UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 6, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT ON THE 477TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, NOVEMBER 2—4, 2000, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Meserve:

During its 477th meeting, November 2-4, 2000, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and letters. In addition, the Committee authorized Dr. Larkins, Executive Director, ACRS, to send you the memorandum noted below:

REPORTS

- Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated November 8, 2000)

- License Renewal Guidance Documents (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated November 15, 2000)

LETTERS

- Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated November 20, 2000)

- BWROG Proposal to Use Safety Relief Valves and Low Pressure Systems as a Redundant Safe Shutdown Path to Satisfy the Requirements of 10 CFR 50, Appendix R (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated November 20, 2000)

MEMORANDUM

- Draft Safety Evaluation Report Related to "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-Through Steam Generators" (BAW-2374, Rev.0, July 2000) (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated November 13, 2000)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Revised Report of the Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

The Committee heard presentations by and held discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) and the Institute for Resource and Security Studies (IRSS) on the revised report of the final technical study of spent fuel pool accident risk at decommissioning nuclear power plants. The Committee discussed the staff's responses to the concerns identified in the April 13, 2000, ACRS report to the Commission. The concerns included the inappropriate use of the Regulatory Guide 1.174 risk acceptance criterion for large early release frequency (LERF) , the use of an ignition temperature based on data from fresh cladding, the failure to consider uncertainties in plume dispersion parameters, the initial plume energy, and the assessment of the seismic risk.

The Committee discussed NEI's assertions that (1) the bounding estimate of seismic risk should not be used to justify retention of operating plant requirements, (2) opportunities to apply practical risk insights are lost if operating plant requirements are retained, and (3) hypothetical phenomena should not be used to determine consequences.

The Committee heard a presentation by Dr. Gordon Thompson, IRSS. He stated that the potential for pool fires could be almost completely eliminated by combining low-density pool storage and dry storage. Dr. Thompson recommended that the NRC declare a moratorium on any decisions or licensing actions that could increase the risk of a radioactive release from any spent fuel pool, pending the completion of new studies on spent fuel pool accident risk.

Conclusion

The Committee sent a report dated November 8, 2000, to Chairman Meserve on this matter.

2. Risk-Informed Regulation Implementation Plan (RIRIP)

The Committee heard a presentation by and held discussions with representatives of the NRC staff concerning the staff's proposed update to the Risk-Informed Regulation Implementation Plan (RIRIP). The Committee and staff discussed the purpose of the RIRIP — to serve as a roadmap for risk-informed regulation and for implementing the NRC Strategic Plan in each strategic arena (nuclear reactor safety, nuclear materials, nuclear waste safety, and international nuclear safety support). The Committee considered the staff's draft screening criteria for evaluating risk-informed initiatives and the role of the RIRIP in communicating planned activities, schedules, and milestones. The Committee and staff extensively discussed challenges to the risk-informed approach, including probabilistic risk assessment quality, the availability of licensee risk analysis for public scrutiny, the need for stakeholder confidence, and the development of safety goals for the nuclear materials and waste arenas.

Conclusion

This briefing was for information only. No Committee report was required.

3. Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50

The Committee heard a presentation by and held a discussions with representatives of the NRC staff and its contractors regarding Attachment 1 to SECY-00-0198. The title of the attachment is "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50." The Committee discussed the purpose of the proposed Option 3 framework as staff guidance for evaluating candidate regulations and in developing risk-informed alternatives. The Committee considered the framework as a "work in progress" that will need to be updated as experience is gained in evaluating candidate regulations such as 10 CFR 50.44 (combustible gas control systems) and 10 CFR 50.46 (emergency core cooling systems). The Committee discussed the framework's consideration of selected issues including defense in depth (structural versus rational approaches), the quantification of safety margins in terms of probabilities, and the definition and implementation of the concept of accident-initiating events (anticipated, frequent, and rare initiators).

Conclusion

The Committee sent a letter dated November 20, 2000, to the Executive Director for Operations, on this matter.

4. Differing Professional Opinion (DPO) on Steam Generator Tube Integrity

Dr. Powers, the Chairman of the Ad Hoc Subcommittee on the DPO, reported to the Committee on the status of the Subcommittee's review of the technical merits of the DPO issues. This review was undertaken at the request of the NRC Executive Director for Operations (EDO). The Ad Hoc Subcommittee, established during the September 2000 ACRS meeting, consists of Dr. Powers (Chairman), Dr. Bonaca, Dr. Kress, Mr. Sieber, and Dr. Ballinger (Massachusetts Institute of Technology). Drs. Catton and Higgins, from the EDO's office, are the DPO consultants to the Subcommittee. The Subcommittee held a meeting October 10—14, 2000, to discuss the DPO issues with the DPO author and the NRC staff. The Subcommittee Chairman is compiling a report based on the information presented during the Subcommittee meeting and the comments made by the Subcommittee members and consultants.

The DPO author and the NRC staff briefed the full Committee during the November ACRS meeting. The DPO author recommended that —

- Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," be rescinded
- All plants that do not meet the 40% plugging criterion be shut down until all their tubes have been plugged

Representatives of the NRC staff discussed past, present, and future efforts to resolve the DPO and the steam generator issues, including the issuance of the DPO Consideration document, several NUREG reports, and research efforts to address the jet cutting tubes.

Conclusion

The Ad Hoc Subcommittee plans to submit a report to the full Committee. The report during the December 2000 ACRS meeting will reflect incorporation of the internal peer review comments. The Committee has approved the report, and will send it to the EDO for use in resolving the DPO issues.

5. Performance-Based, Risk-Informed Fire Protection Standard for LWRs and

Related Issues

The Committee heard presentations by and held discussions with representatives of the Boiling Water Reactor Owners Group (BWROG), NRC staff, the National Fire Protection Association (NFPA), and NEI about the BWROG proposal to use the safety relief valves (SRVs) and the low-pressure system (LPS) as redundant safe shutdown path and about the NFPA 805 standard. The Committee discussed the BWROG proposal. The BWROG's position is that use of SRVs and LPS to support the 10 CFR Part 50, Appendix R safe shutdown requirement is consistent with the original design basis for BWRs. The proposal specifies a technically acceptable and safe method of achieving and maintaining either hot or cold shutdown. The representatives of BWROG also stated that same method is specified in emergency operating procedures as a means to achieve cold shutdown after small-break loss-of-coolant accidents in BWRs. The NRC staff supports the BWROG's position on the use of SRVs and the LPS as a redundant path to achieve safe shutdown. The NRC staff plans to publish the safety evaluation report on this issue in the near future.

Mr. Fred Emerson, NEI, presented the industry's views on implementation of the NFPA 805 standard. The NFPA 805 standard has six chapters on: (1) goals, performance objectives, and performance criteria, (2) a general approach for establishing a fire protection program and fire protection requirements, (3) determination of fire protection systems and features, (4) fire protection during decommissioning and permanent plant shutdown, and (5) a summary of referenced NFPA publications. There are six appendices on: (1) explanations of matters in the body of the standard, (2) nuclear safety assessment, (3) the application of fire modeling to nuclear power plants, (4) the use of PSA methods, (5) a deterministic approach for plant fire damage and business interruption, and (6) referenced publications. The NFPA membership vote is scheduled for November 15, 2000. The membership can accept the standard as it is, accept it as amended, return part of the standard to the NFPA 805 Committee, or return the entire standard to the Committee. If the Committee approved, the Standards Council will issue the standard on January 13, 2001. In concluding, Mr. Emerson said that the NFPA standard offers potential benefits to plants and that the industry has extensively supported its development.

Mr. Denis Shumaker presented the preliminary results of the pilot use of the NFPA 805 standard at Salem Generating Plant. He concluded that the NFPA 805 standard adequately explains fire risk and manage it. It provides a basis for assessing fire risk actual plant configurations and modifications and will allow licensees to focus of resources on the most important fire risks.

The NRC staff presented a brief overview of the NFPA 805 standard. The standard changes the existing Appendix R requirements. The performance criterion for NS allow the use of ADS and LPS for shutting down BWRs and specifies feed-and-bleed as the only shutdown method for PWRs. Performance-based, risk-informed allow the recovery of SSCs vs free of fire damage. The 72-hour cold shutdown requirement, alternative or dedicated shutdown, and 8-hour emergency lighting requirement has been eliminated from the NFPA 805 standard. Technical and implementing issues need to be resolved before the standard can be adopted by the NRC staff.

Conclusion

The Committee sent a letter dated November 17, 2000, to the EDO on BWROG proposal to use SRVs or LPS as redundant shutdown paths.

6. ABB/CE and Siemens Digital Instrumentation and Control (I&C) Applications

The Committee received a report on the results of October 31, 2000, meeting of the Plant Subcommittee on ABB/CE and Siemens Digital I&C Applications. During the Subcommittee meeting with the representatives of the NRC staff, Westinghouse Nuclear Automation (formerly known as ABB/CE), and Siemens Corporation the safety evaluations of the Westinghouse and Siemens topical reports on digital I&C applications was discussed.

The NRC staff stated that, based on information from Westinghouse on the topical report and the staff's review the design of the Common Qualified (Common Q) platform meets the relevant NRC regulatory requirements and is acceptable for safety-related I&C applications in nuclear power plants, subject to the satisfactory resolution of the generic open items. The staff had also reviewed the safety system design descriptions in the Siemens topical report for conformance to the guidelines in the regulatory guides and industry codes and standards applicable to these systems. The staff concluded that the applicant adequately identified the guidelines applicable to these systems.

Conclusion

The Committee will continue its discussion of the topical reports during future ACRS meetings.

7. License Renewal Guidance Documents

The Committee heard presentations by and held discussions with the NRC staff and NEI regarding draft guidance documents for preparing and reviewing license renewal applications. The staff discussed the changes incorporated into the latest drafts of the guidance documents, how it disposed of stakeholders' comments, and the status of unresolved issues. NEI said the industry is concerned that the license renewal process may be used to impose additional unnecessary programs on licensees. The Committee and the staff discussed the aging of electrical cables and the use of emergency operating procedures in the scoping process. They also discussed updating the guidance document to incorporate lessons learned and to recognize new editions of codes and standards.

Conclusion

The Committee sent a report dated November 15, 2000, to Chairman Meserve.

8. Research Report to the Commission

The Committee discussed the 2001 ACRS report to the Commission on NRC safety research program. The Committee will continue to take an active role in reviewing ongoing and proposed research activities and provide comments and make recommendations to the Commission.

The Committee members discussed the format and content of the report and stated that the report should focus on longer-term research to ensure the Commission's carry out its safety mission efficiently and effectively in the future.

Conclusion

The Committee will continue discussing and preparing the 2001 ACRS report to the Commission on the NRC safety research program at the December 6-9, 2000 ACRS meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated October 23, 2000, to the ACRS comments and recommendations included in the ACRS report dated September 8, 2000, concerning the proposed high-level guidelines for performance-based activities.

  The Committee decided that it was satisfied with the EDO's response.

- The Committee discussed the response from the EDO, dated October 25, 2000, to ACRS comments and recommendations included in the ACRS report dated September 13, 2000, concerning proposed risk-informed revisions to 10 CFR 50.44, Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

The Committee decided that it was satisfied with the EDO's response.

- The Committee discussed the response from the EDO, dated October 25, 2000, to ACRS comments and recommendations included in the ACRS report dated September 7, 2000, concerning assessment of the quality of probabilistic risk assessments.

  The Committee decided that it was satisfied with the EDO's response.

- The Committee discussed the response from the EDO dated October 30, 2000, to ACRS comments and recommendations included in the ACRS report dated September 14, 2000, concerning the pre-application review of the AP1000 standard plant design-phase 1.

  The Committee decided to continue its discussion of the issues included in its report and the adequacy of the EDO's response during future meetings.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from October 5 through October 31, 2000, the following Subcommittee meetings were held:

- Ad Hoc Subcommittee - October 10-14, 2000

  The Subcommittee met to discuss the technical merits of the Differing Professional Opinion Issues associated with steam generator tube integrity.

- Fire Protection Subcommittee - October 16-17, 2000

  The Subcommittee met to review the revised draft NFPA 805 Performance Standard for Fire Protection for Light Water Reactor Electric Generating Plants, Draft Regulatory Guide on Fire Protection for Operating Nuclear Power Plants, post-fire safe shutdown circuit analysis, and other fire protection related issues.

- Reactor Fuels Subcommittee - October 18, 2000

  The Subcommittee met to discuss the status of the staff's effort regarding the revised report of a technical study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, and related matters.

- Plant License Renewal Subcommittee - October 19-20, 2000

  The Subcommittee met to review drafts of the Standard Review Plan for license renewal, the Generic Aging Lessons Learned (GALL) Report Sections 2, 3, 4, 5 through 8, the associated regulatory guide, and NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54, The License Renewal Rule."

- Plant Systems Subcommittee - October 31, 2000

  The Subcommittee met to discuss the safety evaluation reports on the topical reports for ABB/CE and Siemens Digital I&C Applications.

- Planning and Procedures Subcommittee - October 31, 2000

  The Subcommittee met to discuss proposed ACRS activities, practices, and procedures for conducting Committee business and organizational and personnel matters relating to ACRS and its staff.

- Safety Research Program - November 1, 2000

  The Subcommittee met to discuss the NRC safety research programs and hear the views of the NRC staff in preparation of the 2001 ACRS report to the Commission.

FOLLOWUP ITEMS

- The Committee plans to review the staff's safety evaluation report on the BWR Owners Group proposal to use safety relief valves and low-pressure systems as a redundant method to achieve safe shutdown as required by 10 CFR Part 50, Appendix R.

- The Committee plans to review additional refinements to the framework document associated with risk-informed changes to the technical requirements of 10 CFR Part 50.

- The Committee plans to review the staff's validation that the artificially aged cables used in the accelerated aging studies conducted to address the issues of GSI-168 are representative of 30-40 year old cables along with its review of the proposed resolution of GSI-168.

- The Committee plans to continue its discussion of issues included in its September 14, 2000 report on the pre-application review of the AP1000 design and the adequacy of the EDO's response dated October 30, 2000, during future meetings.

PROPOSED SCHEDULE FOR THE 478TH ACRS MEETING

The Committee agreed to consider the following topics during the 478th ACRS meeting, December 7-9, 2000:

Issues Associated with Core Power Uprates
Briefing by and discussions with representatives of the NRC staff regarding issues associated with core power uprates, including: staff plans for developing a Standard Review Plan Section for power uprate reviews; staff position regarding the need for applying risk-informed decisionmaking in the review of significant power uprate applications; and other related matters.

Differing Professional Opinion (DPO) on Steam Generator Tube Integrity
Report by the Chairman of the Ad Hoc Subcommittee on DPO regarding conclusions and recommendations of the Ad Hoc Subcommittee on the technical merits of the DPO issues. Discussion with representatives of the NRC staff and the DPO author, as needed, regarding additional information on DPO issues.

Subcommittee Report
Report by the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the status of review of the GE Nuclear Energy TRACG best-estimate thermal-hydraulic code.

Subcommittee Report
Report by the Chairman of the Plant Systems Subcommittee regarding ABB/CE and Siemens digital I&C applications and insights gained from meeting with the RSK on digital I&C in Germany during November 2000.

Meeting with NRC Commissioner Diaz
Discussion with Commissioner Diaz regarding the NRC Safety Research Program and other items of mutual interest.

South Texas Project Exemption Request
Briefing by and discussions with representatives of the NRC staff and South Texas Project Nuclear Operating Company (STPNOC) regarding the STPNOC's exemption

request to exclude certain components from the scope of special treatment requirements in 10 CFR Part 50 and the associated NRC staff's Draft Safety Evaluation Report.

Control Room Habitability
Briefing by and discussions with representatives of the NRC staff and the nuclear industry regarding issues associated with control room habitability and the staff and industry efforts in resolving those issues.

Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"
Briefing by and discussions with representatives of the NRC staff regarding the proposed final Regulatory Guide DG-1053, including the staff's resolution of public comments.

Proposed Modifications to the Commission's Safety Goal Policy Statement for Reactors
Briefing by and discussions with representatives of the NRC staff regarding the proposed modifications to the Commission's Safety Goal Policy Statement for reactors.

NRC Safety Research Program
Discussion of the 2001 ACRS report to the Commission on the NRC Safety Research Program.

Response to Commission Request
Discussion of the response to the Commission request that the ACRS provide a detailed discussion of how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and provide more specific recommendations on how those weaknesses should be addressed.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

CERTIFIED

Date Issued: 12/27/00
Date Certified: 1/8/01

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NOVEMBER 2-4, 2000

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APPENDICES

I. *Federal Register Notice*
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MINUTES OF THE 477TH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS NOVEMBER 2-4, 2000 ROCKVILLE, MARYLAND

The 477th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on November 2-4, 2000. Notice of this meeting was published in the *Federal Register* on October 20, 2000 (65 FR 63104) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting was kept and is available in the NRC Public Document Room at the One White Flint North Building, Mail Stop 1F-15, Rockville, MD, 20852-2738. [Copies of the transcript are available for purchase from Ann Riley & Associates, Ltd., 1025 Connecticut Avenue, N.W., Suite 1014, Washington, D.C. 20036, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

ACRS Members: Dr. Dana A. Powers (Chairman), Dr. George Apostolakis (Vice Chairman), Dr. Mario V. Bonaca, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. William J. Shack, Dr. Robert L. Seale, Mr. John D. Sieber, and Dr. Graham B. Wallis. Dr. Uhrig was not in attendance during this meeting. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

Dr. Dana A. Powers, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

II. Revised Report of the Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (Open)

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

Dr. Thomas Kress, ACRS, stated that the purpose of this meeting was to discuss the NRC's staff effort regarding the revised technical study of SFP accident risk at decommissioning plants. The Committee also heard the views of the Nuclear Energy Institute (NEI) and the Institute for Resource and Security Studies (IRSS) representatives on this issue. Dr. Kress stated that in a Staff Requirements Memorandum dated December 21, 1999, the Commission requested the ACRS to perform a technical review of the validity and risk objectives of the draft technical study prepared by the NRC staff regarding the spent fuel pool (SFP) risk assessment. During the 471st meeting of the ACRS (April 5-7, 2000), the Committee reviewed the draft technical study and issued a report to the Commission. The Committee in its report expressed concern regarding the study and recommended the following:

- The integrated rulemaking on decommissioning should be put on hold until the staff provides the technical justification for the proposed acceptance criterion for fuel uncovery frequency. In particular, the staff needs to incorporate the effects of enhanced release of ruthenium under air-oxidation conditions.

- The technical basis underlying the zirconium-air interactions and the criteria for ignition needs to be strengthened. In particular, the potential impact of zirconium-hydrides in high burnup fuel and the susceptibility of the clad to breakaway oxidation need to be addressed.

- Uncertainties in the risk assessment need to be quantified and made part of the decision-making process.

NRC STAFF PRESENTATION

Mr. Timothy Collins, NRR, stated that the staff had previously prepared a draft technical study (dated February 2000) to address the SFP accident risk at decommissioning plants. In this draft study, the staff estimated that after one year following permanent shutdown, the generic frequency of events leading to zirconium fires to be less than 3×10^{-6} per year for a plant that implements the design and operational characteristics assumed in the risk assessment performed by the staff. This frequency was estimated based on the assumption that the industry decommissioning commitments (IDCs) plus additional staff assumptions would be implemented. The staff recognized that this estimate could be much higher for a plant that does not implement these operational characteristics. The staff noted in the draft study that the most significant contributor to the SFP risk issue is a seismic event which exceeds the design basis earthquake. However, the staff indicated that the overall frequency of this event is within the developed SFP performance guideline for large radionuclide releases (related to zirconium fire) of $1x10^{-5}$ per year.

On October 12, 2000, the staff completed its revision of the technical study. The revised technical study indicate that while the risk at SFPs is low, it is not markedly lower than that for operating reactors especially in the earliest years after shutdown. Even though the likelihood of a zirconium fire is very low, the consequences in terms of both the integrated dose to the public and the early fatalities can be comparable to a large early release frequency (LERF) from an operating plant during a potential severe core damage accident. The revised study indicates that the analysis of early fatality risk shows that the range of the SFP risk estimates, which address seismic, source term, and thermal hydraulic uncertainties, overlap with the range of operating reactor risk estimates during the first few years after shutdown. The analysis of latent cancer fatality risk shows that the range of possible SFP risk continues to overlap with the range of operating reactor risk until the time when ad hoc accident management recovery actions can be credited to suppress the SFP risks. The staff stated that the effects of a significant ruthenium and fuel fines release, as suggested by the ACRS, was notable, but not so important as to result in consequences larger than those associated with a reactor accident large early release. Thus, the staff concluded that the original spent pool performance guideline (PPG) of $1x10^{-5}$ per year is deemed appropriate. Using either the Lawrence Livermore National Laboratory (LLNL) or the Electric Power Research Institute (EPRI) seismic hazard curve, the staff concluded that the risk is well below the safety goal quantitative health objectives (QHO) for both the individual risk of early fatality and the individual risk of latent cancer fatality. However, the risks are not dramatically reduced relative to operating reactor risks as estimated in NUREG-1150.

The staff has reevaluated appropriateness of temperature criteria considering zirconium reaction kinetics, hydriding, fuel damage testing, fission product release data, and materials interactions. The staff concluded that for assessing the onset of fission product release under transient conditions, to establish the critical decay time for determining availability of 10 hours to evacuate, it is acceptable to use a temperature of 900 °C if fuel and cladding oxidation occurs in the air. If steam kinetics dominate the transient heat-up case, as it would in many boildown and drain down scenarios, then a suitable temperature criterion would be around 1200 °C. For establishing long term equilibrium conditions for fuel pool integrity during SFP accidents which preclude significant fission product release it is necessary to limit temperatures to values of 600 °C to 800 °C.

Mr. Jason Schaperow, Office of Nuclear Regulatory Research (RES), briefed the Committee regarding the consequence assessment for SFP accidents. Mr. Schaperow stated that it was initially thought that at one year after final shutdown, the radiological consequences from an SFP accident might be negligible. If consequences were negligible, requirements for emergency planning and insurance could be eliminated. Therefore, RES performed offsite radiological consequence calculations with MACCS (for 30 days, 90 days, and 1 year after final shutdown) to quantify the consequences. The issues examined were reduced inventory (at 1 year), early vs. late evacuation (at 1 year), importance of cesium and ruthenium, number of assemblies releasing fission products, fission product release fractions, plume heat content, plume spreading, decay times beyond 1 year, and reassessment of source term. The results of large number (about 300) of MACCS calculations were used to understand decommissioning risk in the staff's generic study. The effect of reduced inventory is that early fatalities was reduced by a factor of 2 from 30 days to 1 year. The cancer fatalities and societal dose was unaffected. The effect of reduced decay heat (early evacuation) is that early fatalities was reduced by up to a factor of 100, and the cancer fatalities and societal dose was unaffected.

Mr. Schaperow also discussed the effect of the number of fuel assemblies releasing fission products. The original staff's calculations assumed entire SFP inventory of Millstone 1 was involved in heatup and release (3.5 cores). The revised calculations, depending on reductions in decay heat from radioactive decay, assumed less fuel may be involved in heatup. The staff performed MACCS calculations for two cases: entire SFP inventory (3.5 cores), and inventory in final core offload. Mr. Schaperow stated that the calculations showed that smaller consequence reduction for cases with large ruthenium release because most ruthenium is in final core offload due to its one year half-life.

Other issues such as the effect of plume heat content was analyzed by the staff. The potential for plume heat content to be higher than that of a reactor accident was considered. The staff performed sensitivity calculations using different plume heat contents. The base case was plume heat content from NUREG-1150 (3.7 MW). The staff estimated plume heat content to be about 256 MW for complete oxidation of one core in 30 minutes. As part of an international cooperative effort on consequence assessment codes, experts provided updated values for the dispersion parameters σ_y and σ_z. Experts provided distributions instead of point estimates.

Mr. Schaperow stated that the revised technical study included atmospheric and consequence determination. Instead of relying on a LERF surrogate, the results can be directly compared with the prompt and latent fatality safety goals. Based on the sensitivity study, the staff adopted a revised source term with a ruthenium release fraction of 0.75 and an actinide release fraction of 0.035.

Dr. Robert Palla, NRR, briefed the Committee regarding the risk analysis results and conclusions. He stated that for the first 1 to 2 years, the early fatality risk for a SFP fire is low, but comparable to that for a severe accident in an operating reactor. At 5 years following shutdown, the early fatality risk for SFP accidents is approximately two orders of magnitude lower than for a reactor accident. Societal risk for a SFP fire is also comparable to that for a severe accident in an operating reactor, and does not exhibit a substantial reduction in time due to the slower decay of fission products. Changes to emergency preparedness requirements affect only the cask drop accident, and do not substantially impact either the total risk or the margin between SFP risk, and operating reactor risk due to the low frequency of cask drop accidents.

Dr. Palla stated that the revised technical study used a less conservative method that made use of a typical high confidence of low probability of failure (HCLPF) for a plant. The staff combined the HCLPF with both the LLNL and EPRI seismic hazard curves to estimate the seismic risk. Both the individual early fatality risk and the individual latent cancer fatality risk are about 1 to 2 orders of magnitude lower than the Commission's Safety Goal, depending on assumptions regarding the SFP accident source term and seismic hazard:

- At upper end (LLNL seismic hazard estimates and high ruthenium source term) the risks are somewhat lower than the corresponding risks for reactor accidents, and about a decade lower than the safety goal.
- At lower end (EPRI seismic hazard estimates and low ruthenium source term) the risks are lower than those for reactor accidents, and about 2 decades lower than the safety goal.

The staff stated that a lower zirconium ignition temperature would shorten the time to a release, but this was found not to be significant in early years because of the already short times available. Partial drain down scenarios result in restricted air flow which can be important to insurance considerations. The staff summarized its findings as follows:

- The risk at decommissioning plant SFPs is low, and within the range of operating reactor risk for at least the first few years after shutdown.
- Relaxation of offsite emergency planning a few months after shutdown results in a small change in risk and is consistent with staff guidelines for small changes in risk.

- Insurance requirements could be considered as a function of time available for implementation of accident management measures, but are not recommended in the first five years.

- As long as spent fuel is present in the SFP, some level of safeguards and security is necessary.

- Research regarding source term generation in an air environment is recommended.

NEI PRESENTATION

Ms. Lynnette Hendricks briefed the Committee regarding industry views on risk informing decommissioning regulations. She stated that the industry envision the use of risk insights to adapt deterministic rules for operating plants to decommissioning plants. The Commission's principles on risk informing must be adapted to address different types of consequences, lower probability, and a different type of system (e.g., passive, robust, slowly evolving sequences).

Ms. Hendricks noted that best estimates should be used, and consequences should not be based on phenomena that have not been validated through NRC's severe accident program. She added that more efforts should be devoted to the probability side of risk equations, and if the probability of an SFP fire is acceptably low there are diminishing returns on efforts to refine consequences.

Industry characterizes huge seismic events that are background risk factors for operating plants to dominate risk profile for decommissioning plants. In addition, seismic risk should be treated in the same manner for decommissioning plants as for operating plants.

In conclusion, Ms. Hendricks stated the following:

- Bounding estimates of seismic risk should not be used to justify retention of operating plant requirements intended for a much broader scope of initiating events.

- Overly conservative treatment of seismic risk leads to the conclusion that operating plant requirements should be retained.

- Opportunities to apply practical risk insights are lost if operating plant requirements are retained.

- Speculative phenomena should not be used to determine consequences.

IRSS PRESENTATION

Dr. Gordon Thompson, IRSS, stated that the potential for pool fires could be almost completely eliminated by storing spent fuel using a combination of low-density pool storage and dry storage. The potential for a runaway exothermic reaction of cladding in a high-density spent fuel pool, following water loss, has been known since the late 1970's. Dr. Thompson indicated that the potential for a pool fire can exist at any high-density pool but may be especially significant for pools at operating plants due to the presence of recently discharged fuel with a high decay heat and the potential for a reactor accident to initiate a pool accident. Dr. Thompson stated the following:

- Pool fires have not been studied to the same extent as reactor accidents (e.g., NUREG-1150, IPEs).

- There are major gaps in knowledge about the probability of pool fires, their phenomenology, and their consequences.

- Pool fires deserve attention because they could contaminate large areas of land with comparatively long-lived radioisotopes (Cesium-137), leading to significant health and economic impacts.

- Pools generally have a low inventory of short-lived radioisotopes, and as a result pool fires would generally have a comparatively low potential for causing early fatality.

Dr. Thompson cited the NRC Safety Goals, “Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.” Dr. Thompson stated that the NRC staff's analysis has not addressed land contamination, which is the most important indicator of pool risk, and accordingly the analysis does not provide a credible basis for decisionmaking.

In conclusion, Dr. Thompson provided the following steps:

- The NRC should declare a moratorium on any decisions or licensing actions that could increase the risk of a radioactive release from any spent fuel pool, pending the completion of new studies on pool accident risk.

- The NRC should perform studies and supporting experiments, to at least the depth of NUREG-1150, on the probability of pool fires, their phenomenology, and their consequences (for operating plants, this work should address interactions between reactor accidents and pool fires)

- Licensees should be required to extend their individual plant examinations and individual plant examination of external events to address pool fires.

Conclusion

The Committee sent a report dated November 8, 2000, to Chairman Meserve on this matter.

III. Risk-Informed Regulation Implementation Plan (RIRIP) (Open)

[Note: Mr. Michael T. Markley was the Designated Federal Official for this portion of the meeting.]

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment introduced the topic to the Committee. He stated that the purpose of this meeting was to review the NRC staff's Risk-Informed Regulation Implementation Plan (RIRIP). He noted that the RIRIP was developed in response to a March 1999 General Accounting Office report that recommended that the NRC Commissioners direct the staff to develop a comprehensive strategy for the transition to risk-informed regulation. He stated that the staff last briefed the Committee in May 2000 on its initial effort to transform the probabilistic risk assessment (PRA) Implementation Plan into the RIRIP. He noted that the proposed framework document is organized according to the strategic arenas in the agency's Strategic Plan, namely: Nuclear Reactor Safety, Nuclear Materials Safety, Nuclear Waste Safety, and International Nuclear Safety Support.

NRC Staff Presentation

Mr. Thomas King and Ann Ramey-Smith, RES, led the presentation for the NRC staff. Messrs. Martin Virgilio, NMSS, and Mark Rubin, NRR, provided supporting discussion. Significant points made during the presentation include:

- SECY-00-0213, "Risk-Informed Regulation Implementation Plan," represents a first attempt by the staff to put together a complete, multi-office implementation plan that provides an integrated roadmap for translating the NRC Strategic Plan and PRA Policy Statement into planned risk-informed actions and schedules.

- The RIRIP provides plans for internal communications and staff training. It provides the staff's approach to resolving impediments to risk-informed regulation and serves as a tool for communicating with external stakeholders.

- Draft screening criteria are provided to facilitate resolving safety concerns, make NRC and Agreement State activities more efficient and effective, reduce unnecessary regulatory burden, and address technical issues related to risk data and modeling.

- Challenges to the risk-informed approach include PRA quality, unavailability of licensee risk analysis for public scrutiny and the need for stakeholder confidence, and development of safety goals for the nuclear materials and waste arenas.

Dr. Powers questioned how one would recognize "state-of-the-art" risk analysis and how the staff's "roadmap" provided for an evolution of technology. The staff stated that advances in technology are expected as risk-informed approaches become more broadly used and that the RIRIP would be revised to accommodate advances in analysis techniques, modeling, and computer technology.

Dr. Apostolakis questioned how one might know when something is sufficiently risk-informed. He also questioned what criteria is needed for making regulatory decisions in terms of inputs, modeling issues, and uncertainties. The staff stated that they have developed a list of questions as part of the general guidance and suggested that the resolution of these questions would characterize the information needed for particular decisions. The staff also noted that many key questions are addressed in satisfying the cornerstones of the revised reactor oversight process (RROP).

Mr. Leitch and Dr. Bonaca expressed the view that the communication plan focuses mostly on internal communications and suggested that more could be done to enhance the involvement of external stakeholders. Mr. Leitch also questioned the emphasis on reducing unnecessary regulatory burden relative to the focus on enhancing safety. The staff agreed that more emphasis could be placed on the importance and involvement of external stakeholders. The staff noted that the RIRIP's was designed to focus regulatory attention and resources on risk-significant issues. The staff also noted that the focus on reducing unnecessary regulatory burden is consistent with and supported by the PRA Policy Statement.

Conclusion

This briefing was for information only. No Committee report was required.

IV. Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Open)

[Note: Mr. Michael T. Markley was the Designated Federal Official for this portion of the meeting.]

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment, introduced this topic to the Committee. He stated that the purpose of this meeting was to review Attachment 1 to SECY-00-0198 entitled, "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50." He noted that the proposed Option 3 framework is intended for use by the staff in evaluating candidate regulations and in developing risk-informed alternatives. He also noted that it is a "work in progress" and will need to be updated as experience is gained in its application to risk-informing regulations.

NRC Staff Presentation

Mr. Thomas King and Mary Drouin, RES, led the discussion for the NRC staff. Mr. Alan Kuritsky, RES, and NRC contractors Messrs. Alan Camp, Sandia National Laboratories, Trevor Pratt, Brookhaven National Laboratories, and Eric Haskin, ERI Consulting, provided supporting discussion. Significant points made during the presentation include:

- The initial version of the framework document, described in SECY-00-0062, was developed and tested for proposed revisions to 10 CFR 50.44 related to combustible gas control systems. Additional changes to the framework will evolve as risk-informed alternatives are considered for other regulations, e.g., 10 CFR 50.46 concerning emergency core cooling systems.

- The framework focuses on revising existing regulations with particular emphasis on initiating events that have the potential to lead to core damage. The risk-informed alternative regulations would be voluntary and the staff will consider performing backfit analysis for safety enhancements if the regulations are found to be insufficient in certain areas.

- The proposed framework is structured after the revised reactor oversight (RROP) cornerstones of safety. The framework utilizes "strategies" of accident prevention and mitigation and employs "tactics" related to design, construction, and operation. Tactics include safety margins; redundancy, diversity, and independence; general design criteria, special treatment, etc.

- Quantitative guidelines have been established to enhance the consistency and predictability of the regulatory decision-making process, e.g., accident initiating event frequency, conditional core damage probability, conditional large release probability, and conditional individual fatality probability.

Dr. Powers questioned why the staff had not taken a "clean approach" such that the framework could accommodate a new or innovative reactor design. He noted that the guidelines in the proposed framework are well-suited for core damage events at light-water reactors but questioned how the framework would treat a reactor design with a positive reactivity coefficient, e.g., CANDU. The staff stated that they were directed by the Commission to focus on existing regulations for the current generation of nuclear plants.

Dr. Apostolakis questioned the role of external events in the framework document. He stated that the framework appears only to address internal events. The staff stated that the framework is intended to address full-scope PRA, including external events, but acknowledged that completeness may be an issue depending on the particular regulation and its application.

Dr. Apostolakis questioned why defense in depth was not included in the list of "tactics." Dr. Apostolakis complimented the staff for the description of safety margins in Section 4.2 of the framework document but questioned the sufficiency of the definition of safety margins. In particular, he questioned how much safety margin is too much? How much is too little? What is the right amount of safety margin and defense in depth and how do they relate to the quantitative guidelines? The staff stated that the tactics are an option to apply when the quantitative objectives are not met. Dr. Apostolakis suggested that the staff consider enhancing the framework document to more fully consider the Committee's May 19, 1999 report, concerning defense in depth (structuralist versus rationalist approaches). The staff agreed to consider this suggestion. Dr. Apostolakis also suggested several changes to the quantitative guidelines which the staff also agreed to consider.

Conclusion

The Committee sent a letter on this matter to the Executive Director for Operations dated November 14, 2000.

V. Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open)

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

Dr. Powers, Chairman of the Ad Hoc Subcommittee on DPO, provided a report to the Committee regarding the status of the Subcommittee's review of the technical merits of the DPO issues. Significant points made by Dr. Powers include the following:

Background

In a memorandum dated July 20, 2000, the NRC Executive Director for Operations (EDO) requested that the ACRS function as the equivalent of an ad hoc panel, under Management Directive 10.159, "Differing Professional Views or Opinions," to review the DPO issues and provide a report documenting the conclusions and recommendations relative to the pertinent technical issues. During its September 2000 meeting, the Committee decided to undertake the review of the DPO issues as requested by the EDO and established an Ad Hoc Subcommittee, composed of Dr. Powers (Chairman), Dr. Bonaca, Dr. Kress, Mr. Sieber, and Dr. Ballinger (Massachusetts Institute of Technology), to review this matter. In a memorandum dated September 11, 2000, the Committee informed the EDO that the Ad Hoc Subcommittee will function under the provisions of the Federal Advisory Committee (FACA) and will use the consultants (Drs. Carron and Higgins, which the EDO agreed to provide) and other consultants, as needed, to obtain technical support in reviewing the DPO issues.

Subcommittee's Review of the DPO Issues

The Ad Hoc Subcommittee held a meeting on October 10-14, 2000, to review the technical merits of the DPO issues noted below.

- Accident Analysis
  - design basis accidents
  - severe accidents
- Limitations on non-destructive examination (NDE) methods
- Corrosion and Cracking Phenomena
- Leakage and Burst Phenomena
- Damage Progression

- crack opening
- crack unplugging
- jet cutting of tubes

- Source Term
  - iodine spiking
  - aerosol behavior

The Subcommittee members and consultants reviewed all relevant documents that were essential to understand and analyze the technical merits of the DPO issues. The Subcommittee heard presentations by and held discussions with representatives of the NRC staff, the DPO author, and other interested persons. Dr. Powers commended the outstanding presentations provided by the staff and the DPO author during the meeting.

Dr. Powers summarized the main issue of contention, stating that the NRC staff issued Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." This generic letter allows licensees to use voltage indications rather than crack depth as a criterion for repairing cracks confined to the regions of the steam generator tube support plates. The DPO author contends that the voltage-based repair criteria are complex and non-scientific, based on poorly written material and insufficient technical information, and on hidden or unsupported assumptions. The DPO author recommended that Generic Letter 95-05 be rescinded and that all plants that do not meet the 40% plugging criteria be shutdown and all tubes plugged accordingly.

Dr. Powers stated that during the Ad Hoc Subcommittee meeting, the staff presented its efforts to resolve the DPO and steam generator issues, including the following:

- DPO and steam generator issues have been given serious attention.

- Assessment of the DPO and steam generator tube integrity issues have been documented (e.g., DPO Consideration Document and NUREG reports).

- Regulatory framework for steam generator tube integrity is being developed in coordination with NEI.

- NRR and RES are working closely to resolve several issues (e.g., tube cutting/erosion, vibration).

- A new generic issue is being considered to address the vibration issue.

- Lessons learned from the Indian Point 2 steam generator tube rupture event will be evaluated and a decision will be made with regard to improvements that need to be done to ensure steam generator tube integrity.

Schedule for Completing the ACRS Report on DPO

Dr. Powers stated that he is in the process of preparing a report (NUREG form) with input from the subcommittee members and consultants. After completion of the peer review, he plans to submit the report to the full Committee for review and approval during the December 2000 ACRS meeting.

Presentation by the DPO Author - Dr. J. Hopenfeld

Dr. Hopenfeld provided a brief presentation to the full Committee, reiterating the main points made to the Ad Hoc Subcommittee. Key points made by Dr. Hopenfeld include:

- Risk to the public from not removing the degraded steam generator tubes from service is at least 100 times greater than has been reported.

- Eddy current probe has inherent limitations to detect cracks. The probability of detection of 0.6 allowed by the NRC is arbitrary, totally unfounded, and non-conservative.

- The NRC voltage methodology for predicting steam generator tube leakage is grossly nonconservative. Cracks that exhibit a low signal to noise ratio may result in steam generator tube failure with catastrophic consequences. The NRC predictions of leakage is several orders of magnitude lower than that which can be expected during main steamline break accidents.

- The NRC cannot support the use of Generic Letter 95-05 for leakages of more than 1000 gpm. Jet erosion, steam generator tube vibrations, bending and buckling during main steamline break events can lead to leakages of thousands of gallons per minute.

In response to a question regarding the basis for steam generator tube leakage exceeding 1000 gpm, Dr. Hopenfeld said that it was estimated by RES subsequent to a tube leakage event at the Trojan Nuclear Power Plant. Some consultants from outside the NRC predicted a leakage of about 600 gpm.

Regarding the applicability of Generic Letter 95-05 to the Indian Point Unit 2 steam generator tube rupture event, Dr. Powers said that Generic Letter 95-05 does not apply to the Indian Point event because the crack occurred at the tube bend region rather than at the tube support plate.

Presentation by the Staff

Mr. J. Strosnider, NRR, discussed the staff's efforts associated with the DPO and steam generator issues. He reiterated the main points made at the October 10-14, 2000, Ad Hoc Subcommittee meeting. He said that a project integrated steam generator plan is expected to be submitted to the EDO in November 2000. He noted that the staff would continue to provide technical support to the ACRS in reviewing the technical merits of the DPO issue.

Conclusion

The Committee plans to discuss and approve the report of the Ad Hoc Subcommittee during the December 2000 ACRS meeting.

VI. Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues

[Mr. Amarjit Singh was the Designated Federal Official for this portion of the meeting.]

Mr John D. Sieber, Chairman of the ACRS Subcommittee on Fire Protection, introduced this topic to the Committee. He stated that the subcommittee heard presentations by and held discussions with representatives of the Boiling Water Owners Group (BWROG), NRC staff, National Fire Protection Association (NFPA), and Nuclear Energy Institute (NEI) regarding the BWROG proposal for using the safety relief valves (SRVs) and low pressure system (LPS) as a redundant safe shutdown path and NFPA 805 Standard and other fire protection related issues.

BWROG Presentation

Mr. Thomas A. Gorman, led the BWROG's discussion on the use of SRVs and an LPS as a redundant method to achieve safe shutdown as required by 10 CFR Part 50, Appendix R. The BWROG proposal states that use of SRVs and an LPS in support of Appendix R safe shutdown requirement, is consistent with the original design basis for BWRs. The proposal specifies a technically acceptable and safe means of achieving and maintaining either hot or cold shutdown. Mr. Gorman also stated that the same method is specified in the plant's emergency operating procedures as a means to achieve cold shutdown upon the occurrence of small break loss of coolant accidents in BWRs. The NRC staff supports the BWROG position on the use of SRVs and an LPS as a redundant path to achieve safe shutdown. The NRC staff plans to issue the safety evaluation report on this issue in the near future.

NEI Presentation

Mr. Fred Emerson, NEI, presented the industry views on implementation of the NFPA 805 Standard. The NFPA 805 standard includes six chapters and six appendices. The six chapters provide goals, performance objectives and criteria, a general approach for establishing a fire protection program and requirements, determination of fire protection systems and features, fire protection during decommissioning and permanent plant shutdown, and summary of referenced NFPA publications. The appendices provide explanatory material for the body of the standard, nuclear safety assessment, application of fire modeling in nuclear power plants, use of PSA methods, a deterministic approach for plant fire damage/business interruption, and referenced publications. The NFPA membership vote is scheduled for November 15, 2000. The membership can accept the standard as is, accept it as amended, return a portion of the standard to the NFPA 805 Committee, or return the standard to the committee. If approved, the Standards Council will issue the Standard on January 13, 2001. Mr. Emerson concluded the NFPA standard offers potential benefits to plants and the industry and has provided extensive support for its development.

Mr. Dennis Shumaker, presented the preliminary results of the Pilot use of NFPA 805 standard at Salem Generating Plant. He concluded that the NFPA 805 Standard provides the ability to understand and manage fire risk. It provides a basis for reviewing fire protection attributes by assessing actual plant configurations for risk and the modifications to identify risk implications and benefits which allows allocation of resources to achieve most risk benefits.

NRC Staff Presentation

Mr. Mark Salley, NRR, presented the brief overview of their perspective on the NFPA 805 Standard. There are changes from the existing Appendix R requirements. Performance criteria for nuclear safety (NS0 allows use of ADS/LPS for BWRs and feed and bleed for PWRs as only shutdown method). Performance-based, risk-informed criteria allows recovery of SSCs versus free of fire damage. The 72 hour cold shutdown, alternative/dedicated shutdown, and 8 hour emergency lighting requirements have been eliminated from the NFPA 805 Standard. There are outstanding technical and implementing issues that need to be resolved before the Standard can be adopted by the NRC staff.

Conclusion

The ACRS sent a letter to the EDO dated November 17, 2000, in the area of BWROG proposal on the use of SRVs/LPS as a redundant shutdown path.

VII. ABB/CE and Siemens Digital I&C Applications

[Mr. Amarjit Singh was the Designated Federal Official for this portion of the meeting.]

Dr. Powers, Acting Chairman, Plant Systems Subcommittee provided the ACRS a report regarding the October 31, 2000, meeting on ABB/CE and Siemens Digital I&C Applications. Discussions during the Subcommittee meeting with the representatives of the NRC, Westinghouse Nuclear Automation (formerly known as ABB/CE), and Siemens Corporation centered on the safety evaluations of their respective topical reports for Digital I&C Applications. Dr. Powers stated that the reasons for replacing the digital equipment includes: (1) analog equipment are becoming obsolescent and replacement is difficult to obtain; (2) plant components are aging and maintenance costs are increasing since vendors are not supporting replacement analog equipment; (3) digital equipment and components are readily available, with the potential for performance and reliability improvements; and, (4) nuclear utility replacement includes reactor protection systems, engineering safety feature actuation systems, monitoring systems, and balance of plant control and electrical systems. The NRC staff's safety evaluation reports for the Westinghouse and Siemens topical reports concluded that based on information provided by Westinghouse on the topical report and the review conducted the by the NRC staff, the design of the Common Qualified (Common Q) platform and TXS system meets the relevant NRC regulatory requirements and is acceptable for safety-related instrumentation and control (I&C) applications in nuclear power plants, subject to the satisfactory resolution of the generic open items.

Conclusion

The Committee will continue its discussion of the topical reports during future ACRS meetings.

VIII. License Renewal Guidance Documents

[Note: Mr. Noel F. Dudley was the Designated Federal Official for this portion of the meeting.]

Dr. Mario Bonaca, Chairman of the Plant License Renewal Subcommittee, noted that the staff and industry had revised the license renewal guidance documents for preparing and reviewing license renewal applications. He noted that the Subcommittee reviewed these revised documents during the October 19-20, 2000 ACRS Subcommittee meeting.

Mr. Christopher Grimes, NRR, stated that the public comment period had ended and that the staff was resolving the comments that have been received. He noted that the staff was scheduled to meet with the Commission on December 4, 2000 to provided a status of the guidance documents. Dr. Samson Lee, NRR, presented the changes that had been incorporated by the latest revisions to the standard review plan (SRP), the generic aging lessons learned (GALL) report, draft regulatory guide DG-1104, and NEI 95-10. The staff explained the disposition of stakeholders' comments.

Mr. Stephen Hoffman, NRR, presented the staff's position with the concern as to whether the emergency operating procedures (EOPs) are within the scope of the license renewal rule. He also explained how licensee voluntary commits will be handled during the period of extended operations. Ms. Tamara Bloomer, NRR, provided an example of how the guidance documents are structured to require one-time inspections. Mr. Jitendra Vora, RES, described the testing performed to resolve generic safety issue GSI-168, "Environmental Qualification (EQ) of Low-Voltage Instrumentation and Control Cables," and the implication of the test results for aging management of electrical cables during the period of extended operation. Mr. Barry Elliot, NRR, explained the aging management programs for metal components and for neutron embrittlement of reactor vessel internals.

The Committee and the staff discussed that the GALL report is not used during the scoping process, components affected by irradiation-assisted stress corrosion cracking, and programs for identifying potential aging effects on reactor vessel internals. They discussed the validity of using artificially aged electrical cables to represent 20 to 30 year old cables and aging management programs for electrical cables. They also discussed the use of EOPs and severe accident management guidelines in identifying components that may be within the scope of the license renewal rule.

Mr. Douglas Walters, NEI, explained that the industry is concerned about the license renewal process being used to impose additional unnecessary programs on licensees. He stated that aging management programs should manage the aging effect and not the causes of material degradation. Mr. Walters provided the examples of the additional license renewal requirements concerning inspections of inaccessible areas, maintaining the water chemistry program, and conducting one-time inspections.

The Committee and Mr. Walters discussed what information in the license renewal application is submitted under oath, crevice corrosion issues, and aging managing programs for cables.

<u>Conclusion</u>

The Committee provided a report to Chairman Meserve dated November 15, 2000, concerning this issue.

IX. Research Report to the Commission

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

The Committee discussed the 2001 ACRS report to the Commission regarding the NRC Safety Research Program. The Committee will continue to take an active role in reviewing the ongoing and proposed research activities and provide comments and recommendations to the Commission.

