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

September 30, 2004

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

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT — 515th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SEPTEMBER 9-11, 2004 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 515th meeting, September 9-11, 2004, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report and memoranda:

REPORT:

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

- Report on the Safety Aspects of the License Renewal Application for the Dresden 2 and 3 and Quad Cities 1 and 2 Nuclear Power Stations, dated September 16, 2004

MEMORANDA:

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

- Proposed Regulatory Guide 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," dated September 10, 2004

- Draft Regulatory Guide, DG-1.XXX, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," dated September 14, 2004

HIGHLIGHTS OF KEY ISSUES

1. Final Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Plants

The Committee met with the NRC staff and representatives of the Exelon Generation Company (Exelon) to review and discuss the results of the staff evaluation of the license renewal application for Dresden 2 & 3 and Quad Cities 1 & 2 Nuclear Power Stations and the associated final Safety Evaluation Report. The applicant has requested approval for continued operation of these plants for a period of 20 years beyond the current license expiration dates. The operating licenses for Dresden 2 and 3 expire on December 22, 2009 and January 12, 2011. The Quad Cities 1 and 2 licenses expire December 14, 2012.

Committee Action

The Committee issued a report to the NRC Chairman dated September 16, 2004, concluding that the programs instituted and committed to by Exelon, to manage age-related degradation, are appropriate and provide reasonable assurance that the Dresden and Quad Cities Nuclear Power Stations can be operated in accordance with the current licensing bases for the period of extended operation without undue risk to the health and safety of the public. The Committee recommended that the Exelon application for renewal of the operating license for Dresden and Quad Cities Nuclear Power Station be approved once the staff requires that the steam dryers for the Dresden and Quad Cities plants are included within scope of license renewal and once the staff has confirmed that the applicant will conduct an evaluation of applicable operating experience prior to entering the period of extended operation.

2. Proposed Changes to the License Renewal Program

The Committee met with representatives of NRR to discuss proposed changes to the scoping and screening reviews of license renewal applications. The staff described an assessment of it's scoping and screening review process and a proposed approach for sampling the systems to be reviewed as part of a license renewal application. The team performing this assessment recommended that audits and inspections be better coordinated to eliminate duplication of effort and that the inconsistencies among license renewal guidance documents be corrected. A methodology in which only a portion of the auxiliary systems and the steam and power conversions systems would be reviewed was also proposed. The selection of systems to be reviewed in detail would consider inherent risks and experience from previous reviews of license renewal applications. The staff believes that this proposed sampling would improve the effectiveness and efficiency of reviews while providing reasonable assurance that the applicant has identified all structures, systems, and components required by license renewal regulations.

Committee Action

This briefing was for information only. No committee action is necessary.

3. Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity

The Committee met with representatives of NRR to discuss the staff's review of more performance-based technical specifications (TS) for steam generator tube integrity. The staff stated that all outstanding issues regarding the proposed TS have been resolved and a safety evaluation for the lead plant (Farley) will be issued on September 17, 2004. The proposed TS will set new Limiting Conditions for Operation (LCOs) and establish a new administrative TS for a Steam Generator Program. The LCO for operational leakage will be lowered from 500 gpd to 150 gpd for each SG to provide added assurance that the plant can be shut down before tube rupture. The Steam Generator Program TS will specify performance criteria for tube integrity, provisions for monitoring, repair criteria, and provisions for inspection. The structural integrity performance criteria specify safety factors for normal and design basis accident (DBA) loads. The accident leakage performance criterion limits DBA leakage to one gpm for all SGs. The SG tube repair criteria and methods are consistent with those currently approved. The scope, method, and frequency of tube inspections shall be done to ensure that integrity is maintained until the next inspection. The staff is reviewing a Generic License Change Package for these new TS and is preparing a Generic Letter on Steam Generator Tube Inspections.

A representative of the Nuclear Energy Institute (NEI) stated that industry has already committed to implementing NEI 97-06, "Steam Generator Program Guidelines," the associated Electric Power Research Institute documents, and the General License Change Package.

Committee Action

This was an information briefing. No Committee action was necessary. The members plan to provide the staff with a list of topics to be discussed at a future Materials and Metallurgy Subcommittee meeting regarding the resolution of technical issues raised by the staff regarding NEI 97-06.

4. Safeguards and Security Matters

The Committee heard briefings by and held discussions with representatives of the NRC staff and NEI regarding safeguards and security matters. **Note: This session was 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).**

Committee Action

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

5. Assessment of the Quality of the Selected NRC Research Projects

The Office of Nuclear Regulatory Research (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 developed a strategy for reviewing the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During the September 9-11, 2004 ACRS meeting, the Committee discussed the preliminary results of the cognizant ACRS members' assessment of the quality of the NRC research projects on Sump Blockage and on MACCS Code.

Committee Action

The Committee plans to discuss the assessment of the quality of the research projects on Sump Blockage and MACCS Code during October 7-9, 2004 ACRS meeting.

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

In an April 28, 2003 Staff Requirements Memorandum (SRM), resulting from the April 11, 2003 ACRS meeting with the Commission, it was 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 the draft of a white paper to be used by the ACRS in responding to the Commission. During the September 9-11, 2004 ACRS meeting, the Committee discussed the draft White Paper regarding divergence in regulatory approaches 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 October 7-9, 2004 ACRS meeting.

7. Trip Report — AP1000 Workshop in China

Dr. Thomas Kress, Chairman of the ACRS Subcommittee on Future Plant Designs, provided a trip report to the Committee regarding a workshop held in Beijing, China on July 26-28, 2004. He stated that the workshop was the direct result of a proposal from the Chinese regulatory authorities made to Chairman Diaz during his visit in April 2004. The purpose of the workshop was to acquaint the Chinese with the U.S. nuclear regulatory process, particularly for design certification, and specifically with the activities associated with the design certification of the AP1000. The workshop team from the NRC was headed by Mr. Ashok Thadani. Other team members included Dr. Stephen Bajorek, Mr. Jerry Wilson, and Mr. Kevin Burke.

The team members presented an overview of the NRC, the differences between the 10 CFR Parts 50 and 52 licensing processes, specifics on the design certification process, an overview of the AP1000 design, a description of the vendor's test program and analytical models, a summary of the NRC's treatment of severe accidents, a summary of several policy and technical issues, and the role of the ACRS in the design certification process.

All of the presentations were very well received and good, probing questions were asked by the attendees. The team members exhibited great competence in their presentations and answers to questions.

8. Trip Report — Chalk River Facility in Canada

In response to the ACRS Chairman's request, Dr. Dana Powers provided a brief summary of his visit to the Chalk River facility in Canada. He stated that the ACRS is aware of the pre-application efforts for the Advanced Candu Reactor (ACR)-700 design. Originally, the ACRS Planning and Procedures Subcommittee has had an interest in examining a possible schedule to visit the Chalk River facility.

Currently, it seems that the detailed analyses have not been completed of either design basis accidents or severe accidents for the ACR-700. Design. Dr. Powers recommended postponing the site visit at a later date, and noted some of the issues that the ACRS will encounter with the certification of the ACR-700 design. Such issues could include the adequacy of the staff's review of thermal hydraulic analyses, adequacy of the staff's application to ACR-700 of rules and regulations established originally for boiling water and pressurized water reactors, and fire safety issues.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of July 30, 2004, to conclusions and recommendations included in the ACRS letter of April 27, 2004, concerning 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 Committee decided that it was satisfied with the EDO's response.

  The staff will consider the need to develop criteria that could be used to monitor and to limit any changes in risk associated with late containment failure as part of a future revision to RG 1.174.

- The Committee considered the EDO's response of July 20, 2004, to conclusions and recommendations included in the ACRS letter of June 9, 2004, concerning Digital Instrumentation and Controls Research Program.

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

- The Committee considered the EDO's response of July 21, 2004, to conclusions and recommendations included in the ACRS letter of June 15, 2004, concerning Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors." The Committee decided that it was satisfied with the EDO's response.

  The staff committed to keeping the ACRS apprized of the insights gained from the trial use of Regulatory Guide 1.201 in light of the issues and concerns highlighted in the Committee's June 15, 2004 report.

- The Committee considered the EDO's response of August 10, 2004, to conclusions and recommendations included in the ACRS report of July 20, 2004, concerning a Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP1000 Passive Plant Design.

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

- The Committee considered the EDO's response of August 25, 2004, to conclusions and recommendations included in the ACRS letter of May 21, 2004, concerning Resolution of Certain Items identified by the Advisory Committee on Reactor Safeguards in NUREG-1740, "Voltage-Based Alternative Repair Criteria." In general the Committee was satisfied with the EDO's response. However, the staff disagreed with the Committee recommendation regarding the best estimation of the heat transferred from the vessel to the steam generator during severe accidents. The Committee plans to prepare a response to the EDO on this matter during the October 2004 meeting.

  The staff committed to keep the Committee informed of activities related to open items in the Steam Generator Action Plan and to provide a copy of the screening analysis for Generic Issue-197 "Iodine Spiking Phenomena." The Committee plans to review steam generator tube pullout force data when provided by the staff.

<u>OTHER RELATED ACTIVITIES OF THE COMMITTEE</u>

During the period from July 7, 2004 through September 7, 2004, the following Subcommittee meetings were held:

- <u>Planning and Procedures</u> - September 7, 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.

<u>LIST OF MATTERS FOR THE ATTENTION OF THE EDO</u>

- The Committee plans to provide the staff with a list of topics to be discussed at a future Materials and Metallurgy Subcommittee meeting regarding the resolution of technical issues raised by the staff during their review of proposed technical specifications for ensuring steam generator tube integrity.

- The staff committed to revising the standard review plan and inspection procedures to incorporate knowledge gained from previous reviews of license renewal applications. This guidance will assist the staff in selecting the systems to be examined in detail as part of the review of license renewal applications.

- The Committee decided to review the draft Regulatory Guide DG-1XXX, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," after the public comments have been resolved.

- The Committee plans to discuss the assessment of the quality of the research projects on Sump Blockage and MACCS Code during October 7-9, 2004 ACRS meeting.

PROPOSED SCHEDULE FOR THE 516th ACRS MEETING

The Committee agreed to consider the following topics during the 516th ACRS meeting, to be held on October 7-9, 2004:

- Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance
- Pre-Application Safety Assessment Report for the Advanced CANDU 700 (ACR-700) Design
- Proposed Recommendations for Resolving GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs"
- Mitigating System Performance Index Program
- Response to the August 25, 2005 EDO Response to the May 21, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria"
- Technology Neutral Framework for Future Plant Licensing
- Assessment of the Quality of the NRC Research Projects
- Divergence in Regulatory Approaches and Requirements Between U.S. and Other Countries

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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SEPTEMBER 9-11, 2004

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

- Report on the Safety Aspects of the License Renewal Application for the Dresden 2 and 3 and Quad Cities 1 and 2 Nuclear Power Stations, dated September 16, 2004

MEMORANDA:

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

- Proposed Regulatory Guide 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," dated September 10, 2004

- Draft Regulatory Guide, DG-1.XXX, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," dated September 14, 2004

APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
V. List of Documents Provided to the Committee

CERTIFIED

MINUTES OF THE 516th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SEPTEMBER 9-11, 2004ROCKVILLE, MARYLAND

The 516th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on September 9-11, 2004. Notice of this meeting was published in the *Federal Register* on August 26, 2004 (65 FR 52530) (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. George E. Apostolakis, 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. 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 Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Plants (Open)

[Note: Mr. Marvin D. Sykes was the Designated Federal Official for this portion of the meeting.]

The Committee met with the NRC staff and representatives of the Exelon Generation Company to review and discuss the results of the staff evaluation of the license renewal application for the Dresden and Quad Cities Nuclear Power Stations and the associated final Safety Evaluation Report (SER). The applicant has requested approval for continued operation of the plant for a period of 20 years beyond the current license expiration dates. The operating licenses for Dresden 2 and 3 expire on December 22, 2009 and January 12, 2011. The Quad Cities 1 and 2 licenses expire December 14, 2012.

The license renewal application for Dresden and Quad Cities was submitted by a letter dated January 3, 2003. The application included an outline of specific actions that have been or will be taken to manage the effects of aging on the structures and components subject to aging management reviews (AMRs) such that the intended functions will be maintained consistent with the current licensing basis during the term of the renewed operating license.

There were **5 open and 16 confirmatory items** identified in the SER with Open Items provided to the ACRS on February 27, 2004. In the final SER issued July 2004, the staff stated that these open and confirmatory issues had been adequately addressed. The staff concluded in the final SER that the applicant had satisfied the requirements of 10 CFR 54 and imposed two general license conditions requiring the applicant to include the UFSAR Supplement in the next UFSAR update required by 10 CFR 50.71(e) following issuance of the renewed license and complete future inspections and analyses identified in the UFSAR Supplement in accordance with the schedules specified in Appedix A of the final SER.

Committee Action

The Committee issued a letter report to the NRC Chairman dated September 16, 2004 recommending that the Exelon application for renewal of the operating license for Dresden and Quad Cities be approved once the Committee's recommendations are accepted by the staff. Specifically, the Committee recommended that the staff require, prior to entering the period of extended operation, Exelon to conduct an evaluation to ensure that operating experience at extended power uprate (EPU) levels is properly addressed by the aging management programs. This evaluation should be reviewed and approved by the staff. The Committee also recommended that the steam dryers be included in the scope of license renewal for Dresden and Quad Cities.

III. Proposed Changes to the License Renewal Program (Open)

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

The Committee met with representatives of the Office of Nuclear Reactor Regulation (NRR) to discuss proposed changes to the scoping and screening reviews of license renewal applications.

Mr. Gillespie, NRR, began the discussion by describing the staff's efforts to improve the license renewal review process. Based on an anticipated schedule for issuing renewed licenses, Mr. Gillespie estimated that half of the industry could benefit from these improvements.

The next presentation made by Mr. Yerokun, NRR, described some of these proposed improvements. The first part of the presentation summarized an assessment of the staff's scoping and screening review process. This assessment was performed by a team comprised of NRR and regional staff experienced with the license renewal process. The objective of the assessment was to develop recommendations for improving the review process. The team examined audits of the scoping and screening methodologies, reviewed the scoping and screening results, and inspected the implementation of the scoping and screening results. The assessment found that the reviews were being implemented in accordance with regulations but there was duplication of effort and inconsistencies in license renewal guidance documents. The assessment team recommended that audits of the scoping and screening methodology and inspections of the implementation of the scoping and screening results be coordinated to eliminate duplication of effort that resulted in no added value to the review. The team also recommended that unique plant systems or systems that fall under 10 CFR 54.4(a)(2) be verified through inspection instead of requests for additional information (RAIs). Mr. Gillespie added that an inspection in which a reviewer is able to see the layout and relative spacing between non-safety related components and safety related components located in the same compartment is more effective than reviewing drawings obtained through RAIs. A recommendation was also made for the formation of a regional center of excellence that would plan and schedule all license renewal inspections. Finally, the assessment team recommended that inconsistencies among license renewal documents be corrected. Mr. Yerokun concluded this portion of his presentation by stating that a plan to implement these recommendations has been developed.

Mr. Leitch asked if steam dryers should be included in the scope of license renewal. Mr. Yerokun responded that those types of determinations were not within the scope of the team's assessment.

The second part of Mr. Yerokun's presentation described an approach for sampling the systems to be reviewed as part of a license renewal application. The systems to be considered for this sampling are the auxiliary systems and the steam and power conversion systems that fall under 10 CFR 54.4 (a)(1) and 10 CFR 54.4 (a)(2). The staff proposes to perform a detailed review of all the components in at least 50% of these systems. The selection of systems to be reviewed would consider inherent risks, experience from previous reviews of license renewal applications, and insights from operating experience. The selection of systems would also be such that the applicant would not be able to predict which systems would be selected. Mr. Yerokun concluded his presentation by stating that these proposed changes will improve the effectiveness and efficiency of the reviews of license renewal application while providing reasonable assurance that all structures, systems, and components should be identified as part of an aging management program.

Dr. Powers asked how the experience from earlier reviews of license renewal applications will be incorporated into the sampling methodology given that documentation of this information is incomplete and knowledge from experienced reviewers will eventually be lost. **Mr. Gillespie committed to revising the standard review plan and inspection procedures to incorporate the knowledge gained from previous reviews of license renewal applications.**

Dr. Kress noted that an obvious criticism of this approach is that the review is incomplete because not all of the systems were examined by the staff. In response to a question, Mr. Gillespie stated that reviewing only 50% of these systems would result in a significant savings of staff resources. Mr. Kuo, NRR, added that the first plant likely to be reviewed using this proposed sampling methodology is Brunswick.

Committee Action

This briefing was for information only. No committee action is necessary.

IV. Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity (Open)

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

The Committee met with representatives of NRR to discuss the staff's review of proposed technical specifications for steam generators (SG) tube integrity.

Mr. Ford opened the discussion by stating that the last Committee briefing on this topic was in December 2001 and at that time there were still a number of technical issues to be resolved.

The first presentation by Ms. Lund, NRR, reviewed the staff's efforts to revise the regulatory framework for the SG program. The industry documented its own initiative to develop a more performance-based program for SGs in an Nuclear Energy Institute (NEI) document number 97-06, "Steam Generator Program Guidelines." The staff is reviewing a Generic License Change Package (GLCP) and plant specific submittals implementing these NEI 97-06 guidelines. In response to a question, Mr. Riley, NEI, stated that the industry has already committed to implementing NEI 97-06, the associated Electric Power Research Institute (EPRI) documents, and the GLCP.

