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

April 4, 2003

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 500th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MARCH 6-8, 2003 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 500th meeting, March 6-8, 2003, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and a letter. In addition, the Committee authorized Dr. John T. Larkins, Executive Director, ACRS, to transmit the memoranda noted below:

REPORTS:

The following reports were issued to Richard A. Meserve, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft Review Standard, RS-002: "Processing Applications for Early Site Permits," dated March 12, 2003
- Reactor Oversight Process, dated March 13, 2003
- Report on the Safety Aspects of the License Renewal Application for the Peach Bottom Atomic Power Station Units 2 and 3, dated March 14, 2003

LETTER:

The following letter was issued to William D. Travers, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft Final Regulatory Guide DG-1119, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," dated March 13, 2003

MEMORANDA:

The following memorandum was issued to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Draft Regulatory Guide DG-1079, "Criteria for Power Systems for Nuclear Power Plants" (Proposed Revision 3 to Regulatory Guide 1.32), dated March 11, 2003

The following memorandum was issued to Mr. Robert H. Leyse, a member of the public, from John T. Larkins, Executive Director, ACRS:

- Petition for Rulemaking (to address the Impact of Fouling on the Performance of Heat Transfer Surfaces Throughout the Licensed Nuclear Power Plants) and Related Issues, dated March 14, 2003

HIGHLIGHTS OF KEY ISSUES

1. Peach Bottom License Renewal Application

The Comrnittee heard presentations by, and held discussions with, representatives of the NRC staff and Exelon Generation Company, LLC (Exelon) regarding the staff's final Safety Evaluation Report (SER) for the Peach Bottom Atomic Power Station Units 2 and 3 license renewal application. The staff presented the resolution of open items identified in the draft SER. The staff and the applicant presented information regarding the reactor scram event that occurred at Peach Bottom Unit 2 on December 21, 2002.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated March 14, 2003 in which it recommended that the application be approved. The Committee also noted that the scram at Peach Bottom Unit 2 that occurred on December 21, 2002, highlighted a number of weaknesses in the current corrective action and preventive maintenance programs. The Committee expects that ongoing corrective actions committed to by the licensee will resolve these weaknesses. The Committee also stated that during inspections, the NRC staff should assess the effectiveness as well as the adequacy of implementation of these programs.

2. Reactor Oversight Process

The Committee heard presentations by and held discussions with representatives of the NRC staff on the Reactor Oversight Process (ROP) that focused on a summary of the Subcommittee meeting held on January 21, 2003 to address the issues raised in a Staff Requirements Memorandum (M011205B) dated December 20. 2001.

The staff discussed the differences between the risk-informed and performance-based elements of the ROP, consistency and transparency of the thresholds related to performance indicators (PIs) and the significance determination process (SDP), the yellow-red performance indicator thresholds, and cross-cutting issues.

The staff believes the risk-informed and performance-based aspects of the ROP are addressed. The Committee does not agree. The ROP basis document addresses the consistency and transparency of the PIs and the SDP. The current yellow-red thresholds will be maintained and adjustments made as ongoing changes are made to the PI program. Cross-cutting issues are being developed.

Committee Action

The Committee issued a report to the NRC Chairman dated March 13, 2003, indicating that there are still significant disagreements between the staff and the Committee regarding the ROP.

3. Vessel Head Penetration Cracking and Degradation

The Committee heard presentations by and held discussions with representatives of the Electric Power Research Institute Materials Reliability Program (EPRI/MRP) regarding vessel head penetration cracking and degradation issues.

The MRP provided data on the results of plant inspections conducted to date and the inspection plans for the spring 2003 outages.

The total reactor vessel head (RVH) nozzle population includes 3871 control rod drive mechanism (CRDM) nozzles, 1090 control element drive mechanism (CEDM) nozzles, and 94 in-core instrument (ICI) nozzles, at 69 PWR units. Bare-metal visual and/or non-visual non-destructive examination inspections have been performed on about 81% of the RVH nozzles.

Following the spring outages, 97% of the total PWR nozzle population will have been inspected. After the fall 2003 outages, it is expected that all 69 units will have been inspected or the heads replaced.

Committee Action

This was an information briefing and no Committee action was taken. The Committee plans to continue its discussion of this matter during future meetings.

4. Draft Final Revision 1 to Regulatory Guide 1.180 (DG-1119) "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems"

The Committee heard presentations by and held discussions with representatives of the NRC's Office of Nuclear Regulatory Research (RES) concerning draft final Regulatory Guide (DG-1119), "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems." Draft Final Revision 1 to Regulatory Guide 1.180 (DG-1119) will replace Regulatory Guide 1.180 which was issued in January 2000. DG-1119 provides guidance to address the recent changes in the electromagnetic and radio-frequency interference (EMI/RFI) envelopes. Some of these changes include endorsing Military Standard (MIL-STD- 461E) and the International Electrotechnical Commission (IEC-61000) series of EMI/RFI test methods.

Oak Ridge National Laboratory (ORNL) has been engaged by RES to assist in developing the technical basis for regulatory guidance on the EMI/RFI immunity and power surge withstand capability. ORNL also assisted in developing the technical basis for test criteria, test methods, and operating envelopes applicable for ensuring immunity to conducted disturbances along signal lines in safety-related I&C systems.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations dated March 13, 2003, recommending that DG-1119 be issued.

5. Early Site Permit Process

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the early site permit (ESP) process. The staff has developed a draft ESP review standard (RS-002, "Processing applications for early site permits") to provide guidance to the staff and information to stakeholders on the review of an ESP application. Use of existing guidance to the extent feasible was the basic premise in the development of the ESP review standard. The staff was tasked to review NUREG-0800," Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and applicable sections of NUREG-1555, " Standard Review Plans for Environmental Reviews for Nuclear Power Plants", Regulatory Guides, Information Notices, and Generic Letters in the development of the draft ESP review standard. Currently, the draft ESP RS-002 does not contain security, quality assurance, and

accident analysis sections.

Three utilities have notified the NRC of intent to submit ESP applications. These are Exelon (Clinton site)in June 2003, Entergy (Grand Gulf site) in June 2003, and Dominion (North Anna site), in September 2003. The Nuclear Energy Institute is proposing a plant parameter envelope (PPE) approach for ESP applications that would include a set of postulated design parameters that bound the characteristics of a reactor that might later be deployed at a site. The PPE, together with the description of the site characteristics, allows the NRC to evaluate the technical and environmental suitability of the site.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated March 12, 2003, concluding that the proposed ESP review standard is appropriate for review of ESP applications and will accommodate the industry's proposed use of the PPE concept.

6. Overview of the Format and Content of the Fort Calhoun License Renewal Application

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the improved license renewal application (LRA) format and review approach. The staff described standardized methods for implementing the license renewal rule (10 CFR Part 54), as well as techniques used by the staff in evaluating applications for license renewals. Starting with Fort Calhoun, all LRAs will follow the new format.

Committee Action

This was an information briefing. No Committee action was taken.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

The Committee did not receive any response from the Executive Director for Operations for reconciliation at this meeting.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from February 6, 2003 through March 5, 2003, the following Subcommittee meetings were held:

- Planning and Procedures - March 5, 2003

The 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 MATTERS FOR THE ATTENTION OF THE EDO

- The Committee plans to review the draft final Regulatory Guide DG-1079, "Criteria for Power Systems for Nuclear Power Plants," after reconciliation of public comments.

- The Committee plans to continue its discussion of vessel head penetration cracking and degradation during future meetings.

- The Committee plans to hold further discussions with the NRC staff regarding the ACRS concerns associated with the reactor oversight process.

PROPOSED SCHEDULE FOR THE 501st ACRS MEETING

The Committee agreed to consider the following topics during the 501st ACRS meeting, to be held on April 10-12, 2003:

- Draft Final Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors"
- Draft Final Regulatory Guide, DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities"
- Control Room Habitability
- Meeting with the NRC Commissioners
- Significant Recent Operating Events
- Subcommittee Report on the Interim Review of the License Renewal Application for the St. Lucie Nuclear Power Plant
- Subcommittee Report on AP1000 Thermal-Hydraulic Matters
- Safety Culture
- Advancement of PRA Technology to Improve Risk-Informed Decisionmaking

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

CERTIFIED

Date Issued: 4/17/2003
Date Certified: 4/25/2003

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V. Draft Final Revision 1 to Regulatory Guide 1.180 (DG-119) "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems" (Open)

VI. Early Site Permit Process (Open)

VII. Overview of the Format and Content of the Fort Calhoun License Renewal Application (Open)

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A. Reconciliation of ACRS Comments and Recommendations

B. Report on the Meeting of the Planning and Procedures Subcommittee Held on March 5, 2003 (Open)

C. Future Meeting Agenda

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REPORTS:

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LETTER:

The following letter was issued to William D. Travers, Executive Director for Operations, from Mario V. Bonaca, Chairman, ACRS:

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MEMORANDUM:

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- Draft Regulatory Guide DG-1079, "Criteria for Power Systems for Nuclear Power Plants" (Proposed Revision 3 to Regulatory Guide 1.32), dated March 11, 2003

APPENDICES

I. *Federal Register Notice*
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MINUTES OF THE 500th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
March 6-8, 2003
ROCKVILLE, MARYLAND

The 500th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on March 6-8, 2003. Notice of this meeting was published in the *Federal Register* on February 21, 2003 (65 FR 8529) (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 is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc. 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: ACRS Members: Dr. Mario V. Bonaca (Chairman), Dr. Graham B. Wallis (Vice Chairman), and Mr. Stephen L. Rosen, Member-at-Large, Dr. George E. Apostolakis, Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Dr. Victor H. Ransom, Dr. William J. Shack, and Mr. John D. Sieber. 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. Mario V. Bonaca, 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. Peach Bottom License Renewal Application (Open)

[Note: Mr. Timothy J. Kobetz was the Designated Federal Official for this portion of the meeting and Mr. Ramin Assa was the Cognizant Staff Engineer]

Mr. Leitch noted that the Peach Bottom license renewal application was previously discussed during the October 30, 2002 Subcommittee meeting, and November 2002 full Committee meeting. He stated that the staff had issued the final Safety Evaluation Report (SER) on February 5, 2003. Mr. Leitch also referred to a scram that had happened on December 20, 2003, at Peach Bottom and asked the staff and the applicant to describe the event and explain whether it would have any impact on the license renewal process.

Mr. Bohlke, Senior Vice President for Nuclear Services of Exelon Corporation, began the applicant's presentation.

Mr. Polaski provided a brief description of the Peach Bottom plant and its surrounding area. He also provided a background of the application and staff's safety evaluation report (SER). He stated that the updated final safety analysis report supplement, which includes the summary description of the applicant's aging management programs, will be implemented in the next update of the final safety analysis report in April 2005. He noted that all commitments are documented in Exelon's Commitment Tracking System and procedures have been annotated for on-going commitments. Mr. Polaski displayed a list of aging management activities and described how this list, along with plant commitments, will be used to track aging management activities. As an example, he reviewed commitments and procedures related to the fire protection system aging inspections at Peach Bottom.

Dr. Bonaca, ACRS Member, asked the applicant how they intended to communicate unexpected aging degradation to the rest of the industry. Mr. Kuo, Office of Nuclear Reactor Regulation (NRR), responded that licensees file event reports which are reviewed by the staff who revise Generic Aging Lessons Learned (GALL) accordingly.

Mr. Polaski informed the Committee that the configuration change control procedures were being updated to include physical plant changes, operational changes, water chemistry conditions, and other changes to the current licensing basis. Exelon's Records Management System will retain documents, such as scoping packages and boundary drawings, that are generated during the license renewal application process.

Mr. Bohlke started the presentation by conveying the December 20, 2002 scram. He reported that the scram had initiated because of a failed electronic hydraulic control (EHC) card. Because EHC card failures are random and unpredictable, Exelon had initiated a preemptive replacement program. The card that failed and caused the scram was one of those that were replaced. However, the burn-in testing had not detected the latent manufacturing defect in the card. Of the 157 circuit cards in the EHC system the applicant had focused its replacement efforts on the 27 critical cards that could result in de-ratings or scrams. The applicant is pursuing dynamic testing of the cards to evaluate their condition before installation. Additionally, Mr. Polaski stated that they have requested General Electric to aggressively control sub-component parts.

Dr. Bonaca asked the applicant to describe their corrective action programs to identify other latent failures that became evident as a result of this scram. Mr. Stathes, Exelon, explained that following the scram, the plant had a Reactor Core Isolation Cooling (RCIC) flow oscillation. Two secondary containment isolation valve dampers, that did not close within the required stroke time, and a start-up feedwater control valve did not operate properly. A common cause analysis of the event revealed that overall, the preventive and corrective maintenance programs at Peach Bottom are adequate; however, the timeliness of corrective actions needs improvement. Each of the equipment that contributed to this event, except RCIC, were identified as needing maintenance and were scheduled for corrective action. In the case of RCIC, the technicians had adjusted the gain setting for a test mode but had not returned it to normal after the test. The applicant has modified its procedures to include a more robust review of any changes that occur during the maintenance testing process.

Mr. Kuo asked Mr. Gillespie to address the Committee's concerns regarding the integration of operational data into NRC programs. Mr. Gillespie informed the Committee that the staff is preparing a Commission paper to describe how inspection results and operating experiences should be used by the NRC. He also noted that the result of the analysis should be in a form that can easily be used by the end-users in the day-to-day decision making process. Mr. Kuo committed to meet with the Committee in the near future and discuss event analysis in general.

Mr. Solorio, NRR, briefed the Committee on resolution of the open items. He provided a brief status of BWR Vessel and Internals Projects (BWRVIP) that the staff had relied on to prepare the final SER. He noted that two license conditions were included in the SER because review of several BWRVIPs were not completed.

Mr. Solorio provided the status of torus inspections and noted that section 3.0.3.7 of the SER credits a primary containment in-service inspection (ISI) program for inspecting the torus. Mr. Polaski stated that the torus downcomer bellows were in scope, part of the containment boundary, and inspected in accordance with the ISI program. He noted that the bellows did not have any sign of corrosion.

Mr. Solorio discussed the diesel fuel oil tanks inspection activities and reported that the licensee had not found any confirmed wear during the last inspection in 1996. He then proceeded to review the standby gas treatment (SBGT) inspection results and noted that the NRC inspectors did not identify any age related degradation during a visual examination of all accessible surfaces of the system. Mr. Fulvio, Exelon, reported that the annual inspection of the SBGT filters and fire header spray nozzles have not shown any evidence of condensation or degradations. Mr. Solorio informed the Committee that fuse holder issue and top guide beams inspection issue were satisfactorily resolved.

Dr. Powers, ACRS Member, raised a concern regarding the addition of license renewal requirements to the existing programs before the extended operations period and possibly challenging licensee's staff and negatively impact current programs. Mr. McMurtrie, Peach Bottom Senior Resident Inspector, responded that the licensee (PECO) had successfully implemented all of Exelon's processes and procedures and he did not see the new license renewal requirements as a higher challenge. Mr. Bohlke agreed and added that the new requirements are being added in layers and Exelon has been able to adjust its resources to accommodate these new requirements.

Mr. Rosen, ACRS Member, questioned the insights gained by the reactor oversight process (ROP). Mr. McMurtrie, responded that the ROP tells the NRC that the applicant does not have any safety-related or risk-significant components that is significantly degraded. Mr. McMurtrie continued to say that inspectors found some low-tier issues that the licensee was not identified in their corrective action program.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated March 14, 2003 in which it recommended that the application be approved. The Committee also noted that the scram at Peach Bottom Unit 2 that occurred on December 21, 2002, highlighted a number of weaknesses in the current corrective action and preventive maintenance programs. The Committee expects that ongoing corrective actions committed to by the licensee will resolve these weaknesses. The Committee also

stated that during inspections, the NRC staff should assess the effectiveness as well as the adequacy of implementation of these programs.

III. Reactor Oversight Process (Open)

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting]

Dr. F. Peter Ford, Chairman of the Materials and Metallurgy Subcommittee, introduced this topic to the committee. The purpose of this meeting was to hear information regarding vessel head penetration cracking and degradation.

Industry Presentation

The industry presentation was made by Mr. Larry Mathews, Southern Nuclear Operating Corporation (SNOC) and Chairman of the MRP Alloy 600/82/182 Issue Task Group, Electric Power Research Institute Materials Reliability Program (EPRI/MRP).

Mr. Mathews provided data on the results of plant inspections conducted to date and the inspection plans for the Spring 2003 outages. The data showed the extent to which the fleet has been inspected, the refueling outage schedules, and current head replacement plans. It also showed the extent of detected cracking leakage, and wastage correlated with effective degradation time and position on the head.

The total reactor vessel head (RVH) nozzle population includes 3871 control rod drive mechanism (CRDM) nozzles, 1090 control element drive mechanism (CEDM) nozzles, and 94 in-core instrument (ICI) nozzles at 69 pressurized water reactor (PWR) units. Bare-metal visual and/or non-visual non-destructive examination inspections have been performed on about 81% of the RVH nozzles. Approximately 47 nozzles have been found to be leaking.

Following the spring outages, 97% of the total PWR nozzle population would have been inspected. After the fall 2003 outages, it is expected that all 69 units would have been inspected or the heads replaced.

The industry is developing a comprehensive safety assessment report as the basis for a revised MRP inspection plan. They expect to be prepared to discuss the contents of the safety assessment report and inspection plan in the summer of 2003.

Committee Action

This was an information briefing and no Committee action was taken. A future subcommittee meeting is planned to discuss these and other issues in more detail.

IV. Vessel Head Penetration Cracking and Vessel Degradation (Open)

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting]

Mr. John D. Sieber, Chairman of the ACRS Subcommittee on Plant Operations, introduced this topic to the committee. The purpose of the meeting was to discuss with the staff, a summary of the Subcommittee meeting held on January 21, 2003 relating to Staff Requirements Memorandum (SRM) M011205B, dated December 20, 2001. The SRM directed the staff, with ACRS input, to provide recommendations for resolving, in a transparent manner, apparent conflicts and discrepancies between aspects of the revised reactor oversight process that are risk-informed (e.g., significance determination process) and those that are performance based (e.g., performance indicators).

NRC Staff Presentation

The staff presentation on the ROP was made by Ms. Cynthia Carpenter, Mr. Mark Satorius, and Mr. Ronald Frahm, all of NRR.

The staff discussed the differences between the risk-informed and performance-based elements of the ROP. They also discussed the consistencies and transparency regarding the performance indicator (PI) and significance determination process (SDP) thresholds, the threshold values for the yellow-red performance indicator levels, and cross-cutting issues.

The staff indicated that, while there are differences between the risk-informed and performance-based aspects of the ROP, they believe these issues are appropriately addressed. The Committee does not agree.

The ROP basis document was developed to help reduce concerns regarding consistency and transparency of the PIs and the SDP. The staff stated that they will continue to adjust and improve the thresholds to ensure a consistent regulatory response.

The staff indicated that while it is highly unlikely that the yellow-red thresholds would be crossed, the existing thresholds provide a gauge of the relative risk of a particular indicator and demonstrate the safety margin associated with it. The staff plans to

maintain the current thresholds and consider adjusting them as ongoing changes are made to the PI program.

The staff acknowledged the concern regarding the cross-cutting issues and is developing a response.

Committee Action

The Committee wrote a letter to Chairman Meserve indicating that there are still significant disagreements between the staff and the Committee regarding the ROP. These differences continue to be those delineated in Committee letters of October 12, 2001 and February 13, 2002. The differences include, risk-informed and performance-based issues, yellow-red thresholds, cross-cutting issues, and the Action Matrix.

V. Draft Final Revision 1 to Regulatory Guide 1.180 (DG-1119) "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems" (Open)

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

Mr. J. Sieber, Chairman of the Subcommittee on Human Factors, stated that the NRC's Office of Nuclear Regulatory Research (RES) would discuss concerns regarding the draft final Regulatory Guide (DG-1119), "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems."

Mr. Sieber commented that electromagnetic interference (EMI), radio-frequency interference (ERI), and power surges have been identified as environmental conditions that can affect the performance of safety-related electrical equipments. 10 CFR Part 50 states that structures, systems, and components (SSC) important to safety in a nuclear power plants be designed to accommodate the effects of environmental conditions (i.e., remain functional under all postulated service conditions) and that design control measures, such as testing, be used to check the adequacy of design.

DG-1119 will replace Regulatory Guide 1.180 which was issued in January 2000. DG-1119 provides guidance to address the recent changes in the electromagnetic and radio-frequency interference (EMI/RFI) envelopes. Some of these changes include endorsing Military Standard (MIL-STD- 461E) and the international electrotechnical Commission (IEC-61000) series of EMI/RFI test methods, extending the guidance to cover signal line testing, incorporating frequency ranges where portable communication devices are experiencing increase use, and relaxing the operating envelopes (test levels) when experience and confirmatory research warrants.

Oak Ridge National Laboratory (ORNL) has been involved with RES in developing the technical basis for regulatory guidance on the EMI/RFI immunity and power surge withstand capability. The recommendation by ORNL on test criteria, test methods, and operating envelopes was significantly influenced by the military standards issued by the Department of Defense. ORNL also assisted in developing the technical basis for electromagnetic compatibility testing for conducted susceptibility along interconnecting signal lines.

Committee Action

The Committee issued a letter to William D. Travers, NRC Executive Director for Operations, dated March 13, 2003, recommending that DG-1119 provide appropriate guidance and should be issued. The Committee also stated in its letter that the public comments received by the NRC staff were appropriately dispositioned.

VI. Early Site Permit Process (Open)

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

Dr. Thomas Kress, Chairman of the Subcommittee on Future Plant Designs, stated that the staff has developed a draft early site permit (ESP) review standard (RS-002, "Processing Applications for Early Site Permits") to provide guidance to the staff, and information to stakeholders on the review of an ESP application. Use of existing guidance to the extent feasible was the basic premise in the development of the ESP review standard (RS).

Mr. Ronaldo Jenkins, NRR, stated that the staff was tasked to review NUREG-0800," Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and applicable sections of NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," Regulatory Guides, Information Notices, and Generic Letters in the development of the draft ESP RS. The staff's modifications are generally of the form of strikeouts of text not applicable to ESP and revisions to bring portions of the standard up to date. Some review issues that require knowledge of items that are design-specific, such as source terms, will be accommodated by bounding values. The current draft ESP RS-002 does not contain security, quality assurance, and accident analysis sections.

Subpart A of 10 CFR Part 52 provides a process for applicants considering construction of a nuclear power plant to resolve site issues separately from application for a combined license. Three utilities have notified the NRC of intent to submit ESP

applications. They are Exelon (Clinton site) in June 2003; Entergy (Grand Gulf site) in June 2003; and, Dominion (North Anna site), in September 2003.

The Nuclear Energy Institute (NEI) believes that 10 CFR Part 52.17(a)(2) requires an environmental report that focuses on the effects of construction and operation of a reactor, or reactors, which have characteristics that fall within the postulated site parameters. NEI is proposing a plant parameter envelope (PPE) approach for ESP applications. The PPE is a set of postulated design parameters that bound the characteristics of a reactor that might later be deployed at a site. The PPE, together with the description of the site characteristics, allows the NRC to evaluate the technical and environmental suitability of the site.

Committee Action

The Committee issued a report to Chairman Meserve on this matter dated March 12, 2003, concluding that the proposed ESP RS is appropriate for review of ESP applications and will accommodate the industry's proposed use of the PPE concept.

VII. Overview of the Format and Content of the Fort Calhoun License Renewal Application (Open)

[Note: Mr. Timothy J. Kobetz was the Designated Federal Official for this portion of the meeting]

The Committee heard discussions with the NRC staff regarding the improved, and significantly different, license renewal application (LRA) format and review approach. The staff described standardized methods for implementing the license renewal rule (10 CFR Part 54), as well as techniques used by the staff in evaluating applications for license renewals. Starting with Fort Calhoun, all LRAs will use this guidance in the following documents:

- NUREG-1800 – Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants
- NUREG-1801 Volume 1 – Generic Aging Lessons Learned (GALL) Report – Summary
- NUREG-1801 Volume 2 – Generic Aging Lessons Learned (GALL) Report – Tabulation of Results
- Regulatory Guide 1.188 – Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses
- Nuclear Energy Institute (NEI) 95-10, Revision 3 – Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule

Mr. William Burton, Project Manager, NRR, initiated the presentation by noting that St. Lucie is the last facility that will seek license renewal and not specifically follow the above guidance (generically referred to as GALL) for its LRA. St. Lucie has incorporated several aspects of GALL in its application, however, starting with Fort Calhoun, all future LRAs will use the GALL guidance. In general, the changes to the LRA involve the following:

- Section 2, "Scoping and Screening Methodology and Results." The tables will contain specific links to corresponding information in Section 3.
- Section 3, "Aging Management Reviews." This section is now formatted by component in lieu of by system as previous applications were formatted. For instance, all of the reactor system components get combined into a series of reactor system tables. The individual components for those systems are no longer presented. Several of the Committee Members were concerned that the new format would make the review process more cumbersome.
- Appendix B, "Aging Management Activities." The format of this Appendix has changed significantly.

With the new format, Section 3 of the LRA will resemble the tables in GALL for compatibility. The new format allows the applicant to take credit for aging management reviews previously evaluated in GALL. Section 3 is now grouped into three sets of tables (1) structures and components (SCs) evaluated in GALL, (2) not evaluated in GALL, and (3) SCs not evaluated in GALL, but GALL could be credited with managing component aging.

There is no significant changes in the format or in the review approach in Sections 2 and 4. In Section 2, the goal remains to ensure that the applicant has identified all the structural SCs that are within the scope of the license renewal rule. The tables in Section 2 have been reformatted to provide direct links to information in Section 3. Similarly, in Section 4, which provides the Time-Limited Aging Analysis (TLAA), the staff's review ensures any analyses or calculations that are time-dependent on the initial 40 year nuclear plant life span, and will be valid for the entire 60 years as provided by license renewal.

Next, Mr. Burton stepped through several examples of how the new format is being used for reviewing the Fort Calhoun LRA and answered a variety of format and processing questions from the Committee Members. During this discussion, Mr. Burton also discussed the formatting changes to Appendix B which describes the Aging Management Programs.

The staff continues to have discussions with NEI regarding the LRA formatting improvements. In addition, during the Fort Calhoun LRA review the staff has been identifying lessons learned that will also be used to modify the format.

Dr. Ford questioned whether any lessons learned have been identified that affect previously approved LRAs. Mr. Burton acknowledged there had been, however, they have not been related to the use of GALL. When new issues arise that need to be addressed, the staff initiates an Interim Staff Guidance document that must be reviewed in accordance with 10 CFR 50.109 for backfit implications.

Committee Action

This was an information briefing. No Committee action was requested or is planned.

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 did not receive any comments or recommendations for reconciliation from the NRC Executive Director for Operations for this meeting.

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

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on March 5, 2003. The following items were discussed:

- Review of the Member Assignments and Priorities for ACRS Reports and Letters for the March ACRS meeting

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

- Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through May 2003 was considered. 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

- Update on the Budget

The continuing resolution has ended, and we now have our budget. However, given the increase in workload and several special events (quadripartite meeting, 500th meeting, etc.) funds for the remainder of this fiscal year (September 30, 2003) are limited, particularly in the travel category.

- ACRS Meeting with the NRC Commissioners

A meeting between the ACRS and the NRC Commissioners is scheduled for April 11, 2002, between 9-11:00 a.m. to discuss items of mutual interest. Items approved by the Commission for this meeting are as follows:

- Overview by the ACRS Chairman (MVB)
  - 500th meeting celebration
  - Quadripartite Meeting
  - License Renewal Activities
  - Core power uprates
  - Future ACRS activities - including "sunset" of activities that do not necessarily need to be reviewed.

(2) Advanced Reactor Designs (TSK)
  - Early Site Permit Process
  - Options paper for resolving policy issues
  - AP1000 review activities

(3) Pressurized Thermal Shock Technical Basis Reevaluation Project (WJS)

(4) ACRS Report on the NRC Safety Research Program (FPF)

The Commission has approved the above agenda with the recommendation that the PTS discussion be limited to about 5-10 minutes.

- Role and Use of PRA in the Regulatory Decisionmaking Process

Mr. Fleming, under a contract with NRC/ACRS, submitted a draft report on the role and use of PRA in the risk informed decisionmaking process to the Committee. The Committee discussed this report with Mr. Fleming during its February 2003 meeting. During the February 2003 ACRS meeting, Dr. Nourbakhsh, ACRS Senior Fellow, presented an outline of a draft "White Paper" and recommendations to be included in the paper based on Mr. Fleming's report. Dr. Nourbakhsh has worked with Carl Fleming to edit his report and is in the process of having it published as a NUREG. We anticipate this being issued in the next month. Dr. Apostolakis has prepared a draft report to the Commission instead of a White Paper as previously discussed in February.

- Comments by Mr. Robert H. Leyse, a Member of the Public, Regarding Fouling in Reactivity Insertion Accidents

Mr. Leyse, a member of the public, submitted a petition for rulemaking (pp. 10-16) to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants. This includes all heat transfer surfaces, fuel elements, steam generators, condensers, fan coolers, etc. The new regulations must also require the inclusion of fouling considerations in the NRC-funded test programs such as the Rod Bundle Heat Transfer Program at the Penn State University. Some of the comments provided by Mr. Leyse for evaluation by the ACRS are as follows:

- Even though fouling of nuclear reactor fuel elements has a greater impact on the Reactivity Insertion Accidents and LOCAs than high-burnup, the NRC and its collaborators continue to totally ignore the impact of fouling.

- There was no discussion of fouling either at the October 9, 2002 ACRS Subcommittee meeting on Reactor Fuels or at the October 2002 full Committee meeting. In addition, the October 17, 2002 ACRS letter on the Confirmatory Research Program on High-Burnup Fuel did not include any discussion on fouling.

- The Rod Bundle Heat Transfer Program at the Penn State University includes no provisions for the impact of fouling on the heat transfer surfaces. Surprisingly, the ACRS did not point out the gross inadequacy of this Program. Copies of the information provided by Mr. Leyse were sent to Dr. Powers and Dr. Wallis.

The Subcommittee discussed several options for handling the concerns raised by Mr. Leyse, including the following:

- Should this matter be referred to the EDO, as has been done in the past on similar issues?

- Should Mr. Leyse be notified by the ACRS Executive Director that the ACRS does not get involved in the petition for rulemaking process and that the Reactor Fuels Subcommittee plans to invite him in to discuss his concerns at the next meeting of the Reactor Fuels Subcommittee?

- Committee Review of License Renewal Applications (Peach Bottom Event)

There have been a number of e-mails exchanged regarding the ACRS review of license renewal applications related to plants whose current operating experience raises the question of whether their license renewal applications should be approved. Most recently, this issue has come up in connection with the license renewal application for Peach Bottom Units 2 and 3, which was discussed by the Committee at its March 2003 meeting.

The applicant, NRC staff, and the Senior Resident Inspector addressed the reactor scram event which occurred at Peach Bottom in December 2002 along with the corrective action programs at the March meeting.

The ACRS has previously moved ahead with its review of license renewal applications in spite of current operating experience with the idea that these conditions would be corrected prior to entering into the renewal period.

- Insights/Safety Culture

During the February 2003 meeting, the Committee discussed a draft report on Safety Culture written by ACRS Members Rosen and Apostolakis, and requested that Mr. Rosen revise this report incorporating the comments from the members. It was suggested that the letter be reasonably short and minimize recommendations to suggesting that the Commission encourage the staff to stay aware of developments in the Safety Culture arena both domestically and internationally.

It was further suggested that the ACRS meet with the NRC staff, First Energy Nuclear, NEI, and others (i.e., David Collins and Howard C. Whitcomb) later in the year at a subcommittee meeting to discuss initiatives related to assessing safety culture and its implications on the safe operations of nuclear facilities.

- Member Issues

  - Emergency Notification to Members Regarding Subcommittee Cancellations

Recent events have noted the importance of having some process in place to notify members in the event of a severe weather condition which would inhibit travel to and from the Washington area. A process is being put in place to address this issue.

- An issue has been raised in terms of how the Committee should handle differing technical views among the NRC staff after the Committee has approved a particular staff action such as DG-1077 Equipment Qualification of Microprocessor-Based Equipment.

- The ACRS Executive Director was successful in getting an exemption to the Commission reappointment policy in having Dr. Thomas Kress appointed for a fourth term to begin at the end of his current term (September 3, 2003). The Commission, in the SRM, approved the reappointment and noted that this exception to its policy will be done sparingly.

- The NRC staff is planning a one week tour of a test facility associated with SWR 1000 (located in Germany) in early June 2003. The NRR staff is expecting possibly 2-3 NRC managers to attend, and asked if an ACRS member (maybe Dr. Kress) would be interested in attending such a tour. Framatome is sponsoring the tour, however, Framatome is not responsible for the travel and hotels.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 501st ACRS Meeting, April 10-12, 2003.

The 500th ACRS meeting was adjourned at 12:00 p.m. on March 8, 2003.

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

April 25, 2003

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

FROM: Mario V. Bonaca
Chairman

SUBJECT: CERTIFIED MINUTES OF THE 500th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MARCH 6-8, 2003

I certify that based on my review of the minutes from the 500th 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.

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

April 17, 2003

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 500th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - MARCH 6-8, 2003

Enclosed are the proposed minutes of the 500th 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, which will be distributed within six (6) working days from the date of this memorandum.

Attachment:
As stated

such as electronic delivery of modified wage decisions directly to the user's desktop, the ability to access prior wage decisions issued during the year, extensive Help Desk Support, etc.

Hard-copy subscriptions may be purchased from: Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402, (202) 512–1800.

When ordering hard-copy subscription(s), be sure to specify the State(s) of interest, since subscriptions may be ordered for any or all of the six separate Volumes, arranged by State. Subscriptions include an annual edition (issued in January or February) which includes all current general wage determinations for the States covered by each volume. Throughout the remainder of the year, regular weekly updates will be distributed to subscribers.

Signed at Washington, DC, this 13th day of February 2003.

Carl J. Poleskey,

Chief, Branch of Construction Wage Determinations.

[FR Doc. 03–4135 Filed 2–20–03; 8:45 am]

BILLING CODE 4510–27–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 70–143–MLA–2; ASLBP No. 03–810–02–MLA]

Nuclear Fuel Services, Inc.; Designation of Presiding Officer

Pursuant to delegation by the Commission, *see* 37 FR 28,710 (Dec. 29, 1972), and the Commission's regulations, *see* 10 CFR §§ 2.1201, 2.1207, notice is hereby given that (1) a single member of the Atomic Safety and Licensing Board Panel is designated as Presiding Officer to rule on petitions for leave to intervene and/or requests for hearing; and (2) upon making the requisite findings in accordance with 10 CFR § 2.1205(h), the Presiding Officer will conduct an adjudicatory hearing in the following proceeding: Nuclear Fuel Services, Inc., Erwin, Tennessee, (Material License Amendment–2).

The hearing will be conducted pursuant to 10 CFR part 2, subpart L, of the Commission's Regulations, "Informal Hearing Procedures for Adjudications in Materials and Operator Licensing Proceedings." This proceeding concerns requests for hearing submitted on February 6, 2003, by Friends of the Nolichucky River Valley, Inc., the State of Franklin Group of the Sierra Club, the Oak Ridge Environmental Peace Alliance, the Tennessee Environmental Council, and Kathy Helms-Hughes. The requests were filed in response to an NRC staff notice of receipt of a second request from Nuclear Fuel Services, Inc. (NFS), to amend its special nuclear materials license to support downblending and conversion of high-enriched uranium material to low-enriched uranium oxides. This amendment would allow processing operations at its Erwin, Tennessee Blended Low-Enriched Uranium Preparation Facility to prepare low-enriched uranium solutions for a new complex that would manufacture low-enriched nuclear reactor fuel. The notice of receipt of amendment request and opportunity for a hearing were published in the **Federal Register** on January 7, 2003 (68 FR 796).

The Presiding Officer in this proceeding is Administrative Judge Alan S. Rosenthal. Pursuant to the provisions of 10 CFR §§ 2.722, 2.1209, Administrative Judge Richard F. Cole has been appointed to assist the Presiding Officer in taking evidence and in preparing a suitable record for review.

All correspondence, documents, and other materials shall be filed with Judges Rosenthal and Cole in accordance with 10 CFR § 2.1203. Their addresses are: Administrative Judge Alan S. Rosenthal, Presiding Officer, Atomic Safety and Licensing Board Panel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; Administrative Judge Richard F. Cole, Special Assistant, Atomic Safety and Licensing Board Panel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

Issued at Rockville, Maryland, this thirteenth day of February 2003.

G. Paul Bollwerk, III,

Chief Administrative Judge, Atomic Safety and Licensing Board Panel.

[FR Doc. 03–4111 Filed 2–20–03; 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 (ACRS) will hold a meeting on March 6–8, 2003, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 20, 2002 (67 FR 70094).

Thursday, March 6, 2003

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

8:35 a.m.–10:15 a.m.: Peach Bottom License Renewal Application (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Exelon Generation Corporation, LLC, regarding the license renewal application for the Peach Bottom Nuclear Plant, Units 2 and 3 and the associated NRC staff's final Safety Evaluation Report.

10:30 a.m.–12:30 p.m.: Reactor Oversight Process (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding staff's views concerning ACRS recommendation on thresholds for performance indicators (PIs); staff's efforts associated with risk-based PIs and improvements to the significance determination process (SDP); and staff's activities for resolving apparent conflicts and discrepancies between aspects of the revised reactor oversight process that are risk informed (*e.g.*, SDP) and those that are performance based (*e.g.*, PIs).

1:30 p.m.–3:30 p.m.: Vessel Head Penetration Cracking and Vessel Head Degradation (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding industry responses to NRC Bulletin 2002–02, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," inspection requirements and programs for reactor pressure vessel heads and vessel head penetration nozzles, Electric Power Research Institute's Materials Reliability Program's proposed inspection program, wastage research, and related matters.

3:45 p.m.–5 p.m.: Draft Final Revision 1 to Regulatory Guide 1.180 (DG–1119) "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference In Safety-Related Instrumentation and Control Systems" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final revision 1 to Regulatory Guide 1.180 (DG–1119) and the staff's resolution of public comments.

5:15 p.m.–7:15 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting,

as well as a proposed ACRS report on Safety Culture.

Friday, March 7, 2003

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

8:35 a.m.–10:30 a.m.: Early Site Permit Process (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding draft review standard RS–002, "Processing Applications for Early Site Permits," and related matters.

10:45 a.m.–11:45 a.m.: Overview of the Format and Content of the Fort Calhoun License Renewal Application (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff on the format and content of the Fort Calhoun Nuclear Plant license renewal application, the first plant to submit a license renewal application using the generic license renewal guidance documents.

12:45 p.m.–1:30 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, including anticipated workload and member assignments.

1:30 p.m.–1:45 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 p.m.–4 p.m.: Preparation for Meeting with the NRC Commissioners (Open)—The Committee will discuss proposed topics for discussion during the ACRS meeting with the NRC Commissioners which is scheduled to be held on Friday, April 11, 2003, between 9 a.m. and 11 a.m.

4:15 p.m.–7 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, March 8, 2003

8:30 a.m.–1 p.m.: Proposed ACRS Reports (Open)—The Committee will continue to discuss proposed ACRS reports. In addition, the Committee will discuss a draft white paper prepared by the ACRS Senior Fellow on the role and use of PRA in the regulatory decisionmaking process.

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

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 11, 2002 (67 FR 63460). In accordance with those 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. Persons desiring to make oral statements should notify the Associate Director for Technical Support named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Associate Director prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Associate Director if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Dr. Sher Bahadur, Associate Director for Technical Support (301–415–0138), between 7:30 a.m. and 4:15 p.m., ET.

ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., ET, 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 and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: February 13, 2003.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 03–4109 Filed 2–20–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on March 5, 2003, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows: Wednesday, March 5, 2003—1:30 p.m. until the conclusion of business.

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301/415–7364) between 7:30 a.m. and 4:15 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

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

February 24, 2003 (REVISED)

SCHEDULE AND OUTLINE FOR DISCUSSION
500th ACRS MEETING
MARCH 6-8, 2003

THURSDAY, MARCH 6, 2003, 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 (MVB/JTL/SD)
   - 1.2) Items of current interest (MVB/SD)

2) 8:35 - ~~10:15~~ 10:30 A.M. Peach Bottom License Renewal Application (Open) (GML/MVB/RRA/TJK)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and the Exelon Generation Corporation, LLC, regarding the license renewal application for the Peach Bottom Nuclear Plant, Units 2 and 3 and the associated NRC staff's final Safety Evaluation Report.

~~10:15~~ - ~~10:30~~ 10:30 - 10:45 A.M. ***BREAK***

3) ~~10:30~~ 10:45 - 12:30 P.M. Reactor Oversight Process (Open) (JDS/GEA/MWW)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding staff's views concerning ACRS recommendation on thresholds for performance indicators (PIs); staff's efforts associated with risk-based PIs and improvements to the significance determination process (SDP); and staff's activities for resolving apparent conflicts and discrepancies between aspects of the revised reactor oversight process that are risk informed (e.g., SDP) and those that are performance based (e.g., PIs).

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

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

4) 1:30 - 3:30 P.M. Vessel Head Penetration Cracking and Vessel Head Degradation (Open) (FPF/MWW)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff regarding industry response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," inspection requirements and programs for reactor pressure vessel heads and vessel head penetration nozzles, Electric Power Research Institute's Materials Reliability Program's proposed inspection program, wastage research, and related matters.

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

3:30 - 3:45 P.M. ***BREAK***

5) 3:45 - ~~5:00~~ 5:10 P.M. Draft Final Revision 1 to Regulatory Guide 1.180 (DG-1119) "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference In Safety-Related Instrumentation and Control Systems" (Open) (JDS/MME)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision 1 to Regulatory Guide 1.180 (DG-1119) and the staff's resolution of public comments.

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

~~5:00~~ - ~~5:15~~ 5:10-5:25 P.M. ***BREAK***

6) 5:15 - 7:15 P.M Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

5:30-5:50 6.1) Peach Bottom License Renewal Application (GML/MVB/RRA/TJK)

5:30-6:15 6.2) Reactor Oversight Process (JDS/GEA/MWW)

6.3) Vessel Head Penetration Cracking and Vessel Head Degradation (FPF/MWW)

6:30-7:00 6.4) Draft Final Revision 1 to Regulatory Guide 1.180 (DG-1119) (JDS/MME)

6.5) Safety Culture (SLR/MWW)

6:15-7:00 6.6) White Paper on the Role and Use of PRA in the Regulatory Decisionmaking Process (GEA/HN/SD)

FRIDAY, MARCH 7, 2003, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - ~~10:30~~ 10:10 A.M. Early Site Permit Process (Open) (TSK/MME)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff regarding draft review standard RS-002, "Processing Applications for Early Site Permits", and related matters.

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

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

9) 10:45 - ~~11:45~~ 12:00 A.M. Overview of the Format and Content of the Fort Calhoun License Renewal Application (Open) (MVB/TJK)

9.1) Remarks by the Subcommittee Chairman

9.2) Briefing by and discussions with representatives of the NRC staff on the format and content of the Fort Calhoun Nuclear Plant license renewal application, the first plant to submit a license renewal application using the generic license renewal guidance documents.

~~11:45~~ - ~~12:45~~ 12:00 - 1:00 P.M. ***LUNCH***

10) ~~12:45~~ 3:15 - ~~1:30~~ 4:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/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, including anticipated workload and member assignments.

11) 1:30 - 1:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, 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.

1:45 - 2:00 P.M. ***BREAK***

12) 1:00 - 2:50 ~~2:00~~ - ~~4:00~~ P.M. Preparation for Meeting with the NRC Commissioners (Open) (MVB/JTL)
The Committee will discuss proposed topics for discussion during the ACRS meeting with the NRC Commissioners which is scheduled to be held on Friday, April 11, 2003 between 9:00 a.m. and 11:00 a.m.

~~4:00 - 4:15 P.M. ***BREAK***~~

13) 4:15 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

- 13.1) Peach Bottom License Renewal Application (GML/MVB/RRA/TJK)
- 4:15-5:55 13.2) Reactor Oversight Process (JDS/GEA/MWW)
- 13.3) Vessel Head Penetration Cracking and Vessel Head Degradation (FPF/MWW)
- 10:30-10:40 13.4) Draft Final Revision to Regulatory Guide 1.180 (DG-1119) (JDS/MME)
- 6:15-7:00 13.5) Safety Culture (SLR/MWW)
- 5:55-6:04 13.6) Early Site Permit Process (TSK/MME)
- ~~13.7) White Paper on the Role and Use of PRA in the Regulatory Decisionmaking Process (GEA/HN/SD)~~

SATURDAY, MARCH 8, 2003, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - ~~1:00~~ 12:00 P.M. Preparation of ACRS Reports (Open)
(10:00-10:15 A.M. BREAK) Continue discussion of the proposed ACRS reports listed under Item 13, as well as a draft white paper prepared by the ACRS Senior Fellow on the role and use of PRA in the regulatory decisionmaking process.

15) 1:00 - 1:30 P.M. Miscellaneous (Open) (MVB/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.