The Committee members discussed the format and content of the report and indicated that the focus of the report should be on what the longer term research program should be to assure the Commission's mission can be carried out efficiently and effectively in the future

X. Executive Session (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated October 23, 2000, to the ACRS comments and recommendations included in the ACRS report dated September 8, 2000, concerning the proposed high-level guidelines for performance-based activities.

  The Committee decided that it was satisfied with the EDO's response.

- The Committee discussed the response from the EDO, dated October 25, 2000, to ACRS comments and recommendations included in the ACRS report dated September 13, 2000, concerning proposed risk-informed revisions to 10 CFR 50.44, Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

  The Committee decided that it was satisfied with the EDO's response.

- The Committee discussed the response from the EDO, dated October 25, 2000, to ACRS comments and recommendations included in the ACRS report dated September 7, 2000, concerning assessment of the quality of probabilistic risk assessments.

  The Committee decided that it was satisfied with the EDO's response.

- The Committee discussed the response from the EDO dated October 30, 2000, to ACRS comments and recommendations included in the ACRS report dated September 14, 2000, concerning the pre-application review of the AP1000 standard plant design-phase 1.

  The Committee decided to continue its discussion of the issues included in its report and the adequacy of the EDO's response during future meetings.

B. Report on the Meeting of the Planning and Procedures Subcommittee

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the November ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the November ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed.

Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through February 2001 was discussed. The objectives were:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate.
- Manage the members' workload for these meetings.
- Plan and schedule items for ACRS discussion of topical and emerging issues.

CY2000 Self Assessment

The ACRS will hold its annual planning meeting in January 2001 and conduct its CY 2000 self assessment. Dr. Savio was assigned the task of selecting a small group of

ACRS work products that would be the subject of critical analysis by the ACRS members during the January 2001 planning meeting. The focus will be on selecting activities that would provide lessons learned.

ACRS Retreat for 2001

During the October meeting, the Committee agreed to have a retreat locally. A decision will be made on the dates for the retreat.

Proposed ACRS Meeting Dates for CY 2001

The proposed dates for CY 2001 ACRS meetings listed below were distributed to the members during the October 2000 ACRS meeting.

ACRS Meeting No.	Proposed Meeting Dates for 2001
--	January 2000 - No meeting
479	February 1-3, 2001
480	March 1-3, 2001
481	April 5-7, 2001
482	May 10-12, 2001
483	June 6-8, 2001
484	July 11-13, 2001
--	August 2001 - No meeting
485	September 5-7, 2001
486	October 4-6, 2001
487	November 8-10, 2001
488	December 6-8, 2001

ACRS Action Plan for CY 2001-2002

During the May 2000 ACRS meeting, the Committee approved the development of an ACRS Action Plan for CY 2001-2002. A draft Action Plan was prepared by the ACRS staff for review and comment by the ACRS members. This draft incorporates preliminary comments provided by the Planning and Procedures Subcommittee members. Subsequent to receiving the comments from the members, a revised draft will be prepared incorporating, as appropriate, the members' comments. The revised draft will be discussed by the Planning and Procedures Subcommittee during its December meeting. Subject to Subcommittee concurrence, it will be submitted to the full Committee for approval at the December 2000 ACRS meeting. Subsequently, the ACRS Action Plan and Operating Plan will be forwarded to the Commission.

Estimation of Resources for FY 2001

Due to the anticipated high workload facing the ACRS in FY 2001, it is important to plan how to use each member's time efficiently and effectively. Assuming the number of ACRS members remains constant throughout FY 2001, the maximum member time that will be available is 1,300 days.

During October's Planning and Procedures Subcommittee meeting, the need to manage better the number of Subcommittee meetings and the number of members participating in Subcommittee meetings was discussed. Senior staff engineers with input from Subcommittee chairmen were asked to revise the estimate of the number of Subcommittee meetings for FY 2001. The current estimate shows 36 Subcommittee meetings, 10 full Committee meetings and 1 retreat, consuming a total of approximately 1155 days. During the October ACRS meeting, the Planning and Procedures Subcommittee informed the Committee that it plans to scrutinize these proposed Subcommittee meetings to assess where some cuts might be made or combining of Subcommittee meetings might be done. This is to make sure that the maximum days available for members to work are not exceeded.

Election of Officers for CY 2001

The election of Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee will be held during the December 2000 ACRS meeting. In accordance with Section 8.4 of the ACRS Bylaws, those members who do not wish to be considered for any of the above offices should notify the ACRS Executive Director in writing at least two weeks prior to the December meeting.

C. **Future Meeting Agenda**

Appendix IV summarizes the proposed items endorsed by the Committee for the 478th ACRS Meeting, December 6-9, 2000.

The 477th ACRS meeting was adjourned at 1:30 p.m. on November 4, 2000.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 27, 2000

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

Sherry Meador

SUBJECT: PROPOSED MINUTES OF THE 477th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - NOVEMBER 2-4, 2000

Enclosed are the proposed minutes of the 477th meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate.

Attachment:
As stated

NATIONAL SCIENCE FOUNDATION

Advisory Panel for Methods, Cross-Directorate and Science and Society; Notice of Meeting

In accordance with the Federal Advisory Committee Act (Pub. L. 92–463, as amended), the National Science Foundation (NSF) announces the following meetings of the Advisory Panel for Methods, Cross-Directorate and Science and Society (1760):

Date/Time: December 4–5, 2000, 8 a.m.–5 p.m.

Place: National Science Foundation, 4201 Wilson Blvd., Room 920, Arlington, VA.

Contact Person: Bonney H. Sheahan, Program Director for Cross Directorate Programs; National Science Foundation, 4201 Wilson Boulevard, Arlington, VA 22230. Telephone: (703) 292–8763.

Agenda: To review and evaluate REU proposals as part of the selection process for awards.

Date/Time: December 13–15, 2000; 8 a.m.–5 p.m.

Place: National Science Foundation, 4201 Wilson Blvd., Rm. 365/920, Arlington, VA.

Contact Person: Paul Chapin, Program Director for Cross Directorate Programs; National Science Foundation, 4201 Wilson Boulevard, Arlington, VA 22230. Telephone: (703) 292–1733.

Agenda: To review and evaluate Infrastructure proposals as part of the selection process for awards.

TYPE OF MEETINGS: Closed.

PURPOSE OF MEETINGS: To provide advice and recommendations concerning support for research proposals submitted to the NSF for financial support.

REASON FOR CLOSING: The proposals being reviewed include information of a proprietary or confidential nature, including technical information; financial data, such as salaries; and personal information concerning individuals associated with the proposals. These matters are exempt under 5 U.S.C. 552b(c), (4) and (6) of the Government in the Sunshine Act.

Dated: October 16, 2000.

Karen J. York,

Committee Management Officer.

[FR Doc. 00–26985 Filed 10–19–00; 8:45 am]

BILLING CODE 7555–01–M

NORTHEAST DAIRY COMPACT COMMISSION

Notice of Meeting

AGENCY: Northeast Dairy Compact Commission.

ACTION: Notice of meeting.

SUMMARY: The Compact Commission will hold its regular monthly meeting to consider matters relating to administration and enforcement of the price regulation, including the reports and recommendations of the Commission's standing Committees.

DATES: The meeting will begin at 10:00 a.m. on Wednesday, November 1, 2000.

ADDRESSES: The meeting will be held at the Centennial Inn, Armenia White Room, 96 Pleasant Street, Concord, New Hampshire.

FOR FURTHER INFORMATION CONTACT: Daniel Smith, Executive Director, Northeast Dairy Compact Commission, 34 Barre Street, Suite 2, Montpelier, VT 05602. Telephone (802) 229–1941.

Authority: 7 U.S.C. 7256.

Dated: October 16, 2000.

Daniel Smith,

Executive Director.

[FR Doc. 00–26973 Filed 10–19–00; 8:45 am]

BILLING CODE 1650–01–M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards will hold a meeting on November 2–4, 2000, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Thursday, October 14, 1999 (64 FR 55787).

Thursday, November 2, 2000

8:30 A.M.–8:35 A.M.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 A.M.–10:45 A.M.: Proposed Final Report of the Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the revised version of the report and the staff's response to previous ACRS concerns.

11:00 A.M.–12:30 P.M.: Risk-Informed Regulation Implementation Plan (RIRIP) (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the update to the RIRIP.

1:30 P.M.–2:30 P.M.: Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed NRC framework for risk-informed changes to the technical requirements of 10 CFR Part 50 described in SECY–00–0198, Attachment 1.

2:30 P.M.–4:30 P.M.: Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open)—The Committee will hear a report by the Ad Hoc Subcommittee Chairman regarding the outcome of the October 10–14, 2000 subcommittee meeting and hold discussions with the DPO author and representatives of the NRC staff, as needed, on additional information related to DPO issues.

4:30 P.M.–5:30 P.M.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

5:30 P.M.–7:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Friday, November 3, 2000

8:30 A.M.–8:35 A.M.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 A.M.–10:30 A.M.: Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff, Nuclear Energy Institute (NEI), and National Fire Protection Association (NFPA) on the revised NFPA 805 standard, post-fire safe shutdown circuit analysis, and other related fire protection issues.

10:45 A.M.–12:00 Noon: ABB/CE and Siemens Digital I&C Applications (Open)—The Committee will hear a report by the Subcommittee Chairman on a subcommittee meeting on this matter and his recommendation regarding further review by the full Committee.

1:00 P.M.–3:00 P.M.: License Renewal Guidance Documents (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed Standard Review Plan for license renewal, Generic Aging Lessons Learned Report, Regulatory Guide, and NEI 95–10, "Industry Guidelines for Implementing the

Requirements of the License Renewal Rule."

3:15 P.M.–4:30 P.M.: Research Report to the Commission (Open)—The Committee will discuss the current status of the draft report.

4:30 P.M.–5:00 P.M.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

5:00 P.M.–5:15 P.M.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

5:15 P.M.–6:00 P.M.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

6:00 P.M.–7:30 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, November 4, 2000

8:30 A.M.–1:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

1:00 P.M.–1:30 P.M.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 11, 2000 (65 FR 60476). In accordance with these procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Mr. James E. Lyons, ACRS, five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting Mr. James E. Lyons prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Mr. James E. Lyons if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Mr. James E. Lyons (telephone 301–415–7371), between 7:30 a.m. and 4:15 p.m., EDT.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at http://www.nrc.gov/ACRSACNW.

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., EDT, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: October 16, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–26990 Filed 10–19–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Safety Research Program; Notice of Meeting

The ACRS Subcommittee on Safety Research Program will hold a meeting on November 1, 2000, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, November 1, 2000—8:30 a.m. until the conclusion of business

The Subcommittee will discuss the 2001 draft ACRS report to the Commission regarding the NRC Safety Research Program and related matters. In addition, it will meet with representatives of the NRC Office of Nuclear Regulatory Research to discuss the ongoing and proposed research activities, as needed. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Dr. Medhat El-Zeftawy (telephone 301/415–6889) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes in the proposed agenda, etc., that may have occurred.

Dated: October 16, 2000.

Sam Duraiswamy,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 00–26991 Filed 10–19–00; 8:45 am]

BILLING CODE 7590–01–P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 17, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
477TH ACRS MEETING
NOVEMBER 2-4, 2000

THURSDAY, NOVEMBER 2, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open)
   - 1.1) Opening statement (DAP/JTL/HJL)
   - 1.2) Items of current interest (DAP/NFD/HJL)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/HJL)

2) 8:35 - ~~10:45~~ 11:30 A.M. Revised Report of the Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (TSK/MME/MWW)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding the revised version of the report and the staff's response to previous ACRS concerns.

   Representatives of the nuclear industry will provide their views, as appropriate.

11:30 - 11:45
~~10:45~~ - 11:00 A.M. ***BREAK***

3) ~~11:00~~ 11:45 - 12:30 P.M. Risk-Informed Regulation Implementation Plan (RIRIP) (Open) (GA/MTM)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the update to the RIRIP.

   Representatives of the nuclear industry will provide their views, as appropriate.

12:40 - 1:40
~~12:30 - 1:30~~ P.M. ***LUNCH***

4) ~~1:30~~ 1:40 - 2:30 P.M. Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Open) (GA/MTM)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed NRC framework for risk-informed changes to the technical requirements of 10 CFR Part 50 described in Attachment 1 to SECY-00-0198.

   Representatives of the nuclear industry will provide their views, as appropriate.

5) 2:30 - 4:30 P.M. Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open) (DAP/SD/US)

5.1) Report by the Chairman of the Ad Hoc Subcommittee on DPO Issues regarding the outcome of the October 10-14 subcommittee meeting, proposed subcommittee recommendations, schedule for completing the review, and related matters.

5.2) Briefing by and discussions with the DPO author and representatives of the NRC staff, as needed, on additional information related to DPO issues.

4:55

6) 4:30 - ~~5:30~~ P.M. Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

4:55-6:50

7) ~~5:30~~ - ~~7:00~~ P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

7.1) Framework for Risk-Informed Changes to 10 CFR Part 50 (GA/MTM)

5:00-6:50 7.2) Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (TSK/MME/MWW)

~~7.3) Risk-Informed Regulation Implementation Plan (GA/MTM)~~

7.4) DPO on Steam Generator Tube Integrity (DAP/SD/US)

FRIDAY, NOVEMBER 3, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

8) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (DAP/JTL)

9) 8:35 - 10:30 A.M. Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues (Open) (JDS/DAP/AS)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff, Nuclear Energy Institute, and National Fire Protection Association (NFPA) on the revised NFPA 805 standard, post-fire safe shutdown circuit analysis, and other related fire protection issues.

10:30 - 10:45 A.M. *****BREAK*****

10) 10:45 - 12:00 Noon ABB/CE and Siemens Digital I&C Applications (Open) (REU/AS)
Report by the Subcommittee Chairman on a subcommittee meeting on this matter and his recommendation regarding further review by the full Committee.

12:10 - 1:10

~~12:00 - 1:00~~ P.M. *****LUNCH*****

11)	1:10 - 3:15 ~~1:00~~ - ~~3:00~~ P.M.	License Renewal Guidance Documents (Open) (MVB/RLS/NFD) 11.1) Remarks by the Subcommittee Chairman 11.2) Briefing by and discussions with representatives of the NRC staff regarding proposed Standard Review Plan for License Renewal, Generic Aging Lessons Learned Report, Regulatory Guide, and NEI 95-10, Industry Guidelines for Implementing the Requirements of the License Renewal Rule.. Representatives of the nuclear industry will provide their views, as appropriate.
	3:00 - 3:15 P.M.	*****BREAK*****
12)	3:30 - 5:30 ~~3:15~~ - ~~4:30~~ P.M.	Research Report to the Commission (Open) (DAP/MME) Discussion of the status of the draft ACRS report on the NRC Safety Research Program. Representatives of the NRC staff will provide their views, as appropriate.
13)	4:00 - 4:25 ~~4:30~~ - ~~5:00~~ P.M.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/HJL) 13.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings. 13.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.
14)	5:45 - 5:55 ~~5:00~~ - ~~5:15~~ P.M.	Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./HJL, et al.) Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.
15)	5:15 - 6:00 P.M.	Break and Preparation of Draft ACRS Reports Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.
16)	6:00 - ~~7:30~~ 7:45 P.M.	Proposed ACRS Reports (Open) Discussion of proposed ACRS reports on: 16.1) Framework for Risk-Informed Changes to 10 CFR Part 50 (GA/MTM) 5:55-6:50 16.2) Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (TSK/MME/MWW) Final 16.3) Risk-Informed Regulation Implementation Plan (GA/MTM) 16.4) DPO on Steam Generator Tube Integrity (DAP/SD/US) 16.5) Performance-Based, Risk-Informed Fire Protection Standard (JDS/DAP/AS) Final

11/4/00

16.6) Research Report to the Commission (DAP/MME)

11:00 - 12:40 16.7) License Renewal Guidance Documents (MVB/RLS/NFD) Final

<u>SATURDAY, NOVEMBER 4, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

17) 8:30 - 1:00 P.M. <u>Proposed ACRS Reports</u> (Open) - The Committee will continue its discussion and preparation of proposed ACRS reports listed under item 16.

18) 1:00 - 1:30 P.M. <u>Miscellaneous</u> (Open) (DAP/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**

- **Number of copies of the presentation materials to be provided to the ACRS - 35.**

APPENDIX III: MEETING ATTENDEES

477TH ACRS MEETING
NOVEMBER 2-4, 2000

NRC STAFF (November 2, 2000)
A. Levin, OCM/RAM
T. Hsia, OCM/NJD
J. Beall, OCM/EM
I. Schoenfeld, OEDO
S. Rosenberg, OEDO
G. Hubbard, NRR
K. Gibson, NRR
E. Throm, NRR
T. Collins, NRR
D. Diec, NRR
B. Huffman, NRR
D. Barss, NRR
D. Wrona, NRR
G. Kelly, NRR
G. Parry, NRR
J. Lehning, NRR
G. Bagchi, NRR
P. Ray, NRR
J. Hannon, NRR
E. McKenna, NRR
S. Magruder, NRR
J. Sebrosky, NRR
C. Carpenter, NRR
J. Staudenmeier, NRR
W. Lyon, NRR
X. Orechwa, NRR
J. Strosnider, NRR
W. Bateman, NRR
K. Raglin, HR
S. Basu, RES
J. Flack, RES
F. Eltawila, RES
A. Ramey-Smith, RES
A. Kuritzky, RES
J. Ibarra, RES
S. Arndt, RES
M. Drouin, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
C. Fleming, Winston & Strawn
R. Kennedy, RPK Structural Mechanics Consulting
O. Payne, FEMA
L. Hendricks, NEI
G. Thompson, IRSS
A. Wyche, SERCH Licensing/Bechtel
B. Henry, FAI
A. Heymer, NEI
B. Bradley, NEI
P. Negus, GE
A. Camp, SNL
E. Haskin, ERI Consulting
W. Pratt, BNL

NRC STAFF (November 3, 2000)
E. Weiss, NRR
J. Hannon, NRR
S. Dinsmore, NRR
S. West, NRR
M. Sallay, NRR
P. Koltay, NRR
J. Hyslop, NRR
M. Rubin, NRR
A. El-Bassioni, NRR
T. Ulses, NRR
P. Lain, NRR
L. Whitney, NRR
J. Dozier, NRR
P. T. Kuo, NRR
K. Rin, NRR
S. Lee, NRR
A. Hiser, NRR
J. Davis, NRR
F. Brubelicil
B. Elliot, NRR
W. Liu, NRR
C. Grimes, NRR
J. Straisha, NRR
S. Koenick, NRR
J. Peralta, NRR
S. Hoffman, NRR
C. Grattan, NRR
S. Mitra, NRR
D. Tabataba, NRR
T. Bloomer, NMSS
N. K. Stablein, NMSS

M. Wegner, RES
J. Vora, RES
J. Mitchell, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

C. Pragma, BWROG
T. Gorman, BWROG
F. Emerson, NEI
D. Shumaker, PSEG-Nuclear
B. Najafi, EPRI
J. Kenny, BWROG
P. Negus, GE
H. Fonticella, Dominion
J. Keys, Bechtel
R. Lofaro, BNL
Y. Liu, ANL
A. Mario, NEI
D. Walters, NEI

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 13, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
478TH ACRS MEETING
DECEMBER 6-9, 2000

WEDNESDAY, DECEMBER 6, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 1:00 - 1:05 P.M. Opening Remarks by the ACRS Chairman (Open)
   - 1.1) Opening statement (DAP/JTL)
   - 1.2) Items of current interest (DAP/NFD/SD)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/SD)

2) 1:05 - 3:00 P.M. Issues Associated with Core Power Uprates (Open) (MVB/GBW/PAB/AWC)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding issues associated with core power uprates, including: staff plans for developing a Standard Review Plan Section for power uprate reviews; staff position regarding the need for applying risk-informed decisionmaking in the review of significant power uprate applications; and other related matters.

   Representatives of the nuclear industry will provide their views, as appropriate.

3:00 - 3:15 P.M. ******BREAK******

3) 3:15 - 4:45 P.M. Differing Professional Opinion (DPO) on Steam Generator Tube Integrity (Open) (DAP/SD/US)
   - 3.1) Report by the Chairman of the Ad Hoc Subcommittee on DPO regarding conclusions and recommendations of the Ad Hoc Subcommittee on the technical merits of the DPO issues.
   - 3.2) Discussion with representatives of the NRC staff and the DPO author, as needed, regarding additional information on DPO issues.

4) 4:45 - 5:00 P.M. Subcommittee Report (Open) (GBW/PAB)
   Report by the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the status of review of the GE Nuclear Energy TRACG best-estimate thermal-hydraulic code.

5) 5:00 - 5:15 P.M. Subcommittee Report (Open) (REU/AS)
Report by the Chairman of the Plant Systems Subcommittee regarding ABB/CE and Siemens digital I&C applications and insights gained from meeting with the RSK on digital I&C in Germany during November 2000.

5:15 - 5:30 P.M. ***BREAK***

6) 5:30 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
6.1) Issues Associated with Core Power Uprates (MVB/GBW/PAB/AWC)
6.2) Response to the Commission request for a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and for more specific recommendations on how those weaknesses should be addressed (GBW/PAB)

THURSDAY, DECEMBER 7, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:35 A.M. Opening Statement by the ACRS Chairman (Open) (DAP/SD)

8) 8:35 - 9:30A.M. Meeting with NRC Commissioner Diaz (Open) (DAP/AS)
8.1) Remarks by the ACRS Chairman
8.2) Meeting with NRC Commissioner Diaz regarding the NRC Safety Research Program and other items of mutual interest.

9:30 - 9:45 A.M. ***BREAK***

9) 9:45 - 11:45 A.M. South Texas Project Exemption Request (Open) (JDS/GA/MWW)
9.1) Remarks by the Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff and South Texas Project Nuclear Operating Company (STPNOC) regarding the STPNOC's exemption request to exclude certain components from the scope of special treatment requirements in 10 CFR Part 50 and the associated NRC staff's Draft Safety Evaluation Report.

11:45 - 12:45 P.M. ***LUNCH***

10) 12:45 - 2:15 P.M. Control Room Habitability (Open) (TSK/PAB/AS)
10.1) Remarks by the Subcommittee Chairman
10.2) Briefing by and discussions with representatives of the NRC staff and the nuclear industry regarding issues associated with control room habitability and the staff and industry efforts in resolving those issues.

2:15 - 2:30 P.M. ***BREAK***

11) 2:30 - 4:00 P.M. Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Open) (WJS/NFD)
- 11.1) Remarks by the Subcommittee Chairman
- 11.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed final Regulatory Guide DG-1053, including the staff's resolution of public comments.

Representatives of the nuclear industry will provide their views, as appropriate.

12) 4:00 - 5:00 P.M. Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

13) 5:00 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
- 13.1) South Texas Project Exemption Request (JDS/GA/MWW)
- 13.2) Control Room Habitability (TSK/PAB/AS)
- 13.3) Proposed Final Regulatory guide DG-1053 (WJS/NFD)
- 13.4) Issues Associated with Core Power Uprates (MVB/GBW/PAB/AWC)
- 13.5) Response to the Commission request for a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and for more specific recommendations on how those weaknesses should be addressed(GBW/PAB)
- 13.6) DPO on Steam Generator Tube Integrity (DAP/SD/US)

FRIDAY, DECEMBER 8, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 8:35 A.M. Opening Statement by the ACRS Chairman (Open) (DAP/JTL)

15) 8:35 - 10:00 A.M. Proposed Modifications to the Commission's Safety Goal Policy Statement for Reactors (Open) (GA/MTM)
- 15.1) Remarks by the Subcommittee Chairman
- 15.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed modifications to the Commission's Safety Goal Policy Statement for reactors.

Representatives of the nuclear industry will provide their views, as appropriate.

10:00 - 10:15 A.M. ***BREAK***

16) 10:15 - 11:30 A.M. NRC Safety Research Program (Open) (DAP/MME)
- 16.1) Discussion of the draft ACRS report to the Commission on the NRC Safety Research Program.
- 16.2) Discussion with representatives of the NRC staff, as needed.

11:30 - 1:00 P.M. *****LUNCH*****

17) 1:00 - 1:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/SD)

17.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

17.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

18) 1:30 - 1:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

19) 1:45 - 2:15 P.M. Election of ACRS Officers for CY 2001 (Open) (JTL)
Election of a Chairman and Vice Chairman for the ACRS and a Members-at-Large for the Planning and Procedures Subcommittee for CY 2001.

20) 2:15 - 3:15 P.M. Break and Preparation of Draft ACRS Reports
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

21) 3:15 - 7:00 P.M. Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:

21.1) Proposed Modification to the Commission's Safety Goal Policy Statement for Reactors (GA/MTM)

21.2) South Texas Project Exemption Request (JDS/GA/MWW)

21.3) Control Room Habitability (TSK/PAB/AS)

21.4) Proposed Final Regulatory guide DG-1053 (WJS/NFD)

21.5) Issues Associated with Core Power Uprates (MVB/GBW/PAB/AWC)

21.6) Response to the Commission request for a detailed discussion on how the perceived weaknesses with industry-developed thermal-hydraulic codes may adversely affect the NRC's regulatory role and for more specific recommendations on how those weaknesses should be addressed(GBW/PAB)

21.7) DPO on Steam Generator Tube Integrity (DAP/SD/US)

21.8) Research Report to the Commission (DAP/MME)

<u>SATURDAY, DECEMBER 9, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

22) 8:30 - 1:00 P.M. Proposed ACRS Reports (Open)
The Committee will continue its discussion and preparation of proposed ACRS reports listed under item 21.

23) 1:00 - 1:30 P.M. Miscellaneous (Open) (DAP/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**

- **Number of copies of the presentation materials to be provided to the ACRS - 35.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
477th ACRS MEETING
November 2-4, 2000

[Note: Some documents listed below may have been provided or prepared for Committee use only. These documents must be reviewed prior to release to the public.]

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman
	1. Items of Interest, dated November 2-4, 2000
2	Revised Report of the Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants
	2. Spent Fuel Pool AccidentRisk Study presentation by NRR [Viewgraphs]
	3. Risk Analysis Results and Conclusions presentation by NRR [Viewgraphs]
	4. Consequence Assessment for Spent Fuel Pool Accidents presentation by RES [Viewgraphs]
	5. The Response of the Spent Fuel Pool to Postulated Accident Conditions presentation by R. Henry, Fauske & Associates, Inc. [Viewgraphs]
	6. Risks Associated with Spent Fuel Storage in High-Density Pools presentation by G. Thompson, Institute for Resource and Security Studies [Viewgraphs]
3	Risk-Informed Regulation Implementation Plan (RIRIP)
	7. Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50 presentation by M. Drouin, A. Kuritzky, RES; A. Camp, SNL; E. Haskin, ERI Consulting; T. Pratt, BNL [Viewgraphs]
	8. Risk-Informed Regulation Implementation Plan presentation by T. King, RES [Viewgraphs]
	9. Industry Perspectives on Risk Informing Decommissioning Regulations presentation by L. Hendricks, NEI [Viewgraphs]
4	Differing Professional Opinion (DPO) on Steam Generator Tube Integrity
	10. Report by the Chairman of the Ad Hoc Subcommittee [Viewgraphs]
	11. Differing Professional Opinion on Steam Tube Integrity Issues presentation by Dr. Joram Hopenfeld [Viewgraphs]
5	Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues
	12. NRR Redundant SRV/LPS Shutdown Activities presentation by E. Weiss, NRR [Viewgraphs]
	13. NRR Fire Protection Inspection Activities presentation by L. Whitney, NRR [Viewgraphs]
	14. Attachment 71111.05 [Handout]

7 License Renewal Guidance Documents

15. Memorandum from C. Grimes, NRR, to N. Dudley, ACRS/ACNW, Subject: ACRS Subcommittee Follow-up Actions, dated November 1, 2000 [Handout]
16. License Renewal Guidance Documents presentation by M. Bonaca, Chairman of the License Renewal Subcommittee [Viewgraphs]

13 Report of the Planning and Procedures Subcommittee

17. Future ACRS Activities - 478th ACRS Meeting, December 7-9, 2000 [Handout #13-2]
18. Final Draft Minutes of Planning and Procedures Subcommittee Meeting - October 31, 2000 [Handout #13.2]

14 Reconciliation of ACRS Comments and Recommendations

19. Reconciliation of ACRS Comments and Recommendations [Handout 14.1]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
Model	
2	Revised Report of the Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants
	1. Table of Contents
	2. Proposed Schedule
	3. Status Report, dated November 2, 2000
	4. Staff Requirements Memorandum, dated December 21, 1999
	5. ACRS Report dated April 13, 1999
	6. EDO Response dated May 26, 2000
	7. EDO Letter to the Commission, dated September 11, 2000
	8. NEI e-mail, regarding Appendix 2.b
	9. Gordon Thompson Report
3	Risk-Informed Regulation Implementation Plan
	10. Table of Contents
	11. Proposed Schedule
	12. Status Report dated November 2, 2000
	13. Draft Commission paper received October 25, 2000, entitled “Risk-Informed Regulation Implementation Plan (pre-decisional)
4	Proposed Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50
	14. Table of Contents
	15. Proposed Schedule
	16. Status Report
	17. SECY-00-0198 and Attachment 1
	18. ACRS report dated September 13, 2000, on 10 CFR 50.44
	19. ACRS report dated May 19, 2000, on defense in depth
5	Differing Professional Opinion (DPO) on Steam Generator Tube Integrity
	20. Table of Contents
	21. Proposed Schedule
	22. Status Report dated November 2, 2000
	23. Memorandum from W.D. Travers, EDO, to J. T. Larkins, ACRS Executive Director, dated July 20, 2000
	24. Memorandum from D. A. Powers, ACRS Chairman, to W. D. Travers, EDO, dated September 11, 2000
	25. Memorandum from D. Hopenfeld to W. D. Travers, EDO dated July 28, 2000
	26. Excerpt from the Differing Professional Opinion Consideration Document
	27. Agenda for the Ad Hoc Subcommittee Meeting on DPO, October 10-14, 2000
	28. List of Contentious Issues Reviewed by the Ad Hoc Subcommittee
	29. List of Documents Provided to the Members

9 <u>Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues</u>

30. Table of Contents
31. Proposed Schedule
32. Status Report dated November 3, 2000
33. Attachment 71111.05 Inspection Procedures for Fire Protection Baseline Inspection dated April 3, 2000
34. Draft NFPA 805 Subject: Performance-based standard for fire protection for Light Water Reactor Electric Generating Plants

10 <u>ABB/CE and Siemens Digital I&C Applications</u>

35. Table of Contents
36. Proposed Schedule
37. Status Report dated November 3, 2000

11 <u>License Renewal Guidance Documents</u>

38. Table of Contents
39. Proposed Schedule
40. Status Report dated November 3, 2000
41. License Renewal Guidance for ACRS Review of Generic Documents
42. Section from the Working Minutes for the Materials and Metallurgy Subcommittee Meeting, October 19-20, 2000
43. C. Chen, Apollo Consulting, Inc., "Report to USNRC ACRS on the Independent Review of SRP-LR and GALL Report for Containment Structures," dated October 8, 2000 [Internal Use Only]
44. S. P. Carfagno, Consultant, "Review of Adequacy of Staff Guidance for Reviewing License Renewal Applications," dated October 12, 2000 [Internal Use Only]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NOVEMBER 2-4, 2000
Date(s)

NOVEMBER 2, 2000
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
477TH FULL COMMITTEE MEETING

NOVEMBER 2, 2000
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NOVEMBER 2-4, 2000
Date(s)

NOVEMBER 3, 2000
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NOVEMBER 2-4, 2000
Date(s)

NOVEMBER 3, 2000
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
477TH FULL COMMITTEE MEETING

NOVEMBER 3, 2000
Today's Date

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Alex Marion	NEI
DOUG WALTERS	NEI

(1)

ITEMS OF INTEREST

477th ACRS MEETING

NOVEMBER 2 - 4, 2000

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
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NOVEMBER 2-4, 2000

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MISCELLANEOUS

NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF PUBLIC AFFAIRS

Office of Public Affairs **Telephone: 301/415-8200**
Washington, DC 20555-001 **E-mail: opa@nrc.gov**
Web Site: http://www.nrc.gov/OPA

No. S-00-20 October 4, 2000

[PDF Version (43 KB)]

THE REVISED REACTOR OVERSIGHT PROCESS

- - THE FIRST SIX MONTHS

DR. RICHARD A. MESERVE
CHAIRMAN
UNITED STATES NUCLEAR REGULATORY COMMISSION

at the

NEI STRATEGIC ISSUES ADVISORY COMMITTEE MEETING

OCTOBER 4, 2000

INTRODUCTION

Good evening. I am pleased to have the opportunity to address this distinguished group of nuclear industry leaders and would like to thank Ralph Beedle for his invitation. As Ralph requested, my remarks are focused on the revised reactor oversight process. Before I begin my remarks on this subject, however, I would like to reflect briefly on some of the events of the past year.

Overview

I have just completed my 11th month as Chairman of the Nuclear Regulatory Commission. Although the time has flown by swiftly, I am struck by the significant changes within the nuclear power industry in this brief period. As the economic deregulation of electric utilities proceeds, we are seeing significant restructuring among our licensees and the start of the consolidation of nuclear generating capacity among a small group of operating companies. This has no doubt brought significant changes to the lives of many of those in this room.

Even more striking than industry consolidation is the changing attitude, at least in the business world, toward nuclear power. Only a short time ago, pundits claimed that the deregulation of electricity markets would result in the premature decommissioning of many nuclear plants. Now, in contrast, a great deal of attention is focused on reactor license extension. We now expect that as much as 85 percent of the current fleet will be the subject of applications for license extensions. If these are successful, our existing plants will contribute to our Nation's energy security well into this century. In the last few weeks, there even has been talk of new construction in the United States. In short, in the course of a single a year, we have seen a remarkable change in the attitude toward nuclear generation in this country.

Unfortunately, I cannot claim that these developments in the nuclear industry can be causally connected to my assumption of the chairmanship of the NRC. The credit must go to an industry that has achieved remarkable gains in both economic and safety performance over the past decade.

Nonetheless, I believe that the NRC has played a role in setting the stage for the change we are observing. We have tried to establish a regulatory system that is fair, that is understandable, that is predictable, and that reaches its decisions with reasonable dispatch. I hope this regulatory environment has helped to facilitate valuable change. Perhaps equally important for the longer term, we have embarked on a journey to reexamine our regulatory foundations in a fashion that should allow further improvement in our processes. It is this process of change on which I will focus this evening.

NRC Priorities

Before I turn to what is changing, however, let me first emphasize the unchanging bedrock on which we must build our regulatory system. The fulfillment of the promise of nuclear energy is crucially and absolutely dependent on the maintenance of safe operations. The NRC's -- and the industry's -- highest priority must be the protection of public health and safety. If we fail in this joint obligation, the emerging optimism about nuclear generation will quickly disappear.

To accomplish its mission in the coming years, the Commission has established a set of four strategic objectives: to maintain safety, to increase regulatory effectiveness and efficiency, to reduce unnecessary regulatory burden, and to increase public confidence. In order to define how to achieve these objectives, the Commission recently published its Strategic Plan for Fiscal Years 2000 to 2005. The plan describes how we intend to accomplish our mission in terms of fundamental principles and strategies, and sets out both goals and measures to enable us to gauge our performance. The first and highest priority-- maintaining safety -- reflects our commitment to ensuring that good safety practices are utilized in the management and operation of nuclear facilities. This will be a significant challenge for the NRC and for our licensees during a time of consolidation and increased economic pressures.

To address the second and third objectives -- increasing effectiveness and efficiency and reducing unnecessary regulatory burden -- the NRC is seeking to focus attention on issues of the highest safety significance. To accomplish this goal, the Commission is utilizing probabilistic risk assessments, sometimes called probabilistic safety assessments, as tools to "risk-inform" our activities and regulations. These tools are not free of uncertainties and thus they are used to inform our processes and decisions, not to provide the sole basis for them. I will say more about this effort in a moment.

Finally, we must recognize that building and maintaining public trust is critical to the achievement of success. The NRC must both be and be perceived to be an independent, open and conscientious regulator. To achieve this aim, we must make public participation in the regulatory process more accessible and we must be objective in our examination of nuclear power plant performance.

Achieving these objectives presents special challenges in a time of transition. We must be ready to adapt, as appropriate, to the effects of changing financial pressures on our licensees -- pressures to cut costs coupled with pressures to achieve improved operating performance. NRC's focus on our mission and our performance goals as articulated in the Strategic Plan should serve as our guide through this turbulent period. Because we intend for the Plan to be a living document that will allow us to accommodate and adapt to changing circumstances, I invite your further comment and advice on it.

Informing Decisions with Understanding of Risks

As I mentioned, one of the key strategies for accomplishing our goals is to risk-inform our regulations through the use of Probabilistic Risk Analyses or PRAs. In addition to the revision of the oversight program that I will discuss in detail in a moment, we have initiated a program to evaluate the technical bases that underlie the requirements in 10 CFR Part 50 and to modify them, as appropriate, to focus on safety-significant issues. For example, we are moving forward with risk-informing so-called "special treatment" requirements, such as equipment seismic specifications and environmental qualifications. Other ongoing initiatives include the revision of the regulations or regulatory guidance governing decommissioning and fire protection. I envision a decade or more of work to apply safety insights in the reform of our regulatory requirements.

As we move forward with increased use of risk-informed techniques, we must also undertake the effort to explain our activities. Any modification of our regulatory processes cannot be satisfactorily achieved without acceptance of the approach by our staff and by our stakeholders. That is why the NRC is conducting mandatory PRA training for staff, holding workshops with the industry and the public, and generally reaching out to ensure our efforts in this area are both visible and understandable. We need to establish an understanding of our approach so that our stakeholders, including the general public, have confidence that our efforts to modify regulations are not whimsical, or designed to favor or to harm licensees, but rather are firmly based on the best information that is now available using the best analytical tools.

The NRC is committed to work to resolve the issues associated with risk-informing our regulations on a priority basis and to develop solutions in collaboration with our stakeholders. My vision for the final product of this complex process is a regulatory structure that is more aligned with safety, more internally consistent, and easier for our licensees and the public to understand and our staff to implement. As the process moves forward, I believe that the overall regulatory burden will be reduced without sacrificing safety.

Reactor Oversight Informed by Risk

The NRC's Revised Reactor Oversight Process is an outstanding example of what can be accomplished through the collaborative work of the NRC and its stakeholders. We have made significant progress over the last several years in the development, pilot testing, and initial implementation of this process. In light of the fact that we have completed the first six months of the initial implementation, it perhaps is now appropriate to reflect on our progress and the areas that have been identified as requiring further refinement.

As you know, the NRC has been widely criticized over the years for the way in which it has evaluated the performance of licensees. The evaluations were often viewed as subjective; licensees were at times surprised by the NRC findings and believed that NRC's conclusions were not supported by objective indicators of performance. Licensees perceived that inspectors imposed additional requirements that went beyond regulatory requirements. And the process was seen as too "retrospective," often producing outdated assessments of licensee performance. As a result, licensees believed that they were not given due credit for the current performance of facilities. Moreover, the public did not understand our inspection process, with the consequence that the process did not serve to inform public opinion adequately.

In response to these criticisms, the NRC chose to develop a new process for assessment of licensee performance. The goal was to have a process that would provide a more objective and understandable evaluation of plant performance, with a focus on operational aspects that were of the highest safety significance. The development of the revised reactor oversight process involved a significant effort by the NRC, NEI, nuclear utilities, and other external stakeholders, including public interest groups. As a result, the new oversight process can properly be seen as the product of a collaborative effort.

Nonetheless, when the revised process was approved for use across the fleet of plants in April of this year, the Commission described its action as "initial implementation." This was a carefully chosen phrase, which was intended to capture the fact that adjustments and mid-course corrections would be necessary and appropriate. Minor adjustments have been made. But we are also aware that more substantial adjustments may be necessary to further improve the oversight process.

Our review of progress to date, and those of various stakeholders, have identified four areas that warrant additional consideration. These areas include performance indicators, fire protection, reactor security and the documentation of cross-cutting issues. I will discuss each in turn. Let me simply note, however, that at the time of the implementation of the new program, I had expected far more problems than we have in fact encountered. Although there are issues to be addressed and problems to be corrected, the relatively smooth initial period of implementation is a credit to the foresight of the staff, the industry, and the other stakeholders in designing the system.

Performance Indicators

Performance Indicators -- or PIs as they are often called -- have proven in general to be useful tools for assessment of licensee performance. Comments from the industry indicate that the program is manageable without undue effort. The results of our inspections have shown that licensee personnel generally understand the guidance documents and the reporting requirements. Our inspections in this area have not identified significant problems, which gives us confidence in the accuracy of the data we are receiving.

While we are satisfied with the overall concept of PIs, we recognize that further improvements should be made. The goal is to have indicators that provide data which, when combined with inspection results, serve to represent overall licensee performance accurately, while at the same time not leading to unintended consequences. PIs associated with initiating events and mitigating systems have been identified as requiring further improvement. While we are actively working with stakeholders to develop improved PIs, the process of revising a PI is expected to take at least 6 to 8 months so as to assure that any new PIs do not create new problems.

For example, we have been working with an industry group formed by NEI to revise two PIs, both of which deal with reactor scrams. Some in industry expressed concern at the time of initial implementation that the original PIs sent the wrong message to plant personnel, potentially providing incentives for an operator to make decisions with adverse safety consequences. The revised indicators will be subject to pilot testing at about 20 sites in the near future. NRC staff is assured that these revised PIs continue to meet the intent of the original indicators, so that information adequate for assessing performance will still be obtained, but will not provide unintended incentives. Of course, external stakeholder input will be solicited in the development and piloting of revised PIs.

Another indicator that needs to be changed is the PI that tracks scrams followed by a loss of normal heat removal. With certain plant designs, an uncomplicated reactor trip can result in the isolation of several of the "normal heat removal" systems. At these facilities, although the plant might respond to a shutdown as designed, the event would nonetheless count against the PI. The original formulation of this PI unnecessarily penalized certain licensees because of such design features, and the PI will be changed accordingly.

The PI associated with the initiating events cornerstone, "Unplanned Power Changes," is also seen to have potential unintended consequences. While a revision to that PI is not as far along as the two scram-related PIs, the staff is working with the industry to develop an alternative that can be pilot tested in the near future.

Another issue relating to Performance Indicators concerns the unavailability of safety systems. Valid questions have been raised regarding the way in which we count safety-system out-of-service time against both the PI and the Maintenance Rule goals and the way in which unavailability is calculated. There are also inconsistencies in the way in which "unavailability" is assessed in the maintenance rule, in the PI, and in the counterpart WANO indicator. An industry working group sponsored by NEI has been established to address these problems. The NRC will continue to work collaboratively with the group and other stakeholders to develop solutions to these issues.

Fire Protection

The second area associated with the reactor oversight process that warrants additional consideration is fire protection. Questions have been raised regarding inspections to examine the effect of electrical faults on equipment associated with safe shutdown. As you may know, NEI and the BWR Owner's Group are engaged in an initiative to enable better definition of electrical fault characteristics related to fire protection and safe shutdown. As a result, the NRC has decided to postpone inspections in this area and to take no enforcement action while work is in progress to resolve these circuit analysis issues.

A second issue in the area of fire protection that warrants attention concerns the use of the Significance Determination Process (or SDP) for fire protection findings. Although the fire protection SDP is considered to be sufficient for evaluating findings, additional guidance is needed to ensure that it is applied consistently and appropriately. It appears that the staff at times has used overly conservative assumptions and unrealistic fire scenarios in characterizing the potential impact of fire-related inspection findings. Better guidance is being developed to clarify these issues with the goal of ensuring the SDP is utilized in a consistent and predictable manner.

Reactor Security

A third area that warrants further consideration concerns the treatment of reactor security. I am aware that stakeholders have raised a number of issues as a result of the manner in which Operational Safeguards Response Evaluations (or OSREs) are conducted and results are evaluated.

Let me begin by saying that the Commission recognizes that a substantial amount of work remains to be done in connection with the NRC's approach to security, wholly apart from issues related to inspection. I am particularly mindful of the fact that our policy on security matters has not been transparent and that we have not been consistent in our requirements. Although the design-basis threat

defined in our regulations (10 CFR Part 73.1) has been fairly stable, the adversary characteristics that define the details were revealed to licensees in the past only in the context of an OSRE and have varied over time and from site to site. In short, we have not had a disciplined process within the NRC to define the fundamental obligations of our licensees and we have not clearly and consistently communicated our expectations.

As a first step, the Office of Nuclear Reactor Regulation has sought to communicate a common set of guidelines that will be used for future OSREs. For example, the staff has developed and transmitted the specific list of adversary characteristics to the industry. We have received positive feedback as a result of this action and believe it has helped to clarify the agency's expectations. The staff is also working diligently with its stakeholders to enable the agency's endorsement of an acceptable Safeguards Performance Assessment Program, which could replace the OSRE program as an interim pilot program. In short, we are working with our stakeholders to bring predictability to the existing program. We will also continue to efforts to improve communications in this area.

For the longer term, the Commission is engaged in rethinking our fundamental policies in the area of security requirements. The Commission is working with the staff in developing a process for the systematic evaluation of the design basis threat and the adversary characteristics to which our licensees are expected to respond. We now await a rulemaking plan from the staff on the revision of the regulation that defines licensee obligations for security (10 CFR Part 73.55). I expect that the Commission will devote considerable effort in this area over the coming months.

Let me now turn to the classification of the findings from the OSREs in the reactor oversight process. The original approach for determining the significance of the OSRE findings was to use the reactor SDP to assess the significance of the equipment disabled by the adversary force. It has turned out, however, that this approach was somewhat misguided; we did not appropriately consider some of the unique aspects of OSRE exercises and their impact on traditional risk analysis. As a result, the staff is currently reviewing alternative approaches for determining the significance of these security-related findings. I expect adjustments to be made in this area as well.

Cross Cutting Issues

A fourth aspect of the new reactor oversight process that is undergoing consideration is the documentation of cross-cutting issues as so called "no-color" findings. From the start of the revised process, the Commission recognized that some issues should be documented even though they could not be evaluated under a specific cornerstone and its associated SDP. To address this concern, the new program contemplated that substantive cross-cutting issues -- such as those relating to human performance, problem identification and resolution, or a safety-conscious work environment -- could be documented in inspection reports. Since these issues are not typically processed for risk characterization by the SDP, they are not assigned a color to reflect the seriousness of the finding.

The staff and the Commission are sensitive to the fact that findings relating to cross-cutting issues have the potential to inject subjectivity into the inspection process. Moreover, I am aware that there have been some inconsistencies in the use of no-color findings. As a result, the staff is revising the guidance so that cross-cutting issues will be documented only in situations that involve findings that are more than minor in nature and that can be evaluated by the significance determination process.

In sum, we recognize that there are some important areas in which the new oversight process would benefit from revision. We are conducting mid-cycle workshops to obtain feedback, including one in which Oliver Kingsley and I participated yesterday. The staff is developing internal metrics to assess the performance of the program. Moreover, a review panel, comprising both NRC staff and external stakeholders, will evaluate the initial implementation of the program. Our goal is to define more precisely those areas to which more attention should be given, as well as to develop possible solutions. While the first six months of the Oversight program have generally been successful, we are aware of issues that warrant our attention. We intend to improve the program as it goes forward.

Longer Range Issues

Before closing, I would like to share with you some preliminary thoughts about two longer range issues that we face together. I will only touch on these now, with the modest objective of getting them on the table for your thoughtful consideration.

One lesson I learned quickly when I became Chairman is that too often we are forced by events to focus our attention on the day's most immediate problems. We have too little time, if any, to step back from the current storm to consider the larger climate, how it might change in the future, and what we need to do to prepare for it. My point is that with urgent issues to be addressed every day, we often push consideration of longer range issues to the future, often not worrying if we will get to the issues before the future gets to us. I suspect that everyone in this audience has similar experiences.

We have an obligation, however, to make time now to consider longer range issues so that we -- and our successors -- will be better able to manage the day-to-day issues that arise in the future. I want to describe two such issues for you now: managing low activity wastes and maintaining the core technical competence of the NRC. Both issues are, I would argue, in the vital long-term interests of the nuclear power industry.

First, on the question of low activity wastes, let me simply note that the future of low level waste disposal in this country is precarious. Our policies for low level waste disposal are simply not working. Even establishing a policy on release of slightly radioactive materials when risks are negligible (currently being considered under the rubric of "clearance") is proving to be difficult. As a Nation, we need to take a fresh look at waste issues with the aim of identifying alternative management strategies -- disposal and reuse -- that have better chances of success. We have to address these problems sooner or later.

Second, on the question of maintaining the core technical competence of the NRC, let me note that it is in both the public interest and the regulated industries' interest that the NRC have the capacity to reach sound technical judgments efficiently. To be able to respond to changing environments -- not just in the nuclear power industry but in other civilian uses of radioactive materials, such as in nuclear medicine -- the NRC has to be both sophisticated and agile. Your operations depend, for example, on our ability to write technically sound, risk-informed rules; to make sound licensing decisions without undue delay; and to conduct fair and meaningful oversight. The public depends on our ability to reach independent judgments on safety. We all benefit from a core NRC staff that is technically competent in the performance of these tasks and that is recognized as such.

