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

July 30, 2004

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 514th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JULY 7-9, 2004 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 514TH meeting, July 7-9, 2004, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letter, and memoranda:

REPORT:

Report to Nils J. Diaz, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP1000 Passive Plant Design, dated July 20, 2004.

LETTER:

Letters to Luis A. Reyes, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs, dated July 19, 2004.

MEMORANDA:

Memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Draft Final Revision to 10 CFR 50.55a, "Codes and Standards," dated July 13, 2004

- Deferral of ACRS Review of Draft Regulatory Guide, DG-1128, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," (Revision 4 to Regulatory Guide 1.97), dated July 15, 2004

HIGHLIGHTS OF KEY ISSUES

1. Final Safety Evaluation Report (SER) Associated with the AP1000 Design Certification

The Committee heard presentations by and held discussions with representatives of the NRC staff and Westinghouse regarding the safety aspects of the Westinghouse Electric Company application for certification of the AP1000 passive plant design. The Committee reviewed the application which consists of the AP1000 design control document (DCD), and the probabilistic risk assessment. The ACRS viewed the AP1000 design in concert with all the ACRS review activities conducted for certification of the AP600 design. The ACRS focused on the changes from the AP600 design made to accommodate the increased power level and ensured that such changes did not pose any new safety considerations or result in an unacceptable increase in risk. The new phenomena identification and ranking table was also reviewed to determine if any new phenomena were identified and that there were no significant changes in ranking of events. The ACRS reviews did not address security related issues.

During the AP1000 review, the ACRS identified technical issues that needed additional discussions such as the automatic depressurization system (ADS)-4 squib valve function, assurance of long-term cooling (strainer blockage), code deficiencies, range of pi-groups values, in-vessel retention/fuel-coolant interaction, organic iodine production, and catastrophic failure of a free-standing steel containment. The ACRS agreed with the resolutions proposed by the staff of all but two of these issues. For the in-vessel retention and organic iodine production, the ACRS developed its own arguments for the resolution. All ACRS issues have been resolved. The Committee also discussed concerns expressed by a member of the public. Most of these concerns are process related and are within the purview of the staff. The Committee considered one technical item raised by the individual. This item concerned the effect of solar heating on the passive containment cooling system's ability to deal with design basis accidents. The Committee found Westinghouse's assumption for this item to be sufficiently conservative.

Committee Action

The Committee issued a report to Chairman Nils J. Diaz on this matter, dated July 20, 2004. The report provides a brief description of the design and summarizes the Committee's review. The Committee in its report concluded that the AP1000 design is robust and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public.

2. Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs

The Committee heard presentations from the NRC staff and from the Nuclear Energy Institute (NEI) concerning the proposed draft final generic letter (GL) related to the potential impact of debris blockage on emergency recirculation during design basis accidents at pressurized water reactors (PWRs). The staff presented a version of the GL to the Thermal-Hydraulic Phenomena Subcommittee on June 22-23, 2004, together with its resolution of the public comments from various stakeholders. This version of the proposed GL was "action-oriented"

and directed licensees to perform analyses and take corrective action to resolve identified discrepancies in accordance with the schedule approved by the Commission for resolving Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance." At the full Committee meeting on July 7, the staff provided the Committee with a different version of the GL, which removed many of the action-oriented requirements, and returned the letter to the format that was originally issued for public comment. The staff explained that the Office of the General Counsel (OGC) could not accept the "action-oriented" provisions of the letter, because they imposed new requirements on licensees, and that a GL could not be used for this purpose. Therefore, the version that was discussed on July 7 was more of a request for information from licensees, to allow the staff to determine whether they were in compliance with 10 CFR 50.46.

Subsequently, on July 9, the staff returned to the meeting, and informed the Committee that its discussions with OGC had continued, and it appeared that it might be able to return some of the "action-oriented" provisions to the GL. The staff could not make a commitment that this would actually occur, but it left the Committee with the impression that it would continue to work with OGC to develop a GL that would be as "action-oriented" as possible, given the legal constraints of the GL process. The Committee also understood that the Committee to Review Generic Requirements (CRGR) would also review the proposed GL, and would ensure that it complied with the process requirements for generic communications.

During the discussion of the GL, the Committee questioned the staff about the need and utility of issuing the GL before the review of the industry guidance document is complete. Because the GL references the guidance document, and is intended to be used to ensure its implementation, the Committee did not understand why the staff's safety evaluation for the guidance document and the GL should not be issued simultaneously. The staff explained that it wanted the industry to see the GL as soon as possible and there was no reason to delay its issuance until the guidance had been reviewed.

Mr. Pietrangelo (NEI) commented on July 7 that he had not seen the version of the GL that the Committee was considering, but from the discussion, it appeared that it was very similar to the version that had been issued for public comment. He urged the staff not to issue the GL in that form and he noted that the industry comments consistently asked the staff to issue the GL in a form that is more "action-oriented," and that acknowledges that the position that the staff is taking is a backfit. He explained that the industry is fully prepared to make any necessary modifications, but they would prefer a process that followed the one used to resolve this issue for BWRs, rather than the one proposed in the original draft GL.

Committee Action

The Committee issued a letter to the Executive Director for Operations (EDO) on this matter, dated July 19, 2004, recommending that a GL be issued, with the format and process to be defined by the staff. Also, the staff should continue confirmatory research in areas where the technical basis of the guidance is uncertain, and on issues such as chemical and downstream effects that are not directly addressed by the guidance proposed by NEI. The Committee will consider the technical issues associated with the industry guidance document at a September Subcommittee meeting, and during its October 2004 meeting.

3. Risk-Informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"

The Committee met with representatives of the NRC staff to discuss risk-informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." The briefing focused on the staff's approach to responding to the Commission's July 1, 2004, Staff Requirements Memorandum (SRM) and the upcoming *Federal Register Notice* to solicit public comment on the staff's conceptual framework for risk-informing 10 CFR 50.46. The Office of Nuclear Regulatory Research (RES) also briefed the Committee on the status of its expert elicitation in support of the proposed rulemaking. The staff concluded that LOCA frequency estimates can be sensitive to the method used to analyze panelists' input.

Committee Action:

This was an information briefing. The Committee plans to review the proposed rule to risk-inform the requirements addressing large break loss-of-coolant accidents in November 2004.

4. Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries

In an April 28, 2003, SRM on the April 11, 2003 meeting with the ACRS, the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements, The Commission should be informed." Dr. Nourbakhsh, ACRS Senior Staff Engineer, has prepared a draft white paper which is to be used by the ACRS in responding to the Commission. During the July 7-9, 2004,ACRS meeting, the Committee was briefed by Dr. Nourbakhsh regarding his draft White Paper on differences in regulatory approaches and requirements between U.S. and other countries.

Committee Action

The Committee plans to discuss the draft final of the white paper on differences in regulatory approaches and requirements between U.S. and other countries during the September 8-11, 2004 ACRS meeting.

5. Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants

The purpose of this session was to hear presentations from the staff and industry regarding a proposed generic communication on Ultrasonic Flow Measurement (UFM) devices.

The staff's presentation described the basic principles of UFM technology, their application in nuclear power plants, and several overpower events caused by inaccuracies in UFM instruments. It was noted that these overpower incidents are not a safety issue because the uncertainties in these devices are small compared to existing safety margins. The staff proposes to issue a bulletin which will (1) advise addressees of operating experience with UFM

devices that have not provided the intended accuracy needed to maintain plant operation within licensed thermal power, (2) advise addresses that there are potential questions regarding the use of UFM devices because of sensitivities to plant configuration and lack of data to support instrument performance, (3) recommend the licensees confirm UFM accuracy by comparisons with standard tests of known accuracy, and (4) require addresses provide a written response that verifies actions taken to ensure plants are not operated above licensed thermal power or outside the licensed design basis.

A member of the staff presented another approach for addressing this problem. Since inaccuracies in UFM devices are not safety significant nor a generic issue, the staff should issue an Information Notice or Regulatory Information Summary instead of a Bulletin.

A representative from Caldon, a vendor of UFM instruments, read a prepared statement to the Committee. Caldon believes that certain types of UFM instruments have a measurement uncertainty which is greater than that approved by the staff in Safety Evaluation Reports (SER). It was noted that the staffs approval was based upon proprietary information contained in a topical reports.

Committee Action

The Committee has deferred action to write a letter until after it has reviewed additional documents and had further discussions with the staff and industry.

6. Status of the ACRS Members' Assessment of the Quality of Selected NRC Research Projects

RES is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in assessing the effectiveness and utility of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During the July 7-9, 2004 ACRS meeting, the Committee discussed the status of the activities of cognizant ACRS members associated with the assessment of the quality of the research projects on Sump Blockage and on MACCS Code.

Committee Action

The Committee plans to discuss the preliminary assessment of the quality of the research projects on Sump Blockage and MACCS Code during the September 8-11, 2004 ACRS meeting.

7. Future Plant Designs Subcommittee Report

The Chairman of the ACRS Subcommittee on Future Plant Designs provided a report to the Committee regarding the NRC staff's proposed "Regulatory Structure for New Plant Licensing, Part 1: Technology-Neutral Framework" document that was discussed at the June 24, 2004, Subcommittee meeting. The staff has developed a set of four protective strategies: initiating

event frequency, barrier integrity, protective systems, and accident management. The objective of this document is to develop and implement a risk-informed regulatory structure. To meet this objective, four tasks are proposed: development of a technology-neutral framework, formulation of proposed content, development of guidance on a technology-specific basis, and formulation of regulatory guides.

The expected regulatory structure will have desired characteristics to establish acceptance criteria of the technology-neutral framework. These characteristics include reproducible, traceable, understandable, defensible, flexible, risk-informed, performance-based, completeness, uncertainty, defense-in-depth, and consistency.

The proposed framework document has three major parts. Part 1 represents framework for a technology-neutral regulatory structure that describes framework road map, safety fundamentals-protective strategies, risk guidelines and design/construction/operation expectations, treatment of uncertainties, and development of technology-neutral requirements. Part 2 (content of technology-neutral requirements) and Part 3 (framework for a technology-specific regulatory structure) have not been written yet.

Committee Action

The staff's briefing was provided for information only. The Subcommittee will follow-up on the progress of this matter during future meetings.

8. Thermal-Hydraulic Phenomena Subcommittee Report

Dr. Wallis, Chairman of the Thermal-Hydraulic Phenomena Subcommittee, reported that the Subcommittee met on June 22-23, 2004, to discuss the technical guidance methodology developed by NEI to address PWR sump blockage during large, break LOCAs (GSI-191). He noted that it included several conservative features, supplemented by potential "refinements" that could be applied by individual licensees, and also included a risk-informed proposal to treat breaks above a particular size using assumptions about equipment operation and operator actions that would not normally be allowed in a strict design-basis method. He commented that it appeared that the staff would need to perform a large number of plant-specific reviews of the implementation of this methodology, and the ACRS would consider it in more detail after the staff has completed its evaluation.

9. ACRS Plant and Region Visit

The ACRS Plant Operations Subcommittee members and staff visited the Donald C. Cook Nuclear Plant on June 9, 2004 and Region III on June 10, 2004.

The annual plant visits provide the members an opportunity to tour the facility and hear presentations from plant management and staff to gain first hand knowledge of plant operations and issues. In addition to the plant tour, topics discussed were the D.C. Cook Improvement Plan, reactor vessel head inspection, debris multi-disc screens, digital turbine controls, and PRA model improvements.

The Plant Operations Subcommittee meetings with the Regions are also held annually to allow ACRS members to gain valuable information regarding issues related to and the status of the plants in the region. Some topics discussed at the meeting were organization and challenges, region plant performance (inspections, performance indicators, cross cutting issues, etc.), materials issues (reactor head inspections, pressurizer issues, and power uprates/license renewal/steam dryer issues), the reactor oversight process (SRAs and regional/resident inspectors), and fire protection.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of June 7, 2004, to conclusions and recommendations included in the ACRS report dated April 27, 2004, concerning the draft plan for implementing the Commission's phased approach to probabilistic risk assessment (PRA) quality.

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to review the draft NUREG document that provides guidance for performing bounding, sensitivity, and uncertainty analyses as described in the staff's plan for implementation of the Commission's phased approach to PRA quality.

- The Committee considered the EDO's response of June 17, 2004, to observations and recommendations included in the ACRS report dated April 22, 2004, concerning Options and Recommendations for Policy Issues Related to Licensing Non-Light Water Reactor Designs.

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to hold further discussions with the staff after the staff has developed its positions, including how the staff has included the ACRS views and issues in its evaluation of the treatment of integrated risk.

- The Committee considered the EDO's response of June 18, 2004, to conclusions and recommendations included in the ACRS letter of May 13, 2004, concerning Good Practices for Implementing Human Reliability Analysis.

  The Committee decided that it was satisfied with the EDO's response, although the staff did not commit to a peer review by domestic and international experts as recommended by the ACRS. The staff plans to brief the Committee on human reliability analysis good practices document in fall 2004, before the final report is issued.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from June 2, 2004, through July 6, 2004, the following Subcommittee meetings were held:

- Plant Operations Subcommittee - June 10, 2004

The Subcommittee held discussions with representatives of NRC Region III staff regarding matters related to regional operations.

- Thermal-Hydraulic Phenomena Subcommittee - June 22-23, 2004

The Subcommittee discussed the ongoing staff review associated with GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." Representatives from NEI presented a description of their guidelines for use by licensees. The staff presented their initial assessment of guidelines, and the results of the public comments on the draft generic letter regarding PWR sump blockage. RES provided initial results of experiment al programs to investigate chemical phenomena in PWR sumps.

- Future Plant Designs Subcommittee - June 24, 2004

The Subcommittee reviewed and discussed the NRC staff's proposed technology-neutral framework document for future plant licensing.

- Future Plant Designs Subcommittee - June 25, 2004

The Subcommittee reviewed the AP1000 Final Safety Evaluation Report (FSER) and the resolution of any remaining open items and ACRS concerns.

- Planning and Procedures - July 6, 2004

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 and discuss Options and Recommendations for Policy Issues Related to Licensing Non-Light Water Reactor Designs once the NRC staff has developed its positions.

- The Committee plans to review the proposed regulatory structure for new plant licensing technology-neutral framework document, once it is completed, during future meetings.

- The Committee plans to write a lessons learned letter, as a result of the review of the AP1000 design, during the October 2004 ACRS meeting.

- The Committee plans to review the draft final report on Good Practices for Implementing Human Reliability Analysis in the fall 2004.

- The Committee decided to postpone its site visit to the Chalk River facility used for the ACR-700 design. The Committee, however, plans to review the pre-application documents during the October 2004 ACRS meeting.

- The Committee plans to review the proposed rule to risk-inform the requirements addressing large break loss-of-coolant accidents in November 2004.

- The Committee plans to review the draft NUREG document that provides guidance for performing bounding, sensitivity, and uncertainty analyses as described in the staff's plan for implementation of the Commission's phased approach to PRA quality.

- The Committee plans to meet with the staff and the Industry to discuss staff and industry activities associated with the resolution of steam dryer cracking events.

- The Committee plans to review the draft final Regulatory Guide, DG-1128, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," (Revision 4 to Regulatory Guide 1.97) after reconciliation of public comments.

PROPOSED SCHEDULE FOR THE 515th ACRS MEETING

The Committee agreed to consider the following topics during the 515th ACRS meeting to be held on September 7-9, 2004:

- Final Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Plants
- Proposed Changes to the License Renewal Program
- Trip Report - AP1000 Workshop in China
- Trip Report - Chalk River Facility in Canada
- Safeguards and Security Matters
- Assessment of the Quality of the Selected NRC Research Projects
- Divergence in Regulatory Approaches Between U.S. and Other Countries

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

CERTIFIED

Date Issued: 8/9/2004
Date Certified: 8/23/2004

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REPORT:

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APPENDICES

I. *Federal Register Notice*
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MINUTES OF THE 514th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
July 7-9, 2004
ROCKVILLE, MARYLAND

The 514th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on July 7-9, 2004. Notice of this meeting was published in the *Federal Register* on June 22, 2004 (65 FR 34694) (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), Mr. Stephen L. Rosen, (Member-at-Large), Dr. F. Peter Ford, Dr. Thomas S. Kress, Dr. Dana A. Powers, Dr. Victor H. Ransom, Dr. William J. Shack, and Mr. John D. Sieber. Dr. George E. Apostolakis and Mr. Graham M. Leitch did not attend this meeting. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

Dr. 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. Final Safety Evaluation Report (SER) Associated with the AP1000 Design Certification (Open)

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

Dr. Thomas Kress, Future Plant Designs Subcommittee Chairman, stated that the purpose of this session was to hear a presentation by representatives of the NRC staff and Westinghouse regarding the safety aspects of the Westinghouse Electric Company application for certification of the AP1000 passive plant design.

Westinghouse submitted its application in accordance with Subpart B, "Standard Design Certification" of 10 CFR Part 52. The application consists of the AP1000 design control document (DCD), and the probabilistic risk assessment. Westinghouse originally submitted the DCD on March 28, 2002. The DCD Tier 1 information contains inspection, tests, analyses, and acceptance criteria, and Tier 2 information describes the AP1000 design.

Mr. J. Segala, Office of Nuclear Reactor Regulations (NRR), stated that the AP1000 design is similar in concept to the AP600 design, but provides higher power levels. Many systems and components were increased in size and/or capacity over those of AP600 to accommodate the higher power. Some of the AP1000 design enhancement features included an improved reactor core design, a large reactor vessel, a large pressurizer, an in-containment refueling water storage tank, an automatic depressurization system, a digital microprocessor-based instrumentation control system, hermetically sealed canned motor coolant pumps, and increased battery capacity.

The NRC staff and Westinghouse agreed to a three-phased approach to the AP1000 standard plant design review. Phase 1, which identified the key review issues was completed and resulted in the review of four key issues:

- Acceptability of the proposed use of design acceptance criteria (DAC) for particular parts of the design review.
- Acceptability of certain exemptions that Westinghouse intends to request.
- Acceptability of the AP600 test program to the AP1000 design.
- Applicability of the AP600 analyses codes to the AP1000 design.

The purpose of the Phase-2 review was for the staff to develop positions on the above four key issues. The staff has completed the Phase-2 review. On March 14, 2002, the ACRS issued a report to the NRC Chairman, stating that the staff has made a competent and thorough review of the Phase-2 issues. The ACRS also agreed that the proposal by Westinghouse to use DAC for the piping design should be approved.

Westinghouse, in its application, requested exemptions from the regulations in three areas:

(a) Section 50.34 (f)(2)(iv) requires a "safety parameter display console that will display to operators a minimum set of parameters defining the safety status... displaying a full range of important plant parameters..., and capable of indicating when process limits are being approached or exceeded."

(b) Section 50.62(c)(1) requires that equipment be available to ensure the automatic startup of the auxiliary feedwater system under ATWS conditions.

(c) GDC 17 of 10 CFR Part 50 Appendix A requires two physically independent offsite power sources.

The staff granted Westinghouse the above three exemptions. The ACRS agreed with the staff's positions that item (a) will be part of the DAC for control room design; item (b) is satisfied because the AP1000 does not have an auxiliary feedwater system as the emergency core cooling system requirement and is met by the passive residual heat removal system automatic initiation under ATWS; and, item (c) is satisfied because with the passive emergency core cooling system, the AP1000 does not need offsite power to make its safety case.

The NRC staff has reviewed the design certification application and issued its draft safety evaluation report (DSER) on June 16, 2003. The DSER originally contained 174 open items.

During the 510th ACRS meeting (March 3-6, 2004) the Committee met with the NRC staff and Westinghouse representatives and discussed the status of the open items as well as issues previously raised by the ACRS. The Committee reviews have not addressed security matters and their impacts on the AP1000 design.

On March 17, 2004, the Committee issued a letter to Dr. William D. Travers, former Executive Director for Operations (EDO). The Committee outlined seven technical issues in which the ACRS have comments related to the certification of the AP1000 design. These seven issues are:

Issue 1- Automatic Depressurization System (ADS)-4 Squib Valve Function.
Issue 2- Assurance of Long-Term Cooling (Strainer Blockage).
Issue 3- Code Deficiencies.
Issue 4- Range of Pi-Group Values.
Issue 5- In-Vessel Retention/Fuel-Coolant Interactions (FCI).
Issue 6- Organic Iodine Production.
Issue 7- Catastrophic Failure of a free-Standing Steel Containment.

On May 18, 2004, the NRC staff responded to the above 7 issues.

For Issue 1-- the staff indicated that the AP1000 DCD included an ITAAC that ensures the ADS-4 squib valves will perform their function. Tests or type tests will be performed to demonstrate the capability of the ADS-4 squib valves to operate under their design conditions. The staff concludes that the performance characteristics of the ADS-4 squib valves will be adequately verified.

For Issue 2-- the staff stated that Westinghouse revised the COL action item in the DCD to include the evaluation of chemical debris. This COL action item will capture any impact of chemical effects on the ability of the affected components to accommodate anticipated debris loadings identified during the resolution of GSI-191. The staff concludes that Westinghouse has resolved the concern related to additional debris that can be caused by chemical reactions in the containment.

For Issue 3-- the staff indicated that for the AP1000, an acceptable solution was to conservatively bound the calculations. The TRACE code was not used in the AP1000 review, but is currently being assessed using APEX-Ap1000 test data. Currently, there are two ongoing experimental studies to help correct code deficiencies. New thermal-hydraulic models will be developed and implemented into TRACE as deficiencies are identified. The experimental programs, along with the associated TRACE assessment, will be discussed as a generic issue at a future thermal-hydraulic Subcommittee meeting.

For Issue 4-- the staff indicated that an acceptance range of 0.5 to 2.0 for various Pi-groups was selected. The staff noted that the appropriate Pi-groups is generic and does not represent an issue that is specific only to AP1000 designs. The staff, however, as a long term effort and on a generic basis, will work to develop and document a procedure to define an appropriate Pi-group range for scaling integral test facilities.

Materials-- the ACRS commented that ongoing and future studies may suggest material and environmental changes that will be addressed at the COL stage. The staff notes that any future and environmental changes that would be requested at the COL stage would have to follow the change processes set forth in paragraph VIII.C of the DCD.

Severe Accidents-- the ACRS commented and questioned the technical justification for the aerosol removal coefficient (lambda) for containment. The staff performed an independent dose analysis with the median aerosol removal coefficient values from the staff's uncertainty analysis. Along with other analysis parameters and the bounding hypothetical atmospheric dispersion factors provided by Westinghouse, the results are within the dose criteria of 10 CFR 50.34 and General Design Criteria 19. The staff concluded that, while the staff and Westinghouse diverge on values for the intermediate steps in the dose calculations, the overall conclusion that the AP1000 dose results are acceptable.

For Issue 5-- the staff notes that it has performed a reasonably large number of sensitivity analysis and found that the ex-vessel FCI for the AP1000 is of no greater concern than that for the AP600 design. The ex-vessel FCI analysis for the AP600 indicated that the containment integrity would not by challenged by the FCI load.

For Issue 6-- Westinghouse plans to provide the staff with additional information regarding the pH of the water film on the inside of the containment wall where acidification could produce organic iodine.

For Issue 7-- the staff stated that even if an event progressed to an intermediate or late release, it would likely involve a vented release rather than a catastrophic containment failure, since the AP1000 design will include the capability to vent the containment. The non-safety related containment spray function could also be effective in reducing re-suspended fission products following containment failure.

On June 25, 2004, the Future Plant Designs Subcommittee held a meeting with the NRC staff and Westinghouse representatives regarding the above seven issues. The staff provided additional discussion regarding Issue# 6, "Organic Iodine" production. The water film pH determines the iodine behavior. For pH value less than 7, this will lead to production of elemental iodine, some of which is subsequently converted into organic iodine. To prevent organic iodine production, the film pH should be maintained above 7. Westinghouse's calculations determined that the film pH is maintained above 7, assuming the amount of CsOH present in the design basis accident source term. The staff found Westinghouse's calculations to be acceptable. Currently, the staff

determined that all DSER open items have been resolved, and the staff is on schedule to issue the FSER by September 13, 2004.

Westinghouse representatives stated that the AP1000 is smaller and dramatically simpler than evolutionary plants. Some of the improved general arrangements included improved separation (e.g., fire areas inside containment); improved inspection programs and maintenance; and improved access to containment (e.g., equipment hatches access from auxiliary building). The AP1000 also has improved construction such as major reduction in bulk materials and field labor. A construction schedule of 36 months is anticipated. Westinghouse representatives stated that the AP1000 design comfortably meets the NRC and industry safety standards for future plants for both deterministic and probabilistic reviews. The AP1000 final design approval will enable the next steps to realizing new plant construction as proposed by Nu-Start Energy consortium.

The ACRS viewed the AP1000 design in concert with all the ACRS review activities conducted for certification of the AP600 design. The ACRS reviewed the changes from the AP600 design made to accommodate the increased power level and ensured that such changes did not pose any new safety considerations or result in an unacceptable increase in risk. The new phenomena identification and ranking table was also reviewed to determine if any new phenomena were identified and that there were no significant changes in the ranking of events. The ACRS reviews did not address security related issues.

The ACRS agreed with the resolutions proposed by the staff of all but two of these issues. For the in-vessel retention and organic iodine production, the ACRS developed its own arguments for the resolution. All ACRS issues have been resolved. The Committee also discussed concerns expressed by a member of the public. Most of these concerns were process-related and are within the purview of the staff.

Committee Action

The Committee issued a report to NRC Chairman Nails J. Diaz on this matter dated July 20, 2004. The Committee, in its report, concluded that the AP1000 design is robust and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public.

III. Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs (Open)

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

The Committee heard presentations from the NRC staff and from the Nuclear Energy Institute (NEI) concerning the proposed generic letter related to the potential impact of debris blockage on emergency recirculation during design basis accidents at pressurized water reactors (PWR). The staff presented a version of the generic letter (GL) to the Thermal-Hydraulic Subcommittee on June 22-23, 2004, together with its resolution of the public comments from various stakeholders. This version of the proposed GL was "action-oriented," and directed licensees to perform analyses and take corrective action to resolve identified discrepancies, in accordance with the schedule that was approved by the Commission for resolution of this generic safety issue. At the Committee meeting on July 7, the staff provided the Committee with a different version of the GL, which removed many of the action-oriented requirements, and returned the letter to the format that was originally issued for public comment. The staff explained that the Office of the General Counsel (OGC) could not accept the "action-oriented" provisions of the letter because they imposed new requirements on licensees, and that a GL could not be used for this purpose. Therefore, the version that was discussed on July 7 was more of a request for information from licensees, to allow the staff to determine whether they were in compliance with 10 CFR 50.46.

Subsequently, on July 9, the staff returned to the meeting, and informed the Committee that its discussions with OGC had continued, and it appeared that it might be able to return some of the "action-oriented" provisions to the GL. The staff could not make a commitment that this would actually occur. The staff left the Committee with the impression that it would continue to work with OGC to develop a GL that would be as "action-oriented" as possible (given the legal constraints of the GL process). The Committee also understood that the CRGR would also review the proposed GL and would ensure that it complied with the process requirements for generic communications.

During the discussion of the GL, the Committee questioned the staff about the need and utility of issuing the GL before the review of the industry guidance document is complete. Because the GL references the guidance document, and is intended to be used to ensure its implementation, the Committee did not understand why the staff's safety evaluation report and the GL should not be issued simultaneously. The staff

explained that it wanted the industry to see the GL as soon as possible, and there was no reason to delay its issuance until the guidance had been reviewed.

Mr. Pietrangelo (NEI) commented on July 7 that he had not seen the version of the GL that the Committee was considering, but from the discussion it appeared that it was very similar to the version that had been issued for public comment. He urged the staff not to issue the GL in that form, and he noted that the industry comments consistently asked the staff to issue the GL in a form that is more "action-oriented," and acknowledges that the position the staff is taking is a backfit. He explained that the industry is fully prepared to make any necessary modifications, but they would prefer a process that followed the one used to resolve this issue for boiling water reactors (BWR), rather than the one proposed in the original draft GL.

Committee Action

The Committee agreed that it should limit its inquiries to technical issues, and inasmuch as the GL is primarily a process document. The Committee focused its advice on recommending that a GL be issued with the format and process to be defined by the staff. The Committee will consider the technical issues associated with the industry guidance document at an August subcommittee meeting, and at the September 2003 full Committee meeting.

IV. Risk-Informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" (Open)

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

Dr. Bill Shack, the cognizant Committee member for this issue, introduced the topic. Dr. Shack stated that the Committee would be briefed on the NRC staff's approach for responding to the Commission's SRM, dated July 1, 2004, concerning issues related to proposed rulemaking to risk-inform requirements related to large break loss-of-coolant accident.

NRC Staff Presentation

Dr. Sheron, NRR, provided background on the NRC staff's efforts to develop a risk-informed alternative break size. Dr. Sheron said that he was heading an interoffice review group to provide more focus and accountability to the effort. Dr. Sheron stated that the NRC staff issued a *Federal Register Notice* in July which provided a conceptual outline and summary description for the rule.

Dr. Sheron said that one of the first tasks was to determine if there was an adequate basis for establishing revised design basis break sizes for both PWRs and BWRs. Dr. Sheron said that break sizes less than the revised break size would be treated the same. Breaks greater than the transition break size would not have to use an evaluation model with all the conservatisms. One could use the best estimate model but they would not necessarily have to consider 95 percent certainty. Dr. Sheron said that when the steering group began they laid out some fundamental ground rules. The main premise was to meet the Commission's schedule and base their decisions on information available. Dr. Sheron said this was an enabling rule. Some licensees may choose to uprate power. Others may choose to increase peaking factors or change allowed outage times for equipment. Licensees will have to demonstrate that they meet the guidelines of 1.174 with regard to delta CDF and delta LERF.

Dr. Sheron then said that the Commission's previous SRM talked about updating the LOCA frequency every ten years. This requirement has been removed. The staff agrees with this position because of the staff's plans to continuously monitor the data; and, if there is new data that comes later in two years, then the staff is not going to wait. Dr. Sheron concluded by going over point-by-point the Committee's report to the Commission on SECY 04-0037. Lee Abramson, RES, then gave an update of RES' expert elicitation findings and the sensitivity analyses that RES has done. Mr. Gary Hammer then discussed how NRR intends to use the results in the break size selection. Mr. Abramson then discussed what remained to be done.

During the above discussions, the NRC staff and the ACRS Members made the following points:

- Dr. Shack asked what was the criteria if it's not frequency-based for choosing the maximum break size. Mr. Sheron said one might pick a probability of a frequency that corresponds to a break of X inches in diameter. Then one asks what things were not considered (e.g., heavy load and seismic). He said another approach might be what are the largest pipes below the main coolant pipes, which for PWRs is the pressurizer surge line.

- Dr. Wallis asked what was meant by mitigative capability. Dr. Sheron responded that one definition being considered was maintaining a coolable geometry. Right now the staff would say that in the absence of any additional data 2,200 degrees and 17 percent oxidation is sufficient to demonstrate coolable geometry. That does not mean that other criteria could not be used to demonstrate coolable geometry. He wanted to consider more recent work.

- Dr. Wallis asked how can GSI-191 be resolved as a 50.46 compliance issue when you are changing 50.46. Dr. Sheron said that he asked his staff the same question. He stated that the break size that the staff would let the industry choose would have to be bounded by those being considered for risk-informing 50.46. Dr. Sheron acknowledged that there is a slight potential for inconsistency but he said that NRC management is well aware of both efforts.

- Dr. Ford asked whether you can correlate qualitative rationale with a quantitative estimate. Dr. Chokshi responded that the base case was an example. By using two different probabilistic fracture mechanic models to confirm two different looks at the service-based experience. These estimates provided a lot of insights. As far as the qualitative rationale the panelists were asked why it was like this as compared to the quantitative base case. How does this compare with another judgment made, what is driving this, and what do is important about this.

Committee Action:

This was an information briefing. A Subcommittee will review the staff's conceptual frame work and proposed rule language on October 28, 2004, and the full Committee will review and comment upon the proposed rule language during its November meeting.

V. Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open)

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

In an April 28, 2003, SRM on the April 11, 2003 meeting with the ACRS, the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements, the Commission should be informed." Dr. Nourbakhsh, ACRS Senior Staff Engineer, prepared a draft of a white paper that will be used by the Committee in responding to the Commission. The Committee was briefed by Dr. Nourbakhsh regarding the draft white paper on differences in regulatory approaches and requirements between U.S. and other countries.

Committee Action

The Committee plans to discuss the draft final of the white paper on differences in regulatory approaches and requirements between U.S. and other countries during the September 2004 ACRS meeting.

VI. Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants (Open)

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

The Committee met with representatives of NRR to discuss the staff's proposal to issue generic communication regarding the use of Ultrasonic Flow Measurement (UFM) systems for measuring feedwater flow rates.

Evangelos Marinos, NRR began the discussion. He described the key points of his presentation, which included describing the basic principles of UFM technology, their application in nuclear power plants, and recent thermal overpower events caused by miscalculations of reactor thermal power based on non-conservative UFM system outputs. He noted that the uncertainty in reactor thermal power calculations is significantly influenced by the uncertainty in measuring steam generator feedwater flow.

Mr. Marinos described the two types of UFM devices that are currently in use in U.S. reactor plants. The Leading Edge Flow Meter (LEFM) instruments developed by Caldon, Inc. use transit time technology to calculate fluid flow by measuring the time difference between sonic waves traveling in opposite directions through the piping systems. Both clamp-on and in-line type LEFMs are available. The second type of measurement system uses cross correlation technology to calculate fluid flow rates. The flow rate is calculated by measuring the time difference between ultrasonic signals modulated by eddies in the fluid. This technology was developed by Westinghouse Electric Company LLC/Advanced Measurement and Analysis Group, Inc. (W/AMAG) and is currently available in clamp-on type units only.

Mr. Marinos also described several events in which licensed thermal power was exceeded due to inaccurate reactor thermal power calculations. These incidents involved instruments using both types of UFM technologies. On February 2, 2004, the NRC formed a task group to address concerns regarding the accuracy of UFM devices. The task force completed their review and issued a report on June 7, 2004. Based on

recommendations from the task force, the staff has proposed generic correspondence that will (1) advise addressees of operating experience with UFM devices that have not provided the intended accuracy needed to maintain plant operation within licensed thermal power, (2) advise addressees that there are potential questions regarding the use of UFM devices because of sensitivities to plant configuration and a lack of data to support instrument performance, (3) recommend licensees confirm UFM accuracy by comparisons with standard tests of known accuracy, and (4) require addressees to provide a written response that verifies these actions have been taken to ensure plants are not operated above licensed thermal power or outside the licensed design basis.

Jose Calvo, NRR, recommended another approach for addressing this issue. Mr. Calvo stated that since inaccuracies noted in UFM devices are not considered safety significant nor a generic issue, the staff should issue an information notice or a regulatory information summary instead of a bulletin. UFM devices provide input to a process computer which calculates thermal power. Plant operators independently verify these calculations against other secondary plant parameters before making any changes to the power level. Mr. Sieber stated that these secondary parameters are subject to many external variables and cannot provide accurate information regarding thermal power.

Cal Hastings, CEO and President of Caldon, read prepared notes which contended that even though the previously discussed overpower incidents may not create a significant risk to public health and safety, they are still serious events because they violate regulations. Also, power uprates that credit reduced feedwater flow measurement uncertainty can be achieved without violating these regulations. Mr. Hastings noted that Caldon believes that the use of CrossFlow instruments developed by W/AMAG have a measurement uncertainty that may be as high as 3% because of its sensitivity to velocity profile effects.

John McInerney, W/AMAG, followed Mr. Hastings and reassured the ACRS that Westinghouse stands behind the cross flow technology and the integrity of the system and that Westinghouse chooses not to comment on the capability of their competitors relative to any type of nuclear component or service.

Christopher Grimes, NRR, explained to the Committee that the observed performance of the UFM systems is not a public safety issue because the uncertainties in the accuracy of UFM devices are small compared to existing plant safety margins. However, there is an issue with public confidence when UFM devices are improperly calibrated causing plants to exceed their licensed thermal power levels.

Committee Action

The Committee has deferred action on this issue pending review of previously undisclosed documents and additional discussions with the staff and industry.

VII. Status of the ACRS Members' Assessment of the Quality of Selected NRC Research Projects (Open)

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

RES is required to have an independent evaluation of the quality of its research programs. This evaluation is mandated by the Government Performance and Results Act (GPRA) and needs to be in place during the next fiscal year. The Committee has agreed to assist RES in assessing the effectiveness and utility of the NRC research programs. The Committee has previously approved the strategy for the review of the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During this meeting, the Committee discussed the status of the activities of cognizant ACRS members associated with the assessment of the quality of the research projects on Sump Blockage and on MACCS Code.

Committee Action

The Committee plans to discuss the preliminary assessment of the quality of the research projects on Sump Blockage and MACCS Code during the September 2004 ACRS meeting.

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/EDO Commitments

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

The Committee discussed the response from the NRC Executive Director for Operations (EDO) to ACRS comments and recommendations included in recent ACRS reports:

- The Committee considered the EDO's response of June 7, 2004, to conclusions and recommendations included in the ACRS report dated April 27, 2004, concerning the draft plan for implementing the Commission's phased approach to probabilistic risk assessment (PRA) quality.

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to review the draft NUREG document that provides guidance for performing bounding, sensitivity, and uncertainty analyses as

described in the staff's plan for implementation of the Commission's phased approach to PRA quality.

- The Committee considered the EDO's response of June 17, 2004, to observations and recommendations included in the ACRS report dated April 22, 2004, concerning Options and Recommendations for Policy Issues Related to Licensing Non-Light Water Reactor Designs.

  The Committee decided that it was satisfied with the EDO's response. The Committee plans to hold further discussions with the staff after the staff has developed its positions, including how the staff has included the ACRS views and issues in its evaluation of the treatment of integrated risk.

- The Committee considered the EDO's response of June 18, 2004, to conclusions and recommendations included in the ACRS letter of May 13, 2004, concerning Good Practices for Implementing Human Reliability Analysis.

  The Committee decided that it was satisfied with the EDO's response, although the staff did not commit to a peer review by domestic and international experts as recommended by the ACRS. The staff plans to brief the Committee on human reliability analysis good practices document in fall 2004, before the final report is issued.

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 July 6, 2004. The following items were discussed:

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

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

- Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through October 2004 were addressed. 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

- Staff Requirements Memorandum Resulting from the ACRS Meeting with the NRC Commissioners

In a SRM dated June 30, 2004, which resulted from the ACRS meeting with the Commissioners in June, the Commission stated the following:

- The ACRS should work with the staff to resolve outstanding issues with respect to PWR Sump Performance, and make a recommendation for a practical solution within a reasonable period of time. Both the ACRS and the staff should focus their attention, resources, and additional research, if needed, on evaluating realistic scenarios rather than all possible scenarios.

- The Commission commends the ACRS' efforts in the security area. The Commission noted that the ACRS has closely followed the Commission's guidance contained in an October 31, 2003 SRM and that those provisions remain in place.

- The staff and the ACRS should engage with the Commission prior to making arrangements similar to the one recently agreed to between RES and the ACRS, in which the ACRS will assess the quality of NRC research projects. This interaction should focus on the benefits and resource implications of such work and consistency of the proposals with previous Commission direction.

- Operating Events

In an e-mail dated June 24, 2004, Mr. Leitch pointed out the recent reactor scrams that occurred at Dresden 3, Palo Verde 1, 2, and 3, Vermont Yankee, and Limerick 2 nuclear power plants. An Augmented Inspection Team (AIT) had been sent to investigate the event at Palo Verde. He suggested that the Committee hear a presentation from the staff on the results of the AIT investigation of the Palo Verde

event as well as the staff's views on the effectiveness of the corrective actions taken by the industry in response to similar events that occurred last year.

Dr. Powers agreed with Mr. Leitch's suggestion and added that the Committee should also hear a presentation from the staff on the general issue of grid reliability.

In an e-mail dated June 29, 2004, Mr. Sieber stated the following:

- Agreed with the issue raised by Mr. Leitch and his conclusions.
- Stated that deregulation has a cut-investment in transmission and substation capability in a dramatic fashion.
- Lower investment in power plants and transmission and distribution capabilities means more risk of power failures and more risk of nuclear plants suffering loss-of-offsite power events.
- Discussed the issues raised by Mr. Leitch and Dr. Powers, and that the Committee should discuss the issue of increasing loss-of-offsite power probability.

In response to Mr. Sieber's e-mail, Dr. Powers suggests (pp. 16-17) that the Planning and Procedures Subcommittee establish an ad hoc Subcommittee, Chaired by Mr. Sieber, to discuss the issues noted above with the staff and develop a report for consideration by the Committee.

- Tour of the Chalk River Facility Used for the ACR-700 Design

The tour of the Chalk River Facility in Canada and a joint meeting of the ACRS Subcommittees on Future Plant Designs and on Materials and Metallurgy are scheduled for September 27-30, 2004. Drs. Ford (tentative), Kress, Ransom, and Wallis have expressed interest in touring the Chalk River Facility and attending the meeting. Since AECL and the NRC staff are investing enormous time and efforts in arranging the tour and preparing for the subcommittee meeting, more members should participate. Unless a majority of the members participate, the Committee should consider canceling the tour and the meeting.

- Workshop on the AP1000 Design

An NRC staff delegation (RES, NRR, and the Chairman's Office) headed by Mr. Ashok Thadani, will be participating in an international workshop regarding the AP1000 design in China on July 26-29, 2004. Dr. Kress was invited to join the NRC panel and participate in such workshop. The Department of State is supporting this meeting. The Committee approved Dr. Kress' participation in this workshop.

- Appointment of New ACRS Member (Closed)

The Commission has approved the appointment of Dr. Richard S. Denning to the ACRS. Subsequent to completing the security clearance process and resolving conflict-of-interest issues, if any, he will be officially appointed to the Committee.

- Safeguards and Security Matters (Closed)

The Subcommittee discussed the Commission's views in an SRM dated June 30, 2004, and other issues, related to safeguards and security.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 515th ACRS Meeting, September 8-11, 2004.

The 514th ACRS meeting was adjourned at 11:35 a.m. on July 9, 2004.

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

August 23, 2004

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

FROM: Mario V. Bonaca
Chairman

SUBJECT: CERTIFIED MINUTES OF THE 514th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JULY 7-9, 2004

I certify that based on my review of the minutes from the 514th 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

August 9, 2004

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 514th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - JULY 7-9, 2004

Enclosed are the proposed minutes of the 514th 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

3. *How often the collection is required:* Applications for new licenses and amendments may be submitted at any time. Applications for renewal are submitted every 10 years. Reports are submitted as events occur.

4. *Who is required or asked to report:* Applicants for and holders of specific licenses authorizing the use of licensed radioactive material for radiography.

5. *The number of annual respondents:* 282 (62 NRC licensees and 220 Agreement State licensees).

6. *The number of hours needed annually to complete the requirement or request:* 244,048 hours. The NRC licensees total burden is 48,335 hours (85 reporting hrs [an average of 1.3 hours per response] plus 48,250 recordkeeping hours [an average of 383 hours per recordkeeper]). The Agreement State licensees total burden is 195,713 hours (299 reporting hours [an average of 1 hour per response] plus 195,414 recordkeeping hours [an average of 430 hours per recordkeeper]).

7. *Abstract:* 10 CFR part 34 establishes radiation safety requirements for the use of radioactive material in industrial radiography. The information in the applications, reports and records is used by the NRC staff to ensure that the health and safety of the public is protected and that licensee possession and use of source and byproduct material is in compliance with license and regulatory requirements.

Submit, by August 23, 2004, comments that address the following questions:

1. Is the proposed collection of information necessary for the NRC to properly perform its functions? Does the information have practical utility?

2. Is the burden estimate accurate?

3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4. How can the burden of the information collection be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the draft supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O–1 F21, Rockville, MD 20852. OMB clearance requests are available at the NRC World Wide Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html.* The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Comments and questions about the information collection requirements may be directed to the NRC Clearance Officer, Brenda Jo. Shelton (T–5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, by telephone at 301–415–7233, or by Internet electronic mail to *infocollects@nrc.gov.*

Dated at Rockville, Maryland, this 16th day of June, 2004.

For the Nuclear Regulatory Commission.

Brenda Jo. Shelton,

NRC Clearance Officer, Office of the Chief Information Officer.