The next presentation was by Mr. Murphy, NRR. Mr. Murphy stated that all of the outstanding issues regarding the GLCP submitted by NEI have been resolved and the staff plans to issue a safety evaluation for the lead plant (Farley) by September 17, 2004. Some of the key issues that have been addressed since December 2001 include inspection intervals, performance criteria, repair limits, and repair methods. Mr. Murphy stated that the current technical specifications (TS) call for a specified number of SG tubes to be inspected at a specified frequency. Mr. Murphy characterized these SG inspection and repair requirements as too prescriptive while not focusing on ensuring tube integrity. In response, Mrs. Lund stated that the current TS were based on a wastage degradation mechanism and over time new degradation mechanisms have emerged. In addition, operating experience has shown that the previous requirements are not adequate and need to be changed.

The proposed TS will set a new Limiting Condition for Operation (LCO) for operational leakage and establish a new administrative TS for a steam generator program. The LCO for operational leakage change from 500 gpd to 150 gpd for each SG would provide added assurance that the plant can be shut down before tube rupture. The steam generator program TS specify performance criteria for tube integrity, provisions for monitoring, repair criteria, and provisions for inspection. The structural integrity performance criteria specify safety factors (SFs) for normal and design basis accident (DBA) loads. In general the most limiting DBA for SG tubes is a main steamline break. The accident leakage performance criteria states that DBA leakage should not exceed values assumed in accident analyses and limits DBA leakage to 1 gpm for all SGs. The tube repair criteria and methods are consistent with those currently approved. The scope, method, and frequency of inspections should be done to ensure that integrity is maintained until the next inspection. At the first refueling outage all SG tubes must be inspected. A maximum time interval between inspections is determined based upon the results of the previous inspection and the SG tube material.

Questions were raised regarding the bases for the SFs specified in the structural integrity performance criteria and the failure probabilities associated with them. The staff stated that the SFs were derived from the stress limits in the ASME Code and no failure probability is associated with them. The staff added that the SFs are conservative.

Mr. Murphy concluded his presentation by describing future actions by the staff. After the GLCP has been reviewed, the draft safety evaluation will be issued for public comment. The staff is also preparing a generic letter (Steam Generator Tube Inspections) requesting licensees to describe their plans for modifying their SG program. This generic letter will be issued for public comment in the fall of 2004.

Mr. Ford requested that the staff provide a briefing to the Materials and Metallurgy Subcommittee describing how the technical issues raised by the staff during their review of the proposed TS were resolved. Mr. Bateman, NRR, asked the Committee to provide a specific list of topics they wish to discuss at this meeting.

Committee Action

This briefing was for information only. The members plan to provide the staff with a list of topics to be discussed at a future Materials and Metallurgy Subcommittee meeting regarding the resolution of technical issues raised by the staff during their review of the proposed TS for ensuring SG tube integrity.

V. Safeguards and Security Matters (Closed)

[Note: Dr. Richard P. Savio was the Designated Federal Official for this portion of the meeting.]

The Committee heard briefings by and held discussions with representatives of the NRC staff and NEI regarding safeguards and security matters. **Note: This session was 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).**

Committee Action

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

VI. Assessment of the Quality of the Selected NRC Research Projects (Open)

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

The Office of Nuclear Regulatory Research (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 developed a strategy for reviewing the quality of selected research projects. This strategy is to be tried during FY 2004 and refined in FY 2005. During the September 9-11, 2004 ACRS meeting, the Committee discussed the preliminary results of the cognizant ACRS members' assessment of the quality of the NRC research projects on Sump Blockage and on MACCS Code.

Committee Action

The Committee plans to discuss the assessment of the quality of the research projects on Sump Blockage and MACCS Code during October 7-9, 2004 ACRS meeting.

VII. Divergence in Regulatory Approaches 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 Staff Requirements Memorandum (SRM), resulting from the April 11, 2003 ACRS meeting with the Commission, it was 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 the draft of a white paper to be used by the ACRS in responding to the Commission. During the September 9-11, 2004 ACRS meeting, the Committee discussed the draft White Paper regarding divergence in regulatory approaches 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 October 7-9, 2004 ACRS meeting.

VIII. Trip Report — AP1000 Workshop in China

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

Dr. Thomas Kress, Chairman of the ACRS Subcommittee on Future Plant Designs, provided a trip report to the Committee regarding a workshop held in Beijing, China on July 26-28, 2004. He stated that the workshop was the direct result of a proposal from the Chinese regulatory authorities made to Chairman Diaz during his visit in April 2004. The purpose of the workshop was to acquaint the Chinese with the U.S. nuclear regulatory process, particularly for design certification, and specifically with the activities associated with the design certification of the AP1000. The workshop team from the NRC was headed by Mr. Ashok Thadani. Other team members included Dr. Stephen Bajorek, Mr. Jerry Wilson, and Mr. Kevin Burke.

The team members presented an overview of the NRC, the differences between the 10 CFR Parts 50 and 52 licensing processes, specifics on the design certification process, an overview of the AP1000 design, a description of the vendor's test program and analytical models, a summary of the NRC's treatment of severe accidents, a summary of several policy and technical issues, and the role of the ACRS in the design certification process.

All of the presentations were very well received and good, probing questions were asked by the attendees. The team members exhibited great competence in their presentations and answers to questions.

IX. Trip Report — Chalk River Facility in Canada

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

In response to the ACRS Chairman's request, Dr. Dana Powers provided a brief summary of his visit to the Chalk River facility in Canada. He stated that the ACRS is aware of the pre-application efforts for the Advanced Candu Reactor (ACR)-700 design. Originally, the ACRS Planning and Procedures Subcommittee has had an interest in examining a possible schedule to visit the Chalk River facility.

Currently, it seems that the detailed analyses have not been completed of either design basis accidents or severe accidents for the ACR-700. Design. Dr. Powers recommended postponing the site visit at a later date, and noted some of the issues that the ACRS will encounter with the certification of the ACR-700 design. Such issues could include the adequacy of the staff's review of thermal hydraulic analyses, adequacy of the staff's application to ACR-700 of rules and regulations established originally for boiling water and pressurized water reactors, and fire safety issues.

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 considered the EDO's response of July 30, 2004, to conclusions and recommendations included in the ACRS letter of April 27, 2004, concerning 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 Committee decided that it was satisfied with the EDO's response.

  The staff will consider the need to develop criteria that could be used to monitor and to limit any changes in risk associated with late containment failure as part of a future revision to RG 1.174.

- The Committee considered the EDO's response of July 20, 2004, to conclusions and recommendations included in the ACRS letter of June 9, 2004, concerning Digital Instrumentation and Controls Research Program.

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

- The Committee considered the EDO's response of July 21, 2004, to conclusions and recommendations included in the ACRS letter of June 15, 2004, concerning Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors." The Committee decided that it was satisfied with the EDO's response.

The staff committed to keeping the ACRS apprized of the insights gained from the trial use of Regulatory guide 1.201 in light of the issues and concerns highlighted in the Committee's June 15, 2004 report.

- The Committee considered the EDO's response of August 10, 2004, to conclusions and recommendations included in the ACRS report of July 20, 2004, concerning a Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP1000 Passive Plant Design.

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

- The Committee considered the EDO's response of August 25, 2004, to conclusions and recommendations included in the ACRS letter of May 21, 2004, concerning Resolution of Certain Items identified by the Advisory Committee on Reactor Safeguards in NUREG-1740, "Voltage-Based Alternative Repair Criteria." In general the Committee was satisfied with the EDO's response. However, the staff disagreed with the Committee recommendation regarding the best estimation of the heat transferred from the vessel to the steam generator during severe accidents. The Committee plans to prepare a response to the EDO on this matter during the October 2004 meeting.

  The staff committed to keep the Committee informed of activities related to open items in the Steam Generator Action Plan and to provide a copy of the screening analysis for Generic Issue 197 "Iodine Spiking Phenomena." The Committee plans to review steam generator tube pullout force data when provided by the staff.

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 September 8, 2004. The following items were discussed:

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

Member assignments and priorities for ACRS reports and letters for the September 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 November 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

Proposed ACRS Meeting Dates for CY 2005

Proposed ACRS meeting dates for CY 2005 were discussed and summarized below.

Meeting No.	Dates
---	January 2005 (No meeting)
519	February 10-12, 2005
520	March 3-5, 2005
521	April 7-9, 2005
522	May 5-7, 2005
523	June 1-3, 2005
524	July 6-8, 2005
---	August 2005 (No meeting)
525	September 7-10, 2005
526	October 6-8, 2005
527	November 3-5, 2005
528	December 1-3, 2005

The Committee needs to approve the meeting dates for CY 2005 either during the October 2004 ACRS meeting.

Summary Matrix of ACRS Reports and Letters

In accordance with a Commission SRM, the ACRS Office needs to submit to the Commission, along with the ACRS Operating Plan, a summary matrix of ACRS reports and letters. To preclude violation of the ACRS Bylaws, the Committee should authorize the ACRS Executive Director and/or his designee to summarize the comments and recommendations included in the ACRS reports and letters that were issued in FY 2004. Upon completion, a copy of the summary matrix will be provided to the members for review and comment.

Draft Final ACRS Action Plan

A draft final version of the ACRS Action Plan was sent to the members and ACRS staff engineers on August 2, 2004. The current version of the Action Plan reflects incorporation of the comments received from most of the members and staff engineers. Subsequent to the Committee's approval, this Action Plan will be published.

ACRS Retreat in 2005

The Committee needs to decide whether it intends to hold a retreat in 2005. If decided to have a retreat, it should decide on topics, location, and dates. Also, the Committee should assign a lead member to work with the ACRS Executive Director to develop an agenda.

ACNW Working Group Meeting on Radiation Protection

The ACNW Working Group on Radiation Protection plans to hold a meeting on October 19, 2004, in the NRC Auditorium. The purpose of this meeting is to review the proposed recommendations by ICRP in the area of radiation protection. Since these recommendations may have some impact on 10 CFR Part 20, the NRC plans to provide comments on the proposed ICRP recommendations. The ACNW comments will be factored into the agency comments and sent to ICRP.

The ACNW would like to have participation by interested ACRS members in this Working Group meeting.

FY 2005 NRC Budget

The NRC budget for FY 2005 is not expected to be approved prior to the beginning of FY 2005. As a result, the agency has begun contingency planning in anticipation of operating under continuing resolution through the first half of FY 2005. All NRC Offices will receive funding at the FY 2004 level. As in the past, all purchases made by using the office bankcard must have prior approval from Tanya Winfrey while the continuing resolution is in effect.

Public Interest in Risk-informing 10 CFR 50.46

Two members of the public informed Dr. Powers, ACRS member, that they have some issues regarding risk-informing 10 CFR 50.46. Dr. Powers suggested hiring these individuals as ACRS consultants to participate in the Committee's review of a conceptual framework for risk-informing 10 CFR 50.46 and the expert elicitation to estimate LOCA frequencies. Dr. Shack, Chairman of the ACRS Subcommittee on Regulatory Policies and Practices, does not believe it is a good idea to hire these individuals as ACRS consultants since it will pave the way for other experts to seek the same treatment. However, he does not object to hearing their views at an ACRS Subcommittee meeting.

If these two individuals would like to provide their views on risk-informing 10 CFR 50.46, they can do so by attending an ACRS Subcommittee meeting dealing with that issue (they should pay for their own expenses) or by providing their comments in writing.

Member Issues

- Mr. Sieber suggested that the Committee hear a briefing from the staff on differences in regulatory requirements for cable separation related to Appendix R versus Regulatory Guide 1.75.

- In an e-mail to Dr. Bonaca dated August 24, 2004, Mr. David Collins, Engineering Analyst, Dominion Nuclear Connecticut, stated that an effective, integrated safety culture management methodology is a new and complex concept to understand, but none of the various supporting concepts are individually new or difficult to understand. He is working on developing an automated voice-narrated power point presentation that breaks the concept down into the individual components which will hopefully make it understandable to everyone, not just human performance professionals. He would like to brief the ACRS regarding his views on safety culture.

  On August 30, 2004, the Commission issued an SRM on staff action related to Safety Culture, which clearly defines the NRC activities in this area. The emphasis of the SRM was for the staff to use its inspection program and other indicators currently available to fully address safety culture. The staff should develop tools that allow inspections to rely more on objective findings and should be properly trained in the area of safety culture. Also, the Commission noted that in making any changes, the staff should involve stakeholders, which includes ACRS.

– Dr. Graham Wallis has been reviewing, in detail, work in the area of PWR Sump Performance which is being done by the NRC staff and its contractor. Dr. Wallis has identified a number of technical deficiencies in the work characterizing the debris blockage phenomena and associated pressure drop across the pump screen. Members' comments on these issues are solicited. This review is part of the Subcommittee's activities and responsive to the Commission's June 30, 2004, SRM requesting the ACRS to work with the staff to resolve outstanding issues with respect to PWR Sump Performance, and make recommendations for a practical solution within a reasonable period of time.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 516th ACRS Meeting, October 7-9, 2004.

The 515th ACRS meeting was adjourned at 11:00 am on September 11, 2004.

From: <JDSIEBER@aol.com>
To: <SAM@nrc.gov>
Date: 10/18/04 3:13PM
Subject: Minutes for 515 ACRS meet9ing

Sherry,

I received the minutes for the 515 meeting over the weekend. I have a comment:

On page 13 -- "Member issues," The sentence should read:

"Mr. Sieber suggested that the Committee hear a briefing from the staff on differences in regulatory requirements for cable separation related to Appendix R versus Regulatory Guide 1.75.

CC: <jtl@nrc.gov>

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

October 12, 2004

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 515th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - September 9-11, 2004

Enclosed are the proposed minutes of the 515th 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

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

October 20, 2004

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

FROM: Mario V. Bonaca
Chairman

Mario V. Bonaca

SUBJECT: CERTIFIED MINUTES OF THE 515th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), SEPTEMBER 9-11, 2004

I certify that based on my review of the minutes from the 515th 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.

Signed at Washington, DC, on August 20, 2004.

John L. Henshaw,

Assistant Secretary of Labor.

[FR Doc. 04–19532 Filed 8–25–04; 8:45 am]

BILLING CODE 4510–26–M

MEDICARE PAYMENT ADVISORY COMMISSION

Commission Meeting

AGENCY: Medicare Payment Advisory Commission.

ACTION: Notice of meeting.

SUMMARY: The Commission will hold its next public meeting on Thursday, September 9, 2004, and Friday, September 10, 2004, at the Ronald Reagan Building, International Trade Center, 1300 Pennsylvania Avenue, NW., Washington, DC. The meeting is tentatively scheduled to begin at 10 a.m. on September 9, and at 9 a.m. on September 10.

Topics for discussion include initial findings on congressionally mandated studies including: specialty hospitals; certified registered nurse first assistants; physician practice expenses; risk adjustment and other issues related to the adjusted average per capita cost (AAPCC); and beneficiary cost sharing in private plans. Additional presentations will include analysis on post-acute care outcomes and state lessons from the Medicare prescription drug card program. The Commission will also discuss work plans for a study on skilled nursing facility quality measures and home health quality.

Agendas will be e-mailed approximately one week prior to the meeting. The final agenda will be available on the Commission's Web site (*www.MedPAC.gov*).

ADDRESSES: MedPAC's address is: 601 New Jersey Avenue, NW., Suite 9000, Washington, DC 20001. The telephone number is (202) 220–3700.

FOR FURTHER INFORMATION CONTACT: Diane Ellison, Office Manager, (202) 220–3700.

Mark E. Miller,

Executive Director.

[FR Doc. 04–19528 Filed 8–25–04; 8:45 am]

BILLING CODE 6820–BW–M

NATIONAL FOUNDATION ON THE ARTS AND THE HUMANITIES

National Endowment for the Arts; Arts Advisory Panel

Pursuant to Section 10(a)(2) of the Federal Advisory Committee Act (Pub. L. 92–463), as amended, notice is hereby given that a meeting of the Fellowships Advisory Panel, Literature section (Poetry Fellowships category) to the National Council on the Arts announced for September 21–23, 2004 in Room 716 at the Nancy Hanks Center, 1100 Pennsylvania Avenue, NW., Washington, DC, 20506, will be held as a meeting of the Arts Advisory Panel. All other information regarding this meeting remains unchanged.

Dated: August 19, 2004.

Kathy Plowitz-Worden,

Panel Coordinator, Panel Operations, National Endowment for the Arts.

[FR Doc. 04–19497 Filed 8–25–04; 8:45 am]

BILLING CODE 7537–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–313]

Entergy Operations, Inc.; Notice of Withdrawal of Application for Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Entergy Operations, Inc., (the licensee) to withdraw its April 2, 2003, application as supplemented by letters dated November 21 and December 31, 2003, for proposed amendment to Renewed Facility Operating License No. DPR–51 for the Arkansas Nuclear One, Unit No. 1, located in Pope County, Arkansas.

The proposed amendment would have revised the technical specifications pertaining to the fuel enrichment, the spent fuel pool (SFP) boron concentration and criticality analysis, the SFP regions (including the use of Metamic poison panels in a portion of the SFP) and loading restrictions, and the loading patterns in the new fuel storage racks.

The Commission had previously issued a Notice of Consideration of Issuance of Amendment published in the **Federal Register** on May 13, 2003 (68 FR 25651). However, by letter dated June 24, 2004, the licensee withdrew the proposed change.

For further details with respect to this action, see the application for amendment dated April 2, 2003, as supplemented by letters dated November 21 and December 31, 2003, and the licensee's letter dated June 24, 2004, which withdrew the application for license amendment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams/html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, or 301–415–4737 or by email to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 19th day of August 2004.

For the Nuclear Regulatory Commission.

Thomas W. Alexion,

Project Manager, Section 1, Project Directorate IV, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 04–19506 Filed 8–25–04; 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 September 9–11, 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).

Thursday, September 9, 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 Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Plants (Open)—The Committee will hear presentations by and hold 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.

10:45 a.m.–11:45 a.m.: Proposed Changes to the License Renewal Program (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed changes to the license renewal program related to scoping and screening processes.