In my judgment, the current NRC staff has the necessary qualifications and skills. The future, however, is uncertain. We have experienced declining real budgets over a number of years (until the slight upturn in this fiscal year). Moreover, we have had a loss of technically skilled personnel not only because of the loss of Full Time Equivalents (FTEs) in the budget, but also because budgetary retrenchment adversely affects morale. Further, we confront an aging demographic profile among our scientists and engineers. Our financial inability to make grants and contracts to universities has reduced opportunities for access to that community, as well as for the education and training of future nuclear scientists and engineers. And the government is challenged in recruiting the best and brightest. Combined, these circumstances should raise red flags.

I do not offer solutions to either the nuclear waste or the technical competency problems to you this evening. I mention them now only because such matters should be on the agenda for both the NRC and our stakeholders and I hope to stimulate your thoughts about them.

Conclusion

I would like to close by emphasizing again that, although the means by which we seek to attain our objectives may be changing, our fundamental mission -- the achievement of reasonable protection of public health and safety -- remains our abiding preoccupation. Our success is dependent on continuous and open dialogue with those we regulate and with the general public. I therefore welcome the opportunity to interact with you.

It has been a pleasure meeting with you this evening. Thank you.

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"The Role of Research in a Changing Environment"

Remarks of
Dr. Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission

at the

Water Reactor Safety Information Meeting

28th Annual Meeting
Bethesda, Maryland

Good morning. It gives me great pleasure to add my welcome to all of you. This is the 28th year that the Water Reactor Safety Information Meeting has been held, but it is the first that I have had the pleasure of attending. I am pleased to be able to address this opening session, particularly since the panel on the WASH-1400 study that follows this talk includes several friends. I am looking forward to hearing their reflections on that landmark effort.

The topic of my talk this morning is "The Role of Research in a Changing Environment." I hope to give you a sense of where I see the nuclear industry heading over the next several years, what the change means for the Nuclear Regulatory Commission, and the essential and vital role that research must play in ensuring that the NRC is equipped to deal with the challenges ahead.

The Changing Environment

The electric utility industry as a whole, and the nuclear sector of that industry in particular, is encountering a period of profound change. For the nuclear industry, the current turbulence is certainly greater than at any time since the Three Mile Island accident, and it may be unequaled in the history of civilian nuclear power electric production. The driving force for these changes is the deregulation of electricity pricing. In a competitive and deregulated market, the economics of generation is the essential consideration, and reliable nuclear power plants - particularly those for which the capital costs have been largely amortized - have become increasingly valuable assets. The changed view of nuclear generating assets is driving a number of initiatives: industry consolidation, plant sales, and license renewal. We are even beginning to see the first stirring of interest in construction of new nuclear power plants in the United States. These developments have significant implications for the NRC in general, and for our research program in particular.

The Role of Research in the Near Term

In the near term, NRC-sponsored research has a key role in developing the regulatory tools that the NRC will need to deal with the changing environment. The industry's focus on economics has a number of potential consequences. During a time of change, it is important to maintain vigilance so as to assure that safety is maintained. I am optimistic, however, that the changed economic circumstances could in fact lead to safety improvements. Industry consolidation has the potential to enhance nuclear plant safety as companies with many plants apply best practices and lessons learned across their entire fleets. Perhaps even more important is the reality that safe operation and economic operation should go hand-in-hand. A safe and well-run plant is reliable, stays on-line, and is able to avoid extended shutdowns, either as a result of the need to fix problems or because of regulatory action on the NRC's part to address a significant safety deficiency.

How do these developments affect the NRC? The NRC's statutory mandate, and our foremost obligation, is to provide reasonable assurance of adequate protection of public health and safety and the environment. We must never allow economic considerations to compromise our commitment to fulfill that obligation. However, that does not mean that we should not strive to operate as efficiently and effectively as possible. The price deregulation of the electric generation business means that the cost of safety regulation - both direct, from fees charged to licensees to recover the cost of the NRC's operations, and indirect, from the costs of regulatory compliance - come directly off the bottom line. Just as we owe the public the assurance that their health and safety are protected, we owe our licensees the assurance that the regulatory obligations that we impose on them minimize unnecessary burdens. We must therefore sharpen our focus to those areas that are safety-significant.

As you are undoubtedly aware, the NRC has embarked on a fundamental re-examination of our reactor regulations to consider risk explicitly. This move to risk-informed regulation builds on the foundation that has been established through NRC-sponsored research, beginning with the WASH-1400 study and continuing to the present day, to develop and apply quantitative methodologies for the assessment of reactor risk. The current focus of the agency's efforts in this area include risk-informing the technical bases of our reactor regulations and supporting the efforts to risk-inform the so-called "special treatment" requirements, such as quality assurance, environmental qualification, and technical specifications. We have also made substantial changes in our reactor oversight program, with a focus on safety and objectivity. Our research programs support these initiatives through evaluation of plant operational experience and development of risk-based performance indicators, thereby helping us to sharpen the safety focus of the oversight process.

The process of risk-informing our regulations requires that our tools for assessing technical issues be as realistic as possible. This move away from a traditional conservative, bounding approach has been made possible through a combination of operating experience, which now comprises more than 2000 reactor years in the U.S. alone, and experimental and analytical programs nurtured by NRC-sponsored research to develop better models of the behavior of a reactor during design-basis and beyond-design-basis accidents. One recent product of this research was an NRC-approved alternate source term for more realistic assessment of radiological consequences. Other ongoing research programs in this same vein include upgrading of the NRC's thermal-hydraulic codes to support review of industry-sponsored "best-estimate" accident analysis codes, and revisions to the pressurized thermal shock rule, based on a better understanding of radiation-induced embrittlement and fluid-structure interactions in reactors.

The drive for improved economic performance of operating plants is also manifesting itself in other ways. One outgrowth of the application of more realistic analyses is that the margins between calculated plant conditions and operational or regulatory safety limits are larger than previously demonstrated.

Licensees are naturally inclined to make use of these additional margins in ways that allow improved economic performance, such as by increasing fuel burnups, changing core power distributions, and increasing reactor power. (We refer to these as power uprates.) The research program on high-burnup fuels, along with the improved analytical techniques for accident analyses, are essential elements of the NRC's capability to review such initiatives. Licensees are also bringing on-line new technologies, such as digital I&C systems, that have the potential to increase plant reliability; the programs to assess the potential impacts of these new technologies are needed to ensure that the NRC is not an impediment to the appropriate deployment of these technologies.

The developments that I have just covered are extremely important both to the industry and to the NRC. However, I believe that the most significant near-term impact of the new environment is the widespread interest in nuclear plant license renewal. A few years ago, pundits claimed that a large number of nuclear plants would shut down prematurely. But the changed economic circumstances now make it worthwhile for a generating company to take steps to keep a plant operating beyond the term of the original 40-year license if the plant can operate safely and reliably for an extended period. As a result, we are seeing a strong interest in license renewal. We have renewed the licenses of two plants, Calvert Cliffs and Oconee, and are currently reviewing the applications for three other plants -- Hatch, ANO-1, and Turkey Point. Five more applications are expected in the current fiscal year, and the number in the years beyond 2001 continues to grow. About 40 percent of operating plants have indicated their intention to seek license renewal, and that fraction may ultimately reach 85 percent or more. If license renewal can appropriately be granted, nuclear power from existing plants will continue to make a significant contribution to our energy supply well into this century.

The core question is whether license renewal is appropriate. Fortunately, the NRC has been working on various aging-related issues for many years. As a direct consequence of these research programs, we have the technical bases to approach license renewal in a manner that focuses appropriately on the effects and management of aging. We were able to complete comprehensive assessments of the first two applications that we received for license renewal within the targeted schedule of 30 months. The challenge is to maintain this record as more applications are submitted. I believe we are up to the challenge, with the help of the tools that the NRC research program has helped to provide. As you may know, the NRC recently published its Generic Aging Lessons Learned, or GALL, report, reflecting insights gained as a result of our work to date on license renewal. (The report is available on the NRC's website.) There were many contributors to this important compilation of lessons learned, but a significant portion of the information is derived from reports prepared as part of our Nuclear Plant Aging Research Program. Without that technical foundation, I suspect that we would not be in the position to respond to the applications for license renewal with the depth of knowledge that we can now bring to bear.

Long-Term Developments and the Role of Anticipatory Research

I have concentrated thus far on areas that are of current or near-term interest to the industry and the NRC. Now, I would like to take out my crystal ball and speculate about what the future might hold for the industry, and discuss how the NRC's research programs with a longer-term focus support future NRC regulatory needs.

The overall environment for nuclear power is changing, in addition to the economic environment. Concern about global warming, for example, should focus attention on power technologies, such as nuclear, that minimize the emission of carbon dioxide and other potential "greenhouse gases." Similarly, consideration of energy security is seen to justify the support of a portfolio of energy technologies. The renewed interest in such matters may bring about a national reconsideration of the role of nuclear technology.

Perhaps as a natural reflection of these changes, the Department of Energy has begun to increase its research expenditures for civilian nuclear power technology after a period of essentially zero funding. The current program has several components. The Nuclear Energy Plant Optimization program, or "NEPO," focuses on existing plants, with research projects to develop new technologies to increase reliability, availability, and efficiency. By contrast, the Nuclear Energy Research Initiative, or "NERI," is to overcome scientific and technical obstacles to the future use of nuclear energy in the U.S. Many of the projects in the NERI program involve what is referred to as "Generation IV" reactor designs -- plants that might offer improved safety, lower capital and operating costs, proliferation resistance, and reduced waste production. A separate Nuclear Engineering Education Research (NEER) Program has funds that are earmarked for university research; a number of the projects supported by this program also deal specifically with advanced reactor concepts and related technology.

What might all of this mean for the future use of nuclear power? Again, I must offer an impressionistic and distant view. The NRC does not have a promotional role, and must remain agnostic on the question of whether the nuclear path should be resuscitated. Nonetheless, we must watch developments so that our processes do not serve as a needless impediment. As I said earlier, we are beginning to see the first stirring of interest among our licensees in constructing new plants. Given these circumstances, the NRC must prepare to deal with future demands.

Several years ago, we developed a licensing process for standardized plant designs. The idea was to permit the certification of a design in a fashion in which many key technical issues could be resolved once and for all, thereby stabilizing and streamlining the plant licensing process. An application to build a plant based on a certified design would not require examining issues that had been resolved during the certification. Upon approval of such application, a single combined construction permit and operating license would be issued. We have certified three standardized plant designs: General Electric's Advanced Boiling Water Reactor, the System 80-plus design of Combustion Engineering, which is now under the BNFL umbrella, and Westinghouse's AP600 passive plant design, which is also now a BNFL product. We have recently begun a review of Westinghouse's AP1000 design for possible certification. We have not received any applications to build these plants in the U.S., but I must note that two ABWRs are operating in Japan, and several more are planned.

I would also like to mention that the confirmatory testing and analysis programs conducted by the Office of Research were a key element in the review of the AP600 design. While these projects were specific to the AP600 review, they also contributed to the more general objective of upgrading the NRC's thermal-hydraulics codes, and initiated development of advanced risk assessment techniques that should ultimately contribute to risk-informed regulation for both current and future plants.

Some longer-term needs have already been defined for us. The end of the Cold War and the move toward reductions in nuclear weapons stockpiles have resulted in the need to manage significant amounts of weapons-grade plutonium. The strategy selected for this task involves using a portion of that material to create mixed-oxide fuel to be burned in commercial nuclear power reactors. We have already begun to prepare for the licensing of a MOX fuel fabrication plant, and have a research program to develop a technical basis for reviewing the license amendments that will be required to permit licensees to burn that fuel in their reactors.

Other longer-term issues are perhaps not so clear cut. We are following DOE's work on NERI and Generation IV reactors, so that we can understand the primary features of potential advanced reactor concepts. We recognize that our current reactor regulations may not translate well to the licensing of new reactor designs, particularly if the new designs are not water-cooled. Some of these issues may be resolved by our efforts to risk-inform our regulations, but, in other cases, the best approach may well be to start with a clean sheet of paper. This challenge is clearly a considerable one, but we must ensure that our research program has adequate resources to prepare us for the future. If we do not start now, we may find it extremely difficult to respond when we are called upon to begin to review these advanced designs.

Resources and Other Research Issues

My reference to "adequate resources" brings me to my next topic: research funding within the NRC. This is a subject that tends to generate a significant amount of discussion, especially among our licensees, since their fees currently pay our costs, including those for research.

Earlier this year, I spoke to a meeting of the Nuclear Energy Institute. The topic of the meeting was "change," and I stated that our research programs provide the basic technical capabilities that allow us to master change rather than to be its victim. I hope that I have conveyed throughout this talk how our research effort provides the technical "backbone" of the NRC's regulatory requirements. Our research program also plays a major role in maintaining the NRC's core technical competencies. This is essential not only from the standpoint of our relationship with our licensees, but also for developing and maintaining public confidence and trust in the NRC as a competent, technically knowledgeable regulator.

Despite the vital contributions of research to the NRC's activities, however, I must also acknowledge that over nearly the last two decades, the research budget has been significantly reduced. Accordingly, I - with the support of my colleagues on the Commission - have taken action to stabilize the budget to ensure that we have adequate resources for key research initiatives. I would also like to note that the bill containing the appropriation for the NRC's 2001 budget includes a provision to remove 10 percent of the NRC's total budget from our fee base, in 2 percent increments over a five-year period. We requested this provision in recognition that some of our activities, while valuable to the NRC's overall mission, do not directly affect the activities of our current licensees, but are of a more general benefit to the public. Instead of license fees, these funds would be supplied from general revenues. I am hopeful that this initiative will ease some of the pressure on our budget in future years.

The strain on the research budget is also occurring in other countries. Under such circumstances, international cooperation becomes essential so as to sustain major research initiatives that are beyond the means of any single country. We have many important international collaborations. I note that our international research partners are well-represented at this conference, and I would particularly like to acknowledge the contributions that you make to further our common understanding.

Our cooperative research efforts extend to the nuclear industry, as well. While we are mindful of the need to conduct independent assessments of important safety issues, there are times in which it is appropriate pool our resources and work with the industry to develop research programs. These include, for example, facility designs and test plans, with each party performing an independent analysis of the results. We have developed memoranda of understanding on the conduct of cooperative research with both the Electric Power Research Institute and the Department of Energy. I would like to acknowledge the value of these programs, as well.

We are also taking other steps to address the issue of resources and the broader question of the direction of the research program. A few months ago, we convened a group of experts drawn from a wide range of disciplines - academia, the nuclear industry, the public, Congressional staff, and other government agencies - to review the research program and provide suggestions regarding the role, funding, and focus of the research program. The initial reports of the participants were recently submitted and I very much appreciate the group's efforts. I note that several of the members of this group will be participating in a panel session on Wednesday morning to discuss their views on these important questions.

I have been able to touch upon only a portion of the research-related activities that are underway. Fortunately, some of the matters that I did not have time to address are the subject of later sessions. For example, you will hear presentations dealing with reactor decommissioning, dry cask storage, the transportation of spent fuel, and PWR sump blockage issues. The fact that I was not able to discuss these programs, and many others, in the course of this talk, does not mean that I ascribe any less value to them. I hope you will take the opportunity to learn about them first-hand during the remainder of the meeting.

<u>Conclusion</u>

Let me conclude by emphasizing once again the crucial role that our research programs play in meeting our current regulatory challenges and in preparing the NRC to deal effectively and efficiently with issues that may confront us in the future. Whether we are considering operating plants, new reactor designs that may be deployed a few years down the road, or other aspects of the nuclear power enterprise, such as decommissioning and waste disposition, we depend on the results of our research to establish the technical foundation for our regulatory activities. The organizational agility and responsiveness demanded by the rapidly changing environment in the electric utility industry is possible only if we have that firm technical foundation. I am proud of the past record of NRC's research efforts and am committed to sustaining the program in the future.

Thank you.

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Remarks of

Jeffrey S. Merrifield
Commissioner
U.S. Nuclear Regulatory Commission

at the

Water Reactor Safety Information Meeting

28th Annual Meeting
Bethesda, Maryland
October 25, 2000

Good Morning. Thank you very much for the opportunity to speak to you today. It is a pleasure to be here.

I would like to begin by reflecting on the speech I gave a year ago, and share with you my current views on the state of the NRC's research program. I also want to spend some time looking at the future and the role research will have in shaping our regulatory landscape. Frankly, my view of this landscape is remarkably different today than it was just one year ago.

Let me begin by reflecting on what I said last year and by giving you my current impressions of the NRC's research program. For the sake of those who are not familiar with my comments last year, I'll briefly summarize them. I challenged our Office of Research in 5 critical areas:

1. First, I stated that the growing economic pressures facing the NRC and our licensees would result in even greater scrutiny of each and every research dollar we spend. Given the fact that these economic pressures are undoubtably here to stay, I challenged our research staff to adapt to a higher standard of fiscal accountability and to more effectively demonstrate to their stakeholders that the NRC's research activities represent a valuable and prudent use of agency resources.
2. Second, I challenged our staff to reinvent the way in which they defend their research activities. Contrary to popular belief, good research does not speak for itself. I stated that if we have a

defendable research program, our staff must learn to market it, sell it, and clearly make the case for why it should be funded. If research activities are not important to the NRC's mission or closely linked to the agency's strategic and performance goals, then the NRC should sunset these activities and move on to higher agency priorities.

3. Third, I told our staff that while it is important to have a research program that is visionary in its approach and capable of providing an independent view on important agency matters, that independence must be carefully managed so that it does not lead to isolation. I challenged the research staff to work closely with our program offices - the primary end users of the research - to ensure that these parties share similar priorities and a consistent, or at least a compatible, vision of the future.
4. Fourth, I challenged our research staff and our stakeholders to stop their fixation with the bottom line of the research budget. From my perspective, the fact that the NRC's reactor research budget declined from over $100M in the early 1990s to around $40M in FY 2000 is not relevant to the decisions we are tasked with today. Budget realities dictate that we approach our research budget, line item by line item. I challenged those who argue that our research budget is too big, or too small, to move beyond the bottom line and instead make the case for either adding research initiatives that we **should** be doing but aren't, or for eliminating research initiatives that we **are** doing but shouldn't.
5. Fifth, I challenged our staff to seek ways to expand their efforts to capitalize on research work being conducted by the international nuclear community. As economic pressures drive greater fiscal restraint, we must leverage our international research efforts and not foolishly aspire to be the premier nuclear research agency in every discipline.

I believe the challenges I laid out last year were clear and meant to be constructive. However, some who attended the conference viewed my speech as an attack on research - somehow reflecting a lack of appreciation on my part for the contribution our research program makes to the effective fulfillment of our safety mission. With all due respect, I would argue that anyone who left last year's conference with that impression either did not listen carefully, felt threatened by the challenges, or did not recognize the realities we face. Let me make one thing perfectly clear - I believe our research program is absolutely essential to the long-term viability and success of our agency. However, if the program can't be managed properly, if its value can't be adequately conveyed to internal and external stakeholders, or if its links to the agency's strategic goals can't be clearly demonstrated, I assure you the agency will lose its ability to control the program's destiny. Others will decide that destiny for us. Like it or not, this is our reality.

With that said, let me now shift my focus to where I think our research program currently stands.

As I assess our research program today, I am pleased to say that it is healthier than it was just a year ago. Ashok and his management team deserve credit for what they have been able to accomplish in such a short period. While it is far too early to declare victory, the program has become more responsive to stakeholders, more fiscally disciplined, and frankly, more defendable. Given the importance of this matter, I believe it is essential that I articulate my thoughts more thoroughly.

First, let me focus on our external environment. The financial challenges facing our agency are greater today than they were last year, and I anticipate that these challenges will continue to intensify as our licensees - those that pay our fees - face greater competitive challenges associated with a deregulated electric market. This situation will only be compounded by the trend toward fewer reactor owners. It would be naive to think that distributing the fees associated with our research program among far fewer licensees will not bring with it an escalation of external scrutiny.

In regard to the research program itself, the Commission recently completed its review of the agency's research budget for FY 2002. As I promised at last year's conference, I vigorously challenged the merits of every line item in that budget. I am pleased to say that my expectations were exceeded. There were clear links between proposed research activities and the NRC's strategic and performance goals. There was a clear and defensible articulation of why each research project was necessary. There was less focus on the bottom line and greater focus on the merits of each project. In fact, without divulging too much about the agency's internal matters, the Commission, with my full support, approved a research budget virtually unchanged from that requested by our staff. Nobody in this room should underestimate the significance of that action.

As you know, I am a lawyer, not an engineer. Nonetheless, I understand the hazards associated with trying to identify a trend from a single data point, and I recognize that the recent budget cycle was but one data point. For me, another significant data point came during a recent visit I made to the Argonne National Laboratory, a lab that performs about $5.5M of research annually for the NRC. As you might expect, I was briefed on the status of the research initiatives they are conducting for the NRC. To my surprise, however, I was also briefed on how these initiatives are linked to the strategic and performance goals of the agency, and how the Argonne staff is exercising the fiscal discipline necessary to obtain the greatest return from every dollar the NRC spends. To me, this was especially gratifying because it demonstrated that the expectation of greater fiscal accountability that I and the other members of the Commission have been preaching has been embraced not only by our staff but also by our contractors.

A third data point came during a recent trip I made to Norway where I had the opportunity to visit the Halden Reactor Project. Over 100 nuclear organizations from around the world participate in research activities at Halden on such important matters as high burn-up fuel, MOX fuel, material properties, and human performance. While we spend less than one million dollars annually on research at Halden, our participation provides us with access to tens of millions of dollars of international research activities. My experience at Halden left me with little doubt that our staff is placing greater emphasis on leveraging our research dollars by looking for opportunities to capitalize on the research carried out by our international counterparts.

Data point #4 is not so encouraging because it represents a challenge that remains unanswered - a challenge requiring greater management attention. I voice this as constructive criticism in the hope that significant progress can be made this coming year. Despite efforts by our research staff, our attempts to reach out to stakeholders have resulted in limited success. Frankly, some of our internal and external stakeholders still do not have an appreciation of the value provided by our research initiatives. When the research management team attempts to articulate the value of the agency's research program, they are met with significant skepticism among our stakeholder communities - skepticism that is centered around the critical question, "Valuable to whom?" The accuracy of the perception is irrelevant. When you are dealing with stakeholders, perception **is** reality and thus it cannot be ignored.

Let me give you an example that illustrates my point.

In the May 8th edition of **Inside NRC**, Oliver Kingsley, Unicom's President of Nuclear Generation, provided his views of the NRC's research program. Mr. Kingsley stated that he does not support more money for the NRC's research program. More importantly, Mr. Kingsley added, "What would [the] NRC need research for? We've been operating plants for decades. Unless there's some type of advanced reactor program, I don't see a great deal of need [to fund NRC research]." Now, I have not talked to Mr. Kingsley about the article or the context in which his comments were made, but, assuming the article is accurate, the NRC cannot afford to underestimate the significance of his comments. As most of you know, Mr. Kingsley is responsible for the largest commercial nuclear program in the U.S.; a stakeholder that is well-respected throughout the industry for his emphasis on operational safety and technical excellence. The fact that such a well-informed and respected stakeholder does not see a need to fund NRC research should serve as a wake-up call to our agency. The fact that he made those comments in the same article that he discussed license renewal, the new reactor oversight process, and risk-informed regulation - all matters in which NRC research initiatives were instrumental - only serves to highlight just how high a hurdle our research program must overcome.

The message I want to leave today is that the NRC's research team has been successful in meeting many of the challenges I put before them last year. Nevertheless, challenges remain. Maintaining fiscal discipline and accountability requires continuous vigilance. Cultural changes of this magnitude typically take years before sustainable benefits are recognized. Our research staff must redouble their efforts to ensure that our stakeholders understand the value the agency hopes to derive from each and every research initiative. Frankly, if we are not successful in clearly defining the value of our research program, our critics will undoubtably define it for us. I am not willing to accept such a scenario.

The Future Landscape

I'm now going to change course and share my views on the future research needs of the agency. From my perspective, the future landscape of the nuclear industry, and the research associated with it, look much different today than just a few years ago. There are challenges looming on the horizon that could serve to reshape the commercial nuclear industry in the United States - challenges that will tax the NRC's technical capabilities. While some of these challenges may never come to fruition, I believe it is essential that the Commission assess our staff's readiness for them, and take the steps necessary to develop our capabilities at a rate commensurate with the pace of change we face. I'll take a few minutes to discuss some of these challenges.

1. If you have been reading the trade press, I am sure you are aware that several utilities are exploring the option of building new nuclear plants in the United States. Joe Colvin, the President of the Nuclear Energy Institute, recently told a gathering in London that a new plant may be ordered in the United States within 5 years, but that conditions for doing so may be ready in as little as 2 years. I am not prepared to address the likelihood of such an initiative, and I certainly do not want to give the impression that I am promoting it - as I am not. As a Commissioner of the NRC, to do so would be irresponsible. However, it would be just as irresponsible for us not to take the initial steps necessary to ensure that the staff is prepared to carry out its responsibilities should new plant orders emerge. We must critically assess our staff's technical and licensing capabilities to ensure that we can effectively and efficiently carry out our responsibilities. Given that we have not overseen the construction of a new plant in many years, we must assess our inspection assets to determine where there are gaps in knowledge and expertise. We must also critically assess the quality and stability of the regulatory infrastructure supporting Part 52. These tasks simply cannot be accomplished overnight. Thus, the NRC cannot wait until a licensee knocks on our door with an application. I believe the Commission must act soon to reallocate the funds necessary to at least assess whether the agency is up to the challenges associated with new plant orders. Clearly, the Office of Research will play a critical role in this effort.

2. We must also be prepared to address advanced reactor designs. It is not inconceivable that one day it may be more appropriate to call this conference the Water and Pebble Bed Reactor Safety Meeting. Again, I am not prepared to address the likelihood of such an eventuality, nor am I promoting the ongoing Pebble Bed initiatives; however, it would be irresponsible for us to stick our head in the sand and ignore reality. The reality associated with this issue is that one of our licensees, PECo Energy (PECo), is actively involved in Pebble Bed reactor initiatives in South Africa. According to recent comments attributed to Corbin McNeill, PECo's President and CEO, PECo could apply for a design certification in as few as 15 months. Such a development would be a real challenge for the NRC. The fact is, expertise associated with such a new reactor technology cannot be developed overnight. We must take steps now to develop this expertise so that we do not one day find ourselves incapable of carrying out our responsibilities associated with Part 52. I believe that our Offices of Research and NRR must, at a minimum, follow the activities in South Africa so that we can gradually build a prudent regulatory foundation and an appropriate level of expertise commensurate with the rate of progress made on the Pebble Bed initiative. One should not underestimate the safety and public confidence ramifications of falling short in our preparations.

   Clearly, our responsibilities in the area of new plant designs will not be limited to the Pebble Bed reactor. As you know, the NRC has already been approached by Westinghouse on an AP-1000 design. With escalating global warming concerns and the growing emphasis being placed around the world on energy independence, there is little doubt in my mind that domestic and international initiatives related to advanced reactor designs will intensify and that the NRC will be called upon to play a significant role in the safety reviews associated with these designs.

3. Another area that undoubtably will dot our landscape is the issue of extended power uprates. As many of you know, Alliant Energy is pursuing a 15% power uprate for their Duane Arnold facility. In addition, it appears that the Dresden and Quad Cities plants may submit similar

licensing amendment requests in late 2000 and that the Brunswick plant may do the same in 2001.

I am confident that the NRC is prepared to meet the technical challenges associated with 15% uprates. However, we should not kid ourselves that this represents the limit of future uprate requests. In a deregulated environment, our licensee's will look to squeeze as many megawatts as prudently possible out of their existing nuclear plants. How this incentive will manifest itself in the power uprate arena, I simply do not know. However, I do not believe it is unrealistic to expect that licensees could seek power uprates that extend beyond 15%. Should we face uprate requests of this magnitude, we have an obligation to all of our stakeholders to maintain safety and carry out our regulatory responsibilities in an effective, efficient, and realistic manner. In order to do that, we must ensure that our engineering analyses, our thermal-hydraulic code expertise, and our understanding of plant systems and safety margins, are sound. It is clear to me that our research program must be at the forefront of the NRC's efforts to address the realities we likely will face in the power uprate arena.

4. Steam generator research must also be a significant component of the NRC's research program in the future. It is essential that both we and our licensees develop better tube inspection methods, improve the accuracy of our data evaluation processes, and make further progress in our understanding of flaw growth predictions. Our goal must be to prevent, with greater certainty, tube failure events like the one that recently occurred at Indian Point 2. Now, some may argue that the Indian Point event was <u>not</u> of particularly high risk significance and thus preventing such events should <u>not</u> receive higher priority by the agency. I could not disagree more, and here's why. While we can argue risk numbers until we are blue in the face, I believe it would be irresponsible to assess the significance of such events so narrowly. This event certainly was significant to the public. It certainly was significant to the media. It certainly was significant to the New York Congressional delegation. It certainly was significant to our staff who faced the wrath of stakeholders and who ultimately will spend thousands of hours conducting event follow-up activities. It certainly was significant to ConEd, which is not only bearing the financial implications of an extended plant shutdown, but also the heavy burdens associated with facing a public that has lost confidence in their ability to operate the plant safely. So, as the NRC and our licensees go about assessing risk in the traditional safety sense, we must not ignore the enormous business, social, and political risks associated with a steam generator tube failure. Events like the one at Indian Point 2 could damage our credibility as a regulator and serve to erode public, Congressional, and to some extent, regulatory confidence in each of the 103 reactors operating throughout the U.S. Therefore, I believe we owe it to our staff and our stakeholders to continue the valuable steam generator research we are sponsoring at Argonne and to provide the resources necessary to further enhance our knowledge and capabilities in this very important area.

Our research program will also face challenges associated with the growing use of risk insights to support operational and maintenance decisions, licensing actions, and regulatory reforms. While we have started down the road toward risk-informing Part 50, I believe we are just now scratching the surface. At some point, licensees will undoubtably attempt to use risk-insights in applications that we cannot even imagine today, and the NRC will be called upon to effectively and efficiently carry out its regulatory responsibilities related to those applications. The NRC's research program must ensure that the agency's risk capabilities are sound and evolve in a manner commensurate with the applications they are being called upon to support. Our research program must proactively identify vulnerabilities and knowledge gaps, and ensure that our program offices recognize them, respect them, and compensate for them in their regulatory decisions. Let's face it, the use of risk insights is here to stay. The NRC can either manage them, or be managed by them. From my perspective, I believe our research program must be especially robust in this area so that our capabilities and expertise stay one step ahead of the applications we are being called upon to address. One should not underestimate the safety implications or the difficulty of this task.

5. Last but not least, I believe that the time has come for our research program to reassess whether the NRC's quality assurance (QA) requirements are continuing to produce outcomes

that are consistent with the agency's performance goals. As most of you know, Appendix B to Part 50 lays out the quality assurance criteria for nuclear power plants. It is a regulation that has served an important role in our regulatory framework for many years. However, during my visits to 60 nuclear units over the last 2 years, it has been common to see maintenance activities involving the replacement of plant components and equally common to hear licensee concerns over the difficulty they face finding suppliers that maintain an Appendix B QA program. During a recent briefing I received from our staff, I learned that the number of suppliers with Appendix B QA programs has declined. I also learned that this type of problem is not new to the nuclear industry. In our discussions on related matters like the ASME Code and the N-stamp process, I learned that during the 1989 time-frame, a number of utilities experienced difficulties obtaining replacements for components that were originally constructed in accordance with Section III of the ASME Code. In that case, the NRC was compelled to issue Generic Letter 89-09 to provide appropriate regulatory relief.

Here's my concern. Are the agency's quality assurance requirements inappropriately discouraging high-quality component suppliers from participating in the U.S. nuclear market, and if so, do we fully understand the consequences? Are these requirements unwittingly inhibiting potential safety enhancements? More broadly, are the agency's QA requirements consistent with our performance goals of maintaining safety, reducing unnecessary regulatory burden, increasing public confidence, and carrying out our responsibilities more effectively, efficiently, and realistically? I understand the commercial-grade dedication process and I am familiar with our ongoing efforts in the risk-informed arena. While these are important initiatives, I believe the time has come to take a more fundamental look at our quality assurance requirements to determine whether they are effectively and efficiently achieving their intended outcomes.

I believe our staff should take a fresh look at Appendix B and our regulatory framework surrounding quality assurance. The staff should also assess whether there are insights that can be drawn from more widely utilized national and international quality standards. For example, the ISO 9000 family of standards has become one of the most widely utilized quality standards in the world, already adopted by thousands of organizations, many of which have outstanding quality records. While I understand the staff has conducted some limited comparisons between Appendix B and ISO 9001, quite frankly, that's simply not enough. I want to know why ISO banners are rapidly going up as Appendix B banners are coming down. I want a better understanding of what is driving suppliers away from Appendix B quality assurance programs. We owe it to our stakeholders to critically assess Appendix B, compare it to more widely accepted quality standards like ISO 9001, identify where there are differences, and assess whether these differences are meaningful in our efforts to protect public health and safety. If particular Appendix B requirements cannot be linked to safety or to the NRC's performance goals, we should consider eliminating them. To the extent feasible and prudent, we must seize opportunities to bring Appendix B in line with widely accepted quality standards. Simply put, I believe the Commission must provide the resources necessary to ensure the agency's quality assurance requirements <u>are not</u> inappropriately driving high-quality component suppliers from the U.S. nuclear market, <u>are</u> aligned with our performance goals, and <u>are</u> in the best interests of the American people.

In closing, these are very dynamic times for the NRC and the U.S. nuclear industry, and the future promises to be even more dynamic. As I have outlined, there are many challenges on the horizon - challenges that bring with them opportunities. For us to seize these opportunities, the NRC must have the vision and leadership to not only recognize them, but to be prepared for them. Our research program must play an instrumental role in this process. It must be visionary in its approach and must provide the technical foundation necessary to support the bold decisions our agency will be called upon to make. I believe the next 10 years will prove to be some of the most challenging and rewarding our research program has ever faced. Winston Churchill once said, "A pessimist sees the difficulty in every opportunity; an optimist sees the opportunity in every difficulty". I am an optimist and I truly see tremendous opportunities embedded in the difficulties facing our research program. As a Commissioner, I believe I have an obligation to ensure that our research program and our staff are well-positioned to seize these opportunities. I assure you, I take that obligation very seriously.

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"Perspectives on Research's Role in Regulation"

The Honorable Greta Joy Dicus
Commissioner
U.S. Nuclear Regulatory Commission

at the

Water Reactor Safety Information Meeting

28th Annual Meeting
Bethesda, Maryland
October 24, 2000

Good morning ladies and gentlemen. I am very pleased to have the opportunity to speak to you at this conference. Today I intend to provide my perspectives on some of the activities within Research which I believe are a very important part of the NRC mission. In particular, my remarks will be focused on the following: (1) how important I perceive the office of Research's role to be; (2) current initiatives which benefit from Research's support; and (3) challenges which provide opportunities to shape Research's future, But first I would like to recall what Congress had in mind when it formed the office of Research.

The Energy Reorganization Act of 1974 stipulated that the Director of Nuclear Regulatory Research shall perform such functions as the Commission shall delegate including: (1) developing recommendations for research deemed necessary for performance by the Commission of its licensing and related regulatory functions, and (2) engaging in or contracting for research which the Commission deems necessary for the performance of its licensing and related regulatory functions.

Of note as stipulated in the Act, was that the head of every other Federal Agency shall cooperate with respect to the establishment of priorities for the furnishing of such research services as requested by the Commission for the conduct of its functions. This is a mandate that we should continue to exploit to the maximum benefit for our research activities. As I'm sure many of you have heard from those within the NRC the research budget has decreased from a high of over 200 million in the past to about 42 million in the last fiscal year. This is due in part to the fact that the nuclear industry has matured. This has provided challenges for the NRC to get the most from each research dollar to support both short term and longer term activities that support the Agency's mission.

As I hope that most of you know by now, in meeting this challenge the NRC has adopted a strategic plan that articulates four primary objectives: (1) to maintain safety; (2) to improve public confidence; (3) to make our regulatory processes more effective, efficient, and (4) to reduce unnecessary regulatory burden. In the process of meeting these objectives I believe we are benefitting in that we are focusing our research efforts to gain the maximum benefit for the stakeholders we serve.

Over our recent history the NRC has been challenged to redefine or at least re-examine Research's role and future direction. It pretty much began with an issue paper, Direction Setting Issue 22 written in 1996, which posed fundamental questions about what role research should play in meeting the Agency's mission and it also provided several recommendations. Since then there have been several status reports to the Commission and one of the outcomes of NRC's efforts to increase its efficiency and effectiveness has been to fold many of the responsibilities previously charged to the NRC Office of Analysis and Evaluation of Operational Data into the Office of Research.

I'm sure some of you may have heard that recently a panel was convened to review what role research should have in our current and future regulatory environment in an effort to gain input from stakeholders. And I will do a little advertising and mention that tomorrow, my fellow Commissioners Merrifield and McGaffigan will be part of a discussion on this subject. I'm pleased by the diversity that has been brought to the panel which is chaired by former Commissioner Kenneth Rogers and includes membership from academia, public interest, industry, other federal agencies, former NRC executive managers, as well as, congressional and senate staff representation. I have studied some of their preliminary recommendations and I understand that they are only about half way through their study; but I am intrigued by the scope of their individual recommendations. And while the focus of the panel so far has not specifically identified the role of research with respect to materials issues, I am sure this panel will give appropriate consideration to those research activities because there are many materials challenges that go hand in hand with the future of nuclear power in the U.S. Also, I noted a question posed by several members of the panel was whether the Offices of Nuclear Reactor Regulation and Nuclear Materials Safety and Safeguards should also be solicited to provide input. However, even if these offices do not participate as part of this panel, I am confident that any future changes to the direction of our research programs would surely be weighted in on by all NRC stakeholders at the appropriate juncture.

One particular aspect I would hope to see as an outcome of this effort would be recommendations regarding what minimum staffing level or minimum core areas of research might be necessary to maintain research's ability to respond to future challenges. Recently, I read where the technology boom in the Silicon Valley and other similar technology centers is taking the best and brightest from government research laboratories. It can only stand to reason that the same might hold true for our University expertise base. Because the chance to become an internet millionaire is very alluring, I think we might need to start looking at ways to ensure our current base of technical expertise which we frequently draw upon, the national laboratories, does not become too watered down. One thing I am very mindful of every time I review the NRC's budget is, what level of funding will ensure that RES can efficiently and effectively function to support the NRC mission while maintaining highly qualified respected technical staff who produce high quality products.

CHALLENGES THAT TRANSLATE TO OPPORTUNITIES

Regulatory Initiatives

One of NRC's management challenges is to develop and implement a risk-informed, performance based regulatory oversight program. We are answering this challenge by working with industry on risk-informing 10 CFR Part 50 through several initiatives focusing on what has been referred to as "special treatment" requirement and piloting risk-informing regulations such as 10 CFR Part 50.44. Years ago when research for much of today's regulatory framework was conducted using experience, testing programs, defense-in-depth philosophy and engineering margins incorporated to account for areas of uncertainty, we didn't have the benefit of quantitative estimates of risk. This framework has served our nation quite well for many years, and we don't expect to throw it out and start over. Rather, given that the margin of safety is a recurring issue in the implementation of risk-informed regulation we must not lose sight of the benefits of research to identify which margins do -- and which do not -- contribute to safety. As we move into the 21st century, continued research directed at quantifying margins should NOT be confused with the perception that while reducing regulatory burden, to support risk-informed regulation we are also improving safety. Remember we now have much commercial operating experience and research to consider as a result of the ensuing years of inquiry and challenges the nuclear industry has brought us all -- and we should try and benefit from this knowledge in every way possible.

We must also be mindful of the impact of industry deregulation and license transfers on those we regulate. While we will always conduct our activities so as to be true to our mission to regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment -- that does not mean that we cannot support industry initiatives such as the development of technical basis to support license renewal, or risk inform our current regulatory requirements and appropriately reconcile these requirements to allow licensee's to more efficiently and effectively focus their resources in those areas where their impact on improving safety will have the greatest result.

I believe the NRC has been responding to the changing environment well, but I'll be the first to agree we can continue to do more. And I believe that the staff is up to this challenge. For example, earlier this year we launched implementation of the new power reactor oversight program for all plants. If you will recall, last year we piloted the new program with a few plants, made adjustments and subsequently initiated the program for all reactor licensees in April of this year. A key part of this initiative is that risk insights were used and we are making every aspect of it transparent as possible -- one just needs to visit the revised reactor oversight program webpage accessible through the NRC's homepage to see what I mean. And while there is agreement that lessons learned since its recent wide scale implementation suggest that more changes to the program will probably be necessary, I think we all can agree that overall the effort has been a success to a large part because of stakeholder input. And I think experience gained through research has contributed to this effort and we are currently looking to Research in conducting studies aimed at developing data and methods to risk-inform the various performance measures.

Decommissioning is another area where we have been working with stakeholders to remedy inefficiencies in our current regulatory framework which was largely established from the perspective of operating reactors. As a result, in the power reactor area, the NRC is taking a formal look at our whole approach to decommissioning to see if we need to create a new regulatory framework, and to see if we can focus on the areas of greatest risk. This year the staff proposed an integrated rulemaking plan and has been discussing its recommendations with stakeholders. Research is contributing by examining various analytical tools and studying the viability of possible approaches to decommissioning, such as entombment.

Participation & Communication

Closer involvement and improved dialog with the industry and all stakeholders is required in order to better define and focus NRC research efforts. Only through such interactions will it be possible to obtain broader support for research programs. And meetings like this one are just one of the many ways we can actively achieve education of and input from all of our stakeholders. Looking at the various topics that will be discussed I see there are papers from both the staff and industry experts which give me the impression that we are making progress toward working together on challenging technical issues. Another way to raise consciousness for the value of research is to ensure that our research products provide relevant recommendations toward improving our regulatory structure.

I think if we are going to be successful in making the case for maintaining the current funding levels or perhaps even increasing funding we will have to get better at communicating and demonstrating how research dollars have benefitted safety and are providing products to support concerns such as license renewal, power up-rates, increased fuel burnup, and mixed oxide fuels. To quote Mr. Thadani "we would have had a difficult time moving as rapidly as we did on license renewal without anticipatory research." Much of which contributed significantly to the beginnings of the first Generic Aging Lessons Learned report. Obviously, explaining to stakeholders the costs of such efforts in terms of anticipatory research dollars should increase confidence in what we consider to be forward thinking research activities.

Timeliness of Our Activities

However, there is one aspect with respect to our research activities that I am very sensitive to, which is timeliness of outcomes. Frequently, we find real world uses for our anticipatory research, but we end up taking many years to see the results to fruition. Our research programs must be timely and responsive to both internal and external stakeholders. I suppose resources could be part of this mix, but I would also argue that management oversight might also be a contributing factor. I believe one way to ensure we can improve performance in this area is to get input early on from all stakeholders. I can assure you that while I am on the Commission I will be very critical of research activities that lend themselves to improving our regulatory infrastructure but do not have an aggressive schedule for seeing their contribution through to improving our regulatory framework.

Cooperation with Independence

As resources for research become more subject to challenge, I think we can really benefit by maintaining our existing relationships and looking to develop new relationships and cooperative agreements with our Federal colleagues, private sector stakeholders, and international colleagues. For example, I noted that with respect to one of the topics that will be discussed, digital instrumentation and control, the research staff have identified that digital failure assessment methods are currently used by defense and aerospace industries to determine types of failures and their impact on overall safety. Also, the railroad industry has experience with systems which we foresee as being potentially viable for the nuclear industry. Obviously the practical experience and research results from these parties could serve as a minimum -- as a starting point as the NRC begins to determine and gather information on digital instrumentation and control failure rates to better assess the risk from the increased use of this type of equipment. Another example that has already yielded significant results is the successful collaboration between the NRC and industry in the 1980's on research projects under the auspices of the Nuclear Plant Aging Research Program which lead to development of much of the basis for our conclusions that license renewal was viable. And just recently at the conference of the International Atomic Energy Agency the U.S. and France signed an agreement on scientific and technological cooperation for developing an advanced type of nuclear reactor. Under the agreement, the two countries will cooperate in developing an advance type of nuclear reactor, establishing research programs in materials and combustibles for future reactors and in developing medical and industrial uses for radio-isotopes. Another very good example of working to achieve unique solutions as the nuclear industry moves to a deregulated environment is the Research-Energy Power Research Institute memorandum of understanding which advocates sharing available data and sharing costs of generating new data, when required. I would hope this would go a long way towards ending disagreements over data which has traditionally been one area where contentions arise between the staff and industry when facing new challenges. This is especially useful as those facilities which the NRC has traditionally relied upon are scaling down or closing down as the need for research in new areas has dwindled as the industry has matured and also in the face of declining budgets. In the area of cooperation aimed at risk-informing regulations, I noted that last month the NRC's PRA Steering Committee and the NEI Risk-Informed Regulation Working Group held their second meeting to discuss the various initiatives which could be used to support the framework for risk-informing 10 CFR Part 50.

While working with the industry is becoming more of a reality in our current environment we must also remain vigilant to insure that the public's confidence in the NRC's independence is not eroded by blindly accepting results from others. Confirmatory research or anticipatory research for industry initiatives has been, is, and will always be necessary to insure we maintain our charge as an independent regulator. I think upon reflection of the lessons we have learned from Millstone and those we are still learning from Indian Point Unit 2, I am convinced that communicating what we do and how we do it in a way that is open to all stakeholders is very important to maintaining public confidence.

Research's overall budget has decreased. However, as I just stated the NRC has a management challenge to redefine the role of research in a mature industry I think we can't be too short sighted as we implement this challenge. If you look at the curmet challenges facing Ford and Firestone I think you will agree that the consequence of not aggressively investigating suspicious safety problems has resulted in a significant loss of credibility for both these companies. We cannot allow that to happen to the NRC. There are many past and recent examples which demonstrated the benefits of being a forward thinking organization and I will use remarks made by the Chairman which I whole heartedly agree with, to illustrate my point. " . . . Virtually every major new initiative that the agency has undertaken over the past few years, license renewal, risk-informed regulation, design certification of advanced reactor designs, assessment of digital instrumentation and control systems, steam generator tube integrity programs, and the new source term, have required technical guidance derived from our research programs. I do not believe that the NRC would have either the reputation that it enjoys as a world leader in nuclear regulation, or the credibility and the technical wherewithal to proceed with the implementation of a risk-informed regulatory structure, were it not for the contributions of the Office of Research."

We are hearing rumblings today that utilities are beginning to explore the possibility of building a new reactor in the United States. I can't see how the NRC can wait until we see an application at the door to begin exploring what new regulatory requirements might be necessary if an application was received. At some point, as soon as the picture focuses a little more on this issue, we might need to embark on what some might perceive to be anticipatory research. Performing the research now to better understand where the uncertainties lie with possible new technologies will not only provide short term benefits but long term benefits if and when we see future power plant applications.

CONCLUSIONS

In closing, I would just like to add that my vision of the NRC Office of Research would be a center of excellence and source of expertise. This center would maintain a cadre of reactor and materials safety specialists in various key areas, with independent and unbiased expertise across a broad spectrum of advanced nuclear technology, to provide the technical basis for robust and transparent regulatory decisions. Experimental facilities and resources would be maintained to ensure our ability to respond in a timely manner to new or emerging issues. The office would complement the front-line regulatory activities of the agency and independently examine evolving technology and anticipated issues. While I am pleased to see that we are soliciting stakeholders more in what we do, I would expect we do more and focus on making what we produce more timely and more useful.

One final thought that I would like to leave with you regards the issue of funding. The current funding process of NRC research through users fees has the unintended impact of discouraging user support in the face of economic pressures. As a result, some are starting to pose the question as to where the NRC's research activities, if not the anticipatory activities, should be funded from the general fund rather than from those we regulate, since the public at large benefits from activities such as establishment of new regulatory requirements to support new reactor designs for example. I find this proposition very interesting and must study it more before I reach my final conclusion, but nevertheless I appreciate new ideas from our stakeholders as we continue to explore the future role of research and what mix of anticipatory and confirmatory research is optimum.

Thank you for your attention, I would be pleased to answer any questions you might have at this time.

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"Short and Long-term Prospects for U.S.-Japan Nuclear Cooperation"

Remarks of
DR. RICHARD A. MESERVE
CHAIRMAN
UNITED STATES NUCLEAR REGULATORY COMMISSION

at the

U.S.-Japan Workshop on Nuclear Energy

October 5, 2000
Decatur House, Washington, DC

Introduction

Thank you, Mr. Chairman. I am pleased to participate in the U.S.-Japan Workshop on Nuclear Energy. This is the first time I have had the pleasure of addressing members of the Santa Fe Energy Seminar hosted by Washington Policy and Analysis. However, as I look at the participants in the audience I see many whom I had the pleasure of meeting at the Japan Atomic Industrial Forum conference held last April in Tokyo, and others with whom I have worked in the United States.

