October 14, 1997 SECY-97-234

FOR: The Commissioners

FROM: L. Joseph Callan /s/
Executive Director for Operations

SUBJECT: QUARTERLY STATUS FOR THE PROBABILISTIC RISK ASSESSMENT IMPLEMENTATION PLAN

PURPOSE:

This quarterly report presents the status of activities for the Probabilistic Risk Assessment (PRA) Implementation Plan, including the development of risk-informed standards and guidance. The report also serves to provide responses to Staff Requirements Memoranda (Attachment 1) dated May 28, 1997, June 5, 1997, and June 13, 1997, which include, respectively:

(1) actions the staff has taken to expedite (a) the use of IPE results to prioritize inspection activities; (b) improvements in regional capabilities for the use of PRA and risk insights; and (c) provision of related inspector training;

(2) the staff's plans for training NRC staff on (a) the risk-informed regulatory approach(es) contained in the regulatory guidance and standard review plan documents and (b) overall PRA methods and techniques; and

(3) an update on the staff's efforts to work with industry to address shortfalls and limitations in the data on reliability and availability of risk-significant systems to be provided to the staff voluntarily.

CONTACT:
Ashok Thadani, OEDO
415-1705

BACKGROUND:

In a memorandum dated January 3, 1996, from the Executive Director for Operations to Chairman Jackson, the staff committed to submitting quarterly reports on the status of its development of risk-informed standards and guidance. Previous quarterly reports were sent to the Commission on March 26, June 20, and October 11, 1996, and on January 13, April 3, and July 22, 1997. This quarterly report covers the period July 1, 1997 to September 30, 1997.

DISCUSSION:

Attachments 2 and 3 provide this quarter's implementation plan update. Significant achievements in the past quarter include the following:

- The staff incorporated proposed resolutions of the policy, technical, and process issues in drafts of the application-specific Regulatory Guide (RG) and Standard Review Plan (SRP) for inservice inspection (ISI), and discussed these new drafts with the Advisory Committee on Reactor Safeguards (ACRS) and the Committee to Review Generic Requirements (CRGR). Both the ACRS and the CRGR have reviewed the guidance and concurred in the staff's proposal to issue the guidance for comment by the public. On August 20, 1997, the staff forwarded the draft guidance documents to the Commission and requested its approval for issuing the documents for comment. Commission approval was received in an October 1, 1997, SRM.

  To facilitate solicitation of public comments on the ISI RG and SRP, the staff will conduct a workshop during the comment period to explain the draft documents and answer questions. The workshop will be held late November or early December, at the Marriott Hotel in Bethesda, Maryland.

  In completing the draft RG and SRP for risk-informed inservice inspection, the staff has found that a greater than expected effort was required to incorporate all points of view and gain a consensus on draft guidance. With this experience, the staff projects that the schedule for issuing the final ISI RG and SRP will slip from February 1998 to April 1998.

  For risk-informed ISI programs, the industry had identified three pilot plants that would submit requests for authorization for the use of risk-informed ISI methodology; these applications have not yet been received. In a letter from NEI, dated August 29, 1997, the industry requested to add two additional plants to the list of pilot applications and identified an aggressive schedule for all the pilot plants. The scope for the two new pilot plants is limited to Class-1 piping (primary coolant system piping only). To date, only one application has been submitted by one of the pilot plants with a limited scope RI-ISI program.

  Due to industry's delays in submitting the applications, and the addition of two plants as pilots, the staff is unable to develop an integrated review schedule for the pilot plants at this time. This schedule is contingent on information regarding actual timing of submittals, the quality of the submittals, and the ability of the pilot plant licensees to commit the resources necessary to respond to the staff's requests for additional information (RAIs). The staff continues to hold working meetings with industry to facilitate the development of regulatory guidelines.

- The staff completed ten more maintenance rule baseline inspections, which included inspection of licensee methods for using PRA in maintenance programs and in inspection of safety assessments performed by licensees when removing equipment from service for maintenance in accordance with Paragraph (a)(3) of the Maintenance Rule. As of September 30, 1997, the staff has completed 36 inspections.

- The NRC staff briefed the Commission in May 1997 on the Individual Plant Examination (IPE) insights report, draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance." As a result of this briefing, the staff received a SRM dated May 28, 1997, requesting the staff "to expedite activities in the following areas: (1) using IPE results to prioritize inspection activities; (2) improving regional capabilities for the use of PRA and risk insights; and (3) providing related inspector training." The staff has been active in accomplishing each of the three items as described in item (3) of Attachment 2.

- In a June 5, 1997, Staff Requirements Memorandum (SRM), the Commission requested information on the plans for training NRC staff on 1) the risk-informed regulatory approach(es) contained in the regulatory guidance and Standard Review Plan documents and 2) overall PRA methods and techniques. Attachment 4 provides the staff’s response to the SRM. The attachment describes the training that will be necessary to implement the initiatives discussed in the draft RGs and SRPs for risk-informed regulation.

- The staff has developed responses to all the public comments it has received on draft NUREG-1560 and where appropriate, draft NUREG-1560 has been revised. Attached for information are present drafts of the executive summary of NUREG-1560 (Attachment 5) and Appendix C of NUREG-1560 which documents resolution of comments from the public (Attachment 6). The final version of NUREG-1560 will be published in November 1997.

- A draft interim report has been developed that provides preliminary perspectives and summarizes the information presented in the first 24 Individual Plant Examination for External Events (IPEEE) submittals reviewed by the staff. This interim report will be sent to the Commission by the end of November 1997. A summary of the significant preliminary perspectives from the first 24 IPEEE reviews is presented in Attachment 7.

- In an SRM dated June 13, 1997, the Commission requested that the staff periodically report on their efforts to work with industry to address shortfalls and limitations in the data on reliability and availability of risk-significant systems to be provided to the staff voluntarily. The staff's quarterly report on this activity is provided in item (10) of Attachment 2.

- A three-day public workshop was held on August 11-13, 1997, on the following draft Regulatory Guides, Standard Review Plans, and NUREG report:

  - General Guidance (DG-1061 and SRP)
  - Inservice Testing (DG-1062 and SRP)
  - Graded Quality Assurance (DG-1064)
  - Technical Specifications (DG-1065 and SRP), and
  - The Use of PRA in Risk-Informed Applications (NUREG-1602)

  The workshop was well attended by industry representatives. They offered a number of constructive comments, some criticisms, and some suggestions for changing the guidance. Overall, the comments indicated general support for pursuing risk-informed regulation but in a manner which would necessitate modifications to the draft guidance. The significant issues raised at the workshop are summarized in item (1) of Attachment 2.

- In a letter to the NRC dated August 21, 1997, the Nuclear Energy Institute (NEI) made a proposal for three new risk-informed pilot applications of PRA in support of changes to the licensing basis of operating nuclear power plants. The staff met with NEI on September 17, 1997, to discuss the proposal, including potential NRC activities. The pilots would use a full scope PRA to assess risk versus the regulatory requirements and plant operating and maintenance costs. The staff has concluded that, in concept, the initiative is worthwhile and plans to meet with NEI in November to discuss plans for pursuing the initiative.

- In June, 1997, NRC staff met with representatives of the American Society of Mechanical Engineers (ASME) to discuss cooperation with both industry and professional societies to develop new codes and standards, as directed in the SRM on Direction Setting Issue (DSI) 13, dated March 7, 1997 (see SRM in Attachment 1). The development of PRA standards was one subject of this meeting. At the meeting ASME indicated their interest and is convening an ad hoc committee that will have the responsibility to develop such a standard. This committee will be comprised of ASME personnel, NRC staff, national laboratory, academic, and industry personnel.

  A charter for this committee is now being drafted, and will describe the goals and objectives of the committee, committee membership and associated responsibilities, schedules, and milestones. An addition, the charter will include anticipated scope of the standard (e.g., Level 1, 2 and 3 PRA, including core damage accidents initiated by internal and external events during full power operation) and the level of PRA modeling and analysis appropriate for different PRA uses. The Commission will be informed of progress on this development work in the quarterly updates of the PRA Implementation Plan.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objections to its issuance.

L. Joseph Callan
Executive Director
for Operations

Attachments:
As stated

DOCUMENT NAME: A:\SEP97_UP.WPD

To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	DST/PRAB	DST/PRAB	DD/DST	D/DST	NRR/SPSB
NAME	MCaruso:bgj	MCunningham	TKing	MWHodges	JFlack
DATE	/ /	/ /	/ /	/ /	/ /
OFFICE	NRR/DSSA	NRR/DE	NRR/DRCH	NRR/DISP	D/NRR
NAME	GHolahan	GLainas	LSpessard	FGillespie	BSheron
DATE	/ /	/ /	/ /	/ /	/ /
OFFICE	D/NRR	D/NMSS	AEOD	AEOD	AEOD
NAME	SCollins	CPaperiello	PBaranowsky	KRaglin	TMartin
DATE	/ /	/ /	/ /	/ /	/ /
OFFICE	OGC	D/RES	EDO	EDO	
NAME	STreby	MKnapp	AThadani	LJCallan	
DATE	/ /	/ /	/ /	/ /	/ /

OFFICIAL RECORD COPY
(RES File Code) RES 2C-5

Attachment 2

QUARTERLY STATUS UPDATE OF THE AGENCY-WIDE
IMPLEMENTATION PLAN FOR PROBABILISTIC RISK ASSESSMENT (PRA)
(from June 30, 1997 to September 30, 1997)

SUMMARY OF SIGNIFICANT PROGRESS

(1) Regulatory Guide (RG) and Standard Review Plan (SRP) Development (Tasks 1.1 and 2.1)

On April 8, 1997, the staff sent to the Commission SECY-97-077, "Draft Regulatory Guides, Standard Review Plans and NUREG Document In Support of Risk-Informed Regulation for Power Reactors." SECY-97-077 requested Commission approval to publish for comment four draft Regulatory Guides (RGs), three draft Standard Review Plan (SRP) sections, and one draft NUREG series report that support implementation of risk-informed regulation for power reactors. By Staff Requirements Memorandum (SRM) dated June 5, 1997, the Commission approved publication of the draft documents. A notice was placed in the Federal Register announcing availability of the documents and requesting public comment on them.