12:45 p.m.–1:45 p.m.: Safety Evaluation for Proposed Amendment to Technical Specifications for Farley Units 1 and 2—Steam Generator Program (Open)—The Commission will hear presentations by NRC staff regarding the safety evaluation for a proposed amendment to technical specifications for Farley Units 1 and 2—Steam Generator Program.

2 p.m.–5:45 p.m.: Safeguards and Security (Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding safeguards and security matters.

6 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Friday, September 10, 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.: Assessment of the Quality of the Selected NRC Research Projects (Open)—The Committee will discuss the preliminary results of the cognizant ACRS members' assessment of the quality of the NRC research projects on Sump Performance and on MACCS Code.

10:45 a.m.–11:45 a.m.: Divergence in Regulatory Approaches Between U.S. and Other Countries (Open)—The Committee will discuss a draft White Paper prepared by Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding divergence in regulatory approaches between U.S. and other Countries.

12:45 p.m.–1:45 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

1:45 p.m.–2 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

2 p.m.–2:30 p.m.: Trip Report—AP1000 Workshop in China (Open)—The Committee will hear a report by and hold discussions with Dr. Kress, ACRS member, who attended the International Workshop on AP1000 that was held in China on July 26–29, 2004.

2:45 p.m.–3:15 p.m.: Trip Report—Chalk River Facility in Canada (Open)—The Committee will hear a report by and hold discussions with Dr. Powers, ACRS member, who visited the Chalk River Facility in Canada.

3:15 p.m.–4:15 p.m.: Draft Final ACRS Action Plan (Open)—The Committee will discuss the draft final ACRS Action Plan.

4:15 p.m.–6:30 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Saturday, September 11, 2004, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–12 Noon: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

12 Noon–12:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities 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.

In accordance with Subsection 10(d) P.L. 92–463, I have determined that it is necessary to close portions of this meeting noted above to discuss and protect information classified as national security information as well as safeguard information pursuant to 5 U.S.C. 552b(c)(1) and (3).

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: August 20, 2004.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 04–19507 Filed 8–25–04; 8:45 am]

BILLING CODE 7590–01–P

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 III: MEETING ATTENDEES

515TH ACRS MEETING
SEPTEMBER 9-11, 2004

NRC STAFF (September 9, 2004)

M. Heath, NRR
A. Lee, NRR
T. Lee, NRR
J. Yenokim, NRR
D. Merzke, NRR
G. Gillett, NRR
K. Corp, NRR
P.T. Kuo, NRR
G. Makar, NRR
P. Klein, NRR
S. Long, NRR
L. Land, NRR
K. Karwoski, NRR
S. Peters, NRR
W. Bateman, NRR
L. Olshan, NRR
K. Kavanagh, NRR
J. Davis, RES
S. Wong, NRR
T. Tjader, NRR
L. Rossback, NRR
R. Auluck, NRR
T. Valentine, NRR
M. Rock, NRR
J. Rawley, NRR
B. Elliot, NRR
J. Honcharlk, NRR
T. Ford, NRR
J. Eads, NRR
A. Pal, NRR
J. Strnisha, NRR
R. Jullat, NRR
G. Suber, NRR
S. Lee, NRR
P. Y. Chen, NRR
P. Patnalk, NRR
M. Hartzman, NRR
S. Bailey, NRR
S. Coffin, NRR
A. Black, NRR
B. Smith, NRR
A. Stubbs, NRR
D. Shum, NRR
D. Reddy, NRR
C. Wu, NRR
Y. Diaz, NRR
R. Hernandez, NRR
R. Reyes, NRR
H. Asher, NRR
S. Mitra, NRR
J. Ma, NRR
P. Qualls, NRR
N. Iqbal, NRR
T. Liu, NRR
S. Hoffman, NRR
J. Dixon-Herrity, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

B. Hoffman, Public Citizen
F. Polaski, Exelon
R. Stochniak, Exelon
J. Nosko, Exelon
W. Bohlke, Exelon
E. Flick, Exelon
W. Porter, Exelon
T. Raueh, Exelon
K. Jury, Exelon
A. Fluvio, Exelon
S. Traiforo, LINK
A. Tabatabai, LINK
C. Willbanks, ATL Intl
M. Hayse, Exelon
D. Tubbs, Exelon
S. Dolby, McGraw-Hill
F. Emerson, NEI
J. Riley, NEI

515th ACRS Meeting
September 9-11, 2004

Attendees (continued)

<u>NRC STAFF</u> (September 10, 2004)
N. Sui, NRR

<u>ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC</u>

S. Traiforos, LINK
J. Butler, NEI
C. Reid, Bechtel
L. Collins, Westinghouse

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

September 15, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
516th ACRS MEETING
OCTOBER 7-9, 2004

THURSDAY, OCTOBER 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 - 10:45 A.M. Safety Evaluation of the Industry Guidelines Related to Pressurized Water Reactor (PWR) Sump Performance (Open) (GBW/RC)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute regarding the staff's evaluation of the industry guidelines associated with the resolution of Generic Safety Issue (GSI)-191, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at PWRs," and related matters.

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

3) 11:00 - 12:30 P.M. Pre-Application Safety Assessment Report for the Advanced CANDU 700 (ACR-700) Design (Open) (TSK/MME)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's Safety Assessment Report related to the pre-application review of the ACR-700 design and related matters.

   Representatives of the Atomic Energy of Canada Ltd. may provide their views, as appropriate.

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

4) 1:30 - 3:00 P.M. Proposed Recommendations for Resolving GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs" (Open) (VHR/RC/MRS)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff and its contractors regarding the proposed recommendations for resolving GSI-185.

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

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

5) 3:15 - 4:45 P.M. Mitigating System Performance Index Program (Open) (JDS/MWW)

5.1) Remarks by the ACRS Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the Mitigating System Performance Index Program.

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

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

6) 5:00 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Pre-Application Safety Assessment Report for the ACR-700 Design (TSK/MME)

6.2) Safety Evaluation of the Industry Guidelines Related to PWR Sump Performance (GBW/RC)

6.3) Proposed Recommendations for Resolving GSI-185 (VHR/RC/MRS)

6.4) Mitigating System Performance Index Program (JDS/MWW)

6.5) Response to the August 25, 2004 EDO Response to the May 21, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)

FRIDAY, OCTOBER 8, 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 - 10:00 A.M. Technology Neutral Framework for Future Plant Licensing (Open) (TSK/MME)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding the technology neutral framework for licensing of future plant designs.

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

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

9) 10:15 - 11:30 A.M. Assessment of the Quality of the NRC Research Projects (Open) (DAP/SLR/TSK/RC/HPN)

9.1) Remarks by the Subcommittee Chairman

9.2) Discussion of the preliminary results of the cognizant ACRS members' assessment of the research projects on Sump Blockage and on MACCS code.

10) 11:30 - 12:15 P.M. Divergence in Regulatory Approaches and Requirements Between U.S. and Other Countries (Open) (DAP/HPN/SD)

10.1) Remarks by the Subcommittee Chairman

10.2) Discussion of the draft Final White Paper prepared by Dr. Nourbakhsh, ACRS Senior Staff Engineer, regarding divergence in regulatory approaches and requirements between U.S. and other countries.

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

11) 1:15 - 2:15 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)

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

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

12) 2:15 - 2:30 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.

2:30 - 2:45 P.M. *****BREAK*****

13) 2:45 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of the proposed ACRS reports on:

13.1) Pre-Application Safety Assessment Report for the ACR-700 Design (TSK/MME)

13.2) Safety Evaluation of the Industry Guidelines Related to PWR Sump Performance (GBW/RC)

13.3) Proposed Recommendations for Resolving GSI-185 (VHR/RC/MRS)

13.4) Mitigating System Performance Index Program (JDS/MWW)

13.5) Response to the August 25, 2004 EDO Response to the May 21, 2004 ACRS Letter on Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria" (GBW/FPF/CS)

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

SATURDAY, OCTOBER 9, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 2:00 P.M.		Preparation of ACRS Reports (Open)
(12:00-1:00 P.M. - LUNCH)		Continue discussion of proposed ACRS reports listed under Item 13.
15) 2:00 - 2:30 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
515TH ACRS MEETING
SEPTEMBER 9-11, 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 September 8-11, 2004

2 Final Review of the License Renewal Application for the Dresden and Quad Cities Nuclear Plants
- 2. Dresden Nuclear Power Station, Quad Cities Nuclear Power Station presentation by Exelon Nuclear [Viewgraphs]
- 3. Dresden and Quad Cities Nuclear Power Station License Renewal Application presentation by T.J. Kim, Project Manager, NRR [Viewgraphs]
- 4. ACR-700 Prepared for the ACRS by Link Technologies [Handout]

3 Proposed Changes to the License Renewal Program
- 5. License Renewal Program Improvements presentation by J. Yerokun, RES [Viewgraphs]

4 Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity
- 6. Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity presentation by L. Lund, NRR [Viewgraphs]
- 7. Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity presentation by E. Murphy, NRR [Viewgraphs]

8 Assessment of the Quality of the Selected NRC Research Projects
- 8. The Value Tree for Finished Projects presentation by G. Apostolakis [Viewgraphs]

9 Divergence in Regulatory Approaches Between U.S. and Other Countries
- 9. Differences in Regulatory Approaches and Requirements Between U.S. and Other Countries (Progress Report on the White Paper) presentation by H. Nourbakhsh [Viewgraphs]

10. Future ACRS Activities/Report of the Planning and Procedures Subcommittee
- 10. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - September 8, 2003 [Handout #10.1]

11. Reconciliation of ACRS Comments and Recommendations
- 11. Reconciliation of ACRS Comments and Recommendations [Handout #11.1]

12 <u>Trip Report - AP1000 Workshop in China</u>

12. Trip Report - Workshop in China on Design Certification Process of AP1000 [Handout 12-1]

13 <u>Trip Report - Chalk River Facility in Canada</u>

13. Trip Report: Visit to Chalk River and the Certification of the ACR-700 [Handout 13-1]

Appendix V
515th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Review of the Plant License Renewal Application and Final SER for the Dresden and Quad Cities Nuclear Power Stations
	1. Table of Contents
	2. Meeting Schedule
	3. Status Report dated September 9, 2004
3	Proposed Changes to the License Renewal Program
	4. Table of Contents
	5. Proposed Meeting Schedule
	6. Status Report dated September 9, 2004
	7. Memorandum to Richard Barrett, NRR/DE, Suzanne Black, NRR/DSSA, Bruce Boger, NRR/DIPM, Ledyard Marsh, NRR/DLPM, and David Mathews, NRR/DRIP, from Paotsin Kuo, NRR, Program Director, License Renewal and Environmental Impact Programs, Subject: Results of the Assessment of the Nuclear Regulatory Commission's Review of the Scoping and Screening of License Renewal Applications, April 28, 2004
	8. Memorandum to David Mathews, Director, DRIP, from Suzanne Black, Director, DSSA, Subject: Sampling Approach for the Review of the Scoping and Screening of License Renewal Applications
4	Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity
	9. Table of Contents
	10. Proposed Meeting Schedule
	11. Status Report
	12. Memorandum to A. Louise Lund, Chief, Steam Generator Integrity and Chemical Engineering Section, to Mary Jane Ross-Lee, Acting Chief, Project Directorate Section II-1A, Subject: Safety Evaluation for Proposed Amendment to Technical Specifications for Farley Units 1 and 2 - Steam Generator Program, August 12, 2004

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

515th FULL COMMITTEE MEETING
SEPTEMBER ~~8~~ 9-11, 2004

[redacted]
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
MAURICE HEATH	NRR/DRIP/RLEP
Arnold Lee	NRR/DE/EMEB
Tommy Lee	NRR/DRIP/RLEP
Jimi Yerokun	NRR/RES
DANIEL MERZKE	NRR/DRIP/RLEP
Greg Galletti	NRR/DIPM/IPSB
Kimberley Corp	NRR/DRIP/RLEP
P T Kuo	NRR/DRIP/RLEP
Greg Makar	NRR/DE/EMCB
Paul Klein	NRR/DE/EMCB
Steve Long	NRR/DSSA/SPSB
Louise Lund	NRR/DE/EMCB
Ken Karwoski	NRR/DE/EMCB
Sean Peters	NRR/DLPM/LPDII-1
William Bateman	NRR/DE/EMCB
L.N. OLSHAN	NRR/DLPM/PDII-1
KERRI KAVANAGH	NRR/DIPM/IROB
James A. Davis	RES/DET/MEB
See-Meng Wong	NRR/DSSA/SPSB
T.R. TJADER	NRR/DIPM/IROB-A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

515th FULL COMMITTEE MEETING
SEPTEMBER ~~8~~ 9-11, 2004

[redacted]
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
LARRY ROSSBACH	NRC/DLPM
Raj Auluck	NRC/NRR/DRIP/RLEP
Theresa Valentine	NRC/NRR/DSSA/SPSB
MOLLIE ROCK	NRC/NRR/DIPM/IPSB
[illegible]	[illegible]
BARRY ELLIOT	NRC/NRR/EMCB
John Honcharik	NRC/NRR/DE-EMCB
Tanya Ford	NRC/NRR/SRXB/DSSA
Johnny Eads	NRC/RLEP
AMAR PAL	NRC/NRR/DE/EEIB
JIM STRNISHA	NRC/NRR/DE/EMEB
Ram Subbaratnam	NRC/NRR/RLEP
GREGORY SUBER	NRC/NRR/RLEP A
SAM LEE	NRC/NRR/RLEP
P Y Chen	NRC/NRR/DE/EMEB
PAT PATNAIK	NRR/DE/EMCB.
MARK HARTZMAN	NRR/DE/EMEB
Stewart Bailey	NRR/DE/EMEB
Stephanie Coffin	NRR/DE/EMCB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

515th FULL COMMITTEE MEETING
SEPTEMBER ~~8~~ 9-11, 2004

[redacted]
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Allison Black	NRC/DE/NRR/EMCB
Brian Smith	NRR/DIPM/IIPB
ANGELO STUBBS	NRR/DSSA/SPLB
DAVID SHUM	NRR/DSSA/SPLB
DEVENDER REDDY	NRR/DSSA/SPLB
CHENG-IH (JOHN) WU	NRR/DE/EMEB
Yoira Diaz	NRR/RLEP/
Raul Hernandez	NRR/DSSA/SPLB
Ruth Reyes	NRR/DSSA/SPLB
Hans Ashar	NRR/DE/EMEB
S.K. MITRA	NRR/DRIP/RLEP
J S Ma	NRR/DE/EMEB
Phil QUALLS	NRR/~~[illegible]~~DSSA/SPLB
Naeem Iqbal	NRR/DSSA/SPLB
TILDA LIU	NRR/~~DSSA~~ DRIP/RLEP
Steve Hoffman	NRR/DRIP/RLEP
J. L. DIXON-HERRITY	NRR/DSSA/SPLB
[redacted]	
TOMMY LE	NRR/DRIP/RLEP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

515th FULL COMMITTEE MEETING

SEPTEMBER ~~8~~ 9-11, 2004

[redacted]
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
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NAME	AFFILIATION
Fred Polaski	Exelon
Rob Stachniak	Exelon
John Nosko	Exelon
WH Bohlke	Exelon
Elliott Flick	Exelon
William Porter	Exelon
Tim Rausch	Exelon
Keith Jury	Exelon
Al Fulvio	Exelon
Spyros Traiforos	LINK
Ali Tabatabai	LINK
Charles Willbanks	ATL Intl
Michael A Hayse	Exelon
David C Tubbs	MidAmerican Energy
Steven Dolley	McGraw-Hill
Fred Emerson	NEI
Jim Riley	NEI

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

515th FULL COMMITTEE MEETING
SEPTEMBER ~~8~~-11, 2004
9-

[redacted]
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
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NAME	AFFILIATION
SPYROS TRAIFOROS	LINK
John BUTLER	NEI
CAL REID	BECHTEL
Leslie Collins	Westinghouse
NATHAN SIU	NRC/RES

①

ITEMS OF INTEREST

515th ACRS MEETING

SEPTEMBER 8-11, 2004

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
515th MEETING
September 8-11, 2004

NUCLEAR NEWS FLASHES

U.S. NEWS:

International

August 30, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary

SUBJECT: STAFF REQUIREMENTS - SECY-04-0111 - RECOMMENDED STAFF ACTIONS REGARDING AGENCY GUIDANCE IN THE AREAS OF SAFETY CONSCIOUS WORK ENVIRONMENT AND SAFETY CULTURE

The Commission has approved Option 1A to engage stakeholders by noticing the draft document in the Federal Register for a brief comment period, subject to the changes noted in the attachment. Options 1B and 1C are disapproved. Although the document is being issued for public comment, there should be no further discussion on whether to issue the document. It should be clear to stakeholders that the comments should address the content of the document only. The content of any notice attached to the document should explicitly reflect the connection between a Safety Conscious Work Environment and Safety Culture. At a minimum the staff should explain as it did in the paper that SCWE is an attribute of Safety Culture.
(EDO) (SECY Suspense: 9/24/04)

The Commission has approved Option 2C to continue to monitor industry efforts to assess Safety Culture and ensure the Commission remains informed of industry efforts and progress. Of particular note was the progress made by INPO to address recent industry issues in this area. As industry works to develop guidance in this area, the staff should use its resources to ensure that it has programs and procedures in place that encourage licensees to establish strong Safety Culture programs. Options 2A and 2B are disapproved.

The Commission has approved Option 3B to enhance the Reactor Oversight Process (ROP) treatment of cross-cutting issues to more fully address Safety Culture. The staff should not use surveys of licensee personnel, but rather should rely on inspector observations and other indicators already available to the NRC. Consequently, the staff should develop tools that allow inspectors to rely on more objective findings. The staff should consider including enhanced problem identification and resolution initiatives as part of this effort. Most important, the staff should ensure that the inspectors are properly trained in the area of Safety Culture. The staff should consider developing an enhanced training program for its inspectors and resident inspectors on Safety Culture that uses both insights from INPO's work in this area and insights from the international community. The staff should consider if the cross-cutting issues in the enhanced ROP treatment may be more appropriately labeled Safety Management rather than Safety Culture. In making any changes, the staff should follow the established processes for revising the ROP, in particular the process for involving stakeholders.