[FR Doc. 04–14036 Filed 6–21–04; 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 July 6, 2004, 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 the internal personnel rules and practices of the 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:

Tuesday, July 6, 2004—1:30 p.m.—3:30 p.m.

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. (e.t.) 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.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (e.t.). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: June 16, 2004.

Ralph Caruso,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 04–14037 Filed 6–21–04; 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 July 7–9, 2004, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 21, 2003 (68 FR 65743).

Wednesday, July 7, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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.: Final Safety Evaluation Report (SER) Associated With the AP1000 Design Certification (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Westinghouse Electric Company regarding the final SER associated with the certification of the AP1000 design, resolution of any unresolved issues previously raised by the ACRS, and related matters.

10:45 a.m.–12: 15 p.m.: Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors (PWRs) (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft final Generic Letter on PWR sump blockage and the staff's resolution of public comments on the proposed version of this Generic Letter.

1:15 p.m.–3:45 p.m.: Risk-Informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed rule language for risk-informing 10 CFR 50.46 and the

sensitivity studies on large-break loss-of-coolant accident frequency reevaluation performed in support of risk-informing 10 CFR 50.46.

4 p.m.–5 p.m.: Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open)—The Committee will hear a presentation by and hold discussions with Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding his draft White Paper on differences in regulatory approaches and requirements between U.S. and other countries.

5:15 p.m.–6:45 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Thursday, July 8, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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.: Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants.

10:45 a.m.–11:45 a.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.

11:45 a.m.–12 noon: 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.

1 p.m.–2 p.m.: Status of the ACRS Members' Assessment of the Quality of Selected NRC Research Projects (Open)—The Committee will discuss the status of the activities of the cognizant ACRS members associated with the assessment of the quality of the research projects on Sump Blockage and on MACCS Code.

2:15 p.m.–6:30 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Friday, July 9, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–3 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

3 p.m.–3:30 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 16, 2003 (68 FR 59644). 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 Cognizant ACRS staff 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 Cognizant ACRS staff 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 Cognizant ACRS staff 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 Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), 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: June 16, 2004.

Kenneth R. Hart,

Acting Secretary of the Commission.

[FR Doc. 04–14035 Filed 6–21–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY: Nuclear Regulatory Commission.

DATES: Weeks of June 21, 28, July 5, 12, 19, 26, 2004.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of June 21, 2004

There are no meetings scheduled for the Week of June 21, 2004.

Week of June 28, 2004—Tentative

There are no meetings scheduled for the Week of June 28, 2004.

Week of July 5, 2004—Tentative

There are no meetings scheduled for the Week of July 5, 2004.

Week of July 12, 2004—Tentative

Tuesday, July 13, 2004

2:15 p.m. Discussion of Security Issues (Closed—Ex. (1)

Week of July 19, 2004—Tentative

Wednesday, July 21, 2004

9:30 a.m. Meeting with Advisory Committee on Nuclear Waste (ACNW)

APPENDIX II

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 17, 2004 (REVISED)

SCHEDULE AND OUTLINE FOR DISCUSSION
514th ACRS MEETING
JULY 7-9, 2004

WEDNESDAY, JULY 7, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)
   - 1.1) Opening Statement
   - 1.2) Items of current interest

2) 8:35 - ~~10:30~~ 10:55 A.M. — Final Safety Evaluation Report (SER) Associated with the AP1000 Design Certification (Open) (TSK/MME)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and Westinghouse Electric Company regarding the final SER associated with the certification of the AP1000 design, resolution of any unresolved issues previously raised by the ACRS, and related matters.

7/8/04
11:10-11:45 am

~~10:30 - 10:45 A.M.~~ 10:55 - 11:15 — ***BREAK***

3) ~~10:45~~ 11:15 - ~~12:15~~ 12:50 P.M. — Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs (Open) (GBW/RC)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final Generic Letter on PWR sump blockage and the staff's resolution of public comments on the proposed version of this Generic Letter.

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

~~12:15 - 1:15 P.M.~~ 12:50 - 1:52 — ***LUNCH***

4) ~~1:15~~ 1:52 - ~~3:45~~ 4:25 P.M. — Risk-Informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" (Open) (WJS/MRS)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the office of Nuclear Reactor Regulation regarding the proposed rule language for risk-informing 10 CFR 50.46.

4.3) Briefing by and discussions with representatives of the Office of Nuclear Regulatory Research regarding sensitivity studies on large-break loss-of-coolant accident frequency reevaluation performed in support of risk-informing 10 CFR 50.46.

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

4:25-4:45
~~3:45~~ - ~~4:00~~ P.M. *****BREAK*****

5) 7/8/04 ~~4:00~~ - ~~5:00~~ P.M. 12:50-2:00

Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open) (DAP/HPN/SD)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding his draft White Paper on differences in regulatory approaches and requirements between U.S. and other countries.

5:00 - 5:15 P.M. *****BREAK*****

6) 5:15 - ~~6:45~~ 6:10 P.M.

Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) AP1000 Design Certification (TSK/MME)

6.2) Proposed Rule Language for Risk-Informing 10 CFR 50.46 (WJS/MRS) (Tentative)

6.3) Draft Final Generic Letter on PWR Sump Blockage (GBW/RC)

THURSDAY, JULY 8, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) ~~8:30~~ - ~~8:35~~ 8:35-8:40 A.M.

Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)

8) ~~8:35~~ - ~~10:30~~ 8:40-10:50 A.M.

Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants (Open) (JDS/MWW/CS)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants.

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

10:50-11:10
~~10:30~~ - ~~10:45~~ A.M. *****BREAK*****

9) 2:00 - 3:25 ~~10:45~~ - ~~11:45~~ A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)

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

9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

9.3) Subcommittee Reports (Open)

a) Plant Operations (JDS/MWW)
Report by and discussions with the Chairman of the Plant Operations Subcommittee regarding the visit to the D.C. Cook Nuclear Plant and meeting with the NRC Region III personnel on June 9-10, 2004.

b) Thermal-Hydraulic Phenomena (GBW/RC)
Report by and discussions with the Chairman of the Thermal-Hydraulic Phenomena Subcommittee regarding the ongoing staff activities associated with the resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," that were discussed during the June 22-23, 2004 meeting.

c) Future Plant Designs (TSK/MME)
Report by and discussions with the Chairman of the Future Plant Designs Subcommittee regarding the NRC staff's proposed technology-neutral framework document for future plant licensing that was discussed at the June 24, 2004 meeting.

10) 11:45 - 11:50 ~~12:00~~ Noon 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.

11:50 - 12:50 ~~**12:00 - 1:00 P.M.**~~ ***LUNCH***

11) 3:40 - 4:10 ~~1:00~~ - ~~2:00~~ P.M. Status of the ACRS Members' Assessment of the Quality of Selected NRC Research Projects (Open) (SLR/TSK/RC/HPN)
Discussion of the status of the activities of the cognizant ACRS members associated with the assessment of the quality of the research projects on Sump Blockage and on MACCS Code.

3:25 - 3:40 ~~**2:00 - 2:15 P.M.**~~ ***BREAK***

12) 3:40 - 6:15 ~~2:15~~ - ~~6:30~~ P.M. Preparation of ACRS Reports (Open)
Discussion of the proposed ACRS reports on:

12.1) AP1000 Design Certification (TSK/MME)

12.2) Proposed Rule Language for Risk-Informing 10 CFR 50.46 (WJS/MRS) (Tentative)

12.3) Draft Final Generic Letter on PWR Sump Blockage (GBW/RC)
12.4) Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices (JDS/MWW/CS)

FRIDAY, JULY 9, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) ~~8:30~~ 10:00 - ~~3:00~~ 11:35 P.M. — Preparation of ACRS Reports (Open)
(12:00-1:00 P.M. LUNCH) — Continue discussion of proposed ACRS reports listed under Item 12.

14) 3:00 - 3:30 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.~~

Adjourn 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) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

514TH ACRS MEETING
July 7-9, 2004

NRC STAFF (July 7, 2004)

J. Segala, NRR
F. Gillespie, NRR
E. Throm, NRR
L. McQuinones-Navarro, NRR
J. Colaccino, NRR
M. Hart, NRR
J. Lyons, NRR
S. Bajorek, RES
A. Drozd, NRR
J. Wilson, NRR
J. Hannon, NRR
T. Hafeia, NRR
J. Lamb, NRR
S. Black, NRR
J. Uhle, NRR
M. Marshall, NRR
B. Kemper, OIG
B. P. Jain, RES
D. Cullison, NRR
D. Lew, RES
M. Kowal, NRR
K. Parczewski, NRR
L. Kerr, NRR
G. Menlinsky, NRR
A. Hiser, RES
D. Terao, NRR
J. Fair, NRR
G. Hammer, NRR
L. Abramson, RES
B. Thomas, NRR
N. Chokshi, RES
M. Tschiltz, NRR
C. Ader, RES
R. Mathew, NRR
J. Mitchell, RES
R. Architzel, NRR
S. Unikewicz, NRR
R. Taylor, NRR
J. Beall, OC/EM

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

E. Cummins, Westinghouse
R. Vijuk, Westinghouse
B. Hammersley, Westinghouse
T. Schulz, Westinghouse
C. Reid, Bechtel
P. Negus, GE
S. Traiforos, LINK
C. Brinkman, Westinghouse
J. Weil, McGraw-Hill
S. Sterrett, Duke University
Y. Rodriguez, FCT Consulting
T. Pietrangelo, NEI
M. Knapik, McGraw-Hill
A. Einham, ICF

NRC STAFF (July 8, 2004)

W. Lyon, NRR
J. Wermiel, NRR
C. Doutt, NRR
B. Macow, NRR
C. Grimes, NRR
S. Arndt, RES
E. Marinos, NRR
P. Redstock, NRR
J. Calvo, NRR
I. Ahmed, NRR
G. Dick, NRR
H. Li, NRR
A. Lee, NRR
H. Garg, NRR
B. Marcus, NRR
D. Thatchet, NRR
G. Cwalina, NRR
J. Page, RES
D. Kern, OEDO
M. Waterman, RES
W. Held, NMSS
V. Hall, OIP
J. Segale, NRR
L. Quinones-Navarro, NRR
J. Colaccino, NRR
J. Mitchell, RES
J. Beall, OC/EM

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

T. Long, Southern Nuclear
B. Turkowski, Westinghouse
J. McInerney, Westinghouse
C. French, Wesitnghouse
A. Lopez, AMAG
Y. Askari, AMAG
F. Bursic, AMAG
W. Slagle, Westinghouse
H. Thompson, Talis Mau
C. Hastings, Caldon
B. Horin, Winston & Strawn
P. Campbell, Winston & Strawn
H. Estrada, Caldon
P. Negus, GE
N. Chapman, SERCH/Bechtel
M. Knapik, McGraw-Hill
S. Traifores, LINK
R. Vijvk, Westinghouse
S. Sterrett, Duke University

NRC STAFF (July 9, 2004)

J. Hannon, NRR
R. Elliott, NRR

APPENDIX IV

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

August 23, 2004

REVISED

SCHEDULE AND OUTLINE FOR DISCUSSION
515th ACRS MEETING
SEPTEMBER 9-11, 2004

THURSDAY, SEPTEMBER 9, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD
   - 1.1) Opening Statement
   - 1.2) Items of current interest

2) 8:35 - 10:30 A.M. — Final Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Plants (Open) (MVB/MDS/CS)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the Exelon Generation Company, LLC and the NRC staff regarding the license renewal application for the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2, as well as the associated final Safety Evaluation Report prepared by the NRC staff.

10:30 - 10:45 A.M. — ***BREAK***

3) 10:45 - 11:45 A.M. — Proposed Changes to the License Renewal Program (Open (MVB/SD/CS)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding proposed changes to the license renewal program related to the review of scoping and screening processes.

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

11:45 - 12:45 P.M. — ***LUNCH***

4) 12:45 - 1:45 P.M. Proposed Technical Specifications For Ensuring Steam Generator Tube Integrity (Open) (FPF/CS)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff regarding proposed technical specifications associated with steam generator tube integrity.

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

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

5) 2:00 - 5:45 P.M. Safeguards and Security Matters **(Closed)** (MVB/RPS/RKM)

5.1) Remarks by the ACRS Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding Safeguards and Security matters.

[NOTE: This session will be closed to protect information classified as national security information as well as safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).]

5:45 - 6:00 P.M. ***BREAK***

6) 6:00 - 7:00 P.M. Preparation of ACRS Report (Open)

Discussion of proposed ACRS report on:

6.1) License Renewal Application for Dresden and Quad Cities Nuclear Plants (MVB/MDS/CS)

FRIDAY, SEPTEMBER 10, 2004, CONFERENCE ROOM T-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 A.M. Assessment of the Quality of the Selected NRC Research Projects (Open) (DAP/SLR/TSK/RC/HPN)

8.1) Remarks by the Subcommittee Chairman

8.2) Discussion of the preliminary results of the cognizant ACRS members' assessment of the quality of the NRC research projects on Sump Performance and on MACCS Code.

	10:30 - 10:45 A.M.	*****BREAK*****
9)	10:45 - 11:45 A.M.	Divergence in Regulatory Approaches Between U.S. and Other Countries (Open) (DAP/HPN/SD) 9.1) Remarks by the Subcommittee Chairman 9.2) Discussion of a draft White Paper prepared by Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding divergence in regulatory approaches between U.S. and other Countries.
	11:45 - 12:45 P.M.	*****LUNCH*****
10)	12:45 - 1:45P.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)	1:45 - 2:00 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.
12)	2:00 - 2:30 P.M.	Trip Report - AP1000 Workshop in China (Open) (TSK/MME) Report by and discussions with Dr. Kress, ACRS member, who attended the International Workshop on AP1000 that was held in China on July 26-29, 2004.
	2:30 - 2:45 P.M.	*****BREAK*****
13)	2:45 - 3:15 P.M.	Trip Report - Chalk River Facility in Canada (Open) (DAP/MME) Report by and discussions with Dr. Powers, ACRS member, who visited the Chalk River Facility in Canada.
14)	3:15 - 4:15 P.M.	Draft Final ACRS Action Plan (Open) (MVB/JTL/MWW) Discussion of the draft final ACRS Action Plan.

15) 4:15 - 6:30 P.M.

Preparation of ACRS Reports (Open)
Discussion of the proposed ACRS reports on:
15.1) License Renewal Application for Dresden and Quad Cities Nuclear Plants (MVB/MDS/CS)
15.2) Divergence in Regulatory Requirements Between U.S. and Other Countries (DAP/HPN/SD)

SATURDAY, SEPTEMBER 11, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16) 8:30 -12:00 Noon
(10:30-10:45 A.M. BREAK)

Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 15.

17) 12:00 - 12:30 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) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
514th ACRS MEETING
July 7, 2004

[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 July 7-9, 2004
2	Final Safety Evaluation Report (SER) Associated with the AP1000 Design Certification
	2. AP1000 Safety - Westinghouse Electric Company (slides)
	3. AP1000 Review Status (slides)
	4. Letter to Dr. Susan Sterrett, Asst. Professor, Duke University from James Lyons, NRR/NRC: Response to Concerns About the AP1000 Design Certification
	5. Letter to ACRS from Dr. Susan Sterrett, Asst. Professor, Duke University: AP1000 Fluid System Design & QA Procedures
	6. Letter to ACRS Subcommittee on Future Plant Design from Dr. Susan Sterrett, Asst. Professor, Duke University: Heat of Solar Radiation and AP1000 Ultimate Heat Sink
	7. ACR-700 Prepared for the ACRS
3.	Draft Final Generic Letter on Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs
	7. GSI-191 Generic Letter 2004-xx (slides)
	8. Redline-strikeout version of Generic Letter 2004-xx: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors
	9. (Attachment 1 to ?) Generic Letter 2004-xx: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors
	18.
4.	Risk-Informing 10 CFR 50.46, Acceptance Critria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors
	10. Risk-Informing 50.46 ECCS Acceptance Criteria
	11. Passive System LOCA Frequencies for Risk-Informed Revision of 10 CFR 50.46

5. Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries

8. Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants

   12. NRR Presentation to ACRS Ultrasonic Flow Meter (UFM) for Measuring Feedwater Flow Used in Determining Reactor Thermal Power (handout)
   13. Statement to ACRS Ultrasonic Flow Meter Input for Thermal Power Calculation Issue in Perspective (handout)

9. Future ACRS Activities/Report of the Planning and Procedures Subcommittee

   14. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - June 3, 2004 [Handout 1]
   15. ACRS/AACNW Full Committee and ACRS Subcommittee Meetings

10. Reconciliation of ACRS Comments and Recommendations

   16. Reconciliation of ACRS Comments and Recommendations [Handout #2]

11. Status of the ACRS Members' Assessment of the Quality of Selected NRC Research Projects

14. Metrics for Evaluating the Quality of the NRC Research Programs

   17. The Value Tree for Starting Projects [Viewgraph]

Appendix V
513th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Draft Final 10 CFR 50.69, "Risk-Informed categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" [Handout]

1. Table of Contents
2. Proposed Agenda/Schedule
3. Project Status Report, dated June 2, 2004 [Internal Committee Use Only: Predecisional Material Attached]
4. Memorandum dated May 17, 2004 to John T. Larkins from Catherine Haney, Director, Reactor Policy Rulemaking Program, DRIP/NRR, Subject: Final Rule - Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants" (with attachments)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING

JULY 7-9, 2004

July 7, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
John Segala	NRR/DRIP/RNRP
Frank Gillespie	NRR/DRIP
Edward D Throm	NRR/DSSA/SPSB
Lauren M Quinones-Navarro	NRR/DRIP/RNRP
Joseph Colaccino	NRR/DRIP/RNRP
Michelle Hart	NRR/DSSA/SPSB
Jim Lyons	NRR/DRIP/RNRP
Steve Bajorek	RES / DSARE
Andre Drozd	NRR/DSSA
Jerry Wilson	NRR/ RNRP
John Hannon	NRR / DSSA
Thomas Hafera	NRR/DSSA
John G. Lamb	NRR/DLPM
Suzanne Black	NRR/DSSA
Jennifer Uhle	NRR/DSSA/SRXB
Michael Marshall	NRR/DLPM/PDII-2
BILL KEMPER	NRC/OIG
B. P. Jain	NRC/RES
DAVID CULLISON	NRR/DSSA/SPLB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING

JULY 7-9, 2004

July 7, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Mark Kowal	NRR/DSSA/SPLB
Kris Parczewski	NRR/DE/EMCB
Leslie Kerr	NRR/RPRP/DRIP
GEORGE MENCINSKY	NRR/DRIP/RPRP
Allen Hiser	RES/DET/EMCB
David Terao	NRR/DE/EMEB
John Fair	NRR/DE/EMEB
Gary Hammer	NRR/DE/EMEB
Lee Abramson	RES/DRAA/PRAB
BRIAN THOMAS	NRR/DRIP/FRAS
Nilesh Chokshi	RES/DET/MEB
Mike Schiller	NRR/DSSA/SPSB
Charles Ader	RES/DRAA
Roy MATHEW	NRR/ADPT
Jocelyn Mitchell	RES/BSARE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING
JULY 7-9, 2004

July 7, 2004
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
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NAME	AFFILIATION
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING
JULY 7-9, 2004

July 7, 2004
Today's Date

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David Lew	NRC/RES/DRAA
Andy Ahlum	ICF

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING

JULY 7-9, 2004

July 8, 2004
Today's Date

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Joel Page	NRC/RES/MEB
DAVE KERN	NRC/OEDO

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING

JULY 7-9, 2004

July 8, 2004
Today's Date

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NAME	AFFILIATION
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Victor Hall	OIP
John Segala	NRR/DRIP/RNRP
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Joseph Colaccino	NRR/DRIP/RNRP
Jocelyn Mitchell	RES/SARE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING
JULY 7-9, 2004

July 8, 2004
Today's Date

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John McInerney	Westinghouse
Chip French	Westinghouse
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Frank Bursic	AMAG
William Slagle	Westinghouse
Hugh Thompson	TALIS MARU
Cal Hastings	Caldon
Bill Horin	Winston + Strawn
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Nancy Chapman	SERCH / Bechtel
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING
JULY 7-9, 2004

July 8, 2004
Today's Date

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Susan G. Sterrett	self (Prof., Duke University)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

514th FULL COMMITTEE MEETING

JULY 7-9, 2004

July 9, 2004
Today's Date

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Rob Elliott	NRR/DSSA

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ITEMS OF INTEREST

514th ACRS MEETING

JULY 7-9, 2004

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
514th MEETING
July 7-9, 2004

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- Staff Requirements - Meeting with ACRS, 1:30 p.m., Wednesday, June 2, 2004, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance), dated June 30, 2004 4

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July 1, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - SECY-04-0037 - ISSUES RELATED TO PROPOSED RULEMAKING TO RISK-INFORM REQUIREMENTS RELATED TO LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LOCA) BREAK SIZE AND PLANS FOR RULEMAKING ON LOCA WITH COINCIDENT LOSS-OF-OFFSITE POWER

The Commission has approved the development of a proposed rule, subject to the additional comments and clarifications noted below, to risk-inform the requirements addressing large break loss-of-coolant accidents (LOCA). The staff should provide the Commission with a proposed rulemaking package in six months. The staff should ensure that quality and safety are not compromised in order to meet the six-month schedule. The staff should keep the Commission fully and currently informed of any significant issues that arise and any delays in this schedule.

(EDO) (SECY Suspense: 12/31/04)

The staff should use the initiating event frequencies from the expert elicitation process, supported by historical data and fracture mechanics analysis and other relevant information, to guide the determination of an appropriate alternative break size. In addition, the staff should use (or require licensees to use) the approach and guidance in Regulatory Guide (RG) 1.174 to assure that the selection of the maximum break size is risk-informed and conforms to the RG 1.174 safety principles. For example, a frequency of 1 occurrence in 100,000 reactor years is an appropriate mean value for the LOCA frequency guideline for selecting the maximum design-basis LOCA since it is complemented by the requirement that appropriate mitigation capabilities, including effective severe accident mitigation strategies, must be retained for the beyond design-basis LOCA category.

The proposed rule package should allow operational as well as design changes. However, the scope of changes should be constrained in areas where engineering margins should be retained to satisfy the safety principles of RG 1.174 (e.g., containment design pressure, and severe accident mitigation capability). Finally, this scope should be constrained in areas where the current design requirements contribute significantly to the "built-in capability" of the plant to resist security threats.

Licensees should be required, by regulation, to retain the capability to successfully mitigate the full spectrum of LOCAs for break sizes between the new maximum break size and the double-ended guillotine break of the largest pipe in the reactor coolant system. The mitigation capabilities for beyond design basis events, and any changes to these capabilities, should be

controlled by NRC requirements commensurate with the safety significance of these capabilities and not by voluntary means.

The low risk contribution of the large break LOCA, which allows removal of the large break LOCA from the design basis event category, should weigh heavily in the types of requirements that would be imposed in this area. Because of the low safety significance of the large-break LOCA, a high level criterion in the rule should include the requirement for the licensee to provide effective mitigation capabilities, including severe accident mitigation strategies directed at break sizes greater than the alternate maximum break size permitted by the rule, to maintain the core in a coolable geometry. Consistent with the approach taken in the 10 CFR 50.69 rulemaking on treatment and commensurate with the low safety significance of these capabilities, the staff should ensure that capabilities are provided in a performance-based manner and not in a prescriptive manner. Furthermore, to address the potential consequences from a beyond design basis LOCA, the staff should include a requirement for containment integrity.

The level of regulatory oversight, including the required level of detail and conservatism of the supporting analyses, should be commensurate with the categorization (i.e., as design-basis events or beyond design-basis events). For example, design-basis LOCA analysis should continue to meet the requirements of 10 CFR 50.46 (either the Appendix - K requirements or the 95th percentile of the realistic alternative), while the appropriate mitigation capabilities for beyond design-basis LOCAs need not meet the single failure criterion nor would the models used to demonstrate mitigation capabilities need to be 50.46 evaluation models.

A change process for proposed plant changes using the rule should follow existing regulations and guidance, (e.g., 10 CFR 50.59 and 50.90, and RG 1.174) and should ensure that the review mechanisms for such changes provide for adequate NRC oversight. If specific requirements to control changes are needed for this rule, including reversibility considerations, then an appropriate change process should be part of the rule.

The proposed rule should be structured such that a backfit analysis is not necessary for plant changes resulting from LOCA frequency increases identified by a periodic re-evaluation of LOCA frequencies. Backfit analyses should not be required where restorations to the design basis and other actions are necessary because the licensee is unable to maintain compliance with the relevant large break LOCA criteria as a result of changes in plant design and operating characteristics (or new information such as revised frequency estimates). The re-estimation of LOCA frequencies should build on the existing information at the time and should not involve a complete repeat of the expert elicitation process. Stability and reliability of the process should be important considerations. Additionally, licensees should be aware that changes or other adjustments may be necessary if estimated LOCA frequencies increase as a result of the re-estimation or review.

The proposed rule should encourage the use of realistic LOCA methods, but the safety benefits to be gained by the re-definition of the large-break LOCA should not be delayed by requiring that the implementation of the rule be coupled to other activities that might be desirable but are not critical to addressing the safety issues. Licensees should not be required to reanalyze small break LOCAs with best-estimate models. Since the existing analytical models for small breaks are adequate and conservative, there is no need to make those best estimate models a regulatory requirement for small break LOCA analyses.
The staff should continue to consider how future plant designs would be covered by the rule;

however, the rulemaking for future plants should be pursued on a separate (and slower) path from rulemaking for existing plants. The rulemaking for future plants should not be constrained by the decisions made on the rulemaking for existing plants.

The Commission has approved the staff's recommendation to review the Boiling Water Reactor Owners Group (BWROG) pilot exemption request and subsequently proceed with rulemaking on loss-of-coolant accident/loss of offsite power. However, the staff should be ready to proceed with rulemaking if the BWROG efforts encounter significant delays (i.e., delays of six months or more). The BWROG should be informed of the potential impacts, consistent with staffing issues, that delays in submitting the pilot exemption request could have on the staff's ability to perform the review prior to completing the rulemaking.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

IN RESPONSE, PLEASE
REFER TO: M040602B

June 30, 2004

MEMORANDUM FOR: John T. Larkins
Executive Director, ACRS/ACNW

Luis A. Reyes
Executive Director for Operations

FROM: Annette Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ACRS, 1:30 P.M., WEDNESDAY, JUNE 2, 2004, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the Advisory Committee on Reactor Safeguards on the following topics:

1. Overview
2. PWR Sump Performance
3. PRA Quality for Decisionmaking
4. Risk-Informing 10 CFR 50.46
5. ACRS 2004 Report on the NRC Safety Research Program
6. ESBWR Pre-Application Review
7. Interim Review of the AP1000 Design

The Commission requested the ACRS work with the staff to resolve outstanding issues with respect to PWR Sump Performance, and make a recommendation for a practical solution within a reasonable period of time. Both the ACRS and the staff should focus their attention, resources, and additional research, if needed, on evaluating realistic scenarios rather than all possible scenarios.

The Commission commends ACRS' efforts in the security area. The Commission notes that the ACRS has closely followed the Commission's guidance contained in the October 31, 2003 Staff Requirements Memorandum (SRM M031002), and that those provisions remain in place.

The staff and the ACRS should engage with the Commission prior to making arrangements similar to the one recently agreed to between the Office of Nuclear Regulatory Research (RES) and the ACRS for ACRS to assess the quality of NRC research projects. This interaction should focus on the benefits and resource implications of such work and consistency of the proposals with previous Commission direction.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
OIG
OPA

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The 3rd Annual Homeland Security Summit

Session on, "The Best-Laid Plans:
A Case Study in Preparedness Planning"

The Very Best-Laid Plans
(the NRC's Defense-in Depth Philosophy)

Remarks of Nils J. Diaz
Chairman, U.S. Nuclear Regulatory Commission

June 3, 2004

Preparedness and Defense-in-Depth

As the title of this session suggests, "the best-laid plans of mice and men often go astray" -- I won't attempt Robert Burns' original Scottish version -- but, I believe that the *very best-laid plans* can accommodate many imperfections, weaknesses, or failings. The NRC has long had a philosophy that accommodates this reality. We call it our "defense-in-depth philosophy". It is really more than a philosophy: it is an action plan, an approach to ensuring protection. The concept of "defense-in-depth" is a centerpiece of our approach to ensuring public health and safety, and it goes beyond pieces of equipment. It calls for, among other things, high quality design, fabrication, construction, inspection, and testing; plus multiple barriers to fission product release; plus redundancy and diversity in safety equipment; plus procedures and strategies; and lastly, emergency preparedness, which includes coordination with local authorities, sheltering, evacuation, and/or administration of prophylactics (for example, potassium iodide tablets). This approach addresses the expected as well as the unexpected; it actually accommodates the possibility of failures. The NRC's defense-in-depth has recently been strengthened by incorporating the dynamics of risk-informed and performance-based decision making.

Integrating Safety, Security, and Emergency Preparedness

The events of 9/11 brought to this country a new recognition of the importance of physical security and emergency preparedness in the world of 21st century America. In the case of the NRC and the nuclear industry, this awareness had already come decades ago, and to that extent, we were, so to speak, ahead of the curve. For a generation, our regulations had postulated the existence of a terrorist threat, as part of the defense-in-depth approach. Thus the kind of drastic changes in security seen in the airline industry, for example, were not required for nuclear plants, because we had put those structures in place long ago. To be sure, we faced new realities, and significant enhancements were made after 9/11. Security orders were issued on February 25, 2002, that tightened existing policies and procedures in the light of the most cu[illegible] information, but it was not a wholesale revamping of our entire regulatory structure. We then continued to make ad[illegible]l improvements. We were among the best prepared then; we still are among the best prepared now.

What the post-9/11 review of security issues highlighted is how tightly interconnected are reactor safety, security and

emergency preparedness. Many of the same issues are involved in avoiding and mitigating reactor accidents as in preventing and mitigating acts of terrorism. Though the initiating events may differ, defense-in-depth applies in very similar ways to both.

Si[illegible]11, the whole of the U.S. government, state and local authorities, and many elements of the private sector have res[illegible]ed in a manner that increases our security. The NRC has worked closely with the Homeland Security Council, the Department of Homeland Security, NORTHCOM, the FBI and other agencies to enhance the nation's overall detection, prevention, mitigation and response capabilities. Federal action at the airports and on airliners, for example, surely reduces the likelihood of terrorists using commercial aircraft against nuclear facilities or any other targets. It is clear that we have made significant progress in the past year toward achieving an integrated response program for the defense of nuclear facilities. For our part, the NRC has required enhanced security measures for the defense of nuclear power reactors. These include multiple, but strongly interdependent elements, all directed to one fundamental goal: how best to protect our people, with the appropriate resources placed at the right places. These elements are:

- Enhanced access controls, to prevent unauthorized entry of persons and materials to nuclear facilities;
- Enhanced work and training requirements for security personnel, to increase their capability to detect and respond to threats;
- Enhanced Force-on-Force security exercises at nuclear power plants;
- Revised Design Basis Threat (addressing vehicle bomb threats, land-based and water-based assaults) and associated defensive capabilities;
- Enhanced mitigation procedures and strategies based on the established concept of Severe Accident Management Guidelines and using the results of extensive vulnerability studies;
- Enhanced emergency preparedness.

Also, the NRC has conducted extensive analyses of the potential vulnerability of nuclear power plants to aircraft attacks. While these analyses are classified, the studies confirm that the likelihood of damaging the reactor core and releasing radioactivity that could affect public health and safety is low. The fact is that nuclear reactor design requirements for structures to withstand severe external events (hurricanes, tornadoes, and floods), and for safety systems to include redundant emergency core cooling, redundant and diverse heat removal, fire protection features, and station blackout capabilities, provide built-in means of dealing with attempted terrorist attacks. Existing emergency operating procedures and en[illegible]ed severe accident management guidelines are well suited for mitigating the effects of accidents or intentional attacks o[illegible]ar power plants. In addition, all nuclear power plants have been required to enhance the integration of safety, se[illegible], and emergency preparedness. Given these enhancements, the potential radiological consequences to the public of an aircraft attack are low.

Further, the studies confirm that even in the unlikely event of a radiological release due to terrorist use of a large aircraft, NRC's emergency planning basis remains valid. Defense-in-depth provides the time needed to use the right protective strategies. The people of our country will have the protection they need and deserve. The Nuclear Regulatory Commission is joined by other Federal agencies, led by DHS, and by state and local authorities in assuring that our people will be protected.

The analyses, conclusions, and insights that I just presented for nuclear power plants also apply to spent fuel pools, since they are also well engineered and protected structures, and are amenable to simple and effective mitigative actions, if needed. For a dry spent fuel storage cask, it is highly unlikely that aircraft impact on a cask would cause a significant release of radioactive material. In addition, results to date show that a large commercial aircraft crashing into a transportation cask would not result in a release of radioactive material.

Defense-in-depth works for nuclear facilities. It is definitely a case study in total preparedness planning.

In summary, I believe that the NRC and the industry have done their jobs well, planning for success in safety and security but ever prepared to deal with the expected as well as the unexcepted. We have assessed what needed to be done and we have done it. The NRC, other government organizations, and the licensees have taken action to protect the people of our nation.

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Fifty Years On: Golden Anniversaries, Golden Opportunities

Remarks of Nils J. Diaz
Chairman, U.S. Nuclear Regulatory Commission
before the
American Nuclear Society

Pittsburgh, Pennsylvania
June 14, 2004

President Foulke, distinguished participants in this special plenary panel, ladies and gentlemen: it is a pleasure and an honor to take part in this landmark meeting of the American Nuclear Society. The past few months, as we are all aware, have been a time of anniversaries, remembering milestones in war and peace. Dwight Eisenhower, whose place in wartime history is forever associated with the D-Day landings of 60 years ago, changed the course of peacetime history a decade later with his visionary "Atoms for Peace" speech. This past week, we mourned the passing yet celebrated the achievements of another peace-time visionary, President Ronald Reagan. We at the NRC frequently quote and use one of President Reagan's phrases, "Trust but Verify." This year marks the 50th anniversary of the Atomic Energy Act, in which Congress, after deep study of the practical, legal, and economic factors involved, laid the groundwork for the civilian use of nuclear energy, under civilian control.

It was also in 1954 that the pioneers in nuclear engineering and related fields recognized the need for a nuclear organization through which knowledge would be shared and the public and decisionmakers educated.

This too was a visionary step. One is reminded of the famous coded telegram announcing Enrico Fermi's success in achieving fission beneath Stagg Field at the University of Chicago: "The Italian navigator has landed in the New World." Half a century ago, the nuclear field was still very much a new world, and these pioneers of the American Nuclear Society had the imagination, the foresight, and the wisdom to anticipate future needs, like settlers laying out the design and governance of cities that had yet to be built. I have been privileged to know a few of them and every one of them left a definite impression in my life.

It was said of Sir Christopher Wren, the architect of so many London churches, "If you seek his monument, look around you." The 103 U.S. operating nuclear plants we have today, the nuclear medicine departments in our hospitals, the nuclear power sources in the sea and above, the innumerable uses of radiation in industry -- these are the monuments to these pioneers, and we and the nation owe them a debt of gratitude today.

I would add that the relationship between the Nuclear Regulatory Commission and the ANS has long been close and fruitful. One illustration of this is that the roster of presidents of the ANS includes one former NRC chairman, Joseph Hendrie, and another, the late Nunzio J. Palladino, who led the NRC later in his career. Distinguished chairmen both, they are remembered warmly at the NRC both for their leadership and their human qualities. Nor do I wish to leave out former Commissioner Gail de Planque, who served with distinction at NRC and was the first woman to lead this organization.

Fifty years of achievement in the nuclear field are an impressive beginning. But as the motto engraved on the National Archives says, "What is past is prologue." The accomplishment of the past half century will be only partial, and ultimately disappointing, if it does not serve as prologue to another half century just as fruitful and successful in serving the needs of the American people. Today I would like to talk about what the NRC is doing to prepare the way for that next half century.

Operational Safety

Operational safety is, of course, the alpha and omega of nuclear power plants. It requires constant, unremitting vigilance on the part of licensees and of the NRC. In recent years, we have seen the nuclear industry performing, in general, at an extremely high level of safety, with corresponding economic benefits. Indeed, one of my principal concerns this last year has been to reinforce awareness that the fine safety record we have seen to date is not to be taken for granted, and not a reason for laxity of any kind. Not long ago, we had a reminder, at Davis-Besse, that neither licensees nor regulators can afford to take safety for granted. It should be a wake-up call to everyone connected with nuclear power plants that if we want to maintain the industry's safety record, we need to recognize complacency as perhaps our greatest enemy. Every so often, an unnecessary event seems to raise unwarranted questions about the technology. These preventable events are detrimental to our nation. There can be no slackening of our commitment to the highest standards, and no departure from a permanently questioning and problem solving attitude.

At this point in our history, public acceptance of nuclear power is at a high level, and rightly so; but it is not unconditional, and it is certainly not irrevocable. Public acceptance has to go on being earned.

I've made the comment in the past that even the Three Mile Island accident, the worst crisis in the history of commercial nuclear power in this country, did at least provide some enlightenment. For industry, it reinforced the realization of how interconnected this country's utilities are -- that every nuclear plant major activity or event could be a reflection on every other nuclear plant, for better and for worse. After TMI, we saw the industry, through the creation of the Institute for Nuclear Power Operations, taking an increasingly forward role in ensuring that weaker performers were brought up to a high standard of performance, in the recognition that in the public mind, a problem at any plant casts a shadow over all the others. I think that is something that all of us connected with nuclear power need to bear in mind. The responsibility of every licensee is not only to the public in the vicinity of the plant; not only to the utility's shareholders; but also to the entire industry, to all those associated with it, and to the American public as a whole. The reality is that today America depends upon safe and reliable electricity from nuclear reactors and on the many benefits from uses of radioisotopes to maintain or enhance our quality of life.

Planning for the Future

At the NRC, we have made a number of changes with a view to enhancing the safety of our operating nuclear plants, including the move in the direction of risk-informed and performance-based regulation, which I strongly believe is a necessity. (Those of you who have attended the NRC's Regulatory Information Conferences have been hearing me beat the drum for risk-informed regulation since 1998.) The old system of prescriptive regulation served the nation's nuclear power plants well in their formative decades, but we have gone too far and learned too much to continue in that mold indefinitely. We now have a wealth of operating experience, as well as the benefit of increasingly sophisticated risk analyses that allow us to focus on the key safety issues. We would be neglecting our duty if we did not take advantage of that vastly increased information base and those invaluable analytical tools to modernize our regulatory apparatus.

In other fora, I have recently described two major regulatory steps in that direction, dealing with Special Treatment requirements and Emergency Core Cooling Systems and the Large Break LOCA, and will not repeat that discussion here. (Anyone interested in the details can find them on the NRC website.) Briefly, they are designed to focus where safety counts the most, while allowing flexibility on areas of lesser safety significance. These changes are not, as some have charged, a diminution of safety margins, but rather an optimization of resources that are to be focused on safety, something that all who are concerned about nuclear safety should welcome.

The question is whether these and other steps in the direction of risk-informed and performance-based regulations represent the end of our efforts in this area or a beginning. I personally believe that they should be a good start, and that we need to revise all the NRC's reactor regulations accordingly. Just last month, the Nuclear Energy Institute, which initially was somewhat wary of so major a change in NRC's regulatory approach, endorsed the idea of risk-informing the NRC's regulations in their entirety. I welcome this support.

It may be asked, why do we need to risk-inform the NRC's reactor regulations? Are the current regulations not good enough? Isn't there something to be said for sticking with the tried and true?

I would answer that by saying that they are certainly good enough for the regulation and oversight of currently operating reactors, and even for the evolutionary and advanced reactor designs that NRC has reviewed. But "good enough" should not be our standard, when we have the capacity to be far better than that. We have the know-how and the tools to create regulations that will allow us to incrementally incorporate the best scientific and technical information, and the best methods and approaches, not only for the benefit of existing reactors, but also for future generations of reactors. These regulations can integrate and make more effective and efficient our approach to reactor safety, security, and emergency preparedness. In the objective statement that the Commission is now using in our Strategic Plan, our regulations exist to enable the use and management of radioactive materials and nuclear fuels for beneficial civilian purposes that protects public health and safety and the environment, promotes the security of the nation and provides for regulatory actions that are open, effective, efficient, realistic and timely.

Water reactors have served this country well during the past decades, and will continue to do so in the near future. I believe, however, that the long-term future of power reactors belongs to very high temperature reactors. Fusion power is much further in the future. It therefore makes sense to prepare the way now with regulations that are risk-informed and technology-neutral, providing consistency in our approach to the key regulatory issues. I believe we need to work on an Advanced Notice of Proposed Rulemaking (ANPR) for risk-informing of all the reactor regulations. This ANPR should present in clear and concrete terms the Commission's vision and commitment to this safety initiative. It should encompass what we know and what we should develop in preparing a comprehensive process to provide a path forward.

By doing so, we will also substantially increase our ability to address safety and security matters in an international context. The concept of "international certification," which has been discussed so much in recent years, can be brought closer to realization through a technology-neutral set of risk-informed and performance-based safety requirements.

Conclusion

The future of the nuclear option in this country, and all the promise that it holds for this nation's security in so many senses of the word, depends on a complex of factors. Technological developments, business judgments, and regulatory actions will all play a role. More fundamentally, however, it depends on people. Machines are no better than the men and women who design them, operate them, and regulate them.

For the past five decades, the nuclear option in this country has had the benefit of the expertise, creativity, idealism, and passion of the generation of pioneers who committed themselves to careers in this field when it was young. That first generation, the generation that was present when the American Nuclear Society was created 50 years ago, is now aging; all too many of its members are no longer with us. At the NRC, one of the pressing problems we face is that so many of the experts with decades of experience under their belts are now in retirement or approaching it. Filling their shoes will not be easy. I know that this is an issue of concern throughout the nuclear industry.

That brings us back to the American Nuclear Society. Your work, which has done so much over half a century to inform and enlighten our nation about nuclear energy, is never more needed than today. By instruction and by their example -- a shining example -- the members of this Society can help ensure that in the coming half century, as in the 50 years just past, our country will have the kind of men and women it needs in the nuclear field: people of expertise, dedication, and commitment. The fulfillment of the promise of nuclear energy depends on the success of that effort.

Thank you.

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REMARKS OF CHAIRMAN NILS J. DIAZ
UNITED STATES NUCLEAR REGULATORY COMMISSION
before the
GENERATION IV INTERNATIONAL FORUM REGULATORS MEETING
June 17, 2004
Paris, France

Good Morning. It is a great pleasure and honor to chair this meeting today. I wish to thank NEA, and especially Dr. Magwood, for this opportunity to share some of my thoughts with you. Last September, I had the privilege of meeting with you during the Generation IV International Forum meeting in Toronto, Canada, where I spoke about improving the safety and reliability of nuclear power plants, and the potential to use design certification processes internationally. Today, a question for us to consider in the context of Generation IV is, "are we regulators doing what needs to be done to fulfill our responsibility to enable the safe and beneficial uses of nuclear energy and radioactive materials?" And, of course, I will offer my thoughts on the usefulness of design certification, based on our experience with design certifications reviews.

Before I launch into the theme, let me offer some general comments. Principles often do not change, even when the application changes. As regulators, we must expect and demand from the nuclear industry the same thing we expect and demand of ourselves: an unconditional commitment to safety, security, and preparedness. Moreover, we must require that our actions, both those of the regulators and the regulated, be consistent, predictable, realistic, and appropriately conservative -- using what I call "realistic conservatism."

By "realistic conservatism," I mean the decisions that are informed by the real world of advancing scientific knowledge, technological capabilities, and experience in order to preserve appropriate and prudent safety margins and to regulate in a manner that corresponds to the actual risk presented and not to worst case assumptions. Being realistically conservative is in the best interest of the public safety and the environment, by ensuring that we maintain the right balance between under-regulating -- which puts the public safety and the licensees' investment at risk -- and over-regulating, which could divert resources from important safety issues while increasing costs to licensees and thus to consumers, without a matching safety or security benefit.

Design Certification Process

I would now like to expand further on my Toronto comments, and discuss the need for an effective and efficient regulatory framework that is risk-informed and performance-based, and that could have international use. Looking ahead to the potential technologies that may be employed over the next 30 - 50 years, I fully recognize that some, perhaps many, of our current regulations may not be directly applicable. This implies to me that we will need a regulatory framework that will adequately address design and operational issues associated with future reactors that may be distinctly different from current LWRs.