Public Workshop on Regulatory Guides and Standard Review Plans

To facilitate solicitation of public comments, the staff held a workshop on August 11, 12, and 13, 1997, at the DoubleTree Hotel in Rockville, Maryland to explain the draft documents and answer questions. The workshop was well attended by industry representatives. They offered a number of constructive comments, some criticisms, and some suggestions for changing the guidance. Overall, the comments indicated general support for pursuing risk-informed regulation but in a manner which would necessitate some modifications to the draft guidance. The more significant issues raised during the workshop regarding the general regulatory guidance included:

- how the guidelines on CDF and LERF would be applied when proposed increases in risk are very small;
- the conditions under which a full scope PRA would be necessary;
- what constitutes a "quality PRA" and the role of NUREG-1602 in judging the quality of the PRA supporting an application;
- having separate acceptance guidelines for accident sequences initiated during power operation and sequences initiated during low-power and shutdown operations;
- having new industry/NRC pilot programs to ensure the effectiveness of the guidance issued for use.

The staff is reviewing the comments provided at the workshop and those formal written public comments it has received.

Draft Regulatory Guide and Standard Review Plan for Inservice Inspection

The staff completed new drafts of the application-specific RG and SRP for inservice inspection (ISI) and discussed them with senior agency management, the Advisory Committee on Reactor Safeguards (ACRS) and the Committee to Review Generic Requirements (CRGR) in a number of meetings held over the past three months. Both the ACRS and the CRGR have completed their reviews of the guidance and concur with the staff's proposal to issue the guidance for comment by the public. On August 20, 1997, the staff sent to the Commission SECY-97-190, "Draft Regulatory Guide and Standard Review Plan on Risk-Informed Inservice Inspection of Piping." SECY-97-190 requested Commission approval to publish for comment the RG and SRP that supports implementation of risk-informed inservice inspection programs. Commission approval was obtained in an October 1, 1997, SRM. In completing the draft RG and SRP, the staff has found that a greater than expected effort was required to incorporate all points of view and gain a consensus on draft guidance. With this experience, the staff projects that the schedule for issuing the final ISI RG and SRP will slip from February 1998 to April 1998.

(2) Pilot Applications (Task 1.2)

For risk-informed ISI programs, the industry had identified three pilot plants that would submit requests for authorization for the use of risk-informed ISI methodology; these applications have not yet been received. In a letter from NEI, dated August 29, 1997, the industry requested to add two additional plants to the list of pilot applications and identified an aggressive schedule for all the pilot plants. The scope for the two new pilot plants is limited to Class-1 piping (primary coolant system piping only). To date, only one application has been submitted by one of the pilot plants with a limited scope RI-ISI program. Due to industry's delays in submitting the applications, and the addition of two plants as pilots, the staff is unable develop an integrated review schedule for the pilot plants at this time. This schedule is contingent on information regarding the actual timing of submittals, the quality of the submittals, and the ability of the pilot plant licensees to commit the resources necessary to respond to the staff's RAIs.

As noted in an August 21, 1997, memorandum to the Commission, completion of the RI-IST pilot plant safety evaluation has been delayed. Nevertheless, between July 14 and 18, 1997, the NRC staff and its contractors reviewed PRA models, backup calculations, and data at Comanche Peak Steam Electric Station (CPSES). The review was conducted as part of the staff's evaluation of Texas Utilities Electric Company's (TUE's) proposed RI-IST program and was aimed at determining whether the CPSES PRA is consistent with the quality and scope guidelines in draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis." While the review team identified some minor weaknesses with the CPSES PRA for the RI-IST application (e.g., missing success paths, limited documentation of human error probabilities, optimistic recovery factors for equipment repair, plant-specific performance data not having already been incorporated into the PRA), the review team feels that these issues can be addressed adequately by the licensee. The staff also identified an area in the calculation of sequence success that needs further clarification. The calculated core damage frequency from the licensee's base PRA will approach $1x10^{-4}$ per year when external event initiators and shutdown operations are taken into account. Thus, the licensee's proposed RI-IST program is receiving increased NRC technical and management review in accordance with guidance in DG-1061.

In response to staff questions and concerns, South Texas Project (STP) submitted for staff review another Operational QA Program (OQAP) revision, revised procedures for implementing facets of the graded QA program, a proposed Final Safety Analysis Report (FSAR) revision that would invoke 10CFR50.59 change controls on the GQA implementing procedures, and responses to staff information requests. The staff has prepared a safety evaluation for graded QA based on the reviews performed, which was sent to the Commission via a separate Commission paper in October 1997.

The staff has been working with the Combustion Engineering Owners Group (CEOG) to develop a TS administrative control for a configuration risk management program (CRMP). The CRMP constitutes the third tier of the three-tiered approach the staff has used in reviewing risk-informed TS allowed outage time (AOT) changes. As discussed in SECY 97-095, the staff is requiring licensees to incorporate a commitment to implement a CRMP in the TS as part of the basis for its approval of risk-informed TS AOT changes. Once the staff reaches agreement with the CEOG on a TS administrative control for the CRMP, and commitments are received from the individual pilot licensees, the staff will issue amendments to the lead plant and the other CE pilot licensees that have review results comparable to those for ANO-2, as discussed in SECY-97-085 and endorsed by the Commission in its May 28, 1997, SRM.

(3) <u>Inspections (Task 1.3)</u>

Significant PRA-related technical support has been provided for the agency's Maintenance Rule baseline inspections. As of September 30, 1997, the staff has performed 36 full inspections. These inspections were performed with the support of experienced staff and contractor personnel trained in the use of PRA, using an inspection procedure that focuses on the inspection and assessment of the relevant PRA-related technical aspects of the NRC-approved industry guideline for implementing the rule (i.e., NUMARC 93-01).

New technical guidance on the use of PRA in the power reactor inspection program has been issued with the revision of Inspection Manual Chapter 2515, Appendix C.

The NRC staff briefed the Commission in May 1997 on the Individual Plant Examination (IPE) insights report, NUREG-1560. As a result of this briefing, the staff received a SRM requesting the staff "to expedite activities in the following areas: (1) using IPE results to prioritize inspection activities; (2) improving regional capabilities for the use of PRA and risk insights; and (3) providing related inspector training." The staff has been active in accomplishing each of the three items as discussed below.

Since June 1995, briefings on IPEs have been made by the Office of Nuclear Regulatory Research (RES) staff to all four regional offices. To date, the majority of IPEs reviewed by the staff have been covered in the briefings. In addition, a detailed briefing (tailored for each region) of the results and insights from NUREG-1560 was presented at each region. These briefings (both types) have been attended by the resident inspectors, regional personnel, and plant inspection teams (where applicable). The briefings have been specifically structured to aid in prioritizing inspection activities, and to provide guidance on how to use PRA results. In many cases, Senior Reactor Analysts (SRA), on assignment in RES, participated in the preparation and presentation of the briefings as part of their developmental training. Consequently, the SRAs have gained a solid knowledge of the variety of information contained in a PRA.

The briefings on IPEs have provided valuable insights, particularly in plant-specific inspection activities. However, since many of the licensee's IPEs are out of date, inspectors will need to use supplementary information, including current licensee PRAs, as available,[1] to draw appropriate inspection insights. Consequently, plant-specific briefings based on the submitted IPEs have been discontinued. Instead, the SRAs in each Regional office and in NRR, who are now either fully trained and certified, or are in training, will continue to provide ongoing PRA advice for site-specific activities with support from headquarters offices as needed. Regional and headquarters SRA activities include: providing risk-based inspection prioritization, event assessment and inspection follow up, maintenance rule inspection support, inspection procedure guidance development, and maintenance of the SRA homepage on the NRC intranet.

(4) Accident Management (Task 1.9)

The staff review of the IPE submittals included an assessment of licensee responses to the requests in GL 88-20 and NUREG-1335 related to accident management. Based on IPE insights, the staff has not identified any areas where immediate industry actions related to accident management appear necessary. However, the following accident management areas raised in the IPE submittals warrant further staff evaluation:

- Inhibiting ADS in boiling water reactors (BWRs)
- Use of drywell sprays to prevent Mark I containment liner failure
- Preclude terminating injection to the reactor from external sources
- Effectiveness of external reactor vessel cooling

These follow-up items will be addressed in the staff's evaluation of the BWROG Emergency Procedure and Severe Accident Guidelines (EP/SAG) described in SECY-97-132.

(5) Evaluating IPE Insights To Determine Necessary Follow-up Activities (Task 1.10)

As part of finalizing NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," the staff has defined an initial set of follow-up activities. This initial set of activities consists of:

- Additional analysis of plants identified in NUREG-1560 as having risks approaching the Commission's quantitative health objectives (QHO's), based on a preliminary screening analysis. This additional analysis will use updated information and refined methods to make a better comparison with the QHOs;
- Analysis of plants with accident sequence frequencies greater than $1x10^{-5}$ per

[1] Since there is no requirement for licensees to submit or update PRAs, actual availability to staff in inspection or other activities is on a case-by-case basis. The staff will investigate options for addressing this issue.

reactor year and/or conditional containment failure probabilities greater than 0.1. The analysis will evaluate whether these plants have features which merit backfit consideration. This will be done in a manner consistent with the Safety Goal screening assessment in the Commission's Regulatory Analysis Guidelines;

- Analysis of selected generic issues which may merit further staff evaluation, such as:

  - Contributors to station blackout including grid unreliability

  - RCP seal LOCA and its associated contribution to core damage and large early release frequency,

  - Steam generator tube rupture;

- Follow-up on whether the actions licensees stated they were taking as a result of their IPE have, in fact, been taken;

- Follow-up on selected licensee responses to Containment Performance Improvement questions included in GL-88-20, Supplements 1 and 3.

In conjunction with this effort, the staff is developing a plan for audit of licensee-identified improvements credited in IPE analyses, to determine the effectiveness of licensee actions to reduce risk. The schedule for finalizing the list of items and a program plan to address those items is scheduled for completion in November 1997.

In a May 21, 1996, Staff Requirements Memorandum (SRM), the Commission requested that the staff track the regulatory uses of IPE/IPEEE results. Additionally, the Commission noted that consideration should be given to linking the resulting IPE/IPEEE databases together in a single, integrated, coherent program. This task was placed under item 1.10 of the PRA Implementation Plan in the October 1996 update, and a structure and linking process is under development. The staff will discuss the database content in the next quarterly Implementation Plan update. Due to other staff priorities, such as support for pilot applications and risk-informed regulatory documents, and delays in authorizing contract funds, the target schedule for defining uses for risk information, clarifying regulatory use, and assessing methods of data collection has been revised from December 1997 to May 1998.

