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

July 7, 2000

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

SUBJECT: SUMMARY REPORT - 473rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, ON JUNE 7-9, 2000, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Meserve:

During its 473rd meeting June 7-9, 2000, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report and letters. In addition, the Committee authorized Dr. Larkins, Executive Director of the ACRS, to transmit the memoranda noted below.

REPORT

- Proposed Resolution of Generic Safety Issue 173A, "Spent Fuel Storage Pool for Operating Facilities" (Report to Richard A. Meserve, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, dated June 20, 2000)

LETTERS

- Proposed Final Regulatory Guide and Standard Review Plan Section Associated with the Alternative Source Term Rule (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated June 20, 2000)

- Draft Report, "Regulatory Effectiveness of the Station Blackout Rule" (Letter to William D. Travers, Executive Director for Operations, NRC, from Dana A. Powers, Chairman, ACRS, dated June 22, 2000)

MEMORANDA

- Industry Initiatives in the Regulatory Process (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated June 12, 2000)

- AP1000 Pre-Application Review (Memorandum to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS, dated June 21, 2000)

HIGHLIGHTS OF KEY ISSUES CONSIDERED BY THE COMMITTEE

1. Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning Generic Safety Issue (GSI) 173A. The NRC staff developed a generic action plan for ensuring the safety of spent fuel storage pools in response to two postulated event sequences at two separate plants (Susquehanna and Dresden 1). The principal concerns addressed by the action plan involve the potential for a sustained loss of spent fuel pool cooling and the potential for a substantial loss of spent fuel pool coolant inventory. The action plan also included plant-specific evaluations and regulatory analyses for safety enhancement backfits for plants that are more vulnerable to the GSI-173A concerns.

Conclusion

The Committee issued a report on this matter to Chairman Meserve, dated June 20, 2000.

2. Regulatory Effectiveness of the Station Blackout Rule

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the draft report concerning the regulatory effectiveness of the station blackout (SBO) rule.

The NRC staff presented background information on the SBO rule. The staff discussed the draft report on the "Regulatory Effectiveness of the Station Blackout Rule," which has been issued for public comment. This report reflects the comments made by NRR and the regions. The draft report was provided to the Nuclear Energy Institute (NEI) for comment. The staff stated that the lesson learned from the evaluation of the SBO rule is that regulatory documents have to be reviewed more carefully for consistent interpretation of terms, goals, criteria, and measurements. The draft report concluded that the SBO rule was effective and that industry and NRC resources spent to implement the SBO rule were reasonable.

Conclusion

The Committee issued a letter on this matter to Dr. William D. Travers, Executive Director for Operations, on June 22, 2000.

3. Proposed Final Standard Review Plan Section and Regulatory Guide Associated With the Revised Source Term Rule

The Committee heard a presentation by and held discussions with representatives of the NRC staff regarding the proposed final version of Regulatory Guide 1.XXX (DG-1081), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," and Standard Review Plan, Section 15.0.1, "Radiological Consequences Analysis Using Alternative Source Terms." NRC released these documents for public comment in December 1999. More than 130 comments were received. In response to the public comments, the staff revised DG-1081 relative to fuel gap fraction assumptions, the chemical form for fuel-handling accident calculations, selective implementation processes, and guidance for use of 10 CFR 50.59, along with other technical modifications. Changes were also made to the final version of the regulatory guide in response to the Committee's formal comments on the draft document.

Conclusion

The Committee issued a report on this matter to the Executive Director for Operations dated June 20, 2000.

4. Assessment of the Quality of Probabilistic Risk Assessments

The Committee heard presentations by and held discussions with representatives of the NRC staff and NEI concerning the staff's plans for addressing the issue of probabilistic risk assessment (PRA) quality until the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) standards have been completed. The Committee discussed the schedule for and the status of the ASME and ANS standards development efforts, as well as staff plans to review the subject documents when they are available. The staff highlighted plans to develop guidance on what the staff considers to be technical elements and characteristics of a robust PRA. The staff also discussed plans to clarify the level of analysis and/or performance that might be expected for various applications and the role of the expert panel in risk-informed decisionmaking. The Committee considered NEI's plans to develop an industry-wide PRA certification process guideline to integrate the initiatives of the various

Nuclear Steam Supply System Owners Groups.

Conclusion

The Committee decided to continue its review of this matter during the July 2000 ACRS meeting when the staff's draft Commission paper and the proposed NEI guideline are expected to be available.

5. Performance-Based Regulatory Initiatives

The Committee heard presentations by and held discussions with representatives of the NRC staff concerning the proposed high-level guidelines for performance-based activities. The Committee considered the staff's actions to solicit internal and external stakeholder input in response to the Staff Requirements Memorandum dated September 13, 1999. The Committee and the staff discussed the relationship between the proposed guidelines and the Commission's White Paper on risk-informed, performance-based regulation (SECY-98-144) and the PRA Policy Statement. The Committee considered the revised guidelines issued for public comment on May 9, 2000, including the questions and proposed reconciliation of past comments. The Committee and the staff discussed the possible use of these guidelines in revising regulations to make them performance-based and in considering possible performance-based approaches to meet existing regulatory requirements.

The Committee also met with a representative of Public Citizen, Critical Mass Energy Project, to discuss this matter. The concerns expressed by the Public Citizen representatives were that the staff dismissed public comments without proper consideration, the proposed guidelines would erode regulatory conservatism and plant safety margins, the guidelines focus on economic efficiency rather than safety performance, and implementation of these guidelines would weaken public confidence. The Committee will consider these concerns.

Conclusion

The Committee decided to continue its review of this matter during the July 2000 ACRS meeting when the staff's draft Commission paper is expected to be available.

6. Use of Industry Initiatives in the Regulatory Process

The Committee heard presentations by and held discussions with representatives of the NRC staff and the NEI regarding a proposed Commission paper concerning proposed guidelines for using industry initiatives in the regulatory process. The ACRS members, the staff, and NEI representatives discussed whether the guidelines would impose additional burden on the industry, the steps in the proposed process, and the enforcement of plant-specific commitments associated with industry initiatives.

Conclusion

The Committee authorized the ACRS Executive Director to provide a memorandum to the NRC Executive Director for Operations dated June 12, 2000, stating that the ACRS had no objection to issuing these guidelines for public comment and requested the opportunity to review the proposed final guidelines after resolution of public comments.

7. Safety Culture at Operating Nuclear Power Plants

The Committee heard a presentation by and held discussions with Mr. John Sorensen, an ACRS Fellow, regarding a paper he was preparing concerning safety culture at operating nuclear power plants. He explained what safety culture is, why it is important, and what the NRC staff could do to support it. Mr. Sorensen recommended that the staff identify the essential attributes of safety culture and the associated performance indicators, and ensure that licensees maintain effective root cause analysis processes.

The ACRS members and Mr. Sorensen discussed the use of alternative phrases, such as general culture and quality culture, the correlation between safety culture and potential performance indicators, how to teach and inspect the elements of safety culture, and the importance of a risk-informed, performance-based regulatory system.

Conclusion

The Committee decided to consider preparation of a report to the Commission concerning this matter during the ACRS meeting on July 12-14, 2000.

8. Visit to Davis-Besse Nuclear Power Plant and Meeting With NRC Region III Plant Personnel

The Committee heard a presentation by and held discussions with Mr. Amarjit Singh, an ACRS Senior Staff Engineer, regarding the proposed schedule for touring the Davis-Besse Nuclear Power Plant, the specific areas to be visited, and the proposed topics of discussion with representatives of the licensee and the NRC Region III office. Mr. Singh presented the background information, current operating issues, and the results of the maintenance rule inspection at Davis-Besse. The topics of discussion with Region III representatives included regional organization, reactor program implementation, the new oversight inspection program, the senior reactor analyst program, and other fire protection related issues.

Conclusion

This briefing was held for information only. No Committee action is required.

9. Proposed Plan and Assignments for Reviewing License Renewal Guidance Documents

The Committee heard a presentation by Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, concerning plans for the ACRS to review license renewal guidance documents and ACRS member assignments for reviewing these documents. Dr. Bonaca outlined the scoping and screening processes used for identifying systems and components subject to aging management programs. He explained how the proposed generic aging lessons learned report would be used as a reference for aging management programs acceptable to the staff. The Committee discussed and revised proposed items that would be used to focus its review.

Conclusion

The Committee decided to hear a briefing from the staff at the ACRS meeting on August 30-September 1, 2000, concerning an overview of the proposed license renewal guidance documents.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee discussed the response from the NRC Executive Director for Operations (EDO) dated May 26, 2000, to the ACRS comments and recommendations included in the ACRS report dated April 13, 2000, concerning

the draft final technical study of spent fuel pool accident risk at decommissioning nuclear power plants.

As a result of the ACRS comments and recommendations, the staff has defined additional technical work that needed to be performed and included in the final report. The Committee plans to review and comment on the final report.

- The Committee discussed the response from the EDO, dated May 26, 2000, to ACRS comments and recommendations included in the ACRS report dated April 13, 2000, concerning the NRC program for risk-based analysis of reactor operating experience.

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

- The Committee discussed the response from the EDO dated May 12, 2000, to ACRS comments and recommendations included in its letter dated April 17, 2000, concerning the Commission's Reactor Safety Goal Policy Statement

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

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from May 11 through June 6, 2000, the following Subcommittee meetings were held:

- Planning and Procedures - June 6, 2000

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

LIST OF FOLLOW-UP MATTERS FOR THE EXECUTIVE DIRECTOR FOR OPERATIONS

- The Committee decided to continue its review of the staff's plan for addressing the issue of PRA quality during the July 2000 ACRS meeting when the staff's draft Commission paper and the proposed NEI guideline are expected to be available.

- The Committee decided to continue its review of the staff's proposed high-level guidelines for performance-based activities during the July 2000 ACRS meeting when the staff's draft Commission paper is expected to be available.

- The ACRS/ACNW Joint Subcommittee plans to continue its review of risk-informed regulatory initiatives in NMSS. The Joint Subcommittee decided to review the technical merits of Integrated Safety Assessments (ISAs) during a future meeting.

- The Committee decided to consider preparation of a report to the Commission concerning safety culture in nuclear power plants during the July 12-14, 2000 ACRS meeting.

- The Committee decided to hear a briefing from the staff at the August 30 - September 2000 ACRS meeting concerning an overview of the proposed license renewal guidance documents.

- The Committee had no objection to issuing the NRC staff's guidelines for using industry initiatives in the regulatory process for public comment and would like the opportunity to review the proposed final guidelines after resolution of public comments.

PROPOSED SCHEDULE FOR THE 474th ACRS MEETING

The Committee agreed to consider the following topics during the 474th ACRS Meeting, July 12-14, 2000:

Activities Associated with Risk-Informing 10 CFR Part 50
Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding: (a) proposed revision to 10 CFR 50.44 concerning combustible gas control systems and advance notice of proposed rulemaking (10 CFR 50.69 and Appendix T); and (b) NEI letter dated January 19, 2000.

Assessment of the Quality of the Probabilistic Risk Assessments
Briefing by and discussions with representatives of the NRC staff regarding a draft Commission paper on the assessment of the quality of probabilistic risk assessments (PRAs).

Proposed Final ASME Standard for PRA Quality
Briefing by and discussions with representatives of the American Society of Mechanical Engineers (ASME) regarding the proposed final ASME Standard for PRA quality.

Annual Report to the Commission on the NRC Safety Research Program
Discussion of the format and content of the annual ACRS report to the Commission on the NRC Safety Research Program.

Topics for Meeting with the NRC Commissioners on October 5, 2000
Discussion of topics for meeting with the Commissioners scheduled for October 5, 2000.

Sincerely,

Dana A. Powers
Chairman

Date Issued: 7/26/00
Date Certified: 8/7/00

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APPENDICES

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473rdACRS Meeting
June 7-9, 2000

MINUTES OF THE 473RD MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS JUNE 7-9, 2000 ROCKVILLE, MARYLAND

The 473rd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on June 7-9, 2000. Notice of this meeting was published in the *Federal Register* on May 24, 2000 (65 FR 33584) (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 Gelman Building, 2120 L Street, N.W., Washington, D.C. [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), Mr. John Barton, Dr. Mario V. Bonaca, Dr. Thomas S. Kress, Dr. William J. Shack, Dr. Robert L. Seale, Mr. John D. Sieber, Dr. Robert E. Uhrig, and Dr. Graham B. Wallis. 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. Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities" (Open)

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

Dr. Thomas S. Kress stated that in 1996, the NRC staff developed and implemented a generic action plan for ensuring the safety of spent fuel storage pools (SFPs) in response to two postulated event sequences involving the SFP at two separate plants (Susquehanna and Dresden 1). Generic Safety Issue (GSI) 173 Part A was initiated for operating facilities. The principal concerns addressed by the action plan (and GSI-173A) involve the potential for a sustained loss of SFP cooling and the potential for a sustained loss of spent fuel coolant inventory that could expose irradiated fuel.

On August 9, 1996, the staff briefed the ACRS regarding the action plan and described three courses of action:

- Plant-specific evaluations or regulatory analyses for safety enhancement backfits.
- Rulemaking implementation of the shutdown rule for SFP operations.
- Revision of staff guidance (Standard Review Plan (SRP) Section 9.1.3 and Regulatory Guide (RG) 1.13)

On October 16, 1998, the ACRS issued its letter endorsing the staff's "HIGH" priority ranking of GSI-173A. Currently, the staff is still in the process of revising its guidance and is working with the industry American Nuclear Society ([ANS] Subcommittee) to revise American National Standards Institute/American Nuclear Society [ANSI/ANS]-57.2 standard.

Mr. Christopher Gratton, NRR, briefed the Committee regarding the status of GSI-173A. He stated that the objective of the staff's presentation is to obtain an agreement from the ACRS that GSI-173A should be closed.

The principal concerns included in GSI-173A involved the potential for a sustained loss of SFP cooling capability, which was identified through the report filed with the NRC relating to Susquehanna, and the potential for a substantial loss of SFP cooling inventory, which was given renewed emphasis following the Dresden 1 special inspection. Postulated adverse conditions that may develop

following a loss-of-coolant accident or a sustained loss of power to SFP cooling system components could prevent restoration of SFP decay heat removal. The heat and water vapor added to the building atmosphere by subsequent SFP boiling could cause failure of accident mitigation or other safety equipment and an associated increase in the consequences of the initiating event. Incomplete administrative controls combined with certain design features, particularly at the oldest facilities, may create the potential for a substantial loss of SFP coolant inventory.

Currently, the focus of the staff's work is on plant-specific evaluations or regulatory analyses for safety enhancement backfits. The staff has identified specific operating reactors in each of the following categories for further evaluation:

- Absence of passive antisiphone devices on piping extending below the top of stored fuel
- Transfer tube(s) within the SFP rather than a separate transfer canal
- Piping entering pool below the top of stored fuel
- Limited instrumentation for loss-of-coolant events
- Absence of leak detection capability or absence of isolation valves in leakage detection system piping
- Shared systems and structures at multi-unit sites
- Absence of onsite power supply for systems capable of SFP cooling
- Limited SFP decay heat removal capability
- Infrequently used backup SFP cooling systems
- Limited instrumentation for loss-of-cooling events
- Refueling cavity seals with pneumatic components

For regulatory analyses, the staff used screening criteria for the frequency of uncovery to within 1 foot of the top of the fuel or loss of cooling for 8 hours. The screening criteria were $\leq 10^{-6}$/yr (no action justified); 10^{-6}/yr to 10^{-5}/yr (further evaluation needed); and $\geq 10^{-5}$/yr (proceed to value-impact evaluation). During the staff's review, 12 licensees proposed certain voluntary actions. In general, the staff concluded that existing facilities are in compliance with the regulations, and the plant-specific evaluations identified no additional regulatory actions. The staff's actions for rulemaking and revising the SFP guidance are still being developed.

Conclusion

The Committee issued a report to Chairman Meserve on this matter dated June 20, 2000.

III. Regulatory Effectiveness of the Station Blackout Rule

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

Dr. Mario V. Bonaca, Acting Chairman of the Subcommittee on Regulatory Policies and Practices, introduced this topic to the Committee. He stated that the purpose of this session was to discuss with the representatives of the NRC staff the draft report concerning the regulatory effectiveness of the station blackout (SBO) rule.

NRC Staff Presentation

Mr. Jack Rosenthal led the discussions for the staff. He stated that this draft report is related to the agency's goals; this initiative is to examine the effectiveness of major rules, which are clearly related to maintain safety.

Mr. William S. Raughley presented background information regarding the SBO rule. He stated that on June 21, 1988, NRC published the SBO rule to provide further assurance that a loss of both offsite and onsite emergency ac power systems would not adversely affect public health and safety. In May 1997, the Commission directed the staff to use individual plant examination (IPE) results to assess regulatory effectiveness in resolving major safety issues. The staff issued a draft report, which included a set of baseline expectations that were established from the SBO rule and related regulatory documents in the areas of coping capability, risk reduction, emergency diesel generator reliability, and value impacts. The draft report was provided to the Nuclear Energy Institute (NEI) for comments.

The draft report concluded that the SBO rule was effective and that industry and NRC resources spent to implement the SBO rule were reasonable.

Conclusion

The Committee issued a letter to Dr. William D. Travers, Executive Director for Operations, on June 23, 2000.

IV. Proposed Final Standard Review Plan Section and Regulatory Guide Associated With the Revised Source Term Rule

Dr. Kress, cognizant ACRS member for this issue, introduced this topic to the Committee. He presented a brief history on the development of the alternative source term rule and the associated regulatory guide and SRP section. He said that the Committee reviewed the final version of the rule along with the draft versions of the above documents and issued a letter report on this matter in September 1999.

NRC Staff Presentation

Mr. S. LaVie, NRR, discussed the proposed final regulatory guide (DG-1081) and the SRP section (15.0.1) supporting implementation of the alternative source term rule. He cited the background of this issue, noting that the alternative source term rule (10 CFR 50.67) became effective on January 24, 2000. Public comments on the proposed final versions of these documents were extensive: more than 130 comments were received from six entities and individuals.

In response to these comments, the staff modified the regulatory guide in several areas: fuel gap fractions assumed in analyses, selective implementation pursuant to the use of 10 CFR 50.59, and technical changes in the areas of environmental qualification, steam generator iodine transport, and the containment spray decontamination factor, among others. In addition, the staff modified the regulatory guide to address two recommendations made by the ACRS in its September 1999 report. Dr. Kress said that the staff responses to the Committee's recommendations were appropriate.

Drs. Kress and Powers expressed some concern that the staff had retained items in the regulatory guide that appear to be inappropriate for use with the alternative source term. Dr. Kress said that he had compiled a list of these items along with some minor editorial concerns from his reading of the regulatory guide that the staff should address. He said that he would provide this list to the staff for its consideration.

Conclusion

The Committee issued a report to the Executive Director for Operations on this matter dated June 20, 2000.

V. Assessment of the Quality of Probabilistic Risk Assessments

[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 [PRA], introduced this topic to the Committee. He stated that the purpose of this meeting was to discuss the staff's proposed recommendations to the Commission for addressing the issue of PRA quality until the American Society of Mechanical Engineers (ASME) and the ANS standards have been completed. He noted that the staff's response is due to the Commission by June 30, 2000. He also noted that the staff's proposed Commission paper on this matter is not yet available for ACRS review and suggested that it may be difficult for the Committee to prepare a report during this meeting.

NRC Staff Presentation

Messrs. Thomas King, RES, and Richard Barrett, NRR, led the discussions for the NRC staff. Mr. Mark Cunningham and Ms. Mary Drouin, RES, and Mr. Gareth Parry, NRR, provided supporting discussion. The staff discussed the schedule and status of the ASME and ANS standards development efforts, as well as staff plans to review the subject documents, when available. Significant points made during the presentation include the following:

- Issues associated with the technical acceptability of PRAs include the following:
  - PRA scope and level of analysis required
  - PRA elements and characteristics
  - Peer review
  - PRA application process
  - Expert panels
- Current risk-informed activities in the reactor arena include the following:
  - Proposed revision to 10 CFR Part 50
  - Revised reactor oversight process
  - Operating event assessment review
  - Risk-informed license amendments

- In addressing the issue of PRA quality, the staff will consider proposed licensee approaches for limiting risk, including performance monitoring to complement the level of completeness of PRAs, the robustness of modeling, and quantitative and qualitative risk analysis tools. The staff will also consider industry peer review and certification programs to the extent that they confirm the technical adequacy of PRAs and support integrated decisionmaking.

- The staff plans to develop guidance on what the staff considers to be the technical elements and characteristics of a robust PRA. The staff also plans to clarify the level of analysis and/or performance that might be expected for various applications and the role of expert panels in risk-informed decisionmaking.

- The staff plans to revise RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to endorse the proposed ASME and ANS standards and NEI certification process, as appropriate.

Industry Presentation

Mr. Biff Bradley of NEI made an overview presentation to the Committee. He expressed agreement with the staff's presentation and noted that licensee IPEs have become obsolete. He stated that NEI was working with the Nuclear Steam Supply System (NSSS) Owners Groups to develop a Peer Review Process Guideline (NEI 00-02) to address the quality of PRAs. Mr. Bradley stated that all U.S. nuclear plants are scheduled to undergo the peer review process by the end of Calendar Year 2001. Drs. Apostolakis and Kress requested copies of NEI 00-02 and associated NSSS Owners Group certification processes. The staff informed the Committee that it was currently reviewing the subject documents and agreed to provide copies after the meeting.

Drs. Apostolakis and Wallis suggested that the staff attempt to define what constitutes a "good enough" PRA for making robust decisions. Dr. Kress stated that it could be like "proving a negative" and reiterated the need for a PRA that could provide confidence in decisionmaking. He suggested that there are ways to compensate for varying degrees of PRA quality (e.g., performance monitoring, defense in depth, etc.). The staff stated that it is difficult to identify all the requirements for a quality PRA and expressed preference for clarifying the key elements and characteristics, limitations of PRA, and expectations for analysis and/or performance monitoring.

Dr. Kress questioned whether PRAs would be expected to have robust uncertainty analysis. The staff stated that the NRC plans to identify its expectations of what a PRA should be and suggested that it would be more appropriate for the NEI Peer Review Process to determine the level of uncertainty analysis needed for particular decisions. The staff also stated that expert panels will be a critical part of plant-specific decisionmaking.

Conclusion

The Committee decided to continue its review of this matter during the July 2000 ACRS meeting when the staff's draft Commission paper and the proposed NEI guideline are expected to be available.

VI. Performance-Based Regulatory Initiatives

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

Mr. Jack Sieber, Vice Chairman of the ACRS Subcommittee on Plant Operations, introduced this topic to the Committee. He stated that the purpose of this meeting was to review the proposed high-level guidelines for performance-based activities. He stated that the Committee would consider the staff's actions in response to internal and external stakeholder input and the staff requirements memorandum (SRM) dated September 13, 1999. He noted that the staff's draft Commission paper was not yet available for review by the Committee. He also informed the Committee that a representative of Public Citizen, Critical Mass Energy Project, had requested time to discuss this matter.

NRC Staff Presentation

Mr. Prasad Kadambi, RES, led the discussions for the NRC staff. Messrs. Jack Rosenthal and Farouk Eltawila, RES, provided supporting discussion. Mr. Joseph Birmingham, NRR, also participated. Significant points raised during the staff presentation include the following:

- In the SRM dated September 13, 1999, the Commission supported the ongoing performance-based activities described in SECY-99-176 but disapproved the staff's overall plan because it lacked specificity with respect to detailed activities and schedules. The Commission directed the staff to develop high-level guidelines to identify and assess the viability of candidate performance-based activities.

- The general goal of this effort is to make the regulations and the regulatory process more efficient and effective. The staff noted that this goal was also consistent with the NRC Strategic Plan, the PRA Policy Statement, Regulatory Guide 1.174, and so on.

- The staff issued the proposed guidelines for public comment and held workshops to validate the integration of stakeholder feedback. Overall stakeholder input was generally favorable. However, some public comments expressed concern regarding implementation. The staff plans to pilot test the proposed guidelines over a range of regulatory applications to validate their suitability and to identify candidate performance-based regulatory initiatives.

- Risk information may provide the basis for pursuing selected performance-based initiatives, for example, safety enhancements, reduction in unnecessary regulatory burden, and use of risk metrics for evaluating performance and/or thresholds for regulatory action.

Public Citizen Presentation

Ms. Lisa Gue of Public Citizen made a brief presentation to the Committee. Significant points include the following:

- Public Citizen is concerned that the public comments were dismissed without proper consideration, that the proposed guidelines would erode regulatory conservatism and plant safety margins, that the guidelines focus on economic efficiency rather than safety performance, and that implementation of the guidelines would weaken public confidence.

- Public Citizen believes it is irresponsible to base nuclear safety standards on a probabilistic analysis of risk. Low probability does not justify the devastating consequences of nuclear accidents. The nuclear industry is using PRA to manage costs rather than safety.

- Because of the uncertainties in the entire range of probable and improbable events, particularly in the area of waste management, a performance-based regulatory structure can never be truly risk informed. A performance-based system is deficient because it relies on the lack of adverse data to justify a perception of acceptable performance. It

provides a false sense of comfort and does not preclude safety-significant events from occurring.

Dr. Powers questioned whether the NRC has identified metrics for efficiency and effectiveness. The staff stated that the proposed guidelines are intended to be established at a high level and suggested that lower tier implementing procedures could be developed to provide measures of efficiency and effectiveness. The staff noted that an objective of the performance-based approach is to reduce the prescriptive nature of regulatory requirements in order to provide flexibility for licensees in the conduct of operations.

Dr. Apostolakis questioned why monitoring is required if it can be demonstrated that a licensee meets the regulations. The staff stated that the NRC cannot assume that licensees meet the regulatory requirements and license conditions. The NRC will still continue to perform operational safety verification through its inspection, assessment, and enforcement programs.

Dr. Seale questioned the extent to which internal NRC stakeholder feedback was solicited on the proposed guidelines. In particular, he questioned the involvement of and feedback from regional and resident inspectors. The staff stated that most of the internal feedback was from persons working in the licensing and rulemaking areas. Dr. Seale noted that inspectors often provide the most valuable contributions and suggested that more be done to solicit their feedback.

Dr. Apostolakis questioned how the staff plans to consider uncertainties and defense in depth. The staff stated that it plans to develop examples or case studies to illustrate the possible use of these guidelines in revising regulations to be performance based and in considering possible performance-based approaches to meet existing regulatory requirements.

At the conclusion of the meeting, Dr. Powers expressed the view that the proposed guidelines presented no new information on the subject of performance-based regulation. Dr. Shack and Mr. Sieber noted that the questions used to solicit feedback in selected areas were valuable and suggested that they be considered more prominently in the proposed final version of the guidelines. Dr. Shack stated that Public Citizen does not believe that regulatory burden can be reduced without adversely affecting safety. Mr. Sieber noted that the staff is continuing to solicit feedback on the proposed guidelines by way of an on-line Internet workshop. He also noted that it is not yet apparent what the staff's recommendations to the Commission will be;

therefore, he suggested that the Committee defer preparing a report or a letter on this matter until the staff's draft Commission paper is available.

Conclusion

The Committee decided to continue its review of this matter during the July 2000 ACRS meeting when the staff's draft Commission paper is expected to be available.

VII. Use of Industry Initiatives in the Regulatory Process

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

Mr. John Barton, Acting Chairman of the Regulatory Policy and Practices Subcommittee, stated that the guidelines proposed by the staff for using industry initiatives in the regulatory process contain substantial detail and could preclude, in some cases, the need for regulatory action.

NRC Staff Presentation

Mr. Richard Wessman, NRR, presented background information related to the development of the proposed Commission paper concerning the guidelines. Mr. C. E. Carpenter, NRR, explained that the purpose of the guidelines is to ensure that future initiatives proposed by applicable industry groups would be treated and evaluated in a consistent, controlled, and open manner. He explained the industry initiatives process and presented recommendations and future actions. Mr. Carpenter concluded that the guidelines provided flexibility to licensees while making optimal use of existing regulatory processes. He stressed that the guidelines provide for public participation in the process and for making related information available to all stakeholders.

The ACRS members and the staff discussed whether the guidelines would impose additional regulatory burden on licensees, details of the industry initiatives process, and funding for the staff's review of the industry initiatives process.

Nuclear Energy Institute Presentation

Mr. Alex Marion, NEI, stated that NEI had been involved in the stakeholder meetings and had submitted two sets of written comments on the process. He explained that NEI has a broad spectrum of concerns, including the following:

- using industry initiatives as a substitute or an alternative for regulatory action,
- early determination of whether the issue is technical or regulatory,
- holding early and frequent communications with the industry in public meetings, and
- pursuing complementary sets of activities between the staff and the industry.

Mr. Marion stated that NEI would like the staff to respond to its comments. He explained NEI's position that if a proposed industry initiative would be subject to NRC inspection and enforcement, then the NRC must pursue regulatory action.

The ACRS members, Mr. Marion, and the staff discussed the flow chart used to depict the industry initiatives and the enforcement of commitments associated with industry initiatives.

Conclusion

The Committee had no objection to the staff's issuing these guidelines for public comment and would like the opportunity to review the proposed final guidelines after resolution of public comments.

VIII. Safety Culture at Operating Nuclear Power Plants

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

Dr. Dana Powers, Chairman of the ACRS, introduced Mr. John Sorensen, ACRS Fellow, and noted that the ACRS had been discussing the issue of safety culture at operating nuclear power plants for more than 3 years. Dr. Powers stated that safety culture has been considered an important aspect of safety but has never been quantified. He explained that Mr. Sorensen would provide a clear picture of what is meant by "safety culture."

Mr. Sorensen stated that he had written a white paper concerning safety culture at operating nuclear power plants. He explained what safety culture is, why it is important, and what the NRC staff could do. He recommended that the staff

identify the essential attributes of safety culture and associated performance indicators and ensure that licensees maintain effective root cause analysis processes.

The ACRS members and Mr. Sorensen discussed the use of alternative phases, such as general culture and quality culture, the correlation between safety culture and potential performance indicators, how to teach and inspect the elements of safety culture, and the importance of a risk-informed, performance-based regulatory system.

Conclusion

The Committee decided to consider preparation of a report to the Commission concerning safety culture in nuclear power plants during the ACRS meeting on July 12-14, 2000.

IX. Proposed Plan and Assignments for Reviewing License Renewal Guidance Documents

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

The Committee heard a presentation by Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, concerning plans for the ACRS to review license renewal guidance documents and ACRS ember assignments for reviewing these documents. Dr. Bonaca outlined the scoping and screening processes used for identifying systems and components subject to aging management programs. He explained how the proposed Generic Aging Lessons Learned report would be used as a reference for aging management programs acceptable to the staff. The Committee discussed and revised proposed items that would be used to focus its review.

Conclusion

The Committee decided to hear a briefing by the staff at the ACRS meeting on August 30-September 1, 2000, concerning an overview of the proposed license renewal guidance documents.

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 May 26, 2000, to the ACRS comments and recommendations included in the ACRS report dated April 13, 2000, concerning the draft final technical study of spent fuel pool accident risk at decommissioning nuclear power plants.

  As a result of the ACRS comments and recommendations, the staff has defined additional technical work that needed to be performed and included in the final report. The Committee plans to review and comment on the final report.

- The Committee discussed the response from the EDO, dated May 26, 2000, to ACRS comments and recommendations included in the ACRS report dated April 13, 2000, concerning the NRC program for risk-based analysis of reactor operating experience.

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

- The Committee discussed the response from the EDO dated May 12, 2000, to ACRS comments and recommendations included in its letter dated April 17, 2000, concerning the Commission's Reactor Safety Goal Policy Statement.

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

B. Report on the Meeting of the Planning and Procedures Subcommittee (Open)

The Committee heard a report from Dr. Powers and the Executive Director, ACRS, on the Planning and Procedures Subcommittee meeting held on June 6, 2000. The following items were discussed:

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

Member assignments and priorities for ACRS reports and letters for the June ACRS meeting were included in a separate handout. Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

- Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through September 2000 were 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

The Subcommittee discussed and developed recommendations on the items that require Committee decision.

- Meeting with Industry Representatives

During the January 2000 retreat, the ACRS discussed ways in which the Committee could interact with industry, including NEI, INPO, and utilities, to obtain information on significant industry issues. Dr. Apostolakis and Dr. Savio were tasked with making arrangements with NEI for a discussion between ACRS and NEI at a future ACRS meeting on NEI regulatory initiatives. The discussion has been tentatively scheduled for the October 2000 ACRS meeting.

- Meeting With the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners between on Thursday, October 5, 2000 in the Commissioners' Conference Room, One White Flint North. A list of topics proposed by the ACRS staff is as follows:

- Risk-Informed Regulation
  Option 2 and Option 3 activities
  NEI letter dated January 19, 2000

PRA quality
- Status of ACRS Activities Associated with License Renewal
- Safety Culture at Nuclear Power Plants

• AP1000 Advanced Reactor Design

The staff has started the Phase 1 review of the AP1000 advanced reactor design. During this phase, the staff will identify questions and policy issues that will be evaluated during Phase 2 (feasibility) review. During the May 2000 meeting, the Committee suggested that the members identify issues that they believe should be evaluated by the staff during Phase 2.

• Technical Expertise Needed for Future ACRS Members

The Commission has recently selected Mr. Graham Leitch to be appointed as a new member to the ACRS and he is expected to attend the September ACRS meeting. Regarding future vacancies on the Committee, the Commission has asked the ACRS/ACNW Executive Director to identify specific technical expertise that is needed.

• Proposed Assignment and Guidance for Reviewing License Renewal Guidance Documents

The staff is in the process of preparing a Standard Review Plan, Generic Aging Lessons Learned II (GALL II) Report, and a Regulatory Guide associated with license renewal. The Committee needs to complete its review of these documents in November 2000. Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, has proposed assignments for the members for reviewing these documents.

• ACRS/ACNW Self Assessment

The Commission paper (SECY-00-0102) on the CY 1999 self-assessment of ACRS and ACNW performance was issued on May 5, 2000.

• ACRS Memorandum of Understanding

A draft Memorandum of Understanding between the ACRS and the EDO was provided to the ACRS members during the May ACRS meeting. The EDO will provide this draft to NRR, NMSS, RES, and OGC for review.

- Letter from Gordon Thompson on Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

Mr. Gordon Thompson forwarded a letter to Dr. Powers raising issues on Spent Fuel Pool Accidents. In his letter, comparisons are made between the NRC staff's technical study for decommissioning plants and operating plants. He recommended that the ACRS take two actions: (1) independently investigate both operating and decommissioning plants and (2) recommend that the NRC immediately initiates a comprehensive investigation of the state of knowledge on scientific issues relevant to the risk posed by spent fuel pools. Also, he provided a report to the NRC's Licensing Board and volunteered to discuss these matters with the ACRS.

- Other Issues

Based on discussion of other issues, the Subcommittee makes the following recommendations:

- The trip to Germany to meet with RSK previously scheduled for June 2000 has been postponed.

- Drs. Apostolakis and Bonaca will attend the ASME workshop scheduled for June 27, 2000, in Rockville to discuss the proposed final ASME Standard for PRA Quality.

- The members should develop a list of issues associated with power uprates to be sent to the NRC staff for discussion at a future meeting of the Thermal-Hydraulic Phenomena Subcommittee.

- Mr. Sieber should recommend whether he and Mr. Singh should attend the Fire Protection Conference in London scheduled for February 12-14, 2001. Attendance would be contingent upon presenting a paper during this conference.

- Dr. Apostolakis will propose assignments for the members for reviewing the proposed final ASME Standard (Phase 1) and the proposed ANS Standard (Phase 2) for PRA Quality, which are scheduled for discussion during the July and September 2000 meetings, respectively.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 474th ACRS Meeting, July 12-14, 2000.

The 473rd ACRS meeting was adjourned at 12:45 p.m. on June 9, 2000.

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

August 7, 2000

MEMORANDUM TO: Sherry Meador, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Dana A. Powers, Chairman
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFIED MINUTES OF THE 473rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JUNE 7-9, 2000

I certify that based on my review of the minutes from the 473rd ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

Dana A. Powers

Dana A. Powers, Chairman

August 7, 2000
Date

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

July 26, 2000

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 473rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - JUNE 7-9, 2000

Enclosed are the proposed minutes of the 473rd 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

videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: May 17, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–13061 Filed 5–23–00; 8:45 am]

BILLING CODE 7590–01–P

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 June 7–9, 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).

Wednesday, June 7, 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.–10:00 A.M.: Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed resolution of Generic Safety Issue-173A.

10:15 A.M.–11:45 A.M.: Regulatory Effectiveness of the Station Blackout Rule (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the results of the review performed by the staff to determine the regulatory effectiveness of the Station Blackout Rule.

12:45 P.M.–2:15 P.M.: Proposed Final Standard Review Plan Section and Regulatory Guide Associated with the Revised Source Term Rule (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed final Standard Review Plan Section and Regulatory Guide associated with the application of the revised source term for operating nuclear power plants.

2:30 P.M.–4:30 P.M.: Assessment of the Quality of Probabilistic Risk Assessments (PRAs) (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the staff's proposal to address PRA quality until the industry standards have been completed, including the potential role of industry PRA certification process.

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.

Thursday, June 8, 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.–10:00 A.M.: Performance-Based Regulatory Initiatives (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding a draft Commission Paper associated with performance-based regulatory initiatives and related matters.

10:15 A.M.–11:30 A.M.: Use of Industry Initiatives in the Regulatory Process (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding use of industry initiatives in the regulatory process.

11:30 A.M.–12:00 Noon: Safety Culture at Operating Nuclear Power Plants (Open)—The Committee will hear a presentation by and hold discussions with Mr. Sorensen, ACRS Senior Fellow, regarding the safety culture at operating nuclear power plants.

1:00 P.M.–1:30 P.M.: Visit to Davis Besse Nuclear Power Plant and Meeting with NRC Region III Personnel (Open)—The Committee will hear a presentation by and hold discussions with Mr. Singh, ACRS Senior Staff Engineer, regarding the proposed schedule for touring the Davis Besse Nuclear Power Plant, specific plant areas to be visited, proposed topics for discussion with representatives of the licensee, and the NRC Region III Office.

1:30 P.M.–2:00 P.M.: Proposed Plan and Assignments for Reviewing License Renewal Guidance Documents (Open)—The Committee will discuss the proposed plan and member assignments for reviewing the license renewal guidance documents (Standard Review Plan, Regulatory Guide, and Generic Aging Lessons Learned II Report).

2:00 P.M.–2:15 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.

2:15 P.M.–3:00 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.

3:00 P.M.–4:00 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.

4:00 P.M.–7:00 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Friday, June 9, 2000

8:30 A.M.–2:30 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

2:30 P.M.–3:00 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 September 28, 1999 (64 FR 52353). 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. Sam Duraiswamy, 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 this 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

contacting Mr. Sam Duraiswamy 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. Sam Duraiswamy 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. Sam Duraiswamy (telephone 301/415–7364), 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 3:00 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: May 18, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–13060 Filed 5–23–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Joint Meeting of the Subcommittees on Plant Operations and Fire Protection; Notice of Meeting

The ACRS Subcommittees on Plant Operations and Fire Protection will hold a joint meeting on June 14, 2000, NRC Region III Office, 801 Warrenville Road, Lisle, Illinois.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, June 14, 2000—8:30 a.m. Until the Conclusion of Business

The Subcommittees will discuss items of mutual interest with the representatives of NRC Region III Office, including plant performance review process, implementation challenges associated with the revised inspection and assessment programs, and fire protection issues. 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 and 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 Subcommittees, their 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 Subcommittees, along with any of their consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittees will then hear presentations by and hold discussions with representatives of the NRC Region III Office, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and 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, Mr. Amarjit Singh (telephone 301/415–6899) 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 to the agenda, etc., that may have occurred.

Dated: May 17, 2000.

Richard K. Major,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 00–13062 Filed 5–23–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of May 22, 29, June 5, 12, 19, and 26, 2000.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of May 22

Thursday, May 25

8:30 a.m. Briefing on Operating Reactors and Fuel Facilities (Public Meeting) (Contact: Joe Shea, 301–415–1727)

10:15 a.m. Briefing on Status of Regional Programs, Performance and Plans (Public Meeting) (Contact: Joe Shea, 301–415–1727)

1:25 a.m. Affirmation Session (Public Meeting)

a: Hydro Resources, Inc., Docket No. 40–8968–ML, Memorandum and Order (Financial Assurance for Decommissioning Issues), LBP–99–13, 49 NRC 233 (March 9, 1999); and Memorandum and Order (Motion to Hold in Abeyance), LBP–99–40 (October 19, 1999); and,

b: Final Rule: "Elimination of the Requirement for Noncombustible Fire Barrier Penetration Seal Materials and Other Minor Changes" (10 CFR Part 50) (WITS 199800128) (Contact: Ken Hart, 301–415–1659)

1:30 p.m. Briefing on Improvements to 2.206 Process (Public Meeting) (Contact: Andrew Kugler, 301–415–2828)

Week of May 29—Tentative

Tuesday, May 30

9:25 a.m. Affirmation Session (Public Meeting) (If needed)

Week of June 5

There are no meetings scheduled for the Week of June 5.

Week of June 12—Tentative

Tuesday, June 13

9:25 a.m. Affirmation Session (Public Meeting) (If needed)

9:30 a.m. Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting) (Contact: Paul Lohaus, 301–415–3340)

1:00 p.m. Meeting with Korean Peninsula Energy Development

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001
June 1, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
473rd ACRS MEETING
JUNE 7-9, 2000

WEDNESDAY, JUNE 7, 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/SD)
   - 1.2) Items of current interest (DAP/NFD/SD)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/SD)

2) 8:35 - 10:00 A.M. Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities" (Open) (TSK/MME)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed resolution of Generic Safety Issue-173A.

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

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

3) 10:15 - ~~11:45~~ 11:30 A.M. Regulatory Effectiveness of the Station Blackout Rule (Open) (MVB/NFD/AS)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the results of the review performed by the staff to determine the regulatory effectiveness of the Station Blackout Rule.

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

~~11:45~~ 11:30 - ~~12:45~~ 12:05 P.M. ***LUNCH***

4) ~~12:45~~ 1:05 - ~~2:15~~ 2:00 P.M. Proposed Final Standard Review Plan Section and Regulatory Guide Associated with the Revised Source Term Rule (Open) (TSK/PAB/MWW)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed final Standard Review Plan Section and Regulatory Guide associated with the application of the revised source term for operating nuclear power plants.

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

2:00
~~2:15~~ - **2:30 P.M.** *****BREAK*****

4:25
5) 2:30 - ~~4:30~~ P.M. Assessment of the Quality of Probabilistic Risk Assessments (PRAs) (Open) (GA/MTM)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's proposal to address PRA quality until the industrial standards have been completed, including the potential role of industry PRA certification process.

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

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.

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

5:30-6:00
6:10-7:00 +
6:00-6:05

7.1) Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities" (TSK/MME)

7.2) Proposed Final Standard Review Plan Section and Regulatory Guide Associated with the Revised Source Term Rule (TSK/PAB/MWW)

7.3) Regulatory Effectiveness of the Station Blackout Rule (tentative) (MVB/NFD)

THURSDAY, JUNE 8, 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/SD)

9) 8:35 - 10:00 A.M. Performance-Based Regulatory Initiatives (Open) (JDS/MTM)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding a draft Commission Paper associated with performance-based regulatory initiatives and related matters.

9:37-10:00 9.3) Lisa Gue (Public Citizens Critical Mass)

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

10:20
10:00 - ~~10:15~~ A.M. *****BREAK*****

10:20-11:18
10) ~~10:15~~ - 11:30 A.M. Use of Industry Initiatives in the Regulatory Process (Open) (JJB/NFD)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding use of industry initiatives in the regulatory process.

11:00-11:18 Representatives of the nuclear industry will provide their views, as appropriate.

11:20-12:25
11) ~~11:30~~ - ~~12:00~~ Noon Safety Culture at Operating Nuclear Power Plants (Open) (GA/NFD/JS)

11.1) Remarks by the Subcommittee Chairman

11.2) Briefing by and discussions with Mr. Sorensen, ACRS Senior Fellow, regarding the safety culture at operating nuclear power plants.

Representatives of the NRC staff will provide their views, as appropriate.

12:25-1:30
~~12:00~~ - ~~1:00~~ P.M. ***LUNCH***

1:30-1:55
12) ~~1:00~~ - ~~1:30~~ P.M. Visit to Davis Besse Nuclear Power Plant and Meeting with NRC Region III Personnel (Open) ~~(JJB/AS)~~ → Sr Staff Engineer briefed Committee

12.1) Remarks by the Subcommittee Chairman

12.2) Briefing by and discussion with Mr. Singh, ACRS Senior Staff Engineer, regarding the proposed schedule for touring the Davis Besse Nuclear Power Plant, specific plant areas to be visited, proposed topics for discussion with representatives of the licensee, and the NRC Region III Office.

1:57-3:15
13) ~~1:30~~ - ~~2:00~~ P.M. Proposed Plan and Assignments for Reviewing License Renewal Guidance Documents (Open) (MVB/NFD)

Discussion of the proposed plan and member assignments for reviewing the license renewal guidance documents (Standard Review Plan, Regulatory Guide, and Generic Aging Lessons Learned II Report).

3:15-3:25
14) ~~2:00~~ - ~~2:15~~ P.M. Reconciliation of ACRS Comments and Recommendations (Open) (DAP, et al./SD, et al.)

4:00-4:08

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

4:08-5:10
15) ~~2:15~~ - ~~3:00~~ P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/SD)

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

15.2) 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.

3:25
16) ~~3:00~~ - 4:00 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.

5:10-5:45 Break

5:45 - 7:25

17) ~~4:00~~ - ~~7:00~~ P.M. Discussion of Proposed ACRS Reports

6/9 Final

Discussion of proposed ACRS reports on:

6/8

17.1) Assessment of PRA Quality (GA/MTM)

17.2) Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities" (TSK/MME)

9:50-10:15 Final

17.3) Regulatory Effectiveness of the Station Blackout Rule (tentative) (MVB/NFD)

Final

17.4) Proposed Final Standard Review Plan Section and Regulatory Guide Associated with the Revised Source Term Rule (TSK/PAB/MWW)

3:30-3:45

17.5) Performance-Based Regulatory Initiatives (tentative) (JDS/MTM)

3:40-3:45 9:20-9:30

17.6) Use of Industry Initiatives in the Regulatory Process (JJB/NFD)

3:50-4:00

17.7) Safety Culture at Nuclear Power Plants (tentative) (GA/NFD/JS)

FRIDAY, JUNE 9, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

12:45

18) 8:30 - ~~2:30~~ P.M. Discussion of Proposed ACRS Reports (Open)

Continue discussion of proposed ACRS reports listed under Item 17.

8:55-9:20 Discuss topics for meeting w/Commission in October

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

19) 2:30 - 3:00 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 III: MEETING ATTENDEES

473RD ACRS MEETING
June 7-9, 2000

NRC STAFF (June 7, 2000)
G. Millman, OEDO
E. Throm, NRR
G. Kelly, NRR
G. Hubbard, NRR
S. Lee, NRR
T. Eaton, NRR
T. Collins, NRR
F. Gallardo, NRR
J. Lening, III, NRR
J. Lee, NRR
S. LaVie, NRR
M. Blumberg, NRR
R. Emch, NRR
R. Barrett, NRR
J. Williams, NRRJ. Hyslop, NRR
I. Jung, NRR
G. Parry, NRR
S. Dinsmore, NRR
J. Ibarra, RES
H. VanderMolen, RES
B. Roughley, RES
F. Eltawila, RES
T. Wolf, RES
J. Rosenthal, RES
A. El-Bassioni, RES
M. Drouin, RES
J. Lanz, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
F. Saba, NUSIS
A. Wyche, SERCH/Bechtel
B. Youngblood, ISL
A. Marion, NEI
P. Negus, GE
J. Meyer, ISL
M. Radvansky, GPU
M. Khakb-Rabba, ERI
K. Cozens, NEI
P. Campbell, Winston & Strawn
B. Bradley, NEI

NRC STAFF (June 8, 2000)
J. Rosenthal, RES
N. Kadambi, RES
F. Eltawila, RES
B. Schoenfeld, RES
J. Kramer, RES
G. Lanik, RES
J. Birmingham, NRR
S. West, NRR
C. Carpenter, NRR
D. Wessman, NRR
B. Hermann, NRR
P. T. Kuo, NRR
B. Jasinski, OPA

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
B. Youngblood, ISL
L. Gue, Public Citizen
H. Fonticella, Dominion Generation
F. Saba, NUSIS
D. Raleigh, SERCH/Bechtel
A. Marion, NEI
P. Negus, GE
J. Riccio, Public Citizen

NRC STAFF (June 9, 2000)
J. Persensky, RES

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

June 21, 2000

SCHEDULE AND OUTLINE FOR DISCUSSION
474TH ACRS MEETING
JULY 12-14, 2000

WEDNESDAY, JULY 12, 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/SD)
   - 1.2) Items of current interest (DAP/NFD/SD)
   - 1.3) Priorities for preparation of ACRS reports (DAP/JTL/SD)

2) 8:35 - 10:30 A.M. Activities Associated with Risk-Informing 10 CFR Part 50 (Open)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding:
     - a) Proposed revision to 10 CFR 50.44 concerning combustible gas control system and advance notice of proposed rulemaking (10 CFR 50.69 and Appendix T) (WJS/MTM)
     - b) NEI letter dated January 19, 2000 (TSK/NFD)

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

3) 10:45 - 11:45 A.M. Assessment of the Quality of the Probabilistic Risk Assessments (Open) (GA/MTM/MWW)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding a draft Commission paper on the assessment of the quality of probabilistic risk assessments (PRAs).

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

11:45 - 1:15 P.M. ***LUNCH***

4) 1:15 - 3:15 P.M. Proposed Final ASME Standard for PRA Quality (Open) (GA/MTM)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the American Society of Mechanical Engineers (ASME) regarding the proposed final ASME Standard for PRA quality.

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

3:15 - 3:30 P.M. *****BREAK*****

5) 3:30 - 4:30 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.

6) 4:30 - 7:00 P.M. Discussion of Proposed ACRS Reports (Open)
Discussion of proposed ACRS reports on:
6.1) Proposed Revision to 10 CFR 50.44 concerning combustible gas control system and advance notice of proposed rulemaking (10 CFR 50.69 and Appendix T) (WJS/MTM)
6.2) NEI Letter on Risk-Informing 10 CFR Part 50 (TSK/NFD)
6.3) Assessment of the Quality of PRAs (GA/MTM/MWW)
6.4) Proposed final ASME Standard for PRA Quality (GA/MTM)
6.5) Safety Culture at Nuclear Power Plants (GA/NFD/JS)

THURSDAY, JULY 13, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 9:30 A.M. Annual Report to the Commission on the NRC Safety Research Program (Open) (DAP/MME)
Discussion of the format and content of the annual ACRS report to the Commission on the NRC Safety Research Program.

9) 9:30 - 9:45 A.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.

10) 9:45 - 10:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (DAP/JTL/SD)
10.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
10.2) 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.

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

11) 10:45 - 11:45 A.M. Break and Preparation of Draft ACRS Reports (Open)
Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

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

12) 12:45 - 6:00 P.M. Discussion of Proposed ACRS Reports

Discussion of proposed ACRS reports on:

12.1) Proposed Revision to 10 CFR 50.44 concerning combustible gas control system and advance notice of proposed rulemaking (10 CFR 50.69 and Appendix T) (WJS/MTM)

12.2) NEI Letter on Risk-Informing 10 CFR Part 50 (TSK/NFD)

12.3) Assessment of the Quality of PRAs (GA/MTM/MWW)

12.4) Proposed final ASME Standard for PRA Quality (GA/MTM)

12.5) Safety Culture at Nuclear Power Plants (GA/NFD/JS)

FRIDAY, JULY 14, 2000, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) 8:30 - 11:30 A.M. Discussion of Proposed ACRS Reports (Open)

(10:00-10:15 A.M. BREAK) Continue discussion of proposed ACRS reports listed under Item 12.

14) 11:30 - 1:30 P.M. Topics for Meeting with the NRC Commissioners on October 5, 2000

Discussion of topics for meeting with the Commissioners scheduled for October 5, 2000.