I will begin my talk by addressing the state of the nuclear industry and regulation in the United States. I will then turn to our collaborations with Japan and what we might be able to do in the future, as the global nuclear enterprise evolves.

The Dynamic Environment in the United States

In the United States -- and to a great extent also in Japan -- the regulatory environment is now extraordinarily dynamic. We are in a period of transition in several dimensions, probably experiencing more rapid change now than at a time since the beginning of the almost 50-year history of civilian nuclear power.

While focusing on today's regulatory environment is essential, the rate of change we are experiencing strongly suggests that the future will continue to impose increasing demands on us all. I firmly believe that we, government and industry, have an important obligation to prepare for the future to which today's changes are moving us.

I do not pretend to be able to predict the future with certainty. Who would feel secure in forecasting in light of the changes of the past few years? Nonetheless, we all know of issues that will surely be with us in the long run if we do not act to resolve them in the interim. I believe that together we can positively affect change so that the regulatory environment of tomorrow is even better suited to assure excellence in nuclear safety than that of today.

Economic Regulation

The most important agent of change in the U.S. today is the price deregulation of electricity generation. Engineering and technology developments of the past two decades have made it possible to decouple electricity generation from transmission and delivery, so that it is no longer technologically necessary to include generation *per se* as part of the public utility function.

In the past few years, more and more states have initiated actions to deregulate electricity prices. One consequence has been a rapid restructuring of the U.S. nuclear industry, characterized by mergers, consolidation, joint operating agreements, and other changes. We have also seen a steadily increasing interest in nuclear plant license renewal. Whereas only a few years ago, the conventional wisdom was that nuclear power was an industry with limited, if any, future in this country, industry observers now speak of the future in optimistic tones. Only a few years ago, the NRC expected most, if not all, plants would be decommissioned at or before the end of their 40-year license terms. Now we hear estimates from industry leaders that licensees of up to 85 percent of U.S. plants will seek extension of their licenses. As a result, the existing fleet of nuclear plants may contribute to our energy security well into this century. Industry leaders are also beginning to consider strategies for the development of new plants, a thought that was almost unthinkable only a few years ago. In short, we are seeing a publicly unnoticed renaissance in nuclear power in the U.S.

Safety Regulation

What else has changed? The U.S. approach to safety regulation. Our national economic system is based on free, open markets that are moderated by government to achieve social objectives that are not valued by markets. Markets do not ordinarily value public health and safety or environmental protection, and the generation of electricity is increasingly no longer the responsibility of a public utility. Therefore, the government will continue to regulate nuclear activities to achieve external social objectives.

Government can, however, take advantage of what has been learned over the past four decades about nuclear operations and safety to do a more efficient job of regulation. And that is what we are trying to do.

As the industry has evolved, we have accumulated data and developed new tools for analyzing data so that today we have a much better understanding of the nature and magnitude of the risks to public health and safety that arise from nuclear operations. We are applying that accumulated understanding to inform our activities, with the goal of focusing attention on specific features commensurate with the risks that they pose. Government regulation always comes at a cost, and ideally those costs should only burden markets to the extent of the benefits that society derives as a consequence.

This aspect of the current dynamic environment is NRC's own creation. Under the existing regulatory regime, the U.S. nuclear industry has accumulated an impressive safety record. But keeping our regulatory system up to date with technical developments serves to meet a fundamental obligation to the public, to industry, and to government. It is for this reason that we have started the significant and necessary task of reform. We are seeking to examine our regulatory system-much of which was enacted on a deterministic basis in the early days of nuclear power-in order to adopt new regulations based on risk insights.

Let me be more specific. In the early 1990s, the Commission determined that the science of quantitative risk assessment had matured sufficiently to permit the use of probabilistic safety assessments in "risk-informing" our regulations. By "risk-informed," we mean that risk insights are considered, along with more traditional deterministic assessments, in evaluating licensee performance and proposed actions, such as in-service inspection and technical specification changes. We are also making our regulations more "performance-based," so that licensees are given more latitude in how they meet regulatory requirements. We have already overhauled our plant oversight process, using performance indicators along with risk-informed inspection techniques, to provide a better focus on safety. And other regulatory requirements, such as those governing special treatment requirements -- requirements imposed on nuclear equipment that go beyond commercial standards -- are now under revision. We are embarked on a decade of work to bring our regulations up to date with the best current knowledge.

The NRC'S Approach to Nuclear Safety Assurance

In order to provide a foundation for our regulatory activities, we have established a set of four strategic objectives for our regulatory program:

- to maintain safety;
- to increase effectiveness and efficiency;
- to reduce unnecessary regulatory burden; and
- to increase public confidence.

The objective of maintaining safety is and must remain our most fundamental goal. But we are hopeful that the reform of our regulatory system will enable us to maintain our focus on safety, while simultaneously increasing effectiveness and efficiency and reducing increasing burden. With the benefit of risk insights we can determine which parts of our regulatory system should be enhanced or which should be reduced or eliminated.

The fourth objective, to increase public confidence, may be the most challenging task of all. I cannot stress too strongly the need for all of us to communicate effectively with the national and international public about nuclear technology. It is essential that our regulatory actions both be fair and be perceived as fair. A key to achieving this perception of fairness is to be open and accessible. Initiatives we have undertaken to strengthen public confidence include establishing a website through which the public may get information about our activities, and increasing interactions at all levels with our "stakeholders." These interactions include public meetings, workshops, and other outreach efforts.

To summarize, we believe our efforts to risk-inform our regulations will serve to focus our regulatory activities on the issues of highest significance for safety, while also satisfying our other strategic objectives. In this way, we expect to meet the challenge of the changing economic environment for nuclear power in the U.S. and to assure that our licensees maintain a vigilant approach to nuclear safety.

We could not accomplish our objectives, however, without the participation of our international partners. As each of you is well aware, nuclear technology is international in scope. Over 400 nuclear power plants are now operating in more than thirty nations, supplying about one-sixth of the world's electricity. In several countries, nuclear power supplies over 70% of domestic electricity production. New nuclear capacity is planned or is being considered in a range of nations: some with established civil nuclear programs, such as Japan, France and the Republic of Korea; some with mid-size programs, such as India and China; and some that do not currently have nuclear power, such as Bangladesh, and Vietnam. Regulation, construction, and ownership all have international components. Regulators leverage research money through joint international activities. Construction consortia, drawn from multiple countries, build the plants. And, foreign ownership of plants, while often limited by national laws, is becoming more common.

Whether or not to use nuclear power; the number, size, and location of the plants; and the methods used both by plant operators and regulatory agencies to ensure their safe operation and public protection are matters that each Nation must decide for itself. But there is a vital need for international cooperation to ensure that safety is <u>*the*</u> fundamental consideration in the use of nuclear technology. As we have all experienced, a nuclear accident anywhere has consequences that transcend national borders. If nuclear power is to continue to make a significant contribution to the world's energy supply in the coming century, we -- utilities, vendors, researchers, regulators, and policy makers -- must all work together to ensure that those who use the technology have safety as their primary goal, and that they have the necessary resources and technical capabilities to achieve that goal.

Prospects for U.S.-Japan Nuclear Cooperation

This leads me to answer the question implicit in the title of this segment of the Workshop, "What are the short and long term prospects for U.S.-Japan nuclear cooperation?" In my view the prospects are excellent, and are visible in every aspect of our respective nuclear programs. The U.S. and Japan have, and will continue to work effectively to enhance nuclear safety at home and abroad through international and national legal frameworks, regulatory cooperation, and commercial enterprise.

Our governments already coordinate closely in connection with the international legal instruments that provide the basis for cooperative programs. For example, the United States and Japan acted together on the recently negotiated conventions on nuclear safety, liability, and the safety of spent fuel management and the safety of radioactive waste management. These instruments effectively serve to acknowledge that, although the decision to employ nuclear power is a sovereign decision, there are legitimate transnational interests in assuring that the technology is used in safe and responsible manner. The cooperative programs which are enabled by these legal instruments are, in turn, implemented through an interconnected web of multilateral nuclear safety organizations and bilateral activities in which both our countries are actively engaged.

Cooperation between our national regulatory agencies has grown and, in my view, should continue and expand. The exchange of information between the United States and Japan on operating experiences and regulatory issues helps to promote good safety practices and to discourage poor ones. I am firmly committed to continuing the NRC's active role in cooperative exchanges with Japan. NRC staff members participate with their Japanese colleagues in international conferences, such as professional society meetings and on many international working groups, such as those organized by the International Atomic Energy Agency and the Nuclear Energy Agency. On the Commission level, my fellow Commissioners and I have met with our Japanese regulatory and industry counterparts to discuss perspectives on nuclear regulation and ways in which to promote adherence to the highest degree of safety assurance. The NRC's Office of International Programs coordinates technical information exchange agreements, including an active program with Japan. One of the most valuable methods for sharing information and experiences is through the assignment of staff to other organizations, and the NRC is proud to have hosted many regulatory staff from the Japanese Ministry of International Trade and Industry and from the Science and Technology Agency. We have also sent our regulatory staff to Japan to learn from the valuable experiences of our international colleagues.

The nuclear industry also clearly recognizes the need for and value of international cooperation and technical information exchange, and hosts forums to promote free and open discussion of research, operational experiences, emerging technical and safety issues, and other related topics. As the first country to build and operate an Advanced Boiling Water Reactor (ABWR), which is a product of a cooperative venture between Japan's Toshiba and Hitachi and GE Nuclear Energy of the United States, Japan is a leader among nations in establishing the environment for the future of nuclear power generation. In fact, leaders of the industry in the U.S. have been quoted recently as looking toward the experience with the ABWR in Japan as providing a basis for eventual development of new plants in the U.S. In April I had the opportunity to visit the ABWR plants at Kashiwazaki-Kariwa, and was impressed with what I saw.

One other subject in the area of U.S.-Japan collaboration deserves special attention: the role of our cooperative research programs. The contributions of our international research partners are essential to the vitality of the NRC's research program. One unfortunate aspect of the changing environment -- in the United States, in Japan, and almost everywhere -- is the tightening of the available budget, in general, and of the research budget in particular. However, the need for research continues. It provides the technical foundation for new regulatory initiatives, such as risk-informed regulation. It positions the NRC and our regulatory counterparts to deal with new technology and new industry initiatives. Research enables the development of state-of-the-art analytical tools and the ability to respond to the emerging technical and safety issues that arise as our operating reactors grow older.

While I could not possibly list all of the international cooperative programs in which the NRC takes part, among the most prominent is our very valuable collaboration with the Japan Atomic Energy Research Institute (JAERI). One example is the confirmatory testing program conducted in the ROSA-Large Scale Test Facility at JAERI's Tokai laboratory for the NRC certification of Westinghouse's AP600 design. This extensive series of tests, simulating design-basis accidents and transients, as well as multiple-failure scenarios, provided valuable data for the validation of the NRC's thermal-hydraulic analysis codes, and provided the NRC staff with insights into the way in which the AP600's unique passive safety systems would behave during such events. Another program of note is the ongoing testing program on high-burnup fuel in JAERI's Nuclear Safety Research Reactor. During my April trip to Japan I visited the JAERI facilities and observed tangible evidence of the tremendous value of our international cooperation with Japan.

A Lesson Drawn: Embrace and Prepare for Change

What lessons can we derive from this brief sketch of the current dynamic nuclear environment? Perhaps the most fundamental is that change is an inevitable consequence of current activity. As we go through life we gain experience and our universe changes. A Greek philosopher, once wrote that "you [can] not step twice into the same rivers; for other waters are ever flowing on to you."[(1)] We cannot ignore change.

It is human nature to seek to avoid change, and organizational settings exacerbate that tendency. A mind set against change exists in all organizations, whether in the United States or in any country around the world. Our responsibility, however, is to embrace change, to engender the attitude among our colleagues and the public that change offers opportunities for doing our work better, and to prepare for the future.

That means that we -- all of us -- must accept the responsibilities not only of maintaining our institutional capacities to meet current needs, but also of building the capabilities to meet the changing needs that will be thrust upon us. The NRC not only must be effective and efficient as a regulator, but also must be an agile agency, dynamically responsive to changes in the communities that it regulates and anticipating those communities' future needs. The same holds true for Japan's regulatory institutions. And, each can accomplish its goals more readily if we help each other through our cooperative activities.

It is for this reason that I have sought to provide the NRC's perspectives -- and my own -- of the value of U.S.-Japan nuclear safety cooperation. We share a common obligation to assure the responsible use of nuclear technology. Working together, we can meet that obligation.

Thank you.

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NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF PUBLIC AFFAIRS, REGION II

61 Forsyth Street, Suite 23T85, Atlanta, GA 30303

No. II-00-56 September 22, 2000

CONTACT: Ken Clark (404)562-4416/e-mail: kmc2@nrc.gov
Roger D. Hannah (404)562-4417/e-mail: rdh1@nrc.gov

NRC STAFF TO MEET WITH HATCH NUCLEAR OFFICIALS TO DISCUSS INITIAL PLANT LICENSE RENEWAL INSPECTION

Officials from the Nuclear Regulatory Commission will meet with Southern Nuclear Operating Company management at 1:00 p.m. (EDT) on Thursday, October 5 at the Hatch nuclear power plant visitor center near Baxley, Georgia, to discuss results of the NRC's initial inspection of the plant's license renewal program.

The meeting is open to observation by the public, and NRC officials will be available at its conclusion to answer questions from interested observers.

NRC officials said the agency's initial inspection of the Hatch license renewal program, conducted from September 11-15 of this year, indicated that the company has properly set the scope of key plant systems, structures and components and is correctly implementing the methodology for screening them as described in its renewal application, submitted to the NRC this past February 29, for the two-unit plant.

NRC officials will discuss the license renewal process and the schedule for plant Hatch.

Luis A. Reyes, administrator of the NRC's Region II office in Atlanta, said the agency's initial inspection is the first of three license renewal reviews and was conducted to verify that the company's license renewal program is implemented consistent with its license renewal application and pertinent regulations. He said subsequent NRC inspections will verify that programs are in place to manage the material condition of the plant's systems, structures and components.

The full inspection report is due to be issued by November 19 and will be posted on the NRC's internet web page at: http://www.nrc.gov/OPA/reports/renewal.htm

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION
Office of Public Affairs **Telephone: 301/415-8200**
Washington, DC 20555-001 **E-mail: opa@nrc.gov**
Web Site: http://www.nrc.gov/OPA

No. 00-163 October 19, 2000

NRC ANNOUNCES OPPORTUNITY FOR EVIDENTIARY HEARING ON TURKEY POINT NUCLEAR POWER PLANT LICENSE RENEWAL

The Nuclear Regulatory Commission has announced the opportunity to request an evidentiary hearing on a request for renewal of the operating licenses for Units 3 and 4 of the Turkey Point Nuclear Power Plant. The facility is operated by Florida Power & Light (FP&L) Company, and is located near Florida City, Florida.

FP&L is seeking a 20-year extension of the current Turkey Point licenses, which expire on July 19, 2012, for Unit 3, and April 10, 2013, for Unit 4. The NRC staff has begun reviewing both safety and environmental aspects of the application, received on September 11.

The deadline for hearing requests is November 13 -- 30 days after the Federal Register notice was published. By that time, petitions requesting a hearing and leave to intervene must be filed by anyone whose interest might be affected by the license renewal and who wishes to participate as a party to the proceeding.

Petitions for a hearing and leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, Attention: Rulemakings and Adjudications Staff. They may also be delivered to the NRC Public Document Room at 11555 Rockville Pike (first floor) Rockville, Maryland, 20855-2738. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, and to Thomas F. Plunkett, President - Nuclear Division, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408-0420.

More information about the opportunity for hearing may be found in a Federal Register notice published on October 12. A complete list of the steps in the license renewal review process and more detailed information about nuclear plant license renewal can be found at: http://www.nrc.gov/NRC/REACTOR/LR/index.html on the NRC web site.

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PNO-IV-00-029 -Nebraska Public Power District

October 18, 2000

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO-IV-00-029

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV staff on this date.

Facility	**Licensee Emergency Classification**
Nebraska Public Power District	__ Notification of Unusual Event
Cooper Nuclear Station	__ Alert
Brownville, Nebraska 68023	__ Site Area Emergency
Docket: 50-298	__ General Emergency
License No.:DRP-46	X Not Applicable

Subject: Reactor Scram with Plant Shutdown Greater than 72 Hours

Description:

On October 14 at 3:24 a.m. CDT, the Cooper Nuclear Station main generator tripped on a Phase C differential current signal when a current transformer on the main transformer failed. A reactor scram followed on turbine governor valve fast closure. Peak vessel pressure was 1068 psig and was controlled with the bypass valves. Safety relief valves did not lift during the event. Safety system actuations included an alternate rod insertion and Containment Isolation Groups 2, 3, and 6 on low reactor water level caused by shrink. Additionally, a feedwater pump trip occurred on high reactor water level caused by excessive feeding and swell. The lowest reactor vessel water level recorded was approximately -20 inches on the wide range indicators. Operators were able to quickly restart Feedwater Pump A and feed the reactor vessel. Feedwater Pump B failed to reset because of high differential pressure across the oil filter.

In addition to repairing the current transformer, licensee mechanics located an approximately 5-foot opening in the boot upstream of the cooling coils on Containment Cooling Fan Unit D. The licensee has summarized that the resulting bypass flow was the primary reason for the high drywell temperatures observed throughout the past summer. Licensee maintenance technicians also performed modifications to the safety-relief valve tailpipe pressure switches. Engineers had previously identified that these switches were not qualified in accordance with the requirements of 10 CFR 50.49.

Licensee personnel have completed the forced outage physical work, with the exception of troubleshooting and repair of the current transformer, and final paperwork review and verifications for restart are being conducted.

The State of Nebraska has been informed.

Region IV received notification of this occurrence by telephone from the Senior Resident Inspector on October 18. Region IV has informed the OEDO, PAO, and NRR.

The information has been verified by the licensee and is current as of 1:00 p.m. CDT, on October 18, 2000.

PNO-II-00-040a - Tennessee Valley Authority

October 10, 2000

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO-II-00-040a

This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by Region II staff (Atlanta, Georgia) on this date.

Facility		**Licensee Emergency Classification**
Tennessee Valley Authority		Notification of Unusual Event
Sequoyah 1		Alert
Soddy-Daisy, Tennessee		Site Area Emergency
Dockets: 50-327, 50-328		General Emergency
	X	Not Applicable

Subject: Unit 1 Reactor Trip Following Loss of a Main Feedwater Pump - Update

On September 25, 2000, Sequoyah Unit 1 tripped following the loss of a main feedwater pump. Unit 1 was returned to service at 04:54 p.m. EDT on October 5, 2000, following repair efforts for the #4 reactor coolant pump (RCP), which had indication of increasing vibration prior to the unit trip. Repair efforts primarily included an inspection of the lower motor bearing with a subsequent balancing of the RCP motor while it was aligned to the RCP pump assembly. At 07:27 a.m. on October 6, 2000, in accordance with a site abnormal operating procedure (AOP), the operating crew began reducing power from 48 percent rated thermal power due to increasing vibration levels in the #4 RCP. At 07:42 a.m., also in accordance with the AOP, the main turbine and reactor were manually tripped from 20 percent power. The unit was stabilized in Mode 3 (Hot Shutdown) and the #4 RCP was secured.

An NRC resident inspector was in the control room during the manual trip and observed the licensee's response to the transient.

At 05:02 a.m. on October 9, 2000, the unit was placed in Mode 5 (cold shutdown) following successful initiation of shutdown cooling. The unit remains in cold shutdown as of the release of this notification.

The licensee is currently evaluating a repair strategy for the #4 RCP, which may include a motor and/or pump internals replacement. The resident inspectors will continue to monitor Unit 1 activities.

The State of Tennessee has been notified.

The information is current as of 3:00 p.m. EDT, on October 10, 2000.

Contact: Paul
Fredrickson
(404) 562-4530

PNO-III-00-039 - Nuclear Management Company, LLC

October 27, 2000

PRELIMINARY NOTIFICATION PNO-RIII-00039

LICENSEE: Nuclear Management Company, LLC

FACILITY: Point Beach
UNIT: 1
Rx INFO: [1] W-2-LP, [2] W-2-LP
DOCKETS: 50-266, 50-301
CITY: Two Rivers
EMERGENCY CLASS: None

EVENT NUMBER: 37463
EVENT DATE: October 27, 2000
REGION: 3
LOCATION CODE: POW

STATE: Wisconsin

Subject: Point Beach Unit 1 Manually Shut down Due to Underwater Diver Communication Problem

Description:

Unit 1 was manually tripped at 10:28 a.m. CDT on October 27, 2000, after control room operators were informed by security personnel that there was some type of problem involving divers inspecting cooling water discharge pipes in the Unit 2 forebay. Unit 2 is shutdown for a scheduled refueling outage. All control rods fully inserted into the core and all safety systems operated as expected.

While performing normal maintenance activities in the Point Beach Unit 2 forebay, divers lost communication with one another. All divers were accounted for except one whose safety tether appeared to be stuck when pulled on by a fellow diver. Because of the unknown status of the diver, control room operators manually scrammed the Unit 1 reactor.

All divers were accounted for and uninjured. The divers were examined by emergency medical personnel. The licensee is investigating this incident.

After the reactor trip, the steam generator atmospheric valves were used to cool the plant because the circulating water system, which is usually used for a plant cooldown, had been secured to reduce water currents in the forebay .

Reactor restart preparations are in progress.

The Nuclear Management Company is planning to issue a news release on the shutdown.

The State of Wisconsin will be informed. The information in this preliminary notification has been reviewed with licensee management.

NRC resident inspectors reviewed plant equipment's response to the scram.

This information is current as of 3:25 p.m. CDT on October 27, 2000.

PNO-IV-00-028A -Omaha Public Power District

October 18, 2000

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO-IV-00-028A

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV staff on this date.

Facility	**Licensee Emergency Classification**
Omaha Public Power District	__ Notification of Unusual Event
Fort Calhoun Station	__ Alert
Omaha, Nebraska 68023	__ Site Area Emergency
Docket: 50-285 License No.:DRP-40	__ General Emergency
	X Not Applicable

Subject: Update on the Reactor Shutdown to Replace Degraded Reactor Coolant Pump Seal Package

Description:

Operators placed the unit in shutdown cooling at 9:00 p.m. on October 15, 2000. Information gathered during the shutdown and depressurization indicated that approximately 115 fuel pins were leaking as opposed to the 90 pin-hole leaks previously reported. Reactor coolant system dose-equivalent iodine peaked at 4.5 micro curies per gram. The chemical and volume control system has been used to restore the reactor coolant radioactivity to within the steady-state Technical Specification limit of less than or equal to 1 micro curie per gram, dose equivalent iodine. Elevated reactor coolant system radionuclides resulted in increased inventory in the waste gas storage tanks. Therefore, the tanks did not have enough capacity to support reactor drain down. The licensee delayed the drain down for approximately 1 day until the tank with the least radioactive content had decayed enough to meet administrative release limits. The release was completed at 12:35 p.m., and operators drained the reactor vessel to reduced inventory at 3:08 p.m., on October 17, 2000.

Additionally, licensee engineers evaluated the indicated parameters for Reactor Coolant

Pumps C and D seals. As a result, the licensee has decided to replace the Pump C seal package during the forced outage, in addition to the packages for Pumps A and B.

The state of Nebraska has been informed.

Region IV received notification of this occurrence by the resident inspectors on October 18, 2000, Region IV has informed OEDO and NRR.

This information has been discussed with the licensee and is current as of 10:00 a.m. CDT, on

October 18, 2000.

CONTACTS: David P. Loveless
(817) 860-8161

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200
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Web Site: http://www.nrc.gov/OPA

No. 00-170 October 26, 2000

NRC NAMES GRAHAM M. LEITCH TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Nuclear Regulatory Commission has appointed Graham M. Leitch to the Advisory Committee on Reactor Safeguards (ACRS).

Mr. Leitch received a Masters of Science in Mechanical Engineering, with an emphasis on nuclear engineering, from Drexel University in Philadelphia.

His 40-year career includes a wide array of executive management and technical experience in all phases of commercial power plant operations, including 25 years in which he was associated with nuclear power. He was the first site vice president of the Limerick Generating Station and vice president of operations at the Maine Yankee Atomic Power Station. He also has been certified as a senior reactor operator at both the Limerick and Dresden plants. Mr. Leitch was partially responsible for the development of the Limerick probabilistic risk assessment effort and its application to the design and operation of the plant.

In 1991, the American Nuclear Society gave him a Meritorious Performance Award in Reactor Operations based on activities related to the successful start-up of the Limerick plant.

From 1988 to 1991, he was a member of the industry review group that guided the Institute of Nuclear Power Operations (INPO) in policy matters related to power plant evaluation and assistance, serving as chairman in the final year. In addition, Mr. Leitch was a member of the Industry Review Group for Training and Education at INPO during the development of the training accreditation process and served as a mentor for the Senior Nuclear Managers Program on three separate occasions.

The ACRS, established in 1957, advises the Commission on the safety aspects of nuclear facilities and the adequacy of safety standards. Members serve for four-year terms, and may serve no more than three consecutive terms.

Other Members of the ACRS are:

CHAIRMAN: Dr. Dana A. Powers, Manager, Nuclear Facilities Safety Department, Sandia National Laboratories, Albuquerque, NM.

VICE-CHAIRMAN: Dr. George Apostolakis, Professor, Nuclear Engineering Department, Massachusetts of Technology, Cambridge, MA.

MEMBER-AT-LARGE: Dr. Mario V. Bonaca, retired Director, Nuclear Engineering Department, Northeast Utilities, CT.

Dr. Thomas S. Kress, retired Head of Applied Systems Technology Section, Oak Ridge National

Laboratory, Oak Ridge, TN.

Dr. Robert L. Seale, Professor Emeritus of Nuclear and Energy Engineering, Department of Nuclear and Energy Engineering, College of Engineering and Mines, University of Arizona, Tucson, AZ.

Dr. William J. Shack, Associate Director, Energy Technology Division, Argonne National Laboratory, Argonne, IL.

Mr. John D. Sieber, retired Senior Vice President, Nuclear Power Division, Duquesne Light Company, Pittsburgh, PA.

Dr. Robert E. Uhrig, Distinguished Professor, Nuclear Engineering Department, University of Tennessee, Knoxville, Tennessee, and Distinguished Scientist, I&C Division, Oak Ridge National Laboratory, Oak Ridge, TN.

Dr. Graham B. Wallis, Professor, Thayer School of Engineering, Dartmouth College, Hanover, NH.

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September 18, 2000

EA-00-165

Mr. Michael A. Balduzzi
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, Vermont 05302-7002

SUBJECT: NOTICE OF VIOLATION (Office of Investigations Case 1-1999-027)

Dear Mr. Balduzzi:

This letter refers to an investigation conducted at the Vermont Yankee Nuclear Power Plant by the NRC Office of Investigations (OI), to determine whether a manager deliberately failed to comply with Vermont Yankee (VY) procedural requirements concerning the control of contract valve technicians during the 1998 refueling outage. Based on the investigation, OI found that the former Mechanical Maintenance Manager deliberately caused a violation of the VY procedure implementing the requirement to control contracted services during the 1998 refueling outage. In an NRC letter dated August 8, 2000, the NRC provided you a factual summary of the OI investigation, including a basis for the finding, and indicated that an apparent violation of 10 CFR 50, Appendix B, Criterion VII, "Control of Purchased Equipment, Materials and Services" was identified and being considered for enforcement action.

On August 25, 2000, a predecisional enforcement conference was conducted in the Region I Office to discuss the apparent violation, including its apparent deliberate nature, its significance, root causes, and your corrective actions. At the conference, you agreed that a violation occurred, but did not agree that it was deliberate. In support of your contention, you indicated that (1) the manager, who was in attendance at the conference, firmly denied that he had told anyone that the purchase order had been changed to allow the contractors to work unsupervised; (2) the day-shift supervisor, when interviewed by your attorneys, stated that the manager did not tell him that the purchase order had been changed to safety-related, and (3) it was the manager, himself, who had initiated an adverse trend event report on valve work deficiencies that were being identified. You reiterated these points in a subsequent written submittal dated September 11, 2000, wherein you provided affidavits of the manager, day-shift supervisor, and night-shift supervisor.

Notwithstanding your contention, the NRC maintains that the violation was deliberate. In support of this conclusion, the NRC notes that the day-shift supervisor, during his sworn testimony to OI, clearly indicated that the manager had told him that the purchase order had changed. The day-shift supervisor stated, "I asked him [manager] and he said that the purchase order was now safety class." Although the manager, during the conference, denied having made such a statement, the manager was much less definitive, and in fact, inconsistent, when previously interviewed by OI. For example, when the manager was presented, during his OI interview, with the day-shift supervisor's testimony, the manager stated, "..... I don't recall saying that. If I did, I made a mistake and miscommunicated....." When the investigator reminded the manager that he was under oath, the manager stated, "I'll say I don't recall telling him that, because I had no reason to tell him that. That wouldn't have made any sense." When asked if he could out-and-out deny saying that, he replied, "I can't out-and-out say that." Furthermore, other members of your staff, including the night-shift supervisor, believed that the purchase order had been changed. While the NRC acknowledges that the manager ultimately wrote an adverse trend event report describing valve work deficiencies, the initiation of that report by the manager does not refute the fact that his initial actions led staff to believe that the purchase order had changed, when, in fact, he knew that it had not.

As a result of this deliberate violation, contract valve technicians performed unsupervised work on a safety-related valve in the reactor core isolation cooling (RCIC) system, and during that time, these

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contractors failed to properly chamfer the wedge seat and body guides of RCIC motor operated valve (MOV) 13-20. Chamfering MOV 13-20 was necessary as part of an assumption used to determine the minimum thrust required for closing the valve against maximum differential pressure. Since the valve was not chamfered, it was not possible to accurately predict the performance of the valve. A non-cited violation was issued on February 29, 2000, for failing to follow the maintenance procedure.

The NRC recognizes that the lack of chamfering would not have prevented the valve from performing its design function because the valve had a motor operator with considerable thrust margin. The NRC also recognizes that the inadequate chamfer was later identified by your staff and corrected prior to the valve's return to service. Nonetheless, the performance of the unsupervised work based on the deliberate actions of the manager constitutes an additional violation and is described in the enclosed Notice of Violation (Notice). The violation, absent deliberateness, would be considered green if assessed by the Significance Determination Process. However, because it was deliberate, it has been categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, at Severity Level III.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $55,000 is considered for a Severity Level III violation. Because a deliberate Severity Level III violation occurred, the NRC considered whether credit was warranted for *Identification* and *Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. In this case, the inadequate chamfer was identified and corrected by your staff prior to the return to service of the valve. Your staff also initiated an adverse trend event report and performed a root cause analysis, which concluded that supervisory oversight and work control were inadequate for a non-nuclear safety-related contractor performing work on safety-related equipment. Therefore, the NRC has determined that credit is warranted for identification. In addition, prompt and comprehensive corrective actions were taken, including but not limited to: (1) revising the contractor control procedure; 2) providing extensive, improved training and subsequent examination of contractors; 3) revising the maintenance procedure for performing valve work; and 4) improving oversight of this area through self-assessments, supervisory observations and Quality Assurance surveillances. Therefore, the NRC has determined that credit is warranted for your corrective actions.

Therefore, to encourage prompt identification and comprehensive correction of violations, and in recognition of the absence of previous escalated enforcement action, I have been authorized, after consultation with the Director, Office of Enforcement, not to propose a civil penalty in this case. However, significant violations in the future could result in a civil penalty.

The NRC has concluded that the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved were already adequately addressed during the predecisional enforcement conference on August 25, 2000, and in your submittal dated September 11, 2000. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, and your response will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Reading Room). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the Public Document Room without redaction.

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Spent Fuel Pool Accident Risk Study

Timothy E. Collins

Deputy Director, DSSA

Advisory Committee On Reactor Safeguards

October 18, 2000

2

Presentation Outline

- February report findings
- Summary of significant comments
- Approach to comment resolution
- Results of re-analysis
- Conclusions

February Report Conclusions

- Frequency of zirconium fire is low
- Consequences comparable to reactor accident large early release
- Seismic events dominate
- EP relaxation after one year is supportable
- Security needed as long as fuel in pool
- Insurance relaxation is more plant specific

Comments On February Draft

- Source term may be non conservative
- Seismic hazard estimates too conservative
- Zr ignition temperature may be too high
- Partial draindown needs more attention
- Results support EP relaxation at 60 days
- Recommendations not risk-informed

Approach To Comment Resolution

- Consequence analyses expanded:
  - Ruthenium and fuel fines
  - Plume parameters
  - Decay times
- Risks assessed using EPRI and LLNL estimates

Approach To Comment Resolution (con't)

- "Small change" analysis per RG 1.174
- Evaluated sequences for likelihood of flow blockage
- Impact of lower temperature criterion examined

Results

- Consequences with ruthenium and fuel fines still comparable to reactor large early release
- Risk is low but in ball park of operating reactors for first years
- Use of EPRI hazard estimate reduces total risk by about a factor of 4

Results (con't)

- EP relaxation after 60 days is "small change" consistent with guidelines
- Obstructed air flow potential precludes generic decay time when "significant release is no longer possible"
- Temperature criterion effect not important due to already short times in first years

Conclusions

- Risk at decommissioning plants is low even in consideration of ruthenium source term
- Relaxation of EP after 60 days is consistent with "small change" in risk guidelines
- New criterion needed if insurance relaxation is to be considered
- Security required as long as fuel is in pool

3

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: Risk Analysis Results and Conclusions

DATE: November 2, 2000

PRESENTER: Robert L. Palla

TITLE/ORG: Sr. Reactor Engineer
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

TELEPHONE: 415-1095

Risk Characterization

- Risk for each accident estimated based on frequency of fuel uncovery and SFP consequence estimates
- Fuel uncovery assumed to result in SFP fire (large release)
- Consequences assigned based on either early or late evacuation cases, depending on factors affecting EP
  - effectiveness of offsite notification
  - fission product release times relative to evacuation times
- Evacuation modeled as follows:

Event	Full EP	Relaxed EP
Seismic	Late	Late
Cask Drop	Early (for t > 4-5 h)	Early (for t > 10 h)
Boildown	Late	Late

Rationale for Evacuation Modeling

- Seismic
  - for ground motion corresponding to SFP failure, there would be extensive collateral damage within the emergency planning zone (electric power, structures, roads, bridges)
  - radiological pre-planning would have marginal impact because of impairment by offsite damage

- Cask Drop
  - unambiguous indication of event; intact infrastructure for emergency response
  - Full EP: evacuation credited when > 4-5 hours delay time (1 year after shutdown and beyond)
  - Relaxed EP: evacuation credited when > 10 hours delay time (5 years after shutdown and beyond)

- Boildown
  - failure paths involve failure to acquire offsite resources to provide SFP makeup
  - failure to contact offsite authorities or implement effective response also expected for the same reasons

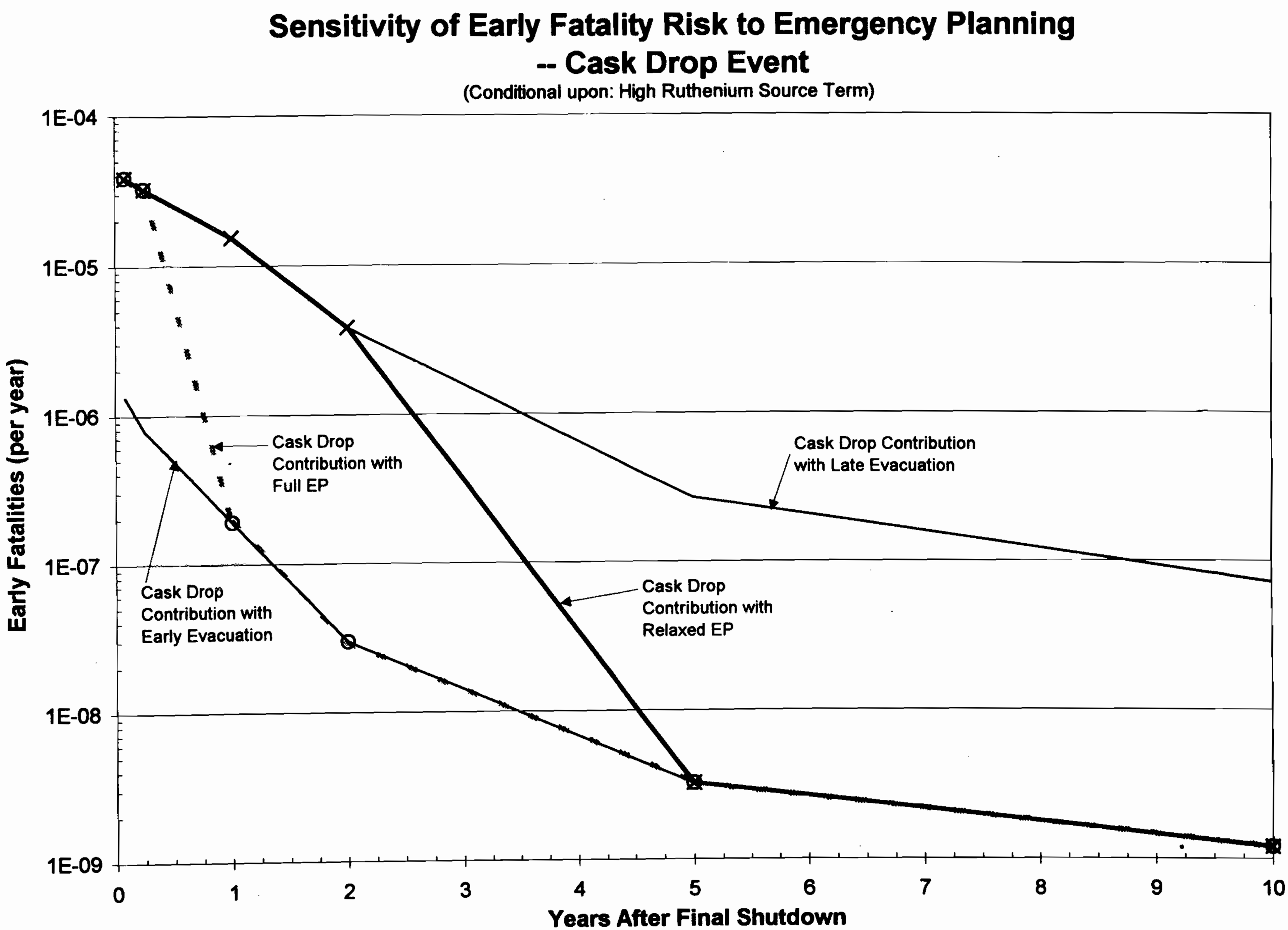

Sensitivity of Early Fatality Risk to Emergency Planning
-- Cask Drop Event
(Conditional upon: High Ruthenium Source Term)
1E-04
1E-05
1E-06
1E-07
1E-08
1E-09
Early Fatalities (per year)
0
1
2
3
4
5
6
7
8
9
10
Years After Final Shutdown
Cask Drop Contribution with Full EP
Cask Drop Contribution with Late Evacuation
Cask Drop Contribution with Early Evacuation
Cask Drop Contribution with Relaxed EP

Figure 3.7-5

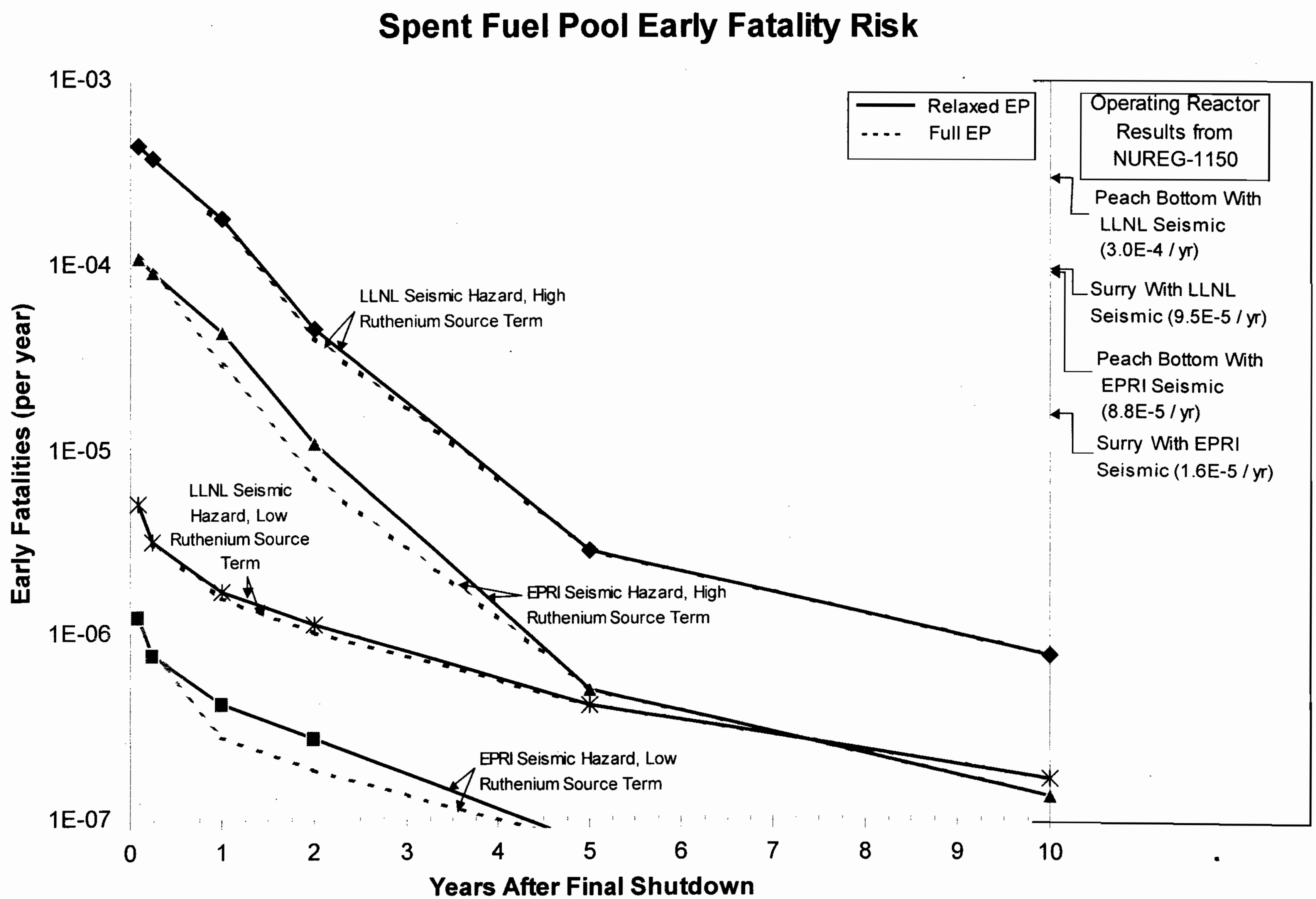
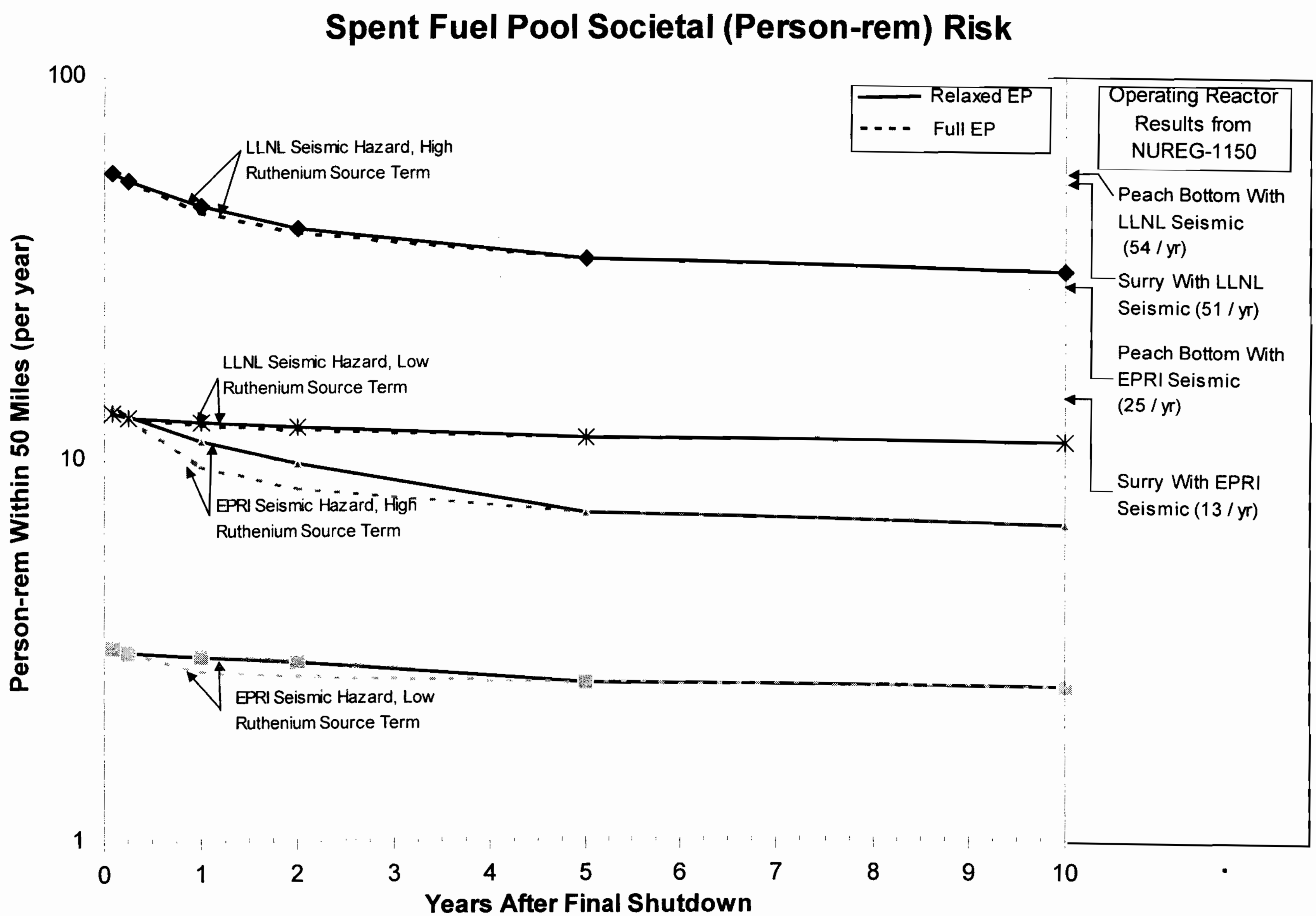

Spent Fuel Pool Early Fatality Risk
Early Fatalities (per year)
1E-03
1E-04
1E-05
1E-06
1E-07
0
1
2
3
4
5
6
7
8
9
10
Years After Final Shutdown
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
Peach Bottom With LLNL Seismic (3.0E-4 / yr)
Surry With LLNL Seismic (9.5E-5 / yr)
Peach Bottom With EPRI Seismic (8.8E-5 / yr)
Surry With EPRI Seismic (1.6E-5 / yr)
LLNL Seismic Hazard, High Ruthenium Source Term
LLNL Seismic Hazard, Low Ruthenium Source Term
EPRI Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term

Figure 3.7-3



Spent Fuel Pool Societal (Person-rem) Risk
Person-rem Within 50 Miles (per year)
100
10
1
0
1
2
3
4
5
6
7
8
9
10
Years After Final Shutdown
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
LLNL Seismic Hazard, High Ruthenium Source Term
LLNL Seismic Hazard, Low Ruthenium Source Term
EPRI Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Peach Bottom With LLNL Seismic (54 / yr)
Surry With LLNL Seismic (51 / yr)
Peach Bottom With EPRI Seismic (25 / yr)
Surry With EPRI Seismic (13 / yr)

Figure 3.7-4

Risk Conclusions

- For the first 1 to 2 years, the early fatality risk for a SFP fire is low, but comparable to that for a severe accident in an operating reactor. At 5 years following shutdown, the early fatality risk for SFP accidents is approximately two orders of magnitude lower than at shutdown

- Societal risk for a SFP fire is also comparable to that for a severe accident in an operating reactor, but does not exhibit a substantial reduction with time due to the slower decay of fission products and the interdiction modeling assumptions that drive long term doses

- Changes to EP requirements affect only the cask drop accident, and do not substantially impact the total risk due to the low frequency of cask drop accidents

Risk Conclusions (continued)

- Use of the low ruthenium source term reduces early fatality risk by about a factor of 100 (relative to the high ruthenium source term) within the first 1 to 2 years, and by about a factor of 10 at 5 years and beyond

- With the low ruthenium source term, the early fatality risk for SFP accidents is about an order of magnitude lower than the corresponding values for a reactor accident shortly following shutdown, and about two orders of magnitude lower at 2 years following shutdown

- With the low ruthenium source term, the societal risk for SFP accidents is also about an order of magnitude lower than the corresponding values for a reactor accident shortly following shutdown, but does not exhibit a substantial reduction with time due to the slower decay of fission products and the interdiction modeling assumptions

- The above observations are valid regardless of whether seismic event frequencies are based on the LLNL or the EPRI seismic hazard study.