(6) Methods Development and Demonstration (Task 2.4)

The Seabrook nuclear power plant is participating in the first trial PRA application of ATHEANA (A Technique for Human Event Analysis). ATHEANA is a human reliability analysis (HRA) method under development in RES which addresses errors of commission as well as omission. It focuses on combinations of performance shaping factors and plant conditions which increase the likelihood of certain human errors. In addition to identifying unsafe acts that will be considered for quantification within the plant PRA model, ATHEANA is showing promise for identifying strategies for improving human reliability.

In response to concerns over the influence of management and organizational factors, safety culture, and downsizing and deregulation on human performance and safe plant operations, RES held a workshop in August 1997 to discuss these issues with nationally and internationally recognized leaders in management and safety issues (including experts from academia, utilities, national laboratories, consulting companies, the NRC, DOE, and NASA). The experts presented papers and results of current research and participated in working sessions on these topics. The products of this workshop will be used to suggest research methods and/or to assess the influences of management and organizational factors, safety culture, and the effects of downsizing and deregulation.

(7) Individual Plant Examination (IPE) and IPE of Externally Initiated Events (IPEEE) Reviews (Task 2.5)

Status of IPE Reviews

The reviews of all 75 original IPE submittals (i.e., not including Browns Ferry, Unit 3) have been completed with a staff evaluation report (SER) issued by RES to NRR for each submittal. With the exception of the Crystal River and Susquehanna IPEs, all IPE submittals have now been found to meet the intent of Generic Letter 88-20. The licensees for Crystal River and Susquehanna plan to submit revised IPEs that would address the staff concerns. It is expected that these two revised IPEs will be submitted to the staff by December 1997 and staff review will be completed by June 1998.

Preliminary review of the recently submitted IPE for Browns Ferry, Unit 3, and responses to a staff request for additional information have been completed. It is expected that RES will issue its SER for the Browns Ferry, Unit 3, IPE by December 1997.

IPE Insights Report

In October 1996, the staff issued draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," for public comment. Comments were received from numerous licensees, individuals and other government organizations. Overall, the comments received were positive in nature. The staff has developed responses to all the comments received and, where appropriate, draft NUREG-1560 has been revised. Attached for information are the executive summary of NUREG-1560 (Attachment 5) and Appendix C of NUREG-1560 which documents resolution of comments from the public (Attachment 6). The final version of NUREG-1560 will be published in November 1997.

Status of IPEEE Reviews

Of the 74 expected IPEEE submittals, the staff has received 63; four of which were not complete. Currently, 49 submittals are under various stages of review. Nine additional submittals are expected to be received by the end of December 1997, one by June 1998, and the submittal date of one IPEEE has yet to be determined. The staff will complete all IPEEE reviews and Staff Evaluation Reports (SERs) by June 1999. Similar to the IPE program, the staff will take prompt action should any significant vulnerabilities or safety insights be identified in these reviews.

An interim report has been developed that provides preliminary IPEEE perspectives and summarizes the information presented in the first 24 IPEEE submittals reviewed by the staff. This interim report will be sent to the Commission by the end of November 1997. A summary of the significant preliminary perspectives from the first 24 IPEEE reviews is presented in Attachment 7. In addition, a draft report that summarizes the findings and perspectives from all IPEEE reviews will be sent to the Commission in June 1999 and released for public comment. After receipt and review of comments, the staff will issue the final IPEEE insights report in December 1999.

(8) Risk-Based Trends and Patterns Analysis Task (3.1)

The Common Cause Failure (CCF) database has been updated with events through 1995. The database and its associated technical reports are being provided on CD ROM to all nuclear utilities in accordance with the INPO agreement regarding distribution of NPRDS proprietary data. Initial draft reports for the initiating event update, loss of offsite power study, the auxiliary feedwater system study, and the Westinghouse reactor protection system study have been received by AEOD from its contractor and reviewed with the ACRS. In July 1997, the BWR high pressure core spray system draft report was distributed to NRC staff for internal peer review. Comments have been received and are being incorporated into the final report.

(9) Accident Sequence Precursor (ASP) (Task 3.2)

The last 1996 preliminary ASP analysis has been sent to the licensee for review. Four 1996 final analyses are completed and have been sent to the respective licensees and have been made publicly available. Seven precursor analyses are under licensee or AEOD final review. Events of significance, i.e., those with a conditional core damage probability (CCDP) greater than $1x10^{-4}$, include a Catawba event (loss of offsite power with diesel failure), a Wolf Creek event (frazil icing of the ultimate heat sink), a Seabrook event (long term unavailability of an emergency feedwater turbine-driven pump), a Prairie Island event (loss of offsite power to safeguards buses at both units), and a Haddam Neck event (potential inadequate residual heat removal pump net positive suction head following a medium or large loss of coolant accident).

(10) Compile Operating Experience Data (Task 3.5)

INPO submitted a revision to the Memorandum of Agreement with the NRC regarding access to the EPIX database. The EDO indicated his agreement with minor modifications and sent it to INPO on August 21, 1997, for signature.

The Sequence Coding and Search System (SCSS) conversion from a mainframe computer to a

PC-based computer has been completed as well as the beta testing of the new system. Direct access capability to SCSS via the Internet is now functional. Training and direct use for all NRC staff will be implemented by December 31, 1997.

(11) <u>Staff Training (Task 3.6)</u>

A new course, “PRA for Technical Managers,” has been added to the curriculum and two presentations were held in FY 1997. This course is designed to provide all levels of staff managers with a basic understanding of PRA methods, strengths, and limitations needed to implement risk-informed, performance-based regulations. Current plans are to present this course seven times in FY 1998 in headquarters.

PRA Level 2 and Level 3 courses have been added to the PRA curriculum. The first presentation of the new PRA Level 2 course, “Accident Progression Analysis,” was held in February 1997. This three-day course addresses accident phenomenology under post core damage conditions and development of PRA models for this severe-accident regime. Based on feedback from the first presentation of the course, the course is undergoing significant modification. The PRA Level 3 course, “Accident Consequence Analysis,” was “dry-run” in early 1997 and a first presentation was given in September 1997. The three-day course addresses environmental transport of radionuclides and the estimation of offsite consequences from core damage accidents. Current plans are to present each course twice a year.

A new course on external events has been completed. This three-day course addresses external events (such as fires, floods, earthquakes, high winds, and transportation accidents) and the development of external-event PRA models such as those used in the IPEEEs. The first presentation of this course was held August 5-7, 1997.

A new course, “PRA Technology and Regulatory Perspectives”, is under development and scheduled for first presentation in January 1998. A pilot presentation of the course was given on September 22-26, 1997. The course was originally scheduled to start in October 1997. However, based on the pilot presentation, further development and refinement of the course necessiated its delay to January 1998. The course will replace the PRA Basics for Regulatory Application course and the Insights Into IPEs course for some basic level users.

In a June 5, 1997, Staff Requirements Memorandum, the Commission requested information on the plans for training NRC staff on 1) the risk-informed regulatory approach(es) contained in the regulatory guidance and Standard Review Plan documents and 2) overall PRA methods and techniques. Attachment 4 of this Commission paper provides the staff’s response to the SRM, and describes the training that will be necessary to implement the initiatives discussed in the draft RGs and SRPs for risk-informed regulation.

REVISIONS TO THE EXISTING PRA IMPLEMENTATION PLAN

(1) Risk-Informed Regulatory Guides and Standard Review Plans (Tasks 1.1 & 2.1)

In completing the draft RG and SRP for risk-informed inservice inspection (ISI), the staff has found that a greater than expected effort was required to incorporate all points of view and gain a consensus on draft guidance. With this experience, the staff projects that the final ISI RG and SRP will be delayed from February 1998 to April 1998.

(2) Risk-Informed Pilot Applications (Task 1.2)

Inservice Inspection

With respect to the risk-informed ISI programs, the staff expected but has not received a formal submittal from the three pilot plants (Surry, ANO-2 and Fitzpatrick). Based on an NEI letter, dated August 29, 1997, the staff anticipates receiving pilot applications to implement RI-ISI programs through the winter of 1998. This includes applications from two new pilot plants (ANO-1 and Vermont Yankee). The staff continues to hold public working meetings with the industry and with Virginia Power on the Surry pilot, in anticipation of receipt of a formal application and to facilitate the development of regulatory guidelines.

Due to industry's delays in submitting the application for Fitzpatrick and ANO-2, and the addition of two plants as pilots, the staff is unable to develop an integrated review schedule for the pilot plants at this time. The schedule is contingent on information regarding the actual timing of submittals, the quality of the submittals, and the ability of the pilot plant licensees to commit the resources necessary to respond to the staff's RAIs.

With respect to the EPRI method (EPRI-TR-106706), the staff has not received responses to its RAIs. The EPRI method is used by all of the pilots except Surry. Delays in receiving responses to the staff's RAIs could also impact the schedule for the review of the pilot plants.

Inservice Testing

In a June 17, 1997, memorandum to the Commission, the staff stated that it expected to be able to issue the safety evaluation (SE) on the Comanche Peak RI-IST program in October 1997. The October 1997 completion date for the Comanche Peak SE was based on the assumption that TU Electric Company (TUE) would respond satisfactorily to both the second and final round requests for additional information by August 8, 1997. The staff's final round RAI asked the pilot plant licensee to describe how their proposed RI-IST program comports with the draft RI-IST RG and to explain their rationale for any differences.

In a letter to the NRC dated July 31, 1997 (amended by letter dated September 12, 1997), TUE stated they need additional time to determine how Comanche Peak's RI-IST program comports with the NRC draft guidance. TUE indicated that the resources required to complete the RAIs are also being used to provide support for (1) the third refueling outage of Comanche Peak Unit 2 in the fall of 1997, and (2) the NRC Maintenance Rule baseline inspection scheduled at Comanche Peak in October 1997. TUE plans to respond to the second and third round RAIs by September 30, 1997. This will delay issuance of the SE on the Comanche Peak RI-IST program until at least late December 1997.

In a letter to the NRC dated August 1, 1997, Arizona Public Service Company (APS) informed the staff that its resources must be diverted from the Palo Verde RI-IST program development effort in order to complete other activities (e.g., the 10 year IST program update and improved technical specification implementation). Therefore, due to the resource constraints and operational priorities discussed above, APS indicated that they will not be in a position to resume supporting the RI-IST implementation effort until mid-1998. At that time, APS will provide the NRC staff with a schedule for responding to the third RAI.

