, protection of the environment, and of the common defense and security

Strategy 8 *We will continue to develop and incrementally use risk-informed and, where appropriate, less prescriptive performance-based regulatory approaches to maintain safety.*

Implementation Activity MS8-5: Develop alternative requirements for safeguards that are risk-informed and/or performance-based. **(NRR)**

The staff has underway a comprehensive review of 10 CFR 73.55; the staff intends to include a requirement for power reactor licensees to conduct drills and exercises to evaluate their protective strategy against a simulated design basis threat. In performing this review, the staff is also seeking opportunities to make the requirements risk-informed, as for example in terms of the equipment that needs to be protected.

In SECY-00-0063, the staff provided its proposed approach for revising this regulation, which was approved by the Commission in an SRM dated April 12, 2000. Subsequently the staff provided a status report of its activities in SECY-00-0142, dated June 26, 2000.

Major Milestones

Proposed rule to Commission 5/2001

Performance Goal: Maintain safety, protection of the environment, and the common defense and security

Strategy 8: *We will continue to develop and incrementally use risk-informed, and where appropriate, performance-based regulatory approaches to maintain safety.*

Implementation Activity MS8-6: Develop the technical basis to improve evaluations of reactor pressure vessel (RPV) integrity. **(RES)**

The staff is working to develop the technical basis to improve the realism of evaluations of reactor pressure vessel (RPV) integrity to support risk-informed modifications to the regulations associated with RPV integrity. The staff is evaluating the application of advanced fracture mechanics concepts to the revision of the regulatory framework for RPV integrity to provide analysis codes and techniques for evaluating licensee submittals pertaining to RPV integrity, particularly as related to pressurized thermal shock (PTS). The staff is also conducting the research and analyses needed to develop a statistically valid generic flaw density and size distribution for reactor vessel welds and plates for use by the staff and licensees in performing probabilistic fracture evaluations of reactor pressure vessels. In addition, the staff is performing experimental program and computer analyses to support rulemaking for PTS and guidance for reactor vessel embrittlement. The results of these efforts will be reflected in review guidance documents and in modifications to the regulations addressing issues associated with reactor pressure vessel integrity such as setting operating pressure-temperature limits and LTOP setpoints, and in applying the 10 CFR 50.61 pressurized thermal shock (PTS) screening criteria. Some specific staff activities include:

-Continuing the mechanistic and statistical assessment of plant embrittlement data

-Completing a NUREG report on effects of heat treatment and chemistry unavailability on embrittlement trends

-Completing the development of the technical bases for revision of RG 1.99

-Completing irradiation of high-Cu, high-Ni welds and validation of embrittlement trend curves

-Conducting an expert elicitation to verify that a generalized flaw size and density distribution can be properly developed for the entire population of U.S. RPVs and to assist in developing a flaw distribution

-Performing calculations to provide technical basis for revising 10 CFR 50.61 (the PTS rule)

-Integrating results of probabilistic fracture calculations and PRA considerations to develop revised PTS screening criteria for incorporation into 10 CFR 50.61

-Characterizing flaw density size distributions for River Bend and Hope Creek vessels

Major Milestones:

Performance Goal: Make NRC activities and decisions more effective, efficient, and realistic.

Strategy 1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions.*

Implementation Activity EER1-1: Incorporate risk insights into reviews of advanced reactor designs. **(NRR)**

As discussed in several Commission policy statements (e.g., Policy Statement on Severe Reactor Accidents regarding future designs and existing plants August 1985), and SECY papers, for advanced reactors, the Commission's objective is that designers of new plants would achieve a higher standard of severe accident performance than prior designs.

In the requirements promulgated in 10 CFR Part 52 for Standard Design Certification applications, NRC included a requirement that a PRA be performed for each design. Further, the application is to demonstrate compliance with technically-relevant requirements set forth in 50.34(f)[added as lessons-learned from the TMI accident]. The staff also assessed performance with respect to other technical and policy issues in SECY-90-016 and SECY-93-087. These issues were derived, in part, from risk insights. Embodied in the change control processes are specific criteria for when NRC review is needed for changes affecting resolution of severe accident issues.