Safety Culture 11:35-12:00
Peach Bottom 8:40-9:30
ESP 9:35-10:10
ROP 10:10-11:35

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.**
- **Thirty-Five (35) copies of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

500TH ACRS MEETING
MARCH 6-8, 2003

NRC STAFF (March 6, 2003)

R. Auluck, NRR
N. Iqbal, NRR
H. Wang, NRR
J. Raval, NRR
S. Coffin, NRR
R. Pettis, NRR
C. Sydnor, NRR
J. Walker, NRR
J. Fair, NRR
P. Kang, NRR
P. T. Kuo, NRR
J. Boska, NRR
D. Nguyen, NRR
T. J. Kim, NRR
S. Bailey, NRR
B. Elliot, NRR
K. Parczewski, NRR
Y. C. (Renee) Li, NRR
C. Li, NRR
I. Rajan, NRR
D. Jeng, NRR
D. Solorio, NRR
M. Markley, NMSS
B. Bateman, NRR
W. Burton, NRR
G. Hatchett, NRR
G. Bagchi, NRR
R. Borchardt, NRR
D. Allsopp, NRR
H. Asher, NRR
M. Marshall, NRR
J. Costello, RES
S. Arndt, RES
C. Antonesau, RES
S. Burnell, OPA
G. Georgiev, NRR
K. Corp, NRR
O. Tabatabai, NRR
S. Lee, NRR
M. Cora, NRR
C. Holden, NRR
R. Franovich
R. Arrighi, NRR
D. Solorio, NRR
J. Medoff, NRR
C. Hunter, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

E. Patel, Exelon
G. Stathes, Exelon
P. Fulvio, Exelon
F. Powaski, Exelon
R. Clemiewicz, Exelon
W. Bohlke, Exelon
R. Janati, PADEP/BRP
D. Dyckman, PADEP/BRP
R. Huston, Lic. Support Serv.
S. Page, ALT
T. McMurtray, NRC/Peach Bottom
A. Wyche, SERCH Lic/Bechtel
M. Woods, Pitts. Post Gazette
L. Mathews, SNC
K. Cozens, NEI
D. Hunter, McGraw-Hill
R. Wood, ORNL
P. Ewing, ORNL

NRC STAFF (March 7, 2003)
R. Jenkins, NRR
E. Throm, NRR
B. Zalcman, NRR
J. Mitchell, RES
L. Brown, NRR
K. Campe, NRR
J. Tapport, NRR
T. Kenyon, NRR
D. Barss, NRR
M. Gamberoni, NRR
J. Wilson, NRR
C. Hinson, NRR
K. Coyne, NRR
P. Prescott, NRR
J. Lee, NRR
S. Klementowicz, NRR
K. Heck, NRR
G. Bagchi, NRR
S. Koenik, NRR
R. Tripathi, RES
J. Strnisha, OE

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
R. Bell, NEI
G. Zinke, Entergy
E. Grant, Exelon
S. Gambhir, OPPD/Fort Calhoun
T. Clementis, Progress Energy
G. Wrobel, Rochester Gas & Electric
D. Wilson, Rochester Gas & Electric
M. Fallin, Constellation Nuc Services
M. Fletcher, Progress Energy

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

March 14, 2003

SCHEDULE AND OUTLINE FOR DISCUSSION
501st ACRS MEETING
APRIL 10-12, 2003

THURSDAY, APRIL 10, 2003, 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 (MVB/JTL/SD)
   1.2) Items of current interest (MVB/SD)

2) 8:35 - 9:45 A.M. Draft Final Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors" (Open) (WJS/MRS/SD)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final risk-informed revisions to 10 CFR 50.44.

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

9:45 - 10:00 A.M. ***BREAK***

3) 10:00 - 12:30 P.M. Draft Final Regulatory Guide, DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities" (Open) (GEA/MRS/SD)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final version of DG-1122, including resolution of public comments.

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

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

4) 1:30 - 3:15 P.M. Control Room Habitability (Open) (DAP/MME)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff regarding four draft regulatory guides concerning control room habitability, dose assessment, meteorological effects, and in-leakage testing as well as a related draft generic letter.

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

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

5) 3:30 - 4:45 P.M. Preparation for Meeting with the NRC Commissioners (Open) (MVB, et al./JTL, et al.)
Discussion of the following topics scheduled for the ACRS meeting with the NRC Commissioners between 9:00 - 11:00 a.m. on April 11, 2003:

a) Overview by the ACRS Chairman (MVB)
- 500th ACRS Meeting Celebration
- Quadripartite Meeting
- License renewal activities
- Core power uprates
- Future ACRS activities - including "sunset" of activities that do not necessarily need to be reviewed

b) Advanced Reactor Designs (TSK)
- Early site permit process
- Options paper for resolving policy issues
- AP1000 review activities

c) Pressurized Thermal Shock Technical Basis Reevaluation Project (WJS)

d) ACRS Report on the NRC Safety Research Program (FPF)

4:45 - 5:00 P.M. ***BREAK***

6) 5:00 - 7:15 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

6.1) Draft final risk-informed revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors" (WJS/MRS/SD)

6.2) Draft final Regulatory Guide, DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities" (GEA/MRS/SD)

6.3) Control Room Habitability (DAP/MME)

6.4) Advancement of PRA Technology to Improve Risk-Informed Decisionmaking (GEA/HN)

6.5) Insights/Safety Culture (SLR/MWW)

FRIDAY, APRIL 11, 2003, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:40 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)

8:40 - 9:00 A.M. ***BREAK***

8) 9:00 - 11:00 A.M. Meeting with the NRC Commissioners (Open) (MVB, et al./JTL, et al.)
The Committee will meet with the NRC Commissioners in the Commissioners' Conference Room, One White Flint North to discuss the matters listed under Item 5.

11:00 - 11:15 A.M. *****BREAK*****

9) 11:15 - 12:15 P.M. Significant Recent Operating Events (Open) (GML/MWW)
The Subcommittee Chairman will discuss significant recent operating events.

12:15 - 1:15 P.M. *****LUNCH*****

10) 1:15 - 2:15 P.M. Refresher Training Course for Members (Open) (MVB/JTL)
Representatives of the Offices of the General Counsel, Administration, and Security will provide a refresher training course for the members regarding the ethics, conflict-of-interest, travel, and security requirements.

11) 2:15 - 2:30 P.M. Subcommittee Report on the Interim Review of the License Renewal Application for the St. Lucie Nuclear Power Plant (Open) (MVB/GML/TJK)
Report by the Plant License Renewal Subcommittee Chairman regarding the Subcommittee's review of the St. Lucie license renewal application and the staff's initial Safety Evaluation Report.

12) 2:30 - 2:45 P.M. Subcommittee Report on AP1000 Thermal-Hydraulic Matters (Open) (GBW/MRS/MME)
Report by the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the Subcommittee's review of the thermal-hydraulic matters associated with the AP1000 passive plant design.

2:45 - 3:00 P.M. *****BREAK*****

13) 3:00 - 3:45 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)
13.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
13.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

14) 3:45 - 4:00 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, 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.

15) 4:00 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
15.1) Draft final risk-informed revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors" (WJS/MRS/SD)

15.2) Draft final Regulatory Guide, DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities" (GEA/MRS/SD)
15.3) Control Room Habitability (DAP/MME)
15.4) Advancement of PRA Technology to Improve Risk-Informed Decisionmaking (GEA/HN)
15.5) Insights/Safety Culture (SLR/MWW)

SATURDAY, APRIL 12, 2003, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16) 8:30 - 1:00 P.M. Preparation of ACRS Reports (Open)
(10:00-10:15 A.M. BREAK) Continue discussion of the proposed ACRS reports listed under Item 15.

17) 1:00 - 1:30 P.M. Miscellaneous (Open) (MVB/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.**
- **Thirty-Five (35) copies of the presentation materials should be provided to the ACRS.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
500th ACRS MEETING
MARCH 6-8, 2003

[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 March 6-8, 2003
2	Peach Bottom License Renewal Application
	2. Peach Bottom Atomic Power Station Plant License Renewal presentation by Exelon [Viewgraphs]
	3. Peach Bottom License Renewal Safety Evaluation Report presentation by NRR [Viewgraphs]
3	Reactor Oversight Process
	4. Letter to the NRC Chairman Richard A. Meserve from the ACRS Chairman George Apostolakis, dated October 12, 2001, subject "The Revised Reactor Oversight Process" [Handout]
	5. Letter to the ACRS Chairman George E. Apostolakis from William D. Travers, Executive Director for Operations, dated January 2002, subject [response] "The Revised Reactor Oversight Process" [Handout]
	6. ACRS Full Committee Meeting, The Reactor Oversight Process presentation by C. Carpenter, M. Satorious, R. Frahm, NRR [Viewgraphs]
4	Vessel Head Penetration Cracking and Vessel Head Degradation
	7. MRP Alloy 600 ITG Reactor Vessel Head WG Status Report, Vessel Head Penetration Cracking and RPV Head Degradation presentation by L. Mathews, SNOC (MRP Alloy 600/82/182 Issue Task Group Chairman) [Viewgraphs]
5	Draft Final Revision 1 to Regulatory Guide 1.180 (DG-1119) "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems"
	8. Overview of DG-1119 Guidelines for Evaluating EMI/RFI in Safety-Related I&C Systems presentation by RES and Oak Ridge National Laboratory [Viewgraphs]

8 Early Site Permit

9. Early Site Permit (ESP) and ESP Review Standard presentation by R. Jenkins and M. Scott, New Reactor Licensing Project Office [Viewgraphs]

9 Overview of the Format and Content of the Fort Calhoun License

10. License Renewal Under the New GALL Regime presentation by NRR [Viewgraphs]

10 Future ACRS Activities/Report of the Planning and Procedures Subcommittee

11. Final Future ACRS Activities/Draft Minutes of Planning and Procedures Subcommittee Meeting - March 5, 2003 [Handout **#10.1**]

11 Reconciliation of ACRS Comments and Recommendations

XX. Reconciliation of ACRS Comments and Recommendations [Handout **#XX**]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Peach Bottom Atomic Station Units 2 and 3 License Renewal Application
	1. Table of Contents
	2. Proposed Schedule
	3. Status Report
	7. Peach Bottom Atomic Power Station - NRC Inspection Report 50-277/02-012, 50-278/02-012
	8. Excerpts from the final Safety Evaluation Report (SER) related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3," February 2003
3	The Reactor Oversight Process
	9. Table of Contents
	10. Proposed Schedule
	11. Status Report
	12. Note to John Larkins from Cynthia Carpenter, Subject: Staff Response to Issues Discussed During the ACRS Subcommittee Briefing on the Reactor Oversight Process on September 9, 2003
	13. Inside NRC, February 24, 2003, NRC Comments of the Signficance Determination Process Modifications (from a February 10, 2003 Briefing on NRR Programs, Performance and Plans)
	14. Inside NRC, February 24, 2003, NRC Comments Taken from the SDP Task Group Report on the SDP in Response to a DPO
4	Vessel Head Penetration Cracking and RPV Head Degradation
	15. Table of Contents
	16. Proposed Schedule
	17. Status Report
	18. Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors, EA-03-009, February 11, 2003
	19. List of PWR Licensees and Susceptibility (attached to Order)
	20. Letter from A. Marion, NEI, to B. Sheron, NRC, Subject, "Questions Concerning NRC Order EA-03-009," February 19, 2003
5	Draft Final Revision 1 to R.G. 1.180 (DG1119)
	21. Table of Contents
	22. Proposed Agenda

23. Status Report
24. NUREG/CR-XXX, "Comparison of U.S. Military and International Electromagnetic Compatibility Guidance," ORNL, June 2002
25. NUREG/CR-5609, "Electromagnetic Compatibility Testing for Conducted Susceptibility Along Interconnecting Signal Lines," ORNL, June 2002
26. Larkinsgram to W.D. Travers, dated July 16, 2002
27. DG-1119, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," February 2003

8 Early Site Permit Process

28. Table of Contents
29. Proposed Agenda
30. Status Report
31. RS-002, "Processing Applications for Early Site Permits," December 23, 2002

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500th FULL COMMITTEE MEETING

MARCH 6-8, 2003

MARCH 6, 2003
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
R. Auluck	NRR/DRIP
NAEEM IQBAL	NRR/DSSA/SPLB
Hai-Boh Wang	NRR/DRIP/RLEP
J. H. Raval	NRR/SPLB
Stephanie Coffin	NRR/DE/EMCB
ROBERT PETTIS	NRR/DIPM
Chris Sydnor	NRR/DE/EMCB
JACQUAN WALKER	NRR/DRIP/RLEP
John Fair	NRR/DE/EMCB
Ramin Assa	ACRS
Peter J. KANG	NRR/DRIP/RLEP.
P T KUO	NRR/DRIP/RLEP
Ram Subramanian	NRR/DRIP/RLEP
John Boska	NRR/DLPM
George Georgiev	NRR/DE/EMCB
KIMBERLEY CORP	NRR/DRIP/RLEP
Omid Tabatabai	" " "
SAM LEE	NRR/DRIP/RLEP
MARIO G. CORA	NRR/DRIP/RLEP
Cornelius Holden	RRR/DE/EEIB
Rani Franovich	NRR/DRIP/RLEP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500th FULL COMMITTEE MEETING

MARCH 6-8, 2003

MARCH 6, 2003
Today's Date

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David Solorio	NRR/DRIP
Jim Medoff	NRR/DE/EMCB
Scott Burnell	OPA
DUC NGUYEN	NRR/DE/EEIB
TJ Kim	NRR/DRIP
Stewart Bailey	NRR/DE/EMEB
BARRY ELLIOT	NRR/DE/EMCB
K. Parczewski	NRR/DE/EMCB
Y.C. (Rene) Li	NRR/DE/EMCB
Chang Li	NRR/DSSA/SPLB
[illegible]	NRR/DRP/RLEP
J. Rajan	NRR/DE/EMEB
David C. Jeng	NRR/DE/EMEB
Bill Bateman	NRR/DE/EMCB
WM BURTON	NRR/DRIP/RLSB
Greg Hatchett	NRR/DSSA
Goutam Bagchi	NRR/DE
R.W. Borchardt	NRR
DAVID ALLSOPP	NRR
Gary Asher	DE/NRR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500th FULL COMMITTEE MEETING

MARCH 6-8, 2003

MARCH ~~7~~ 6, 2003
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
Michael Marshall	NRC\DLPM\PDII
VF Costello	NRC\RES\DET
STEVEN ARNDT	NRC/RES/DET
Christina Antonescu	NRC/RES/DET

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500th FULL COMMITTEE MEETING

MARCH 6-8, 2003

MARCH 6, 2003
Today's Date

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NAME	AFFILIATION
ERACH PATEL	EXELON
GAREY L. STATHES	EXELON
AL FULVIO	EXELON
FRED POWASKI	EXELON
RICHARD CIEMIELWICZ	EXELON
WH BOHLKE	EXELON
RICK JANATI	PA DEP/BRP
Dennis Dyckman	PA DEP/BRP
~~HANS ASOPR~~	~~DEL/NRC~~
ROGER HUSTON	Licensing Support Services
Stephen Page	NLT
Michael Markley	NMSS/IMNS
Tony McMurtray	NRC/Peach Bottom SRI
Altheia Wyche	SERCH Licensing/Bechtel
Mike Woods	Publishing Post Gazette
CS Hunter	AES/OERAB
Larry K. Mathews	SNC
KURT COZENS	NEI
Daniel Horner	McGraw-Hill
Richard Wood	ORNL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500th FULL COMMITTEE MEETING

MARCH 6-8, 2003

MARCH 6, 2003
Today's Date

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NAME	AFFILIATION
Paul Ewing	ORNL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500th FULL COMMITTEE MEETING

MARCH 6-8, 2003

MARCH 7, 2003
Today's Date

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NAME	NRC ORGANIZATION
Ronaldo V. Jenkins	NRR/NRLPO
Edward D Throm	NRR/DSSA/SPLB
BARRY ZALCMAN	NRR/DRIP/RLEP
Jocelyn Mitchell	RES/DSARE/RPEPWM
Leta Brown	NRR/DSSA/SPSB
KAZIMIERAS CAMPE	NRR/DSSA/SPSB
John Lappert	NRR/DRIP/RLEP
Thomas Kenyon	NRR/DRIP/RLEP
DAN BARSS	NRR/DIPM/IEHB/EPHP
Marsha Gamberoni	NRR/ADIP/NRLPO
JERRY Wilson	NRR/NRLPO
Charles Hinson	NRR/DIPM/IEHB/EPHP
Kevin Coyne	NRR/DIPM/IEHB/IQMS
Paul Prescott	NRR/DIPM/IEHB/IQMS
Jay Lee	NRR/DSSA/SPSB
Ramin Assa	ACRS
Stephen Klementowicz	NRR/DIPM
KEN HECK	NRR
Goutam Bagchi	NRR/DE
Stephen Koenick	NRR/NRLPO
Raj Tripathi	RES/DSARE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500th FULL COMMITTEE MEETING

MARCH 6-8, 2003

MARCH 7, 2003
Today's Date

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NAME	AFFILIATION
Russ Bell	NEI
George Zinke	Entergy
Eddie R Grant	Exelon / EXCEL Services
Sudesh Gambhir	OPPD / Fort Calhoun
Talmage Clements	Progress Energy
George Wrobel	Rochester Gas + Electric
David Wilson	Rochester Gas & Electric Corp
Michael Fallin	Constellation Nuclear Services
Michael H. Fletcher	Progress Energy
Jim Strnisha	NRC/OE

ITEMS OF INTEREST

500th ACRS MEETING

MARCH 6-8, 2003

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
500TH MEETING
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Page

SPEECH

- Remarks by Commissioner Dicus on High-Level Waste: Transportation, Disposal, and Yucca Mountain at the National Association of Regulatory Utility Commissioner's Conference, Washington, DC, February 24, 2003 1-5

MISCELLANEOUS

- Chairman Meserve was elected to National Academy of Engineering membership, February 14, 2003 6

- NRC to hold 15th Annual Regulatory Information Conference April 16-18, in Washington, DC., February 13, 2003. 7-16

- Statement Submitted by the U.S. Nuclear Regulatory Commission and Chairman Richard A. Meserve, to the Subcommittee on Clean Air, Climate Change, and Nuclear Safety Committee on Environment and Public Works, United States Senate, for the Oversight Hearing, February 13, 2003 17-25

- A letter from Samuel J. Collins, Director, Office of Nuclear Rector Regulation, to the Holders of Licenses for Operating Pressurized Water Reactors, Subject: "Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors", dated February 11, 2003. 26-42

- Final Significance Determination for a White Finding and Notice of Violation (NRC Inspection Report, Oconee Nuclear Station), to Mr. R.A. Jones, Vice President-Duke Energy Corporation, from Luis A. Reyes, Regional Administrator, February 7, 2003 43-46

- A letter from Luis A. Reyes, Regional Administrator, to Duke Energy Corporation, Mr. D. Jamil, Vice President, McGuire Nuclear Station, Subject: "Notice of Enforcement Discretion for Duke Energy Corporation Regarding McGuire Unit 1, February 11, 2003 47-48

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200
Washington, DC 20555-0001 E-mail: opa@nrc.gov
Web Site: www.nrc.gov

No. S-03-004

High-Level Waste: Transportation, Disposal, and Yucca Mountain

Remarks of

Greta Joy Dicus, Commissioner
United States Nuclear Regulatory Commission

at the

National Association of Regulatory Utility Commissioners Conference

February 24, 2003
Washington, DC

Good morning everyone. I am very pleased to be here today, and to have the opportunity to speak to you at the National Association of Regulatory Utility Commissioners' 2003 Winter Meeting. Let me begin by extending my appreciation to the National Association of Regulatory Utility Commissioners and its subcommittee on Nuclear Issues and Waste Disposal for hosting this very important conference, and to welcome all of you, who are participating in this week's scheduled events. With the number of participants and distinguished representatives here today, it is clear that our domestic utility regulators have a sincere and strong collective interest in the direction the nuclear industry is heading and changes that will be affecting that direction. As most of you are aware, over the last few years the nuclear industry, the public, and the Nuclear Regulatory Commission **(NRC)** have been working together to address many areas of common interests and concerns regarding the NRC's current regulatory system and oversight involvement, and ultimately, to mutually resolve these issues without compromising worker and public health and safety or environmental protection. From my perspective, I view this dialogue as necessary, constructive, and beneficial. When taking on any change, one must remember that the process must always begin with the end-in-mind. Maintaining the safety and health of the public and our stakeholders are that end. Before I begin with my discussion topic, **"High-Level Waste: Transportation, Disposal, and Yucca Mountain,"** I would like to share the results of some of the wide-sweeping changes the NRC has undergone in recent years. These changes

have and are continuing to be integrated in every program throughout the agency, including spent fuel transportation and high-level waste.

Throughout the years, the establishment and implementation of a sound infrastructure to systematically and safely construct, operate, and manage our licensed facilities has been a common goal of both the nuclear industry and the NRC, and overall, we have recognized and shared many successes. While realizing these successes in our programs and efforts, we also have faced a number of shortcomings from which we have gained a great deal of knowledge and experience. Experiencing those short comings, identifying their root-cause, and implementing timely and effective corrective actions have facilitated the maturing of the nuclear industry and have brought us to where we are today. Additionally, advancements in technology and operations, coupled with the industry's continued awareness for improving process safety, personnel training, and management accountability, and most of all, being able to demonstrate safe, effective, and predictable operations, has afforded the NRC the opportunity to assess its existing regulatory and oversight structure and programs to gain better perspective of our own efficiencies and effectiveness. After 25 years of existence and in concert with the nuclear industry, the NRC has undertaken changes of its own and is continually in search of improving its business operations. Over the last few years, the nuclear industry and the public have raised several ideas of interest to the NRC, and have effectively gained the Commission's attention. Ideas that have made us re-think how the NRC currently conducts its business operations, as well as the effectiveness in communicating, "who we are and what we do." As a result, we have put forth and continue to refine a number of initiatives that will allow for a more risk-informed, streamlined, and effective regulatory implementation and oversight process, which ultimately will improve the predictability and objectivity of our regulatory decisions. We have actually embraced many of these initiatives in the NRC Strategic Plan, which also includes performance goals and metrics, so that we can measure and evaluate our performance. Specifically, the four overarching performance goals that are a part of each of our regulatory programs include:

Maintaining safety, protection of the environment, and the common defense and security;

Increasing public confidence;

Making the NRC activities and decisions more effective, efficient, and realistic; and

Reducing unnecessary regulatory burden on stakeholders.

From rulemaking to standard review plan development, including licensing, inspection, and enforcement, the NRC's fuel-cycle, enrichment, reactor, spent-fuel, and waste management activities are being examined to further clarify our regulatory role and to streamline and improve the effectiveness and efficiency of our regulatory and oversight operations. In these same technical areas, the NRC continues its role with respect to external regulation of certain Department of Energy programs, such as high-level waste disposal, and mixed-oxide fuel-fabrication. Along these paths, we have also concentrated on establishing better lines of communication and more openly engaging and responding to both our stakeholders and the public. Regulatory formulation, decision making, and improving our objectivity and response time to submitted questions, comments, and petitions, are areas where the NRC has become more open, efficient, and transparent. From the examples provided, you can see that the NRC is part of and not separate from this changing environment. We are moving toward a less prescriptive, more risk-informed regulatory approach for all of our programs. However, I strictly emphasize, our primary mission, to **"protect public health, safety, and the environment"** and to **"promote common defense and security"** has not, and will not change.

Today, I would like to focus on the latter portion of the nuclear fuel cycle, and provide you with a panoramic look at some of the issues and our progress associated with the transportation of spent nuclear fuel and the disposal of that spent fuel and high-level waste in a licensed repository. First, let's look at spent fuel and high-level waste transportation.

Recently, the press and some stakeholders have raised concerns over the security and safety of transporting spent nuclear fuel and high-level waste. It is worth noting that the federal regulation of spent fuel transportation is shared between the U.S. Department of Transportation ***(DOT)*** and the NRC. DOT regulates the transport of all hazardous materials, including spent fuel, and has established regulations for shippers and carriers regarding, among other things, radiological controls, hazard communication, and training. The NRC regulates spent fuel transportation through a combination of safety and security requirements. We establish the design standards for the casks used to transport spent fuel, and review and certify the cask designs before their use. We also conduct an inspection and enforcement program, and review and approve physical security plans for spent fuel shipments. Over the last 30 years, more than a thousand shipments of commercially generated spent nuclear fuel have been made throughout the United States without causing any radiological releases to the environment or harm to the public. Most of these shipments occur between different reactors owned by the same utility to share storage space for spent fuel, or they may be shipped to a research facility to perform tests on the spent fuel itself. By any measure, this is an excellent safety record. However, this does not preclude the possibility that undetected weaknesses may exist. Because of this possibility, we are continually examining our transportation safety program. As an example, the NRC initiated a Package Performance Study to examine cask performance under severe impact and fire accident conditions. These studies will continue with full-scale cask testing to confirm computer models of cask response to severe accident conditions. As a part of this continuing examination, the staff also analyzes appropriate national transportation accidents, such as the 2001 train tunnel accident in Baltimore, to determine if our transportation requirements need to be modified. Finally, we are also sponsoring a study to update the evaluation of cask response to acts of sabotage. In the future, the Department of Energy will use NRC-certified casks when shipments of high-level waste begin. Next, let me tell you about the progress the NRC has made in preparation to receive a license application for the Yucca Mountain High-Level Waste Repository.

Last year, several events occurred that clear the way for NRC to receive an application from the Department of Energy for the construction of a permanent high-level waste repository at Yucca Mountain. A year ago this month (February), the Secretary of Energy made a formal recommendation in favor of the Yucca Mountain site as the repository for high-level waste. The President endorsed the Secretary's recommendation. Shortly thereafter, the Governor of Nevada gave notice of disapproval of that recommendation, as allowed by law, which transferred the issue to the Congress. In July, the Senate granted the final approval for the consideration of Yucca Mountain as the nation's high-level waste repository. The Department of Energy will now submit an application for construction to the NRC, which is currently planned for December 2004. Once we receive the application from the Department of Energy, the law provides NRC up to four years, including the completion of the administrative proceedings, to decide whether to grant the license.

In preparation to receive the application, the NRC has completed several important steps and actions. First, in late 2001, we issued the technical requirements a repository must meet in order to be licensed by NRC in 10 CFR Part 63. These are the risk-informed regulations encompassing both pre-closure and post-closure repository operations, and include the Integrated Safety Analysis approach for the pre-closure performance period, and the Total System Performance Assessment approach for the post-closure performance period. Second, the staff recently issued, as a draft for public comment, the

Yucca Mountain Review Plan, which when finished, will guide the agency's review of the forthcoming license application. This Plan is designed to ensure the quality and uniformity of our licensing reviews. We have received numerous comments from stakeholders and members of the public on the Plan, and the staff is currently working to address those comments and issue a revised Plan in its final form. Lastly, and to a large part, most important, the NRC has been working to involve stakeholders and members of the public in all aspects of the pre-licensing activities for Yucca Mountain to assure that expressed safety concerns will be addressed during the upcoming license review process.

As you can tell, a lot of work has been done, . . . but remember, we have yet to receive the Department of Energy's application! Once the NRC receives that application, the work will really begin. First, there will an acceptance review, which is a preliminary screening of the application to see whether it contains enough information to establish compliance with the regulations. This does not judge the technical adequacy of the application, but rather, is an evaluation to determine whether the submitted information, if found to be valid, would be sufficient to support granting a license -- in other words, whether the application is ready for the NRC staff to begin its detailed technical review. Even at this early stage of the review, the NRC will face some unique challenges. You may have heard about the 293 agreements in which the Department of Energy committed to provide additional information on technical issues associated with the Yucca Mountain repository. At the time of the acceptance review, the NRC will have to ensure that the information provided under these agreements results in a full and complete license application. All this must be accomplished within 90 days of receiving the application.

CLOSING REMARKS

In closing, you can see that past initiatives the NRC undertook to reform its regulatory approach have resulted in significant changes in all NRC programs, including spent nuclear fuel transportation and high-level waste. A lot has happened for these two programs in the last year. The NRC has accomplished a lot and is well positioned to meet the upcoming challenges as a license application for the construction of the Yucca Mountain repository is received. Throughout our past efforts, the NRC has included our stakeholders and the public and has made publically available, related rulemaking, licensing, and inspection information. The objective of this important step is not to try and please every individual, but to demonstrate that the NRC conducts its business operations in a fair, objective and independent manner, while ensuring adequate protection of public health and safety, and the environment. This approach helps to build public trust, gain public confidence, and demonstrates that the regulatory process is being carried-out in a transparent manner. Establishing and implementing formal public participation mechanisms, such as public meetings and workshops, addressing and reconciling public concerns in a fair manner and with an open mind, using plain language and terminology that are generally understood or recognized, not only help to establish public trust and confidence, but to maintain it as well. Clearly communicating our thoughts and processes to our stakeholders and the public, involving them through formal participation mechanisms, and demonstrating a general effort to be open to constructive criticism, are elements that are essential to effective and successful regulation and program implementation. These interactions provide early signals regarding dominant interests and concerns of those individuals and communities that will be directly or indirectly impacted by the action. By remaining receptive and responsive to those signals, the NRC continues to improve its credibility as an open-minded objective regulator, while at the same time, ensuring a realistic, predictable, and a stable regulatory framework, that is protective of the worker, the public, and the environment.

I hope that the insights and examples I've shared with you today about our spent fuel transportation and high-level waste programs provide a clear picture of some of the challenges the NRC will face, and the progress we've made to address those challenges. Thank you.

NRC NEWS
U.S. NUCLEAR REGULATORY COMMISSION
Office of Public Affairs Telephone: 301/415-8200
Washington, DC 20555-0001 E-mail: opa@nrc.gov
www.nrc.gov

No. 03-022 February 14, 2003

CHAIRMAN MESERVE ELECTED TO NATIONAL ACADEMY OF ENGINEERING

The National Academy of Engineering has elected Richard A. Meserve, Chairman of the Nuclear Regulatory Commission, to Academy membership.

Dr. Meserve was elected specifically "for leadership in the development of national and international programs for strengthening nuclear safety and safeguards," the National Academy declared in a statement today. Membership is considered among the highest professional distinctions accorded to an engineer.

Dr. Meserve was one of 77 newly elected members to the National Academy from the United States and abroad. He became Chairman of the NRC in October 1999 and has announced plans to leave the agency on March 31 to take over as president of the Carnegie Institution of Washington.

Dr. Meserve earned a B.A. from Tufts University in 1966, a J.D. from Harvard Law School in 1975, and a Ph.D. in applied physics from Stanford University in 1976.

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs Telephone: 301/415-8200

Washington, DC 20555-0001 E-mail: opa@nrc.gov

Web Site: www.nrc.gov

February 13, 2003

No. 03-021 February 13, 2003

NRC TO HOLD 15TH ANNUAL REGULATORY INFORMATION CONFERENCE APRIL 16 - 18 IN WASHINGTON, D.C.

The Nuclear Regulatory Commission staff will hold its 15th annual Regulatory Information Conference from Wednesday, April 16, to Friday, April 18, at the Capital Hilton Hotel, 16th and K Streets, NW, Washington, DC. The conference will open at 9:00 a.m. Wednesday and at 8:00 a.m. Thursday and Friday. There is no conference fee and the sessions will be open to the public.

The conference focuses on developing a better understanding of future trends for improved nuclear safety and on challenges shaping NRC policies and programs. "RIC 2003" will bring together NRC management, its regulated utilities, and other interested stakeholders for discussions regarding safety initiatives and regulatory trends, as well as issues such as fire protection and new and advanced reactor designs.

Hotel reservations should be made through the Capital Hilton Hotel at (202) 393-1000 or 1-800-HILTONS. Please mention "NRC Meeting April 16, 17, 18."

Interested parties may register electronically via the NRC website at this address:
http://www.nrc.gov/public-involve/conference-symposia/ric/
or contact JW Associates, 850 Sligo Avenue, Suite 300, Silver Spring, MD 20910; telephone 301-495-9471 or fax 301-495-5989.

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RIC 2003

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www.nrc.gov/public-involve/conference-symposia/ric

= Keynote Speakers

Updated: February 13, 2003 (1:00PM)

****DRAFT** **DRAFT** **DRAFT****

Session No.	Session Information	Contact Information
Wednesday, April 16, 2003		
W1	**Welcome** **Location: Presidential Ballroom** **9:00 - 9:45 am** **William D. Travers, Executive Director for Operations (EDO)/NRC** **Samuel J. Collins, Conference Chairman and Director (D), Office of Nuclear Reactor Regulation (NRR)/NRC**	
W2	**Plenary Session: Presentation/Q&A Session** **Location: Presidential Ballroom** **9:45 - 10:45 am** ***NRC Chairman***	
BREAK 10:45 - 11:15 am		
W3	**Plenary Session: Safeguards & Security** **11:15 am - 12:45 pm** **Location: Presidential Ballroom** **Panel Chair:** Samuel J. Collins D/NRR/NRC **Panel Co-Chair:** Roy P. Zimmerman D/Office of Nuclear Security and Incident Response (NSIR)/NRC **Panelists:** TBD	**Session Coordinators:** Richard P. Rosano RPR@nrc.gov 301-415-7118 Breck W. Henderson BWH@nrc.gov 301-415-7670 Prior to the conference, please share your questions, comments, thoughts or concerns with the session coordinators noted above.
LUNCH 12:45 - 2:15 pm		
W4	**Plenary Session: Stakeholder Forum** **2:15 - 3:15 pm** **Location: Presidential Ballroom** The Stakeholder Forum is your opportunity to discuss issues that may not be covered in other sessions or issues that cut across many different sessions. NRC staff and panel chairs from other sessions will be available to answer questions. We hope this forum will provide a meaningful and productive exchange of ideas, information, and comments.	**Facilitators:** Francis X. "Chip" Cameron FXC@nrc.gov 301-415-1642 Margaret A. Kotzalas MXK5@nrc.gov 301-415-2737 Mindy S. Landau MSL@nrc.gov 301-415-8703
BREAK 3:15 - 3:45 pm		
BREAKOUT SESSIONS (Set of four concurrent sessions) 3:45 - 5:15 pm		

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Updated: February 13, 2003 (1:00PM)

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Session No.	Session Information	Contact Information
Wednesday, April 16, 2003		
W5	**Reactor Oversight Process** **3:45 - 5:15 pm** **Location: Presidential Ballroom** **Panel Chair:** Bruce A. Boger D/Division of Inspection Program Management (DIPM)/NRR/NRC **Panelists:** TBD ***Theme: ROP - Past Insights and Future Challenges*** **Subtopics:** - SDP Task Group - Industry Trends - Process Enhancements - Public Confidence	**Boger:** BAB2@nrc.gov 301-415-1004 **Session Coordinator:** Paul W. Harris PWH1@nrc.gov 301-415-1169
W6	**Steam Generator & Materials Issues** **3:45 - 5:15 pm** **Location: Congressional/Senate Rooms** **Panel Chair:** Richard J. Barrett D/Division of Engineering (DE)/ NRR/NRC **Panelists:** Terence Chan, Section Chief, Materials & Chemical Engineering Branch (EMCB)/DE/NRR/NRC Paul Gunter, Nuclear Information and Resource Service Alex Marion, Director of Engineering, Nuclear Energy Institute (NEI) ***Theme: Closing the Loop on Materials Degradation Issues*** **Subtopics:** - Options available to assure safety in light of significant material degradation problems - Related NRC regulatory initiatives - Industry activities - Stakeholder concerns	**Barrett:** RJB3@nrc.gov 301-415-3298 **Session Coordinator:** Joseph E. Donoghue JED1@nrc.gov 301-415-1131
W7	**License Renewal / Power Uprates** **3:45 - 5:15 pm** **Location: Federal A/B Rooms** **Panel Chair:** John A. Zwolinski D/Division of Licensing Project Management (DLPM)/NRR/NRC **Panelists:** TBD	**Zwolinski:** JAZ@nrc.gov 301-415-1453 **Session Coordinator:** James W. Andersen JWA@nrc.gov 301-415-4010

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Updated: February 13, 2003 (1:00PM)

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Session No.	Session Information	Contact Information
Wednesday, April 16, 2003		
W8	**Allegations / Safety Conscious Work Environment** **3:45 - 5:15 pm** **Location: South American A/B Rooms** **Panel Chair:** Lisamarie L. Jarriel Agency Allegations Advisor NRR/NRC **Panelists:** TBD	**Jarriel:** LLJ@nrc.gov 301-415-8529 **Session Coordinator:** TBD
WEDNESDAY RECESS 5:15 pm		

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Session No.	Session Information	Contact Information
Thursday, April 17, 2003		
BREAKOUT SESSIONS (Set of four concurrent sessions) 8:00 - 9:30 am		
T1	**Building on the Davis-Besse Experience** **8:00 - 9:30 am** **Location: Presidential Ballroom** **Panel Chair:** Samuel J. Collins D/NRR/NRC **Panelists:** TBD	**Session Coordinator:** Darrell J. Roberts DJR@nrc.gov 301-415-1669
T2	**Emergent Technical Issues** **8:00 - 9:30 am** **Location: Congressional/Senate Rooms** **Panel Chair:** Brian W. Sheron Associate Director for Project Licensing & Technical Analysis (ADPT)/NRR/NRC **Panelists:** TBD	**Sheron:** BWS@nrc.gov 301-415-1274 **Session Coordinator:** Mel B. Fields MBF1@nrc.gov 301-415-3062
T3	**Risk Informed Activities** **8:00 - 9:30 am** **Location: Federal A/B Rooms** **Panel Chair:** Christopher I. Grimes Program Director/Policy & Rulemaking/ DIPM/NRR/NRC **Panelists:** TBD ***Theme: Current and Future Risk-Informed Applications*** **Subtopics:** - Brief Overview of NRC Goals / Ongoing Reactor Activities - South Texas Experience Implementing Categorization & Treatment - Risk-Informing ECCS Requirements - Implementing DG-1122 (PRA Quality) - Overarching Issues - Risk-Informed Environment - Improving Coherence Among Regulatory Activities	**Grimes:** CIG@nrc.gov 301-415-1282 **Session Coordinator:** Stewart L. Magruder SLM1@nrc.gov 301-415-3139

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= Keynote Speakers

Updated: February 13, 2003 (1:00PM)

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Session No.	Session Information	Contact Information
Thursday, April 17, 2003		
T4	**Spent Fuel Storage, Transportation, and Disposal** **8:00 - 9:30 am** **Location: South American A/B Rooms** **Panel Chair:** R. William Borchardt Associate Director for Inspection & Programs (ADIP)/NRR/NRC **Panel Co- Chair:** C. William Reamer Deputy Director (DD)/Division of Waste Management (DWM)/Office of Nuclear Material Safety and Safeguards (NMSS) **Panelists:** TBD	**Borchardt:** RWB1@nrc.gov 301-415-1284 **Reamer:** CBR@nrc.gov 301-415-6537 **Session Coordinator:** TBD
BREAK 9:30 - 10:00 am		
T5	**Plenary Session: Presentation/Q&A Session** **Location: Presidential Ballroom** **10:00 - 11:00 am** ***NRC Commissioner Greta Joy Dicus***	
BREAK 11:00 - 11:30 am		
T6	**NEI Luncheon** **Location: Congressional/Senate Rooms** **11:30 am - 12:30 pm** ***NEI Guest Speaker*** **12:30 - 1:00 pm**	
BREAK 1:00 - 1:30 pm		
T7	**Plenary Session: Presentation/Q&A Session** **Location: Presidential Ballroom** **1:30 - 2:30 pm** ***NRC Commissioner Edward McGaffigan Jr.***	
BREAK 2:30 - 2:45 pm		
BREAKOUT SESSIONS (Set of four concurrent sessions) 2:45 - 4:15 pm		
T8	**New Reactors / Advanced Reactors** **2:45 - 4:15 pm** **Location: Presidential Ballroom** **Panel Chair:** James E. Lyons D/New Reactor Licensing Project Office (NRLPO)/NRR/NRC **Panelists:** TBD	**Lyons:** JEL@nrc.gov 301-415-1126 **Session Coordinator:** Ronaldo V. Jenkins RVJ@nrc.gov 301-415-2985

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Session No.	Session Information	Contact Information
Thursday, April 17, 2003		
T9	**Operating Experience Assessments** **2:45 - 4:15 pm** **Location: Congressional/Senate Rooms** **Panel Chair:** David B. Matthews D/Division of Regulatory Improvement Programs (DRIP)/NRR/NRC **Panelists:** TBD **Subtopics:** - Status of NRC's Evaluation of Operating Experience Functions - INPO/NRC Interface - David-Besse Lessons Learned Task Force Report & Role of Operating Experience	**Matthews:** DBM@nrc.gov 301-415-1199 **Session Coordinator:** Terrence Reis TXR@nrc.gov 301-415-3281
T10	**Fire Protection** **2:45 - 4:15 pm** **Location: Federal A/B Rooms** **Panel Chair:** Suzanne C. Black DD/Division of Systems Safety and Analysis (DSSA)/NRR/NRC **Panelists:** TBD ***Theme: Fire Protection - Challenges and Solutions*** **Subtopics;** - Inspection Finding or Backfit - NFPA 805 - Fire Protection Improvement Plan - Advances in Fire Modeling and Fire PRA	**Black:** SCB@nrc.gov 301-415-3226 **Session Coordinator:** Vincent S. Klco VXK@nrc.gov 301-415-8348
T11	**International Issues** **2:45 - 4:15 pm** **Location: South American A/B Rooms** **Panel Chair:** Samuel J. Collins D/NRR/NRC **Panel Co-Chair:** Janice Dunn-Lee D/Office of International Programs (OIP)/NRC **Panelists:** TBD	**Session Coordinator:** Michael C. Cullingford MCC@nrc.gov 301-415-1276
BREAK 4:15 - 4:30 pm		
T12	**Plenary Session: Presentation/Q&A Session** **Location: Presidential Ballroom** **4:30 - 5:30 p.m.** ***NRC Commissioner Jeffrey S. Merrifield***	

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Session No.	Session Information	Contact Information
Thursday, April 17, 2003		
BREAK 5:30 - 5:45 pm		
T13	Plenary Session: International Experience 5:45 - 7:00 pm Location: Presidential Ballroom Panel Chair: TBD Panelists: TBD *Theme: International Experience with Operational Safety Assessment Review Teams (OSARTs) and International Regulatory Review Team (IRRT) Services*	Session Coordinator: Michael C. Cullingford MCC@nrc.gov 301-415-1276
THURSDAY RECESS 7:00 pm		

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Session No.	Session Information	Contact Information
Friday, April 18, 2003		
F1	**Plenary Session: Regulatory Trends** **8:00 - 8:45 am** **Location: Presidential Ballroom**	
BREAK 8:45 - 9:00 am		
BREAKOUT SESSIONS (Set of four concurrent sessions) 9:00 - 10:00 am		
F2	**Region II Breakout** **9:00 - 10:00 am** **Location: Presidential Ballroom** **Panel Chair:** Luis A. Reyes Regional Administrator (RA)/Region II **Panelists:** Victor M. McCree, Deputy Director, Division of Reactor Projects (DRP)/RII	**Reyes:** LAR1@nrc.gov 404-562-4410 **Session Coordinator:** Paul E. Fredrickson PEF@nrc.gov 404-562-4530
F3	**Region III Breakout** **9:00 - 10:00 am** **Location: Congressional/Senate Rooms** **Panel Chair:** James E. Dyer RA/Region III **Panelists:** Chris Bakken, Senior Vice President and Chief Nuclear Officer, D.C. Cook Nuclear Power Plant, American Electric Power (AEP)	**Dyer:** JED2@nrc.gov 630-829-9657 **Session Coordinator:** Sonia D. Burgess SDB2@nrc.gov 630-829-9752
F4	**Region IV Breakout** **9:00 - 10:00 am** **Location: Federal A/B Rooms** **Panel Chair:** Ellis W. Merschoff RA/Region IV **Panelists:** Gregg Overbeck, Senior Vice President, Arizona Public Power Company	**Merschoff:** EWM@nrc.gov 817-860-8225 **Session Coordinator:** Troy W. Pruett 817-860-8182 TWP@nrc.gov
F5	**Region I Breakout** **9:00 - 10:00 am** **Location: South American A/B Rooms** **Panel Chair:** Hubert J. Miller RA/Region I **Panelists:** R. William Borchardt, ADIP/NRR/NRC	**Miller:** HJM1@nrc.gov 610-337-5299 **Session Coordinator:** Tom R. Hipschman 610-337-5328 TRH@nrc.gov
BREAK 10:00 - 10:15 am		

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Session No.	Session Information	Contact Information
Friday, April 18, 2003		
F6	**Plenary Session: Inter-Regional** **10:15 - 11:15 am** **Location: Presidential Ballroom** **Panel Chair:** Samuel J. Collins D/NRR/NRC **Panelists:** James E. Dyer, RA/RIII Ellis W. Merschoff, RA/RIV Hubert J. Miller, RA/RI Luis A. Reyes, RA/RII	**Session Coordinator:** Darrell J. Roberts DJR@nrc.gov 301-415-1669
F7	**Plenary Session: Conference Closing** **11:15 - 11:45 am** **Location: Presidential Ballroom**	

STATEMENT SUBMITTED
BY THE
UNITED STATES NUCLEAR REGULATORY COMMISSION
TO THE
SUBCOMMITTEE ON CLEAN AIR, CLIMATE CHANGE,
AND NUCLEAR SAFETY
COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS
UNITED STATES SENATE

FOR THE
OVERSIGHT HEARING

SUBMITTED BY
RICHARD A. MESERVE
CHAIRMAN

Submitted: February 13, 2003

U.S. NUCLEAR REGULATORY COMMISSION
OVERSIGHT HEARING

Introduction

Mr. Chairman, and members of the Subcommittee, it is a pleasure to appear before you today with my fellow Commissioners to discuss the Nuclear Regulatory Commission's programs. We appreciate the past support that we have received from the Subcommittee and the Committee as a whole, and we look forward to working with you in the new Congress.

Mr. Chairman, I believe that fiscal year 2002 and the first four and one-half months of fiscal year 2003 have been marked by significant achievements by our agency in the face of great challenges. Let me enumerate a few of our achievements and the challenges. I will not go into great detail here because I submit a monthly report on our activities to you and our other authorization and appropriations subcommittees.

Security

Over the past 17 months, the Commission has undertaken a comprehensive review of safeguards and security programs, in close consultation with the Department of Homeland Security and other Federal agencies and with significant involvement by State agencies. Out of that review has come a series of interim compensatory measures (ICMs) to strengthen nuclear security at power reactors, Category I fuel cycle facilities, decommissioning reactors, research and test reactors, independent spent fuel storage facilities, the two gaseous diffusion plants, and the conversion facility, as well as in the transportation of spent fuel. Last August we put in place a five-tier threat advisory system compatible with the Homeland Security Advisory System, and we have used that system twice, including just last week, to improve security measures at our licensed facilities. We have issued Orders to strengthen our access authorization programs at power reactors. We have drafted proposed Orders to strengthen guard training and address guard fatigue. We have provided revised design basis threats (DBTs) for comment to other Federal agencies, the States and cleared industry personnel. We have been conducting enhanced table-top security exercises at our reactor facilities and will by the end of this month begin enhanced force-on-force exercises at these facilities. We will conduct force-on-force exercises on a 3-year cycle and have requested the resources to do this in our fiscal year 2004 budget. We have defined the actions that we need to take to ensure better control of high risk radioactive sources containing radioactive isotopes of the most concern for potential use in a radiological dispersal device.