Graded Quality Assurance

Task 1.2 of the PRA Implementation Plan states that the target schedule for completing the graded QA safety evaluation (SE) for STP is July 1997. The South Texas Project (STP) is the only graded QA volunteer plant that submitted a revised graded QA program for staff review and approval. The staff has prepared safety evaluation for the STP program that will be transmitted to the Commission in a separate Commission paper in October 1997. Dialogue with STP on several issues as well as competing priorities for staff resources delayed completion of the safety evaluation from July to September 1997. Staff monitoring of activities at all three volunteer plants (STP, Grand Gulf, and Palo Verde) will continue in order to observe the results of equipment categorization for additional systems, and the results of the application of graded QA controls and to assess the integrity of the corrective action and operational performance feedback programs. This monitoring effort is expected to continue for an extended period (several years) to provide the staff with lessons learned.

For the purposes of the PRA Implementation Plan, this phase of volunteer plant interactions will be considered complete when the GQA RG and inspection procedure (IP) are issued in final form. In the future, the staff will continue to monitor the volunteer plant GQA implementation, gain feedback to revise the RG and IP as warranted, and evaluate GQA implementation strategies for other licensees who choose to pursue GQA. Although issuance of the SER for the STP GQA program is expected in October 1997, the completion date for the GQA pilot application remains March 1998 to reflect the expected schedule for issuance of the final GQA inspection procedure.

New Pilot Applications

In parallel with the NEI initiatives to study the risk and cost of regulated activities (see Task 2.7 “Whole-Plant” Risk Studies), the pilot plants will be submitting license amendment applications related to diesel generator start time and hydrogen control. Specific schedules will be established when each application is received.

(3) Inspections (Task 1.3)

As discussed just above, work has been initiated on an inspection procedure for Graded Quality Assurance (GQA). However, because of higher priority work on the South Texas GQA program safety evaluation, the schedule for completing a draft inspection procedure for GQA has been changed from September 1997 to December 1997 with completion of the final guide in March 1998.

Due to personnel being reassigned to higher priority activities, such as development of the PRA for Regulatory Applications course, the completion date for the review of core inspection procedures has been revised to October 1997.

(4) Application of IPE/IPEEE to Generic Issue Resolution (Task 1.6)

The completion target for identifying generic issues to be audited and selection of plants to be audited has been revised from "TBD" to December 1997.

(5) "Whole-Plant" Risk Study (Task 2.7-New)

In a letter to the NRC, dated August 21, 1997, the Nuclear Energy Institute (NEI) made a proposal for three new risk-informed pilot applications of PRA in support of changes to the licensing basis of operating nuclear power plants. The staff met with NEI on September 17, 1997, to discuss the proposal, including potential NRC activities. The pilots would use a full scope PRA to assess risk versus the regulatory requirements and plant operating and maintenance costs. The staff has concluded that, in concept, the initiative is worthwhile and plans to meet with NEI in November to discuss plans for pursuing the initiative. In parallel with the "whole-plant " risk studies, the pilot plant licensees will be submitting license amendment applications on issues such as diesel generator start time and hydrogen controls. These items will be tracked under PRA Implementation Plan Task 1.2 "Pilot Applications for Risk-Informed Regulatory Initiatives."

(6) PRA Standards Development (Task 2.8-New)

In June, 1997, NRC staff met with representatives of the American Society of Mechanical Engineers (ASME) to discuss cooperation with both industry and professional societies to develop new codes and standards, as directed in the SRM on Direction Setting Issue (DSI) 13, dated March 7, 1997 (see SRM in Attachment 1). The development of PRA standards was one subject of this meeting. At the meeting ASME indicated their interest and is convening an ad hoc committee that will have the responsibility to develop such a standard. This committee will be comprised of ASME personnel, NRC staff, national laboratory, academic, and industry personnel.

A charter for this committee is now being drafted, and will describe the goals and objectives of the committee, committee membership and associated responsibilities, schedules, and milestones. An addition, the charter will include anticipated scope of the standard (e.g., Level 1, 2 and 3 PRA, including core damage accidents initiated by internal and external events during full power operation) and the level of PRA modeling and analysis appropriate for different PRA uses. The Commission will be informed of progress on this development work in the quarterly updates of the PRA Implementation Plan.

(7) Low Power Shutdown Risk Reevaluation (Task 2.9-New)

RES has been assigned responsibility to further investigate methods for estimating the risk of severe accidents initiated during low power and shutdown operational states. The results of this investigation could include, for example, staff activities such as the development of new analysis methods or performance of experiments.

The staff intends to complete planning for this investigation in FY1988. Consistent with agency resources allocations, defined work will begin in FY1999.

(8) Revision of Safety Goal Policy Statement (Task 2.10-New)

SECY-97-208 discusses a number of issues relating to possible revision of the Safety Goal Policy Statement, including the possible elevation of core damage frequency to a fundamental safety goal. The staff recommended that additional discussions with ACRS be undertaken, with a goal of providing a Commission paper by March 31, 1998, which would include the staff's analysis, conclusions, and recommendations. This item has been inserted into the Implementation Plan; however no specific actions will be taken until the SRM on SECY-97-208 is received.

(9) Risk Based Trends and Patterns (Task 3.1)

The date for the component studies (Task 3.1) has been delayed because the cognizant engineer has been detailed to the Millstone Project. The dates for the systems studies have been delayed due to difficulties in applying models to the various system designs in a manner consistent with the reportability of failures and demands in multiple train systems. The delay in the initiating events update is due to difficulty in interpreting the extent of loss of offsite power and the nature of some initiating events from LERs.

(10) Accident Sequence Precursor Program (Task 3.2)

Schedules for development of low power/shutdown models and external events (earthquake and fire) models for use in the Accident Sequence Precursor Program are currently being revised to reflect NRC staff comments on the initial models and staff assigned to higher priority work.

(11) Risk-Based Performance Indicators (Task 3.4)

The delay in the development and implementation of risk-based performance indicators (Task 3.4) is due to the delays in the component and system studies. The outputs of these tasks serve as basic inputs for risk-based PIs.

(12) Risk Assessment of Material Uses (Task 4.4)

The work for developing PRA methods (Task 4.1) for use in evaluating medical devices containing nuclear material has been subsumed into the larger risk assessment of material uses (Task 4.4). A working group of NRC and Agreement States personnel has been chartered to:

- identify and document a technical basis for a risk-informed approach to the regulation of nuclear byproduct material, and

- develop plans for a graded regulatory approach for nuclear byproduct materials, based on risk information.

There was an initial meeting of the working group in mid-June 1997. Additional meetings were held in July, August, and September 1997 and are expected to continue about monthly through September 1998. Contractor support is planned to be available by November 1997 to assist the working group in its activities. The expected completion date of working group activities is September 1998.

(12) Nuclear Material Licensing and High-Level Waste Issues (Task 4.5)

In the SRM of April 15, 1997, about risk-informed, performance-based regulation (DSI-12) the Commission directed the staff to (1) reexamine its risk-informed, performance-based or risk-informed, less prescriptive (RIPB) approaches with regard to nuclear material licensees and to high-level waste issues, to ensure that the needs of those licensees and those areas receive adequate consideration; (2) review the basis for nuclear materials regulations and processes to identify and prioritize those areas that are or, with minimal additional staff effort and resources, could be made amenable to RIPB regulation; and (3) develop a framework for applying PRA to nuclear material uses, similar to the one developed for reactor regulation (SECY-95-280), where appropriate. In a paper that will be transmitted to the Commission in October 1997, the staff will reexamine preliminarily the RIPB approaches that it has identified in the PRA Implementation Plan, primarily those for nuclear materials licensees and high-level waste issues, but also those for low-level wastes, spent fuel storage facilities, and transportation (the other activities included in the PRA implementation plan). Also, the staff will identify preliminarily other NMSS areas that are or, with minimal resources, can be made amenable to RIPB approaches. Finally, the staff will provide a plan for developing a framework for applying RIPB approaches in NMSS regulation.

REVISED TASK TABLES

Attachment 3 provides updates to reflect the progress and revisions to the PRA Implementation Plan from July 1 to September 30, 1997.

ATTACHMENT -3
REVISED PRA IMPLEMENTATION PLAN
TASK TABLE (September 1997)

1.0 REACTOR REGULATION

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.1 DEVELOP STANDARD REVIEW PLANS FOR RISK-INFORMED REGULATION	Standard review plans for NRC staff to use in risk-informed regulatory decision-making.	* Evaluate available industry guidance. * Develop a broad scope standard review plan (SRP) chapters and a series of application specific standard review plan chapters that correspond to industry initiatives. * These SRPs will be consistent with the Regulatory Guides developed for the industry. * Draft SRPs transmitted to Commission to issue for public comment General IST ISI TS * Issue final SRP General IST ISI TS	 4/97C[2] 4/97C 8/97C 4/97C 12/97 12/97 4/98 12/97	NRR /RES

[1] C = Task Completed

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.2 PILOT APPLICATION FOR RISK-INFORMED REGULATORY INITIATIVES	* Evaluate the PRA methodology and develop staff positions on emerging, risk-informed initiatives, including those associated with: 1. Motor operated valves. 2. IST requirements. 3. ISI requirements. 4. Graded quality assurance. 5. Maintenance Rule. 6. Technical specifications. 6a. Commission Approval 6b. Pilot Amendments Issued 7. Other applications to be identified later. (applications related to diesel generator start times and Hydrogen Control are expected)	* Interface with industry groups. * Evaluation of appropriate documentation (e.g., 10 CFR, SRP, Reg Guides, inspection procedures, and industry codes) to identify elements critical to achieving the intent of existing requirements. * Evaluation of industry proposals. * Evaluation of industry pilot program implementation. * As appropriate, complete pilot reviews and issue staff findings on regulatory requests.	1. 2/96C 2. 12/97 (TUE) TBD (APS) 3. TBD 4. 3/98 5. 9/95C 6a. 5/97C 6b. 12/97	NRR
1.3 INSPECTIONS	* Provide guidance on the use of plant-specific and generic information from IPEs and other plant-specific PRAs.	* Develop IC 9900 technical guidance on the use of PRAs in the power reactor inspection program.	6/97C	NRR
		* Revise IC 2515 Appendix C on the use of PRAs in the power reactor inspection program.	7/97 C	
		* Propose guidance options for inspection procedures related to 50.59 evaluations and regular maintenance observations.	10/97	
		* Review core inspection procedures and propose PRA guidance where needed.	10/97	
		* Complete revision to proposed core inspection procedures	12/97	
		* Issue draft Graded QA Inspection Procedure for public comment	12/97	
		* Issue final Graded QA Inspection Procedure	3/98	
	* Provide PRA training for inspectors.	* Identify inspector functions which should utilize PRA methods, as input to AEOD/TTD for their development and refinement of PRA training for inspectors.	7/96C	NRR
		* Develop consolidated/comprehensive 2-3 week PRA for regulatory applications training course.	10/97	NRR/ AEOD
		* Conduct training for Maintenance Rule baseline inspections	8/96C	NRR
	* Provide PRA training for Senior Reactor Analysts (SRA)	* Conduct training courses according to SRA training programs	Ongoing	NRR/RES
		* Rotational assignments for SRAs to gain working experience	Ongoing	