15) 1:30 - 2:00 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
473RD ACRS MEETING
June 7-9, 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 June 7-8, 2000
2	Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities"
	2. GSI-173A, Generic Spent Fuel Storage Pool Part A: Operating Facilities, presentation by NRR [Viewgraphs]
	3. Briefing to the ACRS on the Draft Report - Regulatory Effectiveness of the SBO Rule, presentation by RES [Viewgraphs]
3	Regulatory Effectiveness of the Station Blackout Rule
	4. Draft Report - Regulatory Effectiveness of the SBO Rule, presentation by RES [Viewgraphs]
4	Proposed Final Standard Review Plan Section and Regulatory Guide Associated with the Revised Source Term Rule
	5. Final Regulatory Guide 1.183 (DG-1081), Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Final SRP 15.0.1, presentation by NRR [Viewgraphs]
5	Assessment of the Quality of Probabilistic Risk Assessments (PRAs)
	6. Response to SRM on PRA Quality, presentation by RES and NRR [Viewgraphs]
9	Performance-Based Regulatory Initiatives
	7. High-Level Guidelines for Performance-Based Activities, presentation by N. Prasad Kadambi, REAHFB, NRR [Viewgraphs]

AGENDA ITEM NO.	DOCUMENTS
10	Use of Industry Initiatives in the Regulatory Process
	8. Industry Initiatives in the Regulatory Process, presentation by C. E. Carpenter, NRR [Viewgraphs]
11	Safety Culture at Operating Nuclear Power Plants
	9. Safety Culture, presentation by J. N. Sorensen, ACRS/ACNW [Viewgraphs]
12	Visit to Davis Besse Nuclear Power Plant and Meeting With NRC Region III Personnel
	10. Site Visit to the Davis-Besse Nuclear Power Plant, June 8, 2000, presentation by Amarjit Singh, ACRS [Viewgraphs]
13	Proposed Plan and Assignments for Reviewing License Renewal Guidance Documents
	11. License Renewal Process, presentation by Mario Bonaca, ACRS [Viewgraphs]
14	Reconciliation of ACRS Comments and Recommendations
	12. Reconciliation of ACRS Comments and Recommendations [Handout #14.1]
15	Future ACRS Activities/Report of the Planning and Procedures Subcommittee
	13. Minutes of Planning and Procedures Subcommittee Meeting - June 6, 2000, 19 [Handout #15.1]
	14. Future ACRS Activities - 473rd ACRS Meeting, June 7-9, 2000, [Handout 15.2]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Proposed Resolution of Generic Safety Issue-173A, "Spent Fuel Storage Pool for Operating Facilities"
	1. Table of Contents
	2. Proposed Schedule
	3. Project Status Report, dated June 7, 2000
	4. Resolution of Spent Fuel Storage Pool Action Plan Issues, dated July 26, 1996
	5. Follow up Activities on the Spent Fuel Pool Action Plan, dated September 30, 1997
	6. ACRS letter to the Executive Director for Operations, dated October 16, 1998
3	Regulatory Effectiveness of the Station Blackout Rule
	7. Table of Contents
	8. Proposed Schedule
	9. Status Report
	10. Letter dated April 14, 2000, from Charles E. Rossi, RES, to David Modeen, NEI, Subject: Draft Report, "Regulatory Effectiveness of the Station Blackout Rule
4	Proposed Final Standard Review Plan Section and Regulatory Guide Associated with the Revised Source Term Rule
	11. Table of Contents
	12. Presentation Schedule
	13. Project Status Report
	14. Letter to Greta Joy Dicus, Chairman, NRC, from Dana A. Powers, Chairman, ACRS, "Proposed Final Rule on Use of Alternative Source Term at Operating Reactors, Draft Regulatory Guide, and Standard Review Plan," dated September 17, 1999
	15. Letter to D. A. Powers, Chairman, ACRS, from W. D. Travers, EDO, Subject: Proposed Final Rule on Use of Alternative Source Term at Operating Reactors, Draft Regulatory Guide, and Standard Review Plan, dated October 26, 1999
	16. Memorandum to J. T. Larkins, ACRS, from G. M. Holahan, NRR, Subject: Transmittal of Final Regulatory Guide 1.XXX (DG-1081), "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear

Power Plants," and Standard Review Plan Section 15.0.1, "Radiological Consequences Analysis Using Alternative Source Terms," dated May 5, 2000

5 Assessment of the Quality of Probabilistic Risk Assessments (PRAs)

17. Table of Contents
18. Proposed Schedule
19. Status Report
20. Memorandum dated April 18, 2000, from A. Vietti-Cook, SECY, NRC, to W. D. Travers, EDO, NRC, Subject: SRM-Briefing on Risk-Informed Regulation Implementation Plan (SECY-00-0062)
21. Letter dated March 25, 1999, from D. A. Powers, Chairman, ACRS, to W. d. Travers, EDO, NRC, Subject: Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (Phase 1)
22. Letter dated May 13, 1999, from W. D. Travers, EDO, NRC, to D. A. Powers, Chairman, ACRS, Subject: Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plants (Phase 1)
23. Letter dated May 3, 1999, from A. Thadani, Director, RES, to Jess Moon, ASME, Subject: Staff Comments on ASME Draft "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," without enclosures

9 Performance-Based Regulatory Initiatives

24. Table of Contents
25. Proposed Schedule
26. Status Report
27. *Federal Register* Notice dated May 9, 2000, request for comments. US NRC, Subject: Revised High-Level Guidelines for Performance-Based Activities
28. Memorandum dated September 13, 1999, from A. Vietti-Cook, SECY, to W. D. Travers, EDO, Subject: SRM-Plans for Pursuing Performance-Based Initiatives

10 Use of Industry Initiatives in the Regulatory Process

29. Table of Contents
30. Proposed Schedule
31. Status Report
32. Proposed Draft Commission Paper, "Industry Initiatives in the Regulatory Process, " received May 30, 2000

11 Safety Culture at Operating Nuclear Power Plants

33. Table of Contents
34. Proposed Schedule
35. Status Report
36. T. Shiel, "The Human Performance Improvement Program at Duke Power Nuclear Stations, Nuclear News, May 2000

13 Proposed Plan and Assignments for Reviewing License Renewal Guidance Documents

37. Table of Contents
38. Proposed Schedule
39. Status Report
40. M. Bonaca, "License Renewal Plan for Reviewing Guidance Documents," updated May 25, 2000
41. M. Bonaca, "ACRS Review Assignments for License Renewal Guidance Documents," dated May 25, 2000
42. M. Bonaca, "License Renewal Guidance for ACRS Review of Generic Documents"

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

473rd FULL COMMITTEE MEETING

June 7, 2000

NRC STAFF SIGN IN FOR ACRS MEETING

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NAME	BADGE NO.	NRC ORGANIZATION
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Glenn Kelly	B6358	" " "
George Hubbard	B6279	NRR/DSSA/SPLB
Jose Ibarra	B8398	RES/DSARE
Samuel Lee	B7501	NRR/DSSA/SPSB
Harold VanderMolen	B7214	RES/DSARE/REAHFB
Tanya Eaton	B8886	NRR/DSSA/SPLB
Jim Collins	A7568	NRR/DSSA
Gil Millman	B6804	EDO
Francisco Gallardo	F6109	NRR/EICB
John Lehninger	C6679	NRR/DSSA/SPLB
Bill Raughley	B-8415	RES/DSARE/HFREB
FAROUK ELTAWILA	A6364	RES/DSARE
TOM WOLF	B-7299	RES/DRAA
F Rosenthal	A6661	RES
Jay Lee	B6664	NRR/DSSA
Steve LaVie	B8172	NRR/DSSA/SPSB
Mark Blumberg	B8170	" " "
Rich Emch	B8575	" " "
RICH BARRETT	A7330	" " "
JOE WILLIAMS	B7257	NRR/DLPM/PDIV
Adel A. El-Bassioni	B6525	RES/

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

_____ FULL COMMITTEE MEETING

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J Stolyslogel	B6290	NRR / SPSB
Ian Jung	B7905	NRR / SPSB
Mary Moore	B7583	NRC / RES
Garth Parry	B8060	NRR / DISA
Stephen Dinsmore	B 7898	NRR/DSSA
J. LANE	B6625	NRC/RES

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

MEETING OF THE 473rd

FULL COMMITTEE MEETING

June 7, 2000

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NAME	AFFILIATION
Farideh E. Saba	NUSIS / Scientech
Altheia Wyche	SERCH Licensing / Bechtel
Bob Youngblood	ISL
Alex Marion	NEI
Paul Negus	GE
Jim Meyer	ISL
Mike Radvansky	Corpu Nuclear
Mohsen Khalil-Rahim	ERI
Kurt Cozens	NEI
Patricia L. Campbell	Winston & Strawn
Biff Bradley	NEI

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

473rd FULL COMMITTEE MEETING

June 8, 2000

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Jack Rosenthal	A6661	RES
N P KADAMBI	B6343	RES.
FAROUK ELTAWILA	A6364	RES
Joe Birmingham	B 6085	NRR
Bakelle Schoenfeld	B 6988	NRR
Steve West	B7258	NRR
C E Carpenter	B6465	NRR
Dirk Wessman	A6827	NRR
Bob Jasinski	B8692	OPA
Bob Hermann	B8586	NRR
Joel Kramer	B8401	RES
George Lainas	B 8402	RES
P T Kuo	B7548	NRR

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_____ FULL COMMITTEE MEETING

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

MEETING OF THE 473rd

FULL COMMITTEE MEETING

June 8, 2000

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NAME	AFFILIATION
BOB YOUNGBLOOD	ISL
Lisa Gue	Public Citizen
H. Fontecilla	Dominion Generation
Farideh Saba	Nusis / Scientech
Dean R. Rahn	SSRCH Bechtel
Alex Marion	NEI
Paige Negus	GE
Vim Riccio	Public Citizen

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

473rd FULL COMMITTEE MEETING

June 9, 2000

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J PERSENSKY	B8413	RES

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1

ITEMS OF INTEREST

473rd ACRS MEETING

JUNE 7-8, 2000

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
473rd MEETING
JUNE 7-9, 2000

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-00-14 May 15, 2000

[PDF Version (55 KB)]

AN OVERVIEW OF MANAGING THE U.S. RADIATION PROTECTION PROGRAM CONCERNING GENERALLY-LICENSED SOURCES AND DEVICES

Commissioner Greta Joy Dicus
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Keynote Presentation

at the

10th Annual International Radiation Protection Association Conference

May 15, 2000
Hiroshima, Japan

INTRODUCTION

Since the mid-1980s, there have been a steady number of lost, stolen or abandoned radioactive sources and devices reported throughout the world. Currently, in the United States (U.S.), there are about 1.8 million devices in use today that contain varying, but limited amounts of radioactive material. These devices are authorized for possession, use, or storage by about 150,000 "general licensees." Because of the relatively small radiation risk imposed by these devices, the U.S. Nuclear Regulatory Commission (NRC or the Commission) did not establish routine a contact or inspection program for these general licensees. Based on data received by the NRC, approximately 375 sources or devices of varying risk have been reported lost

or stolen per year in the U.S. Although the number of reported lost devices has decreased over recent years (NRC data indicates that 286 reports were received in 1999), the NRC has become increasingly concerned about occurrences where generally-licensed devices have not been handled or disposed of properly because of the potential for public exposure or contamination of property. Some generally-licensed devices have been accidentally melted in steel mills causing contamination of the mill, the steel product, and the wastes from the process (i.e., the slag and the baghouse dust). Although only a few exposures have exceeded the public dose limits, there may be a future potential for exposures involving the general public.

Due to the public risk associated with the uncontrolled release of radioactive sources and devices, there has been much attention directed by international safety organizations and national authorities to address any potential radiation and environmental hazards. Ensuring that radiation and environmental safety in the nuclear fuel cycle arena is consistent with societal expectations is a challenge to all concerned. It is clear that we need to establish a balance between the many beneficial uses of radiation sources, as well as to ensure that there exists a "cradle to grave" mechanism for the safe disposal and accountability of used, spent sources. Our challenge in the regulatory community is to have a process in place that will improve the accountability and control over devices and sources of particular concern, so that the responsible party can be contacted or corrective actions taken when the need arises.

The societal impact of incidents involving "generally licensed" radiation sources appears to not make the same strong imprint upon the public that events involving nuclear power reactors or fuel cycle facilities do. One consequence of this is that the public and political pressure for legislative and regulatory action in this area is not always as strong as it is for nuclear facilities. As a result, legislative bodies have not always provided the needed resources to the regulatory authorities so that they can implement more effective radiation safety regulatory programs for all types of radiation sources. Our challenge is to ensure that *all* radiation sources receive appropriate regulatory attention commensurate with their risk in order to protect public health and safety. Consequently, this paper will discuss NRC's proposed risk-based regulations which are planned to ensure greater improvement in the accountability and traceability of generally-licensed devices in the U.S.

BACKGROUND

On February 12, 1959 [24 Federal Register (FR) 1089], the Atomic Energy Commission (AEC) amended its regulations to provide "general licenses" for the use of byproduct material[(1)] contained in certain measuring, gauging, or controlling devices. Under current regulations in U.S. Title 10, Code of Federal Regulations (CFR), Section 31.5, certain persons may receive and use a device containing byproduct material under this general license if the device has been manufactured and distributed according to a specific license issued by the NRC or by an Agreement State.[(2)] A specific license authorizing distribution of generally-licensed devices is issued if the regulatory authority determines that the safety features of the device and the instructions for its safe operation and disposal are adequate and meet regulatory requirements.

The person or firm who receives such a device is a general licensee. These general licensees are subject to requirements for maintaining labels, following instructions for safe use, storing and disposing of the device properly, and reporting transfers and failure of or damage to the device. For some devices, the general licensee must also comply with testing requirements for leakage and for proper operation of on-off mechanisms. General licensees must comply with the safety instructions contained in, or referenced on, the label of the device and must have the testing or servicing of the device performed by an individual who is authorized to manufacture, install, or service these devices except as indicated on the label.

The devices authorized by the general license usually consist of radioactive material, contained in a sealed

source, within shielded housing. The device is designed with inherent radiation safety features so that it can be used by persons with little or no radiation training or experience. The general license simplifies the regulatory licensing process so that a case-by-case determination of the adequacy of the radiation training or experience of each user is not necessary. Unfortunately, individuals or the public, who possess devices under this general license are not always aware of applicable requirements and thus are not always complying with all of these requirements.

U.S. operational experience with radioactive materials includes few accidents with generally-licensed devices, and only five have resulted in potential radiation overexposures to the general public since 1989. The U.S. metal recycling industry has been particularly affected by losses and thefts of radioactive sources, some of which were generally-licensed and have subsequently become mixed with metal scrap destined for recycling. Since 1983, U.S. steel mills accidentally melted radioactive sources on 20 occasions and radioactive sources have been accidentally melted at other metal mills on 10 other occasions. Due to the nature of these incidents (melting of the source) there is no conclusive evidence that the radioactive material was from generally-licensed, specifically-licensed (e.g., large radiography or irradiator sources), or whether it was from any U.S. licensee at all. The NRC's Nuclear Materials Events Database (NMED) indicates that approximately 90% of radiation alarms at scrap recycling facilities are from naturally occurring radioactive material, while the remaining 10% are from licensed byproduct material. While radiation exposures of mill workers and the public have, thus far, been very low and below the international dose limits for members of the public, the financial consequences for a few events have been large. For a smelting event involving a large radiation source (believed to not be a generally licensed device) one U.S. steel mills incurred an average cost of approximately US$ ten million, while yet in another case the cost approached US$ 23 million.

Lost, stolen, and abandoned generally-licensed sources or devices appearing in recycled metals constitute a worldwide problem. Thirty other smelting events have been reported in at least eighteen other countries ([3]). Others may have occurred but have not come to our attention or cannot be confirmed. These events have the potential for international consequences because of the transboundary transport of radioactive effluents from a mill that has accidentally melted a source or as the result of international marketing of mill products and byproducts that have become contaminated, such as cobalt-60 contaminated steel products. Radioactively contaminated products imported into the U.S. have been found on ten occasions ([4]). The sources of contamination in most of these cases are probably radioactive sources that became mixed with the raw materials used to make the products. Although none of these cases resulted in significant exposures of the public, the result of their unexpected appearance in the marketplace has a tendency to raise concerns about the effectiveness of regulatory programs to assure the safety of radiation sources.

To again cite from U.S. experience, although the NRC has a well-developed regulatory program for radioactive sources, the data that had been collected on lost and stolen radioactive sources and on discoveries of uncontrolled sources in the public domain, such as recycled metals, showed a clear need for strengthening of that program. As a result, the NRC conducted a 3-year sampling (1984 through 1986) of general licensees to assess the effectiveness of the general license program. The sampling revealed two major areas of concern: first, many general licensees were unaware of the regulations that apply to the possession of a generally-licensed device and second, approximately 15 percent of the general licensees sampled were unable to account for their devices. NRC concluded that these issues could be resolved by more frequent and timely contact between general licensees and the regulatory authority.

On December 27, 1991 (56 FR 67011), the NRC published a notice of proposed rulemaking concerning the accountability of generally-licensed devices. The proposed rule contained a number of provisions, including a requirement for general licensees to provide information to the NRC upon request, through which a device registry could be developed. The proposed rule also included requirements for other

specific licensees who manufacture or initially transfer generally-licensed devices. Although the public comments received were reviewed and a final rule developed, a final rule was not issued because the resources to fully implement the rule were not available.

The NRC has continued to consider the issues related to the loss of control of generally-licensed material, as well as specifically-licensed devices. In July 1995, the NRC, with assistance from the Organization of Agreement States, formed a working group to evaluate these issues. The working group consisted of both NRC and Agreement State regulatory personnel and encouraged the involvement of all persons having a stake in the process and the working group's final recommendations. All working group meetings were open to the public. A final report was published in October 1996 as NUREG-1551, "Final Report of the NRC-Agreement State Working Group to Evaluate Control and Accountability of Licensed Devices."

In considering the recommendations of this working group, the NRC decided, among other things, to initiate rulemaking to establish an annual registration of devices generally-licensed. Although the registration program would be similar to the program in the 1991 proposed rule, it would only apply to those devices considered to present a higher risk of potential exposure of the public or property damage in the case of loss of control (as compared to other generally-licensed devices).

Initially, NRC used the criteria developed by the working group for determining which sources should be subject to the registration program. Using these criteria, at the time of the proposed rule, it was estimated that the registration requirement would only apply to about 5100 general licensees possessing about 20,000 devices. These criteria were based on considerations of relative risk and are limited to radionuclides used in generally-licensed devices. If quantities of other radionuclides that would present a similar risk are used in these devices in the future, the criteria may be revised to include additional radionuclides.

On December 2, 1998, (63 FR 66492), the Commission directed that changes be made to provide more routine contacts with licensees using radioactive sources to remind them that they are responsible for accounting, control and proper disposal of licensed material. On October 4, 1999, this rule became final (*see* 64 FR 42269, published on August 4, 1999). The NRC intends to use this provision primarily to institute a registration and accounting system for devices containing certain quantities of specific radionuclides that present a higher risk of exposure to the public or a higher risk of property damage if a device were lost.

DISCUSSION

It was the Commission's intent that 1999 final rule provide one of the key elements in improving the accountability and control over devices of particular concern, through the institution of a registration process in allowing U.S. regulatory authorities to track general licensees and the specific devices they possess. Also, the tracking of devices would be useful in disseminating information to general licensees if a generic defect in devices was identified.

Therefore, on July 26, 1999 (64 FR 40295), NRC published a second notice of proposed rulemaking which added specific requirements concerning the registration of devices and provisions for an enhanced regulatory oversight program for all general licensees. Required information about these devices would be verified through a physical inventory and by checking label information. The advantage of including more explicit requirements in the regulation is that information about the registration process would be more clearly defined, available and standardized.

The proposed rule is planned to establish additional requirements for general licensees and distributors. Although the proposed regulations were written for the U.S., they can serve as a model to improve national

regulations and international recommendations for the control of radioactive sources and device. A brief discussion of some of the proposed requirements are explained below.

. Responsible Person. In practice, in order for a general licensee to comply with existing regulations, an individual in the corporation or institution must be aware of the requirements and be authorized to take the required actions. The "person" who holds a general license is usually a corporation, or public or private institution, rather than an individual. This proposed regulation would add a requirement for appointing a specific individual to be responsible for knowing about and taking actions to comply with regulations. Currently, if a device is not subject to testing, there are no routine actions required to be taken, primarily because the requirements are generally restrictions on actions, such as not abandoning the device, or actions to be taken only in the case of non-routine events, such as notification of the regulatory authority of the transfer or failure of the device. It is this type of situation, where knowledge of the nature of the device, the general license, and the associated regulations are unlikely to be maintained and passed on to individuals using the device. The proposed regulations would not require this individual to work on site at the place of use of the device, but would be responsible for ensuring that the general licensee is aware of required actions to be taken.

2. Timeliness of Disposition and Deferral of Testing While in Storage. Past operating experience has shown that when a device is not in use for a prolonged time, it is particularly susceptible to being forgotten and ultimately disposed of or transferred inappropriately. In addition, if a device is being held in storage indefinitely, it is likely that it is being stored to avoid the costs of proper disposal. By having a timeliness requirement in the regulations, if a period of storage exceeds the normal interval for testing, testing would not need to be done until the device is to be put back into use again. This would relieve the burden of unnecessary testing during the period of storage as well as eliminate any unnecessary exposure that could occur during testing for that period.

3. Provisions for Transfers to Specific Licensees. This proposed revision would provide some flexibility to the general licensee in transferring a device while ensuring that it is transferred appropriately. It would allow a general licensee to transfer a device directly to a waste collector for disposal, rather than going through a distributor. It would also allow the transfer of a device to other specific licensees, but would require regulatory authority approval in these cases so that it can ensure that the recipient is authorized to receive the device. The inclusion of a recipient's license number in the report of transfer would better ensure that the general licensee has verified that the recipient is a specific licensee, a waste collection licensee, or a specific licensee under equivalent Agreement State regulations authorized to receive it. It would also supply an additional means for the regulatory authority to identify the recipient, because company names and addresses sometimes change. The addition of the date of transfer will make the transfer easier to track and help to ensure that the general licensee makes the report in a timely manner (required within 30 days of transfer).

4. Change of Address Notification. If general licensees move their operations without notifying the regulatory authority, which has happened repeatedly in the past, they can be difficult to locate. Currently, the quarterly reports currently required of distributors are only intended to provide the U.S. regulatory authorities with the identity of general licensees in their jurisdictions and addresses at which these general licensees can be contacted. This revision would add a requirement to report address changes to the regulatory authority and would only apply to previously supplied mailing addresses and, for portable devices, the mailing address for the primary place of storage.

This simple change of address notification is intended to track moves of the general licensees and keep mailing addresses current. If a general licensee intends to move from one jurisdiction to another, such as from NRC to an Agreement State, it should contact the applicable regulatory authority before doing so to

determine the applicable, current regulations in that jurisdiction.

5. Reports of Device Failures. In the U.S., general licensees are not subject to decommissioning requirements. Under normal circumstances, a general license is granted by regulation and does not involve any termination of license process. If some generally-licensed device fails or is seriously damaged so as to cause significant contamination of the premises or environs, the regulatory authority may need to respond to the notification of an incident to ensure that a facility is properly decontaminated. Following such an incident, the regulatory authority would determine what actions are necessary on a case-by-case basis and, if necessary, would apply the criteria set out in the decommissioning regulations. The provision proposed in this action would require that the general licensee propose to the Commission how it will be shown that the premises are or will be adequately cleaned up. Depending on the nature of the event, the remedial action taken (and reported under existing requirements) along with any confirmatory surveys may be sufficient to complete action on the event.

6. Reporting New General Licensee's Responsible Individual. Consistent with the provision for appointing an individual through whom the general licensee will ensure compliance with the applicable regulations and requirements, and other reporting requirements being proposed, it is more effective for the general licensee to provide the name of the new responsible individual when another general licensee takes over the facility and responsibility for the device.

7. Reporting. The proposed rule would add additional information to the existing quarterly reporting requirement. This information would include the serial and model number of the device, the date of transfer, an indication if the device is a replacement, and the specific reporting period, among other requirements. Including the serial number will allow the regulatory authority to track individual devices in order to contact the responsible party if the need arises.

8. Labeling. The proposed rule would amend the existing labeling requirements to require an additional label on any separable source housing and a permanent label on devices meeting the criteria for registration. The NRC would consider a label "permanent," if, for example, it were embossed, etched, stamped, or engraved in metal. Under these requirements, new distributors would have labels approved as part of obtaining a license; distributors including existing licensees would have the new labeling requirements as conditions of license. Approval of new labels by NRC for existing distributors would not be required. NRC estimates that the impact of this proposed requirement should be minimal. The permanent label for devices requiring registration would provide better assurance that even when a device has been exposed to other than normal use conditions, for example, when a building has been refurbished or demolished with the device in place, the label will be intact and the device may be identified and proper actions can be taken. This may result in a more significant change to the labeling in the manufacturing of these devices.

9. Information to be provided to general licensees. The proposed rule would amend the requirements pertaining to the information distributors must provide to the recipient of the source or device, i.e., the general licensee. Distributors are now required to provide general licensees with a copy of pertinent regulations when the device is transferred. The proposed rule would require that a copy of these regulations be provided *before* transfer. The distributor would also be required to provide copies of additional applicable sections of the regulations, a listing of the services that can only be performed by a specific licensee, and information regarding disposal options for the devices being transferred. The disposal options would include the estimated cost for disposal of the device at the end of its useful life to the extent that the cost information is available to the distributor at the time of the sale of the device.

The Commission believes that the general licensee should be aware of the specific requirements before

purchasing a generally-licensed device, rather than afterward. While the NRC does not want to get involved with details of licensees' business practices, it is the Commission's intent that "prior to transfer" would be before a final decision to purchase so that the information can be considered in making that decision. Information on the estimated cost for disposal of the device at the end of its useful life may be a significant factor in a decision to purchase a device because of the high costs of disposing of radioactive materials. In some cases, the cost of disposal could exceed the purchase price of the device.

In order to offset the cost of the registration and follow-up program, and to comply with Federal Law (NRC is required to recover approximately 100% of its budget from licensees' fees), a proposed registration fee of US$ 420 for each general licensee possessing devices has been estimated to recover the cost of the general license program associated with this group of general licensees in an equitable way, as required by law.

The comment period for this proposed rulemaking closed on October 12, 1999, and the NRC staff is currently in the process of reviewing the 39 comments received, including three from the Agreement States. The NRC staff will perform a detailed analysis of the comments received, compile a summary of how the comments were incorporated into a revised final rule, and provide this final rule and supporting documentation to the Commission by mid-2000.

NATIONAL DATABASE

To supplement the revisions in the proposed rule, the Commission is in the process of developing a new computer database to incorporate information about general licensees and generally-licensed devices. Among other improvements from the earlier system, it will be designed to handle the registration process efficiently with automated features. In doing so, the Commission has given some consideration to whether a national database should be established in which information on the identity of general licensees and device information for all jurisdictions would be maintained, making this information accessible to all Agreement States and the NRC.

There are variations on the exact approach that might be taken particularly with respect to access and update authority. At this time, the Commission has not yet found it practical to resolve all the issues related to having broad access to the database. The Commission would like to give further consideration to establishing such a database that would not require rulemaking. However, if it were to be established, one option would be to change the material transfer reporting requirements so that distributors would report all transfers to the NRC rather than reporting to all jurisdictions into which transfers of devices are made.

A primary advantage of a national database would be the ease of tracing a "found" device back to the general licensee owner responsible for the device. A "found" generally-licensed device would be considered an orphan source until such time as the responsible general licensee is identified and it is returned to the licensee. The Commission is in the process of modifying the Nuclear Materials Events Database (NMED) to accept and track information on orphan sources in the U.S. Access to NMED will only be available to regulatory agencies in the U.S. The Commission will encourage the States to use NMED for this purpose so that this category of information will be shared nationally. However, NMED would rely on reporting of events for its data. In addition, information in a national general license database would be immediately available, and would contain the most complete information about general licensees and generally-licensed devices.

The primary disadvantage to a national database would be the difficulty of maintaining the security of the data, which is primarily made up of proprietary information. A national database would also present more risk to the integrity of the data, because there would be a higher potential for illicit corruption of data.

In considering whether or not to implement a national database and, if so, what the particular approach would be used, there are a number of aspects to be considered including--

(1) Who will maintain the database (the NRC, an independent third party, or each agency maintaining its own data)?
(2) How access to the data would be controlled.
(3) Potential changes to the reporting requirements for transfers.
(4) The ability for the NRC and the Agreement States to protect information of other agencies.
(5) Costs to implement and maintain the system or systems (including training).

Since the Commission has requested comment on the advantages and disadvantages of implementing a national database, it is their intent to review the staff's resolution of comments and publish a final rule which may or may not address the national database issue in calendar year 2000.

ENFORCEMENT

Separate from the proposed rule, the Commission has already established an interim enforcement policy for violations of generally-licensed devices that licensees discover and report during the initial cycle of the registration program. This policy supplements the normal NRC Enforcement Policy in NUREG-1600, Rev. 1. It will remain in effect through one complete cycle of the registration program.

Under this interim enforcement policy, enforcement action normally will not be taken for violations that are identified by the general licensee, and reported to the NRC if reporting is required, provided that the general licensee takes appropriate corrective action to address the specific violations and prevent recurrence of similar problems and otherwise has undertaken good faith efforts to respond to NRC notices and provides requested information. This change from the Commission's normal enforcement policy is to remove the potential for the threat of enforcement action to be a disincentive for the licensee to identify deficiencies. This approach is warranted given the limited NRC inspections of general licensees. This approach is intended to encourage general licensees to determine if applicable requirements have been met, to search their facilities to ensure sources are located, and to develop appropriate corrective actions when deficiencies are found. Under the interim enforcement policy, enforcement action, including issuance of civil penalties and Orders, may be taken where there is:

(a) failure to take appropriate corrective action to prevent recurrence of similar violations;
(b) failure to respond and provide the information required by regulation;
(c) willful failure to provide complete and accurate information to the NRC; or
(d) other willful violations, such as willfully disposing of generally-licensed material in an unauthorized manner.

As noted in the December 2, 1998, proposed rule, the Commission also plans to increase the civil penalty amounts specified in its Enforcement Policy in NUREG-1600, Rev. 1, for violations involving lost or improperly disposed sources or devices. This increase will better relate the civil penalty amount to the costs avoided by the failure to properly dispose of the source or device. Due to the diversity of the types of sources and devices, the Commission is considering the establishment of three levels of base civil penalty for loss or improper disposal. The three levels of base civil penalty would be US$ 5500, US$ 15,000, and US$ 45,000. The higher tiers would be for sources that are relatively costly to dispose of and would be based on approximately three times the average cost of proper transfer or disposal of the source or device.

CONCLUSION

Loss or inadvertent contamination of generally-licensed sources or devices is an issue having international implications. The proposed rulemaking NRC has underway can be considered as a model program for adoption by national regulatory authorities and international bodies to improve the accountability and control over devices and sources of particular concern, so that the responsible party can be contacted or corrective actions taken when the need arises. For this reason, national and international programs to facilitate the international exchange of information and cooperation in control and security of radioactive materials is essential. This international conference is a key step in achieving these objectives for even greater enhancement for the safe use of generally-licensed sources and devices in radiation protection worldwide.

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1. *Byproduct Material* as defined in 10 CFR Part 30.4 means any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or utilizing special nuclear material.

2. An *Agreement State* is one that has signed an agreement with the NRC to assume authority to regulate the use of byproduct material.

3. Dicus, Greta Joy, USA Perspectives: Safety and Security of Radioactive Sources. IAEA Bulletin, Vol. 41, No. 3, 1999, Vienna, Austria.

4. ibid.

From: Network Announcement
To: HQ distribution, Regional Distribution
Date: Wed, May 24, 2000 6:11 PM
Subject: **Senate Confirmation of Commissioner McGaffigan**

On May 24, 2000, the Senate confirmed Edward McGaffigan, Jr., to serve as Commissioner of the Nuclear Regulatory Commission for a second term. Commissioner McGaffigan will be sworn in by the Chairman at NRC Headquarters prior to June 30, when his current term ends. The Commissioner's second term will end on June 30, 2005.

UNITED STATES
NUCLEAR REGULATORY COMMISSION

Announcement No. 040

Date: May 23, 2000

To: **All NRC Employees**

SUBJECT: APPOINTMENT OF JOHN W. CRAIG AS ASSISTANT FOR OPERATIONS, OFFICE OF THE EXECUTIVE DIRECTOR FOR OPERATIONS

I am pleased to announce the appointment of Mr. John W. Craig as Assistant for Operations, Office of the Executive Director for Operations, effective May 28, 2000. He succeeds James L. Blaha who will begin the transition to his new duties as the NRC's Nuclear Safety Attache before assuming the position formally in October.

Mr. Craig began his career with the NRC in the cooperative education program. After receiving a Bachelor of Science degree in Nuclear Engineering from the University of Maryland in 1979, he joined the NRC staff and held progressively more responsible positions in the former Office of Inspection and Enforcement. In 1987, Mr. Craig was appointed to the Senior Executive Service and held several management positions in the Office of Nuclear Reactor Regulation including Chief, Plant Systems Branch; Project Director, Project Directorate III-2; and Project Director, License Renewal Project Directorate. In 1993, Mr. Craig was reassigned to the Office of Nuclear Regulatory Research where he has served as Deputy Director, Division of Engineering and Director, Division of Regulatory Applications. In 1999, he was appointed his most recent position of Director, Division of Engineering Technology. Since December 1999, Mr. Craig has served as Special Assistant for Communications in the Office of the Executive Director for Operations providing leadership in coordinating the implementation of communication activities including developing and implementing a plan to improve communications with internal and external stakeholders.

Prior to joining the NRC, Mr. Craig served in the U.S. Navy's nuclear power program. He is a recipient of the NRC Meritorious Service Award and the Presidential Meritorious Executive Rank Award.

Please join me in congratulating Mr. Craig on his new assignment.

/RA/
William D. Travers
Executive Director for Operations

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

Announcement No. 043

Date: June 1, 2000

To: **All NRC Employees**

SUBJECT: MANAGEMENT ASSIGNMENT CHANGES IN THE OFFICE OF NUCLEAR REGULATORY RESEARCH

I am pleased to announce the following management changes in the Office of Nuclear Regulatory Research effective June 1, 2000.

Michael E. Mayfield has been appointed to the position of Director of the Division of Engineering Technology, Office of Nuclear Regulatory Research.

Mr. Mayfield began his career with the NRC in the Office of Nuclear Regulatory Research as a Materials Engineer in 1985, and has held progressively more responsible positions in RES.

Mr. Mayfield was selected to participate in the SES Candidate Development Program in 1993, and in 1994, he was appointed to the Senior Executive Service as the Chief of the Materials Engineering Branch in RES.

Prior to joining the NRC, Mr. Mayfield worked for Battelle Memorial Institute in Columbus, Ohio, and Materials Engineering Associates in Lanham, Maryland. He received a Bachelor of Science in Physics from Missouri Southern State College in Joplin, Missouri, and a Master of Science in Mechanical and Aerospace Engineering from the University of Missouri-Columbia. He is a recipient of the NRC Meritorious Service Award and the Presidential Meritorious Executive Rank Award.

Mr. Mayfield is located in room T-10E11 and can be reached on 415-6690.

Farouk Eltawila, will be the Acting Director, Division of Systems Analysis and Regulatory Effectiveness. Dr. Eltawila began his career with the NRC in 1975 and has worked in more progressively senior positions dealing with regulation of licensed reactors. In 1990, Dr. Eltawila was appointed to the Senior Executive Service as the Chief of the Accident Evaluation Branch, Division of Systems Research, RES. In 1994, Dr. Eltawila became the Chief of the Reactor and Plant Systems Branch, Division of Systems Technology, RES. From 1999 to present, Dr. Eltawila served as the Chief of the Safety Margins and Systems Analysis Branch, Division of Regulatory Effectiveness and Systems Analysis, RES.

Dr. Eltawila received his Bachelor of Science from the University of Alexandria, Egypt, and a Ph.D. in Nuclear Engineering from Virginia Tech. in 1974.

Dr. Eltawila is located in room T-10E47 and can be reached on 415-5741.

John H. Flack, will be the Acting Chief of the Safety Margins and Systems Analysis Branch, Division of Systems Analysis and Regulatory Effectiveness. Since March 1999, Dr. Flack has been Assistant Branch Chief of the Regulatory Effectiveness Assessment and Human Factors Branch within the Division of Systems Analysis and Regulatory Effectiveness, RES. Dr. Flack was selected for the SES Candidate Development Program in October 1999.

Dr. Flack received degrees from several New York Colleges including the Academy of Aeronautic (A.A.S. in 1969), Richmond College (B.S. in 1972) and Queens College (M.A. in 1974). In 1982, Dr. Flack received a Ph.D. in Physics from the University of Hawaii.

Dr. Flack is located in room T-10F21 and can be reached on 415-6436.

/RA/
C. Thadani, Director
Office of Nuclear Regulatory Research

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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. 00-091 May 31, 2000

NRC RESEARCH EMPLOYEES RECOGNIZED FOR OUTSTANDING ENGINEERING EXPERTISE

Four Office of Regulatory Research employees at the U.S. Nuclear Regulatory Commission have won awards in recognition of their significant technical expertise and career accomplishments.

Two of the employees, Dr. James Costello of the Engineering Research Applications Branch and Dr. Joe Muscara of the Materials Engineering Branch, recently were presented with "Best Technical Paper Awards" at the Eighth International Conference On Nuclear Engineering (ICONE-8) in Baltimore, Md.

The industry group cited Dr. Costello for his technical paper, "Predictability of Steel Containment Response Near Failure." Dr. Muscara was honored for his work on the effects of light water coolant environments on fatigue crack initiation in piping and pressure vessel steels. The conference was sponsored by the American Society of Mechanical Engineers, the Japan Society of Mechanical Engineers and the Societe Francaise de'Energie Nucleaire.

In addition, Dr. Harold Ornstein of the agency's Regulatory Effectiveness Assessment and Human Factors Branch received the City College of New York Engineering School Alumni Association's annual "Career Achievement Award." The honor is presented to a graduate of the school "whose career and achievements are a source of pride to Alma Mater and to his/her fellow alumni." Dr. Ornstein, who accepted the award at a recent ceremony in New York City, has been employed by the NRC for 25 years.

Also, Steven Arndt of the Safety Margins and Systems Analysis Branch was named a Fellow of The Ohio Academy of Science, a honor bestowed upon individuals who have made extensive, productive, scientific, technological or educational contributions to society.

"We are delighted and proud of the major contributions of our colleagues," said Ashok Thadani, Director of the NRC's Office of Nuclear Regulatory Research. "We welcome the well-deserved recognition of their hard work, technical expertise and career accomplishments by these prestigious outside organizations, and share in the celebration of their efforts."

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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. 00-086 May 25, 2000

Senior Managers Brief Commission on Nuclear Power Plant Performance

Senior managers of the Nuclear Regulatory Commission briefed the Commission today on the results of their May 10-11 meeting to review the safety performance of nuclear power plants.

The Commission concurred in the following recommendations made by NRC senior managers:

Three plants will be the subject of agency focus. They are:

- D.C. Cook Units 1 and 2, operated by American Electric Power Company, near Benton Harbor, Michigan. The licensee has made significant progress and has resolved the majority of the technical issues related to a prolonged shutdown. It has also made improvements to its corrective action program and the design control process. However, because all corrective actions necessary for restart and safe plant operation have not been demonstrated, D.C. Cook continues to warrant oversight as an agency focus plant.
- Indian Point 2, operated by the Consolidated Edison Co., near Buchanan, New York. NRC senior managers discussed plant performance associated with two recent events, an August 1999 reactor trip with electrical system complications and a steam generator tube failure in February that led to the declaration of an Alert. These events revealed several interrelated problems: (1) communication and coordination weaknesses; (2) engineering support shortcomings; (3) configuration management/control problems; (4) equipment reliability problems and large corrective action backlogs; and (5) operator knowledge, station training, and procedural weaknesses. Senior managers noted that the current Chief Nuclear Officer has set high standards and improvements are underway, but determined that Indian Point 2 warrants oversight as an agency focus plant.

Three plants discussed at last year's senior management meeting will receive routine oversight. They are:

Millstone Units 2 and 3, operated by Northeast Utilities of Waterford, Connecticut, and Clinton, operated by Illinois Power Company, near Clinton, Illinois. Millstone Unit 3 and Clinton were identified as requiring regional focus last year -- calling for special attention from the appropriate Regional Administrator. Millstone 2 was identified at last year's senior management meeting as requiring agency focus -- calling for the attention of the Executive Director for Operations and/or the Commission.

All the other nuclear power plants will have routine oversight.

No nuclear fuel facilities warranted discussion at the public meeting.

Texts of letters to utilities with plants warranting agency focus will be available at:

http://www.nrc.gov/OPA/assessment.htm on the NRC Internet home page.

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs **Telephone: 301/415-8200**
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No. 00-84 May 23, 2000

NRC Renews Licenses for Oconee Nuclear Power Plant for an Additional 20 Years

The Nuclear Regulatory Commission has renewed the operating licenses for the three units of the Oconee Nuclear Station for an additional 20 years, the second license extension granted to a commercial nuclear power plant.

The Commission unanimously approved the extension of the licenses following a May 2 briefing by the NRC technical staff and a careful review of staff recommendations.

Duke Energy Corporation submitted an application to the NRC in July 1998 to renew the licenses for Oconee Units 1, 2 and 3. The Oconee Nuclear Station is located near Seneca, South Carolina. The current licenses expire on February 6, 2013, for Unit 1; October 6, 2013, for Unit 2; and July 19, 2014, for Unit 3. The NRC conducted an extensive review of the license renewal application in accordance with Parts 51 and 54 of Title 10 of the Code of Federal Regulations.

The NRC's environmental review, under Part 51, is described in a site-specific supplement to the NRC's "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," (NUREG-1437, Supplement 2). In this Final Environmental Impact Statement, issued in December, the staff concluded there were no impacts that would preclude renewal of the licenses for environmental reasons.

In the "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2 and 3," (NUREG-1723) issued in March, the staff concluded that there were no safety concerns that would preclude renewal of the licenses, because the licensee had demonstrated the capability to manage the effects of plant aging.

In addition, the NRC conducted three inspections of the plant to verify information submitted by the licensee.

On March 13, the NRC's Advisory Committee on Reactor Safeguards -- an independent body of technical experts which advises the Commission -- issued its recommendation that the operating licenses for Oconee be renewed. That recommendation is contained in the "Report on the Safety Aspects of the License Renewal Application for Oconee Nuclear Station, Units 1, 2 and 3."

Copies of these documents and others relating to the license renewal are available at: http://www.nrc.gov/OPA/reports/renewal.htm on the NRC's web site. A copy of the staff's recommendation on the renewal of the Oconee licenses, which contains the license conditions for Oconee, is available in the NRC Public Document Room at 2120 L Street, N.W., Washington, D.C. 20555; telephone (202) 634-3273 and has been posted at the same web site.

The NRC is currently reviewing license renewal applications for two other operating nuclear power facilities: Arkansas Nuclear One, Unit 1, operated by Entergy Operations, Inc., near Russelville,

Arkansas; and Hatch Units 1 and 2, operated by the Southern Nuclear Operating Company, near Baxley, Georgia.

The agency renewed the operating licenses for both units of the Calvert Cliffs Nuclear Power Plant for an additional 20 years on March 23.

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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. 00-092 May 31, 2000

NRC Revises Its Regulations for Fire Barrier Penetration Seals

The Nuclear Regulatory Commission is revising its regulations to modify the requirement for fire barrier penetration seal material for nuclear power plants because the present requirement makes a negligible contribution to safety.

As part of the NRC's defense-in-depth philosophy for protecting nuclear power plants through the use of multiple safety features, the plants are divided into separate areas by fire-rated barriers designed to contain a fire and prevent its spread to other areas. Various kinds of materials, some silicone-based, are used to seal openings between fire barriers. The seals provide reasonable assurance that a fire will be confined to the area in which it starts.

NRC has determined that if fire barrier penetration seals are properly designed, tested, configured, installed, inspected and maintained, they are capable of preventing the spread of fire for one, two or three hours, providing sufficient time for automatic systems or firefighters to control and extinguish the fire. Therefore, the NRC's technical staff and Advisory Committee on Reactor Safeguards -- an independent body of technical experts that provides guidance to the Commission -- agreed that the non-combustibility requirement in Title 10 of the Code of Federal Regulations, Part 50, Appendix R, be eliminated because it has a negligible contribution to safety.

The proposed rule was published in the August 18 edition of the *Federal Register* for public comment. Eight letters were received and the NRC staff reflected some of the suggestions in the final rule.

The new rule will become effective 90 days following publication in an upcoming edition of the *Federal Register*. A copy of the rule will be posted at: http://www.nrc.gov/NRC.rule.html on the NRC homepage.

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[**NRC Home Page** | **News and Information** | **E-mail**]

NRC MAY QUEUE PLANTS FOR LICENSE RENEWAL IF INDUSTRY DOESN'T

NRC's license renewal chief says the agency could be forced to push younger plants to the back of the review line if the industry does not set up some order for the anticipated wave of license extension applications.

Christopher Grimes, chief of the NRC license renewal and standardization branch, said the agency's "most serious challenge" is keeping track of the target date of application submittals in order to budget accordingly.

The idea that the agency would queue the applicants is unappealing to the industry, Grimes said, yet the industry is unwilling to manage the demand. "They want instant gratification," he said.

At a May 17 meeting, Grimes said the agency was "on the verge of critical budget decisions" for fiscal 2001 and FY-02. The agency has budgeted for four applications in each of those years. While those assumptions appear to be reasonably accurate at the moment, the agency could be hit with requests to review as many as 10 units between June and September 2001, according to one industry official. Meanwhile, NRC's projections of eight applications each in FY-03 and FY-04 now seems to be a gross overestimate.

The result is that the agency could be slammed with requests in some years and nearly idle in others. Grimes said the agency might start regularly polling companies by phone to get a better idea of when applications are expected.

One recent change came last week when Entergy said it would move up its submittal date for Peach Bottom from December 2001 to July 2001. The five-month advance effectively changes the fiscal year that NRC staff begin reviewing the application.

Mike Tuckman, executive vice president for nuclear generation at Duke Energy Co., said that companies want to come in early with applications so they can satisfy their "economic considerations" relating to capital investments, staffing and planning. He said some hesitated, though, because they initially were worried about NRC making a timely decision. But by the time the staff finishes reviewing the second round of applicants—Arkansas Nuclear One-1 and Hatch—and the ones scheduled in FY-01, the industry should have more confidence, Tuckman said.

Tuckman said the Nuclear Energy Institute would start surveying the industry every four months on license renewal plans. "But we felt we can't regulate the flow," he said.

NRC staff plans to issue for public comment in August three documents important to license renewal: the draft generic aging lessons learned (GALL), standard review plan (SRP), and revised regulatory guide (NEI 95-10). The commission is tentatively expected to approve GALL and the SRP in March 2001.

—Jenny Weil, Washington (jenny_weil@mgh.com)

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Contacts: Jenny Weil, Washington (jenny_weil@mgh.com)

NEI PREFERS AD HOC WORKING RELATIONSHIP, OPPOSES FORMALIZING IT

If it ain't broke, don't fix it. That's the position the Nuclear Energy Institute (NEI) appears to have adopted with regard to industry-NRC interactions.

The nuclear power industry's Washington lobby opposes NRC staff plans to develop guidelines for governing increasingly cooperative efforts between the federal regulator and the regulated community to resolve technical and potential safety issues.

"We acknowledge that the process we have been using is ad hoc, but it's been effective," Alex Marion, NEI's director for programs, told Inside N.R.C. May 16. Marion said every issue dealt with in NRC-industry discussions is unique, making it very difficult to come up with a generic set of procedures to follow. Marion said NEI thinks there would be no benefit, and potential drawbacks, to formalizing the process.

"The first and most important step" when the NRC has a concern "is to meet with the industry," Marion stressed.

Nonetheless, a staff paper containing proposed guidelines is due to commissioners this week and should be released publicly soon. The commissioners have approved, and even encouraged, increased staff cooperation with industry in tackling technical issues.

The agency's generic communication process has been completely revamped—and the number of information notices, generic letters and bulletins dramatically reduced—as the agency has chosen instead to work with industry up front. That cooperation has often led to either the adoption by industry of voluntary initiatives to address the safety concern or the elimination of the concern through better understanding of its technical merits. In either case, the cooperation has obviated the need for NRC to issue generic communications and it has led to an increasing reliance on voluntary industry programs (INRC, 7 June '99, 1). In just one significant example, the commissioners decided not to adopt a shutdown rule and told staff instead to monitor industry's voluntary adoption of NEI guidelines on safe shutdown practices.

But the ad hoc relationship has generated concern by some public participants in NRC regulatory activities

and even controversy. When the NRC staff was drafting a commission paper on reforming generic communications to make them more in accord with industry wishes, the agency's Inspector General (IG) found that NRC staff gave a copy of the paper to NEI two weeks before it was available to anyone else in the public, prompting charges of blatant favoritism or at least unseemly coziness with industry. The IG also found that then-Chairman Greta Dicus misled Rep. Edward Markey (D-Mass.) by telling him that the document was released simultaneously to both NEI and the public when it had not been (INRC, 13 March, 6).

In any case, the commission tasked the staff with developing guidelines for using voluntary industry initiatives in an August 1999 staff requirements memo.

The staff has been working on the guidelines, but has gotten very little public input in general and little support from NEI in particular. Despite repeated pleas for comments, including letters and telephone calls to NEI, the group declined to provide any substantive comments to NRC on the proposed guidelines. NEI has met with the NRC staff to discuss the subject—mostly to convey its opposition.

NEI, NRC Staff Debate Public Participation

An admittedly "frustrated" Brian Sheron, associate director for project licensing and technical analysis in the reactor regulation office, told NEI at a Feb. 17 meeting that he agreed the ad hoc process has worked to date. However, he said, the agency has an obligation to its other "stakeholders" to have a process that is understandable and accessible to all parties, not just the regulated community. Moreover, he pointed out that the NRC has already eliminated some generic communications through the use of voluntary industry initiatives. "I don't know what you guys want. You don't like generic letters. We said, 'fine.' We will give you the opportunity to take these issues on and come back to us and tell us how you think they should be dealt with." Despite that offer, industry was opposing formalizing a process for doing that, Sheron said.

NEI General Counsel Bob Bishop took a legalistic view of the matter at the February meeting and told Sheron that NRC, through a variety of measures, frequently goes beyond its legal requirements for including the public in its decision-making processes. While it is within NRC's purview to decide how to carry out its mandates, strictly speaking, many of its activities to include the public are not required, Bishop said. "I think that is something we kind of need to keep in mind, that all of these things are your decision on how you can best satisfy your mandate. They are not required by law," Bishop said. "I'm hard pressed to think that it's a wise use of resources to say in every case...this is the process we are going to follow....The ad hoc may not be neat, but it seems to be working," Bishop said.

"It think your responsibility is not to make sure 280-million (Americans) vote in favor of anything; it is to figure out what's the right balance of interests here and the need to solve problems," Bishop said.

"We are not trying to get people to love nuclear power," Sheron replied. "We want the public to be able to understand the way this agency operates and how our decisions are made. In other words, we want to operate in a transparent way. You can't really operate in a transparent way if we continue on an ad hoc basis," Sheron said. Otherwise, Sheron said, the perception is one of collusion. "There they go again. They are off meeting. They are off figuring out how...they are going to get this one off the books and the like."

Bishop replied that he didn't believe the public had that perception. Most Americans are "not concerned," he said. Those who are interested in NRC activities, such as public interest groups including the Union of Concerned Scientists and others, understand the ad hoc process, Bishop said. "Somebody who doesn't understand the process might conclude" that NRC is colluding with industry, "but I'm not sure that's a minority that is worth spending a whole lot of resources to satisfy."

But Sheron said there are other reasons to have a formal process. He noted that NRC got frustrated waiting for NEI to pick up the ball on a concern about "small bore piping. It took about five months before you guys

ACRS APPEARS SUPPORTIVE OF RISK-INFORMING TWO TECH SPEC CHANGES

Advisory Committee on Reactor Safeguards (ACRS) members stated general agreement with industry and the NRC staff that risk-informing two technical specification items—missed surveillances and upward power mode changes when restrictions on the next mode would be in place due to equipment problems—would present very little change in plant risk and should probably proceed apace.

The tech spec changes, known as Initiatives 2 and 3 of seven initial planned items in the industry-NRC, risk-informed, tech spec effort, are considered "low-hanging fruit," said Nuclear Energy Institute's (NEI) Biff Bradley. Nonetheless, they are the first to go beyond mere extensions of allowed outage times (AOTs) on equipment out of service, which licensees have been getting based on risk considerations. Therefore, there was interest in them by ACRS, if only for the potential precedent-setting nature of the changes and out of concern for keeping an eye on eroding safety margins, various ACRS members said at a May 11 meeting.

Under Initiative 2, which addresses missed surveillances, a licensee that discovers it has returned to power operation and missed a surveillance would perform a risk analysis to ascertain whether the risk is higher for operating with the missed surveillance or for putting the plant through another shutdown and start-up cycle. Plant risk at shutdown can spike very high for short periods of time.

Risk, in this context, means evaluating the likelihood or probability of a core damage accident, or similar unwanted end state, through examination of complex statistical analyses, which are becoming increasingly routine. Bradley said the risk analyses done in these cases would be "fairly simple."

Bradley said there was "no intention" under Initiative 2 to allow operators to "willfully miss" a surveillance. The risk-informed tech spec would allow the possibility for continuing to operate, after the risk evaluation, while current tech specs require licensees, depending on the equipment involved, to immediately shut down or seek a notice of enforcement discretion from NRC to continue to operate.

NRC staff told ACRS any missed surveillances would still be "reportable" events. Missed surveillances would then fall under the licensees' corrective action program, a program subject to an annual NRC inspection. "We're well aware the oversight program has to mesh with this," said Mark Reinhart, a probabilistic safety assessment branch staffer in the Office of Nuclear Reactor Regulation.

Initiative 3 would risk-inform tech spec 3.0.4, which precludes moving up in power modes when a limiting condition for operation (LCO) would exist in that mode because of equipment problems. NEI's Bradley said each of the vendor owner's groups is developing risk models to use to decide when mode changes could be made despite an LCO on certain equipment. The equipment would still have to be returned to service within the AOT, but the mode change wouldn't have to wait until that work was completed, under the proposal.

—David Stellfox, Washington (dstel@mh.com)

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ACRS REVIEWS CRITICIZED BY INDUSTRY AS TOO COSTLY, OF QUESTIONABLE VALUE

Industry officials are taking aim at the NRC Advisory Committee on Reactor Safeguards (ACRS), criticizing the cost and extent of committee reviews.

At two separate meetings last month, industry representatives complained about the burden of ACRS reviews.

The ACRS' billing practice was one complaint brought up at an April 26 Westinghouse Owners Group (WOG) meeting with NRC senior management. Andy Drake, project manager for the Westinghouse Owners Group (WOG), said the group received a large, and unexpected, bill for an ACRS review on a WOG topical report.

"Nobody else ever got one. I checked with the other (owners groups) and they hadn't gotten one," Drake said.

"They asked us to come in," he said. "Three months later we got a rather sizable bill, and we didn't think ACRS was worth that much, and we had to go figure out how to pay for it."

But Douglas Weiss of NRC's Office of the Chief Financial Officer (CFO) asserted that the NRC has been billing for ACRS since 1985.

John Larkins, executive director for the ACRS, said the staff might not have been breaking out the individual office charges on past bills, which could explain the WOG's surprise."Before it was an aggregate cost," he said. More recently, he said, the CFO has been pushing to break out specific unit costs.