As a further enhancement to the ROP, the staff should include as part of its enhanced inspection activities for plants in the Degraded Cornerstone Column (referred to as Column Three) of the ROP Action Matrix, a determination of the need for a specific evaluation of the licensees Safety Culture. The staff should interact with our stakeholders to develop a process for making the determination and conducting the evaluation. The staff's methodology for using the treatment of cross-cutting issues to more fully address Safety Culture should require a specific determination for plants in the Degraded Cornerstone Column.

With respect to Option 3C, the staff should continue to monitor developments by foreign regulators, as directed in the SRM on SECY-02-0166, but should limit the expenditure of resources in this area to previously programmed levels. Options 3A, 3D, and 3E are disapproved.

The attachment contains recommended revisions to the draft document on "Establishing and Maintaining a Safety Conscious Work Environment", but the staff should feel free to continue to improve this document.

Attachment: Changes to the Federal Register notice in SECY-04-0111

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

Attachment

Changes to the Federal Register notice in SECY-04-0111

1. On page 3, last paragraph, revise line 2 to read ' ... raise concerns; and (c) makes clear that ....'

2. On page 4, bullet 2 at the top of the page, add semi-colon at the end. In bullet 5, add a period at the end.

3. On page 4, 1st full paragraph, line 2, correct the spelling of "principles."

4. On page 4, sub-bullet 4, revise line 2 to read ' ... regarding a ~~an~~ provision of ....' In sub-bullet 5, revise line 1 to read ' ... in, or being ~~is~~ about to ....'

5. On page 5, 2nd full paragraph, revise line 2 to read ' ... quality assurance programs, corrective ....'

6. On page 6, bullet 3, delete "or not".

7. On page 6, paragraph 2, remove the comma after "staff."

8. On page 6, last paragraph, after the title to paragraph B., remove the dash and start a new paragraph with "Aside."

9. On page 7, paragraphs C., D., E., and F., remove the dash and start a new paragraph after the title.

10. On page 7, paragraph E., revise the title to read '**Timely Feedback is** ...' Revise line 1 to read 'Timely f~~F~~eedback should ....'

11. On page 8, paragraphs G., H., and A., remove the dash and start a new paragraph after the title.

12. On page 8, paragraph H., line 10, correct the spelling of "accessibility."

13. On page 8, paragraph A., revise the title to read '**Lessons Learned Evaluations**' Revise line 1 to read 'It may be useful to p~~P~~eriodically evaluate ....'

14. On page 9, paragraphs B., C., and D., remove the dash and start a new paragraph after the title.

15. On page 9, last paragraph under C., revise line 4 to read ' ... above and others may provide some ....'

16. On page 10, 4th full paragraph, revise line 2 to read ' ... work groups or ~~and~~ generic to the ....' Revise line 4 to read ' ... results of a survey or ....'

17. On page 10, paragraphs E., F., and G., remove the dash and start a new paragraph after the title.

18. On page 11, paragraph 1, line 7, consider replacing "reasonableness" with "extent" or "effectiveness."

19. On page 11, paragraphs A., B, and C., remove the dash and start a new paragraph after the title.

20. On page 11, paragraph A., revise lines 6 and 7 to read ' ... subcontractors that the licensee ~~they~~ expects them to ....' Revise line 8 to read ' ... discrimination against ~~of~~ contractor ....' Revise line 9 to read ' ... SCWE, or they adopt and ....'

21. On page 12, paragraph 1, revise line 4 to read ' ... the potential impact the contractor's ~~their~~ actions might ....'

22. On page 12, paragraph D., remove the dash and start a new paragraph after the title.

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No. S-04-013

PDF Version (14 KB)

REMARKS BY CHAIRMAN NILS J. DIAZ,
UNITED STATES NUCLEAR REGULATORY COMMISSION
at the
NRC ANNUAL DIVERSITY DAY CELEBRATION
10:00 A.M. WEDNESDAY
JULY 14, 2004
NRC AUDITORIUM

Good morning, and welcome to NRC's annual observance of Diversity Day. Commissioner McGaffigan, Commissioner Merrifield, and I are pleased to join you this morning and to participate in this annual agency event.

Before we proceed further this morning, I want to acknowledge the many individuals and organizations that have come together to make this day possible. The Commission is pleased to welcome Navy Captain R. Sydney Abernathy III, our keynote speaker; Mr. Jack Julius, who will be performing a little magic for us later in the program; Ms. Kris Gamble of 99.5 FM Radio, who will serve as our master of ceremonies for the entertainment segments of the program this afternoon; and the various musicians, professional societies, arts and craft vendors, and food vendors who will be here either on the Green or in the Exhibit Area throughout the day. In addition, we appreciate the presence of representatives of the State of Maryland and the Government of Montgomery County as well as the embassies of Japan, Kenya, and Thailand. Our special thanks also go to the Office of Small Business and Civil Rights, which organizes and sponsors this event every year, and to the Diversity Day Planning Group that worked out all the details of our program today.

Our purpose this morning is to take time out from our normal activities to appreciate the varied ethnic and cultural backgrounds, perspectives, customs, and cuisines of our workplace colleagues; to recognize the unique American tapestry that the presence of many cultures, languages, and peoples on this land has woven over the course of time; and to acknowledge the extraordinary strength that diversity can bring to any organized activity whether that activity takes place at the neighborhood, community, State, or national levels. Diversity Day encompasses each one of us at the NRC not only because we each have our own particular cultural and ethnic background to value and share with others, but also because we all work together to achieve common goals that are important to our fellow citizens and to the Nation as a whole. This fundamental idea that diversity and unity are compatible with each other is reflected in the theme for today's event - "Diversity: United We Stand."

The belief that diversity and unity can mutually coexist is one of the distinguishing characteristics of modern America. Although nearly every modern society is experiencing significant increases in the diversity of their populations as a result of the growth of the global economy, the spread of information technology, and increasing ease of international travel, few modern societies understand diversity as a virtue. In 21st century America, by contrast, diversity is celebrated and perceived to be an inherent part of the definition of who we are as a people. In fact, today the United States may very well be the most diverse society on the planet. And we are continuing to become more diverse every day.

The agency today is far more diverse than it was in 1975 at the time it was created, and our diversity will continue to grow. We are a small, tightly organized, and highly skilled organization that has succeeded in what we do in part because we have learned to rely upon each other in carrying out our individual responsibilities. We have a common purpose -- protection of the public health and safety; common perceptions -- that each of us, every day, contributes to the achievement of agency objectives; shared experiences -- forging a high quality regulatory program that is recognized as such around the world; and commitment to individual worth -- evidenced by the mutual respect we have for each other in what we do and who we are. I believe that diversity has strengthened the NRC and will continue to do so as long as we continue to maintain our mutual respect, understanding, and appreciation of each other; in other words, as long as the spirit of Diversity Day that we celebrate and enjoy today remains with us every day in all our interactions in the NRC work environment.

One of the ways we seek to improve our performance as a diverse organization is to share experiences with other agencies of the Federal Government. In that regard, we are very fortunate to have with us this morning Captain R. Sydney Abernethy, III, of the United States Navy. Captain Abernethy is a native of Baltimore and grew up in Annapolis, where he graduated from the Naval Academy in 1981. He has had a variety of interesting assignments and experiences in the Navy; for example, he served three six-month deployments to Antarctica in support of scientific research; flew air reconnaissance missions out of Guam and in support of the Seventh Fleet; served on the staff of the Joint Chiefs in the Intelligence Directorate; participated in Operations Desert Storm and Desert Shield in the early 1990's; and served on board the USS John C. Stennis during its deployment to the Arabian Gulf in 1999. He is currently serving as the Special Assistant for Minority Affairs to the Chief of Naval Personnel. He is going to share with us some of the Navy's experience with diversity and diversity management and perhaps some of his personal experiences as a Naval Officer. Please join me in welcoming our guest speaker, Navy Captain R. Sydney Abernethy, III.

DIRECTOR, NAVAL REACTORS

30 July 2004

Dear Dr. Bonaca,

I just returned from directing initial sea trials of the nuclear-powered submarine PCU VIRGINIA (SSN 774), the first ship in the VIRGINIA class. I am proud to report that both the crew and the ship performed exceptionally well.

In this single platform, VIRGINIA combines a unique mix of stealth, endurance, agility, and firepower to fulfill vital national security roles, even in areas denied to other U.S. assets. Reflecting the *Operational Requirements Document* approved in September 1993, VIRGINIA is our Navy's ***only*** major combatant now ready for delivery that was designed with the post–Cold War security environment in mind. Specifically, VIRGINIA is built to dominate the hostile littorals without sacrificing undersea dominance in the open ocean. One significant technological development helping to maintain that dominance is the new propulsion plant with its life-of-the-ship reactor core. The success of these rigorous sea trials confirmed my high expectations about this ship's many capabilities—clearly, VIRGINIA is the finest submarine in the world.

Sincerely,

F. L. BOWMAN
Admiral, U.S. Navy

Dr. Mario V. Bonaca, Chairman
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
11555 Rockville Pike, Room 2E26
Rockville MD 20852

2Q/2004 ROP Inspection Findings Summary

This summary provides the color designation of the most significant inspection findings over the previous 4 quarters. Physical Protection information not publicly available.

Plants	Initiating Events	Mitigating Systems	Barrier Integrity	Emergency Preparedness	Occupational Radiation Safety	Public Radiation Safety
Arkansas Nuclear 1	No Findings	White (1)	Green	No Findings	No Findings	No Findings
Arkansas Nuclear 2	Green	Green	Green	No Findings	Green	No Findings
Beaver Valley 1	Green	Green	No Findings	Green	No Findings	No Findings
Beaver Valley 2	No Findings	Green	Green	Green	No Findings	No Findings
Braidwood 1	No Findings	Green	Green	No Findings	No Findings	No Findings
Braidwood 2	Green	Green	Green	No Findings	No Findings	No Findings
Browns Ferry 2	Green	No Findings	No Findings	No Findings	Green	No Findings
Browns Ferry 3	Green	Green	No Findings	No Findings	Green	No Findings
Brunswick 1	No Findings	Green	No Findings	No Findings	No Findings	No Findings
Brunswick 2	No Findings	White (1)	No Findings	No Findings	No Findings	No Findings
Byron 1	No Findings	Green	Green	No Findings	Green	No Findings
Byron 2	No Findings	Green	Green	No Findings	Green	No Findings
Callaway	Green	Green	Green	No Findings	Green	Green
Calvert Cliffs 1	Green	Green	No Findings	No Findings	No Findings	No Findings
Calvert Cliffs 2	Green	Green	No Findings	No Findings	No Findings	No Findings
Catawba 1	No Findings	Green	Green	No Findings	No Findings	No Findings
Catawba 2	No Findings	Green	Green	No Findings	No Findings	No Findings
Clinton	Green	Green	Green	No Findings	Green	No Findings
Columbia Generating Station	Green	Green	Green	No Findings	Green	No Findings
Comanche Peak 1	No Findings	Green	No Findings	No Findings	No Findings	No Findings
Comanche Peak 2	No Findings	Green	No Findings	No Findings	No Findings	No Findings
Cooper	Green	White (1)	Green	No Findings	No Findings	No Findings
Crystal River 3	Green	Green	Green	No Findings	No Findings	No Findings
D.C. Cook 1	No Findings	Green	Green	No Findings	Green	White (1)
D.C. Cook 2	Green	Green	Green	No Findings	Green	White (1)
Davis-Besse	Green	Yellow (1)	Red (1)	No Findings	No Findings	No Findings

Diablo Canyon 1	No Findings	Green	No Findings	No Findings	Green	No Findings
Diablo Canyon 2	No Findings	Green	No Findings	No Findings	Green	No Findings
Dre[redacted]	Green	Green	No Findings	No Findings	Green	No Findings
Dres[redacted]3	Green	Green	Green	No Findings	Green	No Findings
Duane Arnold	Green	Green	No Findings	No Findings	No Findings	Green
Farley 1	No Findings	Green	No Findings	No Findings	Green	No Findings
Farley 2	No Findings	Green	No Findings	No Findings	No Findings	Green
Fermi 2	Green	Green	No Findings	No Findings	No Findings	No Findings
FitzPatrick	Green	Green	Green	No Findings	No Findings	No Findings
Fort Calhoun	Green	Green	Green	No Findings	Green	No Findings
Ginna	Green	Green	Green	Green	No Findings	No Findings
Grand Gulf 1	Green	Green	No Findings	No Findings	No Findings	No Findings
Harris 1	No Findings	Green	Green	No Findings	No Findings	No Findings
Hatch 1	No Findings	Green	No Findings	No Findings	No Findings	No Findings
Hatch 2	No Findings	Green	No Findings	No Findings	No Findings	No Findings
Hope Creek 1	Green	White (1)	No Findings	No Findings	No Findings	No Findings
Indian Point 2	Green	Green	No Findings	Green	No Findings	No Findings
Indian Point 3	Green	Green	No Findings	Green	No Findings	No Findings
Kewaunee	No Findings	Green	No Findings	Green	No Findings	No Findings
La[redacted]	Green	Green	No Findings	No Findings	Green	No Findings
La Salle 2	Green	Green	No Findings	No Findings	Green	No Findings
Limerick 1	No Findings	Green	Green	No Findings	No Findings	No Findings
Limerick 2	No Findings	Green	Green	No Findings	No Findings	No Findings
McGuire 1	No Findings	Green	Green	No Findings	Green	No Findings
McGuire 2	No Findings	Green	No Findings	No Findings	No Findings	No Findings
Millstone 2	Green	Green	No Findings	No Findings	No Findings	No Findings
Millstone 3	Green	Green	No Findings	No Findings	No Findings	No Findings
Monticello	Green	Green	No Findings	No Findings	No Findings	Green
Nine Mile Point 1	Green	Green	Green	Green	No Findings	No Findings
Nine Mile Point 2	No Findings	No Findings	Green	No Findings	No Findings	No Findings
North Anna 1	Green	Green	No Findings	Green	No Findings	No Findings
North Anna 2	Green	Green	No Findings	Green	No Findings	No Findings
Oyster Creek	No Findings	White (1)	No Findings	Green	Green	No Findings
Oconee 1	Green	White (1)	Green	No Findings	No Findings	Green
Oconee 2	Green	White (1)	No Findings	No Findings	No Findings	Green
Oc[redacted]3	Green	White (1)	No Findings	No Findings	No Findings	Green
Pa[redacted]s	No Findings	Green	Green	No Findings	No Findings	No Findings
Palo Verde 1						

	No Findings	Green	No Findings	Green	No Findings	No Findings
Palo Verde 2	No Findings	Green	Green	Green	No Findings	No Findings
Pal[illegible]3	No Findings	Green	No Findings	Green	No Findings	No Findings
Peach Bottom 2	No Findings	White (1)	No Findings	No Findings	No Findings	No Findings
Peach Bottom 3	No Findings	Green	No Findings	No Findings	Green	No Findings
Perry 1	Green	White (2)	No Findings	White (1)	No Findings	No Findings
Pilgrim 1	Green	Green	No Findings	No Findings	No Findings	No Findings
Point Beach 1	Green	[illegible]	No Findings	Green	No Findings	No Findings
Point Beach 2	Green	[illegible]	Green	Green	Green	Green
Prairie Island 1	Green	Green	Green	No Findings	No Findings	No Findings
Prairie Island 2	Green	Green	Green	No Findings	No Findings	No Findings
Quad Cities 1	Green	Green	Green	No Findings	No Findings	No Findings
Quad Cities 2	Green	Green	No Findings	No Findings	No Findings	No Findings
River Bend 1	Green	Green	Green	No Findings	Green	Green
Robinson 2	No Findings	No Findings	No Findings	Green	Green	No Findings
Saint Lucie 1	No Findings	Green	Green	No Findings	No Findings	No Findings
Saint Lucie 2	Green	Green	No Findings	No Findings	No Findings	No Findings
Salem 1	Green	White (1)	Green	No Findings	No Findings	Green
Salem 2	Green	Green	Green	No Findings	No Findings	No Findings
Sa[illegible]e 2	Green	Green	No Findings	No Findings	Green	No Findings
San Onofre 3	No Findings	Green	No Findings	No Findings	Green	Green
Seabrook 1	Green	Green	No Findings	Green	No Findings	No Findings
Sequoyah 1	Green	Green	No Findings	No Findings	No Findings	No Findings
Sequoyah 2	Green	Green	No Findings	No Findings	No Findings	No Findings
South Texas 1	Green	Green	No Findings	No Findings	Green	No Findings
South Texas 2	Green	Green	No Findings	No Findings	Green	No Findings
Summer	Green	Green	No Findings	No Findings	No Findings	No Findings
Surry 1	No Findings	Green	No Findings	No Findings	No Findings	No Findings
Surry 2	No Findings	Green	Green	No Findings	No Findings	No Findings
Susquehanna 1	Green	Green	Green	No Findings	No Findings	No Findings
Susquehanna 2	Green	Green	Green	No Findings	No Findings	No Findings
Three Mile Island 1	No Findings	Green	Green	No Findings	No Findings	No Findings
Turkey Point 3	No Findings	Green	No Findings	No Findings	No Findings	Green
Turkey Point 4	No Findings	Green	No Findings	No Findings	No Findings	Green
Vermont Yankee	Green	Green	Green	No Findings	No Findings	No Findings
Vo[illegible]	Green	Green	No Findings	No Findings	Green	No Findings
Vog[illegible]2	Green	Green	No Findings	No Findings	Green	No Findings
Waterford 3						

	Green	White (1)	Green	No Findings	Green	No Findings
Watts Bar 1	Green	Green	Green	No Findings	No Findings	No Findings
Wo[illegible]k 1	Green	Green	No Findings	No Findings	Green	No Findings

PI Summary | Action Matrix Summary | Reactor Oversight Process

Last Modified: August 2, 2004

2Q/2004 ROP Performance Indicators Summary

Physical Protection information not publicly available.