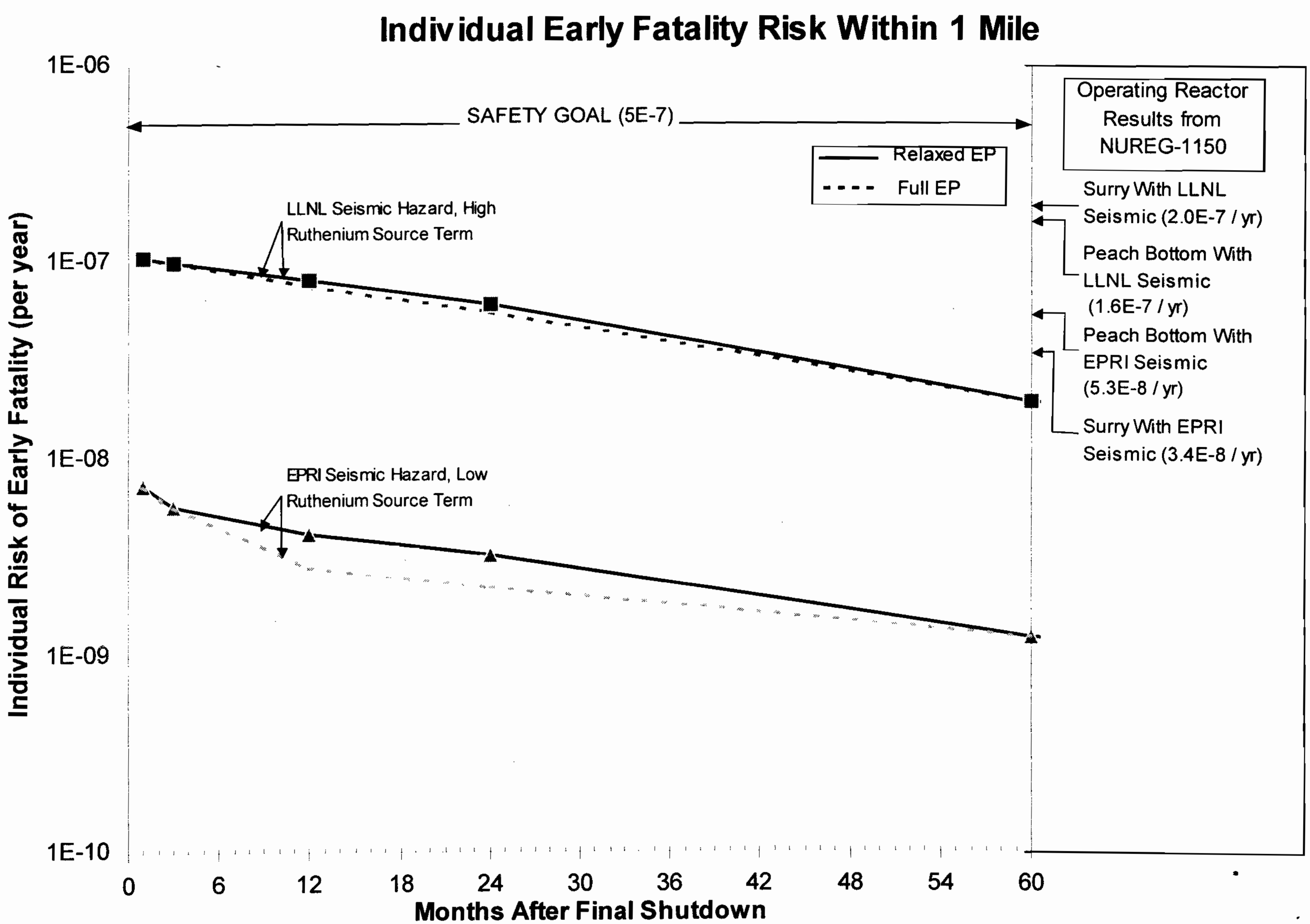
Comparisons to the Safety Goals

- Both the Individual Early Fatality Risk and the Individual Latent Cancer Fatality Risk for a SFP accident are about one to two orders of magnitude lower than the Commission's Safety Goal, depending on assumptions regarding the SFP accident source term and seismic hazard
  - At upper end (LLNL seismic hazard estimates and high ruthenium source term) the risks are somewhat lower than the corresponding risks for reactor accidents, and about a decade lower than the Safety Goal
  - At lower end (EPRI seismic hazard estimates and low ruthenium source term) the risks are lower than those for reactor accidents, and about 2 decades lower than the Safety Goal

- The Individual Early Fatality Risk for a SFP accident decreases with time, and is about a factor of 5 lower at 5 years following shutdown (relative to the value at 30 days)

- The Individual Latent Cancer Fatality Risk is not substantially reduced with time due to the slower decay of fission products and the interdiction modeling assumptions that drive long term doses

- Changes to EP requirements, as modeled, do not substantially impact the margin between SFP risk and the Safety Goals due to the low frequency of events for which EP would be effective



Individual Early Fatality Risk Within 1 Mile
1E-06
1E-07
1E-08
1E-09
1E-10
Individual Risk of Early Fatality (per year)
SAFETY GOAL (5E-7)
Relaxed EP
Full EP
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Operating Reactor Results from NUREG-1150
Surry With LLNL Seismic (2.0E-7 / yr)
Peach Bottom With LLNL Seismic (1.6E-7 / yr)
Peach Bottom With EPRI Seismic (5.3E-8 / yr)
Surry With EPRI Seismic (3.4E-8 / yr)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure 3.7-7

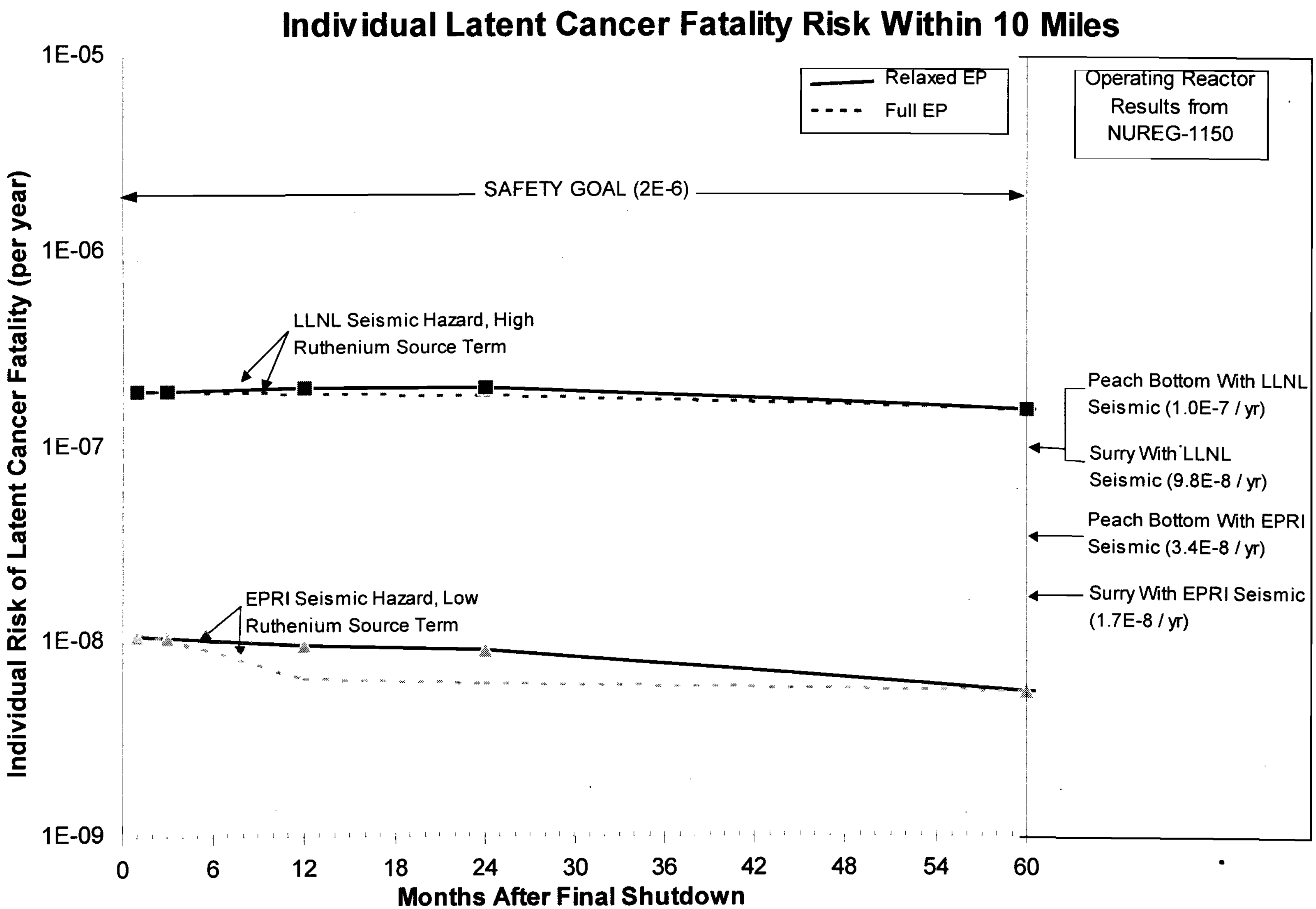

Individual Latent Cancer Fatality Risk Within 10 Miles
Individual Risk of Latent Cancer Fatality (per year)
1E-05
1E-06
1E-07
1E-08
1E-09
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
SAFETY GOAL (2E-6)
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Peach Bottom With LLNL Seismic (1.0E-7 / yr)
Surry With LLNL Seismic (9.8E-8 / yr)
Peach Bottom With EPRI Seismic (3.4E-8 / yr)
Surry With EPRI Seismic (1.7E-8 / yr)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure 3.7-8

Comparison to RG 1.174 Principles
1. Small Increases in Risk

- A SFP facility that conforms with IDCs and SDAs would meet the QHOs by one to two orders of magnitude shortly after shutdown, and with greater margins at later times
- Risk increases associated with EP relaxations are small, even under optimistic assumptions regarding the value of EP in seismic events, and the QHOs continue to be met with margin
- Continued conformance with IDCs and SDAs provides reasonable assurance that the SFP risk and risk increases associated with regulatory changes would remain small

Comparison to RG 1.174 Principles
2. Defense-in-Depth

- Defense-in-depth for accident prevention is provided by robust design of SFP, simple nature of pool support systems, and long times available for corrective actions in response to system failures

- Remaining onsite EP requirements assumed in study, together with the substantial amount of time available for ad hoc offsite emergency response should provide a sufficient level of defense-in-depth for consequence mitigation in SFP accidents

- In the large seismic events that dominate SFP risk, pre-planning for radiological accidents would have marginal benefit due to extensive collateral damage offsite. Accordingly, relaxations in EP requirements are not expected to substantially alter the outcome from such a large seismic event

- In those sequences in which current EP would be effective, such as cask drop accidents, a comparable level of protection should continue to be provided though remaining requirements for on-site EP and the capability to implement offsite protective actions on an ad hoc basis.

Comparison to RG 1.174 Principles
3. Safety Margins

- A SFP facility that conforms with IDCs and SDAs would meet the QHOs by one to two orders of magnitude shortly after shutdown, and with greater margins at later times

- A SFP facility maintained at or below the recommended PPG would continue to meet the QHOs for even the most severe source term.

- The estimated risk increases associated with the EP relaxations are well below the values developed from the RG 1.174 LERF criteria (by about a factor of 10)

- Even under optimistic assumptions regarding the value of EP in seismic events, the change in risk associated with EP relaxations is relatively small
  - increases in early fatalities and individual early fatality risk remain below the maximum allowable for each risk measure
  - population dose and individual latent cancer fatality risk are about a factor of two higher than the allowable value inferred from RG 1.174, however, the increase in individual latent cancer risk represents less than 10 percent of the QHO

Comparison to RG 1.174 Principles
4. Monitoring Program

- The following monitoring should continue following decommissioning in order to assure SFP risk remains low:
  - Performance and reliability monitoring of the SFP systems, heat removal, AC power and inventory should be carried out similar to the provisions of the maintenance rule (10 CFR 50.65)
  - The current monitoring programs identified in licensee's responses to Generic Letter 96-04 with respect to monitoring of the Boraflex absorber material should be maintained by decommissioning plants until all fuel is removed from the SFP (SDA #7)
  - Heavy load activities and load paths should be monitored and controlled by the licensee (IDC # 1)
  - Licensees should continue to provide a level of onsite capabilities to assure prompt notification of offsite authorities, characterization of potential releases, development of protective action recommendations and communication with the public. These capabilities should be monitored by holding periodic onsite exercises and drills
- Continued compliance with the maintenance rule, the IDCs, and the SDAs, together with remaining requirements related to onsite EP provides a reasonable level of monitoring of SFP safety

Consequence Assessment for Spent Fuel Pool Accidents

Presentation to the Advisory Committee on Reactor Safeguards

Jason Schaperow
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

November 2, 2000

F

Overview

Overall risk assessment comprised of three elements:

- **Consideration of initiating event frequencies**
- **Thermal hydraulic analysis to further refine events leading to fuel uncovery and heating**
- ➜ **Consequence assessment for events which led to loss of pool cooling/inventory, fuel heatup and degradation, and significant fission product release**

  Source term issues
  Plume issues
  Evacuation

Overview (cont.)

Issues examined

- **reassessment of source term and release fractions of fission products**
  - **ruthenium**
  - **cesium**
  - **fuel fines**
- **reduced inventory for different decay times**
- **plume spreading**
- **plume heat content**
- **early vs. late evacuation**

Results of large number of MACCS calculations were used to understand decommissioning risk in staff's generic study.

<u>Effect of Ruthenium</u>

Small-scale tests (AECL, ORNL) with an air environment showed significant ruthenium release following cladding oxidation.

MACCS calculations show that release of <u>all</u> ruthenium increases early fatalities by a factor of 20 to 100, because the assumed form (oxide) has a large dose per Ci inhaled due to its long clearance time from the lung.

Mitigating factors for ruthenium releases in spent fuel pool accidents

> **1 year half-life of ruthenium**
>
> **degradation of fuel geometry (e.g., melting, debris bed) may limit air ingress**

PHEBUS test planned to examine effect of air ingress on a larger scale in an integral facility

Effect of Ruthenium (cont.)

Decay Time Prior to Release	Mean Consequences for Surry Population Density (0-100 miles)	
	Early Fatalities	Societal Dose (rem)
1 year	1.01	4.54×10^6
1 year (100% ruthenium release)	95.3	9.53×10^6
1 year (100% ruthenium release)[a]	.13	6.75×10^6

[a]Based on early evacuation.

Effect of Cesium

For cases with small ruthenium release, consequence reduction from decay was modest. As a follow-up, evaluated the effect of cesium.

Cesium release fraction: 1.0
Cesium half-lives: Cs-134, 2 years; Cs-136, 13 days; Cs-137, 30 years

Decay Time Prior to Release	Mean Consequences for Surry Population Density (0-100 miles)	
	Early Fatalities	Societal Dose (rem)
1 year	1.01	4.54×10^6
1 year (without cesium)	0.00	1.46×10^5

Effect of Release Fractions

Case	Release Fraction							Mean Consequences[b] (0-100 miles)	
	I,Cs	Ru	Te	Ba	Sr	Ce	La	Early Fatalities	Societal Dose (rem)
1	1	$2x10^{-5}$	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	1.01	$4.54x10^{6}$
45	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	92.2	$9.50x10^{6}$
45a	1	1	.02	.01	.01	.01	.01	103	$1.33x10^{7}$
45b	.75	.75	.02	.01	.01	.01	.01	54.9	$1.17x10^{7}$
46[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	1.32	$6.84x10^{6}$
46a[a]	1	1	.02	.01	.01	.01	.01	1.54	$8.89x10^{6}$
46b[a]	.75	.75	.02	.01	.01	.01	.01	.543	$7.94x10^{6}$
46c[a]	.75	.75	.75	.01	.01	.01	.01	.544	$7.94x10^{6}$
46d[a]	.75	.75	.75	.75	.01	.01	.01	.544	$7.94x10^{6}$
46e[a]	.75	.75	.75	.75	.75	.01	.01	.644	$1.01x10^{7}$

[a]Based on early evacuation.
[b]Decay time of 1 year.

Effect of Release Fractions (cont.)

Results

- **Increased fuel fines release fraction: increased consequences for cases with early and late evacuation.**

- **Increased tellurium and barium release fractions: no change in consequences due to short half-lives.**

- **Increased strontium release fraction: increased consequences.**

Also evaluated the effect of evacuation percentage (95% vs. 99.5%).

- **Main difference involved early evacuation; factor-of-ten decrease in early fatalities.**

Source Terms

Source Term	Release Fractions								
	noble gases	iodine	cesium	tellurium	strontium	barium	ruthenium	lanthanum	cerium
NUREG/CR-4982	1	1	1	.02	.002	.002	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$
NUREG-1465	1	.75	.75	.31	.12	.12	.005	.0052	.0055
NUREG-1465 (mod)	1	.75	.75	.31	.12	.12	.75[a]	.035[b]	.035[b]

[a]Ruthenium release fraction is that of a volatile fission product.
[b]Fuel fines release fraction is that of the Chernobyl accident (*Chernobyl Ten Years On, Radiological and Health Impact, An Appraisal by the NEA Committee on Radiation Protection and Public Health*, November 1995).

Results using Full NUREG-1465 Source Term (Both In-Vessel and Ex-Vessel)

Case	Decay Time	Mean Consequences (0-100 miles)	
		Early Fatalities	Societal Dose (rem)
77a	30 days	2.21	$7.15x10^6$
77b	90 days	1.37	$6.99x10^6$
77c	1 year	.736	$6.81x10^6$
77d	2 years	.481	$6.65x10^6$
77e	5 years	.192	$6.47x10^6$
77f	10 years	.0778	$6.26x10^6$
78a[a]	30 days	.0720	$5.69x10^6$
78b[a]	90 days	.0461	$5.58x10^6$
78c[a]	1 year	.0301	$5.48x10^6$
78d[a]	2 years	.0208	$5.40x10^6$
78e[a]	5 years	.00882	$5.33x10^6$
78f[a]	10 years	.00400	$5.24x10^6$

[a]Based on early evacuation.

Results using NUREG-1465 (modified) Source Term (Large Ruthenium and Fuel Fines Releases)

Case	Decay Time	Mean Consequences (0-100 miles)	
		Early Fatalities	Societal Dose (rem)
79a	30 days	192	2.62×10^7
79b	90 days	162	2.49×10^7
79c	1 year	76.9	2.15×10^7
79d	2 years	19.2	1.90×10^7
79e	5 years	1.34	1.66×10^7
79f	10 years	.360	1.53×10^7
80a[a]	30 days	6.65	1.60×10^7
80b[a]	90 days	3.95	1.52×10^7
80c[a]	1 year	.951	1.34×10^7
80d[a]	2 years	.149	1.20×10^7
80e[a]	5 years	.0162	1.07×10^7
80f[a]	10 years	.00601	1.00×10^7

[a]Based on early evacuation.

Effect of Number of Fuel Assemblies Releasing Fission Products

- Consequence estimates assumed entire spent fuel pool inventory of Millstone 1 was involved in heatup and release (3.5 cores).

- Depending on reductions in decay heat from radioactive decay, less fuel may be involved in heatup.

- Performed MACCS calculations for two cases: (a) entire spent fuel pool inventory (3.5 cores) and (b) inventory in final core offload.

Effect of Number of Fuel Assemblies Releasing Fission Products (cont.)

Ruthenium Release Fraction	# of cores	Mean Consequences for Surry Population Density[a] (0-100 miles)	
		Early Fatalities	Societal Dose (rem)
$2x10^{-5}$	3.5	1.01	$4.54x10^{6}$
$2x10^{-5}$	1	.014	$3.23x10^{6}$
1	3.5	95.3	$9.53x10^{6}$
1	1	50.5	$7.25x10^{6}$

[a]Decay time of 1 year.

Number of cores reduced for cases with and without large ruthenium release

Smaller consequence reduction for case with large ruthenium release because most ruthenium is in final core offload due to its one year half-life

Effect of Plume Spreading

MACCS uses a Gaussian plume model with the amount of spreading determined by the model parameters σ_y and σ_z.

As part of international cooperative effort on consequence assessment codes, experts provided updated values for σ_y and σ_z.

Experts provided distributions for σ_y and σ_z, instead of point estimates.

SNL performed MACCS calculations based on sampling from the distributions; a total of 300 MACCS calculations were run.

Results: Factor of 1.1 to 15 decrease in early fatalities. Up to 60% increase in cancer fatalities and population dose. (Expect similar effects for reactor accidents.)

Effect of Plume Heat Content

Potential for plume heat content to be higher than that of a reactor accident —> staff performed sensitivity calculations using different plume heat contents

Base Case: plume heat content from NUREG-1150 (3.7 MW)

Bounding estimate of plume heat content of 256 MW based on complete oxidation of one core in 30 minutes

More detailed estimate of plume heat content (about 43 MW)

Effect of Plume Heat Content (cont.)

Case	Release Fraction							Plume Heat Content (MW)	Mean Consequences[b] (within 100 miles)	
	I,Cs	Ru	Te	Ba	Sr	Ce	La		Early Fatalities	Societal Dose (rem)
1	1	$2x10^{-5}$	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	3.7	1.01	$4.54x10^{6}$
45	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	3.7	92.2	$9.50x10^{6}$
47	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	83.0	57.3	$9.24x10^{6}$
49	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	256.0	18.3	$8.24x10^{6}$
46[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	3.7	1.32	$6.84x10^{6}$
48[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	83.0	.00509	$7.28x10^{6}$
50[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	256.0	.00357	$6.96x10^{6}$

[a]Based on early evacuation.
[b]Decay time of 1 year.

Increased plume heat content: main effect is to reduce early fatalities.

Summary

Issues examined

- **reassessment of source term and release fractions of fission products**
  - **ruthenium**
  - **cesium**
  - **fuel fines**
- **reduced inventory for different decay times**
- **plume spreading**
- **plume heat content**
- **early vs. late evacuation**

Results of large number of MACCS calculations were used to understand decommissioning risk in staff's generic study.

(5)

THE RESPONSE OF THE SPENT FUEL POOL TO POSTULATED ACCIDENT CONDITIONS

Presented by:
Robert E. Henry
Fauske & Associates, Inc.

Presented to:
ACRS
Rockville, Maryland

November 2, 2000

MAJOR POINTS

1. Given an accident condition, such as loss of the heat removal function, the response of the pool and the fuel assemblies should be analyzed in a realistic manner.
2. With spent fuel pool water inventory available the pool is adequately cooled for days without the pool cooling function.
3. If the pool water level were to decrease sufficiently to uncover the top of the fuel bundles as a result of the accident condition, the heat removed by boiling and steam flow is important and the power distribution is not important.
4. If the pool is assumed to eventually dry out, the fuel bundle configuration is somewhat influential.
5. If the fuel pins become sufficiently hot that oxidation (chemical energy release) becomes comparable to decay power, the chemical reaction increases the heat generation which in turn increases the reaction rate. With this escalation of the heat generation, the fuel bundle response would be similar to an "at power" case. In this case the zircaloy reaction getters the oxygen resulting in core geometry changes (liquefaction, melting and relocation) comparable to the TMI-2 core response but on a somewhat longer time scale.

APPROACH TO EVALUATIONS

- All evaluations should use a mechanistically identified failure condition.
- Evaluations should assess the results of potential recovery actions consistent with the postulated accident initiator.
- Evaluations should consider all mechanisms for cooling and for energy generation, including the results of vaporization of water in the lower regions of the pool as well as natural circulation of air.

FOCUS FOR ANALYTICAL MODELS

- Spent fuel pool is at atmospheric pressure.
- Flow within the fuel assemblies is laminar, i.e. resistances are well characterized by standard representations.
- Openings in individual fuel assemblies are influential flow paths and should be considered.
- The fuel assembly distribution within the pool does not matter for those accident conditions where the water inventory decreases below the top of fuel until the water is at about 70% of the fuel assembly height. The fuel assembly distribution would matter in the multi-dimensional flow pattern that would develop at lower water levels, i.e. if a thermal plume is developed.

EXAMPLE OF A POSTULATED ACCIDENT CONDITION AND THE RESPONSE BOILDOWN RATE

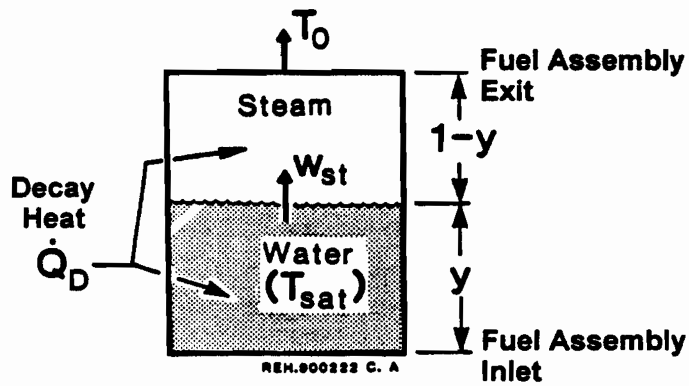
- Assume an average power of 5 kw/assy and 1000 fuel assemblies = 5 MW.
- Assume the pool is 27 ft. (8.2 m) x 23 ft. (7.0 m).
- Boildown rate when the water level is above the fuel is about 5.4 in/hr. (14 cm/hr).
- If the water level progresses into the fuel assembly, this rate is then about 9 in/hr. (23 cm/hr.).
- This boildown can be stopped with a water addition rate of about 35 gpm.

ESTIMATION OF PEAK CLADDING TEMPERATURE FOR ASSUMED ACCIDENT CONDITIONS WHERE THE TOP OF THE FUEL IS UNCOVERED – ASSUMPTIONS –

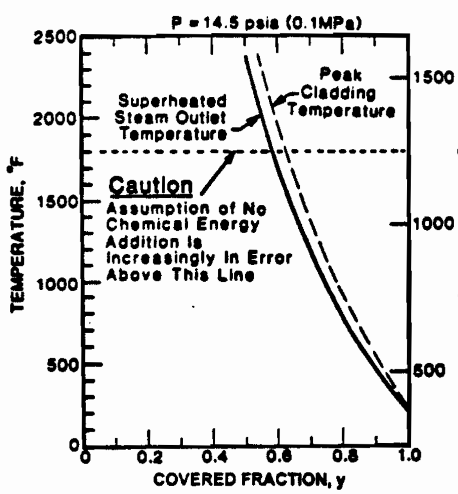
1. The process is quasi-steady.
2. Steam and water are the only fluids in the core.
3. The inlet water is at the saturation temperature T_{sat}.
4. The decay heat (Q_D) is constant along the fuel pin length.
5. The collapsed water level (y) can be used to represent the covered portion of the fuel assemblies.
6. The cladding temperatures remain low enough that the energy released by Zircaloy oxidation is an insignificant fraction of the decay heat.
7. This results in

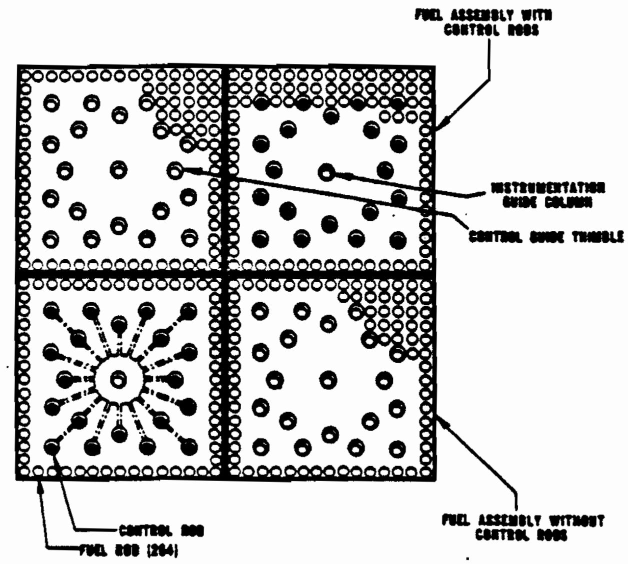
$$T_o - T_{sat} = \left[\frac{1-y}{y}\right]\frac{h_{fg}}{c_{pg}}$$

QUASI-STEADY HEAT REMOVAL



QUASI-STEADY CLADDING TEMPERATURE FOR A PARTIALLY UNCOVERED GROUP OF FUEL ASSEMBLIES





ESTIMATE OF NATURAL CIRCULATION COOLING BY AIR

$$\Delta P = f\left(\frac{L_l}{D}\right)\frac{\bar{\rho}\, U^2}{2} = \Delta\rho\, g\, L_h$$

$$f = \frac{64}{N_{Re}}; \quad \Delta\rho = \frac{\Delta\rho_{max}}{2}; \quad \dot{Q}_D = \bar{\rho}\, A_F\, U\, c_p\, \Delta T_{max}$$

$$\Delta T_{max} = \bar{T}\left[\frac{\dot{Q}_D}{A_F}\left(\frac{L_l}{L_h}\right)\right]^{1/2}\left[\frac{64\,\nu}{g\,D^2}\,\frac{R}{P\,M_w\,c_p}\right]^{1/2}$$

CONCLUSIONS

1. Each evaluation should have a well defined failure condition and recovery actions.
2. For the spent fuel pool there are long intervals available for recovery actions to be implemented.
3. For postulated accident conditions that preclude any recovery actions, the fuel assemblies would eventually increase in temperature sufficient for significant Zircaloy clad reaction. Under these conditions the chemical energy release would dominate the fuel bundle response and this would be similar to those accident conditions considered for "at power" states.

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(6)

ILLUSTRATIONS ACCOMPANYING
A PRESENTATION

BY

GORDON THOMPSON

TO

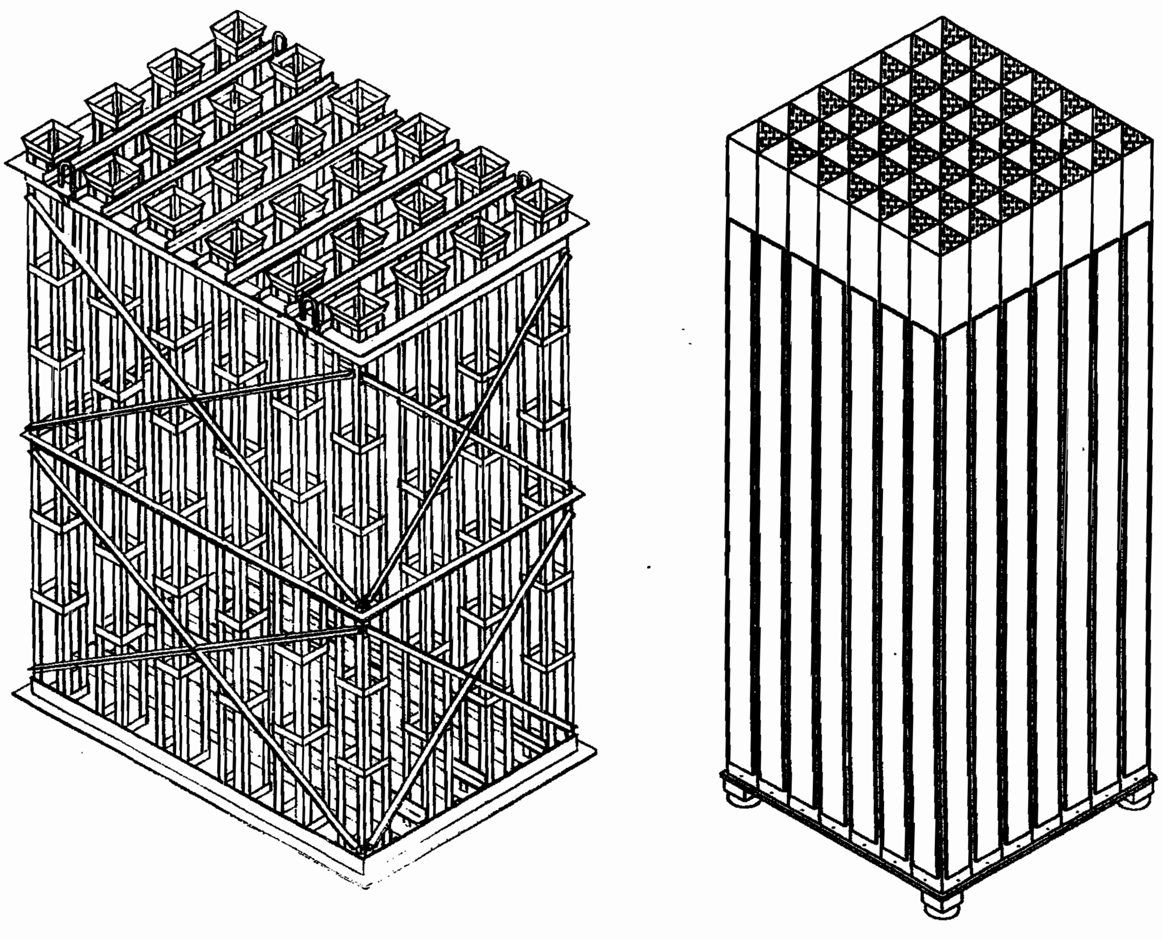
THE ADVISORY COMMITTEE
ON REACTOR SAFEGUARDS

ON

2 NOVEMBER 2000

REGARDING

RISKS ASSOCIATED WITH SPENT FUEL STORAGE
IN HIGH-DENSITY POOLS



(1) LOW AND HIGH-DENSITY RACKS FOR POOL STORAGE OF SPENT FUEL

• The potential for a runaway exothermic reaction of cladding in a high-density spent fuel pool, following water loss, has been known since the late 1970s. (For convenience, this event is described here as a "pool fire".)

• The potential for a pool fire can exist at any high-density pool but may be especially significant for pools at operating nuclear power plants, due to: (a) the presence of recently-discharged fuel with a high decay heat; and (b) the potential for a reactor accident to initiate a pool accident.

• Pool fires have not been studied to the same extent as reactor accidents (e.g., NUREG-1150, IPEs).

• There are major gaps in knowledge about the probability of pool fires, their phenomenology, and their consequences.

• Pool fires deserve attention because they could contaminate large areas of land with comparatively long-lived radioisotopes (e.g., Cesium-137), leading to significant health, economic, social and political impacts.

• Pools generally have a low inventory of short-lived radioisotopes; as a result, pool fires would generally have a comparatively low potential for causing early fatalities.

• The potential for pool fires could be almost completely eliminated by storing spent fuel using a combination of low-density pool storage and dry storage.

(2) SOME OBSERVATIONS ABOUT POOL FIRES

REACTORS

• WASH-1400 core inventory of Cs-137:	4.7 MCi
• WASH-1400 release fraction of Cs: (PWR2 release category)	0.5
• NUREG-1150 release fraction of Cs: (Surry, containment bypass, mean)	0.2
• Chernobyl release of Cs-137: (Livermore estimate)	2.4 MCi

POOLS

• Illustrative pool inventory of Cs-137: (1,000 PWR assemblies @ 0.05 MCi/assy at discharge, av. age 15 yrs)	35 MCi
• Pool fire release fraction of Cs: (NUREG/CR-4982)	1.0

CONCLUSION

• The release of Cesium-137 from a pool fire could exceed the release from a reactor accident, by a factor of 10 or more.

(3) POTENTIAL RELEASES OF CESIUM-137

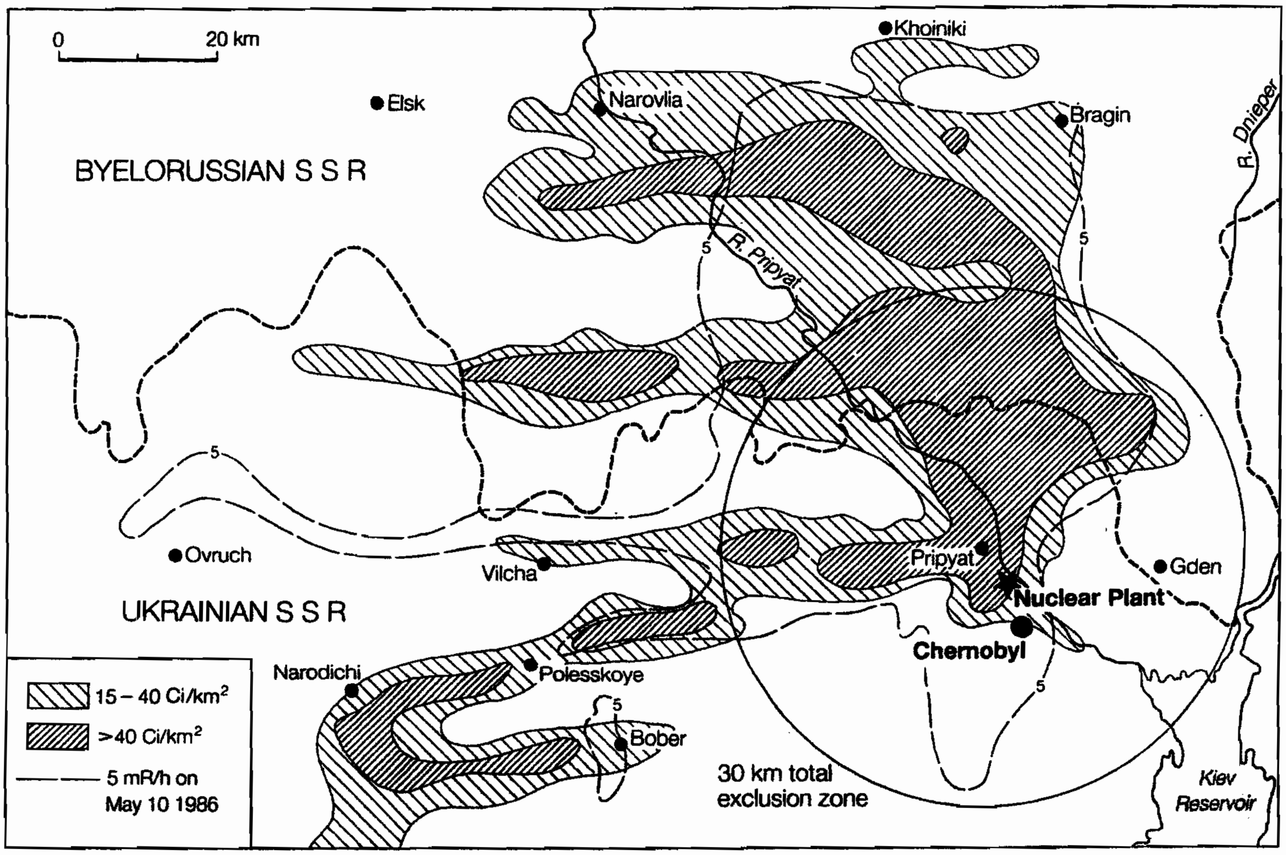
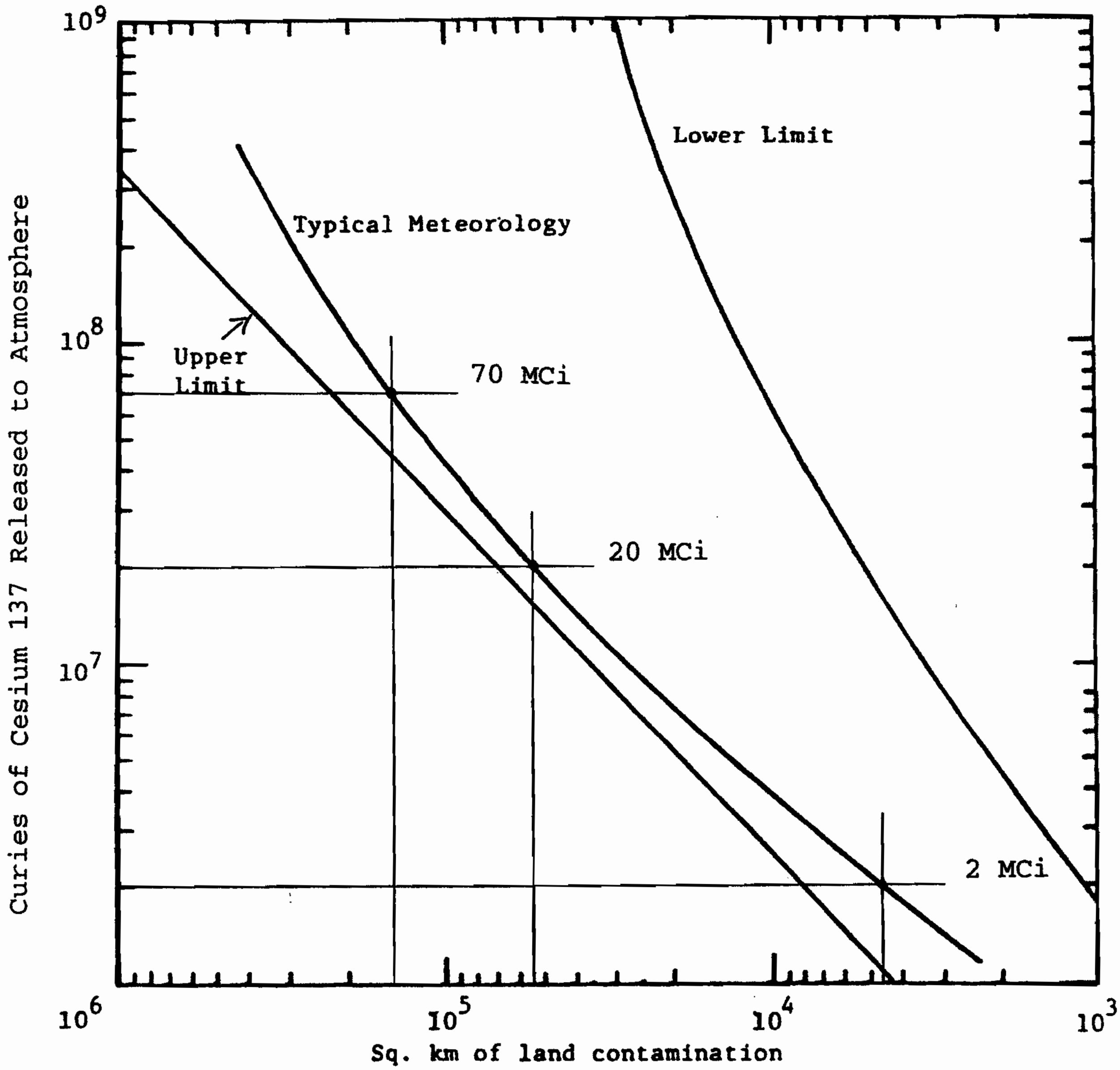


Figure 3.3b Areas of heavy contamination around the exclusion zone (marked by a 30 km radius circle) with the caesium-137 as measured during 1988. Only two levels are indicated. The contour marked by isolines indicates the territory which was contaminated above 5 mR/h of gamma radiation on 10 May, 1986.

(from Medvedev, 1990)

(4) LAND CONTAMINATION BY CESIUM-137 IN THE VICINITY OF CHERNOBYL



(from Beyea, 1979)

(5) ESTIMATED AREA OF CONTAMINATION BY Cs-137
(THRESHOLD OF 10 REM PER 30 YR, SHIELDING FACTOR 0.25)

ESTIMATED LIFETIME RISK PER 100,000 PERSONS EXPOSED TO 1 mSv (0.1 REM) PER YEAR, CONTINUOUSLY THROUGHOUT LIFE

	Males	Females
• Point estimate of excess mortality	520	600
• 90 percent confidence limits	410-980	500-930
• Normal expectation	20,560	17,520
• Excess as percent of normal	2.5	3.4
• Average years of life lost per excess death	16	18

(6) EXCESS CANCER MORTALITY FROM CONTINUOUS EXPOSURE TO RADIATION: BEIR V ESTIMATE

NRC SAFETY GOALS

- "Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health."

- "Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."

NRC STAFF ANALYSIS ON POOL RISK AT DECOMMISSIONING PLANTS

- The NRC Staff's analysis has not addressed land contamination, which is the most important indicator of pool risk; accordingly, the analysis does not provide a credible basis for decision making.

NEXT STEPS

- The NRC should declare a moratorium on any decisions or licensing actions that could increase the risk of a radioactive release from any spent fuel pool, pending the completion of new studies on pool accident risk.

- The NRC should perform studies and supporting experiments, to at least the depth of NUREG-1150, on the probability of pool fires, their phenomenology, and their consequences; for operating plants, this work should address interactions between reactor accidents and pool fires.

- Licensees should be required to extend IPEs and IPEEEs to address pool fires.