As suggested last September, I believe that the path forward should include the development and international adoption of a regulatory framework that can establish the appropriate safety requirements, compatible with the on-going evolutionary nature of today's nuclear technologies. The NRC has been putting in place a regulatory framework that should allow for the

safe and beneficial use of nuclear power in the future.

In this regard, the NRC has developed a design certification process under 10 CFR Part 52 to bring about enhanced safety and the early resolution of licensing issues. It provides a more stable and predictable licensing process by: (1) resolving safety issues for an essentially complete nuclear power plant design; (2) placing these resolutions under a restrictive change process that applies to both the regulator and the applicant for design certification and thereby reducing licensing uncertainty; and, (3) extending the approval duration to 15 years. This process resolves safety and environmental issues before authorizing construction, thus reducing licensees' financial risk while allowing for timely and meaningful public participation.

By using rulemaking for these certifications, the Commission assures license applicants who reference a certified design that the safety issues already resolved will not be reconsidered during the plant licensing process. This has led to the development of licensing processes which are ready to be used, and the NRC has issued rules certifying standard designs - the Advanced Boiling Water Reactor (ABWR), System 80+, and the AP-600. And soon, the AP-1000 design may be added to this list.

The design certification review (DCR) process examines: (1) an essentially complete design, thus facilitating standardization; (2) the final design information, which is equivalent to the information in a Final Safety Analysis Review (FSAR); (3) the postulated site parameters; (4) interface requirements; and (5) inspections, tests, analyses, and acceptance criteria (ITAAC). It does not review site safety issues, like seismology, environmental impact issues, operational programs, site-specific design features, or selected design areas. Site-specific issues are bounded to allow for separation of siting reviews from the design reviews. We have developed a design certification process that has been tested and proven to meet our needs. It is the design certification process that could serve as a benchmark for the potential acceptance of a certified design by a group of signatory countries participating in its development. The site-specific issues could be resolved by each country's regulatory authority in accordance with their own regulatory framework.

These siting reviews lead to Early Site Permits (ESP), which allow for the "banking" of a site for up to 20 years. The review looks at site safety, environmental impact issues, and emergency preparedness and reduces licensing uncertainty by resolving site-related issues.

A utility can elect to use a certified design in conjunction with a combined license application. A Combined License (COL) combines both a construction permit and an operating license with conditions for a nuclear power plant that has a 40-year duration. It can reference the ESP, DCR, both, or neither, and is the fundamental licensing process in Part 52 for reducing regulatory uncertainties. The review process includes extensive opportunities for public participation as well as independent technical advice from the ACRS. The DCR regulatory framework is risk-informed and performance-based, and could be adopted for international use.

Technology-Neutral Framework

Substantial progress has been made on the development of a technology-neutral framework for new plant licensing, including its overall structure, scope, approach, and content. Allow me to briefly describe what the NRC has been developing in this area.

The framework provides the technical basis for developing a hierarchical, risk-informed technology-neutral set of requirements for new plant licensing on both a generic and a plant-specific basis, utilizing the Commission's 1986 Safety Goal Policy Statement as the top-level expression of the NRC's expectations for safety. It establishes an overall safety objective that defines the region that may be deemed to provide adequate protection and beyond, if appropriate. That is, future regulatory requirements can be written to achieve the necessary level of safety defined by the Commission's Safety Goal Policy.

The framework's technical approach is a risk-informed approach that blends probabilistic and deterministic criteria in establishing four "protective strategies" -- barrier integrity, limit initiating event frequency, protective systems, and accident management -- to address uncertainties through the use of a defense-in-depth philosophy and provide reasonable assurance that the overall mission of protecting the public health, safety, security, and preparedness is met. It identifies those areas that need to be addressed to ensure plant design, construction, and operation meet the safety objective.

Administrative requirements will need to be established to ensure that the implementation of the technical requirements by licensees is done in a consistent, controlled, and documented fashion. These will include format and content of applications, PRA quality, reporting and record keeping, change control, and license amendments.

I would like to note that defense-in depth no longer includes just an array of structures, systems, and components capable of performing the intended safety function - it is broader in philosophy and in practice. It now also incorporates design, engineering, and operating experience, and is complemented by risk-informed and performance-based decision making. As a parallel issue, I have been emphasizing lately the "big three" interrelated components of defense-in-depth -- safety, security and emergency preparedness. All three are essential components of regulatory predictability and are factors on which the NRC and our licensees depend for adequate assurance of safety.

Summary

We have developed a design certification process that has been tested and proven to meet our needs, allowing for a more stable and predictable licensing process. We are making considerable progress in developing a technology-neutral framework that will ensure an acceptable level of safety for future designs, irrespective of the type of reactor being licensed.

I believe that the design certification process and technology-neutral framework I have described offers a considerable opportunity that is available to the international community. I offer for your consideration the option of reviewing and further developing this process as a potential basis for an international standard that could ensure the safety and global utilization of any reactor technology through-out the first half of this new century.

Thank You.

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Maintaining the Safety Management Perspective for Licensee's Contractors

REMARKS OF NILS J. DIAZ
CHAIRMAN, U.S. NUCLEAR REGULATORY COMMISSION

Before the
CNRA Regulatory Industry Forum

Paris, France
June 17, 2004

It is a great pleasure and honor to participate in this international regulatory forum and to share my perspective on the timely subject of ensuring operational safety and, specifically, ensuring operational safety when utilizing contractors and technical support services. I value our interactions and technical exchanges as nuclear regulators. Each of our countries has a unique array of plants to regulate and different regulatory frameworks. While we may approach and resolve safety issues in different ways, we know that these differences do not equate to different goals or results. We are all focused on ensuring nuclear safety. That is why this forum is so important. It gives us an opportunity to share the paths that we have traveled and discuss the road ahead.

This forum's topic of "Contractor and Technical Support Services" is an important one at any time. However, I believe it is particularly important as we see the interest in building new nuclear power plants gaining momentum in the U.S. and other countries. At one time, we may have expected to look inside our own borders for the fabrication and construction of the majority of the equipment for a nuclear power plant. Today that is not the case. We are all linked by the international suppliers of nuclear equipment and contractor technical expertise. For example, France, Italy, Japan, Spain, and Canada have all supplied steam generators or reactor vessel heads to U.S. nuclear power plants. It appears that this trend will continue. As regulators, we must ensure that the regulatory framework and industry practices are aligned with each other.

I would like to start my remarks today with a story that you may have heard before. It has several versions but the point of the story always remains the same. It is a story about four people named Someone, Anyone, Everyone, and No one. There was an important job that had to be done -- Someone should have done it, Anyone could have done it, and Everyone thought that Someone would do it. In the end, No one did it. At the beginning, the responsibility was not assigned.

In the nuclear arena, we can not afford the luxury of allowing a job to go undone or be poorly done. It is of paramount importance for the lines of responsibility to be clearly defined. The regulator must develop a clear framework to ensure the responsibilities of the regulator, licensee, and contractors are assigned and discharged, consistently. A clear framework starts with a common goal -- here, the common goal is safety. With a framework in place, each party can then determine how best to fulfill its responsibilities in the most effective and efficient manner. With a clear framework, the actions taken by industry and the regulator should be transparent and predictable.

Safety Management

There are many issues and challenges in today's nuclear safety landscape: license renewals, power uprates, materials degradation, risk-informed and performance-based approaches to operation and regulation, and the management of safety. Both the industry and the regulator have responsibilities to address the safety landscape with a comprehensive safety management approach. And good safety management must apply to contractors and vendors as well as to the licensees; there should be no gaps in the assignment or in the discharge of responsibilities.

Everyone probably has their own definition of safety management; they all share some fundamental elements. Here is my version. Safety management is, at least, the collective product of three essential, interactive elements that are actively managed:

1. A functional and executable commitment to operational, maintenance and engineering safety, imbedded in every activity of the organization,

2. a technical expertise that is applied where and when it should be; able to receive, process, form, and communicate technical issues; cognizant of safety functions and safety systems, with licensing and regulation as boundary conditions but taken beyond them by the pursuit of safety and reliability, and

3. management of the people, programs, and processes to implement a safety program effectively.

Let me elaborate on these three essential elements of safety management.

The first element -- commitment to safety -- includes the desire to do things right; a questioning attitude and a receptiveness to questioning attitudes; a willingness and ability to learn; and the experiential awareness of how indispensable safety is.

The second element -- the application of technical expertise -- involves using realistic conservatism in safety analysis; quality engineering based on state-of-the-art information; and, operational safety and maintenance founded in science, engineering, technology, and operating experience.

Our technical and regulatory know-how is increasing. As regulators, our technical competency has to match industry's. As technology and regulation progresses, both industry and the regulator must learn on a parallel curve. An important part of the nuclear industry's technical know-how depends on the capability of the contractors and technical support services. From cradle to grave, the industry utilizes contractors and support services with the technical know-how to manufacture fuel and safety equipment and to perform specialized services, such as refueling operations, accident analyses, and security operations. There must be appropriate checks by the licensee and regulator to ensure the performance and the technical competency of contractors as well as ensuring the quality of their work. In the U.S., the responsibility of contractor oversight resides with the plant operator but must also be part of the regulatory framework. I believe it is also important that licensees and contractors alike be cognizant of regulatory requirements.

Last but not least, the third element is management. It is my long-standing position that the management of nuclear power plants is the licensee's responsibility and prerogative. Part of management is monitoring performance. Just as any employee's performance must be monitored, contractor performance must also be monitored. Again, there must be criteria against which contractors are measured for quality and performance; and there must also be accountability for not meeting expectations.

The U.S. Framework

The U.S. nuclear industry is a mature industry, and maturity has a lot to do with the present high performance of the U.S. nuclear power plants. In the nuclear business, maturity also requires learning, awareness of the old and new, and the appropriate application of know-how, especially for emerging issues. However, there have been lapses in performance. Extended shutdown of U.S. reactors have occurred too frequently, and there have been a few avoidable events of safety significance because the requisite technical expertise and safety management criteria were not applied in a timely manner to the resolution of design, operational, and maintenance problems. This is not acceptable. Whether poor performance comes from the licensee's staff or its contractor, the licensee bears the responsibility and must accept the consequences.

The NRC has increased its safety focus on licensing and oversight activities by applying a balanced combination of experience, deterministic models, and probabilistic analysis. We call this approach risk-informed and performance-based regulation. This enhanced safety focus is used by our licensees and by the agency in a concerted effort to ensure adequate

protection of public health and safety with a more quantitative and up-to-date technical basis. It has resulted in significant improvements in the effectiveness and efficiency of our licensing and oversight activities. Some of the most important licensing activities as you know, are extended power uprates and license renewals. These have become key contributors to the [illegible]lity of the U.S. nuclear infrastructure.

Aside from licensing activities, we independently conduct assessments and inspections to verify that adequate safety margins are maintained. Thus, the safety framework includes both the licensees' multiple programs for conducting safe operations, implemented through operational safety programs, and the government's clear role in providing independent analysis and oversight for assurance of safety. This is true for licensees and for their contractors. The licensee is responsible for ensuring that the quality of the contractor's work meets the requirements of our regulatory framework. The NRC holds the licensee accountable for the performance of its contractors. If the NRC finds a significant violation of the regulatory requirements by a licensee's contractor, the violation is issued to the licensee. As a result, each licensee is required to and motivated to ensure that its contractors are meeting the NRC regulations. In some rare cases of willful violation of regulations, the NRC can take and has taken direct action against contractors. In most cases of a program weakness or deficiency, the NRC will take action against the licensee.

In addition, as part of our framework, there are NRC regulations that apply directly to contractors and companies that supply technical services. Part 21 of Title 10 of the U.S. Code of Federal Regulations is titled "Reporting of Defects and Non-compliance." This regulation applies to all licensees, and to all contractors and technical support service companies that affect the safety function of a facility licensed by the NRC. If a contractor becomes aware of a defect which could create a substantial safety hazard, it is required to report it to the NRC. It is subject to a civil penalty if it knowingly fails to report the defect. As the regulator, we act on the reported defect and non-compliance information in several ways: first we look at the immediate corrective actions performed by licensees that are known to be affected. Next, we examine the information for generic applicability to determine if the information should be communicated to all licensees for appropriate action. This is part of our framework for the oversight of licensee contractors.

In the U.S., the amount of regulatory oversight of licensee contractors has varied in response to the maturity of the industry. As utility companies have become more experienced in nuclear operations, they have become more capable of overseeing the technical activities of their contractors. At one time, the NRC had a vendor inspection branch dedicated to the oversight of licensee contractors. Today, we monitor licensees in their oversight of their contractors. One means of co[illegible]or oversight by licensees is the Nuclear Procurement Issues Committee (NUPIC) audits. U.S. licensees and other int[illegible]onal partners that are NUPIC members participate in a coordinated approach on nuclear vendor quality assurance oversight. As regulators, we will take an active role in emerging situations whenever prudent, to inspect licensee's contractors and to satisfy ourselves that the contractor meets our safety requirements.

For today and looking ahead, the need for clear responsibilities is critical as contractors and technical support companies compete in a global marketplace for nuclear goods and services. International cooperatives are now a common practice for the nuclear industry. For design certification reviews, the NRC performed quality assurance inspections at facilities across the U.S. and in Japan, Italy, and Canada. Specialized inspection services, such as reactor vessel head inspections and steam generator tube inspections, are being performed by international contractors. The IRIS advanced reactor is being designed by an international consortium consisting of vendors, energy companies, and universities from many countries, including Brazil, Italy, Japan, Mexico, Spain, the United Kingdom, and the United States.

In summary, for the U.S., the NRC's technical and regulatory framework establishes high standards that ensure adequate protection of safety, but places the primary responsibility of safety management on the licensee. For other countries, the boundaries may be defined in another manner. Exactly where the boundaries of responsibility are defined is secondary in importance to the overarching necessity to define clear boundaries so the regulatory process can be predictable and transparent to all parties involved.

I believe in the importance of a framework to define responsibilities clearly, and the importance of defining responsibilities driven by a safety management program. Simply stated, safety management embodies using commitment, technical expertise, and good management practices to achieve the requisite "adequate protection" standard that regulators demand, the reliability that licensees need, and both the protection and the reliability that the public deserves.

I cannot pass on the opportunity today to quote one of my favorite persons. The late former-President Ronald Reagan once said, "There are no great limits to growth because there are no limits of human intelligence, imagination, and wonder." In thi[illegible]stry, safety, reliability, and growth should not be limited by our know-how, our management, and our imagination. We[illegible] to keep them at work, day in and day out. We seek to learn and improve our methods in ensuring the safe operation of nuclear power plants world-wide.

Thank you for the opportunity to speak to you this morning on this important topic. I look forward to discussing the subject with you as the forum continues.

June 8, 2004

The Honorable James M. Jeffords
United States Senate
Washington, D.C. 20510

Dear Senator Jeffords:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter of April 22, 2004, in which you raised specific questions concerning the accountability for two irradiated spent fuel rod segments presumed missing from the Vermont Yankee Nuclear Power Station (Vermont Yankee). Responses to your specific questions are enclosed. The NRC staff is conducting a special inspection to investigate the missing fuel rod segments. More information may become available as the special investigation progresses.

In March 2004, the NRC's Resident Inspector at Vermont Yankee performed an inspection of Entergy's (the licensee's) program for control of special nuclear material. The inspection identified concerns with the method used by Entergy to perform the physical inventory of the two spent fuel rod segments. In response to the Resident Inspector's concerns, Entergy used an underwater video device to determine if the two spent fuel rod segments were in the storage container in the spent fuel pool, but could not locate them. On April 21, 2004, Entergy notified the NRC that the spent fuel rod segments were missing. Entergy initiated an investigation consisting of an additional underwater video inspection of the entire spent fuel pool, a review of Entergy's radioactive material shipment records, personnel interviews, and identification of possible disposal locations. On May 19, 2004, Entergy reported to the NRC that the underwater video inspection of the spent fuel pool was essentially complete. This second underwater video inspection did not find the unaccounted for fuel rod segments.

The NRC staff has closely monitored Entergy's actions and investigation since Entergy reported that it could not locate the two spent fuel rod segments. During the special inspection, the NRC will review the results of Entergy's investigation, assess Entergy's root cause evaluation, determine if Entergy is in compliance with applicable regulations, and identify which findings may have generic implications. The NRC will consider any generic findings in its determination of the need for actions by all licensees. The NRC staff has discussed this issue with individuals representing the States of Vermont, New Hampshire, Massachusetts, Washington, South Carolina, and New York, and will continue to engage them in the NRC's response to the issue. The NRC staff has also briefed members of your staff on this issue.

In your April 22, 2004, letter, you also raised concerns about the steam dryer at Vermont Yankee and requested that the NRC expeditiously release the results of NRC inspections performed in this area. Status information about the steam dryer inspection is included in the enclosed responses.

Additionally, I responded on May 4, 2004, to your letter dated March 31, 2004. As stated in that letter, I responded to the Vermont Public Service Board (PSB) by letter dated May 4, 2004, regarding the PSB's request for an independent engineering assessment of Vermont Yankee. My letter to the PSB discussed an engineering inspection of Vermont Yankee that the Commission believes is appropriate for addressing our oversight responsibilities and is also responsive to the PSB's concerns.

The Commission shares your concerns regarding the two missing spent fuel rod segments. While it is premature to make any conclusions on their location, we believe it is highly unlikely that the material is in the public domain. The NRC views the control of spent nuclear fuel to be of great importance. The NRC will continue to monitor and evaluate the licensee's response to this issue to assess actions to be taken to preclude future similar events.

If you have further comments or questions related to this matter, please contact me.

Sincerely,

/RA/

Nils J. Diaz

Enclosure: Questions and Answers

Identical letter sent to:

The Honorable James M. Jeffords
United States Senate
Washington, D.C. 20510

The Honorable Patrick Leahy
United States Senate
Washington, D.C. 20510

The Honorable Bernard Sanders
United States House of Representatives
Washington, D.C. 20515

Questions and Answers

Q. "When was the most recent NRC [Nuclear Regulatory Commission] inspection concluding that the missing pieces were in the spent-fuel pool?"

A. The NRC has never performed an inspection at the Vermont Yankee Nuclear Power Station (Vermont Yankee) specifically to verify that the two missing spent fuel rod segments in question were in the spent fuel pool. However, the NRC has conducted materials control and accounting (MC&A) inspections at Vermont Yankee.

In general, the NRC conducts MC&A inspections to determine whether licensees have limited their possession and use of special nuclear material (SNM) to the locations and purposes authorized by their operating licenses. In addition, during these inspections, the NRC determines whether licensees have implemented adequate and effective programs to account for and control the SNM in their possession. As part of MC&A inspections, NRC inspectors also verify the physical location of a sample of spent fuel assemblies in the spent fuel pool. Findings from MC&A inspections at power reactors prior to 1988 did not indicate that there were major deficiencies in power reactor licensees' MC&A programs. At that time, the NRC considered there was low risk of improper storage of spent fuel at a power reactor since its physical and radiological characteristics made it highly unlikely that spent fuel could be safely removed from the fuel pool without proper equipment and procedures. Therefore, in 1988, the NRC chose to allocate inspection resources to other more risk-significant issues. In 2001, the NRC staff examined MC&A vulnerabilities as part of the comprehensive review of the NRC's Safeguards and Security Program that was undertaken in response to the terrorist activities of September 11, 2001, and in response to the issue of two missing fuel rods at the Millstone Nuclear Power Station Unit 1.

In November 2000, the licensee for the Millstone Nuclear Power Station Unit 1 reported to the NRC that two fuel rods were missing from the spent fuel pool. As part of the lessons learned from that event, the NRC staff developed a new Temporary Instruction (TI) 2515/154, "Spent Fuel Material Control and Accounting at Nuclear Plants," to enhance the NRC's inspection of licensees' MC&A programs. The TI provides specific inspection guidance to NRC inspectors and consists of three phases. The first phase determines if a licensee has ever removed irradiated fuel rods from a fuel assembly. If the answer is yes, phase two of the TI is then implemented. Phase two of the TI determines, through detailed questions, if a licensee's MC&A program is adequate to account for items located in the spent fuel pool. If it is determined that a licensee's MC&A program has deficiencies, then Phase three of the TI is implemented. Phase three is a much more detailed inspection of the MC&A program, which includes verifying the location in the spent fuel pool of all spent fuel rods that have been separated from their parent fuel assemblies.

In March 2004, the NRC's Resident Inspectors conducted an inspection at Vermont Yankee using TI 2515/154. Phases one and two of the TI have been completed at Vermont Yankee and documented in NRC Integrated Inspection Report

No. 05000271/2004002, dated May 3, 2004 (ADAMS Accession No. ML041240438)[1]. While performing these inspections, the NRC's Resident Inspectors identified concerns with the method used by Entergy to perform the physical inventory of the two spent fuel rod segments, which are about one-half inch in diameter and about 7 inches and 17 inches in length, respectively. These two fuel rod segments were placed in a container in the spent fuel pool in 1980 after they had broken off from full length fuel rods during repair work on a fuel assembly. In response to the Resident Inspectors' concerns, Entergy used an underwater video device to check if the two spent fuel rod segments were in the storage container in the spent fuel pool, but could not locate them.

Upon identification of the discrepancy in the location of the two spent fuel rod segments, Entergy initiated an investigation. Entergy's investigation consists of an underwater video inspection of the spent fuel pool, a review of Entergy's radioactive material shipment records, personnel interviews, and identification of possible disposal locations. On May 19, 2004, Entergy reported to the NRC that the visual inspection of the spent fuel pool was essentially complete and that the underwater video inspection did not find the fuel rod segments.

The NRC staff has closely monitored Entergy's actions and investigation since Entergy formally made a report to the NRC staff on April 21, 2004, that the two spent fuel rod segments were not in their storage container. The NRC is conducting a special inspection at Vermont Yankee to investigate the missing spent fuel rod segments. During the inspection, the NRC will review the results of Entergy's investigation, assess Entergy's root cause determination, determine if Entergy is in compliance with applicable regulations, and identify which findings may have generic implications.

The NRC will consider the results of the ongoing special inspection at Vermont Yankee and the findings from MC&A inspections at other facilities to determine the need for generic communications, enforcement, inspection, or other regulatory actions.

Q. "We understand that these pieces came from damaged fuel rods and are highly radioactive. Such materials should have set off radiation detectors if moved outside the plant. When was the most recent NRC inspection concluding the detectors were working? Are there circumstances in which could this material have moved off site without alerting the detectors?"

A. As part of the NRC's Baseline Inspection Program, the NRC staff conducts focused inspections of Vermont Yankee's on-site radiation monitors and Entergy's program for the control of the shipment and disposal of radioactive material. These inspections are performed every 2 years by regional health physics specialists during several on-site visits. The last period for inspection at Vermont Yankee was January 1, 2002, through December 31, 2003. Documentation of the inspections of the on-site radiation monitors performed in 2003 can be found in NRC Inspection Report Nos. 2003-002, 2003-005,

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[1]The inspection report is available for public inspection at the NRC's Public Document Room, located at One White Flint North, File Public Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. It is also accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html.

and 2003-006[2]. Results of NRC inspections indicate that Vermont Yankee's on-site radiation monitors, as well as the controls for the shipment and disposal of radioactive material, meet regulatory requirements.

The NRC staff is confident that the radiation detectors in the area of the spent fuel pool would have alarmed if the spent fuel rod segments or any other highly radioactive material stored in the pool were inadvertently removed from the spent fuel pool in an unshielded container. Besides the radiation detectors located within the plant, there are additional radiation detectors located at personnel exit points. These detectors monitor each worker exiting the plant's protected area. These are very sensitive detectors designed to alarm at radiation levels slightly above natural background. These detectors help ensure that workers do not leave the site with contaminated clothing, equipment, or radioactive material.

Even if the spent fuel rod segments were inadvertently mixed with other radioactive waste that was stored in the spent fuel pool, and then removed from the site, they would be subject to rigorous controls and oversight. The waste is processed, packaged, and prepared for shipment in a specially shielded container. The shielded container is loaded onto a transport truck and a radiation survey is performed to verify that the radiation levels from the container meet NRC and U.S. Department of Transportation (DOT) regulatory requirements. Once in a shielded container, the two spent fuel rod segments would not be detected by the required radiation survey as long as the radiation levels meet NRC and DOT regulatory requirements.

Results of our inspections indicate that Vermont Yankee's on-site radiation monitors and the controls for the shipment and disposal of radioactive material make it highly unlikely that the spent fuel rod segments are in the public domain.

Q. "If these materials remain in the bottom of the spent fuel pool, do they pose any operational risks at the facility?"

A. If the spent fuel rod segments were located at the bottom of the spent fuel pool, they pose no operational risks to the facility.

With the exception of fuel and water transfer to the reactor vessel during refueling operations, the spent fuel pool is essentially isolated from the remainder of the power plant. The fuel segments are much denser than the cooling water and, therefore, would remain at the bottom of the pool. Spent fuel cooling would be unaffected. Spent fuel assemblies are stored in the spent fuel pool in specially designed racks that hold the assemblies several inches above the bottom of the spent fuel pool. Cooling water enters each assembly through cooling holes which are several inches in diameter and are cut through the bottom plate of the storage rack. Even if the fuel segments did not remain on the pool bottom, the fuel segments are too small to interfere with this cooling flow.

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[2]These inspection reports are available for public inspection at the NRC's Public Document Room and are also accessible from the ADAMS Public Electronic Reading Room using Accession Nos. ML031250446, ML032110574, and ML033140011, respectively.

Q. Status of the Vermont Yankee Steam Dryer

A. For the past year, the NRC staff has been closely following events at certain boiling-water reactor plants involving cracks in welds on the steam dryer inside the reactor pressure vessel. Although the steam dryer does not perform a safety-related function, it must retain its structural integrity under all loading conditions to ensure that loose parts that might be generated from its structural failure do not adversely impact the ability of safety-related components to perform their intended function. During the recent April 2004 refueling outage, Entergy identified some cracks in Vermont Yankee's steam dryer. Entergy repaired the cracks that were in external portions of the dryer. In addition, Entergy analyzed the cracks that were not repaired and determined that the unrepaired cracks could not cause loose parts or impact dryer performance. The NRC staff has inspected Entergy's analysis and did not identify any safety concerns for operation of the plant at the current licensed power level. The results of the inspection will be documented in an inspection report which is scheduled to be issued in August 2004. The inspection report will be publicly available.

The NRC staff is continuing its review of Entergy's request to increase Vermont Yankee's licensed operating power level. The NRC has informed Entergy that Entergy will need to address comprehensively the impact of the proposed increased power level on Vermont Yankee's steam dryer performance.

Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

June 24, 2004

The Honorable Edward J. Markey
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Markey:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter of April 22, 2004, in which you raised a number of questions concerning the accountability for two irradiated spent fuel rod segments missing from the Vermont Yankee Nuclear Power Station (Vermont Yankee). Responses to your questions are enclosed. The NRC staff is conducting a special inspection to investigate the missing fuel rod segments. More information may become available as the special inspection progresses.

In March 2004, the NRC's resident inspector at Vermont Yankee performed an inspection of Entergy's (the licensee's) program for control of special nuclear material. The resident inspector used an inspection procedure that was recently updated based on lessons learned from a similar event which occurred at the Millstone Nuclear Power Station in 2000. The inspection identified concerns with the method used by Entergy to perform the physical inventory of the two spent fuel rod segments. On April 20, 2004, in response to the resident inspector's concerns, Entergy used an underwater video device to determine if the two spent fuel rod segments were in the storage container in the spent fuel pool, but could not locate them. On April 21, 2004, Entergy notified the NRC that the spent fuel rod segments were missing. Entergy is conducting an investigation that includes an additional underwater video inspection of the entire spent fuel pool, a review of Entergy's radioactive material shipment records, personnel interviews, and identification of possible disposal locations. On May 19, 2004, Entergy reported to the NRC that the underwater video inspection of the spent fuel pool was essentially complete but had not located the spent fuel rod segments.

The NRC staff has closely monitored Entergy's actions and investigation since its formal report to the NRC staff on April 21, 2004, that the two spent fuel rod segments were not in their documented location in the spent fuel pool. During our special inspection, the NRC will review the results of Entergy's investigation, assess the root cause evaluation, determine if Entergy is in compliance with applicable regulations, and identify which findings may have generic implications. The NRC will use the results of the ongoing special inspection at Vermont Yankee and the findings from materials control and accounting inspections at other facilities to determine the need for generic communications, inspection, enforcement, or other regulatory actions. The NRC staff has discussed this issue with individuals representing Vermont and neighboring States, as well as Washington and South Carolina, where low level radioactive waste is shipped for final burial.

The Commission shares your concerns regarding the two missing spent fuel rod segments. Spent fuel rods are required to be secured and safely stored. While it is premature to reach any conclusions about their location, we believe it is highly unlikely that the material is in the public domain. The NRC will continue to monitor and evaluate Entergy's response to this issue to assess actions to be taken.

Please feel free to contact us if you have any questions.

Sincerely,

/RA/

Jeffrey S. Merrifield
Acting Chairman

Enclosure:
Questions and Answers

Questions and Answers

QUESTION 1. "Please describe how it was discovered that the two Vermont Yankee fuel rods were missing? When were these fuel rods last accounted for?"

ANSWER 1.

In March 2004, the Nuclear Regulatory Commission's (NRC's) resident inspector at Vermont Yankee performed an inspection of Entergy's program for control of special nuclear material (SNM). The inspection identified concerns with the method used by Entergy to perform the physical inventory of the two spent fuel rod segments. On April 20, 2004, in response to the resident inspector's concerns with the inventory methods, Entergy performed an underwater video inspection of the storage container for the two spent fuel rod segments and found that they were not in the container, which was in the spent fuel pool. Entergy is conducting an investigation that includes an underwater video inspection of the spent fuel pool, a review of Entergy's radioactive material shipment records, personnel interviews, and identification of possible disposal locations. On May 19, 2004, Entergy reported to the NRC that the visual inspection of the spent fuel pool was essentially complete and that the underwater video inspection did not locate the spent fuel rod segments.

The two spent fuel rod segments are about one-half inch in diameter and about 9 inches and 17.75 inches in length, respectively. They had broken off from full-length fuel rods during repair work on a fuel assembly in 1980. According to Entergy documents prepared at the time, the segments were placed in a 5-gallon stainless steel container, which is fitted with two vertical stainless steel pipes to hold the fuel rod segments, and is stored on the bottom of the spent fuel pool. Entergy last physically verified the two spent fuel rod segments in January 1980. Entergy's records indicate that the two spent fuel rod segments were placed in the special container in the bottom of the spent fuel pool following the 1980 repair work on a fuel assembly.

The NRC has conducted materials control and accounting (MC&A) inspections at Vermont Yankee in the past. However, prior to the most recent inspection, the NRC had not performed an inspection at Vermont Yankee specifically to verify that the two spent fuel rod segments in question were in the spent fuel pool.

In general, the NRC conducts MC&A inspections to determine whether licensees have limited their possession and use of SNM to the locations and purposes authorized by their operating licenses. In addition, during these inspections, the NRC determines whether licensees have implemented adequate and effective programs to account for and control the SNM in their possession. Findings from MC&A inspections at power reactors prior to 1988 did not indicate that there were major deficiencies in power reactor licensees' MC&A programs. At that time, the NRC considered there was low risk of improper storage of spent fuel at a power reactor since physical and radiological characteristics of spent fuel made it highly unlikely that spent fuel could be safely removed from the fuel pool without proper equipment and procedures. Therefore, in 1988, the NRC chose to allocate inspection resources to other more risk-significant issues. In 2001, the NRC staff conducted a reexamination of MC&A vulnerabilities as part of a comprehensive review of the NRC's Safeguards and Security Program which was conducted in response to the terrorist activities of September 11, 2001, and in response to the report of two missing fuel rods at the Millstone Nuclear Power Station (Millstone) Unit 1.

Enclosure

In November 2000, the licensee for Millstone Unit 1 reported to the NRC that two fuel rods, each about 12 feet in length and containing about 300 curies of radioactive material, were missing from the spent fuel pool. As part of the lessons learned from that event, the NRC staff developed Temporary Instruction (TI) 2515/154, "Spent Fuel Material Control and Accounting at Nuclear Plants," dated November 26, 2003, to enhance the NRC's inspection of licensees' MC&A programs. The TI provides specific inspection guidance to NRC inspectors and consists of three phases. The first phase requires the NRC resident inspector to determine, through interviews, if a licensee has ever removed irradiated fuel rods from a fuel assembly. If the answer is yes, phase two of the TI is then implemented. Phase two of the TI determines, through detailed questions, if a licensee's MC&A program is adequate to account for items located in the spent fuel pool. If it is determined that a licensee's MC&A program has deficiencies, then phase three of the TI is implemented. Phase three is a much more detailed inspection of the MC&A program, which is conducted by experienced MC&A inspectors and includes verifying the location in the spent fuel pool of all spent fuel rods that have been separated from their parent fuel assemblies.

QUESTION 2. "What is the Commission doing to ascertain the whereabouts of the Vermont Yankee fuel rods? Please describe all investigative actions taken or planned to be taken."

ANSWER 2.

The NRC staff has closely monitored Entergy's actions and investigation since Entergy formally reported to the NRC staff on April 21, 2004, that the two spent fuel rod segments were not in their storage container in the spent fuel pool. The NRC is currently in the process of conducting a special inspection at Vermont Yankee to investigate the missing spent fuel rod segments. During the inspection, the NRC will review the results of Entergy's investigation, assess the root cause determination, determine if Entergy is in compliance with applicable regulations, and identify which findings may have generic implications. The charter for the special inspection was issued on April 30, 2004.[1]

QUESTION 3. "What is the Commission doing to obtain an inventory of all spent nuclear fuel at all nuclear reactors in the U.S.? If no such inventory is planned, why not, since it is clear from both the Vermont and Connecticut cases that this could be an industry-wide problem?"

ANSWER 3.

The NRC's regulations at Title 10 of the *Code of Federal Regulations* (10 CFR), Part 74, Section 74.19, require licensees to perform an annual physical inventory of all SNM in their possession, including spent fuel. Licensees are also required to maintain current records of spent fuel pool inventories. The NRC's regulations at 10 CFR 74.13 require nuclear reactor licensees to submit an annual report of their inventory of SNM to the Nuclear Materials

[1]The special inspection charter is available for public inspection at the NRC's Public Document Room, located at One White Flint North, File Public Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. It is also accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html, using Accession No. ML041240493.

Management and Safeguards System (NMMSS). NMMSS is operated for both the NRC and the Department of Energy. Also, 10 CFR 74.15 requires licensees to submit a report to NMMSS of any transfer of SNM.

The NRC is currently conducting inspections at all U.S. commercial nuclear power plants to determine if there are any MC&A issues at other plants. As previously stated, the NRC issued TI 2515/154 for this purpose.

QUESTION 4. "The licensee of the Millstone nuclear reactor was fined only $288,000 for its failure to keep track of its spent fuel. How much will Entergy be fined for its failure to keep track of the spent nuclear fuel at Vermont Yankee?"

ANSWER 4.

The NRC's special inspection is still in progress. It is premature to speculate on what enforcement outcome is appropriate.

QUESTION 5. "10 CFR 70.51(c) states that 'a power reactor licensee is required to establish, maintain and follow written material control and accounting procedures that are sufficient to enable the licensee to account for the special nuclear material (SNM) in its possession.' In light of the fact that Vermont Yankee is unable to account for the whereabouts of these two missing fuel rods, do you believe that the licensee has complied with this requirement? Why or why not?"

ANSWER 5.

The NRC is conducting a special inspection at Vermont Yankee and is still in the process of discovery. Part of the special inspection charter is to review Entergy's compliance with all applicable MC&A regulatory requirements. The NRC will formally evaluate and document the inspection findings when the inspection is complete. Note that 10 CFR 70.51(c) has been replaced by 10 CFR 74.19(b).

QUESTION 6. "10 CFR 70.51(d) states that a power reactor licensee is required to conduct a physical inventory of all SNM in its possession at intervals not to exceed 12 months.' Given the fact that the two fuel rods apparently were not identified as missing in any physical inventory conducted by Entergy, do you believe that Entergy has complied with this requirement? Why or why not?"

ANSWER 6.

As indicated in the response to Question 5 above, the NRC's special inspection is ongoing at Vermont Yankee. Part of the special inspection charter is to review Entergy's compliance with all applicable MC&A regulatory requirements. The NRC is still in the process of discovery. An annual inventory was being performed at Vermont Yankee by the licensee. However, the effectiveness of the inventory is in question. Note that 10 CFR 70.51(d) has been replaced by 10 CFR 74.19(c).

QUESTION 7. "According to the Commission's February 1, 2001 letter to me regarding the Millstone missing spent fuel case, (see http://www.house.gov/markey/Issues/iss nuclear ltr010201.pdf), a variety of civil and criminal penalties can be imposed for violations of Commission regulations, including fines of up to $100,000 per day prior to 1986 and fines of up to $110,000 beginning in 1986. What would be the maximum civil monetary penalty incurred by Entergy in this case, assuming full application of the $100,000-110,000 per day civil penalty mentioned in your letter?"

ANSWER 7.

As previously stated, the NRC's special inspection at Vermont Yankee is ongoing. The NRC has not arrived at any enforcement decisions, thus, it is premature to speculate on what enforcement outcome is appropriate.

In accordance with the NRC's Enforcement Policy, fines can be up to $100,000 per day for violations identified before 1996, $110,000 per day for violations identified between 1996 and October 4, 2000, and $120,000 per day for violations identified after that date. In general, the NRC considers civil penalties for commercial nuclear power plants if the violation was (1) a significant, willful violation, (2) a significant violation for impeding or impacting the regulatory process, or (3) a significant violation associated with actual consequences, such as a radiation overexposure to the public or plant personnel above regulatory limits. If a violation does not meet any of these descriptions, it is addressed through the NRC's Significance Determination Process (SDP) within the Reactor Oversight Process. If a violation associated with findings that the SDP evaluates as having "greater than very low" safety significance is identified, it would be cited in a Notice of Violation requiring a written response. Typically, in such cases, no civil penalty is assessed and the finding is entered into the Action Matrix to determine the type of inspection the NRC will conduct to verify that the finding was properly addressed. The more significant the inspection finding, the more extensive the NRC's inspection/verification process will be. The NRC reserves the use of discretion in assessing civil penalties for particularly significant violations.

QUESTION 8. "In your February 1, 2001 letter, the Commission stated that 'following the completion of the NRC's inquiry [into the Millstone matter], we will consider whether industry-wide generic action is warranted.' Did you conclude that industry-wide generic action was warranted? If so, what action? If not, why not, and will you take such action now that a second such case has been revealed?"

ANSWER 8.

Yes. The NRC decided that industry-wide generic action was warranted.

Based on lessons learned from the Millstone Unit 1 event, the NRC issued TI 2515/154. The NRC is currently conducting inspections at all U.S. commercial nuclear power plants in accordance with the TI to determine if there are any MC&A issues at other plants. The development of this TI began in 2001; however, the events of September 11, 2001, caused a delay in completing the TI.

The NRC will determine the need for further generic communications, inspection, enforcement or other regulatory actions based on the results of the ongoing special inspection at Vermont Yankee and the findings from the TI 2515/154 MC&A inspections at other facilities.

QUESTION 9. "In your February 1, 2001 letter, you said that it is unlikely that the two spent fuel rods were stolen, because 'The very high radiation level of the material makes theft difficult, dangerous, and very unlikely' and 'amount and chemical form of the fissile material contained in the two spent fuel rods make it unlikely, in our judgment, that the rods could be used to assist in the manufacture of a weapon.' However, the September 11th terror attacks have demonstrated that terrorists may be willing to commit suicide in order to cause harm to America, and may be willing to devote many years to the planning and execution of such an attack.

a) Have you evaluated the possibility that the fuel rods may have been stolen or diverted?
b) Isn't it possible that rather than trying to use the fissile material from these weapons for a nuclear explosive device or weapon, terrorists might want to use it for a crude radiological weapon, or 'dirty bomb' aimed at dispersing radioactive materials in a populated area?
c) What would be the worst-case public health, safety, and environmental consequences of detonation of a 'dirty bomb' fabricated from the two Vermont Yankee spent fuel rods?"

ANSWER 9(a).

Yes. As stated in response to Question 2 above, the NRC is in the process of conducting a special inspection at Vermont Yankee to investigate the circumstances surrounding the missing spent fuel rod segments. The NRC continues to believe that the missing spent fuel rod segments do not pose a threat to public health and safety as it is highly unlikely that the material is in the public domain. Given the security of the site and the extensive array of radiation detectors at the site, it is highly likely that the unaccounted for fuel segments are in a location designed to deal with radioactive waste. They could only have been removed from the site in heavily shielded containers, which would be shipped directly to other controlled, safe locations.

The NRC staff is confident that the radiation detectors in the area of the spent fuel pool would have alarmed if the spent fuel rod segments or any other highly radioactive material stored in the pool were inadvertently removed from the spent fuel pool in an unshielded container. Besides the radiation detectors located within the plant, there are additional radiation detectors located at personnel exit points. These detectors monitor each worker exiting the plant's protected area. These are very sensitive detectors designed to alarm at radiation levels slightly above natural background. These detectors help ensure that workers do not leave the site with contaminated clothing, equipment, or radioactive material.

Even if the spent fuel rod segments were inadvertently mixed with other radioactive waste that was stored in the spent fuel pool, and then removed from the site, they would be subject to rigorous controls and oversight. The waste is processed, packaged, and prepared for shipment in a specially shielded container. The shielded container is loaded onto a transport truck and a

radiation survey is performed to verify that the radiation levels from the container meet NRC and U.S. Department of Transportation (DOT) regulatory requirements. Once in a shielded container, the two spent fuel rod segments would not be detected by the required radiation survey as long as the radiation levels meet NRC and DOT regulatory requirements. However, rigorous shipping controls are used for packages containing radioactive waste to ensure that packages shipped from plants are received by the waste disposal facilities.

ANSWER 9(b).

First, we would note that there is a very small amount of fissile material in these rod segments, far less than needed for a nuclear weapon. Furthermore, extracting that material from the rod segments would require chemical processing capabilities generally only possessed by governments.

As for the use of these rod segments in a crude radiological dispersal device (RDD), note that the shorter-lived radionuclides have decayed significantly in the almost quarter century since the rod segments were in the reactor's core. The rod segments are also in a particularly non-dispersable form. If used in a crude RDD, it is most likely the rod segments would break up into relatively large discrete pieces, which would not travel far from the site of the explosion making them easily identifiable for retrieval. Contamination could extend over a few city blocks or more. Processing the material into a more dispersable form would require considerable effort and then could not be considered a crude RDD.

ANSWER 9(c).

If radioactive material typical of a fuel rod segment were used in a dirty bomb, the radioactive material could contaminate an area of a few city blocks or more, depending on the size of the explosive, the amount of radioactive material used, and weather conditions. It is unlikely that significant, immediate health effects or prompt fatalities would result, other than from the explosion itself. Over the long-term, people who were contaminated or exposed to elevated radiation levels may have a very small increased risk of cancer.

EA-04-120 - Cooper Nuclear Station (Nebraska Public Power District)

June 25, 2004

EA-04-120

Randall K. Edington, Vice
President-Nuclear and CNO
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - NRC SUPPLEMENTAL INSPECTION REPORT 05000298/2004-011 AND NOTICE OF VIOLATION

Dear Mr. Edington:

The Nuclear Regulatory Commission (NRC) conducted the onsite portion of a supplemental inspection at Cooper Nuclear Station from April 5-9, 2004, and an in-office portion April 12 through May 12, 2004. Inspection debriefs were held onsite on April 8, by telephone on April 14, and May 5, and an exit meeting was held onsite with your staff on May 12, 2004. The enclosed report documents the inspection findings, which were discussed with you and other members of your staff.

The NRC issued a White inspection finding in a letter dated March 24, 2004, "Cooper Nuclear Station - NRC Inspection Report 05000298/2004-009 Biennial Licensed Operator Requalification Inspection - Final Significance Determination for a White Finding." This finding involved a high failure rate on the licensed operator biennial requalification written examinations. The performance deficiency associated with this finding involved the failure to adequately implement the systems approach to training process required by 10 CFR 55.59, "Requalification." Failure to adequately implement the systems approach to training is notable because training and testing deficiencies resulted in a decline in licensed operator knowledge over time. The NRC found that this decline in operator knowledge was evident in both plant operating experience and biennial requalification examination performance.