Plants	IE 01	IE 02	IE 03	MS 01	MS 02	MS 03	MS 04	MS 05	BI 01	BI 02	EP 01	EP 02	EP 03	OR 01	PR 01
Arkansas Nuclear 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Arkansas Nuclear 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Beaver Valley 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Beaver Valley 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Braidwood 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Braidwood 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Bro[illegible]erry 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Browns Ferry 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Brunswick 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Brunswick 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Byron 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Byron 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Callaway	W	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Calvert Cliffs 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Calvert Cliffs 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Catawba 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Ca[illegible]	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G

Clinton	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Columbia Generating Station	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Comanche Peak 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Comanche Peak 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Cooper	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Crystal River 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
D.C. Cook 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
D.C. Cook 2	G	W	G	G	G	G	G	G	G	G	G	G	G	G	G
Davis-Besse	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Diablo Canyon 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Diablo Canyon 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Dresden 2	W	G	G	G	G	T	G	G	G	G	G	G	G	G	G
Dr[illegible]	G	G	G	G	W	T	G	G	G	G	G	G	G	G	G
Duane Arnold	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Farley 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Farley 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Fermi 2	G	G	G	W	G	G	G	G	G	G	G	G	G	G	G
FitzPatrick	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Fort Calhoun	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Ginna	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Grand Gulf 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Harris 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Hatch 1	G	G	G	G	G	G	G	G	G	G	G	G	U	G	G
Ha[illegible]	G	G	G	G	G	G	G	G	G	G	G	G	U	G	G

Hope Creek 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Indian Point 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Indian Point 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Kewaunee	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
La Salle 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
La Salle 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Limerick 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Limerick 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
McGuire 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
McGuire 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Millstone 2	W	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Millstone 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Monticello	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Nine Mile Point 1	G	G	G	G	G	T	G	G	G	G	G	G	G	G	G
Nine Mile Point 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
North Anna 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
North Anna 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Oconee 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Oconee 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Oconee 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Oyster Creek	G	G	G	G	U	T	G	G	G	G	G	G	G	G	G
Palisades	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Palo Verde 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Pal[illegible]	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G

Palo Verde 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Peach Bottom 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Peach Bottom 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Perry 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Pilgrim 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Point Beach 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Point Beach 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Prairie Island 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Prairie Island 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Quad Cities 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Quad Cities 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
River Bend 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Robinson 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Saint Lucie 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Saint Lucie 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Salem 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Salem 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
San Onofre 2	G	W	G	G	G	G	G	G	G	G	G	G	G	G	G
San Onofre 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Seabrook 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Sequoyah 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Sequoyah 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
South Texas 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
South Texas 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G

Summer	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Surry 1	G	G	G	W	G	G	G	G	G	G	G	G	G	G	G
Sur[illegible]	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Susquehanna 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Susquehanna 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Three Mile Island 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Turkey Point 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Turkey Point 4	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Vermont Yankee	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Vogtle 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Vogtle 2	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Waterford 3	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Watt[illegible] 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G
Wolf Creek 1	G	G	G	G	G	G	G	G	G	G	G	G	G	G	G

Legend: R=Red   W=White   T=Thresholds under development   N=Not Applicable
Y=Yellow   G=Green   I=Insufficient data to calculate PI   U=Unique Design

IE01 = Unplanned Scrams per 7000 Critical Hours
IE02 = Scrams with Loss of Normal Heat Removal
IE03 = Unplanned Power Changes
MS01 = Emergency AC Power System
MS02 = High Pressure Injection System
MS03 = Heat Removal System
MS04 = Residual Heat Removal System
MS05 = Safety System Functional Failures
BI01 = Reactor Coolant System Specific Activity
BI02 = Reactor Coolant System Leakage
EP01 = Drill/Exercise Performance
EP02 = ERO Drill Participation
EP03 = Alert and Notification System
OR01 = Occupational Exposure Control Effectivene
PR01 = RETS/ODCM Radiological Effluent

Inspection Findings Summary | Action Matrix Summary | Reactor Oversight Process

ast modified : July 28, 2004

Repeated white findings on pumps move Perry into ROP's column 4

Inside NRC
Volume 26 / Number 17 / August 23, 2004

The NRC moved Perry this month into the "multiple/repetitive degraded cornerstone" column of the reactor oversight process (ROP) action matrix, the second-worst performance category in NRC's five-column scheme.

Perry, which is operated by FirstEnergy Nuclear Operating Co. (Fenoc), is the sixth unit to enter that category in the four years since NRC adopted the ROP.

Under the ROP, one way for plants to move into the fourth column is by having at least two "white" findings—denoting low to moderate safety significance—in one of the cornerstones for five or more consecutive quarters. Perry met that criterion through three white findings at various points during the period in the area of "mitigating systems," or safety equipment. They involved the failure of a high-pressure core spray pump, which produced a white "inspection finding" that extended from fourth quarter 2002 through fourth quarter 2003; the failure of an emergency service water (ESW) pump, which resulted in a white finding starting in third quarter 2003 and extending into the current quarter; and one involving air binding of a residual heat removal and low-pressure core spray waterleg pump, which led to a white finding that began in fourth quarter 2003.

NRC Region III spokesman Jan

Strasma said that white findings normally
remain posted on a reactor's
wide reorganization. In its June 24 announcement of the reorganization, Fenoc said it was seeking to "operate more efficiently and effectively and continue the progress we have already made toward improving the performance at each of our nuclear plants." The announcement also said, "The overall size of the company is expected to be somewhat smaller through attrition and staff reductions, bringing Fenoc more in line with other top-performing nuclear utilities."

But Schneider said the goals of the reorganization were broader. He said Fenoc had benchmarked the best-performing companies in "performance, operations, and safety" as well as "employee head count."

He also said Davis-Besse performance since restart (Nucleonics Week, 20 May, 5) "shows Fenoc's capability." Grobe said he didn't think there was a connection between the current problems at Perry and the management shuffle in response to Davis-Besse. He said he didn't see the Perry troubles as a "management issue." The problems at Perry have been "nagging" issues, of relatively low significance, that didn't "seem to get fixed," he said. The fact that such issues could propel Perry into the second-worst category of the action matrix is "one of the beauties" of the ROP, because the system increases agency and operator focus on the problems before they create serious safety issues, he said.

But Dicus, now an independent consultant, said the current situation at Perry validates the concern she expressed at the briefing. When moving people within the organization, it is important for a company to be "planning it right and doing it right, and making sure someone is in charge," she said. The problems Fenoc is having indicate "a system-wide concern," she said. She said such problems are a potential concern for multi-unit operators. The more consolidation there is in the nuclear industry, "the more there needs to be protection" against such problems, she said.

But she emphasized she was not objecting to consolidation, which, she said, "is a better way to go" because plants can be run more inexpensively and can share resources. However, a company with multiple plants has to be able to carry out that kind of coordinated management, and "maybe Fenoc hasn't gotten there yet," she said.
—*Daniel Horner, Washin*gton

Industry to develop guidance on license renewal SAMA questions

Inside NRC

Volume 26 / Number 17 / August 23, 2004

The Nuclear Energy Institute (NEI) plans to develop a guidance document to streamline responses to NRC requests for additional information (RAIs) associated with license renewal applications, NEI representatives said at a meeting last week.

Industry representatives said at an Aug. 19 meeting that the guidance document would help with NRC's review of severe accident mitigation alternatives, known as SAMAs.

NRC regulations (10 CFR 51.53) require that "if the staff has not previously considered severe accident mitigation alternatives for the applicant's plant in an environmental impact statement or related supplement or in an environmental assessment, a consideration of alternatives to mitigate severe accidents must be provided" in the environmental review accompanying a power reactor license renewal application.

"The analysis of SAMAs includes the identification and evaluation of alternatives that reduce the radiological risk from a severe accident by preventing substantial core damage, i.e. preventing a severe accident, or by limiting releases from containment in the event that substantial core damage occurs, i.e. mitigating the impacts of a severe accident," NRC said in Nureg-1555, Supplement 1, published in October 1999.

SAMAs have long been a point of contention between NRC and industry. In 1999, NEI petitioned NRC to eliminate the SAMA requirement, claiming it was not directly related to component aging or other license renewal issues and not required by various NRC rulings and federal court decisions. In January 2001, the commission denied NEI's petition based on staff's recommendation in an October 2000 paper, Secy 00-210.

SAMA expenses

Industry now accepts the SAMA requirement but says it is burdened by the number and complexity of SAMA-related requests for RAIs issued by NRC staff, Bill Watson of

Dominion said in a presentation at the Aug. 19 meeting. "Applicants are having to apply more and more resources to the SAMA analysis, both pre- and post-submittal," Watson said, estimating that "40% to 50% of [environmental review] resources are spent on SAMA analysis, sometimes a little bit higher." Industry representatives said Exelon had spent $250,000 to answer RAI requests on the Dresden-Quad Cities license renewal application, and Dominion spent $250,000 to $300,000 on SAMA analysis for Millstone's application. Tim Abney of the Tennessee Valley Authority (TVA) said TVA spent more than $320,000 responding to RAIs on Browns Ferry's SAMA analysis.

Watson said Entergy's Arkansas Nuclear One-2 and Dominion's Millstone had each expended 1,750 personhours on RAI responses, and TVA's Browns Ferry had expended 2,300 to 2,500 person-hours. This "seem like a lot of resources on just one item in the overall environmental report," Watson said.

NRC staff replied that RAIs remain necessary because applicants are still not providing sufficient information in their initial applications. As a result, staff "seem to see some of the same RAIs over and over again," said John Tappert of NRC's division of regulatory improvement programs. RAIs issued by staff "are very similar from plant to plant," agreed Bob Palla of NRC's division of systems safety and analysis.

"The industry has tried to respond to NRC's requests to address past RAIs" in subsequent license renewal applications, "but there appears to be an evolving list of new and more detailed RAIs," Watson said, adding that "a more standard approach to what information is needed would seem beneficial."

Industry RAI concerns

Watson identified probabilistic risk assessments (PRAs) of SAMAs as one area where "significant resources are being expended" by industry. "Many RAIs focus on the impact of uncertainties" in SAMA PRAs, Watson said. "However, significant conservatism is built into much of the SAMA analysis, which should adequately account for uncertainties. In many cases this conservatism overwhelms the uncertainties."

"The cumulative impact of uncertainties can be quite large, and we want to test the robustness of [PRA] conclusions," Palla responded. Often a proposed SAMA is "not a clear go or no-go" because there is "some fuzz in cost-benefit analyses," he said. "We need to be pretty comfortable you haven't missed something."

Mark Rubin of the division of systems safety and analysis said staff does not perform an exhaustive "PRA quality audit" during its SAMA reviews. Staff review is "more in the line of higher-level questions, similar to other risk-informed regulatory actions," because NRC is "trying to get a sense [that] there's sufficient basis for yes-no cost-benefit decisions on [implementing] SAMAs," Rubin said.

Watson identified the analysis of external events such as fires as another difficult area in SAMA reviews. "RAIs concerning external events methodology require extensive new research" and applicants are not sure "how to address the impact of external events when a detailed analysis does not exist," Watson said. Rubin replied that applicants might choose to perform an external events PRA because "bounding studies are less conducive to quantitative decision making." But "there is no cookbook" for external events analysis, and licensees need to make the case for their selected approach, Rubin said.

Industry also has concerns about cost-benefit analysis of SAMAs. "Many RAIs focus on significant new plant equipment, such as hardened vents, new feedwater pumps, [and] new electrical support," he said. When NRC does propose less expensive SAMAs, such as non-safety-grade pumps and emergency generators, NRC's cost estimates sometimes "are considerably lower than the applicants' estimates. This results in conflicting opinions about the cost-benefit of the SAMAs," Watson said in his presentation.

Palla said NRC staff doesn't look only at "gold-plated" alternatives. Sometimes less expensive options "will get the majority of the risk-reduction benefit," he said. However, "not every SAMA should be investigated for cheap alternatives," Rubin cautioned.

NEI to develop guidance document

NEI's Fred Emerson said his institute was willing to develop a guidance document to assist industry in preparing SAMA analyses. The document would draw on lessons learned from RAIs issued by NRC staff in previous license renewal applications. However, "licensees should have some confidence that staff won't be throwing a lot of new stuff at them if they use the guidance document," Emerson emphasized.

Palla said that SAMA analyses "will still need a review and there will still be questions, but a document articulating issues can only help." Richard Emch of NRC's division of regulatory improvement programs said that such a guidance document "must come out pretty quickly" if it is to be useful. NRC has approved license renewal for 26 units, and requests to extend the licenses of 18 other reactors are under review.

Emerson said NEI would try to supply NRC with a schedule and outline for developing the guidance document within a month. Dominion's Watson said industry and NRC should consider whether the guidance document ought to be incorporated into industry's guidelines for preparing license renewal applications, NEI 95-10.

—Steven Dolley, Washington

Navy admiral's name floated as possible NRC contender

Inside NRC
Volume 26 / Number 17 / August 23, 2004

Retired Rear Adm. William Jeremiah "Jerry" Holland is the latest name mentioned in political circles as a possible candidate for NRC commissioner.

The 71-year-old Iowa City native was commissioned as an ensign after graduating from the U.S. Naval Academy and climbed the ranks over his 32-year service in the Navy. Early in his military career he was assigned to the Atomic Energy Commission's (AEC) Schenectady Naval Reactor Operations Office & Naval Nuclear Power Training Unit in New York to learn about the technical aspects of nuclear propulsion plants. Later, he became a chief engineer for that Schenectady Naval Reactors Operations's unit in Windsor, Conn.

He spent several months in the AEC division of naval reactors in Washington, D.C. to receive instruction on nuclear power plant operations. He was deputy director of the National Military Command Center in the Office of Joint Chiefs of Staff, served as a submarine commander, and was deputy director of Space, Command and Control in the Navy's Office of the Chief Nuclear Officer before retiring in July 1987.

There has been talk about the White House being interested in making a possible recess appointment to fill one of two NRC commission vacancies.

But one Hill staffer said Bush administration officials have given "zero indication" that a nomination is imminent. "I don't think it's high on the priority list," the staffer said, adding that it is unlikely the administration would want to wage a battle that the recess appointment would engender. But a battle would also ensue if the White House sends up a Republican nominee for approval through the normal Senate confirmation process, given that Democrats, led by Sen. Harry Reid of Nevada, would also push for confirmation of Gregory Jaczko, a nominee that is anathema to the nuclear industry.—***Jenny Weil, Washington***

NRC to ask licensees for proof proper SG tube probe used

Inside NRC

Volume 26 / Number 17 / August 23, 2004

Questioning whether all steam generator tube inspection practices ensure compliance with NRC requirements, a final version of a generic letter (GL) the agency made available electronically last week would ask licensees for a description of their last tube inspections to ensure flaws weren't missed because the wrong probe was used. The staff notified the commission (Secy 04-141) that it would issue the letter this month. The staff said that the agency's Committee to Review Generic Requirements had endorsed the letter after the committee's comments were addressed.

The GL, which closely resembles an NRC notice in the May 14, 2003 Federal Register announcing the proposed generic communication, cited cases in which "tube inspections with a specialized probe near the top of the tubesheet clearly indicated the potential for circumferential cracks to occur deeper into the tubesheet, beyond the region inspected with the specialized probes. In each case the licensee was aware of the potential for such cracks to exist deeper into the tubesheet," but didn't inspect there with specialized probes because its safety analysis concluded such cracks didn't have safety implications, the GL stated.

"If licensees do not use probes capable of detecting flaws that may potentially be present, licensees would be allowing flaws to remain in service which may exceed the applicable TS [technical specification] acceptance criteria (i.e., tube repair or plugging limit)," staff stated. "Even when a probe is capable of finding flaws potentially present, flaws may be inadvertently missed for a variety of reasons (e.g., the flaw size is below the threshold of detection). However, missing a flaw is different from using a probe which is not capable of detecting the forms of degradation that may be present," the staff letter said.

The GL said licensees would have 60 days from the date of issuance to provide NRC with descriptions of the steam generator tube inspections performed at their plants. Licensees not using inspection methods capable of detecting specific flaws should provide the agency with an assessment of how their tube inspections meet the TS inspection requirements, it said.

The letter added that if licensees "conclude that full compliance with the TS in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B, requires corrective actions, they should discuss their proposed corrective actions (e.g., changing inspection practices consistent with the NRC's position or submitting a TS amendment request with the associated safety basis for limiting the inspections) to achieve full compliance." At a plant where the inspection doesn't comply, "the licensee should submit a safety assessment (i.e., a justification for continued operation based on maintaining tube structural and leakage integrity) that addresses any differences between the licensee's inspection practices and those called for by the NRC's position," the GL stated. "Safety assessments should be submitted for all areas of the tube required to be inspected by the TS, where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used, and should include the basis for not employing such inspection techniques." NRC posted the document Aug. 18 on its Web site (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2004/secy2004-0141/2004-0141scy.pdf), rejecting earlier public comments that industry's proposed generic changes to steam generator technical specifications obviated the need for NRC action. In a response document, NRC staff said it agreed that completion of the ongoing initiative with industry would upgrade the existing TS. However, it added, "there are potential safety implications if licensees' interpretations of the applicable regulatory requirements are different than the NRC's in that a condition (e.g., circumferential cracking) could exist at a given location such that SG [steam generator] tube structural or leakage integrity could be impaired, given conditions at a particular plant."