7

Framework for Risk-Informed Changes to the Technical Requirments of 10 CFR 50

Presented to
Advisory Committee on Reactor Safeguards

Presented by
Mary Drouin, Alan Kuritzky
U.S. Nuclear Regulatory Commission

Allen Camp
Sandia National Laboratories

Eric Haskin
ERI Consulting

Trevor Pratt
Brookhaven National Laboratory

November 2, 2000

Outline

- ❐ Background
- ❐ Objective
- ❐ Scope and Limitations
- ❐ Framework
- ❐ Future plans

Background

- NRC Policy Statement on PRA use
- NRC's Strategic Plan
- SECY-98-300
- June 8, 1999 SRM
- SECY-99-264
- February 2, 2000 SRM
- SECY-00-0198

Objective

- Mechanism for systematic study of technical requirements of 10 CFR 50, consistent with Commission policy

Scope and Limitations

- Emphasis on regulations impacting existing plants
- Regulations impacting core damage accidents
- Voluntary alternatives to current requirements
- Backfit analysis on safety enhancement options
- Entire set of plants; e.g., industry-wide risk impact
- Staff use, not for licensees

Framework

Approach Builds Upon:

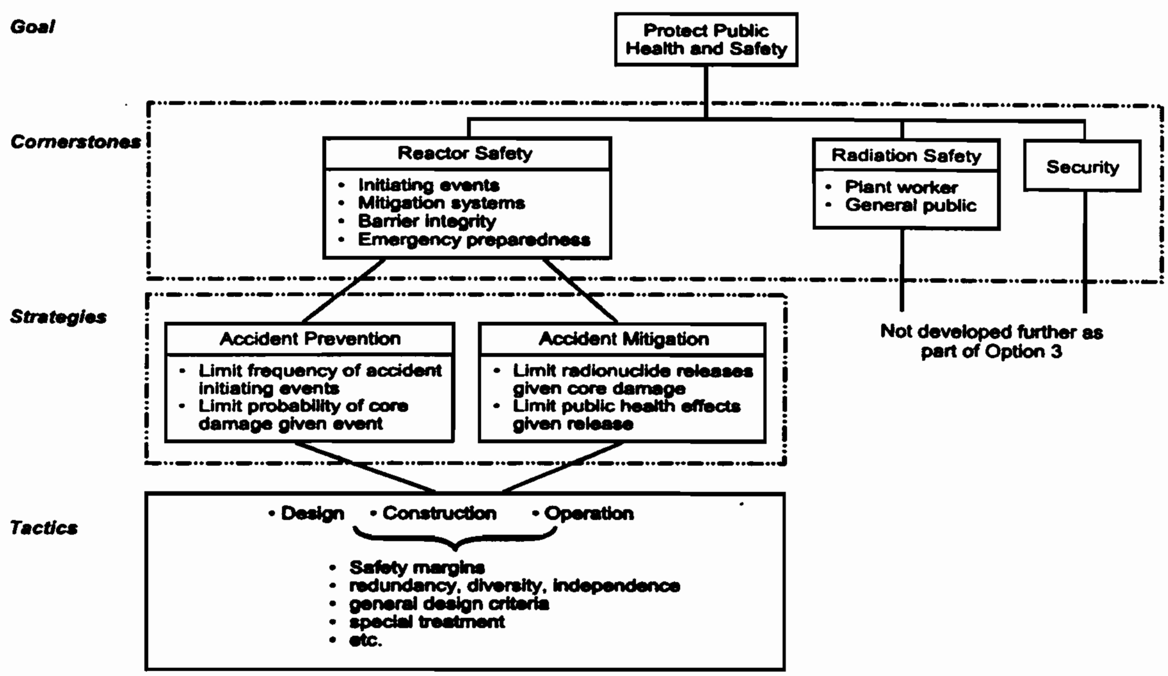
- Commission' White Paper
- Reactor Oversight Cornerstones
- ACRS Recommendations
- Principles of RG 1.174

Approach

Four Elements

- ❐ Goal of protecting public health and safety
- ❐ Cornerstones for safe nuclear power plant operation
- ❐ Strategies of accident prevention and mitigation
- ❐ Tactics to formulate regulations

Framework



Framework

Risk-Informed,Defense-in-Depth

- ❒ Elements dependent on risk insights
  - ★ Balance among the strategies
  - ★ Safety function
- ❒ Elements employed independent of risk insights
  - ★ Prevention and mitigation
  - ★ Reliance on programmatic activities
  - ★ Barriers
  - ★ General Design Criteria in Appendix A to 10 CFR 50

Quantitative Guidelines

- ❒ Need for guidelines
- ❒ Issues needed to be addressed
- ❒ Bases for selecting quantitative values

Quantitative Guidelines

Prevention-Mitigation Method

Accident Prevention	Accident Mitigation
Core Damage Frequency $\leq 10^{-4}$/year	Conditional Large Early Release Probability $\leq 10^{-1}$

Initiator-Defense Method

	Limit frequency of accident initiating events	Limit probability of core damage given event	Limit radionuclide releases given core damage	Limit public health effects given release
	Initiator Frequency	Conditional core damage probability	Conditional large release probability	Conditional individual fatality probability
Frequent initiators	≥ 1/year	$\leq 10^{-4}$	$\leq 10^{-1}$	*
Infrequent initiators	$\leq 10^{-2}$/year	$\leq 10^{-2}$	$\leq 10^{-1}$	*
Rare initiators	$\leq 10^{-5}$/year	*	*	*

Quantitative Guidelines

- Product across each row
- Single type of initiators
- Fourth strategy
- Rare initiators
- Large late release

Uncertainties

- ❐ Sources of
- ❐ Approaches; example:
  - ★ Conservatisms
  - ★ Safety margins

Future Plans

- ❐ Modify framework as needed
- ❐ Continue with 50.46, special treatment requirements, other regulations
- ❐ Public meetings and workshop
- ❐ ~June 2001, paper to Commission

Risk-Informed Regulation Implementation Plan

Presentation to ACRS
November 2, 2000
T.L. King, RES

Background

- PRA Policy Statement
- PRA Implementation Plan
- GAO recommendation

Purpose of RIRIP

- Implement Strategic Plan strategies
- Roadmap to risk-informed regulation
  - Where we are going
  - How to get there
- Communication regarding RIR

Organization of the RIRIP

- Part I – General
- Part II – Arena activities
- Part III – Training and communications
- Scope –primarily activities initiated since 1995 PRA Policy

General Guidance

- **Where do we want to go?**
  - **Vision- 1995 Policy Statement**
  - **Application of Criteria for selection of activities to be risk-informed**

General Guidance

- Important considerations
  - Defense-in-depth
  - Safety margins
  - ALARA
  - Safety goals

General Guidance

- Implementation Issues
  - Performance-based
  - Voluntary versus mandatory
  - Selective implementation
  - Regulatory oversight

Draft Screening Criteria

Would risk-informing:

- **Resolve a safety concern**
- **Make the NRC (or Agreement States) regulatory process more efficient, effective or realistic**
- **Reduce unnecessary regulatory burden on the applicant or licensee**
- **Help to effectively communicate a regulatory decision or situation**
- **Rely on existing risk data and analytical models (or data and models that could be developed)**
- **Have a net benefit**
- **Not encounter factors that would preclude changing the regulatory approach and therefore limit the utility of implementing a risk-informed approach**

Strategic Plan Safety Arenas

- Nuclear Reactor Safety
- Nuclear Materials Safety
- Nuclear Waste Safety

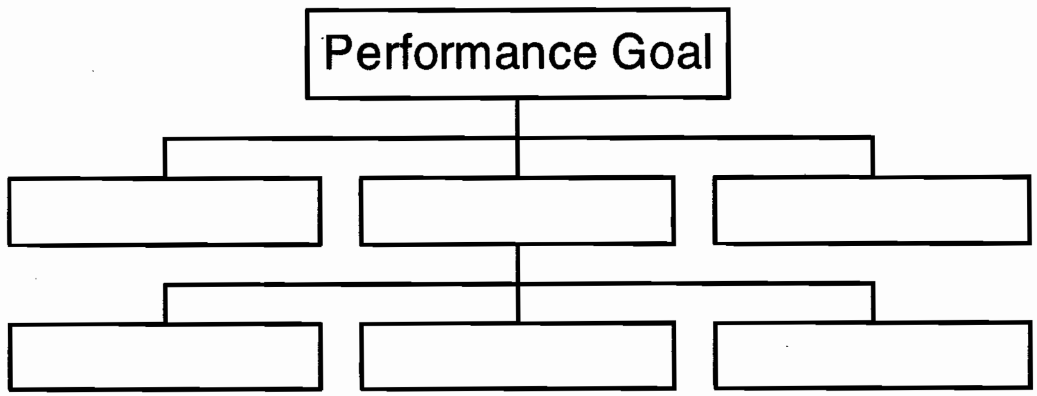
Strategic Plan Performance Goals

- Maintain safety
- Increase public confidence
- More efficient, effective, and realistic
- Reduce unnecessary regulatory burden

General Structure

Strategies

Activities



Infrastructure
Training
Communication

Reactor Arena

- Maintain Safety
  - Strategy 1: We will sharpen our focus on safety to include a transition to a revised NRC reactor oversight program for our inspection, assessment, and enforcement activities.
  - Strategy 3: We will evaluate operating experience and the results of risk assessments or safety implications.
  - Strategy 5: We will ensure that changes to operating licenses and exemptions to regulations maintain safety and meet regulatory requirements.
  - Strategy 8: We will continue to develop and incrementally use risk-informed and, where appropriate, less-prescriptive regulatory approaches to maintain safety.

Reactor Arena

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

Reactor Arena

- **Reduce Unnecessary Regulatory Burden**
  - **Strategy 1: We will utilize risk information and performance-based approaches to reduce unnecessary regulatory burden.**
  - **Strategy 3: We will improve our reactor oversight process by redirecting resources from those areas less important to safety.**

Communication

- Describe RIR and the RIR-IP
- Key Messages:
  - Safety is first priority
  - RIR helps focus on safety
  - Bases for change well grounded
  - Where are we going? (implements Strategic Plan)
- Stakeholder participation:
  - Solicit input and feedback

Communication

- **Describe the planned activities, major milestones, and status of implementation activities**

Communication Milestones

- Issue yellow announcement (December 2000)
- Add RIRIP to NRC website (February 2001)
- Stakeholder meetings (TBD)

Key Challenges

- PRA quality
- Public availability of risk information
- Stakeholder confidence
- Development of materials and waste safety goals

Future Activities

- Solicit internal and external feedback
  - ACRS/ACNW
  - Website
  - Workshops (NRC staff and the public)

Future Activities

- Apply criteria
- Develop integrated schedule
  - Critical path items
- Identify additional needs and activities
  - Infrastructure
  - Training

Summary

- Progress continues to be made within the 3 safety arenas in implementing risk-informed regulation
- RIRIP will be updated every 6 months

9

Industry Perspectives on Risk Informing Decommissioning Regulations

presented by:

Lynnette Hendricks, NEI

November 2, 2000 ACRS Meeting

NEI

Scope (per 12/99 SRM)

- Develop an integrated, risk informed rulemaking addressing EP, FP, Security, Backfit and Operator Training (applicability of maintenance rule, fitness for duty, station blackout, fire protection, etc. to D&D plants will benefit from risk insights)

NEI

Commission principles for risk informing must be adapted to address:

- Different type of consequences
- Risk is dominated by single very low probability event
- Different defense in depth considerations, e.g., passive, robust, slowly evolving sequences
- Very short risk period
- Few plants at risk during a given time period

NEI

What is needed?

- In the end the Commission must make informed judgements on continued applicability of operating plant requirements to decommissioning plants (no magic formula)
- Informed judgement requires best estimates of risk using realistic scenarios

NEI

Defense in Depth Considerations

- Robustness of Pool Structure
- Simplicity of operation
- Slow evolution of all but 2 sequences
- By comparison operating plant address have 100's of sequences for internal events

NEI

Perspective Needed for Risk Driven by Rare Seismic Events

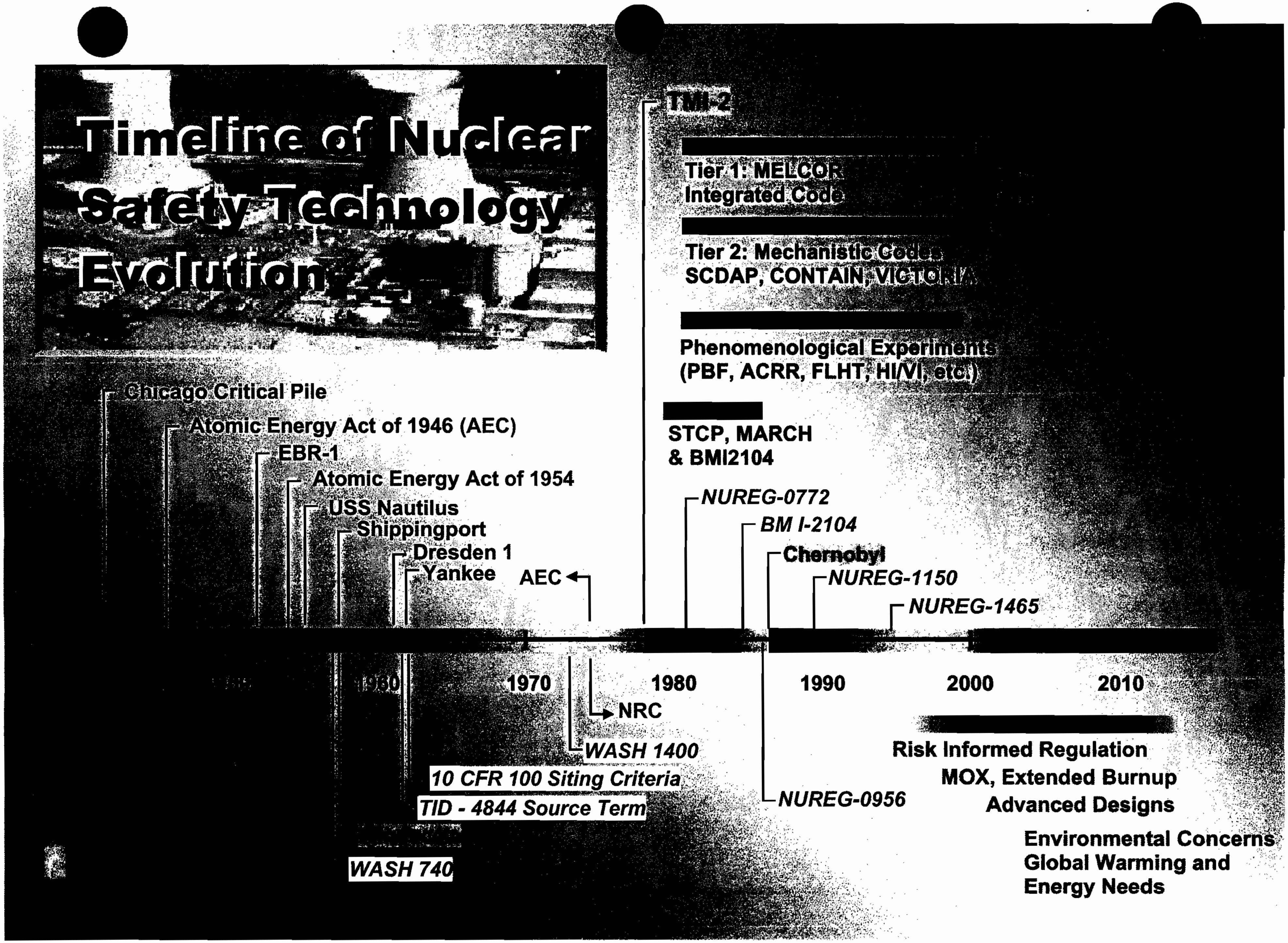
- Extremely large seismic events that are background risk factors for operating plants, dominate risk profile for decommissioning plants
- NUREG 1150:
  - avoided including offsite dose consequences from seismic events
  - recommended placing reactor induced accident losses in context of overall losses from the seismic event (report observes nuclear losses likely to be very small)
- Seismic risk can be addressed deterministically:
  - Checklist demonstrates seismic capacity of pool at 2-3 SSE

NEI

Conclusions

- Use of bounding estimates will not allow proper risk informed decisions
- Opportunities to apply practical risk insights to spent fuel pool risk for decommissioning plants may be lost if operating plant requirements are retained

NEI


Timeline of Nuclear Safety Technology Evolution
TMI-2
Tier 1: MELCOR
Integrated Code
Tier 2: Mechanistic Codes
SCDAP, CONTAIN, VICTORIA
Phenomenological Experiments
(PBF, ACRR, FLHT, HI/VI, etc.)
STCP, MARCH
& BMI2104
Chicago Critical Pile
Atomic Energy Act of 1946 (AEC)
EBR-1
Atomic Energy Act of 1954
USS Nautilus
Shippingport
Dresden 1
Yankee
AEC
NUREG-0772
BM I-2104
Chernobyl
NUREG-1150
NUREG-1465
1960
1970
1980
1990
2000
2010
NRC
WASH 1400
10 CFR 100 Siting Criteria
TID - 4844 Source Term
NUREG-0956
WASH 740
Risk Informed Regulation
MOX, Extended Burnup
Advanced Designs
Environmental Concerns
Global Warming and
Energy Needs

BACKGROUND

EDO REQUEST TO ACRS, JULY 20, 2000

- ACRS ASSISTANCE IN THE TECHNICAL RESOLUTION OF A DPO ASSOCIATED WITH STEAM GENERATOR TUBE INTEGRITY

- ACRS FUNCTION AS THE EQUIVALENT OF AN AD HOC PANEL UNDER MANAGEMENT DIRECTIVE (MD) 10.159, TO REVIEW THE DPO ISSUES AND PROVIDE A REPORT TO THE EDO

ACRS RESPONSE TO EDO REQUEST

- DURING THE SEPTEMBER 2000 MEETING, THE ACRS AGREED TO THE EDO'S REQUEST AND SENT A MEMORANDUM TO THE EDO DATED SEPT. 11, 2000

- ESTABLISHED AN AD HOC SUBCOMMITTEE ...

  - FUNCTION UNDER THE PROVISIONS OF THE FEDERAL ADVISORY COMMITTEE ACT

10

AD HOC SUBCOMMITTEE MEMBERS

D. POWERS, CHAIRMAN
M. BONACA
T. KRESS
J. SIEBER
R. BALLINGER (MIT)

CONSULTANTS PROVIDED BY EDO TO THE SUBCOMMITTEE

I. CATTON (UCLA)
J. HIGGINS (BNL)

OBJECTIVES OF THE AD HOC SUBCOMMITTEE

- GATHER AND SYSTEMATIZE INFORMATION

  - areas of contention
  - data and analyses to support positions
  - applicability of data
  - validation of models & applicability
  - risk significance of issues

- DEVELOP DRAFT POSITIONS FOR CONSIDERATION BY THE ACRS

  - Full Report

    - NUREG format
    - internal peer review

  - Draft Letter

ASSIGNMENTS

SUBCOMMITTEE	TOPIC	REVIEWER
D. A. POWERS	CHAIRMAN Iodine Spiking	R. Seale
M. Bonaca	Risk Human Factors	G. Apostolakis
T. S. Kress	Severe Accidents Thermalhydraulics	G. Wallis
R. Ballinger (MIT)	Metallurgy	Richard Ricker (NIST)
J. Sieber	NDE	R. Uhrig

Consultants: Prof. I. Catton (UCLA) and James Higgins (BNL)

Support Staff: Undine Shoop and Sam Duraiswamy

CONTENTION CATEGORIES

- ACCIDENT ANALYSIS
  - design basis accidents
  - severe accidents
- LIMITATIONS OF **NDE** METHODS
- CORROSION AND CRACKING PHENOMENA
- CORRELATION OF **NDE** RESULTS AND PHENOMENA
- LEAKAGE AND BURST PHENOMENA
- DAMAGE PROGRESSION
  - crack opening
  - crack unplugging
  - jet cutting of tubes
- SOURCE TERM
  - iodine spiking
  - aerosol behavior

5

DATA GATHERING

SUBCOMMITTEE MEETING, OCTOBER 10-14, 2000

- REVIEWED A LARGE VOLUME OF DOCUMENTS
- DOCUMENTS REVIEWED BY THE AD HOC SUBCOMMITTEE HAVE BEEN SENT TO ALL ACRS MEMBERS
- MET WITH PRINCIPALS
  - Dr. J. Hopenfeld, author of DPO
  - Robert Spence, PE, RES
  - NRC Staff

ISSUE SUMMARY

- STAFF HAS ISSUED **GL 95-05**, "VOLTAGE-BASED REPAIR CRITERIA FOR WESTINGHOUSE STEAM GENERATOR TUBES," ALLOWING LICENSEES TO USE VOLTAGE INDICATIONS RATHER THAN CRACK DEPTH AS A CRITERION FOR REPAIRING CRACKS IN STEAM GENERATORS CONFINED TO THE REGIONS OF THE STEAM GENERATOR TUBE SUPPORT PLATES.

- **DPO** CONTENDS THAT THERE IS INSUFFICIENT TECHNICAL SUPPORT FOR THIS ALLOWED DEGRADATION OF THE PROTECTION PROVIDED FOR THE PUBLIC HEALTH AND SAFETY.

  - contentions raised concerning the specific issues of the generic letter
  - contentions raised concerning the evaluation of the risk status of the plant

- SUBCOMMITTEE GATHERED DATA ON ALL THE CONTENTIONS

- DPO AUTHOR RECOMMENDS:

  - GL 95-05 should be rescinded
  - All plants that do not meet the 40% plugging criteria should be shut down and all tubes plugged accordingly

STAFF EFFORTS TO RESOLVE THE DPO/STEAM GENERATOR ISSUES

DURING THE AD HOC SUBCOMMITTEE MEETING, THE STAFF STATED THE FOLLOWING:

- DPO/STEAM GENERATOR ISSUES HAVE BEEN GIVEN SERIOUS ATTENTION.

- ASSESSMENT OF THE DPO AND STEAM GENERATOR ISSUES HAVE BEEN DOCUMENTED (E.G., DPO CONSIDERATION DOCUMENT AND NUREG REPORTS).

- REGULATORY FRAMEWORK FOR STEAM GENERATOR TUBE INTEGRITY IS BEING DEVELOPED IN COORDINATION WITH THE NUCLEAR ENERGY INSTITUTE.

- NRR AND RES ARE WORKING CLOSELY TO RESOLVE SEVERAL ISSUES (E.G., TUBE CUTTING/EROSION, VIBRATION) ASSOCIATED WITH STEAM GENERATOR TUBE INTEGRITY.

- A NEW GENERIC ISSUE IS BEING CONSIDERED TO ADDRESS THE VIBRATION ISSUE.

- LESSONS LEARNED FROM THE INDIAN POINT 2 STEAM GENERATOR TUBE RUPTURE EVENT WILL BE EVALUATED AND A DECISION WILL BE MADE WITH REGARD TO IMPROVEMENTS THAT NEED TO BE DONE TO ENSURE STEAM GENERATOR TUBE INTEGRITY.

SCHEDULE

• INFORMATION GATHERING	OCTOBER 10-14
• DRAFT REPORT TO AD HOC SUBCOMMITTEE MEMBERS	NOVEMBER 6
• COMMENTS FROM AD HOC SUBCOMMITTEE MEMBERS	NOVEMBER 10
• DRAFT REPORT TO PEER REVIEWERS	NOVEMBER 13
• COMMENTS FROM PEER REVIEWERS	NOVEMBER 21
• FINAL REPORT TO ACRS	DECEMBER 1
• DRAFT LETTER TO ACRS	DECEMBER 1

DIFFERING PROFESSIONAL OPINION ON STEAM TUBE INTEGRITY ISSUES

477TH ACRS MEETING

November 2, 2000

Presented by Dr. Joram Hopenfeld

Differing Professional Opinion (DPO) on Steam Generator Tube (SGT) Integrity

Conclusions From the DPO Ad-Hoc Subcommittee Meeting (October 10-14, 2000)

In my introductory remarks to the ACRS Ad Hoc Subcommittee on DPO issues, I pointed out that the risk to the public from not removing degraded steam generator tubes from service is at least a hundred times larger than has been reported to the public. This is the crux of the DPO. Briefly I would like to discuss the risk to the public in terms of four key factors: (1) Instrument Capabilities, (2) Primary to Secondary Leakage Predictions, (3) Operator's Response to Main Steam Line Break (MSLB) Accidents, and (4) The NRC Process.

<u>(1) Instrumentation Capabilities.</u> The ability to detect the threshold of defects which could lead to catastrophic SGT failure is based only on laboratory tests. After more than ten years of research, large cracks with small voltage readings are missed even in the laboratory environment. Actual plant experience such as the recent Indian Point 2 event demonstrates that in the field, significant defects will not be detected. The Probability of Detection (POD) of 0.6 allowed by the NRC is arbitrary, totally unfounded, and non conservative.

<u>(2) Primary to Secondary Leakage Predictions.</u> The NRC voltage methodology for predicting leakage is grossly non conservative. The correlations between voltage and leakage are inconsistent with basic physical laws governing the flow of fluids through cracks. After more than ten years of research, large flaws with small voltage readings are missed in both the laboratory environment and in the field. Cracks which exhibit a low signal to noise ratio may result in SGT failure with catastrophic consequences. The tube leakage and burst database which is used to correlate the voltage with leakage was not obtained under realistic and valid conditions. The NRC predictions of the leakage are several orders of magnitude lower than those that can be expected during MSLB accidents.

<u>(3) Operator Response.</u> The NRC assigns 99.9% probability of success to an operator's ability to depressurize and cool down the primary coolant system before the reactor core is uncovered. Operating experience of 10 steam generator tube ruptures, which were relatively mild in comparison to MSLB events, do not authenticate such optimistic predictions.

<u>(4) The NRC Process</u>. The NRC regulatory process primarily protects the financial interests of the nuclear industry. Public safety takes a backseat to necessary corrective actions which would be costly to the industry. The Generic Safety Issue Program and other research activities create the appearance that the NRC is concerned with plant safety and is effectively resolving safety issues. In fact these programs delay the implementation of necessary and urgently needed corrections. The attached Table shows that it takes up to 17 years to resolve high priority safety issues. Existing design safety margins are being drastically and dangerously reduced under the masquerade of "Risk Informed Regulations".

At the October 13 ACRS Sub-committee meeting, Mr. Joe Donoghue described how the technical specifications for Braidwood-1 and Byron 1 were relaxed using a modified RELAP5 code even though inappropriate data was used to modify the code. Mr. Steve Long described how the Farley plant was allowed to skip a scheduled inspection ignoring the fact that there was no data considered to show how small cracks could rapidly and catastrophically propagate SGT damage by jet erosion. The NRC OIG recently documented that inexperienced NRC engineers poorly supervised and constrained from conducting free discussions with licensees are responsible for major safety determinations and decisions on steam generator tube integrity. The NRC has also been successful in preventing the staff from identifying safety issues through the DPO process and in delaying the DPO process.

In summary, uncertainties in instrumentation, leakage predictions, operator response, and inadequate NRC oversight of licensee submittals, substantiate the conclusion that leaving degraded SG tubes in service can easily lead to catastrophic consequences. Even with the unrealistically optimistic assumption that an operator will be 90% successful in controlling the accident, the actual risk to the public is 100 times larger than predicted by the NRC.

At the subject ACRS Sub-committee meeting, the NRC for the first time admitted that GL-95-05 is not valid for SGT leakage of more than 30gpm. The NRC can not support the use of GL-95-05 where existing predictions indicate that the leakage could exceed 1000gpm. Jet erosion, SGT vibrations, bending and buckling during MSLB events can lead to leakage of thousands of gallons per minute. At the ACRS meeting, the NRC also failed to address the compatibility of present ATWS rules with allowing degraded steam generator tubes to remain in service. During unmitigated ATWS events, the design differential pressure of the tube sheet is exceeded by at least two times. Also the database for leakage was obtained for pressures up to 2600psi, which are lower by at least 600 psi from the present anticipated ATWS pressure (>3200psi).

The draft report on "Regulatory Effectiveness of the Anticipated Transient Without Scram Rule" made public on October 18, 2000, addresses higher peak pressures and up to 37% unfavorable exposure times that lessen mitigative functions required by the ATWS rule, especially if there is no diverse scram system in Westinghouse reactors. This is scheduled to be addressed in a February 2001 ACRS meeting. The percentage of the fuel cycle during which a Westinghouse ATWS is unmitigated proportionally increases the risk. High peak pressures within less than 10 seconds from an unmitigated ATWS will also have an affect on tube sheet cladding separation and tube weld cracks, similar to the Robinson 2 cold hydro. This high pressure rise is an additional mechanism for steam generator leakage with containment bypass during a severe accident.

I was pleased to learn yesterday that NRR has recently questioned licensee requests under GL95-05 for increasing the repair voltage to 3V. These questions were related to the issues that I raised at the subject meeting.

At the October meeting, the NRC Division of Research strongly advocated the need for additional research. If research is still needed after 10 years in order to prove the validity of GL-95-05 and to provide Tech Spec relief for the 40% through wall tube plugging, it shows that this document has no valid technical basis. The alternate repair criteria as specified in GL-95-05 should be rescinded immediately without further delay. Failure to do so will continue to mislead the public concerning the safety of allowing plants to operate with severely and unacceptably defective steam generator tubes.

REACTOR GSIs RESOLVED BY FISCAL YEAR: FY-1983 TO FY-2000

ISSUE NUMBER	TITLE	PRIORITY	RESOLUTION PRODUCT	DATE APPROVED FOR RESOLUTION	DATE RESOLVED	TIME TO RESOLVE AFTER PRIORITIZATION (YEARS)
171	ESF Failure from LOOP Subsequent to a LOCA	HIGH	No Req.	06/16/95	12/98	3.58
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	No Req.	NRR OP FY-83	03/99	16.50
158	Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions	MEDIUM	Staff Report (No Req.)	01/26/1994	08/1999	5.58
165	Spring-Actuated Safety and Relief Valve Reliability	HIGH	Staff Report (No Req.)	11/26/1993	06/1999	5.58
FY-2000						0
23	Reactor Coolant Pump Seal Failures	HIGH*	Staff Report (No Req.)	NRR OP FY-83	11/1999	17.17
145	Actions to Reduce Common Cause Failures	HIGH*	Regulatory Issue Summary 99-03 (No Req.)	02/11/1992	10/1999	7.75
190	Fatigue Evaluation of Metal Components for 60-Year Plant Life	HIGH*	Staff Report (No Req.)	08/26/1996	12/1999	3.33
B-17	Criteria for Safety-Related Operator Actions	MEDIUM	Staff Report (No Req.)	03/22/82	03/2000	18.00
B-55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	Staff Report (No Req.)	NRR OP FY-83	12/1999	17.25
TOTAL TIME TO RESOLVE ALL 145 GSIs:						667.89

* Previously listed as Nearly-Resolved but changed to HIGH in SECY-98-166

NOTES:
1. ***The average time to resolve a GSI was 4.61 years***
2. ***The computation for HIGH-priority GSIs is skewed by past management decisions to change the priority of Nearly-Resolved GSIs to HIGH***

RECOMMENDATIONS

- RESCIND GL 95-05

- All plants that do not meet the 40% plugging criteria should be shut down and all tubes plugged accordingly.

NRR REDUNDANT SRV/LPS SHUTDOWN ACTIVITIES

ERIC WEISS
OFFICE OF NUCLEAR REACTOR REGULATION
301-415-3264

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NOVEMBER 2000

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REDUNDANT USE OF SRV/LPS

- IN SEPTEMBER, 1999, BWROG SUBMITTED A DOCUMENT ON THE USE OF SAFETY RELIEF VALVES AND LOW PRESSURE SYSTEMS (SRV/LPS) AS A MEANS OF REDUNDANT POST-FIRE SAFE SHUTDOWN

- REGULATORY BACKGROUND

  - SRV/LPS IN BWR DESIGN BASIS ACCIDENT LICENSING BASIS

- SRV/LPS IN BWR NORMAL S/D GDC 34 SINGLE FAILURE LIC. BASIS SINCE 1975

- SRV/LPS WIDELY APPROVED BY THE STAFF AS A MEANS OF ALTERNATIVE SHUTDOWN (WITH DETECTION AND SUPPRESSION IN THE FIRE AFFECTED AREA IAW APPENDIX R SECTION III.G.3)

- THE STAFF'S REGULATORY AND CORE THERMO-HYDRAULIC ANALYSES SUPPORT USE OF SRV/LPS AS A REDUNDANT SAFE SHUTDOWN METHOD

- ON 4/25/00 THE STAFF MET WITH THE BWROG TO PROVIDE TECHNICAL, REGULATORY, RISK AND LEGAL FEEDBACK

- SEVEN MAJOR SUB-ISSUES WERE ADDRESED BY THE STAFF DURING AND SUBSEQUENT TO THAT MEETING:

  - THE EXISTENCE OF PLANT SPECIFIC LIC. BASES IN WHICH THE STAFF HAS APPROVED SRV/LPS AS A <u>REDUNDANT</u> MEANS OF POST-FIRE SAFE SHUTDOWN (5 EXAMPLES IDENTIFIED TO DATE)

- WHETHER AN SRV/LPS "HOT SHUTDOWN" PROCEDURE EXISTED. THE BWROG PROVIDED A HOT SHUTDOWN PROCEDURE NARRATIVE BASED ON EPG 4
    - INCLUDED LIKELY DEPRESSURIZATION AT TOP OF ACTIVE FUEL
    - HOT S/D MAINTENANCE CAPABILITY FROM 200 TO 212 DEGREES F
- NON-APPLICABILITY OF APPENDIX R SECTION III.L PERFORMANCE CRITERIA

- NON-APPLICABILITY OF SINGLE FAILURE CRITERIA
- POTENTIAL RISK INCREASE FROM REMOVAL OR ABANDONMENT OF DETECTION AND SUPPRESSION GENERALLY SMALL AS DEFINED IN RG 1.174 (FIRE AREA OUTLIERS MAY EXIST AT AT SOME PLANTS)

- NUMBER OF PROTECTED SRVS FOR CORE THERMO-HYDRAULIC SAFETY DURING DE-PRESSURIZATION (BASED ON PLANT SPECIFIC ANALYSES), AND

- VESSEL MATERIAL CONCERNS RELATED TO COOLDOWN RATE >100 DEGREES F/HR (DEPRESSURIZATION COUNTERACTS THERMAL STRESSES, AND VESSEL FATIGUE ADDRESSED BY LIMITING THE NUMBER OF STRESS CYCLES)

- UPON FINAL CONSIDERATION OF THE ISSUES, THE STAFF EXPECTS TO ISSUE AN SER ON THE BWROG SRV/LPS TOPICAL DURING NOVEMBER, 2000.

NRR FIRE PROTECTION INSPECTION ACTIVITIES

LEON WHITNEY
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NOVEMBER 2000

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BASELINE FIRE PROTECTION INSPECTION PROGRAM

- COMMENCED APRIL 2000 IAW SECY 99-140 (FPFI FINAL REPORT) AFTER 3 PILOTS
- FIRE RISK COMPARABLE TO TOTAL RISK FROM INTERNAL EVENTS
- BASELINE INSPECTION TECHNIQUES DERIVED FROM FPFI PROGRAM

FIRE PROTECTION
SIGNFICANCE DETERMINATION PROCESS

- BASED ON FP DEFENSE-IN-DEPTH
- FIRE SCENARIO MUST BE DEVELOPED (NO MORE "WALL TO WALL" FIRE ASSUMPTIONS)
- RISK SIGNIFICANCE OF FP FEATURE DEGRADATIONS ASSESSED
- DELTA CDF COMPUTED

BASELINE PROCEDURE CONTENT

- MONTHLY/ANNUAL RESIDENT INSPECTION
  - COMBUSTIBLES AND IGNITION SOURCES
  - DETECTION AND SUPPRESSION
  - MANUAL FIRE FIGHTING
  - PASSIVE FP FEATURES/FIRE BARRIERS
  - FIRE BRIGADE CAPABILITY AND PERFORMANCE
  - COMPENSATORY MEASURE ADEQUACY
  - RCP OIL COLLECTION

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM INSPECTION OF POST-FIRE SAFE SHUTDOWN CAPABILITY
  - ELECTRICAL, RX/MECHANICAL SYSTEMS, AND FP INSPECTORS
  - 2-3 DAY INFORMATION GATHERING VISIT
  - 1-2 WEEKS OF ONSITE INSPECTION WITHIN DESIGN AND LICENSING BASES

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY:
  - FIRE AREA BOUNDARY DESIGN
  - SS/D SYSTEMS SELECTION ADEQUACY
  - HOT S/D SYSTEMS SEPARATION
  - SS/D CIRCUIT PROTECTION ANALYSIS
  - ALTERNATIVE SHUTDOWN

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):
  - COMMUNICATIONS
  - EMERGENCY LIGHTING
  - FIRE PROTECTION SYSTEMS, EQUIPMENT AND FEATURES
  - FIRE SUPPRESSION DAMAGE ASSESSMENT

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):

- OPERATOR RECOVERY ACTIONS
  - SMOKE REMOVAL
  - DEWATERING
  - CONTROLLED RE-ENERGIZATION
  - RETURN TO SERVICE

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):

  - ASSOCIATED CIRCUITS OF CONCERN (INTERFERING CIRCUITS AS OPPOSED TO INTEGRAL SS/D CIRCUITS) [NOTE: CIRCUIT ANALYSIS ENFORCEMENT SUSPENDED INDEFINITELY BY EGM 98-002 REV 2 OF 2/2/00 AWAITING INDUSTRY RESOLUTION EFFORTS]

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM ASSOCIATED CIRCUITS LINES OF INQUIRY (CONTINUED):
  - COMMON POWER SUPPLY CONCERN (MULTIPLE HIGH IMPEDANCE FAULTS AND FUSE/BREAKER COORDINATION)
  - COMMON ENCLOSURE CONCERN (ELECTRICAL FAULT PROTECTION FROM NON-ESSENTIAL CIRCUITS)

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM ASSOCIATED CIRCUITS LINES OF INQUIRY (CONTINUED):
    - SPURIOUS SIGNAL CONCERN
        - HOT SHORTS
        - SHORTS TO GROUND
        - OPEN CIRCUITS

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):

  - SS/D SYSTEM SELECTION ADEQUACY
  - INDEPENDENCE OF REMOTE S/D PANEL FROM THE MAIN CONTROL ROOM
  - S/D CAPABILITY WITH AND W/O OFFSITE POWER
  - EFFECT OF FIRE-INDUCED CIRCUIT FAULTS ON TRANSFER OF CONTROL

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):

  - OPER. TRNG (OBSERVE ASD SIMULATOR)
  - SHUTDOWN STAFFING (ONSITE STAFF EXCLUSIVE OF FIRE BRIGADE)
  - PERIODIC OPERATIONAL TESTS OF ALTERNATIVE TRANSFER CAPABILITY
  - PROCEDURES

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):
  - TIMELINE (THERMO-HYDRAULIC ANALYSIS)
  - COMMUNICATION PLANS
  - HUMAN FACTORS
    - NUMBER OF MANUAL ACTIONS
    - FEASABILITY
    - HABITABILITY
    - ACCESS ROUTES INDEPENDENCE

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):

- PERIODIC OPERATIONAL TESTS OF REMOTE SHUTDOWN PANEL INSTRUMENTATION AND CONTROL FEATURES

- PORTABLE AND FIXED COMMUNICATIONS:
  - OPERABLE/AVAILABLE/RELIABLE
  - CLEAR WITH FULL COVERAGE

BASELINE PROCEDURE CONTENT (CONTINUED)

- TRIENNIAL TEAM LINES OF INQUIRY (CONTINUED):
  - COLD SHUTDOWN REPAIRS
    - DAMAGE SPECIFIC REPAIR PROCEDURES
    - DEDICATED ONSITE REPAIR EQUIPMENT AND MATERIALS
    - REPAIRS FEASIBLE WITHIN APPLICABLE TIME REQUIREMENTS

TRIENNIAL INSPECTION TRAINING

- ONE WEEK BNL/NRR CONDUCTED REGIONAL INSPECTOR TRAINING CLASSES CONDUCTED IN MARCH AND JUNE, 2000

- ONE DAY REGIONAL INSPECTOR REFRESHER TRAINING CONDUCTED IN EACH REGION SEPTEMBER, 2000

TRIENNIAL INSPECTION RESULTS

- NINE TRIENNIAL INSPECTION RESULTS SETS STUDIED
  - OVERALL RESULTS: 19 ISSUES, NO ISSUES AT TWO PLANTS

TRIENNIAL INSPECTION RESULTS OF INTEREST

- IN 92-18 "MECHANISTIC" (VERSUS FUNCTIONAL) DAMAGE PHENOMENON CONTESTED, AND NO LICENSEE ANALYSIS CONDUCTED

- "SINGLE SPURIOUS ACTUATION" ASSUMPTION MADE, BUT APPARENTLY NOT ACTUALLY APPLIED IN THE LICENSEE ANALYSIS (THEREFORE NO ISSUES DURING INSPECTION)

TRIENNIAL INSPECTION RESULTS OF INTEREST (CONTINUED)

- VARIOUS INCOMPLETE CIRCUIT ANALYSES, AND INCOMPLETE TRANSLATIONS OF SAFE SHUTDOWN ANALYSIS INTO PROCEDURES

- ALT. S/D CAPABILITY NOT INDEPENDENT OF FIRE AREA (VCT AND RWST VALVE CONTROL CABLES PLUS CHARGING PUMP POWER CABLES) - THREE PLANT AREAS

TRIENNIAL INSPECTION RESULTS OF INTEREST (CONTINUED)

- ALT. S/D CAPABILITY DID NOT ENSURE PRIMARY COOLANT INTEGRITY (LOSS OF RCP SEAL INJECTION W/O TEMPERATURE INDICATION FOR OPERATOR RCP TRIP) - THREE PLANT AREAS

RECENT CHANGE IN FP BASELINE INSPECTION SCOPE

- DIRECT ASSOCIATED CIRCUITS INSPECTION SUSPENDED UNTIL COMPLETION OF VOLUNTARY INDUSTRY CIRCUIT ANALYSIS INITIATIVE (FY 2001)

  - GENERAL ASSOCIATED CIRCUITS, IN 92 - 18, AND MHIF REVIEWS NOT TO BE CONDUCTED

RECENT CHANGE IN FP BASELINE INSPECTION SCOPE (CONTINUED)

- UNAVOIDABLE ("BYPRODUCT) ASSOC. CKTS ISSUES TEMPORARILY URIs

- INSPECTOR CAN STILL REVIEW:

  - ASSOCIATED CIRCUITS CALCULATIONS
  - PLANT CONFIGURATION ASSUMPTIONS
  - FUSE/BREAKER COORDINATION (NON-CONTROVERSIAL ISSUE)

RECENT CHANGE IN FP BASELINE INSPECTION SCOPE (CONTINUED)

- CHANGE RATIONALE:
  - RECENT UNDERSTANDING OF WIDE VARIABILITY IN LICENSING BASES, SOME AT VARIANCE WITH GL 86-10 ASSOCIATED CIRCUIT ANALYSIS CRITERIA
  - THEREFORE, ASSOC. CKTS ISSUES UNRESOLVABLE BY INSPECTION TEAM

(14)

ATTACHMENT 71111.05

INSPECTABLE AREA: Fire Protection

CORNERSTONES: Initiating Events (10%)
Mitigating Systems (90%)

INSPECTION BASES: Fire is generally a significant contributor to reactor plant risk. In many cases, the risk posed by fires is comparable to or exceeds the risk from internal events. The fire protection program shall extend the concept of defense in depth (DID) to fire protection in plant areas important to safety by (1) preventing fires from starting, (2) rapidly detecting, controlling, and extinguishing those fires that do occur, and (3) providing protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by fire suppression activities will not prevent the safe shutdown of the reactor plant. If DID is not maintained by an adequately implemented fire protection program, overall plant risk can increase.

This inspectable area verifies aspects of the Initiating Events and Mitigating Systems cornerstones for which there are no performance indicators to measure licensee performance.

The scope of this procedure has been reduced while criteria for review of fire-induced circuit failures of associated circuits is the subject of a voluntary industry initiative. Temporarily, the inspector is not required to address associated circuits issues as a direct line of inquiry nor develop associated circuits inspection findings (with certain exceptions contained in Section 02.03 of this procedure). However, in certain instances, associated circuits issues may arise unavoidably and indirectly during the inspector's review of safe shutdown system selection, redundant train separation, and the provision of independent alternative shutdown capabilities ("byproduct" associated circuits issues). These byproduct associated circuits issues shall be documented as unresolved items (URIs) awaiting generic resolution of the related associated circuits issues. The inspection report should reflect the temporary limitation in inspection scope, and the potential for "byproduct" associated circuits issues to exist as long-term (>180 day) unresolved items (URIs).

LEVEL OF EFFORT: Routine Inspection: The resident inspector will tour six to twelve plant areas important to reactor safety (on a plant specific basis) each calender quarter to observe conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational lineup, and operational effectiveness of fire protection systems, equipment and features; and (3) the material condition and operational status of fire barriers used to prevent fire damage or fire propagation.

Annual Inspection: In addition, for approximately two hours each year, the resident inspector will observe a plant fire drill.

Triennial Inspection: Every 3 years, an inspection team consisting of a fire protection specialist, a reactor systems engineer, and an electrical engineer will select three to five fire areas (fire zones where applicable) and conduct a design-based, plant specific, risk-informed, onsite inspection of the DID elements used to mitigate the consequences of a fire.

Identification and Resolution of Problems: Effort will include a review of licensee's problem identification and resolution of fire protection program.

71111.05-01 INSPECTION OBJECTIVES

01.01 The resident inspector inspection objective is to determine if the licensee has implemented a fire protection program that adequately controls combustibles and ignition sources within the plant, provides effectively maintained fire detection and suppression capability, maintains passive fire protection features in good material condition, and puts adequate compensatory measures in place for out-of-service, degraded or inoperable fire protection equipment, systems or features. The resident inspector approaches this effort from an operational status and material condition point of view.

01.02 The triennial team inspection objective is to assess, whether the licensee has implemented a fire protection program that adequately controls combustibles and ignition sources within the plant, provides adequate fire detection and suppression capability, maintains passive fire protection features in good material condition, puts adequate compensatory measures in place for out-of-service, degraded or inoperable fire protection equipment, systems or features, and ensures that procedures, equipment, fire barriers, and systems exist so that the post-fire capability to safely shut down the plant is ensured. The triennial team approaches this effort from a design point of view, as well as from the operational status and material condition points of view.

71111.05-02 INSPECTION REQUIREMENTS

02.01 Routine Inspection. The resident inspector will tour six to twelve plant areas important to safety (not necessarily limited to the top few contributors to overall plant fire risk) to assess the material condition of reactor plant active and passive fire protection systems and features, their operational lineup and operational effectiveness. For the areas selected, as applicable to the area of concern, conduct the following lines of inspection inquiry:

a. Control of Transient Combustibles and Ignition Sources

1. Observe if any transient combustible materials are located in the area. If transient combustible materials are observed, verify that they are being controlled in accordance with the licensee's administrative control procedures.

2. Observe if any welding or cutting (hot work) is being performed in the area. Verify that hot work is being done in accordance with the licensee's administrative control procedures.

b. Fire Detection Systems. Observe the physical condition of the fire detection devices and note any that show physical damage. Determine from licensee administrative controls the known material condition and operational status of the system, and verify that any observed conditions do not affect the operational effectiveness of the system (see compensatory measures section below).

c. Fire Suppression Systems

1. Sprinkler Fire Suppression Systems. Observe that sprinkler heads are not obstructed by major overhead equipment (e.g., ventilation ducts). Verify through visual observation or surveillance record review that the water supply control valves to the system are open and that the fire water supply and pumping capability is operable and capable of supplying the water supply demand of the system. Observe any material conditions that may affect performance of the system, such as mechanical damage, painted sprinkler heads, corrosion, etc.

2. Gaseous Suppression Systems. Observe that the gaseous suppression system (e.g. Halon or CO2) nozzles are not obstructed or blocked by

plant equipment such that gas dispersal would be significantly impeded. Observe and verify that the suppression agent charge pressure is within the normal band, extinguishing agent supply valves are open, and that the system is in the automatic mode. Observe and verify that the dampers/doors are unobstructed so that they will be permitted to close automatically upon actuation of the gaseous system. Observe and verify that the room penetration seals are sealed and in good condition. Observe and note any material conditions that may affect performance of the system, such as mechanical damage, corrosion, damage to doors or dampers, open penetrations, or nozzles blocked by plant equipment.

d. Manual Fire fighting Equipment and Capability

1. Fire Extinguishers. Ensure that portable fire extinguishes are provided at their designated locations in or near the area being inspected, and that access to the fire extinguishers is unobstructed by plant equipment or other work related activities. Observe and verify that the general condition of fire extinguishes is satisfactory (e.g., pressure gauge reads in the acceptable range, nozzles are clear and unobstructed, charge test records indicate testing within the normal periodicity).

2. Hose Stations and Standpipes. Observe that fire hoses are installed at their designated locations. Observe and verify that the general condition of hoses and hose stations is satisfactory (e.g., no holes in or chafing of the hose, nozzle not mechanically damaged and not obstructed, valve hand wheels in place). Observe and verify that the water supply control valves to the standpipe system are open and that the fire water supply and pumping capability is operable and capable of supplying the water flow and pressure demand. Ensure that access to the hose stations is unobstructed by plant equipment or work-related activities.

e. Passive Fire Protection Features

1. Electrical Raceway Fire Barrier Systems. Observe the material condition of electrical raceway fire barrier systems (e.g. cable tray fire wraps) and determine if there are any cracks, gouges, or holes in the barrier material, that there are no gaps in the material at joints or seams, and that banding, wire tie, and other fastener pattern and spacing appears appropriate. Where the fire barrier is a wrap or blanket-type material, observe that the material has no tears, rips, or holes in any of the visible layered material, that there are no gaps in the material at joint or seam locations, and that banding spacing is such that the material is held firmly in place. If plant modifications have recently been

conducted, establish that fire barriers removed as interference have been restored.

2. <u>Fire Doors</u>. Observe the material condition of the fire door in the area being inspected. Observe that selected fire doors close without gapping (e.g. due to fire door damage from previous obstructions), and that the door latching hardware functions securely.

3. <u>Ventilation System Fire Dampers</u>. To the extent practical and safe, directly observe the condition of the accessible ventilation fire dampers in the areas being inspected (to ensure fusible link fire dampers are not prematurely shut or obstructed). For those dampers which can not be readily observed in the selected plant areas, review the licensee's surveillance efforts directed towards verifying the continuing operability of ventilation fire dampers.

4. <u>Structural Steel Fire Proofing</u>. Observe the material condition of the structural steel fire-proofing (fibrous or concrete encapsulation) within the areas being inspected. Observe that this material is installed and that the structural steel is uniformly covered (no bare areas).

5. <u>Fire Barrier and Fire Area/Room/Zone Electrical Penetration Seals</u>. Tour plant areas being inspected and observe accessible electrical and piping penetrations. Observe whether any seals are missing from locations in which they appear to be needed to complete a fire barrier or area/room/zone wall, and determine that seals appear to be properly installed and in good condition.

6. <u>Reactor Coolant Pump Oil Collection Systems</u>. If applicable, verify that the licensee has installed a reactor coolant pump oil collection system which is designed to and does collect oil leakage and spray from all potential reactor coolant pump oil system leakage points.

f. <u>Compensatory Measures</u>. Verify that adequate compensatory measures are put in place by the licensee for out-of-service, degraded or inoperable fire protection equipment, systems or features (e.g. detection and suppression systems and equipment, passive fire barrier features, or safe shutdown functions or capabilities). Short term compensatory measures should be adequate to compensate for the degraded function or feature until appropriate corrective action can be taken. Review licensee effectiveness in returning the equipment to service in a reasonable period of time (typically days or weeks).

02.02 Annual Inspection. During the annual observation of a fire brigade drill in a plant area important to safety, evaluate the readiness of the licensee's personnel to prevent and fight fires, including the following aspects:

a. Protective clothing/turnout gear is properly donned.

b. Self-contained breather apparatus (SCBA) equipment is properly worn and used.

c. Fire hose lines are capable of reaching all necessary fire hazard locations, that the lines are laid out without flow constrictions, the hose is simulated being charged with water, and the nozzle is pattern (flow stream) tested prior to entering the fire area of concern.

d. The fire area of concern is entered in a controlled manner (e.g., fire brigade members stay low to the floor and feel the door for heat prior to entry into the fire area of concern).

e. Sufficient fire fighting equipment is brought to the scene by the fire brigade to properly perform their firefighting duties.

f. The fire brigade leader's fire fighting directions are thorough, clear, and effective.

g. Radio communications with the plant operators and between fire brigade members are efficient and effective.

h. Members of the fire brigade check for fire victims and propagation into other plant areas.

i. Effective smoke removal operations were simulated.

j. The fire fighting pre-plan strategies were utilized.

k. The licensee pre-planned the drill scenario was followed, and that the drill objectives acceptance criteria were met.