This supplemental inspection was conducted to provide assurance that the root and contributing causes of the White inspection finding were understood and to provide assurance that the corrective actions were sufficient to address the causes, and prevent recurrence of the problems. Detailed observations, assessments, and conclusions of the inspection are presented in the enclosed inspection report. The inspection also reviewed aspects of the licensed operator requalification training program to determine if the program was implemented using a systems approach to training as defined in 10 CFR 55.4 and NUREG-1220, "Training Review Criteria and Procedures."

The inspection concluded that your root cause analyses of the finding was appropriately evaluated and understood. The corrective actions identified as a result of your evaluations addressed the root and contributing causes, and should adequately address correction of the requalification program weaknesses if the corrective actions are consistently implemented. However, the inspection also concluded that your extent of condition and extent of cause evaluations of the high failure rate were not completed at the time of the inspection, and that other areas of the root cause lacked in-depth evaluation, including the adequacy of operator knowledge and the establishment of objective criteria to evaluate effectiveness of the corrective actions. The inspection also concluded that the analysis and evaluation elements of a systems approach to training, described in NUREG-1220, were implemented with significant weaknesses, and that the evaluation element was inadequate during the 2-year requalification program cycle beginning February 2002.

The NRC has also determined that the failure to consistently implement all elements of a systems approach to training in the licensed operator requalification program is a violation of 10 CFR 55.59(c), as cited in the attached Notice of Violation. The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation would be considered escalated enforcement action because it is associated with a White finding, however, since the White finding was previously issued in NRC letter dated March 24, 2004, this Notice of Violation is not considered to be a separate escalated enforcement action.

Nevertheless, you are required to respond to this letter and should follow the instructions specified in the enclosed Notice of Violation when preparing your response.

The NRC also identified one finding that was evaluated under the risk significance determination process as having very low safety significance (Green). The NRC also determined that there was a violation associated with the finding. The violation is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy. The noncited violation is described in the subject inspection report. If you contest the violation or significance of the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station facility.

In a telephone conversation on June 25, 2004, Anthony Gody of my staff discussed the apparent violation of 10 CFR 55.59 (c) with Joe Waid, Training Manager. Mr. Waid indicated that Cooper Nuclear Station declined a predecisional enforcement conference and stated that no written response would be provided prior to issuance of the violation.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and 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).

Sincerely,

/RA/

Dwight D. Chamberlain, Director
Division of Reactor Safety

Docket: 50-298
License: DPR 46

Enclosure:
1. Notice of Violation
2. NRC Inspection Report 05000298-2004011

cc w/enclosures:
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ENCLOSURE 1

NOTICE OF VIOLATION

Nebraska Public Power District
Cooper Nuclear Station

Docket No. 50-298
License No. DPR-46
EA-04-120

During an NRC inspection conducted on April 5 through May 12, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

Section 10 CFR 55.59(c) provides, in part, that "The requalification program must meet the requirements of paragraphs (c)(1) through (7) of this section. In lieu of paragraphs (c)(2), (3), and (4) of this section, the Commission may approve a program developed by using a systems approach to training." Section 10 CFR 55.4 defines a systems approach to training as "a training program that includes the following five elements. . . ." Element (4) is "Evaluation of trainee mastery of the objectives during training."

Through Generic Letter 87-07 and the licensee's notification dated August 13, 1987, the NRC approved the licensee's requalification program, developed using a systems approach to training.

Cooper Training Program Procedure 201, "CNS Licensed/SRO Certified Personnel Requalification Program," Revision 25, Step 4.1.1 requires that, "Cycle examinations shall be used to evaluate comprehension of training subjects presented during LOR (licensed operator requalification) training. . . ." Step 2.1.7 defines a cycle written examination as, "A written exam to demonstrate proficiency on material covered during cycle(s) training." The licensee divided the biennial requalification training program into 12 training cycles, each of which was approximately 6 weeks in duration.

Contrary to the above, during the biennial requalification program period from February 25, 2002, through January 11, 2004, the licensee's use of cycle written examinations was not adequate to evaluate comprehension of training subjects presented during LOR training. During this biennial requalification program period, the licensee administered a total of three cycle written examinations. Two of the cycle examinations were administered following two cycles of training. The third cycle examination was administered following six cycles of training (a period of approximately 36 weeks) and failed to test comprehension of several training subjects, including, for example, changes to the severe accident management guidelines and modifications to the reactor vessel level control system.

This violation is associated with a White significance determination process finding that was previously issued in an NRC letter of March 24, 2004.

Pursuant to the provisions of 10 CFR 2.201, Nebraska Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, and a copy to the NRC Resident Inspector at Cooper Nuclear Station of this Notice of Violation, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-04-026" and should include for each violation: (1) the reason for the violation or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice of Violation, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within 2 working days.

Dated this 25th day of June 2004

VERMONT YANKEE'S SAFETY MARGINS WILL BE "`SIGNIFICANTLY" REDUCED in three areas if NRC approves a requested 20% power uprate, according to two individuals who said they have done an in-depth review of plant documents and NRC regulations. "The technical terms for these three criteria are defense in depth, protection against single failures and the independence of three barriers preventing the release of radioactive material in the event of an accident," wrote Arnie Gundersen, who has been working for New England Coalition, a group that wants to shut Vermont Yankee, and Paul Blanch, an energy consultant. Gundersen and Blanch outlined their concerns in a letter today to the Vermont Public Service Board, which held a meeting on the proposed uprate. Blanch criticized NRC's decision not to publish its review criteria for the upcoming engineering inspection, saying it appears the agency is avoiding "openness."
(*NUCLEAR NEWS FLASHES - June 28, 2004*)

--AREVA HAS INSTALLED THE FIRST DIGITAL "THYRIPART" EMERGENCY DIESEL generator excitation system in the U.S. at TXU Power's Comanche Peak-1. Areva's joint subsidiary with Siemens, Framatome ANP, described the system as "a load dependent static excitation system powered by the generator voltage and current" that includes a passive analog power circuit and a digital voltage regulation circuit. "While the two circuits are designed to work together, the analog circuit alone is capable of keeping the generator output within the design basis limits," Areva said today. It also replaced 12 emergency diesel generator protection relays with a single digital protection relay. "Electrical system upgrades are increasing in the power industry because of equipment obsolescence and a lack of replacement parts," Areva said.
(*NUCLEAR NEWS FLASHES - June 28, 2004*)

--NRC HAS APPROVED THE RETRANSFER OF U.S.-ORIGIN URANIUM TO JAPAN for research into how uranium oxide and mixed-oxide (MOX) would react at high- burnup rates. An NRC notice in today's Federal Register said roughly 594 grams of the material would be transferred from the Studsvik Research Center in Sweden to the Japan Atomic Energy Research Institute (Jaeri) as reactor fuel rod segments that Jaeri will subject to a post-irradiation examination. Jaeri will take ownership of the material when it leaves Sweden. The material will be handled as waste and stored in Japan once that work is completed
(*NUCLEAR NEWS FLASHES - June 28, 2004*)

Industry looks to commission for help on 50.69 rule

Inside NRC
Volume 26 / Number 12 / June 14, 2004

The industry is growing more convinced that it will have to appeal to the commission to fine-tune the staff's proposed final rule that would risk-inform the agency's special treatment requirements for procuring safety-related systems, structures, and components (SSCs)—10 CFR 50.69.

At a June 2 briefing to the Advisory Committee on Reactor Safeguards (ACRS), the NRC staff said it is on schedule to deliver a final rulemaking package to the commission by June 30. All program offices had concurred with the final rule package, although the agency's Office of the General Counsel was still "actively looking" at the final package, according to Timothy Reed of the Office of Nuclear Reactor Regulation. Reed told the ACRS that the staff expects the commission to make the rulemaking package public and to seek feedback.

Some NRC sources have indicated that the staff consensus on the final rule was fragile. "We've gone about as far as we can go, without [that consensus] breaking apart," said one source. The industry has not seen any of the final rule language, but fears that the staff, to get consensus, has "muddied the waters" to the point that many licensees may choose not to use the new and voluntary 50.69 rule, said one industry source. "We hope the commission doesn't have patience" for this staff approach, he added.

The NRC staff told the ACRS June 2 that it believes a regulatory guide (RG 1.201) that would endorse the Nuclear Energy Institute's (NEI) guidance on categorization of SSCs based on their safety significance (NEI 00-04) should be

issued for trial use based on lessons learned from pilot projects. One pilot (at Dominion's Surry) will test a new regulatory guide on the quality of probabilistic risk assessments (PRA) in the context of categorizing SSCs under the new 50.69 rule. Sources said the ACRS will probably endorse the staff's recommendation.

NEI's Anthony Pietrangelo told the ACRS June 2 that a trial-use period for RG 1.201 would send "the wrong message." The success of 50.69, Pietrangelo said, depends on broad application throughout the industry.
But the trial-use issue may not be the most significant disagreement NEI has with how the NRC staff has written the final rulemaking package.

Although there is significant agreement between the NRC staff and NEI on the wording of requirements in the key NEI SSC categorization guide, a May 24 letter from the staff indicated that there are still some key technical issues that need to be resolved or clarified before it will endorse that guide (NEI 00-04) with few, if any, exceptions.

A key difference emerged regarding NEI's proposed guidance for monitoring failures of low safety significant SSCs that had been procured under the relaxed quality assurance (QA) requirements of the 50.69 rule. The NEI guidance suggested that some level of increased component failure could be tolerated (although it does not expect such increases to occur). It proposed that QA treatment of the component not be altered unless the number of failures after 50.69 implementation was greater than the multiplier used in the risk sensitivity study that was part of the original categorization process.

The NRC staff called this approach "overly simplistic and technically not acceptable." The staff went on to say that "an acceptable process would need to have a focused cause analysis when a RISC-3 [low safety significant] SSC failed to determine if this failure was due to the reduction in treatment and/or an indication of a potential common cause failure or degradation mechanism. If there is indication that one of these factors is the cause of the failure, then the applicant or licensee should have a process for immediately expanding testing to similar SSCs to demonstrate their functionality and for entering a corrective action to the treatment and/or categorization processes."

The staff said that until NEI came up with a "technically defensible approach," the NRC staff would "review the applicant's or licensee's approach and process as part of the application requesting to implement 50.69."
—Michael Knapik, Washington

PSEG presents plans for overhaul of Salem-Hope Creek safety culture

Inside NRC
Volume 26 / Number 13 / June 28, 2004

PSEG Nuclear this month briefed NRC on the company's plans at Salem and Hope Creek to follow up on the findings of three recent assessments of the safety conscious work environment (SCWE) and related issues at the two plants, saying work management and the corrective actions will be two areas of principal focus.

The corrective action program is tied to the site's SCWE, the assessments suggest, because the company's willingness to defer needed maintenance sends a signal that keeping the plant in topnotch condition is not a priority.

One of the steps PSEG said it would take is to institute a corrective action review board, to ensure that the required actions are carried out in a timely and conscientious way.

Also highlighted at several points in PSEG's presentation during the June 16 meeting in Swedesboro, N.J. was the need for improved communication. Frank Cassidy, the president/chief operating officer (COO) of PSEG Power, the parent of PSEG Nuclear, said that one example of change in that area would be more frequent visits by corporate management to the plant site. Cassidy said he has started spending about one day per week on the Lower Alloways Creek, N.J. Salem-Hope Creek site,

rather than in his office in Newark, N.J. The goal of such visits, he said, is to increase each side's understanding of the other's concerns.

The so-called "corporate-site interface" was one area discussed in an "independent assessment team" (IAT) report commissioned by PSEG in response to a Jan. 28 letter from the NRC expressing concerns about SCWE at the plant. The IAT said that in the interviews it conducted with plant personnel, no one reported "direct knowledge" of a corporate decision to defer long-standing equipment issues as a cost-cutting measure or to emphasize production and schedule over conservative decision-making. Nevertheless, the IAT said, corporate policies and practices may have had the "unintended consequence" of deferring maintenance or focusing too heavily on production and scheduling. One of the IAT's recommendations was to mprove the direct communications link from corporate to the site."

Similar issues apparently have come up with regard to actions by on-site managers. James O'Hanlon, the former Dominion Energy president/COO who is leading the IAT, recounted a case in which a senior plant official had a three hour meeting with a member of the operations department about keeping the plant shut down. O'Hanlon said, "People may be taking an action for a reason they thought was right," but it was "clear the message was quite different." In an interview the next day, A. Christopher Bakken, PSEG Nuclear's senior vice president for nuclear operations, offered a similar assessment of the three-hour meeting. He said the decisions by plant management were correct, but the way they were handled left "residual concerns" among plant staff.

Regarding the perception that management was overemphasizing production, NRC Region I Administrator Hubert Miller said in an interview, "We worry about perceptions." Moreover, he said, the equipment problems "are not [just] perceived—they're real." Management, he said, has to "reearn" the confidence of the plant staff and make them feel "not just empowered but required" to raise safety issues.

Charting progress

During the meeting, Bakken said that as part of PSEG's reform effort, it would use certain "metrics" to chart its progress and would publicly release data on the company's progress against those yardsticks. Michael Brothers, PSEG Nuclear's vice president for site operations, said the SCWE metrics would include surveys of employees' perceptions of management commitment, supervisor communication effectiveness, and other issues. But the measurements also will include categories such as tallies of nuclear condition report operations that are overdue and unplanned entries into limiting conditions of operations, he said.

A. Randolph Blough, the director of the division of reactor projects in Region I, said he was glad to see an extensive use of metrics, but warned PSEG against "becom[ing] so obsessed with numbers" that elements that are not easily quantified are neglected. Miller reinforced that point, saying that at other plants that have had to make changes on the scale of those that PSEG is now attempting, "we have seen tendency to focus on metrics that are easy to track." At such plants, Miller said, there is sometimes a tendency to underestimate the amount of work on the "softer" issues—that is, those related more to improving environment and attitudes than to repairing hardware problems. "We'll be suspicious of anything that involves quick fixes," he said. In too many cases, he said, NRC has seen plants rapidly make progress at first and then "fall back." Miller said he wants to see "steady foundations," even if that means a downturn in some metrics in the short term.

"There is a legacy here that is quite long—both on the hardware side and the people side," he said. Both Salem units were in NRC-monitored shutdown in the mid-1990s, and plant management pledged to make safety-related improvements in the way it ran the plants. But in the letter the NRC sent PSEG this January, the agency, citing interim results of "a special review" launched in late 2003, referred to SCWE-related information from Salem to Hope Creek "received in various allegations and inspections over the past few years." In the agency's last three semi-annual assessments of the plants, NRC found a "cross-cutting issue" in PSEG's problem identification and resolution, including its corrective action.

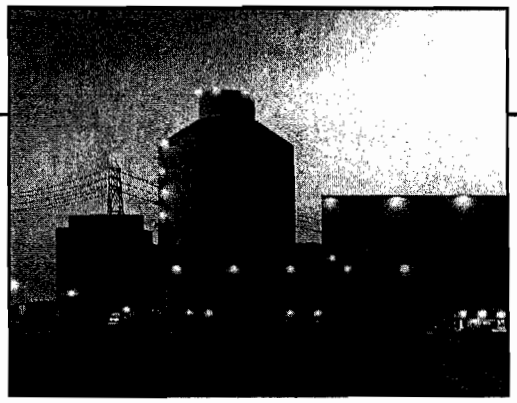
Bakken acknowledged the plants' history, saying PSEG was planning to release the data on its efforts so that NRC and the public could "assess results" rather than relying on promises. "There have been lots of plans at meetings like this over the past few years," he said.

Bakken said PSEG would submit a "commitment letter" to NRC by June 25. David Lochbaum of the Union of Concerned Scientists said that since NRC views such commitments as unenforceable, the agency should issue a confirmatory order or a confirmatory action letter.

Miller commended PSEG for its openness in making so much information public. Lochbaum also has praised PSEG for putting a great deal of information—including the IAT report and surveys by the Utilities Service Alliance and Synergy Consulting Services Corp.—into the public domain. But at the same time, Lochbaum has criticized PSEG's record and called on NRC to shut the plants down. Miller, in comments during the meeting and the interview, said NRC was not going to take that step "based on what we see now." But he said, "If at any time, there is a condition that crosses the line, as we judge it, we will step up." As for the criticism that the NRC had not been forceful enough in pressing PSEG to make the necessary changes, he said he believed the agency was having an impact. "Just look at our inspections the last few years," he said, characterizing them as "aggressive" and "way above average." But he acknowledged, "I completely understand why people are skeptical."
At the meeting, Kymn Harvin, a former PSEG nuclear employee who has alleged that she lost her job in retaliation for her safety advocacy, said that PSEG should shut down the plant voluntarily and "[u]se the time to fix what is broken, in the culture and equipment." She said plant culture has tolerated mediocrity—or even encouraged it, as long as it is cost-effective—but that a voluntary shutdown would "clearly demonstrate a willingness to replace mediocrity with excellence."

She also asked why the IAT did not speak with her and other whistle-blowers and posed several sharp questions to Bakken and the other PSEG officials. Bakken said he was not in a position to answer them, explaining later that he was constrained in what he could say because Harvin is pursuing litigation against PSEG. But Harvin said the issues she raised were broader ones, not specifically tied to her case.

Bakken said he had "reached out" twice to talk to her, but that she had declined. She confirmed that point, citing the pending litigation.
—Daniel Horner, Swedesboro, N.J. and Washington



AP1000

AP1000 Safety

Westinghouse Electric Company

Presentation to

Advisory Committee on Reactor Safeguards

July 7, 2004

BNFL Westinghouse

AP1000

AP1000 Safety

- AP1000 Comfortably Meets NRC Regulations and Industry Safety Standards
  - **Both deterministic and probabilistic**

BNFL Slide 2 Westinghouse

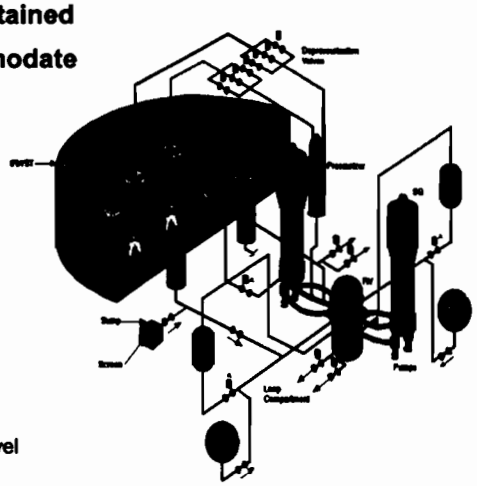
AP1000 Approach to Safety

- **Passive Safety-Related Systems**
  - Use "passive" process only, no active pumps, diesels, ...
    - One time alignment of valves
    - No support systems required after actuation
      - No AC power, cooling water, HVAC, I&C
  - Greatly reduced dependency on operator actions
  - Mitigate design basis accidents without nonsafety systems
  - Meet NRC PRA safety goals without use of nonsafety systems
- **Active Nonsafety-Related Systems**
  - Reliably support normal operation
    - Redundant equipment powered by onsite diesels
  - Minimize challenges to passive safety systems
  - Not required to mitigate design basis accidents

BNFL Westinghouse

AP1000 Passive Core Cooling System

- **AP600 System Configuration Retained**
- **Capacities Increased to Accommodate Higher Power**
  - PRHR HX capacity increased
    - Larger pipe, more/longer tubes
  - CMT volume & flow increased
  - ADS 4 flow increased
    - Larger pipe/valves
  - IRWST injection flow increased
    - Larger pipe/valves
  - Cont. recirc. flow increased
    - Larger pipe/valves, higher flood level
- **System Performance Maintained**

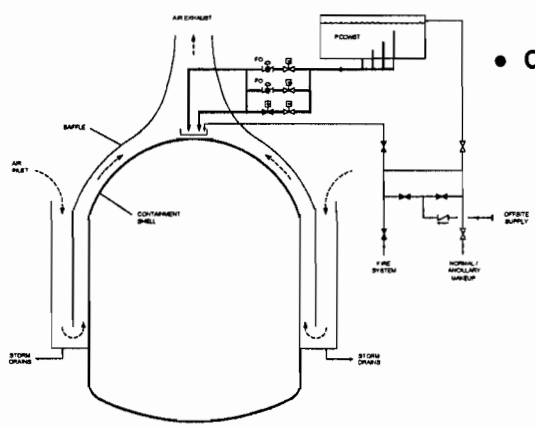


BNFL Slide 4 Westinghouse

Containment Recirc Debris

- **AP1000 Has Robust Containment Recirc Design**
  - Delayed recirc (typically 5 hr), deep flood levels (>10' above screen)
  - Low velocities in containment pools and at screen face
  - No spray to wash down containment surfaces and increase velocities
  - Tall recirc screens (10-13') located well above floors (2')
  - Horizontal plates located just above recirc screens
  - Folded screen designs with large surface area (140 ft2 ea)
  - Cross-connected recirc screens so both always operate
  - Metal reflective insulation used in LOCA blowdown damage zone
  - High density, nonsafety coatings used inside containment
- **COL Item Requires Cleanliness Program to Limit Debris**
- **COL Item Addresses Anticipated New Information**
  - Site specific resident debris data and chemical corrosion test results

Passive Containment Cooling



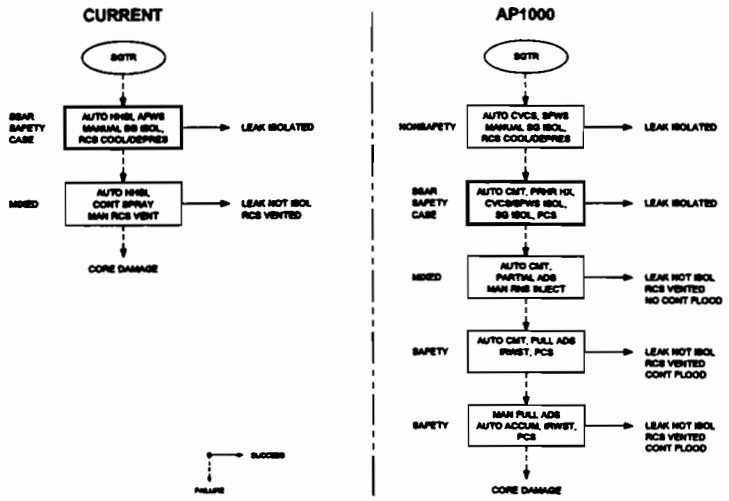
- **Same Configuration as AP600**
  - Except added 3rd diverse valve
    - Addresses T&H uncertainty air cooling
- **Capacities Increased**
  - Containment volume & design pres
  - PCS Water Storage Tank
    - Provides 72 hr drain
      - Afterwards use on/offsite water
      - Air only cooling prevents/delays failure in PRA sequences
    - Flow decreases with time
      - Uses 4 standpipes
  - PCS Flow Rates
    - Same high initial flow
      - Rapidly forms water film
      - Effectively reduces cont pressure
    - Later flows increased
      - Match decay heat

AP1000 Safety Margins - deterministic

	Typical Plant	AP600	AP1000
• Loss Flow Margin to DNBR Limit	~ 1 - 5%	~16%	~19%
• Feedline Break (°F) Subcooling Margin	>0	~170	~140
• SG Tube Rupture	Operator actions required in 10 min	Operator actions **NOT** required	Operator actions **NOT** required
• Small LOCA	3" LOCA core uncovers PCT ~1500°F	< 8" LOCA **NO** core uncovery	< 8" LOCA **NO** core uncovery
• Large LOCA (°F)	2000 - 2200	1676	2124
• ATWS, Pres (psig) UET (% core life)	3200 5-10%	3200 10%	2800 0%

BNFL Slide 7 Westinghouse

AP1000 Has More Levels of Defense (SG Tube Rupture Example)



BNFL Westinghouse

AP1000 CDF and LRF Summary

- **Meets US NRC Safety Goals with High Margin & Low Uncertainty**
  - Demonstrates effectiveness of passive safety features
    - Reduced dependency on operator actions and nonsafety features
    - Low safety risk from floods and fires
  - Severe accidents addressed by design

	Core Damage Frequency		Large Release Frequency	
	At-Power	Shutdown	At-Power	Shutdown
Internal Events	2.41E-07 /yr	1.23E-07 /yr	1.95E-08 /yr	2.05E-08 /yr
Internal Floods	8.80E-10 /yr	3.22E-09 /yr	7.10E-11 /yr	5.40E-10 /yr
Internal Fires	5.61E-08 /yr	8.52E-08 /yr	4.54E-09 /yr	1.40E-08 /yr
Sub-Totals	2.98E-07 /yr	2.11E-07 /yr	2.41E-08 /yr	3.50E-08 /yr
Grand-Totals	5.09E-07		5.92E-08	
NRC Safety Goals	1 E-4		1 E-6	

Organic Iodine Production

Westinghouse Electric Company

Presentation to

Advisory Committee on Reactor Safeguards

July 7, 2004

AP1000

Organic Iodine Production (Issue #6)

- **Acidification of Steam Condensate Draining Down Containment Shell Could Lead to Increased Production of Organic Iodine**
- **AP1000 Can Accommodate Uncertainties in Iodine Production**
  - AP1000 Meets NRC Safety Goals with Significant Margin

	NRC Goal	AP1000	Ratio
Risk Prompt Fatality	5.0E-07	8.4E-11	0.02%
Risk Latent Cancer Fatality	2.0E-06	8.6E-10	0.04%

  - A small amount of CsOH maintains film pH > 7 - no dose impact
    - Only 270 g, 0.1% of available CsOH needed to neutralize acids
  - Even without any CsOH, DBA dose criteria still met
    - Independent of acid production, transport
    - Organic iodine in containment increases from 0.15% to < 0.33%
    - Largest impact is on Control Room dose
      - Dose increase < 5.6%

AP1000

CsOH Expected to Neutralize Acids

- **Severe Accident Sequence Considered**
  - 3BE-1, representative DVI LOCA with failure of 2 ADS 4 vent paths
- **Water Film Residence Times Calculated**
  - Varies from 40 to 230 sec over first 10 hr after accident
- **Additional Acid Sources Considered**
  - Nitric acid from irradiation of water containing dissolved air
  - Hydrochloric acid from cable jacket (Hypalon) degradation
- **CsI Deposition Calculated**
  - Varies over first 10 hr after accident
  - Deposition essentially complete in this time period
- **Small Amount of CsOH Would Neutralize This Acid**
  - 270 g or 0.1% of available CsOH would raise pH > 7
  - Would prevent production of additional elemental / organic iodine
- **No Dose Impact**

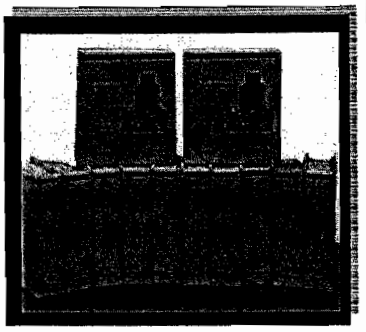
Without CsOH, Dose Impact Small

- **All Iodine Transported to Containment Water Film is Assumed to Instantly Become Elemental Iodine**
  - Acid production / transportation are unimportant
- **Elemental Iodine is Assumed to Instantly Partition into Aqueous / Gaseous Concentrations Based on the Water Film Temperature**
  - Water film is conservatively assumed to be saturated, maximizes I_2 gas
- **3% of Elemental Iodine in Containment Atmosphere is Converted into Organic Iodine**
  - Organic iodine increases from 0.15% to < 0.33%
- **Impact on Doses is Accommodated by Margins to Limits**

Site Boundary	24.7 rem increases to < 24.71 rem
LPZ	22.8 rem increases to < 23.16 rem
Control Room	4.8 rem increases to < 5.07 rem

AP1000 Design Certification Nearing Completion

Past Milestones	Target Schedule	
1. W Submits Application	3/28/02	☑
2. Staff Issues Requests for Additional Info (RAI)	9/30/02	☑
3. W Provides Responses to All RAI's	12/2/02	☑
4. NRC Identifies Potential Open Items	2/28/03	☑
5. W Addresses Potential Open Items	4/15/03	☑
6. NRC Issues Draft Safety Evaluation Report	6/16/03	☑
Future Milestones	**NRC Target**	
7. NRC Issues Final SER	September 2004	
8. NRC Issues Final Design Approval	September 2004	
9. NRC Issues Design Certification	August 2005	



AP1000 Review Status

July 7, 2004

ACRS Full Committee Meeting

John Segala, Senior Project Manager
Office of Nuclear Reactor Regulation

Purpose

- Provide overview of the staff's review
- Provide current status of the AP1000 project
- Discuss major schedule milestones
- Discuss two ACRS interim letter issues

Previous Review Milestones

- March 2002 - Completed pre-application review
- March 28, 2002 - Westinghouse (W) submitted DC application
- June 25, 2002 - NRC accepted the application for docketing
- June 16, 2003 - NRC issued DSER with 174 Open Items
- May 18, 2004 – NRC provided responses to the issues in the ACRS interim letter
- May 25, 2004 – Sent advanced copy of FSER to ACRS

Past ACRS Meetings

Subcommittee Meetings

Thermal Hydraulic Phenomena
5 Meetings
March 2001- February 2004

Future Plant Designs
3 Meetings
February 2002 - June 2004

Reliability and Probabilistic Risk Assessment
1 Meeting
January 2003

Full Committee Meetings

9 ACRS Meetings

August	2000 (Pre-Application)
April	2001 (Pre-Application)
March	2002 (Pre-Application)
November	2002
February	2003
April	2003
October	2003
March	2004
June	2004
July	2004

Remaining Schedule Milestones

- July 17, 2004 – ACRS issues final letter
- August 6, 2004 – Division Directors Concurrence
- August 13, 2004 – OGC no legal objection
- August 30, 2004 - EDO memo to Commission w/FSER/FDA attached
- September 13, 2004 – FSER/FDA issued
- December 2005 – Final Design Certification Rule issued

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Sun, Summer
Talbot, Frank
Tardiff, Albert
Throm, Edward
Trehan, Narinder
Unikewicz, Steven
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Ward, Leonard
Wilson, Jerry
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Contractors

- Brookhaven National Laboratory
  - Mechanical and Civil/Structural Engineering
- Carl J. Costantino Engineering Consultants
  - Structural and Earthquake Engineering
- Energy Research, Inc.
  - Severe Accidents
- Information Systems Laboratories, Inc.
  - RELAP5 Input Development
- Sandia National Laboratory
  - Aerosol Removal

RAIs

- We issued 742 RAIs
  - General - **3**
  - Mech. Eng - **70**
  - Structural Eng. - **19**
  - Seismology - **23**
  - Hydrol. and Meteor - **5**
  - Geotech. Eng. - **3**
  - Inservice Inspection - **3**
  - Component Integrity - **29**
  - Materials Application - **12**
  - QA and RAP - **8**
  - Emergency Preparedness - **3**
  - Containment Systems - **11**
  - Technical Specifications - **53**
  - ITAAC - **1**
  - Initial Test Program - **18**
  - Fire Protection - **11**
  - Chemical Technology - **4**
  - Auxiliary Systems - **22**
  - Instrumentation & Controls - **48**
  - Electric Power - **15**
  - Reactor Systems - **189**
  - Meteorology - **8**
  - Effluent Treatment - **11**
  - Radiological Impact - **13**
  - Radiation Protection - **11**
  - Human System Int. - **44**
  - PRA - **99**
  - Generic Issues - **6**

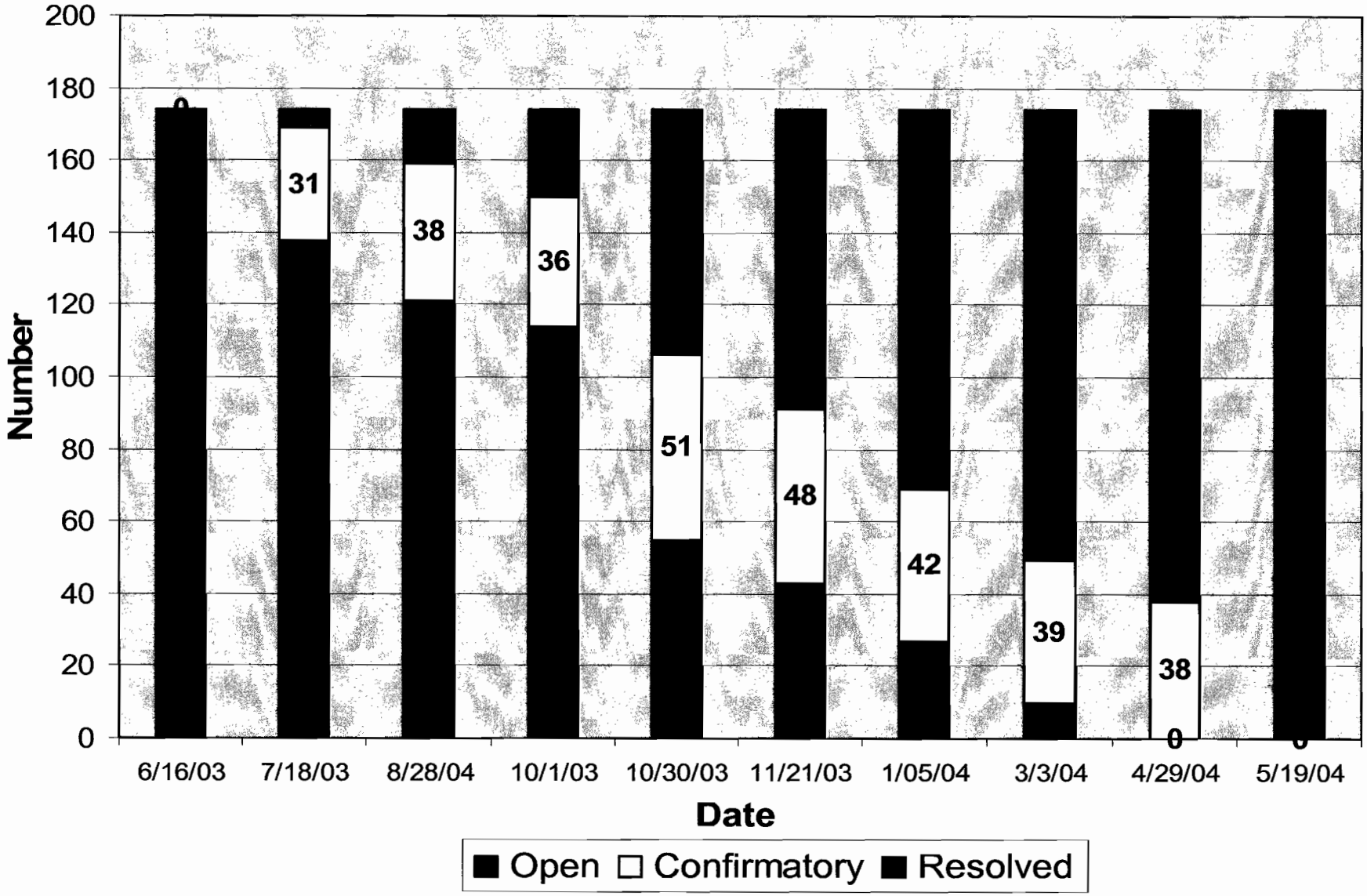
DSER Open Items

174 Open Items

- (as compared to over 1300 for AP600 DSER)

- Chapter 1 (Introduction) - - - - - - - - - - - - - - 3
- Chapter 2 (Site Envelope Char) - - - - - - - - - 6
- Chapter 3 (Structures, Comp., Equip.) - - - 30
- Chapter 4 (Reactor) - - - - - - - - - - - - - - - - - 3
- Chapter 5 (Reactor Coolant System) - - - - 3
- Chapter 6 (Engineered Safety Features) - - 9
- Chapter 7 (I & C) - - - - - - - - - - - - - - - - - - - 0
- Chapter 8 (Electric Power Systems) - - - - - - 1
- Chapter 9 (Auxiliary Systems) - - - - - - - - - - 7
- Chapter 10 (Steam and Power Conv.) - - - - 3
- Chapter 11 (Radioactive Waste Man.) - - - - 0
- Chapter 12 (Radiation Protection) - - - - - - 0
- Chapter 13 (Conduct of Ops) - - - - - - - - - - - 3
- Chapter 14 (Verification Progs) - - - - - - - - - 43
- Chapter 15 (Transient & Acc. Anal.) - - - - - 6
- Chapter 16 (Technical Specs) - - - - - - - - - - 3
- Chapter 17 (Quality Assurance) - - - - - - - - - 5
- Chapter 18 (Human Factors) - - - - - - - - - - - 7
- Chapter 19 (Severe Accidents) - - - - - - - - - 36
- Chapter 20 (Generic Issues) - - - - - - - - - - - 2
- Chapter 21 (Testing & Comp Code Eval.) - - 4
- Chapter 22 (RTNSS) - - - - - - - - - - - - - - - - - 0
- Chapter 23 (Review by the ACRS) - - - - - - 0
- Chapter 24 (Conclusions) - - - - - - - - - - - - - 0

AP1000 DSER Open Item Status



ACRS Interim Letter Issues

- Issue 1 - Automatic Depressurization System (ADS)-4 Squib Valve Function
- Issue 2 - Assurance of Long-Term Cooling (Strainer Blockage)
- Issue 3 - Code Deficiencies
- Issue 4 - Range of Pi-Group Values
- Issue 5 - In-Vessel Retention/Fuel-Coolant Interactions
- Issue 6 - Organic Iodine Production
- Issue 7 - Catastrophic Failure of the Steel Containment

Issue 2 - Assurance of Long-Term Cooling (Strainer Blockage)

- <u>ACRS Interim Letter Issue</u>:
  - AP1000 is a robust design to prevent screen blockage.
  - ACRS recommended ITAAC to ensure compliance with GSI 191
- <u>Staff Conclusion</u>:
  - Robust design which is less susceptible to debris blocking of the screens
  - Regulatory process exists to address significant adverse findings from continuing resolution of GSI-191

Open Items

- <u>6 DSER open items related to</u>:
  - Debris loading of IRWST screens and recirculation screens
  - Debris through reactor coolant system break
  - Chemical effects

Robust Screen Design

- Recirculation screen robust design to prevent screen blockage
  - Folded screen design with large surface area (140 ft^2)
  - Cross-connection pipe between the two recirculation screens
  - Tall recirculation screens (10 ft and 13 ft)
  - Bottom of recirculation screens 2 ft above the floor
  - Horizontal steel plates (10 ft in front and 7 ft to its side) located above recirculation screens
  - Screens protected by a trash rack
  - Low transport velocities in containment pool
  - Low flow velocities at the screen surface

Robust Screen Design (cont.)

- No safety-related containment sprays
- Metal reflective insulation used in LOCA blowdown damage zone (20/45 inside pipe diameters)
- High density, nonsafety coatings used inside containment
- Long time (up to 5 hours) before recirculation
- Deep containment floodup levels (~ 25 ft)
- Water level at beginning of recirculation is about 10 ft above top of recirculation screens (i.e. floating debris will be clear of screens)
- Backflow initially through recirculation screens due to higher water level in IRWST

ITAAC

- ITAAC verifies as-built screen design
  - location of the plates above containment recirculation screens
  - screen surface area
  - location of bottom of containment recirculation screens
  - type of insulation
  - dry film density of the coatings

COL Action Items

- COL applicant will develop a containment cleanliness program to limit debris
- COL applicant will perform an evaluation consistent with RG 1.82, Rev. 3 to confirm adequate long term cooling
  - Site specific resident debris
  - Post accident water chemistry
  - Applicable research and testing

NRC Review

- Staff review based on current state of knowledge
- Staff plans to:
  - Issue FSER and FDA on September 13, 2004
  - Complete design certification rulemaking by December 2005
  - Complete GSI 191 by December 2007

NRC Regulatory Change Process

- If new information is identified post FDA
  - Pre-design certification rulemaking:
    - 10 CFR Part 52, Appendix O, Item 5 and 10 CFR 50.109
      - Adequate protection, or
      - Compliance backfit
    - COMSECY 94-003
  - Post-design certification rulemaking/pre-COL:
    - Generic Backfit – 10 CFR 52.63(a)(1)
      - Adequate protection, or
      - Compliance backfit
  - Post-COL application:
    - Plant Specific Orders - 10 CFR 52.63(a)(3)
      - Adequate protection, or
      - Compliance backfit, and
      - Special Circumstances

Screen Design Conclusion

- Robust design which is less susceptible to debris blocking of the screens
- Regulatory process exists to address significant adverse findings from continuing resolution of GSI-191

Issue 6 - Organic Iodine Production

- <u>ACRS Interim Letter Issue</u>:
  - Water film pH determines iodine behavior
    - pH < 7 leads to production of elemental iodine some of which is subsequently converted into organic iodine
    - To prevent organic iodine production the film pH should be maintained above 7

Organic Iodine Production (cont.)

- W calculations determined:
  - Film pH is maintained above 7, assuming the amount of CsOH present in the DBA source term
  - A minimum of 270 g of CsOH (0.1% of available CsOH) is sufficient to keep pH above 7
  - The DBA dose criteria still met, even if assume no CsOH present
    - Increased amount of assumed organic iodine in containment from 0.15% to 0.33%
- Staff audited W calculations
  - Staff found the calculations to be acceptable
  - Staff agreed with W conclusions

AP1000 Summary

- All DSER open items resolved
- All ACRS issues addressed
- NRC staff still on schedule to issue FSER by September 13, 2004
- Questions or comments?

Background Slides

COL Action Items

- DCD Section 6.3.8.1, "Containment Cleanliness Program"
  - The Combined License applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages.
  - The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation consistent with COL item 6.3.8.2.

COL Action Items (cont.)

- DCD Section 6.3.8.2, "Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA"
  - The Combined License applicants referencing the AP1000 will perform an evaluation consistent with Regulatory Guide 1.82, revision 3, to demonstrate that adequate long-term core cooling is available considering debris resulting from a LOCA together with debris that exists before a LOCA.
  - As discussed in DCD subsection 6.3.2.2.7.1, a LOCA in the AP1000 does not generate fibrous debris due to damage to insulation or other materials included in the AP1000 design.

    The evaluation will consider resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program.
  - The determination of the characteristics of such resident debris will be based on sample measurements from operating plants.
  - The evaluation will also consider the potential for the generation of chemical debris (precipitants).
  - The potential to generate such debris will be determined considering the materials used inside the AP1000 containment, the post accident water chemistry of the AP1000, and the applicable research/testing.

(4)

April 20, 2004

Dr. Susan G. Sterrett, Assistant Professor
Department of Philosophy
Duke University
201 C West Duke Building
Durham, NC 27708

SUBJECT: RESPONSE TO CONCERNS ABOUT THE AP1000 DESIGN CERTIFICATION

Dear Dr. Sterrett:

The purpose of this letter is to respond to your concerns regarding the NRC's design certification review of the AP1000 advanced reactor plant. Specifically, in your letters dated July 30 and July 31, 2003, you expressed concerns with the AP1000 fluid systems design and quality assurance (QA) procedures and the impact of solar radiation heat on the ultimate heat sink. Both of these letters were addressed to the Nuclear Regulatory Commission (NRC) Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Future Plant Designs. We appreciate your initiative in writing these letters and letting us know of your concerns regarding AP1000 design certification.

Public involvement is a key element in NRC's reactor licensing process. The NRC is committed to enhancing public confidence, and we regard public outreach as a top priority in the work we do. Meetings with the licensees or applicants on safety issues are typically open to the public and documented for public access. We wish to make your letters (as well as all electronic communications between you and the NRC staff) publically available in the Agencywide Documents Access and Management System (ADAMS). If you have any concerns or reservations about your letters, and e-mail communications being made available to public, please contact Raj Anand by phone at 301-415-1146 or by e-mail to rka@nrc.gov by April 23, 2004.