At another meeting held the same day between Nuclear Energy Institute (NEI) and NRC senior managers, the question was raised about ACRS reviews on matters unrelated to safety standards.

Tony Pietrangelo, NEI's licensing director, asserted that the regulatory guide on 10 CFR 50.59 could be completed much faster if the ACRS did not have to review the guide again after the comment period closes. "There are no major issues left," he said. "We question going back to the ACRS."

Pietrangelo said the remaining issues have nothing to do with safety standards but rather are more of a "regulatory threshold" issue. The ACRS reviews "add time" to the process, he stressed.

"We seem to spend an inordinate amount of time" preparing for ACRS meetings, Pietrangelo said. "It's

getting more and more difficult to support these meetings. It might be time to reexamine when the ACRS is needed."

NRC Office of Nuclear Reactor Regulation Director Sam Collins agreed to "carry that question back" to the agency.

While the industry might be taking a critical look at the necessity of ACRS reviews, the advisory committee's contributions have not gone unnoticed within the agency. NRC Commissioner Edward McGaffigan called the ACRS an "important part" of the license renewal reviews. He scolded the staff at a May 2 meeting for failing to include a summary of the ACRS findings specific to the Oconee review and several generic issues relating to the management of aging systems, structures, and components.

Christopher Grimes, chief of the license renewal and standardization branch, agreed that the ACRS has added "substantial" value to the generic and plant-specific license renewal reviews. He also said he believes the ACRS "is more in tune with their purpose and role" than a decade or so ago.

For the moment, the ACRS is helping the staff to develop a standardized license renewal process. It will be closely involved in the review of plant-specific safety evaluation reports during the so-called initial application phase, a period expected to extend until the agency receives the first application for each of the four vendor designs. The ACRS also has been asked to evaluate the effectiveness of the license renewal process and to look at any issues raised during the review process that merit special attention.

In addition, the ACRS will aid staff in resolving generic safety issues and owners group license renewal topical reports. It also will be reviewing the Generic Aging Lessons Learned report, the standard review plan, the regulatory guide and the industry implementation guidelines (NEI 95-10)—collectively called the "implementation guidance"—before the staff finalizes each document.

Larkins said the ACRS is contemplating drafting a lessons-learned report. And the ACRS has developed a plan to streamline its review from four planned subcommittee and full committee meetings down to two for each application. The move from a four-step to a two-step process is expected to occur following the issuance of the "implementation guidance"—anticipated in 2002.

Larkins acknowledged the ACRS was involved in an unusually large number of reviews last year, but most were at the behest of the commission for work related to the long- and short-term goals on the agency's tracking list it now sends monthly to Congress.

He estimates the committee was doing 40%-50% more reviews than in past years because of the tracking list. "We're trying to get back to a more realistic schedule," he said.

—Jenny Weil (jenny_weil@mgh.com) and David Stellfox (dstel@mh.com), Washington

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs **Telephone: 301/415-8200**
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No. 00-81 May 12, 2000

NRC Schedules Washington, D.C. Workshop on Fuel Facility Oversight Program

The NRC has scheduled a workshop May 24-25 in Washington, D.C., to obtain suggestions from the public on revising the agency's oversight program for nuclear fuel cycle facilities.

This workshop is one of several to be held over a period of two years to factor public and other stakeholder comments into the revised fuel facility oversight program. At last month's meeting, cornerstones for meeting the NRC safety and national security missions were discussed.

This month's workshop will focus on:

- A proposed communication plan for informing and obtaining views of stakeholders about revision of the oversight program;
- The objective and scope of safety and national security-related cornerstones for meeting the NRC's mission;
- Key performance attributes for achieving each cornerstone;
- Licensee performance attributes the NRC needs to monitor and risk-inform inspections to ensure cornerstone objectives are met;
- Criteria for selection of performance indicators;
- Measurable parameters and methods for performance indicators; and
- Thresholds for performance.

The goals of the new initiative are to focus oversight on activities where the potential risks are greatest, obtain more objective indicators of risk-related performance, increase public confidence in the NRC's oversight program, and increase regulatory effectiveness, efficiency and realism. This initiative will employ lessons learned from the recently revised commercial nuclear reactor oversight program.

The nuclear fuel cycle begins with the milling of uranium ore to produce uranium concentrate called "yellowcake." The yellowcake is converted into uranium hexafluoride gas at a special facility and loaded into cylinders. The cylinders are sent to a gaseous diffusion plant, where uranium is enriched for use as reactor fuel. The enriched uranium is then converted into oxide powder, fabricated into fuel pellets, loaded into fuel rods, and bundled into reactor fuel assemblies at a fuel fabrication facility. Assemblies are then transported to nuclear power plants, non-power research reactor facilities, and naval propulsion reactors for use as fuel.

The NRC currently inspects these fuel facilities several times a year in a variety of technical areas, such as chemical process safety, fire protection, nuclear criticality safety, radiation safety, and nuclear material safeguards. Results of these NRC inspections are available to the public.

The workshop, which is open to the public, will be held from 9:00 a.m. until 5:00 p.m. both days at the Nuclear Energy Institute, located at 1776 I Street, N.W. (Republic Place). Visitor parking is limited; however, the building can be reached via the Red Line Metro to the Farragut North station.

Those who seek background information on this initiative may obtain transcripts of past meetings at http://www.nrc.gov/NMSS/FCSS/FCOB/INSP/REVISED/fcindex.htm on the NRC's web site.

Interested persons can also access a related NRC paper, SECY 99-188, "Evaluation and Proposed Revision of the Nuclear Fuel Cycle Safety Inspection Program," from the agency's web site, at http://www.nrc.gov/NRC/COMMISSION/SECYS/index.html or from the Public Document Room, telephone 202-634-3273.

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2000 Utility W[illegible]ing Conference
August 6-10, 2000
Amelia Island Plantation, Amelia Island, Florida

SUNDAY, AUGUST 6	**Meeting Registration**	3:00 - 7:00 p.m.	• *Amelia Foyer*
	Opening Reception and Dinner	6:00 - 9:00 p.m.	• *Amelia 3 & 4*

MONDAY, AUGUST 7	**Meeting Registration**	7:00 a.m. - 4:30 p.m.	• *Amelia Foyer*

8:30 a.m.

Plenary Session • *Amelia 4*

"MANAGING THE BUSINESS OF NUCLEAR POWER"

Introduction

Leon Oliver, Senior Vice President and Chief Nuclear Officer, Northeast Utilities

Speakers

Richard Meserve, Chairman, U.S. Nuclear Regulatory Commission
William Magwood IV, Director, Office of Nuclear Energy, Science and Technology, U.S. Department of Energy
Jerry Yelverton, President and Chief Executive Officer, Entergy Nuclear
James O'Hanlon, President and Chief Executive Officer, Virginia Power
Edward J. Tirello, Jr., Managing Director, Deutsche Banc Alex Brown
Lucian Conway, President, Conway Consulting

Awards Luncheon 12:00 p.m. - 1:30 p.m. • *Amelia 1*

1:30 p.m.

Regulatory Relations	Supply Chain	Maintenance	Executive Focus-Information Technology	Business	Engineering
"Reactor Oversight Program"	"Bringing the Bookends Together: Opportunities not Obstacles with the Strategy Behind Merging Independent Supply Chains of Two Generating Giants"	"Human Performance"	"Maximizing: Efficiency and Flexibility in a Deregulated Environment"	"Mergers Acquisitions, and Alliances"	"Control Room Habitability"
Amelia 2 & 3	*Amelia 4*	*Cumberland C*	*Talbot*	*Sapelo*	*Ossabaw*

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TUESDAY, AUGUST 8

Meeting Registration 7:00 a.m. - 4:30 p.m. • *Amelia Foyer*

8:30 a.m.	**Regulatory Relations** Risk-Informed "Regulation" *Amelia 2 & 3*	**Executive Focus-Information Technology** "Building Solutions" *Talbot*	**Supply Chain & Engineering** "Reverse Engineering and Products Standardization" *Amelia 4*	**Business** "Business Aspects of License Renewal" *Sapelo*	**Maintenance** "Outage Preparation and the Work Management Interface with Maintenance" *Cumberland C*

Luncheon - Vendor Displays 11:30 a.m. - 1:30 p.m. • Amelia 1

1:30 p.m.	**Regulatory Relations** "Licensee Renewal" *Amelia 2 & 3*	**Business** "Work Talent Retention, Replenishment and Renewal" *Sapelo*	**Supply Chain** "Performance Indicators for Nuclear Supply Chain" *Amelia 4*	**Engineering** "50.59 Rule Change Effect on Engineering" *Ossabaw*	**Operations** "On the Job Training: Status, Best Practices, & Future Development" *Talbot*	**Maintenance** "Maintenance Roundtable" *Cumberland C*

WEDNESDAY, AUGUST 9

Meeting Registration 7:00 a.m. - 11:00 a.m. • *Amelia Foyer*

8:30 a.m.	**Regulatory Relations** "Maintenance Rule" *Amelia 2 & 3*	**Operations** "Crew Behaviors: Is It Really a Team?" *Talbot*	**Supply Chain** "Methods of Reducing the Total Cost of Ownership in the Nuclear Environment" *Amelia 4*	**Engineering** "Non-Traditional Methods of Accomplishing Engineering Work" *Ossabaw*	**Business** "Business Impacts of Deregulation" *Sapelo*	**Maintenance** "Maintenance Standards" *Cumberland C*

Wrap-Up Luncheon 11:30 a.m. - 1:30 p.m. • Amelia 1

ANS PROFESSIONAL DEVELOPMENT WORKSHOP "MAINTENANCE RULE AND CONDITION MONITORING"

THURSDAY, AUGUST 10, 2000
8:30 a.m. - 4:30 p.m. • Ossabaw Room

Workshop Organizer: **Carolyn Heising, PhD,** Professor of Industrial, Mechanical and Nuclear Engineering, Iowa State University of Science and Technology, Office of Nuclear Energy, U.S. Department of Energy (1999-2000)

8:00 a.m. Introduction: "Importance of Maintenance and the NRC Maintenance Rule to NPPs"

Introductory Remarks: Carolyn Heising, PhD, Professor of Industrial, Mechanical and Nuclear Engineering, Iowa State University of Science and Technology, Office of Nuclear Energy, U.S. Department of Energy (1999-2000)

8:30 a.m. "The Maintenance Rule: An NRC Perspective", Richard Correia, U.S. Nuclear Regulatory Commission

9:30 a.m. "The Maintenance Rule: A Legal Perspective", Sheldon Trubatch and James Curtiss, Counsels, Winston & Strawn

1[illegible].m. Coffee Break

10:45 a.m. "The Maintenance Rule: Industry Perspective and Approaches to Meeting the New 9 (4)" Tony Pietrangelo, Director of Licensing, Nuclear Energy Institute

11:30 a.m. "The Maintenance Rule and Condition Monitoring in the Nuclear Industry", Carolyn Heising, PhD, Professor of Industrial, Mechanical and Nuclear Engineering, Iowa State University of Science and Technology, Office of Nuclear Energy, U.S. Department of Energy (1999-2000)

12:00 p.m. Group Luncheon

Samuel Collins, Director, Office of Nuclear Reactor Regulations, U.S. Nuclear Regulatory Commission

1:00 p.m. "Condition Monitoring Applications in Support of the Maintenance Rule at Ft. Calhoun NPP", Sumeet Sarma and Ashok Kumar, Iowa State University and Daniel Bye and Joe Gasper, PhD, Omaha Public Power District, Ft. Calhoun NPP

3:00 p.m. Coffee Break

3:15 p.m. "Virtual Reality Models of NPPS: An Advanced Tool for Predictive Maintenance", Mark Bryden, PhD, Assistant Professor, Mechanical Engineering Department, Iowa State University

4:00 p.m. Panel Discussion (All Speakers)

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"Managing the Business of Nuclear Power"
August 6-10, 2000
Amelia Island Plantation
Amelia Island, FL

MEETING REGISTRATION FORM

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UTILITY WORKING CONFERENCE REGISTRATION

(Includes one (1) ticket to the following events: Sunday Evening Dinner/Reception; Monday, Tuesday and Wednesday Luncheons and a copy of the Conference Proceedings)

	ANS Member	Non-Member
Pre-Registration (Before July 21, 2000) Full meeting registration *2000 Utility Working Conference ONLY*	(01) ❒ $470	(02) ❒ $570
On-Site Registration (or After July 21, 2000) Full meeting registration *2000 Utility Working Conference ONLY*	(03) ❒ $570	(04) ❒ $670
		ANS Mbr / Non Mbr
ANS Professional Development Workshop ***"Maintenance Rule and Condition Monitoring"*** **Thursday, August 10th** *(includes Thursday lunch and meeting materials)*		(05) ❒ $150

Additional Tickets	**#/Tickets**	**Total**
Sunday Evening Dinner/Reception	____ (21) × $60 =	$ _____
Monday Luncheon	____ (22) × $40 =	$ _____
Tuesday Luncheon	____ (23) × $40 =	$ _____
Wednesday Luncheon	____ (24) × $40 =	$ _____
Thursday Luncheon–*Professional Development Workshop*	____ (25) × $40 =	$ _____
Meeting and Workshop Subtotal:		$ _____
Additional Ticket(s) Subtotal:		$ _____

Grand Total

$ _____

Additional tickets available at registration for Sunday Dinner/Reception ($60 each) and luncheons ($40 each). Please indicate if you would like to purchase additional tickets in advance. Cost of extra tickets will be added to the total meeting price.

Mail payment, payable to the American Nuclear Society and the conference registration form to **ANS Registrar, American Nuclear Society, P.O. Box 97781, Chicago, IL 60678-7781.** If you are using a major credit card (VISA, Mastercard, American Express, or Diners Club), you may fax your registration to the ANS Registrar at 708/579-8314 (ANS telephone number: 708/579-8316).

CANCELLATIONS: Registrations **canceled prior to July 21, 2000,** will be refunded minus a $75 processing fee. Cancellations **received after July 21, 2000,** will NOT be refunded. You may send a substitute.

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2000 ANS Utility Working Conference
August 6-10, 2000

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1 Bedroom Suite:	$163 per night	_____	1 Bedroom Suite:	$123 per night	_____
2 Bedroom Villa:	$277 per night	_____	2 Bedroom Villa:	$174 per night	_____
3 Bedroom Villa:	Available on Request	_____	3 Bedroom Villa:	$240 per night	_____

Arrival Date: _____ (check in time: 4:00 p.m.)	**Departure Date:** _____ (check-out time 12 noon)

Payment:
❒ Check ❒ VISA ❒ MasterCard ❒ Discover ❒ American Express

Credit Card # _____ Expiration Date _____
Credit Card Holder Signature _____

A deposit paid hereunder shall be refunded if notice of cancellation is received by the Group Reservations Department (10) ten days prior to the arrival date.

Amount of Deposit (one night's rate for each Unit) $_____

Visit the Amelia Island Plantation website at www.aipfl.com

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

Christopher Gratton
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

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GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Purpose of the Presentation
- Background - Identification of Issues
- Evaluation of Issues
- Follow Up Actions
- Closing GSI-173A

GSI-173A

Purpose of the Presentation

- Review the Status of Spent Fuel Storage Issues for Operating Plants
- Obtain Agreement that GSI-173A Should Be Closed

GSI-173A

Background

- Initial Issue Identified
  - Susquehanna 10 CFR Part 21 Report
- Generic Spent Fuel Storage Pool Action Plan
  - Technical Review Identified and Evaluated Design Features Regarding Spent Fuel Storage
- AEOD Study on Spent Fuel Pool Cooling
  - Evaluated Likelihood and Consequence of Extended Cooling Loss

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Technical Review Focused on SFP Design Features and Safety Functions
- Coolant Inventory
  - Lacks Passive Anti-siphon Protection
  - Potential for Draining Through the Fuel Transfer System
  - Potential for Draining Through Interfacing Systems
  - Absence of a Direct Low Level Alarm
  - Lacks Liner Leakoff Isolation

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Coolant Temperature
  - Structural Intergrity and Leak Tightness Under Normal, Accident and Environmental Loading
  - Coolant Purification Cannot Be Performed at Elevated SFP Temperatures
  - Environmental Effects of High SFP Temperature - Multi-Unit Issues
  - Cooling System Reliability and Capability

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Fuel Reactivity
  - Solid Neutron Absorbers
  - Soluable Boron

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Results of the Technical Review:
  - Existing Facilities Meet Regulations
  - Follow Up on Certain Plant Specific Design Features
  - Other Planned Actions
- ACRS Presentation - August 9, 1996
  - Presented the Results of the Review of Spent Fuel Storage Issues to Full Committee
  - Committee Satisfied with Staff Actions
  - Letter Not Deemed Necessary at the Time

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Follow Up Actions on Plant Specific Design Features
  - Group 1: Plant Specific Regulatory Analyses (7)
    - Antisiphon
    - Transfer Tube in the SFP
    - Piping Entering SFP Below Fuel
    - Limited Level Instrumentation
    - Shared Systems and Structures at Multi-Unit Sites
    - Absence of On-site Power for SFP Cooling Systems
    - Limited Instrumentation for Loss of Cooling

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Follow Up Actions on Plant Specific Design Features (Cont.)
  - Group 2: Evaluate Additional Information (4)
    - Absence of Leak Detection/Isolation for Liner
    - Limited DHR Capability
    - Infrequently Used Backup Systems
    - Refueling Cavity Seal

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Overall Results of Plant Specific Follow Up Activities:
  - Group 1: Design Features Requiring Regulatory Analysis
    - Five design features did not meet screening criteria
    - One design feature closed by voluntary actions
    - One design feature exceeded minimum screening criteria and was subject to further evaluation
  - Group 2: Design Features Requiring Further Evaluation
    - Based on the review of additional information, the staff determined further regulatory analysis not warranted for these four design features

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Plant Specific Regulatory Analysis Results - Design Features Meeting Screening Criteria
  - Draining Through the Fuel Transfer System
  - Draining Through Interfacing Systems
  - Absence of a Direct Low Level Alarm
  - Absence of On-site Power for SFP Cooling Systems
  - Limited Instrumentation for Loss of Cooling

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Plant Specific Regulatory Analysis Results - Design Feature Not Meeting Screening Criteria
  - Shared Systems and Structures at Multi-Unit Sites
    - 13 plants with this design feature
    - Staff visited at Hatch and Dresden
    - Evaluation expanded to include all plants for this issue
    - Onsite power for SFP cooling system - Low likelihood of sustained boiling
    - Staff concluded no further regulatory action warranted for this design feature

GSI-173A

Generic Spent Fuel Storage Pool
Part A: Operating Facilities

- Results of the Review of Additional Information for Plant Specific Design Features
  - Administrative Controls
  - Plant Specific System Design and Operation
  - Previous NRC Evaluations

GSI-173A

Summary - Generic Safety Issue 173A

- Evaluation of GSI 173A Technical Issues Concluded - Existing Facilities Are In Compliance with the Regulations
- Follow Up Actions on Plant Specific Design Features Identified No Additional Regulatory Actions
- Close GSI-173A

BRIEFING TO THE ACRS ON THE DRAFT REPORT - REGULATORY EFFECTIVENESS OF THE SBO RULE

BY
BILL RAUGHLEY
OFFICE OF REGULATORY RESEARCH

JUNE 7, 2000

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BACKGROUND

- Report will provide basis to respond to Commission
- SBO definition and risks
- SBO rule historical highlights
  - Regulatory requirements and guidance before the SBO rule
  - SBO rule evolution
  - SBO rule technical basis and regulatory analysis
- SBO rule, 10 CFR 50.63
  - SBO coping duration based on individual plant design factors
  - Procedures to cope with an SBO
  - Modifications, if any
- Related documents
  - Initially RG 1.155 (NSAC -108, NUMARC 87-00)
  - Subsequently maintenance rule documents from resolution of GSI B-56, "EDG Reliability"

ASSESSMENT

- **Regulatory Effectiveness**
  -A regulation is effective if expectations are being achieved
  -Regulation includes rule, regulatory guide, and inspection documents

- **Scope**
  -Is the SBO rule effective and if any areas need attention
  -Plant specific problems not addressed
  -GSI-23 on RCP seal failure previously addressed

- **Method**
  -Compared the expectations to the outcomes using objective measures in areas of risk, value-impact, coping time, and EDG reliability
  -Used operating experience for trends in LOOP frequency and duration

- **Data**
  -Expectations from NRC documents
  -Outcomes from NRC PRA/IPE databases, LERs, SBO safety evaluations, and NRC reliability studies

RESULTS

- **Industry Wide Risk Reduction in Mean SBO CDF**
  - -3.2E-05/RY reduction which is more than 2.6E-05/RY expected
  - -Plants that had the greatest vulnerability did the most
  - -Plant shutdown with power supply unavailability may increase risk

- **Value-Impact**
  - -Outcome is just within range of expected averted rem
  - -NRC underestimated value and impact from added power supplies

- **Minimum Acceptable Coping Time**

- **EDG Reliability**
  - - Based on unit average safety performance from DRAFT INEL 95/0035
    - 0.95 target reliabilities met with and without MOOS at power
    - 0.975 target reliabilities difficult considering MOOS at power
  - -Plant followup found differences in analytical and site EDG reliabilities
  - -Many licensees could demonstrate additional SBO CDF reductions

EDG PERFORMANCE BASIS

- **NUREG-1032**
  - -MOOS while at power and non-power small (0.006)
  - -Equipment and system boundary included load sequencer
  - -Used actual test/unplanned demands to count valid starts/load-runs
- **RG 1.155**
  - -Minimum individual target reliability of 0.95 or 0.975 excluding MOOS (0.007)
  - -Notes that MOOS must not be excessive to achieve EDG failure rates
  - -Endorses NUMARC 87-00-monitor EDG unavailability versus industry
- **RG 1.160, Rev 2**
  - -Use as a goal or performance criterion
    - RG 1.155 target reliabilities, or
    - IPE unavailability assumptions in comparison to industry, or
    - Maintenance preventable failures.
  - -Balance reliability and unavailability
  - -Endorses NUMARC 93-01- allows PRA/IPE EDG performance assumptions

EDG PERFORMANCE BASIS

- **RG 1.9, Rev 3**
  - -Valid starts/load runs includes conditional failures from maintenance
  - -System boundary used to count failures excludes load sequencer
- **SBO rule and maintenance rule inspection documents use NUMARC 87-00, Rev 1, Appendix D triggers values for assessing compliance to RG 1.155 target reliabilities**

INSIGHTS FROM OPERATING EXPERIENCE

- Modifications Due to the SBO rule have been used to provide for safe shutdown and economic benefit
- Generally favorable LOOP frequency and duration; EDG(s) worked when needed
- Provide additional defense in depth from changing offsite power trends due to deregulation
- Potential SBO Alternate ac power source unavailability

CONCLUSIONS

- SBO rule was effective and the costs were reasonable

- Opportunities to improve the clarity of regulatory documents
  -RG 1.155, RG 1.9, and RG 1.160 EDG reliability calculation terms revisions
  -Inspection documents
  -RG1.93

- Lesson learned- to extent that regulatory documents are revised to be risk-informed and performance based, need to ensure consistent interpretation and use of terms, goals, criteria, and measurements.

FOR COMMENT

PROPOSED ACRS ISSUES RELATED TO THE AP1000 DESIGN

1. Guidance for acceptable scaling methods, such as the Code Scaling , Applicability, and Uncertainty (CSAU) evaluation methodology, and for acceptable utilization of integral test data for the validation of computer codes should be developed.

2. Establish the scope of additional analyses needed for the SSAR Chapter 15 accidents. Revised codes used in the analyses may need to be revalidated.

3. Clear identification in the SSAR is needed of the inadequacies in the NOTRUMP code and the steps taken to compensate for them. A rigorous demonstration of the applicability of the revised NOTRUMP code to the AP1000 design is needed.

4. More experiments or analyses will be required before in-vessel core debris retention can be credited as part of the licensing basis.

5. Since in-vessel retention is widely considered to be an important accident management strategy for operating reactors, the impact of intermetalic exothermic reactions on this strategy should be reassessed.

6. Westinghouse has chosen the thermal-hydraulic conditions of a specific sequence (i.e., a direct vessel injection line break) for use with the DBA source term to take credit for diffusiophoresis and thermophoresis. It is not clear that the thermal-hydraulic conditions of the selected sequence is consistent with the desired generality of the source term.

7. The applicability of the AP600 probabilistic risk analysis to the AP1000 must be demonstrated. Consideration should be given to the following:

   a. In-containment aerosol behavior and the effects of particle charging on aerosol behavior,

   b. Catastrophic failure of the steel shell containment,

   c. Containment bypass accident sequences and especially mitigation of steam generator tube rupture accidents,

   d. Reactor coolant system depressurization reliability, and the

   e. Efficacy and reliability of external cooling of the containment shell,

PLEASE PROVIDE YOUR COMMENTS TO NOEL OR SAM.

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- Draft -

Possible Uses and Limits of Risk Assessment in the Nuclear Industries

Author: Lothar Hahn

Presentation on the EC-JRC International Workshop on

Promotion of Technical Harmonization on Risk-Based Decision Making

May 22-24, 2000
Stresa, Italy

Content

Part 1 Presentation

Part 2. Transparencies

Part 3. Answers to the Questions

Part 1: Presentation

1.1. Introduction

Risk-based regulation is a highlight in current discussions in many industrial and technical areas. One prerequisite of any decision making process in the context of risk-based regulation is the probabilistic risk assessment. This term will mostly be used in this presentation and be abbreviate by PRA. Sometimes, especially for plant specific analysis without assessment of the consequences the term probabilistic safety assessment (PSA) is used because it is largely spread in Germany.

The following presentation summarizes the author's view of experiences, status, profiles, possible uses and limits of risk and safety assessment in the nuclear industry.

1.2 Lessons learned from Probabilistic Risk Assessment in the Past

Existing Probabilistic Risk Assessments (PRA) have substantially promoted the knowledge on safety risks of nuclear plants. The insights have been generic as well as plant-specific. The most important generic insights are related to the phenomena during severe accidents. Within the assessment of the consequences of accidents it was necessary to quantify the source term as precise as possible, because the knowledge of nature, time and amount of radioactive releases the most important parameters for the determination of consequences of a nuclear accident. Whereas in early generic risk studies as the U.S. Reactor Safety Study <1> or in the German Risk Study, Phase A <2> it was assumed that radioactive releases and the potential of catastrophic consequences would be an unrealistic worst-case-scenario (due to steam explosion), recent investigations in the eighties revealed additional phenomena with the risk of early containment failure. Examples are the effects of Direct Containment Heating, which was described in NUREG 1150 <3>, or the high pressure core melting and the hydrogen problem, which both have been discussed in the German Risk Study, Phase B <4>. With these results the knowledge about the risks of nuclear reactor accidents was extended dramatically.

Another lesson from generic risk assessments was the result of french studies that the risk of shutdown condition should no longer be neglected.

Conclusions were drawn from the results of these generic risk assessments, for example the implementation of accident-management-measures, the installation of catalytic hydrogen combiners or the revision of the instructions for the shutdown states of the plants.

Also the results of plant specific probabilistic safety assessments revealed important insights. Particularly a lot of safety related weaknesses and vulnerabilities were found in nearly every plant. In many cases these facts were caused by noncompliance with the deterministic safety requirements. Depending on their probabilistic significance different kinds of backfitting measures were provided. Maybe more important than probabilistic numbers in the fact, that through the analysis the operator got a more profound understanding of his plant and that deficiencies were found which were corrected spontaneously.

It can be stated that existing PRA have led to important progress in the knowledge of the risks of nuclear accidents as well as of individual plants safety characteristics. In many cases the results of PRA were used for improvements and backfitting measures so that a significant gain in safety was achieved. But all this had nothing to do with risk-based regulation.

1.3 Risks of misuse of PRA

There is no doubt that PRA results have a high importance particularly for relative or comparing evaluations of different technical solutions. On the other hand PRA results can result in a loss of safety, if the relevance of absolute probabilistic numbers is overestimated. This can be the case if probabilistic results are used as justification to tolerate deterministic safety deficiencies or cases of non-compliance with fundamental requirements. There are also cases in which probabilistic numbers are used as argument to classify corrective measures as not urgent even if fundamental deterministic principles are concerned. Numerous cases are known from the past in which probabilistic arguments were misused in the described sense. Three examples:

- In one plant probabilistic assessments were used to declare the missing of physical separation of the cables of redundant parts of the safety system as not safety relevant.
- In another case it was argumented that by reasons of a low probability of the safe shutdown necessary backfittings of the seismic design were not urgent respectively could not be required by the authority.

- In a third case the low probability of certain primary system breaks was used as argument for not installing devices to circulate hydrogen inside the containment.

Such argumentations are problematic if absolute numbers are used in order to tolerate relevant noncompliance with the requirements or deficiencies within the defence-in-depth concept. In addition it must be recognized that in most countries probabilistic criteria for the evaluation of deterministic deficits do not exist or do not have a legal basis.

1.4 Risks of manipulation in PRA

In the case that probabilistic criteria for the evaluation of deterministic facts exist, it cannot be excluded that people attempt to obtain the "desired" result by choosing the "appropriate" assumptions, methods or data. It is true that the methodology to perform PRA has reached a considerable degree of maturity; nevertheless the results of the analysis of the same matter may differ significantly if the investigations are performed by different teams. Differences of one to two orders of magnitude are possible only due to different approaches in modelling or selecting data. More than ever are such differences possible if one tries to get "good" results by selecting optimistic assumptions, methods and data.

This is another fact that diminishes the value of absolute probabilistic numbers. On the other hand are probabilistic comparisons, for example for the evaluation of competing technical solutions, more resistant against manipulation provided that the analysis is transparent.

1.5 Insufficient accuracy and quality of existing PRA

Beside the considerable progress in PRA methodology in the last decade there still remain inherently large uncertainties in PRA results. In addition to the uncertainties induced by the spread of input data uncertainties due to modelling effects must be considered. Furthermore in some areas different expert options exist on the appropriate methodological approach, for example in the fields of common-mode-failure and of human reliability. Other potentially risk relevant factors such as the influence of organization, management and safety culture are normally not considered in current PRA. These facts lead to the conclusion, that even from a methodological point of view

the maturity of PRA is not sufficient in order to serve as an adequate basis for risk-based decision making processes.

In addition to these facts it must be considered that in practice the PRA for existing plants do widely vary in quality, scope and completeness. Examples for different PRA scope are the treatments of fire, flooding, external hazards and shut down conditions.

Standards are missing to guarantee that the PRA results for different plants are comparable.

1.6 The relation of probabilistic and deterministic safety concepts

A fundamental impediment for the application of risk-based decision making in the regulation is the fact, that the existing plants are designed, licensed and constructed on the basis of a body of regulations which were not formulated to be risk-based. It is true that each individual regulation as well as the whole system of deterministic requirements can be examined by probabilistic evaluations. But in practice it is not possible to replace the deterministic system by an equivalent probabilistic system.

The deterministic principle relies on the approach to guarantee the completeness of design by a set of representative design basis accidents and to fill up the remaining gaps with safety margins and conservations assumptions. Contrarily the probabilistic approach must try to be complete by analyzing all possible sequences and failure combination individually and as realistic as possible. This difference results in manifold difficulties if deterministic and probabilistic approaches are mixed. One of such difficulties arises if PRA shall be used to reduce "unnecessary conservatism" associated with current (deterministic) regulatory requirements. The decision, if a requirement is unnecessary conservative or not, can only be reliable, if the analyst has explicitly in mind the full purpose of the requirement. This means for example that the analyst must reflect all elements of the background of the conservatism. Without a detailed, complete and transparent analysis of the means of a conservative assumption, a correction of deterministic requirements by probabilistic arguments seems not to be reliable. Changes at deterministic requirements should be justified primarily by deterministic arguments rather than by probabilistic arguments. In cases of doubt deterministic principles as the defence-in-depth concept must override probabilistic considerations. Probabilistic is an excellent element to supplement deterministic but can never replace it. And vice versa.

1.7 Missing of probabilistic acceptance criteria

A further obstacle for the introduction of risk-based regulation is the fact that quantitative risk-acceptance criteria for the range of regulatory objectives do not exist. What does exist in the international context are the two IAEA/INSAG-targets of 10^{-4}/a for the core damage frequency and 10^{-5}/a for the large radioactive releases. Risk-based decision-making and risk-based regulation is not possible without quantitative risk-based criteria at least for the mean elements of the regulatory requirements. These were in general not formulated to be risk-based. This is one of the main reasons that the US Nuclear Regulatory Commission is not going to risk-based regulation but is following the way to so-called risk-informed regulation.

1.8 Insufficient homogeneity of existing PRA

A further argument against a comprehensive use of PRA for risk-based regulation is the fact that PRA exist for many plants but the results are not or not easily comparable. One of the conditions for a uniform risk-based regulation is the existence of a homogeneous database which includes PRA for all plants whit comparable result. This is not the case in most countries because appropriate standards do not exist until now.

Beside the facts discussed in chapter 1.5 one has to keep in mind the following observations for the example of Germany:

- For some plants not any or at least no up-to-date PRA exists.
- A PRA guideline has been formulated in Germany only a few years ago.
- The present PRA guidelines are not very precise and allow a high degree of freedom for the realisation of a PRA.
- PRA are not yet obligatory be atomic law.
- The existing PRA vary widely in qualities, scope and completeness.

Therefore a further condition for a reliable risk-based regulation is missing. For similar reasons the USNRC does not rely on risk-based regulation.

1.9 Recommendations for an optimal use of probabilistic risk and safety assessment

In the preceding chapters several arguments against the introduction of risk-based regulation in nuclear industry in the present time and for existing nuclear power plants have been formulated. This does not mean that the profit of PRA for the nuclear safety is disregarded or underestimated by the author. In contrary PRA are regarded as an excellent tool to gain deeper insights in the safety and risk characteristics from a generic point of view as well as for individual plants. Particularly for the identification of weakness and vulnerabilities of plant design features PRA are a modern and highly effective instrument. This is also valid for its application in comparative decisions and for prioritisation of different measures.

For these applications PRA can be used in regulation processes. So far the further development and higher a degree of standardisation is not only desirable but also necessary. Corresponding efforts should be supported as far as possible. It should be an important goal so that for each nuclear power plant in the world will be stated a comprehensive and high quality PRA as soon as possible and that this PRA will renewed continuously in the sense of living PSA. With this approach the highest profit of the probabilistic tools can be achieved.

On the contrary it should be avoided to stress the probabilistic tools in areas where PRA have weakness or to pose questions on PRA which cannot be answered by technical experts but only by society and policy. Absolute probabilistic numbers are so long useless as society and policy do not have defined what risk is acceptable or not. PRA experts can only analyse and give answers on technical questions. They cannot replace risk acceptance criteria. PRA cannot replace deterministic principles as defence-in-depth concept, PRA can only complement them. Probabilistic tools are most useful and effective for enhancing the safety. PRA should not be misused to justify deterministic deficiencies; this would limit the profit of probabilistic tools and their future importance drastically.

Part 2: Transparencies

(Transparencies itself will be delivered on Monday, 8.5.2000

HISTORY OF PRA IN THE NUCLEAR INDUSTRY

- 1975 Reactor Safety Study (WASH 1400)
- 1979 German Risk Study, Phase A
- 1989 German Risk Study, Phase B
- 1990/91 Risk Study for Five US Reactors (NUREG 1150)
- 1992 German BWR Study

- Plant Specific PRA or PSA
  in USA (e.g. IPE, IPEEE)
  in Germany since 1987

- Precursor Studies

MAIN INSIGHTS AND MERITS OF PRA

Generic Insights

- Accident Phenomena
- Accident Consequences
- General Vulnerabilities
- Shutdown Risks
- Risk Characteristics

Plant Specific Insights

- Design Deficiencies
- Ranking of the risk contribution of individual initiating events, sequences, failures, failure modes and failure combinations
- Prioritisation of backfitting measures
- Absolute safety level

DEFICIENCIES OF CURRENT PSA

- Poor database in some areas
- No consensus on the appropriate model in some areas
- Human performance not treated full scope
- Organisational and safety culture factors not explicitly covered
- Large uncertainties
- Precise standards are missing
- Existing PRA vary largely in quality, scope and completeness

ARGUMENTS AGAINST RISK-BASED REGULATION

- Missing of probabilistic risk acceptance criteria
- Existing plants are licensed and constructed on the basis of regulations which are not risk-based
- Individual PRA do not exist for all plants
- Existing PRA vary largely in quality, scope and completeness
- Standards are missing which guarantee that the individual PRA results are comparable
- Risks of misuse and manipulation exist

PROBABILISTIC APPROACH VS. DEFENCE-IN-DEPTH?

Deterministic design concept

- Representative Design Basis Accidents
- Conservative assumptions
- Safety Margins
- Defence-in-depth-concept
  - high quality of construction and operation
  - control, limiting, protection
  - safety functions, accident procedures
  - measures and accident management in case of severe conditions

Probabilistic evaluation

- All significant risk contributions to be analysed
  ⇒ completeness
- Realistic instead conservative estimates
  ⇒ conservatism not always quantitative
- Data uncertainties
- Model uncertainties

CONCLUSIONS AND RECOMMENDATIONS

- Use PRA for better Systems understanding
- Use PRA for detecting design and procured deficiencies and vulnerabilities
- Use PRA for relative evaluations of technical solutions, backfitting measures etc.

- Don't stress PRA for justifying deterministic deficiencies on the basis of absolute numbers

- Improve the standards
- Promote harmonisation
- Guarantee transparency

- Make use of the merits for safety improvements
- Do not hinder to future of PRA methodology by misuse for lowering the safety

Part 3: Answers to the questions

3.1 Topic 1: Development, purpose and general principles of risk assessment use

Question 1.1: First PRA in USA 1975 (WASH 1400);

German Risk Study, Phase A 1979;

German Risk Study, Phase B 1989;

PSA as part of periodic safety assessment since end of the eighties;

Standardisation by PSA guideline 1997;

Question 1.2: Complement to defence-in-depth;

Analysis of deficiencies and vulnerabilities;

Evaluation of harmonic design;

assessment of safety level;

Question 1.3: General requirements by PSA guideline;

Question 1.4: Only general requirements; no specific standards;

Question 1.5: General recommendation for living PSA;

Updating frequencies every 10 years;

Question 1.6: No regulation;

Question 1.7: Authorities / Experts / Reviews;

3.2: Topic 2: Terminology

Question 2.1: Yes, in a very general sense;

Question 2.2: Yes, in a very general sense;

Question 2.3.: We distinguish three levels of PSA:

Level 1: Probability of core damage states;

Level 2: Probability and amount of radioactive releases;

Level 3: Probability and amount of consequences;

(Level 1+): Probability of core damage states and active containment failure;

Step I and Step II: Level 1;

Step III and Step IV-A: Level 3;

Step IV-B and Step V: not established;

3.3 Topic: Step I: Hazard Identification

Question 3.1 and 3.2: Identification of the areas with the highest radioactive inventories, mostly reactor core and spent fuel pool; also storage facilities and auxiliary systems as filters;

Question 3.3: Radioactive inventories whit barriers and containments which can fail due to internal and external hazards, technical and human failures;

Question 3.4: Prompt and stochastic health effects;
Land contaminations (Level 3);
Radioactive releases (Level 1a and 2);

Question 3.6 Mostly: Event tree analysis;

Question 3.7: Engineering judgement review;

Question 3.8: Review process;

Question 3.9: PSA guideline;
National and international experience and standards;

Question 3.12: Uncertainty analysis;

Question 3.13: Both;

Question 3.14: Minor importance in this step;

3.4 Topic 4: Step II – Event scenario Assessment

Question 4.1 and 4.2: Identification of initiating events;
Event sequence analysis;
Description of the final states;
Event tree analysis;

Question 4.3: No boundaries;
Men, Nature, Land;

Question 4.6 and 4.7: PSA Guideline;

Question 4.8: Only internal procedures;

Question 4.9: Uncertainly analysis;

Question 4.11: Medium importance; should be treated transparently;

Question 4.12: Specific models like THERP; not completely covered;

(5)

Final Regulatory Guide 1.183 (DG-1081)

Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

Final SRP 15.0.1

Radiological Consequence Analyses Using Alternative Source Terms

Stephen F. LaVie

NRR/DSSA/SPSB

Background

- ☆ 06/30/98 Rulemaking plan for implementation of revised source terms at currently operating reactors (SECY-98-158)
  - ☆ New rule section § 50.67
  - ☆ Miscellaneous conforming changes
  - ☆ New regulatory guide
    - ☆ Supersedes RG 1.3, 1.4, 1.5, 1.25, 1.77
  - ☆ New standard review plan section
- ☆ 06/30/98 Re-baselining study (SECY-98-154)

Background

- 03/11/99 Draft rule published in Federal Register *(ACRS 11/98)*
- 12/23/99 Final Rule published in Federal Register *(ACRS 9/99)*
  - Includes announcement of public comment period on Draft Guide DG-1081 and Draft SRP 15.0.1
- 01/24/00 Final rule effective
- 03/31/00 Public comment period ends

Public Comments

- 6 Comment letters
  - NEI
  - NUGEQ
  - Duke Energy
  - Virginia Power
  - STPNOC
  - FPC
- Several informal comments (e-mail, etc.)
- Addressed ACRS recommendations (10/99 ltr)
- 138 total comments

Significant Changes

- ☆ Fuel Gap Fraction
- ☆ Fuel Handling Accident Chemical Form
- ☆ Selective Implementation
- ☆ 50.59 Guidance
- ☆ Other Technical Changes

Fuel Gap Fractions

- ☆ Industry suggested changes
  - ☆ Gap fraction ranging from 3% at 50 GWD/MTU to 9.3% at 75 GWD/MTU
  - ☆ Allow licensee to vary gap fraction across core
  - ☆ Address fuel heatup impact separately
- ☆ Staff Concerns
  - ☆ Insufficient experimental data to support iodine gap fractions above 65 GWD/MTU
    - ☆ Majority of industry data for low burnup; few data points above 62 GWD/MTU
  - ☆ Uncertainties in gap fraction data
    - ☆ Data from actual fuel; no operational transients addressed
    - ☆ Current fuel management more aggressive than that under which data collected
    - ☆ In majority of experiments, iodine is not measured directly

Gap--Staff Approach

- ☆ Use NUREG-1465 data for LOCA (0-62 GWD/MTU)
- ☆ Use RG 1.77 data for reactivity insertion accidents
- ☆ Work by PNNL to address environmental impact of change in fuel burnup from 60 to 62 GWD/MTU reported in this period
  - ☆ Will be documented as update to NUREG/CR-5009
- ☆ PNNL Analyses
  - ☆ Core average and peak rod average at 35, 60, and 65 GWD/MTU
  - ☆ FRAPCON-3 Code w/ Missah release model
  - ☆ Best estimate approach
  - ☆ No operational transients addressed
  - ☆ Core inventories calculated to 75 GWD/MTU
- ☆ Base RG on PNNL analyses with some adjustments
- ☆ Balance uncertainty in gap fractions with other analysis conservatisms (e.g., peak burnup rod in peak power position)

Fuel Gap Fractions

	DG-1081	LOCA	Non-LOCA		RIA
Burnup, GWD/MTU=	0-62	0-62	0-40	40-62	0-62
I-131	0.12	0.05	0.05	0.08	0.1
Kr-85	0.15	0.05	0.06	0.1	0.3
Other NG	0.1	0.05	0.03	0.05	0.1
Other Halogen	0.1	0.05	0.03	0.05	0.1
Alkali Metals	0.1	0.05	0.08	0.12	0.0

- ☆ Gap fraction associated with peak rod burnup in core is used with rod inventory adjusted for maximum radial peaking factor
- ☆ Optional for accidents other than FHA, RIA, LOCA: if location of damaged fuel in core can be projected with reasonable certainty, can use assembly-specific gap fraction and radial peaking factors (not less than 1.0) on assembly-by-assembly basis

Fuel Handling Accident Iodine Species

- ☆ DG-1081 retained Safety Guide 25 species--99.75% elemental, 0.25% organic
- ☆ Industry suggested using NUREG-1465 species
- ☆ Staff concerned about low pH of fuel pool, lack of transport data (e.g., pool DF)
- ☆ Final guide:
  - ☆ Release from fuel is 95% CsI, 4.85% elemental, 0.15% organic
  - ☆ CsI completely dissociates in pool water, re-evolving as elemental iodine
  - ☆ With pool effective DF=200, release from pool is 57% elemental, 43% organic
  - ☆ Justifiable mechanistic treatments considered on case-by-case basis

Selective Implementation

- ☆ DG-1081 required licensees desiring to extend an approved AST implementation to another application in the plant to request prior approval
- ☆ Final Guide:
  - ☆ Subsequent modifications using the approved AST characteristics incorporated into the design basis possible under § 50.59.
  - ☆ Prior staff approval under § 50.67 required if:
    - ☆ Use of AST characteristics or TEDE criteria not in the approved design basis.
    - ☆ Changes to previously approved AST characteristics.
- ☆ Revised position consistent with new § 50.59 guidance and § 50.67

§ 50.59 Guidance

- Issue:
  - Under guidance of DG-1081, it is possible to have some design basis calculations based on TID14844 and the traditional whole body and thyroid doses, while others are based on AST and TEDE.
  - Guidance requires each these analysis to be updated to AST and TEDE if it is recalculated for any reason in future.
  - How does one determine increase in consequences when dose quantities and criteria are different?
- Solution:
  - Guidance added to final guide to convert prior analysis results before making § 50.59 comparison.
  - $TEDE_{prior} = WB_{prior} + (Thyroid_{prior} * 0.03)$

Other Technical Changes

- EQ
  - Since GSI not resolved, text was added to final guide to allow use of TID14844 or AST for re-analyses required under the guide.
  - Added general guidance for calculation of EQ doses outside of containment
- Steam Generator iodine Transport
  - Corrected error regarding decontamination credit when tubes are uncovered.
- Containment Spray DF
  - Corrected error related to maximum spray DF when using SRP model
- Building Mixing Credit for FHA
  - All credit on case-by-case basis with suitable justification; guidance provided
- Passive ECCS leakage (50 gpm) Evaluation (w/ LOCA)
  - Requirement deleted

ACRS Recommendation

- ☆ ACRS recommended that the requirement to have prior NRC approval for *"changes...that result in a reduction in safety margins"* should be re-evaluated for removal in light of both the analytical assessments performed by RES and the results of the pilot applications of the alternative source term. (ACRS discussion identified § 50.59 as alternative.)
- ☆ Staff committed to re-evaluating this requirement during public comment period in light of § 50.59 guidance.
- ☆ Staff has retained language in guide.
  - ☆ Guide applies to initial implementation for which § 50.59 is not applicable.
  - ☆ Rebaseling and pilots provided useful insights, but the limited sample of plants considered does not provide an a priol basis to summarily disposition all potential plant-specific and modification-specific impacts
- ☆ Staff added language referencing § 50.59 for subsequent plant modifications.

ACRS Recommendation

- ☆ ACRS recommended that the staff should modify the proposed redefinition of the source term to eliminate the connotation that the release is necessarily to the containment but should retain the wording *"...release from the RCS..."*
- ☆ Staff declined to change rule language; committed to reviewing the draft guide to ensure that our description of the alternative source term for the LOCA does not misrepresent the NUREG-1465 basis
  - ☆ § 50.2 definition has to address accidents, other than a LOCA, that result in fuel damage, but may not involve RCS or containment
  - ☆ Accident-specific appendices in DG-1081 provide source term guidance.
  - ☆ Final guide is a standalone document. User does not need to refer to NUREG-1465 to be able to use the guide.
- ☆ Clarifications added to the final guide to ensure that the release for a LOCA is defined as the release from the RCS to the containment.

United States
Nuclear Regulatory Commission

Response to SRM on PRA Quality

Mary Drouin, Mark Cunningham Division of Risk Analysis and Applications Office of Nuclear Regulatory Research	Richard Barrett, Gareth Parry, Michael Cheok Division of System Safety and Analysis Office of Nuclear Reactor Regulation

Presentation to

Advisory Committee on Reactor Safeguards

June 7, 2000

OUTLINE

- Background
- Proposed Response to SRM
- Contents of Attachment
- Schedule/Milestones

Background

- Staff faced with issue of the PRA quality needed for risk-informed regulatory activities and the need for an integrated, uniform approach
  - Response to April 18, 2000, SRM

    "The staff should provide its recommendations to the Commission for addressing the issue of PRA quality until the ASME and ANS standards have been completed, including the potential role of an industry PRA certification process."
  - Review of Option 2 documents, including NEI certification submittal

Background (cont.)