02.03 Triennial Inspection. Every three years, an inspection team will conduct risk-informed inspection of selected aspects of the licensee's fire protection program. The inspection will emphasize the review of post-fire safe shutdown capability, including the fire protection features provided to ensure that selected aspects the post-fire safe shutdown success path is maintained free of fire damage.

On a temporary basis, while certain associated circuits issues are the subject of an ongoing, voluntary industry initiative, the inspection team leader shall direct the triennial team inspectors, to NOT conduct direct and purposeful inspection of associated circuits issues. Associated circuits are defined in the

"Associated Circuits of Concern" section of the Generic Letter 81-12 Clarification Letter: Mattson to Eisenhut of March 22, 1982 "Fire Protection Rule - Appendix R." Certain exceptions to this temporary restriction are discussed in Section 02.03b.3 below.

a. Inspection Preparation. Select three to five fire areas (fire zones where applicable) important to risk for review. Obtain necessary information for determining post-fire safe shutdown capability and the fire protection features for maintaining post-fire safe shut down path free of fire damage.

b. Inspection Conduct. For the plant areas selected for review, conduct the following inspection efforts:

1. Systems Required to Achieve and Maintain Post-fire Safe Shutdown

Consider whether the licensee's shutdown methodology has properly identified the components and systems necessary to achieve and maintain safe shutdown conditions for each fire area, room and/or zone selected for review. Specifically determine the apparent adequacy of the systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring and support system functions.

If the above high level performance criteria are not met, review the licensee's engineering and/or licensing justifications (e.g., NRC guidance documents, license amendments, technical specifications, SERs, exemptions, deviations).

To the extent that it is confirmed that a postulated fire in an area under consideration can cause the loss of offsite power, verify that hot and cold shutdown from outside the control room can be achieved and maintained with off-site power not available.

2. Fire Protection of Safe Shutdown Capability

Evaluate the separation of systems, including power, control and instrumentation cables necessary to achieve safe shutdown, and verify that fire protection features are in place to satisfy the separation and design requirements of Section III.G of Appendix R (or, for reactor plants reviewed under the Standard Review Plan, license specific separation requirements).

Verify that the fire detectors and automatic fire suppression systems, associated with 1-hour fire barriers and/or 20 foot areas free of intervening combustibles required by Section III.G.2 of Appendix R (or, for reactor plants reviewed under the Standard Review Plan, license specific requirements), have been adequately installed. Review licensee evaluations which confirm, and verify through observation in the reactor plant, that selected installed automatic detection and suppression systems are installed in accordance with the code of record and would adequately control and suppress fires associated with the hazards of each selected area.

For the plant areas selected, when applicable, verify that redundant trains of systems required for hot shutdown located in the same fire area are not subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems. Determine each of the following:

(a) How the licensee has addressed whether a fire in a single location may, indirectly, through the production of smoke, heat, or hot gases, cause activation of potentially damaging fire suppression for all redundant trains,

(b) How the licensee has addressed whether a fire in a single location (or inadvertent actuation or rupture of a fire suppression system) may, through local fire suppression activity, indirectly cause damage to all redundant trains (e.g., sprinkler-caused flooding of other than the locally affected train), and

(c) How the licensee has addressed whether a fire in a single location may cause damage to all redundant trains through the utilization of manually controlled fire suppression systems.

For the plant areas selected, review the adequacy of the design (fire rating) of fire area boundaries (i.e., able to contain the fire hazards of the area), raceway fire barriers, equipment fire barriers, and fixed fire detection and suppression systems.

Evaluate licensee operator recovery action capabilities, plans and timing estimates for smoke removal, dewatering of spaces, controlled re-energization, and return to service of equipment in fire-affected areas) for fires in each plant area under consideration.

If a fire brigade drill is observed, consider the lines of inspection inquiry of Section 02.02 above.

3. <u>Post-fire Safe Shutdown Circuit Analysis</u>

Verify that safety-related and non-safety-related cables for selected post fire safe shutdown equipment in selected fire areas have been identified by the licensee and analyzed to show that they would not prevent safe shutdown because of hot shorts, open circuits, or shorts to ground.

The inspector is not precluded from developing findings related to purely deficient licensee performance in these areas. Thus for example, findings are not precluded where they are associated with mathematical errors or invalid plant configuration assumptions. Neither is the inspector precluded from developing findings in the specific associated circuits area of fuse/breaker coordination. However, the restriction does extend to IN 92-18 and multiple high impedance fault (MHIF) concerns (subjects of the current voluntary industry initiative).

Inspect the licensee's electrical systems and electrical circuit analyses with respect to the following:

(a) Common Power Supply/Bus Concern

(1) On a sample basis, for the safe shutdown equipment and cables located in the fire area, verify that circuit breaker coordination and fuse protection have been analyzed, provided and are acceptable as means of protecting the power source of the designated redundant or alternative safe shutdown equipment.

4. Alternative Shutdown Capability

Determine whether the licensee's alternative shutdown methodology has properly identified the components and systems necessary to achieve and maintain safe shutdown conditions for each fire area, room and/or zone selected for review. Specifically determine the apparent adequacy of the systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring and support system functions.

If the above high level performance criteria are not met, review the licensee's engineering and/or licensing justifications (e.g., NRC guidance documents, license amendments, technical specifications, SERs, exemptions, deviations).

Verify that hot and cold shutdown from outside the control room can be achieved and maintained with off-site power available or not available.

Verify that the transfer of control from the control room to the alternative location has been demonstrated to not be affected by fire-induced circuit faults (e.g. by the provision of separate fuses and power supplies for alternative shutdown control circuits).

5. Operational Implementation of Alternative Shutdown Capability

Verify that the training program for licensed and non-licensed personnel has been expanded to include alternative or dedicated safe shutdown capability.

Verify that personnel required to achieve and maintain the plant in hot shutdown following a fire using the alternative shutdown system can be provided from normal onsite staff, exclusive of the fire brigade.

Verify that adequate procedures for use of the alternative shutdown system exist. Verify the implementation and human factors adequacy of the alternative shutdown procedures by independently "walking through" the procedural steps. Ensure that adequate communications are available for the personnel performing alternative or dedicated safe shutdown. Verify that the operators can reasonably be expected to perform the procedures within applicable shutdown time requirements.

Establish whether the licensee conducts periodic operational tests of the alternative shutdown transfer capability and instrumentation and control functions. In addition, establish whether these tests are adequate to show that if called upon, the alternative shutdown capability would be functional upon transfer.

6. Communications

Verify through inspection of the contents of designated emergency storage lockers and review of alternative shutdown procedures, that portable radio communications and/or fixed emergency communications systems are available, operable, and adequate for the performance of alternative safe shutdown functions. Assess the capability of the communication systems to support the operators in the conduct and coordination of their required actions (e.g., consider ambient noise levels, clarity of reception, reliability, coverage patterns, and survivability). If specific, risk-significant issues arise relating to alternative shutdown communications adequacy, then, on a not-to-interfere with operational safety basis, observe licensee conducted communications tests in the subject plant area or areas.

7. Emergency Lighting

Review emergency lighting provided, either in fixed or portable form, along access routes and egress routes, at control stations, plant parameter monitoring locations, and at manual operating stations:

(a) If emergency lights are powered from a central battery or batteries, verify that the distribution system contains protective devices so that a fire in the area will not cause loss of emergency lighting in any unaffected area needed for safe shutdown operations.

(b) Review the manufacturer's information to verify that battery power supplies are rated with at least an 8-hour capacity.

(c) Determine if the operability testing and maintenance of the lighting units follow licensee procedures and accepted industry practice.

(d) Verify that sufficient illumination is provided to permit access for the monitoring of safe shutdown indications and/or the proper operation of safe shutdown equipment.

(e) Verify that emergency lighting unit batteries are being properly maintained (observe the unit's lamp or meter charge rate indication, and specific gravity indication).

8. Cold Shutdown Repairs

Verify that the licensee has dedicated repair procedures, equipment, and materials to accomplish repairs of damaged components required for cold shutdown, that these components can be made operable, and that cold shutdown can be achieved within time frames specified by Appendix R to 10 CFR Part 50 (or, for reactor plants reviewed under the Standard Review Plan, license specific requirements). Verify that the repair equipment, components, tools, and materials (e.g., pre-cut cable connectors with prepared attachment lugs) are available on site.

9. Fire Barrier and Fire Area/Zone/Room Penetration Seals

Selectively verify through review of installation records that material of an appropriate fire resistence rating (equal to the overall rating of the barrier itself) has been used to fill the opening/penetration .

10. Fire Protection Systems, Features and Equipment

In selected plant locations, review the material condition, operational lineup, operational effectiveness and design of fire detection systems, fire suppression systems, manual fire fighting equipment, fire brigade capabilities, and passive fire protection features. Establish that selected fire detection systems, sprinkler systems, gaseous suppression systems, portable fire extinguishers and hose stations are installed in accordance with their design, and that their design is adequate given the current equipment layout and plant configuration.

11. Compensatory Measures

Verify that adequate compensatory measures are put in place by the licensee for out-of-service, degraded or inoperable fire protection and post-fire safe shutdown equipment, systems or features (e.g. detection and suppression systems and equipment, passive fire barrier features, or pumps, valves or electrical devices providing safe shutdown functions or capabilities). Short term compensatory measures should be adequate to compensate for the degraded function or feature until appropriate corrective action can be taken. Review licensee effectiveness in returning the equipment to service in a reasonable period of time (typically days or weeks).

02.04 Identification and Resolution of Problems. During routine (quarterly and annual) resident inspection and triennial team inspection, verify that the licensee is identifying issues related to this inspection area at an appropriate threshold and entering them in the corrective action program. For a sample of selected issues documented in the corrective action program, verify that the corrective actions are appropriate. See Inspection Procedure 71152, "Identification and Resolution of Problems," for additional guidance.

71111.05-03 INSPECTION GUIDANCE

General Guidance

Routine Inspection. See Attachment 1.

The main focus of the resident inspector's activities is on the material condition and operational status of fire detection and suppression systems and equipment, and fire barriers used to prevent fire damage or fire propagation. The six to twelve plant areas to be inspected should be selected on the basis of site-specific risk worksheets.

Triennial Inspection

Objective. The triennial inspection is primarily a risk-informed look at the mitigation elements of fire protection defense in depth (DID) (i.e., detection, suppression, and confinement of fires through passive barriers, and the fire protection features and procedures which establish the licensee's ability to achieve and maintain post-fire safe shutdown conditions during and after a fire). The triennial inspection is that portion of the baseline inspection program that focuses on the design of reactor plant fire protection and post-fire safe shutdown systems, features, and procedures. The inspection team leader will manage and coordinate the conduct of an inspection emphasizing post-fire safe shutdown. The team will use plant-specific risk, event, and technical information (including the results of licensee self-assessments) to confirm that selected aspects of one train of safe shutdown equipment (capable of providing reactivity control, reactor coolant makeup, reactor heat removal, and process monitoring and support functions) is free of potential fire damage.

Inspection Team and Responsibilities. The team assigned to conduct the multi-disciplinary triennial fire protection inspection would include a fire protection inspector, an electrical inspector, and a reactor systems/mechanical systems inspector.

1. Reactor Systems/Mechanical Systems Inspector (RSI). The reactor systems/mechanical systems inspector (RSI) will assess the capability of reactor and balance-of-plant systems, equipment, operating personnel, and procedures to achieve and maintain post-fire safe shutdown and minimize the release of radioactivity to the environment in the event of fire. Therefore, the inspection team leader will ensure that he is knowledgeable regarding integrated plant operations, maintenance, testing, surveillance and quality assurance, reactor normal and off-normal operating procedures, and BWR and/or PWR nuclear and balance-of-plant systems design.

2. Electrical Inspector (EI). The EI will identify electrical separation requirements for redundant train power, control, and instrumentation cables. He will review alternative shutdown panel electrical isolation design to establish the panels' electrical independence from postulated fire areas. Therefore, the inspection team leader will ensure that he is knowledgeable regarding reactor plant electrical and instrumentation and control (I&C) design and is familiar with industry ampacity derating standards.

3. Fire Protection Inspector (FPI). The FPI will work with other team members in determining the effectiveness of the fire barriers and systems that establish the reactor plant's post-fire safe shutdown configuration and maintain it free of fire damage. He will determine whether suitable fire protection features (suppression, separation distance, fire barriers, etc.) are provided for the separation of equipment and cables required to ensure plant safety. Therefore, the inspection team leader will ensure he is knowledgeable regarding reactor plant fire protection systems, features and procedures.

Regulatory Requirements and Licensing Bases. The regulatory requirements and licensing bases against which post-fire safe shutdown capability is assessed are as follows:

1. Plants licensed before January 1, 1979. Effective February 17, 1981, the NRC amended its regulations by adding Section 50.48 and Appendix R to 10

CFR Part 50 to require certain provisions for fire protection in nuclear power plants licensed to operate before January 1, 1979. This action was taken to resolve certain contested generic issues in fire protection safety evaluation reports (SERs), and (1) to require all applicable licensees to upgrade their plants to a level of fire protection equivalent to the technical requirements in Sections III.G, J, L, and O of 10 CFR Part 50, Appendix R, and (2) to require all applicable licensees to meet all other requirements of Appendix R to the extent that comparable items had not been closed out in pre-Appendix R SERs (under Appendix A of the Branch Technical Position). Licensees were required to meet the separation requirements of Section III.G.2, the alternative or dedicated shutdown capability requirements of Sections III.G.3 and III.L, or to request an exemption in accordance with 10 CFR 50.48. Alternative or dedicated safe shutdown capabilities were required to be submitted to the Office of Nuclear Reactor Regulation (NRR) for review. NRR approvals are documented in SERs.

2. Plants licensed after January 1, 1979. These plants are subject to requirements similar to those in 10 CFR part 50, Appendix R, as specified in the conditions of their facility operating license, commitments made to the NRC, or deviations granted by the NRC. These reactor plants licensed after January 1, 1979, are subject to 10 CFR 50.48 (a) and (e) only.

   The fire hazards analysis (FHA) ("Fire Protection Review, Fire Protection Evaluation") document of the reactor plants licensed after January 1, 1979, may have been reviewed under Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear power Plants Docketed Prior to July 1, 1976," of August 23, 1976 (in which case, the licensee conducted an Appendix R comparison and justified final safety analysis report (FSAR) or FHA differences from the specific provisions of Appendix R). It is possible also that licensee submittals for plants licensed after January 1, 1979, were reviewed under the Standard Review Plan, NUREG-0800, and Branch Technical Position (BTP) CMEB 9.5-1 (formerly BTP ASB 9.5-1), "Guidelines for Fire Protection for Nuclear Power Plants," Rev. 2 (July 1981) (in which case, licensee submittals were reviewed according to requirements that closely paralleled the provisions of Appendix R).

   The actual fire protection requirements applicable to a given reactor plant licensed after January 1, 1979, arise from the specific license conditions in the facility operating license. These license conditions possibly refer to SERs and their supplements. Section 9.5 of such an SER delineates which licensee submittals were reviewed (e.g., a fire hazards analysis would be such a submittal).

3. All changes to fire protection license conditions which have been placed in the reactor plant's FSAR/USAR may be conducted under 10 CFR 50.59.

Inspection Process

1. Licensee Notification Letter. The licensee should be notified of the triennial inspection in writing at least three months in advance of the onsite week. The information gathering visit shall be conducted no fewer than three weeks in advance of the onsite inspection week. The letter should discuss the scope of the inspection, request an information-gathering visit to the licensee reactor site/engineering offices, discuss documentation and licensee personnel availability needs during the onsite inspection week, and request a pre-inspection conference call to discuss administrative matters and finalize inspection activity plans and schedules. A template for an NRC to licensee triennial fire protection baseline inspection notification letter is provided as Attachment 2.

2. Information-gathering Site Visit. The inspection team leader should conduct a two to three day information gathering site visit. The purposes of the information gathering site visit are to (1) gather site-specific information important to inspection planning, and (2) conduct initial discussions with licensee representatives regarding administrative items and inspection activity plans and schedules. In advance of the information-gathering site visit, the team leader should provide the licensee with a list of information and documents that may be needed for the team to prepare for and conduct the triennial inspection, as well as a list of any planned requests for licensee conducted evolutions (e.g., emergency lighting tests, communication tests, fire drills, shutdown walkthroughs, etc.).

2. Information Required/Preparation. The team members should gather sufficient information to become familiar with the following during preparation period:

   (a) The reactor plant's design, layout, and equipment configuration.

   (b) The reactor plant's current post-fire safe shutdown licensing basis through review of 10 CFR 50.48, 10 CFR Part 50 Appendix R (if applicable), NRC safety evaluation reports (SERs) on fire protection, the plant's operating license, updated final safety analysis report (UFSAR), and approved exemptions or deviations.

   (c) The licensee's strategy and methodology, and derivative procedures, for accomplishing post-fire safe shutdown conditions. Among the sources of information are the updated final safety analysis report (UFSAR), the latest version of the fire hazards analysis (FHA), the latest version of the post-fire safe shutdown analysis (SSA), fire protection/post-fire safe-shutdown related 10 CFR 50.59 and Generic Letter 86-10 review documentation and modification packages, plant drawings, emergency/abnormal operating procedures, and the results of licensee internal audits (e.g., self assessments and quality assurance (QA) audits in the fire protection and post-fire safe shutdown areas).

(d) The historical record of plant-specific fire protection issues through review of plant-specific documents such as previous NRC inspection results, internal audits performed by the reactor licensee (e.g., self-assessments and quality assurance audits), corrective action system records, event notifications submitted in accordance with 10 CFR 50.72, and licensee event reports (LERs) submitted in accordance with 10 CFR 50.73.

(e) The safe shutdown systems and support systems credited by the licensee's analysis for each fire area, room, or zone for accomplishing of the required shutdown functions (e.g., reactivity control, reactor coolant makeup, reactor heat removal, and process monitoring and support functions) as necessary to comply with the safe shutdown requirements of 10 CFR 50.48(a) and plant-specific licensing requirements. The shutdown logic for each area, room, or zone to be inspected must be thoroughly understood by the team members.

(f) The licensee's analytical approach for electrical circuits separation analyses, and the licensee's methodology for identification and resolution of associated circuits of concern. The team's electrical review should include addressing the assumptions and boundary conditions used in the performance of the licensee's analyses.

Specific Guidance

03.01 Routine Inspection. The resident inspector should not attempt to address all plant areas each inspection. The routine plant tour should focus on six to twelve plant areas important to risk. The resident inspector should note transient combustibles and ignition sources (and compare these with the limits provided in licensee administrative procedures). The resident inspector should also note the material condition and operational status (rather than the design) of fire detection and suppression systems, and fire barriers used to prevent fire damage or fire propagation.

03.02 No specific guidance provided

03.03 Triennial Inspection

1. Prior to the inspection information gathering trip, the team leader should contact the regional senior reactor analyst (SRA) to obtain summary of plant specific fire risk insights (e.g., fire risk ranking of the rooms/plant fire areas, conditional core damage probabilities (CCDPs) for those rooms and areas, and transient

sequences for these rooms). After considering the focus of past fire protection and post-fire safe shutdown inspections, the team leader should select three to five fire areas important to risk for inspection

2. The fire protection and post-fire safe shutdown information gathered should focus on the samples selected.

3. After the information gathering site visit, the team leader should use the SRA developed fire risk insights, as well as technical input from the other team members, to develop an inspection plan addressing (for the selected three to five fire areas, zones, as applicable) post-fire safe shutdown capability and the fire protection features for maintaining one success path of this capability free of fire damage.

Inspection Requirement 02.03b2 Short term compensatory measures should be adequate to compensate for the degraded function or feature until appropriate corrective action can be taken.

03.04 Identification and Resolution of Problems. No specific guidance is provided.

71111.05-04 RESOURCE ESTIMATE

The resource to perform this inspection procedure is estimated to be, on average, 33 hours per year for routine inspection including approximately 2 hours for annual observation of a fire drill and 200 hours every 3 years for the triennial inspection regardless of the number of reactor units at the site.

71111.05-05 REFERENCES

The SDP Guideline "Appendix 4 - Determining Potential Risk Significance of Fire Protection and Post-fire Safe Shutdown Inspection Findings."

Appendix H of the Fire Protection Supplemental Inspection Procedure (FPSI) "Guidance for Making a Qualitative Assessment of Fire Protection Inspection Findings, Fire Protection Risk Significance Screening Methodology" [FPRSSM])

Inspection Procedure 71152, "Identification and Resolution of Problems."

Generic Letter 91-18 "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Non-conforming Conditions and on Operability."

Information Notice 97-48 "Inadequate or Inappropriate Interim Fire Protection Compensatory Measures," July 9, 1997

NRC Internal Memorandum dated August 17, 1998, from John N. Hannon to Arthur T. Howell titled "Response to Region IV Task Interface Agreement (TIA) (96TIA008) - Evaluation of Definition of Continuous Fire Watch (TAC No. M96550).

Individual Plant Examination of Externally Initiated Events(IPEEE)

END

ATTACHMENT 1
ROUTINE INSPECTION GUIDANCE TABLE

CORNERSTONE	RISK PRIORITY	EXAMPLES
INITIATING EVENTS	Equipment or actions that could cause or contribute to initiation of fires in plant areas important to safety or near equipment required for safe shutdown.	Transient combustibles (rags, wood, ion exchange resin, lubricating oil, or Anti-Cs) are not in areas where transient combustibles are prohibited. Transient combustible amounts in other areas do not exceed administrative controls. Ignition sources (welding, grinding, brazing, flame cutting) have a fire watch. Planning includes precautions and additional fire prevention measures where these activities are near combustibles.

MITIGATING SYSTEMS	Functionality of fire barriers in plant areas important to safety. Functionality of detection systems in plant area important to safety. Functionality of automatic suppression systems in plant areas important to safety. Fire brigade manual suppression effectiveness. Compensatory measures for degraded fire detection systems, fire suppression features, and barriers to fire propagation.	Doors and dampers that prevent the spread of fires to/or between plant areas important to safety remain in place and are functional. Electrical raceway fire barriers and penetration seals that protect the post-fire safe-shutdown train are not damaged. Fire detection and alarm system is functional for plant areas important to safety. Automatic suppression system sprinklers are functional and their sprinkler head patterns are not blocked by plant equipment. Fire brigade performance indicates a prompt response with proper fire fighting techniques for the type of fire encountered. Manual fire suppression equipment is of the proper type and has been tested. Degraded fire detection equipment, suppression features and fire propagation barriers are adequately compensated for on reasonably short-term bases.

Mr. , President
Licensee Nuclear Department
Licensee Corporation or Company
Address

SUBJECT: SELECTED NUCLEAR POWER STATION, UNITS 1 AND 2 - NOTIFICATION OF CONDUCT OF A TRIENNIAL FIRE PROTECTION BASELINE INSPECTION

Dear Mr. :

The purpose of this letter is to notify you that the U.S. Nuclear Regulatory Commission (NRC) Region # staff will conduct a triennial fire protection baseline inspection at Selected Nuclear Power Station, Units 1 and 2 in Month, 20##. The inspection team will be lead by First Last, a fire protection specialist from the NRC Region # Office. The team will be composed of personnel from NRC Region #, and Contracted National Laboratory. The inspection will be conducted in accordance with IP 71111.05, the NRC's baseline fire protection inspection procedure

The schedule for the inspection is as follows:

- Information gathering visit - Month ##-##, 20## [Note - this date is pre-coordinated with the licensee]
- Week of onsite inspection - Month ##, 20##.

The purposes of the information gathering visit are to obtain information and documentation needed to support the inspection, to become familiar with the Selected Nuclear Power Station, Units 1 and 2 fire protection programs, fire protection features, and post-fire safe shutdown capabilities and plant layout, and, as necessary, obtain plant specific site access training and badging for unescorted site access. A list of the types of documents the team may be interested in reviewing, and possibly obtaining, are listed in Enclosure 1.

During the information gathering visit, the team will also discuss the following inspection support administrative details: office space size and location; specific documents requested to be made available to the team in their office spaces; arrangements for reactor site access (including radiation protection training, security, safety and fitness for duty requirements); and the availability of knowledgeable plant engineering and licensing organization personnel to serve as points of contact during the inspection.

We request that during the onsite inspection week you ensure that copies of analyses, evaluations or documentation regarding the implementation and maintenance of the Selected Nuclear Generating Station, Units 1 and 2 fire protection program, including post-fire safe shutdown capability, be readily accessible to the team for their review. Of specific interest are those documents which establish that your fire protection program satisfies NRC regulatory requirements and conforms to applicable NRC and industry fire protection guidance. Also, personnel should be available at the site during the inspection who are knowledgeable regarding those plant systems required to achieve and maintain safe shutdown conditions from inside and outside the control room (including the electrical aspects of the relevant post-fire safe shutdown analyses), reactor plant fire protection systems and features, and the Selected Nuclear Power Station fire protection program and its implementation.

Your cooperation and support during this inspection will be appreciated. If you have questions concerning this inspection, or the inspection team's information or logistical needs, please contact First Last, the team leader, in the Region # Office at ###-###-####.

Sincerely,

Docket Nos.: 50-###
and 50-###

Enclosure: As stated (1)

Reactor Fire Protection Program Supporting Documentation

[Note: This is a broad list of the documents the NRC inspection team may be interested in reviewing, and possibly obtaining, during the information gathering site visit.]

1. The current version of the Fire Protection Program and Fire Hazards Analysis.

2. Current versions of the fire protection program implementing procedures (e.g., administrative controls, surveillance testing, fire brigade).

3. Fire brigade training program and pre-fire plans.

4. Post-fire safe shutdown systems and separation analysis.

5. Post-fire alternative shutdown analysis.

6. Piping and instrumentation (flow) diagrams showing the components used to achieve and maintain hot standby and cold shutdown for fires outside the control room and those components used for those areas requiring alternative shutdown capability.

7. Plant layout and equipment drawings which identify the physical plant locations of hot standby and cold shutdown equipment.

8. Plant layout drawings which identify plant fire area delineation, areas protected by automatic fire suppression and detection, and the locations of fire protection equipment.

9. Plant layout drawings which identify the general location of the post-fire emergency lighting units.

10. Plant operating procedures which would be used and describe shutdown from inside the control room with a postulated fire occurring in any plant area outside the control room, procedures which would be used to implement alternative shutdown capability in the event of a fire in either the control or cable spreading room.

11. Maintenance and surveillance testing procedures for alternative shutdown capability and fire barriers, detectors, pumps and suppression systems.

12. Maintenance procedures which routinely verify fuse breaker coordination in accordance with the post-fire safe shutdown coordination analysis.

13. A sample of significant fire protection and post-fire safe shutdown related design change packages (including their associated 10 CFR 50.59 evaluations) and Generic Letter 86-10 evaluations.

14. The reactor plant's IPEEE, results of any post-IPEEE reviews, and listings of actions taken/plant modifications conducted in response to IPEEE information.

15. Temporary modification procedures.

16. Organization charts of site personnel down to the level of fire protection staff personnel.

17. If applicable, layout/arrangement drawings of potential reactor coolant/recirculation pump lube oil system leakage points and associated lube oil collection systems.

18. A listing of the SERs and actual copies of the 50.59 reviews which form the licensing basis for the reactor plant's post-fire safe shutdown configuration.

19. Procedures/instructions that control the configuration of the reactor plant's fire protection program, features, and post-fire safe shutdown methodology and system design.

22. A list of applicable codes and standards related to the design of plant fire protection features and evaluations of code deviations.

23. Procedures/instructions that govern the implementation of plant modifications, maintenance, and special operations, and their impact on fire protection.

24. The three most recent fire protection QA audits and/or fire protection self-assessments.

25. Recent QA surveillances of fire protection activities.

26. A listing of open and closed fire protection condition reports (problem reports/NCRs/EARs/problem identification and resolution reports).

27. Listing of plant fire protection licensing basis documents.

28. A listing of the NFPA code versions committed to (NFPA codes of record).

29. A listing of plant deviations from code commitments.

30. Actual copies of Generic Letter 86-10 evaluations.

END

11/3 @ 11:00

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ACRS MEETING HANDOUT

Meeting No. 477TH	Agenda Item 11	Handout No.: 11.1

Title: License Renewal Guidance Documents

Author: Mario V. Bonaca

List of Documents Attached

Memorandum from Christopher Grimes, NRR, to Noel Dudley, ACRS/ACNW, Subject: ACRS Subcommittee Follow-up Actions, dated November 1, 2000.

11

Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person
N. Dudley

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 1, 2000

Note to: Noel Dudley, Senior Staff Engineer
Advisory Committee on Reactor Safeguards

FROM: Christopher Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs, NRR

CIGrimes

Subject: ACRS SUBCOMMITTEE FOLLOW-UP ACTIONS

During the ACRS subcommittee meeting on October 19-20, 2000, we made several commitments related to background information and follow-up actions. The purpose of this note is to provide the background information and confirm the commitments for future actions.

In response to the specific request from Dr. Shack, attached are samples of typical Nuclear Energy Institute (NEI) comments on pre-August draft Generic Aging Lessons Learned (GALL) report and draft Standard Review Plan (SRP) for License Renewal attached. Industry comments on the August draft of GALL and SRP from Union of Concern Scientist (UCS) are included in ADAMS; the accession number for UCS comments is ML003763009. We shall provide you with the accession number for NEI comments dated, October 13, 2000, as soon as we have confirmed it is in ADAMS. If you prefer we can provide you with the hard copies of those comments.

During the subcommittee meeting, we committed to take the following additional actions relative to the improved renewal guidance in GALL and the SRP:

1. We will review the transcript of the subcommittee meeting to identify ACRS suggestions for improvements of the, for example (a) clarifying table of contents, (b) expanding the description of dams in the table, and (c) clarifying where one-time inspections are recommended;

2. We will be prepared to explain the treatment of the FSAR supplement, technical specifications and the environmental review in more detail during the ACRS subcommittee meeting on the ANO-1 application;

3. We will plan on publication of GALL and SRP in loose-leaf form to facilitate future updates;

4. We will share the summary of all the public comments on the improved renewal guidance that will be prepared for the Commission meeting on December 4, 2000, as soon as available.

If the subcommittee has any questions or comments about our plans, please contact m

ATTACHMENT

SAMPLE NEI PRE-AUGUST COMMENTS ON SRP

Comment No.	Page No.	SRP Paragraph	Comment and Basis	Recommendation
1	2.1-1	2.1.1.2	Delete "and (2)." 54.21(a)(2) is the methodology requirement	Rewrite as "The methodology used by the applicant to implement the "screening" requirements of 10CFR54.21(a)(1) is reviewed."
2	2.1-3	2.1.3, 4, and 5	The LR Rule is deterministic not probabilistic. 60FR22468: "... [T]he Commission concludes that it is inappropriate to establish a licensee renewal scoping criterion, ..., that relies on plant-specific probabilistic analyses. Therefore, within the construct of the final rule, PRA techniques are of very limited use for license renewal scoping."	Delete paragraphs 4 and 5. . Renumber the following paragraphs as "4." and "5." Also remove the example referring to the IPEEE on page 2.1-5 in 2.1.3.1.1
3	2.1.1 2	Table 2.2-1	Emergency Operating Procedures are for mitigating DBE's and not for design purposes.	Delete Emergency Operating Procedure

ATTACHMENT

SAMPLE NEI PRE-AUGUST COMMENTS ON GALL

Comment No.	GALL Page No.	Comment and Basis	Recommendation
1	VI A-0	Electric Cables is too broad a term since, by its name, it does not distinguish it from grounding system conductors and transmission conductors. The critical distinguishing factor for electric cables is whether they are insulated or uninsulated. It makes sense to review all insulated cables together since they have similar functions to maintain related to the insulating materials. The term "insulated cables" would also distinguish it from other, non-electric cables since non-electric cables (e.g., crane cables) are not insulated.	Change "Electric Cables" to "Insulated Cables".
2	VI A-4	The paragraph for case (ii) states "..., *and the period of time prior to the end of qualified life when the reanalysis will be completed.*" Case (ii) are those TLAAs that "have been" projected. The reanalyses have already been performed at the time of application. This language was probably meant for case (iii).	Remove this statement from the case (ii) discussion.
3	IIA1-5,9	The Aging Management Programs imply that there are additional requirements for in-service inspection of inaccessible areas when there are no indications of degradation for accessible areas.	These implications should be removed. Basis: implying such requirements is equal to additional rule-making over and above 10 CFR 50.55a without adhering to the rule-making process.

UNITED STATES NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

November 3, 2000

LICENSE RENEWAL GUIDANCE DOCUMENTS

6

- AGENDA -

TOPIC		PRESENTER
I.	Introduction	M. Bonaca, ACRS Chairman of Subcommittee
II.	Improved License Renewal Guidance Documents	Sam Lee, NRR
	A. Changes Between August 2000 and Previous Drafts of Guidance Documents	
	B. Future Updates	
III.	Stakeholders Comments on Previous Drafts	
	A. Nuclear Energy Institute	Jerry Dozier, NRR
	B. Union of Concerned Scientists	
	C. License Renewal Issues Inventory	Steve Koenick, NRR
IV.	License Renewal Rule	Steve Hoffman, NRR
	A. Scope of Rule Regarding EOPs	
	B. Continuation of Voluntary Commitments	
V.	Generic Aging Lessons Learned Report	
	A. Example of One-Time Inspection Guidance	Tamara Bloomer, NRR
	B. Cable Aging	Jit Vora, RES
	C. Neutron Embrittlement of Reactor Vessel Internals	Barry Elliot, NRR
VI.	Nuclear Energy Institute Comments	D. Walters, NEI
VII.	Discussion	M. Bonaca, ACRS

IMPROVED LICENSE RENEWAL GUIDANCE DOCUMENTS

Generic Aging Lessons Learned (GALL) Report

December 1999 version →→→ August 2000 version

- **Relocated Chapter 1, "Introduction," to Volume 1 summary**
- **New Chapters I, X, and XI**
- **Deleted:**

Mark I concrete containment	Chapter II
Fan cooler systems	Chapter V
Other than cables and connectors	Chapter VI
Liquid waste disposal system	Chapter VII
Inservice testing program	Chapters V, VII, VIII

- **Added:**

Carbon steel components (Boric acid and atomospheric corrosion)	Chapters V, VII, VIII
Closure bolting	Chapters V, VII, VIII

- **Modified GALL locations of containment isolation valves**

August 2000 version →→→ March 2001 version

- **Single page format with extensive Chapter XI**

IMPROVED RENEWAL GUIDANCE DOCUMENTS (Continued)

Generic Aging Lessons Learned (GALL) Report

Table of Contents for Volume 2 (Tabulation of Results):

IMPROVED RENEWAL GUIDANCE DOCUMENTS (Continued)

Standard Review Plan

September 1997 version →→→ April 2000 version

- Complete rewrite to incorporate lessons learned and GALL report

April 2000 version →→→ August 2000 version

- Incorporated August 2000 version of GALL report

Regulatory Guide

August 1996 version →→→ August 2000 version

- Removed exceptions

IMPROVED RENEWAL GUIDANCE DOCUMENTS (Continued)

<u>NEI 95-10</u>

Revision 0 (March 1996) →→→ Revision 2 (August 2000)

- Incorporated standard format of application
- Incorporated 10-element program review process
- Removed examples

<u>Future Updates</u>

- Guidance documents are living documents capturing lessons learned
- Frequency of update to be determined

STAKEHOLDERS COMMENTS ON PREVIOUS DRAFTS

Nuclear Energy Institute Comments

- Use of GALL report for scoping
- Use of minimum programs
- Use minimum program descriptions
- Applicable aging effects
- Inaccessible areas

STAKEHOLDERS COMMENTS ON PREVIOUS DRAFTS (Continued)

Union of Concerned Scientists Comments

- UCS provided 5 reports for consideration as input to GALL
- Component/aging effects were identified from the reports and compared to GALL
- The jet pump sensing line and separator support ring were added to the August version of GALL

DISPOSITION OF LICENSE RENEWAL INVENTORY

- Inventory based on September 1997 Draft Standard Review Plan

- Letter to NEI and UCS, "Disposition of License Renewal Issue Inventory," dated May 4, 2000

    - August 2000 Draft SRP incorporated Generic Aging Lessons Learned Report

    - License renewal inventory incorporated into revised guidance documents

    - Aggressive publication schedule encompasses inventory

    - Stakeholders agreement on process for disposition of inventory with provision for additional feedback through public comment period

LICENSE RENEWAL RULE

- Scope of the rule regarding emergency operating procedures (EOPs)
- Continuation of voluntary commitments during the period of extended operation

EXAMPLE OF ONE-TIME INSPECTION GUIDANCE

- Some aging effects of specific structures and components may need verification to the effectiveness of the aging management program

- One way that has been postulated within GALL and SRP is a one-time inspection

EXAMPLE OF ONE-TIME INSPECTION GUIDANCE (Continued)

Example: Spent Fuel Pool Cooling and Cleanup---Corrosion

- GALL, Chapter VII, Section A3, page A3-4, Item A3.2.1: The "Aging Management Program" column (page A3-5) states: "An acceptable verification program consists of a one-time inspection..." and the "Further Evaluation" column states: "Yes, detection of aging effects should be further evaluated"

- SRP, Section 3.3, Subsection 3.3.2.2.1, page 3.3-3: "A one-time inspection of select components and susceptible locations is an acceptable method . . ."

- SRP, Section 3.3, Table 3.3-2, page 3.3-18, FSAR supplement: "One- Time Inspection: To verify the effectiveness of the reactor water chemistry program consists of a one-time inspection of . . ."

A3.1.1	Piping	Closure Bolting	Carbon Steel (CS) Low Alloy Steel (LAS)	Air, Leaking Chemically Treated Borated Water	Loss of Material	Boric Acid Corrosion (BAC)	NRC GL 88-05. ASME Section XI, 1989 or later Edition as approved in 10 CFR 50 55a. NRC IN 86-108.
A3.2.1	Filter	Housing	Carbon Steel (CS) with Lining	Chemically Treated Borated Water	Loss of Material	Pitting and Crevice Corrosion	EPRI TR-105714. (Rev. 3 or later updates or revisions of the report)
A3.2.1	Filter	Housing (External Surface)	CS	Air, Leaking Chemically Treated Borated Water	Loss of Material	BAC	*Same as for the effect of Boric Acid Corrosion on Item A3.1.1 piping closure bolting.*

The AMP relies on implementation of NRC Generic Letter 88-05 and inservice inspection (ISI) in conformance with ASME Section XI (1989 edition), Subsection IWC, Table IWC 2500–1, to monitor the condition of the reactor coolant pressure boundary for occurrence of borated water leakage.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M5, "Boric Acid Corrosion."	No
The AMP relies on the water chemistry program which consists of monitoring and control of water chemistry based on EPRI guidelines of TR-105714 for primary water chemistry in PWRs to manage the effects of loss of material due to crevice or pitting corrosion. However, crevice or pitting corrosion may occur at locations of stagnant flow conditions, and verification of the effectiveness of the chemistry control program should ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. An acceptable verification program consists of a one-time inspection of select component and susceptible locations in the system.	For the evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M11, "Water Chemistry."	Yes, detection of aging effects should be further evaluated
Same as for the effect of Boric Acid Corrosion on Item A3.1.1 piping closure bolting.	*Same as for the effect of Boric Acid Corrosion on Item A3.1.1 piping closure bolting.*	No

3.3.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL report indicates that further evaluation should be performed for:

3.3.2.2.1 Loss of Material From General, Microbiologically Influenced, Galvanic, Pitting, and Crevice Corrosion

Loss of material from general, microbiologically influenced, pitting, and crevice corrosion could occur in carbon steel piping, valve bodies, pump casing, tanks, heat exchangers, and ion exchangers in the spent fuel pool cooling and cleanup system (BWR and PWR), and the shutdown cooling system (older BWR). The water chemistry program relies on monitoring and control of reactor water chemistry based on EPRI guidelines of TR-103515 for water chemistry in BWRs, TR-105714 for primary water chemistry in PWRs, and TR-102134 for secondary water chemistry in PWRs to manage the effects of loss of material from crevice or pitting corrosion. However, high concentrations of impurities at crevices and locations of stagnant flow conditions could cause crevice or pitting or microbiologically influenced corrosion. Therefore, verification of the effectiveness of the chemistry control program should be performed to ensure that corrosion is not occurring. The GALL report recommends further evaluation of programs to manage loss of material from general, microbiologically influenced, pitting, and crevice corrosion to verify the effectiveness of the water chemistry program. A one-time inspection of select components and susceptible locations is an acceptable method to ensure that corrosion is not occurring and the component's intended function will be maintained during the period of extended operation.

Table 3.3-1. Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of the GALL Report

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/ PWR	Components in spent fuel pool cooling and cleanup	Loss of material from general and pitting and crevice corrosion	Water chemistry	Yes, detection of aging effects should be further evaluated (see subsection 3.3.2.2.1)
BWR/ PWR	Valve lining in spent fuel pool cooling and cleanup system, and seals in ventilation systems	Materials degradation from cracking, wear, or hardening from loss of strength	Plant-specific	Yes, plant-specific (see subsection 3.3.2.2.2)
BWR/ PWR	Diesel fuel oil strainer and tanks	Loss of material from corrosion, buildup of deposit from biofouling	Fuel oil chemistry	Yes, detection of aging effects should be further evaluated (see subsection 3.3.2.2.9)

Fuel oil chemistry (BWR/PWR)	The AMP relies on a combination of surveillance and maintenance procedures. Monitoring and controlling of fuel oil contamination in accordance with the guidelines of ASTM Standards D975, D270, D1796, D2276, and D2709 maintains the fuel oil quality. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic cleaning/draining tanks and by verifying the quality of new oil before its introduction into the storage tanks.	Existing program
Inservice inspection (BWR/PWR)	The program consists of periodic volumetric, surface, and/or visual examination of components and their supports for signs of degradation, assessment, and corrective actions. This program is in accordance with ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.	Existing program
One-time inspection	To verify the effectiveness of fuel oil program, a one–time thickness measurement of the tank bottom is performed. To verify the effectiveness of the reactor water chemistry program consists of a one–time inspection of internal surfaces of carbon steel piping, valve bodies, pump casing, and tanks, is performed using suitable techniques at the most susceptible locations is performed to ensure that corrosion is not occurring. To verify the effectiveness of fire protection program, a one-time visual inspection for the bottom half of the inside of the tank is an acceptable option is performed to ensure that corrosion is not occurring.	The inspection should be completed before the period of extended operation.

CABLE AGING
Condition Monitoring (CM) Methods

Issues Involving CM Methods

- Evaluating bulk or localized properties of insulating materials
- Evaluating integrity of a cable system, end-to-end
- Determining residual life/service life

Technical Challenge

- Detect and locate incipient defects and localized anomalies prior to failures in an installed cable system
- Condition monitoring method(s) needs to be non-intrusive, reliable and cost effective
- Effective ground planes are not readily available for scanning unshielded low-voltage I&C cables. This makes electrical tests at system's level more challenging.

Condition Monitoring (CM) Methods (Continued)

CM Methods Evaluated as Part of Research Program for Resolution of GSI-168

- Visual Inspection
- Elongation-at-Break
- Oxidation Induction Time
- Oxidation Induction Temperature
- Fourier Transform Infrared Spectroscopy
- Indenter
- Hardness
- Dielectric Loss
- Insulation resistence
- Functional Performance Test
- Voltage Withstand Test

To date, the staff has not identified one single CM method that is effective to scan an entire length of a cable system, end-to-end, that meets the criteria of "non-intrusive," "reliable," and "cost effective."

Cable Aging and Resolution of GSI-168
(In context of license renewal - 60 years)

- For license renewal, EQ of cables is considered a time-limited aging analysis (TLAA)

- The requirements of 10CFR 54.21(c) provides three options to demonstrate continued EQ during the renewed license term

- For EQ equipment licensees are expected to continue to comply with the requirements of 10CFR 50.49 (EQ rule) during the renewal period

- CLB carries forward during the renewed license term. The outcome of the GSI-168 resolution for the current license term in turn applies to license renewal

- As discussed during the staff's presentation to the ACRS on GSI-168 on October 6, the staff is still evaluating various options for the resolution of GSI-168

- For non-EQ cables, licensees are expected to propose an appropriate aging management program for license renewal

NEUTRON EMBRITTLEMENT OF REACTOR VESSEL INTERNALS

- Reactor vessel materials neutron embrittlement threshold is 10E17 according to Appendix H to Part 50

- Reactor vessel internals consist of wrought and cast stainless steels and welds

  - Staff proposes screening criterion of 10E17

  - Most susceptible locations should be inspected

(17)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.: ✓
477th	13	13-2

Title

FUTURE ACRS ACTIVITIES

Authors

John T. Larkins

List of Documents Attached

Future ACRS Activities - 477th ACRS Meeting, November 2-4, 2000

13

Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person

John T. Larkins

ANTICIPATED WORKLOAD
November 2-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	Leitch	Markley	Risk-Informed Regulation Implementation Plan	Report (tentative)		P&P 10/31 (P.M.)
Apostolakis	Shack	Markley	Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3)	Report		
Bonaca	Seale	Dudley	License Renewal GuidanceDocuments: SRP, GALL, Regulatory Guide, and NEI 95-10	Report		P&P 10/31 (P.M.) M&M 11/16
Kress	--	El-Zeftawy/ Weston	Spent Fuel Pool Accident Risk at Decommissioning Plants	Report	SAM 11/15	TH 11/13-14 M&M 11/16
Powers	All Members	El-Zeftawy	Research Report to the Commission	Report	P&P 10/31 (P.M.)	
		Duraiswamy/Shoop	Differing Professional Opinion on Steam Generator Tube Integrity			

ANTICIPATED WORKLOAD
November 2-4, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Sieber	Powers –	Singh	Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues	Report	10/16-17	SAM 11/15
		Singh	ABB/CE and Siemens Digital I&C Applications (Subcommittee Report)	–	PS 10/31	

ANTICIPATED WORKLOAD
December 7-9, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG. CHAIR.	SUBC. MTG. MEMBER
Apostolakis	Kress	Markley	Modifications to the Reactor Safety Goal Policy Statement	Report		
Bonaca	Wallis	Boehnert	Central Issues Related to Core Power Uprate Reviews	Report	–	P&P 12/6 (P.M)
Kress	–	Boehnert	Control Room Habitability	Report	–	
Powers	–	Duraiswamy/Shoop	DPO on Steam Generator Tube Integrity	Report	P&P 12/6 (P.M)	
	–	El-Zeftawy	Research Report to the Commission [possible finalization of letter @ retreat?] (draft)	Report		
		Larkins	**Commissioner Diaz-** Periodic meeting with the ACRS			
Shack	–	Dudley	Proposed Final Regulatory Guide DG-1053, "Calibration and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"	Report		

ANTICIPATED WORKLOAD
December 7-9, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Sieber	Apostolakis	Weston	Draft Safety Evaluation for South Texas Project Exemptions from special treatment requirements	Report		
Uhrig		Singh	ABB/CE and Siemens Digital I&C Applications (Subcommittee Report)	Report		
Wallis		Boehnert	Report on Nov. 13-14 T/H Phenomena Subcommittee Meeting - Review of TRACG Code	--		
		Boehnert	Response to Commission SRM regarding how NRC should address observed weaknesses with T/H Codes	Report		

ANTICIPATED WORKLOAD
FEBRUARY 1-3, 2001

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis		Markley	ANS Standard on PRA External-Events	Report		ACRS/ACNW Joint Sub.1/19/01
Powers		El-Zeftawy	Research Report to the Commission	Report		
		Larkins	**Chairman Meserve-** Meeting with the ACRS			
Seale		Singh	Management Directive 6.4 to address ACRS Concerns Associated with the Generic Safety Issue Process	Report		
		Dudley	Fitness for Duty (Scope) Rulemaking	Report		
Sieber		Singh	GSI-152 ,"Reprioritization of Valves Subject to Blowdown Loads"	Report		
Sieber		Singh	MOX Fuel Fabrication Facility	--		

II. ITEMS REQUIRING COMMITTEE ACTION

11. GSI-152, Design Basis for Valves that Might Be Subject to Significant Blowdown Loads (Open) (REU/AS) ESTIMATED TIME: 1 hour

    Purpose: Review and Comment

    NRC staff review request. [Owen Gormley, RES]. This issue was identified by the Office of Nuclear Regulatory Research (RES) following ACRS concerns raised during the 355th meeting regarding the resolution of GSI-87, "HPCI Steam Line Break Without Isolation." GSI-87 addressed the design bases for those MOVs that isolate the HPCI, RCIC, and RWCU systems in BWRs. These design bases required that the MOVs close against loads imposed by a double-ended pipe break at design basis flow conditions.