In short, we have a comprehensive and aggressive program to enhance security. Nuclear facilities had very significant security before September 11th and that security has been greatly strengthened in the aftermath of the attacks.

Reactor Safety Programs

The past 17 months have seen the maturing of our new reactor oversight process.

We and most stakeholders believe that this new program is a significant improvement over our old inspection, enforcement and assessment processes. One of its strongest factors is its transparency and accessibility to members of the public. You will find on our web page performance indicators and inspection findings for every power reactor, as well as our current assessment of that reactor's overall performance. The transition to the new process has gone remarkably well, although it is still a work in progress on which we will make further improvements.

Overall the industry has performed very well. As of the end of 2002, there was one plant designated for the highest level of scrutiny, the Cooper plant in Nebraska, and one other plant, the Davis-Besse plant in Ohio, which is effectively being treated similarly under our Manual Chapter 0350 restart process. The Cooper plant has received significant attention from both our Region IV and headquarters staffs, and we are confident that it is on a path to resolving long-standing problems.

The Davis-Besse plant has been our greatest recent challenge. Mr. Chairman, you have followed this matter in detail and we have had meetings about this. But let me try to summarize the issues for your colleagues.

In February 2001, Duke Energy, the licensee at the Oconee Nuclear Station, conducted a vessel head inspection at its Unit 3. The vessel head is the very large steel structure that serves as the top of the reactor pressure vessel. Duke found circumferential cracking in several control rod drive mechanism penetration nozzles within the vessel head. The NRC staff immediately recognized the significance of these inspection findings -- the possibility of the ejection of the control rod drive mechanism -- and initiated a series of actions to ensure that any similar cracking would be promptly detected and repaired at other pressurized water reactors.

The Davis-Besse plant was one which the staff and the industry believed potentially had high susceptibility to such cracking. The staff's August 2001 bulletin called for such plants to conduct vessel head inspections by December 31, 2001, unless a later time could be justified. Davis-Besse petitioned for additional time (until April 2002) to complete the inspection. The staff initially planned to issue a shutdown Order, but decided in November 2001 to grant Davis-Besse a 46-day extension on the vessel head inspection requirement. When Davis-Besse shut down and conducted the required inspection, they found no through-wall circumferential cracking in the CRDM penetration welds, but, as they began to repair the axial cracks, they unexpectedly found a large cavity in the carbon steel of the reactor head. The cavity had been caused by corrosion due to the presence of boric acid.

This degradation was preventable, and the licensee's actions leading up to the discovery of the corrosion in March 2002 are unacceptable. This discovery has led to investigations, which are ongoing, of the licensee's actions. It has led us to focus large inspection resources on the facility as it seeks to restart its reactor with a new vessel head. It has also caused the Commission's staff to focus on mistakes the NRC made in dealing with boric acid corrosion issues in the 1990s. In this connection, an internal lessons-learned task force has made a comprehensive set of recommendations related to inspections, assessment of operating experience, NRC staff training and experience, and the assessment of stress corrosion cracking, boric acid corrosion, and barrier integrity requirements. The NRC staff is now developing action plans to implement the highest priority recommendations on an aggressive schedule. On Tuesday of this week NRC issued orders to all 69 pressurized water reactor licensees outlining much tougher vessel head inspection requirements than those previously required by our regulations and by industry codes.

The Commission staff has devoted significant resources to the Davis-Besse plant and to the broader issues raised by the Davis-Besse incident. Davis-Besse will only return to operation after the staff is convinced through intensive inspections both that the plant is physically ready to operate, and, perhaps more importantly, that the safety culture at the plant, which the licensee has identified as the main root cause of this event, is on the path to recovery.

Reactor Licensing Programs

Let me now turn to significant achievements in our reactor licensing programs. Four reactors -- Hatch 1 and 2 in Georgia and Turkey Point 3 and 4 in Florida -- have had their

licenses renewed to operate for twenty additional years. That brings the total of renewed licenses to ten. The staff currently has license renewal applications under review for twenty additional units. In every instance, the staff has met its timeliness goals in carrying out the safety and environmental reviews required by our regulations. This is truly a remarkable achievement. Today we expect almost all of the 104 reactors licensed to operate to apply for renewal of their licenses. The staff will continue to face an increasing workload in this area for the next several years as a bow wave of license renewal applications are submitted (echoing the bow wave of nuclear reactor construction in the 1970s).

The Commission also carefully reviews requests to raise the maximum power level at which a plant may be operated. These so-called power uprates range from requests for small increases based on better flowmeter technology, to large requests in the 15 to 20 percent range that require substantial hardware modifications at the plants. In all instances, staff must be satisfied that safety margins are maintained. In 2001 and 2002, the NRC approved 40 power uprates, which have added approximately 1800 megawatts electric to the Nation's generating capacity -- the equivalent of two large power plants. We expect a similar pace of uprates in the years ahead.

The staff has similarly processed a series of license transfer applications that have allowed significant consolidation within the nuclear power industry. Most of these transfers were processed within a 6-month target, and, with one exception, the NRC was not the last regulatory agency to grant the necessary approval.

The NRC staff is preparing for potential new reactor and reactor design applications. The staff is on target to conduct a timely review of the Westinghouse AP-1000 design certification. It is preparing to review three early site permit requests expected later this year. The staff is also in the pre-application phase in dealing with potential design certifications for several additional reactor designs. And the staff is making infrastructure improvements to prepare for a potential combined operating license request. These are resource-intensive activities, and our fiscal year 2004 budget request provides for the necessary significant growth to meet this challenge.

Materials Program

Mr. Chairman, the NRC in partnership with thirty-two Agreement States also conducts a comprehensive program to ensure the safe use of radiological materials in a variety of medical and industrial settings.

In the last 17 months, the Commission has completed a complex rulemaking on medical use of byproduct material -- a rulemaking on which there was significant interaction with the Congress. We now face the challenge of implementing that rule and assuring that compatible regulations are adopted in the thirty-two Agreement States.

The Commission has also been implementing a major rule change relating to large fuel cycle facilities. This rule requires the submission of an integrated safety assessment for all new licenses and license renewals that applies risk insights to the regulation of these facilities. Several major licensing reviews underway or soon to be submitted will test the new rule. Substantial new construction of fuel cycle facilities is planned in the near future, including a mixed oxide (MOX) fuel fabrication facility in South Carolina as part of the Department of Energy's program to dispose of excess weapons grade plutonium, as well as two new gas centrifuge enrichment facilities, one in Tennessee proposed by Louisiana Energy Services (LES) and one in Ohio proposed by U.S. Enrichment Corporation. The staff is also providing support to our Russian colleagues at Gosatomnadzor (GAN) regarding the licensing of a Russian MOX facility, which will have an identical design to the U.S. facility.

Nuclear Waste Programs

The Commission staff has made progress on a wide array of programs relating to the safe disposal of nuclear waste.

A central focus of this program is the preparation for the Department of Energy's (DOE's) application to construct a high-level waste repository at Yucca Mountain, Nevada. That application is currently expected in December 2004. Over the past year the staff has issued a draft Yucca Mountain Review Plan for public comment and has conducted numerous public meetings with DOE in anticipation of its application. Preparations are now underway for the conduct of the licensing proceeding, including the creation of an information technology system to handle the large number of complex documents that will be involved. This licensing proceeding will present the NRC with a formidable challenge. The technical issues involved will be substantial. Moreover, no single NRC decision or set of decisions since the response to Three Mile Island accident is likely to be scrutinized as closely as those concerning this one-of-a-kind facility.

Yucca Mountain is by no means the sole activity in our waste program. The Commission staff has a substantial effort underway in the area of dry cask storage of spent reactor fuel. Storage and transport casks continue to be certified. Independent Spent Fuel

Storage Installations (ISFSIs) continue to be licensed. The Atomic Safety and Licensing Board panel will soon issue its final decisions on the Private Fuel Storage (PFS) ISFSI in Utah. And the Surrey ISFSI in Virginia is the lead facility for ISFSI license renewal. Indeed, our workload related to ISFSIs and dry cask storage in general will increase substantially in the years ahead based on licensees' plans to adopt dry cask storage at their sites. We also have a major research program underway, the Package Performance Study (PPS), which will conduct full-scale integrity tests of both truck and rail casks under stringent conditions. The PPS test protocols are being issued for public comment.

The NRC staff is also continuing to make significant progress in decommissioning contaminated sites. The staff has identified several issues requiring Commission attention, particularly in the area of making the restricted release and institutional control provisions in our license termination rule work in practice.

Human Capital

The NRC is very dependent on a strong and capable work force for the effective execution of its activities. The Commission's human capital planning integrates strategies for finding and attracting new staff, and for promoting employee development, succession planning, and retention. In this connection, the Commission has developed and implemented a strategic workforce planning system to identify and monitor its human capital assets and needs. This includes the development of an agency-wide online skills and competency system which is used to identify gaps in needed skills and to address critical skills shortages; the development of a restructuring initiative to more closely align NRC's organizational structure with its human capital goals; and the development of a web-based vacancy announcement system that includes online application, rating, ranking, and referral features. The agency has also implemented two leadership competency development programs to select high-performing individuals and train them for future mid-level and senior-level leadership positions. In addition, the agency has continued to support its fellowship and scholarship programs and identified a significant number of highly qualified entry level candidates through participation in recruitment events and career fairs.

NRC is utilizing a variety of incentives to remain competitive with the private sector. So far we have been successful in attracting new staff, particularly at entry-levels. Nonetheless, it is likely to become more difficult for the Commission, as for many Federal agencies, to hire and retain personnel with the knowledge, skills, and abilities to conduct the safety reviews, licensing, research, and oversight actions that are essential to our safety mission. Moreover, the number

of individuals with the technical skills critical to the achievement of the Commission's safety mission is rapidly declining in the Nation, and the educational system is not replacing them. The maintenance of technically competent staff will continue to challenge governmental, academic, and industry entities associated with nuclear technology for some time to come.

Budget

The NRC has proposed a Fiscal Year 2004 budget of $626.1 million. This represents approximately a 7 percent ($41.1 million) increase over the Fiscal Year 2003 budget. This budget proposal will allow the NRC to continue to protect the public health and safety, promote the common defense and security, and protect the environment, while providing sufficient resources to address increasing personnel costs and increasing workloads. Approximately 25 percent of the budget growth is for personnel costs, primarily the pay raise that the President has authorized for Federal employees. The remaining increase serves several other needs. First, the NRC's proposed FY 2004 budget supports enhanced security efforts to protect public safety and security. Toward that end, the NRC is strengthening its safeguards and security programs for nuclear reactors, other NRC-regulated facilities, and radioactive materials. Second, the proposed budget addresses the growing interest in building new nuclear power plants. It strengthens the capability of the NRC to conduct reviews of new reactor designs and early site permit applications. Third, the budget enables the agency to process the increasing flow of applications for license renewal. Finally, with Presidential and Congressional approval of the proposed Yucca Mountain site for a HLW repository, the pace of the NRC's high-level waste program is increasing, and the proposed budget enables the NRC to continue its preparations for the license application that the U.S. Department of Energy plans to submit in late 2004. In short, we have important new work and there is strong justification for the budget increase that we seek.

Conclusion

Mr. Chairman, the NRC obviously has many important initiatives underway. This reflects the reality that we are in a time of striking change. Fortunately the NRC is up to the challenges before it.

I have had the privilege of leading the Commission for over 3 years. I can tell you that I am proud of the people with whom I work. They are dedicated to ensuring the safe use of nuclear technology for the benefit of the Nation. You will not find a more technically competent and hard-working workforce in the Federal government. Thanks to them the NRC has

accomplished many milestones during my tenure and will accomplish many more after I step down.

We appreciate the opportunity to appear before you today. My colleagues and I welcome the opportunity to respond to your questions.

February 11, 2003

EA-03-009

Holders of Licenses for Operating
Pressurized Water Reactors as listed in
Attachment to the Enclosed Order

SUBJECT: ISSUANCE OF ORDER ESTABLISHING INTERIM INSPECTION REQUIREMENTS FOR REACTOR PRESSURE VESSEL HEADS AT PRESSURIZED WATER REACTORS

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Order that modifies the current license for your facility. The Order requires specific inspections of the reactor pressure vessel (RPV) head and associated penetration nozzles at pressurized water reactors (PWRs). The Order applies to all Addressees listed in the Attachment to the enclosed Order.

The discovery of leaks and nozzle cracking at the Davis-Besse Nuclear Power Station and other PWR plants has made clear the need for more effective inspections of RPV heads and associated penetration nozzles. The current RPV head inspection requirements require visual examination of the insulated surface or surrounding area for signs of leakage. Such inspections are not sufficient to reliably detect circumferential cracking of RPV head nozzles and corrosion of the RPV head. Circumferential cracking of RPV head nozzles and corrosion of the RPV head pose a safety concern because of the possibility of a nozzle ejection or loss-of-coolant accident if the conditions are not detected and repaired. The enclosed Order establishes interim requirements to ensure that licensees implement and maintain appropriate measures to inspect and, as necessary, repair RPV heads and associated penetration nozzles.

To address the immediate concerns raised by the increasing discovery of problems with RPV heads at PWRs, the NRC issued a series of bulletins and other communications, including Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," dated August 9, 2002. Bulletin 2002-02 included the NRC request for licensees to provide information about their inspection programs and any plans to supplement existing visual inspections with volumetric and surface examinations. Most licensees responded to Bulletin 2002-02 with descriptions of their near-term inspection plans or with a schedule to submit such descriptions. Many of the licensees' responses to the Bulletin did not describe long-term inspection plans. Instead, licensees stated that they would follow guidance being developed by the industry-sponsored Materials Reliability Program (MRP).

The long-term resolution of this issue is expected to involve changes to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and will involve changes to the NRC regulations in 10 CFR 50.55a, "Codes and standards." The development of the NRC regulations, whether the rule adopts the ASME Code standards or defines separate requirements, will likely benefit from additional operating experience, continuing assessments, and research being conducted by the NRC and the MRP.

Although licensees' actions to date have provided reasonable assurance of adequate protection of public health and safety for the near-term operating cycles, the NRC deems it appropriate to establish a clear regulatory framework pending the revision of 10 CFR 50.55a. The enclosed Order imposes enhanced requirements for PWR licensees to inspect RPV heads and related penetration nozzles pending the revision of 10 CFR 50.55a. The requirements of the Order are immediately effective and are expected to remain in place until superceded by changes to 10 CFR 50.55a.

The enclosed Order requires responses and actions within specified time frames. Please contact your Licensing Project Manager regarding any issues related to compliance with the requirements in the enclosed Order, or if you have any other questions. In addition, the NRC is arranging a public meeting, tentatively scheduled for February 24, 2003, to discuss the Order with licensees and other stakeholders.

The enclosed Order has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosure: Order

cc w/encl: See next page*

* For a complete listing off all parties who received courtesy copies, see ADAMS Accession number ML030410402. In order to reduce size and weight of documents mailed, the enclosed plant mailing list is that associated with the specific facility noted.

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7590-01-P

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)
	) Docket Nos. (as shown in Attachment 1)
ALL PRESSURIZED WATER	) License Nos. (as shown in Attachment 1)
REACTOR LICENSEES	) EA-03-009

ORDER MODIFYING LICENSES (EFFECTIVE IMMEDIATELY)

I.

The Licensees identified in the Attachment to this Order hold licenses issued by the U.S. Nuclear Regulatory Commission (NRC or Commission) authorizing operation of pressurized water reactor (PWR) nuclear power plants in accordance with the Atomic Energy Act of 1954 and 10 C.F.R. Part 50.

II.

The reactor pressure vessel (RPV) heads of PWRs have penetrations for control rod drive mechanisms and instrumentation systems. Nickel-based alloys (e.g., Alloy 600) are used in the penetration nozzles and related welds. Primary coolant water and the operating conditions of PWR plants can cause cracking of these nickel-based alloys through a process called primary water stress corrosion cracking (PWSCC). The susceptibility of RPV head penetrations to PWSCC appears to be strongly linked to the operating time and temperature of the RPV head. Problems related to PWSCC have therefore increased as plants have operated for longer periods of time. Inspections of the RPV head nozzles at the Oconee Nuclear Station, Units 2 and 3 (Oconee), in early 2001 identified circumferential cracking of the nozzles above the J-groove weld, which joins the nozzle to the RPV head. Circumferential cracking above the

J-groove weld is a safety concern because of the possibility of a nozzle ejection if the circumferential cracking is not detected and repaired.

Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), which is incorporated into NRC regulations by 10 C.F.R. 50.55a, "Codes and standards," currently specifies that inspections of the RPV head need only include a visual check for leakage on the insulated surface or surrounding area. These inspections may not detect small amounts of leakage from an RPV head penetration with cracks extending through the nozzle or the J-groove weld. Such leakage can create an environment that leads to circumferential cracks in RPV head penetration nozzles or corrosion of the RPV head. In response to the inspection findings at Oconee and because existing requirements in the ASME Code and NRC regulations do not adequately address inspections of RPV head penetrations for degradation due to PWSCC, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated August 3, 2001. In response to the Bulletin, PWR Licensees provided their plans for inspecting RPV head penetrations and the outside surface of the heads to determine whether any nozzles were leaking.

In early March 2002, while conducting inspections of reactor vessel head penetrations prompted by Bulletin 2001-01, the Licensee for the Davis-Besse Nuclear Power Station (Davis-Besse) identified a cavity in the reactor vessel head near the top of the dome. The cavity was next to a leaking nozzle with a through-wall axial crack and was in an area of the reactor vessel head that the Licensee had left covered with boric acid deposits for several years. On March 18, 2002, the NRC issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," which requested PWR Licensees to provide information on their reactor vessel head inspection and maintenance programs, the material condition of their reactor vessel heads, and their boric acid inspection programs. In their responses, the Licensees provided information about their boric acid

inspection programs and their inspections and assessments to ensure that their respective plant did not have reactor vessel head degradation like that identified at Davis-Besse.

The experience at Davis-Besse and the discovery of leaks and nozzle cracking at other plants reinforced the need for more effective inspections of RPV head penetration nozzles. The absence of an effective inspection regime could, over time, result in unacceptable circumferential cracks in RPV head penetration nozzles or in the degradation of the RPV head by corrosion. These degradation mechanisms increase the probability of a more significant loss of reactor coolant pressure boundary through ejection of a nozzle or other rupture of the RPV head. The NRC issued Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," dated August 9, 2002, requesting that Licensees provide information about their inspection programs and any plans to supplement existing visual inspections with additional measures (e.g., volumetric and surface examinations). Licensees have responded to Bulletin 2002-02 with descriptions of their inspection plans for at least the first refueling outage following the issuance of Bulletin 2002-02 or with a schedule to submit such descriptions before the next refueling outage. Many of the Licensees' responses to Bulletin 2002-02 did not describe long-term inspection plans. Instead the Licensees stated that they would follow guidance being developed by the industry-sponsored Materials Reliability Program.

Inspections performed at several PWR plants in late 2002 found leakage and cracks in nozzles or J-groove welds that have required repairs or prompted the replacement of the RPV head. In addition, as discussed in NRC Information Notice 2003-02, "Recent Experience with Reactor Coolant System Leakage and Boric Acid Corrosion," issued January 16, 2003, leakage has recently occurred at some plants from connections above the RPV head and has required additional assessments and inspections to ensure that the leakage has not caused significant degradation of RPV heads.

III.

Based on recent experience, current inspection requirements in the ASME Code and related NRC regulations do not provide adequate assurance that reactor coolant pressure boundary integrity will be maintained for all combinations of construction materials, operating conditions, and operating histories at PWRs. The long-term resolution of RPV head penetration inspection requirements is expected to involve changes to the ASME Code and NRC regulations, specifically 10 C.F.R. 50.55a. Research being conducted by the NRC and industry is increasing our understanding of material performance, improving inspection capabilities, and supporting assessments of the risks to public health and safety associated with potential degradation of the RPV head and associated penetration nozzles. These research activities are important to the long term development of revisions to the ASME Code and NRC regulations.

The operating history of PWRs supports a general correlation among certain operating parameters, including the length of time plants have been in operation, and the likelihood of occurrence of PWSCC of nickel-based alloys used in RPV head penetration nozzles. Bulletin 2002-02 presented a three-tier categorization of susceptibility to RPV head penetration nozzle degradation based on reactor operating durations and temperatures. Licensees' responses to the Bulletin included an estimate of the effective degradation years (EDY) and the appropriate categorization of each plant into one of the three susceptibility categories. Each Licensee proposed an inspection plan for RPV head penetrations based upon the susceptibility to degradation via PWSCC (as represented by the value of EDY calculated for the facility). In addition, recent operating experience has shown that, under certain conditions, leakage from mechanical and welded connections above the RPV head can lead to the degradation of the low alloy steel head by boric acid corrosion.

Revising the ASME Code and subsequently the NRC regulations will take several years. The Licensees' actions to date in response to the NRC bulletins have provided reasonable assurance of adequate protection of public health and safety for the near term operating cycles, but cannot be relied upon to do so for the entire interim period until NRC regulations are revised. Additional periodic inspections of RPV heads and associated penetration nozzles at PWRs, as a function of the unit's susceptibility to PWSCC and as appropriate to address the discovery of boron deposits, are necessary to provide reasonable assurance that plant operations do not pose an undue risk to the public health and safety. Consequently, it is necessary to establish a minimum set of RPV head inspection requirements, as a supplement to existing inspection and other requirements in the ASME Code and NRC regulations, through the issuance of an Order to PWR Licensees.

It is appropriate and necessary to the protection of public health and safety to establish a clear regulatory framework, pending the development of consensus standards and incorporation of revised inspection requirements into 10 C.F.R. 50.55a, directly or through reference to a future version of the ASME Code. In order to provide reasonable assurance of adequate protection of public health and safety for the interim period, all PWR Licenses identified in the Attachment to this Order shall be modified to include the inspection requirements for RPV heads and associated penetration nozzles identified in Section IV of this Order. The NRC requirements imposed by this Order are based on the body of evidence available through February 2003. Continuing research and operating experience may support future changes to the requirements imposed through this Order. In addition, pursuant to 10 C.F.R. 2.202, I find that in the circumstances described above, the public health, safety, and interest require that this Order be immediately effective.

IV.

Accordingly, pursuant to Sections 103, 104b, 161b, 161i, 161o, 182, and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 C.F.R. § 2.202 and 10 C.F.R. Part 50, IT IS HEREBY ORDERED, **EFFECTIVE IMMEDIATELY**, THAT ALL LICENSES IDENTIFIED IN THE ATTACHMENT TO THIS ORDER ARE MODIFIED AS FOLLOWS:

A. To determine the required inspection(s) for each refueling outage at their facility, all Licensees shall calculate the susceptibility category of each reactor vessel head to PWSCC-related degradation, as represented by a value of EDY for the end of each operating cycle, using the following equation:

$$EDY = \sum_{j=1}^{n} \left\{ \Delta EFPY_j \exp\left[-\frac{Q_i}{R}\left(\frac{1}{T_{head,\,j}} - \frac{1}{T_{ref}} \right) \right] \right\}$$

where:

EDY	=	total effective degradation years, normalized to a reference temperature of 600°F
$\Delta EFPY_j$	=	operating time in years at $T_{head,j}$
Q_i	=	activation energy for crack initiation (50 kcal/mole)
R	=	universal gas constant (1.103×10^{-3} kcal/mole°R)
$T_{head,j}$	=	100% power head temperature during time period j (°R = °F + 459.67)
T_{ref}	=	reference temperature (600°F = 1059.67°R)
n	=	number of different head temperatures during plant history

This calculation shall be performed with best estimate values for each parameter at the end of each operating cycle for the RPV head that will be in service during the subsequent operating cycle. The calculated value of EDY shall determine the susceptibility category and the appropriate inspection for the RPV head during each refueling outage.

B. All Licensees shall use the following criteria to assign the RPV head at their facility to the appropriate PWSCC susceptibility category:

High (1) Plants with a calculated value of EDY greater than 12, OR

(2) Plants with an RPV head that has experienced cracking in a penetration nozzle or J-groove weld due to PWSCC.

Moderate Plants with a calculated value of EDY less than or equal to 12 and greater than or equal to 8 AND no previous inspection findings requiring classification as High.

Low Plants with a calculated value of EDY less than 8 AND no previous inspection findings requiring classification as High.

C. All Licensees shall perform inspections of the RPV head [1] using the following techniques and frequencies[2]:

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[1] This Order imposes additional inspection requirements. Licensees are required to address any findings from these inspections (i.e., perform analyses and repairs) in accordance with existing requirements in the ASME Code and 10 C.F.R. 50.55a. The NRC has issued guidance to address flaw evaluations for RPV head penetration nozzles (see letter dated November 21, 2001, from J. Strosnider, NRC, to A. Marion, Nuclear Energy Institute) and will, as necessary, issue revised guidance pending the updating of the ASME Code and related NRC regulations.

[2] The requirements of this Order are generally consistent with inspection plans that the NRC staff accepted in letters to some Licensees regarding their responses to Bulletin 2002-02. If the NRC staff has already accepted a specific variation from the requirements of this Order (e.g., inspections to less than two (2) inches above the J-groove weld), the Licensee may continue with the previously accepted inspection plan for the next refueling outage after issuance of this Order, provided that in its response to this Order the Licensee identifies all discrepancies between the requirements of this Order and the previously accepted inspection plan. Licensees proposing to deviate from the requirements of this Order for subsequent refueling outages shall seek relaxation of this Order pursuant to the procedure specified at the end of this Section.

(1) For those plants in the High category, RPV head and head penetration nozzle inspections shall be performed using the following techniques every refueling outage.[3]

(a) Bare metal visual examination of 100% of the RPV head surface (including 360° around each RPV head penetration nozzle), AND

(b) Either:

(i) Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove weld to the bottom of the nozzle and an assessment to determine if leakage has occurred into the interference fit zone, OR

(ii) Eddy current testing or dye penetrant testing of the wetted surface of each J-Groove weld and RPV head penetration nozzle base material to at least two (2) inches above the J-groove weld.

(2) For those plants in the Moderate category, RPV head and head penetration inspections shall be performed such that at least the requirements of 2(a) <u>or</u> 2(b) are performed each refueling outage. In addition the requirements of 2(a) <u>and</u> 2(b) shall each be performed at least once over the course of every two (2) refueling outages.

(a) Bare metal visual examination of 100% of the RPV head surface (including 360° around each RPV head penetration nozzle).

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[3] For repaired RPV head penetration nozzles that establish a new pressure boundary, the ultrasonic testing inspection shall include the weld and at least one (1) inch above the weld in the nozzle base material. For RPV head penetration nozzles or J-groove welds repaired using a weld overlay, the overlay shall be examined by either ultrasonic, eddy current, or dye penetrant testing in addition to the examinations required by (1)(b)(i) or (1)(b)(ii).

(b) Either:

(i) Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove weld to the bottom of the nozzle and an assessment to determine if leakage has occurred into the interference fit zone, OR

(ii) Eddy current testing or dye penetrant testing of the wetted surface of each J-Groove weld and RPV head penetration nozzle base material to at least two (2) inches above the J-groove weld.

(3) For those plants in the Low category, RPV head and head penetration nozzle inspections shall be performed as follows. An inspection meeting the requirements of 3(a) must be completed at least every third refueling outage or every five (5) years, whichever occurs first. If an inspection meeting the requirements of 3(a) was not performed during the refueling outage immediately preceding the issuance of this Order, the Licensee must complete an inspection meeting the requirements of 3(a) within the first two (2) refueling outages following issuance of this Order. The requirements of 3(b) must be completed at least once over the course of five (5) years after the issuance of this Order and thereafter at least every four (4) refueling outages or every seven (7) years, whichever occurs first.

(a) Bare metal visual examination of 100% of the RPV head surface (including 360° around each RPV head penetration nozzle).

(b) Either:

(i) Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove

weld to the bottom of the nozzle and an assessment to determine if leakage has occurred into the interference fit zone, OR

(ii) Eddy current testing or dye penetrant testing of the wetted surface of each J-Groove weld and RPV head penetration nozzle base material to at least two (2) inches above the J-groove weld.

D. During each refueling outage, visual inspections shall be performed to identify potential boric acid leaks from pressure-retaining components above the RPV head. For any plant with boron deposits on the surface of the RPV head or related insulation, discovered either during the inspections required by this Order or otherwise and regardless of the source of the deposit, before returning the plant to operation the Licensee shall perform inspections of the affected RPV head surface and penetrations appropriate to the conditions found to verify the integrity of the affected area and penetrations.

E. For each inspection required in Paragraph C, the Licensee shall submit a report detailing the inspection results within sixty (60) days after returning the plant to operation.[4] For each inspection required in Paragraph D, the Licensee shall submit a report detailing the inspection results within sixty (60) days after returning the plant to operation if a leak or boron deposit was found during the inspection.

F. In the response required by Section V of this Order, all Licensees shall notify the Commission if: (1) they are unable to comply with any of the requirements of Section IV, or (2) compliance with any of the requirements of Section IV is unnecessary. Licensees proposing to deviate from the requirements of this Order shall seek relaxation of this Order pursuant to the procedure specified below.

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[4] This reporting requirement supercedes the 30-day reports requested by NRC Bulletin 2002-02.

The Director, Office of Nuclear Reactor Regulation, may, in writing, relax or rescind any of the above conditions upon demonstration by the Licensee of good cause. A request for relaxation regarding inspection of specific nozzles shall also address the following criteria:

(1) The proposed alternative(s) for inspection of specific nozzles will provide an acceptable level of quality and safety, or

(2) Compliance with this Order for specific nozzles would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for relaxation associated with specific penetration nozzles will be evaluated by the NRC staff using its procedure for evaluating proposed alternatives to the ASME Code in accordance with 10 C.F.R. 50.55a(a)(3).

V.

In accordance with 10 C.F.R. § 2.202, the Licensee must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, within twenty (20) days of the date of this Order. Where good cause is shown, consideration will be given to extending the time to request a hearing. A request for extension of time in which to submit an answer or request a hearing must be made in writing to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. The answer may consent to this Order. Unless the answer consents to this Order, the answer shall, in writing and under oath or affirmation, specifically set forth the matters of fact and law on which the Licensee or other person adversely affected relies and the reasons as to why the Order should not have been issued. Any answer or request for a hearing shall be submitted to the Secretary, Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, ATTN: Rulemakings and Adjudications Staff, Washington, DC 20555. Copies shall also be sent

to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555; to the Assistant General Counsel for Materials Litigation and Enforcement at the same address; to the Regional Administrator for NRC Region I, II, III, or IV, as appropriate for the specific plant; and to the Licensee if the answer or hearing request is by a person other than the Licensee. Because of possible disruptions in delivery of mail to United States Government offices, it is requested that answers and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301-415-1101 or by e-mail to hearingdocket@nrc.gov and also to the Assistant General Counsel for Materials Litigation and Enforcement either by means of facsimile transmission to 301-415-3725 or by e-mail to OGCMailCenter@nrc.gov. If a person other than the Licensee requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 C.F.R. § 2.714(d).[5]

If a hearing is requested by the Licensee or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

Pursuant to 10 C.F.R. § 2.202(c)(2)(i), the Licensee may, in addition to demanding a hearing at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

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[5] The version of Title 10 of the *Code of Federal Regulations*, published January 1, 2002, inadvertently omitted the last sentence of 10 C.F.R. 2.714 (d) and paragraphs (d)(1) and (d)(2) regarding petitions to intervene and contentions. For the complete, corrected text of 10 C.F.R 2.714 (d), please see 67 FR 20884, April 29, 2002.

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Dated this 11th day of February 2003

EA-02-243 - Oconee 3 (Duke Energy Corporation)

February 7, 2003

EA-02-243

Duke Energy Corporation
ATTN: Mr. R. A. Jones
Site Vice President
Oconee Nuclear Station
7800 Rochester HighwayLSeneca, SC 29672

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT NO. 50-269/03-07, 50-270/03-07, 50-287/03-07, OCONEE NUCLEAR STATION)

Dear Mr. Jones:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC) final significance determination for a finding involving the inadequate installation of the "red" and "black" connectors on the pre-staged Unit 3 high pressure injection (HPI) pump emergency power supply cable from the auxiliary service water (ASW) switchgear. The finding was documented in NRC Inspection Report No. 50-269,270,287/02-15, dated November 21, 2002, and was assessed under the significance determination process as a preliminary White issue (i.e., an issue of low to moderate safety significance, which may require additional NRC inspection). The cover letter to the inspection report informed Duke Energy Corporation (DEC) of the NRC's preliminary conclusion, provided DEC an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for this finding.

In lieu of a regulatory conference, DEC submitted a written response dated January 10, 2003, which conveyed the results of their own root cause investigation. In summary, the response indicated that DEC did not identify a specific performance deficiency which led to the event, and that the associated NRC Inspection Report No. 50-269,270,287/02-15 provided information which differed with DEC's conclusions. The differing points included DEC's conclusions that the "red" connector was partially threaded on and would not have failed; the ground detection relays would protect the HPI pump motor from damage due to overheating caused by an electrical phase imbalance; and the other HPI pump would be available to provide flow to the reactor coolant system. DEC's response also documented corrective actions that have been taken and/or planned to preclude recurrence.

After considering the information developed during the inspection and the information DEC provided in its response, the NRC has concluded that the final inspection finding is appropriately characterized as White for Unit 3, in the mitigating systems cornerstone.

NRC Inspection Report No. 50-269,270,287/02-15 documented the performance deficiency as improper installation of the connectors (i.e., two parts of the connector not being screwed together). Although DEC could not identify a specific activity that resulted in the as-found condition of the connectors, the NRC concluded that the improper installation of the connectors was reasonably within the licensee's ability to foresee and correct and, as such, should have been prevented. By not ensuring proper cable connector installation, DEC failed to provide reasonable assurance that the pre-staged emergency power supply cable from the auxiliary service water switchgear to an HPI pump would be available for postulated events that involve High Energy Line

Break and/or tornado event recovery.

At the time of the inspection, the NRC interviewed the DEC technician who originally identified the issue. Based on this discussion, the NRC concluded that the "red" connector was not connected properly, and as such, the NRC did not have reasonable assurance that the connector would perform as designed. The information provided in DEC's response of January 10, 2003, did not provide sufficient information to cause the NRC to modify this conclusion.

Regarding DEC's conclusions that the ground detection relays would protect the HPI pump motor from damage due to overheating caused by an electrical phase imbalance, and the other HPI pump would be available to provide flow to the reactor coolant system, the NRC concluded that these factors were previously considered in its risk analysis (under "Recovery Considerations" attached to NRC Inspection Report No. 50-269, 270, 287/02-15). The information you provided does not alter the NRC's conclusion that, although there would be ample time, recovery actions would consist of complex, diagnostic actions of a troubleshooting nature without a clearly defined procedure and under high stress conditions. Accordingly, based on information developed during the inspection and in light of the above discussion, we conclude that the final risk significance of the inspection finding is appropriately characterized as White.

You have ten business days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Supplement 2.

The NRC also determined that a violation occurred involving the requirements of Technical Specification 5.4.1, in that DEC failed to properly implement the vendor's written instructions for attaching the electrical connectors on the pre-staged Unit 3 HPI pump emergency power supply cable from the ASW switchgear. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the Notice is considered escalated enforcement action because it is associated with a White finding. The NRC also notes that, as documented in Inspection Report No. 50-269,270,287/02-15, and confirmed in DEC's letter of January 10, 2003, corrective actions have been taken to preclude recurrence of a similar problem on either of the Oconee units' pre-staged HPI emergency power supply cables from the ASW switchgear. Additionally, it is understood that DEC is planning to inspect the normal cable connectors on all three units' HPI pump motors for tightness and apply tape to cables and elbows to prevent loosening.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved is adequately addressed on the docket in NRC Inspection Report No. 50-269,270,287/02-15, and in DEC's letter of January 10, 2003. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

Because plant performance for this issue has been determined to be in the increased regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this finding. We will notify you, by separate correspondence, of that determination.

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

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 50-269,270,287/03-07, and the above violation is identified as VIO 50-287/03-07-01: Inadequate Installation of Connectors on the Unit 3 HPI Pump Emergency Power Supply Cable from the ASW Switchgear. Accordingly, the associated apparent violation AV 50-287/02-15-01 is closed.

Should you have any questions regarding this letter, please contact Robert Haag, Chief, Reactor Projects Branch 1, at 404-562-4550.

Sincerely,

/RA by Bruce S. Mallett for/

Luis A. Reyes
Regional Administrator

Docket Nos: 50-269, 50-270, 50-287
License Nos: DPR-38, DPR-47, DPR-55

Enclosure: Notice of Violation

cc w/ encl:

Compliance Manager (ONS)
Duke Energy Corporation
Electronic Mail Distribution

R. Mike Gandy
Division of Radioactive Waste Mgmt.
S. C. Department of Health and
Environmental Control
Electronic Mail Distribution

Lisa Vaughn
Legal Department (PB05E)
Duke Energy Corporation
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County Supervisor of
Oconee County
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Peggy Force
Assistant Attorney General
N. C. Department of Justice
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NOTICE OF VIOLATION

Duke Energy Corporation
Oconee Nuclear Station
Unit 3

Docket Nos.: 50-287
License Nos.: DPR-55
EA-02-243

During an NRC inspection completed on November 21, 2002, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy), the violation is listed below:

> Technical Specification 5.4.1 requires that written procedures shall be established implemented and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Section 9, Procedures for Performing Maintenance, requires that maintenance which can affect the performance of

safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances.

Contrary to the above, prior to May 30, 2002, the licensee failed to adequately implement the vendor's written instructions for attaching the "Elastimold" electrical connectors on the "Black" and "Red" phases of the Unit 3 high pressure injection (HPI) pump emergency power supply cable from the auxiliary service water switchgear. Consequently, the "Elastimold" connectors on these two phases were found to be improperly installed (i.e., not screwed on), resulting in the possible loss of HPI pump function during a postulated high energy line break/tornado event recovery.

This violation is associated with a White SDP finding for Unit 3 only.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in NRC Inspection Report No. 50-269,270,287/02-15 and in DEC's letter of January 10, 2003. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region RII, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 7th day of February 2003

Privacy Statement | Site Disclaimer
Last revised Thursday, February 13, 2003

NOED-03-2-003 - McGuire 1 (Duke Energy Corporation)

February 11, 2003

Duke Energy Corporation
ATTN: Mr. D. Jamil
Vice President
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: NOTICE OF ENFORCEMENT DISCRETION FOR DUKE ENERGY CORPORATION REGARDING MCGUIRE UNIT 1 - (NOED NO. 2003-2-003)

Dear Mr. Jamil:

By letter dated February 7, 2003, you formally documented a verbal request made on February 6, 2003, for discretionary enforcement concerning McGuire Nuclear Station Unit 1 Technical Specification (TS) 3.7.5, Auxiliary Feedwater System and TS 3.6.3, Containment Isolation Valves. Your letter addressed the information previously discussed with the NRC in a telephone conference on February 6, 2003, at 3:00 p.m., EST. The principal NRC staff members who participated in that telephone conference included: B. S. Mallett, Deputy Regional Administrator, Region II (RII); H. N. Berkow, Director, Project Directorate II (LPD2), Office of Nuclear Reactor Regulation (NRR); L. R. Plisco, Director, Division of Reactor Projects (DRP), RII; H. O. Christensen, Deputy Director, Division of Reactor Safety (DRS), (RII); D. Terao, Section Chief, Mechanical and Civil Engineering Branch (EMEB), Division of Engineering (DE), NRR; P. E. VanDoorn, DRS, RII; R. E. Martin, Project Manager, LPD2-1, NRR; T. G. Scarbough, EMEB, NRR; and M. Franovich, Probabilistic Safety Assessment Branch, Division of Systems Safety and Analysis, NRR.

Because of a failure of an auxiliary feedwater (AFW) isolation valve, 1CA-42B, you stated that on February 7, 2003, at 8:26 a.m., McGuire Unit 1 would not be in compliance with TS 3.7.5, Limiting Condition for Operation (LCO), Action B.1, which requires restoration of a failed AFW train to operable status within 72 hours, and TS 3.6.3 LCO, Action C.1, which requires isolation of the affected penetration within 72 hours. As such, per TS 3.7.5, Action C.1, and TS 3.6.3, Action F.1, Unit 1 would be required to be in Mode 3 (Hot Standby) in 6 hours. You requested that a Notice of Enforcement Discretion (NOED) be issued pursuant to the NRC's policy regarding exercise of discretion for an operating facility, set out in Section VII.c, of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, and be effective for an additional 72 hours to complete a temporary modification, testing, and the restoration of valve 1CA-42B to operable status. An additional telephone conference was held on February 8, 2003, to discuss the results from your post-repair testing and additional technical information you obtained regarding the design of valve 1CA-42B. This letter documents our verbal issuance of the NOED for an additional 72 hours during the telephone conference on February 6, 2003. We understand that the condition causing the need for this NOED was corrected by you causing you to exit from TS 3.7.5 and TS 3.6.3 and from this NOED at 4:58 a.m., on February 9, 2003.

McGuire Unit 1 was operating at 100 percent power when the stem for AFW valve 1CA-42B failed during post-maintenance testing. While electrically stroking the valve in the open direction, the stem sheared due to an open limit switch malfunction in combination with an improperly set open torque switch setting. This caused the valve actuator to generate excessive loads in the open direction and failure of the valve stem. The safety basis in your NOED request letter included a discussion of compensatory measures and an evaluation of the potential impact on the public health and safety and the environment. Your evaluation concluded that the request for an

additional 72 hours to restore valve 1CA-42B to an operable status represented no net increase in radiological risk. In addition you concluded that no significant hazard consideration was involved. The compensatory measures that were in place until 1CA-42B was returned to service were integral to your no net increase in risk determination. These compensatory measures included: (1) no unnecessary maintenance or testing for valves 1CA-40B or 1CA-39, which provided an alternate means of isolating AFW flow to the "D" steam generator (S/G); (2) no unnecessary maintenance or testing that would render other Unit 1 AFW components unavailable; (3) no unnecessary work that would render risk-significant equipment unavailable or have a high likelihood of causing a plant trip or transient; (4) procedural guidance available to provide alternatives to manually isolate AFW flow should it be necessary; and (5) operators were briefed on existing procedures to address control of AFW following a loss of instrument air.

We have reviewed your request and agreed that maintaining the plant stable in Mode 1 for an additional 72 hours was preferable to the potential for a plant transient that could occur during a plant shutdown to Mode 3. Also, we agree that your compensatory measures, risk analysis, and safety basis considerations were adequate to demonstrate that the additional 72 hours would not involve a net increase in radiological risk and would not adversely affect public health and safety or the environment. Our decision was based primarily on the fact that the safety function for valve 1CA-42B to close and isolate flow to a S/G could be accomplished by two other valves and your repair plans minimizing the time that valve 1CA-42B would be closed.

Based on the above considerations, the staff concluded that Criterion B.2.1.1.a and the applicable criteria in Section C.4 to NRC Manual Chapter 9900, "Technical Guidance, Operation - Notices of Enforcement Discretion" were met. Criterion B.2.1.1.a states that for an operating plant, the NOED is intended to avoid unnecessary transients as a result of compliance with the license condition and, thus, minimize potential safety consequences and operational risks.

On the basis of the staff's evaluation of your request and the information provided in your letter dated February 7, 2003, we concluded that issuance of this NOED is consistent with the Enforcement Policy and staff guidance, and has no adverse impact on public health and safety. Therefore, it is our intention to exercise discretion not to enforce compliance with TS 3.7.5 and TS 3.6.3 for inoperable AFW valve 1 CA-42B for the period from February 7, 2003, at 8:26 a.m., (EST) until February 10, 2003, at 8:26 a.m., (EST).

However, as stated the Enforcement Policy, action will be taken, to the extent that violations were involved, for the root cause or causes that led to the request for this NOED.