- Review of draft standards, endorsement of final standards

	Draft	Final
ASME	6/00	1/01
ANS - Seismic	6/00	9/00
ANS - LPS*	9/00	12/00

*ANS anticipates delay

Proposed Response to SRM

- Summarize what staff is now doing
- Propose/recommend/inform Commission of additional activities
- Develop staff documents
- Update RG 1.174, SRP Chapter 19
- Review submittals against positions
- Review standards and NEI guidelines for possible endorsement

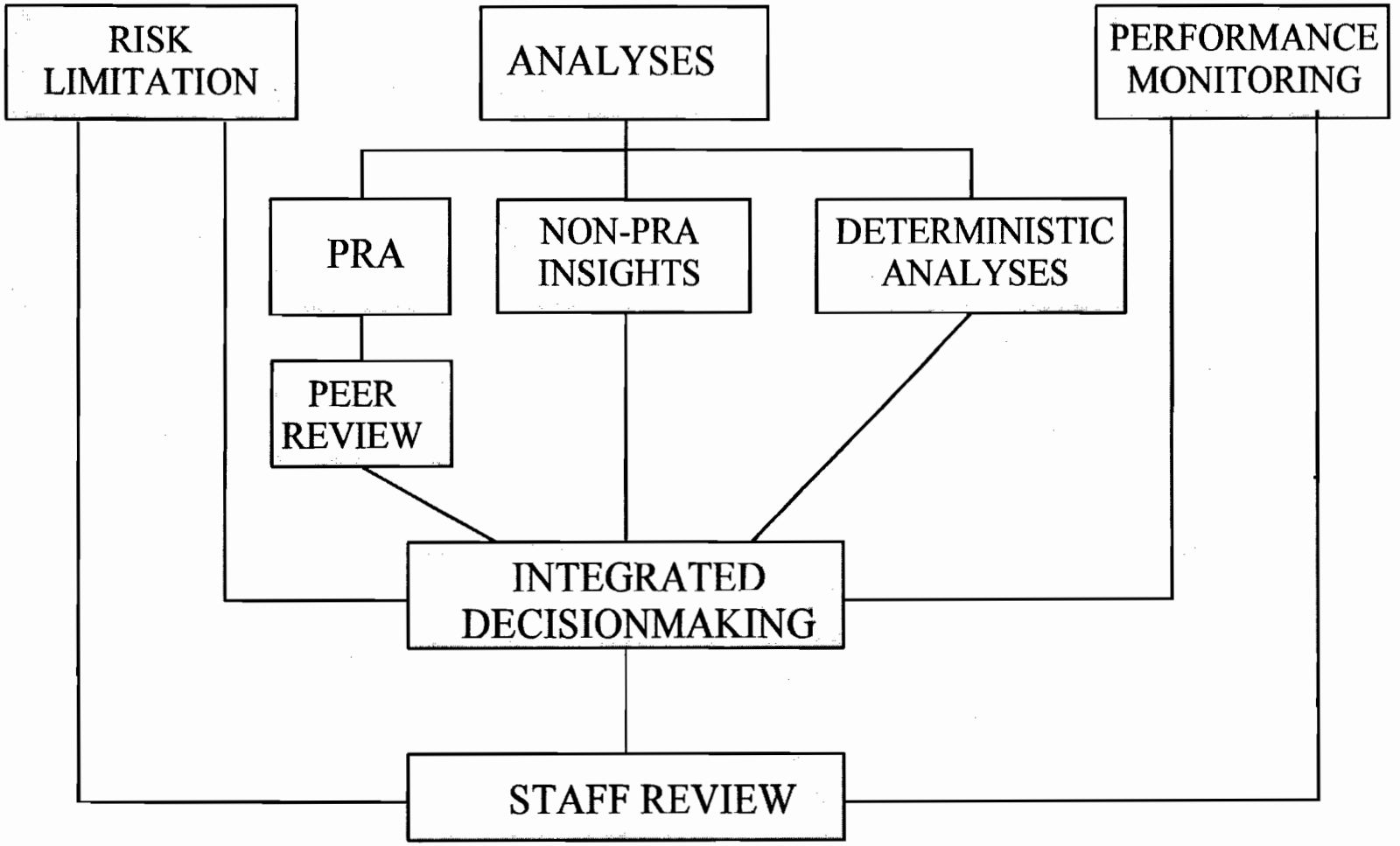
Proposed Response to SRM -- Attachment

- Risk-Informed Reactor Activities
- NRC Decision-making Process
- PRA Technical Acceptability
  - PRA scope and level of analysis
  - PRA elements and characteristics
  - Peer review
  - PRA application process
  - Expert panel

RISK-INFORMED REACTOR ACTIVITIES

- Risk-Inform 10CFR Part 50
- Plant Oversight Process
- Operating Events Assessment
- License Amendments

NRC DECISION-MAKING PROCESS



Factors Controlling Potential Increases in Risk

RISK LIMITATION

- Nature of the application

  "Categorization of equipment for the purpose of determining regulatory treatment"

- Extent of the application

- Controls and backstops

  Configuration risk management / Maintenance rule a(4)

  Functionality

- Limits and focuses the need for analysis

PERFORMANCE MONITORING

Attributes

- Measures and criteria
- Timely detection
- Margin of safety

Examples

- Maintenance rule monitoring
- Steam generator tube condition monitoring

ANALYSIS

- PRA
  - Quantitative risk estimates
    - Focused on relevant issues
  - Standards
  - Peer review / Certification

- Non - PRA Risk Insights
  - Generic results
  - Qualitative findings

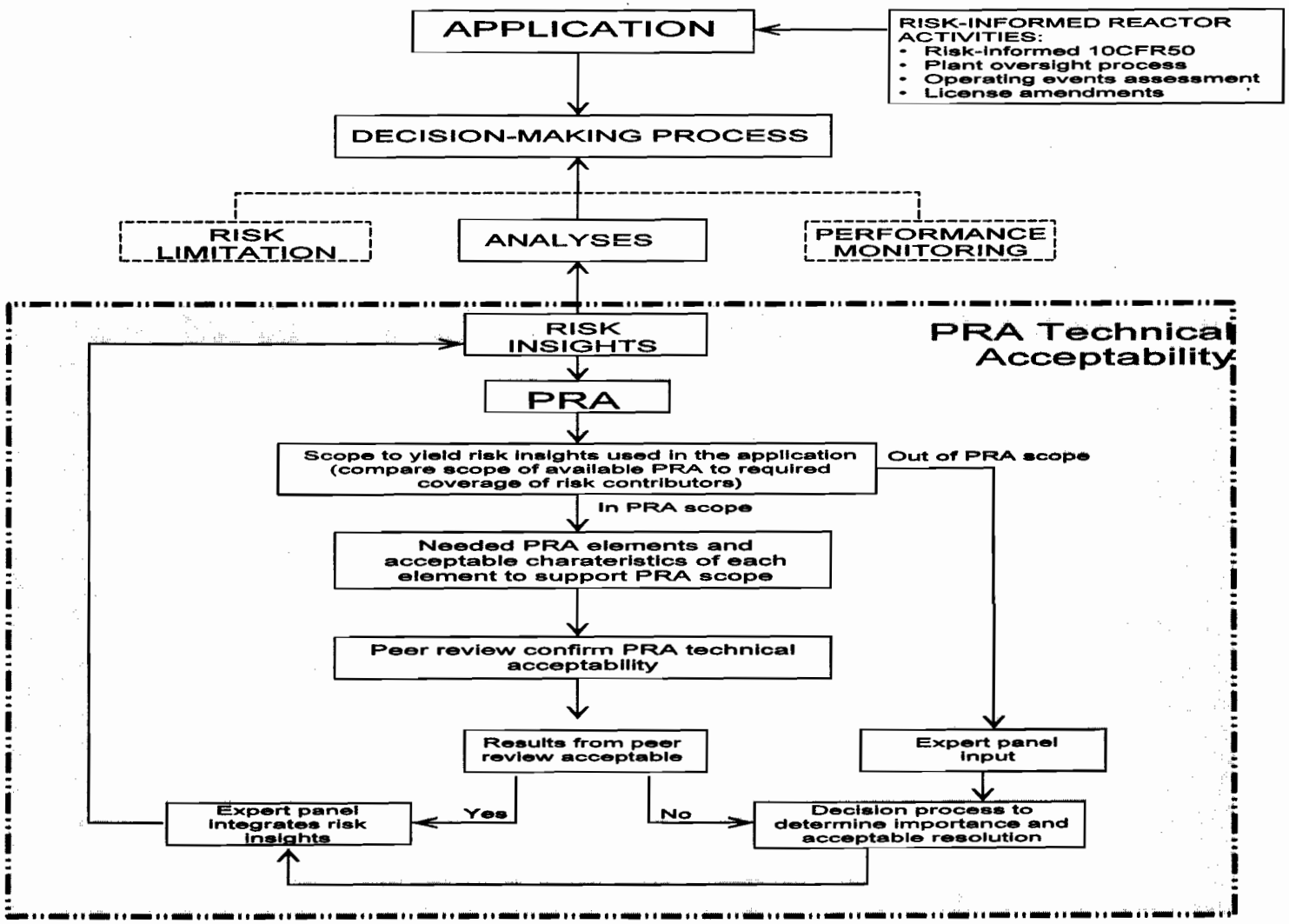
- Deterministic Analysis
  - Defense in depth
  - Margins

- Integrated decision making

ROLE OF FACTORS IN RISK-INFORMED APPLICATIONS

	RISK LIMITATION	ANALYSES	PERFORMANCE MONITORING
Tech spec AOT	high	high	med
RI-ISI	high	med	low
SG tubes	low	med	high

PROCESS FOR ESTABLISHING PRA TECHNICAL ACCEPTABILITY



SCOPE AND LEVEL OF ANALYSIS OF A PRA

Scope/Level Definition	Requirements
POS	full and low power, hot and cold shutdown
Initiating Events	internal • transients • LOCAs • floods • fires external • seismic • high wind • others
Risk Characterization	Level 1: core damage frequency
	Level 2: LERF and late containment failures
	Level 3: not required

TECHNICAL ELEMENTS AND CHARACTERISTICS OF A PRA

- Identify the PRA results used in the decision-making process (e.g., CDF, LERF, dominant accident sequences and associated contributors)
- Specify the technical elements (e.g., initiating event analysis)
- Identify the needed characteristics and attributes for each element

Element	Required Characteristics and Attributes
Initiating Event	• thorough identification and characterization of initiators • grouping of individual events according to plant response and mitigating requirements
Success Criteria	• based on best-estimate engineering analyses applicable to the actual plant design
Accident Sequence Development	• defined in terms of hardware, operator action, and timing requirements • includes all necessary and sufficient equipment (safety and non-safety) reasonably expected to be used to mitigate initiators • includes functional, phenomenological, and operational dependencies and interfaces

PEER REVIEW

- A peer review process can be used to help confirm the technical acceptability of a PRA
- Specify the necessary elements of a peer review process
  - team qualifications
  - peer review process
  - documentation
- Identify the needed characteristics and attributes for each element

Element	Required Characteristics and Attributes
Team Qualifications	• independent with no conflicts of interest • expertise in all the technical elements of a PRA including integration • knowledge of the plant design and operation • knowledge of the peer review process
Documentation	• describe the peer review team qualifications • describe the peer review process • document where PRA does not meet requirements of PRA standard • document adequacy of PRA maintenance process to incorporate plant modifications • assess and document significance of deficiencies

PRA APPLICATION PROCESS

- Required PRA scope and technical requirements may vary depending on application

- Specify the necessary elements of an application process
  - Define the application
  - Examine PRA scope and level of detail
  - Determine sufficiency of PRA "standard"
  - Determine sufficiency of PRA
  - Resolve insufficiencies and differences

- Identify the needed characteristics and attributes for each element

Element	Required Characteristics and Attributes
Examine PRA Scope and Level of Detail	▸ determination of the existing PRA scope ▸ determination of the existing PRA modeled SSCs ▸ determination of resolution for insufficient aspects
Determine Sufficiency of PRA Standard	▸ determination of sufficiency of PRA standard to assess the application under consideration ▸ determination of resolution for insufficient aspects
Determine Sufficiency of PRA	▸ identification of differences between PRA and the standard's requirements ▸ determination of the importance of the differences ▸ determination of resolution for important differences

EXPERT PANEL

- PRA results may be integrated into the decision-making process by an expert panel
- Expert panel may be used to evaluate the importance of missing risk information
- Specify the necessary elements of a expert panel process
  - Decision-making process
  - Technical information bases
  - Incorporation of non-PRA Modeled Components
  - Identification of Limitations
  - Panel Member Qualifications
  - Documentation
- Identify the needed characteristics and attributes for each element

Element	Required Characteristics and Attributes
Decision-making Process	• decision-making process appropriate • appropriate information available • evaluation of risk significance represents appropriate consideration of issues
Incorporation of non-PRA Modeled Components	• evaluate in a systematic manner the safety significance of components not modeled in the PRA but affected by a proposed application
Identification of Limitations	• process applied by the licensee to overcome limitations of PRA is appropriate • decisions made that do not follow straightforwardly from the PRA require a technical basis that shows how the PRA information and the supplementary information validly combine to support the finding, and • no findings contradict the PRA in a fundamental way

Key Short-Term Milestones

▸ Discuss general approach with RILP	June 1
▸ Discuss approach, draft Commission paper with PRA Steering Committee	June 15
▸ Commission paper to EDO	June 27

HIGH-LEVEL GUIDELINES
FOR
PERFORMANCE-BASED ACTIVITIES

PRESENTATION TO ACRS FULL COMMITTEE

JUNE 8, 2000

OFFICE OF NUCLEAR REGULATORY RESEARCH

N. Prasad Kadambi, REAHFB

J. E. Rosenthal, Branch Chief, REAHFB

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OUTLINE

- OVERVIEW
- HISTORICAL BACKGROUND
- SRM TO SECY-99-176
- ACTIONS TAKEN FOR INTERNAL AND EXTERNAL STAKEHOLDER INPUT
- USE OF RISK INFORMATION FOR PERFORMANCE-BASED INITIATIVES
- DISCUSSION OF HIGH-LEVEL GUIDELINES
- DISCUSSION OF STAFF'S PLAN
- CONCLUSION

OVERVIEW

- THE STAFF IS MAKING STEADY PROGRESS TO DEVELOP PERFORMANCE-BASED APPROACHES CONSISTENT WITH COMMISSION DIRECTION

- THE DEVELOPMENT OF THE HIGH-LEVEL GUIDELINES AND THEIR VALIDATION REPRESENT A SIGNIFICANT MILESTONE IN PROGRESS TOWARD ACHIEVING THE GOALS OF THE PERFORMANCE-BASED REGULATION INITIATIVE.

- THE GUIDELINES WILL BE VALIDATED AND TESTED OVER A RANGE OF REGULATORY ISSUES TO GAIN CONFIDENCE IN THEIR USE AND IDENTIFY KEY CHALLENGES WHICH MAY LIMIT THEIR APPLICATION.

- THE STAFF WILL EVENTUALLY INTEGRATE THE PERFORMANCE-BASED ACTIVITIES INTO THE MAINSTREAM OF REGULATORY IMPROVEMENT ACTIVITIES WHICH CURRENTLY HAS A MULTITUDE OF RISK-INFORMED EFFORTS.

HISTORICAL BACKGROUND

- THE COMMISSION HAS EXPRESSED A FIRM COMMITMENT TO INSTITUTING PERFORMANCE-BASED APPROACHES WHEREVER FEASIBLE STARTING WITH THE DIRECTION SETTING PAPERS FROM 1996 ON THROUGH THE LATEST DRAFT OF THE STRATEGIC PLAN.

- WHILE SIGNIFICANT PROGRESS WAS BEING MADE ON RISK-INFORMED INITIATIVES THE FOCUS OF THE PERFORMANCE-BASED INITIATIVES WAS ON THOSE ISSUES "NOT AMENABLE TO PRA" (SRM TO SECY-98-132).

- THE MOST RECENT PAPER FROM THE STAFF, SECY-99-176, WAS NOT RECEIVED FAVORABLY BY THE COMMISSION BECAUSE THE PLANS LACKED SPECIFICITY AND THE MAGNITUDE OF PROGRESS IT REPRESENTED WAS INSUFFICIENT.

- ACRS LETTER OF JUNE 10, 1999 CALLED FOR FOCUSING OF DIVERSE ACTIVITIES ON PERFORMANCE-BASED REGULATION

- THE SRM TO SECY-99-176 EXPLICITLY PROVIDES COMMISSION EXPECTATIONS AND DIRECTS THE STAFF TO TAKE THE ACTIONS DESCRIBED IN THIS PRESENTATION.

SRM TO SECY-99-176

- THE COMMISSION DIRECTED THE STAFF TO:
  - "... develop high-level guidelines to identify and assess the viability of candidate performance-based activities."
- IN SECY-99-176, THE STAFF HAD PROPOSED GUIDELINES AS A DOWNSTREAM ACTIVITY. THE COMMISSION ADVANCED THE SCHEDULE SIGNIFICANTLY.
- THE SRM INCLUDED THE FOLLOWING ELEMENTS:
  - The guidelines should be developed with input from stakeholders and the program offices.
  - The guidelines should include discussion on how risk information might assist in the development of performance-based initiatives.
  - The guidelines should be provided to the Commission for information.
  - The staff should periodically update the Commission on its plans and progress in identifying and developing performance-based initiatives.
- THE PROPOSED GUIDELINES WILL PROVIDE THE FRAMEWORK FOR FOCUSING ACTIVITIES AS ACRS HAD SOUGHT TO DO.

INTERNAL AND EXTERNAL STAKEHOLDER INPUT

- CREATION OF THE PERFORMANCE-BASED REGULATION WORKING GROUP (PBRWG) FROM ALL AFFECTED PROGRAM OFFICES.
- FEDERAL REGISTER NOTICES ISSUED ON JANUARY 24 AND FEBRUARY 17, 2000.
- FACILITATED WORKSHOP HELD ON MARCH 1, 2000.
- WRITTEN COMMENTS RECEIVED FROM A RANGE OF EXTERNAL AND INTERNAL STAKEHOLDERS.
- FEDERAL REGISTER NOTICE OF MAY 9, 2000, WITH RESPONSE TO COMMENTS.
- ON-LINE WORKSHOP OF JUNE 8, 2000
- STAFF CHARACTERIZES STAKEHOLDER INPUT AS BEING NOT NECESSARILY UNFAVORABLE PROVIDED CERTAIN “IMPLEMENTATION” AND “TRUST” ISSUES ARE ADDRESSED.

USE OF RISK INFORMATION

- RISK INFORMATION MAY PROVIDE THE BASIS FOR UNDERTAKING AN INITIATIVE
  - SAFETY ENHANCEMENT
  - REDUCTION OF UNNECESSARY BURDEN
  - CHANGES RESULTING FROM RISK-INFORMED REGULATION (OPTIONS 2 & 3) WILL CONSIDER USING A PERFORMANCE-BASED APPROACH
- RISK INFORMATION IS USED FOR METRICS, THRESHOLDS AND/OR REGULATORY RESPONSE
- INITIATIVES MAY BE CLASSIFIED AS "NOT AMENABLE TO PRA", BUT WOULD BE CONSIDERED AS A PERFORMANCE-BASED INITIATIVE.

HIGH-LEVEL GUIDELINES

I. VIABILITY

A. MEASURABLE OR CALCULABLE PARAMETER

(a) Directly measured and related to safety objective
(b) Calculated and related to safety objective
(c) Ready access to data
(d) Monitored periodically

B. OBJECTIVE CRITERIA

(a) Use risk insights, deterministic analysis or performance history

C. FLEXIBILITY

(a) Programs and processes at licensee's discretion
(b) Encourage and reward improved outcomes

D. NO IMMEDIATE SAFETY CONCERN IF CRITERION NOT MET

(a) Sufficient safety margin
(b) Time for corrective action
(c) Capability to detect and correct performance degradation

HIGH-LEVEL GUIDELINES (Contd)

II. ASSESS IMPROVEMENT

A. MAINTAIN SAFETY
   (a) Safety plays primary role
   (b) Adequacy of safety margins assured by assessing conservatism and treatment of uncertainty

B. INCREASE PUBLIC CONFIDENCE
   (a) Assess impact of results and objective criteria with public participation

C. INCREASE EFFECTIVENESS, EFFICIENCY AND REALISM
   (a) Methodology and assumptions consistent with accounting for uncertainty and defense-in-depth
   (b) Assess placement in performance hierarchy

D. REDUCE UNNECESSARY BURDEN

E. TEST FOR OVERALL NET BENEFIT
   (a) Merits of pursuing change
   (b) Assess NRC or licensee benefits from change
   (c) Simplified assessment preferred

HIGH-LEVEL GUIDELINES (Cntd)

F. INCORPORATION INTO REGULATORY FRAMEWORK
- (a) CFR; Reg Guide; NUREG; SRP; TS; Inspection Guidance
- (b) One or more components considered for change
- (c) Justified by proponent; feedback from stakeholders
- (d) Inspection and enforcement considerations (including reduced NRC scrutiny) addressed early

G. ACCOMMODATE NEW TECHNOLOGY
- (a) Difficulties due to change in technology
- (b) New technology provides better solutions

III CONSISTENCY WITH REGULATORY PRINCIPLES

A. CONSISTENT AND COHERENT WITH OVERRIDING GOALS
- (a) Principles of Good Regulation; PRA Policy Statement; RG 1.174; Strategic Plan
- (b) Defense-in-Depth Philosophy; treatment of uncertainties

PROPOSED PLAN

- THE OBJECTIVE OF THE PLAN IS TO BUILD ON THE PROGRESS MADE IN THE STAFF'S RESPONSE TO THE ELEMENTS OF THE SRM

- AS CONFIDENCE IS DEVELOPED IN THE USE OF THE GUIDELINES THE PLANNING, BUDGETING AND PERFORMANCE MEASUREMENT PROCESS WILL BE USED TO INCORPORATE THE ACTIVITIES INTO OPERATING PLANS AND BUDGET RESOURCES AS APPROPRIATE.

- BY SIX MONTHS AFTER ISSUANCE OF SRM:

  - HIGH-LEVEL GUIDELINES WILL BE VALIDATED AND TESTED FOR ONE ISSUE IN THE REACTOR ARENA AND ONE IN THE MATERIALS OR WASTE ARENA

  - PROVIDE OBSERVATIONS ON INTEGRATION OF INITIATIVES IN THE RISK-INFORMED AND PERFORMANCE-BASED AREAS AND PROPOSE LONGER TERM IMPLEMENTATION

- STAFF RECOMMENDS THAT THE COMMISSION APPROVE THE ELEMENTS OF THIS PLAN BECAUSE IT PROVIDES MILESTONES AND DELIVERABLES; LINKAGES AMONG THE ACTIVITIES; EFFECTIVELY AND EFFICIENTLY USES RESOURCES.

CONCLUSIONS

- STAFF HAS RESPONDED TO THE ELEMENTS OF THE SRM
- ADVISORY COMMITTEES' INPUTS WILL BE REFLECTED IN THE PAPER TO BE ISSUED BY AUGUST 21, 2000
- INPUT SO FAR FROM INTERNAL AND EXTERNAL STAKEHOLDERS FAVORABLE TO ADOPTING THE HIGH-LEVEL GUIDELINES FOR PERFORMANCE-BASED ACTIVITIES
- ADVISORY COMMITTEES WILL RECEIVE REPORTS FOR INFORMATION

INDUSTRY INITIATIVES IN THE REGULATORY PROCESS

Presentation to the ACRS

June 8, 2000

R. H. Wessman
C. E. Carpenter
R. A. Hermann

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AGENDA

- PURPOSE
- BACKGROUND
- PROPOSED GUIDELINES
- RECOMMENDATIONS AND FUTURE ACTIONS
- CONCLUSIONS

PURPOSE

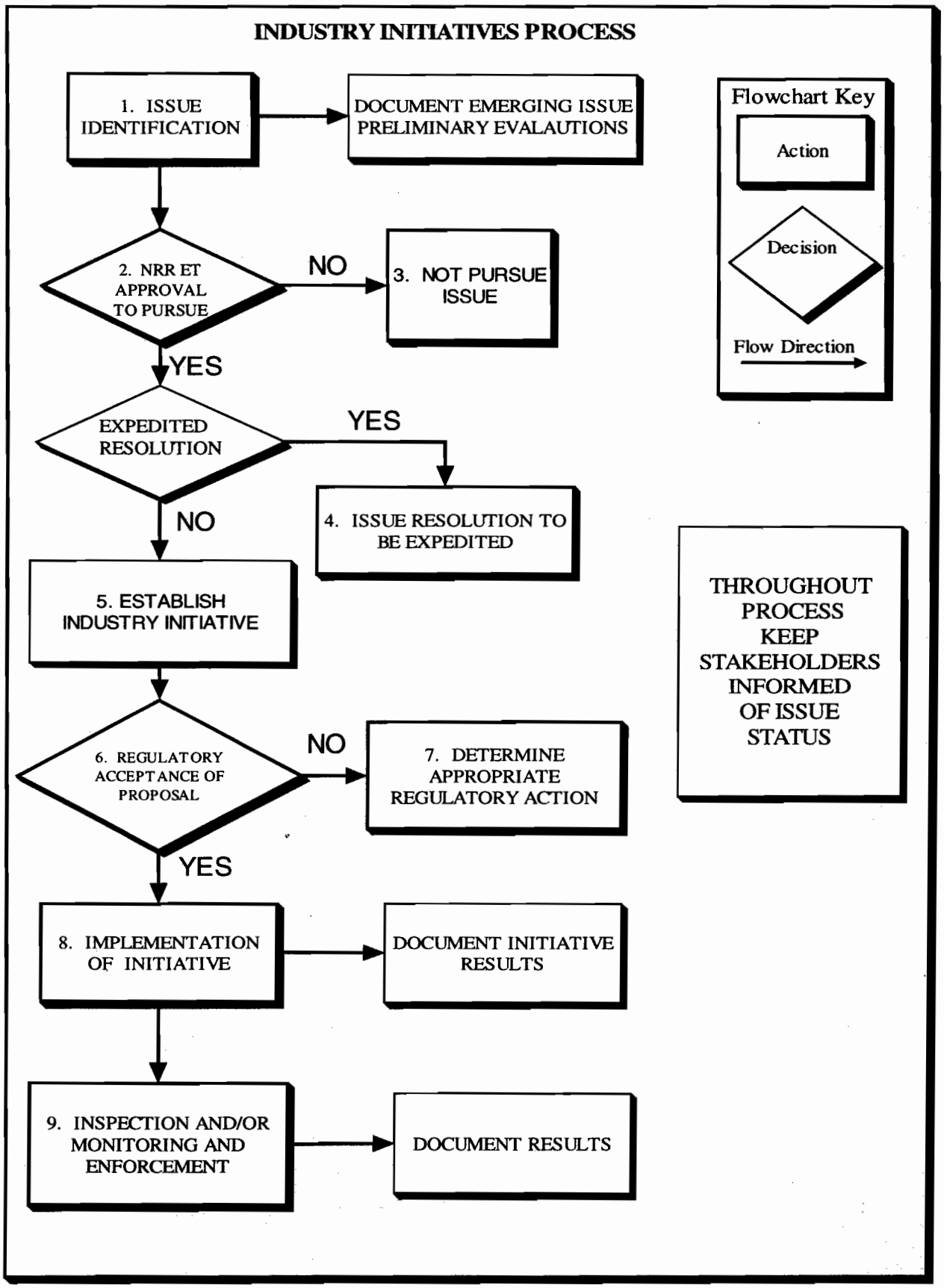
- Proposed Guidelines Intended To Ensure That Future Initiatives Proposed By Applicable Industry Groups (AIGs) Would Be Treated And Evaluated In A Consistent, Controlled And Open Manner and will
  - → Maintain Safety,
  - → Reduce Unnecessary Regulatory Burden,
  - → Improve Efficiency, Effectiveness, and Realism, and
  - → Improve Public Confidence

BACKGROUND

- Direction Setting Initiative 13, "The Role of Industry"
- SECY-99-063, "The Use by Industry of Voluntary Initiatives in the Regulatory Process," and Associated SRM
- Actions to Develop Proposed Guidelines
  - → Staff Met with Industry, NEI, and Other Stakeholders
  - → Staff Developed Web Page to Provide Information on Guidelines
  - → Staff Issued Federal Register Notice (FRN) (64 FR 69574) Soliciting Stakeholder Comments on Both Technical and Regulatory Aspects Related to Development of Guidelines to Allow Drafting of Regulatory Framework from Interested Stakeholders
  - → Final Proposed Guidelines Provided to Commission in SECY-00-0116, "Industry Initiatives in the Regulatory Process," dated May 30, 2000

PROPOSED GUIDELINES

- Definitions
  - → Type 1 and Type 2 Industry Initiatives:
    - ◊ Type 1: those developed by AIG(s) in response to some issue of potential regulatory concern (a) to substitute for or complement regulatory actions for issues within existing regulatory requirements, or (b) which are potential cost beneficial safety enhancement issues outside existing regulatory requirements;
    - ◊ Type 2: those that are initiated and developed by AIG(s) to address issues of concern to the AIG(s) but that are outside existing regulatory requirements and are not cost beneficial safety enhancements, or that are used as an information gathering mechanism
  - → Applicable Industry Group(s) (AIGs) could be the members of one or more Owners Groups, an industry organization (e.g., the Nuclear Energy Institute or the Electric Power Research Institute), or two or more licensees


INDUSTRY INITIATIVES PROCESS
1. ISSUE IDENTIFICATION
DOCUMENT EMERGING ISSUE PRELIMINARY EVALAUTIONS
2. NRR ET APPROVAL TO PURSUE
NO
3. NOT PURSUE ISSUE
YES
EXPEDITED RESOLUTION
YES
4. ISSUE RESOLUTION TO BE EXPEDITED
NO
5. ESTABLISH INDUSTRY INITIATIVE
6. REGULATORY ACCEPTANCE OF PROPOSAL
NO
7. DETERMINE APPROPRIATE REGULATORY ACTION
YES
8. IMPLEMENTATION OF INITIATIVE
DOCUMENT INITIATIVE RESULTS
9. INSPECTION AND/OR MONITORING AND ENFORCEMENT
DOCUMENT RESULTS
Flowchart Key
Action
Decision
Flow Direction
THROUGHOUT PROCESS KEEP STAKEHOLDERS INFORMED OF ISSUE STATUS

PROPOSED GUIDELINES

- Other Items
  - → Project Management
  - → Public Participation
  - → Communications Plan
  - → Resource Planning
  - → Fees
  - → Tracking of Commitments Consistent with Existing Regulatory Processes
  - → Enforcement Guidelines Consistent with Reactor Oversight Process Improvements

- Stakeholder Comments
  - → NEI's Views Regarding Proposed Process

RECOMMENDATIONS AND FUTURE ACTIONS

- Staff Requesting Commission's Approval To Issue Proposed Guidelines For Public Comment

- After Considering Further Stakeholder Comments, Staff Will Communicate Final, Revised Guidelines And Implement For Future Industry Initiatives

- Expected milestones are:
  - → Commission Approval to Issue Guidelines for Public Comment -- July 31, 2000
  - → Guidelines Issued for 45-day Public Comment -- August 31, 2000
  - → Comments Resolved and Final Guidelines Issued -- January 5, 2001

CONCLUSIONS

- Proposed Guidelines For Including Industry Initiatives In The Regulatory Process Provide Flexibility In The Form That Initiatives Might Take While Making Optimal Use Of Existing Regulatory Processes To Provide A Framework For The Efficient And Effective Use Of Initiatives To Resolve Issues And Maintain Safety

- Guidelines Provide For Public Participation In Process And For Making Information Related To Industry Initiatives Readily Available To All Stakeholders

Safety Culture

Presentation to the Advisory Committee on Reactor Safeguards

June 8, 2000

J. N. Sorensen

Viewgraphs - Rev. 9, 6/7/00

Safety Culture

What is it?

- IAEA/INSAG view

Why is it important?

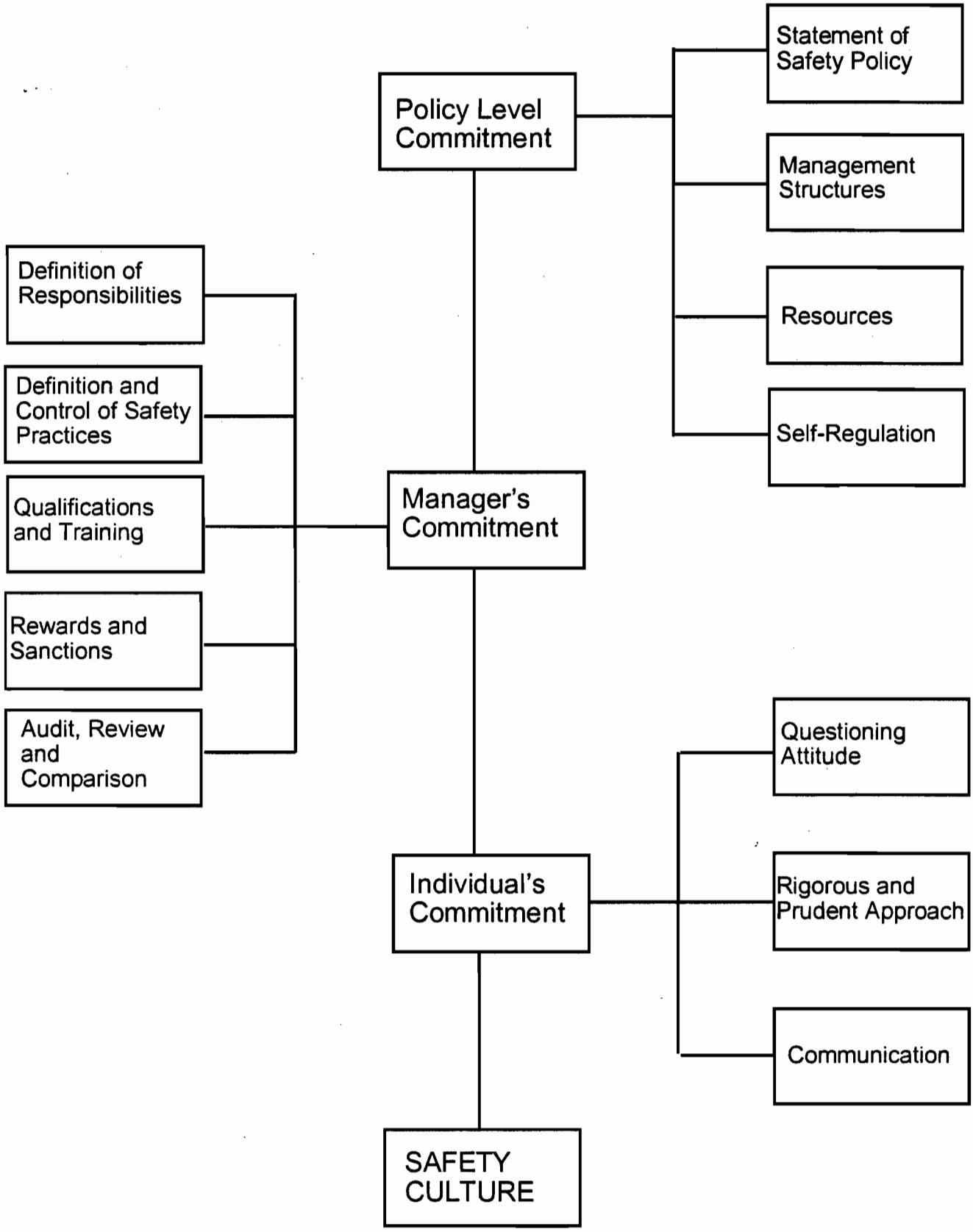
- Human performance improvement
- Latent errors
- ATHEANA needs

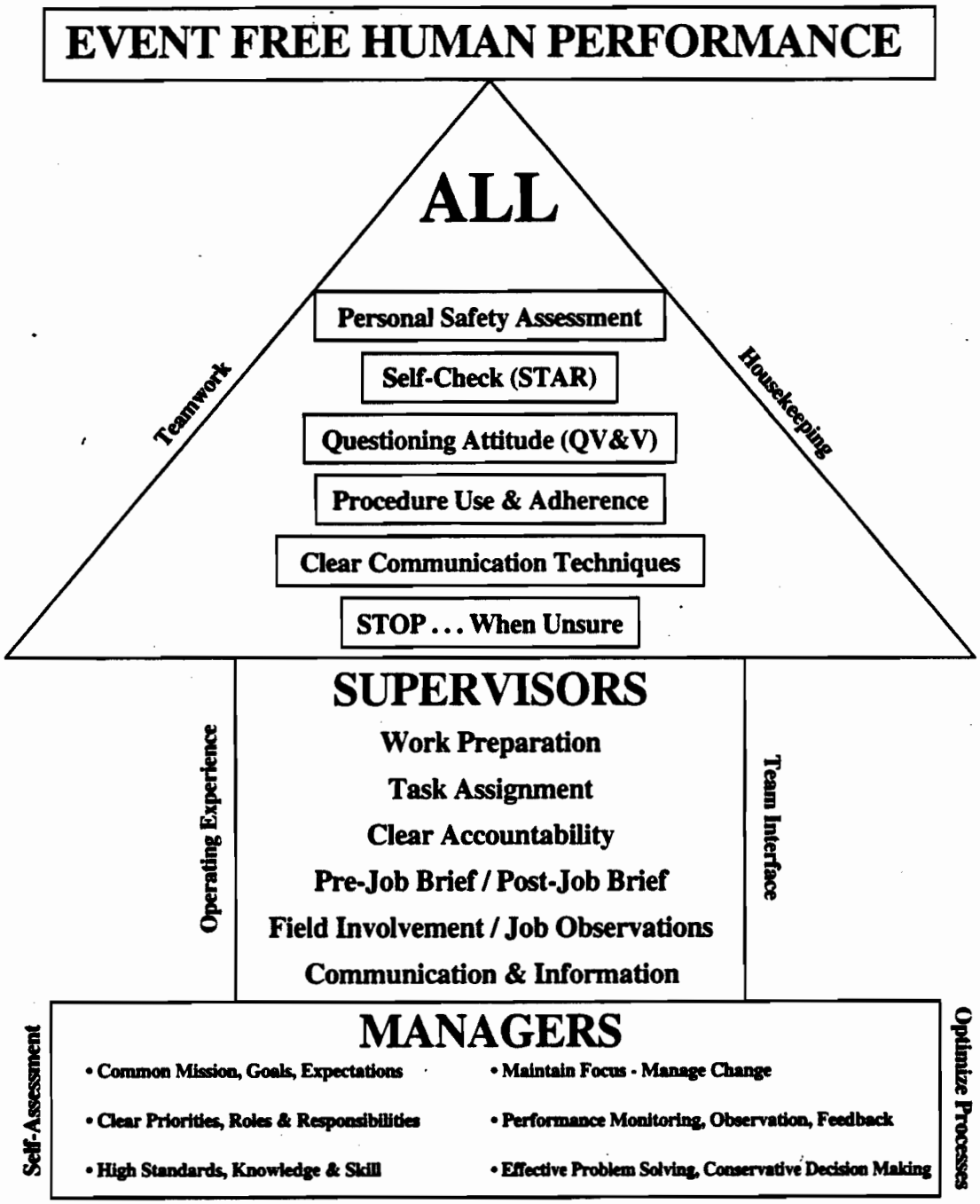
What can NRC do about it?

- Identify performance indicators
- Expand root cause analysis

ILLUSTRATION OF THE PRESENTATION OF SAFETY CULTURE

Figure 1 from INSAG-4, Safety Culture





Human Performance Model. This graphic is worn by workers at all three Duke Power–operated nuclear stations. The concepts identified on the perimeter of the arrow are intended to support the tools inside that section of the arrow. QV&V™ is a registered trademark of Performance Improvement International (PII). (Source: Duke Power)

Source: "The Human Performance Improvement Program at Duke Power Nuclear Stations," by Tom Shiel, Nuclear News, May 2000, American Nuclear Society

Duke Power
Human Performance Improvement Program

"If you analyze an entire event, . . . you'll find it wasn't just one mistake - - it was five, six or seven mistakes that occurred and there weren't enough contingencies or barriers built in to prevent the event from happening."

"This common cause assessment identified the need for focused human error reduction training for technicians and supervisors."

Quantitative Analysis of Risk Associated with Human Performance

- Study performed by Idaho National Engineering and Environmental Laboratory
- One objective was to identify the influence of human performance in significant operating events
- Analyzed 35 operating events, 20 using PRA methods
- Event importance ranged from 1.0E-6 to 5.2E-3 (Wolf Creek drain-down event)

INEEL Analysis and Findings

Most identified errors were latent - no immediate observable impact. Ratio of latent to active errors was 4:1

Latent Errors

- **Failure to correct problems**
  Known deficiencies, failure to respond to notices
- **Engineering problems**
  Design, design change testing, engineering evaluations were sources of failure
- **Maintenance problems**
  Maintenance practices, post-maintenance testing, work package QA & use.

Active Errors

- **Failures in command and control**
  Wrong actions, right people not present, loss of phone communications, actions independent of control room
- **Incorrect operator actions**
  Incorrect line-ups, failure to take actions when automatics fail, actions without procedural guidance, delay in performing cooldown

Important Management & Organization Factors
(Weil & Apostolakis, 1999)

- Communications
- Formalization
- Goal Prioritization
- Problem Identification
- Roles & Responsibilities
- Technical Knowledge

Work Process Analysis
(Weil & Apostolakis)

"The potential for organizational factors to lead to common cause failures is strongly suspected . . . ."

Poor work prioritization, for example, can lead to the failure of dissimilar components.

Important Safety Culture Indicators
ASCOT Guidelines

IAEA, through INSAG-4 and ASCOT guidelines, attempts to identify important aspects of safety culture and a process for finding tangible evidence of good safety culture.

INSAG-4 suggests ~150 questions regarding government, operating organization, and support organizations such as design & research. ASCOT adds ~ 300 guide questions

SKi STUDY

Used Expert Opinion to Identify Five Performance Indicators:

- Safety-significant Error Rate
- Maintenance Problem Rate
- Ratio of Corrective to Preventive Maintenance
- Rate of Problems with Repeated Root Cause
- Rate of Plant Changes Not Documented

Wolf Creek Drain-Down

Selected elements from ATHEANA analysis

- Incompatible work activities
- Compressed outage schedule
- Poor mental model of system valves
- Heavy reliance on control room crew to identify potential problems
- Inadequate review of procedures prior to use

New Reactor Oversight Program: Technical Framework for Licensee Performance Assessment

Cross Cutting Issues

- Human Performance
- Safety Conscious Work Environment
- Problem Identification & Corrective Action

"Risk-informed, performance-based regulation will ... involve a shift in the NRC role from improving human reliability to one of monitoring human reliability."

RECOMMENDATIONS

- Identify essential attributes of safety culture
- Identify associated performance indicators
- Ensure an effective root cause analysis process

Advisory Committee On Reactor Safeguards

SITE VISIT TO THE DAVIS-BESSE NUCLEAR POWER PLANT

June 8, 2000

Briefing by

Cognizant Staff Engineer:

Amarjit Singh

10

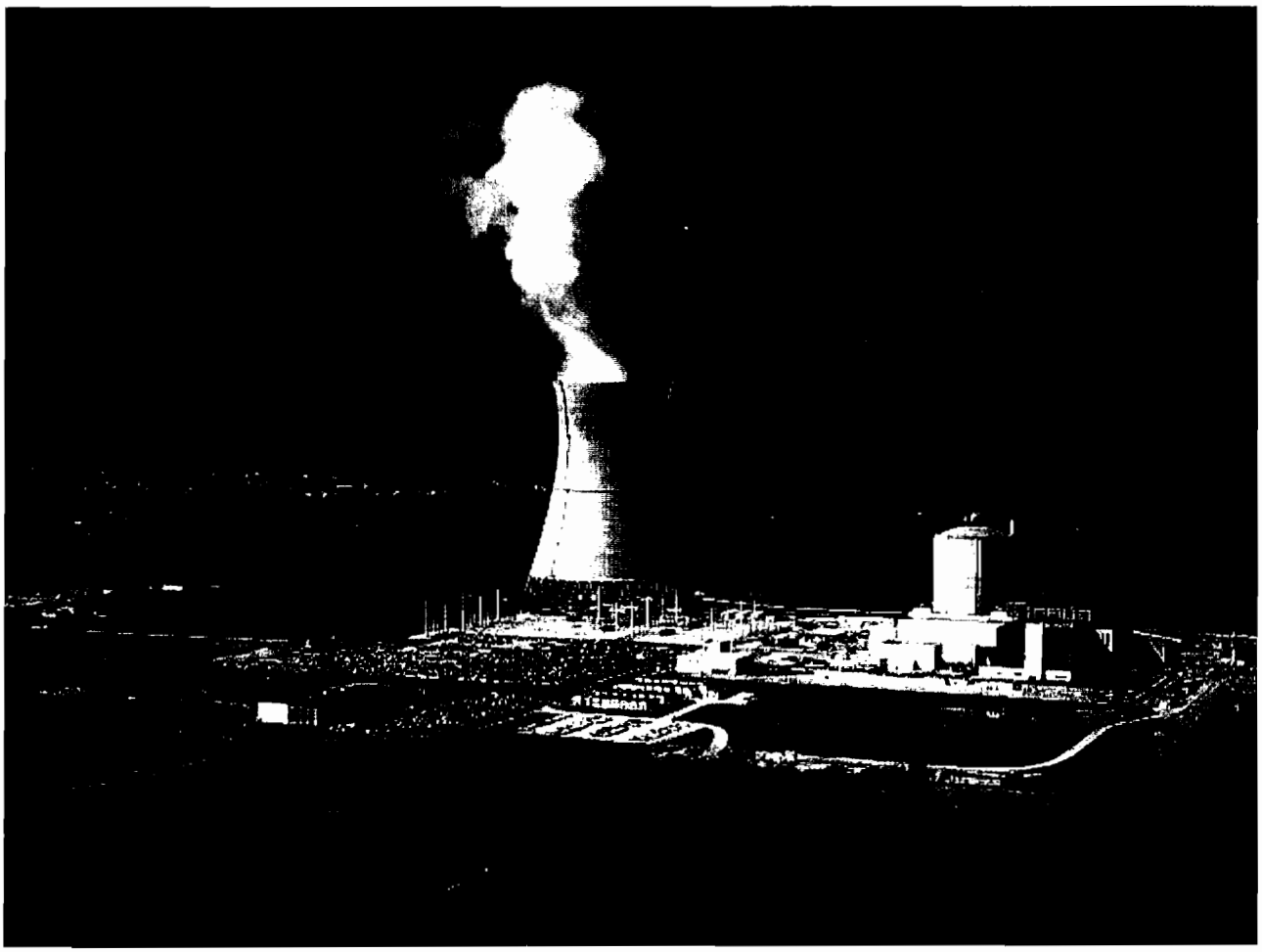
Plant Visit By ACRS
Purpose:

- To demonstrate that the advice provided by ACRS to the Commission is based on knowledge of plant operations encompassing more than "on paper" reviews
- To obtain the licensee's perspectives on issues which come before the ACRS for review

Plant Visit By ACRS

Purpose:

- To stay abreast of operational issues at the plant level including: technological enhancements; IPEEE validity; expert panels; spent fuel storage; and fire protection



Davis-Besse Nuclear Power Plant Background:

- Single unit site, consisting of a Babcock & Wilcox Pressurized Water Reactor, with a 873 MWe nominal rating
- Unit 1 original fuel load - April 27, 1977; commercial operation July 31, 1978
- Unit 1 operating at 100% power
- Unit 1, 50 day refueling outage ended 5/20/00

David-Besse Nuclear Power Plant Operating Issues:

- Licensee plugged 46 steam generator tubes and performed the chemical cleaning of the steam generators during the outage
- Licensee committed to submit Improved Technical Specification (ITS) by mid December 2000
- Implementation of Thermo-Lag Corrective actions completed in December 1998

Davis-Besse Nuclear Power Plant Operating Issues:

- Lacks full core offload capability necessary for reactor vessel in-service inspection
- Use of advanced fuel cladding and structural material (M5) in the reactor fuel starting next cycle
- The licensee plans to re-rack the existing spent fuel pool with the high density racks in the near future

Davis-Besse Nuclear Power Plant Operating Issues:

- Integrated containment leak rate test, 10-year in-service inspection, bearing replacement of #1 and seal replacement of #2 reactor coolant pump were performed during recent outage
- October 1999, #2 component cooling water pump had a short and was tripped, degradation mechanism most likely intrusion of ground water, however root cause remains undetermined

Findings Of Maintenance Rule Inspection:

Strengths

- The material condition of the plant systems examined was very good
- The licensee's self-assessment activities were appropriately conducted in and identified worthwhile issues in the area of quality assurance
- Use of independent personnel in the 1995 self-assessment was considered a strength

Findings Of Maintenance Rule Inspection:

Strengths

- System engineers were actively involved in maintenance rule implementation and generally viewed the maintenance rule as a useful tool
- Training for systems engineers was good

Findings Of Maintenance Rule Inspection:

Weaknesses

- The risk matrix used to assess plant risk while at power was viewed as weak in the following areas:
  - making use of outdated Individual Plant Examination (IPE) risk information
  - providing user guidance

Findings Of Maintenance Rule Inspection:

Weaknesses

- Even though the structures, systems, and components (SSCs) at Davis-Besse were properly scoped, documentation to support scoping decisions were weak
- Outdated IPE information and inadequate documentation of the expert panel's determinations were being utilized

Mid-cycle Plant Performance Review (PPR)

- Most recent PPR issued on September 30, 1999, indicated overall improvements in operational performance were achieved and sustained during the assessment period
- Performance in the plant support area, including radiological controls, chemistry, radiological environmental monitoring, and transportation of radioactive material, were positive
- Engineering department personnel provide good oversight of activities associated with their system

Individual Plant Examination Of External Events (IPEEE) Review

Fire Protection

- IPEEE submittal is under review in the area of fire protection
- Staff expects to issue the Safety Evaluation Report by November 2000

LICENSE RENEWAL PROCESS

Scoping

- Safety related SSCs
- Non safety related SSCs whose failure could prevent other SSCs from performing a safety function
- SSCs that perform safety functions in regulated events (EQ, ATWS, fire protection, PTS & SBO rules)

Screening

- Identification of intended function
- Identification of long lived passive structures and components

S/C Subject to Aging Effects?

- Possible but not expected	Confirmatory (one-time?) inspections
- Expected, CLB involves TLAA	Demonstrate CLB valid to 60 yrs, or Extend validity of CLB to 60 yrs, or Monitor/Inspect/Manage aging
- Expected	Map to CLB Programs/Activities: - Existing Programs, or - Modified Programs, or - New Program

GUIDANCE FOR ACRS REVIEW OF LICENSE RENEWAL STANDARD REVIEW PLAN AND SUPPORTING DOCUMENTS

1. **Are the SRP, the GALL II report and the NEI implementation documents effectively integrated? Do they provide a consistent and understandable process? Does the SRP provide a user friendly map of how these documents come together?**

2. **Is guidance adequate to support effective scoping/screening of older plants? Are the lessons learned from the review of the OCONEE LRA adequately conveyed to future reviewers?**

3. **Is guidance adequate to support effective scoping/screening of plants with a risk-informed classification of SSCs?**

4. **Is review of plant specific operating experience adequately emphasized by the SRP? Is guidance adequate to evaluate the effectiveness of plant programs dealing with unique types of plant specific aging degradation?**

5. **Does the SRP provide a process for raising questions regarding the adequacy of the CLB? (Aging reviews lead to question the bases for plant programs based on old analyses, for example, EQ of electrical equipment in harsh environment. If the reviewer finds the bases inadequate, is his burden to re-open the CLB issue too great?)**

6. **Have the SRP and supporting documents taken into proper consideration the issues and concerns raised by all stakeholders?**

7. **Has the interactive involvement of the industry resulted in the expected improved process without loss of effectiveness?**

8. **Are the license renewal generic issue resolutions adequately reflected in the guidance documents?**

PROPOSED LICENSE RENEWAL SUBCOMMITTEES

Subcommittee A	Subcommittee B
M. Bonaca Chairman	R. Seale, Chairman
G. Apostolakis	T. Kress
J. Sieber	D. Powers
W. Shack	R. Uhrig
	G. Wallis

G Drive:PP

ACRS MEETING HANDOUT (13)

Meeting No.	Agenda Item	Handout No:
473rd	15	15.1

Title MINUTES OF PLANNING & PROCEDURES SUBCOMMITTEE MEETING - JUNE 6, 2000

Authors JOHN T. LARKINS

List of Documents Attached

15

Instructions to Preparer

1. Punch holes
2. Paginate attachments
3. Place copy in file box

From Staff Person

JOHN T. LARKINS

June 6, 2000

MINUTES OF THE
PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
TUESDAY, JUNE 6, 2000

The ACRS Subcommittee on Planning and Procedures held a meeting on June 6, 2000, in Room 2 B1, 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 4:00 p.m.

ATTENDEES

D. A. Powers, Chairman
G. Apostolakis
M. Bonaca

ACRS STAFF

J. T. Larkins
H. Larson
R. P. Savio
S. Duraiswamy
C. Harris
S. Meador

EDO Staff
G. Millman

DISCUSSION

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the June ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the June 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 June 2000 ACRS meeting be as shown in the handout. Since the Commission paper on PRA Quality has not been provided to the Committee by the staff, the Committee should defer its report on this matter to the July ACRS meeting. The Committee should try to complete the discussion of proposed ACRS reports by close of business Thursday, June 8, 2000.

On Friday, after approving all ACRS reports, as needed, the Committee should discuss:

- Risk-informed regulation - general
- NEI letter
- Safety Culture (need to define a Committee approach to the issue)
- Agenda for the meeting with the Commission in October (are there topics we are prepared to "advocate" to the Commission?)

Time permitting, it might be useful also to discuss:

- Power uprates and the meaning of the loss of margin
- Strategies for the review of research; who can do what, when
- Need for risk-informed changes to the regulation to pass the backfit rule

2) Anticipated Workload for ACRS Members

The anticipated workload of the ACRS members through September 2000 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. Section II of the Future Activities list and the Subcommittee's recommendations are attached to the anticipated workload document.

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee needs to decide on the Subcommittee's recommendations on items listed in Section II of the Future Activities.

3) Meeting with Industry Representatives

During the January 2000 retreat, the ACRS discussed ways in which the Committee could interact with industry, including NEI, INPO, and utilities, to obtain information on significant industry issues. Dr. Apostolakis and Dr. Savio were tasked with making arrangements with NEI for a discussion between ACRS and NEI at a future ACRS meeting on NEI regulatory initiatives. The discussion has been tentatively scheduled for the October 2000 ACRS meeting.

The ANS is sponsoring its annual Utility Working Conference 2000 at Amelia Island Plantation, Florida, August 6-10, 2000 [Note: The members are scheduled to visit Naval Reactor Training Center at Charleston, SC, on August 7, 2000].

RECOMMENDATION

The Subcommittee recommends that Dr. Apostolakis and Dr. Savio continue to work with NEI and develop an agenda. Topics proposed by the Subcommittee members include: Industry Standards for PRA Quality, and NEI views on the Role of ACRS in the regulatory process.

The Subcommittee recommends that a member and/or an ACRS staff attend the Utility Working Conference 2000.

4) . Meeting With the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners between 9:30 and 11:30 a.m. on Thursday, October 5, 2000 at the Commissioners' Conference Room, One White Flint North. During July, we need to send a list of topics for Commission feedback and approval. A list of topics proposed by the ACRS staff is as follows:

- Risk-Informed Regulation
  - Option 2 and Option 3 activities
  - NEI letter dated January 19, 2000
  - PRA quality
- Status of ACRS Activities Associated with License Renewal
- Safety Culture at Nuclear Power Plants

Dr. Powers has agreed to develop an alternative list of topics for consideration by the Committee.

RECOMMENDATION

The Subcommittee recommends that the Committee approve a list of topics, and that teams be assigned to develop a presentation package on each subject.

5) AP1000 Advanced Reactor Design

The staff has started the Phase 1 review of the AP1000 advanced reactor design. During this phase, the staff will identify questions and policy issues that will be evaluated during the Phase 2 (feasibility) review. During the May 2000 meeting, the Committee suggested that the members identify issues that they believe should be evaluated by the staff during Phase 2.

On May 25, 2000, Mr. Dudley compiled a list of issues that were raised by the Committee during its review of the AP600 design and sent it to the members for review

and comment. The attached list (pp. 1-3) reflects incorporation of comments received from the members so far.

RECOMMENDATION

As recommended by the Subcommittee, Mr. Dudley prepared a consolidated list of issues that was distributed to the members on Wednesday, June 7, 2000.
The Subcommittee recommends that the Committee discuss and approve the list of issues to be sent to the NRC staff by the ACRS/ACNW Executive Director.

6) Technical Expertise Needed for Future ACRS Members

The Commission has recently selected Mr. Graham Leitch to be appointed as a new member to the ACRS. Subsequent to completing all necessary paper work, he will be appointed to the ACRS. Mr. Leitch is expected to attend the September ACRS meeting either as a member or as an observer.

The Commission also asked the ACRS/ACNW Executive Director to identify the specific technical expertise that will be sought to fill the existing and upcoming vacancies on the Committee. Based on input from the Commissioners and the members, we have identified such expertise (Attachment p. 4).

RECOMMENDATION

The Subcommittee recommends that the Committee provide feedback on the specific expertise identified by the ACRS/ACNW Executive Director.

7) Proposed Assignment and Guidance for Reviewing License Renewal Guidance Documents

The staff is in the process of preparing a Standard Review Plan, Generic Aging Lessons Learned II (GALL II) Report, and a Regulatory Guide associated with license renewal. The Committee needs to complete its review of these documents in November 2000. Dr. Bonaca, Chairman of the Plant License Renewal Subcommittee, has proposed assignments for the members for reviewing these documents. These assignments were distributed to the members for comment during the May ACRS meeting. The attached revised list of assignments (pp.5-6) reflects incorporation of comments received from the members so far.

During the May meeting, the Committee also suggested that Dr. Bonaca identify "high level" issues as well as develop a guidance for use by the members in reviewing these documents. The "high level" issues and guidance developed by Dr. Bonaca and Mr. Dudley are attached (pp. 7-8).

RECOMMENDATION

The Subcommittee recommends that the Committee approve these assignments and guidance during the June meeting. After receiving the documents in August, the members should review portions of the documents assigned to them and provide comments to Dr. Bonaca prior to the October full Committee meeting.

In view of the anticipated heavy workload associated with license renewal, Dr. Bonaca should propose two teams of members for reviewing license renewal applications and associated matters in the future. The two teams proposed by Mr. Dudley and Dr. Bonaca are attached (pp. 9). The Committee should comment on this proposal. In addition, Dr. Bonaca should identify technical areas associated with the license renewal guidance documents for which the Committee will need to augment its expertise by using consultants.

8) ACRS/ACNW Self Assessment

The Commission paper (SECY-00-0102) on the CY 1999 self-assessment of ACRS and ACNW performance was issued on May 5, 2000. The commitments from this self assessment that will require follow-up and the proposed assignments of lead responsibility are attached (pp. 11-13). Commitments from the ACNW self-assessment are attached for information. Dr. Larkins and Dr. Savio will track actions related to these commitments and provide periodic reports to the Planning and Procedures Subcommittee.

Recommendation

The Subcommittee recommends that the Committee endorse the commitments and lead assignments on this ACRS list.