Three standard design certifications are complete. The staff is performing a lessons-learned review of Part 52 based on these three reviews. As part of this effort, the staff also plans to add a requirement that an applicant/operator of a standard-design reactor would maintain, update and use the PRA ("living PRA") beyond the design phase for the life of the facility.

Major Milestones

Proposed revisions to Part 52 to Commission . . . . spring 2001

Final rule to Commission 2002

Performance Goal: Make NRC activities and decisions more effective, efficient, and realistic.

Strategy 1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions.*

Implementation Activity EER1-2: Develop standards for the application of risk-informed, performance-based regulation in conjunction with national standards committees. **(RES and NRR)**

The increased use of probabilistic risk assessments (PRA) in the regulatory decision-making process requires consistency in the quality, scope, methodology and data used in such analyses. These requirements apply to PRAs developed by industry to support specific, risk-informed licensing actions as well as to PRAs developed by NRC staff to analyze specific technical issues or to support risk-informed Commission decisions. To this end, NRC has been working with the American Society of Mechanical Engineers (ASME) to develop a national consensus standard setting forth specific guidance regarding the construction and execution of a PRA covering internal initiating events (Level 1). When developed, such a standard will help to ensure that PRAs developed in accordance with this standard are robust, consistent, and defensible and are documents upon which regulatory decisions can confidently be made. While the ASME maintains overall responsibility for this effort, active NRC and industry participation has been, and will continue to be, essential to the development of such a standard. In parallel, the staff has been working with the National Fire Protection Association (NFPA) to develop standards for fire risk analysis (See activity MS8-4).

More recently, the NRC staff has been working with the American Nuclear Society (ANS) to develop a companion standard covering probabilistic analyses that would include the progression of severe accidents, the impacts of external events on plant risk, and risk-significant events that could occur when a plant is operating at low power or when shutdown (LP/SD).

As discussed under activity MS-8-1, the staff is also reviewing the NEI peer review process for PRA, in particular for use with that activity.

In addition, the staff is active with the Institute of Electrical and Electronic Engineers (IEEE) in developing an industry standard for conducting human reliability analysis at nuclear power generating stations.

Major Milestones

Recognizing that control of these projects properly rests with the standards committees, the following milestones have been established by these organizations:

Final PRA standard issued by ASME 3/2001

Final fire PRA standard issued by NFPA 3/2001

Final PRA standards issued by ANS
    Seismic 6/2001

LP/SD 6/2001

Final HRA Standard issued by IEEE 1/2005

Performance Goal: Make NRC activities and decisions more effective, efficient, and realistic.

Strategy1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions.*

Implementation Activity EER1-3: Develop improved methods for calculating risk in support of risk-informed regulatory decision making **(RES)**

From the ground-breaking work of the WASH-1400 study and the NUREG-1150 reactor risk studies through the individual plant examinations and present risk studies, tremendous advancements have been made in PRA methods. Consistent with the direction provided in the 1995 PRA policy statement, the NRC is continuing to develop methods needed to better support realistic, risk-informed decision making. Current PRA methods do not adequately address certain key aspects of plant risk, including the effects of quality assurance, human reliability, fire, low power and shutdown operations, degraded SSCs, and digital instrumentation and control failures. Uncertainty concerning the nature and magnitude of the contributions of these aspects to plant risk, particularly as they relate to agency decision making processes and acceptance criteria, will limit progress in risk-informed regulation by requiring conservative decisions to be made to account for large uncertainties. The new methods will complement the methods developed to-date, further reducing uncertainties and improving realism. The quality of risk assessments is also highly dependent upon the quality of the engineering analyses (e.g., thermal-hydraulic, severe accident, structural) that is used to calculate plant performance and success criteria. Although not included in this plan, work to improve and ensure the analytical tools used for these analyses are realistic and readily useable is vital to the success of risk-informed regulation.

Decisions to pursue development of methods and models are made based on three general considerations: (1) the importance of new methods to risk informing our regulations; (2) the adequacy of existing methods for understanding the risk implications of experimental findings and operational experience; and (3) the availability of methods for assessing the risk associated with the introduction of new technologies and new reactor designs. These criteria are associated with the issue of PRA model completeness and the degree to which PRA models adequately characterize risk-important failure modes and mechanisms. Thus, the more complete our understanding of plant risk, the more free are we to identify and remove unnecessary conservatism from our regulations and decision making.