Sincerely,

/RA/

Luis A. Reyes
Regional Administrator

Docket No.: 50-369
License No.: NPF-9

cc:
C. J. Thomas
Regulatory Compliance Manager (MNS)
Duke Energy Corporation
Electronic Mail Distribution

M. T. Cash, Manager
Regulatory Issues & Affairs
Duke Energy Corporation
526 S. Church Street
Charlotte, NC 28201-0006

Lisa Vaughn
Legal Department (EC11X)
Duke Energy Corporation

Exelon Nuclear

Peach Bottom Atomic Power Station
Plant License Renewal

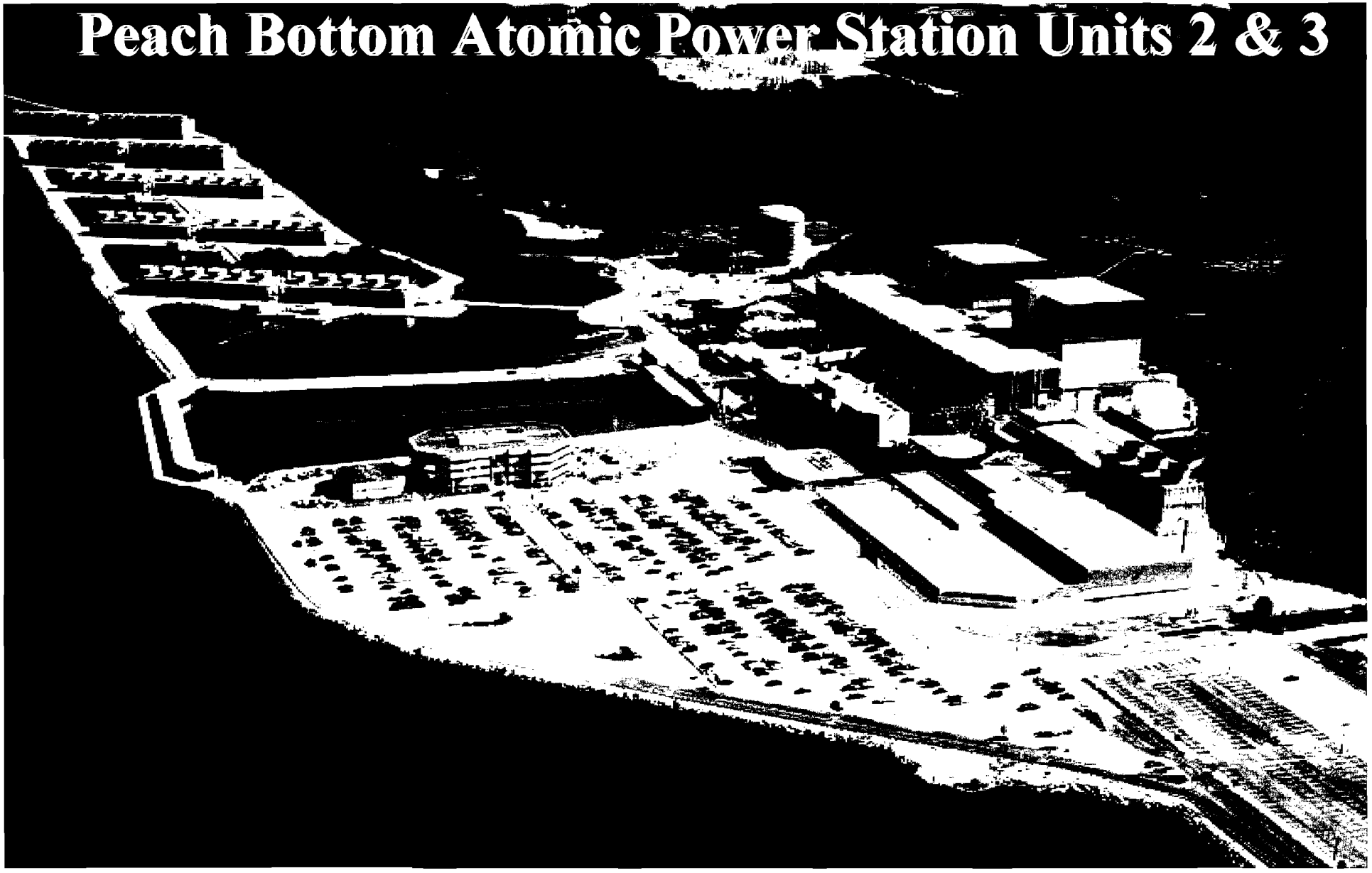
Presentation to
Advisory Committee on Reactor Safeguards

Bill Bohlke
Fred Polaski
March 6, 2003

Exelon SM

Nuclear

Peach Bottom Atomic Power Station Units 2 & 3



Plant Description

- PBAPS Units 2 & 3 are General Electric BWR4's with Mark I containment
- ~2340 Mwe total generating capacity
- Initial licenses expire in 2013 and 2014

Agenda

- Application Background
- Post License Look-ahead

Application Background

- July 2001 – Application submitted
- December 2002 – Supplemental Environmental Impact Statement issued
- February 2003 – Safety Evaluation Report issued
- February 2003 – Regional Administrator letter issued

Post License Look-Ahead

- UFSAR supplement will be implemented in the next update
- Aging Management Activity commitments as defined in UFSAR supplement will be completed
- Plant changes will be evaluated to ensure commitments are maintained

Implementing Commitments

- All commitments are documented in the Exelon commitment tracking system
- Commitment tracking numbers have been created for each Aging Management Activity
- Implementing procedures have been annotated for on-going commitments
- Future actions have been identified on commitment tracking evaluations

App A	EXISTING AGING MANAGEMENT ACTIVITIES	Commitment Tracking No.	Future Action**
1.1	Flow Accelerated Corrosion Program	T04338	
1.2	Reactor Coolant System Chemistry	T04336	
1.3	Closed Cooling Water Chemistry	T04336	
1.4	Demineralized Water and Condensate Storage Tank Chemistry Activities	T04336	
1.5	Torus Water Chemistry Activities	T04336	
1.6	Fuel Pool Chemistry Activities	T04336	
1.7	High Pressure Service Water Radioactivity Monitoring Activities	T04341	
1.8	Inservice Inspection (ISI) Program	T04343	E19
1.9	Primary Containment Inservice Inspection Program	T04345	
1.10	Primary Containment Leakage Rate Testing Program	T04325	
1.11	Inservice Testing (IST) Program	T04335	
1.12	Reactor Materials Surveillance Program	T04346	E08
1.13	Corrective Action Program	T04473	
1.14	Crane Inspection Activities	T04328	
1.15	Conowingo Hydroelectric Plant (Dam) Aging Mgmt Program		
1.16	Maintenance Rule Structural Monitoring Program	T04344	E20, E21
1.17	Electrical Cables not subject to 10 CFR 50.49 Environmental Qualification Requirement used in Instrumentation circuits	T04476	E14
	ENHANCED AGING MANAGEMENT ACTIVITIES		
2.1	Lubricating and Fuel Oil Quality Testing Activities	T04326, T04327	
2.2	Boraflex Management Activities	T04330	
2.3	Ventilation System Inspection and Testing Activities	T04324	
2.4	Emergency Diesel Generator Inspection Activities	T04332	
2.5	Outdoor, Buried, and Submerged Component Insp Activities	T04329	E23
2.6	Door Inspection Activities	T04331	E24
2.7	Reactor Pressure Vessel and Internals ISI Program	T04347	E15, E18
2.8	Generic Letter 89-13 Activities	T04333	E25
2.9	Fire Protection Activities	**T04342**	E05, **E06**, E22
2.10	HPCI and RCIC Turbine Inspection Activities	T04340	
2.11	Susquehanna Substation Wooden Pole Inspection Activity	T04348	E17
2.12	Heat Exchanger Inspection Activities	T04339	

App A	NEW AGING MANAGEMENT ACTIVITIES	Commitment Tracking No.	Future Action**
3.1	Torus Piping Inspection Activities	T04349	E07
3.2	FSSD Cable Inspection Activity	T04337	E16
3.3	Non-EQ Accessible Cable Aging Management Activity	T04475	E11
3.4	One-time Piping Inspection Activities	T04474	E26, E27, E28
3.5	Inaccessible Medium-Voltage Cables not subject to 10 CFR 50.49 Environmental Qualification Requirements	T04477	E12
3.6	Fuse Holder Aging Management Activity	T04478	E13
	TIME-LIMITED AGING ANALYSES ACTIVITIES		
4.1	Environmental Qualification Activities	T04412	
4.2	Fatigue Management Activities	T04411	E09, E29

** All future actions are tracked on commitment tracking evaluations on Action Request A1329928

Exelon Nuclear

*** PLANT COMMITMENT ***

COMMITMENT NBR: T04342 PRINT DATE: 11/26/02
TYPE : CT LAST UPDATE: 01/11/02
UNIT NUMBER : PB STATUS DATE: 06/01/01
COMMITMENT SATISFIED: NO STATUS: INIT REV #: 0000
TOPIC: PBLR FIRE PROTECTION ACTIVITIES

1. Document Information.

	TITLE	SECTION	DOCUMENT DATE	REV	CODE	NUMBER
SOURCE:	LICENSE RENEWAL APPL.				NLTN	010702
SOURCE:						
SOURCE:						
SOURCE:						

CM ESSENTIAL ELEMENT NBR: PBLR

2. System/Component Information.
CLASS: MODE: __ __ __ __ __ __
SYSTEM/TOPIC:

3. Statement of Commitment/Essential Element.
THE PEACH BOTTOM LICENSE RENEWAL APPLICATION (LRA), APPENDIX A AND B, IDENTIFIED THE ACTIVITIES THAT ARE CREDITED FOR COMPLIANCE WITH THE LICENSE RENEWAL RULE FOR AGING MANAGEMENT OF PASSIVE, LONG-LIVED COMPONENTS AND STRUCTURES WITHIN THE SCOPE OF LICENSE RENEWAL. AGING MANAGEMENT REVIEWS (AMRS) DEVELOPED IN SUPPORT OF THE PBAPS LRA PROVIDED FURTHER DETAIL REGARDING THE AGING MANAGEMENT ACTIVITIES IDENTIFIED IN THE LRA APPENDICES.
****
STATEMENT OF COMMITMENT:
THE FOLLOWING ACTIVITIES ARE CREDITED FOR AGING MANAGEMENT:
LRA APP. A, SECTION A.2.9 AND APP. B, SECTION B.2.9 CREDIT FIRE PROTECTION ACTIVITIES FOR MANAGING AGING EFFECTS IN THE FIRE PROTECTION SYSTEMS.
EXISTING FIRE PROTECTION ACTIVITIES PROVIDE FOR INSPECTIONS, SYSTEM MONITORING, AND/OR SYSTEM PERFORMANCE TESTS OF:
- FP SYSTEM PUMPS, PIPING, SPRINKLERS, AND VALVES,
- DIESEL DRIVEN FIRE PUMP FUEL OIL SYSTEM PUMPS, VALVES, PIPING AND TUBING
- BURIED FIRE MAIN PIPING AND VALVES,
- OUTDOOR FIRE HYDRANTS, HOSE CONNECTIONS, AND HOSE STATION BLOCK VALVES, AND
- FIRE BARRIER PENETRATION SEALS, FIRE BARRIER DOORS, AND FIRE WRAPS EXPOSED TO SHELTERED AND OUTDOOR ENVIRONMENTS.

THE SCOPE OF FP ACTIVITIES WILL BE ENHANCED TO:
- REQUIRE ADDITIONAL INSPECTION REQUIREMENTS FOR DELUGE VALVES IN THE POWER BLOCK SPRINKLER SYSTEMS
- PERFORM FUNCTIONAL TESTS OF SPRINKLER HEADS THAT HAVE BEEN IN SERVICE FOR 50 YEARS
- INSPECT DIESEL DRIVEN FIRE PUMP EXHAUST SYSTEMS,
- INSPECT DIESEL DRIVEN FIRE PUMP FUEL OIL SYSTEM FLEXIBLE HOSES, AND
- ADD SPECIFICITY FOR DETECTION OF LOSS OF MATERIAL IN REQUIREMENTS FOR INSPECTION OF FIRE DOORS.
- A ONE TIME TEST OF CAST IRON FIRE PROT. COMPONENT TO DETECT LOSS OF MATERIAL DUE TO SELECTIVE LEACHING

PERFORMANCE TESTING OF FIRE PROTECTION SYSTEMS AND COMPONENTS ARE BASED ON NFPA 24 STANDARDS.

Exelon Nuclear

*** PLANT COMMITMENT ***

COMMITMENT NBR: T04342 PRINT DATE: 11/26/02
TYPE : CT LAST UPDATE: 01/11/02
UNIT NUMBER : PB STATUS DATE: 06/01/01
COMMITMENT SATISFIED: NO STATUS: INIT REV #: 0000
TOPIC: PBLR FIRE PROTECTION ACTIVITIES

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THE FOLLOWING AGING EFFECTS ARE MANAGED:
- FP SYSTEM PIPING, SPRINKLERS, AND VALVES - VISUAL INSPECTIONS TO DETECT LOSS OF MATERIAL, CRACKING, AND FLOW BLOCKAGE
- FIRE PUMPS - VISUAL INSPECTIONS DURING CORRECTIVE MAINTENANCE ACTIVITIES TO DETECT LOSS OF MATERIAL AND FLOW BLOCKAGE
- DIESEL DRIVEN FIRE PUMP FUEL OIL SYSTEM PUMPS, VALVES, PIPING, AND TUBING - VISUAL INSPECTIONS TO DETECT LOSS OF MATERIAL AND CRACKING
- BURIED FIRE MAIN PIPING AND VALVES - CONTINUOUS MONITORING OF SYSTEM PRESSURE FOR LEAKAGE DETECTION
- BURIED MAIN PIPING AND VALVES, OUTDOOR FIRE HYDRANTS, HOSE CONNECTIONS, AND HOSE STATION BLOCK VALVES - FLOW TESTS TO DETECT BLOCKAGE AND COMPONENT DEGRADATION
- FIRE BARRIER PENETRATION SEALS, FIRE BARRIER DOORS, AND FIRE WRAPS - VISUAL INSPECTIONS TO DETECT CHANGE IN MATERIAL PROPERTIES, CRACKING, DELAMINATION, SEPARATION, AND LOSS OF MATERIAL. THIS WILL CONSIST OF INSPECTING SPECIFIED SAMPLE QUANTITIES OF PENETRATION SEALS EVERY 24 MONTHS WITH THE ENTIRE POPULATION BEING INSPECTED EVERY 16 YEARS.
- POWER BLOCK DELUGE VALVES - VISUAL INSPECTIONS TO DETECT LOSS OF MATERIAL, CRACKING, AND FLOW BLOCKAGE
- SPRINKLER HEADS IN SERVICE FOR 50 YEARS - FUNCTIONAL TESTING TO DETECT FLOW BLOCKAGE. THIS WILL CONSIST OF FUNCTIONAL TEST OF A REPRESENTATIVE SAMPLE
- DIESEL DRIVEN FIRE PUMP EXHAUST SYSTEM - VISUAL INSPECTIONS TO DETECT LOSS OF MATERIAL
- DIESEL DRIVEN FIRE PUMP FUEL OIL SYSTEM FLEXIBLE HOSES - VISUAL INSPECTIONS TO DETECT CHANGE IN MATERIAL PROPERTIES
- FIRE DOORS - VISUAL INSPECTIONS FOR LOSS OF MATERIAL
- CAST IRON FIRE PROT COMPONENTS - TESTED FOR LOSS OF MATERIAL DUE TO SELECTIVE LEACHING.
****
REFERENCES:
AMR-M01, GAS ENVIRONMENTS
AMR-M03, OIL/FUEL OIL ENVIRONMENT
AMR-M04, RAW WATER
AMR-M08, EXTERNAL ENVIRONMENTS
AMR-S06, HAZARDS BARRIERS AND ELASTOMERS

4. Implementing Activities.

___ 001 DOCUMENT CODE: M 037-001 SECTION: _____
DOC. DATE: _____ REV NUMBER: _____
ANNOTATED: Y IMPLMNT ORGN: P COMPLT: _____
ACTIVITY: CHECK VALVE MAINT MK 249

___ 002 DOCUMENT CODE: M 037-002 SECTION: _____
DOC. DATE: _____ REV NUMBER: _____
ANNOTATED: Y IMPLMNT ORGN: P COMPLT: _____
ACTIVITY: DIESEL FIRE PUMP MAINT.

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*** PLANT COMMITMENT ***

COMMITMENT NBR: T04342 PRINT DATE: 11/26/02
TYPE : CT LAST UPDATE: 01/11/02
UNIT NUMBER : PB STATUS DATE: 06/01/01
COMMITMENT SATISFIED: NO STATUS: INIT REV #: 0000
TOPIC: PBLR FIRE PROTECTION ACTIVITIES

==========

4. Implementing Activities.

___ 093 DOCUMENT CODE: ST O-37B-362-2 SECTION:
DOC. DATE: REV NUMBER:
ANNOTATED: Y IMPLMNT ORGN: P COMPLT:
ACTIVITY: SBGT FILTER TRAIN B DELUGE SYS FUNC.

___ 094 DOCUMENT CODE: ST O-37B-381-2 SECTION:
DOC. DATE: REV NUMBER:
ANNOTATED: Y IMPLMNT ORGN: P COMPLT:
ACTIVITY: UNDERGROUND FIRE MAIN FLOW TEST

___ 095 DOCUMENT CODE: ST O-37C-330-2 SECTION:
DOC. DATE: REV NUMBER:
ANNOTATED: Y IMPLMNT ORGN: P COMPLT:
ACTIVITY: MOTOR DRIVE FIRE PP FLOW RATE

___ 096 DOCUMENT CODE: ST O-37D-340-2 SECTION:
DOC. DATE: REV NUMBER:
ANNOTATED: Y IMPLMNT ORGN: P COMPLT:
ACTIVITY: DIESEL DRIVEN FIRE PP FLOW RATE

___ 097 DOCUMENT CODE: ST O-37D-370-2 SECTION:
DOC. DATE: REV NUMBER:
ANNOTATED: Y IMPLMNT ORGN: P COMPLT:
ACTIVITY: DIESEL DRIVEN FIRE PP OPERABILITY TEST

___ 098 DOCUMENT CODE: ST S-037-350-2 SECTION:
DOC. DATE: REV NUMBER:
ANNOTATED: Y IMPLMNT ORGN: P COMPLT:
ACTIVITY: SAFETY RELATED FIRE BRRIER DOOR INSP.

___ 099 DOCUMENT CODE: RT PLACE KEEPER SECTION: A1329928 E06
DOC. DATE: 06/15/12 REV NUMBER:
ANNOTATED: N IMPLMNT ORGN: PEPB COMPLT:
ACTIVITY: SPRINKLER HEADS IN SERVICE FOR 50 YEARS INSPECTION

___ 100 DOCUMENT CODE: RT PLACE KEEPER SECTION: A1329928 E05
DOC. DATE: 01/05/08 REV NUMBER:
ANNOTATED: N IMPLMNT ORGN: PEPB COMPLT:
ACTIVITY: FIRE PROT. COMP. SELECTIVE LEACHING TESTING

========== END OF COMMITMENT ==========

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5.2.2 Visually **EXAMINE** the following components for damage, excessive wear, cracks, corrosion, pitting, erosion, evidence of Asiatic Clams or other abnormalities. **RECORD** As Found in CREM. **CM-1**

- Cover
- Hanger
- Hanger pin
- Hanger plugs
- Disc
- Valve body including seat ring
- All bolting material

_____ WV

5.2.3 Using 0.003" Feeler Gauge, **CHECK** disc to valve body seat to verify 360° contact. **RECORD** results in CREM.

SAT ___ UNSAT ___ _____ WV

NOTE: If lapping is performed, valve body and internals shall be vacuumed and thoroughly cleaned.

5.2.4 If required, **LAP** disc seat to in-body seat ring.

Lapping performed: YES___ NO___ _____ WV

1. Using 0.003" Feeler Gauge, **CHECK** disc to valve body seat to verify adequate contact. **CONTINUE** lapping if required. **RECORD** results in CREM.

_____ WV

2. IF 360° contact CANNOT be obtained, THEN **CONTACT** Maintenance Supervision.

_____ _____
Person Contacted WV

5.2.5 **REPLACE** valve body seat ring as follows:

1. **UNTHREAD** seat ring (13) AND **REMOVE** from valve body (1).

2. **APPLY** Never Seez to threads AND **INSTALL** new seat ring (13) into valve body (1).

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5.3.8 **INSTALL** valve cover gasket (5) onto valve body (1).

5.3.9 **INSTALL** valve cover (4) onto valve body (1).

5.3.10 **LUBRICATE** threads with Never Seez AND **INSTALL** valve cover studs (3) and nuts (2).

DOUBLE VERIFICATION

5.3.11 **TORQUE** studs to 160 (150 to 170) ft-lbs.

_____ WV _____ DV

5.3.12 **INITIAL** AND **DATE** indicating subsection complete._____ WV

6.0 **RETURN TO NORMAL**

6.1 **ENSURE** the following:

- Tools and equipment are returned to designated storage location and work area is clean.
- M&TE numbers and date used are recorded in WO.
- Name and initials of each person who initialed procedure steps are entered on last page of this procedure.
- Procedure is complete and attached to WO.

_____ WV

7.0 **REFERENCES**

7.1 Plant Procedures/Drawings

7.1.1 A-C-33, Process for Verification of Quality

7.1.2 MAG-CG-301, Control of Bolting/Torquing/Tensioning

7.1.3 M-112-15, Walworth Company, 12" 175# I.B.B.M. Swing Check Valve

7.1.4 M-318, Sheet 3, P & I Diagram, Fire Protection System

7.2 Commitments

7.2.1 CM-1, Peach Bottom License Renewal Fire Protection Activities (T04342)

Exelon Nuclear

*** ACTION REQUEST ***

PAGE: 01

A/R TYPE : AT CTAI
REQUEST ORG : SL
REQUEST DATE: 01AUG01
REQUESTED BY: MOYER KL

A/R NUMBER : A1329928
A/R STATUS : PLNNED
STATUS DATE: 07SEP01
LAST UPDATE: 12FEB03
PRINT DATE : 19FEB03

==========

EVALUATION NBR: 06
EVALUATING ORG: PEPB
EVAL ASIGND TO: STURNIOLO (LRA) *T
EVAL REQUEST ORG: SEL
EVAL REQUESTOR: LAKE, FD
EVAL RETURNED BY:

ORIG DATE ASSIGNED:
EVAL DUE DATE: 15JUN12
DATE ASSIGNED: 21AUG01

EVAL STATUS : ACCEPT

IMPORTANCE CODE: 1 OEAP: SCHEDULE CODE: DATE FIXED: T

EVAL DESC: TESTING OF SPRINKLER HEADS IN SERVICE > 50 YEARS
THE PB LICENSE RENEWAL APPLICATION (LRA), APPENDIX A & B, FDL2 21AUG01
IDENTIFIED THE ACTIVITIES CREDITED FOR COMPLIANCE WITH FDL2 21AUG01
THE LICENSE RENEWAL RULE FOR AGING MANAGEMENT OF PASSIVE FDL2 21AUG01
LONG-LIVED COMPONENTS AND STRUCTURES WITHIN THE SCOPE OF FDL2 21AUG01
LICENSE RENEWAL. AGING MANAGEMENT REPORTS (AMR) DEVELOPED FDL2 21AUG01
IN SUPPORT OF THE LRA PROVIDED FURTHER DETAIL REGARDING FDL2 21AUG01
AGING MANAGEMENT ACTIVITIES IDENTIFIED IN THE LRA APPEND. FDL2 21AUG01
THE FOLLOWING ACTIVITY IS CREDITED FOR AGING MANAGEMENT: FDL2 21AUG01
LRA APPENDIX A, SECTION A.2.9, AND APPENDIX B, SECTION FDL2 21AUG01
B.2.9, COMMITTED TO PERFORMING FUNCTIONAL TESTING FOR FDL2 21AUG01
FLOW BLOCKAGE OF SPRINKLER HEADS IN SERVICE FOR 50 YEARS. FDL2 21AUG01
TESTING OF SPRINKLER HEADS WITH LONG SERVICES LIVES WILL FDL2 21AUG01
BE REQUIRED DURING THE EXTENDED PERIOD OF PLANT SERVICE FDL2 21AUG01
TO MEET THE REQUIREMENTS OF NFPA 25, SECTION 2-3.1.1. FDL2 21AUG01
. FDL2 21AUG01
A REPRESENTATIVE SAMPLE OF SPRINKLER HEADS THAT HAVE BEEN FDL2 21AUG01
IN SERVICE FOR 50 YEARS ARE REQUIRED TO BE FUNCTIONALLY FDL2 21AUG01
TESTED FOR FLOW BLOCKAGE AND VERIFICATION OF PROPER OPER- FDL2 21AUG01
ATION. FDL2 21AUG01
. FDL2 21AUG01
THIS PROGRAM MUST BE IMPLEMENTED PRIOR TO ANY SPRINKLER FDL2 21AUG01
HEADS ACHIEVING 50 YEARS OF SERVICE LIFE. FDL2 21AUG01
THIS COMMITMENT IS CONTAINED IN T04342, PBLR FIRE PROTECT FDL2 21AUG01
ION ACTIVITIES. ONCE THE PROGRAM IS DEVELOPED AS REQUIRE FDL2 21AUG01
BY THIS A/R EVAL, THE T04342 COMMITMENT MUST BE UPDATED FDL2 21AUG01
TO REFLECT THE NEW PROCEDURE/PM NUMBER. THE COMMITMENT FDL2 21AUG01
CAN ONLY BE DELETED OR CHANGED BY FOLLOWING THE GUIDANCE FDL2 21AUG01
IN LS-AA-110, COMMITMENT MANAGEMENT. FDL2 21AUG01
REFERENCE: AMR M04, RAW WATER FDL2 21AUG01

==========END OF ACTION REQUEST==========

Evaluating Plant Changes

- Configuration change control procedures will be updated to address license renewal requirements
  - Included will be physical plant modifications, operational changes, and changes to CLB
- UFSAR update process will address 10 CFR 54.37(b) requirements

Maintaining Records

- Exelon Records Management system will retain documents, drawings, and aging management reviews that support the Peach Bottom license renewal application
- Future self-assessments and NRC inspections will validate commitment management

3

Peach Bottom License Renewal Safety Evaluation Report

Staff Presentation to ACRS

Project Manager: David L. Solorio

Office of Nuclear Reactor Regulation

March 6, 2003

Agenda

- Opening Remarks.....G. Leitch
- Staff Introduction.....P. Kuo
- Applicant Overview.....F. Polaski
- Safety Evaluation Report.....D. Solorio

Background

- July 2, 2001, Exelon submitted LRA
- Initially a 25 month review → shortened to 22 months
- September 16, 2002 issued safety evaluation report with open items
- October 30, 2002, ACRS subcommittee presentation on SER with open items
- February 6, 2003 issued safety evaluation report
- Three inspections completed and Regional Administrator letter issued February 14, 2003

Safety Evaluation Report Topics

- Scoping of housings
- Scoping of non-safety related equipment
- Status of BWRVIPs (license conditions)
- Torus inspection program
- Diesel generator fuel oil tank
- SGTS buried piping
- Fuse holders
- Top guide inspections
- Inspection program (future commitments)
- December 21, 2002, event

Chapter 2: Scoping and Screening

- Scoping of housings
  - Open Items 2.3.2.7.2-2, 2.3.3.8.2-1, 2.3.3.8.2-2 Scoping of HVAC components (housings)
- Scoping of non-safety related components
  - Open Item 2.3.3.19.2-1

Status of BWRVIPs for Review

Component	Reference	SER Date	Accession # for SER
Reactor pressure vessel components	BWRVIP-74	10/18/01	ML012920549
Vessel shells	BWRVIP-05	03/07/00	ML003690281
Shroud support and attachments	BWRVIP-38	03/01/01	ML010600211
Shroud	BWRVIP-76	the end of 2003	N/A
Nozzle safe ends and piping	BWRVIP-75	09/15/00	ML003751105
Core support plate	BWRVIP-25	12/07/00	ML003775989
Core ΔP/SLC line and nozzle	BWRVIP-27	12/20/99	ML993630179
Core spray, jet pump riser brace, and other attachments	BWRVIP-48	01/17/01	ML010180493
Core spray lines and spargers	BWRVIP-18	12/07/00	ML003775973
Top guide	BWRVIP-26	12/07/00	ML003776119
Jet pump assemblies	BWRVIP-41	05/01/01	ML011570560
CRDH stub tubes and guide tubes, ICM housing guide tubes and penetrations	BWRVIP-47	12/07/00	ML003775765
Instrument penetrations	BWRVIP-49	03/31/02	NUDOCS
Integrated Surveillance Program Plan	BWRVIP-78	02/01/02¶ (40 years)	ML020380691
Intergrated Surveillance Program: Implementation Plan	BWRVIP-86	02/01/02¶ (40 years)	ML020380691

License Conditions

- Integrated surveillance program
  - BWRVIP resolution or propose plant specific solution
- Core shroud inspection and flaw evaluation
  - Review of BWRVIP-76 not complete
  - BWRVIP resolution or propose plant specific solution
- Incorporate UFSAR Summaries into UFSAR
  - Update in accordance with 50.71(e)
- Implementation date for future inspections
  - Prior to extended period of operation

Torus

- SER Section 3.0.3.7 credits primary containment in-service inspection program for inspecting torus (also reviewed during AMR inspection)
- Torus water chemistry activities (SER Section 3.0.3.5) also credited for aging management
- Inspections are performed in accordance with subsection IWE of ASME Section XI
- Inspections confirmed wear rate is low
- Torus condition monitored during subsequent inspection

Diesel Fuel Oil Tanks

- SER Sections 3.0.3.18 and 3.3.16
- Credit lubricating and fuel oil quality testing activities (SER Section 3.0.3.18)
- Inspect once every 10 years
- Last inspection in 1996 confirmed no wear

Standby Gas Treatment

- SER Section 3.2.7
- Aging effect for ducting not expected because ambient air inside and outside of ducting is of similar temperature
- Should not have water leakage into SGTS unit (3 locked series valves upstream of deluge nozzles)
- Carbon steel SGTS piping managed by outdoor, buried, and submerged component inspection activities (SER Section 3.0.3.13)
- Inspection reviewed SGTS condition and records
- Technical specification surveillance (filter and nozzles)

Chapter 3: AMR Results

- Fuse Holders
  - CI 3.6.2.2.2.-1
  - Applicant committed to implement resolution of interim staff guidance ISG-04

Chapter 4: Time Limited Aging Analysis

- Top guide (Open Item 4.5.2-1)
  - Top guide beams inspections performed at the same time control rod drive housing (CRDH) guide tubes inspected (in high fluence region)
  - Sample size and frequency for CRDH guide tubes is a 10% sample of the total population within 12 years; one half (5%) to be completed within six years
  - Enhanced visual examination utilized to examine for cracks
  - Implemented prior to extended period of operation

Future Inspections

- Manual Chapter 2516, Inspection Procedure 71003
  - Post-approval site inspection for license renewal
  - Verify outstanding commitments identified at the time a renewal license was granted (as documented in the plant-specific attachment to this IP) have been met
  - Prior to extended period of operation

Unit 2 Reactor Trip

- December 21, 2002, Unit 2 tripped
- Cause attributed to card failure
- Special inspection conducted
- Review of LER (02-001) did not identify any failures of passive components
- Special inspection report in preparation

(4)

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

October 12, 2001

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

SUBJECT: THE REVISED REACTOR OVERSIGHT PROCESS

Dear Chairman Meserve:

During our 485th meeting on September 5-7, 2001, the Advisory Committee on Reactor Safeguards met with representatives of the NRC staff to discuss the revised Reactor Oversight Process (ROP). We continued our deliberations during our 486th meeting on October 4-6, 2001. This matter was also discussed during meetings of the ACRS Plant Operations Subcommittee on December 6, 2000, May 9, 2001, and July 9, 2001. In addition, the ACRS Subcommittees on Plant Operations and Fire Protection held meetings with licensees on June 13, 2000, and June 27, 2001, and held meetings with Regions III and IV on June 14, 2000, and June 28, 2001, respectively. During our review, we had the benefit of the documents referenced.

BACKGROUND

The ROP utilizes the results of performance indicators (PIs) and baseline inspection findings to determine the appropriate regulatory action to be taken in response to a licensee's performance. The escalation of the regulatory responses is specified in the action matrix, which the staff developed as part of the ROP. This ROP has been in effect for nearly all licensees for about one year. The staff has conducted an assessment of the state of the ROP and recognizes that it is still a process in development.

The ACRS has previously commented on various aspects of the ROP and provided recommendations to the staff regarding potential process improvements. We remain convinced that the ROP is more objective and understandable than the former oversight process and represents a significant improvement. This report discusses some specific questions that the Commission raised to the ACRS, and offers some additional thoughts on potential improvements in the ROP.

In the Staff Requirements Memorandum dated April 5, 2000, the Commission requested the ACRS to:

(1) Review the use of PIs in the ROP to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety.

(2) Review the initial implementation of the significance determination processes (SDPs), and assess the technical adequacy of the SDP to contribute to the ROP.

The current PIs do provide meaningful insight into plant performance. However, there is a need to redefine the thresholds for some of the PIs to provide better input to the ROP. In particular, the numerical values for the white/yellow and yellow/red thresholds for the initiating event and mitigation system PIs are not useful and should be revised. The color bands for the PIs and SDPs associated with all the cornerstones have similar implications with respect to agency action and, therefore, the thresholds should be commensurate with their respective safety significance.

The most immediate and pressing need for the ROP is to improve the SDP tools. Some SDPs are incomplete and, in cases such as fire protection, overly subjective. The technical adequacy of the risk-based SDPs depends on the availability and quality of a relevant probabilistic risk assessment (PRA). Thus, the SDP for at-power situations provides meaningful risk information. For routine findings that are predominantly of very low, low, and moderate safety significance, the process is probably adequate. The threshold values for the risk-based SDPs are appropriate.

We continue to believe that a documented review of the SDP worksheets and SPAR models (as well as the underlying SAPHIRE computer code) is essential to public confidence in the ROP.

An SDP based on low-power and shutdown PRAs or other shutdown management tools is needed to characterize findings during these modes of operation. In addition, the fire protection SDP involves very qualitative inputs to a quantification process of uncertain pedigree. This SDP is probably useful for its intended purpose, however, it may be hard to defend and justify to the public. Even though this SDP calculates the change in core damage frequency (CDF), the SDP is really intended to provide an indication of the degradation of defense in depth for fire protection as defined in 10 CFR Part 50, Appendix R.

Presently, concurrent performance deficiencies are assessed collectively, as applicable, to determine the total change in CDF, but each performance deficiency is assigned a color individually. There may be instances in which conclusions could be altered if the results are considered collectively, and thus such collective results should be considered in the action matrix.

DISCUSSION

An important premise of the ROP is that there should be a graded regulatory response to inspection findings and PI results. Although a graded response to oversight findings is a desirable attribute, the inputs to the action matrix that implements this response must be produced in a way that justifies the resulting response. This is especially true for the right-hand columns of the matrix which could lead to severe regulatory responses.

The current ROP uses different technical bases to establish the thresholds for the PIs and inspection findings. In particular:

- On the basis of its review of recent operating history, the staff set the green/white thresholds for the PIs for initiating events and mitigating systems at the 95th percentile of peer performance for the given PI. By contrast, the staff based the white/yellow and yellow/red thresholds on an assessment of the value of a PI corresponding to increases in CDF of 10^{-5} and 10^{-4} per reactor year, respectively.

- The staff set the PI thresholds for barriers, emergency preparedness, occupational radiation safety, public radiation safety, and physical protection by considering technical specification limits, the number of noncompliances with regulatory requirements, and other absolute measures.

- The staff based the green/white, white/yellow, and yellow/red thresholds for SDP results on increases in CDF of 10^{-6}, 10^{-5}, and 10^{-4} per reactor year, respectively. This is true for the initiating event, mitigating system, and fire protection cornerstones. The other SDPs do not have a PRA basis and take a deterministic and defense-in-depth approach to establish thresholds for safety significant issues.

These different bases for defining the various thresholds raise questions regarding the kinds of information that the PIs and SDPs provide and the consistency of the meaning of the thresholds across the PIs and SDPs. These different thresholds are based on expert judgment that the degradation in performance associated with each color band is appropriately linked to a corresponding regulatory response[1].

It is from this viewpoint that we believe it is necessary to reconsider the definitions of the white/yellow and yellow/red thresholds for initiating events and mitigating systems, which as we noted above were based on an attempt to assess the value of a PI corresponding to increases in CDF.

We have noted previously that it is difficult to generically assess the risk impact of changes in a PI. The associated changes in risk tend to depend strongly on plant-specific features. This approach, however, has a deeper, more intractable flaw. Specifically, it focuses on the change in CDF that results from changes in a single, isolated parameter assuming that all other factors that can affect CDF remain constant. A realistic assessment of the change in CDF cannot be related to the change in a single PI. Thus, in some cases, the use of this approach to select white/yellow and yellow/red thresholds has led to values for these thresholds that, in our judgment and that of many of the staff and the industry, are too high to be meaningful. Regulatory attention would increase at much lower levels.

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[1] The color bands for the ROP are called "constructed scales" in decision analysis. Ensuring the consistency of the bands of these scales is what decision analysts commonly call "performing sanity checks," and such checks are among the most important steps in a decisionmaking process. In our report on the NRC Safety Research Program (NUREG-1635, Vol. 4), we recommended that the staff initiate a program of research to investigate how best to use formal decisionmaking methods in regulatory decisions.

The white/yellow and yellow/red thresholds for the PIs for initiating events and mitigating systems should be set in terms of an expert judgment of what values should in fact trigger the regulatory response associated with the threshold. Although general considerations for the selection of thresholds for PIs and SDPs are discussed in SECY-99-007, the expert judgment process that the staff used to develop the initial values for the thresholds for the non risk-based PIs and SDPs and the corresponding equivalency of the combination of findings in the action matrix have not been well documented. The NRC has been a pioneer in the use of scrutable expert judgment processes, and it is unfortunate that the use of expert judgment in a process as central to the NRC's mission as the ROP lacks the traceability of other NRC uses of expert judgment. Formal decision analysis could be helpful in making the selection of thresholds and the action matrix more objective and scrutable.

In assessing the need to revise the current PIs and develop new PIs, we believe that the staff responsible for the ROP should consider the work being done in other parts of the agency. For example, the review of operating experience for the reactor core isolation cooling (RCIC) system for BWRs (NUREG/CR-5500, Vol. 7) shows that the dominant failure modes involve system failures while running and human failures to recover the system (i.e., failures that are not part of the unavailability calculations that the ROP requires). In analyzing the operating experience, the analysts distinguished between two contexts of RCIC system operation: (1) short-term missions (less than 15 minutes), in which the system must inject water into the reactor vessel following a scram with feedwater available and the main isolation valves open, and (2) long-term missions, in which the system must inject water into the reactor vessel following a scram with feedwater unavailable and/or the reactor vessel isolated. The average system unreliability in these two contexts differs by a factor of 2. The ROP green/white threshold for RCIC system unavailability is 0.04 and makes no distinction between the two contexts identified in the study driven by operating experience. Since unreliability is a metric that includes all potential failure modes, it should be included in the PIs.

We continue to believe that it is important that there be consistency in the definition of terms like "unavailability" which are used in the PIs. Inconsistencies in technical terms that the agency uses in several major activities make comparisons and communication, both internally and externally, difficult.

The ROP is an evolving process. The staff has done an excellent job establishing the basic framework in a relatively short period of time considering the scope of this project. We look forward to continued interactions with the staff on this very important matter.

Additional comments by ACRS Members George E. Apostolakis, Thomas S. Kress, and Steven L. Rosen are presented below.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Staff Requirements Memorandum dated April 5, 2000, from Annette L. Vietti-Cook, Secretary, NRC, to Dr. John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting on March 2, 2000, with ACRS on Risk Informing 10 CFR Part 50.
2. Letter dated March 15, 2000, from Dana A. Powers, ACRS Chairman, to Richard A. Meserve, Chairman, NRC, Subject: Revised Reactor Oversight Process.
3. NRC Inspection Manual, Manual Chapter 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, February 5, 2001.
4. NUREG/CR-5500, Vol. 7, Reliability Study: Reactor Core Isolation Cooling System, 1987 - 1993, Idaho National Engineering and Environmental Laboratory, September 1999.
5. U. S. Nuclear Regulatory Commission, SECY-99-007, Recommendations for Reactor Oversight Process Improvements, January 8, 1999.
6. U. S. Nuclear Regulatory Commission, SECY-99-007A, Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY 99-007), March 22, 1999.
7. U. S. Nuclear Regulatory Commission, SECY 01-0114, Results of the Initial Implementation of the New Reactor Oversight Process, June 25, 2001.
8. U. S. Nuclear Regulatory Commission, SECY 00-0049, Results of the Revised Reactor Oversight Process Pilot Program, February 24, 2000.
9. U. S. Nuclear Regulatory Commission Inspection Manual, Manual Chapter 0305, Operating Reactor Assessment Program, March 23, 2001.
10. NRC Inspection Manual, Manual Chapter 0609, Significance Determination Process, February 27, 2001.
11. Advisory Committee on Reactor Safeguards, NUREG-1635, Vol 4, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission, May 2001.

ADDITIONAL COMMENTS BY ACRS MEMBERS GEORGE E. APOSTOLAKIS, THOMAS S. KRESS, AND STEPHEN ROSEN

We agree with the recommendations and comments of our colleagues. The intent of our comments is to elaborate on the expert judgment process.

In any decisionmaking situation, the most important requirement is that the decisionmaker's judgments be consistent. This is particularly important for the ROP because the bases for the inputs to the action matrix are different.

One of the columns of the action matrix treats two white inputs and one yellow input (for one degraded cornerstone) as being equivalent. This means that the staff's judgment is that two white inputs signify a certain degradation in performance which is about the same as that corresponding to one yellow finding in the sense that the resulting regulatory response should be the same. For consistency in defining these color bands, one would have to address questions such as the following:

- Does the yellow band for the initiating event PI indicate a degradation in performance that is similar to that indicated by the yellow band for a mitigating system PI?
- Is the yellow band of a PI twice as important as its white band?
- Is a yellow finding from an SDP of equal significance as a finding that a PI is in its yellow band?

We appreciate that judgments such as "of equal significance" and "twice as important" are subjective. Our argument is that attempting to answer questions such as these removes a good deal of the subjectivity and, in fact, will be very helpful when the thresholds are determined. This argument acquires additional significance in the present case in which the action matrix does not represent the judgments of a single individual but those of the agency. In other words, communication among the experts who make these judgments would be enhanced.

(5)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 10, 2002

Dr. George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: THE REVISED REACTOR OVERSIGHT PROCESS

Dear Dr. Apostolakis:

This letter responds to the October 12, 2001, letter from the ACRS that provided the results of the Committee's review of the adequacy of performance indicators (PI) and the significance determination process (SDP) as used in the Reactor Oversight Process (ROP). Most of the issues addressed by the ACRS were discussed at the various briefings that we provided to the Committee during the first year of ROP implementation. We will continue to interact with the ACRS on these matters as the ROP evolves. Your specific comments have been summarized and grouped below into three general categories: Significance Determination Process, Performance Indicators, and Licensee Assessment Using the Action Matrix. The comments along with our response follows.

Significance Determination Process

ACRS Comment
The most immediate and pressing need for the ROP is to improve the SDP tools. Some SDPs are incomplete and, in cases such as fire protection, overly subjective. An SDP based on low-power and shutdown PRAs or other shutdown management tools is needed to characterize findings during these modes of operation. In addition, the fire protection SDP involves very qualitative inputs to a quantification process of uncertain pedigree. We continue to believe that a documented review of the SDP worksheets and SPAR models (as well as the underlying SAPHIRE computer code) is essential to public confidence in the ROP.

Staff Response
The staff recognizes that the most immediate and pressing need for the ROP is to improve the SDP. We agree that the threshold values (delta CDF and delta LERF) for the risk-informed SDPs are appropriate, based on our experience gained to date. We have established a detailed strategy and associated plan to achieve near term and long term improvements to the SDP. The plan includes improvements of specific SDP tools, but also includes changes to the SDP process to provide for earlier management involvement with potentially significant findings, greater understanding of staff roles and responsibilities as inspection findings are processed, improved SDP timeliness, and improved coordination of the SDP with the Accident Sequence Precursor (ASP) program.

The staff has and will continue to incorporate lessons to improve the risk-informed SDPs. Based on our experience to date, the fire protection SDP generated the greatest concern from internal and external stakeholders. Some successes in utilizing this SDP have occurred but in

other instances resolutions of fire protection findings were delayed due to the complexity of the probabilistic considerations inherent in fire risk analysis and the need for higher confidence in certain influential engineering input assumptions. The staff acknowledges the Committee's concerns regarding the use of qualitative, and/or subjective input to develop quantitative fire protection SDP assessments. The staff has formed a fire protection SDP task group to examine existing concerns and develop options for improvement. This task group will engage industry and public stakeholders as part of its efforts.

The staff is currently developing shutdown SDP tools similar in level of detail and usage to the reactor safety phase 2 SDP. We plan to test and validate this guidance via plant visits and interactions with the regional inspection staff over the next 16 months.

The staff agrees that a documented review of the SDP worksheets and the SPAR models will enhance public confidence with respect to their use in the ROP. We expect that the phase 2 worksheets will be more consistent with the results of detailed PRAs following revisions based on an ongoing benchmarking effort. Approximately 12 notebooks have been benchmarked to date. The SDP notebook benchmarking reviews are currently scheduled to be completed in FY 2006. In the interim inspectors will continue to utilize the most current revision of the phase 2 notebooks. The results of the phase 2 analysis will be reviewed by NRC risk analysts prior to use.

Additionally, we are conducting a systematic review of the Level 1, Revision 3i SPAR models. The review includes an onsite review of the specific model and is being performed in conjunction with NRR's benchmarking of the plant's SDP phase 2 notebook. The SPAR model reviews are scheduled for completion in FY 2004.

The staff recently proposed revisions to the Emergency Preparedness and Radiation Protection SDPs to incorporate lessons learned from the first year of implementation. The revised SDPs were issued for comment to internal and external stakeholders.

ACRS Comment

Concurrent performance deficiencies are assessed collectively but each performance deficiency is assigned a color individually. There may be instances in which conclusions could be altered if the results are considered collectively, and thus such collective results should be considered in the Action Matrix.

Staff Response

The staff agrees that Action Matrix response can differ depending on whether concurrent degradations in plant performance are viewed as probabilistically independent or are the result of a common performance deficiency. Concurrent findings with common underlying cause are analyzed using the SDP as a single finding and are assigned a single color appropriate to the combined risk. Concurrent findings that are determined to be the result of independent causes are analyzed separately, and each receives a color based on the calculation of its risk significance as if the other finding did not overlap. The Action Matrix considers the SDP outcome of findings with common underlying cause as a single input, while the results of independent concurrent deficiencies are considered as individual inputs.

Performance Indicators

ACRS Comment
A realistic assessment of the change in CDF cannot be related to the change in a single PI. This approach has produced threshold values that are too high to be meaningful. Specifically, the numerical values for the white-yellow and yellow-red thresholds for the Initiating Events and Mitigating Systems performance indicators are not useful and should be set by expert judgment to trigger the regulatory response associated with the particular threshold.

Staff Response
The white-yellow and yellow-red thresholds for the Initiating Events and Mitigating Systems performance indicators were risk-informed using the models available at the time. Some of these thresholds are high, which indicates that the risk significance of those individual PIs, taken alone, is low. The staff believes that the values for these thresholds are representative of the risk to the plant due to problems in the area monitored by the respective indicators individually. The staff does not deem it necessary or appropriate to attempt to monitor combinations of performance problems in a single PI. This would require plant models with a degree of accuracy much greater than those available. It would also require a degree of complexity beyond the intent and the purpose of the indicators. Rather, the green-white and some of the white-yellow thresholds are set low enough to identify problems in individual plant areas that will trigger NRC engagement long before any serious decline in overall plant risk occurs.