As you are aware, the NRC is responsible for licensing and regulating the operation of nuclear power plants. The following provides a general overview of the standard design certification process. The NRC certifies and approves a standard plant design through a rulemaking, independent of a specific site. An application for a standard design certification must contain information and proposed inspections, tests, analyses, and acceptance criteria (ITAAC) for the standard design. Additionally, the application must demonstrate how the applicant complies with the Commission's applicable regulations. The ACRS reviews each application for a standard design certification, together with the NRC staff's safety evaluation report, in a public meeting. Upon determining that the application meets the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations, the Commission utilizes the rulemaking process to issue a standard design certification in the form of a rule which becomes an appendix to the 10 CFR Part 52 regulations. In addition to participating in the design certification rulemaking, members of the public may submit written or oral comments on the proposed design certification rule. The Commission may, at its discretion hold a hearing. The issues that are resolved in a design certification rulemaking are subject to a more

restrictive change process than issues that are resolved through the issuance of a license. The NRC can only change certified design requirements in limited circumstances.

Several of your concerns were related to the differences between the AP600 and AP1000 design reviews. The NRC considers the review and design certification of the AP1000 to be largely independent of the previous AP600 design review. The AP1000 design is not a power uprate of the AP600 design. The NRC's review of the AP1000 design is conducted in accordance with 10 CFR Part 52 requirements, and in accordance with the applicable review procedures, and acceptance criteria of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." In some cases, the staff referenced AP600 design information; however, there is an independent review of all AP1000 safety parameters to verify that they meet NRC regulations.

The safety review of the AP1000 application is based primarily on the information submitted by the applicant under oath and affirmation. An application must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design. In general terms, a design certification application should provide an essentially complete nuclear plant design, with the exception of site-specific design features such as intake structures and the ultimate heat sink. The application presents the design criteria and design information for the proposed reactor and includes information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of structures, systems, and components of the facility as a whole. The scope and contents of the application are equivalent to the level of detail found in a final safety analysis report (FSAR) for a current operating plant. The NRC staff prepares a safety evaluation report (SER) which describes how the staff performed the review of the plant design, and how the design meets applicable regulations.

The following sections address your specific concerns.

AP1000 Fluid Systems Design

You were concerned whether the NRC verifies or asks for proof that the system parameters reported in the AP1000 design certification application (and used in the analyses) are actually justified by a detailed design, rather than only by a conceptual AP1000 system design or by preliminary equipment sizing calculations.

To support design certification, system parameter information must be at a level of detail such that the NRC can make a determination of reasonable assurance of safety at the time the design is certified. The NRC requirement for the level of design information detail supporting an application for design certification is set forth in 10 CFR 52.47(a)(2). Specifically, the applicant must provide sufficient information to the Commission to reach a final conclusion on all safety questions associated with the design before the certification is granted.

In addition, should an applicant reference the AP1000 design, the NRC will conduct independent design verification inspections during the pre-combined license review phase. In SECY-94-294, "Construction Inspection and ITAAC Verification," the staff stated that design descriptions and functional system drawings available for review during the design certification and combined license application stages are adequate for licensing reviews and final safety determinations, but not for actual construction or construction inspection activities. The NRC will inspect and review the adequacy of licensee design engineering early in a construction project, possibly beginning soon after receipt of a licensing application; first-of-a-kind

engineering for the lead plant of each certified design will be assessed during these inspections. The NRC will also assess the effectiveness of the licensee's design change process in maintaining the fidelity of high-level certified design information that is translated into construction drawings.

One specific concern you raised was that head loss coefficients (e.g. L/D criteria) should be provided for systems whose layout is to be finalized at a later date, and that "proof-of-design" calculations should be provided for calculations whose layout is determined at this stage. With regard to piping design, Westinghouse is proposing to use design acceptance criteria (DAC) in lieu of detailed design information for design certification. The NRC staff defines DAC as a set of prescribed limits, parameters, procedures, and attributes upon which the NRC staff relies, in a limited number of technical areas, in making a final determination to support a design certification. The Commission found that approach acceptable for General Electric Advanced Boiling Water Reactor (ABWR) and ABB-Combustion Engineering System 80+ designs. Therefore, the NRC did not expect signed-off, proof-of-design calculations to be completed when the design control documents (DCDs) were submitted. However, piping design calculations will need to be completed to support construction and the NRC will verify the calculations through appropriate use of ITAAC of the design and construction activities. The acceptance criteria for DAC become the acceptance criteria for ITAAC, which are part of the design certification. In addition, the NRC reviews the applicant's safety analysis report, which describes the plant's final design, safety evaluation, operational limits, anticipated response of plant to postulated accidents, and plans for coping the emergencies to determine whether there is reasonable assurance of safety.

Quality Assurance Design Control Measures for the AP1000 Plant

You also raised concerns about the design control process used by Westinghouse for the preparation of AP1000 design documents. Specifically, in your February 11, 2004, statement to the ACRS you questioned who is entitled to make the decision about which features, calculations, and documents for the AP600 need to be reviewed for changes in uprating of the AP1000. In accordance with 10 CFR Part 52, an application for design certification is required to provide a description of the quality assurance (QA) program to be applied to the design of systems, structures, and components (SSCs). In its application for design certification of the AP1000 plant, Westinghouse stated that a continuous QA program spanning the AP600 design and the AP1000 design has been used. Since March 31, 1996, activities affecting the quality of items and services for the AP1000 project during design, procurement, fabrication, inspection, and/or testing have been performed in accordance with the quality plan described in "Westinghouse Energy Systems Business Unit - Quality Management System." The Quality Management System (QMS) establishes design control measures for preparing, reviewing, and approving design documentation for safety-related SSCs. As documented in an NRC evaluation letter dated February 23, 1996, from S. Black (NRC) to N. J. Liparulo, the Westinghouse QMS was reviewed by the staff and found to meet the requirements of 10 CFR Part 50, Appendix B. Subsequent revisions to the QMS have also been reviewed by the staff and found to be acceptable.

To provide additional assurance that Westinghouse implemented the measures described in the QMS, the staff performed a quality assurance implementation inspection at the Westinghouse engineering offices in Monroeville, Pennsylvania, during the week of September 15, 2003. The inspection activities included a sampling review of QMS activities related to the QA organization, supplier evaluation and qualification, the corrective action program, audits and self-assessments, and the design control process. The results of this

inspection were documented in NRC Inspection Report No. 99900404/03-01, dated November 4, 2003. The inspection report can be found on the NRC public Website http://www.nrc.gov/reading-rm/adams.html under ADAMS Accession No. ML033090510. With the exception of the methods used to evaluate and qualify certain suppliers of safety-related design analyses, the staff determined that Westinghouse complied with the requirements of 10 CFR Part 50, Appendix B, for the areas reviewed. Subsequent to the inspection, Westinghouse described the corrective actions taken to evaluate, and qualify suppliers of safety related items and services for the AP1000 design. As noted in a January 13, 2004, letter from T. Quay (NRC) to W. Cummins, the staff determined that the corrective actions were responsive to the staff's concerns.

During the QA implementation inspection, the staff determined that Westinghouse established project-specific quality-related procedures to implement the QMS requirements for the AP1000 project. These project-specific procedures established a design control process for AP1000 that included preparation, review, and approval of AP1000 design information. Although the AP1000 design was derived from the AP600 design, the AP1000 project-specific design control process specified that all documents generated to describe, portray, specify, or report on the AP1000 design were subject to independent verification and approval reviews. Independent verification was intended to confirm that the design document accurately reflected supporting design information. The NRC inspectors concluded that the design control measures described in the AP1000 project specific quality procedures met the design control requirements of 10 CFR Part 50, Appendix B.

Although certain weaknesses in the areas of QA audits and self-assessments were identified during the inspection, the staff concluded that, in general, internal audits and self-assessments for the AP1000 project met the requirements of the QMS. Following the inspection, Westinghouse provided additional docketed information related to the conduct of QA audit and self-assessment activities for the AP1000 project. The staff reviewed this information and concluded that Westinghouse had identified a reasonable cause and corrective actions for the failure of QA audit activities to identify weaknesses in the supplier qualification program. It should be noted that the quality assurance requirements of 10 CFR Part 50, Appendix B, do not include the performance of self-assessments. Furthermore, the staff does not rely on the performance of self-assessments to provide reasonable assurance in an applicant's design control processes. Therefore, although the staff noted certain weaknesses in self-assessment activities, reasonable assurance of the adequacy of design control measures is provided by the applicant's compliance with the quality assurance requirements 10 CFR Part 50, Appendix B.

Impact of Solar Radiation Heat on the Ultimate Heat Sink

You raised a concern about the impact of solar radiation heat on the passive containment cooling system (PCCS) water tank located on top of the containment building.

The passive containment cooling water tank is a large reinforced concrete structure located above the containment building. The tank contains a large volume of water (800,000 gallons). Because the volume of water in the tank is so large, the rise in the water temperature due to solar radiation heat is negligible. The Westinghouse analyses and evaluations assumed initial containment conditions of 120 °F and 1.0 psig. The PCCS water temperature is also assumed

to be 120 °F. The analyses provided conservative peak calculated containment pressure and temperature response following postulated design basis accidents. The PCCS is designed to meet the requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 38, "Containment heat removal," and GDC 40 "Testing of heat removal system." Technical Specification (TS) 3.6.6, "Passive Containment Cooling System" verifies the water storage tank temperature remains less than 120 °F every 24 hours during normal operation of the plant. If the temperature is observed higher than 120 °F, the PCCS would be declared inoperable, and the licensee will be required to take corrective actions to restore water storage tank to operable status within 8 hours or be in hot standby within 6 hours and in cold shutdown within an additional 78 hours. In reviewing the detailed design information and the analyses for the PCCS, we believe the TS requirements and actions taken will bound any possible solar radiation effects.

In summary, the NRC has followed all of the applicable rules, regulations, and technical review processes in reviewing the AP1000 design certification application. Please contact us if you have additional information regarding the AP1000 design certification that you believe we have not captured through our review process.

Thank you again for your interest in AP1000 design certification. If you have any questions regarding this correspondence, please feel free to contact Raj Anand, Project Manager. He can be reached at 301-415-1146 or by e-mail to rka@nrc.gov.

Sincerely,

/RA/

James E. Lyons, Program Director
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 52-006

cc: See next page

Department of Philosophy
201C West Duke Building
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Durham, NC
sterrett@duke.edu

July 30, 2003

To: ACRS Subcommittee on Future Plant Designs

Subject: AP1000 Fluid Systems Design & QA Procedures

1. Purpose

At the July 18th Meeting of the ACRS Subcommittee Meeting on Future Plant Designs held in Monroeville, Pennsylvania, I took advantage of the opportunity afforded members of the public to remark on a topic discussed at that meeting: the NRC's review of QA control of processes used in the AP1000 design currently under licensing review. At that meeting, the NRC staff (Ms. Joelle Starefos) responded by saying that the NRC staff would reply in a letter.

As I did not know which open items were going to be discussed, my remarks were impromptu and I did not have a prepared text. The purpose of this letter is to provide a written statement of the concerns I expressed at that meeting, which made reference to concerns I had expressed earlier, at the 501st ACRS meeting. (References 6 and 7) For completeness, I also include a chronology of the questions and responses already received from other members of the NRC staff in sections 2.1 and 2.2 below. The statement incorporating the concerns I raised at the July 18th, 2003 ACRS meeting appears in section 2.3 below.

According to the policy on Advisory Committee Meetings (10CFR7.12 (b)), " Any member of the public who wishes to do so shall be permitted to file a written statement with an NRC advisory committee regarding any matter discussed at a meeting of the committee." I am filing this letter as such a written statement, as a member of the public, unaffiliated with any organization.

I am currently a professor of philosophy at Duke University in Durham, North Carolina. Prior to my academic career, I worked in the nuclear power industry, including a few years in the mid-nineties on the AP600 fluid systems design as a consultant to Westinghouse. My involvement with the nuclear power industry ended in early 1998 when I began my academic career in philosophy full-time.

I began following the NRC licensing review of the AP1000 in mid-2002 by reading the information publicly available via the NRC's electronic reading room. My knowledge about the AP1000 design and licensing review comes from reading these publicly available documents. I decided to make use of the provisions for public participation in the AP1000 licensing process (References 8, 9) in part because, according to the 10CFR52 licensing process under which the AP1000 is being licensed, opportunities for public participation are extremely limited once design certification is granted. Thus, as a member of the public, providing this input about the AP1000 design and licensing review is a "now-or-never" situation.

2. Chronology of Questions and Statements

2.1 Two Issues Raised with NRC Staff in July 2002 -- Systems Design & AP1000 QA

In mid-2002 (July 10), after the AP1000 design certification submittal, I asked questions about the general 10CFR52 process and the AP1000 licensing review in particular in an email exchange with Jerry Wilson of the NRC. (Reference 3) One question was: what ensures that, by the close of the licensing process, the design process for some components was not still at the stage wherein only preliminary sizing of components had been performed.? In particular, I asked:

> "(i) Are there supposed to be signed-off, proof-of-design calculations, (using the actual piping sizes, equipment parameters, and layout) for the flows reported for all the systems in the AP1000 DCD submitted? Or, performance analyses for the more complex pieces of equipment such as the pressurizer, the steam generator, large control and relief valves, etc.?
>
> (ii) Does the submittal of the DCD imply that the things in (i) are done?
>
> (iii) Does the NRC verify or ask for proof that the things above are in fact completed and signed off by the appropriate functional groups, and that they justify the design details in the DCD? If so, when does this occur? [Reference 3]

In reply, Jerry Wilson cited 10 CFR 52.47(a)(2), and explained that the level of detail required for a DCD (design control document) submittal was sufficient information to support a safety finding in any technical area, and that this level of information corresponds to the level that, under the previous two-step 10CFR50 process, was available at the operating license stage. However, he qualified this by saying that, since design acceptance criteria were to be used in the "piping design area", that "we [NRC staff] didn't expect that signed-off, proof-of-design calculations will need to be completed to support construction." [Reference 10]

This reply made me wonder whether the NRC was in practice approving delaying performing the proof-of-design calculations for system flows, temperatures, and pressures to later stages as well, without explicitly meaning to do so. The rationale for accepting the (DAC) approach for "the piping design area", which was articulated in SECY-02-0059 [Reference 2], was based on the ability to specify piping stress and piping structural analysis acceptance criteria; that rationale does not support delaying the fluid system design to the later COL stage. It is in fact important that the finalized fluid system design be performed prior to or in conjunction with specifying pipe sizes and valve characteristics to be used in the final design. It is always possible to use preliminary calculations to size piping, valves and equipment in order to obtain values to be placed in a design certification application. Proof-of-design calculations differ from preliminary sizing calculations in that they are a set of calculations chosen to take into account all the system criteria that must be met in order for the system to perform the capabilities that are claimed for it. As explained in followup emails, in lieu of using complete piping layout information as input to "proof-of-design" calculations, L/D criteria can be specified based upon "proof-of-design" calculations; these can then be used in piping layout to ensure that the considerations underlying the "proof-of-design" calculations are met. This kind of criteria would be the fluid systems design analogue of piping DAC. My worry was that unless some attention was paid to ensuring that the "proof-of-design" kinds of analyses are done, whether in the form of calculations using "as-built" data or in the form of L/D layout criteria, that the NRC would actually be certifying a design that was based on preliminary sizing considerations rather than on proof-of-design calculations that document that the various fluid systems have actually been designed to provide the system capabilities claimed for them. Since such fundamental things as the classification of initiating events assumes that even many non-safety systems actually do provide the capabilities attributed to them by the design documents, the issue is related to the safety basis of the plant even for the design of non-safety systems.

The problem is particularly acute on the AP1000 because much of the AP1000 makes reference to AP600 documentation. This makes it especially difficult to discern whether a particular pipe size and equipment parameter is merely inherited from the AP600 design or whether final "proof-of-design" kinds of calculations specific to the AP1000 have been performed to support it. Further, there is the danger of making the false assumption that if a system configuration has not changed, the fluid system performance has not changed either. This is not always true; a system temperature or pressure in one system can affect the fluid system performance in another. Thus reasoning about the similarity to AP600 layout that applies for piping stresses and loads does not necessarily extend to fluid systems performance. A comprehensive review of the AP1000 fluid systems designs is called for, similar to the kind of review appropriate when reviewing an extended power uprating.

In further email exchanges with the NRC (Jerry Wilson and Larry Burkhart), I tried to clarify my first question about the fluid system design. These emails are references 11 and 12 and are attached to this letter.

The second question I asked in my July 10, 2002 email to Jerry Wilson concerned the QA program covering the engineering design processes. I wrote there:

> The AP1000 design processes cannot be exactly the same as for the AP600, simply in virtue of the fact that the AP1000 refers to so many design documents for the previously certified, yet different, AP600 design. If the quality assurance program covers the engineering design processes, it seems it needs to be looked at (and maybe revised or supplemented) to ensure that it appropriately covers the case of producing a new design that references another, different, certified design, and to explicitly state what is required in such a case. Here's why I think it is a very important issue:
>
> The AP1000 DCD claims that many of the AP600 documents are applicable to the AP1000. The crucial question is, who (in Westinghouse) makes the determination that a particular AP600 document does in fact apply for the AP1000? It seems to me crucial that the same engineering functional group (preferably the same individual engineer) that was responsible for producing and signing off the document for the AP600 pass judgement on its applicability to the AP1000. Is there a guarantee of this? If not, I suggest that there be such a requirement and that it be made explicit.
>
> Otherwise, there is a gigantic loophole that can be used to circumvent the whole intent of the quality assurance provisions covering the engineering design process -- i.e., otherwise, individuals in other functional groups such as marketing, licensing, or project management, can circumvent the engineering process by simply stating that a certain AP600 engineering report or design document applies to the AP1000. (I don't think I need to explain the conflict of interest involved were this to be permitted.) [Sterrett to Wilson July 10, 2002 Reference 3]

Jerry Wilson replied to this question as well [Reference 10]. He referred me to the NRC's letter on the AP1000 Design Certification Review Schedule [Reference 4], and explained that the NRC staff did plan to inspect Westinghouse's implementation of its design control program for the AP1000 design "in the future." Mr. Lyons's letter of July 12, 2002 stated that the NRC planned to perform these inspections "as necessary", adding that "These inspections will be coordinated with Westinghouse to support the design certification schedule." [Reference 4 , p. 4]

2.2 Clarification & Discussion of Issues with NRC Staff -- December 2002

In December 2002 Larry Burkhart, who was then the NRC's AP1000 Project Manager, held a telecon to discuss my questions. Jerry Wilson, Dave Terao, and other members of the NRC technical staff were present. In this telephone conference call, I clarified my question about fluid system design. Nothing was resolved other than the clarification of the question. However, it was agreed that we should get in contact again to revisit the issues closer to the time the DSER was about to be issued.

Subsequently, after unsucessful attempts to reach Larry Burkhart in March 2003, I learned that there had been a change in management of the NRC's AP1000 Licensing team. The entire team had been replaced with the current team (John Segala, Joelle Starefos and Joseph Colaccino).

2.3 Concerns Raised at ACRS Meetings (April & July 2003)

Soon thereafter, I requested time to speak at the 501st ACRS meeting held on April 11th, where I read a statement presenting the first question I had raised in the original July 10th email. My oral presentation followed the draft text of my comments fairly closely [Reference 7 , included as Attachment II to this letter] and was included in the summary report for the 501st ACRS meeting [Reference 6].

The second question raised in my original email (regarding quality control procedures governing the design processes used in the AP1000) was brought up at an ACRS Subcommittee on Future Plant Designs held on July 18th, 2003, shortly after the NRC issued the Draft Safety Evaluation Report (DSER), and almost a year after I sent the original email expressing concerns about the QA process on the AP1000.

The list of AP1000 DSER Open Items included Open Item17.3.2-2, which reads in part:

> Westinghouse stated that a project-specific quality control plan was used to implement the requirements of the Westinghouse QMS program. The staff plans to conduct an inspection of the implementation of the project-specific quality plan to verify that design activities conducted for the AP1000 project complied with the Westinghouse QMS and the requirements of 10CFR Part 50, Appendix B. [Reference 5]

However, the "project-specific quality control plan" Westinghouse refers to is just the AP600 plan. Although Open Item 17.3.2-2 indicates "N/A" for the original RAI corresponding to the open item, there was an RAI about the AP1000-specific quality assurance plan [RAI 260.008-1 dated May 13, 2003]. Westinghouse's response to that RAI had been to claim that the AP600 document applied to the AP1000. The rationale given in Westinghouse's response to RAI 260.008-1 was:

> As the DCD identifies: " The plan ... is applicable to work performed for the AP1000 design." Westinghouse considers that it has identified a project specific quality plan (i. e., WCAP- 12600) for the AP1000 design.

There is also a discussion of the use of the AP600 project quality plan in Chapter 17 of the DSER, which states:

> A project-specific quality plan was issued to supplement the quality management system document and the topical reports for design activities affecting the quality of structures, systems, and components for the AP6OO project . . . This plan addresses the NQA-1-1989 edition through NQA-lb-1991 addenda and is applicable to work performed for the AP1000 design. [Reference 1, page 17-1]

These statements raise concern, for the reasons mentioned in my original July 10, 2002 email and excerpted in section 2.1 above. When I attended the ACRS Subcommitte Meeting on Future Plant Designs held on July 17th and 18th, I did not anticipate that the subject open item would be mentioned, and did not request time to speak beforehand. However, when I saw that the NRC's presentation included mention of the issue of an inspection of Westinghouse's QA plan during the meeting, I asked to make some impromptu remarks along the lines of the concern raised in my email. There was not time to gather the previous correspondence, relevant Open Items, RAIs, and RAI responses at that time. Therefore, I provide a more complete statement of the situation and my concerns about it here.

My concerns regarding QA of the AP1000 design process are:

A. Integrity of design process for the singular kind of project that the AP1000 is

The kind of process by which the AP1000 design was produced resembles an uprating in some ways, in spite of the fact that it is not regarded as an uprating. That is, one constraint was to use the AP600 design details insofar as possible. An uprating involves activities and considerations not addressed by the kind of design control procedures intended to address design of a plant where the design process starts with the specification of plant parameters and detail is filled in as the design progresses from functional specifications to detailed equipment specifications. Thus I would not expect the AP600 design control procedures to cover all the design processes on the AP1000.

<u>Of special concern is QA control of the overall plant parameters, both in terms of the design process by which they were obtained, and the design processes that use them as input.</u> (Perhaps this question was dealt with in the pre-application phase, but in case not, I raise it here.) I believe the generation of overall plant

parameters, whether for a new plant design, an uprating, or other changes to an existing plant design, is typically very tightly controlled, with oversight by an interdisciplinary committee whose membership is established independently of any particular project.

An important question here that needs to be asked is whether there are additional oversight or formal procedures over and above those addressed in the AP600 QA plan that would be appropriate for an uprating in that they would assure that the parameters are communicated to the affected functional design areas, would see that the right agents identify the specific changes that are required, and would keep track of their implementation. My worry is that due to its special nature (the criterion of keeping the AP600 design details as much as possible), the implementation of the AP1000 project plant parameters would really call for the additional oversight or the kinds of procedures applicable to an uprating.

If design control procedures intended for new plant designs were used in implementing the AP1000 plant parameters, rather than the design control procedures written to cover upratings, this raises a concern about the way that the AP600 information was used on the AP1000 project. This is because, for an uprating, the plant parameters are an input into a design process where an already existing plant is modified under the constraints of keeping much of the design unchanged. All kinds of QA design control questions arise in this case: for instance, who determines what information originally generated for the AP600 applies to the AP1000 or whether it needs to be reviewed? And who reviews it? Whose decision is final? It seems to me that the integrity of the design process relies upon keeping the design functions separate from project management functions. When a design group reports administratively to the project management and on a matrix basis to engineering management, the integrity of the design process depends upon the matrix connection being strong enough to ensure that technical aspects of management initiatives receive their due.

This kind of situation is not explicitly addressed in 10CFR50 Appendix B, but there is a statement on the general topic of who gets to decide such things in the event of design changes: "Design changes . . . shall be approved by the organization that performed the original design unless the applicant designates another responsible organization." Now, on the AP1000, where so many AP600 features are to be inherited, there is a kind of implicit change to an unspecified number of system capabilities in that the plant parameters have changed. Meeting the spirit of the subject criteria would mean that the judgement as to whether an AP600 design or document applies to the AP1000 or not should be made by those responsible for that design or document on the

AP600 design. Since the DCD references many AP600 documents, it is not always clear that the author of the AP600 document or design has approved its applicability to the AP1000. I think an important question is: who has determined that a certain AP600 document is applicable to the AP1000?

B. Organizational Differences Between AP600 and AP1000 affecting design control

The AP600 design control procedures reflected the involvement of ARC, the Advanced Reactor Corporation, a consortium of electrical utilities. I do not have access to the relevant procedures, but I recall from my previous involvement with the AP600 project that representatives of the ARC did have a formal role in the approval of design changes. Thus, beyond the straightforward point that the design control procedure for the AP1000 can not be exactly the same as the AP600 in terms of the letter of the law, there is the more significant point that the involvement of such an agency provided checks and balances on the AP600 project that may not exist on the AP1000 project.

There may be other organizational changes since the AP600 QA inspection was performed that affect the quality and the strength of the ties between technical and engineering design personnel in the AP1000 organization and the technical department managers reported to on a matrix basis. It would seem to me that these would need to be examined in order for the NRC's review of Quality Control to conclude that the assurance provided by the procedure when applied on the AP1000 project is the same as the assurance it provided on the AP600 design.

C. It seems late in the process should problems be detected

The NRC Letter accepting the Design Certification application dated July 12 2002 (Reference 4) stated that QA inspection would be done "as needed".

The fact that a QA inspection is an open item is reassuring in that it means this item will be tracked. However, the fact that it is an open item is cause for concern as to whether the appropriate inspections were performed "as needed" in the area of review of the fluid systems design. It is a concern because of the possibility that the QA inspection might reveal that some design activities need to be performed. Should these design activities result in design changes, it is very late in the process. Further, it seems that the comprehensive fluid system design of the AP1000 plant --- deriving the basic plant parameters from the AP600 design --- as well as the design details of specific systems appropriately designed for the AP1000, should be covered by this item.

The issue here is the QA control on information that is in the DCD: was there design control guaranteeing that the generation and implementation of the basic plant parameters for the AP1000, as well as the fluid systems design details (e.g., equipment parameters, piping size, valve specifications) were the result of design work of the appropriate kind (i.e., not merely preliminary sizing calculations), performed in a context where there was proper control of design information input into the design process, and where there were the appropriate checks and balances that provide assurance of the integrity of the design process? If it turns out there were areas where it was not, it seems there is not a lot of time to allow review and comment on the required design changes if the design certification schedule is to be adhered to.

3. Additional Remarks -- Schedule for Resolution of DSER Open Items and Role of Public Review and Participation

In general, the AP1000 design certification schedule seems to permit a number of potentially significant open items at the DSER stage. This limits the time available for review and comment by the public after the open item is resolved. Considering the finality of a design certification, it seems that the time available for public review and comment should not be abbreviated in the only stage provided for it.

Respectfully submitted,

Susan G. Sterrett
Assistant Professor of
Philosophy
Duke University, Durham, NC

Attachment I Email correspondence Sterrett to NRC dated September 15, 2003.

Attachment II Draft Text of Comments Read at 501st ACRS Meeting --Dr. S. G. Sterrett

References:

1. AP1000 Draft Safety Evaluation Report (DSER)
Chapter 17 "Quality Assurance"

2. SECY-02-059 April 1, 2002
"Use of Design Acceptance Criteria for the AP1000 Standard Plant Design"
William D. Travers to The Commissioners

3. Email S. G. Sterrett to J. N. Wilson,
"AP1000 Review/ 10CFR Part 52 Process"
Wednesday, July 10, 2002

4. Letter July 12, 2002 James E. Lyons to W. E. Cummins
"AP1000 Design Certification Review Schedule (TAC NO. MB4682)"

5. Letter May 27, 2003 James E. Lyons to W. E. Cummins
"Westinghouse AP1000 Draft Safety Evaluation Report
Potential Open Items Chapter 17 Quality Assurance"

6. Letter May 7, 2003 M. Bonaca to Nils J. Diaz
"Summary Report of 501st Meeting of the Advisory Committee
on Reactor Safeguards, April 10 - 12, 2003"

7. Draft text of comments by S. G. Sterrett at 501st ACRS Meeting
(Attachment II to this letter. No transcript of April 11th meeting was made; oral statement followed written draft closely)

8. Nuclear Regulatory Commission Policy Statement
"Enhancing Public Participation in NRC Meetings; Policy Statement"
Federal Register May 28, 2002 (Volume 67, Number 102; Page 36920-36924)

9. 10 CFR7.12 Public Participation in and public notice of advisory committee meetings.

10. Email J. N. Wilson to S. G. Sterrett
"Re: Followup on Questions: AP1000 Review/ 10CFR Part 52 Process"
August 13, 2002.

11. Email S. G. Sterrett to L. J. Burkhart
"Thanks for RAIs"
September 15, 2002
(included in Attachment I to this letter)

12. Email S. G. Sterrett to J. N. Wilson
"Piping Layout L/D Criteria for Fluid System Performance"
September 15, 2002
(included in Attachment I to this letter)

cc:

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ATTACHMENT I

Emails Sterrett to NRC (L. J. Burkhart; J. N. Wilson) dated September 15, 2003

.....

This first email clarifies a question sent earlier to Jerry Wilson and discussed by telephone with Larry Burkhart . In it, I explain why the question is not addressed by the considerations provided in the rationale used in accepting DAC for the AP1000, nor covered by the RAIs sent to Westinghouse as of that date. The email below is followed by a longer one addressed to Jerry Wilson and cc'd to Larry Burkhart and Marsha Gamberoni.

Date: Sun, 15 Sep 2002 16:21:36 -0400 (EDT)
From: sterrett@duke.edu
To: Lawrence Burkhart <LJB@nrc.gov>
Subject: Thanks for RAIs

Dear Larry,

I have looked over the RAIs, and don't see any that address the question I asked Jerry Wilson about paying attention to fluid system performance in doing the piping layout. The RAIs do mention thermal-hydraulic loads, but that isn't what I meant; thermal-hydraulic loads are still related to the mechanical loads on the piping and concern the piping structural-mechanical analyis.

What I meant is the fluid system performance -- flowrates, pressures and temperatures that are achieved by the combination of driving head and fluid piping resistance. The fluid piping resistance is affected by the piping layout. In an email to Jerry Wilson, which I put you on cc for, and which I will send immediately after this one, there is more explanation. The bottom line is that even though the piping layout isn't final, the piping resistance criteria ("L/D criteria") for the AP1000 should be computed and provided at this point. In that email, following this one, there is also an explanation as to why the L/D criteria for the AP1000 will be different in many cases from the AP600.

In our conversation, you mentioned that the AP1000 is so similar to the AP600. That may be, but the question is, should the piping layout really be so similar? It is the fluid system's performance that sets the requirements of the design, and the layout has to meet those criteria. That's the point. One has to check, not just assume it will all turn out okay.

I imagine that there are people at the NRC whose reviews will address this, perhaps on a system-by-system basis. And whether or not the L/D criteria (piping resistance layout criteria) differ much for the AP1000 vis a vis the AP600 for a particular system may be a design detail. However, the overall point that L/D criteria for the AP1000 should be calculated at the DCD application stage is a plant-level issue. It's a very general point. In the email that follows, I explain why I think it is a policy issue about the new licensing process.

I am asking these questions as an individual member of the public, unaffiliated with any organization.

Sincerely,
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.....

The "email that follows" referred to in the above email is appended below. It is: Email dated September 15, 2002 from Sterrett to NRC staff (Jerry Wilson, cc to Larry Burkhart and Marsha Gamberoni)

.....

Date: Sun, 15 Sep 2002 16:46:21 -0400 (EDT)
From: sterrett@duke.edu
To: Jerry Wilson <JNW@nrc.gov>
Cc: LJB@nrc.gov, MKG@nrc.gov, sterrett@duke.edu
Subject: Piping Layout L/D Criteria for Fluid System Performance

To: Jerry Wilson, Senior Policy Analyst, NRC
cc: Larry Burkhart, AP1000 Project Manager, NRC
Marsha Gamberoni, Deputy Director, New Reactor Licensing

Subject: Piping Layout L/D Criteria for Fluid System Performance

Dear Jerry,

In a previous email, you responded to a question I asked regarding whether proof-of-design calculations of fluid system performance were performed for the AP1000. This email is to (a.) clarify the question I was asking, and (b) explain why I think L/D criteria is an issue of policy regarding the 10CFR52 design process, not merely a minor design or schedule detail.

In spite of the length of this email, the two points are simple; I am just including the text of the things I reference to avoid any possible ambiguity.

(a) Clarification of Question Re: Calculations Supporting Fluid System Performance

To recapitulate, the question I asked (July 10) was:

``1. What point of maturity is the design supposed to have at the stage the AP1000 application is presently at? I take it that by the time a design is certified, it is not supposed to be one for which only preliminary sizing calculations have been performed to size the equipment. What ensures this doesn't happen?

(i) Are there supposed to be signed-off, proof-of-design calculations, (using the actual piping sizes, equipment parameters, and layout) for the flows reported for all the systems in the AP1000 DCD submitted? Or, performance analyses for the more complex pieces of equipment such as the pressurizer, the steam generator, large control and relief valves, etc.?

(ii) Does the submittal of the DCD imply that the things in (i) are done?

(iii) Does the NRC verify or ask for proof that the things above are in fact completed and signed off by the appropriate functional groups, and that they justify the design details in the DCD? If so, when does this occur?" [excerpt from email of July 10, 2002 Sterrett to Wilson]

In your response (August 13) you explained why proof-of-design calculations for fluid system performance were _not_ expected to have been performed at the time of DCD submittal:

``With regard to question #1,
the Commission expects that when submitted, the design maturity is equivalent to the level of design information available at the operating license stage under the old 2-step process in Part 50 (Final Safety Analysis Report). The NRC's requirement for the level of detail of design information supporting an application for design certification is set forth in 10 CFR 52.47(a)(2). Specifically, it is sufficient information to support a safety finding in any technical review area. However, with regard to piping design, Westinghouse is proposing to use design acceptance criteria in lieu of detailed design information for design

certification. The Commission found that approach acceptable for the ABWR and System 80+ designs. Therefore, for questions #1(i) and (ii), we did't expect that signed-off, proof-of-design calculations were complete when the DCD was submitted. However, piping design calculations will need to be completed to support construction and the NRC will do verification inspections of the design andconstruction activities [#1(iii)]. ``
[excerpt from email of August 13, 2002 Wilson to Sterrett]

I would like here to clarify my earlier question: by ``proof-of-design calculations", I was referring to proof-of-design calculations for fluid system performance, rather than to piping design calculations. By ``piping design calculations", I assume you are referring to calculations concerning things such as piping stress, fatigue and mechanical loads. But, of course, the proper flow performance of fluid systems sets another kind of criterion: that is, in addition to the criteria that aim to ensure that the structural/mechanical behavior of the piping is acceptable, piping layout activities also have to take into account criteria that ensure that the piping flow resistances will result in the flows through the system called for by the fluid system design (and for which the design of numerious interfacing systems may take credit). In addition, pressures (and, sometimes, temperatures) in the system at various key points, such as at heat exchangers and control valves, are influenced by the piping layout. And here i am including normal system operation. Your response to the question of whether there have been proof-of-design calculations for fluid flow performance was that you did not expect them to be done, because the piping layout wasn't final.

However, if the piping layout isn't far enough along to permit proof-of-design calculations to be performed, the calculations related to fluid system performance should still be done -- the only difference is that they would result in piping fluid flow resistance criteria, or ``L/D criteria."

From your response, I wasn't sure if ``L/D criteria", or piping fluid resistance criteria were included in the DAC. After looking at various meeting transcripts and the RAIs regarding DAC attached to the meeting notice for September 9, 2002 (Reference 3), it doesn't appear to me that the ``L/D criteria" are addressed in these places.

So, the question is whether L/D criteria have been provided for the AP1000 fluid systems. Even if the piping layout for the AP1000 were _exactly_ the same as the AP600 layout, new L/D criteria would need to be calculated for the AP1000. For, anytime the design flowrate for a system changes, the

L/D criteria need to be re-calculated, since piping flow resistances vary with flowrate. Even for those systems, if any, where the fluid flowrate of the system is exactly the same for the AP1000 as it was for the AP600, there is still the question whether there are differences in the inlet or outlet pressures -- i.e., in the pressure in the system or piece of equipment to which it connects and from which the fluid enters the fluid system or to where it discharges. Hence the fluid flow performance would be different for the same layout. Thus, the layout criteria would differ between the AP1000 and the AP600 for cases where a system's inlet or discharge pressures differ. (An example here of such a difference in the AP1000 is the significant change in main steam pressure: obviously L/D criteria will be different between the AP600 and the AP1000 for the inlet piping to the steam relief valves, for example.)

Thus, to rephrase the question in my July email:

``(i) Are there supposed to be signed-off, L/D criteria and supporting calculations, (using the AP1000 fluid system functional requirements and equipment parameters) for the system flows and pressures reported for all the systems in the AP1000 DCD submitted? Or, L/D criteria for the piping associated with the more complex pieces of equipment such as the pressurizer, the steam generator, large control and relief valves,etc.?

(ii) Does the submittal of the DCD imply that the things in (i) are done?

(iii) Does the NRC verify or ask for proof that the things above are in fact completed and signed off by the appropriate functional groups, and that they justify the design details in the DCD? If so, when does this occur?"

This is the question I have now, given your repsonse that you did not expect ``proof-of-design calculations" to be performed due to the fact that the piping layout is not final at the DCD application stage.

(b) Previous process versus new 10CFR52 process

It is simply good common sense to provide L/D criteria for the preliminary piping layout, in order to have confidence that when the final piping layout is in fact completed, the design will be such that the fluid performance functional requirements of the system are in fact met, avoiding major changes to the preliminary layout. As you may be aware, this is the process that was followed on the Westinghouse standard plants.

As I see it, requiring that L/D criteria for performance of fluid system functional requirements be provided at the DCD submittal stage in the AP1000 design process is also a _policy_ issue. Here is why: under the older process, L/D criteria were provided to the architect-engineer for use in laying out piping, that is, in the preliminary layout. Thus they were performed PRIOR to the application for an operating license under the old process. L/D criteria can be provided now, as they do not depend upon the piping layout, much less on the piping layout being final. (They are criteria calculated for use in laying out piping such that the fluid system functional requirements (which should be final at the DCD submittal stage) are met.) The L/D criteria are criteria that apply for _preliminary_ layout as well as final layout.

Certainly the ITAACs and other operational tests are going to provide a checkpoint where deficiences in system performance are found, but, I trust, it certainly isn't the intent of the new 10CFR52 process to increase the surprises encountered during operational testing! I assume that everyone agrees that the intent is to have confidence that the certified design results in fluid systems that meet their functional requirements in terms of flowrates, pressures, and temperatures, even if the piping layout for the certified design may not be final in every detail.

Thus, it seems clear that the L/D criteria should be provided at the DCD submittal stage in the 10CFR52 process. It's an issue of policy because, otherwise, the 10CFR52 process would result in the NRC certifying a design for which there was less confidence in the design than existed under the old process at a comparable stage.

It would be great to hear the answer that L/D criteria for all the AP1000 systems have in fact been calculated and provided, but, in any case, I look forward to your reply. As with my previous inquiry, I am asking these questions as an individual member of the public, unaffiliated with any organization.

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ATTACHMENT II

Draft of Remarks by Dr. S. G. Sterrett - 501st ACRS meeting, April 11th, 2003, Rockville, MD

I'm Susan G. Sterrett. I am currently a professor at Duke University in Durham, North Carolina. I should perhaps mention that, prior to my academic career, I worked as a design engineer in the commerical nuclear power plant industry, including on fluid system design of the AP600 and EPP plants in the mid-nineties. I am making these remarks as a member of the public, unaffiliated with any organization.

I'm here today because I have some questions about the NRC's review of the AP1000. Put briefly, my question is whether the NRC verifies or asks for proof that the system parameters reported in the AP1000 design certification application (and used in the analyses) are actually justified by a detailed design, as opposed to the AP1000 system designs being at the stage of conceptual system design or justified only by preliminary equipment sizing calculations. I'd like a few minutes to explain the relevance and the significance of the question.

According to the rules under which the AP1000 is being licensed by the NRC, the level of design information required in a design certification application is, with a few explicit exceptions, the level of information that was required at the operating license stage under the previous two-step licensing process. I think this requirement makes sense, too, inasmuch as what the NRC is licensing in approving the AP1000 is an actual plant design that is certified to be constructed and operated.

In following some of the AP1000 licensing activities via the NRC's website, I have noticed that much is often made of the similarities between the AP1000 systems and the AP600 systems. This can be misleading: the performance of the various fluid systems in the plant -- that is, the flows, temperatures, and pressures that obtain at various points within a system are affected by many kinds of differences in a plant design. As I am sure everyone here realizes:

--- Anytime a system flowrate changes, pressure drops in the system will change.

--- Likewise, anytime the pressure at some point in a system changes, flowrates in it or some other system can be affected.

--- Thus, even for those systems that are exactly the same physically speaking (i.e., same pipe size and layout) for the AP1000 as for the AP600, there is still the question of whether there are differences in the inlet or outlet pressures in a system or piece of equipment to which it connects. Different inlet or outlet pressures will result in differences in fluid system performance.

For example, suppose the main steam system pressure is different on the AP1000; then, on the AP1000, there would be a different driving head for lines connected to it than there was on the AP600. So, even if the system hardware and layout of a system connected to the main steam system, say, is exactly the same on the AP1000 as it was for the AP600, the resulting values of major fluid system parameters -- e.g., the mass and volume flowrates and the pressures that result -- could be quite different. Obviously the effects on things like the flow capability of relief valve piping and valve arrangements would need to be looked at. Accomodating these changes could require resizing piping or control valves in order to achieve the flowrate claimed for the system.

I've given the main steam system as an example, but the general point holds for every system in the plant. To infer from the fact that the hardware and layout on an AP1000 system is exactly the same as on the AP600, to the conclusion that the performance is the same, is incorrect. The various AP1000 analyses now under review are only as valid as the assumptions made in them about the performance of the plant systems.

What does this point mean for the review of the AP1000 design, which makes frequent appeal to the certified AP600 design? In many aspects of the safety analyses, the NRC has been very alert to the differences between the AP1000 and the AP600. The point of my examples is that this awareness ought to be extended to plant fluid system performance, specifically, that some reassurances should be sought that the fluid system design details for all the plant systems have been properly attended to, and that, given that the level of detail required at this stage is supposed to be the same as that at the operating license stage, these should not be just preliminary sizing calculations. I worry about the complacency with which the AP600 design is referenced in justifying the AP1000 system designs.

The AP1000 is sometimes referred to as an uprating of the AP600 design. Of course this would be significantly larger than any uprating that the NRC has licensed so far, and of course it differs from most upratings in that there is no AP600 operating experience to draw upon. To the extent that thinking of the AP1000 as an uprating of the AP600 is appropriate, however, it would make sense to require that all the plant system reviews that would be required for an extended power uprating be performed for the AP1000. As there is now a draft review standard for extended power uprates that could be used to guide such a review of the AP1000 (RS-001, dated December 2002), this seems a natural thing to do. I wonder whether there has in fact been a review of this sort for the AP1000. So let me ask: has there?

For those systems whose layout is finalized at this stage of the AP1000 design certification application, there should be formally signed-off engineering calculations justifying the claims that the AP1000 system flow, temperature, and pressure parameters will actually be achieved using the AP1000 equpment and layout. These are often referred to as fluid system "proof-of-design" calculations. I gather from the NRC's approval of the use of DAC (design acceptance criteria) for structural piping analysis on the AP1000 that

there may be some systems for which the layout details will not be completed until after design certification. For those systems, what is needed as far as ensuring proper fluid system performance is to provide layout criteria related to the piping flow resistance, so that the fluid flowrates claimed for the system will actually be achieved. Such criteria are commonly called "L/D criteria" and are considered part of the fluid system design. In fact, for the Westinghouse standard plant designs licensed under the previous two-step process, L/D criteria were provided for various fluid systems prior to construction so that the architect engineer could properly perform the piping layout. As I see it, at least this level of design detail is required at the time of the DCD submittal.

Why not just rely on the ITAACs (Inspections, Tests, Analysis, and Acceptance Criteria) to provide such reassurance? Certainly the ITAACs and other operational tests provide a checkpoint where some deficiences in the plant design would show up. However, I trust that it isn't the intent of ITAACs to relieve the designer of the responsibility of the engineering design work of designing the plant systems so that the system parameters crucial to safety are achieved. Certainly increasing the number of surprises encountered during plant testing is not part of the intent of the new one-step licensing process! I assume that everyone agrees that the intent of design certification is to provide confidence that the certified design will result in fluid systems that meet their stated functional requirements in terms of flowrates, pressures, and temperatures, even if the piping layout for the certified design may not be final in every detail.