9) ACRS Memorandum of Understanding

A draft Memorandum of Understanding between the ACRS and the EDO was provided to the ACRS members during the May ACRS meeting and the members were asked to provide comments to Dr. Larkins or Dr. Savio before the June ACRS meeting. No comments have been received. The EDO coordinator to the ACRS has provided some editorial comments and the MOU has been modified. A copy of the revised MOU is attached (pp. 14-19). The EDO will now provide this draft to NRR, NMSS, RES, and OGC for review. Dr. Larkins and Dr. Savio will keep the Committee informed.

RECOMMENDATION

The revised draft MOU is provided for information. No Committee action is required.

10) Letter from Gordon Thompson on Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

Mr. Gordon Thompson forwarded a letter to Dr. Powers raising issues on Spent Fuel Pool Accidents (pp. 20-99). In his letter, he makes comparisons between the NRC staff's technical study for decommissioning plants and operating plants. He recommends that the ACRS take two actions: (1) independently investigate both operating and decommissioning plants and (2) recommend that the NRC immediately initiates a comprehensive investigation of the state of knowledge on scientific issues relevant to the risk posed by spent fuel pools. He provided a report that he prepared for the NRC's Licensing Board. Additionally, he volunteers to discuss these matters with the ACRS.

RECOMMENDATION

The Subcommittee recommends that the ACRS Chairman with the support of the ACRS/ACNW Executive Director respond appropriately to Mr. Gordon Thompson. Also, Mr. Thompson should be invited to brief the ACRS at the September 2000 meeting during which the Committee is scheduled to discuss the spent fuel pool accident risk at decommissioning plants.

11) OTHER ISSUES

Based on discussion of other issues, the Subcommittee makes the following recommendations:

a. The trip to Germany to meet with RSK previously scheduled for June 2000 has been postponed. The Committee should find out which members are interested in attending this meeting when scheduled. Also, the topics for this meeting should be identified.

b. Drs. Apostolakis and Bonaca will attend the ASME workshop scheduled for June 27, 2000, in Rockville to discuss the proposed final ASME Standard for PRA Quality.

c. The members should develop a list of issues associated with power uprates to be sent to the NRC staff for discussion at a future meeting of the Thermal-Hydraulic Phenomena Subcommittee. This list should be provided to the Planning and Procedures Subcommittee for discussion during its September meeting.

d. Mr. Sieber should recommend whether he and Mr. Singh should attend the Fire Protection Conference in London scheduled for February 12-14, 2001. Attendance would be contingent upon presenting a paper during this conference.

e. Dr. Apostolakis will propose assignments for the members for reviewing the proposed final ASME Standard (Phase 1) and the proposed ANS Standard (Phase 2) for PRA Quality, which are scheduled for discussion during the July and September 2000 meetings, respectively.

Document Name: G:\PlanPro\ppmins.473.wpd

FOR COMMENT

PROPOSED ACRS ISSUES RELATED TO THE AP1000 DESIGN

1. Guidance for acceptable scaling methods, such as the Code Scaling , Applicability, and Uncertainty (CSAU) evaluation methodology, and for acceptable utilization of integral test data for the validation of computer codes should be developed.

2. Establish the scope of additional analyses needed for the SSAR Chapter 15 accidents. Revised codes used in the analyses may need to be revalidated.

3. Clear identification in the SSAR is needed of the inadequacies in the NOTRUMP code and the steps taken to compensate for them. A rigorous demonstration of the applicability of the revised NOTRUMP code to the AP1000 design is needed.

4. More experiments or analyses will be required before in-vessel core debris retention can be credited as part of the licensing basis.

5. Since in-vessel retention is widely considered to be an important accident management strategy for operating reactors, the impact of intermetalic exothermic reactions on this strategy should be reassessed.

6. Westinghouse has chosen the thermal-hydraulic conditions of a specific sequence (i.e., a direct vessel injection line break) for use with the DBA source term to take credit for diffusiophoresis and thermophoresis. It is not clear that the thermal-hydraulic conditions of the selected sequence is consistent with the desired generality of the source term.

7. The applicability of the AP600 probabilistic risk analysis to the AP1000 must be demonstrated. Consideration should be given to the following:

   a. In-containment aerosol behavior and the effects of particle charging on aerosol behavior,

   b. Catastrophic failure of the steel shell containment,

   c. Containment bypass accident sequences and especially mitigation of steam generator tube rupture accidents,

   d. Reactor coolant system depressurization reliability, and the

   e. Efficacy and reliability of external cooling of the containment shell,

PLEASE PROVIDE YOUR COMMENTS TO NOEL OR SAM.

- INTERNAL USE ONLY -

FOR COMMENT

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PLEASE PROVIDE YOUR COMMENTS TO NOEL OR SAM.

- INTERNAL USE ONLY -

ACRS ISSUES RELATED TO THE AP1000 DESIGN

The following issues were included in ACRS letters related to the AP600 advanced reactor design. The letters were issued on March 22, 1999; July 23, June 15, April 9, February 19, 1998; and June 17, 1997.

1. Guidelines on the acceptable quality of documentation submitted by the applicant and on the lead times necessary for staff reviews should be established and enforced.

2. Safety evaluation reports should include more of the technical rationale leading to the regulatory decision.

3. Guidance for acceptable scaling methods, such as the Code Scaling , Applicability, and Uncertainty (CSAU) evaluation methodology, and for acceptable utilization of integral test data for the validation of computer codes should be developed. [Note: The staff has developed a Standard Review Plan Section and draft regulatory guide to address this matter.]

4. The probability of containment failure for the AP600 is approximately 0.01. Deformation of the pressurized containment vessel during severe accidents conditions and its interaction with the shield building could also induce leakage and further increase the likelihood of failure.

5. Additional information is needed on the results of the break area sensitivity study for one of the severe small-break LOCAs to ensure that the process for compensating for exclusion of momentum flux terms in the NOTRUMP code is robust for a range of blowdown rates.

6. A description in SSAR Chapters 4 and 15 is needed regarding the interrelationships among the LOFTRAN, THINC-IV, and WESTAR codes and the test data. Clear identification is needed of channel-to-channel mixing coefficients to be used.

7. Clear identification in the SSAR is needed of the inadequacies in the NOTRUMP code and the steps taken to compensate for them. A rigorous demonstration of the applicability of the NOTRUMP code the AP1000 design is needed. The code documentation appears in need of significant revision.

8. More experiments and analyses will be required before in-vessel core debris retention can be credited as part of the licensing basis. [Note: Current severe accident research indicates that ex-vessel debris retention is problematic for 1000 MWe designs.]

9. Since in-vessel retention is widely considered to be an important accident management strategy for operating reactors, the impact of intermetalic exothermic reactions on this strategy should be reassessed.

10. The NRC staff should ensure that the methodologies used by the Combined License applicant to derive χ/Q not mask the effects of any unique site meteorological characteristics related to topology, geographical location, directed wind flows during specific times of the day, or any peculiar atmospheric inversion characteristics.

11. Westinghouse has chosen the thermal-hydraulic conditions of a specific sequence (i.e., a direct vessel injection line break) for use with the DBA source term to take credit for diffusiophoresis and thermophoresis. It is not clear that the thermal-hydraulic conditions of the selected sequence is consistent with the desired generality of the source term.

12. We believe, however, that the spray design concept suggested by the staff is marginally adequate.

ADDITIONAL ISSUES

The following additional comments were provided by Dr. Wallis:

13. The AP600 PRA led to a CDF of 1.7 X 10^{-7}. The SAMDA document contains a claim by Westinghouse that the value of a "100% Effective Design Alternative" was $47. In other workds the net risk cost to society of having one of theses things run for its lifetime with its present design is only $47, or less than the risk costs of a teenaged male driving a sports car for about a week. I think this is a ludicrous claim.

    Moreover, since the failure probability of containment is usually around 0.1, it seems that the present containment is only worth about $470 to society, so that the risk-benefit balance is off by a factor of around a million. Looked at another way, if all present reactors were AP600s the NRC would be administering a total societal risk of around $5,000 for the life of the plants and would only need a part time employee. Something appears to be just palin wrong and it undermines the credibility of the whole exercise.

14. Westinghouse claims that it will use the same codes as for AP600 and that "revalidation is not required." During the review of the AP600 the NOTRUMP code had all sorts of typos in the equations in its documentation. It also contained a lot of judgment-based assumptions with no justification except that the author chose to make them. Is this still acceptable?

The following additional comments were provided by Dr. Powers:

15. In-containment aerosol behavior and the effects of particle charging on aerosol behavior,

16. Catastrophic failure of steel shell containments: experimental data vs. analysis,

17. Ability to control pH in containment sump under accident conditions,

18. Fire risk assessment,

19. Assessment of risk during low power and shutdown operations,

20. Facility security and safeguards,

21. Containment bypass accident sequences and especially mitigation of steam generator tube rupture accidents,

22. RCS depressurization reliability,

23. Analysis of containment vulnerability to hydrogen combustion events including transitions from deflagrations to detonations, and shock wave focusing,

24. Efficacy and reliability of external cooling of the containment shell,

25. Control rod sticking in high burnup fuel assemblies; handling of fuel especially at high burnup,

26. Automation versus human actions in EOPs and beyond design basis accidents, and

27. Common mode failures of instrumentation and control systems.

AREAS AND [illegible]TING COVERAGE BY ACRS MEMBERS

	Existing Member Technical Expertise[1]								
3	TSK	GML	DAP	RLS	WJS[3]	JDS	REU	GBW	CONSULTANT USE[4]
		X		x		X	X		
	x	X	x	x	x	X	X	x	
	X	X	X	X	x	X	X		
	x		x	x	x		X		
	x	x	x					x	x
		x		x		x			x
	x			x				X	
			x			x			
			x	x	X				
							X		x
			X	x	x				x
	X		X			x	x	x	
	X		X						
					x	x			X
					x				X

ndicates a working knowledge in the area. In the case of consultants, a lower case "x" needed to support the ACRS members.

· ACRS member William Shack has conflicts of interest which limit his involvement in

he ACRS.

ting plant experience, detailed knowledge of plant systems and operating procedures

LICENSE RENEWAL
PLAN FOR REVIEWING GUIDANCE DOCUMENTS

The staff plans to brief the Plant License Renewal Subcommittee in October 2000 concerning drafts of the Standard Review Plan, Generic Aging Lessons Learned II Report, and Regulatory Guide related to preparation and review of license renewal applications. The Nuclear Energy Institute (NEI) has revised its NEI 95-10 Report , which provides guidance to licensees concerning implementation of the requirements for preparing a license renewal application. The staff plans to review and endorse NEI 95-10 in a regulatory guide.

The staff has held meetings with NEI and the industry concerning these documents and plans to issue draft documents for public comment in August 2000. The staff plans to hold a public workshop in September and brief the Committee at the November 2000 ACRS meeting.

All ACRS members should participate in the review of these generic guidance documents, since they may become members of the Plant License Renewal Subcommittee in the future. To ensure the Committee members have sufficient time to conduct a thorough and integrated review of these document, the following course of action is recommended:

- assign Members primary responsibilities for reviewing specific portions of the documents,
- provide pre-draft documents to the members in May 2000,
- discuss Committee approach for reviewing generic documents at the June ACRS meeting,
- provide draft public comment generic documents to the members in August 2000,
- schedule a half an hour session at the September ACRS meeting to discuss reviewing the documents (NRR will provide an overview),
- members attend the September NRC workshop,
- schedule a half an hour session at the October ACRS meeting to discussed members' issues and concerns,
- Plant License Renewal Subcommittee meeting in October to review generic documents,
- review and comment on the documents at the November 2000 ACRS meeting, and
- review proposed final documents at the March 2001 ACRS meeting.

Attachments:
1. ACRS Assignments for License Renewal Guidance Documents
2. Guidance for ACRS Review of Generic Documents
3. Image of Presentation Slide

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5

ACRS REVIEW OF LICENSE RENEWAL SAFETY EVALUATION REPORTS

REVIEW ITEMS	Assigned Member	GALL II Chapters	Regulatory Guide	Standard Review Plan Sections		
Introduction or Administrative Information	MVB RLS	I		1.0		
Scoping and Screening Methodology	MVB RLS			2.1		
Plant	JDS DAP			2.2		
Reactor Coolant System	WJS RLS	IV		2.3	3.2	4.2
Engineered Safety Features	MVB TSK	V		2.3	3.3	4.2
Auxiliary Systems	JDS TSK	VII		2.3	3.4	
Steam and Power Conversion	JDS TSK	VIII		2.3	3.5	
Structures	JDS DAP	III		2.4	3.6	
Electrical and I&C	GA REU	VI		2.5	3.7	4.4
Time-Limiting Aging Analyses	MVB DAP					4.1 4.8
Reactor Vessel	WJS RLS	IV				4.2
Containment	WJS GBW	II				4.6
Branch Tech. Positions	ALL					

LICENSE RENEWAL
GUIDANCE FOR ACRS REVIEW OF GENERIC DOCUMENT

The SRP provides guidance on an acceptable method for applying the scoping and screening criteria to identify the long lived passive structures and components. The SRP provides guidance on how to reference the GALL II report to identify the aging effects and the acceptable aging management programs or activities. See the "Industry Process" side for a diagram of this process.

Items to consider during review of the license renewal generic documents and the proposed licenses renewal process:

- Relevance of the guidance for applying the scoping and screening criteria to identify the passive long lived structures and components to older plants. How lessons learned from the review of the Oconee license renewal application will be conveyed to future reviewers.

- How operating experience should be used on a plant specific basis. How should identified component degradation or aging be used to determine the adequacy of an existing aging management program.

- How to credit or validate engineering analyses that justify the existing licensing basis. An example is the adequacy of harsh environments analyses that have not been revised based on identified errors in plant specific early analyses.

- How well the Standard Review Plan, Generic Aging Lesson Learned II report, and NEI implementing documents provide a consistent and understandable process.

- Relationship and consistency between the generic license renewal guidance documents.

INDUSTRY PROCESS

Plant Systems Structures Components

Apply Scoping/Screening Criteria

Identify Long Lived Passive Structures and Components

↓

Identify Aging Effects and Functions

↓↑↓↑

Map to CLB Programs/Activities

Existing Program/Activity

Modified Program/Activitiy

New Program/Activitiy

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PROPOSED LICENSE RENEWAL SUBCOMMITTEES

The following is the proposed membership of two Subcommittees that will review license renewal applications beginning in fiscal year 2002.

Subcommittee A

M. Bonaca Chairman
J. Sieber
W. Shack
G. Apostolakis

Subcommittee B

R. Seale, Chairman
D. Powers
T. Kress
G. Wallis
R. Uhrig

Commitments from CY99 ACRS Self Assessment

1) Modify ACRS/ACNW Operating Plan in accordance with new NRC planning initiatives, draft FY2000-2005 Strategic Plan, and FY2001 Performance Plan, with incorporation of self assessment information and metrics. (Larkins/ Savio/ Gallo-- Issue to Commission by 11/30/2000)

2) Develop action plan that will identify and allocate resources for ACRS and ACNW review of selected decommissioning issues and use the Joint ACRS/ACNW Subcommittee to help coordinate work on decommissioning issues. (Outlined in split of ACRS and ACNW responsibility decommissioning paper and will be developed in ACRS/ACNW Operating Plan. Decommissioning paper will be updated as needed to account for new initiatives and schedule changes. Activities will be incorporated in Future Activities scheduling using existing processes.))

3) Maintain awareness of need to preserve independence, re. early involvement in the development of NRC staff positions. (P&P Subcommittee oversight)

4) Return to a mode of operation that will afford more in-depth review of issues when warranted. (P&P Subcommittee oversight)

5) Look for more opportunities to increase involvement in important technical issues and minimize involvement in routine matters such as rules and regulatory guides addressing routine technical or process issues. The examples of important technical issues given in the self assessment SECY were:

a) risk-informed initiatives for improving regulation (10 CFR Part 50, pressurized thermal shock, and decommissioning)
b) future NRC research needs
c) risk-based performance indicators
d PRA quality standards
e) human performance
f) digital I&C
g) transient and accident analyses code certification
h) emerging uses of mixed-oxide and high-burnup fuels

To conserve resources ACRS would end its review efforts when technical issues have been satisfactorily resolved and staff is addressing implementation. (P&P Subcommittee prioritization and scheduling of ACRS activities)

6) Systematically assess how ACRS, as a Commission-level advisory committee, can add value to an issue prior to agreeing to reviewing the issue. (P&P Subcommittee oversight of

proposed ACRS activities, with increased use of identified review objectives and action plans providing an assessment of resource use)

7) Test and refine streamlined process for ACRS review of license renewal application. (Plant License Renewal Subcommittee)

8) Take actions to maintain and improve ACRS awareness of plant operations issues. (Larkins, Savio, and Plant Operations Subcommittee)

9) Solicit and address feedback on how annual research report can be made more useful to Commission and staff. (Safety Research Subcommittee)

Commitments from CY99 ACNW Self Assessment

1) Modify ACRS/ACNW Operating Plan in accordance with new NRC planning initiatives, draft FY2000-2005 Strategic Plan, and FY2001 Performance Plan, with incorporation of self assessment information and metrics. (Larkins/ Savio/ Gallo-- Issue to Commission by 11/30/2000)

2) Develop action plan that will identify and allocate resources for ACRS and ACNW review of selected decommissioning issues and use the Joint ACRS/ACNW Subcommittee to help coordinate work on decommissioning issues. (Outlined in split of ACRS and ACNW responsibility decommissioning paper and will be developed in ACRS/ACNW Operating Plan. Decommissioning paper will be updated as needed to account for new initiatives and schedule changes. Activities will be incorporated in Future Activities scheduling using existing processes.)

3) Continue to work with NMSS staff in keeping informed through meetings with NMSS staff, through attendance at public meetings, and through access to predecisional documents (Lead ACNW member/ technical staff).

4) Address Commissioners comments related to agenda for and presentations at public Commission meetings. (ACNW Chairman/Larkins)

5) Use video teleconferencing to provide enhanced interactions with the Nevada and other external stakeholders (Lead technical staff)

MEMORANDUM OF UNDERSTANDING

PARTIES: Advisory Committee on Reactor Safeguards (ACRS) ~~- ACRS Chairman~~
Executive Director for the Advisory Committee on Reactor Safeguards (ACRS) and the Advisory Committee on Nuclear Waste (ACNW)

Nuclear Regulatory Commission Staff - Executive Director for Operations (EDO)

SUBJECT: ~~ACRS REVIEW AND COMMENT ON NUCLEAR SAFETY MATTERS~~
ACRS PARTICIPATION IN THE DEVELOPMENT OF NRC RULES, POLICY, AND SAFETY-RELATED GUIDANCE

~~PURPOSE:~~

~~The purpose of this memorandum is to establish procedures for ACRS and NRC interaction in the ACRS review of nuclear safety matters under development by the NRC staff~~.
The purpose of this Memorandum of Understanding (MOU) is to establish a process for ensuring that ACRS views are solicited in the early development of NRC rules, policy, and safety-related guidance. This ~~memorandum is to~~: MOU:

- ~~Specify~~ Identifies those ~~matters~~ areas ~~that are~~ within the ~~purview of the~~ ACRS. scope of responsibility.

- Establishes ~~the~~ a process~~es~~ ~~which will be used to~~ for keeping the ACRS informed of matters within its ~~purview.~~ responsibility.

- Establishes a proce~~dure~~ss for ensuring that ACRS review of matters within its ~~purview~~ responsibility is at a sufficiently early stage to permit effective and efficient interaction.

- ~~Provide guidance which will~~ Establishes a process to enable the ACRS and the NRC staff to establish plans and schedules that ~~satisfy~~ address the needs of the Commission, the NRC staff, and the ACRS~~, the NRC staff, and the Commission~~.

~~These procedures facilitate the NRC staff and ACRS interactions. Deviations from these procedures may at times be needed to carry out the NRC's mission. When this occurs, the procedures can be altered consistent with the needs of the NRC and the ACRS. Such changes will be implemented after being mutually agreed upon by the EDO and the ACRS/ACNW Executive Director.~~ {Moved to Section 7}

1. AREAS WITHIN THE ACRS SCOPE OF RESPONSIBILITY

a. <u>NRC Regulations</u>

The scope of ACRS responsibility encompasses ~~matters relating to~~ the following parts of NRC's regulations ~~(~~found in Title 10 of the Code of ~~f~~Federal ~~r~~Regulations~~)~~.

- Part 20 Standards for Protection Against Radiation
- Part 21 Reporting of Defects and Noncompliance
- Part 26 Fitness for Duty Programs
- Part 50 Domestic Licensing of Production and Utilization Facilities
- Part 52 Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants
- Part 54 Requirements for Renewal of Operating Licensees for Nuclear Power Plants
- Part 70 Domestic Licensing of Special Nuclear Material
- Part 72 Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste
- Part 73 Physical Protection of Plants and Materials
- Part 76 Certification of Gaseous Diffusion Plants
- Part 100 Reactor Site Criteria

b. Regulatory Activities

Regulatory activities that are within the ACRS scope of responsibility include:

- Reactor safety-related policy matters and rules
- Reactor safety-related regulatory guides and other regulatory guidance
- Prioritization and resolution of generic safety issues
- License applications and applications for license renewals
- Risk-informed and performance-based regulation
- NRC sponsored research
- Reactor transient and accident analysis code certification
- Reactor licensee performance assessment and the analysis of plant operating experience
- Regulatory burden reduction initiatives
- Development of regulatory requirements associated with the use of new technology

2. NRC STAFF/ACRS COORDINATORS

NRC staff coordinators ~~contacts~~ will be established in NRR, RES, and NMSS to coordinate the provisions of this MOU for their office. An individual from the OEDO will be assigned the overall responsibility for coordinating implementation of this MOU ~~for~~ by EDO offices. ~~Meetings with~~ ~~t~~The OEDO and office coordinators will ~~be scheduled~~ meet before each ACRS meeting to discuss ~~during which~~ the preparation of ACRS agendas for the subsequent and following two meetings. ~~will be discussed. The NRR, RES, and NMSS contacts will attend these meetings, as needed.~~

The ACRS staff engineer that support~~ing~~s the ACRS Subcommittee ~~that has~~ with the responsibility for ~~the~~ a matter under review by the ACRS will normally serve as the ACRS staff contact for day-to-day interactions on th~~ose~~at matter~~s of interest~~. The NRC staff coordinator for the responsible office will coordinate interactions with the ACRS staff ~~for that office~~. This does not preclude necessary interaction between the responsible ACRS staff engineer and the NRC

staff individual who has the day-to-day responsibility for the matter under ACRS review, as long as the office coordinator is ~~kept apprized~~ informed of any decisions. ~~made.~~

3. EARLY INTERACTION AND SELECTION OF MATTERS FOR THE ACRS REVIEW

The EDO will ~~take necessary steps to~~ ensure that matters requiring ACRS consideration are identified in the early stages of development and that sufficient time is allowed to permit effective and efficient review by the ACRS. Accordingly, when a safety matter, in an area of ACRS ~~purview~~ responsibility, is under consideration by the NRC staff, the cognizant NRC ~~staff~~ office~~, through the NRC staff coordinator,~~ will inform the ACRS of the anticipated staff action (e.g., proposed rulemaking, issuance of a regulatory guide, or issuance of a Commission paper) while ~~the basic requirements are~~ it is being formulated. This will be accomplished, following ~~through~~ discussions between the NRC staff coordinators and the cognizant ACRS staff, ~~and~~ by adding the anticipated staff action, with an appropriate description of the activity, to the list of proposed ACRS agenda items provided in the EDO's monthly memorandum on proposed agenda items for the ACRS and the ACNW. The ACRS will inform the cognizant NRC staff office and~~/or~~ the EDO's office on a timely basis as to whether it intends to review a specific matter. Decisions as to whether to review a specific matter will be made in accordance with Commission guidance, the needs of the EDO, and the recommendations of the responsible ACRS Subcommittee Chairman and the ACRS Planning and Procedures Subcommittee.

The NRC staff will give the ACRS staff insight about papers being developed by the NRC staff for which no ACRS review or information briefing is to be requested. The OEDO will send the ACRS a 90-day projection of such papers. The NRC staff will work with the ACRS to enable the ACRS to determine its review interest.

The ACRS will sometimes take up a matter for review on its own initiative. The ACRS will inform the EDO and the cognizant staff office when these activities are initiated and will coordinate these activities with the responsible NRC OEDO and EDO coordinators and staff

4. ESTABLISHING A SCHEDULE FOR THE ACRS REVIEW

If the ACRS decides to review a specific matter, the review will be performed prior to a Commission decision on the matter so that the Commission can have the benefit of the Committee's advice. When the EDO has the authority for making the regulatory decision, the ACRS review will be performed prior to the EDO making this decision. When a proposed regulatory action is to be published for public comment, the ACRS may review the matter both before and after public comment, as is appropriate for the particular case. There may be circumstances in which the ACRS will prefer to defer its review of a specific matter until after public comments have been received and addressed by the staff. In such cases, the EDO will be notified by the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW.

The cognizant NRC staff office will ensure that the schedule~~s~~ for ~~the~~ development of a specific

matter includes sufficient time (normally about 60 days) for ACRS review prior to the date by which ACRS comments are desired. The documents which the ACRS needs for a full Committee discussion will normally be provided to the ACRS at least four weeks prior to the scheduled full Committee discussion. When the ~~needed~~ documents cannot be provided ~~at least four weeks prior to the Committee discussion~~ in this time-frame, the discussion will only be scheduled after agreement by the EDO and the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW. Documents needed for discussion of a matter at a Subcommittee meeting will be provided no later than two weeks prior to the Subcommittee meeting. Absent some extraordinary need, the Subcommittee meeting will not be held if the documents cannot be provided two weeks prior to the meeting. Exceptions will be made only with the agreement of the EDO, the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW, and the Chairman of the cognizant Subcommittee~~Chairman~~. When the documents are of ~~such~~ a nature ~~as~~ to preclude adequate Committee review in four weeks, ~~(~~or Subcommittee review in two weeks~~)~~, the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW will consult with the EDO and establish other arrangements.

When~~, for whatever reason,~~ a choice must be made between timely submission of documents to the Commission or submission first for ACRS review, the EDO will consult with the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW and the Secretary of the Commission. It is expected that this will occur only in very unusual circumstances and that in these cases the Commission will make the decision as to the appropriate course of action.

~~RESOLVING ACRS COMMENTS~~

~~ACRS comments will be forwarded to the Commission or to the EDO, as appropriate, with copies to the cognizant NRC staff contact. The NRC staff contact will ensure that copies are provided to other NRC staff members, as appropriate.~~

~~The EDO will respond to ACRS comments in a timely manner. On all matters except those where Commission priorities or safety concerns demand action to the contrary, the EDO will respond to ACRS comments on a specific matter prior to taking final action on that matter, or prior to submitting it for Commission approval. Commission papers, if any, should address all ACRS comments including those not endorsed by the staff. The EDO may elect to consider ACRS comments on proposed or draft documents (e.g., proposed rules, draft regulatory guides) following the close of the public comment period within the context of resolution of public comments.~~ {Moved to Section 6}

5. SUBMITTING DOCUMENTS FOR ACRS REVIEW AND INFORMATION

a. Submittal of Documents

Twenty copies of documents related to a specific matter will be ~~provided~~ transmitted to the ACRS by the cognizant NRC office director through ~~staff contact/project engineer with~~ a memorandum addressed to the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW requesting appropriate ACRS action. When sending a specific matter to the ACRS for review, the cognizant staff office ~~(NRC staff contact)~~ will ensure that the ACRS is provided with copies ~~of other~~ related documents, such as public comments and the staff's resolution of these comments, ~~and~~ CRGR comments, ~~as appropriate. The cognizant staff will also include~~ and any

directly related differing professional opinions, and~~/or~~ differing professional views.

Five copies of documents related to a specific matter will also be provided to the ACRS for information by the NRC staff ~~contact/project engineer~~ contact at the following stages, when applicable, with a memorandum addressed to the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW, indicating that they are sent for ACRS information:

- When it is sent to the Federal Register to be published for public comment.
- When it is sent to the Federal Register to be published as an effective document.

The cognizant ACRS staff engineer and other ACRS/ACNW staff designated by the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW will be allowed "viewer" access rights in ADAMS for all documents within the purview of the ACRS when the documents are placed in the concurrence process.

~~[Note: ADAMS is designed to replace the transmittal of multiple paper copies of documents and at some point the requirements stated above can be modified, as appropriate. We will have to discuss what policy we are going to have for very large documents and documents which use color for graphs and charts, etc.]~~

b. ~~DEALING WITH PREDECISIONAL AND PROPRIETARY DOCUMENTS~~
HANDLING OF PREDECISIONAL AND PROPRIETARY DOCUMENTS

In those instances in which a safety-related matter is considered predecisional, and is not otherwise a matter which is exempt from the open meeting requirements of the Federal Advisory Committee Act, cognizant NRC staff will participate in open ACRS Subcommittee or full Committee meetings considered necessary to such reviews. In those cases where discussion of controlled internal documents~~, including predecisional documents,~~ is required during an open meeting, approval of the cognizant office director or regional administrator shall be obtained by the office transmitting the document to the ACRS. Discussion of a predecisional document at an open meeting should require approval by the Commission in the case of a document where the Commission itself is the final decisionmaker. For other predecisional documents originating from the staff, the EDO's approval should be obtained.

ACRS meetings can be closed for review of proprietary material under the exemptions allowed by the Federal Advisory Committee Act and external stakeholders can make such requests for closed meetings. The closing of ACRS meetings requires a written request to the Chairman of the Commission, or the Chairman's designee, and review by the Office of the General Counsel, in accordance with 10 CFR 7.15.

When requests of this type are received by the ACRS, the ACRS staff may need the assistance of NRC staff technical experts on a expedited basis to make accurate judgments as to what information should be protected. To provide for protection in accordance with the provisions of the Freedom of Information Act, documents transmitted to the ACRS by the NRC staff that are considered to be predecisional or proprietary will be identified as such by an appropriate marking and in the accompanying transmittal letter.

6. RESOLVING ACRS COMMENTS [Moved from earlier section in ACRS proposed draft]

ACRS comments will be forwarded to the Commission or to the EDO, as appropriate, with copies to the cognizant NRC office coordinator and staff contact. The NRC staff contact will ensure that copies are provided, as appropriate, to other NRC staff members~~, as appropriate~~.

The EDO will respond to ACRS comments in a timely manner. On all matters except those where Commission priorities or safety concerns demand action to the contrary, the EDO will respond to ACRS comments on a specific matter prior to taking final action on that matter, or prior to submitting it for Commission approval. Commission papers, if any, should address all ACRS comments including those not endorsed by the staff. The EDO may elect to consider ACRS comments on proposed or draft documents (e.g., proposed rules, draft regulatory guides) following the close of the public comment period within the context of resolution of public comments. The NRC staff will ensure that ACRS views on all rules and policy statements pertaining to nuclear safety matters are reflected in final SECY papers prior to the papers being forwarded to the Commission. The Commission should have the ACRS views on major topics at the same time it receives the staff views and recommendations.

7. DEVIATIONS FROM THIS MOU [Moved from earlier in ACRS proposed draft]

These procedures facilitate the NRC staff and ACRS interactions. Deviations from these procedures may at times be needed to carry out the NRC's mission. When this occurs, the procedures can be altered consistent with the needs of the NRC and the ACRS. Such changes will be implemented after being mutually agreed upon by the EDO and the ~~ACRS/ACNW~~ Executive Director for ACRS/ACNW.

_____ (Date)

_____
William Travers,
Executive Director for Operations

_____ (Date)

_____
~~Dana Powers, Chairman~~
~~Advisory Committee on Reactor Safeguards~~
John T. Larkins,
Executive Director for ACRS/ACNW

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15 May 2000

Dr Dana A Powers
Chairman
Advisory Committee on Reactor Safeguards
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr Powers:

Re: The Potential for Release of Radioactive Material from Spent Fuel Pools

I have noted your letter of 13 April 2000 to NRC Chairman Meserve, conveying the views of the ACRS on the NRC Staff's February 2000 Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.

In its study, the NRC Staff has set forth technical findings on the accident risk posed by spent fuel pools, and has recommended a regulatory position. Your letter shows that the Staff has done this without having first acquired an effective understanding of the relevant scientific issues.

Spent fuel pools contain large inventories of long-lived radioactive material. If a substantial fraction of the radioactive inventory of a pool were released to the atmosphere, the offsite consequences could considerably exceed the consequences from the 1986 Chernobyl accident. Thus, it is the NRC's duty to regulate spent fuel pools in a manner that draws upon the best attainable scientific understanding of the risk posed by these pools. That duty applies equally to pools at operating and decommissioning plants.

Although your letter was prepared in the context of decommissioning plants, the scientific issues that you address are equally applicable to pools at operating plants. Thus, your comments about the inadequacy of the Staff's February 2000 Draft Study could equally apply to previous Staff studies that have been prepared in the context of pools at operating plants. If those previous studies were found to exhibit the same inadequacies as you identify in the February 2000 Draft Study, it would follow that the NRC lacks an effective scientific basis for any of its regulations that affect the accident risk posed by spent fuel pools.

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In light of these considerations, the ACRS should use the powers and resources at its command to take actions of two types. First, the ACRS should independently investigate the state of scientific understanding of the accident risk posed by spent fuel pools at operating and decommissioning plants. Second, the ACRS should recommend that the NRC immediately initiates a comprehensive investigation of the state of knowledge on scientific issues that are relevant to the accident risk posed by spent fuel pools at operating and decommissioning plants. Both investigations should draw upon scientific capabilities outside the ACRS and the NRC.

As a source of relevant information, I enclose a February 1999 report that I prepared for Orange County, North Carolina, and which has been filed with the NRC Licensing Board in support of contentions that seek an EIS on the risks posed by expanded spent fuel storage at the Harris nuclear power plant. Although the report was prepared in the context of the Harris plant, it contains material that has generic applicability. The report has a limited scope, and does not purport to provide definitive analysis on the issues that it addresses. Nevertheless, the report identifies two issues that significantly affect the accident risk posed by spent fuel pools, but have been neglected by the NRC.

One issue is the effect on spent fuel pool accident scenarios of situations in which fuel is partially exposed to air. If water is lost from a pool by draining or evaporation, there must be a period in which fuel is partially exposed. In fact, scenarios for water loss could involve a falling, static or rising water level at various times, potentially leading to extended periods of partial exposure. Convective heat transfer by air will be inhibited during partial exposure of fuel assemblies packed at high density, and the steam-zirconium reaction will be more significant than the air-zirconium reaction. These factors have been neglected in NRC studies.

The second issue is the potential for a spent fuel pool accident to be initiated by a degraded-core reactor accident with containment failure or bypass. The reactor accident could involve a loss of cooling to an associated spent fuel pool, and the radioactive material released by the reactor accident would almost certainly preclude access by personnel for the purpose of restoring pool cooling. Water would then be lost from the pool by evaporation. Scenarios of this type have been neglected in NRC studies.

If you or other members of the ACRS wish to discuss any of the abovementioned matters with me, I would be pleased to do so.

Thank you for your attention.

Sincerely,

G. R. Thompson

Gordon Thompson
Executive Director

Enclosure: "Risks and alternative options associated with spent fuel storage at the Shearon Harris nuclear power plant", a report prepared by Gordon Thompson for Orange County, NC, February 1999.

cc (with enclosure): Richard A Meserve
Chairman
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Stuart A Richards
Director
Project Directorate IV and Decommissioning
Division of Licensing Project Management
US Nuclear Regulatory Commission
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RISKS AND ALTERNATIVE OPTIONS ASSOCIATED WITH SPENT FUEL STORAGE AT THE SHEARON HARRIS NUCLEAR POWER PLANT

A report

prepared for

Orange County
North Carolina

by

Gordon Thompson

February 1999

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Acknowledgements

This report was prepared as part of a program of work by the Institute for Resource and Security Studies (IRSS) pursuant to a contract between IRSS and Orange County, North Carolina. The report was written by Gordon Thompson, the executive director of IRSS.

The author acknowledges help with the acquisition of information and documents, from Diane Curran, David Lochbaum, Mary MacDowell and the staff of the NRC public document room in Washington, DC. Paul Thames, county engineer of Orange County, has provided efficient oversight of the contract between IRSS and Orange County. Paula Gutlove of IRSS has assisted in the preparation of this report. Gordon Thompson is solely responsible for the content of the report.

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About the author

Gordon Thompson is the executive director of IRSS. He received an undergraduate education in science and mechanical engineering, in Australia. Subsequently, he studied at Oxford University and received from that institution a doctorate of philosophy in mathematics in 1973.

During his professional career, Dr Thompson has performed technical and policy analyses on a range of issues related to international security, energy supply, environmental protection, and the sustainable use of natural resources. Since 1977, a significant part of his work has consisted of technical analyses of safety and environmental issues related to nuclear facilities. These analyses have been sponsored by a variety of nongovernmental organizations and local, state and national governments, predominantly in north America and western Europe. Dr Thompson has provided expert testimony in legal and regulatory proceedings, and has served on committees advising US government agencies.

About IRSS

The Institute for Resource and Security Studies is an independent, non-profit corporation. It was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting international security and sustainable use of natural resources. IRSS projects always reflect a concern for practical solutions to resource, environment and security problems, and can range from detailed technical studies to preparing educational materials accessible to the public. IRSS actively seeks collaborative relationships with other organizations as it pursues its goals.

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Abstract

Orange County, North Carolina, commissioned this report because the licensee of the Shearon Harris nuclear plant has requested an amendment of its operating license. The amendment would permit the activation of two currently unused spent fuel pools at Harris.

This report examines the risks and alternative options associated with spent fuel storage at Harris. The report identifies a potential for severe accidents at the Harris pools. Such accidents could release to the atmosphere an amount of cesium-137 an order of magnitude larger than the release from the 1986 Chernobyl accident. A severe accident at the Harris PWR, with containment failure or bypass, can be expected to initiate a large release from the fuel pools.

Alternative, safer options for spent fuel management are available. These options include dry storage of spent fuel, which is a well-established practice.

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1. Introduction

Carolina Power & Light Company (CP&L) requested, in December 1998, an amendment of its operating license for the Shearon Harris nuclear plant. The amendment, if granted by the Nuclear Regulatory Commission (NRC), would permit the activation of two currently unused spent fuel pools at Harris. In January 1999, Orange County commissioned this report, which examines the risks and alternative options associated with spent fuel storage at Harris.

Structure of this report

This report has two major components. One component is a main report which is comparatively brief and is intended for a non-specialist audience. The second component is a set of five appendices. These appendices contain detailed, technical material and citations to technical literature. Unless otherwise indicated, discussion in the main report rests upon the more detailed discussion in the appendices.

What is spent fuel?

Figure 1 shows a fuel assembly of the type that is used in the Harris reactor.[1] The fuel rods are 12 feet long, and the assembly is 8.4 inches square. After a fuel assembly is discharged from a reactor, it is "spent" in the sense that it can no longer be used to generate power. However, at this point in its life the assembly is much more dangerous than when it entered the reactor. It emits heat and intense radiation, and contains a large inventory of radioactive material.

Remainder of this report

The remainder of this main report begins with descriptions of the Harris plant (Section 2) and CP&L's intentions regarding the fuel pools at Harris (Section 3). Then, categories of potential accident at Harris are identified (Section 4), followed by descriptions of potential design-basis (Section 5) and severe (Section 6) accidents at the Harris pools. The offsite consequences of potential pool and reactor accidents are addressed in Section 7. Alternative options for spent fuel management are presented (Section 8), followed by a discussion of regulatory processes (Section 9). Conclusions are presented in Section 10.

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[1] Figure 1 is adapted from: A V Nero, A Guidebook to Nuclear Reactors, University of California Press, 1979, page 79.

2. Present status of the Harris nuclear plant

The Harris plant features one pressurized-water reactor (PWR). The core of this reactor contains 157 fuel assemblies, with a center-center distance of about 8.5 inches. The Harris plant was to have four units but only the first unit was built. (A unit consists of a reactor, a turbine-generator and associated equipment.) A fuel handling building was built to serve all four units. This building contains four fuel pools (A, B, C, D), a cask loading pool and three fuel transfer canals, all interconnected but separable by gates.

These pools and transfer canals allow spent fuel to be moved around and stored while remaining under water. The water provides cooling and also shields personnel and equipment from the radiation emitted by the fuel. Shipping casks can carry spent fuel to or from Harris. Casks are loaded and unloaded while submerged in the cask loading pool.

Pools A and B

Pools A and B contain fuel racks, and are in regular use. CP&L says that fresh fuel, and spent fuel recently discharged from the Harris reactor, is stored in pool A. Fuel examination and repair are performed in an open space in pool B. At present, pools C and D are flooded but do not contain racks. The cooling and water cleanup systems for pools C and D were never completed.

Currently, pools A and B store spent fuel from the Harris reactor and from CP&L's Brunswick plant and Robinson plant. The Brunswick plant has two boiling-water reactors (BWRs) while the Robinson plant has one PWR. Shipment of spent fuel from Brunswick and Robinson to Harris is said by CP&L to be necessary to allow sufficient capacity in the pools at Brunswick and Robinson so that the entire core can be removed from the reactor.

Pools A and B now have a combined, potential capacity of 3,669 fuel assemblies. The center-center distance in the racks in pools A and B is 10.5 inches for PWR fuel and 6.25 inches for BWR fuel. This is a much more compact pool storage configuration than was used when nuclear plants first entered service. The United States has no national storage site or repository for spent fuel, so CP&L is currently obliged to store fuel at its plant sites. Compact storage in the existing pools is a comparatively cheap option for on-site storage.

3. Proposed activation of fuel pools C and D

CP&L seeks an amendment to its operating license so that it can activate pools C and D at Harris. By activating these pools, CP&L expects to have sufficient storage capacity at its three nuclear plants to accommodate all the spent fuel discharged by the four CP&L reactors (the Harris and Robinson PWRs and the two Brunswick BWRs) through the ends of their current operating licenses.

Capacity and configuration of pools C and D

CP&L plans to install racks in pool C in three campaigns (approximately in 2000, 2005 and 2014), to create a total capacity in this pool of 3,690 fuel assemblies. Thereafter, CP&L plans to install racks in pool D in two campaigns (approximately in 2016 and at a date to be determined), to create 1,025 spaces. Thus, the ultimate capacity of pools C and D will be 4,715 fuel assemblies. The center-center distance in the racks used in these pools will be 9.0 inches for PWR fuel and 6.25 inches for BWR fuel. In pool C, the space between the outermost racks and the pool wall will be 1-2 inches.

The PWR racks in pools C and D will have a smaller center-center distance than the racks in pools A and B (9.0 inches instead of 10.5 inches). This highly compact arrangement allows more PWR fuel to be placed in a given pool area but also has adverse implications for safety.

Cooling and electrical supply for pools C and D

The water in a spent fuel pool must be cooled and cleaned. Cooling is performed by circulating pool water through heat exchangers, where its heat is transferred to a secondary cooling system. At Harris, the secondary cooling system is the component cooling water (CCW) system. When the Harris plant was designed, the intention was that pools C and D would be cooled by the CCW system for Unit 2. Also, electricity would have been supplied to the circulating pumps at pools C and D from the electrical systems of Unit 2. However, Unit 2 was never built and its CCW and electrical systems do not exist.

CP&L's current plan is to cool pools C and D by completing their partially built cooling systems and connecting those systems to the Unit 1 CCW system. Electricity will be supplied to pools C and D from the electrical systems of Unit 1. The Unit 1 CCW system already provides cooling to pools A and B and serves other, important safety functions. For example, the Unit 1 CCW system provides cooling for the residual heat removal (RHR) system and reactor coolant pumps of the Unit 1 reactor.

Independent support systems for pools C and D

During CP&L's planning for the activation of pools C and D, the company considered the construction of an independent system to cool these pools. Within that option, CP&L considered the further possibility of providing dedicated emergency diesel generators to meet the electrical needs of pools C and D if normal electricity supply were unavailable. Construction of an independent cooling system for pools C and D, supported by dedicated emergency diesel generators, could provide the level of safety that was associated with the original design concept for Harris. However, CP&L has not proceeded with this option.

Capacity of the Unit 1 CCW system

In its present form, the Unit 1 CCW system cannot absorb the additional heat load that will ultimately arise from activation of pools C and D. Over the first few years of pool use, while the heat load is comparatively small, CP&L proposes to exploit the margin in the Unit 1 CCW system. Subsequently, CP&L intends to upgrade the Unit 1 CCW system so that it can accommodate the full heat load from pools C and D, and can also accommodate an anticipated power uprate for the Unit 1 reactor.

Safety implications

In order to exploit the margin in the existing CCW system so as to cool pools C and D, CP&L may be obliged to require its operators to divert some CCW flow from the RHR heat exchangers during the recirculation phase of a design-basis loss-of-coolant accident (LOCA) event at the Harris reactor. This is a safety issue because, during the recirculation phase of a LOCA, operation of the RHR system is essential to keeping the reactor core and containment in a safe condition. CP&L's exploitation of the margin in the existing CCW system is deemed by CP&L and NRC to constitute an "unreviewed safety question".

Lack of QA documentation

Activation of pools C and D will require the completion of their cooling and water cleanup systems, and the connection of their cooling systems to the Unit 1 CCW system. CP&L states that approximately 80 percent of the necessary piping was completed before the second Harris reactor was cancelled. However, some of the quality assurance (QA) documentation for the completed piping is no longer available. Much of the completed piping is embedded in concrete and is therefore difficult or impossible to inspect. To

address this situation, CP&L proposes an "alternative plan" to demonstrate that the previously completed piping and other equipment is adequate for its purpose. Nevertheless, the cooling systems for pools C and D will not satisfy prevailing code requirements.

4. Types of potential accident at the Harris plant

Most of the radioactive material at the Harris plant is either in the reactor or in the spent fuel pools. Thus, these locations are of primary concern when one considers the potential for accidents. This report focusses on the potential for accidents in the reactor or the pools. At present, pools C and D at Harris pose no accident potential, because they are unused.

Some potential accidents could cause injury to plant personnel, without causing any offsite effects. Other potential accidents could release radioactive material beyond the plant boundary, causing offsite effects. The radioactive material could be released as an atmospheric plume, or into ground or surface waters. This report focusses on accidents that release an atmospheric plume which travels beyond the plant boundary. Such a plume will contain radioactive material in the form of gases and small particles. As the plume travels downwind, the small particles will be deposited onto land, bodies of water, structures and vegetation.

Design-basis and severe accidents

A nuclear plant is designed to accommodate the effects of a specified set of accidents, known as "design-basis" accidents. If the plant is properly designed and constructed, if its equipment and operators function in the required manner, and if external influences (e.g., earthquakes) do not exceed specified levels, then the offsite effects of a design-basis accident will be small. Design-basis accidents and their anticipated effects are described in a Final Safety Analysis Report (FSAR) prepared and regularly updated by the licensee.

In the early years of the nuclear industry, some people equated design-basis accidents with "credible" accidents. However, research and operating experience soon revealed that accidents more severe than the design basis are credible. The first systematic study of the potential for severe accidents was the Reactor Safety Study, completed and published by the NRC in 1975. "Severe" accidents are conventionally defined as accidents involving substantial damage to fuel, with or without a substantial release of radioactivity to the environment.

The Three Mile Island (TMI) reactor accident of 1979 was a demonstration of the potential for severe accidents. Soon thereafter, the NRC promulgated

regulations which require an emergency response plan for each nuclear plant. These plans allow for large releases of radioactive material, of the kind that were identified in the Reactor Safety Study. The Chernobyl reactor accident of 1986 further demonstrated the potential for severe accidents. While the TMI accident released a small fraction of the reactor core's inventory of radioactivity, the release fraction during the Chernobyl accident was large.

Since the TMI accident, the NRC's safety regulation of nuclear plants has been guided by a hybrid set of assumptions. Many areas of safety regulation rely upon the assumption that accidents will remain within the design basis. Other areas, such as emergency response planning, assume that severe accidents can occur.

Pool-reactor interactions

At the Harris plant, the reactor and the fuel pools are adjacent, and they share support systems such as the Unit 1 CCW system and the emergency diesel generators. Thus, it is important to understand if an accident at the Harris reactor could accompany, initiate or exacerbate an accident at the Harris pools, or vice versa. The NRC has been slow to examine the potential for safety interactions between reactors and fuel pools. Neither CP&L nor the NRC has assessed the potential for these interactions at Harris.

PRAs and IPEs

A discipline known as probabilistic risk assessment (PRA) has been developed to examine the probabilities and consequences of potential accidents at nuclear facilities. PRA techniques are most highly developed in their application to reactor accidents, but can be applied to fuel pool accidents. Appendix B describes the characteristics, strengths and limitations of PRA.

CP&L has prepared a Level 2, internal-events PRA for the Harris reactor, in the form of an Individual Plant Examination (IPE). Also, CP&L has performed a limited assessment of the vulnerability of the Harris reactor to earthquakes and in-plant fires, in the form of an Individual Plant Examination for External Events (IPEEE).

The Harris IPE and IPEEE could be extended to encompass fuel pool accidents as well as reactor accidents. Such an extension would be logical, because there are various ways in which a severe accident or a design-basis accident at the Harris reactor might accompany, initiate or exacerbate an accident at the Harris fuel pools, or vice versa. However, there is no current indication that CP&L will extend the IPE or IPEEE, or will otherwise apply PRA techniques to potential accidents at the Harris fuel pools.

5. Design-basis pool accidents

The Harris FSAR considers two types of design-basis accident in the Harris fuel pools. One type of accident involves the dropping of a fuel assembly, while the other type involves the dropping of a shipping cask (but not into a fuel pool). In both cases, the FSAR estimates that the release of radioactivity would be relatively small. This report does not review the FSAR analysis.

In its license amendment application, CP&L has considered some other potential accidents, including the dropping of a rack or a fuel pool gate.[2] CP&L's analysis of these accident scenarios is limited in scope. Accidents of this type may be in an intermediate class of severity, and that potential class deserves further analysis.[3] This report focusses on the potential for severe accidents.

It should be noted that the use of pools C and D at Harris will involve many additional cask, fuel and rack movements. These additional movements will increase the cumulative probability of accidents associated with such movements.

6. Severe pool accidents

Spent fuel is stored in a compact, high-density configuration in pools A and B at Harris. CP&L's proposed activation of pools C and D will involve an even higher density of storage. Such high-density configurations inhibit heat loss from the fuel if water is partially or totally lost from a pool. As a result, partial or total loss of water can lead to an exothermic (heat-producing) reaction of the fuel cladding with air or steam. Such a reaction could liberate a large amount of radioactive material from the fuel.

Thus, two questions become important. First, what circumstances could cause a partial or total loss of water? This question is addressed in Appendix C. Second, will an exothermic reaction be initiated if water is lost? That question is addressed in Appendix D.

Potential for loss of water

A variety of events could cause partial or total loss of water from the Harris pools. These events deserve the level of analysis that would be provided by a thorough PRA. Performing a pool accident PRA is beyond the scope of our

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[2] License amendment application, Enclosure 7.

[3] A potential accident in this class, which deserves analysis, would involve the placement of a low-burnup or high-enrichment PWR assembly in the racks in pools C or D.

present work for Orange County. Here, the focus is on two types of event -- a reactor accident, and a sabotage/terrorism event. Consideration of these events demonstrates clearly that loss of water from the Harris pools is a credible accident.

The Harris IPE -- prepared by CP&L -- examines the potential for severe accidents at the Harris reactor. It identifies a category of severe accidents that would involve failure or bypass of the reactor containment. The IPE estimates the collective probability of accidents in this category to be 1 per 100,000 reactor-years.[4] Occurrence of accidents in this category would contaminate the plant with radioactivity, to the point where personnel access would almost certainly be precluded. Water would then be evaporated from the fuel pools, and fuel would be uncovered after a delay of perhaps 10 days.

A credible sabotage/terrorism event at Harris would involve a group taking control of the fuel handling building, shutting down the pool cooling systems, and siphoning water from the pools. The group would require military skills and equipment to take control of the fuel handling building. Siphoning water from the pools would be a comparatively easy task. Escape by the group would be difficult but not impossible. The probability of this event cannot be predicted by PRA techniques.

Initiation of exothermic reactions, given water loss

Since the late 1970s, the NRC has sponsored and performed a variety of studies that have examined the outcomes of a loss of water from a fuel pool. These studies have focussed almost entirely on the instantaneous, total loss of water from a pool. Computer models have been developed to investigate this situation. For a high-density pool configuration, current models suggest that an exothermic reaction will be initiated in fuel aged up to 1-2 years after discharge from a reactor. These models have not been applied to the specific configuration of the Harris pools.

Partial loss of water can be expected in many scenarios, rather than instantaneous, total loss of water. Partial loss of water can be a more severe situation, because convective heat transfer from fuel assemblies is inhibited. The NRC has neglected this issue. Preliminary analysis suggests that partial water loss could initiate an exothermic reaction in fuel aged 10 years after discharge.

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[4] This probability estimate should be accompanied by a range of uncertainty. Even with the inclusion of uncertainties, PRA-derived estimates represent lower bounds to actual accident probabilities.

An exothermic reaction could propagate from one set of fuel assemblies to an adjacent set of assemblies that might not otherwise suffer such a reaction. The NRC's studies of propagation are incomplete, but they acknowledge the potential for propagation.

Exothermic reactions in the Harris pools

CP&L representatives have stated that spent fuel assemblies will not be placed in pools C and D at Harris until the assemblies have aged for 5 years after discharge. However, there is nothing in CP&L's license amendment application that prohibits the placement of more recently-discharged fuel in pools C and D. In any case, preliminary analysis suggests that partial water loss could initiate an exothermic reaction in fuel aged 10 years after discharge. Thus, exothermic reactions could occur in pools C and D.

For the purpose of estimating the potential consequences of a pool accident at Harris, this report considers two scenarios for exothermic reactions. One scenario involves fuel aged up to 3 years after discharge from a reactor, while the second scenario involves fuel aged up to 9 years after discharge from a reactor. In both cases, it is assumed that the entire inventory of cesium in the affected fuel assemblies would be released to the atmosphere. This assumption is consistent with NRC studies.