The yellow-red threshold for the Unplanned Scrams per 7,000 Critical Hours PI has been the subject of much comment. This PI assumes that all scrams are uncomplicated; therefore, a large number of such scrams may occur before the increased initiating event frequency results in a change in CDF of 10^{-4}.[1] The NRC will take action long before the number of scrams approaches the yellow-red threshold of 25. The staff will engage with a supplemental inspection when a plant's scram PI crosses the green-white threshold of 3 per 7,000 critical hours. If the white-yellow threshold of 6 is exceeded, the NRC will, as a minimum, take the appropriate action for a Degraded Cornerstone in the Action Matrix to ensure that effective corrective actions are carried out. While it is highly unlikely that any plant would cross the yellow-red threshold, the staff believes the threshold does reflect the risk significance of the types of events monitored in this PI. Therefore, the threshold does provide a gauge to measure the relative risk of the number of scrams reported by a licensee.

ACRS Comment
The thresholds for both the performance indicators and the SDP should be commensurate with their respective safety significance and consistent with each other.

Staff Response
The staff agrees with this comment. The staff has attempted to provide consistency in the safety significance of PIs and inspection findings to the degree possible with the plant models

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[1] The more complicated, and consequently more risk significant scrams, such as those associated with losses of offsite power or steam generator tube ruptures, are not included in the indicators and may be assessed for related performance deficiencies through a special inspection.

available and considering that the PIs are not intended to cover all aspects of licensee performance or to measure plant risk. The difference is primarily at the green-white threshold, where the PIs identify outliers while the SDP identifies an increase in CDF of 10^{-6}. However, the assessment process is designed to provide for a regulatory response at the same level for either a white PI or a white inspection finding.

The current set of PIs were not intended to be permanent. Recognizing the limitations of performance indicators, the staff implemented an initial set and has expended a great deal of effort to improve those PIs where possible and to replace them where necessary. The PI program of the ROP is a work in progress.

ACRS Comment
The risk impact of changes in a PI depend strongly on plant-specific features.

Staff Response
The staff agrees with this comment. As discussed with the ACRS, the staff has long recognized the need to include plant specific PI thresholds where possible. For over seven months we have been working with industry to develop and test unavailability and unreliability indicators with plant-specific thresholds.

ACRS Comment
Initial values for thresholds for non-risk-based performance indicators and SDPs have not been well documented; formal decision analysis could be helpful in making the selection of thresholds and the Action Matrix more objective and scrutable.

Staff Response
The staff agrees that the process used to develop thresholds for the non-risk-based performance indicators could have been documented more thoroughly. In developing thresholds, the staff formed multi-discipline task groups which used expert judgement, risk insights, and analysis of historical plant performance data. The thresholds were reevaluated after the pilot program and thresholds were for the most part found to be appropriate.

Only one threshold required a slight adjustment. The staff has been investigating the application of formal decision analysis in the regulatory process. Upon completion of the effort the staff will consider possible applications of these techniques in future threshold determinations.

ACRS Comment
In assessing the need to revise current performance indicators or develop new ones, consider the work being done in other parts of the agency, specifically the system reliability studies performed in the Office of Nuclear Reactor Research. The review of operating experience for the reactor core isolation cooling (RCIC) system shows the dominant failure modes involve failures while running and human failures to recover the system. The analysts distinguished between short-term missions (less than 15 minutes) and long-term missions. The average system unreliability differs by a factor of 2. The ROP green-white threshold makes no such distinction.

Staff Response
The staff agrees with the ACRS regarding the benefit of using the results of the work being done in other parts of the agency in developing and improving the performance indicators and other aspects of the ROP. Some of these results have already been incorporated into the process. The Initiating Events Study was used in the development of the unplanned scrams with loss of normal heat removal PI. The system reliability studies are being used in the Industry Trends Program. Data and methodology from the system reliability studies have been used in the unreliability indicator soon to be tested. The SPAR Rev. 3 models currently under development by RES have and will continue to serve a key role in enabling us to develop and implement the enhanced unreliability and unavailability indicators, as well as to improve the Accident Sequence Precursor program.

With regard to the RCIC reliability study, the current RCIC system PI only monitors system unavailability, which does not distinguish between failures to start and failures to run. The future unreliability indicator will include this difference.

<u>Licensee Performance Assessment Using the Action Matrix</u>

ACRS comment
One of the columns of the Action Matrix treats two white inputs and one yellow input (for one degraded cornerstone) as being equivalent. This means that the staff's judgement is that two white inputs signify a certain degradation in performance which is about the same as that corresponding to one yellow finding in the sense that the resulting regulatory response should be the same.

Staff Response
This is an accurate statement. An underlying principle of the ROP is that the crossing of a performance indicator (PI) or inspection finding threshold will have similar meaning with respect to safety significance and subsequent NRC action. An expert panel was organized for development of the assessment program in order to determine the appropriate level of NRC actions for a given number of safety significant PIs and inspection findings.

The expert panel considered the uncertainty associated with the number of white findings that would require an agency response that was proportional to the response for a single yellow finding. Additionally, the panel considered the appropriateness of previous action taken by the agency under the previous assessment process. Consideration was also given to the fact that any white finding could have a delta CDF that was almost green (e.g. 1.1E-6) or almost yellow (9.9E-6) and that an appropriate risk-based response would be complicated when combining multiple white issues. The group determined that it would be inappropriate to attempt to quantitatively define the agency's response when considering these uncertainties. Conservatively, the panel adopted a risk-informed approach that two white PIs or inspection findings in the same cornerstone represents a condition whose approximate agency response is similar to that of a single yellow finding or PI.

A major success of the assessment program so far is that it has been very predictable but allows some flexibility in determining the agency's response within a pre-determined range of actions. This flexibility is provided by the range of allotted supplemental inspection hours which may be customized for the individual circumstances of the performance deficiencies and the range of regulatory actions in the Multiple/Repetitive Degraded Cornerstone column of the Action Matrix.

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

ACRS Full Committee Meeting
The Reactor Oversight Process

March 6, 2003

Cynthia Carpenter

Mark Satorius

Ronald Frahm

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Staff Understanding of Committee Concerns

- Risk-Informed and Performance-Based Elements Are Not Quantifiably Equitable
- PI and SDP Thresholds are not Fully Consistent and Transparent
- Yellow-Red Thresholds for the Scram Indicators are not Meaningful
- Cross-Cutting Issues are not Adequately Addressed in Current ROP

Risk-Informed and Performance-Based Differences

- Acknowledge Differences Between Risk-Informed and Performance-Based Aspects of the ROP
- ROP Uses Performance-Based Thresholds Linked to Appropriate Regulatory Response, Incorporating Risk Insights If Available
- Meet Competing ROP Objectives to Remain Predictable, Understandable, Risk-Informed, and Objective, and the 4 Strategic Performance Goals
- Staff Believes the ROP Appropriately Addresses Risk-Informed and Performance-Based Issues

Consistency and Transparency of PI and SDP Thresholds

- Recognized Concern and Developed ROP Basis Document to Consolidate More Transparent Basis for PIs and SDPs
- Thresholds Developed By Expert Judgment with Significant Stakeholder Input and Buy-In
- Continue to Adjust PI and SDP Thresholds to Ensure a Consistent Regulatory Response
- Seek Continued Improvements Through the ROP Self-Assessment and Feedback Processes and Interactions with Other Stakeholders

Yellow-Red Thresholds for the Scram Indicators

- Recognize that Highly Unlikely Any Plant Would Cross These Yellow-Red Thresholds
- Existing Thresholds Provide a Gauge of the Relative Risk and Demonstrate the Safety Margin
- Plan to Maintain Current Thresholds and Consider Adjusting the Thresholds for the Scram PIs Along With Ongoing Changes to the PI Program

Addressing Cross-Cutting Issues in ROP

- Recognized Concern and are Developing Prescribed Response to Substantive Cross-Cutting Issues
- Incorporated into SDP Improvement Initiative Based on Recommendation from the SDP Task Group
- Response May Include Redirection of Resources, Management Meetings, and/or Docketed Licensee Response

Staff Conclusions

- Current ROP is Working, Plants Are Receiving Appropriate Level of Oversight
- Continued Improvements are Warranted and Anticipated Based on Lessons Learned
- No Current Plans to Revise Fundamental Basis of ROP Regarding Handling of Risk-Informed and Performance-Based Elements
- ROP Must Remain Transparent and Consistently Maintained Via the ROP Basis Document

MRP Alloy 600 ITG Reactor Vessel Head WG Status Report

Advisory Committee on Reactor Safeguards

Vessel Head Penetration Cracking and RPV Head Degradation

March 6, 2003

Larry Mathews, SNOC
MRP Alloy 600/82/182
Issue Task Group Chairman

EPRI

Topics

- Overview Table of Inspection Results by Plant
  - Detected Circumferential Cracks
  - Summary Statistics
- Inspection Plans for Spring 2003 Outages

Overview Table of Inspection Results

Toward head periphery →

EDYs →

Summary of Key Design, Operation, Inspection and Repair Information Related to RPV Head Nozzle PWSCC

EDYs MRP-48 (12/2001)	Outage Head Temp.	Unit Name	Material Supplier[1]	Vessel Fabricator[2]	Max Nozzle Dp[3]	Insulation Type	Refueling Outage Fall 02	Spr 03	Fall 03	Spr 04	EDYs Reported in Response to Bulletin 2002-01 EDYs	Date	Replacing Head	Number of CRDM/CEDM Nozzles	Inspection Methods Since 11/2000 None	Visual	Nozzle NDE	Weld NDE
22.1	602.0	Oconee 1	B	B&W	1.5	Reflective Horizontal			X		30.2	Fall 2003	Fall 2003	69		BMV	VT (23), ET (8), PT (3)	PT (3)
22.0	602.0	Oconee 2	B	B&W	1.5	Reflective Horizontal	Oct-02				32.7	Oct-2002	Spring 2004	69		BMV	VT, PT (8), ET (6)	PT (9)
21.7	602.0	Oconee 3	B	B&W	1.5	Reflective Horizontal		X			22.5	Spring 2003	Spring 2003	69		BMV	VT, ET (14), PT (12)	PT (12)
19.5	602.0	ANO 1	B/H	B&W	1.5	Reflective Horizontal	Oct-02				19.5	Oct-2002	Fall 2006	65		BMV	ET, PT (3)	PT (3)
19.4	600.1	North Anna 1	S	RDM	1.2	Reflective Stepped		Mar-03			21.4	May-2003	Spring 2003	65		BMV	ET (18), UT (8), PT (4)	PT (4)
19.0	598.0	Robinson 2	H	CE	3.0	Blanket Contoured	Oct-02				>19	Since MRP-44		69		BMV	ET, UT (17)	ET
18.6	597.8	Surry 1	B	B&W/RDM	1.2	Reflective Stepped		Apr-03			20.5	Apr-2003	Spring 2003	65		BMV	ET (14)	PT (16)
18.4	597.8	Surry 2	B/S	B&W/RDM	1.2	Reflective Stepped			Sep-03		20.9	Sep-2003	Fall 2003	65		BMV		
18.3	605.1	North Anna 2	S	RDM	1.2	Reflective Stepped	Sep-02				19.8	Sep-2002	Fall 2002	65		BMV	ET, UT (35)	ET (39), PT (6)
17.9	605.2	Davis Besse	B/H	B&W	1.5	Reflective Horizontal					9.0	Fall 2002	Fall 2002	69		BMV	VT	
17.3	601.0	TMI 1	B	B&W	1.5	Reflective Horizontal			X		Note 4	Note 4	Fall 2009	68		BMV	UT (12), PT (12)	PT (12)
14.7	594.4	Turkey Point 3	H	B&W	1.5	Blanket Contoured		Mar-03			18.3	May-2003	Assessing	65		BMV		
14.6	594.4	Turkey Point 4	H	B&W	1.5	Blanket Contoured			Oct-03		18.6	Oct-2003	Assessing	65		BMV		
13.8	594.3	Farley 1	W/B	B&W/CE	3.0	Reflective Stepped		Mar-03			17.3	May-2003	Fall 2006	69		BMV		
13.6	602.0	Crystal River 3	B	B&W	1.5	Reflective Horizontal			X		Note 4	Note 4	Fall 2003	69		BMV	VT (9)	PT (1)
13.1	589.2	Ginna	H	B&W	1.5	Block Contoured Contoured			X		Note 4	Note 4	Fall 2003	37	5			
14.9	591.6	Point Beach 2	W/B	B&W/CE	3.0	Modified for BMV			Sep-03		16.4	Sep-2003	Assessing	49		BMV		
14.7	590.6	St. Lucie 1	H	CE	3.0	Modified for BMV	Sep-02				13.8	Sep-2002	Assessing	69		BMV	VT	
14.5	596.9	Farley 2	B/H	B&W/CE	3.0	Reflective Stepped	Sep-02				15.6	Sep-2002	Fall 2002	69		BMV	VT	
14.5	590.9	San Onofre 2	S/S/H	CE	3.0	Modified for BMV				Feb-04	15.3	Aug-2002	Assessing	91		BMV	VT	ET (46), PT (1)
14.4	590.6	San Onofre 3	S/S/H	CE	3.0	Modified for BMV		Jan-03			15.2	Aug-2002	Assessing	91		BMV	ET, UT	ET
14.2	593.7	Calvert Cliffs 1	H	CE	3.0	Removable Collars				Mar-04	16.1	Mar-2004	Assessing	65		BMV		
14.1	593.7	Waterford 3	S/S/H	CE	3.0	Modified for BMV			X		Note 4	Note 4		91		BMV		
13.8	593.7	Calvert Cliffs 2	H	CE	3.0	Blanket Contoured		Feb-03			15.1	Feb-2003	Assessing	65		BMV (8 ICI)		
12.5	591.6	Point Beach 1	H	B&W	1.5	Modified for BMV	Sep-02				14.5	Sep-2002	Assessing	49		BMV	VT	PT (1)
13.0	600.7	Cook 2	W	CE	4.0	Reflective Stepped		Apr-03			14.6	Apr-03		78	6	BMV	ET (33), ET (10)	ET (10), PT (1)
12.4	595.0	Beaver Valley 1	W/B	B/W/CE	3.0	Reflective Stepped		Mar-03			14.0	May-2003	Spring 2006	65		BMV		
12.3	595.6	St. Lucie 2	S/S/H	CE	3.0	Modified for BMV		Apr-03			14.0	Apr-2003	Assessing	91		BMV		
10.8	588.0	Fort Calhoun	H	CE	3.0	Reflective Stepped			Sep-03		11.8	Sep-2003	Maybe Fall '04	41		BMV		
10.8	589.1	Kewaunee	W/B	B/W/CE	3.0	Reflective Panels		X			<12	Spring 2002		40		BMV		
10.6	589.5	Indian Point 3	H	CE	3.0	Will Be Modified for BMV		Mar-03			11.2	Mar-2003		78				
10.6	594.7	Salem 1	H	CE	3.0	Reflective Horizontal	Oct-02				11.9	Oct-2002		79		BMV		
10.3	594.8	ANO 2	S/S/H	CE	3.0	Collars Around Nozzles			X		Note 4	Note 4		81			UT	
10.5	593.9	McGuire 2	H	CE	3.0	Collars Around Nozzles			Oct-03		12.7	Oct-2003	Assessing	69			UT	PT (3)
9.9	580.2	Prairie Island 1	CE	CE	3.5	Reflective Horizontal	Nov-02				<11	Winter 2002		40		BMV		
9.9	580.2	Prairie Island 2	A	CE	3.5	Reflective Horizontal			X		<11	Fall 2003		40		BMV		
9.6	595.0	Diablo Canyon 2	H	CE	3.0	Reflective Stepped		X			10.9	Feb-2003		78				
9.5	578.0	Cook 1	H	CE	3.0	Reflective Stepped			Oct-03		10.3	Oct-2003		79		BMV	VT (63), ET (28), PT (2)	ET (29), PT (2)
9.4	593.0	Palo Verde 1	S/S/H	CE	3.0	Will Be Modified for BMV	Oct-02				11.2	Fall 2002		97		BMV (26)	ET, UT	ET (13)
9.1	591.7	Palo Verde 2	S/S/H	CE	3.0	Will Be Modified for BMV			X		10.3	Spring 2002		97		BMV (26)	ET, UT	ET (19)
9.1	591.7	Palo Verde 3	S/S/H	CE	3.0	Will Be Modified for BMV		X			11.0	Spring 2003		97				
8.9	594.4	Palisades	H	CE	3.0	Blanket Contoured		X			<10	Spring 2003		65	7			
8.4	595.1	Diablo Canyon 1	H	CE	3.0	Reflective Stepped				Feb-04	10.2	Feb-2004		79		BMV		
8.2	595.0	Beaver Valley 2	H	CE	3.0	Reflective Stepped			Sep-03		10.1	Sep-2003		65		BMV		
8.3	594.1	Salem 2	H	CE	3.0	Reflective Horizontal			Oct-03		10.3	Oct-2003		78		BMV		
7.1	585.5	Indian Point 2	H	CE	3.0	Modified for BMV	Oct-02				9.0	Oct-2002		97		BMV	VT (92), ET (56)	PT (1)
11.1	561.0	South Texas 2	W/B	CE	3.0	Reflective Stepped	Oct-02				5.3	Oct-2002		74	8	BMV		
10.7	561.0	South Texas 1	H	CE	3.0	Reflective Stepped		Mar-03			4.5	Mar-2003		74	8			
2.3	554.4	Callaway	H	CE	3.0	Reflective Stepped	Oct-02				2.5	Oct-2002		78		BMV		
2.3	557.3	Summer	B	CH	4.0	Reflective Stepped			X		2.5	Apr-2002		65		BMV (64)		
2.2	557.0	McGuire 1	H	CE	3.0	Reflective Horizontal	Sep-02				3 to 3	Fall 2003		78		BMV		
2.1	557.0	McGuire 2	S	RDM	1.2	Reflective Horizontal			X		3 to 3	Fall 2003		78		BMV		
2.2	560.0	Vogtle 1	H	CE	3.0	Reflective Stepped			Sep-03		2.7	Sep-2003		78		BMV		
2.2	558.0	Wolf Creek	H	CE	3.0	Reflective Stepped			X		2.4	Sep-2002		78		BMV		
2.1	557.0	Catawba 1	S	RDM	1.2	Reflective Stepped			X		3 to 3	Fall 2003		78		BMV		
2.0	558.0	Braidwood Harris	B	CH	4.0	Reflective Stepped		Apr-03			2.1	Sep-2002		65				
1.9	557.0	Catawba 2	H	CE	3.0	Reflective Stepped		Mar-03			3 to 3	Fall 2003		78				
1.9	560.0	Vogtle 2	H	CE	3.0	Reflective Stepped	Oct-02				2.2	Oct-2002		78		BMV		
1.7	561.0	Comanche Peak 1	H	CE	3.0	Reflective Stepped	Oct-02				2.1	Oct-2002		78		BMV		
1.6	551.0	Byron 1	B	B/W	1.5	Reflective Horizontal			Sep-03		2.0	Sep-2003		78	9			
1.6	558.4	McGuire 1	H	CE	3.0	Reflective Horizontal	Sep-02				1.8	Sep-2002		78		BMV		
1.6	558.9	Seabrook	H	CE	3.0	Reflective Stepped			Oct-03		1.8	Aug-2002		78		BMV		
1.5	556.0	Braidwood 1	B	B/W	1.5	Reflective Horizontal		Apr-03			1.7	Apr-2003		78				
1.4	552.0	Braidwood 2	B	B/W	1.5	Reflective Horizontal			Oct-03		1.7	Oct-2003		78		BMV		
1.4	550.4	Byron 2	B	B/W	1.5	Reflective Horizontal	Sep-02				1.8	Sep-2002		78		BMV		
1.3	561.0	Comanche Peak 2	H	CE	3.0	Reflective Stepped			X		1.8	Fall 2003		78		BMV		
1.3	547.0	Sequoyah 1	S	RDM	1.4	Reflective Horizontal		X			1.5	Spring 2003		78				
1.3	547.0	Sequoyah 2	S	RDM	1.4	Reflective Horizontal			X		1.6	Fall 2003		78		BMV		
0.7	537.3	Watts Bar 1	S	RDM	1.2	Reflective Horizontal			X		1.1	Fall 2003		78				

Summary of Cumulative CRDM and CEDM Nozzle Inspections Remote Since 11/2000

NOTES

1. Key for Material Suppliers: B = B&W Tubular Products, H = Huntington, S = Sandvik, SS = Standard Steel, W = Westinghouse (Huntington), CL = C.L. Inglby, A = Avbest et Orval
2. Key for Vessel Fabricators: BW = B&W, CH = Chicago Bridge & Iron, CE = Combustion Engineering, RDM = Rotterdam Dockyard, CL = C.L. Inglby
3. Maximum Superfluid Dynamical Interference Pit (mils)
4. Not reported in Response to Bulletin 2002-02 as of January 2003 (e.g., head replacement)
5. Because of damaged head insulation, performed special inspections in spring 2002:
   100 % visual above insulation, removed insulation in two suspect areas, no evidence of leakage,
   100 % visual of seal weld joint between CRDM adaptor and CRDM, no indications, and
   UT at center of head and four instrument port penetration locations, through thickness direction of head to assure no voids similar to Davis-Besse experience
   Also 100% inspection of ID surface of all head penetrations by ET in 1999 with no recordable indications
6. Crack found on nozzle 75 (October 1994)
7. Visual inspection in 1995 and ET for all ICI nozzles in 1995
8. Modifications to EDY calculation based on re-evaluation of head temperature
9. Visual inspection in the area of the Head Vent Nozzle (not not BMV), roughly 20% of the head surface was examined.

A = Auxiliary Head Adapter, D = Dye Gap Nozzle, I = ICI Nozzle, T = Thermocouple Nozzle, V = J-Groove type Head Vent Nozzle, W = Weld inspected by PT or ET

Inspection Status Code

- No nozzle associated with this location
- No eye inspected (since the CEDM is larger was discovered in December 2000)
- Visual not meeting 100% BMV requirements (e.g., < 100% nozzle inspected by BMV, insulation not removed or like it)
- Bare Metal Visual (BMV) inspection with no leaks
- Non-visual inspection (UT or ET) of nozzle base metal with no leaks or cracks (indications being detectable depth)
- Non-visual inspection (UT, ET or PT) of nozzle base metal with cracks requiring repair
- Leaking nozzle (even though BMV has not including marked nozzles found not to be leaking by UT/ET)
- Circumferential crack above or at top of J-groove weld
- Significant wastage of low alloy steel head material

MRP-48 EDYs, head temp., unit name, and basic design info

Types of inspections performed

Color-coded nozzle-by-nozzle inspection results (CRDM/CEDM, vent, other)

Outage schedule, BL 2002-02 EDYs, and head replacement plans

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Overview Table of Inspection Results

- The overview table graphically shows:
  - The extent to which the fleet has been inspected
  - The extent of detected cracking, leakage, and wastage correlated with effective degradation time (EDYs) and position on the head
  - Key operating and design data
  - Refueling outage schedule and current head replacement plans
- The overview table complements more detailed outage-specific and defect-specific inspection results tables that are used to generate statistical (i.e., Weibull) fits
- The MRP plans to release a revision to the table at the end of each outage season

Results for Inspections that Detected Cracking

No.	Unit	NSSS Supplier	Approx. EDYs at Inspection	Date	Number of Nozzles on Head				Nozzles Inspected by Non-Visual		No. Leaking Nozzles/ Welds		No. Cracked Nozzles/ Welds		No. Nozzles with Base Metal Cracks	No. Nozzles with Weld Metal Cracks	No. Nozzles with Axial Cracks	No. Nozzles with Circ. Cracks
					CRDM	CEDM	ICI	Total	Total	Total %	Leaking	% of Total Inspected	Cracked	% of Total Inspected				
1	ANO 1	B&W	19.6	Mar-2001	69			69	1	1.4%	1	100.0%	1	100.0%	1	0	1	1
2	ANO 1	B&W	21.1	Oct-2002	69			69	69	100.0%	1	1.4%	8	11.6%	8	8	8	0
3	Cook 2	W	13.9	Jan-2002	78			78	78	100.0%	0	0.0%	2	2.6%	2	0	2	0
4	Crystal River 3	B&W	16.2	Oct-2001	69			69	9	13.0%	1	11.1%	1	11.1%	1	1	1	1
5	Davis-Besse	B&W	19.2	Apr-2002	69			69	69	100.0%	3	4.3%	5	7.2%	5	0	5	1
6	Millstone 2	CE	11.2	Feb-2002		69	8	77	77	100.0%	0	0.0%	3	3.9%	3	1	3	2
7	North Anna 1	W	20.0	Oct-2001	65			65	30	46.2%	0	0.0%	6	20.0%	6	0	6	0
8	North Anna 2	W	19.0	Nov-2001	65			65	3	4.6%	3	100.0%	3	100.0%	3	3	3	0
9	North Anna 2	W	19.7	Sep-2002	65			65	65	100.0%	6	9.2%	42	64.6%	7	42	1	6
10	Oconee 1	B&W	21.8	Nov-2000	69			69	18	26.1%	1	5.6%	1	5.6%	1	1	1	0
11	Oconee 1	B&W	23.2	Mar-2002	69			69	5	7.2%	1	20.0%	3	60.0%	3	1	3	0
12	Oconee 2	B&W	22.2	Apr-2001	69			69	4	5.8%	4	100.0%	4	100.0%	4	4	4	1
13	Oconee 2	B&W	23.7	Oct-2002	69			69	69	100.0%	7	10.1%	15	21.7%	15	5	10	0
14	Oconee 3	B&W	21.7	Feb-2001	69			69	18	26.1%	9	50.0%	10	55.6%	10	0	10	5
15	Oconee 3	B&W	22.5	Nov-2001	69			69	52	75.4%	5	9.6%	7	13.5%	7	2	7	2
16	Surry 1	W	19.1	Oct-2001	65			65	16	24.6%	2	12.5%	6	37.5%	0	6	0	0
17	TMI 1	B&W	18.1	Oct-2001	69			69	12	17.4%	5	41.7%	7	58.3%	7	4	7	0
Totals for Inspections Since First U.S. Leakage (11/2000)					3871	1090	94	5055	1462	28.9%	47	3.2%	120	8.2%	82	75	71	19

NOTE: The table does not reflect the small-diameter thermocouple nozzles found to be cracked and leaking at Oconee 1 and TMI 1. (These are the only two plants that have this type of nozzle.)

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Nozzle Circ Cracks At or Above the J-groove Weld

Unit	NSSS Design	Nozzle ID	Nozzle Angle (°)	Inspection Results						
				Date	Approx. EDYs	OD/ ID	Axial Location	Circ. Angle (°)	Depth (in)	TW Depth (%)
Crystal River 3	B&W	32	26.2	Oct-01	16.2	OD	above weld	91	0.29	47%
Davis-Besse	B&W	2	8.0	Mar-02	19.2	OD	above weld	34	0.31	50%
North Anna 2	W	15	19.8	Sep-02	19.7	OD	≥1.12" below root	5	0.23	36%
		41	33.1			OD	≥0.52" below root	46	0.10	16%
		54	38.6			OD	≥0.04" below root	79	0.23	36%
						OD	≥0.28" below root	32	0.16	25%
		59	40.0			OD	≥0.31" below root	76	0.15	24%
						OD	≥0.32" below root	50	0.15	24%
		65	42.6			OD	≥0.32" below root	72	0.15	24%
						OD	≥0.20" below root	30	0.08	12%
		67	42.6			OD	≥0.80" below root	44	0.09	15%
Oconee 2	B&W	18	18.2	Apr-01	22.2	OD	above weld	36	0.07	11%
Oconee 3	B&W	11	16.2	Feb-01	21.7	OD	over weld	153	0.36	57%
						OD	over weld	113	0.25	40%
		23	23.2			OD	above weld	66	0.22	35%
		50	35.1			OD	above weld	165	0.62	pin holes
		56	35.1			OD	above weld	165	0.62	100%
		2	8.0	Nov-01	22.5	OD	above weld	48	0.18	29%
		26	24.7			OD	over weld	44	0.07	11%

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Summary of Inspection Results Statistics

- The total RVH nozzle population includes 3871 CRDM nozzles, 1090 CEDM nozzles, and 94 ICI nozzles at 69 units
- Bare-metal visual (BMV) and/or non-visual NDE inspections have now been performed on about 81% of the RVH nozzles
  - About 47 nozzles have been found to be leaking
- Almost 8% of the nozzles in B&W plants have leaked, but leakage in non-B&W plants is limited to North Anna 2 and Surry 1 leakage, which is primarily due to weld cracking
- Non-visual examinations have been performed on:
  - About half of the “>12 EDY” nozzles and a third of the “8-12 EDY” nozzles
  - About two-thirds of the nozzles in B&W plants and 25% of the nozzles in non-B&W plants

Summary of Inspection Results Statistics (cont'd)

- About 19% of the inspected B&W plant nozzles show base metal cracking
- Base metal cracking in non-B&W plants is limited to Millstone 2 (3 nozzles) and Cook 2 (1 nozzle), although North Anna 1 and 2 may have experienced some base-metal initiated cracking (Sandvik material)
- About 8% of the J-groove welds have been examined by ET or PT
- Weld experience ranges from no indications in a relatively high EDY plant (Robinson) to relatively extensive weld cracking in another high EDY plant (North Anna 2)
- To date, weld cracking has been limited to vessels fabricated by Rotterdam Dockyards and B&W-designed units

Inspection Plans for Spring 2003

- 20 units have refueling outages this spring:
  - Oconee 3, North Anna 1, and Surry 1 will replace their heads with new heads having Alloy 690 material
  - All 17 other units will perform 100% BMV and/or non-visual inspections
  - All the plants having greater than 12 EDYs will have performed a non-visual baseline examination by the end of the spring outage season
- The spring 2003 outage season mainly concludes the initial set of inspections following Bulletin 2001-01. After this spring:
  - All but two units (< 2 EDYs) will have completed 100% BMV and/or non-visual inspections (97% of the total nozzle population)
  - 20 of the 28 units with > 12 EDYs (as of February 2001) will have completed baseline non-visual examinations or head replacement

Inspection Plans for Spring 2003 (cont'd)

- After fall 2003, it is expected that:
  - All 69 units will have completed 100% BMV and/or non-visuals (or head replacement)
  - 27 of the 28 units with > 12 EDYs (as of February 2001) will have completed baseline non-visual examinations or head replacement (28th unit plans such an inspection at its next RFO in spring 2004)
- Upon the conclusion of the spring outage season, the MRP will again look for correlations between cracking and factors such as EDYs, tubing material supplier, and vessel head fabricator

Process for Revising the MRP Inspection Plan

Advisory Committee on Reactor Safeguards

Vessel Head Penetration Cracking and
RPV Head Degradation

Craig Harrington, TXU
MRP Alloy 600/82/182 ITG
RV Head Working Group Chairman

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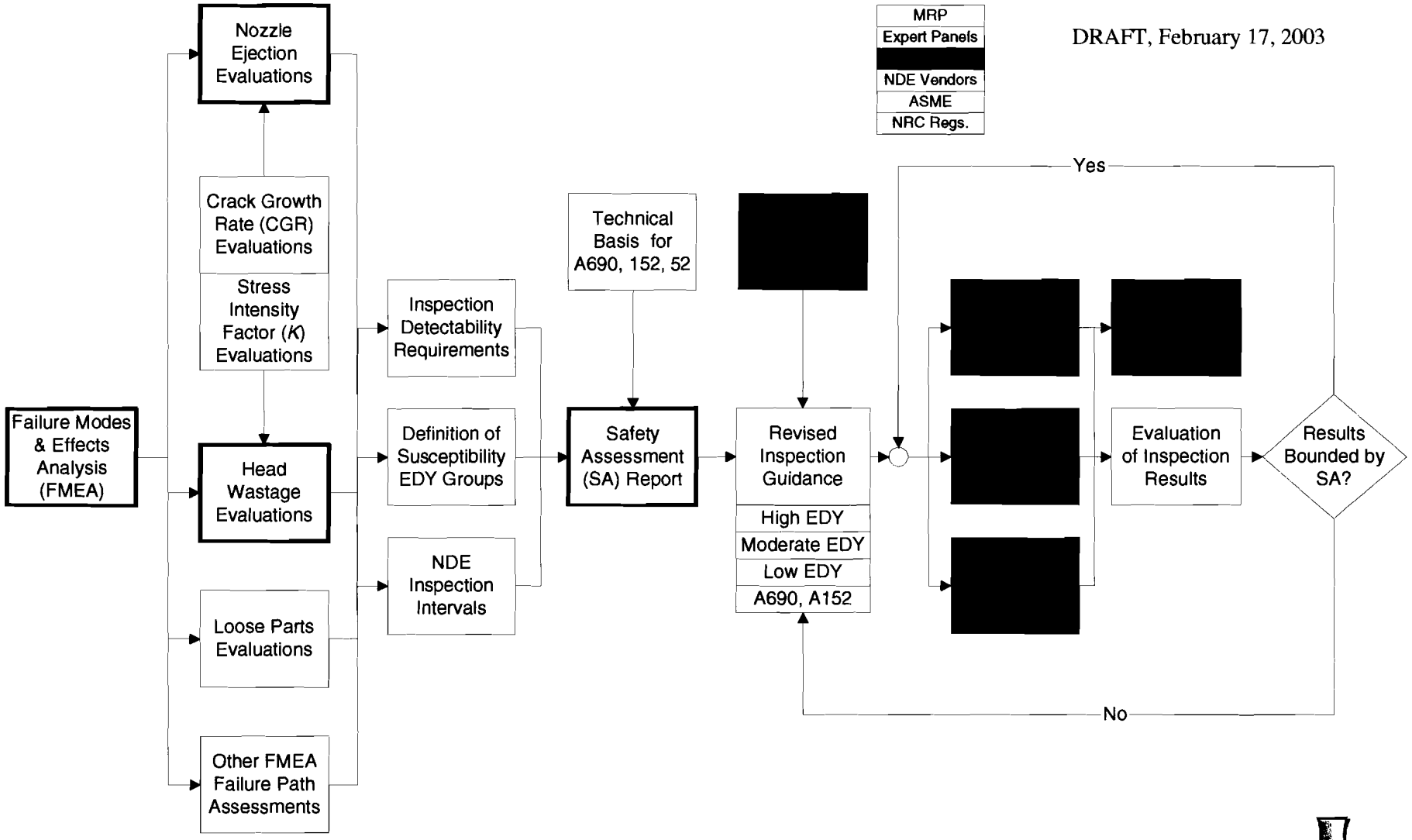
Topics

- Overall Safety Assessment Process
- Transition to Combination Baseline Inspections with Inspection Intervals Chosen to Ensure Safety
- Failure Modes and Effects Analysis
- Schedule for Issuing Revised Inspection Plan and Safety Assessment Report

Overall Process Flowchart

DRAFT, February 17, 2003



Safety Assessment Process: Key Points

- The MRP approach is transitioning to ensuring safety through "combination" baseline inspections at all plants with:
  - The timing for the baseline inspection and the re-inspection interval based on the technical evaluations and
  - More frequent bare metal visual (BMV) inspections providing backup to the program of periodic combination inspections
- The revised MRP inspection plan will be formed on the basis of a comprehensive safety assessment (SA) report
- The SA report:
  - Begins with a failure modes and effects analysis (FMEA) to anticipate the possibility of failure modes that have not been observed in the field and
  - Includes the analysis tools previously developed and described in MRP-75

Safety Assessment Process: Key Points (cont'd)

- The results of the FMEA are used to establish the required technical evaluations and ultimately the inspection detectability requirements
- Existing calculations show that non-visual inspections do not have to be performed every refueling outage to ensure safety
  - Extremely low probability of nozzle ejection and significant wastage
  - Extremely small consequential increase in core damage frequency, consistent with NRC Reg. Guide 1.174

Combination Baseline Inspections

- Subsequent to the release of the MRP-75 inspection plan and technical bases and in light of the most recent inspections results, the MRP has released a letter to the industry recommending a transition to combination baseline inspections
- Three types of combinations inspections:
  - (UT/BMV) UT of the base metal from the tube ID and bare-metal visual (BMV)
  - (UT/ET) UT of the base metal from the tube ID and ET/PT of the weld surface
  - (ET/ET) ET of the base metal ID and OD and ET/PT of the weld surface
- The timing of the baseline inspection and the inspection interval will be based on the technical evaluations to ensure safety

Combination Baseline Inspections (cont'd)

- Time at temperature (EDYs) will continue to form the basis for the susceptibility groups
- It is expected that high susceptibility plants will perform the combination baseline inspection by the next refueling outage
- It is expected that moderate susceptibility plants will perform the baseline inspection by approximately 2005 at the latest
- It is expected that low susceptibility plants will perform the baseline inspection by approximately 2007 at the latest

Failure Modes and Effects Analysis: Introduction

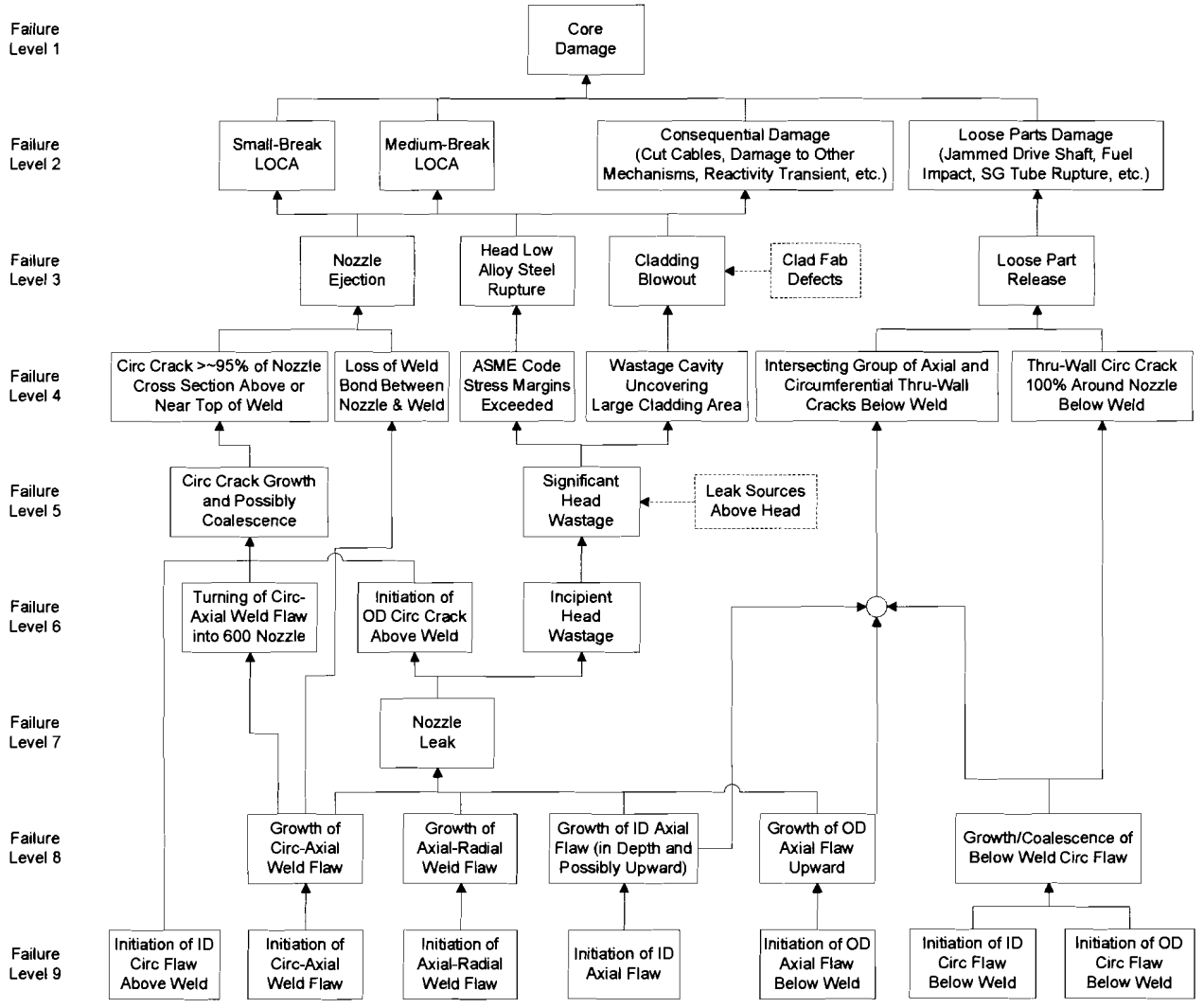
- FMEA is a technique of TQM (Total Quality Management) to ensure product reliability
- Typically, a table of the following characteristics of the possible failure modes is prepared:
  - Cause
  - Effect (consequence)
  - Detectability
  - Frequency of Occurrence
- Relationships among the failure modes are illustrated using a block diagram
- FMEA is a tool that helps anticipate new failure modes

Failure Modes and Effects Analysis: Application to RVH Nozzles

- For RVH penetrations, there are three principal failure modes:
  - Nozzle Ejection Due to Net Section Collapse
  - Cladding Blowout Due to Wastage
  - RCS Damage Due to Loose Parts Generation
- There are several levels in the failure process for these modes:
  - PWSCC initiation (nozzle ID, nozzle OD below weld, weld surface)
  - PWSCC growth (axial and circ in nozzle, axial-radial and circ-axial in weld; weld to nozzle and nozzle to weld; turn from axial to circ)
  - Leakage to annulus (new crack initiation and low-alloy steel wastage)
  - Growth to allowable size / wastage until code allowable stresses are reached
  - Growth to net section collapse or loose parts release / wastage to cladding blowout
  - Small/medium LOCA and possible consequential damage / loose parts damage
  - Effect on core damage frequency (CDF)

Failure Modes and Effects Analysis: Simplified Block Diagram

PRELIMINARY



Failure Modes and Effects Analysis: Classification of Failure Conditions

- Each failure condition will be classified as:
  - Not credible,
  - Not actionable, or
  - Actionable
- A classification as “not credible” will require a strong technical argument and thorough documentation with a high threshold
- A classification as “not actionable” requires that adequate protection be provided at a higher level in the failure process
- Conditions classified as “actionable” will be inputs to the probabilistic and deterministic evaluations and will ultimately shape the detectability requirements specified in the inspection plan

Failure Modes and Effects Analysis: Additional Higher Order Factors

- Additional factors being considered in the FMEA include:
  - Environmental fatigue
  - Fabrication practices such as nozzle straightening or nickel plating
  - Surface and imbedded flaws produced during fabrications such as welding lack of fusion and hot cracking
  - The condition of the inside surface cladding
  - Primary water chemistry factors such as resin intrusions
  - Leaks from sources above the head
  - Plant-specific differences in the air flow across the head top surface

Failure Modes and Effects Analysis: Frequency of Occurrence

- Weibull reference curves based on the latest inspection results
  - Plant experience may support different curves for different nozzle material suppliers and different weld fabricators
- Crack growth rates based on MRP-55 and stress intensity factor calculations
- Existing small- and medium-break LOCA analyses
- Consequential damage assessments
- Loose parts damage assessments

Deliverables and Schedule

- A comprehensive safety assessment (SA) report will form the basis for a revised MRP inspection plan
- As appropriate, the SA report will reference other reports (e.g., the MRP report on crack growth rates of Alloy 600—MRP-55)
- Some calculations remain to be revised and extended, but much of the material to be incorporated into the SA report has already been completed in support of MRP-75
- Data developed subsequent to the initial release of the SA report will be evaluated for consistency with the SA evaluations once such data become available
- The MRP expects to be prepared to discuss the contents of the SA and the revised inspection plan summer 2003
- In the meantime, technical discussions with the NRC staff will continue

Status of Reactor Vessel Head Penetration Inspection Activities

Tom Alley, Duke Energy
Chair, Alloy 600 Inspection WG

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Presentation Outline

- CRDM Issue Background
- Top-of-head Visual Exam Guidance
- MRP Approach to NDE Demonstration
- 2001 Demonstration Process & Results
- 2002 Demonstration Process & Results
- Future Demonstration Activities
- Other Future Inspection Committee Activities
- Summary

CRDM Head Penetration NDE Background

- Original (97-01) demonstrations addressed cracks initiating on the inside surface of the penetration only
- Discovery of tube OD and weld cracking identified the need to modify the NDE demonstration program
  - Inspection technology required rapid development, deployment and field adaptation of existing inspection equipment
- Visual evidence of leakage vastly different from originally postulated
- First phase of MRP demonstrations was available to support fall 2001 inspections
  - Detection of "safety-significant" flaws in the tube
- Second phase performed to support fall 2002 inspections
  - J-groove weld flaws
  - More base metal flaws to evaluate depth sizing

2001 Demo Mock-ups - Oconee Specimens

Axially Oriented PWSCC

Axially Oriented PWSCC

- Specimen #56
  - OD-initiated PWSCC
    - Range of sizes & locations
  - Off-axis flaws (~45 degrees) are representative of circumferential flaw in outermost penetration

- Specimen #50
  - ID-initiated PWSCC

2002 Mockups – Tiger Team Goals

- Blind Mock-up
- Demonstrate sizing capabilities
- Full Scale Mock-up
- Establish Inspection Thresholds
- No POD
- Practice Blocks and Blind Blocks
- Include Effects of Crazed Cracking

MRP Demonstration Process

- All CRDM Head Penetration NDE demonstrations had the following characteristics:
  - Blind
    - supported by non-blind preparation phases
  - Procedure demonstration
    - No acceptance criteria
    - Demonstration best available techniques
      - ASME code will probably develop technique/personnel qualifications
  - Measurements of flaw detection capability and limits
    - No acceptance (pass-fail) criteria

MRP Demonstration Process

- Demonstration protocol
  - Vendor collects data on mockups & reports findings
    - evaluates measured vs. true values
    - Detection (# detected/total flaws)
    - Location with respect to pressure boundary
    - Sizing results documented
    - False call performance
  - NDE Center documents procedure essential variables
    - Allows verification that the techniques used are the same techniques that were demonstrated
  - Analysis process used in the demonstration and must be captured in the procedure
  - Results are provided to utilities

MRP Demonstration Process - Overview

- Complicated examination volume
- Vendor UT inspection procedures include <u>many</u> technique options and probe combinations, examples:
  - Open-tube probes
  - Blade probes
    - Probes are designed to accomplish specific objectives:
      - Specific volumes
      - Flaw orientations, e.g., circumferential or axial flaws
      - Detection technique, e.g., corner trap or tip diffraction
      - Sizing technique
- MRP Demonstrations document performance of individual probes/scans
  - More than one probe may be required to examine the specified inspection volume to detect/size specified flaw locations and orientations

2002 Demonstrations – Summary

- **Blade-Probe UT of penetration tube**
  - Flaws ranging ~ 15 to 100% TWE detected when flaws are oriented perpendicular to beam direction
  - Flaws ranging ~15 to 100% TWE detected when flaws are oriented parallel to beam direction
- **Open-tube "Rotating" probe of penetration tube**
  - Flaws ranging ~ 10 to 100% TWE detected when oriented perpendicular
  - Flaws ranging ~15 to 100% TWE detected when flaws are oriented parallel to beam direction
- **Open-tube "Rotating" probe of tube/weld interface**
  - Tube/weld interface flaw detected when flaw length extended to triple-point
  - Weld metal flaws that did not extend to the triple point were not detected.