    In resolving Issue 87, the staff issued Generic Letter No. 89-10 which required licensees to identify safety-related valves that might not perform adequately under design basis conditions. However, the ACRS believed that the design basis for the HPCI steam line valves and other valves in some plants might not specify this type of heavy duty. Thus, it was possible that heavy duty loads might not be considered for these valves by licensees in response to Generic Letter No. 89-10. The ACRS recommended in its letter of November 20, 1989, that the staff amend the generic letter to require licensees to examine their design bases to determine if safety-related valves, including but not limited to MOVs, were capable of operating against blowdown loads that might not have been considered (by licensees) in their original designs. However, the staff chose to identify a new generic issue instead, GSI-152, because, unlike GSI 87, the question was the adequacy of the design bases rather than the ability of the valves to meet the requirements set forth in the design bases.

    The staff plans to provide the resolution package to ACRS in early January 2001 and brief the Committee in February 2001.

    The Planning and Procedures Subcommittee recommends that Dr. Uhrig propose a course of action.

12. SECY-00-0145,"Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning" (Open) (TSK/MME) ESTIMATED TIME: 1 ½ hours

    Purpose: Review and Comment

    Review requested by the ACRS. The subject SECY, "Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning" issued on June 28, 2000, requests Commission approval to proceed with developing an integrated rulemaking for nuclear power plant decommissioning. The regulatory areas addressed by this rulemaking plan are emergency planning, insurance, safeguards, staffing and training, and backfit.

The staff briefed the ACRS in April 2000 regarding the draft technical study on spent fuel pool accident risk at decommissioning nuclear power plants. The ACRS issued its report on April 13, 2000 regarding this issue and the first recommendation was "The integrated rulemaking on decommissioning should be put on hold **until** the staff provides technical justification for the proposed acceptance criterion for fuel uncovery frequency. ....." The staff, however, on page 3 of SECY-00-0145 (second paragraph) states that "The staff believes that the ACRS comments **will not** impact the overall conclusions of the staff's risk study."

SECY-00-0145 describes sample regulatory languages for emergency planning, insurance, security, operator staffing and training, and applying the backfit rule. The staff also did not approve NEI's request for adapting 10 CFR Part 50 to decommissioning plants. The subject SECY provides two options on this issue; namely:

- **Option 1**, approval of this rulemaking could be placed on **hold** until the staff has provided the Commission a more comprehensive assessment of decommissioning regulatory improvements.

- **Option 2,** approve the initiation of the integrated rulemaking plan.

The staff indicated its preference for Option 1. However, the staff's reason seems to be the absence of any anticipated nuclear power plant decommissionings in the near future, rather than the importance of the ACRS comments and the inadequacy of the technical study. The Commission returned SECY-00-0145 to the staff without vote, pending further developments in this area, and directed the staff to submit a revised paper to the Commission by January 31, 2001 (SRM dated September 27, 2000).

Dr. Kress plans to provide his views on the need for the Committee to review this matter following the staff's presentation on the revised technical study of spent fuel pool accident risk at decommissioning plants during the November 2000 ACRS meeting.

13. Topical Report BAW-2374 Concerning Eliminating LOCAs in Licensing Basis for Once-Through Steam Generators (Open)(WJS/NFD) ESTIMATED TIME: 1 ½ hours

Purpose: Review and Comment

ACRS requested opportunity to review this issue in a Larkinsgram dated July 17, 2000 [S. Bailey, NRR]. The Committee considered Topical Report BAW-2374, "Justification For Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis For Existing and Replacement Once-Through Steam Generators," during the July 12-14, ACRS meeting. The Committee decided that it would like the opportunity to review this matter after the staff prepares the safety evaluation.

The staff plans to provide the ACRS a copy of the proposed safety evaluation by November 3, 2000. The approval of BWR-2374 is necessary to support replacement of the Oconee steam generators during the outage that begins on November 23, 2000.

The Planning and Procedures Subcommittee recommends that Dr. Shack propose a course of action.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 27, 2000

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards
Advisory Committee on Nuclear Waste

FROM: John W. Craig
Assistant for Operations
Office of the Executive Director for Operations

SUBJECT: PROPOSED AGENDA ITEMS FOR THE ACRS AND THE ACNW MEETINGS

Attached is a list of proposed agenda items for the ACRS (December 2000 - April 2001) and the ACNW (November 2000 - February 2001). This list was compiled based upon information received from (1) NRR, NMSS, RES, and IRO in response to the EDO request for the monthly update of proposed agenda items, and (2) the ACRS/ACNW staffs at a meeting held on October 24, 2000 with the OEDO, NRR, NMSS, and RES ACRS/ACNW coordinators [OEDO, I. Schoenfeld; NRR, M.G. Crutchley; NMSS, R.H. Turtil; RES, J.A. Mitchell and S.R. Nesmith].

A copy of the Work Items Tracking System (WITS) list for December 2000 - February 2001 is also attached. This list includes a projection of office originated Commission papers that may be of interest to the ACRS/ACNW. Please provide timely feedback on your interest for briefings on particular items identified from the projected Commission papers that were not planned for formal review or information briefings but that are of interest to the Committees.

Attachments: As stated

ML003764124

PROPOSED AGENDA FOR
ACRS MEETINGS
(December 2000 - February 2001)

ACRS MEETING — December 7 - 9, 2000				
Item #	**Title/Issue**	**Purpose**	**Priority**	**Documents**
1	DG-1053; Dosimetry and NeutronTransport Contact: W. Jones, DET/RES	Review and Comment	High	Draft regulatory guide was provided 10/23/00.
2	Control Room Habitability Contact: J. Hayes, DSSA/NRR	Review and Comment	High	Revisions to NEI 99-03 were provided to staff and ACRS (P. Boehnert) on 10/13/00. NRC holding public meeting to discuss revisions on 11/25/00. Extensive comments will be provided to NEI with a copy to ACRS on 11/30/00.
3	South Texas Exemption from Scope of Special Requirements Contact: J. Nakoski, DLPM/NRR	Review and Comment	High	Draft SER to be provided by 11/3/00.
4	Central Issues Related to Core Power Update Reviews Contact: T. Kim, DLPM/NRR	Information Briefing	Medium	None.
5	Safety Goal Policy Contact: J. Murphy, RES	Review and Comment	High	Draft paper to be provided 11/7/00.

ACRS MEETING ---- FEBRUARY 2001				
Item #	Title/Issue	Purpose	Priority	Documents
1	Status of MD 6.4, "Generic Issues Program" Contact:H. Vandermolen, DSARE/RES	Review and Comment	Medium	Draft SECY paper on MD 6.4 to be provided by January 9, 2001
2	Effectiveness of the ATWS Rule Contact: W. Raughley, DSARE/RES	Review and Comment	Medium	Draft ATWS report provided to ACRS in late September 2000.
3	GSI-152, Reprioritization of Valves Subject to Blowdown Loads Contact: O. Gormley, DET/RES	Review and Comment	High	Documents to be provided by 1/7/01.
4	Siemens S-RELAPS Appendix K Small-Break LOCA Code Contact: R. Caruso/R.Landry, DSSA/NRR	Review and Comment	High	SER on Code to be provided mid-December.
5	Fitness for Duty (Scope) Rulemaking Contact: G. West, DIPM/NRR	Review and Comment	High	Proposed rule to be provided in January 2001.
6	Overview of Licensing of Mixed Oxide Fuel Fabrication Facility Contact: A. Persinko, FCSS/NMSS	Information Briefing	Low	None.
7	GSI-152 Contact: O. Gormlet, RES	Review and Comment	High	Document January 5, 2001.

ACRS MEETING ---- MARCH 2001				
Item #	**Title/Issue**	**Purpose**	**Priority**	**Documents**
1	Waterhammer Issues	Review and Comment	High	EPRI interim report to be provided by 2/1/01.
	Contact: J. Tatum, DSSA/NRR			
2	ANO License Renewal	Review and Comment	Medium	SER with open items to be provided by 2/8/01
	Contact: S. Hoffman, DRIP/NRR			

ACRS MEETING ---- APRIL 2001				
Item #	**Title/Issue**	**Purpose**	**Priority**	**Documents**
1	NEI 97-06 Steam Generator Program Guidelines	Review and Comment	Medium	Draft SER to be provided by mid-March.
	Contact: E. Sullivan, DE/NRR			
2	Hatch License Renewal	Review and Comment	Medium	SER with open items to be provided by 1/10/01.
	Contact: E. Sullivan, DE/NRR			
3	Risk-Based Performance Indicators	Review and Comment		ACRS received the draft report on the results of Phase 1 development of risk-based indicators on October 16, 2000.
	Contact: S. Mays, DRAA/RES			

PROPOSED AGENDA FOR
ACNW MEETINGS
(November 2000 - January 2001)

ACNW MEETING —— NOVEMBER 27-29, 2000 (San Antonio)				
Item #	Title/Issue	Purpose	Priority	Documents
1	Staff Site Recommendation - Strategy and Guidance Contact: B. Reamer, DWM/NMSS	Information Briefing	High	Guidance on Site Recommendation.
2	Research Plan for the Waste Strategic Arena Contact: C. Trottier, DRAA/RES	Information Briefing	Medium	Draft Research Plan to be provided by 11/1/00.

ACNW MEETING —— DECEMBER 2000				
Item #	Title/Issue	Purpose	Priority	Documents
	No scheduled meeting.			

ACNW MEETING —— JANUARY 16 - 18, 2001				
Item #	Title/Issue	Purpose	Priority	Documents
1	Institutional Control Status Contact: L. Camper, DWM/NMSS	Information Briefing	Medium	None.
2	Division of Waste Management Overview Director's Briefing Contact: J. Greeves, DWM/NMSS	Information Briefing	Low	None.

/3

ACNW MEETING —— FEBRUARY 2001

Item #	Title/Issue	Purpose	Priority	Documents
	No scheduled meeting.			

ACRS MEETING HANDOUT

G Drive:PP

(18)

Meeting No.	Agenda Item	Handout No:
477th	15	15.2

Title MINUTES OF PLANNING & PROCEDURES SUBCOMMITTEE MEETING - OCTOBER 31, 2000

Authors JOHN T. LARKINS

List of Documents Attached

13

Instructions to Preparer

1. Punch holes
2. Paginate attachments
3. Place copy in file box

From Staff Person

JOHN T. LARKINS

G:PlanPro:ppmins.477
Nov. 1, 2000

SUMMARY MINUTES OF THE PLANNING AND PROCEDURES SUBCOMMITTEE MEETING TUESDAY, OCTOBER 31, 2000

The ACRS Subcommittee on Planning and Procedures held a meeting on October 31, 2000, in Room 2B1, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 1:00 p.m. and adjourned at 4:30 p.m.

ATTENDEES

D. A. Powers, Chairman
G. Apostolakis
M. Bonaca

ACRS STAFF

J. T. Larkins
J. Lyons
R. P. Savio
H. Larson
S. Duraiswamy
C. Harris
S. Meador
Maggelean Weston
Ethel Barnard

NRC STAFF

I. Schoenfeld

DISCUSSION

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the November ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the November ACRS meeting are included in a separate handout. Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the November 2000 ACRS meeting be as shown in the handout. After the staff's presentation on the

Risk-Informed Regulation Implementation Plan, the Committee should decide whether to write a letter on this matter. Also, a letter on the ABB/CE and Siemens digital I&C applications should be prepared during the December ACRS meeting.

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through February 2001 is included in a separate handout. The objectives are to:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate.

- Manage the members' workload for these meetings.

- Plan and schedule items for ACRS discussion of topical and emerging issues.

During this session, the Subcommittee discussed and developed recommendations on the items that require Committee decision, which are included in Section II of the Future Activities list.

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee needs to consider the Subcommittee's recommendations on items listed in Section II of the Future Activities.

3) CY2000 Self Assessment

The ACRS will be holding its annual planning meeting in January 2001 and conducting its CY 2000 self assessment. Dr. Savio was assigned the task of selecting a small group of ACRS work products that would be the subject of critical analysis by the ACRS members during the January 2001 planning meeting. The focus would be on selecting activities that would provide lessons learned. A proposed list along with proposed metrics for use in identifying lessons learned is included in the attachment (pp. 1-2). A list of key ACRS work processes that could be included in the ACRS' CY 2000 self assessment is also included in the attachment on pp. 1-2.

RECOMMENDATION

The Subcommittee recommends the following:

- The proposed list of work products and metrics should be revised incorporating the Subcommittee members' comments and submitted to the ACRS for comment. (The included attachment has been modified.) The ACRS members should provide feedback to Dr. Savio.

- A member should be assigned to each work product to lead the discussion with regard to lessons learned, effectiveness, timeliness, and quality and should make recommendations for improvements.

- Dr. Savio should review the transcript of the ACRS meeting with NEI in October and develop a list of issues raised by NEI and follow-up items resulting from that meeting. This list will be provided to the Planning and Procedures Subcommittee during its December 6th meeting.

- Dr. Savio should assess the letters and reports issued by the Committee during 2000 and identify those letters and reports, if any, that could have been handled through Larkinsgrams. Also, those letters and reports, which endorsed the staff positions should be identified.

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4) ACRS Retreat for 2001

During the October meeting, the Committee agreed to have a retreat locally. A decision needs to be made on the dates for the retreat.

RECOMMENDATION

okd

The Subcommittee recommends that the Committee select January 22-24, 2001 as the dates for the retreat. Also, it should assign a lead member to work with the ACRS Executive Director to develop an agenda for the retreat.

5) Proposed ACRS Meeting Dates for CY 2001

The proposed dates for CY 2001 ACRS meetings listed below were distributed to the members during the October 2000 ACRS meeting.

RECOMMENDATION

The Subcommittee recommends that the Committee approve these dates listed below during the November ACRS meeting.

ACRS Meeting No.	Proposed Meeting Dates for 2001
--	January 2000 - No meeting
479	February 1-3, 2001
480	March 1-3, 2001
481	April 5-7, 2001
482	May 10-12, 2001
483	June 6-8, 2001
484	July 11-13, 2001
--	August 2001 - No meeting
485	September ~~5-7, 2001~~ 6,7,8
486	October 4-6, 2001
487	November 8-10, 2001
488	December 6-8, 2001

OKd

6) ACRS Action Plan for CY 2001-2002

During the May 2000 ACRS meeting, the Committee approved the development of an ACRS Action Plan for CY 2001-2002. A draft Action Plan (pp. 3) prepared by the ACRS staff is attached for review and comment by the ACRS members. This draft incorporates preliminary comments provided by the Planning and Procedures Subcommittee members. Subsequent to receiving the comments from the members, a revised draft will be prepared incorporating, as appropriate, the members' comments. The revised draft will be discussed by the Planning and Procedures Subcommittee during its December meeting. Subject to Subcommittee concurrence, it will be submitted to the full Committee for approval at the December 2000 ACRS meeting. Subsequently, the ACRS Action Plan and Operating Plan will be forwarded to the Commission. We anticipate comments on the Action Plan.

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the draft Action Plan to the ACRS staff engineer, Maggalean Weston, by November 19, 2000.

7) Estimation of Resources for FY 2001

Due to the anticipated high workload facing the ACRS in FY 2001, it is important to plan how to use member time most efficiently and effectively. Assuming the number of ACRS members remains constant throughout FY 2001, the maximum member time that will be available is 1,300 days.

During last month's Planning and Procedures Subcommittee meeting, we discussed the need to manage better the number of Subcommittee meetings and the number of members participating in Subcommittee meetings. Senior staff engineers with input from Subcommittee chairmen were asked to revise the estimate of the number of Subcommittee meetings for FY 2001. The current estimate shows 36 Subcommittee meetings, 10 full Committee meetings and 1 retreat, consuming a total of approximately 1155 days. During the October ACRS meeting, the Planning and Procedures Subcommittee informed the Committee that it plans to scrutinize these proposed Subcommittee meetings to assess where some cuts might be made or combining of Subcommittee meetings might be done. This is to make sure we do not exceed the maximum days available for members to work and also not to overburden members.

RECOMMENDATION

The Subcommittee recommends the following:

- Cognizant Subcommittee Chairman should evaluate the need for frequent Subcommittee meetings and should provide a clear justification for the need for a Subcommittee meeting.

- Subcommittee meetings should be held only to review complex technical issues and/or issues of contention. Otherwise, the matter should be scheduled for discussion during a full Committee meeting.

- Subcommittee meetings should not be scheduled to review preliminary draft documents that are expected to be revised extensively.

- Subcommittee Chairman should consider holding informal meetings with the staff on certain issues to gather information rather than holding a Subcommittee meeting.

- As a general practice, if a Subcommittee meeting is held, a product (letter or report) should result from that meeting.

The Planning and Procedures Subcommittee will prioritize the Subcommittee meetings, as warranted, to manage the budget and the members' workload.

8) Election of Officers for CY 2001

The election of Chairman and Vice-Chairman for the ACRS and Member-at-Large for the Planning and Procedures Subcommittee will be held during the December 7-9, 2000, ACRS meeting. In accordance with Section 8.4 of the ACRS Bylaws, those members who do not wish to be considered for any of the above offices should notify the ACRS Executive Director in writing at least two weeks prior to the December meeting.

9) ACRS Annual Christmas Party

The ACRS has over the past several years sponsored a Christmas Party for ACRS/ACNW staff and other selected invitees (e.g., Commissioners, EDO, etc.). The Committee should decide whether it wants to continue this tradition and sponsor a Christmas party during the December 2000 ACRS meeting.

RECOMMENDATION

The Subcommittee recommends that the Committee sponsor a Christmas Party to be held on December 8, 2000. ACRS members should provide $60.00 for the Christmas Party to Jenny Gallo (T 2 E10) as soon as possible.

10) Other Issues

- The Subcommittee recommends that those members who are scheduled to attend the meeting with RSK and visit the Siemens facility in Germany in November prepare a report, outlining the issues discussed and follow-up items resulting from that meeting. In view of the significant expenses involved in attending foreign meetings, the members should assess the value added by this meeting and recommend whether such meetings should be held in the future.

- Information gathered during the visit of Siemens, should be factored, as appropriate, into the preparation of the ACRS report at the December 2000 meeting on the ABB/CE and Siemens topical reports on digital I&C applications.

- Committee should encourage attendance by the NRC staff at the conference on Innovation in Government to be held at the Kennedy School of Government at Harvard, Cambridge, Massachusetts. Dr. Powers has agreed to contact the NRC Chairman's office on this matter. The ACRS staff should provide Dr. Powers with additional information on this conference.

CY 2001 SELF ASSESSMENT

Select a small group of ACRS work products for critical analysis by the ACRS during its CY 2001 self assessment. The focus will be on lessons-learned from analysis of the particular activity and not on success or failure. (Chose six to eight of the following work products and construct specific questions to be used as tools for exploring the issues. --- I have used asterisks to identify my choices.) Identify metrics (questions) that would be included in this evaluation. Separately from this, identify key ACRS committee processes that could be self evaluated.

WORK PRODUCTS

An ACRS member will be assigned the lead for providing a critique at the retreat of each work product selected.

Low power shutdown operations risk *
Proposed modifications to the Safety Goal Policy *
Transient and accident analysis code review *
License renewal *
Review of reactor operating experience *
Spent fuel pool fires *
Risk-informed Part 50 *
120 month ISI ASME code updates *

MOX and HBU fuels activities
Power uprates
Joint ACRS and ACNW work on defense in depth/ NMSS risk-informed regulation
Revised reactor oversight process

KEY ACRS PROCESSES

ACRS interactions and communications with its stakeholders
Process for ACRS selection of self-initiated work
Selection of ACRS tasks
Meeting preparation (Agenda planning, information dissemination, etc)
Writing Committee reports
Joint ACRS/ACNW Subcommittee and joint ACRS/ACNW work
Communication with individual Commissioners
Preparation for Commission Briefings
Strategy and process for producing the annual research report
Annual visits to a Region office and operating plant
Lessons learned from license renewal review process

METRICS

Did the work result in important ACRS advice (letter report, oral communication with the Commission, etc) or some other important work product?

What was the guiding ACRS regulatory philosophy and is it being consistently applied in ACRS advice ? (To expand on the meaning, does the ACRS as a committee have a consensus-based regulatory philosophy or do individual reports reflect the philosophy of the members having the most expertise on the subject and, whatever the answer, what is preferred ? As examples, is the approach to the application of defense-in-depth consistent, are recommendations for regulatory changes risk-informed, and do recommendations for regulatory changes met the requirements of the Backfit Rule)

Did the ACRS make a persuasive case, and why or why not? (Did the ACRS state its position clearly, explain its rationale clearly, and where needed make its arguments in a manner such that they could not be ignored ?)

Was the ACRS review informed in that it considered the relevant information, stakeholder views, and focused on the relevant regulatory safety issues ?

Did the ACRS influence the regulatory decision in a significant way, and why or why not?

Was the review efficient in its use of ACRS and stakeholder resources?

NOTE: Redline/strickeout changes represent changes suggested by Drs. Apostolakis, Bonaca, Powers, and Larkins. Staff comments has not been reconciled yet. These will be dispositioned after P&P resolution.

Inside Front Cover

The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory Committee to the Atomic Energy Commission (AEC) by a 1957 amendment to the Atomic Energy Act of 1954. The functions of the Committee are described in Sections 29 and 182b of the Act. The Energy Reorganization Act of 1974 transferred the AEC licensing functions to the U. S. Nuclear Regulatory Commission (NRC), and the Committee has continued in the same advisory role to the NRC.

The ACRS reports directly to the Commission. It provides the Commission with independent reviews of, and advice on, the safety of proposed or existing NRC licensed reactor facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible and the safety- and risk-significant NRC regulations and guidance relating to these facilities. On its own initiative, the ACRS may conduct reviews of specific generic matters or nuclear facility safety- and risk-significant items. The Committee also advises the Commission on safety- and risk-significant policy issues, and performs other duties as the Commission may request. Upon request from the DOE, the ACRS provides advice on U.S. Naval reactor designs and hazards associated with DOE nuclear activities and facilities. Upon request, the ACRS provides technical advice to the Defense Nuclear Facilities Safety Board.

ACRS operations are governed by the Federal Advisory Committee Act (FACA), which is implemented through NRC regulations at 10 CFR Part 7. ACRS operational practices encourage the public, industry, state, and local governments, and other stakeholders to become involved in Committee activities. The ACRS and the ACNW work cooperatively in reviewing matters of interest to the Commission and, where the Committees' responsibilities overlap, divide work in the manner that best serves the interests of the Commission.

10/26/00

This plan provides guidance and direction to the Advisory Committee Reactor Safeguards (ACRS) in the year 2001 and beyond for focusing on issues most important to the U. S. Nuclear Regulatory Commission in carrying out its mission of protecting public health and safety, promoting the common defense and security, and protecting the environment. It also provides the ACRS mission, goals, objectives, and priorities consistent with NRC's Strategic Plan.

SCOPE OF ACRS ACTIVITIES

The Committee reports to and advises the Commission on technical matters related to nuclear reactor safety and safeguards. The basis of ACRS reviews includes, in part, 10 CFR Parts 20, 21, 26, 50, 51, 52, 54, 55, 70, 72, 73, 76, 100, and other applicable legislation and regulations. Current regulatory activities that are within the scope of ACRS responsibilities include license renewal, application of risk-informed and performance based regulations, reactor operations, rulemaking, codes and standards, generic safety issues, research, and other regulatory activities issues as requested by the Commission. The Committee interacts with representatives of the NRC, stakeholders, the public, DOE, Advisory Committee on Nuclear Waste (ACNW), other Federal agencies, State, Tribal and local governments, as well as private, international, and other organizations as appropriate to fulfill its responsibility.

ACRS MISSION

The mission of the ACRS is to provide the Commission with useful, independent, and timely technical advice on issues of public safety related to nuclear reactors and reactor safeguards to support the NRC in conducting an efficient regulatory program that enables the Nation to use nuclear power in a safe manner for civilian purposes.

ACRS VISION

ACRS envisages safety regulation of nuclear power plants based on a coherent set of necessary and sufficient requirements securely founded on science, engineering, and quantitative risk assessment. The ACRS wants to be looked upon to provide advice and recommend solutions that are: (a) relevant, effective, and timely, (b) technically sound and reflect state-of-the-art knowledge, (c) balanced and unbiased, (d) address safety significant issues, (e) forward looking, (f) can be implemented, and (g) reflect the need to balance risk, benefit, and cost to society to enable the safe use of nuclear power.

ACRS OPERATING PRINCIPLES

The ACRS strives to ensure that Commission and EDO priorities are understood and adequately considered in setting the Committee's agenda. It makes its letters and reports clear and concise. The ACRS continues to believe that early involvement is, on balance, the best approach for the resolution of complex issues and that early involvement provides ACRS input when it is most efficient and effective. It believes, also, that it is most effective when it involves itself in the resolution of broad technical issues. The Committee will continue to maintain its independence as it reviews issues.

OUTCOMES AND COMMITMENTS

The Committee aspires to achieve the following outcomes:

1 Provide advice in adequate time for consideration by the Commission in making regulatory decisions.
2. Alert the Commission to potential challenges that may be averted by taking interim action.
3. Forewarn the Commission of emerging issues that may require action at a later time.

4. Advice reflects state-of-the-art technology, yet is practical, and allows for incorporation into NRC technical approaches, regulations, and guidance.
5. Advice is clear and concise.
6. Advice reflects an understanding of inherent risks and considers the need to balance risk, cost and benefit in all of NRC's decisions.
7. ACRS advice is valued by the Commission, the NRC staff, DOE, the public, and other stakeholders.
8. ACRS is trusted by the public for providing frank, open advice, and for offering a forum for public participation in the regulatory process.
9. ACRS assists in resolving conflicts between NRC and other stakeholders by encouraging communication and providing a neutral forum for interaction.

The Committee will carry out the following commitments to accomplish its mission.

1. Be responsive to Commission needs.
2. Focus on nuclear safety.
3. Maintain technical excellence and independence.
4. Foster an atmosphere of mutual problem-solving with the NRC staff.
5. Challenge the status quo, as appropriate, thereby becoming an "agent for change."
6. Remain flexible, be responsive to change, and consider various options and contingencies.
7. Identify in advance those issues that could impact NRC's ability to achieve its mission.
8. Focus on risk by asking, "what is the risk, what are the important contributors to risk, and what are the uncertainties associated with the risk?
9. Keep abreast of international trends and developments that could affect NRC regulatory practices or approaches and factor international experience into Committee advice, where appropriate.
10. Seek to improve approaches for public involvement.

GOALS AND OBJECTIVES

In keeping with its mission, the ACRS has developed goals and objectives consistent with the performance goals in the NRC Strategic Plan. The objectives reflect current regulatory needs.

Goal 1 Provide useful advice to the Commission that will support the NRC in responding to the evolutions and challenges to the safe use of nuclear power.

Objective 1 Advise the Commission in a timely fashion on issues of a technical nature that may require regulatory changes in the following areas:

- use and implementation of risk-informed and performance-based, safety regulations.
- the agency effort relating to the revised oversight process.
- relevant plant operations and significant operational events for emerging safety issues.
- age-related degradation safety issues.
- research efforts that provide the technical bases for NRC regulatory decisions.

Objective 2 Recommend to the Commission solutions to issues that may pose challenges for the NRC or the public if not given adequate attention in the following areas:

- applications for license renewals
- the implementation of the revised reactor oversight process
- risk-informed and performance based activities steam generator tube integrity issues
- the safety research program.
- Review and comment on generic safety issues
- rulemaking and regulatory guidance.

Goal 2 Provide support to the NRC in building and maintaining public trust by involving the public in its review process of nuclear power safety and safeguards.

Objective 1 Ensure opportunities for meaningful public involvement in the regulatory process through the FACA process and through other communication initiatives to keep the public informed.

Objective 2 Foster an open, accessible, and clear, yet independent review process.

Objective 3 Assist the NRC in ensuring that agency decision making is a more transparent process by making sure that agency documentation reviewed by the Committee is thorough, clear, and readily understandable.

Goal 3 Support the effectiveness and efficiency of NRC operations.

Objective 1 Advise the NRC on how to increase its reliance on risk insights as a basis for decision making, including using risk assessment methods for the safe use of nuclear power, that (1) implement a risk-informed approach, (2) quantify and reveal uncertainties, and (3) are consistent across programs, where possible.

Objective 2 Propose approaches to gain a better understanding of the inherent risks associated with nuclear power and the relationship between regulations, cost, and safety.

Objective 3 Support the increased use of information technology and other media to improve stakeholder input to the regulatory process.

Objective 4 Propose technically sound and realistic approaches for resolving new and emerging issues related to the safe operation of nuclear power plants.

Goal 4 Support NRC use of state-of-the-art technology in resolving key safety issues in an effort to help reduce unnecessary regulatory burden on stakeholders

Objective 1 Keep abreast of challenges of new technologies being developed and utilized worldwide and changing regulatory demands.

Objective 2 Recommend ways to utilize risk-informed, performance based approaches to reduce unnecessary burden.

Objective 3 Recommend ways to use the revised reactor oversight process to gain efficiencies in the assessment of nuclear power plant operations.

Objective 4 Advise the Commission of projected needs for additional NRC technical capabilities that could enhance the agency's ability to address safety issues effectively.

Goal 5 Improve the effectiveness and efficiency of ACRS Operations.

Objective 1 Increase the value of ACRS advice to the Commission and the staff.

Objective 2 Maintain innovative and sound business practices that are focused on outcomes and provide effective tools for establishing goals.

Objective 3 Improve and modify operational procedures for reporting on program accomplishments and matters of accountability.

Objective 4 Build upon mutually beneficial relationships with the NRC staff and stakeholders to enhance the effectiveness of the review process.

CRITERIA FOR SELECTING PRIORITY ISSUES

The following criteria are used in the determination of the priority of issues that the ACRS reviews.

- Issues that are required by law and by regulations.
- Issues that have immediate impact on nuclear safety.
- Issues that are risk significant or for protection of health and safety.
- Issues that have the potential for or likelihood to pose undue risk or costs to society.
- Issues that are requested by the Commission or the Commissioners.
- Issues that are requested by the EDO.
- Issues for which the ACRS review is self initiated.
- Issues of timeliness related to Commission schedule and when the advice would be of greatest benefit to aid in the Commission's regulatory decisions.
- Issues that relate to the NRC Strategic Plan, including trends and direction in regulatory practice.
- Issues that arise from strategies and activities of licensees and applicants.

PRIORITY ISSUES

License Renewal

10 CFR 54.25 requires that each license renewal application be referred to the ACRS for a review and report. An ACRS review is essential given the safety implications of extending power operation of a significant number of plants for 20 years beyond their current licensed terms. ACRS involvement is also important because congressional and industry interests have made license renewal a high-priority item for the Commission. This places significant pressure on the NRC staff to expedite the review process and to reduce demonstration and documentation requirements at the very time that the interpretation of the License Renewal Rule and detailed requirements and guidance for future applications are being finalized. ACRS involvement will help in the ongoing development of a standardized license renewal process.

The ACRS will play a valuable role by:

- Participating in the development of a standardized license renewal process to ensure that detailed requirements for license renewal applications are ~~necessary and~~ sufficient to provide reasonable assurance that plants will operate safely for up to 60 years.

- Identifying significant issues and focusing attention on the way these issues are addressed in individual applications.

- Providing to the Commission independent views on contested interpretation of the rule, such as the issue of credit for existing programs.

- Identifying issues, as appropriate, that may be outside the narrow confines of the rule, for example, using risk information to further improve the license renewal process.

The ACRS has performed expedited reviews of the applications for renewal of licenses for the Calvert Cliffs Nuclear Power Plant Units 1 and 2 and the Oconee Nuclear Station Units 1,2, and 3 and provided timely advice to the Commission.

The ACRS will continue to play a significant role in license renewal area, including:

- Reviewing each license renewal application.

- Reviewing license renewal guidance documents (Standard Review Plan, Regulatory Guide, Generic Aging Lessons Learned Report, NEI 95-10, Industry Guidelines for Implementing the Requirements of the License Renewal Rule).

- Reviewing selected industry topical reports.

- Visiting plants, as needed and as resource permits, to gather information on the changes made to structures, systems, and components to support the extended plant operation, adequacy of the aging management programs, and other significant activities related to license renewal.

- Implementing an efficient process to ensure timely completion of the ACRS review of license renewal applications and related matters.

Risk-Informed and Performance-Based Regulation

The ACRS has been a strong advocate of the Agency's move toward establishing a risk-informed and performance-based regulatory system. On numerous occasions in the past, the ACRS ~~had~~ encouraged the use of risk information in the regulatory decisionmaking process and also provided comments and recommendations on consistent use of PRA, impact of PRA results and insights on the regulatory system, and coherence in the regulatory process. The ACRS has been playing a major role in assisting the staff and providing valuable advice to the Commission in developing a risk-informed and performance-based regulatory approach. The ACRS has made significant contributions in this area, including the following:

- Performed a participatory review and assisted the staff in the development of several regulatory guides, especially Regulatory Guide 1.174, "An approach for using probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," and an associated Standard Review Plan Chapter 19 (General Guidance). These documents provide the foundation for risk-informed regulatory philosophy that can better focus resources and can lead to a more coherent regulatory structure.

- Identified impediments to the increased use of risk-informed regulation

- Commented on the role of defense in depth in a risk-informed regulatory system

- Addressed the treatment of uncertainties versus point values in the risk-informed decisionmaking process

- Evaluated the importance measures that are being contemplated for risk-informing 10 CFR Part 50

- Commented on the use of defense in depth in risk-informing NMSS activities

- Commented on the NEI proposal for risk-informing 10CFR Part 50

- Addressed the industry and staff activities associated with the PRA quality

- Commented on proposed Options 1-3 pertaining to the development of risk-informed regulatory approach

- Reviewed the proposed risk-informed revisions to 10 CFR 50.44 regarding combustible gas control systems (Option 3)

The ACRS will continue to add value to the development of a risk-informed and performance-based regulatory structure. It will review and provide timely advice to the Commission on the activities associated with the risk-informed and performance-based regulatory system, including:

- Proposed NRC framework document for risk-informing 10 CFR Part 50 (Option 3)

- Proposed final risk-informed revisions to 10 CFR 50.44 regarding combustible gas control systems and 10 CFR 50.46 regarding emergency core cooling system requirements (Option 3).

- Proposed 10 CFR 50.69 and Appendix T (Option 2) associated with special treatment requirements.

- Proposed ASME, NFPA 805, and ANS Standards for PRA quality as well as the proposed industry PRA-certification process.

- Adequacy of implementation of regulatory guidance documents associated with risk-informed regulation, and the need for potential revisions to these documents.

- Risk-informed performance indicators.

<u>Rules and Regulatory Guidance</u>

10 CFR 2.809 states that when a rule involving nuclear safety matters within the purview of the ACRS is under development by the NRC staff,

the staff will ensure that the ACRS is given an opportunity to provide advice at appropriate stages and to identify issues to be considered during rulemaking hearings. A memorandum of understanding between the ACRS and the EDO delineates procedures for ACRS participation in the development of rules and regulatory guidance documents [e.g., Regulatory Guides, Standard Review Plans, and Regulatory Issue Summary Reports (Generic Letters).] The ACRS has made significant contributions in assisting the staff in formulating and/or revising numerous rules and regulatory guidance documents.

Since its inception, the ACRS has been reviewing all safety-significant rules and regulatory guidance documents that are within its purview (10 CFR Parts 20, 21, 26, 50, 51, 52, 54, 55, 70, 72, 73, 76, and 100), including the General Design Criteria. Recently, the ACRS played a major role in assisting the staff in the development of, or revisions to several important rules and regulatory guidance documents, including those listed below:

- Proposed amendment to 10 CFR 50.55a "Codes and Standards," regarding elimination of the requirement for updating ISI and IST programs every 120 months.

- Proposed Rule, Regulatory Guide, and Standard Review Plan Section associated with the use of Alternative Source Term at Operating Reactors.

- Proposed Revision to 10 CFR 50.59 (Changes, Tests and Experiments)

- Proposed Final Revision to 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.

- Proposed Rulemaking for Shutdown and Fuel Storage Pool Operations at Nuclear Power Plants.

- Proposed Revisions to 10 CFR Parts 50 and 100 and Proposed Regulatory Guide relating to Reactor Siting Criteria.

- Proposed Revision of Appendix K to 10 CFR Part 50.

- Regulatory Effectiveness of the Station Blackout Rule.

- Proposed Rule and Regulatory Guide for Fracture Toughness Requirements for LWR Pressure Vessels.

The ACRS will continue to review and comment on proposed rules and regulatory guidance documents as well as revisions to existing rules and guidance documents that are within its purview, including those associated with risk-informed and performance-based regulatory structure.

Safety Research Program

In a Staff Requirements Memorandum dated September 9, 1997, the Commission requested that the ACRS provide a report to the Commission annually on the NRC Safety Research Program, documenting its views on: the need, scope, and balance of the research program; whether the research programs provide the needed information to the research user offices; anticipation of research needs; prioritization and planning of research in the changing regulatory and technological environment.

- Since the Commission request in 1997, the ACRS provided three reports (NUREG-1635, Vols 1, 2, and 3, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," which included valuable advice to the Commission and the staff.

- NUREG-1635, Vol. 1 included ACRS comments and recommendations resulting from its comprehensive review of the NRC Safety Research Program. It included ACRS comments and recommendations on engineering the reactor safety research program as well as on several specific research activities, including those on: PRA, Human Factors, Fire Protection, Severe Accidents, Containment Integrity, Thermal Hydraulics, Advanced Instrumentation and Control, Reactor Fuels, Reactor Pressure vessel Integrity, and Plant Aging. RES was generally responsive to ACRS comments and recommendations.

- In NUREG-1635, Vol. 2, the ACRS provided additional comments and recommendations on significant research activities addressed in NUREG-1635, Vol. 1. It emphasized the need for PRA model development in the areas of severe accidents, human factors, fire protection, low-power and shutdown operations, and instrumentation and control systems. Also, the ACRS addressed the need for maintaining in-house capability for independent verification of regulatory criteria and resolution of complex technical issues associated with the integrity of reactor vessel and stream generator tubes. It reiterated the need for a strong research program to support the transition to risk-informed and performance-based regulatory structure. In general, RES agreed to consider ACRS comments and recommendations.

- In NUREG-1635, Vol. 3, the ACRS examined the internal and external contexts that together determine the needs for research and the corresponding responses of the agency. It discussed how the NRC research has evolved and how it may develop in the future. Also, it presented specific evaluations of research requirements in response to more significant future issues. The Committee's comments and recommendations were very well received by the Commission and RES.

- In the ensuing report, to be published as NUREG-1635, Vol. 4, the Committee plans to evaluate the ongoing and proposed major research activities and provide comments and recommendations on the continuing need for research in certain areas, whether certain research could be or should be done by the industry, and whether the

NRC could use information developed through cooperative international research activities instead of performing research in certain areas. Also, it will identify long-range NRC research needs.

The ACRS will continue to take an active role in reviewing the ongoing and proposed research activities and provide comments and recommendations to the Commission in its annual report. Also, it plans to look at the inhouse capability to perform research as well as provide views on RES ability to plan and carry out the long-range research program. The ACRS plans to interact with the Advisory Committees of other countries, as appropriate and as resources permit, to ensure that it remains well informed of the developments in international research and bring them to the attention of the Commission, as needed.

Generic Safety Issues

The ACRS has a long-standing interest in Generic Safety Issues (GSIs). For several years, the ACRS maintained a separate list of GSIs which were identified by the ACRS during its review of applications for construction permits and operating licenses and of significant operating events. Recognizing the additional burden on the staff in keeping track of GSIs identified by the ACRS and those identified by the staff, the ACRS and the NRC had agreed in early 1980s to combine the ACRS list of GSIs with the NRC staff's list.

The ACRS made significant contributions to the GSI process, including the following:

- Played a major role in assisting the staff in the development of the methodology for prioritizing GSIs, which is included in NUREG-0933, "A Prioritization of Generic Safety Issues."

- Reviewed the adequacy of the priority rankings (HIGH, MEDIUM, LOW, and DROP) for more than 800 GSIs. Where the ACRS had disagreed, the staff in most cases had reassessed the priority rankings and resolved the ACRS concerns.

- Reviewed the resolution of essentially all of the Unresolved Safety Issues (USIs) and most of the GSIs. ACRS concerns and disagreements on the adequacy of the resolution were resolved by the staff through additional and/or improved analyses which, in turn, resulted in a technically sound resolution.

- The ACRS expressed concern that several GSIs prioritized about 15 years ago still remain to be resolved. Subsequently, schedule for resolving these GSIs has been included in the Chairman's Tasking Memorandum, which is submitted to Congress every year. Since the concern expressed by the ACRS, some of these GSIs have been resolved.

The ACRS will continue to add value to the GSI process by reviewing the:

- Adequacy of the proposed priority rankings and resolution of GSIs.

- Effectiveness of using Management Directive 6.4 and associated Handbook to implement the revised GSI resolution process.

- Validity of the assumptions and analyses used in prioritizing and resolving GSIs.

- Operational events to determine whether they warrant reassessment of those GSIs previously assigned with a "Low" priority ranking and those classified as "RESOLVED."

- Adequacy of the resolution of certain GSIs by the licensees through the IPE and IPEEE programs.

- Adequacy of the resolution of the GSIs identified by MSRP.

- Effectiveness of the revised GSI process.

Reactor Operations

The ACRS has made significant contributions in this area by reviewing safety significant issues associated with operating plants and several other related matters, including the following:

- Revised Reactor Oversight Process (RROP), including initial implementation of RROP.

- Proposed improvements to the inspection and assessment programs, generic communication process, and revision to the enforcement policy.

- Differing Professional Opinion issues associated with steam generator tube integrity.

- Steam generator tube and reactor pressure vessel integrity and steam generator tube repair limits.

- Spent fuel pool accident risk.

- Insights gained from risk-informed pilot applications including those from pilots for ISI, extension of allowed outage times, and online maintenance.

- BWR strainer blockage.

- Yankee Rowe Reactor Pressure Vessel Integrity.

- Reliability of emergency AC power at nuclear power plants.

- IPE and IPEEE programs.

- Physical security requirements.

- Lessons learned from the investigation of significant operating events (e.g., steam generator tube rupture event and reactor trip and loss of offsite power event at Indian Point Unit 2, and reactor trip event at Hatch nuclear plant).

- Power uprates for Fermi, Hatch, and Monticello nuclear plants.

- Highlights of events that occurred at foreign nuclear plants during 1997 and 1998 and the associated safety significance.

- Loss of spent fuel cooling folowing a loss of coolant accident at the Susquehanna nuclear plant.

The ACRS will continue to make significant contributions to the safe operation of nuclear plants. It plans to review several matters in this area, including the following, and provide valuable and timely advice to the Commission:

- As requested by the Commission, the ACRS will review the use of performance indicators (Pis) in the RROP to ensure that Pis provide meaningful insight into aspects of plant operation that are important to safety and the initial implementation of the Significance Determination Processes (SDPs) and assess the technical adequacy of the SDP to contribute to the RROP.

- Significant operating events.

- Issues that led to the shutdown of plants for more than a year and the associated corrective action programs.

- PWR strainer blockage issues.

- Risk-based analysis of reactor operating experience and risk-based technical specifications.

- Reevaluation of the PTS screening criterion.

- Significant issues associated with power uprates and applications for power uprates (more than 5 percent).

- Impact of deregulation on operating plant safety.

- Synergisms among changes in nuclear plants (high burnup fuel, power uprates, plant life extension, and use of "best-estimate" or "more-realistic" analyses) and their potential impact on plant safety.

- Each year, the ACRS will visit one of the NRC Regional Offices and a plant in that Region and meet with the licensee and the Regional Staff to obtain information on significant issues being dealt with. Insights gained from this meeting will be used by the Committee in its review of significant regulatory issues or brought to the attention of the EDO or the Commission.

<u>Transient and Accident Analysis Codes</u>

Analytical computer codes have become the major tools used for calculation of reactor system behavior during transients and accidents. The current codes in use have an ancestry that dates back ~30 years, and a corresponding development process that had not been transparent. As such, the codes must be carefully compared to relevant experimental data and only used within their range of applicability of this data. Given the above, ACRS review of these codes is necessary because:

- The move to risk-informed regulation will result in use of more "realistic" or "best estimate" codes. Use of such Codes requires more quantitative evaluation of model uncertainties and development of acceptance criteria.

- Part of the regulatory process relies heavily on the results of calculations done for the NRC by the national laboratories or other contractors by using thermal-hydraulic codes (e.g., RELAP5, TRAC, and severe accident codes (e.g., SCDAP, MELCOR).

- Code documentation must be acceptable to knowledgeable, impartial observers. Review of codes to date indicates that documentation needs to be improved. Code quality must be adequate to support regulatory decisions and increase public confidence.

The ACRS has significant interest in the transient and accident analysis codes. The ACRS has made major contributions in this area by providing formal and informal comments and recommendations.

- Reviewed several thermal-hydraulic codes and severe accident codes used by the NRC and/or its contractors such as RELAP5, TRAC, SCDAP, MELCOR, and industry codes such as RETRAN-3D, WGOTHIC, and NOTRUMP.

- Identified shortcomings associated with several of these codes.

- Pointed out the inadequate and incomplete code documentation.

- Based on its limited review of the EPRI RETRAN-3D code, the ACRS has identified problems with the momentum equation and inapplicability of several of the correlations.

- Questioned whether the code calculations are sufficiently independent of the noding for full-scale application.

- Recommended that the NRC staff should independently verify the validity of industry codes.

- Urged the staff to develop documents to guide the content of code submittals as well as to establish procedures for use by the staff in reviewing the industry codes.

The ACRS is in the process of reviewing several codes, including Siemens S-RELAP5, GE Nuclear Energy TRACG, and the EPRI RETRAN-3D, as well as the Standard Review Plan Section and Regulatory Guide that will guide content of code submittals and include procedures for use by the staff in reviewing industry codes.

Other Regulatory Activities

The ACRS will continue to or plans to review several other regulatory activities, including: safety issues associated with the extended burnup of reactor fuels; use of Phenomena Identification and Ranking Table (PIRT) for high burnup fuel; safety issues associated with the use of mixed oxide (MOX) fuel in commercial light water reactors; MOX fuel fabrication facility, and acceptance criteria for high burnup fuels. The ACRS encouraged the NRC participation in the experimental studies being performed at CABRI reactor in France and plans to follow-up on those performed at the NSRR reactor in Japan. In addition, the ACRS will review reactor pressure vessel embrittlement issues, control room habitability, decommissioning activities, fire-protection issues including NFPA-805 Standard, human factors, application for uranium enrichment facilities, safeguards, and transportation of radioactive materials.

Special Project

The ACRS has performed an independent review of each major nuclear propulsion plant (NPP) design proposed by the Naval Reactors (NR) organization of DOD/DOE. This review was initially requested by the NR organization and subsequently required by a Presidential Directive issued under Section 91b of the Atomic Energy Act. The ACRS performed a review of the adequacy of the SEAWOLF submarine design in 1994. The ACRS will review the safety aspects of the proposed VIRGINIA class submarine during 2002. The ACRS has added significant value as noted below.

- ACRS reviewed the safety aspects of the proposed NPP designs and provided independent views on the adequacy of these designs. Since these designs are classified, they are not subject to public scrutiny. Independent evaluation of the adequacy of these designs by the ACRS provided credibility as well as aided the NR organization to justify the technical adequacy of these designs in front of the Congress.

- The ACRS comments and recommendations on the NR Training Program were very helpful to the NR organization to enhance the effectiveness of this program.

- During its review of te Moored Training Ship Demonstration Project in 1987, the ACRS recommended that NR organization apply PRA methodology and severe accident analysis to the NPP design. Subsequently, NR organization initiated the practice of performing a PRA, including severe accident analysis, for all succeeding NPP designs.

The Committee will continue to review the safety aspects of the proposed NPP designs and provide independent views on the technical adequacy of these designs. In addition, it will review the Westinghouse AP1000 Standard Plant Design, Differing Professional Opinion on Steam Generator Tube Integrity, and Spent Fuel Fire Risk.

MEASURES OF SUCCESS

An assessment of the extent to which the goals and objectives of this Plan has been achieved, including the effectiveness, efficiency, quality, timeliness, and rate of success in contributing to the regulatory process will be addressed in the annual ACRS Operating Plan.

ACRS ACTION PLAN UPDATE

The ACRS will update this plan periodically, as necessary. Revisions to the plan will be based on ACRS recognition of the need to update the plan, input from the Commission, changes to the Strategic Plan, changes in direction of NRC programs, results from stakeholder surveys and self-assessments, external events and factors, and available resources.