7. Consequences of potential pool and reactor accidents

This report focusses on accidents that release an atmospheric plume which travels beyond the plant boundary. The consequences of such a release can be estimated by site-specific computer models. Here, a simpler approach is used, but this approach is adequate to show the nature and scale of expected consequences. The approach is described in Appendix E.

The role of cesium-137

The consequences of a pool accident can be adequately illustrated by examining a release of only one radioisotope -- cesium-137. This isotope has a half-life of 30 years and is liberally released from damaged fuel. It dominates the offsite radiation exposure from the 1986 Chernobyl accident, and is a major contributor to radiation exposure attributable to fallout from the atmospheric testing of nuclear weapons in the 1950s and 1960s.

Three atmospheric releases of cesium-137 are postulated here for the purpose of examining consequences. First, a release of about 2 million Curies (2 MCi) corresponds to the most severe reactor accident identified in the Harris IPE. Second, a release of about 20 million Curies (20 MCi) corresponds to a pool

accident affecting fuel aged up to 3 years after discharge from a reactor. Third, a release of about 70 million Curies (70 MCi) corresponds to a pool accident affecting fuel aged up to 9 years after discharge from a reactor.

Land contamination by cesium-137

Accident consequences are illustrated here by estimating the area of land that would be contaminated by cesium-137 to a level such that inhabitants would suffer an external radiation dose in excess of 10 rem over 30 years.[5] An exposure of 10 rem over 30 years would represent about a three-fold increase above the typical level of background radiation (which is about 0.1 rem/year). In its Reactor Safety Study, the NRC used a threshold of 10 rem over 30 years as an exposure level above which populations were assumed to be relocated from rural areas. The same study used a threshold of 25 rem over 30 years as a criterion for relocating people from urban areas, to reflect the assumed greater expense of relocating urban inhabitants.

In an actual case of land contamination in the United States, the steps taken to relocate populations and pursue other countermeasures (decontamination of surfaces, interdiction of food supplies, etc.) would reflect a variety of political, economic, cultural, legal and scientific influences. It is safe to say that few citizens would calmly accept a level of radiation exposure which substantially exceeds background levels.

For typical meteorology, a release of 2 MCi would contaminate 4,000-5,000 square kilometers of land, A release of 20 MCi would contaminate 50,000-60,000 square kilometers. Finally, a release of 70 MCI would contaminate about 150,000 square kilometers of land. Note that the total area of North Carolina is 136,000 square kilometers and the state's land area is 127,000 square kilometers.

Health effects of radiation

There is ongoing debate about the health effects of radiation at comparatively low doses. According to estimates by the National Research Council's BEIR V committee, a continuous exposure throughout life at a rate of 0.1 rem/year (above background) will increase the number of fatal cancers, above the normally expected level, by 2.5 percent for males and 3.4 percent for females, with an average of 16-18 years of life lost per excess death. If the dose-response function were linear, it would follow that continuous, lifetime exposure to 1 rem/year would increase the number of fatal cancers by 25

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[5] Without countermeasures such as interdiction of food supplies, the internal dose could be of a similar magnitude to the external dose.

percent for males and 34 percent for females. The shape of the dose-response function is a subject of debate.

8. Alternative options for spent fuel management

The present mode of spent fuel storage in Harris pools A and B poses a major hazard. This hazard will be substantially increased if pools C and D are activated. CP&L has not properly characterized the present and potential hazard, nor has the company provided a systematic assessment of alternative options.

A situation like this calls for a systematic, comprehensive assessment of alternative options and their impacts. A full range of alternatives should be identified, and their impacts and other characteristics should be assessed. Performance of such an analysis is beyond the scope of the author's current work for Orange County. An abbreviated discussion is presented here.

Options not reviewed here

One option would be to cease operation of CP&L's nuclear plants. That option, which could be combined with other options for storage of CP&L's present stock of spent fuel, is not reviewed here. Another set of options would employ high-density pool storage but would introduce technical measures that sought to increase the reliability of the cooling systems for some or all of the Harris pools, or to decrease the potential for safety interactions between the pools and the reactor. Independent support systems for pools C and D, as mentioned in Section 3, would be in this class of options. Such options are not reviewed here.

Options reviewed here

This report focusses on two classes of options for spent fuel storage. One class involves dry storage of spent fuel, using proven technology. The second class, which could complement dry storage, involves low-density storage in pools. A combination of dry storage and low-density pool storage could offer a practical, proven means of dramatically decreasing the hazard posed by high-density pool storage at Harris.

Dry storage

The NRC has approved a variety of designs for the dry storage of spent fuel. These designs are described in Table 1, and their current use by licensees is

described in Table 2.[6] It will be noted from Table 2 that a dry storage installation is licensed at CP&L's Robinson plant. This installation employs eight NUHOMS-7P modules, each of which can hold 7 fuel assemblies. All eight modules are fully loaded.[7]

Dry storage could be implemented at any of CP&L's three plant sites. This report does not recommend any particular design, but notes that the designs vary in their level of safety and other features. For example, some designs are more resistant to sabotage than others.

All of the approved dry storage designs are safe in the event that access to the plant site is precluded by the release of radioactive material during a reactor accident. None of the designs requires active cooling, electricity or operator attention. A sabotage/terrorism event at a dry storage installation could release only a small fraction of the radioactive material that could be released by a sabotage/terrorism event at the Harris pools in their present and proposed configuration. Overall, dry storage poses a much lower level of hazard than high-density pool storage, for the same quantity of fuel.

At present, the NRC licenses dry storage installations for only 20 years. However, the technology is capable of storing fuel for much longer periods. If CP&L employs the dry storage option, they should choose a design that has this capability. This choice, properly documented and supported by ongoing testing, would establish the basis for a license extension in the future.

Low-density pool storage

Spent fuel can be stored in pools in a low-density, open-rack configuration, as was common practice when nuclear plants were first operated. Given a sufficiently low-density configuration, partial or total uncovering of the fuel will not initiate an exothermic reaction in the fuel cladding, even for recently discharged fuel. The fuel would remain vulnerable to consolidation through a cask drop into a pool or a severe earthquake which disrupts the fuel racks. If such consolidation were accompanied by partial or total uncovering, an exothermic reaction could occur in the consolidated region. However, it is unlikely that this reaction would be propagated to other regions of a pool.

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[6] Tables 1 and 2 are adapted from: US Nuclear Regulatory Commission, Information Digest, 1998 Edition, NUREG-1350, Volume 10, November 1998.
[7] M G Raddatz and M D Waters, Information Handbook on Independent Spent Fuel Storage Installations, NUREG-1571, December 1996.

Summary

CP&L could employ a spent fuel storage strategy which combines dry storage with low-density pool storage. Some or all of pools A, B, C and D at Harris would be used in a low-density configuration. If appropriately designed and implemented, this strategy could dramatically reduce the hazard posed by present and proposed fuel storage arrangements at Harris.

9. Addressing risks and alternatives in the regulatory arena

Orange County has requested the NRC to hold a hearing regarding CP&L's license amendment application, and the NRC has established a Licensing Board for this case. These actions have initiated a regulatory process which has been employed many times before. A review of this process is beyond the scope of this report, but some brief observations may be helpful.

The licensing process will typically assume that regulatory decisions taken in the past were correct. Thus, the existing operations at Harris pools A and B might be held to establish a precedent for the proposed operations at pools C and D. However, this report shows that the NRC has not properly analyzed the potential for severe pool accidents at a generic level. This point may or may not influence the NRC's regulatory process, but it deserves continuing emphasis through all available channels.

At Harris, and nationwide, there is a need for a thorough assessment of the hazards associated with high-density pool storage, and of alternative options which could pose a lower hazard. Orange County would provide an important public service if it could persuade the NRC or another body to conduct such an assessment, perhaps in the form of an environmental impact statement. There has been discussion about the US Department of Energy taking title to the nation's spent fuel, while the fuel remains at plant sites. This move could provide an opportunity for a thorough assessment of risks and options, and for the adoption of safer means of fuel storage.

10. Conclusions

C1 Given the present and proposed configuration of spent fuel storage in the Harris pools, partial or total loss of water from the pools could initiate exothermic reactions of fuel cladding, in any or all of pools A, B, C and D.

C2 Partial or total loss of water from the Harris pools could occur through a variety of events including acts of malice, and would be an almost certain outcome of a severe reactor accident at Harris involving containment failure

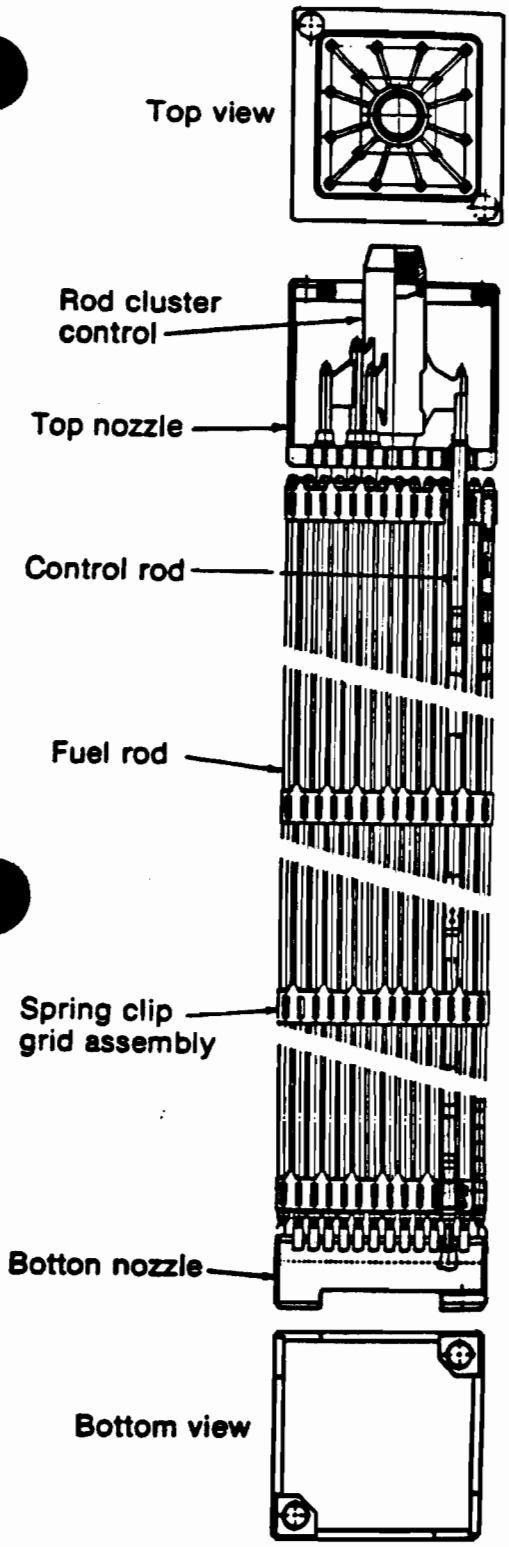
or bypass; CP&L estimates the probability of the latter event as 1 per 100,000 reactor-years.

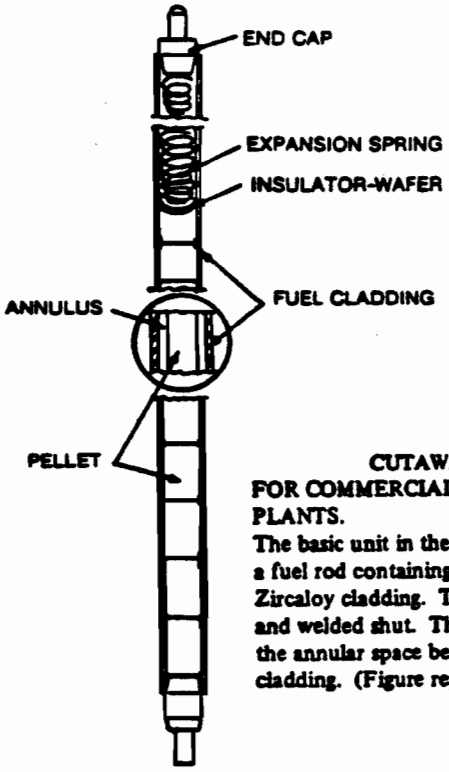
C3 Exothermic reactions in the Harris pools could release to the environment an amount of cesium-137 at least an order of magnitude larger than the amount released by the most severe potential accident at the Harris reactor.

C4 A large release of cesium-137, as could occur from exothermic reactions in the Harris pools, could significantly contaminate an area of land equal to the area of North Carolina.

C5 The probability and magnitude of a potential release from Harris of radioactive material in spent fuel could be dramatically reduced if CP&L adopted a fuel storage strategy which combines dry storage with low-density pool storage; this strategy would employ proven technology.

C6 Activation of pools C and D at Harris could increase the probability and magnitude of design-basis or severe accidents at the Harris fuel pools or reactor.





CUTAWAY VIEW OF OXIDE FUEL FOR COMMERCIAL LWR POWER PLANTS.
The basic unit in the core of a light-water reactor is a fuel rod containing uranium oxide pellets in a Zircaloy cladding. The rod is filled will helium gas and welded shut. The circled portion exaggerates the annular space between the pellet and the cladding. (Figure reproduced from WASH-1250.)

FUEL ASSEMBLY FOR A PRESSURIZED-WATER REACTOR.
In a pressurized-water reactor, fuel rods are assembled into a square array, held together by spring clip assemblies and by nozzles at the top and bottom. The structure is open, permitting flow of coolant both vertically and horizontally. All the assemblies in the reactor may have the same mechanical design, including provision for passage of a control rod cluster (shown in the figure). Where there is no cluster, these positions may have neutron sources, burnable poison rods, or plugs. (Figure reproduced from WASH-1250.)

Figure 1

Fuel for a pressurized-water reactor

Vendor	Storage Design Model	Capacity (Assemblies)	Storage Design Approval Date	Certificate of Compliance Approval Date
General Nuclear Systems, Incorporated	Metal Cask CASTOR V/21 CASTOR X/28 CASTOR X/33	 21 PWR 28 PWR 33 PWR	 09/30/1985 04/22/1994 11/24/1995	08/17/1990
Transnuclear, West Incorporated	Concrete Module NUHOMS-7P	7 PWR	 03/28/1986	
Westinghouse Electric	Metal Cask MC-10	24 PWR	09/30/1987	08/17/1990
PW Energy Applications, Incorporated	Concrete Vault Modular Vault Dry Storage (MVDS)	83 PWR or 150 BWR	03/22/1988	
NAC International, Inc.	Metal Cask NAC S/T	26 PWR	03/29/1988	08/17/1990
NAC International, Inc.	Metal Cask NAC-C28 S/T	28 Canisters (fuel rods from 56 PWR assemblies)	09/29/1988	08/17/1990
Transnuclear, Incorporated	Metal Cask TN-24 TN-32	 24 PWR 32 PWR	 07/05/1989 11/07/1996	11/04/1993
NAC International, Inc.	Metal Cask NAC-128/ST	28 PWR	02/01/1990	
Sierra Nuclear Corporation	Ventilated Cask VSC-24	24 PWR	03/29/1991	05/03/1993
Transnuclear West, Inc.	Concrete Module Standardized NUHOMS-24P NUHOMS-52B	 24 PWR 52 BWR	04/21/1989	01/18/1995
NAC International, Inc.	NAC-STC	26 PWR	07/17/1995	

Note: PWR - Pressurized-Water Reactor; BWR - Boiling-Water Reactor

Table 1

NRC-approved dry spent fuel storage designs

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Reactor Name Utility	Date Issued	Vendor	Storage Model
Surry 1, 2 Virginia Electric & Power Company	07/02/1986	Generals Nuclear Systems, Incorporated	Metal Cask CASTOR V/21 TN-32 NAC-128 CASTOR X/33 MC-10
H. B. Robinson 2 Carolina Power & Light Company	08/13/1986	Transnuclear West, Incorporated	Concrete Module NUHOMS-7P
Oconee 1, 2, 3 Duke Energy Company	01/29/1990	Transnuclear West, Incorporated	Concrete Module NUHOMS-24P
Fort St. Vrain* Public Service Company of Colorado	11/04/1991	FW Energy Applications, Incorporated	Modular Vault Dry Store
Calvert Cliffs 1, 2 Baltimore Gas & Electric Company	11/25/1992	Transnuclear West, Incorporated	Concrete Module NUHOMS-24P
Palisades Consumers Energy	Under General License	Pacific Sierra Nuclear Associates	Ventilated Cask VSC-24
Prairie Island 1, 2 Northern States Power Company	10/19/1993	Transnuclear West, Incorporated	Metal Cask TN-40
Point Beach Wisconsin Electric Power Company	Under General License	Sierra Nuclear Corporation	Ventilated Cask VSC-24
Davis-Besse Toledo Edison Company	Under General License	Transnuclear West, Incorporated	Concrete Module NUHOMS-24P
Arkansas Nuclear One Entergy Operations	Under General License	Sierra Nuclear Corporation	Ventilated Cask VSC-24
North Anna Virginia Electric & Power Company	06/30/98	Transnuclear West, Incorporated	Metal Cask TN-32

*Plant undergoing decommissioning

Table 2

NRC dry spent fuel storage licensees

RISKS AND ALTERNATIVE OPTIONS
ASSOCIATED WITH SPENT FUEL STORAGE AT THE
SHEARON HARRIS NUCLEAR POWER PLANT

Appendix A

Spent fuel management at the Harris plant

1. Introduction

This appendix summarizes present and proposed arrangements for managing spent fuel at the Shearon Harris plant. Carolina Power & Light Company (CP&L), the licensee for the plant, proposes to introduce new arrangements for spent fuel management. For that purpose, CP&L seeks an amendment to the plant's operating license. Unless specified otherwise, information presented here is drawn from CP&L's application to amend the Harris license, from CP&L's Final Safety Analysis Report (FSAR) for the Harris plant, or from viewgraphs shown by CP&L personnel during meetings with staff of the Nuclear Regulatory Commission (NRC).[1]

2. Present and proposed spent fuel storage capacity

The Harris plant features one pressurized-water reactor (PWR). The core of this reactor contains 157 fuel assemblies, with a center-center distance of about 8.5 inches. The Harris plant was to have four units but only the first unit was built. (A unit consists of a reactor, a turbine-generator and associated equipment.) A fuel handling building was built to serve all four units. This building contains four fuel pools (A, B, C, D), a cask loading pool and three fuel transfer canals, all interconnected but separable by gates. Figure A-1 shows a plan view of the interior of the fuel handling building.

Pools A and B

Pools A and B contain fuel racks, and are in regular use. CP&L says that fresh fuel, and spent fuel recently discharged from the Harris reactor, is stored in pool A. Fuel examination and repair are performed in an open space in pool

[1] Meetings between NRC staff and CP&L representatives, to discuss the proposed license amendment, were held on 3 March 1998 and 16 July 1998.

B. Pools C and D are flooded but do not contain racks. The cooling and water cleanup systems for pools C and D were never completed.

Pool A now contains six racks (360 fuel assembly spaces) for PWR fuel and three racks (363 spaces) for boiling-water reactor (BWR) fuel, for a total pool capacity of 723 fuel assemblies. Pool B contains twelve PWR racks (768 spaces) and seventeen BWR racks (2,057 spaces), and is licensed to store one additional BWR rack (121 spaces), for a total, potential pool capacity of 2,946 fuel assemblies. Thus, pools A and B now have a combined, potential capacity of 3,669 fuel assemblies. The center-center distance in the racks in pools A and B is 10.5 inches for PWR fuel and 6.25 inches for BWR fuel.

Pools A and B store spent fuel from the Harris reactor and from CP&L's Brunswick plant and Robinson plant. The Brunswick plant has two BWRs while the Robinson plant has one PWR. Shipment of spent fuel from Brunswick and Robinson to Harris is said by CP&L to be necessary to allow core offload capacity in the pools at Brunswick and Robinson.

Pools C and D

CP&L seeks an amendment to its operating license so that it can activate pools C and D at Harris. By activating these pools, CP&L expects to have sufficient storage capacity at its three nuclear plants to accommodate all the spent fuel discharged by the four CP&L reactors (the Harris and Robinson PWRs and the two Brunswick BWRs) through the ends of their current operating licenses.

CP&L plans to install racks in pool C in three campaigns (approximately in 2000, 2005 and 2014), to create 927 PWR spaces and 2,763 BWR spaces, for a total capacity in this pool of 3,690 fuel assemblies. Thereafter, CP&L plans to install racks in pool D in two campaigns (approximately in 2016 and at a date to be determined), to create 1,025 PWR spaces. Thus, the ultimate capacity of pools C and D will be 4,715 fuel assemblies. The center-center distance in the racks used in these pools will be 9.0 inches for PWR fuel and 6.25 inches for BWR fuel.

The PWR racks in pools C and D have a smaller center-center distance than the racks in pools A and B (9.0 inches instead of 10.5 inches). This arrangement allows more PWR fuel to be placed in a given pool area but also means that PWR fuel in pools C and D is more prone to undergo criticality. In response, CP&L proposes to include in the Technical Specifications for Harris a provision that PWR fuel will not be placed in pools C and D unless it has relatively low enrichment and high burnup.[2]

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[2] License amendment application, Enclosure 5.

Summary

Table A-1 summarizes the present and proposed storage capacity in the Harris pools. At present, pools A and B have a combined, potential capacity of 3,669 assemblies. The proposed, combined capacity of pools C and D will be 4,715 assemblies. Thus, activation of pools C and D will represent an increase of about 130 percent in the number of fuel assemblies that could be stored at Harris.

3. Support services for pools C and D

The water in a spent fuel pool must be cooled and cleaned. Figure A-2 provides a schematic view of typical cooling and cleanup systems. It will be noted that pool water is circulated through heat exchangers, where its heat is transferred to a secondary cooling system. At Harris, the secondary cooling system is the component cooling water (CCW) system. Water in the secondary system is in turn circulated through heat exchangers, where its heat is transferred to a tertiary cooling system. At Harris, the tertiary cooling system is the service water (SW) system.

When the Harris plant was designed, the intention was that pools C and D would be cooled by the CCW system for the second unit. That unit was never built and its CCW system does not exist. Thus, CP&L plans to cool pools C and D by completing their partially built cooling systems and connecting those systems to the CCW system of the first unit. The Unit 1 CCW system already provides cooling to pools A and B and serves other, important safety functions. For example, the Unit 1 CCW system provides cooling for the residual heat removal (RHR) system and reactor coolant pumps of the Unit 1 reactor.

The original design concept for Harris

In the Harris plant's original design concept, pools A and B would have served Units 1 and 4, while pools C and D would have served Units 2 and 3. There would have been a separate, fully-redundant, 100 percent-capacity cooling and water cleanup system for each pair of pools (A+B and C+D). Cooling of pools C and D would have been provided by the CCW system of Unit 2. Electrical power for the pumps that circulate water from the C and D pools through heat exchangers (see Figure A-2) would have been supplied by the Unit 2 electrical systems. Pools A and B would have been supported by the CCW and electrical systems of Unit 1.

During CP&L's planning for the activation of pools C and D, the company considered the construction of an independent system to cool these pools. Within that option, CP&L considered the further possibility of providing dedicated emergency diesel generators to meet the electrical needs of pools C and D if normal electricity supply were unavailable. Construction of an independent cooling system for pools C and D, supported by dedicated emergency diesel generators, could provide the level of safety that was associated with the original design concept for Harris. However, CP&L has not proceeded with this option.

Capacity of the Unit 1 CCW system

According to CP&L's license amendment application, the bounding heat load from the fuel in pools C and D will be 15.6 million BTU/hour (4.6 MW).[3] At present, the Unit 1 CCW system cannot absorb this additional heat load. Thus, CP&L proposes to include in the Technical Specifications for Harris an interim provision that the heat load in pools C and D will not be allowed to exceed 1.0 million BTU/hour.[4] CP&L claims that an additional heat load of 1.0 million BTU/hour can be accommodated by the Unit 1 CCW system, and that the fuel to be placed in pools C and D will not create a heat load exceeding 1.0 million BTU/hour through 2001.

CP&L contemplates a future upgrade of the Unit 1 CCW system, so that this system can accommodate an additional heat load of 15.6 million BTU/hour from pools C and D. This contemplated upgrade is not described in the present license amendment application. Apparently, CP&L intends to perform the upgrade of the Unit 1 CCW system concurrent with a power uprate for the Unit 1 reactor. A 4.5 percent power uprate of the reactor will be associated with steam generator replacement, and will take effect in about 2002. About two years later, there will be a further power uprate of 1.5 percent. CP&L projects that the Unit 1 CCW heat load, including the reactor power uprate and the ongoing use of pools C and D, will substantially exceed the capability of the present CCW system.

To summarize, CP&L's short-term plan (through 2001) for cooling pools C and D is to exploit the margin in the Unit 1 CCW system, so as to accommodate an additional heat load of 1.0 million BTU/hour. CP&L's longer-term plan is to upgrade the CCW system, in a manner not yet specified, so as to accommodate an additional heat load of 15.6 million BTU/hour. The CCW upgrade must also accommodate an increase in the rated power of the Harris reactor. CP&L expects that the design of the CCW

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[3] License amendment application, Enclosure 7, page 5-16.
[4] License amendment application, Enclosure 5.

upgrade will commence in mid-1999 and will be completed in early 2001, one year after the company expects pool C to enter service.

Safety implications

In order to exploit the margin in the existing CCW system so as to cool pools C and D, CP&L may be obliged to require its operators to divert some CCW flow from the RHR heat exchangers during the recirculation phase of a design-basis loss-of-coolant accident (LOCA) event at the Harris reactor.[5] This is a safety issue because, during the recirculation phase of a LOCA, operation of the RHR system is essential to keeping the reactor core and containment in a safe condition. CP&L's exploitation of the margin in the existing CCW system is deemed by CP&L and NRC to constitute an "unreviewed safety question".[6]

In Enclosure 9 of its license amendment application, CP&L provides a brief description of the analysis that is has performed to demonstrate that an additional load of 1.0 million BTU/hour is within the marginal capacity of the Unit 1 CCW system. That analysis is said by CP&L to take the form of a 10CFR50.59 Safety Evaluation. The description in Enclosure 9 raises more questions than it answers, and does not address the practical issues that affect an analysis of a cooling system's thermal margin. For example, CP&L has mentioned elsewhere that exploitation of the margin in the Unit 1 CCW system could involve changes in design assumptions that include fouling factors and tube plugging limits.[7] These matters are not addressed in Enclosure 9.

As background, note that the Unit 1 CCW system has two heat exchangers, each with a design heat transfer rate of 50 million BTU/hour. During the recirculation phase of a design-basis LOCA, the estimated maximum heat load to be extracted from the CCW system by the SW system is 160 million BTU/hour.[8] These numbers suggest that accommodating a design-basis LOCA will already exploit the margin of the CCW system, without any additional load from pools C and D.

Lack of QA documentation

Activation of pools C and D will require the completion of their cooling and water cleanup systems, and the connection of their cooling systems to the

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[5] License amendment application, Enclosure 9.
[6] Ibid; Federal Register: January 13, 1999 (Volume 64, Number 8), pages 2237-2241.
[7] Viewgraphs for presentation by CP&L to the NRC staff, 3 March 1998.
[8] Harris FSAR, section 9.2, Amendment No. 40.

existing CCW system. CP&L states that approximately 80 percent of the necessary piping was completed before the second Harris reactor was cancelled.[9] However, some of the quality assurance (QA) documentation for the completed piping is no longer available. Much of the completed piping is embedded in concrete and is therefore difficult or impossible to inspect. To address this situation, CP&L proposes an Alternative Plan to demonstrate that the previously completed piping and other equipment is adequate for its purpose.[10] Nevertheless, the cooling systems for pools C and D will not satisfy ASME code requirements.

Electrical power

The cooling systems for pools C and D will draw electrical power from the electrical systems of Unit 1. If electricity supply to the cooling pumps for pools C and D is interrupted, the pools will heat up and eventually boil. CP&L says that pools C and D will begin to boil after a time period "in excess of 13 hours", assuming a bounding decay heat load of 15.6 million BTU/hour.[11] To prevent the onset of pool boiling in the event of a loss of offsite power, the Harris operators may be obliged to provide electrical power to pools C and D from the existing emergency diesel generators, which also serve pools A and B and the Unit 1 reactor. In its license amendment application, CP&L does not address the ability of the emergency diesel generators to meet the additional electrical loads associated with pools C and D. CP&L does mention in the Harris FSAR the potential for connecting "portable pumps" to bypass the pool cooling pumps should the latter be inoperable.[12] However, the characteristics, capabilities and availability of such portable pumps are not addressed in the license amendment application.

4. Potential cesium-137 inventory of the Harris pools

For the purposes of Appendix E of this report, it is necessary to estimate the potential inventory of the radioisotope cesium-137 in the Harris pools. As a starting point, consider the inventory of cesium-137 in a typical PWR spent fuel assembly, represented here by an average assembly in batch 16 from the Ginna plant, discharged in April 1987. At discharge, the Ginna assembly contained 1.4×10^5 Curies of cesium-137 per metric ton of heavy metal (MTHM).[13]

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[9] License amendment application, Enclosure 1, page 4.
[10] License amendment application, Enclosure 8.
[11] License amendment application, Enclosure 7, page 5-8.
[12] Harris FSAR, page 9.1.3-6, Amendment No. 48.
[13] V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987, Appendix A.

A Harris PWR assembly has a mass of 0.461 MTHM. Thus, one can estimate that a typical Harris assembly contains, at discharge, 0.65×10^5 Curies of cesium-137. The assembly's content of cesium-137 will decline exponentially, with a half-life of 30 years. At the same age after discharge, a typical BWR assembly in the Harris pools will contain about 1/4 of the amount of cesium-137 in a Harris PWR assembly.[14]

Potential stock of assemblies in the Harris pools

Table A-2 shows CP&L's projection of the stock of assemblies in Harris pools C and D, for the purposes of bounding analysis. A CP&L representative has stated that CP&L will not ship fuel to Harris until it has aged for 3 years, and will not place fuel in pools C and D until it has aged for 5 years.[15] Accepting that fuel aged less than 3 years will not be shipped to Harris, one can assume, to supplement Table A-2, that the Harris pools will contain 456 BWR assemblies aged for 3 years, 172 PWR assemblies aged for 3 years, and 96 PWR assemblies aged for 1 year. Hereafter, these assumptions and Table A-2 are taken to represent the potential stock of fuel assemblies in the Harris pools.

On this basis, the Harris pools' stock of spent fuel aged 3 years or less will be 268 PWR assemblies and 456 BWR assemblies. All of this fuel might be in pools A and B, although there is nothing in CP&L's present or proposed Technical Specifications which prohibits placement of recently discharged fuel in pools C and D. On the same basis, the Harris pools' stock of spent fuel aged 9 years or less will be 784 PWR assemblies and 1,824 BWR assemblies.

Inventory of cesium-137

Now consider the inventory of cesium-137 in the Harris pools. Assume that a newly discharged PWR assembly contains 0.65×10^5 Curies of cesium-137, neglect the difference between Harris and Robinson assemblies, allow for radioactive decay, and assume that a BWR assembly contains 1/4 of the amount of cesium-137 in a PWR assembly of the same age. Then, the Harris pools' stock of spent fuel aged 3 years or less will contain 2.3×10^7 Curies (870,000 TBq) of cesium-137, with a mass of 260 kilograms. Also, the Harris pools' stock of spent fuel aged 9 years or less will contain 7.1×10^7 Curies (2,600,000 TBq) of cesium-137, with a mass of 790 kilograms.

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[14] The ratio of 1/4 derives from the parameters shown in the license amendment application, Enclosure 7, page 5-15.
[15] J Scarola of CP&L, presentation to Orange County Board of Commissioners, 9 February 1999.

CP&L could provide a more precise projection of the cesium-137 inventory in the Harris pools over coming years. However, our estimate will be a reasonable indication of cesium-137 inventory during the next two decades, assuming pools C and D are used as CP&L intends.

For comparison with the pools' inventory of cesium-137, note that the NRC has estimated the inventory of cesium-137 in the Harris reactor core, during normal operation, to be 4.2×10^6 Curies (155,000 TBq, or 47 kilograms).[16] This represents an average inventory of 0.27×10^5 Curies in each of the reactor's 157 fuel assemblies. Note that an average assembly in the core will have a lower cesium-137 content than an assembly at discharge, and that the NRC's estimate may have assumed a relatively low fuel burnup.

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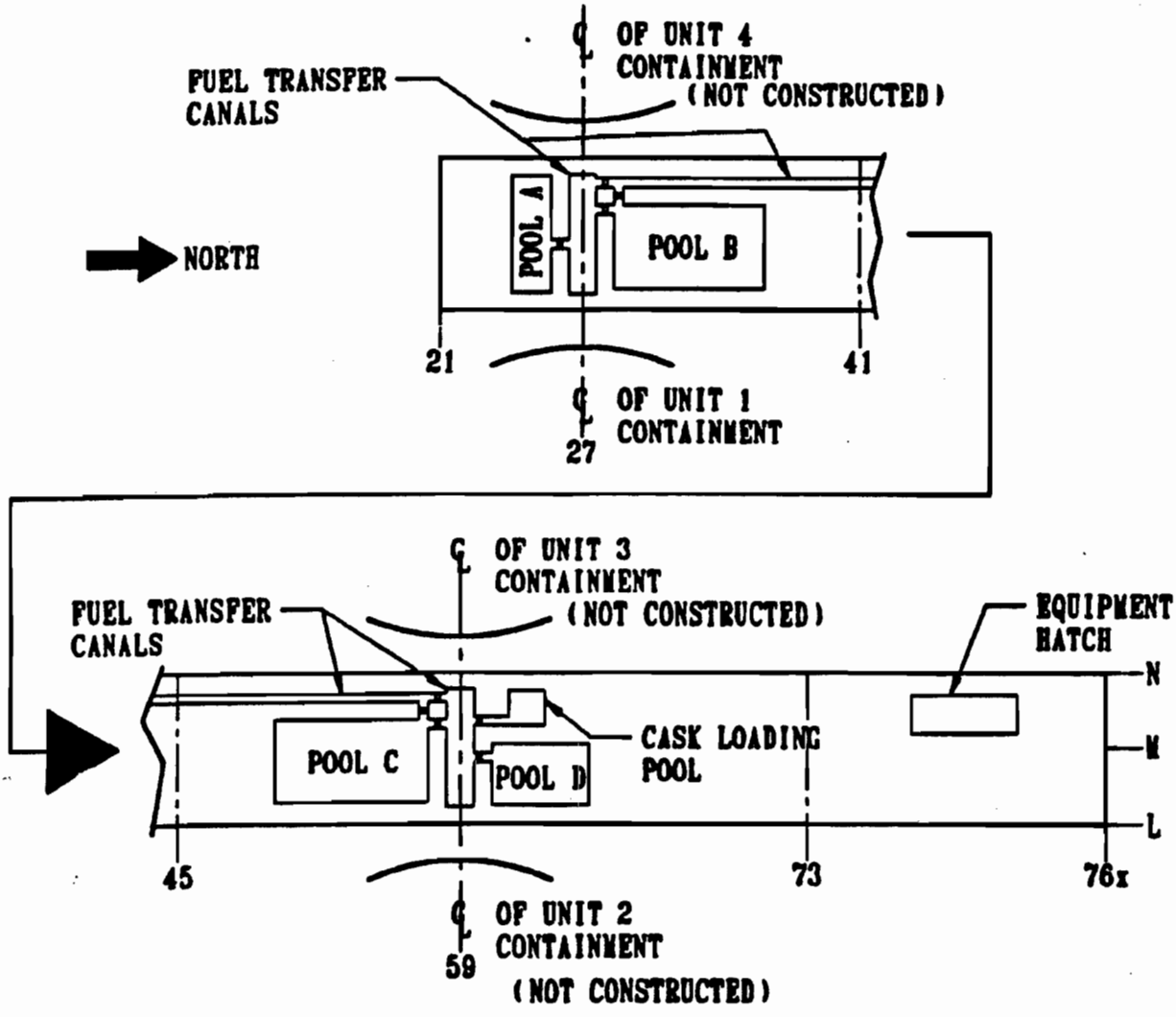
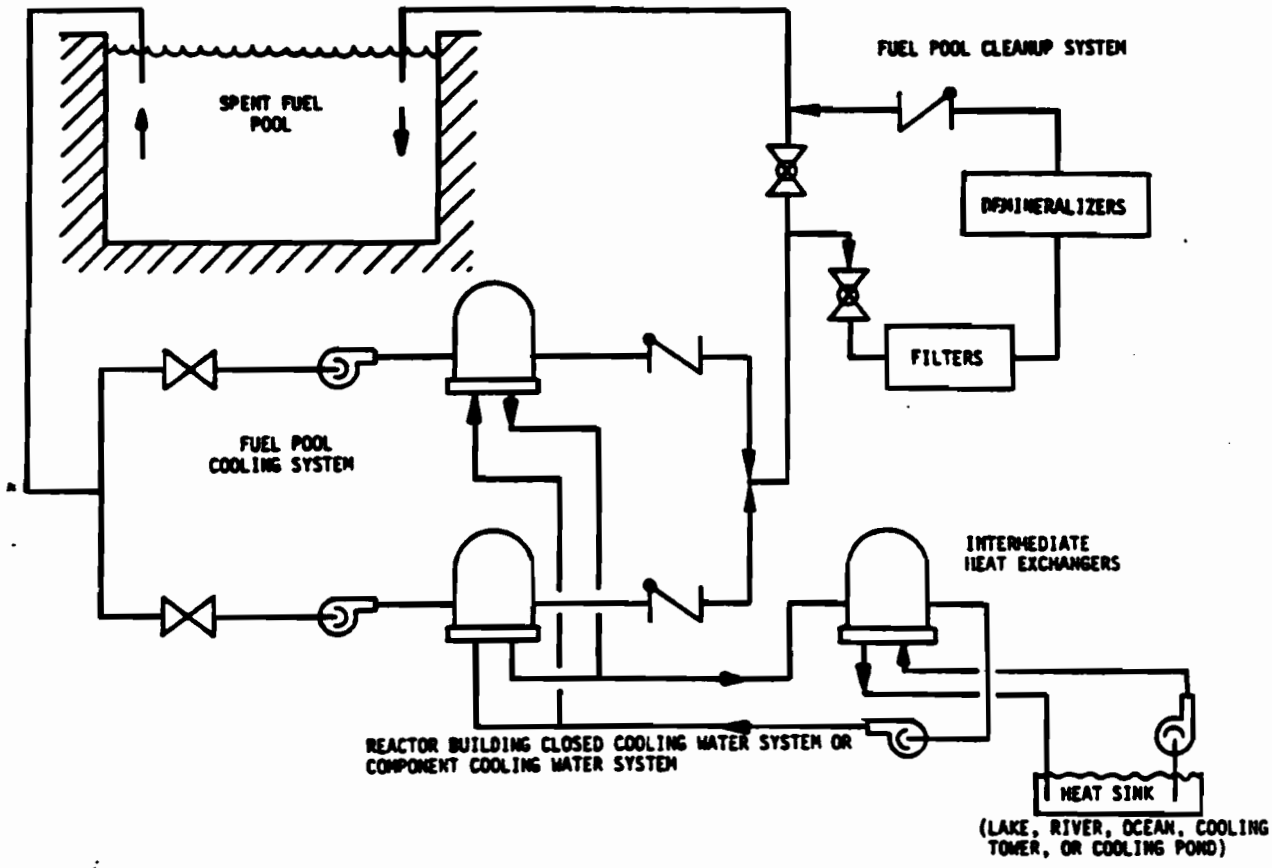
[16] US Nuclear Regulatory Commission, Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2, NUREG-0972, October 1983.



Source: License amendment application

Figure A-1

Interior of the Harris Fuel Handling Building



Source: NUREG-0404

Figure A-2

Typical cooling and cleanup systems for a spent fuel pool

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Pool	PWR spaces	BWR spaces	Total
'A'	360	363	723
'B'	768	2178	2946
'C'	927	2763	3690
'D'	1025	0	1025
Total	3080	5304	8384

Source: License amendment application

Table A-1

Present and proposed storage capacity in the Harris pools

DECAY PERIODS FOR A BOUNDING POOLS C AND D STORAGE CONFIGURATION			
PWR Fuel Assemblies		**BWR Fuel Assemblies**	
Number of Assys	Decay Period	Number of Assys	Decay Period
172	5 years	456	5 years
172	7 years	456	7 years
172	9 years	456	9 years
172	11 years	456	11 years
172	13 years	456	13 years
172	15 years	483	15 years
172	17 years		
172	19 years		
172	21 years		
172	23 years		
232	25 years		

Source: License amendment application

Table A-2

Projected stock of fuel assemblies in Harris pools C and D

RISKS AND ALTERNATIVE OPTIONS
ASSOCIATED WITH SPENT FUEL STORAGE AT THE
SHEARON HARRIS NUCLEAR POWER PLANT

Appendix B

Potential for severe accidents at the Harris reactor

1. Introduction

In examining the risks associated with spent fuel storage at Harris, one must consider the potential for accidents at the Harris reactor. Such consideration is necessary for two reasons. First, a reactor accident could accompany, initiate or exacerbate a spent fuel pool accident. Second, modification of the Harris plant to increase its spent fuel storage capacity could increase the probability or consequences of accidents at the Harris reactor.

This appendix addresses the potential for severe accidents at the Harris reactor. "Severe" reactor accidents have two major defining characteristics. First, they involve substantial damage to the reactor core, with a corresponding release of radioactive material from the fuel assemblies. Second, they extend the envelope of potential accidents beyond the "design basis" accidents that were considered when US reactors were first licensed.

During a severe reactor accident, radioactive material may be released to the environment, as an atmospheric plume or by entry into ground or surface waters. The release may be large or small. In illustration, the 1979 TMI accident and the 1986 Chernobyly accident were both severe accidents, involving substantial damage to the reactor core. However, the TMI release was comparatively small and the Chernobyl release was comparatively large.

2. Probabilistic risk assessment

The probabilities and consequences of potential accidents at nuclear facilities can be estimated through the techniques of probabilistic risk assessment (PRA). Nuclear facility PRAs are performed at three levels. At Level 1, a PRA will estimate the probability of a specified type of accident (e.g., severe core damage at a reactor). At Level 2, which builds upon Level 1 findings, a PRA will estimate the nature of potential radioactive releases from the facility. In

turn, the Level 2 findings can be used in a Level 3 exercise, which will estimate the offsite consequences (health effects, economic effects, etc.) of radioactive releases. For all three levels, a PRA can be performed for "internal" accident-initiating events (equipment failure, operator error, etc.) and for "external" accident-initiating events (earthquakes, floods, etc.).[1]

PRA methodology is used for non-reactor nuclear facilities, but is most highly developed in its application to reactors. The first PRA was the Reactor Safety Study (WASH-1400), which was published by the US Nuclear Regulatory Commission (NRC) in 1975.[2] The present state of the PRA art is exemplified by a study of five nuclear power plants (NUREG-1150) published by the NRC in 1990.[3]

Uncertainty and incompleteness of PRA findings

An in-depth PRA such as NUREG-1150 can provide useful insights regarding a reactor's accident potential. However the findings of any PRA will inevitably be accompanied by substantial uncertainty and incompleteness. Uncertainty arises from the intrinsic difficulties of modelling complex systems, and from limited understanding of some of the physical processes that accompany severe accidents. Incompleteness arises from the potential for unanticipated accident sequences, gross human errors, undetected structural flaws, and acts of malice or insanity.[4] Thus, a PRA's finding about the probability of an accident should be viewed with two caveats. First, the accident probability, as found in the PRA, will fall within some range of uncertainty. Second, the accident probability, as found in the PRA, will be a lower bound to the true probability, which will be impossible to determine.

NUREG-1150 findings for the Surry PWRs

Figures B-1 and B-2 illustrate the findings of NUREG-1150. These figures show the estimated core damage frequency for the Surry nuclear reactors. These reactors are 3-loop Westinghouse pressurized-water reactors (PWRs), as is the Harris reactor. Core damage frequency is shown per reactor-year of

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[1] In PRA practice, it is common for analysis of externally-initiated accidents to build upon previous analysis of internally-initiated accidents.
[2] US Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400 (NUREG-75/014), October 1975.

[3] US Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG-1150 (2 vols), December 1990.
[4] H Hirsch, T Einfalt, O Schumacher and G Thompson, IAEA Safety Targets and Probabilistic Risk Assessment, Gesellschaft fur Okologische Forschung und Beratung, Hannover, August 1989.

operation. Figure B-1 shows core damage frequency for internal events, fires and earthquakes (seismic events). Two estimates are shown for seismic events, one drawing on an estimate of earthquake frequency by Lawrence Livermore National Laboratory, the other on an estimate by the Electric Power Research Institute (EPRI). The bars in Figure B-1 span an estimated uncertainty range from the 5th to the 95th percentile. An alternative portrayal of estimated uncertainty is provided by the probability densities shown in Figure B-2.

The authors of NUREG-1150 made a considerable effort to estimate the uncertainty associated with their findings. However, their uncertainty estimates relied heavily on expert opinion, rather than on a statistical analysis of data. Thus, the uncertainty estimates in NUREG-1150 should be viewed with caution. The reader will observe a cautionary statement attached to Figures B-1 and B-2. Finally, the NUREG-1150 findings of accident probability must be viewed as lower bounds, as explained above.

Acts of malice

Nuclear reactor PRAs do not consider malicious acts such as sabotage, terrorism or acts of war. Such acts are less susceptible to probabilistic analysis than are accident initiators such as human error. Nevertheless, sabotage and terrorism pose a significant threat to US nuclear plants.[5] NRC regulations oblige reactor licensees to take certain precautions against this threat, but these precautions do not preclude the possibility of successful acts of sabotage or terrorism.

The US government is increasing the level of attention and the expenditure that it devotes to the threat of terrorism. Many observers argue that greater effort is required. For example, three authors with high-level government experience have recently written:[6]

> Long part of the Hollywood and Tom Clancy repertory of nightmarish scenarios, catastrophic terrorism has moved from far-fetched horror to a contingency that could happen next month. Although the United States still takes conventional terrorism seriously, as demonstrated by the response to the attacks on its embassies in Kenya and Tanzania in August, it is not yet prepared for the new threat of catastrophic terrorism.

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[5] G Thompson, War, Terrorism and Nuclear Power Plants, Peace Research Centre, Australian National University, October 1996.
[6] A Carter, J Deutch and P Zelikow, "Catastrophic Terrorism", Foreign Affairs, November/December 1998, page 80.

The effectiveness of licensees' arrangements to resist terrorist attacks on nuclear plants has recently been a subject of public debate. According to the head of the NRC's Operational Safeguards Response Evaluation program, plant security arrangements have failed in at least 14 of the 57 mock assaults which the NRC has conducted since 1991. Nevertheless, the NRC intends to weaken its oversight of licensees' antiterrorism efforts.[7]

3. The Harris IPE and IPEEE

The NRC requires each holder of a reactor license to perform an Individual Plant Examination (IPE), to assess the severe accident potential of that reactor. Carolina Power and Light (CP&L) submitted an IPE for the Harris reactor in 1993.[8] This was a Level 2 PRA for internal events, including in-plant flooding but neglecting in-plant fires.

The NRC also requires each licensee to perform an Individual Plant Examination for External Events (IPEEE). CP&L submitted an IPEEE for the Harris reactor in 1995.[9] This study did not follow PRA practice. Instead, it consisted of a seismic margins analysis and a limited analysis of in-plant fires.

IPE estimate of core damage frequency

According to the IPE performed by CP&L, the frequency of severe core damage at Harris is 7×10^{-5} per reactor-year. This must be considered a "point" estimate, because the Harris IPE does not provide an uncertainty band or probability density function of the kind shown in Figures B-1 and B-2. The IPE predicts that accident sequences involving a loss-of-coolant accident (LOCA) will account for 40 percent of Harris' core damage frequency, while sequences involving station blackout (loss of electrical power) will account for 26 percent of the core damage frequency. The 40 percent contribution of LOCAs to core damage frequency is due to LOCAs with injection failure (17 percent) and LOCAs with recirculation failure (23 percent).

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[7] S Allen, "NRC to cut mock raids on atom plants", The Boston Globe, 25 February 1999, page A6.
[8] Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, Unit No. 1: Individual Plant Examination Submittal, August 1993.
[9] Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, Unit No. 1: Individual Plant Examination for External Events Submittal, June 1995.

The NRC has compiled and compared IPE findings for all US commercial nuclear reactors.[10] Some of the results are shown in Figures B-3 and B-4. Figure B-3 shows that the reported core damage frequencies tend to be significantly higher for PWRs than for boiling-water reactors (BWRs). Figure B-4 shows that the reported core damage frequencies tend to be higher for 3-loop Westinghouse (W-3) PWRs than for 2-loop and 4-loop Westinghouse PWRs and PWRs made by Combustion Engineering (CE) and Babcock & Wilcox (B&W). The Harris reactor is a 3-loop Westinghouse PWR.

From its compilation of IPE findings, the NRC concluded that sequences involving LOCAs (especially LOCAs with recirculation failure) and station blackout are major contributors to estimated core damage frequency at 3-loop Westinghouse PWRs. This conclusion is consistent with the Harris IPE findings outlined above. The NRC noted that the 3-loop Westinghouse PWRs exhibit a relatively high dependence of front-line safety systems on service water (SW), component cooling water (CCW) and heating, ventilating & air conditioning (HVAC) systems.

IPEEE findings

The Harris IPEEE consisted of a seismic margins analysis and a limited analysis of in-plant fires. The seismic margins analysis examined the Harris reactor's ability to withstand a review level eathquake (RLE) of 0.3g. Note that the reactor's safe shutdown earthquake (SSE) is 0.15g and its operating basis earthquake is 0.075g. According to the IPEEE, the only actions required to make the Harris reactor safe against the RLE involved housekeeping and minor modifications, and these actions have been taken. The IPEEE did not investigate the implications of an earthquake more severe than the RLE.

A limited analysis of in-plant fires appears in the IPEEE. This analysis identified four fire scenarios as significant contributors to core damage frequency. One scenario would take place in each of switchgear rooms A and B, and two scenarios would take place in the control room. The combined core damage frequency, summed over all four scenarios, would be 1×10^{-5} per reactor-year, but the IPEEE argues that a summation of this kind would be inaccurate without further refinement of the analysis.

Figures B-1 and B-2 illustrate the findings that can be generated by the systematic application of PRA techniques to accident sequences initiated by external events. In comparison, the Harris IPEEE is a relatively crude study.

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[10] US Nuclear Regulatory Commission, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, NUREG-1560 (3 vols), December 1997.

Release of radioactive material

The Harris IPE analyzes the potential for accident sequences to release radioactive material to the environment. The IPE only considers releases to the atmosphere during accident sequences that are initiated by internal events. Potential releases are described by a set of release categories.

Release category RC-5 represents the largest release identified in the IPE. This release would include 100 percent of the noble gas inventory in the reactor core, 59 percent of the CsI inventory, and 53 percent of the CsOH inventory. The IPE does not describe how cesium would be distributed between CsI and CsOH. Thus, one can interpret the RC-5 release as including 59 percent of iodine isotopes in the core and 53-59 percent of cesium isotopes.

Accident sequences contributing to release category RC-5 would involve steam generator tube rupture (SGTR) with a stuck-open safety relief valve (SRV), or an inter-system LOCA (ISLOCA). The SGTR could occur as an accident initiating event or through overheating of steam generator tubes during an accident sequence initiated by some other event. A stuck-open SRV, concurrent with a SGTR, would create a direct pathway from the reactor core to the atmosphere, bypassing the containment. In an ISLOCA sequence, reactor cooling water would be lost from a breach in a piping system outside the containment. This loss of water would initiate the accident, and the water's escape pathway would provide a route for the escape of radioactivity after core damage began.

An accident in release category RC-5 would cause substantial offsite exposure to radioactivity. In addition, the Harris plant and its immediate surroundings would become radioactively contaminated to the point where access by personnel would be precluded. Accidents in other release categories would release smaller amounts of radioactive material, but could also contaminate the Harris plant to the point where access by personnel would be precluded. This matter is addressed further in Appendix C.

The Harris IPE estimates the probability of release category RC-5 as 3×10^{-6} per reactor-year. Note that the overall probability of core damage is estimated to be 7×10^{-5} per reactor-year. Thus, the IPE predicts that 4 percent of core damage sequences would yield a release in category RC-5. Overall, the IPE predicts that 15 percent of core damage sequences would be accompanied by a

significant degree of containment failure or bypass, with a total probability of about 1×10^{-5} per reactor-year.[11]

4. Pool-reactor interactions

Neither CP&L nor NRC have performed an analysis to determine how a severe accident or a design-basis accident at the Harris reactor might accompany, initiate or exacerbate an accident at the Harris fuel pools, or vice versa.[12] Appendix C shows how a severe reactor accident could initiate a pool accident by precluding personnel access. From Appendix E it can be inferred that a pool accident could similarly preclude access to the reactor.

The Harris IPE does not analyze the implications that activation of pools C and D at Harris might have for severe accidents at the Harris reactor. Appendix A points out that activation of pools C and D will raise two safety issues that could increase the probability of core damage at Harris. First, cooling of pools C and D and a planned uprate in reactor power will place an increased heat load on the component cooling water (CCW) system of Harris Unit 1, thus adding stress to operators and equipment at Harris, potentially increasing the probability of core damage. Second, cooling of pools C and D will create an increased load on the electrical systems at Harris, thereby adding stress to operators and equipment and potentially increasing the probability of core damage. Before activation of pools C and D is permitted, these effects should be examined through a supplement to the Harris IPE.