In conclusion, I am asking whether the review of the AP1000 design has included ensuring that the design details upon which the analyses that the ACRS has been reviewing depend, have in fact been attended to. In particular, I think it is clear that L/D criteria should be provided at this stage for systems whose layout is to be finalized at a later date, and "proof-of-design" calculations be provided for those whose layout is determined at this stage. Otherwise, there is no assurance that the analyses you are reviewing so carefully and thoughtfully apply to the plant design you are certifying.

Thank you for listening.

Respectfully submitted,

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(6)

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July 31, 2003

To: ACRS Subcommittee on Future Plant Designs

Subject: Heat of Solar Radiation and AP1000 Ultimate Heat Sink

Although I did not make an oral statement on the subject topic at the ACRS Subcommittee on Future Plant Designs held on July 17th and 18th, 2003, I am taking the opportunity afforded members of the public to file a statement on subjects associated with the topics discussed at ACRS meetings. This statement is related to the AP1000 safety systems and the recently-issued AP1000 Draft Safety Evaluation Report (DSER).

The AP1000, unlike operating PWRs, uses the outside air as the ultimate heat sink. The Passive Containment Cooling System is responsible for transferring heat to the ultimate heat sink in the event of a design basis accident. The question I have is: whether (and if so, how) the heat of solar radiation is taken into account in the design of the AP1000 Passive Containment Cooling System.

As described in the DSER, heat removal from the containment after a design basis accident is to be accomplished by the Passive Containment Cooling System (PCS). The PCS uses the water in the PCS water storage tank located atop the containment, along with the flow of air through the spaces between the primary steel containment and the surrounding concrete building, to cool and depressurize the containment. It is the means by which heat is transferred from the reactor to the ultimate heat sink (the outside air) in the event of a design basis accident.

Thus, the temperature of the water in the PCS water storage tank and the temperature of the concrete walls affect the heat removal capabilities of the PCS. Since the heat of solar radiation can cause the temperature of objects to exceed that of the surrounding air, it seems to me that its effect on:

(i) the temperature of the concrete building, whose walls form the air passages relied upon for the efficacy of cooling by the PCS, and

(ii) the temperature in the PCS water storage tank,

ought to be addressed by the AP1000 design. The effect will vary with geographical location (i.e., one of the coefficients involved is a function of geographical latitude) and will also depend upon the surface geometry, the properties of the concrete and/or the surface coatings used, and the humidity of the outside air.

The site parameters do not include geographical latitute, so I am wondering whether the heat of solar radiation was considered or quantified. I do not see the effect of the heat of solar radiation accounted for explicitly in the DSER. However, it is clear that, unless the heat of solar radiation is shown to make only a negligible contribution, this heat source is relevant to the design of the safety features of the plant. The question does not arise for operating PWR plant designs, since those designs do not use the method of containment cooling employed on the AP1000. It appears to me that some of the regulations and criteria related to ultimate heat sink assume that the ultimate heat sink is a body of water; thus I would not expect them to have specifically addressed the effect of heat of solar radiation on the temperature distribution in concrete walls.

Perhaps this was already addressed at earlier stages of the project. However, even if this is so, there should be some discussion in the DSER of the rationale and assumptions used in making the determination that the effect of the heat of solar radiation on the structures used by the PCS for containment cooling could be neglected.

If in fact the effect of the heat of solar radiation on PCS performance is not determined to be negligible, the assumptions regarding PCS water storage tank temperature and PCS efficacy in heat removal used in the AP1000 PRA (Probabilistic Risk Assessment) Report should also be examined to see if the heat of solar radiation might need to be taken into account in the rationales employed there.

I have raised this question with the NRC staff. I do not know what the response will be. However, due to the late point in the licensing process (the DSER is already issued), the safety significance of the ultimate heat sink, and the finality of design certification which limits opportunities to raise the issue later, I am raising it in a statement to the ACRS now.

Respectfully submitted,

Susan G. Sterrett
Assistant Professor of Philosophy
Duke University, Durham, NC

GSI-191
Generic Letter 2004-XX

Presenters
Dave Cullison, 301-415-1212
NRR/DSSA/SPLB

ACRS Full Committee Meeting
Rockville, MD
July 7, 2004

(1)

Overview

- Purpose:
  - Obtain ACRS endorsement of GSI-191 GL 2004-XX
- Conclusion:
  - Issuance of Generic Letter 2004-XX will confirm the continued compliance with the long term cooling requirement of 10 CFR 50.46 by addressees in light of the new information coming from the efforts to resolve GSI-191

Public Comments

- Union of Concerned Scientists
- Nuclear Energy Institute
- Progress Energy, Inc
- Tennessee Valley Authority
- Westinghouse Owners Group
- Nuclear Utility Backfitting and Reform Group
- Westinghouse
- Florida Power & Light
- Duke Power
- Mr. Lanson Rogers
- Strategic Teaming and Resource Sharing (STARS)
- Dominion Resources
- State of New Jersey

Public Comment Major Issues

- Industry comments - make GL action oriented. Similar to Bulletin 96-03, which dealt with strainer clogging for BWRs
- Emphasis on Compliance
  - UCS - "The NRC must either require compliance determinations or abandon its risk-informed regulatory initiatives."
  - Industry - NRC approach it from a design basis standpoint
  - Industry - Methodology too conservative for compliance confirmation
  - Industry - Plants already comply with current licensing basis
- Backfit
  - Draft GL was not a backfit
  - Industry believes information request a backfit
- Schedule
  - Industry - timeline does not provide for enough time after issuance of letter to respond

Changes to Generic Letter

- Process
  - Purpose
  - Requested Actions/Information
  - Backfit Determination
- Upgrading licensing basis
  - Changing staff position on adequacy of current sump analyses Process:
- Compliance with of 50.46(a)(3)(ii)

Generic Letter - Purpose

- Request that addressees submit information to the NRC to confirm compliance with 10 CFR 50.46(b)(5), which requires long-term reactor core cooling, and other existing regulatory requirements listed in this generic letter. This request is based on the identified potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS) and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

- Require addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

Regulatory Requirements

- 10 CFR 50.46(b)(5) states "Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

- 50.46(a)(3)(ii) ....Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in §§§§ 50.55(e), 50.72 and 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with §§ 50.46 requirements.
  - An exemption from the requirement to take "immediate steps" may be necessary upon determination of noncompliance.

Generic Letter → Requested Action/Information

- Requested Action:
  - The Generic Letter is an information request only
- Within 60 days of the issuance of the guidance for performing the requested evaluation, addressees are requested to provide information regarding their planned actions and schedule to complete the requested evaluation

Generic Letter → Requested Information
Continued

- No later than September 1, 2005, provide the following information:
  - Provide confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements.
  - A general description of and implementation schedule for all corrective actions, including any plant modification that may be necessary to ensure compliance. Initiate actions to implement corrective actions no later than the first refueling outage starting after April 1, 2006; however, all corrective actions should be completed by December 31, 2007. If all corrective actions will not be completed by December 31, 2007, describe how the delays are consistent with the 10 CFR 50.46(a)(3)(ii) requirement to take immediate steps to demonstrate compliance.

Generic Letter → Requested Information
Continued

- September 1, 2005 submittal (continued)
  - A submittal that describes the methodology that was used to perform the analysis.
  - The result of the analysis including the minimum available NPSH margin for the ECCS and CSS pumps with an unblocked sump screen
- Changes to licensing bases
  - Licensing Actions
  - Exemption Requests
- Programmatic controls for material in containment
- Provide JCO, if needed

Backfit Analysis

- Generic Letter requests information only - no backfit
- Simplified cost/benefit analysis is currently being performed at the request of the CRGR

(8)

OMB Control No.:

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555

NRC GENERIC LETTER 2004-XX: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to:

~~(1) Request that addressees perform an evaluation of the emergency core cooling system (ECCS) or containment spray system (CSS) recirculation functions in light of the information provided in this letter and, if appropriate, take additional actions to ensure their compliance with 10 CFR 50.46(b)(5), which requires long-term reactor core cooling, and other existing regulatory requirements listed in this generic letter. This request is based on the identified potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS) and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.~~

(1) Request that addressees submit information~~, as specified in this letter,~~ to the NRC to confirm compliance with 10 CFR 50.46(b)(5), which requires long-term reactor core cooling, and other existing regulatory requirements listed in this generic letter. This request is based on ~~information in this letter concerning~~ the identified potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS) and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

~~(3) Require addressees to inform the NRC of the extent to which they will take the requested actions.~~

(2) Require addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

Background

In 1979, as a result of evolving staff concerns related to the adequacy of PWR recirculation sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." To support the resolution of USI A-43, the NRC undertook an extensive research program, the technical findings of which are summarized in NUREG-0897, "Containment Emergency Sump Performance," dated October 1985. The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985. Although the staff's regulatory analysis concerning USI A-43 did not support imposing new sump performance requirements upon licensees of operating PWRs or boiling-water reactors (BWRs), the staff recommended in GL 85-22 that all affected reactor licensees replace the 50–percent blockage assumption (under which most nuclear power plants had been licensed) with a comprehensive, mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. The 50–percent screen blockage assumption does not require a plant-specific evaluation of the debris-blockage potential and may result in a non-conservative analysis for screen blockage effects. The staff also updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," to reflect the USI A-43 technical findings documented in NUREG-0897.

Following the resolution of USI A-43 in 1985, several events occurred that challenged the conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs:

- On July 28, 1992, at Barsebäck Unit 2, a Swedish BWR, the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray system suction strainers with mineral wool and required operators to shut down the spray pumps and backflush the strainers.

- In 1993, at Perry Unit 1, two events occurred during which ECCS strainers became plugged with debris. On January 16, ECCS strainers were plugged with suppression pool particulate matter, and on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.

- On September 11, 1995, at Limerick Unit 1, following a manual scram due to a stuck-open safety/relief valve, operators observed fluctuating flow and pump motor current on the A loop of suppression pool cooling. The licensee later attributed these indications to a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these ECCS suction strainer plugging events, the NRC issued several generic

communications, including Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994, Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995, and Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. These bulletins requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a loss-of-coolant accident (LOCA). The NRC staff has concluded that all BWR licensees have sufficiently addressed these bulletins.

However, findings from research to resolve the BWR strainer clogging issue have raised questions concerning the adequacy of PWR sump designs. In comparison to the technical findings of the USI A-43 research program concerning PWRs, the research findings demonstrate that the amount of debris generated by a high-energy line break (HELB) could be greater, that the debris could be finer (and, thus, more easily transportable), and that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss than an equivalent amount of either type of debris alone. These research findings prompted the NRC to open Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the ECCS and CSS in recirculation mode at PWRs during LOCAs or other HELB accidents for which sump recirculation is required.

On June 9, 2003, having completed its technical assessment of GSI-191 (summarized below in the Discussion section of this generic letter), the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors." As a result of the emergent issues discussed therein, the bulletin requested an expedited response from PWR licensees as to the status of their compliance on a mechanistic basis, with regulatory requirements concerning the ECCS and CSS recirculation functions. Addressees who were unable to assure regulatory compliance pending further analysis were asked to describe any interim compensatory measures that have been implemented or will be implemented to reduce risk until the analysis could be completed. All licensees have since responded to Bulletin 2003-01. In developing Bulletin 2003-01, the NRC staff recognized that it may be necessary for addressees to undertake complex evaluations to determine whether regulatory compliance exists in light of the concerns identified in the bulletin and that the methodology needed to perform ~~such~~these evaluations was not currently available. As a result, that information was not requested in the bulletin but addressees were informed that the staff was preparing a generic letter that would request this information. This generic letter is the follow-on information request referenced in the bulletin.

In response to Bulletin 2003-01, PWR licensees that chose not to confirm regulatory compliance implemented or ~~plan~~planned to implement compensatory measures to reduce risk or otherwise enhance the capability of the ECCS and CSS recirculation functions. During the process of resolving the potential concerns identified in this generic letter, the revised analysis of sump performance may affect addressees' understanding of their facilities' ECCS and CSS recirculation capabilities, Therefore, addressees may find it necessary to reevaluate the adequacy of their compensatory measures in light of the new information and take further action

as appropriate and necessary. Upon resolution of the potential concerns identified in this generic letter and the completion of any corrective actions resulting from that resolution, addressees may consider continuing, revising, or retiring their compensatory measures as appropriate.

The NRC has developed a Web page to keep the public informed of generic activities on PWR sump performance (http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance.html). This page provides links to information on PWR sump performance issues, along with documentation of NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update this Web page as new information becomes available.

Discussion

In the event of a HELB inside the containment of a PWR, energetic pressure waves and fluid jets would impinge upon materials in the vicinity of the break, such as thermal insulation, coatings, and concrete, causing them to become damaged and dislodged. Debris could also be generated through secondary mechanisms, such as severe post-accident temperature and humidity conditions, flooding of the lower containment, and the impact of containment spray droplets. In addition to debris generated by jet forces from the pipe rupture, debris can be created by the chemical reaction between the chemically reactive spray solutions used following a LOCA and the materials in containment. These reactions may result in additional debris such as disbonded coatings and chemical precipitants being generated. Through transport methods such as entrainment in the steam/water flows issuing from the break and containment spray washdown, a fraction of the generated debris and foreign material in the containment would be transported to the pool of water formed on the containment floor. Subsequently, if the ECCS or CSS pumps were to take suction from the recirculation sump, the debris suspended in the containment pool would begin to accumulate on the sump screen or be transported through the associated system. The accumulation of this suspended debris on the sump screen could create a roughly uniform covering on the screen, referred to as a debris bed, which would tend to increase the head loss across the screen through a filtering action. If a sufficient amount of debris were to accumulate, the debris bed would reach a critical thickness at which the head loss across the debris bed would exceed the net positive suction head (NPSH) margin required to ensure the successful operation of the ECCS and CSS pumps in recirculation mode. A loss of NPSH margin for the ECCS or CSS pumps as a result of the accumulation of debris on the recirculation sump screen, referred to as sump clogging, could result in degraded pump performance and eventual pump failure. Debris could also plug or wear close tolerance components within the ECCS or CSS systems. The effect of this plugging or wear may cause a component to degrade to the point where it may be unable to perform its designated function (i.e. pump fluid, maintain system pressure, or pass and control system flow.)

Assessing the likelihood of the ECCS and CSS pumps at domestic PWRs experiencing a debris-induced loss of NPSH margin during sump recirculation was the primary objective of the NRC's technical assessment of GSI-191. The NRC's technical assessment culminated in a parametric study that mechanistically treated phenomena associated with debris blockage using analytical models of domestic PWRs generated with a combination of generic and plant-specific data. As documented in Volume 1 of NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric

Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002, the GSI-191 parametric study concludes that recirculation sump clogging is a credible concern for domestic PWRs. As a result of limitations with respect to plant-specific data and other modeling uncertainties, however, the parametric study does not definitively identify whether or not particular PWR plants are vulnerable to sump clogging when phenomena associated with debris blockage are modeled mechanistically.

The methodology employed by the GSI-191 parametric study is based upon the substantial body of test data and analyses that are documented in technical reports generated during the NRC's GSI-191 research program and earlier technical reports generated by the NRC and the industry during the resolution of the BWR strainer clogging issue and USI A-43. These pertinent technical reports, which cover debris generation, transport, accumulation, and head loss, are incorporated by reference into the GSI-191 parametric study:

- NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," dated August 2002.
- NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," dated August 2002.
- NUREG/CR-6762, Vol. 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," dated August 2002.
- NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995.

~~In light of the new information identified during the efforts to resolve GSI-191, the NRC staff has determined that the previous guidance used to develop current licensing-basis analyses does not adequately and completely model sump screen debris blockage and related effects. As a result, the staff is revising their guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS. Therefore, the NRC staff determined that it is appropriate to request that addressees perform new, more realistic analyses and submit information to confirm their plant-specific compliance with NRC regulations and other existing regulatory requirements listed in this generic letter pertaining to post-accident debris blockage.~~

In addition to demonstrating the potential for debris to clog containment recirculation sumps, operational experience and the NRC's technical assessment of GSI-191 have also identified three integrally related modes by which post-accident debris blockage could adversely affect the sump screen's design function of intercepting debris that could impede or prevent the operation of the ECCS and CSS in recirculation mode.

First, as a result of the 50-percent blockage assumption, most PWR sump screens were designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, PWR sump screens may not be capable of accommodating the increased structural loadings that would occur due to mechanistically determined debris beds that cover essentially the entire screen surface.

Inadequate structural reinforcement of a sump screen may result in its deformation, damage, or failure, which could allow large quantities of debris to be ingested into the ECCS and CSS piping, pumps, and other components, potentially leading to their clogging or failure. The ECCS strainer plugging and deformation events that occurred at Perry Unit 1 (further described in Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993, and LER 50-440/93-011, "Excessive Strainer Differential Pressure Across the RHR Suction Strainer Could Have Compromised Long Term Cooling During Post-LOCA Operation," submitted May 19, 1993), demonstrate the credibility of this concern for screens and strainers that have not been designed with adequate reinforcement.

Second, in some PWR containments, the flowpaths by which containment spray or break flows return to the recirculation sump may include "choke-points," where the flowpath becomes so constricted that it could become blocked with debris following a HELB. Examples of potential choke-points are drains for pools, cavities, isolated containment compartments, and constricted drainage paths between physically separated containment elevations. Debris blockage at certain choke-points could hold up substantial amounts of water required for adequate recirculation or cause the water to be diverted into containment volumes that do not drain to the recirculation sump. The holdup or diversion of water assumed to be available to support sump recirculation could result in an available NPSH for ECCS and CSS pumps that is lower than the analyzed value, thereby reducing assurance that recirculation would successfully function. A reduced available NPSH directly concerns sump screen design because the NPSH margin of the ECCS and CSS pumps must be conservatively calculated to determine correctly the required surface area of passive sump screens when mechanistically determined debris loadings are considered. Although the parametric study (NUREG/CR-6762, Volume 1) did not analyze in detail the potential for the holdup or diversion of recirculation sump inventory, the NRC's GSI-191 research identified this phenomenon as an important and potentially credible concern. A number of LERs associated with this concern have also been generated, which further confirms its credibility and potential significance:

- LER 50-369/90-012, "Loose Material Was Located in Upper Containment During Unit Operation Because of an Inappropriate Action," McGuire Unit 1, submitted August 30, 1990.

- LER 50-266/97-006, "Potential Refueling Cavity Drain Failure Could Affect Accident Mitigation," Point Beach Unit 1, submitted February 19, 1997.

- LER 50-455/97-001, "Unit 2 Containment Drain System Clogged Due to Debris," Byron Unit 2, submitted April 17, 1997.

- LER 50-269/97-010, "Inadequate Analysis of ECCS Sump Inventory Due to Inadequate Design Analysis," Oconee Unit 1, submitted January 8, 1998.

- LER 50-315/98-017, "Debris Recovered from Ice Condenser Represents Unanalyzed Condition," D.C. Cook Unit 1, submitted July 1, 1998.

Third, debris blockage at flow restrictions within the ECCS recirculation flowpath downstream of

the sump screen is a potential concern for PWRs. Debris that is capable of passing through the recirculation sump screen may have the potential to become lodged at a downstream flow restriction, such as a high-pressure safety injection (HPSI) throttle valve or fuel assembly inlet debris screen. Debris blockage at such flow restrictions in the ECCS flowpath could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, debris blockage at flow restrictions in the CSS flowpath, such as a containment spray nozzle, could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Debris may also accumulate in close tolerance sub-components of pumps and valves. The effect may either be to plug the sub-component thereby rendering the component unable to perform its function or to wear critical close tolerance sub-components to the point at which component or system operation is degraded and unable to fully perform its function. Considering the recirculation sump screen's design function of intercepting potentially harmful debris, it is essential that the screen openings are adequately sized and that the sump screen's current configuration is free of gaps or breaches which could compromise the ECCS and CSS recirculation functions. It is also essential that system components are designed and evaluated to be able to operate with debris laden fluid as necessary post-LOCA.

10 CFR 50.46 (c)(2) defines an evaluation model as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure. While not a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the residual heat removal system are configured properly to provide enough flow to ensure long-term cooling. Therefore, the staff considers the modeling of sump performance as the validation of assumptions made in the ECCS evaluation model.

Based on the new information identified during the efforts to resolve GSI-191, the staff has determined that the previous guidance used to develop current licensing-basis analyses does not adequately and completely model sump screen debris blockage and related effects. The deficiencies in the previous guidance potentially resulted an analytical error that could result in ECCS performance that does not conform with the requirements in 10 CFR 50.46(b)(5). As a result, the staff is revising their guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS. The new information coming from the resolution of GSI, had it been known at the time, would have been included in the original guidance. In light of this new information, the staff has determined that it is appropriate to request that addressees submit information to confirm their plant-specific compliance with NRC regulations and other existing regulatory requirements listed in this generic letter pertaining to post-accident debris blockage. If addressees perform an analysis to confirm compliance, the staff recommends the use of an analysis method that mechanistically accounts for debris generation and transport, post accident equipment and systems operation with debris laden fluid. Since the modeling of sump performance is a boundary calculation for the ECCS evaluation model, the requirements of 10 CFR 50.46(a)(3)(ii) are applicable in this situation. In addition to the reporting requirement,

10 CFR 50.46(a)(3)(ii) requires that affected applicants or licensees shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with 10 CFR 50.46 requirements.

To assist in determining, on a plant-specific basis, whether compliance exists with 10 CFR 50.46(b)(5), addressees may use the guidance contained in Regulatory Guide 1.82 (RG 1.82), Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated November 2003. Revision 3 enhanced the debris blockage evaluation guidance for pressurized water reactors provided in Revision 1 of the regulatory guide to ~~better~~more realistically model sump screen debris blockage and related effects. The NRC staff determined after the issuance of Revision 2 that research for PWRs indicated that the guidance in that revision was not comprehensive enough to ensure adequate evaluation of a PWR plant's susceptibility to the detrimental effects caused by debris accumulation on debris interceptors (e.g., trash racks and sump screens). Revision 2 altered the debris blockage evaluation guidance found in Revision 1 following the evaluation of blockage events, such as the Barsebäck Unit 2 event mentioned above, but for BWRs only. Revision 1 replaced the 50-percent blockage assumption in Revision 0 with a comprehensive, mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. This was in response to the findings of USI A-43. In addition, the NRC staff is reviewing generic industry guidance and will issue a safety evaluation report endorsing portions or all of the generic industry guidance, if found acceptable. Once approved, this guidance may also be used to assist in determining the status of regulatory compliance. Individual addressees may also develop alternative approaches to those named in this paragraph for responding to this generic letter; however, additional staff review may be required to assess the adequacy of such approaches.
~~If the industry guidance will not be available when the generic letter is issued, the NRC will provide additional guidance for determining on a plant-specific basis whether compliance exists with 10 CFR 50.46(b)(5).~~

The time frames for addressee responses in this generic letter were selected to 1) allow adequate time for addressees to perform an analysis, 2) allow addressees to properly design and install any identified modifications, as necessary, 3) allow addressees adequate time to obtain NRC approval, as necessary, for any licensing basis changes, and 4) allow for the closure of the generic issue in accordance with the published schedule. These time frames are appropriate since all addressees have responded to Bulletin 2003-01 and will, if necessary, implement compensatory measures until the issues identified in this generic letter are resolved.

~~During the process of responding to this generic letter, an addressee may determine that they no longer comply with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. In this event, the staff intends to exercise enforcement discretion, consistent with Section VII.B. of the Enforcement Policy provided:~~

- ~~Addressees implement compensatory measures to reduce risk. These measures may include those actions taken in the responses to Bulletin 2003-01.~~
- ~~Addressees complete all corrective actions by December 31, 2007~~

Applicable Regulatory Requirements

NRC regulations in Title 10, of the *Code of Federal Regulations* Section 50.46,(10 CFR 50.46), require that the ECCS must satisfy five criteria, one of which is to provide the capability for long-term cooling of the reactor core following a LOCA. The ECCS must have the capability to provide decay heat removal, such that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core. For PWRs licensed to the General Design Criteria (GDCs) in Appendix A to 10 CFR Part 50, GDC 35 specifies additional ECCS requirements.

Similarly, for PWRs licensed to the GDCs in Appendix A to 10 CFR Part 50, GDC 38 provides requirements for containment heat removal systems, and GDC 41 provides requirements for containment atmosphere cleanup. Many PWR licensees credit a CSS, at least in part, with performing the safety functions to satisfy these requirements, and PWRs that are not licensed to the GDCs may similarly credit a CSS to satisfy licensing basis requirements. In addition, PWR licensees may credit a CSS with reducing the accident source term to meet the limits of 10 CFR Part 100 or 10 CFR 50.67.

10 CFR 50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors, requires nuclear power reactor licenses to notify the NRC of:

> (i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan; or
>
> (ii) Those non-emergency events specified in paragraph (b) of the rule that occurred within three years of the date of discovery.

10 CFR 50.73, Licensee Event Report System, requires that the holder of an operating license for a nuclear power plant submit a Licensee Event Report for any of the events described in the rule within 60 days after the discovery of the event. Unless otherwise specified in the rule, the licensee shall report an event if it occurred within three years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

Criterion XVI (Corrective Action) of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions.

If, in the course of preparing a response to the requested information, an addressee determines that it is not in compliance with ~~their~~its current licensing basis, the addressee is expected to take appropriate action in accordance with requirements of Appendix B to 10 CFR Part 50 and the plant technical specifications to restore the facility to compliance.

Applicable Regulatory Guidance[1]

Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003.

~~Requested Action~~

~~All addressees are requested to take the actions discussed below to confirm their compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This request is based on the NRC staff determination that the previous guidance used to develop current licensing-basis analyses does not adequately and completely model sump screen debris blockage and related effects. As a result, the staff is revising their guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS:~~

~~(1) Using an NRC approved methodology, perform a mechanistic evaluation of the potential for the accumulation of debris to impede or prevent the recirculation functions of the ECCS and CSS following all postulated accidents for which the recirculation of these systems is required. Individual addressees may also use alternative methodologies to those already approved by the NRC; however, additional staff review may be required to assess the adequacy of such approaches.~~

~~(2) Implement any plant modifications that the above evaluation identifies as being necessary to ensure compliance with NRC regulations~~Generic Letter 91-18, Rev. 1," Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," October 8, 1997.

Requested Information

All addressees are requested to provide the following information:

1. Within 60 days of the ~~date~~issuance of the ~~safety evaluation providing the~~staff approved guidance for performing the requested evaluation, addressees are requested to provide information regarding their planned actions and schedule to complete the requested evaluation. The provided information should include the following:

   (a) A description of the methodology used or that will be used to analyze the susceptibility of the ECCS and CSS recirculation functions for your reactor to adverse effects of post-accident debris blockage and operation with debris laden fluids identified in this generic letter. Provide the completion date of ~~the~~any analysis that will be performed.

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[1] The NRC staff is currently reviewing evaluation guidance developed by the industry. The NRC staff intends to document its review in a safety evaluation which licensees can reference as regulatory guidance.

(b) ~~P~~If a mechanistic analysis was or will be performed to confirm compliance, provide a statement of whether or not you plan to perform a containment walkdown surveillance in support of the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of debris blockage identified in this generic letter. Provide justification if no containment walkdown surveillance will be performed. If a containment walkdown surveillance will be performed, state the planned methodology to be used and the planned completion date.

2. Addressees are requested to provide no later than September 1, 2005, information that confirms their compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter~~ once their licensing basis has been updated to reflect the results of the analysis performed above.~~.

(a) Provide confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made.

(b) A general description of and implementation schedule for all corrective actions, if any, including any plant modifications that may be necessary to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. ~~Provide justification for any~~Initiate actions to implement corrective ~~action that will not be completed by the end of~~actions no later than the first refueling outage starting after April 1, 2006~~. The staff's expectation is that~~; however, all corrective actions ~~will~~should be completed by December 31, 2007.~~ The staff does not intend to exercise enforcement discretion as described in the Discussion section of this generic letter for non-compliances with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter past December 31, 2007.~~ If all corrective actions will not be completed by December 31, 2007, describe how the ~~regulatory requirements discussed in the Applicable Regulatory Requirements section will be met until the corrective actions are complete.~~

~~(c)~~delays are consistent with the 10 CFR 50.46(a)(3)(ii) requirement to take immediate steps to demonstrate compliance.

(c) A submittal that describes the methodology that was used to perform the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris laden fluids. The submittal may reference a guidance document (e.g. Regulatory Guide 1.82, Rev 3, industry guidance) or other methodology previously submitted to the NRC. The submittal may also reference the 60-day response ~~to Section 1 above. The~~described above. If a mechanistic analysis was performed to confirm compliance, the documents to be submitted or referenced should include the ~~results of~~methodology for conducting a~~ny~~ supporting containment

walkdown surveillance ~~performed~~used to identify potential debris sources and other pertinent containment characteristics.

(d) ~~T~~If a mechanistic analysis was performed to confirm compliance, the submittal should include, at a minimum, the following information:

(~~i~~i) The minimum available NPSH margin for the ECCS and CSS pumps with an unblocked sump screen.

(ii) The extent of submergence of the sump screen (i.e., partial or full) at the time of the switchover to sump recirculation, and the submerged area of the sump screen at this time.

(iii) The maximum head loss postulated from debris accumulation on the submerged sump screen, and a description of the primary constituents of the debris bed that result in this head loss. In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) and CSS washdown should be considered in the analyses. Examples of this type of debris are disbonded coatings in the form of chips and particulates or chemical precipitants caused by chemical reactions in the pool.

(iv) The basis for concluding that water inventory required to ensure adequate ECCS or CSS recirculation would not be held up or diverted by debris blockage at choke-points in containment recirculation sump return flowpaths.

(v) The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen, such as a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles. The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

(vi) Verification that close tolerance sub-components in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive ~~-~~wear due to extended post accident operation with debris laden fluids.

(vii) If an active approach (e.g. back flushing, powered screens, etc.) is selected in lieu of or in addition to a passive approach to mitigate the effects of the debris blockage, describe the approach and associated analyses.

(e) A general description of and planned schedule for any changes to the plant

licensing bases resulting from any analysis or plant modification done to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included with this submittal.

(f) A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 98-04 to the extent that their responses address these specific foreign material control issues.

(g) If an addressee determines that it will not be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter until the corrective actions identified in their submittal are complete, a submittal justifying continued operation should be provided. Items which may be considered in this submittal are design features, probability of initiating events, operator actions, and the compensatory measures implemented as part of the response to Bulletin 2003-01.

Required Response

In accordance with 10 CFR 50.54(f), the subject PWR addressees are required to submit written responses to this generic letter. This information is sought to verify licensees' compliance with ~~the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter once their~~current licensing basis ~~has been updated to reflect the results of the mechanistic analysis requested in this generic letter. This request is based on the identified potential susceptibility of PWR recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of ECCS and CSS and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage. The~~for the subject PWR addressees. The addressees have two options:

(1) addressees may choose to submit written responses providing the information requested above within the requested time periods, or

(2) addressees who choose not to provide information requested or cannot meet the requested completion dates are required to submit written responses within 30 days of the date of this generic letter. The responses must address any alternative course of action proposed, including the basis for the acceptability of the proposed alternative course of action.

The required written responses should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, Maryland 20852, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, a copy of a response should be

submitted to the appropriate regional administrator.

The NRC staff will review the responses to this generic letter and will notify affected addressees if concerns are identified regarding compliance with NRC regulations and their current licensing bases. The staff may also conduct inspections to determine addressees' effectiveness in addressing the generic letter.

Reasons for Information Request

As discussed above, research and analysis suggests that (1) the potential for the failure of the ECCS and CSS recirculation functions as a result of debris blockage is not adequately addressed in most PWR licensees' current safety analyses, and (2) the ECCS and CSS recirculation functions at a significant number of operating PWRs could ~~potentially~~ become degraded as a result of the potential effects of debris blockage or extended operation with debris laden fluids ~~as~~ identified in this generic letter. An ECCS that is incapable of providing long-term reactor core cooling through recirculation operation would be in violation of 10 CFR 50.46. A CSS that is incapable of functioning in recirculation mode may not comply with GDCs 38 and 41 or other plant-specific licensing requirements or safety analyses. Bulletin 2003-01 requested information to verify addressees' compliance with NRC regulations and to ensure that any interim risks associated with post-accident debris blockage are minimized while evaluations to determine compliance proceed. This generic letter is the follow-on generic communication to Bulletin 2003-01 ~~which is requesting information on the results of the evaluations referenced in the bulletin. This information is sought to verify licensees' compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter once their licensing basis has been updated to reflect the results of the mechanistic analysis~~. Therefore, the information requested in this generic letter~~. This request is based on the identified potential~~ is necessary to confirm plant-specific compliance with 10 CFR 50.46 and other existing regulations in light of the new staff guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS ~~and~~or CSS ~~and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.~~.

The NRC staff will also use the requested information to (1) determine whether a sample auditing approach is acceptable for verifying that addressees have resolved the concerns identified in this generic letter, (2) assist in determining which addressees would be subject to the proposed sample audits, (3) provide confidence that any nonaudited addressees have addressed the concerns identified in this generic letter, and (4) assess the need for and guide the development of any additional regulatory actions that may be necessary to address the adequacy of the ECCS and CSS recirculation functions.

Related Generic Communications

- Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," June 9, 2003.

- Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by

Debris in Boiling-Water Reactors," May 6, 1996.

- Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in the Suppression Pool Cooling Mode," October 17, 1995.
- Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," May 11, 1993.
- Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," February 18, 1994.
- Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," July 14, 1998.
- Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.
- Generic Letter 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985.
- Generic Letter 91-18, Rev. 1," Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," October 8, 1997.
- Information Notice 97-13, "Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants," March 24, 1997.
- Information Notice 96-59, "Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris," October 30, 1996.
- Information Notice 96-55, "Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions," October 22, 1996.
- Information Notice 96-27, "Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation," May 1, 1996.
- Information Notice 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances," February 13, 1996.
- Information Notice 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," October 4, 1995.
- Information Notice 95-47, Revision 1, "Unexpected Opening of a Safety/Relief Valve and

Complications Involving Suppression Pool Cooling Strainer Blockage," November 30, 1995.

- Information Notice 95-06, "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment," January 25, 1995.
- Information Notice 94-57, "Debris in Containment and the Residual Heat Removal System," August 12, 1994.
- Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," April 26, 1993.
- Information Notice 93-34, Supplement 1, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," May 6, 1993.
- Information Notice 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," December 23, 1992.
- Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," September 30, 1992.
- Information Notice 89-79, "Degraded Coatings and Corrosion of Steel Containment Vessels," December 1, 1989.
- Information Notice 89-79, Supplement 1, "Degraded Coatings and Corrosion of Steel Containment Vessels," June 29, 1990.
- Information Notice 89-77, "Debris in Containment Emergency Sumps and Incorrect Screen Configurations," November 21, 1989.
- Information Notice 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," May 19, 1988.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, ~~10CFR 50.109(a)(4)(i)~~ and 10CFR 50.54(f), ~~this generic letter requests that addressees evaluate their facilities to confirm compliance with the existing applicable regulatory requirements previously outlined in~~ this generic letter~~. This generic letter also~~ transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements~~. The staff has determined that, in light of the new information identified during the efforts to resolve GSI-191, the previous guidance used to develop most addressees' current licensing-basis analyses does not adequately and completely model sump screen debris blockage and related effects. This new information, had it been known at the time, would have been included in the original guidance. As a result, the staff revised their guidance for determining the susceptibility of PWR recirculation sump screens to~~ (see the Applicable Regulatory Requirements section of this

generic letter). Specifically, the required information will enable the NRC staff to determine whether the emergency core cooling system (ECCS) and containment spray system (CSS) at reactor facilities are able to perform their safety functions following all postulated accidents for which ECCS or CSS recirculation is required while taking into account the adverse effects of post-accident debris blockage ~~during design basis accidents requiring recirculation operation of the ECCS or CSS. Thus, the information requested by~~and operation with debris laden fluids. No backfit is either intended or approved by the issuance of this generic letter~~is considered a compliance exception to the rule in accordance with 10 CFR 50.109(a)(4)(i)~~, and the staff has not performed a ~~detailed~~ backfit analysis. ~~However, the NRC staff did perform a simplified backfit analysis, which is publicly available in the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession Number MLXXXXXXXXX.~~

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this generic letter is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

- or-

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of the Office of Management and Budget (OMB).

Federal Register Notification

~~The NRC published a~~A notice of opportunity for public comment on this generic letter was published in the ~~*Federal Register*~~
Federal Register (69 FR16980) on March 31, 2004. ~~In addition, the NRC has provided opportunities for public comment at several public meetings. As the resolution of this matter progresses, the NRC will continue to provide opportunities for further public involvement~~Comments were received from ten industry groups, one non-profit organization, one private citizen, and the State of New Jersey. The staff considered all comments that were received. The staff's evaluation of the comments is publicly available through the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession No. xxxxxxxxx.

Paperwork Reduction Act Statement

This generic letter contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB) under approval number XXXX-XXXX which expires on XXX XX, XXXX.

The burden to the public for these mandatory information collections is estimated to average ~~10,000~~5000 hours per response, including the time for reviewing instructions, searching existing

data sources, gathering and maintaining the necessary data, and completing and reviewing the information collections. Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Records Management Branch, Mail Stop T-6 E6, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may neither conduct nor sponsor, and an individual is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contacts or lead project manager listed below.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555

NRC GENERIC LETTER 2004-XX: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to:

(1) Request that addressees submit information to the NRC to confirm compliance with 10 CFR 50.46(b)(5), which requires long-term reactor core cooling, and other existing regulatory requirements listed in this generic letter. This request is based on the identified potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS) and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

(2) Require addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

Background

In 1979, as a result of evolving staff concerns related to the adequacy of PWR recirculation sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." To support the resolution of USI A-43, the NRC undertook an extensive research program, the technical findings of which are summarized in NUREG-0897, "Containment Emergency Sump Performance," dated October 1985. The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985. Although the staff's regulatory analysis concerning USI A-43 did not support imposing new sump

performance requirements upon licensees of operating PWRs or boiling-water reactors (BWRs), the staff recommended in GL 85-22 that all affected reactor licensees replace the 50–percent blockage assumption (under which most nuclear power plants had been licensed) with a comprehensive, mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. The 50–percent screen blockage assumption does not require a plant-specific evaluation of the debris-blockage potential and may result in a non-conservative analysis for screen blockage effects. The staff also updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," to reflect the USI A-43 technical findings documented in NUREG-0897.

Following the resolution of USI A-43 in 1985, several events occurred that challenged the conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs:

- On July 28, 1992, at Barsebäck Unit 2, a Swedish BWR, the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray system suction strainers with mineral wool and required operators to shut down the spray pumps and backflush the strainers.

- In 1993, at Perry Unit 1, two events occurred during which ECCS strainers became plugged with debris. On January 16, ECCS strainers were plugged with suppression pool particulate matter, and on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.

- On September 11, 1995, at Limerick Unit 1, following a manual scram due to a stuck-open safety/relief valve, operators observed fluctuating flow and pump motor current on the A loop of suppression pool cooling. The licensee later attributed these indications to a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these ECCS suction strainer plugging events, the NRC issued several generic communications, including Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994, Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995, and Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. These bulletins requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a loss-of-coolant accident (LOCA). The NRC staff has concluded that all BWR licensees have sufficiently addressed these bulletins.

However, findings from research to resolve the BWR strainer clogging issue have raised questions concerning the adequacy of PWR sump designs. In comparison to the technical findings of the USI A-43 research program concerning PWRs, the research findings

demonstrate that the amount of debris generated by a high-energy line break (HELB) could be greater, that the debris could be finer (and, thus, more easily transportable), and that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss than an equivalent amount of either type of debris alone. These research findings prompted the NRC to open Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the ECCS and CSS in recirculation mode at PWRs during LOCAs or other HELB accidents for which sump recirculation is required.

On June 9, 2003, having completed its technical assessment of GSI-191 (summarized below in the Discussion section of this generic letter), the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors." As a result of the emergent issues discussed therein, the bulletin requested an expedited response from PWR licensees as to the status of their compliance on a mechanistic basis, with regulatory requirements concerning the ECCS and CSS recirculation functions. Addressees who were unable to assure regulatory compliance pending further analysis were asked to describe any interim compensatory measures that have been implemented or will be implemented to reduce risk until the analysis could be completed. All licensees have since responded to Bulletin 2003-01. In developing Bulletin 2003-01, the NRC staff recognized that it may be necessary for addressees to undertake complex evaluations to determine whether regulatory compliance exists in light of the concerns identified in the bulletin and that the methodology needed to perform these evaluations was not currently available. As a result, that information was not requested in the bulletin but addressees were informed that the staff was preparing a generic letter that would request this information. This generic letter is the follow-on information request referenced in the bulletin.

In response to Bulletin 2003-01, PWR licensees that chose not to confirm regulatory compliance implemented or planned to implement compensatory measures to reduce risk or otherwise enhance the capability of the ECCS and CSS recirculation functions. During the process of resolving the potential concerns identified in this generic letter, the revised analysis of sump performance may affect addressees' understanding of their facilities' ECCS and CSS recirculation capabilities, Therefore, addressees may find it necessary to reevaluate the adequacy of their compensatory measures in light of the new information and take further action as appropriate and necessary. Upon resolution of the potential concerns identified in this generic letter and the completion of any corrective actions resulting from that resolution, addressees may consider continuing, revising, or retiring their compensatory measures as appropriate.

The NRC has developed a Web page to keep the public informed of generic activities on PWR sump performance (http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance.html). This page provides links to information on PWR sump performance issues, along with documentation of NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update this Web page as new information becomes available.

Discussion

In the event of a HELB inside the containment of a PWR, energetic pressure waves and fluid jets would impinge upon materials in the vicinity of the break, such as thermal insulation, coatings, and concrete, causing them to become damaged and dislodged. Debris could also be generated through secondary mechanisms, such as severe post-accident temperature and humidity conditions, flooding of the lower containment, and the impact of containment spray droplets. In addition to debris generated by jet forces from the pipe rupture, debris can be created by the chemical reaction between the chemically reactive spray solutions used following a LOCA and the materials in containment. These reactions may result in additional debris such as disbonded coatings and chemical precipitants being generated. Through transport methods such as entrainment in the steam/water flows issuing from the break and containment spray washdown, a fraction of the generated debris and foreign material in the containment would be transported to the pool of water formed on the containment floor. Subsequently, if the ECCS or CSS pumps were to take suction from the recirculation sump, the debris suspended in the containment pool would begin to accumulate on the sump screen or be transported through the associated system. The accumulation of this suspended debris on the sump screen could create a roughly uniform covering on the screen, referred to as a debris bed, which would tend to increase the head loss across the screen through a filtering action. If a sufficient amount of debris were to accumulate, the debris bed would reach a critical thickness at which the head loss across the debris bed would exceed the net positive suction head (NPSH) margin required to ensure the successful operation of the ECCS and CSS pumps in recirculation mode. A loss of NPSH margin for the ECCS or CSS pumps as a result of the accumulation of debris on the recirculation sump screen, referred to as sump clogging, could result in degraded pump performance and eventual pump failure. Debris could also plug or wear close tolerance components within the ECCS or CSS systems. The effect of this plugging or wear may cause a component to degrade to the point where it may be unable to perform its designated function (i.e. pump fluid, maintain system pressure, or pass and control system flow.)

Assessing the likelihood of the ECCS and CSS pumps at domestic PWRs experiencing a debris-induced loss of NPSH margin during sump recirculation was the primary objective of the NRC's technical assessment of GSI-191. The NRC's technical assessment culminated in a parametric study that mechanistically treated phenomena associated with debris blockage using analytical models of domestic PWRs generated with a combination of generic and plant-specific data. As documented in Volume 1 of NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002, the GSI-191 parametric study concludes that recirculation sump clogging is a credible concern for domestic PWRs. As a result of limitations with respect to plant-specific data and other modeling uncertainties, however, the parametric study does not definitively identify whether or not particular PWR plants are vulnerable to sump clogging when phenomena associated with debris blockage are modeled mechanistically.