With these three considerations in mind, the following research efforts have been identified:

- In the effort to risk inform Part 50 requirements, quality assurance requirements were identified as high-priority candidates to be risk informed. A study is being planned to assess the feasibility of modeling the influence of quality assurance activities on plant risk within the context of PRA. Dependent on the results of this study, future work may be pursued to develop such models.

- The development of performance-based fire standards and regulations requires a sound understanding of fire and its contribution to power plant risk. Current fire PRA models are not adequate to support credible, risk-informed changes to these standards and regulations. A fire risk program has been developed and is being implemented to address the complex issues associated with fire risk.

- Level 2 PRA methods address containment performance. An evaluation of the implications of hydrogen research findings on the realism of existing level 2 PRA methods is underway and may suggest needed improvements.

- Regulatory Guide 1.174 provides guidance for making changes to a plant's licensing bases based on total plant risk. Plant risk during all plant operating modes has not yet been calculated, most notably during low power and shutdown operations. The consequences of incomplete risk profiles are that conservative decisions are made to compensate for the lack of risk information and uncertainty exists in our knowledge of defense-in-depth and safety margins during such operating modes. No research is underway at this time to calculate the risk associated with low power and shutdown operations.

Major Milestones

Issue report on fire suppression analysis methods 12/2000

Draft plan for HRA research 12/2000

Draft plan for digital system software risk 12/2000

Revised plan for fire risk 11/2000

Complete feasibility study on developing PRA models on QA effects 9/2001

Develop plan to upgrade MACCS on health effects, land contamination . . . . TBD

Performance Goal: Make NRC activities and decisions more effective, efficient, and realistic.

Strategy 1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions.*

Implementation Activity EER1-4: Develop and maintain analytical tools for staff risk applications **(RES)**

The agency has developed analytical tools that the staff uses in its risk assessments associated with generic safety issues, regulatory backfit reviews, plant operating states, and operational experience. This suite of PRA codes has given the staff the tools it needs to reach risk-informed decisions, independent of licensee analyses. Thus, the staff plans to continue to maintain the SAPHIRE computer code for conducting PRA. The accident sequence precursor (ASP) program uses simplified PRA models as part of the staff's operating event assessments. The staff also plans to continue to develop and improve the ASP SPAR models so the staff can better analyze the risk significance of operating events.

Major Milestones

Complete third set of preliminary (level 1, revision 3) ASP SPAR models 9/2001

Continue SAPHIRE computer code data entry and maintenance 9/2001

Performance Goal: Make NRC activities and decisions more effective, efficient, and realistic.

Strategy 1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions.*

Implementation Activity EER1-5: Organize and conduct an international cooperative research program on PRA to coordinate and share results of research activities. **(RES)**

Recognizing the increasing use worldwide of probabilistic risk assessment (PRA) and probabilistic safety assessment (PSA) in the regulation of nuclear facilities, there is a strong need to maximize progress in addressing key PRA/PSA research and development issues. As such, COOPRA (the International Cooperative PRA Research Program) was formed to:

- Improve the sharing of PRA/PSA information

- Improve efficiency in developing and using needed PRA/PSA tools.

Major Milestones

Convene annual international cooperative PRA program meeting 12/2000

Performance Goal: Make NRC activities and decisions more effective, efficient, and realistic

Strategy 1: *We will use risk information to improve the effectiveness and efficiency of our activities and decisions.*

Implementation Activity EER1-6: Assess regulatory effectiveness using risk information. **(RES)**

The staff will conduct an integrated evaluation of risk information, inspection findings, operating experience, domestic and international research results, and cost data to identify ways to improve the effectiveness of NRC regulatory requirements, guidance, and processes.

Major Milestones:

Evaluate effectiveness of ATWS rule 4/2001

Evaluate effectiveness of USI A-45 resolution 9/2001

Evaluate effectiveness of 10CFR50, App J, Option B 1/2002