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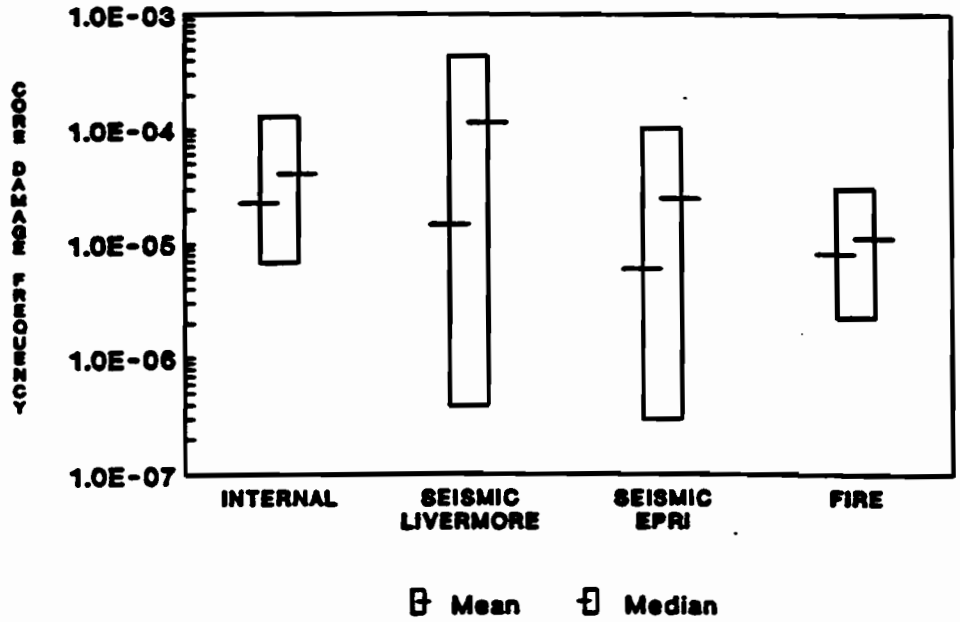
[11] Release categories involving significant containment failure or bypass are, in descending order of estimated probability, RC-4, RC-5, RC-6, RC-1B, RC-4C and RC-3. Each of these categories involves a 100 percent release of noble gases. The CsI release fraction ranges from .001 percent (RC-6) to 59 percent (RC-5).

[12] As examples of literature relevant to potential safety interactions between fuel pools and reactors, see: D A Lochbaum, Nuclear Waste Disposal Crisis, PennWell Books, Tulsa, OK, 1996; and N Siu et al, Loss of Spent Fuel Pool Cooling PRA: Model and Results, INEL-96/0334, Idaho National Engineering Laboratory, September 1996.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Source: NUREG-1150

Figure B-1

Estimated core damage frequency for the Surry PWRs

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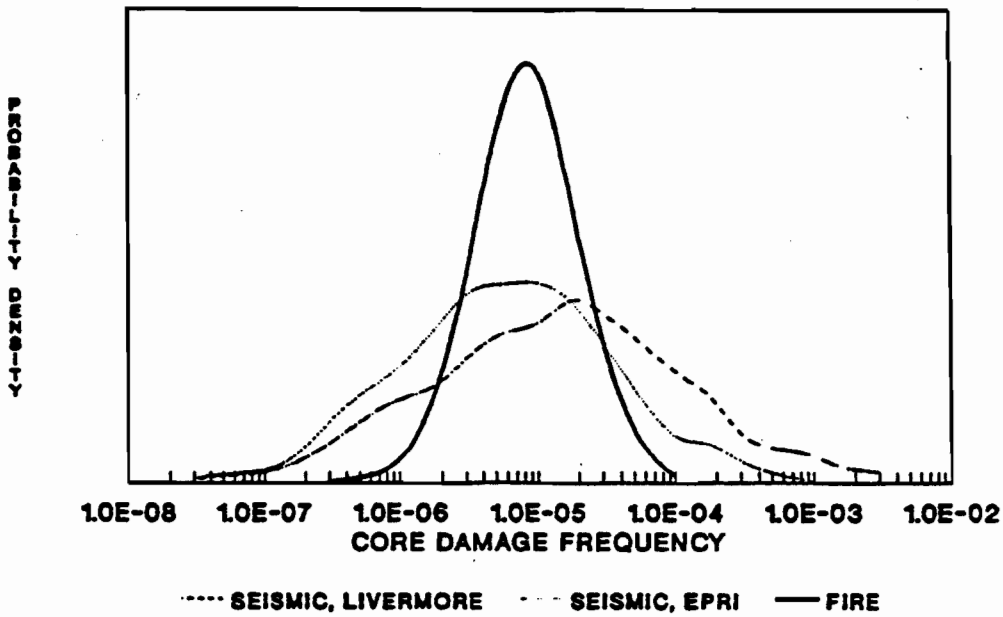
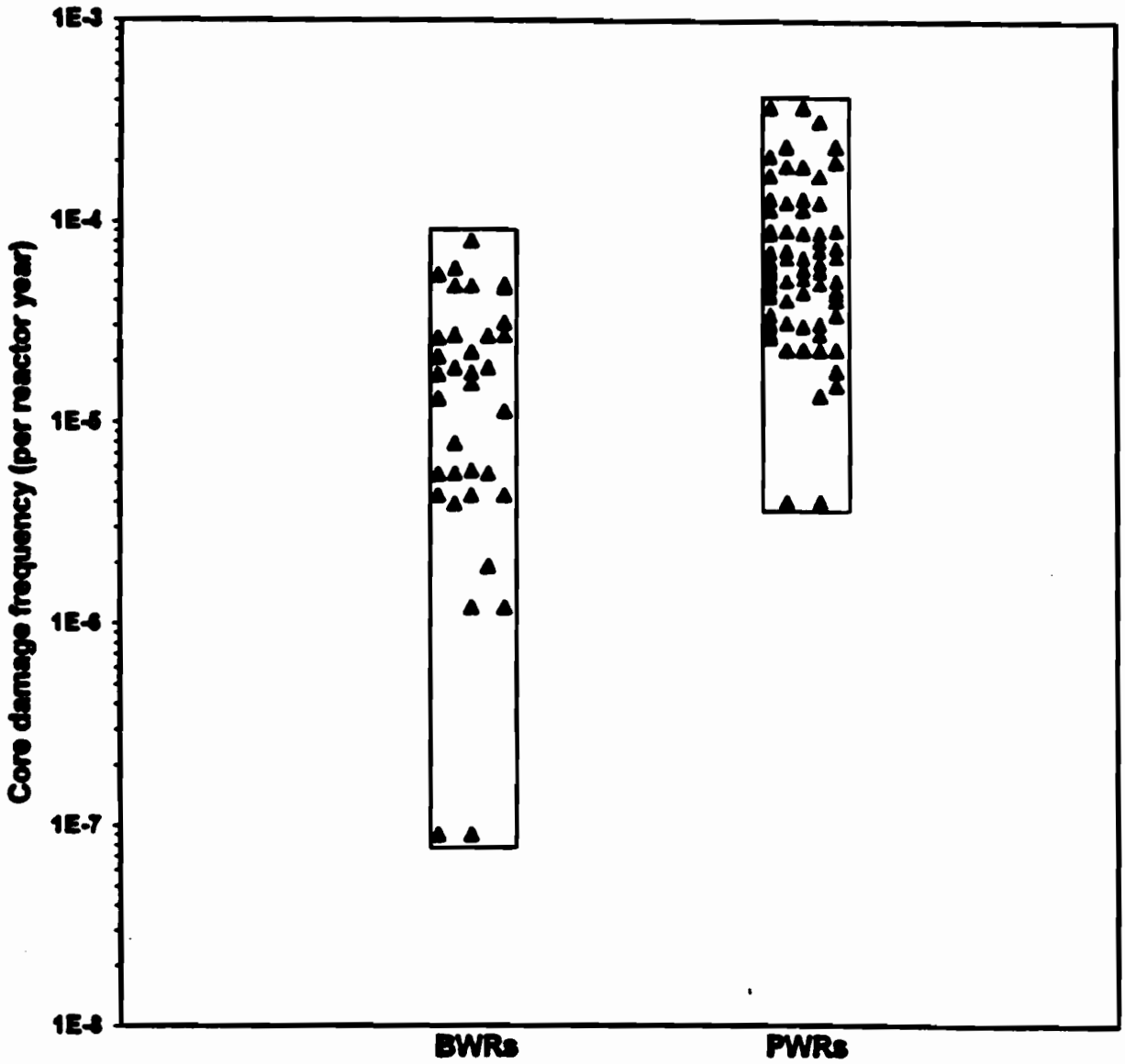


Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Source: NUREG-1150

Figure B-2

Probability density of estimated external-events core damage frequency for the Surry PWRs

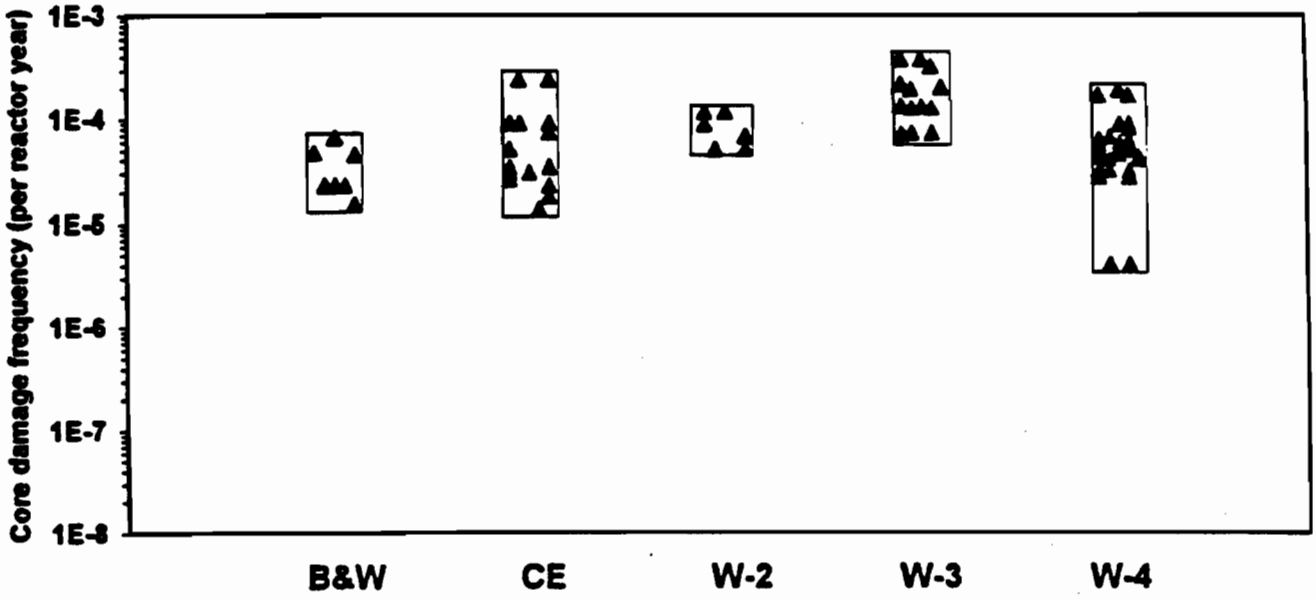


Source: NUREG-1560

Figure B-3

Summary of core damage frequencies as reported in IPEs

67



Source: NUREG-1560

Figure B-4

Core damage frequencies reported in IPEs for types of PWR

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RISKS AND ALTERNATIVE OPTIONS
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SHEARON HARRIS NUCLEAR POWER PLANT

Appendix C

Potential for loss of water from the Harris pools

1. Introduction

This appendix considers the potential for partial or total loss of water from one or more of the Harris fuel pools. The arrangement and use of these pools are described in Appendix A. If a loss of water occurs, then exothermic reactions could occur in the affected pools, as described in Appendix D.

2. Types of event that might cause water loss

A variety of events, alone or in combination, might lead to partial or complete uncovering of spent fuel in the Harris pools. Relevant types of event include:

(a) an earthquake, cask drop, aircraft crash, human error, equipment failure or sabotage event that leads to direct leakage from the pools;
(b) siphoning of water from the pools through accident or malice;
(c) interruption of pool cooling, leading to pool boiling and loss of water by evaporation; and
(d) loss of water from active pools into adjacent pools or canals that have been gated off and drained.

3. Assessing the potential for water loss: the role of PRA

A discipline known as probabilistic risk assessment (PRA) has been developed to examine the probabilities and consequences of potential accidents at nuclear facilities. PRA techniques are most highly developed in their application to reactor accidents, but can be applied to fuel pool accidents. Appendix B describes the characteristics, strengths and limitations of PRA.

Carolina Power & Light Company (CP&L) has prepared a Level 2, internal-events PRA for the Harris reactor, in the form of an Individual Plant

Examination (IPE). CP&L has also performed a limited assessment of the vulnerability of the Harris reactor to earthquakes and in-plant fires, in the form of an Individual Plant Examination for External Events (IPEEE). The findings of the IPE and IPEEE are described in Appendix B.

The Harris IPE and IPEEE could be extended to encompass fuel pool accidents as well as reactor accidents. Such an extension would be logical, because there are various ways in which a severe accident or a design-basis accident at the Harris reactor might accompany, initiate or exacerbate an accident at the Harris fuel pools, or vice versa.[1] However, there is no current indication that CP&L will extend the IPE or IPEEE, or will otherwise apply PRA techniques to potential accidents at the Harris fuel pools.

As an indication of the need for an extended IPE and IPEEE at Harris, covering fuel pool accidents, consider a study performed for the NRC by analysts at the Idaho National Engineering Laboratory.[2] These analysts examined a two-unit boiling-water reactor (BWR) plant based on the Susquehanna plant. They estimated that the plant's probability of spent fuel pool (SFP) boiling events is 5×10^{-5} per year From Appendix B it will be noted that the Harris IPE predicts a core damage frequency of 7×10^{-5} per year. (Years and reactor-years are equivalent for Harris.) The similar magnitudes of these probabilities suggests that pool accidents could be a major contributor to risk at Harris, especially considering the large inventory of long-lived radioisotopes in the Harris pools.

A comprehensive application of PRA techniques to the Harris fuel pools is a task beyond the scope of the author's present work for Orange County. In the remainder of this appendix, selected issues are discussed. These discussions illustrate the need for a comprehensive PRA approach.

4. Analyses of earthquake and cask drop at the Robinson plant

Analysts sponsored by the Nuclear Regulatory Commission (NRC) have examined the effects of a severe earthquake and a cask drop on the fuel pool at CP&L's Robinson plant.[3] The Robinson plant features one pressurized-water reactor (PWR) and a single fuel pool. By examining the vulnerability of

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[1] As examples of literature relevant to potential safety interactions between fuel pools and reactors, see: D A Lochbaum, Nuclear Waste Disposal Crisis, PennWell Books, Tulsa, OK, 1996; and N Siu et al, Loss of Spent Fuel Pool Cooling PRA: Model and Results, INEL-96/0334, Idaho National Engineering Laboratory, September 1996.
[2] N Siu et al, op cit.
[3] P G Prassinos et al, Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants, NUREG/CR-5176, January 1989.

this pool, the NRC sought to obtain knowledge that would be relevant to other PWRs.

Earthquake

The NRC's analysis of the Robinson pool showed that there is high confidence (95 percent) of a low probability (5 percent) of structural failure of the pool in the event of an earthquake of 0.65g. A more severe earthquake could cause structural failure and water loss, and the mean probability of such an event was estimated to be 1.8×10^{-6} per reactor-year.

Cask drop

The NRC's analysts examined a four-foot drop of a 68-ton fuel shipping cask onto the wall of the Robinson fuel pool. They estimated that the wall would suffer significant damage. Cracking of the concrete, yield of reinforcing steel, and tearing of the liner could be expected. Loss of pool water could follow. The probability of this cask drop was not estimated.

Relevance of these findings to Harris

Each nuclear plant has specific design features. Thus, the findings from Robinson cannot be applied uncritically to Harris. Nevertheless, the Robinson findings suggest that the Harris fuel pools may be vulnerable to water loss in the event of a severe earthquake or a cask drop.

The Harris pools are partly below the site's grade level, and the tops of the fuel racks are at grade level. However, there are rooms and passages below the pools. Also, there are three deep cavities adjacent to the fuel handling building, where the containments for Units 2-4 were to have been constructed. Thus, the pools could drain below the tops of the fuel racks, partially or completely, if damaged by an earthquake or cask drop.

Administrative and technical measures are employed at Harris to prevent a cask drop onto a pool wall or into a pool. There is some probability that these measures will fail and a cask drop will occur. No PRA estimate of this probability is available. An NRC-sponsored analysis found the probability of structural failure from a cask drop at the Millstone and Ginna plants, prior to improvements, to be 3×10^{-5} per reactor-year.[4] After improvements, the

[4] V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987, Table 2.10.

probability was estimated to be lower than 2×10^{-8} per reactor-year. Such a low probability is beyond the range of credibility of PRA techniques.

5. A pool accident induced by a reactor accident

The Harris IPE predicts a core damage frequency of 7×10^{-5} per reactor-year. It further predicts that 15 percent of core damage sequences would be accompanied by a significant degree of containment failure or bypass, with a total probability of about 1×10^{-5} per reactor-year.[5] The resulting releases could initiate a pool accident by precluding personnel access.

Radiation levels close to the plant

Figure C-1 shows the estimated whole-body dose to exposed persons following a severe reactor accident.[6] The dose shown is averaged over a range of meteorological conditions and a set of potential atmospheric releases (PWR 1-5) from the NRC's 1975 Reactor Safety Study. Those releases involved a cesium release fraction ranging from 1-50 percent. A similar figure could be drawn for the releases predicted by the Harris IPE, with a qualitatively similar result.

From Figure C-1 it will be seen that an unprotected person one mile from the plant will receive a whole-body dose of about 1,000 rem over one day. Closer to the plant, the dose will be much higher, as shown in Figure C-2.[7] It has been estimated that the dose rate within a reactor containment, following a severe accident, will be 4 million rem per hour.[8] Given containment failure or bypass, doses approaching this level could be experienced outside the containment, in locations such as the fuel handling building.

Health effects of high dose levels

A radiation dose of 500-1,000 rem will normally kill an adult person within a few weeks, due to bone marrow damage. Doses of 1,000-5,000 rem will damage the gastro-intestinal tract, causing extensive internal bleeding and

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[5] Release categories involving significant containment failure or bypass are, in descending order of estimated probability, RC-4, RC-5, RC-6, RC-1B, RC-4C and RC-3. Each of these categories involves a 100 percent release of noble gases. The CsI release fraction ranges from .001 percent (RC-6) to 59 percent (RC-5).

[6] Figure C-1 is adapted from Figure 3.5-10 of: B Shleien, Preparedness and Response in Radiation Accidents, US Department of Health and Human Services, August 1983.

[7] Figure C-2 is adapted from Slide 16 of: J A Martin et al, Pilot Program: NRC Severe Reactor Accident Incident Response Training Manual, NUREG-1210, February 1987, Volume 4.

[8] R P Burke et al, In-Plant Considerations for Optimal Offsite Response to Reactor Accidents, NUREG/CR-2925, November 1982, Table B.2.

death within a few days. Doses above 10,000 rem will lead to failure of the central nervous system, causing death within a day.[9]

Prevention of access, and its implications

It is clear that a severe accident at the Harris reactor, accompanied by containment failure or bypass, would preclude personnel access to the plant. To this author's knowledge, CP&L has made no preparations to maintain pool cooling after such an event. It can be assumed that pool cooling would cease during the accident, and would not resume.

In CP&L's application for a license amendment to activate pools C and D at Harris, the bounding decay heat load for pools C and D is estimated to be 15.6 million BTU/hour (4.6 MW). CP&L states that the mass of water in these two pools, above the racks, will be 2.9 million pounds (1,320 tonnes). Then, CP&L estimates that the pools will begin to boil, if pool cooling systems become inoperative, after a period "in excess of 13 hours".[10] If we assume that cooling remains inoperative, and that 4.6 MW of heat is solely devoted to boiling off 1,320 tonnes of water, then this water will be entirely evaporated over a period of 180 hours (7.5 days). In practice, a slightly longer period will be required, accounting for heat losses.

Thus, a severe reactor accident with containment failure or bypass would lead to uncovering of spent fuel in the Harris pools, after a time delay of perhaps 10 days. Heroic efforts would be needed to restore cooling or to replace evaporated water. If these efforts involved addition of water to the pools after the fuel had been uncovered, they would run the risk of exacerbating the accident by inhibiting convective circulation of air in the pools (see Appendix D).

6. A sabotage/terrorism event involving siphoning

Appendix B discusses the potential for acts of malice at nuclear plants. A potential act of this kind at Harris would involve a group taking control of the fuel handling building, shutting down the pool cooling systems, and siphoning water from the pools. The consequent uncovering of fuel could initiate an exothermic reaction in recently discharged fuel within a few hours (see Appendix D). Once such a reaction was initiated, access to the fuel handling building would be precluded. Over the subsequent hours, exothermic reactions would be initiated in older fuel.

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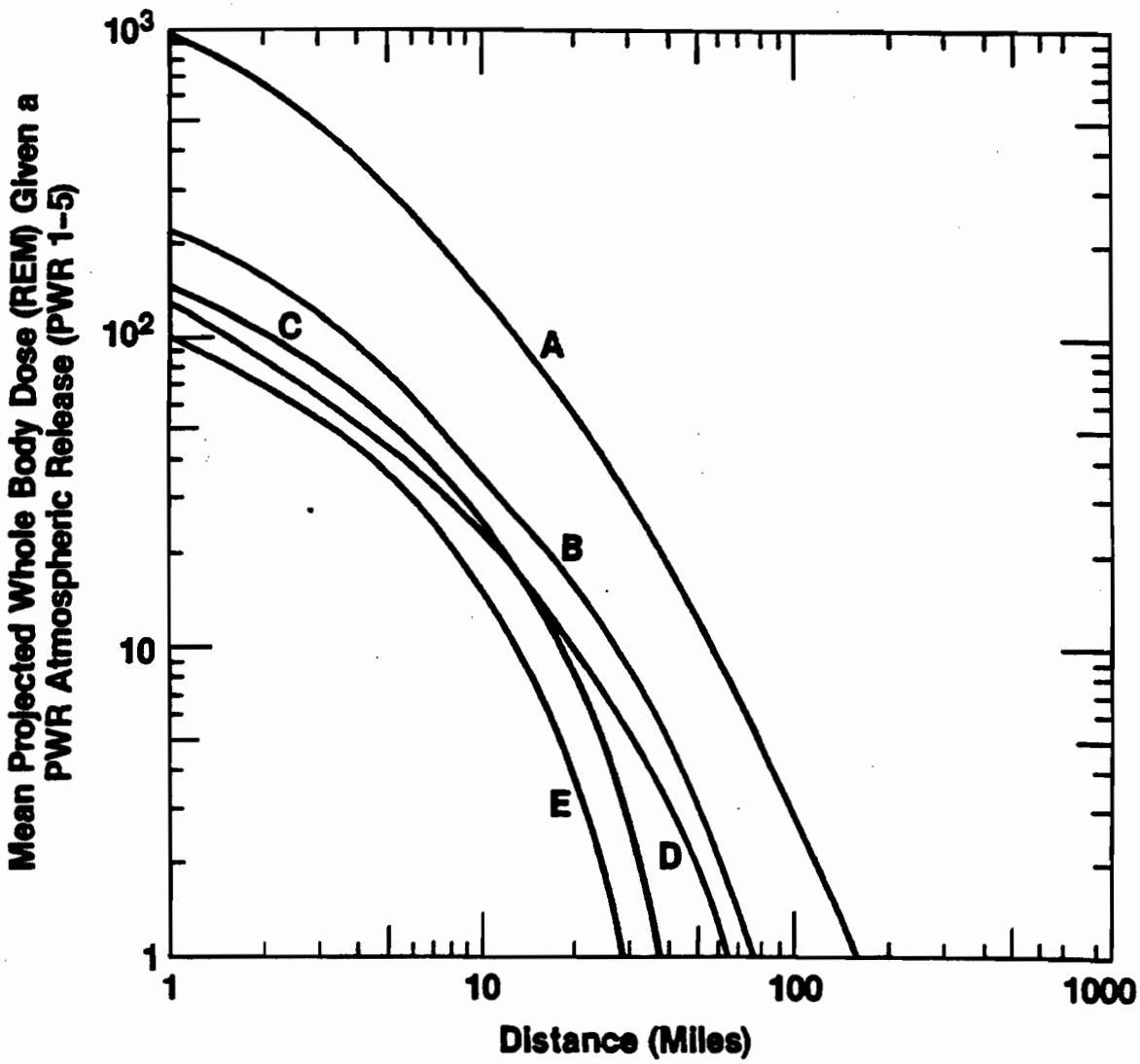
[9] B Flowers et al, Royal Commission on Environmental Pollution, Sixth Report, Cmnd. 6618, Her Majesty's Stationery Office, London, September 1976, page 23.
[10] License amendment application, Enclosure 7, page 5-8.

The group would require military skills and equipment to take control of the fuel handling building. Siphoning water from the pools would be a comparatively easy task. Escape by the group would be difficult but not impossible. The probability of this scenario cannot be predicted by PRA techniques.



Curve A Individual located outdoors without protection. SF's (1.0, 0.7). 1-day exposure to radionuclides on ground.

Curve B Sheltering, SF's (0.75, 0.33), 6-hour exposure to radionuclides on ground.

Curve C Evacuation, 5-hour delay time, 10 mph.

Curve D Sheltering, SF's (0.5, 0.08), 6-hour exposure to radionuclides on ground.

Curve E Evacuation, 3-hour delay time, 10 mph.

Figure C-1

Estimated whole-body dose after a severe PWR accident

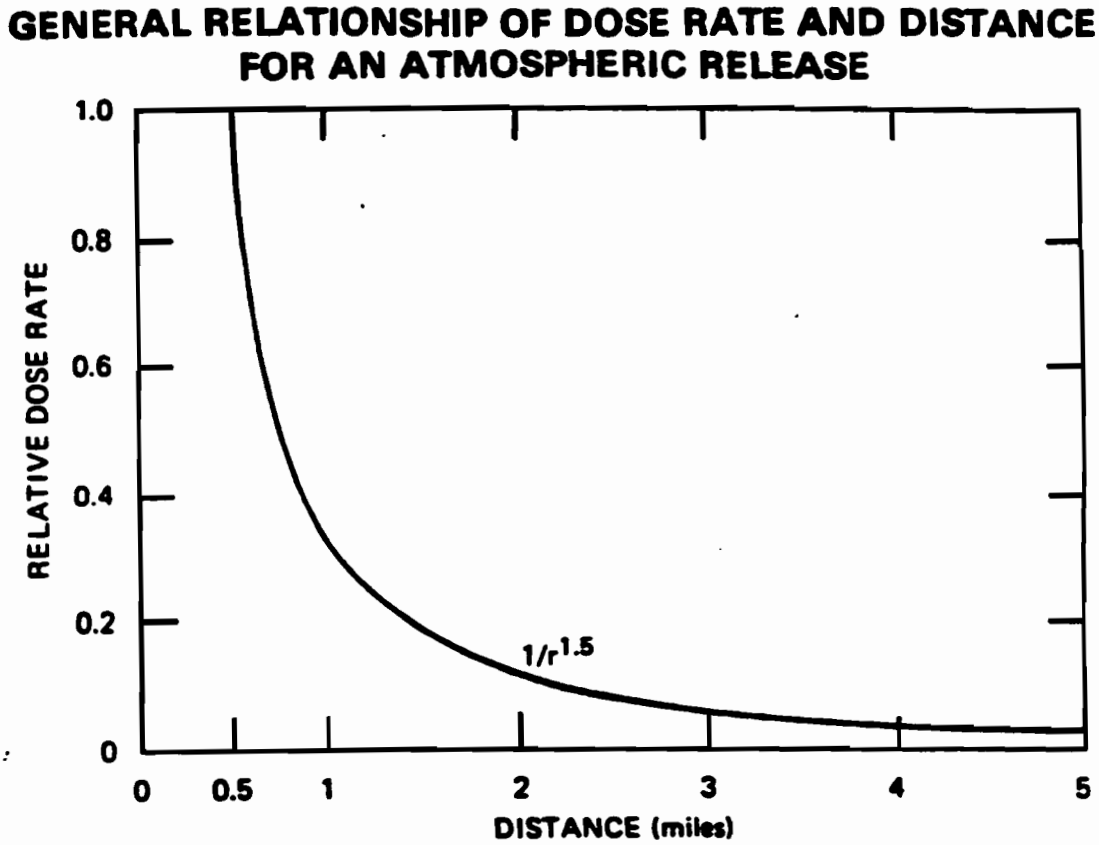


Figure C-2

Dose-distance relationship for a severe reactor accident

RISKS AND ALTERNATIVE OPTIONS
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Appendix D

Potential for exothermic reactions in the Harris pools

1. Introduction

If water is totally or partially lost from one or more of the Harris fuel pools, the potential exists for an exothermic reaction between the fuel cladding and air or steam. The cladding is a zirconium alloy that begins to react vigorously with air or steam when its temperature reaches 900-1,000 degrees C. Partial or total loss of water could cause the cladding to reach this temperature, because water is no longer available to remove decay heat from the fuel. If the cladding temperature reaches 900-1,000 degrees C and air or steam remain available, a runaway reaction can occur. Heat from the exothermic reaction can increase cladding temperature, which will in turn increase the reaction rate, resulting in a runaway reaction.

The steam-zirconium reaction will be familiar to many observers of the 1979 TMI accident. During that accident a steam-zirconium reaction contributed to the partial melting of the reactor core, and generated hydrogen gas. Accumulation of this gas in the upper part of the reactor pressure vessel was a cause of concern during the accident. Hydrogen entered the containment and exploded about 10 hours into the accident, yielding a pressure spike of 28 psig.[1]

The potential for a partial or total loss of water from the Harris pools is addressed in Appendix C. Here, the consequent potential for exothermic reactions is considered. Also, this appendix considers the potential for exothermic reactions to release radioactive material -- especially the radioisotope cesium-137 -- from spent fuel to the atmosphere outside the Harris plant.

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[1] G Thompson, Regulatory Response to the Potential for Reactor Accidents: The Example of Boiling-Water Reactors, Institute for Resource and Security Studies, Cambridge, MA, February 1991.

2. Configuration of the Harris pools

A plan view of the Harris fuel handling building is provided in Figure A-1 of Appendix A. Figure D-1 shows a typical rack used in the Harris fuel pools. Carolina Power & Light Company (CP&L) has not published detailed information about the dimensions and configuration of the Harris racks, claiming that this information is proprietary. The center-center distances in the Harris racks are described in Appendix A.

Figure D-2 shows CP&L's intentions regarding placement of racks in pool C at Harris. It will be noted that the largest gap between the racks and the pool wall will be 2.4 inches, while the gap between racks will typically be 0.6 inches. In other words, the pool will be tightly packed with racks. Moreover, the racks will be tightly packed with fuel.

Effect of pool configuration on convective heat transfer

Examination of Figures D-1 and D-2 shows that convective circulation of air or water through the racks is limited to one pathway. Water (if the pool is full) or air (if the pool is empty) must enter the racks from below and pass upward through the fuel spaces. During Phases I and II of rack placement in pool C, air or water could reach the base of the racks from parts of the pool without racks. After racks are placed in Phase III, air or water must pass downward in the gap (1.4-2.4 inches) between the racks and the pool wall, and then travel horizontally across the bottom of the pool before entering racks from below.

It is further evident that the presence of residual water in the lower part of the pool would prevent convective circulation of air through the racks, in any of the three phases of rack placement. In this case, the only significant source of convective cooling would be from steam rising through the racks. This steam would be generated by the passage of heat from fuel assemblies to residual water, via conduction or thermal radiation.

Heat transfer pathways

Heat will be generated in the fuel assemblies by radioactive decay. Also, heat will be generated by exothermic reactions with zirconium, if these reactions are initiated. In the event of partial or total loss of water from a pool, the following pathways will be available to remove heat from the fuel assemblies, assuming that the assemblies remain intact:

(a) upward convection of air (for total loss of water) or steam (for partial loss of water);
(b) upward or downward conduction along the fuel rods and rack structure;
(c) upward or downward thermal radiation along the narrow passages between fuel rods, and between assemblies and rack walls;
(d) upward thermal radiation from the top of the racks to the interior of the fuel handling building;
(e) downward thermal radiation from the bottom of the racks to the base of the pool or to residual water (if present); and
(f) lateral conduction and thermal radiation across the racks to the pool wall.

For a fuel assembly separated from the pool wall by more than a few spaces, pathway (f) will be ineffective. Thus, only pathways (a) through (e) need to be considered. In the event of total loss of water, the effectiveness of pathway (a) will depend upon the extent of ventilation in the fuel handling building.

3. A scoping approach to heat transfer

To assess the effectiveness of the above-mentioned heat transfer pathways, it is appropriate to begin with a scoping analysis. Detailed calculations, especially if they involve computer modelling, must be guided by physical insight. Scoping calculations can help to provide that insight.

Decay heat output

The first parameter to be considered -- designated here as Q -- is the decay heat in a spent fuel assembly. The unit of Q is kW per metric ton of heavy metal (MTHM) in the assembly. For PWR fuel, Q is about 10 kW/MTHM for fuel aged 1 year from discharge, and about 1 kW/MTHM for fuel aged 10 years.[2]

Upper bound of temperature rise

Now consider a fuel pellet which is in complete thermal isolation. Due to decay heat, this pellet will experience a temperature rise of 11Q degrees C per hour.[3] Thus, if Q=10, the temperature rise will be 110 degrees C per hour (2,640 degrees C per day). A temperature rise of 11Q degrees C per hour is the

[2] For fuel burnups typical of current practice, Q will actually be 10-20 percent higher than the values shown here.
[3] Assuming that a uranium dioxide pellet has a specific heat of 300 J/K per kg of pellet (340 J/K per kg of HM).

upper bound to the temperature rise that could be experienced by a fuel assembly, absent the initiation of an exothermic reaction of the cladding.

Heat transfer by conduction

Next, consider conduction along the fuel rods. A Harris PWR assembly has 264 rods, each containing 1.74 kg of HM. Each rod is 12 ft long, with an outer diameter of 0.374 inches, a cladding thickness of 0.0225 inches, and a pellet diameter of 0.3225 inches.[4] Assume that decay heat is generated uniformly along the length of the rod, conduction along the rod is the only heat transfer mechanism, and the two ends of the rod have the same temperature, Y (degrees C). Then, the temperature at the middle of the rod will be Y+2,000Q degrees C.[5] This result could be viewed as counter-intuitive, because the decay heat in each rod is only 0.48Q Watts per meter of rod.

Convective cooling by steam

Now consider convective cooling of a fuel assembly by upward motion of steam that is generated from residual water at the lower end of the assembly. Neglect other heat transfer mechanisms, assume that decay heat is generated uniformly along the length of the fuel rods, and assume that the temperature of the residual water is 100 degrees C. Define S as the submerged fraction of the assembly and T (degrees C) as the temperature of steam leaving the top of the fuel assembly. Neglect the thermal inertia of the pellets and cladding. Then, the amount of steam generated is proportional to S, while the decay heat captured by this steam is proportional to (1-S). It follows that:[6]

$$T = 100 + (2{,}260/2.1) \times [(1-S)/S]$$

Note that Q does not enter this equation. If one-tenth of a fuel assembly is submerged (S = 0.1), this equation yields a T of 9,800 degrees C. A temperature of this magnitude would not be generated in practice, because of thermal inertia and the operation of other heat transfer mechanisms.[7] However, the calculation establishes an important point. Convective cooling of fuel assemblies by steam from residual water will be ineffective when the submerged fraction of the assemblies is small.

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[4] Harris FSAR, Section 1.3, Amendment No. 30.
[5] Assuming that the cladding's thermal conductivity is 17.3 W/mK, the pellets' conductivity is 1.99 W/mK, and pellets are in perfect contact with each other and the cladding.
[6] Assuming that the latent heat of evaporation of water is 2,260 kJ/kg and the specific heat of steam is 2.1 kJ/kgK.
[7] The singularity of the T equation at S=0 reflects the lack of consideration of other heat transfer mechanisms.

Cooling by thermal radiation

If residual water is present, there remains only one potentially effective mechanism of heat transfer from the mid-length of a fuel assembly -- thermal radiation along the axis of the assembly. Note that a Harris PWR assembly has an active length of 12 feet, a cross-section 8.4 inches square, and contains 264 fuel rods plus other longitudinal structures. In the Harris fuel pools, the assembly will be surrounded by continuous sheets of neutron-absorbing material (Boral), and the center-center distance in pool C will be 9.0 inches. In this configuration, axial heat transfer by thermal radiation will be strongly inhibited. However, calculations more detailed than those above are required to estimate the amount of heat that can be transferred by this pathway.

Note that downward heat transfer by radiation will increase the generation of steam from residual water, thus improving the effectiveness of convective cooling by steam. A detailed analysis should consider such effects through coupled calculations.

Summary

The preceding scoping calculations show that conduction and convective cooling by steam will be relatively ineffective. These cooling mechanisms cannot prevent fuel cladding from reaching a temperature of at least 1,000 degrees C -- the initiation point for a runaway exothermic reaction -- even for fuel aged in excess of 10 years. An estimate of the effectiveness of axial radiation cooling -- the only remaining cooling mechanism if residual water is present -- would require more detailed calculations. However, this author does not expect that such calculations would show axial radiation cooling to be more effective than conduction or convective cooling by steam.

If residual water is not present, a fuel assembly can be cooled by convective circulation of air. Estimation of the effectiveness of this mechanism requires an analysis of convective circulation through the pool and the fuel handling building, reflecting practical factors such as constrictions at the base of fuel racks.

4. Specifications for an adequate, practical analysis

There has been no site-specific analysis of the potential for exothermic reactions in the Harris pools. Generic analyses have been performed for and by the US Nuclear Regulatory Commission (NRC). Before addressing the findings and adequacy of the NRC's generic analyses, let us consider the

ingredients that are necessary if an analysis is to provide practical guidance about the potential for exothermic reactions in the Harris spent fuel pools. Sections 2 and 3 of this appendix provide a basis for specifying those ingredients.

Partial and complete uncovering of fuel

First, the analysis should not be limited to instantaneous, complete loss of water from a pool. Such a condition is unrealistic in any accident scenario which preserves the configuration of the spent fuel racks. If water is lost by drainage or evaporation and no makeup occurs, then complete loss of water will always be preceded by partial uncovering of the fuel. If makeup is considered, the water level could fall, rise or remain static for long periods.

Partial uncovering of the fuel will often be a more severe condition than complete loss of water. As shown above, convective heat loss is suppressed by residual water at the base of the fuel assemblies. As a result, longer-discharged fuel with a lower Q may undergo a runaway steam-zirconium reaction during partial uncovering while it would not undergo a runaway air-zirconium reaction if the pool were instantaneously emptied.

In a situation of falling water level, a fuel assembly might first undergo a runaway steam-zirconium reaction, then switch to an air-zirconium reaction as water falls below the base of the rack and convective air flow is established. In this manner, a runaway air-zirconium reaction could occur in a fuel assembly that is too long-discharged (and therefore has too low a Q) to suffer such a reaction in the event of instantaneous, complete loss of water. Conversely, a rising water level could precipitate a runaway steam-zirconium reaction in a fuel assembly that had previously been completely uncovered but had not necessarily suffered a runaway air-zirconium reaction while in that condition. The latter point is highly significant in the context of emergency measures to recover control of a pool which has experienced water loss. Inappropriate addition of water to a pool could exacerbate the accident.

Computer modelling

An adequate analysis of the potential for exothermic reactions will require computer modelling. The modelling should consider both partial and complete uncovering and the transition from one of these states to the other. Also, the modelling should cover: (a) thermal radiation, conduction, and steam or air convection; (b) air-zirconium and steam-zirconium reactions; (c) variations along the fuel rod axis; (d) radial variations within a representative fuel rod, including effects of the pellet-cladding gap; and (e) clad swelling and

rupture. Experiments will probably be required to support and validate the modelling.

Site-specific factors

The analysis can be strongly influenced by site-specific factors. For convective cooling by air, these factors include the detailed configuration of the racks, the pools and the fuel handling building. All relevant factors should be accounted for. This could be done through site-specific modelling. Alternatively, generic modelling could be performed across the envelope of site-specific parameters, with sensitivity analyses to show the effects of varying those parameters.

Propagation of exothermic reactions to adjacent assemblies

After an exothermic reaction has been initiated in a group of fuel assemblies, this reaction might propagate to adjacent assemblies. Due to their lower Q or to other factors, the adjacent assemblies might not otherwise suffer an exothermic reaction. An analysis of propagation should consider the potential for reactions involving not only the fuel cladding but also material (e.g., Boral) in the fuel racks. The analysis should examine the implications of clad and pellet relocation after a reacting assembly has lost its structural integrity. Those implications include the heating of adjacent assemblies and racks by direct contact, thermal radiation, convection, and the inhibition of air circulation. A bed of relocated material at the base of the pool could have all these effects.

5. The 1979 Sandia study

An initial analysis of the potential for exothermic reactions was made for the NRC by Sandia Laboratories in 1979.[8] This was a respectable analysis as a first attempt. It considered partial drainage of a pool, although it used a crude heat transfer model to study that problem, and neglected to consider the steam-zirconium reaction. It did not address the potential for propagation of exothermic reactions to adjacent assemblies. The Sandia authors were careful to state their assumptions and to specify the technical basis for their computer modelling.

Figure D-3 illustrates the findings of the Sandia study. The three lower curves in Figure D-3 show the sensitivity of convective air cooling to the diameter of the hole in the base of the fuel racks. The next higher curve -- the

[8] A S Benjamin et al, Spent Fuel Heatup Following Loss of Water During Storage, NUREG/CR-0649, March 1979.

"blocked inlets" case -- shows the suppression of convective air cooling due to the presence of residual water. The dashed curve shows the effect of an air-zirconium reaction. The runaway nature of that reaction is evident.

Note that the analysis underlying Figure D-3 assumed a cylindrical rack arrangement with a center-center distance of about 13 inches. Also, the analysis assumed a gap of 16 inches between the racks and the pool wall. The Harris racks are more compact and are packed more tightly into their pools. These factors will tend to inhibit convective air cooling at Harris.

6. Subsequent studies

The 1979 Sandia study could have been the first of a series of studies that moved toward the level of adequacy specified in Section 4. Since 1979 the NRC has sponsored or performed a variety of studies related to the initiation of exothermic reactions in fuel pools.[9] However, the scope of these studies has narrowed, and their potential for building on the 1979 study has not been realized.

Failure to consider partial uncovering

A major weakness of the NRC's studies since 1979 has been their focus on a postulated scenario of total, instantaneous loss of water. This appendix shows clearly that partial uncovering of fuel will often be a more severe condition than complete loss of water. Thus, however sophisticated the NRC's modelling of spent fuel heatup might be, the findings have limited relevance to the practical potential for exothermic reactions.

Brookhaven National Laboratory (BNL) has developed the SHARP code to replace the SFUEL code first developed at Sandia. BNL authors have claimed that the SHARP code can more accurately predict spent fuel heatup in realistic spent fuel pool configurations.[10] A review of the SHARP code is beyond the scope of this report. Applied to spent fuel in a generic, high-density configuration in an instantaneously emptied pool, the SHARP code finds that the fuel cladding will reach a "critical" temperature (565 degrees C) if aged less than 17 months for PWR fuel or 7 months for BWR fuel.[11] The relevance of this finding to the Harris pools is unclear.

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[9] See, for example: V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987; and R J Travis et al, A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants, NUREG/CR-6451, August 1997.
[10] R J Travis et al, page 3-4.
[11] Ibid.

Propagation of exothermic reactions

Pursuant to a Freedom of Information request, the NRC released in 1984 a so-called draft report by MIT and Sandia authors on the propagation of an air-zirconium reaction in a fuel pool.[12] This document has been repeatedly cited in subsequent years, although it should properly be regarded as notes toward a draft report. Those notes were submitted to the NRC after the project ran out of funds; it was never completed.

The MIT-Sandia group concluded from computer modelling and experiments that an air-zirconium reaction in fuel assemblies could propagate to adjacent, lower-Q assemblies. They expressed the view that propagation would be quenched in regions of a pool where fuel is aged 3 years or more, but noted the presence of "large uncertainties" in their analysis.

BNL analysts subsequently reviewed these experiments and conducted their own modelling using the same code (SFUEL). In their modelling the BNL analysts chose to terminate the air-zirconium reaction when the cladding reached its melting point.[13] Neither the MIT-Sandia group nor the BNL group examined the implications of clad and pellet relocation after a reacting assembly has lost its structural integrity. The author is not aware of other analyses which address this problem. Thus, the specifications set forth in Section 4 for analysis of propagation have not been met.

7. The potential for an atmospheric release of radioactive material

Spent fuel at Harris which suffers an exothermic reaction will release radioactive material to the fuel handling building. That building is not designed as a containment structure, and is not likely to be effective in this role, given the occurrence of exothermic reactions in one or more pools. A BNL study has concluded that a reasonable, generic estimate of the release fraction of cesium isotopes, from affected fuel to the atmosphere outside the plant, is 100 percent.[14] This release fraction is used in Appendix E.

The amount of fuel that will suffer an exothermic reaction, given a loss of water from the Harris pools, will depend upon the particular scenario. For scenarios which involve partial uncovering of fuel, the reaction could affect fuel aged 10 or more years. For scenarios which involve total loss of water,

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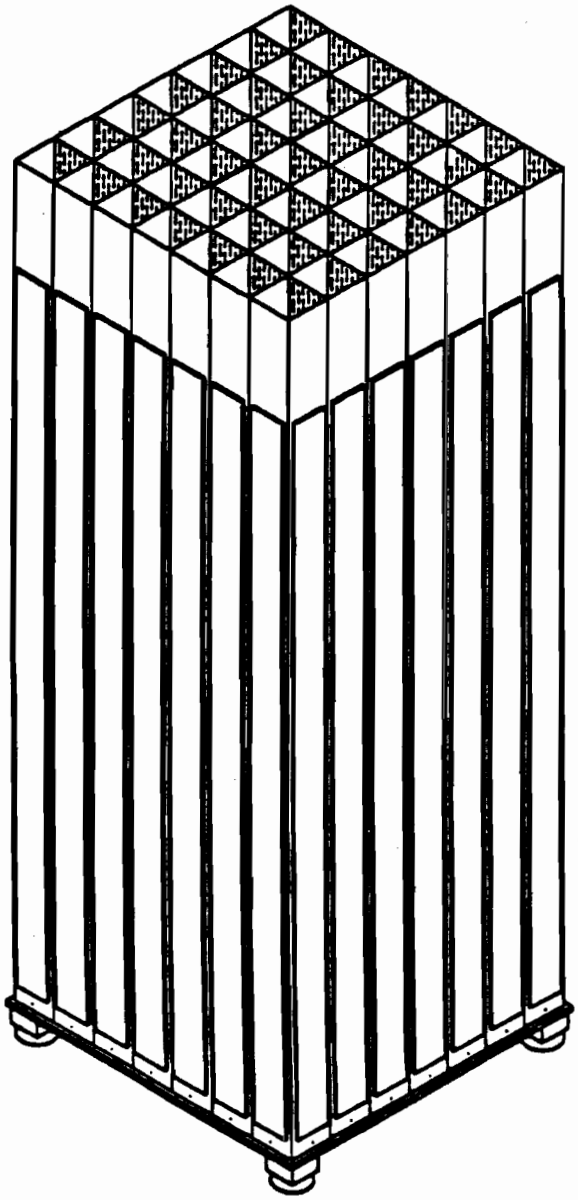
[12] N A Pisano et al, The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool, Draft Report, January 1984.
[13] V L Sailor et al.
[14] Ibid.

the reaction will be initiated only in younger fuel, perhaps aged no more than 1-2 years. However, if clad/pellet relocation is properly factored into a propagation analysis, this analysis may show that a reaction will propagate to much older fuel.

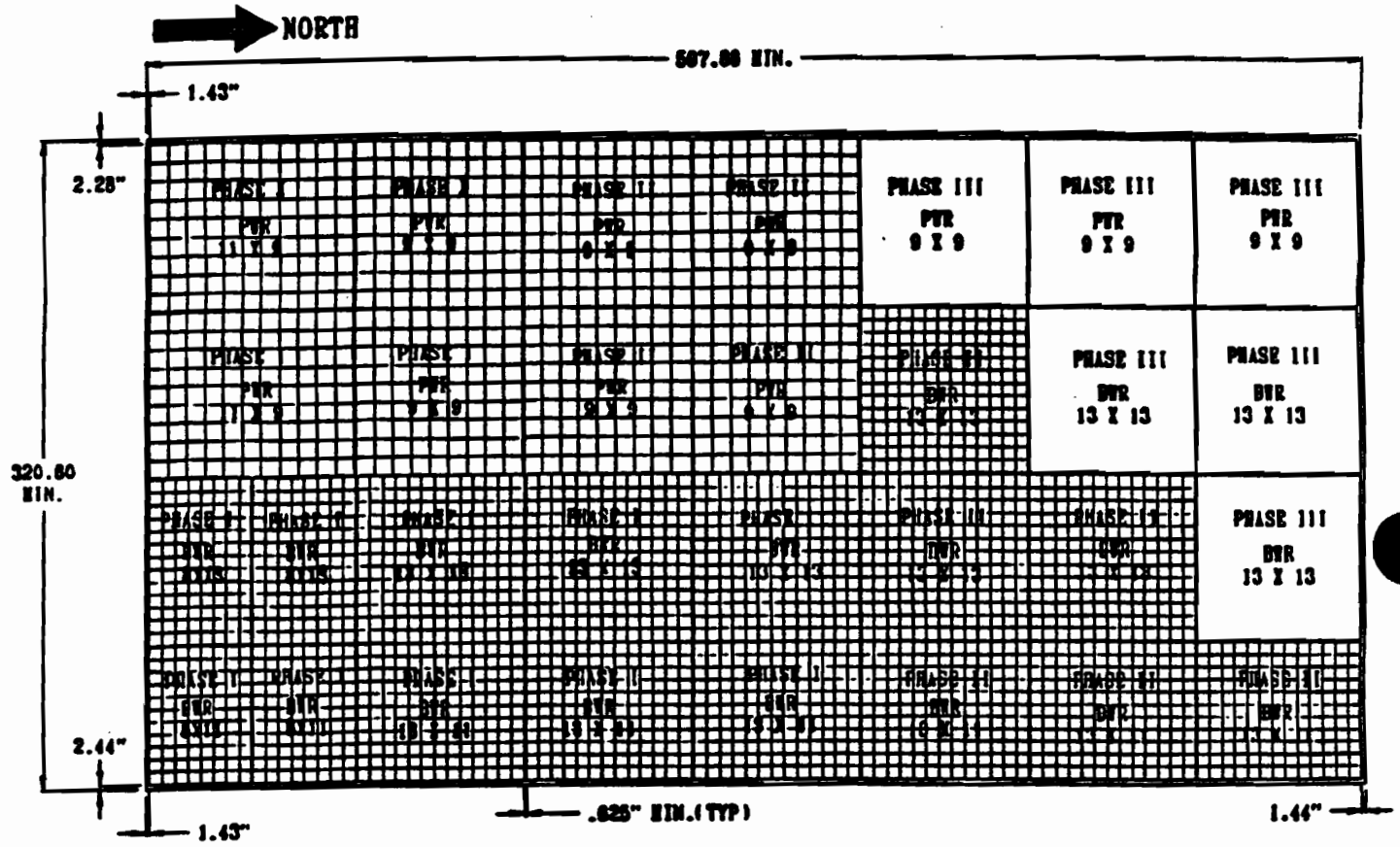
Appendix E considers two potential releases of cesium-137 from the Harris pools. One release corresponds to an exothermic reaction in fuel aged 9 years or less. The other release corresponds to a reaction in fuel aged 3 years or less.



Source: License amendment application

Figure D-1

Typical rack used in the Harris pools

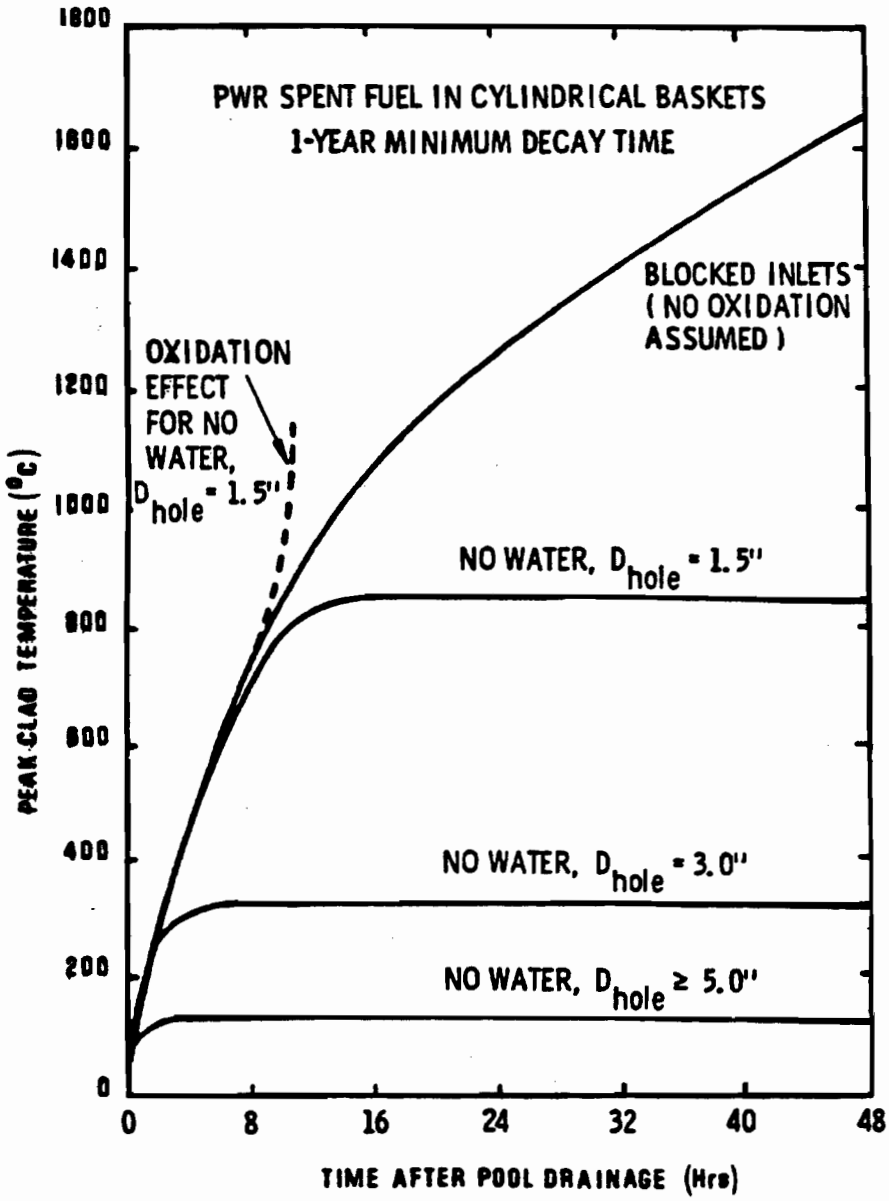


Source: License amendment application

Figure D-2

Proposed rack placement in Harris pool C

89



Source: NUREG/CR-0649

Figure D-3

Estimated heatup of PWR spent fuel after water loss

RISKS AND ALTERNATIVE OPTIONS
ASSOCIATED WITH SPENT FUEL STORAGE AT THE
SHEARON HARRIS NUCLEAR POWER PLANT

Appendix E

Consequences of a large release of cesium-137 from Harris

1. Introduction

This appendix outlines some of the potential consequences of postulated large releases of cesium-137 from the Harris plant to the atmosphere. Such consequences can be estimated by site-specific computer models. A simpler approach is used here, but this approach is adequate to show the nature and scale of expected consequences.