The methodology employed by the GSI-191 parametric study is based upon the substantial body of test data and analyses that are documented in technical reports generated during the NRC's GSI-191 research program and earlier technical reports generated by the NRC and the industry during the resolution of the BWR strainer clogging issue and USI A-43. These pertinent technical reports, which cover debris generation, transport, accumulation, and head loss, are

incorporated by reference into the GSI-191 parametric study:

- NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," dated August 2002.
- NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," dated August 2002.
- NUREG/CR-6762, Vol. 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," dated August 2002.
- NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995.

In addition to demonstrating the potential for debris to clog containment recirculation sumps, operational experience and the NRC's technical assessment of GSI-191 have also identified three integrally related modes by which post-accident debris blockage could adversely affect the sump screen's design function of intercepting debris that could impede or prevent the operation of the ECCS and CSS in recirculation mode.

First, as a result of the 50-percent blockage assumption, most PWR sump screens were designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, PWR sump screens may not be capable of accommodating the increased structural loadings that would occur due to mechanistically determined debris beds that cover essentially the entire screen surface. Inadequate structural reinforcement of a sump screen may result in its deformation, damage, or failure, which could allow large quantities of debris to be ingested into the ECCS and CSS piping, pumps, and other components, potentially leading to their clogging or failure. The ECCS strainer plugging and deformation events that occurred at Perry Unit 1 (further described in Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993, and LER 50-440/93-011, "Excessive Strainer Differential Pressure Across the RHR Suction Strainer Could Have Compromised Long Term Cooling During Post-LOCA Operation," submitted May 19, 1993), demonstrate the credibility of this concern for screens and strainers that have not been designed with adequate reinforcement.

Second, in some PWR containments, the flowpaths by which containment spray or break flows return to the recirculation sump may include "choke-points," where the flowpath becomes so constricted that it could become blocked with debris following a HELB. Examples of potential choke-points are drains for pools, cavities, isolated containment compartments, and constricted drainage paths between physically separated containment elevations. Debris blockage at certain choke-points could hold up substantial amounts of water required for adequate recirculation or cause the water to be diverted into containment volumes that do not drain to the recirculation sump. The holdup or diversion of water assumed to be available to support sump recirculation could result in an available NPSH for ECCS and CSS pumps that is lower than the analyzed value, thereby reducing assurance that recirculation would successfully function. A reduced available NPSH directly concerns sump screen design because the NPSH margin of the

ECCS and CSS pumps must be conservatively calculated to determine correctly the required surface area of passive sump screens when mechanistically determined debris loadings are considered. Although the parametric study (NUREG/CR-6762, Volume 1) did not analyze in detail the potential for the holdup or diversion of recirculation sump inventory, the NRC's GSI-191 research identified this phenomenon as an important and potentially credible concern. A number of LERs associated with this concern have also been generated, which further confirms its credibility and potential significance:

- LER 50-369/90-012, "Loose Material Was Located in Upper Containment During Unit Operation Because of an Inappropriate Action," McGuire Unit 1, submitted August 30, 1990.
- LER 50-266/97-006, "Potential Refueling Cavity Drain Failure Could Affect Accident Mitigation," Point Beach Unit 1, submitted February 19, 1997.
- LER 50-455/97-001, "Unit 2 Containment Drain System Clogged Due to Debris," Byron Unit 2, submitted April 17, 1997.
- LER 50-269/97-010, "Inadequate Analysis of ECCS Sump Inventory Due to Inadequate Design Analysis," Oconee Unit 1, submitted January 8, 1998.
- LER 50-315/98-017, "Debris Recovered from Ice Condenser Represents Unanalyzed Condition," D.C. Cook Unit 1, submitted July 1, 1998.

Third, debris blockage at flow restrictions within the ECCS recirculation flowpath downstream of the sump screen is a potential concern for PWRs. Debris that is capable of passing through the recirculation sump screen may have the potential to become lodged at a downstream flow restriction, such as a high-pressure safety injection (HPSI) throttle valve or fuel assembly inlet debris screen. Debris blockage at such flow restrictions in the ECCS flowpath could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, debris blockage at flow restrictions in the CSS flowpath, such as a containment spray nozzle, could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Debris may also accumulate in close tolerance sub-components of pumps and valves. The effect may either be to plug the sub-component thereby rendering the component unable to perform its function or to wear critical close tolerance sub-components to the point at which component or system operation is degraded and unable to fully perform its function. Considering the recirculation sump screen's design function of intercepting potentially harmful debris, it is essential that the screen openings are adequately sized and that the sump screen's current configuration is free of gaps or breaches which could compromise the ECCS and CSS recirculation functions. It is also essential that system components are designed and evaluated to be able to operate with debris laden fluid as necessary post-LOCA.

10 CFR 50.46 (c)(2) defines an evaluation model as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs,

procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure. While not a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the residual heat removal system are configured properly to provide enough flow to ensure long-term cooling. Therefore, the staff considers the modeling of sump performance as the validation of assumptions made in the ECCS evaluation model.

Based on the new information identified during the efforts to resolve GSI-191, the staff has determined that the previous guidance used to develop current licensing-basis analyses does not adequately and completely model sump screen debris blockage and related effects. The deficiencies in the previous guidance potentially resulted an analytical error that could result in ECCS performance that does not conform with the requirements in 10 CFR 50.46(b)(5). As a result, the staff is revising their guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS. The new information coming from the resolution of GSI, had it been known at the time, would have been included in the original guidance. In light of this new information, the staff has determined that it is appropriate to request that addressees submit information to confirm their plant-specific compliance with NRC regulations and other existing regulatory requirements listed in this generic letter pertaining to post-accident debris blockage. If addressees perform an analysis to confirm compliance, the staff recommends the use of an analysis method that mechanistically accounts for debris generation and transport, post accident equipment and systems operation with debris laden fluid. Since the modeling of sump performance is a boundary calculation for the ECCS evaluation model, the requirements of 10 CFR 50.46(a)(3)(ii) are applicable in this situation. In addition to the reporting requirement, 10 CFR 50.46(a)(3)(ii) requires that affected applicants or licensees shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with 10 CFR 50.46 requirements.

To assist in determining, on a plant-specific basis, whether compliance exists with 10 CFR 50.46(b)(5), addressees may use the guidance contained in Regulatory Guide 1.82 (RG 1.82), Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated November 2003. Revision 3 enhanced the debris blockage evaluation guidance for pressurized water reactors provided in Revision 1 of the regulatory guide to more realistically model sump screen debris blockage and related effects. The NRC staff determined after the issuance of Revision 2 that research for PWRs indicated that the guidance in that revision was not comprehensive enough to ensure adequate evaluation of a PWR plant's susceptibility to the detrimental effects caused by debris accumulation on debris interceptors (e.g., trash racks and sump screens). Revision 2 altered the debris blockage evaluation guidance found in Revision 1 following the evaluation of blockage events, such as the Barsebäck Unit 2 event mentioned above, but for BWRs only. Revision 1 replaced the 50-percent blockage assumption in Revision 0 with a comprehensive, mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. This was in response to the findings of USI A-43. In addition, the NRC staff is reviewing generic industry guidance and will issue a safety evaluation report endorsing portions or all of the generic industry guidance, if found acceptable. Once approved, this guidance may also be used to assist in determining the status of regulatory

compliance. Individual addressees may also develop alternative approaches to those named in this paragraph for responding to this generic letter; however, additional staff review may be required to assess the adequacy of such approaches.

The time frames for addressee responses in this generic letter were selected to 1) allow adequate time for addressees to perform an analysis, 2) allow addressees to properly design and install any identified modifications, as necessary, 3) allow addressees adequate time to obtain NRC approval, as necessary, for any licensing basis changes, and 4) allow for the closure of the generic issue in accordance with the published schedule. These time frames are appropriate since all addressees have responded to Bulletin 2003-01 and will, if necessary, implement compensatory measures until the issues identified in this generic letter are resolved.

Applicable Regulatory Requirements

NRC regulations in Title 10, of the *Code of Federal Regulations* Section 50.46,(10 CFR 50.46), require that the ECCS must satisfy five criteria, one of which is to provide the capability for long-term cooling of the reactor core following a LOCA. The ECCS must have the capability to provide decay heat removal, such that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core. For PWRs licensed to the General Design Criteria (GDCs) in Appendix A to 10 CFR Part 50, GDC 35 specifies additional ECCS requirements.

Similarly, for PWRs licensed to the GDCs in Appendix A to 10 CFR Part 50, GDC 38 provides requirements for containment heat removal systems, and GDC 41 provides requirements for containment atmosphere cleanup. Many PWR licensees credit a CSS, at least in part, with performing the safety functions to satisfy these requirements, and PWRs that are not licensed to the GDCs may similarly credit a CSS to satisfy licensing basis requirements. In addition, PWR licensees may credit a CSS with reducing the accident source term to meet the limits of 10 CFR Part 100 or 10 CFR 50.67.

10 CFR 50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors, requires nuclear power reactor licenses to notify the NRC of:

> (i) The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan; or
>
> (ii) Those non-emergency events specified in paragraph (b) of the rule that occurred within three years of the date of discovery.

10 CFR 50.73, Licensee Event Report System, requires that the holder of an operating license for a nuclear power plant submit a Licensee Event Report for any of the events described in the rule within 60 days after the discovery of the event. Unless otherwise specified in the rule, the licensee shall report an event if it occurred within three years of the date of discovery regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

Criterion XVI (Corrective Action) of Appendix B to 10 CFR Part 50 states that measures shall be

established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions.

If, in the course of preparing a response to the requested information, an addressee determines that it is not in compliance with its current licensing basis, the addressee is expected to take appropriate action in accordance with requirements of Appendix B to 10 CFR Part 50 and the plant technical specifications to restore the facility to compliance.

Applicable Regulatory Guidance[1]

Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003.

Generic Letter 91-18, Rev. 1," Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," October 8, 1997.

Requested Information

All addressees are requested to provide the following information:

1. Within 60 days of the issuance of the staff approved guidance for performing the requested evaluation, addressees are requested to provide information regarding their planned actions and schedule to complete the requested evaluation. The provided information should include the following:

   (a) A description of the methodology used or that will be used to analyze the susceptibility of the ECCS and CSS recirculation functions for your reactor to adverse effects of post-accident debris blockage and operation with debris laden fluids identified in this generic letter. Provide the completion date of any analysis that will be performed.

   (b) If a mechanistic analysis was or will be performed to confirm compliance, provide a statement of whether or not you plan to perform a containment walkdown surveillance in support of the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of debris blockage identified in this generic letter. Provide justification if no containment walkdown surveillance will be performed. If a containment walkdown surveillance will be performed, state the planned methodology to be used and the planned completion date.

2. Addressees are requested to provide no later than September 1, 2005, information that confirms their compliance with the regulatory requirements listed in the Applicable

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[1] The NRC staff is currently reviewing evaluation guidance developed by the industry. The NRC staff intends to document its review in a safety evaluation which licensees can reference as regulatory guidance.

Regulatory Requirements section of this generic letter.

(a) Provide confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made.

(b) A general description of and implementation schedule for all corrective actions, if any, including any plant modifications that may be necessary to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Initiate actions to implement corrective actions no later than the first refueling outage starting after April 1, 2006; however, all corrective actions should be completed by December 31, 2007. If all corrective actions will not be completed by December 31, 2007, describe how the delays are consistent with the 10 CFR 50.46(a)(3)(ii) requirement to take immediate steps to demonstrate compliance.

(c) A submittal that describes the methodology that was used to perform the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris laden fluids. The submittal may reference a guidance document (e.g. Regulatory Guide 1.82, Rev 3, industry guidance) or other methodology previously submitted to the NRC. The submittal may also reference the 60-day response described above. If a mechanistic analysis was performed to confirm compliance, the documents to be submitted or referenced should include the methodology for conducting a supporting containment walkdown surveillance used to identify potential debris sources and other pertinent containment characteristics.

(d) If a mechanistic analysis was performed to confirm compliance, the submittal should include, at a minimum, the following information:

(i) The minimum available NPSH margin for the ECCS and CSS pumps with an unblocked sump screen.

(ii) The extent of submergence of the sump screen (i.e., partial or full) at the time of the switchover to sump recirculation, and the submerged area of the sump screen at this time.

(iii) The maximum head loss postulated from debris accumulation on the submerged sump screen, and a description of the primary constituents of the debris bed that result in this head loss. In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) and CSS washdown should be considered in the analyses. Examples of this type of debris are disbonded coatings in the form of chips and particulates or

chemical precipitants caused by chemical reactions in the pool.

(iv) The basis for concluding that water inventory required to ensure adequate ECCS or CSS recirculation would not be held up or diverted by debris blockage at choke-points in containment recirculation sump return flowpaths.

(v) The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen, such as a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles. The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

(vi) Verification that close tolerance sub-components in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post accident operation with debris laden fluids.

(vii) If an active approach (e.g. back flushing, powered screens, etc.) is selected in lieu of or in addition to a passive approach to mitigate the effects of the debris blockage, describe the approach and associated analyses.

(e) A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modification done to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included with this submittal.

(f) A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 98-04 to the extent that their responses address these specific foreign material control issues.

(g) If an addressee determines that it will not be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter until the corrective actions identified in their submittal are complete, a submittal justifying continued operation should be provided. Items which may be considered in this submittal are design features, probability of initiating events, operator actions, and the compensatory measures implemented as part of the response to Bulletin 2003-01.

Required Response

In accordance with 10 CFR 50.54(f), the subject PWR addressees are required to submit written responses to this generic letter. This information is sought to verify licensees' compliance with current licensing basis for the subject PWR addressees. The addressees have two options:

(1) addressees may choose to submit written responses providing the information requested above within the requested time periods, or

(2) addressees who choose not to provide information requested or cannot meet the requested completion dates are required to submit written responses within 30 days of the date of this generic letter. The responses must address any alternative course of action proposed, including the basis for the acceptability of the proposed alternative course of action.

The required written responses should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, Maryland 20852, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, a copy of a response should be submitted to the appropriate regional administrator.

The NRC staff will review the responses to this generic letter and will notify affected addressees if concerns are identified regarding compliance with NRC regulations and their current licensing bases. The staff may also conduct inspections to determine addressees' effectiveness in addressing the generic letter.

Reasons for Information Request

As discussed above, research and analysis suggests that (1) the potential for the failure of the ECCS and CSS recirculation functions as a result of debris blockage is not adequately addressed in most PWR licensees' current safety analyses, and (2) the ECCS and CSS recirculation functions at a significant number of operating PWRs could become degraded as a result of the potential effects of debris blockage or extended operation with debris laden fluids identified in this generic letter. An ECCS that is incapable of providing long-term reactor core cooling through recirculation operation would be in violation of 10 CFR 50.46. A CSS that is incapable of functioning in recirculation mode may not comply with GDCs 38 and 41 or other plant-specific licensing requirements or safety analyses. Bulletin 2003-01 requested information to verify addressees' compliance with NRC regulations and to ensure that any interim risks associated with post-accident debris blockage are minimized while evaluations to determine compliance proceed. This generic letter is the follow-on generic communication to Bulletin 2003-01. Therefore, the information requested in this generic letter is necessary to confirm plant-specific compliance with 10 CFR 50.46 and other existing regulations in light of the new staff guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS .

The NRC staff will also use the requested information to (1) determine whether a sample auditing approach is acceptable for verifying that addressees have resolved the concerns identified in this generic letter, (2) assist in determining which addressees would be subject to the proposed sample audits, (3) provide confidence that any nonaudited addressees have addressed the concerns identified in this generic letter, and (4) assess the need for and guide the development of any additional regulatory actions that may be necessary to address the adequacy of the ECCS and CSS recirculation functions.

Related Generic Communications

- Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," June 9, 2003.
- Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," May 6, 1996.
- Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in the Suppression Pool Cooling Mode," October 17, 1995.
- Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," May 11, 1993.
- Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," February 18, 1994.
- Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," July 14, 1998.
- Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.
- Generic Letter 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985.
- Generic Letter 91-18, Rev. 1," Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," October 8, 1997.
- Information Notice 97-13, "Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants," March 24, 1997.
- Information Notice 96-59, "Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris," October 30, 1996.
- Information Notice 96-55, "Inadequate Net Positive Suction Head of Emergency Core

Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions," October 22, 1996.

- Information Notice 96-27, "Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation," May 1, 1996.
- Information Notice 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances," February 13, 1996.
- Information Notice 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," October 4, 1995.
- Information Notice 95-47, Revision 1, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," November 30, 1995.
- Information Notice 95-06, "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment," January 25, 1995.
- Information Notice 94-57, "Debris in Containment and the Residual Heat Removal System," August 12, 1994.
- Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," April 26, 1993.
- Information Notice 93-34, Supplement 1, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," May 6, 1993.
- Information Notice 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," December 23, 1992.
- Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," September 30, 1992.
- Information Notice 89-79, "Degraded Coatings and Corrosion of Steel Containment Vessels," December 1, 1989.
- Information Notice 89-79, Supplement 1, "Degraded Coatings and Corrosion of Steel Containment Vessels," June 29, 1990.
- Information Notice 89-77, "Debris in Containment Emergency Sumps and Incorrect Screen Configurations," November 21, 1989.
- Information Notice 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," May 19, 1988.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10CFR 50.54(f), this generic letter transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this generic letter). Specifically, the required information will enable the NRC staff to determine whether the emergency core cooling system (ECCS) and containment spray system (CSS) at reactor facilities are able to perform their safety functions following all postulated accidents for which ECCS or CSS recirculation is required while taking into account the adverse effects of post-accident debris blockage and operation with debris laden fluids. No backfit is either intended or approved by the issuance of this generic letter, and the staff has not performed a backfit analysis.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this generic letter is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

- or-

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of the Office of Management and Budget (OMB).

Federal Register Notification

A notice of opportunity for public comment on this generic letter was published in the Federal Register (69 FR16980) on March 31, 2004. Comments were received from ten industry groups, one non-profit organization, one private citizen, and the State of New Jersey. The staff considered all comments that were received. The staff's evaluation of the comments is publicly available through the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession No. xxxxxxxxx.

Paperwork Reduction Act Statement

This generic letter contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB) under approval number XXXX-XXXX which expires on XXX XX, XXXX.

The burden to the public for these mandatory information collections is estimated to average 5000 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the necessary data, and completing and reviewing the information collections. Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Records

Management Branch, Mail Stop T-6 E6, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may neither conduct nor sponsor, and an individual is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contacts or lead project manager listed below.

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Risk-Informing 50.46 ECCS Acceptance Criteria

Briefing for ACRS Full Committee

Brian Sheron, ADPT/NRR

July 07, 2004

10

Background

- June 99 SRM directed staff to determine how best to proceed with risk-informing Part 50
- June 99 thru March 03 staff performed feasibility studies and technical analyses
- March 03 SRM directed staff to propose a rule providing a risk informed alternative maximum break size for LBLOCA in 12 months
- SECY-04-0037- Staff requested additional policy guidance on alternative break size rule

Staff Actions Pending SRM

- Formed inter-office Steering Committee
  - NRR (Lead), RES, OGC, NSIR
    - Meets weekly to provide guidance and resolve key issues
- Established Division leads for key tasks
  - Break size definition
  - LOCA and PRA success criteria
  - Rule framework
  - Assessment of impacts and potential consequences
  - PRA quality and scope requirements
  - Adequacy of RG 1.174 guidance
  - Security impacts

Staff Actions (con't)

- Established streamlined schedule for rulemaking
  - Meeting with ACRS Full Committee in July and in Fall
    - Several subcommittee meetings also expected
  - Federal Register Notice to be issued early July will discuss conceptual outline and summary description of rule
  - One public meeting for regulatory analysis inputs (July 26th)
  - Package to EDO November 30th
  - Package to Commission December 15th
  - Requesting CRGR waiver of draft rule review

SRM on SECY-04-0037

- Selection of max break size should use RG 1.174 approach, initiating event frequencies from expert elicitation process, and other relevant information
- Allow operational, as well as, design changes
- Restrict changes where engineering margins are necessary to meet RG 1.174 principles or security considerations
- Mitigation of LOCA up to DEGB should be required and changes to this capability should be controlled by regulation, commensurate with risk

SRM on SECY-04-0037 (con't)

- Backfit analyses should not be required to reverse changes needed to maintain compliance
- Use of BE codes should be encouraged, but not required
- Pursue requirements for future plants separately
- Review BWROG pilot exemption before including LOCA/LOOP in rulemaking
- Provide proposed rulemaking package in 6 months

Rule Concept

- Rule will divide break spectrum into 2 regions delineated by break size
- Region I breaks must meet all current 50.46 criteria
- Criteria and analysis assumptions for Region II breaks will be relaxed, but mitigation capability must still be demonstrated up to full DEGB

Rule Concept (con't)

- Break size delineation will be based upon frequency and other considerations
- Changes proposed to plant operations or design as a result of rule must be reviewed by the staff
- Submittals must be risk informed
  - Meet criteria of RG 1.174
  - Meet appropriate PRA quality and scope requirements

Rule Concept (con't)

- If future estimates of LOCA frequencies invalidate basis for plant changes made under the rule, the staff may require compensatory actions without formal backfit process.
- Use of the rule is voluntary

Staff Plan for ACRS Interactions

- **Meet with Full Committee on July 7th and provide high level briefing**
  - **Rule Concept**
  - **Schedule**
- **Propose Subcommittee meetings in July and September to present details**
- **Full Committee Meeting(s) in Fall**
- **Request letter**

ACRS Letter on SECY

- Supports wide range of applications if RG 1.174 criteria are satisfied
- Recommends explicit criteria for mitigative capability up to DEGB
- Recommends explicit criterion for late containment failure be included
- Recommends metric for max break size should be LOCA initiating event frequency

ACRS Letter on SECY (con't)

- Additional criteria and guidance beyond RG 1.1.74 for tracking cumulative risk are not needed
- Expert elicitation results should help provide a basis for a new maximum break size

Next steps

- Finalize conceptual basis for rule
- Federal Register Notice
  - Rule framework
  - Conceptual basis
- ACRS subcommittee on July 23rd
- Public Meeting on July 26th to gather inputs for regulatory analysis

Passive System LOCA Frequencies for Risk-Informed Revision of 10 CFR 50.46

Robert L. Tregoning
Lee Abramson
RES

Charles Hammer
NRR

Advisory Committee on Reactor Safeguards
July 7, 2004

LOCA Frequency Presentation Outline

- Presentation Objectives.
- ACRS Presentation History & Recent Program Milestones.
- Elicitation Findings.
- Sensitivity Analyses.
- Use of Results in Alternative Break Size Selection.
- Remaining Work and Schedule.
- Concluding Remarks.

Presentation Objectives

1. Communicate research conducted since the previous ACRS discussion (April 15, 2004).

2. Describe use of elicitation results in 50.46 rule development.

3. Discuss remaining technical work and schedule related to the elicitation.

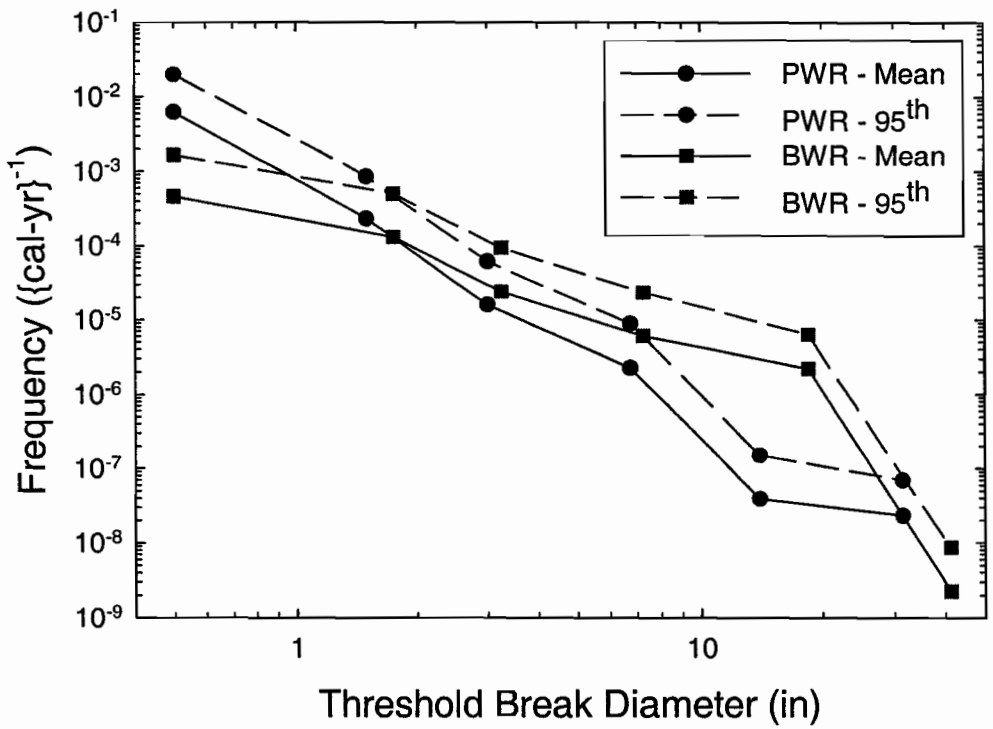
Previous ACRS Briefings and Recent Program Milestones

- Previous ACRS briefings
  - March/April, 2004: RPP Subcommittee and main committee on expert elicitation results.
  - November, 2003: RPP subcommittee on expert elicitation approach and base case development.
  - July, 2003: Main committee on the status and approach of expert elicitation.
  - May, 2002: Combined M&M, THP, R&PRA subcommittee briefing on interim LOCA frequency elicitation and LOCA break size redefinition plans.
  - June, July, November, 2001: Overviews of LOCA frequency and break size redefinition effort provided to outline its importance within 10 CFR 50.46 revision framework.
  - March, 2001: Technical issues necessitating LOCA reevaluation.

- Program milestones since April 2004
  - Completed sensitivity analyses: May 15th.
  - Preliminary NUREG report (main body) sent to elicitation panel members: July 2nd.

Total LOCA Frequencies: SECY-04-0060 Results

- BWR.
  - Decreases are gradual with LOCA size due to IGSCC concerns.
  - Only non-piping failures contribute to largest (41″ effective diameter) breaks.
- PWR.
  - Smallest LOCA frequencies are high due to steam generator and CRDM concerns.
  - Non-piping frequency contributions are also important largest LOCA sizes.
- BWR & PWR expected frequencies are similar for effective break diameters between 1 and 7″.
- BWR and PWR ratios between means and 95th percentiles are similar.



Summary of Important Elicitation Findings

- NRC used formal elicitation process to estimate generic BWR and PWR LOCA frequencies as a function of flow rate and operating time considering both piping and non-piping contributions.
- Developed quantitative estimates for piping and non-piping base cases which were used to anchor subsequent elicitation responses.
- Panelists provided quantitative estimates supported by qualitative rationale.
  - Determined important contributing factors (piping & non-piping systems, degradation mechanisms) governing LOCA frequencies.
  - Provided the relationships between these factors and the base cases.
- Results.
  - Relatively good agreement about important factors contributing to LOCAs.
  - Large individual uncertainty and panel variability in quantifying the frequencies associated with these contributing factors.
  - Results are similar to NUREG/CR-5750 estimates with biggest increase in medium LOCA frequency estimates.

Analysis of Elicitation Responses: Sensitivity Analyses

- Analyze the effect of different assumptions on the calculated LOCA frequency estimates to determine the full range of supportable quantitative results.
- Analysis of individual responses.
  - **Overconfidence adjustment of coverage intervals.**
  - Propagate system uncertainties variance bounds.
    - Lower bound: Independence.
    - Upper bound: Perfect correlation.
- Aggregating expert opinions.
  - Individual vs. group estimates of piping and non-piping contributions.
- Estimating group opinion.
  - **Measures of group opinion for mean, 5th and 95th percentiles.**
  - Measures of panel variability.

Estimating Group Opinion: General Philosophy

- Purpose of elicitation is to estimate the mean, 5th and 95th percentiles of the LOCA frequency distribution.
  - Group results more accurate than any single estimate.
  - Outliers should not dominate results, but should be used as a measure of panelist variability.
  - In general, principal benefit of outliers is to identify issues and variables that other panelists may not have considered.

- The group mean, 5th and 95th percentile estimates were determined in manner consistent with the structure and results of the elicitation process.
  - Panelists provided ratios relative to base cases.
  - Significant variability among panelist responses.

Estimating Group Opinion: Sensitivity Analyses

- Arithmetic mean, median, trimmed geometric mean, and geometric mean of individual results evaluated as measures of group opinion.
- Arithmetic mean results in highest frequencies.
  - Difference from other estimates is a function of the spread among panelist results.
  - Most dependent on the results of the experts with the highest frequencies.
  - Increases generally less than a factor of ten.
  - Occasionally, greater than factor of 10 difference: One high expert.
- The median, trimmed geometric mean, geometric mean values are closer to each other and consistent with the structure of the elicitation.

Analysis of Elicitation Responses: Overconfidence Adjustment

- Experts are generally overconfident about their uncertainty.
  - Demonstrated using almanac-type questions with known answers.
  - Rule of thumb: true coverage level is approximately half the nominal coverage level.
    - Nominal elicitation coverage level: 90% (95th – 5th percentiles)
    - Implication is that true coverage level is about 50% (75th – 25th percentile).
- Evaluate the effect of adjusting the nominal coverage level.
  - No change in the mid value responses
  - Broad adjustment: All responses.
    - Adjust responses to 50% coverage level.
    - Adjust responses to 60% coverage level.
  - Targeted adjustment.
    - Provide greatest adjustment for panelists with the least uncertainty.
    - Vary the adjustment of remaining panelists.
    - Three different adjustments examined.

Overconfidence Adjustment: Sensitivity Analyses

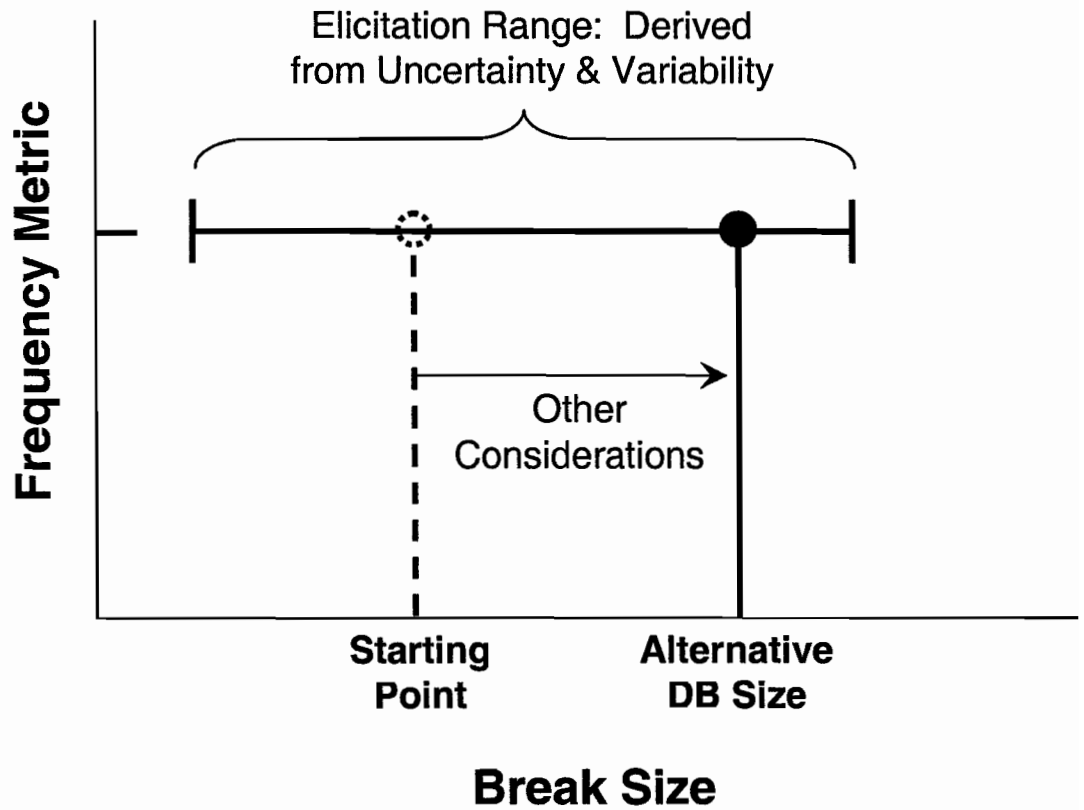
- All blanket adjustment and more conservative targeted adjustment schemes are too severe.
  - Means > 95th percentiles.
  - Unrealistically high frequencies.
- Only a modest targeted adjustment appears to be supported by the results.
  - Convert 90% coverage to 60% coverage for 4 – 5 panelists with lowest uncertainty ratios (< 10).
  - No adjustment for remaining panelists.
- Implication is that other panelists did not underestimate their uncertainty.
  - Elicitation training may have helped.
  - The relative ratio structure of the elicitation may have helped.
- Increases in SECY-04-0060 estimates due to targeted adjustment are generally less than a factor of 3.

Elicitation Results: Use in Alternative Break Size Selection

- Develop acceptable break size ranges from elicitation results.
  - Account for panel variability and individual uncertainty.
  - Consider effects of sensitivity analyses.
  - Select appropriate frequency metric (e.g. mean or 95th percentile) and criteria.
- Use elicitation results as starting point.
- Adjust break size within elicitation range using other considerations.
  - LOCA frequency sources not addressed in the elicitation.
    - Rare event loading (e.g. seismic, severe water hammer).
    - Consequential LOCA contributions (e.g. crane drops).
    - Active System LOCA contributions (e.g. seal LOCAs, ISLOCA).
  - Defense-in-depth considerations.
    - Ensure LPSI trains remain within original design basis.
    - Ensure that changes maintain plant safety & security.

Selection of Alternative Break Size

- Advantages.
  - Risk-informed approach consistent with RG 1.174 framework.
  - Maintains mitigation capability up to DEGB.
- Upcoming decisions.
  - Choose frequency metric and criteria.
  - Finalize elicitation range based on sensitivity analysis results.
  - Quantify the magnitude of the other considerations.



Remaining Work and Schedule

- Confirm elicitation analysis: external review.
  - Evaluate the processing of the elicitation input to calculate LOCA frequencies.
  - Schedule: Completion by August 31st.
- Finish NUREG report.
  - Preliminary report: completed.
  - Review meeting with expert panel: July 20 – 21.
  - Draft for NRR review: August 6th.
  - Incorporation of external review and NRR comments: September 30th.
  - Available for ACRS review: October 4th.
  - Incorporation of ACRS comments: November 30th.
  - Available for public dissemination with the proposed rule: December 2004.

Concluding Remarks

- LOCA frequency estimates can be sensitive to the method used to analyze panelists' input. Key elements are:
  - Overconfidence adjustment.
  - Estimate of group opinion.

- Developed conceptual methodology for risk-informed selection of alternative design basis break size.
  - Consistent with previous risk-informed practice and policy.
  - Work remains to finalize selection criteria.

- Technical basis document for the elicitation study is proceeding in parallel with and in support of proposed rule development.

NRR PRESENTATION TO ACRS ULTRASONIC FLOW METER (UFM) FOR MEASURING FEEDWATER FLOW USED IN DETERMINING REACTOR THERMAL POWER

July 8, 2004

Evangelos C. Marinos, Section Chief
Instrumentation and Controls
Electrical & Instrumentation & Controls Branch
Division of Engineering, NRR
U. S. Nuclear Regulatory Commission

(12)

PRESENTATION TOPICS

1)	Reactor Thermal Power Measurement	Page 3 - 4
2)	Appendix K to Part 50 – ECCS Evaluation Models	Page 5
3)	Ultrasonic Flow Meter Technologies	Page 6 - 8
4)	Non Power Uprate use of Ultrasonic Flowmeters in Nuclear Power Plants	Page 9
5)	Staff Review of Vendor Topicals	Page 10 - 11
6)	Staff Licensing Reviews	Page 12
7)	Public Concerns of UFM Accuracy	Page 13 - 14
8)	Staff Concerns of UFM Accuracy	Page 15 - 16

REACTOR THERMAL POWER MEASUREMENT

PWR Calormetric Calculation Principle Parameters

- Feedwater Flow
- SG Blowdown Flow
- SG Blowdown Liquid Enthalpy
- Feedwater Density
  - Temperature
  - Pressure
- Feedwater Enthalpy
  - Temperature
  - Pressure
- Steam Enthalpy
  - Temperature
  - Pressure
- Net Pump Heat Additon

REACTOR THERMAL POWER MEASUREMENT (Continued)

BWR Heat Balance Calculation Principle Parameters

- Feedwater Flow
- Feedwater Temperature
- Main Steam Pressure
- Reactor Water Cleanup Flow
- Reactor Water Cleanup Temperature
- Control Rod Drive Temperature

APPENDIX K TO PART 50 – ECCS EVALUATION MODELS

Source of heat during the LOCA

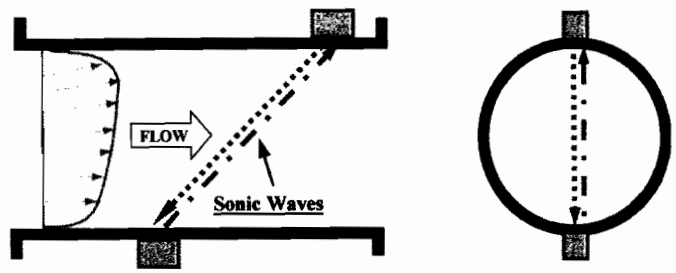
- Reactor assumed operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error)

- An assumed power level lower than 1.02 times the licensed power level may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

ULTRASONIC FLOW METER TECHNOLOGIES

Transit Time Technology - Leading Edge Flow Meter (LEFM) (Caldon)

- **Clamp-On Type Instrument**



$$\text{Angle of the Sonic Wave to Flow} = \alpha$$

$$V = \frac{L(T_1 - T_2)}{(2T_1T_2\cos\alpha)}$$

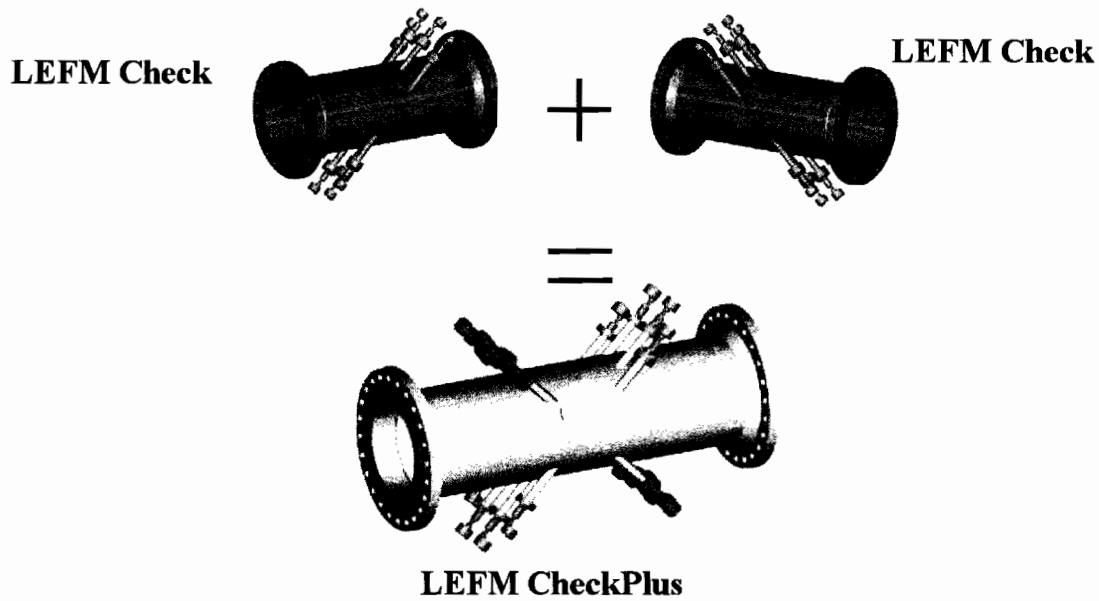
$$\Delta T \cong 1\mu Second$$

- **Fluid velocity (V) is determined by the difference in time of two sonic waves traveling in opposite directions.**

ULTRASONIC FLOW METER TECHNOLOGIES (Continued)

- **In-Line Type LEFM Instrument**

LEFM Check + LEFM Check = LEFM CheckPlus

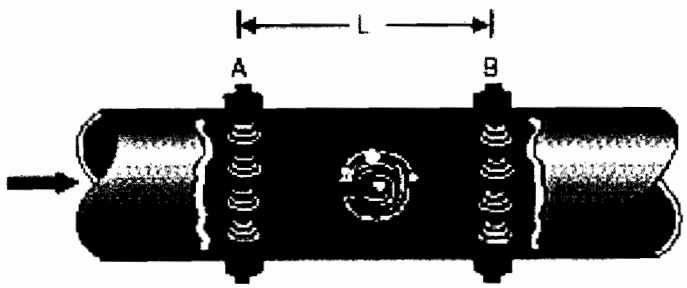


Operates on same principle as Clamp-On Type

ULTRASONIC FLOW METER TECHNOLOGIES (Continued)

Cross Correlation Technology - Westinghouse / Advanced Measurement and Analysis Group (W/AMAG)

- **Clamp-On Type (Crossflow) Instrument**



$$V = L/\tau$$

$$\tau \cong 50ms$$

Fluid Velocity (V) is determined by the time difference of two modulated signals shifted in phase by unique pattern of eddies in the fluid.

NON POWER UPRATE USE OF ULTRASONIC FLOWMETERS IN NUCLEAR POWER PLANTS

One-time venturi calibration.

Power Recovery from fouled venturies.

- **Use of Transit Time Clamp-On-Type, LEFM instrument, resulted in over 2% overpower at River Bend and approximately 1% - 3% at Palo Verde 1 & 2, caused by the instrument not meeting its expected accuracy.**

- **Use of Cross Correlation Clamp-On-Type Crossflow instrument resulted in over 2% overpower at Byron 1 and approximately .8% - 1% at Byron 2 and Braidwood 1&2 caused by an apparent misapplication of the instrument.**

STAFF REVIEW OF VENDOR TOPICALS

Caldon In-line Type LEFM Instrument

- Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using The LEFM✓™ System," Revision 0, dated March 1997, submitted for staff review
  - Staff issued Safety Evaluation Report (SER) dated March 8, 1999, accepting documented accuracy of instrument.
- Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM Check Plus ™ System" Revision 3, dated February 2001, submitted for staff review
  - Staff issued SER dated December 20, 2001 accepting documented enhanced accuracy of instrument.

STAFF REVIEW OF VENDOR TOPICALS (Continued)

Westinghouse / AMAG Clamp-On Type Crossflow Instrument

- Topical Report CENPD-379-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology" Revision 0, dated August 1999 submitted for staff review
  - Staff issued SER dated March 20, 2000 accepting documented accuracy of instrument.

STAFF LICENSING REVIEWS

Plant Applications for Measurement Uncertainty Recapture (MUR) Power Uprates (1.4-1.7%)

- Regulatory Information Summary RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Application" dated January 3, 2002 issued to licensees to provide consistent guidance for application submittals.

- Uprates granted by the staff to plants using Caldon LEFM In-Line Instrument for Feedwater measurement:
  - 21 power plants

Beaver Valley 1	Indian Point 3	Sequoyah 2	Beaver Valley 2	Peach Bottom 2
Comanche Peak 1	Point Beach 1	Susquehana 1	D. C. Cook 1	Peach Bottom 3
Comanche Peak 2	Point Beach 2	Susquehana 2	D. C. Cook 2	River Bend
Indian Point 2	Sequoyah 1	Watts Bar	Grand Gulf	H. B. Robinson
				Waterford 3

- Uprates granted by the staff to plants using Westinghouse / AMAG Crossflow Clamp-On Instrument for Feedwater measurement:
  - 12 power plants

Pilgrim	Hope Creek	San Onofre 2	Hatch 1	South Texas 1
Kewaunee	Salem Unit 1	San Onofre 3	Hatch 2	South Texas 2
Fort Calhoun	Salem Unit 2			

PUBLIC CONCERNS OF UFM ACCURACY

March 8, 2000: NRC met with Caldon in a public meeting to address Caldon's concerns with pending staff SER approving use of W/AMAG CrossFlow instrument.