Vendor A UT Detection Results

	Vendor A – UT Blade & Open Tube Probe Detection Results See Flaw Table 4 and drawing for description of flaw types "A" through "O"					
Field Used UT Techniques	**A, B, & C** ID Axial Flaws	**G, H, & I** ID Circumferential Flaws	**D, E, & F** OD Axial Flaws	**J, K, & L** OD Circumferential Flaws	**M, N, & O** Weld Flaws	**Cluster Flaws** OD Flaws under shallow (< 3 mm deep) ID Cluster Flaws
"Axial Blade" (TOFD UT COAF) (Note 1) **3 degree scan increment**	5%-86% TWE detected (Note 4) Orientation of flaws < 12 % TWE was inconsistent (Note 5)	11%-49% TWE detected Orientation of flaws < 12 % TWE was inconsistent	28%-100% TWE detected 4 flaws < 24% TWE missed: 1-D type flaw, 1-E type flaw, 2-EF type flaws	15%-100% TWE detected 1 K-type false call @ 7% TWE (Note 6)	(Note 7)	100% detection of ID & OD
"Circ Blade" (TOFD UT AOCF) (Note 2) **3 degree scan increment**	11%-86% TWE detected 1 B type flaw < 5% TWE missed Orientation of flaws < 12 % TWE was inconsistent (Note 5)	11%-49% TWE detected Orientation of flaws < 12 % TWE was inconsistent	15%-100% TWE detected 4 flaws < 13% TWE missed: 2-D type flaws, 1-E type flaw, 1-EF type flaw	15%-100% TWE detected 1 K-type false call @ 15% TWE (Note 6)	(Note 7)	100% detection of ID & OD
"Open-Tube" (Note 3) **5 degree scan increment**	5%-86% TWE detected (Note 5)	11%-49% TWE detected	13%-100% TWE detected 3 flaws < 12% TWE missed: 1-D type flaw, 1-E type flaw, 1-EF type flaw	15%-100% TWE detected 1 K-type false call @ 15% TWE (Note 6)	(Note 7)	100% detection of ID & OD
"Open-Tube" (Note 3) **3 degree scan increment**	5%-86% TWE detected (Note 5)	11%-49% TWE detected	13%-100% TWE detected 3 flaws < 12% TWE missed: 1-D type flaw, 1-E type flaw, 1-EF type flaw	15%-100% TWE detected 1 K-type false call @ 15% TWE (Note 6)	(Note 7)	100% detection of ID & OD
Notes:	(1) TOFD UT COAF (Circumferentially Oriented for Axial Flaws) used for detection and sizing of flaws. (2) TOFD UT AOCF (Axially Oriented for Circumferential Flaws) used for detection and sizing of flaws. (3) TOFD UT COAF/AOCF, Pulse/Echo, and 0 degree used for detection and sizing of flaws. (4) Through-wall-extent (TWE) of flaw depth in the tube thickness. (5) Inadequate resolution to separate closely associated (approx. 3 mm spacing) flaws. (6) Appears to be a welding defect at the tube-to-weld-interface. (7) Equipment and procedure were not optimized to resolve indications extending beyond the tube-to-weld interface in the weld volume.					

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2002 Demonstrations ET of Attachment Weld

- Detection is sensitive to weld surface conditions
  - Ground Surface Condition
    - Detected 0.16" long, 0.00031" wide
  - Un-ground (as-welded) Surface Condition
    - Detected 0.55" long, 0.00197" wide
    - Missed; 1.42" long, 0.00591" wide
  - Continue to pursue additional/alternate techniques to improve the detection capabilities

Future Demos

- **Tecnatom**
  - ET of Attachment Weld
    - May 2003
- **Framatome**
  - ET of Attachment weld
    - February 2003
  - "Other" surface method for wetted surface of attachment weld
    - 1st quarter of 2003
- **WesDyne**
  - UT of tube/weld interface
  - ET of attachment weld
  - Thermal imaging
- **B&W Canada**
  - UT of tube/weld interface
    - 1st quarter of 2003
  - ET of attachment weld
    - 1st quarter of 2003

Future Activities

- New mock-ups under construction
  - Existing mock-ups will be made available to vendors for personnel training and technique refinement
- Replacement head inspection
  - Equivalence studies
  - Mock-up drawings
  - Pre-Service Inspection Requirements
- Bottom Head Inspection
  - Document industry (including international) inspection capabilities and experience

Summary

- MRP has organized a comprehensive approach to address recent industry events
- Considerable progress has been made in a short amount of time
- Demonstrations continuing
- Emphasis on examination of attachment weld and increased inspection efficiency

Overview of DG-1119
Guidelines for Evaluating EMI/RFI in Safety-Related I&C Systems

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Engineering Research Applications Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

Richard T. Wood and Paul D. Ewing
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Presented to

Advisory Committee on Reactor Safeguards
Rockville, Maryland
March 6, 2003

Outline of Presentation

- Overview of DG-1119
- Technical Basis for Electromagnetic Compatibility at Nuclear Power Plants
- Summary of DG-1119 Benefits and Value

DG-1119 Endorses Current Electromagnetic Compatibility (EMC) Standards for Safety-Related I&C Systems

- Describes design, installation, and implementation practices to evaluate and minimize the impact of EMI/RFI and power surges on I&C systems
  - Scope covers analog, digital and hybrid equipment
  - Addresses emissions, susceptibility, and surge withstand testing
  - Describes grounding and shielding practices
- Applies to new or modified safety-related I&C equipment in existing and future nuclear power plants
- Endorsement of the guidance in IEC 61000 is introduced
- Technical basis is well documented
  - Enhanced basis for update given in NUREG/CRs 5609 and 6782
  - Basis for existing guide given in NUREG/CRs , 5941, 6431, and 6436

Why is DG-1119 Needed?

- DG-1119 Updates Regulatory Guide 1.180
- Responds to NRR User Need Request 2002-017
- US military standard for EMC has been revised (MIL-STD 461E)
- Alternate testing practices are available through international EMC standard (IEC 61000)
- Guidance to address additional EMC issues is needed
  - Conducted susceptibility along interconnecting signal leads
  - Susceptibility and emissions testing in frequency range above 1 GHz
- Industry has asked for relaxation in EMC testing criteria (e.g., operating envelopes) where technically justified

Resolution of Public Comments

- Four correspondents submitted comments on DG-1119
- Public comments can be grouped into general categories
  - Operating envelopes
  - Testing above 1 GHz
  - Surge testing for signal lines
  - Relation to previous guidance
  - Test methods and exemptions
- DG-1119, Rev. 1, reflects resolution of comments

Technical Basis For Electromagnetic Compatibility Guidance

DG-1119 Employs U.S. Military and Industrial Experiences to Establish a Systematic Approach for EMC and Offers Option for International Standard

- Endorses commercial standards for design and installation practices
  - Grounding and shielding of I&C equipment (IEEE Std 1050)
- Endorses well-established testing standards
  - Power surge withstand (IEEE Stds C62.41 and C62.45 and IEC 61000)
  - Electromagnetic emission and susceptibility requirements for the control of EMI/RFI and test methods to demonstrate compliance (MIL-STD 461 and IEC 61000)
  - Tailored test limits (operating envelopes) to conditions found in nuclear power plants and industrial sites
  - Allows exemption of some test criteria under certain conditions based on technical considerations

DG-1119 Revises the Existing Guidance Provided in RG-1.180

- Updates endorsement of MIL-STD test methods to latest version (MIL-STD 461E)
- Introduces alternate testing options using IEC 61000 test methods
- Permits use of FCC or CISPR certification under specified conditions
- Includes signal line conducted susceptibility test methods
- Extends frequency range for radiated emissions and susceptibility testing above 1 GHz
- Provides enhanced surge withstand capability operating envelopes

Position 1 Outlines the Elements of a Systematic Electromagnetic Compatibility Program for Safety-Related I&C Systems

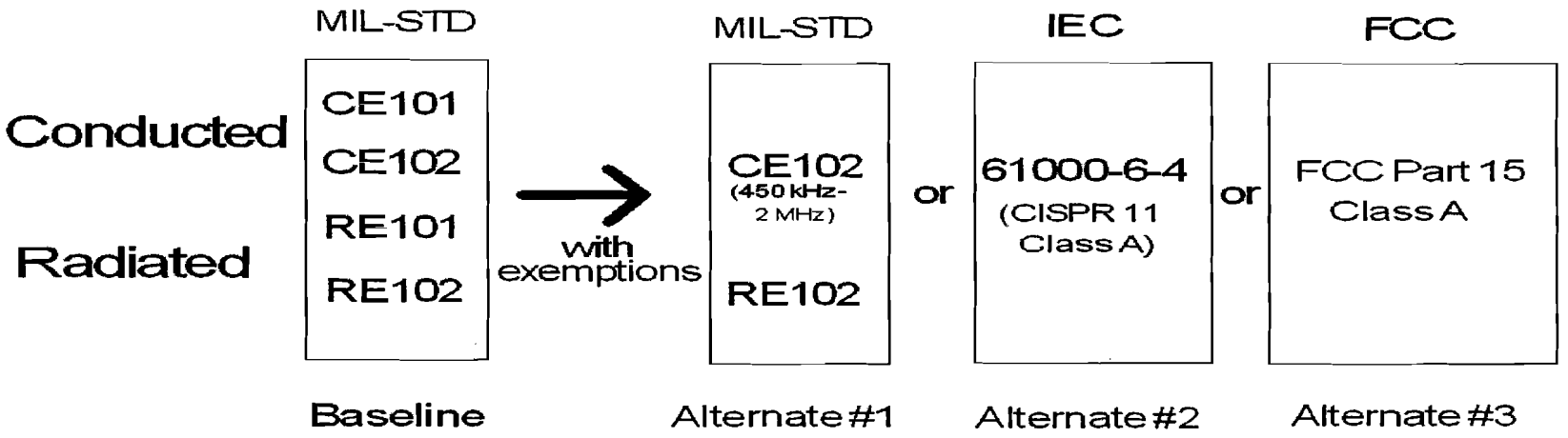
- Design and installation practices to limit the impact of electromagnetic effects
- Testing practices to assess the emissions of equipment and their susceptibility to EMI/RFI
- Testing practices to evaluate the power surge withstand capability of equipment
- Operating envelopes characteristic of the electromagnetic environment in nuclear power plants

Position 2 Endorses Design and Installation Practices in IEEE Std 1050-1996 With One Exception

- Provides guidance on engineering practices needed to minimize the potential impact on safety-related I&C systems exposed to EMI/RFI and power surges
- Addresses grounding and noise-minimization techniques in a generating station environment
- Exception taken to Section 4.3.7.4 on radiative coupling concerning field strength attenuation with distance

IEEE Std 1050-1996 is presently under revision and exception is being submitted for review

Position 3 Endorses EMI/RFI Emissions Tests from MIL-STD 461E, IEC 61000-6, and FCC Part 15, Along With Associated Operating Envelopes



- CE101 exemption is allowed if power quality control is employed
- RE101 exemption is allowed for equipment not intended to be installed in the proximity of magnetic field emitters

Position 4 Endorses EMI/RFI Susceptibility Tests from MIL-STD 461E and IEC 61000-4, Along With Associated Operating Envelopes

	Baseline MIL-STD		Alternate IEC	
	Power	Signal	Power	Signal
Conducted	CS101 CS114	CS114 CS115 CS116	61000-4-6 61000-4-13 61000-4-16	61000-4-6 61000-4-16 61000-4-4 61000-4-5 61000-4-12
Radiated	RS101(*) RS103		61000-4-8(*) 61000-4-9(*) 61000-4-10(*) 61000-4-3	

(*) Exemption based on proximity to magnetic field emitters

Position 5 Endorses IEEE Std C62.41 and IEC 61000-4 Power Surge Withstand Tests, Along With Their Associated Operating Envelopes

- IEEE Std C62.41 Ring Wave or IEC 61000-4-12
  - Simulates high-frequency oscillatory surges
- IEEE Std C62.41 Combination Wave or IEC 61000-4-5
  - Two exponential waveforms
    - open-circuit voltage
    - short-circuit current
  - Represents direct lightning discharges, fuse operation, or capacitor switching
- IEEE Std C62.41 Electrically-Fast Transients or IEC 61000-4-4
  - Emulates local load switching

Power Surge Withstand Capability Operating Envelopes

Table 22 Surge Withstand Levels for Power Lines

Surge Waveform	Category B Low Exposure	Category B Medium Exposure	Category C Exterior
Ring Wave	2 kV	4 kV	N/A
Combination Wave	2 kV / 1 kA	4 kV / 2 kA	6 kV / 3 kA
EFT	2 kV	4 kV	N/A

Position 6 Endorses MIL-STD 461E Testing Above 1 GHz

- RE102 test is applicable above 1 GHz for up to 10 times the highest intentionally generated frequency within the equipment under test
- RS103 test is applicable in the frequency range 1 GHz to 10 GHz, covering the unlicensed frequency bands where much of the high-frequency communications activity is taking place (2.45 GHz and 5.7 GHz)

Position 7 Provides Documentation Guidance for EMC

- Documentation should provide evidence that
  - Safety-related I&C systems are compatible with the electromagnetic environment
  - Acceptable installation practices are followed
  - Administrative emissions controls are in place

Summary of DG-1119 Benefits and Value

Benefits of DG-1119

- Comprehensive guidance on an acceptable method for electromagnetic compatibility of safety-related I&C systems
- Endorsement of current national and international EMC standards
- Specific guidance to address previously unresolved issues
  - Conducted susceptibility along interconnecting signal leads
  - Emissions and susceptibility testing for very high frequency phenomena (> 1 GHz)
- Relaxation of test criteria where technically justified
  - Operating envelopes
  - Opportunity to invoke FCC or CISPR emissions certification

Public Comments Were Addressed In Revised Draft Regulatory Guide

- Identification of IEC test limits was clarified (i.e., use of standard testing levels noted)
- Illustration of alternate test options added for clarity
- Surge operating envelope guidance enhanced to address additional locations and surge environments
- Emissions testing above 1 GHz added

Final Effective Guide Will Support NRC Mission

- Contributes to Achieving NRC Goals
  - Maintaining Safety by providing an enhanced approach for establishing electromagnetic compatibility for safety-related I&C systems in nuclear power plants
  - Reducing Regulatory Burden by providing alternate testing suites and relaxing selected test criteria where technically justified
  - Improving Regulatory Effectiveness by giving comprehensive guidance on acceptable practices for electromagnetic compatibility that utilize current standards

Presentation to the Advisory Committee on Reactor Safeguards

EARLY SITE PERMIT (ESP) AND ESP REVIEW STANDARD

Presented By

Ronaldo Jenkins

Michael Scott

Early Site Permit Project Managers

New Reactors Branch

9

Purpose

- Brief the Committee on status of activities leading up to receipt of first three ESP applications
- Brief the Committee on the contents of the draft ESP Review Standard (RS), in conjunction with Committee review of draft document and potential letter to the Commission
- Discuss future milestones for ESP RS document development and use
- Address Committee questions on [illegible] comments on ESP process

Agenda

- ESP issues and planned activities 30 min
- ESP RS document development approach 10 min
- ESP RS document content 30 min
- Plans for further development and use of ESP RS 10 min

ESP Issues

- NRC staff has met several times with the Nuclear Energy Institute (NEI) and potential applicants to facilitate resolution of ESP issues prior to submittal of applications
- Staff has sent letters to NEI to document staff positions
- Staff is also developing a SECY paper to communicate positions on ESP issues to the Commission
- No additional ESP-related meetings are planned with NEI before applications are submitted
- Emergent issues raised by applicants will be resolved on a case-by-case basis

Staff Positions on ESP Issues

- The staff expects ESP applicants to use applicable QA controls equivalent to those in Part 50 Appendix B for ESP activities that would affect the design of future safety-related systems, structures, and components
- Applicants may use a plant parameter envelope (PPE) to provide the design information required by regulations in lieu of providing a specific plant design
- NRC will [illegible]

Staff Positions on ESP Issues (Continued)

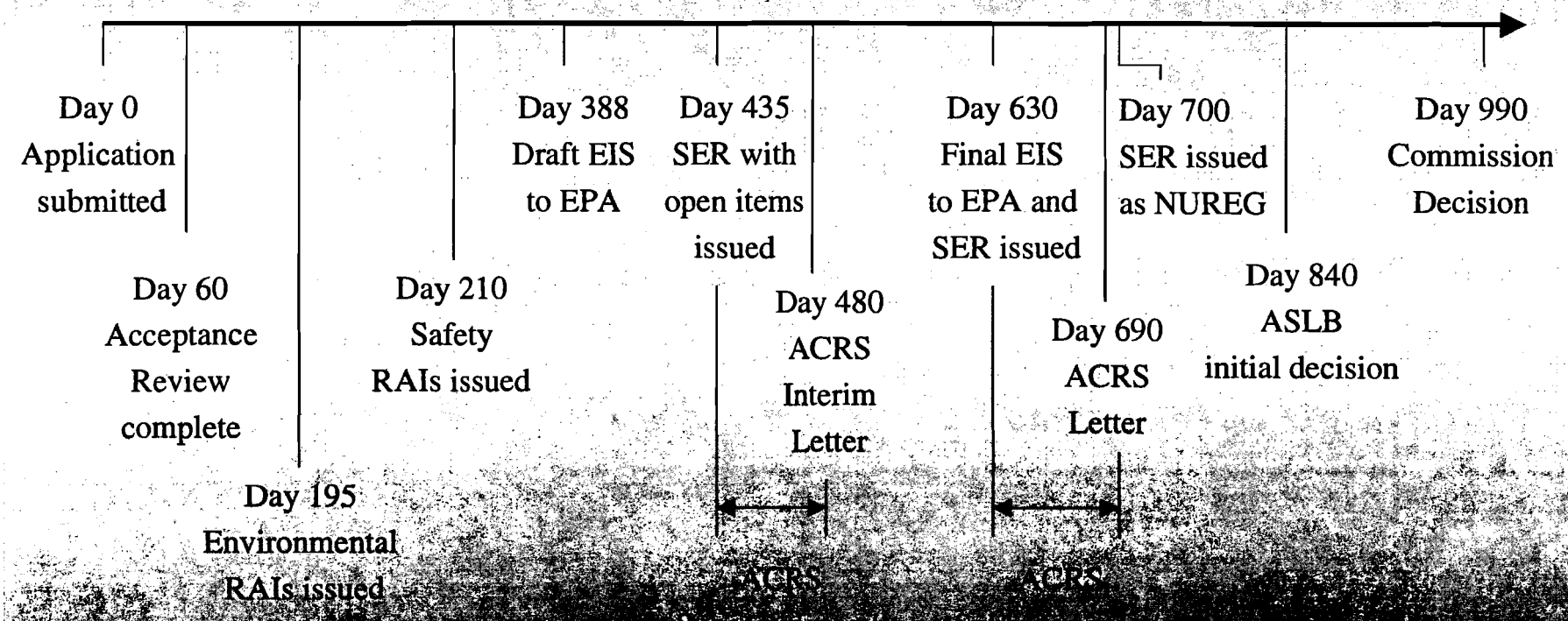
- ESP applicants must provide radiological dose consequence evaluations; they may use the PPE approach
- ESP applicants are expected to evaluate severe accident impacts but may defer severe accident mitigation alternatives to the combined license (COL) stage if information is not available at the ESP stage
- A position on evaluation of alternative sites is [illegible]

Other Activities

- Conduct local public meetings
  - Grand Gulf: Nov. 14, 2002
  - Clinton: Mar. 20, 2003
  - North Anna: Apr. 1, 2003
- Plan additional public meeting in Rockville on May 14, 2003 to inform and receive feedback from public on ESP review process and ESP Review Standard
- Developed nominal ESP review and decision time line in [illegible]

Nominal ESP Timeline

- 22 months from receipt to Commission decision



ESP Review Standard (RS)

- Purpose of ESP RS is to provide guidance to staff and information to stakeholders on review of an ESP application
- Use of existing guidance to the extent feasible was the basic premise in the development of the ESP RS
- While staff sought consistency with draft power uprate RS and license renewal guidance, draft ESP RS and power uprate RS emerged quite different in format and content [illegible] situation [illegible]

Document Development Status

- Draft ESP RS released for interim use and public comment in December 2002; comment period ends March 31, 2003
- Provided copies of document to Committee in January 2003 for review
- Additional guidance being developed on QA, accident analysis, and physical security – scheduled to be released for public comment in March 2003; copies to be provided to Committee

Document Development Approach

- Staff tasked to review for applicability to ESP:
  - NUREG-0800 (1981) (Standard Review Plan, SRP)
  - NUREG-1555 (1999) (Environmental SRP)
  - Regulatory Guides
  - Information Notices
  - Generic Letters
  - Regulatory Issue Summaries
  - Any other potentially applicable regulatory documents (e.g., other NUREGs)

Standard Review Plan Markups

- Primary review branch asked to mark up assigned sections of NUREG-0800 and NUREG-1555 to achieve two results:
  - Strike out text not applicable to ESP to clearly show what is needed and what is not at ESP stage
  - Revise (using highlight and strikeout) existing guidance to bring portions of sections applicable to ESP up to date

Results of Staff Markups

- Most applicable sections of NUREG-0800 are in Chapter 2 (Site Characteristics)
- Certain sections (e.g., 2.4.8 and 2.4.10) were found to not be needed for ESP review, since applicable parts of those sections were adequately addressed in other sections
- Additional sections were found to be applicable:
  - QA (new guidance being developed)
  - Security (new guidance being developed)
  - Site missiles and aircraft hazards
  - Emergency [illegible]
  - [illegible]

Results of Staff Markups (Continued)

- Markups made on all "old" NUREG-0800 sections (most of which date to 1981)
- Markups appended to ESP RS; ESP RS stands alone without reference to NUREG-0800 (except for three much newer NUREG-0800 sections for which markups were not needed)
- Markups apply only to ESP and do not constitute draft revisions to NUREG-0800

Results of Staff Markups (Continued)

- Few changes needed to NUREG-1555, which is much more recent than most of NUREG-0800 and contains specific references to ESP
- Minor clarifications for ESP noted in ESP RS "Applicability Table for Environmental Report" (Attachment 3)
- "Snapshot in time" nature of environmental impact statement reduces need and feasibility of attempting to specify what is needed at each licensing stage

RS Document Contents

- RS-002: ESP review process guidance
- Attachment 1: Review process flow chart
- Attachment 2: "Applicability table" for safety evaluation, followed by NUREG-0800 section markups
- Attachment 3: "Applicability table" for environmental impact statement
- Attachment 4: Sample safety evaluation report text

Issues Addressed During Document Development

- Design information needed and how to discuss in ESP RS
  - Lack of precedent and lack of specificity in 10 CFR Part 52 led to uncertainty about what design information is needed at ESP. Minimum amount needed? Or "the benefit to the applicant depends on the amount of information provided"?
  - Difficult in some NUREG-0800 sections to discern information needs for ESP and COL (gray areas); staff made best effort but will need to consider these areas further after receipt of public comments
  - Elected not to attempt to directly address NEI plant parameter envelope (PPE) approach; called for applicant to specify type and number of plants consistent with "should" statement in 10 CFR 52.17
  - Staff will [illegible]

Issues Addressed During Document Development (Continued)

- Applicability of QA and Appendix B at ESP stage (Note 7 in Attachment 2 to ESP RS; also new section under development)
  - 10 CFR Part 52 does not require Part 50 Appendix B QA program
  - Finality of NRC determinations at ESP stage requires QA controls equivalent in substance to Appendix B be applied
  - New QA section of ESP RS will address this distinction

Issues Addressed During Document Development (Continued)

- Radiation protection for construction workers on new plant adjoining existing plant (Note 6 in ESP RS Attachment 2)
  - Applicant and licensee the same: Licensee ensures compliance with 10 CFR 20 and discusses impacts on construction workers in environmental report
  - Applicant and licensee different: Licensee ensures compliance with 10 CFR 20; applicant discusses impacts on construction workers in environmental report

Issues Addressed During Document Development (Continued)

- Adequacy of coverage for subsurface investigations
  - Reasonable assurance that actual site conditions revealed during excavation will be consistent with model developed for ESP
  - ESP license condition requiring reporting of information having significant implication for public health or safety
- Path forward if inadequate meteorological data not available
  - Application would be denied (change from NUREG-0800, which allowed for [illegible] to obtain [illegible] data)

Next Steps for ESP RS

- Incorporate public and Committee comments
- Develop new draft by June 2003, without use of redline/strikeout
- Incorporate lessons learned from acceptance review of initial ESP applications
- Issue "final" ESP RS (Rev. 0) in December

(11)

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
500	10	10.1

Title **PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES**

Authors
JOHN T. LARKINS

List of Documents Attached

10

Instructions to Preparer	From Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	JOHN T. LARKINS/

INTERNAL USE ONLY

G:PlanPro(ACRS):ppmins.500
March 6, 2003

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
MARCH 5, 2003

The ACRS Subcommittee on Planning and Procedures held a meeting on March 5, 2003, in Room T2B3, 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 1:30 p.m. and adjourned at 3:15 p.m.

ATTENDEES

MEMBERS

M. Bonaca
G. Wallis
S. Rosen
J. Sieber (part time)

ACRS STAFF

J. T. Larkins
S. Bahadur
S. Duraiswamy
R. P. Savio
J. Gallo
S. Meador

NRC STAFF

i. Schoenfeld, OEDO

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the March ACRS meeting

Member assignments and priorities for ACRS reports and letters for the March ACRS meeting are attached (pp. 6-8). 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 March ACRS meeting be as shown in the attachment (pp. 6-8).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through May 2003 is attached (pp. 6-8). 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 will also discuss and develop recommendations on items included in Section II of the Future Activities List (p. 9).

RECOMMENDATION

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

3) Update on the Budget

The continuing resolution has ended, and we now have our budget. However, given the increase in workload and several special events (quadripartite meeting, 500th meeting, etc.) funds for the remainder of this fiscal year (September 30, 2003) are limited, particularly in the travel category.

RECOMMENDATION

The Subcommittee recommends the following:

- Travel for events other than scheduled full Committee and Subcommittee meetings should be approved by John T. Larkins prior to travel.

- All purchases using the Government Bankcard should be approved by Tanya.

4) ACRS Meeting with the NRC Commissioners

A meeting between the ACRS and the NRC Commissioners is scheduled for April 11, 2002, between 9-11:00 a.m. to discuss items of mutual interest. Items approved by the Commission for this meeting are as follows:

(1) Overview by the ACRS Chairman (MVB)
- 500th meeting celebration
- Quadripartite Meeting
- License Renewal Activities
- Core power uprates

- Future ACRS activities - including "sunset" of activities that do not necessarily need to be reviewed.

(2) Advanced Reactor Designs (TSK)
- Early Site Permit Process
- Options paper for resolving policy issues
- AP1000 review activities

(3) Pressurized Thermal Shock Technical Basis Reevaluation Project (WJS)

(4) ACRS Report on the NRC Safety Research Program (FPF)

The Commission has approved the above agenda with the recommendation that the PTS discussion be limited to about 5-10 minutes.

RECOMMENDATION

The Committee needs to approve the slides at this month's meeting with the plan for finalizing them prior to March 28, so that they can be copied and distributed to the Commission a week or so in advance of the meeting.

5) Role and Use of PRA in the Regulatory Decisionmaking Process

Mr. Fleming, under a contract with NRC/ACRS, submitted a draft report on the role and use of PRA in risk informed decisionmaking process to the Committee. The Committee discussed this report with Mr. Fleming during its February 2003 meeting.

During the February 2003 ACRS meeting, Dr. Nourbakhsh, ACRS Senior Fellow, presented an outline of a draft "White Paper" and recommendations to be included in the paper based on Mr. Fleming's report .

Dr. Nourbakhsh has worked with Carl Fleming to edit his report and is in the process of having it published as a NUREG. We anticipate this being issued in the next month.

Dr. Apostolakis had prepared a draft report to the Commission instead of a White Paper as previously discussed in February. Several members commented on the draft report prepared by Dr. Apostolakis

RECOMMENDATION

The Subcommittee recommends that Dr. Apostolakis revise the report to include more explicitly some members' comments and see if there are other issues in the Fleming report that should be included in the letter on enhanced use of PRA in Regulatory Decision making. Additionally, the Committee should decide whether to include a transmittal memorandum with Mr. Fleming's report and if so, what should be the message.

6) Comments by Mr. Robert H. Leyse, a Member of the Public, Regarding Fouling in Reactivity Insertion Accidents

Mr. Leyse, a member of the public, submitted a petition for rulemaking (pp. 10-16) to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants. This includes all heat transfer surfaces, fuel elements, steam generators, condensers, fan coolers, etc. The new regulations must also require the inclusion of fouling considerations in the NRC-funded test programs such as the Rod Bundle Heat Transfer Program at the Penn State University. Some of the comments provided by Mr. Leyse for evaluation by the ACRS are as follows:

- Even though fouling of nuclear reactor fuel elements has a greater impact on the Reactivity Insertion Accidents and LOCAs than high-burnup, the NRC and its collaborators continue to totally ignore the impact of fouling.

- There was no discussion of fouling either at the October 9, 2002 ACRS Subcommittee meeting on Reactor Fuels or at the October 2002 full Committee meeting. In addition, the October 17, 2002 ACRS letter on the Confirmatory Research Program on High-Burnup Fuel did not include any discussion on fouling.

- The Rod Bundle Heat Transfer Program at the Penn State University includes no provisions for the impact of fouling on the heat transfer surfaces. Surprisingly, the ACRS did not point out the gross inadequacy of this Program. Copies of the information provided by Mr. Leyse were sent to Dr. Powers and Dr. Wallis.

The Subcommittee discussed several options for handling the concerns raised by Mr. Leyse, including the following:

- Should this matter be referred to the EDO, as has been done in the past on similar issues?
- Should Mr. Leyse be notified by the ACRS Executive Director that the ACRS does not get involved in the petition for rulemaking process and that the Reactor Fuels Subcommittee plans to invite him to discuss his concerns at the next meeting of the Reactor Fuels Subcommittee, which is tentatively scheduled to be held in October 2003?

RECOMMENDATION

The Planning and Procedures Subcommittee recommended that the ACRS/ACNW Executive Director followup with the EDO to find out what is happening with the petition for rulemaking. Dr. Larkins should draft a response to Mr. Leyse after Drs. Powers and Wallis recommend a proposed course of action.

7) Committee Review of License Renewal Applications (Peach Bottom Event)

There have been a number of e-mails exchanged (see handout) regarding the ACRS review of license renewal applications related to plants whose current operating experience raises the question of whether their license renewal applications should be approved. Most recently, this issue has come up in connection with the license renewal application for Peach Bottom Units 2 and 3, which was discussed by the Committee at its March 2003 meeting.

The applicant, NRC staff, and the Senior Resident Inspector addressed the reactor scram event which occurred at Peach Bottom in December 2002 along with the corrective action programs at the March meeting.

The ACRS has previously moved ahead with its review of license renewal applications in spite of current operating experience with the idea that these conditions would be corrected prior to entering into the renewal period.

RECOMMENDATION

The Subcommittee recommends that based on the information provided by the applicant and the staff, the Committee discuss how it wants to treat the license renewal applications of those plants that had experienced significant operating events recently, which brings into question the performance of the applicant.

8) Insights/Safety Culture

During the February 2003 meeting, the Committee discussed a draft report on Safety Culture written by ACRS members Rosen and Apostolakis and requested that Mr. Rosen revise this report incorporating the comments from the members. It was suggested that the letter be reasonably short and minimize recommendations to suggesting that the Commission encourage the staff to stay aware of developments in the Safety Culture arena both domestically and internationally.

It was further suggested that the ACRS meet with the NRC staff, First Energy Nuclear, NEI, and others (i.e., David Collins and Howard C. Whitcomb) later in the year at a subcommittee meeting to discuss initiatives related to assessing safety culture and its implications on the safe operations of nuclear facilities.

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the draft report prepared by Mr. Rosen and attempt to develop a final draft report during the March 2003 meeting and issue a letter at the April 2003 full Committee meeting.

9) Member Issues

- Emergency Notification to Members Regarding Subcommittee Cancellations

  Recent events have noted the importance of having some process in place to notify members in the event of a severe weather condition which would inhibit travel to and from the Washington area. A process is being put in place to address this issue.

- The issue has been raised in terms of how the Committee should handle differing technical views among the NRC staff after the Committee has approved a particular staff action such as DG 1077 EQ of Microprocessor-Based Equipment.

- The ACRS Executive Director was successful in getting an exemption to the Commission reappointment policy in having Dr. Thomas Kress appointed for a fourth term to begin at the end of his current term (September 3, 2003). The Commission, in the SRM, approved the reappointment and noted that this exception to its policy will be done sparingly.

- The NRC staff is planning a one week tour of a test facility associated with SWR 1000 (located in Germany) in early June 2003. The NRR staff is expecting possibly 2-3 NRC managers will attend, and wondering if an ACRS member (maybe Dr. Kress) would be interested in attending such a tour. Framatome is sponsoring the tour, however, Framatome is not responsible for the travel and hotels.

ANTICIPATED WORKLOAD
MARCH 6-8, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Nourbakhsh/ Duraiswamy	Letter report on the role and use of PRA in the regulatory decisionmaking process	--	--	**Draft**
Bonaca	--	Kobetz	Overview of the format and content of the Ft. Calhoun license renewal application	--	--	--
	All Members	Larkins	Preparation for meeting with the NRC Commissioners, April 11, (9:00-11:00a.m.)	--	--	--
Ford		Weston	Vessel head penetration cracking and vessel head degradation-Information Briefing	--	--	--
Kress		El-Zeftawy	Early Site Permit Process	A	To provide early feedback to the Commission	**Draft 1**
Leitch	Bonaca	Assa/Kobetz	Peach Bottom License Renewal Application-**FINAL REVIEW**	A	To meet the CTM schedule	Draft 1
Rosen	--	Weston	Safety Culture Report	--	--	Draft
Sieber	Apostolakis	Weston	Reactor Oversight Process	A	To respond to Commission SRM	--
		El-Zeftawy	Draft final revision to Reg. Guide 1 .180 (DG-1119), "Review Guidelines for EMI/RFI."	A	To support the staff schedule	--

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ANTICIPATED WORKLOAD
APRIL 10-12, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly/ Duraiswamy	Draft final Regulatory Guide (DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities"	A	To meet the CTM schedule	--
Bonaca	All Members	Larkins	Meeting with the NRC Commissioners - April 11, 2003, 9:00-11:00 A.M.	--	--	--
		Kobetz	St. Lucie license renewal application - INTERIM REVIEW-SUBCOMMITTEE REPORT	--	--	--
Leitch	--	Weston	Significant recent operating events	--	--	--
Powers	--	El-Zeftawy	Reg. Guides and Generic Letter on Control Room Habitability	A	To meet the CTM schedule	--
Rosen	--	Weston	Safety Culture Report	A	To provide Committee's views to the Commission	Draft
Shack	--	Snodderly/ Duraiswamy	Draft final risk-informed revisions to 10 CFR 50.44 regarding combustible gas control systems.	A	To meet the CTM schedule	--
Wallis	--	Snodderly/ El Zeftawy	Subcommittee Report - AP 1000 Thermal-Hydraulic Issues	--	--	--

2

ANTICIPATED WORKLOAD
MAY 8-10, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Ford	--	Weston	Vessel head penetration cracking and vessel head degradation	A	To provide feedback to the Commission	--
Kress	--	Kobetz	Proposed criteria for the treatment of individual requirements in a regulatory analysis	A	To provide feedback to the staff	--
Shack	--	Snodderly/ Duraiswany	Proposed revisions to SRP Section 3.9.8 and Reg. Guide 1.178 for risk-informed inservice inspection of piping	A	To support the staff's schedule	--

II. ITEMS REQUIRING COMMITTEE ACTION

1. Draft Regulatory Guide DG-1079 (Proposed Revision 3 to Regulatory Guide 1.32), "Criteria for Power Systems for Nuclear Power Plants" (Open) (JDS/TJK)
Estimated Time: 1 hour

Purpose: Determine a Course of Action

Review requested by the NRC Staff [S. Aggarwal, RES]. The draft Regulatory Guide DG-1079 (proposed Revision 3 to Regulatory Guide 1.32) endorses IEEE Standard 308-2001, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," with one exception, namely, the NRC does not endorse IEEE Standard 308-2001 criteria for sharing safety-related dc power systems in multi-unit nuclear plants. The staff believes that there is general industry consensus that IEEE Standard 308-2001 provides acceptable guidance for design of safety-related power systems. The staff plans to issue DG-1079 for public comment and requests that the ACRS review the draft final version of this Guide after reconciliation of public comments.

Mr. Sieber recommends that the Committee review the draft final Guide after reconciliation of public comments.

2. Draft Final Amendment to 10 CFR 50.55a to Incorporate by Reference ASME Boiler and Pressure Vessel and OM Code Cases" (Open) (WJS/MRS)
Estimated Time: 1½ hour

Purpose: Determine a Course of Action

Review requested by the NRC Staff [H. Tovmassian, NRR]. The staff proposes to amend its regulations to incorporate by reference certain NRC Regulatory Guides listing Code cases published by the ASME which the NRC staff has reviewed and found to be acceptable for use. These Code cases provide alternatives to requirements in the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants pertaining to construction, inservice inspection, and inservice testing of nuclear power plant components. The amendment incorporates by reference three regulatory guides that address NRC review and approval of ASME published Code cases. As a result, the Code cases in these regulatory guides are incorporated by reference into the NRC's regulations. During the February 7-8, 2002 Meeting, the Committee decided not to review the proposed amendment and asked for an opportunity to review the draft final rule after reconciliation of public comments. The staff is requesting that the ACRS review the draft final rule package which includes resolution of 16 comments..

The Planning and Procedures Subcommittee recommends that Dr. Shack propose a course of action.

9

Robert H Leyse
P O Box 2850
Sun Valley, ID 83353

February 10, 2003

John T. Larkins, Executive Director
ACRS
USNRC
Washington, DC 20555-0001

Dear Executive Director:

This member of the public, in accordance with the provisions of the Federal Advisory Committee Act, submits the following written comments for evaluation by the ACRS The substance of these comments is included in the three documents that are in this letter and that are, or will be, on the NRC's Rulemaking Website

The first document, pages 1-3, is a Petition for Rulemaking that directs the NRC to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants The second document, pages 4-7, NRC Ignores **Fouling** in RIA and LOCA Analyses, is a comment related to the Petition The third document, page 8, NRC Ignores **Fouling** in Reliability and Probabilistic Risk Assessment, is another comment related to the Petition. I have also copied the list of other public comments from the NRC's Rulemaking Website and the ACRS may access and review these as it chooses

Petition and Other Documents

Original File	Size	File Type	Date	HTML	Description
1210-0003.htm	4k	HTML	01/02/03	1210-0003.htm	Federal Register Notice - Extension of Comment Period
1210-0002.wp	8k	Word Perfect	10/31/02	1210-0002.htm	Incoming petition
1210-0001.htm	11k	HTML	10/31/02	1210-0001.htm	Federal Register Notice

Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C., 20555-0001
Attn: Rulemakings and Adjudications Staff

September 2, 2002

Petition

10

1. Brief description of needed regulations:

Regulations are needed to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants. This includes all heat transfer surfaces, fuel elements, steam generators, condensers, fan coolers, etc. The new regulations must also require the inclusion of fouling considerations in NRC funded test programs such as the Rod Bundle Heat Transfer (RBHT) at Penn State. Furthermore, fouling considerations must be added to the vast assortment of NRC produced computer codes such as the RELAP and TRAC series.

2. Petitioner's grounds for and interest in the action requested:

Petitioner is aware that the fouling of heat transfer surfaces is generally not adequately considered in the licensing and the compliance inspections of nuclear power plants.

3. Petitioner's statement of the specific issues involved, views with respect to those issues, relevant technical, scientific, or other data involved:

The following examples cover only a few of the significant heat transfer surfaces within operating nuclear power plants. The NRC must produce a complete inventory of all significant heat transfer surfaces. The regulations must require appropriate reporting of the performance of these surfaces including records of degradation, cleaning procedures and effectiveness, mechanical degradation of the heat transfer assemblies especially the fuel assemblies. Regulations must require detailed reporting that must also be available to the public.

Regulations do not address the significance of severe fouling of the nuclear fuel elements. Currently, Licensing Bases and Technical Specifications do not limit the amount of fouling of fuel elements. In some cases, the fouling is sufficient to induce significant oxidation of the fuel cladding, a situation that has led to "... a debate over (whether) the 17 percent includes the prior oxidation or it's just the oxidation during the ramp-up." (See ACRS Subcommittee transcript, May 31, 2002). In another case, the axial offset anomalies that were traced to fouling of nuclear fuel elements, the situation was evaluated as follows, "They are a(n) annoyance. They affect economics, but they are not safety issues." (See ACRS Subcommittee transcript, April 24, 1998.) Severe fouling of nuclear fuel elements also leads to axial growth of the fuel rods beyond design limits as the operating temperatures of the fuel rods become greater than allowed for in design. The fuel rods may expand sufficiently along their length to become restrained from further axial growth by the fuel assembly end fittings; the rods then bow and contact adjacent rods as well as control rod guide tubes.

While one nuclear power plant continued to operate at power, the need for repeated cleaning of an air cooling heat exchanger was not recognized as a key indicator of a substantial leak in the primary reactor system. As long as the plant operation remained within the technical specifications, there was no basis for the plant operators to perform prudent investigations. Clearly, regulations must address the need for investigating the grossly off-normal performance of this heat exchange equipment.

In several instances, the fouling of steam generator tubes has reduced the heat transfer effectiveness sufficiently to force operation at reduced secondary side pressures in order to maintain heat transfer rates. In the overall pattern, this likely will impact safety issues in addition to being an operating annoyance.

The fouling of main condenser heat transfer surfaces has led to a degradation of heat transfer effectiveness. Furthermore, on occasion these fouling deposits have been released into the coolant stream and then have contributed to fouling of fuel elements.

Shifting to the arena of test programs, during the past several decades the NRC and its defunct predecessors have funded well over one billion dollars of heat transfer test programs that have not included any allowance for the severe fouling of heat transfer surfaces that occurs during the operation of nuclear power plants. These test programs must be thoroughly studied and allowances must be made for a range of fouling of the heat transfer surfaces. Very likely, it will not be possible to produce reliable allowances for a range of degrees of fouling in which case, the results of the prior test programs such as FLECHT, LOFT, Semiscale and others must not be applied to the production of computer codes for reactor heat transfer analyses. Currently, the NRC is spending millions of dollars in heat transfer testing at facilities such as RBHT at Penn State, and these programs must be realigned to cover the cases of several degrees of fouling.

Shifting to the arena of computer codes, during the past several decades the NRC and its defunct predecessors have funded several hundred million dollars of code production related to heat transfer processes in nuclear power reactors. The codes thus produced, several versions of TRAC, RELAP and others, have included no provisions for a range of fouling conditions on the heat transfer surfaces. These codes must not be applied to the licensing of nuclear power plants until reliable allowances for a range of degrees of fouling are incorporated in the codes.

One measure of whether proper regulations are effected will be a determination that under the new regulations, conditions similar to those already reported in certain Licensee Event Reports, will henceforth constitute license violations. Very likely there is a preponderance of documented nuclear power plant operating experience that is worthy of such review; as a possible example, one candidate may be Licensee Event Report 50-458/99-016-00.

The increased attention to detail in plant design, analysis and operations that will be effected via these needed regulations will enhance operating effectiveness, will discourage incomplete and misleading reporting to regulatory authorities and will reduce opportunities for sabotage by insiders. The increased detail of reporting to the public is required to fulfill many obligations including: (1.) The timely disclosure of significant information to professional risk analysts who advise large financial management organizations such as J. P. Morgan, Chase, H&Q. (2.) The timely disclosure of significant information to those individual investors who track available corporate performance data in making their investment decisions. (3.) The timely disclosure of significant information to state agencies that oversee the sale and/or acquisition of nuclear power plants by utility holding companies that operate within their jurisdiction.