That response, as NRC sees it, is tied to the fact that the inspection probe that licensees typically use first—the highspeed bobbin probe—has not been found "to be capable of reliably detecting axial or circumferential flaws in the expanded region of tubing inside the tubesheet." To supplement those inspections, licensees use a slow-moving rotating pancake coil or +Point probe to reinspect locations where a bobbin cannot reliably detect certain degradation, such as circumferential cracks, axial cracks in low-row U bends, and expansion transition, the GL stated. However, the GL added that NRC staff in 2002 learned of several instances in which licensees failed to fully use inspection methods capable of detecting circumferential cracks.

Agency staff also rejected comments that claimed a GL would discourage further advances in inspection technology because utilities would be in violation of their TS each time a technology is improved if they aren't using the new version.

NRC countered that the proposed generic letter does not request a new technology be used but specifies that any technology or methods chosen by a licensee "shall have the objective of detecting flaws of any type that may be present along the length of the tube" to be inspected. In response to other comments, staff added it "did not intend for the GL to imply that the probability of detection...must be 100% for all flaws that meet or exceed the repair criterion."

The nuclear industry has been hoping that license amendments for Duke Power Co.'s Catawba-1 and -2 would pave the way for a fundamental reform of how NRC regulates steam generators (INRC, 17 May, 3). The amendments would codify industry's performance-based criteria for steam generators, serving as a template for other plants' license amendment applications for longer intervals between steam generator inspections, based on the tube's material and condition. ***—Elaine Hiruo, Washington***

Industry uncertain of benefits of proposed 50.46 revisions

Inside NRC
Volume 26 / Number 17 / August 23, 2004

NRC got few concrete responses from industry at an Aug. 17 public meeting on the costs and benefits of changing 10 CFR 50.46 along the lines suggested in a staff concept paper (INRC, 9 Aug., 1). The meeting left some industry representatives thinking that the staff is unlikely to be swayed by industry arguments before a proposed rule goes to the commission by the end of the year.

The rule changes would allow less rigorous analyses of loss-of-coolant accidents (LOCA) above a transition pipe break size; current requirements have licensees analyzing their emergency core cooling systems to respond to largebreak LOCAs using a double-ended guillotine break of the largest pipe in the reactor coolant system.

NRC's Brian Sheron, who chairs an internal NRC committee developing the 50.46 rule proposal, said that the ground rules for the changes were that changes would not need the development of new information or experiments and that the changes couldn't take a "radical departure" from current requirements. He said that any rule change ought to also have "consensus among the staff."

The transition break size (TBS) given in the staff's concept paper was 14 inches for PWRs and 20 inches for BWRs. Sheron, associate director for project licensing and technical analysis in the Office of Nuclear Reactor Regulation (NRR), said those sizes were picked "to provide margin for uncertainties." With those pipe break sizes, NRC would unlikely have to change the TBS in the future based on new information, and that "promotes stability in the regulatory process," Sheron said. If industry wants smaller break sizes, it will have to provide NRC with additional information, Sheron said. "We need a technical basis" to use smaller break sizes, he said.

The NRC staff is clearly sensitive to the importance that Chairman Nils Diaz places on this rule, as indicated by the large number of NRR senior managers who attended last week's meeting in the auditorium at NRC's Rockville, Md. headquarters. One source described this as "a legacy issue" for Diaz, who would very much like the rule to be final before his current term is up at the end of June 2006. Diaz has repeatedly called for a redefinition of the large-break LOCA.

What NRC was hoping to hear Aug. 17 was a discussion of how licensees might use a revised 50.46, including what design changes they might make to their plants, and what the costs and savings might be from implementing those changes. However, the only cost/savings numbers mentioned came from a 3-year-old estimate by the Westinghouse Owners Group, which was based on eliminating analyses of breaks in pipes larger than 5 or 6 inches. That study suggested one-time cost of perhaps $1-million per unit and benefits of $3-million a year.

General Electric's Rick Hill said that BWR owners would have to study the proposal further in other to estimate costs and benefits, and he acknowledged that BWRs might have to "squeeze a little harder to get the juice out of the fruit" of a revised 50.46.

But NRC did hear of some industry concerns with the staff's concept rule paper. Representatives of the Nuclear Energy Institute (NEI) said that any benefits of a rule change may be limited because the concept rule does not make changes to more of the General Design Criteria in Appendix A of 10 CFR Part 50. The staff only proposed changes to GDC 35; the industry suggested that changes to GDC 17 and 44 might also be necessary. Otherwise licensees would be "losing the benefits" of a proposed rule change, said NEI's John Butler.

NEI's Anthony Pietrangelo said that there is a "high potential" for a 50.46 rule change to not be very risk informed unless there is some "regulatory threshold" established. Not every change that a licensee wants to make under a revised rule needs to be submitted to NRC as a license amendment, but should be allowed under other the change mechanisms in other NRC regulations (e.g. 10 CFR 50.59), Pietrangelo said.

After the meeting, some industry representatives raised the possibility that a pilot licensee might come in with an exemption request, using a smaller break size and requesting other changes not contemplated in the staff's concept paper.

One of those sources noted that the other two riskinformed rule changes that NRC has come up with—50.44

(combustible gas control for nuclear reactors) and a new 50.69 (for special treatment requirements)—resulted only after licensees came in requesting exemptions from current regulations. At the very least, the effort to revise 50.46 "cries out for a pilot" to test the staff's proposed changes, said the industry source.—***Michael Knapik, Washington***

BWR Owners Group notes progress on EPU issues to skeptical NRC staff

Inside NRC
Volume 26 / Number 17 / August 23, 2004

Representatives of the BWR Owners Group (BWROG) gave a packed room of NRC staffers an update Aug. 18 on its efforts to get a better handle on potential equipment problems at plants that have gone to extended power uprates (EPU), meaning greater than 5%. But the responses of NRC staffers suggested that the owners group still has a ways to go in convincing the NRC that all the impacts of EPU operation on equipment are understood.

David Terao of the Office of Nuclear Reactor Regulation noted that the recent accident at Kansai Electric Power Co.'s Mihama-3, caused by a break in a secondary-side pipe, has added to public and agency concerns about whether current erosion/corrosion models are still valid under EPU conditions.

David Lochbaum, nuclear safety engineer at the Union of Concerned Scientists, also made the point at the meeting that NRC should be requiring EPU licensees to show how they still satisfying NRC concerns enunciated in past generic communications to the industry.

But the BWROG has taken a number of concrete actions that will result in issuance by the end of the year of a document with EPU lessons-learned and recommendations for strategies to improve the robustness of equipment performance during EPU operations.

A survey of 13 BWRs with EPU experience identified 17 component failures, BWROG told NRC, but many of those equipment performance problems also occurred under non-EPU conditions. But BWROG added that EPU operations does have the potential to decrease the time between failures. BWROG also looked at a power uprate and cycle extension database maintained by the Institute of Nuclear Power Operations. It told NRC that there were 103 BWR/PWR events in the database from January 1992 to January 2004. Of those events, 52 were identified as having power uprates directly or indirectly being a contributing factor. Fifteen of the events were caused by vibration, 18 were due to instruments calibration problems, 12 were due to operational procedural deficiencies, five were due to pre-existing conditions, installation errors, defective components, or some other miscellaneous reason, and two were due to erosion/corrosion.

BWROG told NRC that from its investigations to date the data suggest that the vibration problems at Exelon's Quad Cities "are an anomaly related to high steam velocities and unusually high acoustic vibration levels." BWROG also said that the refinement of BWR steam dryer load methodology would facilitate a "more realistic evaluation of steam dryer structural integrity prior to implementation of EPU." The NRC staff will travel to San Jose, Calif. this week to meet with General Electric (GE) and Vermont Yankee representatives at GE's facility to further discuss current BWR steam dryer structural analysis methodology.

At Quad Cities, Exelon is continuing to operate both units at pre-EPU power levels, and has said it won't return to EPU levels without NRC approval. The utility, with NRC's okay, did run Quad Cities-2 at EPU levels for several hours this month to collect more data.

An Exelon representative said there is no schedule at present for when both units would return to their licensed EPU power levels. (Both units were granted 17.8% power uprates in December 2001, allowing each unit to produce about 148 megawatts more of electricity.)

In May, Exelon promised NRC that it would replace the steam dryers at both Quad Cities units and that it was considering putting a measuring instrument in the new dryer at unit 1. At the Aug. 18 meeting, Exelon's Sharon Eldridge said the company was still "looking hard" at that option The next refueling outage for Quad Cities-1 is spring 2005; for unit 2 it is spring 2006. Exelon is hoping, however, that it may be able to demonstrate that modifications it has made to the steam dryer at unit 2 may allow EPU operation before the steam dryer is replaced.

Exelon and NRC are expected to meet sometime next month to discuss, among other things, the results of Exelon's re-evaluation of previous assessments of the impact of flow-induced vibration under EPU conditions on a number of key plant components.
—Michael Knapik, Washington

STP asks NRC to okay key initiative to increase maintenance flexibility

Inside NRC
Volume 26 / Number 17 / August 23, 2004

A key industry-supported risk-informed initiative designed to give utilities more flexibility in fixing inoperable equipment on line took another step forward earlier this month when STP Nuclear Operating Co. submitted a formal license amendment request to implement a risk-informed process for determining allowed outage times (AOT) for South Texas Project (STP) technical specifications.

To come up with a new AOT, STP is proposing that it use its configuration risk management program (CRMP), which is used in implementing the maintenance rule requirements in 10 CFR 50.65(a)(4).

But before the STP applciation is approved NRC will have to approve Nuclear Energy Institute/Electric Power Research Institute risk management technical specification (RMTS) guidelines. STP's CRMP has to then be judged to be in conformance with those guidelines. In July, NRC sent additional questions to the industry about its RMTS guidelines.

STP told NRC in its Aug. 2 submittal that its proposal was a pilot for one of the industry's main risk-informed technical specifications initiatives, commonly referred to as initiative 4B. The STP submittal is also one of five pilots for the NRC's evaluation of regulatory guide 1.200 on the scope and technical adequacy of utility probabilistic risk assessments (PRA). NRC staffers are expected to visit STP in November to review the adequacy of STP's PRA.
STP said that under its submittal, a new technical specification would introduce the concept of "overall plant configuration management." STP said that if approved, allowable action times would be replaced for affected technical specifications with an action requirement for the overall plant configuration based on the CRMP.

STP said that a "backstop AOT limit of 30 days" would be included to prevent excessively long allowable outage times based on risk analysis. STP said that all of the components within the scope of the proposed change are modeled in STP's PRA such that revised AOTs can be calculated.

Approval of the changes, STP said in its submittal, would allow the plant to concentrate efforts to maintain components while keeping a low overall risk profile and thereby "reducing the likelihood of plant transients." STP said that maintenance actions can be prioritized based on how to most effectively limit or reduce risk due the to the specific plant configuration at a specific time.

Industry also said it believes that NRC staff should like this initiative because it would reduce the need for plants to ask NRC for enforcement discretion if they have to exceed certain AOTs.—***Michael Knapik, Washington***

Nuclear News Flashes

U.S. NEWS:

--SECURITY GUARDS AT NUCLEAR PLANTS IN NEW YORK NOW HAVE AUTHORITY TO USE deadly force to protect the physical facility or defend against theft, trespass, or arson. The legislative measure, co-sponsored by Assemblywoman Sandra Galef (D), whose constituents live near Indian Point, and state Sen. Jim Wright (R), took effect this week. Galef told Platts the statute was needed to close a loophole in state law that prohibited private security companies from using deadly force or granting guards arrest powers. Separately, the legislature approved $450,000 in the budget to buy three marine patrol boats. Two will be used to guard the Ginna station, which is on Ontario Lake in upstate New York, and one will be used to keep watch on Indian Point along the Hudson River. Galef said the boat patrol would be staffed by the state naval militia or the National Guard.
NUCLEAR NEWS FLASHES - Friday, September 3, 2004

--NRC HAS RELEASED A STAFF PAPER ON PWR SUMP SAFETY. The paper (Secy-04-150, dated Aug. 16 and released today) details NRC plans to permit licensees to use alternate risk-informed approaches for resolution of the PWR sump blockage issue, known as GSI-191. The paper also includes the schedule for finalizing NRC's safety evaluation of the Nuclear Energy Institute's sump evaluation methodology. NRC staff said that a forthcoming generic letter on sump safety issues will request a description of and schedule for "all corrective actions, including any plant modifications that may be necessary." This is in contrast to some earlier drafts that requested information from licensees, but no corrective actions. John Hannon of NRC told Platts today that the generic letter is currently being reviewed by the commission and is scheduled to be issued by Sept. 9. Secy-04-150 is on NRC's Web site at http://www.nrc.gov/reading-rm/doc- collections/commission/secys/2004/secy2004-0150/2004-0150scy.pdf. **NUCLEAR NEWS FLASHES - Friday, September 3, 2004**

--CMS ENERGY WILL APPLY FOR A 20-YEAR LICENSE EXTENSION FOR PALISADES, the company announced late yesterday. The current license for the 845-MW PWR expires in 2011. The plant is owned by CMS subsidiary Consumers Energy and operated by Nuclear Management Co. In a statement, Consumers Energy senior vice president Robert Fenech said the company plans to file its renewal application with NRC in first quarter 2005. **NUCLEAR NEWS FLASHES - Friday, September 3, 2004**

--WESTINGHOUSE EXPECTS TO RECEIVE NRC CERTIFICATION FOR ITS AP1000 ADVANCED reactor design this month, Westinghouse spokesman Vaughn Gilbert said today. The passive, 1,100-MW PWR already has stirred some market interest in China, where Westinghouse has had an active presence for years. Westinghouse only had cost estimates for a pair of reactors, which it put at $2.2- to $2.7- billion. **NUCLEAR NEWS FLASHES - Friday, September 3, 2004**

--NEI WILL NAME ADMIRAL FRANK "SKIP" BOWMAN AS ITS NEW PRESIDENT AND CEO. In a press release to be issued tomorrow, the Nuclear Energy Institute (NEI) said that Bowman will begin working Jan. 1 with Joe Colvin, the group's current president and CEO, for a brief transition period. Colvin announced last year that he would retire sometime in early 2005. Bowman is currently director of Naval Nuclear Propulsion and a deputy administrator for naval reactors in DOE's National Nuclear Security Administration. He will leave those posts at the end of the year. In the release, Bowman is quoted as saying that "nuclear energy plays and will continue to play an important role in our nation's energy future. We must take the necessary steps to maintain the high levels of safe and reliable operations at our current plants and ensure that these plants, as well as new reactors, are part of a diverse energy supply for our high-tech, electricity-driven economy."**NUCLEAR NEWS FLASHES - Tuesday, August 24, 2004**

--A SENIOR REACTOR OPERATOR AT PILGRIM FELL ASLEEP ON THE JOB recently during the early morning hours, NRC and plant operator Entergy Nuclear said today. The incident occurred June 29 but was only recently brought to the attention of the NRC, Region I spokesman Neil Sheehan said. The agency responded by conducting "enhanced" control room inspections, meaning more checks and at different hours, Sheehan said. Separately, Entergy management has stepped up observation of the control room on some shifts and also has hired an outside firm to investigate the situation, said plant spokesman Dave Tarantino. Entergy was told of the allegation by NRC last Thursday, said Tarantino, adding that the operator has been removed from his duties in the control room. It is not known how long the operator had been asleep, he said.

INTERNATIONAL:

--THE EC MAY PROPOSE SEPT. 8 NEW DIRECTIVES ON NUCLEAR SAFETY and radwaste management, Dominique Ristori, director for general affairs at the European Commission's (EC) Directorate General for Energy & Transport (DG TREN), told Platts. Initially, the proposal was scheduled to be considered in late August. The first "nuclear package" was flatly rejected in late June, after more than a year of examination, by the Council of Ministers of the European Union. The main reason was that the EC did not consult stakeholders before putting forward its proposals. A coalition of EU member states led by the U.K. and Germany was also strongly opposed to any binding legislation in the fields of nuclear safety and waste management.

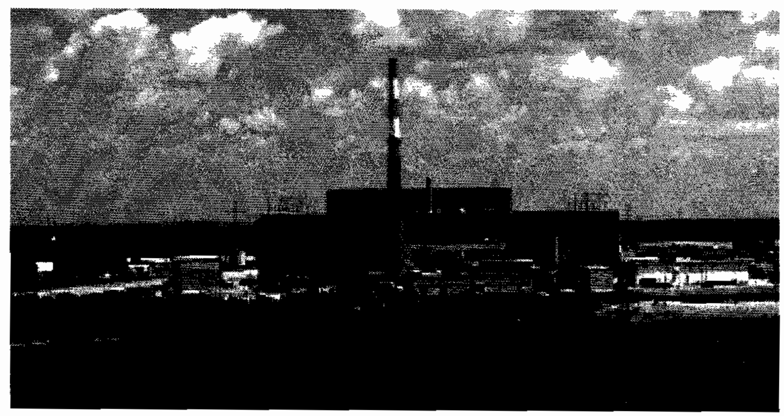
--DOE WILL HAVE ACCESS TO FRANCE'S PHENIX REACTOR UNDER AN AGREEMENT signed today by U.S. Energy Secretary Spencer Abraham and Alain Bugat, the chairman of France's Atomic Energy Commission. According to the DOE announcement, the agreement allows the two agencies to "test various types of fuel loaded with minor actinides under constant conditions "and "acquire data to permit selection of the best-performing fuel for future use in high-level waste transmuting systems." DOE said the agreement was important in part because the U.S. no longer has the capability represented by the Phenix, an experimental fast reactor.