2. Characteristics of postulated releases

Two spent fuel release scenarios are postulated here. The first scenario involves a release of 2.3×10^7 Curies (870,000 TBq) of cesium-137, with a mass of 260 kilograms.[1] This represents the cesium-137 inventory in Harris' stock of spent fuel aged 3 years or less, as estimated in Appendix A. The second scenario involves a release of 7.1×10^7 Curies (2,600,000 TBq) of cesium-137, with a mass of 790 kilograms. This represents the cesium-137 inventory in Harris' stock of spent fuel aged 9 years or less. Note that all of the cesium-137 in the affected fuel is assumed to reach the atmosphere, an assumption which is explained in Appendix D.

Releases of the postulated magnitude could occur as a result of exothermic reactions in the Harris fuel pools. Appendix D discusses the potential for such reactions. Cesium-137 would not be the only radioisotope released to the atmosphere if exothermic reactions occurred in the pools. However, cesium-137 is likely to be the dominant cause of offsite radiological exposure,

[1] 1 Curie is equivalent to 3.7×10^{-2} TBq. 1 TBq of cesium-137 is equivalent to 0.3 grams.

just as it dominates the offsite exposure attributable to the 1986 Chernobyl reactor accident.[2] Note that cesium-137 has a half-life of 30 years.

A severe accident at the Harris reactor could also release cesium-137 to the atmosphere. Appendix A notes that the US Nuclear Regulatory Commission (NRC) has estimated the inventory of cesium-137 in the core of the Harris reactor, during normal operation, to be to be 4.2×10^6 Curies (155,000 TBq, or 47 kilograms). As summarized in Appendix B, an individual plant examination (IPE) study by Carolina Power & Light Company (CP&L) has identified six categories of potential significant release due to severe accidents at the Harris reactor. Release category RC-5, the most severe release category, would involve a release to the atmosphere of 53-59 percent of the cesium isotopes in the reactor core. Thus, given the NRC's estimate of core inventory, release category RC-5 would involve an atmospheric release of 2.2-2.5×10^6 Curies (82,000-92,000 TBq, or 25-28 kilograms) of cesium-137.

Chernobyl and weapons testing releases

For comparison with the above-mentioned potential releases, consider two actual releases -- from the Chernobyl accident and from atmospheric testing of nuclear weapons. The 1986 Chernobyl reactor accident released about 90,000 TBq (27 kilograms) of cesium-137 to the atmosphere, representing 40 percent of the cesium-137 in the reactor core.[3] Through 1980, about 740,000 TBq (220 kilograms) of cesium-137 were deposited as fallout in the Northern Hemisphere, as a result of atmospheric testing of nuclear weapons.[4] Note that the fallout from weapons testing was distributed over a larger area than the fallout from the Chernobyl accident, and a larger fraction of it descended on oceans and lightly inhabited areas.

3. Contamination of land

A useful indicator of the consequences of a cesium-137 release is the area of contaminated land. Here, contamination is measured by the external (whole-body) radiation dose that people will receive if they live in a contaminated area. When cesium-137 is deposited from an airborne plume, it will adhere to the ground, vegetation and structures. From these locations, it will emit gamma radiation which provides an external radiation dose to an exposed person. Cesium-137 will also enter the food chain and water sources, thereby

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[2] US Department of Energy, Health & Environmental Consequences of the Chernobyl Nuclear Power Plant Accident, DOE/ER-0332, June 1987; A S Krass, Consequences of the Chernobyl Accident, Institute for Resource & Security Studies, Cambridge, MA, December 1991.
[3] Krass, op cit.
[4] US Department of Energy, op cit.

providing an internal radiation dose to a person living in the contaminated area. Absent any countermeasures, the internal dose could be of a similar magnitude to the external dose.

Figure E-1 shows the relationship between contaminated land area and the size of an atmospheric release of cesium-137. This figure is adapted from a 1979 study by Jan Beyea, then of Princeton University.[5] The threshold of contamination is an external dose of 10 rem over 30 years, assuming a shielding factor of 0.25 and accounting for weathering of cesium. The "typical meteorology" case in Figure E-1 assumes a wind speed of 5 m/sec, atmospheric stability in class D, a 0.01 m/sec deposition velocity, a 1,000 m mixing layer and an initial plume rise of 300 m (although the results are not sensitive to plume rise). A Gaussian, straight-line plume model was used, providing an estimate of contaminated land area that will approximate the area contaminated during a range of actual meteorological conditions. The lower and upper limits of land contamination in Figure E-1 represent a range of potential meteorological conditions.

The threshold for land contamination

An external exposure of 10 rem over 30 years would represent about a three-fold increase above the typical level of background radiation (which is about 0.1 rem/year). In its 1975 Reactor Safety Study, the NRC used a threshold of 10 rem over 30 years as an exposure level above which populations were assumed to be relocated from rural areas. The same study used a threshold of 25 rem over 30 years as a criterion for relocating people from urban areas, to reflect the assumed greater expense of relocating urban inhabitants.

In an actual case of land contamination in the United States, the steps taken to relocate populations and pursue other countermeasures (decontamination of surfaces, interdiction of food supplies, etc.) would reflect a variety of political, economic, cultural, legal and scientific influences. It is safe to say that few citizens would calmly accept a level of radiation exposure which substantially exceeds background levels.

Land contamination from potential Harris releases

Three potential Harris releases of cesium-137 are shown in Figure E-1.. Releases of 70 million Curies and 20 million Curies correspond to liberation

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[5] J Beyea, "The Effects of Releases to the Atmosphere of Radioactivity from Hypothetical Large-Scale Accidents at the Proposed Gorleben Waste Treatment Facility", in Chapter 3 of Report of the Gorleben International Review, presented (in German) to the Government of Lower Saxony, March 1979.

of cesium-137 from spent fuel aged up to 9 years or up to 3 years, respectively. A release of 2 million Curies corresponds to the most severe reactor accident identified in the Harris IPE.

For typical meteorology, Figure E-1 indicates that a release of 2 million Curies would contaminate 4,000-5,000 square kilometers of land, A release of 20 million Curies would contaminate 50,000-60,000 square kilometers. Finally, a release of 70 million Curies would contaminate about 150,000 square kilometers of land. Note that the total area of North Carolina is 136,000 square kilometers and the state's land area is 127,000 square kilometers.[6]

Potentially exposed population

According to CP&L's Final Safety Analysis Report (FSAR) for the Harris plant, an estimated 1.8 million people will live within 50 miles of the plant in 2000, while 2.2 million people will live within that radius in 2020.[7] A 50 mile-radius circle encompasses an area of 20,300 square kilometers.

If a substantial release of cesium-137 occurs at Harris, the shape and size of the resulting contaminated area will depend on the size of the release and the meteorological conditions during the period of the release. If the wind direction is constant during the release and the atmosphere remains stable, the contaminated area will be comparatively narrow and extended downwind. Changing wind direction during the release period and a less stable atmosphere will produce a more "smeared out" pattern of contamination.

A computer modelling exercise could be performed, to predict patterns of contamination under different meteorological conditions. This exercise could ascribe a probability, assuming a postulated release, that a particular population falls within an area contaminated above a specified threshold.

4. Health effects of radiation

The health effects of exposure to ionizing radiation can be broadly categorized as early and delayed effects. For our postulated releases of cesium-137, early health effects could be suffered by some people in the immediate vicinity of the plant. However, most of the health effects would be delayed effects, especially cancer, which are manifested years after the initial exposure.

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[6] The World Almanac and Book of Facts 1991, Pharos Books, New York, 1990.
[7] Harris FSAR, Section 2.1.3, Amendment No. 2.

Note that a release during a reactor accident (e.g., release category RC-5 at Harris) will contain short-lived radioisotopes as well as cesium-137. Under certain conditions of meteorology and emergency response, the presence of these short-lived radioisotopes in the release could cause many early health effects. Spent fuel contains comparatively small amounts of short-lived radioisotopes. Thus, early health effects are comparatively unlikely if a release occurs from a spent fuel pool.

Table E-1 shows an estimate of the excess cancer mortality attributable to continuous exposure to a relatively low radiation dose rate. This estimate was made by the BEIR V committee of the National Research Council.[8] In Table E-1, a continuous exposure of 1 mSv/year (0.1 rem/year) is assumed to occur throughout life.[9] Such an exposure is estimated to increase the number of fatal cancers, above the normally expected level, by 2.5 percent for males and 3.4 percent for females, with an average of 16-18 years of life lost per excess death. If the dose-response function were linear, it would follow that continuous, lifetime exposure to 10 mSv/year (1 rem/year) would increase the number of fatal cancers by 25 percent for males and 34 percent for females. The shape of the dose-response function is a subject of ongoing debate.

If people continued to occupy urban areas contaminated with cesium-137 to an external exposure level just below 25 rem over 30 years, as was assumed in the Reactor Safety Study, their average exposure during this 30-year period would be 8 mSv/year (0.8 rem/year). An additional, internal exposure would arise from contamination of food and water. After 30 years, rates of external and internal exposure would decline, consistent with the decay of cesium-137. Note that over a period of 300 years (10 half-lives), the activity of cesium-137 will decay to one-thousandth of its initial level.

5. Economic consequences of a release of radioactivity

Computer models have been developed for estimating the economic consequences of large atmospheric releases of radioactive materials. Findings from such models have been used by the NRC to evaluate the cost-benefit ratio of introducing measures to reduce the probabilities or consequences of spent fuel pool accidents.[10] A review of these models, findings and cost-

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[8] National Research Council, Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V, National Academy Press, Washington, DC, 1990. Table E-1 is adapted from Table 4-2 of the BEIR V report.

[9] The exposure of 1 mSv/year is additional to background radiation, whose effects are accounted for in the normal expectation of cancer mortality.

[10] See, for example: E D Throm, Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools, NUREG-1353, April 1989; and J H Jo et al,

benefit analyses is beyond the scope of this report. However, a brief examination of the NRC's literature reveals that findings in this area rest on assumptions and value judgements that are not clearly articulated and deserve thorough public review.

Previous sections of this appendix have shown that potential releases of cesium-137 from the Harris spent fuel pools could lead to the relocation of large populations and ongoing radiation exposure to large, unrelocated populations. Relocation implies abandonment of large amounts of land, other natural resources and fixed capital. Political and social effects would be significant, and would have economic implications. Useful analysis of these matters would require a more sophisticated approach than is evident in literature generated by and for the NRC.

Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools, NUREG/CR-5281, March 1989.

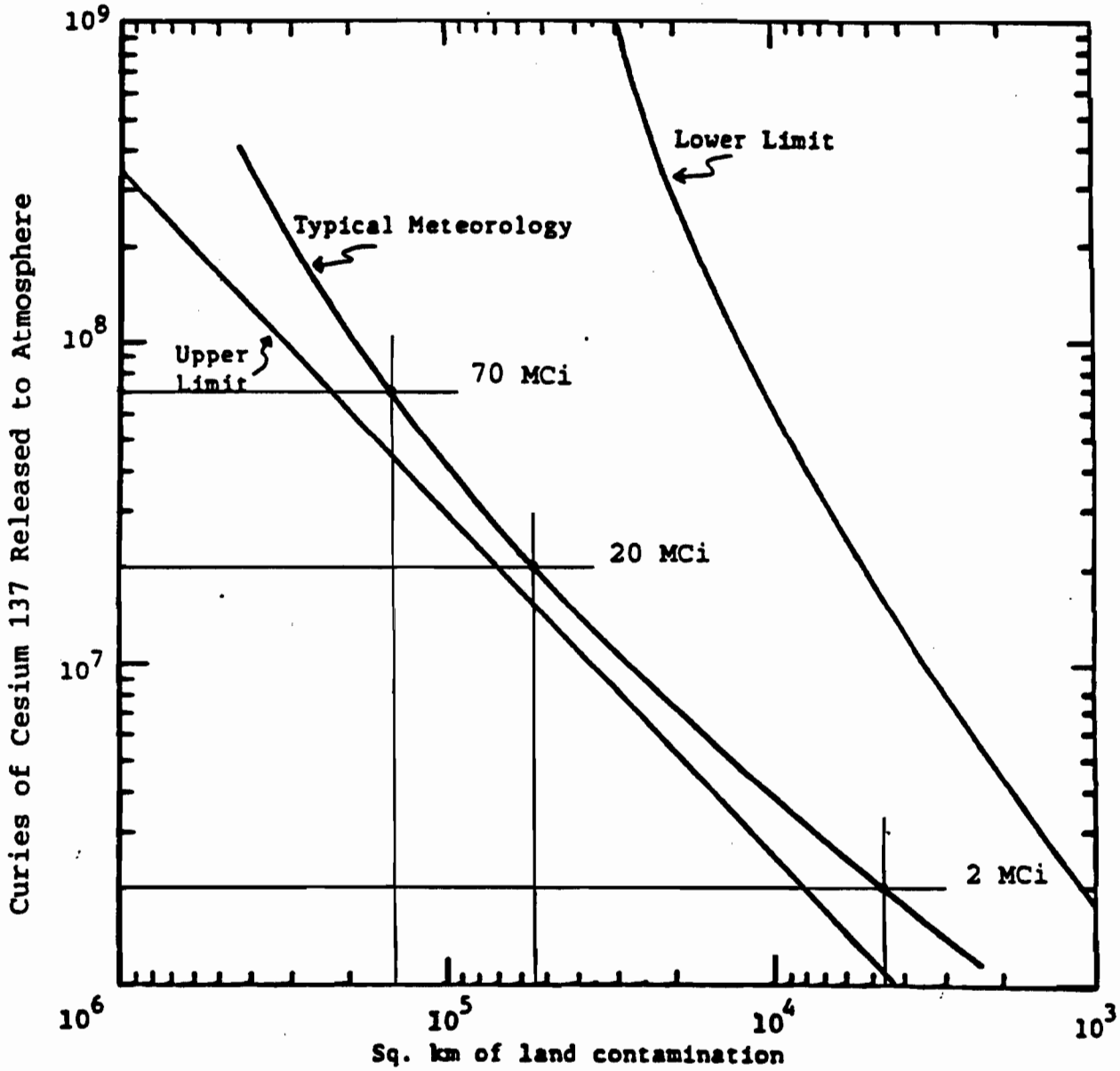

Figure E-1

Contaminated land area as a function of cesium-137 release

ESTIMATED LIFETIME RISK PER 100,000 PERSONS EXPOSED TO 1 mSv 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

Table E-1

Excess cancer mortality from continuous exposure to radiation: BEIR V estimate

1. Improving Decommission Regulations for Nuclear Power Plants

SECY-99-168 describes and approach for the consolidation of a number of ongoing rulemakings related to decommissioning into an integrated risk-informed rule. The SECY also describres a proposal for a comprehensive regulatory review of Title10 to be preformed to determine what regulations are applicable to decommissioning nuclear power plants and to identify where clarifications or modifications are appropriate, based on risk significant differences between operating and decommissioning plants. Decommissioning regulations would be consolidated into a new location in Title 10.

The an risk informed integrated rulemaking will address the following issues.

- Emergency Planning
- Financial indemnity
- Safeguards/Physical Security
- Operator staff and required training
- Backfit rule applicably

These issues were currently being addressed in separate rulemakings actions and consolidating these actions into a single rulemaking will facilitate a consistent approach. As stated, the NRC is to use a risk-informed approach in this integrated rulemaking. The staff is considering including fitness for duty requirements in this integrated rule making.

Milestones:

- Rulemaking Plan on integrated rulemaking issues 6/30/00
- Rulemaking Plan for consolidation of decommissioning regulations 7/15/00

Proposed Lead Committee

- Areas to be addressed in the integrated rulemaking , with the exception of financial indemnity, are in the ACRS area of expertise and traditional responsibility. It is proposed that neither Committee undertake a review of financial indemnity issues. It is proposed that the Joint ACRS/ACNW Subcommittee take the responsibility for the review of the rulemaking plan for consolidated risk-informed rule.

Proposed Action

- ACRS review of rulemaking plan for the integrated rule and subsequent review of all areas to be addressed in the proposed rule with the exception of financial indemnity. Lead ACRS members would review the staff proposals and make recommendations as to what parts of the staff proposals needed to be addressed by the ACRS.
- Joint ACRS/ACNW Subcommittee review of rulemaking plan for consolidated risk-informed rule with subsequent review of the proposed rule. The Joint Subcommittee would refer responsibility for parts the proposed rule to either the ACRS or the ACNW after review of the rulemaking plan. The possible approaches to consolidating and risk-informing decommissioning regulations could be discussed during the ACNW workshop on decommissioning.
-

NEI provided the NRC with a May 17,2000 white paper in which NEI outlines a proposal for consolidating both of these rulemakings into a single risk-informed rulemaking, to be completed in about 24 months. NEI's plan is still under discussion by the staff.

2. Spent Fuel Pool Accident Risk Assessment

Accidents associated with spent fuel pool storage are being examined as a significant source of risk for permanently shutdown nuclear power plants. Loss of spent fuel pool

water with uncovering of the stored fuel and the occurrence of zirconium fires is being examined.

Milestones:

- Discussion during April 5-7, 2000 ACRS meeting and ACRS report 4/00

- NRC staff finalize spent fuel pool accident risk report 5/30/00
  (The NRC's schedule was extended to allow time to address ACRS concerns. It is expected that the ACRS will be able to discuss this with the staff at the September ACRS meeting.)

Proposed Lead Committee

- Assigned to ACRS in 12/21/99 SRM

Proposed action

- ACRS review and comment on content of the NRC staff report and ACRS discussion as to the status of the classification of design basis accidents for decommissioning power reactors
- ACRS followup on issues identified in its 11/12/99 **and 4/13/00 reports**

3. Technical Specifications for decommissioning nuclear power plants

Regulatory oversight by the NRC is accomplished in part through the use of Technical Specifications. The associated needs change when the plant is in the decommissioning process. Standard Technical Specifications (STP) are being developed for decommissioning plants.

Milestones

- Final STP for PWRs FY01
- Proposed STP for BWRs FY00

- Final STP for BWRs FY01

Proposed lead Committee

- Areas to be addressed are in ACRS areas of expertise and traditional responsibility

Proposed Action

- A small team of ACRS members review the documents when available and identify issues for which ACRS review is needed

4. Evaluation of design basis accidents for decommissioning nuclear power plants

Design bases accidents for decommissioning plants be different from those associated with an operating plant. This activity will involve identification and evaluation of these design bases accidents. The NRC staff paper on spent fuel pool fires will partially address this issue.

Milestones

- To be determined

Proposed Lead Committee

- Areas to be addressed are in ACRS area of expertise and traditional responsibility.

Proposed Action

- Explore the NRC staff's plans for and thinking on this issue within the context of the ACRS review of the NRC staff paper on spent fuel pool accident risk and identify any need for further ACRS (or ACNW) involvement.

5. Regulatory Guides, SRPs, and inspection plans for decommissioning of power reactors.

This item covers the following staff activities

- Final Regulatory Guide DG 1067 on decommissioning of nuclear power reactors **- to be discussed at the June 2000 ACNW meeting**
  To ACRS/ACNW 3/00

- Final Regulatory Guide DG 1071, "Standard Format and Content for Post Shutdown Decommissioning Activities Report." **-to be discussed at the June 2000 ACNW meeting**
- To ACRS/ACNW 3/00

- SRP for License Termination Plans 5/00

- Revisions to IMC 2561 "Decommissioning Inspection Program" TBD

- Guidance on Maintenance Rule compliance for decommissioning plants
  To be completed FY2000

- Final Regulatory Guide on fire protection for decommissioning plants, DG-1069
  To be completed Early FY2001

- Guidance for evaluation of safety reviews (10CFR50.59) at permanently shutdown reactors i FY2000

Milestones

- As noted above

Proposed Lead Committee

- As stated under proposed action

Proposed action

- ACRS lead members review of guidance on maintenance rule compliance, fire protection, and 10CFR50.59 reviews and identification of any areas for which ACRS review is appropriate. ACNW review of Regulatory Guides DG1067 and DG-1078. No Committee review of decommissioning inspection guidance. ACNW has reviewed a draft version of the SRP for License Termination Plans. The final version is expected not to be changed in any significant way. ACNW will receive the final version of the SRP for what level of review it believes appropriate.

6. ACRS and ACNW briefing on NRC and utility experience with power reactor decommissioning

It is proposed that a group of ACRS and ACNW members visit a Region office and the site of a decommissioning reactor and receive briefings from Region offices and utility personnel on the issues and lesson-learned associated with the Region and utility experience with decommissioning. This would provide a opportunity for the attendees to learn more about actual field experience and the issues identified.

Milestone

- Schedule in FY2001

Proposed Lead Committee

- Do as a Joint ACRS/ACNW activity with the appropriate ACRS and ACNW members

Proposed action

- Participating members of brief their committee on issues of interest after this visit.

7. NRR Licensing Oversight for Decommissioning reactor Facilities

NRR is currently provides management and licensing oversight for 16 decommissioning reactor facilities at a level commensurate with the associated risk

Milestones

- Ongoing activity

Proposed Lead Committee

- Joint ACRS/ACNW activity

Proposed Action

- Schedule as information briefing, repeated at about two year intervals, during which NRR would brief a Joint ACRS/ACNW group on the status of the NRR work. Participating members would then provide a report to their Committee on insights and issues of interest to that Committee.

Non-Power Reactors Licenses

1. Clearance Rule

The NRC is developing a rulemaking that would set specific requirements on the releases of solid materials. The ACNW was briefed on this issue during its December 1999 meeting and has issued a report. The final of NUREG 1640 will be issued in FY2001

Milestones

- Issue final NUREG 1640 (may be delayed for one year) 1/01

Proposed Lead Committee

- ACNW has the lead

Proposed Action

- ACNW will continued to follow the staff work on this matter as stated in the ACNW report.

2. Rubblized concrete dismantlement

Maine Yankee has expressed a interest in utilizing rubblization in its decommissioning. The process as proposed involves (a) removing all equipment from buildings, (b) some decontamination of the building surfaces, (c) demolishing the above grade structures into concrete rubble, and (f) covering, regrading, and landscaping the site surface.

Milestones

- License Termination Plan review 11/00

Proposed Lead Committee

- ACNW already has the lead and has written a report (1-24-2000)

Proposed action

- ACNW stated in its report that it would continue to interact with the NRC in the development of this option.

3. Entombment

The SRM on SECY 96-068 that addressed DSI-24 requested a NRC staff analyses as to whether they view entombment as a viable option. The staff stated in SECY 98-099 that consideration of entombment as a viable option has merit. In SECY 99-187 the staff stated that they believe that entombment can be a safe and viable option for many situations. The staff based this conclusion in part on PNNL assessment. The staff has conducted a workshop (12/99) during which they solicited stakeholder views.

Milestones

- Staff paper providing recommendations to the Commission 6/00

Proposed Lead Committee

- Areas to be addressed are in the ACNW area of expertise and traditional responsibility. ACRS members with operating reactor expertise could be involved.

Proposed Action

- Review staff paper and report to the Commission. Stakeholder input should be sought on controversial issues.

4. Decommissioning criteria for West Valley

The NRC staff is developing decommissioning criteria for use by DOE for the West Valley Demonstration Project and for any follow-up licensing activities.

Milestones

• SECY proposing a decommissioning criteria policy statement to Commission for approval	8/30/00
• Issue Policy Statement in FR	11/30/00
• Approve specific criteria for West Valley site	TBD

Proposed Lead Committee

- Areas to be addressed an in the ACNW area of expertise and traditional responsibility.

Proposed Action

- Review the Policy Statement and specific criteria for the West Valley site

5. Site Decommissioning Management Plan

The Site Decommissioning Management Plan (SDMP) was developed and submitted to the Commission on March 29, 1990 (SECY-90-121) There are now 26 SDMP sites (proposed 23 in FY2001, 10 in FY2002, and 9 in FY2003)

Milestone

- DandD pilot to evaluate adequacy of screening criteria TDB
- ACNW visit to a SDMP site TBD

Proposed Lead Committee

- ACNW already has the lead

Proposed Action

- Discuss DandD pilot with the NRC staff and visit a selected SDMP site. Object of the site visit would be for ACNW to have a opportunity to familiarize itself with materials site decommissioning field experience and engage in public outreach.
- Shortly after the December 1999 ACNW meeting Richard Major distributed a package with reviews of 6 decommissioning reviews for materials sites. The ACNW should decide if they need to be briefed by the NRC staff.

6. Standard Review Plan for Decommissioning

The NMSS staff is developing a SRP for decommissioning. The document was provided to the ACNW in August, 1999. Assignments were subsequently made to members.

Milestones

- Issue dose modeling SRP 7/00

- Issue SRP 7/00

Proposed Lead Committee

- ACNW already has the lead

Proposed Action

- Review status of members work during the March 2000 ACNW meeting and decide on course of action

7. Pilot for performing decommissioning of a materials site without the submittal of a decommissioning plan

This activity implements the Commission's direction under DSI-9 to initiate a pilot study for decommissioning without the submittal of a decommissioning plan and providing a regulatory framework for encouraging lower cost decommissioning waste disposal options

Milestones

- Status report to the Commission 1/01

Proposed Lead Committee

- ACNW has the lead

Proposed Action

- ACNW should stay informed and make a decision as to if it should review this topic in early FY2001

8. NRC interactions with EPA and ISCORS to resolve issues of mutual concern

Topics addressed in these ongoing interactions include risk harmonization unnecessary duplication of regulatory requirements, mixed waste, recycling, decommission, cleanup, and sewer reconcentration.

Milestones

- Ongoing activity

Proposed Lead Committee

- Areas to be addressed are in the ACNW area of expertise and traditional responsibility

Proposed action

- ACNW should stay generally informed and involve itself only if the Commission requests its involvement or if a related issue arises within the context of ACNW review of some other topic. "Risk Harmonization" is a Second Ten Priority item on the ACNW's CY2000 Action Plan

9. RES work related to decommissioning issues

The work involves code and model development and some data acquisition. (See attachment)

Milestones

- Provide PC version of SEDSS that will implement DandD screening methodology and 1-D flow and transport groundwater pathway 5/00

- Update MARSSIM to incorporate public comments following 2-year testing period 7/00

- Verify and validate testing of 4 SIGHT 10/00

- Develop a probabilistic version of RESRAD and publish NUREG/CR 11/00

- Develop probabilistic version and DandD and publish NUREG/CR 11/00

- Provide draft technical report on test applications of methodology for selecting and testing conceptual models with respect to a specific site 2/01

- Provide PC version of SEDSS with multi-dimensional groundwater pathways 3/01

Proposed Lead Committee

- ACNW already has lead
-

Proposed Action

- ACNW should continue to say informed as to the progress of the staff's work and continue to review this work in the context of its annual RES-sponsored research review.

Activities for which ACRS ACNW review is not recommended - documents will be given to lead committee member for information

1. Decommissioning Project Manager's Handbook

2. NUREG-1628, "Decommissioning Questions and Answers."

3. Revisions to IMG 2561, "Decommissioning Inspection Program" and other decommissioning inspection procedures.

4. Resident Inspector Training and guidance for decommissioning

5. Guidance related to evaluating decommissioning cost and establishing financial indemnity.

6. Guidance on FSAR conversion often permanently ceasing power operation.

(14)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
473rd	15	15-2

Title

FUTURE ACRS ACTIVITIES

Authors

John T. Larkins

List of Documents Attached

Future ACRS Activities - 473rd ACRS Meeting, June 7-9, 2000

15

Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person

John T. Larkins

ANTICIPATED WORKLOAD
JUNE 7-9, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Dudley/Sorensen	Safety Culture-presentation by Sorensen	Report to be completed at the July meeting	--	P&P 6/6
		Markley	Assessment of the Quality of PRAs	--		
Barton		Singh	Davis Besse Plant visit - **Briefing by Mr. Singh on Arrangements**	--	--	--
		Dudley	Use of Industry Initiatives in the Regulatory Process	Report		--
Bonaca	--	Dudley/Singh	Regulatory Effectiveness of the Station Blackout Rule	Report **(Tentative)**	--	P&P 6/6
	Seale	Dudley	License Renewal Documents (SRP, GALL II, Reg. Guide)-**Develop Review Plan.**	--	--	
Kress	--	El-Zeftawy	Proposed Resolution of GSI-173A, Spent Fuel Storage Pool for Operating Facilities	Report	--	--
	--	Boehnert/Weston	Proposed Final Reg. Guide and SRP on Revised Source Term	Report		
Sieber	--	Markley	Performance-Based Regulatory Initiatives	Report **(Tentative)**	--	--

ANTICIPATED WORKLOAD
July 12-14, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley	Proposed Final ASME Standard (Phase 1) for PRA Quality (Dr. Apostolakis to develop assignments for members. Drs. Apostolakis and Bonaca to attend the ASME workshop on June 27)	Report	RPRA 6/28-29	P&P 7/11 PO 6/13-14 (Davis-Besse/ Region III)
	--	Dudley/ Sorensen	**Safety Culture (Presentation completed at the June meeting).**	Report	--	
Kress	Apostolakis	Dudley	NEI Letter on Risk-Informing 10 CFR Part 50 **(Commission Request)**	Report	--	PO 6/13-14 (Davis-Besse/ Region III) RPRA 6/28-29
Powers	--	Larkins/Duraiswamy	Topics for meeting with the Commissioners on October 5, 2000.	--	P&P 7/11	PO 6/13-14 (Davis-Besse/ Region III) RPRA 6/28-29
		El-Zeftawy	Format and content of the Research Report. Dr. Powers to develop a proposal.			
Seale	--	El-Zeftawy	Causes and Significance of Design-Basis Issues[1]	--	--	PO 6/13-14 (Davis-Besse/ Region III) RPRA 6/28-29

[1]The P&P Subcommittee recommends that Dr. Seale propose a course of action.

ANTICIPATED WORKLOAD
July 12-14, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Shack	Apostolakis	Markley	Proposed revision to 10 CFR 50.44 (Option 3) concerning combustible gas control system/advance notice of proposed rulemaking (10 CFR 50.69 and Appendix T)	Report	--	PO 6/13-14 (Davis-Besse/ Region III) RPRA 6/28-29
Wallis	--	Boehnert	Draft SER on RETRAN 3D[2]	Report **(Tentative)**	--	PO 6/13-14 (Davis-Besse/ Region III) RPRA 6/28-29

[2]The P&P Subcommittee recommends that Dr. Wallis propose a course of action after receiving the SER.

ANTICIPATED WORKLOAD
AUGUST 30-SEPTEMBER 1, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Apostolakis	--	Markley	Proposed ANS Standard for PRA Quality (Phase 2) - Dr. Apostolakis to develop assignments for members to be provided at the July meeting.	Report	RPRA 8/29 (A.M.)	FP 8/28
		Markley	Risk-Informing 10 CFR Part 50 (Public comments on Option 2 ANPR changes to Option 3 (10 CFR 50.46 and **(Tentative)**	Report **(Tentative)**		P&P 8/29 (A.M.)
Bonaca	--	Dudley	SERs on BWRVIP Topical Reports	--	--	Charleston 8/7 (Tentative)
	Seale	Dudley	Overview of License Renewal Guidance Documents - **Information Briefing**	--		RPRA 8/29 (P.M.) P&P 8/29 (A.M.)
Kress	--	El-Zeftawy	Spent Fuel Pool Accident Risk at Decommissioning Plants	Report	-- --	Charleston 8/7 THP 8/8 RPRA 8/29 (P.M.)
Seale	--	Weston	Proposed Final Regulatory Guide on Design-Bases Information	Report	--	Charleston 8/7 THP 8/8
		Weston	Operating Events at Indian Point Unit 2	--	--	RPRA 8/29 (P.M.)
Shack	Bonaca	El-Zeftawy	Proposed Final Regulatory Guide on 10 CFR 50.59	--	--	Charleston 8/7

NOTE: Schedule for RPRA, FP, and P&P meetings to be decided.

ANTICIPATED WORKLOAD
AUGUST 30-SEPTEMBER 1, 2000

LEAD MEMBER	BACKUP	ENGINEER	ISSUE	FULL COMM. REPORT	SUBC. MTG.	
					CHAIR.	MEMBER
Sieber	Powers	Singh	Fire Protection Issues (Regulatory Guide, NFPA 805 Standard, etc.)	Report	FP 8/28	Charleston 8/7 RPRA 8/29 (P.M.)
--	--	Boehnert	Proposed final modifications to SRP Chapter 19 and RG 1.174 regarding use of risk-informed decisionmaking in license amendment reviews	Report	—	--
Powers	All Members	Larkins, et.al	Preparation for meeting with the Commissioners	--	P&P 8/29 (A.M.)	Charleston 8/7 THP 8/8 RPRA 8/29 (P.M.)
Uhrig	--	Singh	ABB/CE and Siemens Digital I&C Applications	--	--	Charleston 8/7
Wallis	--	Boehnert Boehnert	Subcommittee report on SRELAP-5 Code AP-1000 Design Review	-- Report	THP 8/8	Charleston 8/7 RPRA 8/29 (P.M.)

II. ITEMS REQUIRING COMMITTEE ACTION

4. SERs on BWR Vessel and Internal Project (BWRVIP) Topical Reports (MVB/WJS/NFD) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review Requested by the ACRS. [Gene Carpenter, NRR] The Committee reviewed and commented on BWRVIP-05, "Boiling Water Reactor Pressure Vessel Shell Weld Inspection Recommendations," in a letter dated September 10, 1997. Since that review, the staff has written safety evaluation reports on dozens of topical reports related to aging of BWR reactor vessel internal components. Many of the topical reports will be used to support license renewal applications.

The staff plans to provide the ACRS with a report concerning the status of the BWRVIP submittals by August 3, 2000, and brief the Committee at the September 2000 ACRS meeting.

Drs. Bonaca and Shack will recommend a course of action after reviewing the SERs.

5. Causes and Significance of Design-Basis Issues (Open) (RLS/MME) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

ACRS initiative. [Stuart Rubin, RES] The NRC staff issued a draft report, "Causes and Significance of Design-Basis Issues at U.S. Nuclear Power Plants," for peer review on May 8, 2000. The report documents the results of a study of design-basis issue trends and patterns. The results of the study revealed that approximately 77% of the 3100 design-basis issues reviewed had either minimal or no risk significance. The insights from this study are intended to assist NRC's and Industry's ongoing efforts to make NRC's regulatory framework and oversight process more risk-informed and performance-based and to reduce unnecessary regulatory burden. The report of the study is available on ADAMS.

Dr. Seale has agreed to recommend a course of action after reviewing the document.

6. Extended Shutdown of Millstone Units 1, 2, and 3 and D.C. Cook Units 1 and 2 (Open) (JJB/MWW) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the ACRS. In June 1996, Millstone was designated as a Category 3 facility (shutdown requiring Commission approval to operate).

The licensee underwent significant management and organizational changes. In November 1996, the NRC established a Special Projects Office (SPO) and an associated Restart Action Plan. SECY-97-003, "Millstone Restart Preview Process," was forwarded to the Committee on January 13, 1997. Millstone Unit 3 restarted on July 30, 1998. Unit 2 resumed full power operation in May 1999. The licensee decided not to restart Unit 1 and has begun decommissioning activities.

In September 1997, the licensee shut down D.C. Cook Units 1 and 2 to address design and operational issues associated with containment sump recirculation cooling during certain accident scenarios. The NRC issued a Confirmatory Action Letter (CAL) specifying issues requiring resolution and actions to be completed prior to restart. In general, the technical issues were considered to be plant-specific to the D.C Cook plants which have a somewhat unique ice condenser containment design. An NRC restart readiness assessment team inspection began on May 8, 2000. The licensee plans to restart both units in mid-2000.

In the past, the ACRS normally heard briefings from licensees whose plants have been shut down for extended periods of time. However, the Committee decided not to hear a briefing from Crystal River 3 and decided to hear briefings on other plants on a case-by-case basis.

The Planning and Procedures Subcommittee and Mr. Barton recommend that the Committee not hear briefings on these matters.

7. Duane Arnold Core Power Uprate (GBW/PAB) ESTIMATED TIME: 1½ hours

Purpose: Decide on a Course of Action

ACRS Initiative. Alliant Energy (AE - Duane Arnold Licensee) is planning to submit a license amendment request to increase core thermal power by 15.3% above the current licensed value (1658MWt to 1912 MWt). With this uprate, core power will be increased by 20% over the initially licensed value of 1593 Mwt. During a May 16 meeting with NRR (see May 18, 2000 P. Boehnert meeting summary), AE stated its intent to submit the uprate request in October 2000, and has requested completion of the staff's review by the end of May 2001 to support power operations. While AE's schedule appears optimistic, the Committee will need to factor this proposed schedule into its review planning. During the May 2000 meeting, the Committee agreed to pursue the issue of the need for development of formal staff guidance for uprate reviews (e.g. Standard Review Plan Section) in tandem with its next power uprate review. Dr. Powers, via a May 23, 2000 E-Mail, has also raised a number of issues he believes need to be explored by the Committee during its review of this matter. (**Note**: NRR and Commonwealth Edison met on May 31, 2000 to discuss proposed power uprates for the Dresden and Quad Cities plants. These uprates are expected to be in the 15%- "plus" range as well.)

The Planning and Procedures Subcommittee recommends that subsequent to receiving the staff's SERs, the Thermal Hydraulic Phenomena Subcommittee hold a meeting to discuss these matters prior to referring them to the full Committee for review.

8. Diablo Canyon Nuclear Power Plant Unusual Event (Open) (JJB/MWW)
ESTIMATED TIME: 1 hour

Purpose: Decide on a Course of Action

Diablo Canyon, Unit 1, a 4 loop Westinghouse plant located near San Luis Obispo, California, declared a "Notice of Unusual Event" following an automatic reactor shutdown at 12:43 a.m. PDT, May 15, 2000. Events at the plant began at about 12:25 a.m. when a fault occurred in electrical cabling inside the Unit 1 turbine generator building. This led to a fire which resulted in a protective shutdown of the main generator followed by the unit 1 reactor trip. Because the reactor tripped from full power, steam was released to the atmosphere. The steam released could have contained minute amounts of radioactivity, but these would be well below the federal release limits. No impact to public health and safety was anticipated. Offsite electric power was restored to safety-related equipment and the "Notice of Unusual Event" was terminated at 9:59 a.m. PDT on Tuesday, May 18, 2000. There were no injuries, all reactor systems functioned properly during the shutdown, and there were no significant complications.

The NRC sent a Special Inspection Team (SIT) to gather information on the electrical fire and automatic reactor trip. The team began work on May 17, 2000 and spent about one week onsite. The team report will be issued about 30 days after completion of the inspection. The SIT will examine the sequence of events, the licensee's root cause determination and corrective actions, the effectiveness of the fire brigade notification and response, the timeliness of the emergency classification and notifications, and whether there are lessons to be learned that could benefit other nuclear plants.

The SIT has exited the site and will conduct a public exit meeting within the next several weeks. They had a technical debriefing with the utility on Friday May 26, 2000.

The Planning and Procedures Subcommittee recommends that the Fire Protection Subcommittee discuss this event during a future meeting and that Mr. Barton provide his reviews on the Planning and Procedures Subcommittee recommendation.

9. RETRAN-3D Transient Analysis Code (Open) (GBW/PAB) ESTIMATED TIME: 1 ½ hours

Purpose: Decide on a Course of Action

Review requested by the NRC staff. As part of its Thermal-Hydraulic (T/H) Code Review Action Plan, NRR initiated review of the EPRI RETRAN-3D thermal-hydraulic transient analysis code. The code is designed for analysis of FSAR Chapter 15 transients (excluding Appendix K LOCA analysis), and plant events. The T/H Phenomena Subcommittee began its review of this code during its December 16-17, 1998 meeting.

NRR had developed a detailed schedule for reviewing the RETRAN-3D code. In accordance with this review schedule, the T/H Phenomena Subcommittee met on March 23, 1999. A Subcommittee report was provided to the Committee during its April 1999 meeting.

Dr. Wallis conducted a detailed review of portions of the RETRAN code documentation. He has identified several issues of a significant nature with the models and correlations used in the "3D" version of the code. NRR has also identified a number of significant issues regarding the code modeling. In addition, EPRI was required to modify its "five equation" flow model to correct known errors. A meeting was held on June 29, 1999 between NRR and EPRI to address these matters. The outcome of the meeting gave indication that a significant amount of work remains before completing the review of this code.

Dr. Wallis provided a report to the Committee during the July 1999 ACRS meeting regarding his concerns. The Committee considered a draft letter to the EDO on this matter, but the letter was tabled. During the September meeting, the Committee discussed the direction to be taken by the ACRS regarding future review of the RETRAN-3D code. It was agreed that the Committee would defer further action on this matter, pending receipt of the staff's review document.

The staff plans to develop a review document subsequent to receiving the EPRI response to a set of RAIs that include questions from Dr. Wallis and the T/H Phenomena Subcommittee (the EPRI response was received in mid-March). Representatives of NRR and EPRI discussed the status of the RETRAN review during the March 15, 2000 T/H Phenomena Subcommittee meeting. Dr. Wallis reported the results of the subcommittee meeting to the ACRS during its April meeting. He said that the subcommittee plans no future action on this matter, subject to further action from the NRR staff. NRR plans to provide the ACRS with a copy of the draft SER in June 2000 and requests to brief the Committee during the July meeting.

The Planning and Procedures Subcommittee recommends that Dr. Wallis propose a course of action after receiving the SER.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 2, 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 DURING JULY - OCTOBER 2000

Attached is a list of proposed agenda items for the ACRS and the ACNW for July - October 2000. 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 May 30, 2000 with the OEDO, NRR, NMSS, and RES ACRS/ACNW coordinators [OEDO, G.C. Millman; NRR, M.G. Crutchley; NMSS, R.H. Turtil; RES, J.A. Mitchell].

A copy of the Work Items Tracking System (WITS) list for July - October 2000 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

AGENDA FOR
ACRS MEETINGS
(July 2000 - October 2000)

ACRS MEETING ---- JULY 12 - 14, 2000				
Item #	Title/Issue	Purpose	Priority	Documents
1	The South Texas Project Graded QA Program Exemption Contact: R. Gramm, DLPM/NRR	Review and Comment	High	Pertinent documents to be provided by 6/9/00.
2	Proposed Update to 10 CFR Part 52 Contact: J. Wilson, DRIP/NRR	Review and Comment	Medium	Proposed rule to be provided by 6/12/00.
3	Part 50.44, Combustible Gas Control (Option 3) Contact: M. Drouin, DRAA/RES	Review and Comment	High	Draft Commission paper to be provided by 6/26/00.
4	Risk Informed Part 50 (Option 2) Contact: T. Bergman, DRIP/NRR	Information Briefing	Medium	Complete package of public comment letters and draft Commission paper will be provided by 6/6/00.
5	RETRAN-3D Transient Analysis Code Contact: R. Caruso, DSSA/NRR	Review and Comment	Medium	SER to be provided by 6/30/00.

ACRS MEETING ---- August 30 - September 1, 2000				
Item #	Title/Issue	Purpose	Priority	Documents
1	Performance-Based, Risk-Informed Fire Protection Standard for LWRs and Related Issues Contact: E. Weiss, DSSA/NRR	Review and Comment	Medium	RG and other relevant documents will be provided by 8/2/00. NFPA 805 will not be revised before meeting.
2	Proposed Final Guidance on Use of Risk Information in License Amendment Reviews Contact: R. Palla, DSSA/NRR	Review and Comment	High	Proposed final SRP and other relevant documents to be provided by 7/19/00.
3	Risk-Informed Part 50 (Option 2) Contact: T. Bergman, DRIP/NRR	Review and Comment	High	Draft SECY paper to be provided by 8/15/00.
4	BWR Vessel and Internal Project Contact: G. Carpenter, DE/NRR	Information Briefing	Medium	Report to be provided by 8/3/00.
5	Operating Events at Indian Point 2 Contact: L. Marsh, DRIP/NRR	Information Briefing	Medium	Pertinent documents will be provided by 8/2/00.
6	ABB-CE and Siemens Digital I&C Applications Contact: E. Marinos, DE/NRR	Information Briefing	High	SE will be provided by the first week of July 2000.
7	Final Regulatory Guide on 50.59 Contact: E. McKenna, DRIP/NRR	Review and Comment	High	Draft SECY paper and proposed final Reg Guide to be provided by 8/14/00.

ACRS MEETING ---- August 30 - September 1, 2000				
Item #	Title/Issue	Purpose	Priority	Documents
8	Spent Fuel Accident Risk at Decommissioning NPPs Contact: G. Hubbard, DSSA/NRR	Information Briefing	High	Draft report to be provided by 8/15/00.
9	Regulatory Guide on Design Basis Information Contact: S. Magruder, DRIP/NRR	Review and Comment	High	Draft final Reg Guide and NEI guidance document to be provided by 8/1/00.

ACRS MEETING ---- October 5 - 7, 2000				
Item #	Title/Issue	Purpose	Priority	Documents
1	DG-1053; Dosimetry and NeutronTransport Contact: E. Hackett, DET/RES	Review and Comment	High	To be provided by 11/2/00.
2	NEI 97-06, Steam Generator Program Guidelines Contact: J. Andersen, DE/NRR	Review and Comment	Medium	Draft SER to be provided by 9/8/00.
3	Pressurized Thermal Shock Technical Basis Reevaluation Project Contact: S. Malik, DET/RES	Review and Comment	High	To be provided by 9/15/00.
4	Control Room Habitability Contact: J. Hayes, DSSA/NRR	Information Briefing	High	Presentation material by 9/5/00.

ACRS MEETING — October 5 - 7, 2000				
Item #	Title/Issue	Purpose	Priority	Documents
5	Risk-Based Performance Indicators	Review and Comment	High	Phase 1 progress report to be provided by 7/31/00
	Contact: P. Baranowsky, DRAA/RES			

ACTIVITIES OF INDIVIDUAL MEMBERS

MEMBER(s)	MEETING	DATE/LOCATION
Kress	Risk Communications Training/ACNW Full Committee Mtg.	10/10-12/99 Las Vegas, NV
Powers, Seale, Uhrig	27th Water Reactor Safety Mtg.	10/25-27/99 Headquarters
ALL	ACRS Retreat	1/24-25/00 Clearwater, FL
Apostolakis, Bonaca, Kress, Powers, Seale, Sieber, Shack, Uhrig	Plant License Renewal Subcte. Mtg.	2/23-24/00 Clemson, SC
Kress	4th PHEBUS Fire Protection Technical Seminar	3/18-23/00 Marseille, France
Uhrig	ESKOM & PBMR	3/23-24/00 South Africa
Apostolakis, Powers	Regulatory InformationConference	3/27-28/00 Washington, DC
ALL	Naval Reactors	4/4/00 Crystal River, VA
Powers	Materials Research Society Mtg.	4/23-26/00 San Francisco, VA
Apostolakis, Kress	Joint ACRS/ACNW Working Group	5//11/00 Headquarters

Letters

Correcting the PRA picture

Having chaired two of the independent studies that dealt with the impact of probabilistic risk assessment (PRA) on the nuclear business in the middle of the 1970s, each of which could fairly be called critical (in the best sense of the word) of the Reactor Safety Study (RSS), I feel an obligation to correct George Apostolakis's picture of that early history, as reported in the March 2000 issue of *Nuclear News* (p. 27). (The American Physical Society study was published in 1975, more or less contemporaneously with the completion of the draft Rasmussen report, the RSS—too soon for a complete review. In any case, that wasn't its mission. The Risk Assessment Review Group study, which did have that mission, was published in 1978. TMI, of course, was later, in 1979.)

George put all the critics into one heap for his picture, blaming them for the long sleep at the Nuclear Regulatory Commission. Specifically, he claimed that the NRC "decided not to use the Reactor Safety Study in its work" as a result of the criticism. That just isn't so.

He further said that the critics attacked the RSS by focusing on the Executive Summary, by playing on the difference between conjectured nuclear accidents and observed airplane accidents, and that this persuaded the NRC to jump ship. That, too, falls short of historical accuracy.

First, let's clear the record about what was really said. The RARG report said in 1978, "In summary we find that the fault-tree/event-tree methodology is sound, and both can and should be more widely used by NRC." I believed that at the time, often testified and spoke to that effect, and believe it to this day. I feel something much stronger than regret that it has not yet happened. (See my Tommy Thompson Award acceptance speech—*NN*, Aug. 1998, p. 125—for the flavor.)

On the matter of the Executive Summary, the RARG said, "[W]e find that the Executive Summary is a poor description of the contents of the report, should not be portrayed as such, and has lent itself to misuse in the discussion of reactor risks." If that had led the NRC to decide not to use the RSS in its work, it would suggest functional illiteracy on their part. In fact, the NRC never made any such decision. What they did decide, consistent with the advice quoted above, was to no longer cite the Executive Summary as a condensed version of the report, or to use the bottom-line probabilities in their statements. The NRC, as a Commission, did not reject PRA, and never has.

Both of these studies were indeed critical of the absolute values of the computed major accident probabilities, and rightly so. There were many imperfections, and it is intrinsically a difficult problem. The APS report said that "based on our experience with problems of this nature involving very low probabilities, we do not now have confidence in the presently calculated absolute values of the probabilities of the various branches." The RARG report was a bit kinder, saying that the numbers "may have been used prematurely as an estimate of the absolute risk of reactor accidents without full realization of the wide band of uncertainties involved." But neither group criticized the comparison between theoretical estimates of nuclear accident probability and statistical knowledge of other accident probabilities.

Finally, then, what did in fact go wrong—so wrong that the NRC is still diligently "studying" how to incorporate PRA into regulation 25 years later?

It was not the Commission that panicked, it was the NRC staff that disavowed PRA in droves. They were set in their deterministic ways, offended by criticism, and dominated by a siege mentality, so they made the default decision to turn their collective back on the unfamiliar methodology. It is hard to make a good recovery from those ailments. Even under the charitable view that the recovery is under way, the convalescence has stretched out far too long.

So the real history is that the advice given to the NRC from the beginning by real critics (not professional antinuclear opportunists) has been to extend and develop the use of PRA in regulation, and to encourage it in the industry. As far as I know, the Commission has never even hinted otherwise, despite the contrary assertion in the interview.

It is also more straightforward to "do" a PRA than to use the output wisely—there are deep conceptual problems. The skills needed exceed those provided in a beginning course in engineering statistics, a fact that remains to be appreciated in many quarters.

Hal Lewis
Santa Barbara, Calif.

LETTERS TO THE EDITOR on any aspect of the contents of *Nuclear News*—or on related nuclear industry issues—are welcome. Letters (which should not exceed 700 words) should be addressed to:

Nuclear News Letters
555 N. Kensington Ave.
La Grange Park, Ill. 60526
Facsimile No.: 708/352-6464
e-mail: gtaylor@ans.org (do *not* send with an attached file)

Letter should include the writer's full name, address, and daytime phone and facsimile numbers.

Reply

It appears that Professor Lewis is upset because I "put the critics into one heap." I certainly did not intend to do this. I just did not think that it was necessary to go into details as to who said what, since these are details of historical interest only. The fact is that the controversy surrounding the Reactor Safety Study, especially its Executive Summary, led the NRC to distance itself from it. In an interview intended for the broad readership of *Nuclear News*, this is sufficient.

The Commission issued a statement dated January 18, 1979, in which it stated that "... the Commission has reexamined its views regarding the Study in light of the Review Group's critique." This reexamination referred to an earlier NRC press release that described the Study as a "realistic assessment ..., providing an objective and meaningful estimate of the present risks associated with the operation of present day light water reactors in the United States." Also in the 1/18/79 statement, the Commission stated: "Executive Summary: The Commission withdraws any explicit or implicit past endorsement of the Executive Summary" and "... the Commission does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident."

Although it is true that the Commission also stated that it "... supports the extended use of probabilistic risk assessment in regulatory decisionmaking," the preceding statements sent a clear message to the staff regarding the use of the study in regulatory matters. In fact, the Commission felt that it was necessary to issue a new policy statement in 1995 to state explicitly that the use of PRA in all regulatory

Candidate Topics for the ACRS Meeting with the Commission October, 2000

1. NEI Letter dated January 19, 2000 ||||
Springboard to discuss Option 3
2. Status report on ACRS activities in License Renewal REQUIRED

3. Spent fuel pool safety ~~||||~~
Cite that current PRAs don't treat
4. PRA Standards

- ASME |||
- ANS - external events
- ANS - low power/shutdown
- NFPA-805
- Staff assessment of PRA Quality Requirements

5. Best Estimate Thermal Hydraulic Codes ~~||||~~ ||
Tie to power uprates; 1.174; "loss of margin"
6. Regulatory Effectiveness of the SBO rule
7. SAFETY CULTURE 1

Note that we will have <u>not</u> sent anything to the Commission on this topic prior to the meeting.