View/Submit Public Comments

Original File	Size	File Type	Date	HTML	Description
1211-0003.pdf	116k	PDF	01/29/03	NONE	Comment letter from D. R. Woodlan. STARS
1211-0002.pdf	19k	PDF	01/14/03	NONE	Industry comments submitted by NEI from Alex Marion
1211-0001.msw	186 4k	MS Word	12/24/02	NONE	NRC Ignores Fouling in RIA and LOCA Analyses from Robert H. Leyse

NRC Ignores **Fouling** in RIA and LOCA Analyses

Even though **Fouling of Nuclear Reactor Fuel Elements (Fouling)** has a greater impact on Reactivity Insertion Accidents (RIAs) and Loss of Coolant Accidents (LOCAs) than high-burnup, the NRC and its collaborators continue to totally ignore the impact of **Fouling**. The following public documents affirm this assertion:

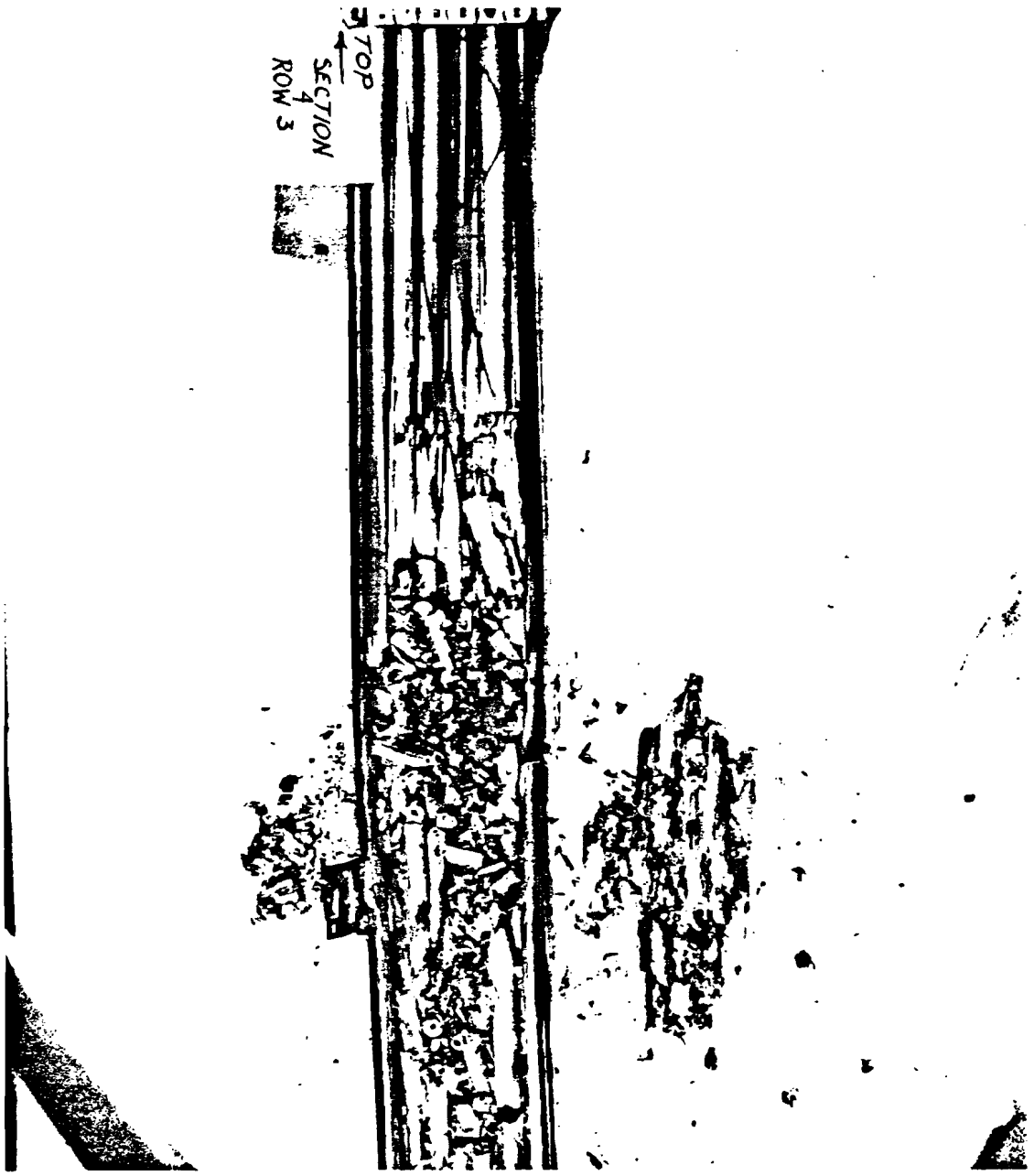
There was no discussion of **Fouling** during the October 9, 2002, the ACRS Subcommittee on Reactor Fuels. The Electric Power Research Institute (EPRI) discussed its unwarranted topical report on the response of high-burnup fuel to reactivity insertion accidents. EPRI disclosed nothing about the impact of **Fouling** and the Subcommittee asked no related questions.

Again, there was no discussion of **Fouling** in follow-on presentations to the Full ACRS on October 10, 2002. The Chairman of the Reactor Fuels Subcommittee described the EPRI presentation as, "... a very delightful presentation from the Electric Power Research Institute in describing its work on Reactivity Insertion Accidents to date." The Full ACRS asked no questions related to **Fouling**.

On October 17, 2002, the Full ACRS sent a letter to the Chairman, NRC, with the subject: CONFIRMATORY RESEARCH PROGRAM ON HIGH-BURNUP FUEL. There is no discussion of **Fouling** in this letter. In fact the ACRS lauds the NRC approaches, "RES has a well-organized and leveraged program of confirmatory research on the behavior of high-burnup fuel under the conditions of reactivity insertion events in pressurized water reactors, design-basis loss-of-coolant accidents (LOCAs), and anticipated transients without scram in boiling water reactors." Surprisingly, the ACRS expressed a somewhat veiled concern as follows: "We remain concerned that the time-temperature conditions used in the study of high-burnup fuel during design-basis LOCAs may not reveal phenomena unique to high-burnup fuel." Well, the ACRS is

invited to view the enclosed photograph that reveals the results (without burn-up or **Fouling**) when otherwise realistic time-temperature-flow conditions are deployed (FLECHT Run 9573).

On November 12, 2002, the ACRS Subcommittee on Thermal-Hydraulic Phenomena discussed the Rod Bundle Heat Transfer (RBHT) program that is being conducted at Pennsylvania State University. The RBHT program includes no provisions for the impact of **Fouling** of the heat transfer surfaces. Thus, this program is useless from the view of realism in evaluating the LOCA characteristics of licensed nuclear power reactors. Surprisingly, the ACRS did not point out this gross inadequacy of the RBHT program. At times, the discussions referred to the NRC's TRAC series of reactor safety analysis codes. These codes include no provisions for **Fouling** and thus are also useless for the realistic evaluation of LOCAs.


TOP
SECTION 4
ROW 3

(NRC_Ignores_Fouling_in_Reliability_and_Probabilistic_Risk_Assessment.doc) have been uploaded to **View/Submit Public Comments** as **1211-0004** under **Robert H. Leyse**
Submitted to the NRC's Rulemking Website at 9:28 PM, February 9, 2003

NRC Ignores **Fouling** in Reliability and Probabilistic Risk Assessment

Presentations by NRC staff, nuclear power equipment suppliers, and nuclear power lobbyists at recent ACRS subcommittee meetings have not included any consideration of the impacts of fouling of heat exchange equipment. These recent subcommittee meetings include:

Date	Subcommittee
01/24/2003	Reliability and Probability Risk Assessment Subcommittee
01/23/2003	Reliability and Probability Risk Assessment Subcommittee
01/22/2003	Reliability and Probability Risk Assessment Subcommittee
01/21/2003	Plant Operations and Reliability & Probabilistic Risk Assessment Subc.

Especially serious is the absence of any consideration of fouling of the nuclear fuel elements.

The transcript of the January 24, 2003 meeting, page 124, lines 13 and 14, includes the following remark by ACRS Member Shack, "The nagging fear is that you have missed something."

Dr. Larkins, related petitions that the ACRS may choose to evaluate are posted on the NRC's Rulemaking Website as follows

Date Posted	**Title**	**PRM No.**
10/12/2001	Robert H. Leyse	PRM-50-73
1/29/2002	Robert H. Leyse. Supplement to a Petition for Rulemaking	PRM-50-73A
8/9/2002	Robert H. Leyse	PRM-50-7

Finally, please provide the ACRS with the LER that is referenced in the following paragraph that I have copied from page 3 of this letter:

One measure of whether proper regulations are effected will be a determination that under the new regulations, conditions similar to those already reported in certain Licensee Event Reports, will henceforth constitute license violations. Very likely there is a preponderance of documented nuclear power plant operating experience that is worthy of such review; as a possible example, one candidate may be Licensee Event Report 50-458 99-016-00.

Thank you for your attention to these matters,

Robert H Leyse

Robert H. Leyse

ACRS BRIEFING

LICENSE RENEWAL UNDER THE NEW GALL REGIME

MARCH 7, 2003

WILLIAM "BUTCH" BURTON
NRR

10

BRIEFING AGENDA

CURRENT STATUS OF GALL PLANTS

NEW LRA FORMAT

NEW REVIEW APPROACH

NEW DOCUMENTATION OF STAFF REVIEW RESULTS

CURRENT STATUS OF GALL PLANTS

St. Lucie is the last pre-GALL plant (although aspects of GALL were incorporated into the St. Lucie LRA)

Starting with Ft. Calhoun, all plants applying for a renewed license will follow the GALL process

Six GALL plants are currently under staff review:

- Ft. Calhoun
- Robinson
- Ginna
- Summer
- Dresden/Quad Cities

NEW LRA FORMAT

SECTION 2: SCOPING AND SCREENING METHODOLOGY AND RESULTS

Section 2 tables contain links to Section 3 tables

SECTION 3: AGING MANAGMENT REVIEW RESULTS

Section 3 format has changed significantly!

SECTION 4: TIME-LIMITED AGING ANALYSES

Section 4 format has not changed

APPENDIX B: AGING MANAGEMENT PROGRAMS

App. B format has also changed significantly!

TABLE 2.3.3.8-1
INSTRUMENT AIR
Component Types Subject to Aging Management Review and Intended Functions

Component Type	Intended Functions	Aging Management Review Results
Accumulators	Pressure Boundary	3.3.1.07 3.3.1.13 3.3.2.23 3.3.2.25
Bolting	Pressure Boundary	3.3.1.05 3.3.1.13
Filter Housing	Pressure Boundary	3.3.2.01 3.3.2.04
Pipes & Fittings	Pressure Boundary	3.3.2.71 3.3.2.75
Tubing	Pressure Boundary	3.3.2.37 3.3.2.40 3.3.2.71 3.3.2.72 3.3.2.71
Valve Body	Pressure Boundary	3.3.2.01 3.3.2.04 3.3.2.05 3.3.2.10 3.3.2.13 3.3.2.14 3.3.2.71 3.3.2.72 3.3.2.71
Valve Operator Bodies	Pressure Boundary	3.3.2.01 3.3.2.04 3.3.1.05 3.3.1.13 3.3.2.23

NEW LRA FORMAT

SECTION 3

INDIVIDUAL SYSTEMS HAVE BEEN ROLLED UP INTO 6 BROAD SYSTEM/STRUCTURAL GROUPS:

3.1 - Reactor Systems
3.2 - ESF Systems
3.3 - Auxiliary Systems
3.4 - Steam and Power Conversion Systems
3.5 - Structures
3.6 - Electrical and I&C

As a result of the system roll-up, individual systems are lost in Section 3

NEW LRA FORMAT

SECTION 3

EACH SYSTEM/STRUCTURAL GROUP CONSISTS OF 3 TABLES:

3.X-1 Structures and components (SCs) evaluated in GALL

3.X-2 SCs not evaluated in GALL

3.X-3 SCs not evaluated in GALL, but GALL could be credited with managing component aging

Note: Robinson, Ginna, and Summer do not have 3.x-3 tables

Section 3 tables include links from Section 2 tables

NEW LRA FORMAT

SECTION 3

3.X-1 TABLES LIST SCs EVALUATED IN GALL

- Component group
- Aging effect/mechanism
- GALL AMPs
- Whether GALL recommends further evaluation
- Discussion

Discussion column identifies component materials and environments, gives information on GALL, and identifies AMPs that are credited for managing component aging

TABLE 3.1-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Row Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.01	Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801 except as noted in item 4 below. 2. The metal fatigue time limited aging analyses are discussed in Section 4.3. 3. Consistent with NUREG-1801, this group includes the low alloy steel and carbon steel with stainless steel cladding, stainless steel, CASS, and nickel alloy in borated treated water; and low alloy steel in deoxygenated water and steam at FCS. 4. Cumulative fatigue damage is not an aging effect requiring management for control element assembly shroud bolts and core support barrel snubber assembly socket head cap screws. These components are preloaded to prevent fatigue cycles.

NEW LRA FORMAT

SECTION 3

3.X-2 TABLES LIST SCs THAT WERE NOT EVALUATED IN GALL

Table format is similar to the traditional 6-column tables in pre-GALL applications

3.X-3 TABLES LIST SCs NOT EVALUATED IN GALL, BUT GALL CAN BE CREDITED FOR MANAGING COMPONENT AGING

Traditional 6 columns, plus 2 additional columns that provide justification for crediting GALL for aging management

Note: Only Ft. Calhoun has 3.X-3 tables

TABLE 3.1-2
FCS REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Row Number	Component Types	Material	Environment	AERMs	Program/Activity
3.1.2.01	External surfaces of stainless steel components in reactor coolant system pressure boundary	Stainless Steel	Ambient Air	None	Not Applicable
3.1.2.02	Pressurizer heater sleeves, steam generator - tubes, ICI nozzles, nozzle safe ends, CEDM and incore instrument housings, reactor head vent pipe, pressurizer bottom head plate cladding, steam generator primary head cladding and shock suppressors & supports, nozzle welds, thermal sleeves	Nickel Based Alloy including Alloy 600	Borated Treated Water	Loss of Material Crevice corrosion in the presence of sufficient levels of oxygen, halogens, sulfates, or copper	Chemistry Program (B.1.2)
3.1.2.03	External surfaces of nickel based alloy components in reactor coolant system pressure boundary	Nickel Based Alloy including Alloy 600	Ambient Air	None	Not Applicable
3.1.2.04	Steam generator lower head and manway cladding and primary side tube sheet	Nickel Based Alloy	Borated Treated Water	Cracking	Chemistry Program (B.1.2)

TABLE 3.3-3 (CONTINUED)
COMPONENTS IN AUXILIARY SYSTEMS NOT EVALUATED IN NUREG-1801 THAT RELY ON AGING MANAGEMENT PROGRAMS IN NUREG-1801 FOR FCS LICENSE RENEWAL

Row Number	FCS Components	FCS Material	FCS Environment	FCS AERMs	FCS Program Activity	Applicable NUREG-1801 Aging Management Review Results Row Number	Justification for applying NUREG-1801 Aging Management Review Results
3.3.3.09	Valve bodies, piping & fittings, duct, damper, bolts, heat exchangers	Cast iron, cadmium plated steel, galvanized steel, copper alloy	Ambient Air	Loss of Material	Boric Acid Corrosion (BAC) Prevention Program (B.2.1)	3.3.1.13	The material is subject to the same environment and aging effect, and managed by the same aging management program as evaluated in Table 3.3-1, Item 3.3.1.13. The aging effect is independent of component type.
3.3.3.10	Fire blocking damper, duct, valve bodies, fan housings	Galvanized steel, cast iron	Ambient Air	Loss of Material	Periodic Surveillance and Preventive Maintenance Program (B.2.7)	3.3.1.05	The FCS components are made of the same material, exposed to the same environment, subject to the same aging effects and managed by the same aging management program as the components evaluated in NUREG-1801, Volume 2, VII.F2.1-a.

NEW LRA FORMAT

APPENDIX B

AGING MANAGEMENT PROGRAMS (AMPs)

AMPs are either:

- Consistent with GALL
- Generally consistent with GALL, but with some deviation from GALL
- Not consistent with GALL

Some AMPs are common (i.e., they are credited with managing aging in components in more than 1 system/structural group)

Some AMPs are system group - specific

NEW REVIEW APPROACH

NO CHANGE IN REVIEW APPROACH IN LRA SECTIONS 2 AND 4

SIGNIFICANT CHANGE IN REVIEW APPROACH IN SECTION 3 AND APP. B (APP. B REVIEW RESULTS ARE DOCUMENTED IN SECTION 3 OF THE STAFF SER)

NEW REVIEW APPROACH

STAFF PERFORMS REVIEW IN THREE PARTS

Part 1 - Review of aging management programs

Part 2 - Review of aging management review results in tables

Part 3 - Review of adequacy of aging management

NEW REVIEW APPROACH

APPENDIX B

PART 1 - REVIEW OF AGING MANAGEMENT PROGRAMS

AMPs THAT APPLICANT CLAIMS ARE CONSISTENT WITH GALL

- **Staff does not review**
- **Claim of consistency is confirmed during the AMR inspection**
- **Reviewer determines whether the FSAR Supplement adequately describes the AMP**

B.1.6 INSERVICE INSPECTION PROGRAM 3.1.6

The Inservice Inspection Program is consistent with XI.M1, "*ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD*," and XI.S3, "*ASME Section XI, Subsections IWF*," as identified in NUREG-1801. The scope of the FCS Inservice Inspection Program includes those plant-specific components identified in Tables 3.1.2 and 3.1.3 of this application for which the Inservice Inspection Program is identified as an aging management program.

Operating Experience:

Review of the plant specific operating experience indicates that the FCS Inservice Inspection Program has been effective in managing the aging effects of components. No significant age related deterioration has been identified in the inspections performed.

Conclusion:

The FCS Inservice Inspection Program provides reasonable assurance that the aging effects will be managed such that the ASME Class 1, 2, and 3 components and their integral supports subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

NEW REVIEW APPROACH

APPENDIX B

PART 1 - REVIEW OF AGING MANAGEMENT PROGRAMS

AMPs THAT APPLICANT CLAIMS ARE CONSISTENT WITH GALL, BUT WITH DEVIATIONS

Reviewer determines:

- **Whether the deviation is acceptable**
- **Whether the AMP, with the deviation, is adequate to manage the aging for which it is credited**
- **Whether the FSAR Supplement adequately describes the modified AMP**

B.2.2 COOLING WATER CORROSION PROGRAM 3.1.9

The FCS Cooling Water Corrosion Program is consistent with XI.M20, "*Open-Cycle Cooling Water System*," and ~~XI.21~~ XI.M21, "*Closed-Cycle Cooling Water System*," as identified in NUREG-1801, with the exception of the enhancements specified in the following table and with the following clarifications:

- XI.M20 - Program Description, 3. Parameters Monitored/Inspected, 4.Detection of Aging Effects, 5. Monitoring and Trending, and 6. Acceptance Criteria

  External coatings are addressed by the FCS General Corrosion of External Surfaces Program.

- XI.M21 - Program Description, 2. Preventative Actions, 5. Monitoring and Trending, 6. Acceptance Criteria, and 7. Corrective Action

  The Chemistry-related portions of the program are addressed in the FCS Chemistry Program.

- The scope of the FCS Cooling Water Corrosion Program includes those plant specific components identified in Tables 3.2.2, 3.3.2, and 3.3.3 of this application for which the Cooling Water Corrosion Program is identified as an aging management program.

The FCS Cooling Water Corrosion Program will also include the following exceptions to NUREG-1801:

- XI.M21 - 3. Parameters Monitored/Inspected, 4. Detection of Aging Effects, and 5. Monitoring and Trending

  The license renewal commitment for these programs relates only to the maintenance of the pressure boundary and not the maintenance of fluid flow. Fluid flow is considered an active function. Performance testing and other active system function testing is not performed on an 18 month or 5 year frequency in accordance with EPRI TR-107396, Closed Cooling Water Chemistry Guideline, because this EPRI document does not address this criteria or specify that testing frequency. Non-destructive testing and heat transfer performance to identify pressure boundary integrity are performed per EPRI TR-107396.

The following enhancements will be made to the Cooling Water Corrosion Program prior to the period of extended operation.

NUREG-1801 Program	Criteria	Enhancement
XI.M20, Open-Cycle Cooling Water System	1. Scope of Program 4. Detection of Aging Effects 5. Monitoring and Trending	Inspections to various raw water components will be added based on FCS' Cooling Water Corrosion Program susceptibility evaluation. These inspection activities will be commensurate with the GALL Program.
XI.M21, Closed-Cycle Cooling Water System	3. Parameters Monitored/ Inspected 4. Detection of Aging Effects	Inspections to various cooling water components will be added based on FCS' Cooling Water Corrosion Program susceptibility evaluation. These inspection activities will be commensurate with the GALL Program.

Operating Experience:

Review of FCS operating experience has identified some component part replacements (and repairs) due to corrosion and cracking in the Component Cooling Water and Raw Water Systems. Appropriate long term corrective actions were implemented based on these experiences. These included material changes, additional preventive maintenance, and increased sample evaluation.

Conclusion:

The FCS Cooling Water Corrosion Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

NEW REVIEW APPROACH

APPENDIX B

PART 1 - REVIEW OF AGING MANAGEMENT PROGRAMS

AMPs THAT ARE NOT CONSISTENT WITH GALL

Reviewer performs a review of the 10 program elements in accordance with Branch Technical Position RLSB-1, in the SRP-LR

Reviewer determines whether the FSAR SUPPLEMENT adequately describes the AMP

B.2.7 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE (PM) PROGRAM

3.1.14

The stated purpose of the PM program is to prevent or minimize equipment breakdown and to maintain equipment in a condition that will enable it to perform its normal and emergency functions. The program and the site administrative control processes provide for a systematic approach in establishing the method, frequency, acceptance criteria, and documentation of results.

The FCS Periodic Surveillance and Preventive Maintenance Program consists of periodic inspections and tests that are relied on to manage aging for system and structural components and that are not evaluated as part of the other aging management programs addressed in this appendix. The preventive maintenance and surveillance testing activities are implemented through periodic work orders that provide for assurance of functionality of the components by confirmation of integrity of applicable parameters.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Program:

The FCS Periodic Surveillance and Preventive Maintenance Program provides for periodic inspection and testing of components in the following systems and structures.

- Auxiliary Building
- Auxiliary Building HVAC
- Auxiliary Feedwater
- Chemical and Volume Control
- Component Cooling
- Containment
- Containment HVAC
- Control Room HVAC and Toxic Gas Monitoring
- Diesel Generator Lube Oil
- Duct Banks
- Emergency Diesel Generators
- Fire Protection
- Fuel Handling Equipment/Heavy Load Cranes
- Intake Structure
- Liquid Waste Disposal
- Containment Penetration, and System Interface Components for Non-CQE Systems
- Reactor Coolant
- Safety Injection and Containment Spray
- Ventilating Air

(2) Preventive Actions:

The Periodic Surveillance and Preventive Maintenance Program includes periodic refurbishment or replacement of components, which could be considered to be preventive or mitigative actions. The inspections and testing to identify component aging degradation effects do not constitute preventive actions in the context of this element.

(3) Parameters Monitored or Inspected:

Inspection and testing activities monitor parameters including surface condition, loss of material, presence of corrosion products, signs of cracking and presence of water in oil samples.

(4) Detection of Aging Effects:

Preventive maintenance and surveillance testing activities provide for periodic component inspections and testing to detect the following aging effects and mechanisms:

- Change in Material Properties
- Cracking
- Fouling
- Loss of Material
- Loss of Material - Crevice Corrosion
- Loss of Material - Fretting
- Loss of Material - General Corrosion
- Loss of Material - Pitting Corrosion
- Loss of Material - Pitting/ Crevice/Gen. Corrosion
- Loss of Material - Wear
- Separation

The extent and schedule of the inspections and testing assures detection of component degradation prior to the loss of their intended functions. Established techniques such as visual inspections and dye penetrant testing are used.

(5) Monitoring and Trending:

Preventive maintenance and surveillance testing activities provide for monitoring and trending of aging degradation. Inspection intervals are established such that they provide for timely detection of component degradation. Inspection intervals are dependent on the component material and environment and take into consideration industry and plant-specific operating experience and manufacturers' recommendations.

The program includes provisions for monitoring and trending with the stated intent of identifying potential failures or degradation and making adjustments to ensure components

remain capable of performing their functions. PM review and update guidelines are provided that include adjustment of PM task and frequency based on the as-found results of previous performance of the PM. In particular, responsible system engineers are required to periodically review the results of preventive maintenance and recommend changes based on these reviews. The program includes guidance to assist the system engineers in achieving efficient and effective trending.

(6) Acceptance Criteria:

Periodic Surveillance and Preventive Maintenance Program acceptance criteria are defined in the specific inspection and testing procedures. They confirm component integrity by verifying the absence of the aging effect or by comparing applicable parameters to limits based on the applicable intended function(s) as established by the plant design basis.

(7) Corrective Actions:

Identified deviations are evaluated within the FCS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation. The FCS corrective action process is in accordance with 10 CFR 50 Appendix B.

(8) Confirmation Process:

The FCS corrective action process is in accordance with 10 CFR 50 Appendix B and includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls:

All credited aging management activities are subject to the FCS administrative controls process, which is in accordance with 10 CFR 50 Appendix B and requires formal reviews and approvals.

(10) Operating Experience:

Periodic surveillance and preventive maintenance activities have been in place at FCS since the plant began operation. These activities have a demonstrated history of detecting damaged and degraded components and causing their repair or replacement in accordance with the site corrective action process. With few exceptions, age-related degradation adverse to component intended functions was discovered and corrective actions were taken prior to loss of intended function.

Conclusion:

The Periodic Surveillance and Preventive Maintenance Program assures that various aging effects are managed for a wide range of components at FCS. Based on the program structure and administrative processes and FCS operating experience, there is reasonable assurance that the credited inspection and testing activities of the Periodic Surveillance and Preventive Maintenance Program will continue to adequately manage the identified aging effects of the applicable components so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

NEW REVIEW APPROACH

SECTION 3

PART 2 - REVIEW OF AGING MANAGEMENT REVIEW RESULTS IN TABLES

REVIEWER EVALUATES INFORMATION PROVIDED IN THE SECTION 3 TABLES

3.X-1 TABLES

Two types of information

1. **Assessment of SCs that are consistent with the GALL evaluation, and for which GALL does not recommend further evaluation**

NEW REVIEW APPROACH

SECTION 3

PART 2 - REVIEW OF AGING MANAGEMENT REVIEW RESULTS IN TABLES

2. **Assessment of SCs that are consistent with GALL, but GALL recommends further evaluation**

NEW REVIEW APPROACH

SECTION 3

PART 2 - REVIEW OF AGING MANAGEMENT REVIEW RESULTS IN TABLES

ASSESSMENT OF STRUCTURES AND COMPONENTS (SCs) THAT ARE CONSISTENT WITH THE GALL EVALUATION, AND FOR WHICH GALL DOES NOT RECOMMEND FURTHER EVALUATION

- Reviewer does not review
- Consistency is confirmed during the AMR inspection
- Reviewers can request the inspectors to look at specific items during the inspection

NEW REVIEW APPROACH

SECTION 3

PART 2 - REVIEW OF AGING MANAGEMENT REVIEW RESULTS IN TABLES

ASSESSMENT OF SCs THAT ARE CONSISTENT WITH GALL, BUT GALL RECOMMENDS FURTHER EVALUATION

- Reviewer does not review claim of consistency
- Consistency is confirmed during the AMR inspection
- Reviewer focuses review on issues that for which GALL recommends further evaluation

TABLE 3.3-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Row Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1.01	Components in spent fuel pool cooling and cleanup	Loss of material due to general, pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	The material identified in NUREG-1801 is not applicable to FCS. These components are addressed in Section 3.3.2 of this application.
3.3.1.02	Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The General Corrosion of External Surfaces Program (B.3.3) manages this aging effect. This program is described in Appendix B of this application. 3. Consistent with NUREG-1801, this group only includes elastomer seals in the ventilation systems exposed to ambient air at FCS.
3.3.1.03	Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems (older BWR)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801 for the chemical and volume control and primary sampling systems. 2. The metal fatigue time limited aging analyses are discussed in Section 4.3.1 of this application.

NEW REVIEW APPROACH

SECTION 3

PART 2 - REVIEW OF AGING MANAGEMENT REVIEW RESULTS IN TABLES

3.X-2 TABLES

Contain assessments of SCs not evaluated in GALL

Reviewer performs traditional pre-GALL review

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Row Number	Component Types	Material	Environment	AERMs	Program/Activity
3.3.2.74	Flow element/orifice body; indicator/recorder housing, orifice plate, pipes & fittings, valve bodies, heat exchanger - tubes	Stainless Steel	Corrosion-inhibited Treated Water	Cracking Due to exposure to halogens and sulfates	Chemistry Program (B.1.2) and Cooling Water Corrosion Program (B.2.2)
3.3.2.75	External surfaces of stainless steel componenets	Stainless Steel	Ambient Air	None	Not Applicable
3.3.2.76	Heat exchanger - tubes	Stainless Steel	Oxygenated Treated Water <200 deg F	Loss of Material • Crevice corrosion due to an oxygenated treated water environment • Pitting corrosion due to exposure to halogens and sulfates	Chemistry Program (B.1.2) and Cooling Water Corrosion Program (B.2.2)
3.3.2.77	Filter strainer housing	Stainless Steel	Raw Water	Loss of Material • Crevice corrosion due to the presence of dis-solved oxygen and impurities • MIC due to exposure to microbiological activity • Pitting corrosion due to exposure to halide ions	Periodic Surveillance and Preventive Maintenance Program (B.2.7)
3.3.2.78	Not used in application				
3.3.2.79	Glass in metal fire penetration barriers	Glass	Ambient Air	None	Not Applicable

NEW REVIEW APPROACH

SECTION 3

PART 2 - REVIEW OF AGING MANAGEMENT REVIEW RESULTS IN TABLES

3.X-3 TABLES (FT. CALHOUN ONLY)

SCs not evaluated in GALL, but GALL could be credited with managing aging of the SCs

Reviewer determines if GALL can be credited with managing aging of the SCs

Note: For GALL plants after Ft. Calhoun, 3.X-3 information is integrated into the 3.X-1 or 3.X-2 tables

TABLE 3.3-3 (CONTINUED)
COMPONENTS IN AUXILIARY SYSTEMS NOT EVALUATED IN NUREG-1801 THAT RELY ON AGING MANAGEMENT PROGRAMS IN NUREG-1801 FOR FCS LICENSE RENEWAL

Row Number	FCS Components	FCS Material	FCS Environment	FCS AERMs	FCS Program Activity	Applicable NUREG-1801 Aging Management Review Results Row Number	Justification for applying NUREG-1801 Aging Management Review Results
3.3.3.07	Valve bodies, piping & fittings, fan housings, bolts, duct, pumps	Cast Iron, cadmium plated steel, galvanized steel	Ambient Air	Loss of Material	General Corrosion of Ext. Surfaces Program (B.3.3)	3.3.1.05	The material is subject to the same environment and aging effect, and managed by the same aging management program as evaluated in Table 3.3-1, Item 3.3.1.05. The aging effect is independent of component type.
3.3.3.08	Electric heater sleeves, tanks, heat exchanger, orifice,	Carbon Steel	Corrosion-Inhibited Treated Water	Loss of Material	Chemistry Program (B.1.2) and Cooling Water Corrosion Program (B.2.2)	3.3.1.14	The FCS components are made of the same material, exposed to the same environment, subject to the same aging effects and managed by the same aging management program as the components evaluated in NUREG-1801, Volume 2, VII.H2.1-a and VII.C2.5-a.

NEW REVIEW APPROACH

SECTION 3

PART 3 - REVIEW OF ADEQUACY OF AGING MANAGEMENT

THIS PART OF THE REVIEW IS REQUIRED TO ENSURE COMPLIANCE WITH THE RULE!!!

Reviewer must determine whether SCs will be adequately managed

This requires tracking the aging management information for each SC from Section 2, through Section 3, to the AMP that is being credited for aging management

The component links are used to perform this portion of the review

NEW REVIEW APPROACH

SECTION 3

PART 3 - REVIEW OF ADEQUACY OF AGING MANAGEMENT

AMR reviewer must communicate with the AMP reviewer to ensure that the components will be adequately managed

TABLE 2.3.3.12-1
CONTROL ROOM HVAC AND TOXIC GAS MONITORING
Component Types Subject to Aging Management Review and Intended Functions

Component Type	Intended Functions	Aging Management Review Results
Blower & Fan Housing	Pressure Boundary	3.3.1.05 3.3.3.07 3.3.3.10
Bolting	Pressure Boundary	3.3.1.05 3.3.3.07 3.3.2.75
Duct	Pressure Boundary	3.3.1.02 3.3.1.05 3.3.3.07 3.3.3.10
Filter/Strainer	Pressure Boundary Filtration	3.3.1.05 3.3.2.75
Heat Exchanger	Pressure Boundary/Heat Transfer	3.3.1.05 3.3.2.29 3.3.2.39 3.3.2.40
Pipes & Fittings	Pressure Boundary	3.3.1.05 3.3.1.14 3.3.2.38 3.3.2.40 3.3.2.75 3.3.2.91
Valve Bodies	Pressure Boundary	3.3.1.05 3.3.2.01 3.3.2.10 3.3.2.15 3.3.2.28 3.3.2.29 3.3.2.38 3.3.2.75 3.3.3.07

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Row Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1.05	Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Periodic Surveillance and Preventive Maintenance (B.2.7), General Corrosion of External Surfaces (B.3.3), and Fire Protection Programs (B.2.5) manage this aging effect. These programs are described in Appendix B of this application. The FCS Fire Protection Program provides guidance for detecting loss of material due to general, pitting, crevice, and microbiologically influenced corrosion (MIC). 3. Consistent with NUREG-1801, this group includes carbon steel, galvanized steel, and copper in air, and carbon steel in diesel engine exhaust gases at FCS.

Table 3.3-1. Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of the GALL Report

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/PWR	Components in spent fuel pool cooling and cleanup	Loss of material due to general, pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated (see subsections 3.3.2.2.1.1 and 3.3.2.2.1.2)
BWR/PWR	Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific (see subsection 3.3.2.2.2)
BWR/PWR	Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems (older BWR)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see subsection 3.3.2.2.3)
BWR/ PWR	Heat exchangers in reactor water cleanup system (BWR); high pressure pumps in chemical and volume control system (PWR)	Crack initiation and growth due to SCC or cracking	Plant specific	Yes, plant specific (see subsection 3.3.2.2.4)
BWR/PWR	Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific (see subsection 3.3.2.2.5)
BWR/PWR	Components in reactor coolant pump oil collect system of fire protection	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection	Yes, detection of aging effects is to be further evaluated (see subsection 3.3.2.2.6)
BWR/PWR	Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated (see subsection 3.3.2.2.7)

Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1, of this standard review plan.)

3.3.2.2.3 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3 of this standard review plan.

3.3.2.2.4 Crack Initiation and Growth due to Cracking or Stress Corrosion Cracking

Crack initiation and growth due to SCC could occur in the regenerative and non-regenerative heat exchanger components in the reactor water cleanup system (BWR) and due to cracking in the high-pressure pump in the chemical and volume control system (PWR). The GALL report recommends further evaluation to ensure that these aging effects are managed adequately. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1, of this standard review plan.)

3.3.2.2.5 Loss of Material due to General, Microbiologically Influenced, Pitting, and Crevice Corrosion

Loss of material due to general, pitting, and crevice corrosion could occur in the piping and filter housing and supports in the control room area, the auxiliary and radwaste area, the primary containment heating and ventilation systems, in the piping of the diesel generator building ventilation system, in the aboveground piping and fittings, valves, and pumps in the diesel fuel oil system and in the diesel engine starting air, combustion air intake, and combustion air exhaust subsystems in the emergency diesel generator system. Loss of material due to general, pitting, crevice, and microbiologically influenced corrosion (MIC) could occur in the duct fittings, access doors, and closure bolts, equipment frames and housing of the duct, due to pitting and crevice corrosion could occur in the heating/cooling coils of the air handler heating/cooling, and due to general corrosion could occur on the external surfaces of all carbon steel structures and components, including bolting exposed to operating temperatures less than 212°F in the ventilation systems. The GALL report recommends further evaluation to ensure that these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1, of this standard review plan.)

3.3.2.2.6 Loss of Material due to General, Galvanic, Pitting, and Crevice Corrosion

Loss of material due to general, galvanic, pitting, and crevice corrosion could occur in tanks, piping, valve bodies, and tubing in the reactor coolant pump oil collection system in fire protection. The fire protection program relies on a combination of visual and volumetric examinations in accordance with the guidelines of 10 CFR Part 50 Appendix R and Branch Technical Position 9.5-1 to manage loss of material from corrosion. However, corrosion may occur at locations where water from wash downs may accumulate. Therefore, verification of the effectiveness of the program should be performed to ensure that corrosion is not occurring. The GALL report recommends further evaluation of programs to manage loss of material due to general, galvanic, pitting, and crevice corrosion to verify the effectiveness of the program. A one-time inspection of the bottom half of the interior surface of the tank of the reactor coolant pump oil collection system is an acceptable method to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation.

B.2.7 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE (PM) PROGRAM

3.1.18

The stated purpose of the PM program is to prevent or minimize equipment breakdown and to maintain equipment in a condition that will enable it to perform its normal and emergency functions. The program and the site administrative control processes provide for a systematic approach in establishing the method, frequency, acceptance criteria, and documentation of results.

The FCS Periodic Surveillance and Preventive Maintenance Program consists of periodic inspections and tests that are relied on to manage aging for system and structural components and that are not evaluated as part of the other aging management programs addressed in this appendix. The preventive maintenance and surveillance testing activities are implemented through periodic work orders that provide for assurance of functionality of the components by confirmation of integrity of applicable parameters.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Program:

The FCS Periodic Surveillance and Preventive Maintenance Program provides for periodic inspection and testing of components in the following systems and structures.

- Auxiliary Building
- Auxiliary Building HVAC
- Auxiliary Feedwater
- Chemical and Volume Control
- Component Cooling
- Containment
- Containment HVAC
- Control Room HVAC and Toxic Gas Monitoring
- Diesel Generator Lube Oil
- Duct Banks
- Emergency Diesel Generators
- Fire Protection
- Fuel Handling Equipment/Heavy Load Cranes
- Intake Structure
- Liquid Waste Disposal
- Containment Penetration, and System Interface Components for Non-CQE Systems
- Reactor Coolant
- Safety Injection and Containment Spray
- Ventilating Air

B.3.3 GENERAL CORROSION OF EXTERNAL SURFACES PROGRAM

3.1.20

The General Corrosion of External Surfaces Program at FCS is credited for aging management of the effects of loss of material and cracking for applicable components, including piping, valves, supports, tanks, and bolting, which are made of cadmium plated steel, carbon steel, cast iron, copper alloy, galvanized steel, low alloy steel, and neoprene.

(1) Scope of Program

The General Corrosion of External Surfaces Program consists of several FCS activities that manage the aging effects of loss of material and cracking for components in the following systems:

- Auxiliary Boiler Fuel Oil
- Auxiliary Building HVAC
- Auxiliary Feedwater (AFW)
- Chemical and Volume Control
- Component Cooling Water (CCW)
- Containment HVAC
- Control Room HVAC
- Diesel Generator Lube Oil
- Diesel Jacket Water
- Starting Air
- Feedwater
- Fire Protection Fuel Oil
- Gaseous Waste Disposal
- Instrument Air
- Main Steam (MS) and Turbine Steam Extraction
- Containment Penetration, and System Interface Components for Non-CQE Systems
- Nitrogen Gas
- Primary Sampling
- Raw Water
- Ventilating Air

(2) Preventive Actions

This program does not prevent aging.

(3) Parameters Monitored or Inspected

Surface conditions of components are monitored through visual observation and inspection to detect signs of external corrosion and to detect conditions that can result in external corrosion, such as fluid leakage.

F1. CONTROL ROOM AREA VENTILATION SYSTEM

F1.1 Duct

F1.1.1 Duct Fittings, Access Doors, and Closure Bolts
F1.1.2 Equipment Frames and Housing
F1.1.3 Flexible Collars between Ducts and Fans
F1.1.4 Seals in Dampers and Doors

F1.2 Air Handler Heating/Cooling

F1.2.1 Heating/Cooling Coils

F1.3 Piping

F1.3.1 Piping and Fittings

F1.4 Filters

F1.4.1 Housing and Supports
F1.4.2 Elastomer Seals

F1. CONTROL ROOM AREA VENTILATION SYSTEM

Systems, Structures, and Components

This section comprises the control room area ventilation system (with warm moist air as the normal environment), which contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, filters, and heating and cooling air handlers. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the control room area ventilation system are governed by Group B Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters and seals are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the control room area ventilation system is the auxiliary and radwaste area ventilation system (VII.F2). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

Table 3. Summary of Aging Management Programs for the Auxiliary Systems Evaluated in Chapter VII of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Item Number in GALL
BWR/ PWR	Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific	VII.F1.1-a, VII.F1.2-a, VII.F1.4-a, VII.F2.1-a, VII.F2.2-a, VII.F2.4-a, VII.F3.1-a, VII.F3.2-a, VII.F3.4-a, VII.F4.1-a, VII.F4.2-a, VII.H1.1-a, VII.H1.2-a, VII.H1.3-a, VII.H2.2-a, VII.H2.3-a, VII.H2.4-a, VII. I1-b.
BWR/ PWR	Components in reactor coolant pump oil collect system of fire protection	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection	Yes, detection of aging effects is to be further evaluated	VII.G.7-a, VII.G.7-b.
BWR/ PWR	Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	VII.H1.4-a, VII.H2.5-a.
BWR	Piping, pump casing, and valve body and bonnets in shutdown cooling system (older BWR)	Loss of material due to pitting and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	VII.E4.1-a, VII.E4.2-a.
PWR	Heat exchangers in chemical and volume control system	Crack initiation and growth due to SCC and cyclic loading	Water chemistry and a plant-specific verification program	Yes, plant specific	VII.E1.7-c, VII.E1.8-b.
BWR/ PWR	Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity and loss of material due to general corrosion (Boral, boron steel)	Plant specific	Yes, plant specific	VII.A2.1-b.

VII Auxiliary Systems
F1. Control Room Area Ventilation System

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
F1.1-a F1.1.1 F1.1.2	Duct Duct fittings, access doors, and closure bolts Equipment frames and housing	Carbon steel (galvanized or painted) bolts: plated carbon steel	Warm, moist air	Loss of material/ General, pitting, crevice corrosion, and microbiologically influenced corrosion (for duct [drip-pan] and piping for moisture drainage)	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.1-b F1.1.3 F1.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.1-c F1.1.3 F1.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Loss of material/ Wear	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.2-a F1.2.1	Air handler heating/ cooling Heating/ cooling coils	Copper/ nickel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.3-a F1.3.1	Piping Piping and fittings	Carbon steel	Hot or cold treated water	Loss of material/ General, pitting, crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
F1.4-a F1.4.1	Filters Housing and supports	Carbon steel, stainless steel	Warm, moist air	Loss of material/ General (only for carbon steel), pitting, and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.4-b F1.4.2	Filters Elastomer seals	Elastomers (Neoprene and similar materials)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific

TABLE 3.3-3 (CONTINUED)
COMPONENTS IN AUXILIARY SYSTEMS NOT EVALUATED IN NUREG-1801 THAT RELY ON AGING MANAGEMENT PROGRAMS IN NUREG-1801 FOR FCS LICENSE RENEWAL

Row Number	FCS Components	FCS Material	FCS Environment	FCS AERMs	FCS Program Activity	Applicable NUREG-1801 Aging Management Review Results Row Number	Justification for applying NUREG-1801 Aging Management Review Results
3.3.3.07	Valve bodies, piping & fittings, fan housings, bolts, duct, pumps	Cast Iron, cadmium plated steel, galvanized steel	Ambient Air	Loss of Material	General Corrosion of Ext. Surfaces Program (B.3.3)	3.3.1.05	The material is subject to the same environment and aging effect, and managed by the same aging management program as evaluated in Table 3.3-1, Item 3.3.1.05. The aging effect is independent of component type.
3.3.3.08	Electric heater sleeves, tanks, heat exchanger, orifice,	Carbon Steel	Corrosion-Inhibited Treated Water	Loss of Material	Chemistry Program (B.1.2) and Cooling Water Corrosion Program (B.2.2)	3.3.1.14	The FCS components are made of the same material, exposed to the same environment, subject to the same aging effects and managed by the same aging management program as the components evaluated in NUREG-1801, Volume 2, VII.H2.1-a and VII.C2.5-a.

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Row Number	Component Types	Material	Environment	AERMs	Program/Activity
3.3.2.27	Heat exchanger - shell	Carbon Steel	Oxygenated Treated Water <200 deg F	Loss of Material • General and crevice corrosion due to dissolved oxygen • Pitting corrosion due to halogens • Galvanic corrosion due to the conductivity of the process fluid and the presence of dissimilar metals in contact	Chemistry Program (B.1.2) and Cooling Water Corrosion Program (B.2.2)
3.3.2.28	Valve bodies	Cast Iron	Gas - Refrigerant (Liquid)	None	Not Applicable
3.3.2.29	Pump casings, valve bodies, pipes & fittings, heat exchanger - channel/channel head	Cast Iron	Corrosion-Inhibited Treated Water	Loss of Material • General and crevice corrosion due to the exposure of cast iron to dissolved oxygen • Pitting corrosion due to exposure to halogens	Chemistry Program (B.1.2) and Cooling Water Corrosion Program (B.2.2)
3.3.2.30	Pump casings, valve bodies, pipes & fittings	Cast Iron	Corrosion-Inhibited Treated Water	Loss of Material Selective leaching due to the exposure of cast iron to dissolved oxygen	Selective Leaching Program (B.3.6)

NEW DOCUMENTATION OF STAFF REVIEW RESULTS

STAFF HAS DEVELOPED AN SER TEMPLATE TO AID IN THE PERFORMANCE AND DOCUMENTATION OF THE STAFF'S REVIEW

- **The template contains boilerplate language (pre-approved by OGC)**
- **Includes 3.X-1 tables to conveniently summarize the GALL review results**
- **Template will be revised based on lessons learned from Ft. Calhoun review**

omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

[I HAVE NO SPECIFIC FORMAT HERE. THE TECHNICAL LEAD SHOULD MAKE SURE THAT THERE IS A CONSISTENT FORMAT FOR EACH SYSTEM WRITEUP. SEE 2.3.4 ABOVE FOR GUIDANCE]

2.5.20.3 Conclusions

The staff reviewed the LRA and to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that [***pending satisfactory resolution of Open Items XXXXX***], there is reasonable assurance that the applicant has adequately identified the bus bar components that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the bus bar components that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.5.21 Evaluation Findings

On the basis of this review, the staff concludes that [***pending satisfactory resolution of Open Items XXXXX***], there is reasonable assurance that the applicant has adequately identified the electrical systems and components that are within the scope of license renewal, in accordance with the requirements of 10 CFR 54.4(a), and that the applicant has adequately identified the electrical systems components that are subject to an aging management review, in accordance with the requirements of 10 CFR 54.21(a)(1).