Dresden Nuclear Power Station
Quad Cities Nuclear Power Station

Presentation to
Advisory Committee on Reactor Safeguards

Exelon Nuclear

September 9, 2004



Agenda

- Plant Description - Bohlke
- Recent Operating Experience - Bohlke
- Major Equipment Replacements & Repairs - Stachniak
- License Renewal Commitments – Polaski

Plant Description

- General Electric BWR-3 with Mark I containment
- Fresh water cooling
- Licensed power level 2957 MWth
- Current Dresden licenses expire in 2009, 2011
- Current Quad Cities licenses expire in 2012
- Extended Power Uprates completed in 2001, 2002
- Dresden Unit 1 is in SAFSTOR condition
  - A portion of the Unit 1 fire equipment supports Units 2 and 3 fire system and is in scope for license renewal

Recent Operating Experience

- All Reactor Oversight Performance Indicators for the four units are Green except for
  - Dresden Unit 3 HPCI Unavailability (White)
  - Dresden Unit 2 Unplanned Scrams (White)

Exelon Nuclear

Steam Dryer Replacement Plan

- New Quad Cities steam dryers planned for 2005
  - New design reduces stress concentrations, increases thickness, and transfers stress away from welds
  - The first dryer replaced will be instrumented to collect data

Exelon Nuclear

Steam Dryer Replacement Plan

- Exelon will conduct inspections of the new dryers during the subsequent refueling outage
- Pending the successful completion of the replacement plan, Exelon will not include the steam dryers within the scope of license renewal

Exelon Nuclear

Major Equipment Replacements

- Reactor water cleanup system piping replacement
- RHR service water system piping replacement (Quad Cities only)
- Reactor recirculation piping replacement (Dresden Unit 3 only)
- Main power transformer replacement
- Underground fire header replacement (Dresden only)
- Hydrogen water chemistry, zinc injection, and noble metals injection applied
- Core shroud repairs

Core Shroud Repair Hardware

- Shroud repairs installed in 1995-7 to structurally replace horizontal core shroud welds
- Repair hardware designed for 40-year life
- Materials included austenitic alloy XM-19 (tie rod), INCONEL X-750, and low carbon Type 316L stainless steel
- Materials were selected for resistance to IGSCC and IASCC
- Vertical shroud welds and shroud repair hardware are inspected per BWRVIP-76

Exelon Nuclear

Future Equipment Replacements/Refurbishments

- Main generator rewind
- Main condenser tube replacements
- Plant process computer upgrades
- LP turbine rotor replacements
- Large motor replacements
- I&C system upgrades to digital

Exelon
Nuclear

Commitment Management

- Exelon's commitment tracking system is controlled by a process consistent with NEI 99-04, Rev 1, "Guidelines for Managing NRC Commitment Changes" (endorsed by the NRC)
- Changes to a commitment require a formal review and evaluation

License Renewal Commitments

- Each Aging Management Program has a unique commitment tracking number
  - Implemented through procedures, work requests and surveillances
  - Aging effects, detection, and inspection criteria
- Implementing steps are annotated as license renewal commitments and are tracked on a station specific basis

Exelon Nuclear

Aging Management Program Implementation

- All procedures, work requests, and periodic surveillances that implement aging management programs will be in place by December 2004

- NRC Region III follow-up inspection of aging management programs concluded that program commitments were accurately tracked

Presentation Summary

ACRS License Renewal Full Committee

Dresden and Quad Cities
Nuclear Power Station
License Renewal Application

September 9, 2004

TJ Kim
Senior Project Manager

6

Overview

- Exelon submitted its application for Dresden and Quad Cities by letter dated January 3, 2003
- General Electric BWR/type 3 reactor, Mark I containment
  - generates 2957 megawatt thermal at both Dresden and Quad Cities
  - generates 912 and 795 megawatt electrical at Dresden and Quad Cities, respectively
- Location of Stations
  - Dresden is on the Illinois and Kankakee Rivers in Grundy County, Illinois.
  - Quad Cities is on the Mississippi River 3 miles north of Cordova, Rock Island County, Illinois.

Overview continued

- Current licenses expire
  - Dresden Unit 2 – December 22, 2009
  - Dresden Unit 3 – January 12, 2011
  - Quad Cities Units 1 & 2– December 14, 2012
- Request license renewal through
  - December 22, 2029 for Dresden Unit 2
  - January 12, 2031 for Dresden Unit 3
  - December 14, 2032 for Quad Cities Units 1 & 2
- Application implemented the generic aging lessons learned (GALL) process

NRC Audits and Inspections

- Scoping and Screening Methodology Audit
  - May 19-23, 2003
- Scoping and Screening Inspection
  - July 28 – August 1, 2003 (Exelon Headquarters)
- Aging Management Program Audit
  - October 7-8, 2003
- Aging Management Review Inspection
  - September 29 – October 3, 2003 (Dresden)
  - October 14-17, 2003 (Quad Cities)
- Optional Third Inspection
  - March 15-17, 2004
- Follow-up to Third Inspection
  - May 25, 2004

AMP Audit

- Date of audit – October 7-8, 2003
- Auditors - 4 Project managers from license renewal, 1 Regional inspector and 5 Contractors
- Concluded AMPS were consistent with GALL except:
  - Three AMPs were revised by making enhancements to the programs for review by the technical staff. The staff found them acceptable.
- AMP Audit Report issued April 23, 2004.

NRC Review Results

- 5 Open Items – all resolved
- 16 Confirmatory Items – all resolved
- Resolution of Open and Confirmatory Items brought into scope and subjected to AMR
  - Several new systems and components
- 4 new AMPs

Open Item

Scoping and Screening Methodology

- OI – 2.1-1
  - The staff identified that there was not sufficient basis for limiting consideration of fluid spray interactions to only those non-safety related SSCs located within 20 ft of an active safety related SSCs.
  - Resolution – The applicant eliminated the 20 ft exception and as a result expanded the license renewal boundaries of 17 plant systems and added 5 non-safety systems to the scope of the license renewal.

Steam Dryers/EPU

- **Steam dryers are generally not in scope for license renewal according to the rule.**

- **Resolution – The applicant has committed to a program plan that will identify the mechanism that has been causing unacceptable steam dryer loads and subsequent loose parts. This is being reviewed by the staff as a current operating reactor issue.**

- **Committed to 10 CFR 54.37(b)**

  After the renewed license is issued, the FSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with § 54.21. This FSAR update must describe how the effects of aging will be managed such that the intended function(s) in § 54.4(b) will be effectively maintained during the period of extended operation.

Open Item

ASME Section XI, Subsection IWF

OI-3.5.2.3.2-1

- The staff identified that the existing IWF program is not consistent with GALL in that it does not include the inspection of Class MC supports and piping supports.

- Resolution – The applicant has committed to perform IWF-2500 for MC supports.

- Resolution – The applicant has committed to perform the same type and quantity of inspections as required by IWF-2500. Structures Monitoring Program has been revised accordingly for MC piping supports.

Aging Management of In-Scope Inaccessible Concrete

	Aggressive Limit	Dresden	Quad Cities
pH	< 5.5	7-9	6.9 - 7.9
Chlorides	> 500 ppm	5 - 30 ppm	< 29 ppm
Sulfates	> 1500 ppm	10 - 30 ppm	< 24 ppm

- Periodic testing to verify chemistry remains non-aggressive
- Below grade soil/water environment non-aggressive

Reactor Vessel Upper Shelf Energy (USE)

Reactor Vessel Beltline Material	Screening Criteria USE (FT-LBS)	Staff Calculated USE (FT-LBS) Dresden		Staff Calculated USE (FT-LBS) Quad Cities	
		Unit 2	Unit 3	Unit 1	Unit 2
Limiting Beltline Plate Material	**≥ 50**	53	54	53	56
Limiting Weld	**≥ 35 (EMA)***	49	47	49	34**

- * EPRI Topical Report – 113596 demonstrated that welds with Charpy USE values of 35 ft-lbs can have margins of safety against fracture equivalent to those required by Appendix G, Section XI of the ASME Code.
- **Open Item Resolution – Applicant prepared a plant specific equivalent margin analysis (EMA) and demonstrated a minimum USE value of 32.4 ft-lbs. meets the criteria of Appendix K, Section XI of the ASME Code. Since 34 ft-lbs exceeds the minimum value, this weld meets the margins of safety against fracture equivalent to those required by Appendix G, Section XI of the ASME Code.

Reactor Vessel USE For Plates

Reactor Vessel Limiting Beltline Material	Capsule	Material	Capsule Neutron Fluence (N/cm^2)	% Drop in Capsule USE	1/4 T Neutron Fluence at EEOL (N/cm^2)	Projected % Drop at EEOL	USE at EEOL (Ft-Lbs)
Dresden Unit 2 Plate	3	A 302B	$1.3x10^{16}$	8	$3.9x10^{17}$	17.5	53
	8	A 302B	$5.2x10^{16}$	10		16	54
Dresden Unit 3 Plate	13	A 302B	$9.3x10^{15}$	4	$3.9x10^{17}$	11	57
	6	A 302B-M	$2.9x10^{16}$	6		15.5	54
	18	A 302B-M	$7.1x10^{16}$	7		11	57
Quad Cities Unit 1 Plate	G2	A 302B	$1.03x10^{16}$	7	$2.9x10^{17}$	16.5	53
	8	A 302B	$5.5x10^{16}$	10		15	54
Quad Cities Unit 2 Plate	13	A 302B	$1.69x10^{16}$	4	$2.9x10^{17}$	12	56
	18	A 302B-M	$6.6x10^{16}$	6		9	58

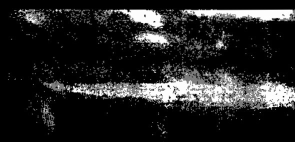
Reactor Vessel USE For Welds

Reactor Vessel Limiting Beltline Material	Capsule	Material	Capsule Neutron Fluence (N/cm^2)	% Drop in Capsule USE	1/4 T Neutron Fluence at EEOL (N/cm^2)	Projected % Drop at EEOL	USE at EEOL (Ft-Lbs)
Dresden Unit 2 Weld	3	ESW	1.3×10^{16}	7	3.9×10^{17}	18.5	49
	8	ESW	5.2×10^{16}	9		16	50
Dresden Unit 3 Weld	13	ESW	9.3×10^{15}	7	2.9×10^{17}	21.5	47
	6	ESW	2.9×10^{16}	9		16	50
	18	ESW	7.1×10^{16}	11		15	51
Quad Cities Unit 1 Weld	G2	ESW	1.03×10^{16}	5	2.9×10^{17}	18.5	49
	8	ESW	5.5×10^{16}	12		12	52
Quad Cities Unit 2 Weld	13	ESW	1.69×10^{16}	15	3.9×10^{17}	32	40
	18	ESW	6.6×10^{16}	28		43	34

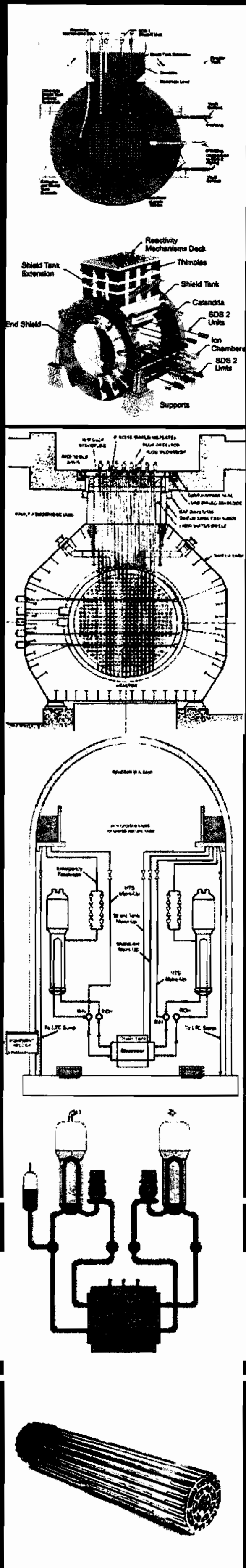
Staff Conclusions

- Actions have been identified and have been or will be taken such that there is reasonable assurance that activities will continue to be conducted in the renewal term in accordance with the current licensing basis as stated in 10 CFR Part 54.
- The applicable requirements of 10 CFR Part 51 have been satisfied.

4



ACR-700



Prepared for the Advisory Committee on Reactor Safeguards (ACRS)

U.S. Nuclear Regulatory Commission
Washington, DC 20555

June 30, 2004

Disclaimer

This report was prepared as background information for the ACRS internal use only, and does not represent or reflect the views of the ACRS.

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ACR-700

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June 30, 2004

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Table of Content

PURPOSE

The purpose of this report is to present the Advisory Committee on Reactor Safeguards (ACRS) with information and analysis that may be of interest and benefit to ACRS members in the context of their review of Advanced Candu Reactor, ACR-700.

Specifically; this paper touches upon three (3) key topics:

1. Chapter I presents the status of the ongoing U.S. Nuclear Regulatory Commission (NRC) pre-application review of the ACR-700 reactor. Specifically, this chapter identifies key safety, regulatory, licensing, and technology-related issues that are being raised in that review through discussions on:

    a. The topics that are being focused on by NRC,
    b. The requests by NRC for additional information from Atomic Energy of Canada Limited (AECL), and
    c. The responses made by AECL.

2. Chapter II presents a discussion of potential engineering and safety issues that ACRS may want to consider (consistent with NRC's key focus areas). The topics addressed are diverse and include:

    a. Class I pressure boundary
    b. Thermal hydraulics performance
    c. Process of on-power fueling
    d. ACR PRA methodology
    e. Severe accident
    f. Negative coolant void coefficient
    g. PIRT process
    h. Fuel design

3. Chapter III explains the Canadian review and approval process and ~~to~~ highlights differences between it and the U.S. process. Specifically;

    a. AECL, the Canadian government-owned nuclear laboratory and reactor developer/designer, has designed the ACR-700, a variant on the CANDU pressure tube reactor that uses slightly enriched (2%) uranium fuel and light-water coolant. As a preliminary step to marketing this reactor, AECL has applied to the Canadian Nuclear Safety Commission (CNSC) to perform a pre-licensing review of the design, while at the same time asking NRC to perform a pre-application review that, if positive, would likely result in a request for a Standard Design Certificate.

    b. The review objective, scope, and schedule of the two reviews are similar. The two regulatory agencies have the same safety objective and a similar basic approach to safety, and the intent of the regulatory requirements in the two countries is similar.

c. The two agencies have agreed to cooperate and collaborate on the review, especially in the areas of common safety concerns, regulatory and technical information exchange, and confirmatory research.

Chapter I. Status of NRC Pre-Application Review of ACR-700

I.1 INTRODUCTION

The purpose of this chapter is to present the status of the NRC pre-application review of the ACR-700 reactor. Specifically, this paper identifies key safety, regulatory, licensing, and technology-related issues that are being raised in that review, including:

- The topics that are being focused on by NRC,
- The requests by NRC for additional information from AECL, and
- The responses made by AECL.

The questions raised by NRC and the responses from AECL thus far are ~~very~~ extensive (several hundred pages) and frequently involve proprietary information. Rather than reproduce the questions and non-proprietary responses here, this paper correlates 'focus issues' with citations to sets of questions and answers (see Table 4 below) so that the reader may access relevant information on a particular issue through the use of the supplied ADAMS ML numbers. This section ends with the current schedule for NRC completion of the pre-application review and the expected contents of the Pre-Application Safety Assessment Report (PASAR) for the ACR-700 reactor design.

I.2 BACKGROUND

In the mid-1990s, the NRC docketed the CANDU-3 reactor for the design certification process. This project was terminated at AECL's request in 1995, due to the unfavorable near-term market at that time for new nuclear plants in the US. Since that time, further CANDU-specific research and development has been performed, formal validation of the computer codes used in the design and safety analysis of CANDU reactors has been completed, and many safety improvements have been made to the ACR design, which AECL stated in September 2002 will address most, if not all, of the issues raised by the NRC staff during the review of the CANDU-3 design.

On June 19, 2002, AECL asked NRC to perform a pre-application review of the AECL ACR-700 reactor design pursuant to NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants." On September 26, 2002, AECL issued a Plan for the NRC review, proposing that NRC conduct the pre-application review in two phases. During Phase 1, AECL expected NRC to familiarize itself with the ACR design and the scope of the available and planned analysis, testing, and operational experience in support of the design. In Phase 2, AECL expected NRC to perform an assessment of the technology base for the design, identify any major technical issues, and provide an estimate of the resources and schedule required for Design Certification. It was expected that both phases would be completed by July 2004.

I.3 PHASE 1

The September 2002 Plan proposed eight ACR-specific technical areas requiring early introduction and focused discussion with the NRC staff:

1. The design of the ACR RCS pressure boundary (i.e., the use of Zr-2.5wt% Nb pressure tubes, the unique fuel channel design, and the role of the fueling machines as components of a Class 1 pressure boundary).

2. The definition of design basis accidents and ACR safety acceptance criteria.

3. The computer codes used in ACR safety analyses.

4. The definition of severe accidents for the ACR and the nature and extent of research and development (R&D) support required.