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
	* Continue to provide expertise in risk assessment to support regional inspection activities and to communicate inspection program guidance and examples of its implementation.	* Monitor the use of risk in inspection reports. * Develop new methodologies and communicate appropriate uses of risk insights to regional offices. * Update inspection procedures as needed. * Assist regional offices as needed. * Conduct Maintenance Rule baseline inspections	Ongoing 7/98	NRR
1.4 OPERATOR LICENSING	Monitor insights from HRAs and PRAs (including IPEs and IPEEEs) and operating experience to identify possible enhancements for inclusion in planned revisions to guidance for operator licensing activities (initial and requalification)	* Revise the Knowledge and Abilities (K/A) Catalogs (NUREGs 1122 and 1123) to incorporate operating experience and risk insights. * Revise the Examiner Standards (NUREG-1021), as needed, to reflect PRA insights.	8/95C 3/97C	NRR NRR
1.5 EVENT ASSESSMENT	* Continue to conduct quantitative event assessments of reactor events while at-power and during low power and shutdown conditions.	* Continue to evaluate 50.72 events using ASP models.	Ongoing	NRR
	* Assess the desirability and feasibility of conducting quantitative risk assessments on non-power reactor events.	* Define the current use of risk analysis methods and insights in current event assessments. * Assess the feasibility of developing appropriate risk assessment models. * Develop recommendations on the feasibility and desirability of conducting quantitative risk assessments.	TBD	NRR
1.6 EVALUATE USE OF PRA IN RESOLUTION OF GENERIC ISSUES	* Audit the adequacy of licensee analyses in IPEs and IPEEEs to identify plant-specific applicability of generic issues closed out based on IPE and IPEEE programs.	* Identify generic safety issues to be audited. * Select plants to be audited for each issue. * Describe and discuss licensees' analyses supporting issue resolution. * Evaluate results to determine regulatory response; i.e., no action, additional audits, or regulatory action.	12/97 12/97 TBD TBD	NRR/RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.7 REGULATORY EFFECTIVENESS EVALUATION	* Assess the effectiveness of major safety issue resolution efforts for reducing risk to public health and safety.	* Develop process/guidance for assessing regulatory effectiveness. * Apply method to assess reduction in risk. * Evaluate result, effectiveness of rules. * Propose modifications to resolution approaches, as needed (SBO rule implementation and RCP seal issue). * Identify other issues for assessment if appropriate.	ongoing ongoing ongoing ongoing TBD	NRR & RES
1.8 ADVANCED REACTOR REVIEWS	* Continue staff reviews of PRAs for design certification applications.	* Continue to apply current staff review process.	Ongoing	NRR
	* Develop SRP to support review of PRAs for design certification reviews of evolutionary reactors (ABWR and System 80+).	* Develop draft SRP to tech staff for review and concurrence. * Finalize SRP.	6/98 12/99	NRR
	* Develop independent technical analyses and criteria for evaluating industry initiatives and petitions regarding simplification of Emergency Preparedness (EP) regulations.	* Reevaluate risk-based aspects of the technical bases for EP (NUREG-0396) using insights from NUREG-1150, the new source term information from NUREG-1465, and available plant design and PRA information for the passive and evolutionary reactor designs.	12/96C	NRR & RES
1.9 ACCIDENT MANAGEMENT	* Develop generic and plant specific risk insights to support staff audits of utility accidents management (A/M) programs at selected plants.	* Develop plant-specific A/M insights/information for selected plants to serve as a basis for assessing completeness of utility A/M program elements (e.g., severe accident training)	TBD	NRR & RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.10 EVALUATING IPE INSIGHTS TO DETERMINE NECESSARY FOLLOW-UP ACTIVITIES	* Use insights from the staff review of IPEs to identify potential safety, policy, and technical issues, to determine an appropriate course of action to resolve these potential issues, and to identify possible safety enhancements.	* Review the report "IPE Program: Perspectives on Reactor Safety and Plant Performance" and identify the initial list of required staff and industry actions (if any), including insights on A/M.	9/97C	NRR & RES
		Finalize list of required staff and industry actions.	11/97	
		* Audit licensee improvements that were credited in the IPEs to determine effectiveness of licensee actions to reduce risk.	TBD	NRR
	* Determine appropriate approach for tracking the regulatory uses of IPE/IPEEE results.	* Define use for information, clarify "regulatory use", and assess the most effective methods for data collection.	5/98	
		* If appropriate, develop approach for linking IPE/IPEEE data bases.	12/98	

2.0 REACTOR SAFETY RESEARCH

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
2.1 DEVELOP REGULATORY GUIDES	Regulatory Guides for industry to use in risk-informed regulation.	* Draft PRA Regulatory Guides transmitted to Commission for approval to issue for public comment. General IST ISI GQA TS * Issue final PRA Regulatory Guides. General IST ISI GQA TS	 C C C C C 12/97 12/97 4/98 12/97 12/97	RES
2.2 TECHNICAL SUPPORT	* Provide technical support to agency users of risk assessment in the form of support for risk-based regulation activities, technical reviews, issue risk assessments, statistical analyses, and develop guidance for agency uses of risk assessment.	* Continue to provide ad hoc technical support to agency PRA users. * Expand the database of PRA models available for staff use, expand the scope of available models to include external event and low power and shutdown accidents, and refine the tools needed to use these models, and continue maintenance and user support for SAPHIRE and MACCS computer codes. * Support agency efforts in reactor safety improvements in former Soviet Union countries.	Continuing Continuing Continuing	RES RES RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
2.3 SUPPORT FOR NRR STANDARD REACTOR PRA REVIEWS	* Modify 10 CFR 52 and develop guidance on the use of updated PRAs beyond design certification (as described in SECY 93-087).	* Develop draft guidance and rule. * Solicit public comment. * Finalize staff guidance and rule.	5/98 11/98 12/99	RES RES RES
2.4 METHODS DEVELOPMENT AND DEMONSTRATION	* Develop, demonstrate, maintain, and ensure the quality of methods for performing, reviewing, and using PRAs and related techniques for existing reactor designs.	* Develop and demonstrate methods for including aging effects in PRAs. * Develop and demonstrate methods for including human errors of commission in PRAs. * Develop and demonstrate methods to incorporate organizational performance into PRAs. * Develop and demonstrate methods for fire risk analysis * Develop and demonstrate methods for assessing reliability/risk of digital systems	9/98 9/98 TBD 9/98 6/99	RES RES RES RES RES
2.5 IPE AND IPEEE REVIEWS	* To evaluate IPE/IEEE submittals to obtain reasonable assurance that the licensee has adequately analyzed the plant design and operations to discover vulnerabilities; and to document the significant safety insights resulting from IPE/IPEEEs.	* Complete reviews of IPE submittals. * Complete reviews of IPEEE submittals. * Continue regional IPE presentations. * Issue IPE insights report for public comment. * Final IPE insights report * Issue preliminary IPEEE insights report * Issue draft final IPEEE insights report	9/97 6/99 C 10/96C 9/97 11/97 12/99	RES RES RES RES RES RES RES
2.6 GENERIC ISSUES PROGRAM	* To conduct generic safety issue management activities, including prioritization, resolution, and documentation, for issues relating to currently operating reactors, for advanced reactors as appropriate, and for development or revision of associated regulatory and standards instruments.	* Continue to prioritize and resolve generic issues.	Continuing	RES
2.7 NEI INITIATIVE TO CONDUCT "WHOLE PLANT" RISK STUDY	* Review NEI initiative to conduct three pilot "whole plant" risk-informed studies of requirements vs. risk and cost	* Agree on ground rules for study * Complete study	1/98 TBD	RES/NRR TBD
2.8 PRA STANDARDS DEVELOPMENT	* work with industry to develop national consensus standard for PRA scope and quality	* Initiate activity * Finalize standard	9/97C TBD	RES
2.9 LOW POWER AND SHUTDOWN BENCHMARK RISK STUDY	*Collect studies of LP&S risk as a benchmark for assessing the need for further staff activities	*Collect and review existing LP&S risk information (domestic and foreign) *Initiate additional work	9/98 10/98	RES
2.10 SAFETY GOAL REVISION	*Assess need to revise Commission's Safety Goal to make core damage frequency a funamental goal and make other changes	*Initiate discussion with ACRS	TBD	RES

3.0 ANALYSIS AND EVALUATION OF OPERATING EXPERIENCE, AND TRAINING

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office
3.1 RISK-BASED TRENDS AND PATTERNS ANALYSIS	* Use reactor operating experience data to assess the trends and patterns in equipment, systems, initiating events, human performance, and important accident sequence.	* Trend performance of risk-important components. * Trend performance of risk-important systems. * Trend frequency of risk-important initiating events. * Trend human performance for reliability characteristics.	12/98 12/98 3/98 TBD	AEOD
	* Evaluate the effectiveness of licensee actions taken to resolve risk significant safety issues.	* Trend reactor operating experience associated with specific safety issues and assess risk implications as a measure of safety performance.	As Needed	AEOD
	* Develop trending methods and special databases for use in AEOD trending activities and for PRA applications in other NRC offices.	* Develop standard trending and statistical analysis procedures for identified areas for reliability and statistical applications. * Develop special software and databases (e.g. common cause failure) for use in trending analyses and PRA studies.	C CCF-C Periodic updates	AEOD
3.2 ACCIDENT SEQUENCE PRECURSOR (ASP) PROGRAM	* Identify and rank risk significance of operational events.	* Screen and analyze LERs, AITs, IITs, and events identified from other sources to obtain ASP events. * Perform independent review of each ASP analyses. Licensees and NRC staff peer review of each analysis. * Complete quality assurance of Rev. 2 simplified plant specific models. * Complete feasibility study for low power and shutdown models. * Complete initial containment performance and consequence models. * Complete development of the Level 2/3 models * Complete the Rev. 3 simplified plant-specific models. * Complete external event models for fire and earthquake * Complete low power/shutdown models	Ongoing Annual report, Ongoing 3/97C 11/96C C 7/99 11/01 TBD TBD	AEOD AEOD RES RES RES RES RES RES RES
	* Provide supplemental information on plant specific performance.	* Share ASP analyses and insights with other NRC offices and Regions.	Annual rpt	AEOD