2.5.22 References

3.0 AGING MANAGEMENT REVIEW

OPPD is the first license renewal applicant to fully utilize the Generic Aging Lessons Learned (GALL) process. The purpose of GALL is to provide the staff with a summary of staff-approved AMPs for the aging of most structures and components that are subject to an AMR. If an applicant commits to implementing these staff-approved AMPs, the time, effort, and resources used to review an applicant's LRA will be greatly reduced, thereby improving the efficiency and effectiveness of the license renewal review process. The GALL report is a compilation of existing programs and activities used by commercial nuclear power plants to manage the aging of structures and components within the scope of license renewal and which are subject to an AMR. The GALL report summarizes the aging management evaluations, programs, and activities credited for managing aging for most of the structures and components used throughout the industry, and serves as a reference for both applicants and staff reviewers to quickly identify those aging management programs and activities that the staff has determined will provide adequate aging management during the period of extended operation.

The GALL report identifies (1) systems, structures, and components, (2) component materials, (3) the environments to which the components are exposed, (4) the aging effects associated with the materials and environments, (5) the AMPs that are credited with managing the aging effects, and (6) recommendations for further applicant evaluations of aging effects and their management for certain specific components types.

In order to determine whether the GALL process would improve the efficiency of the license renewal review, the staff conducted a demonstration project to exercise the GALL process and determine the format and content of a safety evaluation based on the GALL review process. The SRP-LR was prepared based on both the GALL model and the lessons learned from the demonstration project.

3.0.1 The GALL Format for the LRA

The FCS LRA closely follows the standard LRA format, as agreed between NEI and the staff (see letters dated August 9, 1999 and September 22, 1999). This format has been used by previous applicants and will continue to be used by future applicants. However, there are several important changes within the format that reflect the GALL process. First, the tables in LRA Section 2 that identify the structures and components that are subject to an AMR now include a third column which links plant-specific structures and components in the Section 2 tables to generic GALL component groups in Section 3 (this is discussed in more detail below). Second, the tables in LRA Section 3 are different from the Section 3 tables used in previous LRAs. There are no system-specific tables in Section 3 of the FCS LRA. The individual components within a system have been included in a series of system group tables. For example, there are 20 auxiliary systems at FCS. Each system has several components. In previous LRAs, each system had a separate table that listed the components in the system. With the FCS LRA, there are no system tables. Instead all the components in the 20 auxiliary systems are included in any one of three auxiliary system tables. LRA Table 3.3-1 consists of auxiliary system components evaluated in the GALL report, LRA Table 3.3-2 consists of FCS auxiliary systems components not evaluated in the GALL report, and LRA Table 3.3-3 consists of FCS auxiliary systems components that were not evaluated in the GALL report, but the applicant has determined can be managed using a GALL AMR and associated AMP. Similarly, the LRA tables for the other system groups (3.1 - reactor systems, 3.2 - engineered safety feature systems, 3.4 - steam and power conversion systems, 3.5 - structures, and 3.6 - electrical systems) have 3.x-1 LRA tables for components evaluated in the GALL report, 3.x-2 LRA tables for components not evaluated in the GALL report, and 3.x-3 LRA tables for components that were not evaluated in the GALL report, but the applicant has determined can be managed using a GALL AMR and associated AMP.

The 3.x-1 LRA tables have 6 columns. Column 1 identifies the system group, table number, and row number in the table. For example, 3.1.1.1 identifies Table 3.1-1, row 1. This information is repeated in the last column of the Section 2 tables, and allows the staff reviewer to link each plant-specific structure and component identified in the Section 2 tables to the generic structure and component types identified in the Section 3 tables. Column 2 of the 3.x-1 LRA tables lists the generic structure and component types evaluated in GALL. Column 3 identifies the applicable aging effects experienced by the structure or component. Column 4 identifies the AMP that the GALL report credits for managing the aging effect identified in Column 3. Column 5 indicates whether the GALL report recommends further evaluation of the

management of the aging effect(s). Column 6 provides plant-specific information regarding management of the aging effect(s). Columns 2 through 5 of the 3.x-1 LRA tables are taken directly from the associated tables in the SRP and GALL report. Column 6 tells the staff reviewer whether or not the FCS AMP is consistent with GALL and provides information on the material and environment associated with the component group. This column also provides additional information if the aging management differs from what is assumed in GALL, and provides information on any additional evaluations that GALL recommends.

The 3.x-2 LRA tables contain structures and components that were not evaluated in GALL. Because these structures and components were not evaluated in GALL, the staff had to perform a full review, just like those done for past applications.

The 3.x-2 LRA tables also have 6 columns, but the columns are different from those in the 3.x-1 LRA tables. The 3.x-2 LRA tables look very much like the Section 3 tables in previous applications. The first column identifies the system group, table number, and row number in the table. For example, 3.3.2.1 identifies LRA Table 3.3-2, row 1. Column 2 of LRA table 3.x-2 identifies the type of structure or component being evaluated. Column 3 identifies the structure or component material, while Column 4 identifies the environment that the structure or component is exposed to. Column 5 identifies the applicable aging effect, and Column 6 identifies the AMP that's credited for managing the aging effect.

Because these components were not evaluated in GALL, the staff determined the adequacy of the aging management evaluation and programs in the same manner as for previous applications.

The 3.x-3 LRA tables contain structures and components that were not evaluated in GALL, but the applicant has determined that the materials, environments, and aging effects are bounded by the GALL evaluation and that the GALL AMPs can be applied to these structures and components.

The 3.x-3 LRA tables have 8 columns. Columns 1 and 2 are the same as in the other tables. Column 3 identifies the structure or component material. Column 4 identifies the environment to which the structure or component is exposed. Column 5 identifies the applicable aging effect. Column 6 identifies the FCS AMP. Column 7 identifies the applicable GALL AMR evaluation that the applicant credits for managing the aging, and Column 8 provides a justification for applying the GALL AMR evaluation to the structure or component.

For structures and components in the 3.x-3 LRA tables, the staff performed a traditional evaluation of the aging management results and determined whether the GALL evaluation is applicable to the structure or component.

3.0.2 The Staff's Review Process for GALL

The staff's review of FCS LRA was performed in three phases. In Phase 1, the staff reviewed the applicant's AMP descriptions to compare those AMPs for which the applicant claimed consistency with those reviewed and approved in the GALL report. For those AMPs for which the applicant claimed consistency with the GALL AMPs, and for which GALL recommended no further evaluation, the staff conducted an inspection to confirm that the applicant's AMPs were

consistent with the GALL AMPs. Furthermore, in the case of FCS, the staff performed an additional review to determine whether the applicant properly applied the GALL program to its facility. For those AMPs for which the applicant claimed consistency with the GALL AMPs, and for which GALL recommended further evaluation, the staff performed a confirmatory inspection and reviewed the applicant's evaluation to determine whether it addressed the additional issues recommended in the GALL report. In addition, in the case of FCS, the staff performed an additional review to determine whether the applicant properly applied the GALL program to its facility. The staff also reviewed the applicant's evaluation to determine whether the applicable aging effect(s) would be adequately managed during the period of extended operation. For AMPs that were not consistent with GALL, or were not addressed in GALL, the staff's review determined whether the AMPs were adequate to manage the aging effects for which they were credited.

Several FCS AMPs were described by the applicant as being consistent with GALL, but with some deviation from GALL. These deviations are of three types: (1) exceptions to GALL - [*provide definition here*], (2) clarifications to GALL - [*provide definition here*], or (3) enhancements to GALL - [*provide definition here*]. For each AMP that had one or more of these deviations, the staff reviewed each deviation to determine (1) whether the deviation is acceptable and (2) whether the AMP, as modified, would adequately manage the aging effect(s) for which it is credited.

For those AMPs that are not evaluated GALL, the staff evaluated the AMP against the 10 program elements (Branch Technical Position RLSB-1 in Section A-1 of SRP-LR Appendix A).

The staff also reviewed the FSAR supplement for each AMP to determine whether it provided an adequate description of the program or activity, as required by 10 CFR 54.21(d).

The AMRs and associated AMPs in the GALL report fall into two broad categories: those AMRs and associated AMPs that GALL concludes are adequate to manage aging of the components referenced in GALL, and those AMRs and associated AMPs for which GALL concludes that aging management is adequate, but further evaluation must be done for certain aspects of the aging management process. In Phase 2, the staff compared the applicant's AMR results and associated AMPs to the AMR results and associated AMPs in GALL, to determine whether the applicant's AMRs and associated AMPs were consistent with those reviewed and approved in the GALL report. For those AMR results and associated AMPs for which the applicant claimed to be consistent with GALL, and for which GALL did not recommend further evaluation, the staff conducted an inspection to confirm that the applicant's AMRs and associated AMPs were consistent with the GALL AMRs and associated AMPs. For those AMRs and associated AMPs for which GALL recommended further evaluation, in addition to its confirmatory inspection, the staff reviewed the applicant's evaluation to determine whether it addressed the additional issues recommended in the GALL report. Finally, for AMRs and associated AMPs that were not consistent with GALL, the staff's review determined whether the AMRs and associated AMPs were adequate to manage the aging effects for which they were credited.

Once it had determined that the applicant's AMRs and associated AMPs were adequate to manage aging, the staff performed Phase 3 of its review by reviewing plant-specific structures and components to determine whether the applicant has demonstrated that the effects of aging

will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.0.3 Aging Management Programs

Table 3.0.3-1 presents the common aging management programs, the associated GALL program, the system groups that credit the program for management of component aging, and the SER section that contains the staff's review of the program.

Table 3.0.3-1

Common Aging Management Programs

Applicant's AMP (LRA section)	Associated GALL AMP	LRA System Groups that Credit the AMP for Aging Management	Staff Evaluation (SER Section)
Bolting Integrity (B.1.1)	XI.M3, XI.M18	3.1 - RCS 3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion	3.0.3.1
Chemistry (B.1.2)	XI.M2, XI.M21	3.1 - RCS 3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion 3.5 - Structures	3.0.3.2
Containment ISI (B.1.3)	X.S1, XI.S1, XI.S2	3.5 - Structures 4.5 - Concrete and Containment Tendon Pre-Stress TLAA	3.0.3.3
Flow-Accelerated Corrosion (B.1.5)	XI.M17	3.1 - RCS 3.4 - Steam and Power Conversion	3.0.3.4
Inservice Inspection (B.1.6)	XI.M1, XI.S3	3.1 - RCS 3.5 - Structures	3.0.3.5

Boric Acid Corrosion Prevention (B.2.1)	XI.M10,	3.1 - RCS 3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion 3.5 - Structures 3.6 - Electrical	3.0.3.6
Cooling Water Corrosion (B.2.2)	XI.M20, XI.M21	3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion	3.0.3.7
Fatigue Monitoring (B.2.4)	X.M1	3.1 - RCS 4.3 - Metal Fatigue TLAA	3.0.3.8
Fire Protection (B.2.5)	XI.M26, XI.M27	3.3 - Auxiliary 3.5 - Structures	3.0.3.9
Periodic Surveillance and Preventive Maintenance (B.2.7)	Plant-Specific	3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion 3.5 - Structures	3.0.3.10
Structures Monitoring (B.2.10)	XI.S6, XI.S7	3.3 - Auxiliary 3.5 - Structures	3.0.3.11
General Corrosion of External Surfaces (B.3.3)	Plant-Specific	3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion 3.5 - Structures	3.0.3.12
One-Time Inspection (B.3.5)	XI.M32	3.1 - RCS 3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion	3.0.3.13
Selective Leaching (B.3.6)	XI.M33	3.2 - ESF 3.3 - Auxiliary 3.4 - Steam and Power Conversion 3.5 - Structures	3.0.3.14

Table 3.0.3-2 presents the system-specific aging management programs, the associated GALL program, the system groups that credit the program for management of component aging, and the SER section that contains the staff's review of the program.

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Table 3.0.3-2

System-Specific Management Programs

Applicant's AMP (LRA section)	Associated GALL AMP	LRA System Groups that Credit the AMP for Aging Management	Staff Evaluation (SER Section)
Containment Leak Rate (B.1.4)	XI.S1, XI.S4	3.5 - Structures	3.5.2.3.1
Reactor Vessel Integrity (B.1.7)	XI.M31	3.1 - RCS	3.1.2.3.1
Diesel Fuel Monitoring and Storage (B.2.3)	XI.M30	3.3 - Auxiliary	3.3.2.3.1
Overhead Load Handling Systems Inspection (B.2.6)	XI.M23	3.3 - Auxiliary	3.3.2.3.2
Reactor Vessel Internals Inspection (B.2.8)	XI.M13, XI.M16	3.1 - RCS	3.1.2.3.2
Steam Generator (B.2.9)	XI.M19	3.1 - RCS	3.1.2.3.3
Alloy 600 (B.3.1)	XI.M11	3.1 - RCS	3.1.2.3.4
Buried Surfaces External Corrosion (B.3.2)	XI.M34	3.3 - Auxiliary	3.3.2.3.3

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Table 3.0.3-2 (Con't)

System-Specific Management Programs

Applicant's AMP (LRA section)	Associated GALL AMP	LRA System Groups that Credit the AMP for Aging Management	Staff Evaluation (SER Section)
Non-EQ Cable AMP (B.3.4)	XI.E1, XI.E2, XI.E3	3.6 - Electrical	3.6.2.3.1
Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (B.3.7)	XI.M12	3.1 - RCS	3.1.2.3.5

3.0.3.1 Bolting Integrity Program

3.0.3.1.1 Summary of Technical Information in the Application

The applicant's bolting integrity program is discussed in LRA Section B.1.1, "Bolting Integrity Program." The applicant states that the program is consistent with GALL programs XI.M3, "Reactor Head Closure Studs" and XI.M18, "Bolting Integrity," with the exception that the applicant has not identified stress corrosion cracking (SCC) as an aging effect requiring management for high-strength carbon steel bolting in plant indoor air. The applicant also states that it will utilize ASME Section XI, Subsection IWF visual VT-3 inspection requirements rather than volumetric inspections for the inspection of supports.

This AMP is credited with managing aging in bolts in the RCS, ESF, auxiliary, and steam and power conversion systems, as well as in structural bolts.

The applicant performed inspections of bolted components under the FCS ISI program, the boric acid corrosion (BAC) prevention program, and the structures inspection program. Visual inspections conducted under the BAC prevention program included the inspection of bolted components in borated systems. Any indication of boric acid residue or damage is reported and evaluated to determine if a component can remain in service per established procedures. Documentation of operating experience is included as part of the BAC prevention program. On occasion, visual observations have identified BAC damage. These deficiencies were documented in accordance with the FCS corrective action program and resulted in repair or replacement, if required. Review of the plant-specific operating experience indicates that the inspections have been effective in managing the aging effects of bolted components.

In its LRA

On the basis of the above discussion, the applicant concluded that the bolting integrity program provides reasonable assurance that bolting aging effects will be adequately managed.

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the matters described in the GALL report, except to ensure that the material presented in the LRA was applicable, and to verify that the applicant had identified the appropriate programs as described and evaluated in the GALL report. The staff evaluated those aging management issues recommended for further evaluation in the GALL report. The staff also reviewed aging management information submitted by the applicant that was different from that in the GALL report or was not addressed in the GALL report. Finally, the staff reviewed the USAR supplement to ensure that it provided an adequate description of the programs credited with managing aging for the reactor system components.

In LRA Section 3.1, the applicant provided brief descriptions of the reactor systems and summarized the results of its AMR of the reactor systems at FCS.

Table 3.1-1 below provides a summary of the staff's evaluation of components, aging effects/mechanisms, and AMPs listed in LRA Section 3.1 that are addressed in the GALL report.

Table 3.1-1

Staff Evaluation Table for FCS Reactor System Components in the GALL Report

Component Group	Aging Effect/Mechanism	AMP in GALL Report	AMP in LRA	Staff Evaluation
Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)		Consistent with GALL. GALL recommends further evaluation (See Section XXXX below)
Steam generator shell assembly	Loss of material due to pitting and crevice corrosion	Inservice inspection; water chemistry		Consistent with GALL. GALL recommends further evaluation (See Section XXXX below)
Isolation condenser	Loss of material due to general, pitting and crevice corrosion	Inservice inspection; water chemistry		BWR
Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99		Consistent with GALL. GALL recommends further evaluation (See Section XXXX below)
Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance		Consistent with GALL. GALL recommends further evaluation (See Section XXXX below)
Westinghouse and B&W baffle/former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific		N/A (FCS is a CE plant)

NEW DOCUMENTATION OF STAFF REVIEW RESULTS

Because of the increased importance of the AMR inspection results to the staff's findings, future SERs will include the reports from both the scoping and screening inspection as well as the AMR inspection as appendices

LESSONS LEARNED TO DATE

DIFFERENT DEFINITIONS OF WHAT "CONSISTENT WITH GALL" MEANS

- RESOLVED THIS WITH NEI
- BEGINNING WITH NEXT LRA SUBMITTAL, DEFINITION WILL BE CONSISTENT

LRA CAN BE REFORMATTED TO PROVIDE THE GALL INFORMATION MORE EFFICIENTLY

- RESOLVED THIS WITH NEI
- BEGINNING WITH NEXT LRA SUBMITTAL, LRA WILL HAVE REVISED FORMAT

LESSONS LEARNED TO DATE

SER TEMPLATE SHOULD BE TRIMMED DOWN

STAFF IS REVIEWING ALTERNATIVE SER FORMATS

SUMMARY

FT. CALHOUN IS FIRST LICENSE RENEWAL APPLICATION UTILIZING THE NEW GALL PROCESS (ACRS PRESENTATION ON FT. CALHOUN SCHEDULED FOR EARLY JUNE)

6 GALL PLANTS CURRENTLY BEING REVIEWED

STAFF HAS DEVELOPED REVIEW AND DOCUMENTATION PROCESS TO AID IN THE REVIEW AND TO AID STAKEHOLDERS

STAFF WILL REVISIT GALL DOCUMENTS, AS WELL AS LRA AND SER FORMATS, BASED ON LESSONS LEARNED FROM THE FT. CALHOUN REVIEW

INTERNAL USE ONLY

G:PlanPro(ACRS):ppmins.500
March 6, 2003

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
MARCH 5, 2003

The ACRS Subcommittee on Planning and Procedures held a meeting on March 5, 2003, in Room T2B3, 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 1:30 p.m. and adjourned at 3:15 p.m.

ATTENDEES

MEMBERS

M. Bonaca
G. Wallis
S. Rosen
J. Sieber (part time)

ACRS STAFF

J. T. Larkins
S. Bahadur
S. Duraiswamy
R. P. Savio
J. Gallo
S. Meador

NRC STAFF

i. Schoenfeld, OEDO

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the March ACRS meeting

Member assignments and priorities for ACRS reports and letters for the March ACRS meeting are attached (pp. 6-8). 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 March ACRS meeting be as shown in the attachment (pp. 6-8).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through May 2003 is attached (pp. 6-8). 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 will also discuss and develop recommendations on items included in Section II of the Future Activities List (p. 9).

RECOMMENDATION

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

3) Update on the Budget

The continuing resolution has ended, and we now have our budget. However, given the increase in workload and several special events (quadripartite meeting, 500th meeting, etc.) funds for the remainder of this fiscal year (September 30, 2003) are limited, particularly in the travel category.

RECOMMENDATION

The Subcommittee recommends the following:

- Travel for events other than scheduled full Committee and Subcommittee meetings should be approved by John T. Larkins prior to travel.

- All purchases using the Government Bankcard should be approved by Tanya.

4) ACRS Meeting with the NRC Commissioners

A meeting between the ACRS and the NRC Commissioners is scheduled for April 11, 2002, between 9-11:00 a.m. to discuss items of mutual interest. Items approved by the Commission for this meeting are as follows:

(1) Overview by the ACRS Chairman (MVB)
- 500th meeting celebration
- Quadripartite Meeting
- License Renewal Activities
- Core power uprates

- Future ACRS activities - including "sunset" of activities that do not necessarily need to be reviewed.

(2) Advanced Reactor Designs (TSK)
- Early Site Permit Process
- Options paper for resolving policy issues
- AP1000 review activities

(3) Pressurized Thermal Shock Technical Basis Reevaluation Project (WJS)

(4) ACRS Report on the NRC Safety Research Program (FPF)

The Commission has approved the above agenda with the recommendation that the PTS discussion be limited to about 5-10 minutes.

RECOMMENDATION

The Committee needs to approve the slides at this month's meeting with the plan for finalizing them prior to March 28, so that they can be copied and distributed to the Commission a week or so in advance of the meeting.

5) Role and Use of PRA in the Regulatory Decisionmaking Process

Mr. Fleming, under a contract with NRC/ACRS, submitted a draft report on the role and use of PRA in risk informed decisionmaking process to the Committee. The Committee discussed this report with Mr. Fleming during its February 2003 meeting.

During the February 2003 ACRS meeting, Dr. Nourbakhsh, ACRS Senior Fellow, presented an outline of a draft "White Paper" and recommendations to be included in the paper based on Mr. Fleming's report .

Dr. Nourbakhsh has worked with Carl Fleming to edit his report and is in the process of having it published as a NUREG. We anticipate this being issued in the next month.

Dr. Apostolakis had prepared a draft report to the Commission instead of a White Paper as previously discussed in February. Several members commented on the draft report prepared by Dr. Apostolakis

RECOMMENDATION

The Subcommittee recommends that Dr. Apostolakis revise the report to include more explicitly some members' comments and see if there are other issues in the Fleming report that should be included in the letter on enhanced use of PRA in Regulatory Decision making. Additionally, the Committee should decide whether to include a transmittal memorandum with Mr. Fleming's report and if so, what should be the message.

6) Comments by Mr. Robert H. Leyse, a Member of the Public, Regarding Fouling in Reactivity Insertion Accidents

Mr. Leyse, a member of the public, submitted a petition for rulemaking (pp. 10-16) to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants. This includes all heat transfer surfaces, fuel elements, steam generators, condensers, fan coolers, etc. The new regulations must also require the inclusion of fouling considerations in the NRC-funded test programs such as the Rod Bundle Heat Transfer Program at the Penn State University. Some of the comments provided by Mr. Leyse for evaluation by the ACRS are as follows:

- Even though fouling of nuclear reactor fuel elements has a greater impact on the Reactivity Insertion Accidents and LOCAs than high-burnup, the NRC and its collaborators continue to totally ignore the impact of fouling.

- There was no discussion of fouling either at the October 9, 2002 ACRS Subcommittee meeting on Reactor Fuels or at the October 2002 full Committee meeting. In addition, the October 17, 2002 ACRS letter on the Confirmatory Research Program on High-Burnup Fuel did not include any discussion on fouling.

- The Rod Bundle Heat Transfer Program at the Penn State University includes no provisions for the impact of fouling on the heat transfer surfaces. Surprisingly, the ACRS did not point out the gross inadequacy of this Program. Copies of the information provided by Mr. Leyse were sent to Dr. Powers and Dr. Wallis.

The Subcommittee discussed several options for handling the concerns raised by Mr. Leyse, including the following:

- Should this matter be referred to the EDO, as has been done in the past on similar issues?
- Should Mr. Leyse be notified by the ACRS Executive Director that the ACRS does not get involved in the petition for rulemaking process and that the Reactor Fuels Subcommittee plans to invite him to discuss his concerns at the next meeting of the Reactor Fuels Subcommittee, which is tentatively scheduled to be held in October 2003?

RECOMMENDATION

The Planning and Procedures Subcommittee recommended that the ACRS/ACNW Executive Director followup with the EDO to find out what is happening with the petition for rulemaking. Dr. Larkins should draft a response to Mr. Leyse after Drs. Powers and Wallis recommend a proposed course of action.

7) Committee Review of License Renewal Applications (Peach Bottom Event)

There have been a number of e-mails exchanged (see handout) regarding the ACRS review of license renewal applications related to plants whose current operating experience raises the question of whether their license renewal applications should be approved. Most recently, this issue has come up in connection with the license renewal application for Peach Bottom Units 2 and 3, which was discussed by the Committee at its March 2003 meeting.

The applicant, NRC staff, and the Senior Resident Inspector addressed the reactor scram event which occurred at Peach Bottom in December 2002 along with the corrective action programs at the March meeting.

The ACRS has previously moved ahead with its review of license renewal applications in spite of current operating experience with the idea that these conditions would be corrected prior to entering into the renewal period.

RECOMMENDATION

The Subcommittee recommends that based on the information provided by the applicant and the staff, the Committee discuss how it wants to treat the license renewal applications of those plants that had experienced significant operating events recently, which brings into question the performance of the applicant.

8) Insights/Safety Culture

During the February 2003 meeting, the Committee discussed a draft report on Safety Culture written by ACRS members Rosen and Apostolakis and requested that Mr. Rosen revise this report incorporating the comments from the members. It was suggested that the letter be reasonably short and minimize recommendations to suggesting that the Commission encourage the staff to stay aware of developments in the Safety Culture arena both domestically and internationally.

It was further suggested that the ACRS meet with the NRC staff, First Energy Nuclear, NEI, and others (i.e., David Collins and Howard C. Whitcomb) later in the year at a subcommittee meeting to discuss initiatives related to assessing safety culture and its implications on the safe operations of nuclear facilities.

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the draft report prepared by Mr. Rosen and attempt to develop a final draft report during the March 2003 meeting and issue a letter at the April 2003 full Committee meeting.

9) Member Issues

- Emergency Notification to Members Regarding Subcommittee Cancellations

  Recent events have noted the importance of having some process in place to notify members in the event of a severe weather condition which would inhibit travel to and from the Washington area. A process is being put in place to address this issue.

- The issue has been raised in terms of how the Committee should handle differing technical views among the NRC staff after the Committee has approved a particular staff action such as DG 1077 EQ of Microprocessor-Based Equipment.

- The ACRS Executive Director was successful in getting an exemption to the Commission reappointment policy in having Dr. Thomas Kress appointed for a fourth term to begin at the end of his current term (September 3, 2003). The Commission, in the SRM, approved the reappointment and noted that this exception to its policy will be done sparingly.

- The NRC staff is planning a one week tour of a test facility associated with SWR 1000 (located in Germany) in early June 2003. The NRR staff is expecting possibly 2-3 NRC managers will attend, and wondering if an ACRS member (maybe Dr. Kress) would be interested in attending such a tour. Framatome is sponsoring the tour, however, Framatome is not responsible for the travel and hotels.

ANTICIPATED WORKLOAD
MARCH 6-8, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Nourbakhsh/ Duraiswamy	Letter report on the role and use of PRA in the regulatory decisionmaking process	--	--	**Draft**
Bonaca	--	Kobetz	Overview of the format and content of the Ft. Calhoun license renewal application	--	--	--
	All Members	Larkins	Preparation for meeting with the NRC Commissioners, April 11, (9:00-11:00a.m.)	--	--	--
Ford		Weston	Vessel head penetration cracking and vessel head degradation-Information Briefing	--	--	--
Kress		El-Zeftawy	Early Site Permit Process	A	To provide early feedback to the Commission	**Draft 1**
Leitch	Bonaca	Assa/Kobetz	Peach Bottom License Renewal Application-**FINAL REVIEW**	A	To meet the CTM schedule	Draft 1
Rosen	--	Weston	Safety Culture Report	--	--	Draft
Sieber	Apostolakis	Weston	Reactor Oversight Process	A	To respond to Commission SRM	--
		El-Zeftawy	Draft final revision to Reg. Guide 1 .180 (DG-1119), "Review Guidelines for EMI/RFI."	A	To support the staff schedule	--

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ANTICIPATED WORKLOAD
APRIL 10-12, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly/ Duraiswamy	Draft final Regulatory Guide (DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities"	A	To meet the CTM schedule	--
Bonaca	All Members	Larkins	Meeting with the NRC Commissioners - April 11, 2003, 9:00-11:00 A.M.	--	--	--
		Kobetz	St. Lucie license renewal application - INTERIM REVIEW-SUBCOMMITTEE REPORT	--	--	--
Leitch	--	Weston	Significant recent operating events	--	--	--
Powers	--	El-Zeftawy	Reg. Guides and Generic Letter on Control Room Habitability	A	To meet the CTM schedule	--
Rosen	--	Weston	Safety Culture Report	A	To provide Committee's views to the Commission	Draft
Shack	--	Snodderly/ Duraiswamy	Draft final risk-informed revisions to 10 CFR 50.44 regarding combustible gas control systems.	A	To meet the CTM schedule	--
Wallis	--	Snodderly/ El Zeftawy	Subcommittee Report - AP 1000 Thermal-Hydraulic Issues	--	--	--

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ANTICIPATED WORKLOAD
MAY 8-10, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Ford	--	Weston	Vessel head penetration cracking and vessel head degradation	A	To provide feedback to the Commission	--
Kress	--	Kobetz	Proposed criteria for the treatment of individual requirements in a regulatory analysis	A	To provide feedback to the staff	--
Shack	--	Snodderly/ Duraiswany	Proposed revisions to SRP Section 3.9.8 and Reg. Guide 1.178 for risk-informed inservice inspection of piping	A	To support the staff's schedule	--

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II. ITEMS REQUIRING COMMITTEE ACTION

1. Draft Regulatory Guide DG-1079 (Proposed Revision 3 to Regulatory Guide 1.32), "Criteria for Power Systems for Nuclear Power Plants" (Open) (JDS/TJK)
Estimated Time: 1 hour

Purpose: Determine a Course of Action

Review requested by the NRC Staff [S. Aggarwal, RES]. The draft Regulatory Guide DG-1079 (proposed Revision 3 to Regulatory Guide 1.32) endorses IEEE Standard 308-2001, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," with one exception, namely, the NRC does not endorse IEEE Standard 308-2001 criteria for sharing safety-related dc power systems in multi-unit nuclear plants. The staff believes that there is general industry consensus that IEEE Standard 308-2001 provides acceptable guidance for design of safety-related power systems. The staff plans to issue DG-1079 for public comment and requests that the ACRS review the draft final version of this Guide after reconciliation of public comments.

Mr. Sieber recommends that the Committee review the draft final Guide after reconciliation of public comments.

2. Draft Final Amendment to 10 CFR 50.55a to Incorporate by Reference ASME Boiler and Pressure Vessel and OM Code Cases" (Open) (WJS/MRS)
Estimated Time: 1½ hour

Purpose: Determine a Course of Action

Review requested by the NRC Staff [H. Tovmassian, NRR]. The staff proposes to amend its regulations to incorporate by reference certain NRC Regulatory Guides listing Code cases published by the ASME which the NRC staff has reviewed and found to be acceptable for use. These Code cases provide alternatives to requirements in the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants pertaining to construction, inservice inspection, and inservice testing of nuclear power plant components. The amendment incorporates by reference three regulatory guides that address NRC review and approval of ASME published Code cases. As a result, the Code cases in these regulatory guides are incorporated by reference into the NRC's regulations. During the February 7-8, 2002 Meeting, the Committee decided not to review the proposed amendment and asked for an opportunity to review the draft final rule after reconciliation of public comments. The staff is requesting that the ACRS review the draft final rule package which includes resolution of 16 comments..

The Planning and Procedures Subcommittee recommends that Dr. Shack propose a course of action.

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Robert H Leyse

P O Box 2850
Sun Valley, ID 83353

February 10, 2003

John T. Larkins, Executive Director
ACRS
USNRC
Washington, DC 20555-0001

Dear Executive Director:

This member of the public, in accordance with the provisions of the Federal Advisory Committee Act, submits the following written comments for evaluation by the ACRS. The substance of these comments is included in the three documents that are in this letter and that are, or will be, on the NRC's Rulemaking Website

The first document, pages 1-3, is a Petition for Rulemaking that directs the NRC to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants. The second document, pages 4-7, NRC Ignores **Fouling** in RIA and LOCA Analyses, is a comment related to the Petition. The third document, page 8, NRC Ignores **Fouling** in Reliability and Probabilistic Risk Assessment, is another comment related to the Petition. I have also copied the list of other public comments from the NRC's Rulemaking Website and the ACRS may access and review these as it chooses

Petition and Other Documents

Original File	Size	File Type	Date	HTML	Description
1210-0003.htm	4k	HTML	01/02/03	1210-0003.htm	Federal Register Notice - Extension of Comment Period
1210-0002.wp	8k	Word Perfect	10/31/02	1210-0002.htm	Incoming petition
1210-0001.htm	11k	HTML	10/31/02	1210-0001.htm	Federal Register Notice

Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C., 20555-0001
Attn: Rulemakings and Adjudications Staff

September 2, 2002

Petition

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1. Brief description of needed regulations:

Regulations are needed to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants. This includes all heat transfer surfaces, fuel elements, steam generators, condensers, fan coolers, etc. The new regulations must also require the inclusion of fouling considerations in NRC funded test programs such as the Rod Bundle Heat Transfer (RBHT) at Penn State. Furthermore, fouling considerations must be added to the vast assortment of NRC produced computer codes such as the RELAP and TRAC series.

2. Petitioner's grounds for and interest in the action requested:

Petitioner is aware that the fouling of heat transfer surfaces is generally not adequately considered in the licensing and the compliance inspections of nuclear power plants.

3. Petitioner's statement of the specific issues involved, views with respect to those issues, relevant technical, scientific, or other data involved:

The following examples cover only a few of the significant heat transfer surfaces within operating nuclear power plants. The NRC must produce a complete inventory of all significant heat transfer surfaces. The regulations must require appropriate reporting of the performance of these surfaces including records of degradation, cleaning procedures and effectiveness, mechanical degradation of the heat transfer assemblies especially the fuel assemblies. Regulations must require detailed reporting that must also be available to the public.

Regulations do not address the significance of severe fouling of the nuclear fuel elements. Currently, Licensing Bases and Technical Specifications do not limit the amount of fouling of fuel elements. In some cases, the fouling is sufficient to induce significant oxidation of the fuel cladding, a situation that has led to "... a debate over (whether) the 17 percent includes the prior oxidation or it's just the oxidation during the ramp-up." (See ACRS Subcommittee transcript, May 31, 2002). In another case, the axial offset anomalies that were traced to fouling of nuclear fuel elements, the situation was evaluated as follows, "They are a(n) annoyance. They affect economics, but they are not safety issues." (See ACRS Subcommittee transcript, April 24, 1998.) Severe fouling of nuclear fuel elements also leads to axial growth of the fuel rods beyond design limits as the operating temperatures of the fuel rods become greater than allowed for in design. The fuel rods may expand sufficiently along their length to become restrained from further axial growth by the fuel assembly end fittings; the rods then bow and contact adjacent rods as well as control rod guide tubes.

While one nuclear power plant continued to operate at power, the need for repeated cleaning of an air cooling heat exchanger was not recognized as a key indicator of a substantial leak in the primary reactor system. As long as the plant operation remained within the technical specifications, there was no basis for the plant operators to perform prudent investigations. Clearly, regulations must address the need for investigating the grossly off-normal performance of this heat exchange equipment.

In several instances, the fouling of steam generator tubes has reduced the heat transfer effectiveness sufficiently to force operation at reduced secondary side pressures in order to maintain heat transfer rates. In the overall pattern, this likely will impact safety issues in addition to being an operating annoyance.

The fouling of main condenser heat transfer surfaces has led to a degradation of heat transfer effectiveness. Furthermore, on occasion these fouling deposits have been released into the coolant stream and then have contributed to fouling of fuel elements.

Shifting to the arena of test programs, during the past several decades the NRC and its defunct predecessors have funded well over one billion dollars of heat transfer test programs that have not included any allowance for the severe fouling of heat transfer surfaces that occurs during the operation of nuclear power plants. These test programs must be thoroughly studied and allowances must be made for a range of fouling of the heat transfer surfaces. Very likely, it will not be possible to produce reliable allowances for a range of degrees of fouling in which case, the results of the prior test programs such as FLECHT, LOFT, Semiscale and others must not be applied to the production of computer codes for reactor heat transfer analyses. Currently, the NRC is spending millions of dollars in heat transfer testing at facilities such as RBHT at Penn State, and these programs must be realigned to cover the cases of several degrees of fouling.

Shifting to the arena of computer codes, during the past several decades the NRC and its defunct predecessors have funded several hundred million dollars of code production related to heat transfer processes in nuclear power reactors. The codes thus produced, several versions of TRAC, RELAP and others, have included no provisions for a range of fouling conditions on the heat transfer surfaces. These codes must not be applied to the licensing of nuclear power plants until reliable allowances for a range of degrees of fouling are incorporated in the codes.

One measure of whether proper regulations are effected will be a determination that under the new regulations, conditions similar to those already reported in certain Licensee Event Reports, will henceforth constitute license violations. Very likely there is a preponderance of documented nuclear power plant operating experience that is worthy of such review; as a possible example, one candidate may be Licensee Event Report 50-458/99-016-00.

The increased attention to detail in plant design, analysis and operations that will be effected via these needed regulations will enhance operating effectiveness, will discourage incomplete and misleading reporting to regulatory authorities and will reduce opportunities for sabotage by insiders. The increased detail of reporting to the public is required to fulfill many obligations including: (1.) The timely disclosure of significant information to professional risk analysts who advise large financial management organizations such as J. P. Morgan, Chase, H&Q. (2.) The timely disclosure of significant information to those individual investors who track available corporate performance data in making their investment decisions. (3.) The timely disclosure of significant information to state agencies that oversee the sale and/or acquisition of nuclear power plants by utility holding companies that operate within their jurisdiction.

View/Submit Public Comments

Original File	Size	File Type	Date	HTML	Description
1211-0003.pdf	116k	PDF	01/29/03	NONE	Comment letter from D. R. Woodlan, STARS
1211-0002.pdf	19k	PDF	01/14/03	NONE	Industry comments submitted by NEI from Alex Marion
1211-0001.msw	1864k	MS Word	12/24/02	NONE	NRC Ignores Fouling in RIA and LOCA Analyses from Robert H. Leyse

NRC Ignores **Fouling** in RIA and LOCA Analyses

Even though **Fouling of Nuclear Reactor Fuel Elements (Fouling)** has a greater impact on Reactivity Insertion Accidents (RIAs) and Loss of Coolant Accidents (LOCAs) than high-burnup, the NRC and its collaborators continue to totally ignore the impact of **Fouling**. The following public documents affirm this assertion:

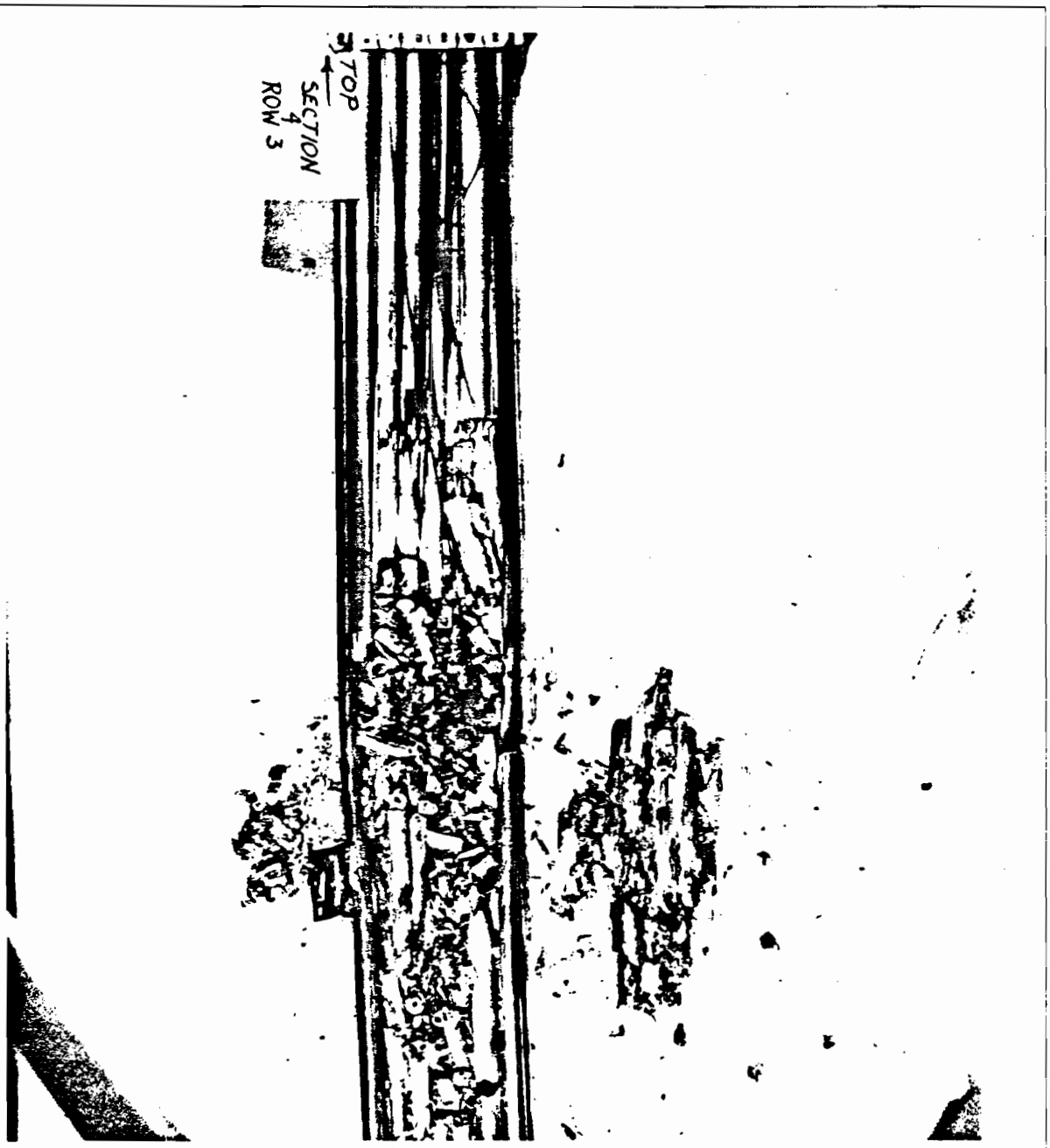
There was no discussion of **Fouling** during the October 9, 2002, the ACRS Subcommittee on Reactor Fuels. The Electric Power Research Institute (EPRI) discussed its unwarranted topical report on the response of high-burnup fuel to reactivity insertion accidents. EPRI disclosed nothing about the impact of **Fouling** and the Subcommittee asked no related questions.

Again, there was no discussion of **Fouling** in follow-on presentations to the Full ACRS on October 10, 2002. The Chairman of the Reactor Fuels Subcommittee described the EPRI presentation as, "... a very delightful presentation from the Electric Power Research Institute in describing its work on Reactivity Insertion Accidents to date." The Full ACRS asked no questions related to **Fouling**.

On October 17, 2002, the Full ACRS sent a letter to the Chairman, NRC, with the subject: CONFIRMATORY RESEARCH PROGRAM ON HIGH-BURNUP FUEL. There is no discussion of **Fouling** in this letter. In fact the ACRS lauds the NRC approaches, "RES has a well-organized and leveraged program of confirmatory research on the behavior of high-burnup fuel under the conditions of reactivity insertion events in pressurized water reactors, design-basis loss-of-coolant accidents (LOCAs), and anticipated transients without scram in boiling water reactors." Surprisingly, the ACRS expressed a somewhat veiled concern as follows: "We remain concerned that the time-temperature conditions used in the study of high-burnup fuel during design-basis LOCAs may not reveal phenomena unique to high-burnup fuel." Well, the ACRS is

invited to view the enclosed photograph that reveals the results (without burn-up or **Fouling**) when otherwise realistic time-temperature-flow conditions are deployed (FLECHT Run 9573).

On November 12, 2002, the ACRS Subcommittee on Thermal-Hydraulic Phenomena discussed the Rod Bundle Heat Transfer (RBHT) program that is being conducted at Pennsylvania State University. The RBHT program includes no provisions for the impact of **Fouling** of the heat transfer surfaces. Thus, this program is useless from the view of realism in evaluating the LOCA characteristics of licensed nuclear power reactors. Surprisingly, the ACRS did not point out this gross inadequacy of the RBHT program. At times, the discussions referred to the NRC's TRAC series of reactor safety analysis codes. These codes include no provisions for **Fouling** and thus are also useless for the realistic evaluation of LOCAs.


TOP
SECTION 4
ROW 3

(NRC_Ignores_Fouling_in_Reliability_and_Probabilistic_Risk_Assessment.doc) have been uploaded to **View/Submit Public Comments** as **1211-0004** under **Robert H. Leyse**
Submitted to the NRC' Rulemking Website at 9:28 PM, February 9, 2003

NRC Ignores **Fouling** in Reliability and Probabilistic Risk Assessment

Presentations by NRC staff, nuclear power equipment suppliers, and nuclear power lobbyists at recent ACRS subcommittee meetings have not included any consideration of the impacts of fouling of heat exchange equipment. These recent subcommittee meetings include:

Date	Subcommittee
01/24/2003	Reliability and Probability Risk Assessment Subcommittee
01/23/2003	Reliability and Probability Risk Assessment Subcommittee
01/22/2003	Reliability and Probability Risk Assessment Subcommittee
01/21/2003	Plant Operations and Reliability & Probabilistic Risk Assessment Subc.

Especially serious is the absence of any consideration of fouling of the nuclear fuel elements.

The transcript of the January 24, 2003 meeting, page 124, lines 13 and 14, includes the following remark by ACRS Member Shack, "The nagging fear is that you have missed something."

Dr. Larkins, related petitions that the ACRS may choose to evaluate are posted on the NRC's Rulemaking Website as follows:

Date Posted	Title	PRM No.
10/12/2001	Robert H. Leyse	PRM-50-73
1/29/2002	Robert H. Leyse: Supplement to a Petition for Rulemaking	PRM-50-73A
8/9/2002	Robert H. Leyse	PRM-50-7

Finally, please provide the ACRS with the LER that is referenced in the following paragraph that I have copied from page 3 of this letter:

One measure of whether proper regulations are effected will be a determination that under the new regulations, conditions similar to those already reported in certain Licensee Event Reports, will henceforth constitute license violations. Very likely there is a preponderance of documented nuclear power plant operating experience that is worthy of such review; as a possible example, one candidate may be Licensee Event Report 50-458 99-016-00.

Thank you for your attention to these matters,

Robert H. Leyse

Robert H. Leyse