- March 15, 2000: Letter from Caldon reaffirming their conclusion that the CrossFlow instrument could not achieve a bounding value of 0.5%.
- March 17, 2000: Follow-up letter from Caldon stated that uncertainties with CrossFlow instrument can be as high as 3%.
- March 17, 2000: Internal NRC memorandum from Jack Cushing to Stuart Richards summarizing public meeting with Caldon on March 8, 2000.

PUBLIC CONCERNS OF UFM ACCURACY

Caldon Engineering Report ER-262, Revision 0, "Effects of Velocity Profile Changes Measured In-Plant on Feedwater Flow Measurement Systems" dated January 2002, submitted to the staff for information.

- Staff informal review confirmed Caldon's determination that accuracy data of the Transit Time Clamp-On LEFM instruments do not invalidate the accuracy of the LEFM In-line instruments approved by the staff.

- On February 12, 2002 Caldon requested formal staff review of ER-262 to address Caldon's concern previously stated that the W / AMAG CrossFlow Clamp-On instrument does not meet the accuracy approved by the staff.

- On September 12, 2002 W / AMAG submitted unsolicited report WCAP-15689-P, Revision 1, "Evaluation of Transit-Time and Cross-Correlation Ultrasonic Flow Measurement Experience with Nuclear Plant Feedwater Flow Measurement" dated September 2002, addressing concerns raised by Caldon in the ER-262 report.

- On January 28, 2003 the staff issued evaluation of ER-262 reaffirming that the accuracy data of the Caldon Clamp-On Transit Time instrument do not invalidate the accuracy of the In-Line LEFMs approved by the staff and the data have no relationship to the W/AMAG Crossflow instrument.

STAFF CONCERNS OF UFM ACCURACY

NRC independent Task Group formed on February 2, 2004 to address concerns regarding accuracy of W/AMAG Crossflow instrument. Task Group considered information available through mid-April, 2004. Reports issued June 7, 2004.

- Identified issues with (1) one time UFM, (2) power recovery, and (3) power uprate applications.
- Crossflow sensitive to plant configurations and has not provided intended accuracy in some installations. The reasons have not been fully demonstrated to the staff.
- Crossflow may provide expected accuracy if properly implemented.
- Some Caldon clamp-on UFMs have not provided intended accuracy. Caldon LEFM√ and LEFM CheckPlus UFMs appear less sensitive to installation configuration than clamp-on designs.
- Task Group does not have information based upon recent insights that demonstrates all UFMs are providing intended accuracy.
- Recommended bulletin to obtain information to demonstrate that UFMs are providing the intended accuracy consistent with the plant license.

STAFF CONCERNS OF UFM ACCURACY (Continued)

Draft Bulletin 2004-XX being proposed to:

- **Advise addressees that plant operating experience at some installations has led the staff to conclude the W/AMAG Crossflow and Caldon LEFM (both Clamp-On-Type) UFM have not provided the intended Feedwater flow rate accuracy to maintain plant operation within the licensed thermal power.**

- **Advise addressees that there are potential questions regarding application of Caldon LEFM ✓ and LEFM Check Plus UFMs, because of plant configuration sensitivity and lack of technical data that can support instrument performance.**

- **Recommend that licensees may confirm UFM accuracy by comparing the instrument performance in operational plant conditions against measurement values of a fully clean ASME flow nozzle or venturi, of known accuracy, or any other standard test of known accuracy, and**

- **Require addresses to provide a written response to the NRC in accordance with the provisions of Section 50.54(f) of Title 10 of the *Code of Federal Regulations* (10 CFR 50.54(f), to verify that actions have been taken to ensure that each addressee's plant(s) is (are) not operated above the licensed thermal power level or outside the licensed design basis.**

Staff Presentation
Nuclear Power Plant Performance Assessment

José Calvo

Office of Nuclear Reactor Regulation

July 8, 2004

Topics

- Safety significance
- Generic Implications
- Licensing/Design basis
- Possible Solutions
- Conclusion

Safety Significance

- Plant comptr. receives UFM & ΔP inputs.
- Thermal Power Calc. Output- Displayed to operator.
- No automatic action.
- Operator verifies calc. thermal power using secondary plant information.
- Operator manually increase or decrease power level.
- Operator verifies that power adjustments provide the expected results.

- UFM is one of many components in a single channel – it cannot be concluded that all malfunctions in that channel would lead to fail-safe conditions.

- Since the operator makes the final decision to adjust the power level, irrespective of the performance of the UFM or venturi/ΔP, it can be concluded that the loss of accuracy of the flow devices can be successfully mitigated and thus the consequences of their failure have no safety significance.

W's flow device

* Byron 1&2 and Braidwood 1&2 overpower – misuse of the instrument.
* Fort Calhoun event discredited.

Caldon flow devices

* River Bend overpower.
* Palo Verde 1&2 overpower.

No other plant information was directly access by the UFM Allegation Task Force.

Insufficient basis to establish generic implications.

- The procedures used by the operator to verify independently thermal power calc by the PC formed the licensing basis.

- Backfit implications since these procedures were not addressed as part of the staff reviews.

- Equipment used is non-safety related, and the staff may have performed only coarsely reviews

- The bulletin suggests new requirements which will not pass the backfit rule.

- Ensure that the accuracy of secondary plant instrumentation readings and expected process values correlated to thermal power are accurate enough to verify the accuracy of the calculated thermal power by the PC.

- Or consider adding another redundant channel for calculating thermal power.

Otherwise overpower conditions may continue to occur.

- There is no safety significance.
- There is no generic implications.
- It is highly improbable that the Bulletin can be legally justified.
- It does not address the potential causes of overpower concerns.

The proposed bulletin is an overkill.

A generic informative communication raising awareness of the questions raised about the application of UFMs in NPPs would be more than sufficient and it could bring to the attention of the licensees the importance of verifying the accuracy of the calculated thermal power before any manual action is taken.

It should also challenge the licensees to determine whether the calorimetric surveillance intervals specified in the TS reflect the reduction in margin between the new thermal power based on the UFM and the 10 CFR 50 Appendix K limit.

(14)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
514	9	9.1

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

Authors
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

9

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

July 7, 2004
G:PlanPro(ACRS):ppmin.514

INTERNAL USE ONLY

MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
July 6, 2004

The ACRS Subcommittee on Planning and Procedures held a meeting on July 6, 2004, in Room T-2B3, 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 2:30 p.m. and adjourned at 4:00 p.m. A portion of this meeting between 3:45 and 4:00 p.m. was closed to discuss personnel and safeguards matters.

ATTENDEES
M. Bonaca
G. Wallis
S. Rosen

ACRS Staff
J. T. Larkins
R. P. Savio
S. Duraiswamy
J. Gallo
M. Snodderly
H. Nourbakhsh
R. Caruso
M. El-Zeftawy

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

Member assignments and priorities for ACRS reports and letters for the July ACRS meeting are attached (pp. 6-9). 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 July ACRS meeting be as shown in the attachment (pp. 6-9).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through October 2004 is attached (pp. 6-9). 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 also discussed and developed recommendations on items included in Section IV of the Future Activities list (pp. 10).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate.

3) Staff Requirements Memorandum Resulting from the ACRS Meeting with the NRC Commissioners

In the Staff Requirements Memorandum (SRM) dated June 30, 2004 (pp. 11-12), which results from the ACRS meeting with the Commissioners on June 2, 2004, the Commission states the following:

- The ACRS should work with the staff to resolve outstanding issues with respect to PWR Sump Performance, and make a recommendation for a practical solution within a reasonable period of time. Both the ACRS and the staff should focus their attention, resources, and additional research, if needed, on evaluating realistic scenarios rather than all possible scenarios.

- The Commission commends the ACRS' efforts in the security area. The Commission notes that the ACRS has closely followed the Commission's guidance contained in the October 31, 2003 SRM and that those provisions remain in place.

- The staff and the ACRS should engage with the Commission prior to making arrangements similar to the one recently agreed to between the Office of Nuclear Regulatory Research (RES) and the ACRS for ACRS to assess the quality of NRC research projects. This interaction should focus on the benefits and resource implications of such work and consistency of the proposals with previous Commission direction.

RECOMMENDATION

The Subcommittee recommends that Dr. Wallis, Chairman of the Thermal-Hydraulic Phenomena Subcommittee, propose a course of action for complying with the Commission directions specified in the SRM with regard to PWR sump performance.

4) Operating Events

In the e-mail dated June 24, 2004 (pp. 13), Mr. Leitch points out the recent reactor scrams that occurred at Dresden 3, Palo Verde 1, 2, and 3, Vermont Yankee, and Limerick 2 nuclear power plants. An Augmented Inspection Team (AIT) had been sent to investigate the event at Palo Verde. He suggests that the Committee hear a presentation from the staff on the results of the AIT investigation of the Palo Verde event as well as the staff's views on the effectiveness of the corrective actions taken by the industry in response to similar events that occurred last year.

Dr. Powers agrees with the suggestion by Mr. Leitch and adds that the Committee should also hear a presentation from the staff on the general issue of grid reliability (pp. 14-15).

In the e-mail dated June 29, 2004 (pp. 16-17), Mr. Sieber states the following:

- He agrees with the issue raised by Mr. Leitch and his conclusions.
- State deregulation has cut investment in transmission and substation capability in a dramatic fashion.
- Lower investment in power plants and transmission and distribution capabilities means more risk of power failures and more risk of nuclear plants suffering loss-of-offsite power (LOOP) events.
- In addition to discussing the issues raised by Mr. Leitch and Dr. Powers, the Committee should discuss the issue of increasing LOOP probability.

In response to Mr. Sieber's e-mail, Dr. Powers suggests (pp. 16-17) that the Planning and Procedures Subcommittee establish an ad hoc Subcommittee, Chaired by Mr. Sieber, to discuss the issues noted above with the staff and develop a report for consideration by the Committee. The ACRS has agreed to look into certain issues proactively, and this issue will fall into that category.

RECOMMENDATION

The Subcommittee recommends that the Plant Operations Subcommittee hold a meeting with the staff to discuss the matters suggested by Mr. Leitch, Dr. Powers, and Mr. Sieber and subsequently, Mr. Sieber, Chairman of the Plant Operations Subcommittee, propose a future course of action.

5) Visit to a Nuclear Plant and Regional Office

Several members (Bonaca, Ford, Leitch, Ransom, Rosen, Sieber, and Wallis) visited the D.C. Cook Nuclear Plant on Wednesday, June 9, 2004, and met with the NRC Region III personnel on Thursday, June 10, 2004, to discuss items of mutual interest. The Planning and Procedures Subcommittee discussed whether to continue to visit different plants in different regions annually or visit certain plants that have applied for license renewals and/or power uprates.

RECOMMENDATION

The Subcommittee recommends the following:

- The members should continue to visit a plant and hold meeting with the NRC Regional Office personnel.
- As circumstances warrant, the members should visit a plant in connection with the ACRS review of the license renewal and/or power uprate applications.

6) Tour of the Chalk River Facility Used for the ACR-700 Design

The tour of the Chalk River Facility in Canada and a joint meeting of the ACRS Subcommittees on Future Plant Designs and on Materials and Metallurgy are scheduled for September 27-30, 2004. Drs. Ford (tentative), Kress, Ransom, and Wallis have expressed interest in touring the Chalk River Facility and attending the meeting. Since AECL and the NRC staff are investing enormous time and efforts in arranging the tour and preparing for the Subcommittee meeting, more members should participate. Unless a majority of the members participate, the Committee should consider canceling the tour and the meeting.

RECOMMENDATION

The Subcommittee believes that it is not worthwhile for the staff and AECL spending their time and efforts in making arrangements for three or four members. As a result, the Subcommittee recommends that the Committee cancel the currently scheduled tour and the Subcommittee meeting, and select alternative dates for such a tour and a meeting if there is sufficient member interest.

7) Workshop on the AP1000 Design

An NRC staff (RES, NRR, and the Chairman's Office) delegation headed by Mr. Ashok Thadani will be participating in an international workshop regarding the AP1000 design to be held in China on July 26-29, 2004. Dr. Kress was invited to join the NRC panel and participate in such workshop. The Department of State is supporting this meeting.

During the May 2004 ACRS meeting, the Committee approved Dr. Kress' participation in this workshop. The presentation slides prepared by Dr. Kress are attached (pp. 18-38)

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the presentation slides prepared by Dr. Kress and that Dr. Kress prepare a trip report subsequent to the workshop.

8) Appointment of New ACRS Member (Closed)

The Commission has approved the appointment of Dr. Richard S. Denning to the ACRS. Subsequent to completing the security clearance process and resolving conflict-of-interest issues, if any, he will be officially appointed to the Committee.

RECOMMENDATION

The Subcommittee recommends that the ACRS Executive Director invite Dr. Denning to attend the September 2004 ACRS meeting as an invited expert.

9) Safeguards and Security Matters (Closed)

The Subcommittee discussed the Commission views in the SRM dated June 30, 2004 and other issues related to safeguards and security.

ANTICIPATED WORKLOAD
JULY 7-9, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Kress	-	El-Zeftawy	Final SER Associated With the AP1000 Design Certification	A	To support the staff schedule	-
			Future Plant Designs Subcommittee Report: Proposed Technology - Neutral Framework Document that was Discussed at the Subcommittee Meeting on June 24, 2004	—	—	—
Rosen/Kress	-	Nourbakhsh/Caruso	Status of Activities Associated With the Assessment of the Quality of NRC Research Projects	—	—	-
Powers		Nourbakhsh/ Duraiswamy	Response to SRM on Divergence in Regulatory Approaches Between U.S. and Several Other Countries	To be completed in September	—	-
Shack	-	Snodderly	Proposed Rule Language for Risk-Informing 10 CFR 50.46, and Sensitivity Studies on LBLOCA Frequency Re-evaluation	(Report as Needed)	-	-
Sieber	-	Weston/Santos	Proposed Generic Communication on the Use of Ultrasonic Flow Measurement Devices for Measuring Feedwater Flow Rates in Nuclear Plants	A	To support the staff schedule	-
	—	Weston	Plant Operations Subcommittee Report - Visit to D.C. Cook and Region III Office on June 9-10, 2004	—	—	—

ANTICIPATED WORKLOAD
JULY 7-9, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. DRAFTS
Wallis	-	Caruso	Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs	A	To support the staff schedule	-
			Thermal-Hydraulic Phenomena Subcommittee Report - Ongoing staff Activities Associated with the Resolution of GSI-191 on PWR Sump Performance that was Discussed at the Subcommittee Meeting on June 24, 2004	—	—	—

ANTICIPATED WORKLOAD
SEPTEMBER 8-11, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	-	Sykes/Santos	Final Review of the License Renewal Application for Dresden and Quad Cities Nuclear Plants	A	To support the staff schedule	-
		Savio/Major	Safeguards and Security matters	A	To provide Committee's views	-
Powers	Rosen/ Kress	Nourbakhsh/ Duraiswamy	Assessment of the Quality of NRC Research on Sump Blockage and on MACCS code	Report to be completed in October	-	-
		Nourbakhsh/ Duraiswamy	Response to SRM on Divergence in Regulatory Approaches Between U.S. and Other Countries	B	To respond to the Commission SRM	-
Ransom	Kress	Caruso	Proposed Resolution of GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs"	A	To support the staff schedule	-
Wallis	-	Caruso	Safety Evaluation Report for the Evaluation Guidelines Regarding Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs	A	To support the staff schedule	-

ANTICIPATED WORKLOAD
OCTOBER 7-9, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	—	Savio/Major	Safeguards and Security Matters (Tentative)	A	To provide Committee's views	—
Kress	—	El-Zeftawy	Status of Early Site Permit Reviews - **INFORMATION BRIEFING**	—	—	—
Powers	Rosen/Kress	Nourbakhsh/ Duraiswamy	Assessment of the Quality of the NRC Research Projects on Sump Performance and on MACCS Code	A	To support the staff schedule	—
		Weston	MOX Fuel Fabrication Facility	A	To support the staff schedule	—
Ransom	—	Caruso	Maximum Extended Load Line Limit Analysis Plus (MELLA+) Licensing Topical Report	B	To provide Committee's views	—
Rosen	—	Sykes	Proposed Rule on Post-Fire Operator Manual Actions	A	To support the staff schedule	—
Sieber		Weston	Mitigating System Performance Index (MSPI) program	B	To provide Committee's reviews	-

Items Requiring Committee Action

1 **Draft Regulatory Guide DG-1128, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants (Revision 4 to RG 1.97)"** (Open)

Member: John Sieber **Engineer:** Ralph Caruso

Estimated Time:

Purpose: Determine a Course of Action

Priority: Medium

Requested by: RES Douglas Tifft

The staff has provided the ACRS with copies of the draft Regulatory Guide DG-1128, which is proposed Revision 4 to RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." Staff plans to issue this Guidance for public comment. The draft RG endorses IEEE Std. 497-2002, which is both simpler and less prescriptive in nature than the existing RG. The Staff has requested that the ACRS defer review of the DG until after public comments have been received. The ACRS needs to determine whether it wants to review this DG before or after public comments are received.

The Planning and Procedures Subcommittee recommends that Mr. Seiber propose a course of action.

2 **Draft Final Amendment to 10 CFR 50.55a, "Codes and Standards"** (Open)

Member: William Shack **Engineer:** Michael Snodderly

Estimated Time: 1 hour

Purpose: Determine a Course of Action

Priority: High

Requested by: NRR Stephen Tingen, NRR

NRR is forwarding a draft final rule package amending 10 CFR 50.55a to incorporate by reference a later edition and addenda of the ASME Code to determine if the Committee's review is warranted. This draft final rule withdraws staff approval of Subsection NH of the 1995 through 2000 Addenda of Section III of the ASME Boiler and Pressure Vessel Code. In the associated statement of considerations, the NRC staff said Subsection NH was inadvertently endorsed and the NRC staff has not performed a technical review of the new subsection. Therefore, the NRC staff is unable to determine if the provisions in Subsection NH ensure adequate protection of public health and safety. The Committee may wish to hear how Subsection NH was inadvertently endorsed despite usually close coordination between the NRC staff and ASME. The statement of considerations also states that the fatigue analysis in Subsection NB may not be applied for some materials above certain temperatures. The application of Subsection NH to existing licenses is limited in that provisions can only be applied to pressurizer heater tubes. The Committee may want to be briefed on why the staff has not reviewed the methodology in Subsection NH for pressurizer heater tube replacement and what analytical methods the staff is approving for pressurizer heater tube replacement.

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

IN RESPONSE, PLEASE
REFER TO: M040602B

June 30, 2004

MEMORANDUM TO: John T. Larkins
Executive Director, ACRS/ACNW

Luis A. Reyes
Executive Director for Operations

FROM: Annette Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - MEETING WITH ACRS, 1:30 P.M., WEDNESDAY, JUNE 2, 2004, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the Advisory Committee on Reactor Safeguards on the following topics:

1. Overview
2. PWR Sump Performance
3. PRA Quality for Decisionmaking
4. Risk-Informing 10 CFR 50.46
5. ACRS 2004 Report on the NRC Safety Research Program
6. ESBWR Pre-Application Review
7. Interim Review of the AP1000 Design

The Commission requested the ACRS work with the staff to resolve outstanding issues with respect to PWR Sump Performance, and make a recommendation for a practical solution within a reasonable period of time. Both the ACRS and the staff should focus their attention, resources, and additional research, if needed, on evaluating realistic scenarios rather than all possible scenarios.

The Commission commends ACRS' efforts in the security area. The Commission notes that the ACRS has closely followed the Commission's guidance contained in the October 31, 2003 Staff Requirements Memorandum (SRM M031002), and that those provisions remain in place.

The staff and the ACRS should engage with the Commission prior to making arrangements similar to the one recently agreed to between the Office of Nuclear Regulatory Research (RES) and the ACRS for ACRS to assess the quality of NRC research projects. This interaction should focus on the benefits and resource implications of such work and consistency of the proposals with previous Commission direction.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

From: <GMLeitch@aol.com>
To: <mvbonaca@snet.net>, <JDSIEBER@aol.com>, <dapower@sandia.gov>, <graham.b.wallis@dartmouth.edu>, <wjshack@anl.gov>, <HistoryArt2004@aol.com>, <TSKress@aol.com>, <FPCTFord@aol.com>, <apostola@mit.edu>, <ransom@ecn.purdue.edu>, <SXD1@nrc.gov>, <JTL@nrc.gov>, <MWW@nrc.gov>
Date: 6/24/04 12:57PM
Subject: Recent Scrams

Colleagues,

It is interesting to note that there seems to be a recent increase in scrams caused by what I call "beyond the generator breaker" events. There have been 6 scrams that I would put in this category. These were all scrams from 100% power.

Dresden 3 5/5/04 Loss of Offsite Power
Palo Verde 1 2 & 3 6/14/04 Grid stability problems resulted in simultaneous tripping of all 3 units. In one case 1 of 2 EDGs failed to load. Alert declared.
Vermont Yankee 6/18/04 Main Transformer Fire
Limerick 2 6/22/04 Switchyard Problems. Details not known to me.

Also during the Palo Verde incident, San Onofre experienced low frequency on the grid. Seems to me we were very close to losing those units too and perhaps precipitating a major blackout. (my opinion)

The NRC is well aware of these events and has sent an AIT to Palo Verde. These events all seem to have been initiated by equipment or activities that are outside the NRC jurisdiction. It's getting to be summertime again and I'm sure we all remember how the number of these events increased last summer. Based on this early summer data, I'm not sure last years corrective actions have been effective.

I continue to believe this is an important safety issue. Grid reliability is not my main concern, but rather it is the impact on the nuclear plant when offsite power is suddenly lost. I know the plants are designed for this, but every time it happens the design is challenged.

I would suggest that Pand P consider inviting the staff to tell us results of Palo Verde AIT and their opinion regarding the effectiveness of last year's corrective actions taken by industry and system operators.

Regards,

Graham L.

From: "Powers, Dana A" <dapower@sandia.gov>
To: "'GMLeitch@aol.com'" <GMLeitch@aol.com>, <mvbonaca@snet.net>, <JDSIEBER@aol.com>, "Powers, Dana A" <dapower@sandia.gov>, <graham.b.wallis@dartmouth.edu>, <wjshack@anl.gov>, <HistoryArt2004@aol.com>, <TSKress@aol.com>, <FPCTFord@aol.com>, <apostola@mit.edu>, <ransom@ecn.purdue.edu>, <SXD1@nrc.gov>, <JTL@nrc.gov>, <MWW@nrc.gov>
Date: 6/26/04 3:05PM
Subject: RE: Recent Scrams

I agree. P&P should schedule something on Palo Verde and on the general issues of grid stability.

Dana

-----Original Message-----
From: GMLeitch@aol.com [mailto:GMLeitch@aol.com]
Sent: Thursday, June 24, 2004 10:55 AM
To: mvbonaca@snet.net; JDSIEBER@aol.com; dapower@sandia.gov; graham.b.wallis@dartmouth.edu; wjshack@anl.gov; HistoryArt2004@aol.com; TSKress@aol.com; FPCTFord@aol.com; apostola@mit.edu; ransom@ecn.purdue.edu; SXD1@nrc.gov; JTL@nrc.gov; MWW@nrc.gov
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Regards,

Graham L.

From: "Powers, Dana A" <dapower@sandia.gov>
To: "'JDSIEBER@aol.com'" <JDSIEBER@aol.com>, <GMLeitch@aol.com>, "Powers, Dana A" <dapower@sandia.gov>, <ransom@ecn.purdue.edu>, <Apostoal@mit.edu>, <mvbonaca@snet.net>, <graham.b.wallis.@dartmouth.edu>, <TSKress@aol.com>, <historyart@computron.net>, <wjshack@anl.gov>, <FPCTFord@aol.com>
Date: 6/29/04 12:42PM
Subject: RE: Recent Scrams

Jack may be correct - he usually is. Why doesn't P&P ask Jack to lead an ad hoc subcommittee to draft up something after talking with the staff et al. This would make a great independent initiative by ACRS - it is something that may not be getting the attention that it deserves in the normal course of work. ACRS is supposed to be on the lookout for such things.

Dana

-----Original Message-----
From: JDSIEBER@aol.com [mailto:JDSIEBER@aol.com]
Sent: Tuesday, June 29, 2004 8:30 AM
To: GMLeitch@aol.com; dapower@sandia.gov; ransom@ecn.purdue.edu; Apostoal@mit.edu; mvbonaca@snet.net; graham.b.wallis.@dartmouth.edu; TSKress@aol.com; historyart@computron.net; wjshack@anl.gov; JDSIEBER@aol.com; FPCTFord@aol.com
Cc: jtl@nrc.gov
Subject: Re: Recent Scrams

Colleagues:

I agree with Graham's report, but I feel even more strongly about his conclusions. Obviously, the US electric power supply, and transmission capacity is the overwhelming issue. More importantly, State deregulation has cut investment in transmission and substation capability in a dramatic fashion. Everybody thought that deregulation of the electric power industry would be a cool thing to do. Customers would save money, the utility industry executives would loose money and the stock holders would take the brunt of the problem. It didn't work out that way. Even a simpleton in economics could easily figure out what would happen. Lower rates mean lower investment in power plants and transmission and distribution capabilities. So, that means more risk of power failures, and more risk of nuclear plants suffering LOOP events. Every day the risk of an unrecoverable LOOP event increases. This result can be laid at the feet of the consumerist and environmental culture in our society. In my opinion, drastic deregulation demands by the States has brought us to the point where reliance on off-site power for nuclear power plant protection should be questioned. Perhaps the assumptions that went into all PRAs about the availability of off-site power should be questioned, and the risk figures adjusted.

I agree with Dana and Graham that the issue should be discussed at the P and P meeting, but I also believe that the issue of increasing LOOP probability deserves more attention than Graham and Dana suggest, which is to ask the Staff to look into it. To me, this is a BIG DEAL. I think that grid instability is a big increase in risk and I blame the State governments (for utility deregulation) and Federal inaction for this increase in risk. I

think that we should put together our independent thoughts and place ourselves on record as to the dangers that we foresee.

Jack Sieber

CC: <jtl@nrc.gov>

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) ROLE IN THE REGULATORY PROCESS

Thomas S. Kress, Ph.D.
ACRS Member
Chairman, ACRS Subcommittee on Future Plant Designs

Presented at:
WORKSHOP ON AP1000
July 26-29, 2004
Beijing, China

ACRS MISSION

- The mission of the ACRS is to provide the Commission with independent and timely technical advice on issues of public safety related to nuclear reactors and reactor safeguards. In so doing, the ACRS supports the NRC in maintaining an efficient regulatory program that enables the Nation to safely use nuclear power for civilian purposes

HISTORICAL PERSPECTIVE

- The organizational history relating to the ACRS began in 1947 when the Atomic Energy Commission (AEC) formed the Reactor Safeguards Committee to serve as an advisory body to its Division of Research

- In 1950, a related committee, the Industrial Committee on Reactor Location Problems, was formed to evaluate the hazards associated with the operation of production facilities

- In 1953, these two Committees were combined under the new name, Advisory Committee on Reactor Safeguards

HISTORICAL PERSPECTIVE (Cont'd)

- ACRS is statutorily mandated by the Atomic Energy Act of 1954, as amended

- In 1974, AEC was reorganized into two separate entities
  - The Nuclear Regulatory Commission
  - The Energy Research and Development Administration (now the Department of Energy)

- The ACRS was assigned to NRC with its statutory requirements intact

ACRS CHARTER

- The ACRS provides the NRC with independent reviews of, and advice on, the safety of proposed or existing reactor facilities and the adequacy of proposed safety standards. The primary purposes of the Committee are to:
  - Review and report on safety studies and reactor facility license and license renewal applications
  - Advise the Commission on the hazards of proposed and existing reactor facilities and the adequacy of proposed reactor safety standards; and
  - Initiate reviews of specific generic matters or nuclear facility safety-related items.

ACRS CHARTER (Cont'd)

- The ACRS is independent of the NRC staff and reports directly to the Commission, which appoints its members (currently 11 members)

- The ACRS is structured to provide a forum where experts representing many technical perspectives can provide independent advice that is factored into the Commission's decisionmaking process

- The ACRS reviews and provides reports on U. S. naval reactor designs, and also advises DOE with regard to the hazards of DOE nuclear activities and facilities consistent with the Energy Reorganization Act of 1974, as amended

ACRS CURRENT ACTIVITIES

- Advanced Reactor Designs
- Risk-Informed Regulation
- Safeguards and Security Matters
- License Renewal Applications
- Extended Power Uprate Applications
- Transient and Accident Analysis Codes
- Operating Plant Issues
- NRC Safety Research Program

ACRS ROLE IN DESIGN CERTIFICATIONS

- 10 CFR 52.53 requires that each application for a standard design certification be referred to the ACRS for a review and report on those portions of the application which concern safety
  - Provide an independent review of the NRC staff's determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations
  - Provide an open forum for public participation in the review process

ACRS REVIEW OF AP1000 DESIGN CERTIFICATION

Thomas S. Kress, Ph.D.
Member of Advisory Committee on Reactor Safeguards (ACRS)
Chairman, ACRS Subcommittee on Future Plant Designs

Presented at:
WORKSHOP ON AP1000
July 26-29, 2004
Beijing, China

ACRS ROLE IN DESIGN CERTIFICATIONS

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  - Provide an open forum for public participation in the review process
  - Provide an independent review of the NRC staff's determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations

ACRS ROLE IN DESIGN CERTIFICATIONS (CONT'D)

- The ACRS review begins early in the licensing process. A series of meetings with the applicant and the NRC staff are held at appropriate times in the review process

- When the Committee has completed its review, its report is submitted to the Commission

ACRS REVIEW PROCESS

- **In-depth reviews are done by the subcommittees**
  - With input from subcommittee members, subcommittee chairman develops proposed ACRS position
- **Presentations are provided to the full Committee. ACRS positions are developed after extensive deliberations by the full Committee**
- **At times, ACRS issues "interim" letters to identify issues of concern and items for which additional information, discussions, and clarifications are needed**

EXTENT OF ACRS REVIEW OF THE AP600 DESIGN

- To appreciate the extent of ACRS review of the AP1000 design, one must consider the reviews for certification of AP600

- The designs are sufficiently similar that most, if not all, of the AP600 review findings are applicable to AP1000

EXTENT OF ACRS REVIEW OF THE AP600 DESIGN (CONT'D)

Subcommittee Meetings

Thermal Hydraulic Phenomena
27 meetings
December 1991- June 1998

Future Plant Designs
9 meetings
January 1995- July 1998

Severe Accidents
one meeting
June 1996

Full Committee Meetings

6 ACRS Meetings

August	1996
June	1997
February	1998
April	1998
May	1998
July	1998

Report to NRC Chairman
July 23, 1998

ACRS REVIEW OF THE AP1000 DESIGN

Subcommittee Meetings

Thermal Hydraulic Phenomena
5 meetings
March 2001- February 2004

Future Plant Designs
3 meetings
February 2002 - June 2004

Reliability and Probabilistic Risk Assessment
one meeting
January 2003

Full Committee Meetings

10 ACRS Meetings

August	2000
April	2001
March	2002
November	2002
February	2003
April	2003
October	2003
March	2004
June	2004
July	2004

Report to NRC Chairman
July 2004

ACRS APPROACH FOR REVIEW OF AP1000 DESIGN

- **Prior to the start of ACRS review, The Committee agreed on its approach in a strategic planning session**
  - Review all design changes from the AP600 design
  - Review experience with the 14 ft long fuel elements (any increased tendency for bowing/blockage)
  - Review scaling analysis to determine applicability of test and analysis program of AP600
  - Determine if separate effects tests for AP600 are applicable to AP1000
  - Request an uncertainty analysis for the codes used for DBA analyses to assess margins
  - Review critical accident sequences for water level and adequacy of the ADS4 squib valve
  - Determine if GOTHIC code is applicable to AP1000
  - Assess defense-in-depth function of containment [seismic, hydrogen stratification, PRA results for CDF and LERF, aerosol behavior]
  - Review PRA quality and results
  - Review staff's DSER and FSER for determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations

LATEST ACRS INTERIM LETTER ON AP1000

- **The latest ACRS "interim" letter identified 7 remaining technical issues to be resolved before the Committee provides its final recommendations on the design certification**
  - ADS4 squib valve reliability [How can this be justified/demonstrated?]
  - In view of the strainer blockage issue, how is AP1000 designed to assure long-term cooling?
  - How to disposition some code deficiencies?
  - Range of "Pi-group" values for appropriate scaling
  - Potential for failure of in-vessel retention and subsequent fuel/coolant interaction potential to fail the containment
  - Organic iodine formation in water films on containment walls
  - Potential for catastrophic failure of free standing steel containment vessel

ILLUSTRATIVE EXAMPLES OF HOW ACRS/STAFF/W RESOLVED SUCH ISSUES

1. **SQUIB VALVE RELIABILITY/PERFORMANCE:**

ISSUE: The ADS4 squib valve reliability/performance assumptions are based on the experience with similar but much smaller valves. How can we be assured of the validity of the reliability/performance claims?

RESOLUTION BASIS:

- Simple design – ASME code section III Class 1
- Redundant and diverse actuation
- Sensitivity studies with PRA
- ITAAC item. To be tested at COL stage

ILLUSTRATIVE EXAMPLES (CONT'D)

2. **STRAINER BLOCKAGE/LONG-TERM COOLING ASSURANCE:**

ISSUE: Barsebäck event in Sweden and other LOCA-like events in U.S. have shown significant debris generation and subsequent tendency to block inlet screens to the recirculation lines that provide for long-term cooling of PWRs. How has AP1000 design dealt with this issue?

RESOLUTION BASIS:

- The screens for AP1000 are more robust than current plants [bigger, higher off floor, significant barriers, known location of the depressurization]

- ITAAC item. This is a generic PWR issue that has not yet been resolved generically. ITAAC will ensure that AP1000 complies with the resolution

ILLUSTRATIVE EXAMPLES (CONT'D)

5. **IN-VESSEL RETENTION/POTENTIAL FOR FCI AND CONTAINMENT FAILURE:**

ISSUE: The defense-in-depth concept of flooding the exterior of the reactor vessel to retain any core melt has not been sufficiently proven. In the event the melt does penetrate, will a subsequent thermal interaction with the subcooled water provide sufficient pressure pulse to fail containment?

RESOLUTION BASIS:

- All core-melt sequences are of such low probability that U.S. NRC Safety Goals are met with large margin even if containment fails.
- Sensitivity studies with current state-of-the-art FCI models show very low conditional probability of containment failure.

CONCLUSIONS

- **ACRS REVIEW OF AP1000 DESIGN IS COMPLETE AND ALL ACRS SAFETY CONCERNS HAVE BEEN ADDRESSED**
  - The staff has done a competent, comprehensive review of acceptability of analytical tools and compliance with the regulations.
  - Most of the ACRS meetings have been open to the public and there has been significant input from members of the public
  - ACRS has raised many issues independent of staff. All these issues have been resolved to the ACRS satisfaction

- **ACRS CONCLUDES THAT THE AP1000 DESIGN IS ROBUST AND THAT IT CAN BE BUILT AND OPERATED WITHOUT UNDUE RISK TO THE HEALTH AND SAFETY OF THE PUBLIC**

REVISED--

FINAL

July 8, 2004

Filed: P:\ACRSMTGS.bjw

Copy to: ACRS/ACNW Members/Staff, S. Schoenmann, Security, T-6E27, Guard's Post TWFN-Lobby, Tim Rollins, ADM, T-7D24

ACRS/ACNW FULL COMMITTEE AND ACRS SUBCOMMITTEE MEETINGS

ACNW Planning and Procedures, July 20, 2004, 11545 Rockville Pike, Rockville, MD, **8:30 a.m. - 10:00 a.m., Room T-2B1**

152nd ACNW Meeting, July 20-22 **(8:30 a.m.)**, 2004, 11545 Rockville Pike, Rockville, MD, **Room T-2B3**

No ACRS Meeting in August 2004

No ACNW Meeting in August 2004

Safeguards and Security, August 11-13, 2004 **dates changed to August 24-26, 2004**, Sandia National Laboratories, Albuquerque, NM (Savio).

Thermal-Hydraulic Phenomena, August 17-18, 2004, 11545 Rockville Pike, Rockville, MD (Caruso), **8:30 a.m. - 5:00 p.m., Room T-2B3.** To review the staff's final safety evaluation report on the NEI guidelines related to resolution of GSI-191 "PWR Sump Strainer Performance". The staff is expected to provide its final evaluation and guidance to licensees on this issue, and will be asking for the Committee to write a letter at the September full Committee meeting. The Committee will also review the final staff resolution of GSI-185 - "Control of Recriticality Following Small-Break LOCAs in PWRs". Lodging will be announced later. Attendance by the following is anticipated:

Wallis
Ford
Kress
Ransom
Sieber

Materials & Metallurgy, Thermal-Hydraulic Phenomena and Reliability and Probabilistic Risk Assessment, August 19, 2004 - **POSTPONED.**

Safeguards and Security **(This Meeting Is Closed To Public)**, August 24-26, 2004, Sandia National Laboratories, Albuquerque, NM (Savio), **8:30 a.m. - 5:00 p.m.** The Subcommittee will meet in closed session to discuss safeguards and security matters. Lodging will be announced later. Attendance by the following is anticipated:

Bonaca
Kress
Powers
Ransom
Shack
Sieber
Wallis

Planning and Procedures, September 7, 2004, 11545 Rockville Pike, Rockville, MD (Duraiswamy), **Time to be determined, Room T-2B1**. The Subcommittee will discuss proposed ACRS activities and related matters. A portion of this meeting 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. Lodging will be announced later. Attendance by the following is anticipated:

Bonaca
Wallis
Rosen

515th ACRS Meeting, September 8-11, 2004, 11545 Rockville Pike, Rockville, MD, Room T-2B3.

ACNW Yucca Mountain Field Trip, September 21, 2004, Amargosa, NV

ACNW Planning and Procedures, September 22, 2004, **8:30 a.m. - 10:00 a.m.,** Sun Coast Hotel, Las Vegas, Nevada

153rd ACNW Meeting, September 22 (**10:30 a.m.**) and 23 **(8:30 a.m.)**, 2004, Sun Coast Hotel, Las Vegas, Nevada

Thermal-Hydraulic Phenomena, September 22, 2004, 11545 Rockville Pike, Rockville, MD (Caruso), **8:30 a.m. - 5:00 p.m., Room T-2B3**. To review the MELLLA+ Topical Report. This report involves an expansion of the BWR operating domain, and expansion of this domain is needed by plants that perform power uprates so that they can perform rod pattern changes and other maneuvers at full power. Lodging will be announced later. Attendance by the following is anticipated:

Wallis
Ford
Kress
Ransom
Leitch

Future Plant Designs and Materials and Metallurgy, September 27-29, 2004, Chalk River Facility, Ottawa, Canada (El-Zeftawy), **8:30 a.m. - 5:00 p.m.** The Joint Subcommittees will tour the Chalk River Facility and hold a meeting in Canada to discuss various aspects of the ACR-700 design, including materials issues. AECL has agreed to accommodate the members' visit to the facility and support the subcommittee meeting. Lodging will be announced later. Attendance by the following is anticipated:

Kress
Ford
Ransom
Wallis

Regulatory Policies and Practices, October 6, 2004, 11545 Rockville Pike, Rockville, MD (Snodderly), **8:30 a.m. - 5:00 p.m., Room T-2B3**. The Subcommittee will review the proposed rule package for risk-informing 50.46. Lodging will be announced later. Attendance by the following is anticipated:

Shack
Apostolakis
Bonaca
Ford
Kress
Powers
Ransom
Rosen
Sieber
Wallis

Planning and Procedures, October 6, 2004, 11545 Rockville Pike, Rockville, MD (Duraiswamy), **Time to be determined, Room T-2B1**. The Subcommittee will discuss proposed ACRS activities and related matters. A portion of this meeting 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. Lodging will be announced later. Attendance by the following is anticipated:

Bonaca
Wallis
Rosen

516th ACRS Meeting, October 7-9, 2004, 11545 Rockville Pike, Rockville, MD, Room T-2B3.

Thermal-Hydraulic Phenomena, October 13, 2004, 11545 Rockville Pike, Rockville, MD (Caruso), **8:30 a.m. - 5:00 p.m., Room T-2B3**. The Subcommittee will review the application for an 8.5% power uprate for the Waterford Nuclear Power Station. Lodging will be announced later. Attendance by the following is anticipated:

Ford
Kress
Ransom
Sieber
Wallis

ACNW Planning and Procedures, October 19, 2004, 11545 Rockville Pike, Rockville, MD, **8:30 a.m. - 10:00 a.m., Room T-2B1**

154th ACNW Meeting, October 19 **(10:30 a.m.)** - 20-21 **(8:30 a.m.)**, 2004, 11545 Rockville Pike, Rockville, MD, **Room T-2B3**

Regulatory Policies and Practices, October 20, 2004, 11545 Rockville Pike, Rockville, MD (Snodderly), **8:30 a.m. - 5:00 p.m., Room T-2B3**. The Subcommittee will review the draft final NUREG documenting the expert elicitation on large break LOCA frequencies. Lodging will be announced later. Attendance by the following is anticipated:

Shack
Apostolakis
Bonaca
Ford
Kress
Powers
Ransom
Rosen
Sieber
Wallis

Planning and Procedures, November 3, 2004, 11545 Rockville Pike, Rockville, MD (Duraiswamy), **Time to be determined, Room T-2B1**. The Subcommittee will discuss proposed ACRS activities and related matters. A portion of this meeting 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. Lodging will be announced later. Attendance by the following is anticipated:

Bonaca
Wallis
Rosen

Plant License Renewal, November 3, 2004, 11545 Rockville Pike, Rockville, MD (Sykes), **1:00 p.m. - 4:30 p.m., Room T-2B3**. The Subcommittee will review the License Renewal Application and associated SER with Open Items for the Joseph A. Farley Nuclear Station. Lodging will be announced later. Attendance by the following is anticipated:

Bonaca
Ransom
Shack
Sieber
Barton
Leitch

517th ACRS Meeting, November 4-6, 2004, 11545 Rockville Pike, Rockville, MD, Room T-2B3.

Thermal-Hydraulic Phenomena, November 16-17 , 2004, 11545 Rockville Pike, Rockville, MD (Caruso), **8:30 a.m. - 5:00 p.m., Room T-2B3**. The Subcommittee will review the application for a 20% power uprate for the Vermont Yankee Nuclear Power Station. Lodging will be announced later. Attendance by the following is anticipated:

Ford
Kress
Ransom
Sieber
Wallis

Reliability and Probabilistic Risk Assessment, November 18, 2004, 11545 Rockville Pike, Rockville, MD (Snodderly), **8:30 a.m. - 5:00 p.m., Room T-2B3**. The Subcommittee will review the NRC staff's regulatory guide which endorses the ANS standard on external events prior to it being issued for public comment. Lodging will be announced later. Attendance by the following is anticipated:

Apostolakis
Bonaca
Ford
Kress
Rosen
Shack

No ACNW Meeting in November 2004

Planning and Procedures, December 1, 2004, 11545 Rockville Pike, Rockville, MD (Duraiswamy), **Time to be determined, Room T-2B1**. The Subcommittee will discuss proposed ACRS activities and related matters. A portion of this meeting 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. Lodging will be announced later. Attendance by the following is anticipated:

Bonaca
Wallis
Rosen

Plant License Renewal, December 1, 2004, 11545 Rockville Pike, Rockville, MD (Santos), **1:00 p.m. - 4:30 p.m., Room T-2B3**. The Subcommittee will review the License Renewal Application and associated SER w/Open Items for the Arkansas Nuclear One, Unit 2. Lodging will be announced later. Attendance by the following is anticipated:

Bonaca
Ford
Rosen
Wallis
Barton
Leitch

518th ACRS Meeting, December 2-4, 2004, 11545 Rockville Pike, Rockville, MD, Room T-2B3.

ACNW Planning and Procedures, December 7, 2004, 11545 Rockville Pike, Rockville, MD, **8:30 a.m. - 10:00 a.m., Room T-2B1**

155th ACNW Meeting, December 7 **(10:30 a.m.)** - 8-9 **(8:30 a.m.)**, 2004, 11545 Rockville Pike, Rockville, MD, **Room T-2B3**