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office
3.3 INDUSTRY RISK TRENDS	* Provide a measure of industry risk that is as complete as possible to determine whether risk is increasing, decreasing, or remaining constant over time.	* Develop program plan which integrates NRR, RES, and AEOD activities which use design and operating experience to assess the implied level of risk and how it is changing. * Update plan for risk-based analysis of reactor operating experience * Implement program plan elements which will include plant-specific models and insights from IPEs, component and system reliability data, and other risk-important design and operational data in an integrated frame work to periodically evaluate industry trends.	C 3/98 6/99	AEOD
3.4 RISK-BASED PERFORMANCE INDICATORS	* Establish a comprehensive set of performance indicators and supplementary performance measures which are more closely related to risk and provide both early indication and confirmation of plant performance problems.	* Identify new or improved risk-based PIs which use component and system reliability models & human and organizational performance evaluation methods. * Develop and test candidate PIs/performance measures. * Implement risk-based PIs with Commission approval.	C 9/00 1/01	AEOD
3.5 COMPILE OPERATING EXPERIENCE DATA	* Compile operating experience information in database systems suitable for quantitative reliability and risk analysis applications. Information should be scrutable to the source at the event level to the extent practical and be sufficient for estimating reliability and availability parameters for NRC applications.	* Manage and maintain SCSS and the PI data base, provide oversight and access to NPRDS, obtain INPO's SSPI, compile IPE failure data, collect plant-specific reliability and availability data. * Develop, manage, and maintain agency databases for reliability/availability data (equipment performance, initiating events, CCF, ASP, and human performance data). * Determine need to revise LER rule to eliminate unnecessary and less safety-significant reporting. * Determine need to revise reporting rules and to better capture ASP, CCF, and human performance events. * Publish revised LER rule.	Ongoing Ongoing 6/98 6/98 10/99	AEOD

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
3.6 STAFF TRAINING	* Present PRA curriculum as presently scheduled for FY 1996	* Continue current contracts to present courses as scheduled. * Maintain current reactor technology courses that include PRA insights and applications. * Improve courses via feedback. * Review current PRA course material to ensure consistency with Appendix C.	Ongoing Ongoing Ongoing Complete	AEOD
	* Develop and present Appendix C training courses.	* Prepare course material based on Appendix C. * Present courses on Appendix C.	C C	RES and AEOD
	* Determine staff requirements for training, including analysis of knowledge and skills, needed by the NRC staff.	* Review JTAs performed to date. * Perform representative JTAs for staff positions (JTA Pilot Program). * Evaluate staff training requirements as identified in the PRA Implementation Plan and the Technical Training Needs Survey (Phase 2) and incorporate them into the training requirements analysis. * Analyze the results of the JTA Pilot Program and determine requirements for additional JTAs. * Complete JTAs for other staff positions as needed. * Solicit a review of the proposed training requirements. * Finalize the requirements.	C C C C C C Ongoing	AEOD
	* Revise current PRA curriculum and develop new training program to fulfill identified staff needs.	* Prepare new courses to meet identified needs. * Revise current PRA courses to meet identified needs. * Revise current and New PRA course to include RegGuide and SRP information * Revise current reactor technology courses as necessary to include additional PRA insights and applications.	Ongoing Ongoing 9/97C Ongoing	AEOD
	* Present revised PRA training curriculum.	* Establish contracts for presentation of new PRA curriculum. * Present revised reactor technology courses. * Improve courses based on feedback.	Ongoing Ongoing Ongoing	AEOD

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
4.1 Validate risk analysis methodology developed to assess most likely failure modes and human performance in the use of industrial and medical radiation devices.	* Validate risk analysis methodology developed to assess the relative profile of most likely contributors to misadministration for the gamma stereotactic device (gamma knife).	* Hold a workshop consisting of experts in PRA and HRA to examine existing work and to provide recommendations for further methodological development. * Examine the use of Monte Carlo simulation and its application to relative risk profiling. * Examine the use of expert judgement in developing error rates and consequence measures.	8/94 C 9/95 C 9/95 C	NMSS
	* Continue the development of the relative risk methodology, with the addition of event tree modeling of the brachytherapy remote afterloader.	* Develop functionally based generic event trees.	TBD	RES/ NMSS
	* Extend the application of the methodology and its further development into additional devices, including teletherapy and the pulsed high dose rate afterloader.	*Develop generic risk approaches.	TBD	RES/ NMSS
4.2 Continue use of risk assessment of allowable radiation releases and doses associated with low-level radioactive waste and residual activity.	* Develop decision criteria to support regulatory decision making that incorporates both deterministic and risk-based engineering judgement.	* Conduct enhanced participatory rulemaking to establish radiological criteria for decommissioning nuclear sites; technical support for rulemaking including comprehensive risk based assessment of residual contamination. *Develop guidance for implementing the radiological criteria for license termination.. * Work with DOE and EPA to the extent practicable to develop common approaches, assumptions, and models for evaluating risks and alternative remediation methodologies. (Risk harmonization).	8/94 PR C Final Rule Published 7/97 C 2/98 Ongoing	RES & NMSS
4.3 Develop guidance for the review of risk associated with waste repositories.	* Develop a Branch Technical Position on conducting a Performance Assessment of a LLW disposal facility.	* Solicit public comments * Publish final Branch Technical Position	5/97 C. TBD, Dependent on Resources	NMSS & RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
4.4 Risk assessment of material uses.	* Develop and demonstrate a risk assessment for industrial gauges containing cesium-137 and cobalt-60 using PRA and other related techniques. * The assessment should allow for modification based on changes in regulatory requirements. * Use empirical data as much as practicable. * Develop and demonstrate risk assessment methods for application to medical and industrial licensee activities.	* Develop and demonstrate methods for determining the risk associated with industrial gauges containing cesium-137 and cobalt-60. * Final report as NUREG *Working Group with contractor assistance to identify and document a technical basis for a risk-informed approach to the regulation of nuclear byproduct material, and to develop plans for a graded approach to nuclear byproduct material regulation based on risk informaiton.	7/98 10/98	
4.5 Framework for Use of PRA in Regulating Nuclear Materials	* develop a framework for applying PRA to nuclear material uses, similar to the one developed for reactor regulation (SECY-95-280), where appropriate.	*Provide plan for developing Framework *Complete Framework	10/97 TBD	NMSS

5.0 HIGH-LEVEL NUCLEAR WASTE REGULATION

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
5.1 REGULATION OF HIGH-LEVEL NUCLEAR WASTE	* Develop guidance for the NRC and CNWRA staffs in the use of PA to evaluate the safety of HLW programs.	* Assist the staff in pre-licensing activities and in license application reviews. * Develop a technical assessment capability in total-system and subsystem PA for use in licensing and pre-licensing reviews. * Combine specialized technical disciplines (earth sciences and engineering) with those of system modelers to improve methodology.	Ongoing	NMSS
	* Identify significant events, processes, and parameters affecting total system performance.	* Perform sensitivity studies of key technical issues using iterative performance assessment (IPA).	Ongoing	NMSS
	* Use PA and PSA methods, results and insights to evaluate proposed changes to regulations governing the potential repository at Yucca Mountain.	* Assist the staff to maintain and to refine the regulatory structure in HLW disposal regulations that pertain to PA. * Apply IPA analyses to advise EPA in its development of a Yucca Mountain regulation * Apply IPA analyses to develop a site-specific regulation for a Yucca Mountain site	Ongoing	NMSS
	* Continue PA activities during interactions with DOE during the pre-licensing phase of repository development, site characterization, and repository design.	* Provide guidance to the DOE on site characterization requirements, ongoing design work, and licensing issues important to the DOE's development of a complete and high-quality license application. * Compare results of NRC's iterative performance assessment to DOE's VA to identify major differences/issues.	Ongoing	NMSS
5.2 APPLY PRA TO SPENT FUEL STORAGE FACILITIES	* Demonstrate methods for PRA of spent fuel storage facilities.	* Prepare user needs letter to RES * Conduct PRA of dry cask storage	4/97C 9/99	RES/NMSS
5.3 CONTINUE USE OF RISK ASSESSMENT IN SUPPORT OF RADIOACTIVE MATERIAL TRANSPORTATION	* Use PRA methods, results, and insights to evaluate regulations governing the transportation of radioactive material.	* Update the database on transportation of radioactive materials for future applications * Revalidate the results of NUREG-0170 for spent fuel shipment risk estimates	End of FY 99 6/99	NMSS

Attachment 4

PRA Training to Support Risk-Informed Regulatory Initiatives

BACKGROUND:

In SECY-97-077, dated April 8, 1997, the staff requested Commission approval to publish risk-informed regulatory guides and standard review plans for public comment. In a June 5, 1997, Staff Requirements Memorandum (SRM), the Commission approved publication of these draft documents and directed the staff to "provide the Commission information on its plans for training NRC staff 1) on the risk-informed regulatory approach(es) contained in the regulatory guidance and standard review plan documents and 2) in overall PRA methods and techniques." The Commission noted that "particular attention should be given to increasing basic user-level knowledge of probabilistic risk assessment (PRA) methods at the regional level." The staff's response to this SRM is provided below.

DISCUSSION:

Risk-informed regulation uses data and insights derived from probabilistic risk analyses to complement and support the traditional engineering analysis approach. To support risk-informed licensing decisions, it is essential that the staff and inspectors be familiar with Commission policy and expectations as well as various aspects of PRA analysis methodologies and results. These aspects include, but are not limited to, strengths and limitations of PRA analysis, the scope of PRA analyses, the use of importance measures, and the effects and sources of uncertainty. Furthermore, the staff also must be familiar with the regulatory framework being established to support risk-informed applications from industry. With these objectives in mind, the staff has designed specific minimum mandatory training programs for technical staff in the Office of Nuclear Reactor Regulation (NRR), inspectors in both NRR and the Regional Offices and NRR, and Regional technical managers that are discussed below.[1]

NRR Technical Staff

All NRR technical staff will be required to attend a newly developed seminar on responsibilities associated with risk-informed regulation. This seminar will orient the staff on the uses risk-informed regulatory initiatives and will be led by a NRR senior manager. The seminar covers the PRA Policy Statement, the scope of risk-informed regulation, staff expectations, responsibilities and acceptance criteria.

All NRR technical staff will be required to complete the four day "PRA Basics for Regulatory

[1]With much of the initial focus of risk-informed regulatory activities being on reactor applications, the initial training focus has been on NRR and regional staff. Training programs for managers and technical staff in other offices are still under development.

Applications" (P-105) course or its equivalent.[2] The Technical Training Division (TTD) of the Office of Analysis and Evaluation of Operational Data (AEOD) has updated this course to include information contained in the risk-informed RGs and SRPs. The target schedule for completion of this training is the end of fiscal year 1999. The staff is currently evaluating resource needs to meet this target schedule.

NRR and Region-Based Inspectors

Regional and NRR Inspectors associated with the regulation of power reactors will be required to complete the "PRA Technology and Regulatory Perspectives" (P-111) course. This is a new basic user PRA course targeted to the specific needs of inspectors. The course curriculum includes extensive practical workshops and case studies applicable to the needs of the inspector. The first offering of this course is scheduled to begin in October 1997. Resident inspectors will be given the highest attendance priority with the goal of having at least one resident at every site complete the training by the end of 1998.

NRR and Regional Technical Managers

Regional and NRR Technical Managers associated with the regulation of power reactors will be required to complete the three day "PRA for Technical Managers" course (P-107). TTD has updated this course to include information contained in the risk-informed regulatory guides and standard review plans. Seven sessions of P-107 are scheduled for fiscal year 1998 and will be sufficient to train two thirds of the agency's technical managers. Additionally, sufficient courses will be available during fiscal year 1999 to permit remaining technical managers, associated with the regulation of power reactors, to complete P-107 by the end of fiscal year 1999. Senior management will establish attendance priority as required to support implementation of risk-informed regulatory activities.

Additional Agency-Wide Technical Training

The training plan described above will provide sufficient training to support implementation of the risk-informed RGs and SRPs; but due to resource limitations, it will not provide staff with all of the basic user level courses and prerequisites recommended in NUREG/BR-0228, "Guidance for Professional Development of NRC Staff in Regulatory Risk Analysis." Consequently, if additional PRA training is needed to support specific risk-informed regulatory applications, NRC managers will be expected to define such training for their staff. For advanced users of PRA, the NRC's current PRA training curricula includes eleven advanced technology courses.

---

[2]NRR technical staff members who have completed basic user level PRA training within the last three years will be exempted from requirement to complete the P-105 course. To ensure that these staff members receive adequate training on the risk-informed documents, they will be required to receive training based on the newly developed risk-informed modules in addition to the risk-informed regulation seminar.

Summary of Results of IPEEE Reviews

On June 28, 1991, the NRC issued Supplement No. 4 to Generic Letter (GL) 88-20, which described the objectives and overall logistics of the Individual Plant Examination of External Events (IPEEE) program, for the evaluation of external events including seismic events, internal fires, and high winds, floods, and other (HFO) external initiators. The primary goal of the IPEEE program has been for licensees to "identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements."

In addition to the principal intent of GL 88-20, the four supporting IPEEE objectives have been for each licensee to:

1. develop an appreciation of severe accident behavior;

2. understand the most likely severe accident sequences that could occur at the licensee's plant under full power operating conditions;

3. gain a qualitative understanding of the overall likelihood of core damage and fission product releases;

4. reduce, if necessary, the overall likelihood of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The recommended guidelines of NUREG-1407, "Procedural and Submittal Guidance for the IPEEE for Severe Accident Vulnerabilities," and Supplement 5 to GL 88-20 have been developed by the NRC to help ensure that each of these objectives is met.

Preliminary perspectives have been obtained from ongoing technical reviews of 24 IPEEE studies that have been submitted by licenses. These results primarily include: (a) summaries of findings reported in IPEEEs, and (b) perspectives on strengths and weaknesses of licensee submittals in achieving the IPEEE objectives. The IPEEE program appears to have been generally successful in meeting the overall intent of GL 88-20. However, the degree and consistency of such success have varied considerably from study to study, and have been strongly dependent on the level of detail, and methods and assumptions employed in the IPEEE analyses.

Based on the continuing review of the first 24 IPEEE submittals, it appears that the IPEEE program has led to an increase in overall licensee appreciation of severe accident behavior for external events. As requested in NUREG-1407, each IPEEE has involved a seismic evaluation, an analysis of internal fires, and an assessment for HFO events. These evaluations have assessed the potential for externally initiated severe accidents, and have evaluated plant-specific behavior in responding to potential severe accidents.

Consistent with the guidance of NUREG-1407, the emphasis in conducting IPEEEs has been on obtaining a qualitative, as opposed to quantitative, understanding. As expected, therefore, the IPEEEs do not generally convey a definitive ranking of the risk-significance of severe accident sequences or of the dominant risk contributors. Rather, by means of systems modeling and screening analysis, licensees have obtained a greater awareness of severe accident sequences and an improved sense as to the most important among those sequences.

By means of IPEEEs, licensees have been able to generally ascertain whether the risk of core damage associated with each external initiator is comparatively negligible (i.e., falling below the 10^{-6} screening threshold), low, moderate, or high. In some cases, this understanding occurred through direct quantification of core damage frequency (CDF), whereas in other cases, this understanding resulted from having knowledge of the hazard in conjunction with an assessment of the plant's ability to withstand the hazard.

It is important to note that, although in many cases licensees have reported numerical risk estimates (for CDF or frequency of significant radiological releases), the accuracy of such estimates is frequently limited due to simplifying assumptions and approximate procedures employed in the analyses. Hence, the results serve only as general indicators of risk level, and a comparison of CDF results between plants is not particularly meaningful.

Based on the first 24 IPEEE submittals, a majority of licensees have implemented or proposed plant modifications that have a beneficial effect on plant safety with respect to external events. Such plant modifications include hardware changes, procedural changes, and implementation of severe accident management guidelines. Consistent with the qualitative nature of the IPEEE program, it is not usually possible to deduce the numerical risk reductions achieved by these modifications. However, some licensees have employed PRA in their IPEEEs as a means for determining whether or not plant modifications are warranted based on cost-benefit rationale.

Licensees have in most cases followed the guidance in NUREG-1407 in performing their IPEEE assessments. The guidance permits alternative methodologies. For example, there are various approaches for the seismic evaluation, including; seismic margin assessment (SMA) using the NRC methodology, SMA using the Electric Power Research Institute (EPRI) methodology, or seismic PRA methodology. Out of the 24 IPEEE submittals reviewed to date, thirteen have been based on seismic PRA methodology, whereas seven have been performed using the EPRI SMA methodology; one has adopted both the seismic PRA and EPRI SMA approaches; one has been based on the NRC SMA methodology; and two have been performed using a site-specific seismic evaluation approach, in consideration of the "Optional Methodologies" provision of NUREG-1407.

NUREG-1407 has also allowed for the implementation of alternative approaches for the evaluation of internal fires and HFO events. For fire IPEEE evaluations, licensees have implemented EPRI's fire-induced vulnerability evaluation (FIVE) methodology, fire PRA methodology, or a combination of these approaches. For evaluation of HFO events, the licensees either demonstrated that criteria of the NRC's 1975 Standard Review Plan (SRP) were met, or conducted at least one of the following forms of analysis: screening assessment, bounding analysis, or PRA methodology.

These observations highlight an important fundamental difference between the IPE process and the IPEEE process. In the IPE process, comparatively detailed PRA investigation has been invariably implemented, whereas in the IPEEE process, a mix of deterministic methods and detailed PRA investigations has been applied, as well as a mix of screening analyses, simplified hazard-based analyses, and/or bounding PRA-based approaches.

Licensees have performed or proposed various IPEEE-related improvements for their plants in seismic, fire, and/or HFO events. In the seismic area, hardware fixes have included items such as: anchoring equipment, bolting cabinets together, improving existing anchorage or supports, installing missing fastener and bolts, installing spacers on battery racks, eliminating potential interaction concerns, and replacing vulnerable relays. Maintenance actions have included the removal of corrosion on equipment anchorages and application of corrosion protection. Maintenance procedural enhancements have included provisions for proper storage of ladders, tools, gas cylinders, etc., and for proper parking of cranes and chain hoists.

Improvements to fire protection systems include hardware modifications and enhancements to, or development of, fire-response procedures. Additionally, improvements have often taken the form of severe accident management guidelines that address specific accident scenarios related to internal fires, potential effects of wind-induced missiles, and external flooding. Implementation of some of the severe accident management guidelines has led to the acquisition or access of temporary or portable equipment (pumps, diesel oil tanker trucks, etc.). One HFO IPEEE reported the strengthening of the stacks of two adjacent fossil-fuel units to reduce the high-wind risk, and refurbishment of a flood wall to reduce flood risk. The IPEEEs have also, in some cases, referenced plant improvements that had been made (or proposed) prior to the IPEEE, since those improvements resulted in a beneficial effect on plant safety in the face of seismic, fire, and/or HFO events. For example, at one plant, the addition of diesel generators was identified as a plant improvement in the IPE, and was correspondingly reported in the IPEEE since it reduces the risk of station blackout for seismic, fire, and HFO events.

A number of important perspectives and insights have been derived from the NRC's overall review activities pertaining to IPEEE submittals. Some of these key observations are described in the following paragraphs for the seismic, fire and HFO aspects of the IPEEE program. It was stated previously that the intent of GL 88-20 appears to be broadly met by the IPEEE submittals; however, the quality of the submittals has varied considerably from plant to plant. Some of the weakness in the submittals are discussed below. When these weakness have been observed during the review of the submittals, the staff has sent requests for additional information (RAI) to the licensees to complete its assessment of the submittal.

Seismic Events

Key observations obtained from a review of seismic IPEEEs include the following:

- A seismic walkdown was performed for each plant, and in most cases, the walkdown identified conditions pertaining to anchorage, interactions, maintenance, and/or housekeeping that required further investigation. As a result, a number of plant-specific fixes have been implemented at many plants.

- In seismic PRA studies, different hazard curves have been used (i.e., 1993 Lawrence Livermore National Laboratories [LLNL], 1989 LLNL, 1989 EPRI, and individual licensee-sponsored contractor results) from plant to plant. Hence, it is difficult to achieve a meaningful comparison of seismic CDFs across plants. However, the ranking of dominant contributors has consistently been reported in seismic IPEEEs as being insensitive to use of the EPRI or LLNL seismic hazard curves.

- Simplifications in systems analyses, unsubstantiated assumptions regarding human error rates, and use of simplified screening fragilities have, in some cases, obscured findings pertaining to dominant seismic risk contributors and produced unrealistic (high or low) CDF estimates.

- Although the analytical methods vary and there are large uncertainties, it appears that the CDF contribution from seismic events can, in some cases, approach that from internal events.

Fire Events

Key observations obtained from a review of fire IPEEEs include the following:

- No fire vulnerabilities have been reported in the first 24 IPEEE submittals; however, fire-initiated accidents have been found to be an important component of the external events CDF contribution.

- While no plants have identified fire vulnerabilities, about half (of the 24 reviewed) have reported some fire-related plant improvement as a result of the IPEEE effort. These improvements include changes to existing procedures, development of new procedures, or plant modifications.

- Overall licensees have expended a considerable level of effort in conducting fire IPEEEs. A few submittals clearly demonstrated the proper application of fire risk methodologies and data. However several weaknesses have been noted in applying the selected methods and data in some of the fire analyses which affect the robustness and completeness of the submittals. Some of these weaknesses are as follows:

  - Operator actions in response to the effects of fire on systems have rarely been modeled in detail.

  - Several submittals have used questionable methods, procedures, or data for fire damage modeling.

  - Several submittals have used the Nuclear Safety Analysis Center (NSAC)/181 and/or the EPRI Fire PRA Implementation Guide documents for which some optimistic guidelines and data have been identified.

  - The possibility of active barrier failure, which may have a significant probability of occurrence, has not been included in most analyses. The significance of active fire barriers is a function of plant layout and separation of redundant trains. Also, the potential for barrier failure associated with large quantities of combustible materials concentrated in one area has not been considered in most of the submittals.

- Although the analytical methods vary and there are large uncertainties, it appears that the CDF contribution from fire events can, in some cases, approach that from internal events.

HFO Events (i.e, high winds, floods, transportation accidents and nearby facility)

Key observations from HFO IPEEE submittals include:

- Transportation and nearby facility accidents have been screened out in all of the 24 IPEEEs that have been reviewed.

- The HFO IPEEE program has had some impact on plant safety. For some plants a greater appreciation of the potential risk impact of high winds/tornadoes and external flooding/dam breaks has resulted from the IPEEE program. Some licensees have implemented or proposed plant improvements with respect to procedural enhancements, severe accident management guidelines, and hardware installation. Procedural enhancements include sandbagging, closing doors, welding doors, hooking up pumps, and creating new circuits to reduce the risk from flooding. In two submittals, development of severe accident management guidance to reduce the risk of high winds is being considered. Hardware improvements include, for example, modifications that enhance flood protection.

- Potential failures of upstream dams, leading to flooding at the site, were considered and screened out in many of the first 24 submittals. However, generic dam failure data has been employed in all cases without considering site-specific information such as dam type and vintage.

- In general, the CDF contribution from HFO is lower than that from internal events.

Generic Issues

The IPEEE program has addressed a number of generic issues (GIs) and unresolved safety issues (USIs) including USI A-45 ("Decay Heat Removal Requirements"), GI-131 ("Potential Interaction Involving the In-Core Flux Mapping System at Westinghouse Plants"), GI-57 ("Effects of Fire Protection System Actuation on Safety Related Equipment"), Sandia fire risk scoping study issues, and GI-103 ("Probable Maximum Precipitation [PMP]"). Some key observations from the review of the first 24 IPEEE submittals include the following:

- In general the seismic and fire evaluations of the IPEEE are capable of addressing USI A-45, without any special additional considerations. Also, no HFO evaluation reported any open issue pertaining to USI A-45.

- For most applicable plants, GI-131 had been addressed through earlier upgrades and analyses. Some IPEEEs evaluated the capability of the in-core flux mapping system for beyond design-basis seismic loads consistent with the IPEEE review level earthquake (RLE).

- Almost all licensees have followed the guidance in FIVE pertaining to the evaluation of the fire risk scoping study issues and GI-57. In a few cases, seismic-fire outliers have been noted. No submittals have reported risk-significant findings associated with either the fire risk scoping study issues or GI-57.

- Most submittals addressed the effects of increased PMP criteria with respect to roof ponding and flooding due to intense local precipitation. In all such cases. the impacts of GI-103 were found to be accommodated by the existing plant design.

- The IPEEE submittals also provide information relevant to some other generic safety issues (GSIs) even though the submittals were not explicitly requested to treat, and the IPEEE program was not originally intended to resolve, such issues. These issues include: GSI-147 ("Fire-Induced Alternate Shutdown/Control Panel Interaction"); GSI-148 (Smoke Control and Manual Fire-Fighting Effectiveness"); GSI-156 ("Systematic Evaluation Program [SEP]"); and GSI-172 ("Multiple System Responses Program [MSRP]"). The IPEEE review process has identified the extent to which the submittals provide information relevant to these GSIs, and how these issues can be considered to be resolved.

<u>QUAD Cities Fire IPEEE</u>

Although not part of the first 24 IPEEE submittals reviewed, the Quad Cities fire IPEEE submittal review has revealed some particularly significant perspectives related to fire risk. A brief summary of the licensee’s fire IPEEE process and findings is provided below.

The licensee’s fire assessment employed EPRI’s Fire-Induced Vulnerability Evaluation (FIVE) method for initial screening and EPRI’s Fire PRA Implementation Guide for detailed evaluations for screened-in fire areas. These evaluations include: assessment of individual sources that can damage safety targets (i.e., safety shutdown equipment); identification of fire scenarios, taking into account fire features, such as detection and suppression; determination of conditional core damage probability for the specific fire targets; and calculation of a scenario-specific core damage frequency (CDF) value. Additionally, multi-compartment fire scenarios were considered in the event that the fire barriers credited in the single compartment analyses are unable to prevent fire propagation in adjacent compartments. Walkdowns were also conducted by Quad Cities plant engineers together with supporting contractors in order to: verify the compartment data; assess the seismic/fire interactions; identify the potential fire sources, safety targets, and locations of fire detection and suppression systems; and inspect the fire barriers.

The licensee estimated a total fire CDF at the Quad Cities to be about $5x10^{-3}$ per reactor year (RY). The licensee reported that the top five accident sequences, which contributed about 40% of the total fire CDF, were all in the turbine building involving postulated oil fires. The licensee stated that, even though the plant is in compliance with the NRC regulations, the lack of separation of certain cables in the turbine building and the complicated procedures needed for recovery actions were responsible for the high CDF number. The licensee used Nuclear Energy Institute's (NEI's) severe accident vulnerability criteria (e.g., CDF exceeds $1x10^{-4}$ per reactor year) and identified fire at the plant as a potential severe accident vulnerability. The licensee has implemented an interim alternate shutdown method involving the use of an independent back-up power supply for both units at Quad Cities to reduce the fire CDF from $5x10^{-3}$ per reactor year to $7x10^{-4}$ per reactor year. Currently, the licensee is evaluating long-term measures to further reduce the fire CDFs and is keeping the staff informed about its progress.

Distribution:
PRAB Subj/File
Central File
HThompson, 0-17G21
AThadani, 0-17G21
PNorry, 0-17G21
JBlaha, 0-17G21
WITS 9700206, 9500047, 9700212
EDO R/F, 0-17G21
KBohrer, 0-12G18
MCullingford, 0-12G18
MKnapp, 0-12G18
RZimmerman, 0-12G18
AChaffee, 0-11E4
MCaruso,T-10E50
MRubin, 0-10E4
MCunningham, T-10E50
Lspessard, 0-9E4
GHolahan, 0-8E2
PMagnanelli, 0-8E2
CBerlinger, 0-8H7
BSheron, 0-7D26
GLainas, 0-7D26
RWessman, 0-7E23
GBagchi, 0-7H15
PBaranowsky, T-4A9
ERossi, T-4A9
JAustin, T-7D13
CPaperiello, T-8A23
TKing, T-10E37
WHodges, T-10E37
MDrouin, T-10E50
JMurphy, T-10F12
BHardin, T-10E50
AElbassioni, T-10E50
JGuttmann, T-10E50
ARubin, T-10E50

OFFICE OF NUCLEAR REGULATORY RESEARCH
DIVISION OF SYSTEMS TECHNOLOGY
PROBABILISTIC RISK ANALYSIS BRANCH

DATE: SEPTEMBER 19, 1997
DOCUMENT: A:SEP97_UP.WPD
ORIGINATOR: MARK CARUSO
TYPIST: JORDAN

SUBJECT: QUARTERLY STATUS UPDATE FOR THE PROBABILISTIC RISK ASSESSMENT (PRA) IMPLEMENTATION PLAN, INCLUDING A DISCUSSION OF FOUR EMERGING POLICY ISSUES ASSOCIATED WITH RISK-INFORMED REGULATION

1.	~~M. Caruso~~	~~Concur~~	~~09 /26 /97~~
2.	M. Cunningham	Concur	/ /97
3.	T. King	Concur	/ /97
4.	M. Wayne Hodges	Concur	/ /97
5.	J. Flack	Concur	/ /97
6.	G. Holahan	Concur	/ /97
7.	G. Lainas	Concur	/ /97
8.	L. Spesard	Concur	/ /97
9.	F. Gillespie	Concur	/ /97
10.	B. Sheron	Concur	/ /97
11.	S. Collins	Concur	/ /97
12.	C. Paperiello	Concur	/ /97
13.	~~P. Baranowsky~~	~~Concur~~	~~09 /24 /97~~
14.	K. Raglin	Concur	/ /97
15.	T. Martin	Concur	/ /97
16.	S. Treby	Concur	/ /97
17.	M. Knapp	Concur	/ /97
18.	A. Thadani	Concur	/ /97
19.	J. Callan	Concur/Sign	/ /97

PARALLEL CONCURRENCE FOR B. Jordan