5. The ACR treatment of safety-related systems, including seismic considerations.

6. The use of Canadian design codes and standards to address unique ACR features.

7. The use of distributed digital control systems and safety critical software.

8. The safeguards aspects of on-power refueling.

Through Phase 1 interactions between AECL and NRC over the following 18 months or so, those technical areas selected for focused review were modified and expanded. (See Table 1)

Table 1 Pre-Application Focus Topics

FOCUS TOPIC	TITLE	OBJECTIVE
1 [KEY]	Class 1 Pressure Boundary Design	The NRC staff accepts the principal design features of the ACR reactor cooling system (RCS) pressure boundary (i.e., the use of Zr-2.5wt%Nb pressure tubes, rolled joints, closure plugs, 403 **stainless steel** end fittings, and fueling machines as components of a Class 1 pressure boundary).
2	Design Basis Accidents and Acceptance Criteria	The NRC staff accepts the definition of ACR design basis accidents and the associated ACR safety acceptance criteria.
3 [KEY]	Computer Codes and Validation Adequacy	The NRC staff accepts the computer codes used in ACR safety analyses and the adequacy of their validation as sufficient for the purpose of providing a safety analysis for the ACR in the US.
4	Severe Accident Definition and Adequacy of Supporting R&D	The NRC staff accepts the definition of severe accidents for the ACR and considers the nature and extent of R&D support provided by the existing and planned R&D program to be sufficient to support the licensing of the ACR in the US.
5	Design Philosophy and Safety-Related Systems	The NRC staff accepts the ACR safety design philosophy and the ACR treatment of safety-related systems, including the approach to seismic considerations.
6	Canadian Design Codes and Standards	The NRC staff accepts the use of Canadian design codes and standards to address the CANDU-unique features of the ACR.
7	Distributed Control Systems and Safety Critical Software	The NRC staff accepts ACR distributed digital control systems and safety critical software.
8 [KEY]	On-Power Fueling (excl. Safeguards)	The NRC staff accepts the ACR CANFLEX fuel design and the process of on-power refueling.
9 [KEY]	Confirmation of Negative Void Reactivity	The NRC staff accepts that the ACR has a negative void reactivity.
10	Preparation for Standard Design Certification Docketing	The NRC staff has a good understanding of the safety aspects of the ACR and has identified any issues that could pose a risk to, or a delay in, licensing the ACR in the US.
11 [KEY]	ACR probabilistic risk assessment (PRA) Methodology	The NRC staff accepts AECL's PRA methodology as sufficient for the purpose of assessing the ACR for licensing in the US.
12	ACR Technology Base	The NRC staff finds the technology base for the ACR to be comprehensive and essentially complete.

FOCUS TOPIC	TITLE	OBJECTIVE
13 [KEY]	Fuel Design	The NRC staff accepts the ACR CANFLEX fuel design (NRC separated this from FT #8).

During Phase 1, AECL conducted 10 familiarization meetings, as indicated in Table 2. The ADAMS reference numbers for the charts used by AECL in those presentations are also tabulated in Table 2. Phase 1 of the pre-application review ended with those meetings.

Subject	Relevant to Focus Topic Numbers	Meeting Date	ADAMS ML References for Handouts
Design and technology base	All	Sept. 25-26, 2002	• ACR-700 Design - Introduction, ML022810586, 16pp. • ACR Overview, ML022810601, 48pp. • Reactor and Fuel Handling, ML022810621, 28pp. • ACR Safety Features, ML022810636, 37pp. • Core Design & Reactor Physics, ML022810624, 31pp. • HT Moderator & Auxiliary Systems, ML022810633, 38pp. • ACR Technology Base RCS Thermal Hydraulics, ML022810646, 27pp. • ACR Moderator Circulation, ML022810649, 17pp. • Developing Technical Insights on ACR-700 ML022810657, 12pp. • ACR Technology Base: Fuel Channel Thermal Hydraulics, ML022810647, 23pp. • ACR Technology Base: Reactor Physics, ML022810639, 22pp. • Qualification Process for Safety Analysis Computer Codes, ML022810656, 24pp. • Technology Base for the ACR: Fuel Channels, ML022810644, 52pp. • ACR Fuel Technology Base, ML022810641, 27pp. • ACR Technical Base: Containment, ML022810653, 33pp. • ACR Technology Base, ML022810617, 59pp.

Subject	Relevant to Focus Topic Numbers	Meeting Date	ADAMS ML References for Handouts
Physics, Fuel Channels, and QA	8 and 1	Dec. 4-5, 2002	• Introduction to the AECL Meeting with the NRC and the CNSC on Fuel Channels, ML030020121, 7pp. • Technology Base for the ACR Fuel Channels, ML030020225, 11pp. • Anticipated ACR Fuel Channel R&D Program, ML030020297, 16pp. • ACR Fuel Channel Fitness for Service, ML030020229, 9pp. • Pressure Tube to End Fitting Rolled Joints, ML030020213, 16pp. • Corrosion and Hydrogen Ingress of Pressure Tubes, ML030020286, 19pp. • Radiation Damage and Deformation, ML030020259, 20pp. • Preliminary Assessment of Physics Toolset for ACR Applications, ML030020094, 20pp. • CANDU Fuel Channel Inspection, ML030020217, 33pp. • Pressure Boundary Codes and Standards Applicable to CANDU Fuel Channels, ML030020140, 16pp. • ACR Physics Tests in ZED-2, ML030020114, 28pp. • Fracture Behavior of Pressure Tubes, ML030020241, 19pp. • Evolution of ACR Physics from CANDU-6, ML030020092, 36pp. • ACR Fuel Channel Design, ML030020132, 43pp. • Reactor- Physics Analysis Basis for Current CANDU, ML030020086, 94pp.
Thermal Hydraulics	12 and 3	Feb. 5-6, 2003	• Part 1 of 2 - ACR Thermal Hydraulics, ML030800377, 60pp. • Part 2 of 2 - ACR Thermal Hydraulic, ML030800378, 53pp. • Part 1 of 4 - List of Open Literature Papers Provided to the NRC Staff at the Meeting on ACR Thermal Hydraulics, ML030800310, 184pp. • Part 2 of 4 - List of Open Literature Papers Provided to the NRC Staff at the Meeting on ACR Thermal Hydraulics, ML030800357, 157pp. • Part 3 of 4 - List of Open Literature Papers Provided to the NRC Staff at the Meeting on ACR Thermal Hydraulics, ML030800372, 195pp. • Part 4 of 4 - List of Open Literature Papers Provided to the NRC Staff at the Meeting on ACR Thermal Hydraulics, ML030800354, 210pp.

Subject	Relevant to Focus Topic Numbers	Meeting Date	ADAMS ML References for Handouts
ACR Safety Design Philosophy, Design Basis Accidents, and Acceptance Criteria	2 and 5	March 27, 2003	• Part 1 of 5, Safety Design Philosophy, including Design Basis Accidents and Acceptance Criteria, ML030870719, 31pp. • Part 2 of 5, Safety Design Philosophy, including Design Basis Accidents and Acceptance Criteria, ML030870723, 48pp. • Part 3 of 5, Safety Design Philosophy, including Design Basis Accidents and Acceptance Criteria, ML030870725, 36pp. • Part 4 of 5, Safety Design Philosophy, including Design Basis Accidents and Acceptance Criteria, ML030870729, 17pp. • Part 5 of 5, 03/27/2003 Meeting Handouts re AECL Technical Presentation on ACR-700 Safety Design Philosophy, including Design Basis Accidents and Acceptance Criteria, ML030870733, 20pp.
SA Methodology and Computer Codes	3 and 2	May 15-16, 2003	• ACR Safety Analysis Methodology and Computer Codes Overview, Meeting Agenda, ML031420424, 55pp. • Summary of Meeting with AECL re: ACR-700 Safety Analysis Methodology and Computer Codes, ML031540528, 9pp.
ACR Severe Accidents and R&D	4 and 12	May 6-7, 2003	• Severe Core Damage Accidents & MAAP4 CANDU, ML031340418, 65pp. • Iodine Behavior in Containment, ML031340410, 72pp. • Severe Flow Blockage, ML031340415, 51pp. • Core Disassembly, ML031340417, 60pp. • Hydrogen Behavior in Containment, ML031340411, 61pp. • Fission - Product Release & Transport in RCS, ML031340408, 60pp. • Sequences of ACR Limited & Severe Core Damage Accidents, ML031340402, 42pp. • Fuel Channel Behavior, ML031340404, 5p6p. • PRA Applications in Canada, ML031340422, 10pp. • External Publications on Iodine, ML031340650, 5pp. • CANDU Fuel Behavior in Limited & Severe Core Damage Accidents, ML031340406, 73pp. • 05/06 - 08/ 2003, Summary of Meeting with AECL re: Limited and Severe Core Damage Accidents and PRA, ML031410880, 11pp.

Subject	Relevant to Focus Topic Numbers	Meeting Date	ADAMS ML References for Handouts
Details of RD-14M Results (in WRL)	3 and 12	June 4-5,2003	• RD-14M Experiments, ML031690504, 56pp. • RD-14M Facility Description, ML031690499, 80pp. • RD-14M Facility Scaling, ML031690498, 37pp. • Use of RD-14/14M Data in Cathena Validation, ML031690496, 18pp. • Hydrogen Experimental Facilities at Whiteshell Labs, ML031690508, 12pp. • RD-14M Quality Assurance Program, ML031690506, 11pp.
PRA Methodology Applied in ACR	11	May 8, 2003	• Level 1 PRA, ML031340426, 45pp. • Level 2 PRA, ML031340427, 9pp.
ACR CANFLEX Fuel Design	13	Sept. 4, 2003	• Introduction to CANDU Fuel, ML032530144, 40pp. • ACR fuel, ML032530141, 5pp. • Experience Base for ACR Fuel, ML032530153, 66pp. • ACR Fuel Design, ML032530149, 15pp. • CANDU Fuel Design and Performance Codes, ML032530148, 33pp. • ACR Fuel Qualification, ML032530157, 30pp. • 09/03-09/04/2003 - Summary of Meeting for ACR-700 Fuel Design and On-Power Fueling Technology, ML032540081, 8pp.
On-Power Fueling	8	Sept.3, 2003	• On Power Fueling Technology Part 1: ACR, ML032530120, 48pp. • On Power Fueling Technology Part 2: Current CANDU Design, ML032530126, 46pp.
SDS Design (incl. Safety Critical software)	5, 7, 10	Cancelled	
ECCS and Containment Design	5 and 10	Cancelled	
ACR Design	All	Cancelled	

I.4 PHASE 2

In February 2004, NRC agreed with AECL's proposed approach to completing the Phase 2 pre-application review of ACR-700, including the focus areas and schedule for meeting the July 2004 completion date, which happens to be in the same timeframe as the proposed ACRS visit to Canada. The NRC accepted Table 3 (originally contained in AECL's July 30, 2003, letter) as the scope for Phase 2 of the ACR pre-application review.

Table 3 Submissions Scope and Schedule for Phase 2 of ACR Pre-Application Review

Report	Submission Dates as of Jan. 2004	Related to Focus Topic
Safety Basis for ACR	July 31, 2003	# 2
ACR PRA Methodology and Scope	July 31, 2003	# 11
Safety Analysis Computer Code Qualification Status and Plans	August 05, 2003	# 12
Safety Analysis, Initial Conditions and Standard Assumptions	August 13, 2003	# 2
Technology of Fuel Channels	August 13, 2003	# 1- Key Focus Topic
Technology of On-Power Fueling	September 12, 2003	# 8 - Key Focus Topic
ACR Severe Accident Progression	September 12, 2003	# 4
Safety Analysis Basis reports:		# 2
- Trip coverage	September 30, 2003	
- Fuel and fuel channels	September 30, 2003	
- Thermal hydraulics (including physics)	January 15, 2004	
- Containment	November 25, 2003	
ACR Anticipatory R&D	September 30, 2003	# 12
Severe Accident Assessment and Mitigation	January 30, 2004	# 4
ACR Design Assist PRA Results	***March 01, 2004(T)***	# 11
Severe Accidents R&D Program	November 14, 2003	# 4
ACR Design Codes and Standards	***March 01, 2004(T)***	# 6
Safety Design Guides	***March 31, 2004(T)***	# 5
CANFLEX Fuel Design for ACR	January 23, 2004	# 13 - Key Focus Topic
R&D Status Report	January 31, 2004	# 12
Report on Safety Analysis Code Validation Methodology (compared to DG-1120)	***March 31, 2004(T)***	# 3 - Key Focus Topic

(T) = Target Date

In addition, given limited NRC resources, AECL requested that priority be given to the following focus topics during Phase 2 of the pre-application:

- Focus Topic #1 - Class 1 pressure boundary design
- Focus Topic #3 - Computer codes and validation adequacy
- Focus Topics #8 and #13 - On-power fueling and fuel design
- Focus Topic #9 - Confirmation of negative void reactivity

The NRC staff agreed that **these focus topics are key areas** in the ACR pre-application review. The high temperature material issues associated with the Class 1 pressure boundary design review are of particular interest to the staff and have been highlighted as a priority area within the Focus Topic #1 pre-application review. The staff is currently planning to provide feedback to AECL on all planned submittals for pre-application review. If resources are limited during the pre-application review, priority will be given to reports associated with key focus topics as identified above and **the following additional focus topics**:

- Focus Topic #4 - Severe accident definition and adequacy of supporting R&D
- Focus Topic #11 - ACR PRA methodology
- Focus Topic #6 - Canadian codes and standards
- Focus Topic #7 - Distributed control systems and safety critical software
- Focus Topic #2 - Design basis accidents and acceptance criteria

The staff's review of the Canadian codes and standards, Focus Topic #6, is focused on the quality assurance (QA) area. The design codes and standards portion of this review extend to the codes and standards associated with the review of the Class 1 pressure boundary design, Focus Topic #1.

In a February 4, 2004, letter to AECL, NRC stated that the planned submission date for several reports supporting key focus topics, as given in Table 3, did not allow sufficient time for the staff to complete a detailed safety review by the requested date of July 30, 2004. A period of six months is required for the staff to review each report and, based on the latest estimate, **the completion date for the staff review will be September 30, 2004, subject to AECL meeting ~~their~~its planned Phase 2 submission schedule.**

AECL has submitted several hundred documents related to the familiarization phase (Phase 1) and the focus topics of Phase 2 of the Pre-Application Review. In the course of its review of those submittals, the NRC staff has generated almost 300 questions (many with subquestions). They have been sent to AECL in letters called Requests for Additional Information (RAIs). The citations for those letters are contained in Appendix A. Table 4 identifies those Focus Topics for which RAIs have been sent to AECL, as well as those RAIs to which AECL has responded as of mid-May 2004.

Table 4 Requests for Additional Information (RAIs)

FOCUS TOPIC	TITLE	RAIs Letter Number (See APPENDIX A)	AECL RESPONSE SUBMITTED (See APPENDIX A)
1 [KEY]	Class 1 Pressure Boundary Design	5	YES (some proprietary)
2	Design Basis Accidents and Acceptance Criteria	6 (CATHENA)	YES (some proprietary); NO, more information to be supplied at Certification step
		7 (Event Categorization) 10 (TH, PIRT)	NO YES
3[KEY]	Computer Codes and Validation Adequacy	2	YES (PROPRIETARY)
4	Severe Accident Definition and Adequacy of Supporting R&D	10 (PIRT)	YES
5	Design Philosophy and Safety-Related Systems	None	
6	Canadian Design Codes and Standards	9 (QA)	NO
7	Distributed Control Systems and Safety Critical Software	None	
8[KEY]	On-Power Fueling (excl. Safeguards)	5 8	YES (some proprietary) YES (PROPRIETARY)
9[KEY]	Confirmation of Negative Void Reactivity	1 10 (PIRT)	YES (PROPRIETARY) NO
10	Preparation for Standard Design Certification Docketing	None	
11[KEY]	ACR PRA Methodology	3, 4	YES
12	ACR Technology Base	1 2	YES (PROPRIETARY) YES (PROPRIETARY)
13[KEY}	Fuel Design	5	YES (some proprietary)
PIRT	Multiple Topics	10	On-going

I.5 PRE-APPLICATION SAFETY ASSESSMENT REPORT

The NRC's pre-application review for the ACR design includes a review of the reports submitted during Phase 1 and Phase 2, resources permitting. The staff's review will result in the development of a Pre-Application Safety Assessment Report (PASAR) for the ACR design. **The planned issue date for the PASAR is September 30, 2004,** or six months after the last Phase 2 report is received from AECL. The six-month review period includes concurrence by the Office of General Counsel (OGC) and a presentation to the ACRS scheduled for September 2004.

For each key focus topic and additional focus topic reviewed, the PASAR will include the following sections:

1. Review Scope: Discussion of what reports were reviewed and what guidance they were reviewed against, to the extent that the guidance exists.

2. Technical Issues: Discussion of technical issues that will require further data, tests, inspections, analyses, or codes.

3. Regulatory Issues: Discussion of regulatory issues, such as rules, rulemaking, or exemptions, that will need to be resolved.

4. Policy Issues: Discussion of policy issues that will need Commission guidance for resolution.

5. Conclusion: Discussion of the feasibility of design certification and the impacts of the issues evaluated.

6. Schedule and Resources: An estimate of the resources required and schedule for completing the review of each specific focus topic area will be provided.

The PASAR will identify licensing issues associated with the pre-application focus topics that must be resolved in order to obtain a design certification for the ACR-700.

Chapter II. Major Issues Within Key Focus Topics

ISSUE 1

Class 1 pressure boundary. What is the effect of the ACR-700 environment on component fatigue and creep life? What are the effects of irradiation damage, aging, and embrittlement? What is the performance of the large number of dissimilar metal welds in the header system? What is the component material behavior under severe accident conditions? How much research will be necessary to satisfy NRC requirements?

BACKGROUND

- This is AECL/NRC Focus Topic #1 [Class 1 Pressure Boundary Design]
- Piping, valves, pressure vessels - designed to ASME
- Feeder pipes - multiple, small diameter, pipes from headers to fuel channels - also designed to ASME
- Fuel Channel - designed to Canadian Standards
  - Designed to meet intent of ASME with accommodation for pressure tube and refueling requirements
  - Material exceptions
    - Zr-2.5%Nb pressure tube
    - Modified 403 SS end fitting
  - Design differences
    - Rolled joint between pressure tube and end fitting
    - Channel closure for refueling
- In terms of experience with pressure tube integrity, there have been **no pressure tube leaks** due to design or material performance since 1986.
- There are areas where you cannot inspect. AECL specifically designed the spaghetti of tubing to enable inspection of the critical piping.
- NRC staff identified the following issues:
  - Basis for fatigue design curves
  - Basis for governing creep equations
  - Sagging of pressure tubes and hydride blister formation
  - Effect of large number of bent pipes >> erosion corrosion, SCC
  - Effect of irradiation damage, aging, and embrittlement
  - Effect of dissimilar metal contacts in typical ACR-700 environment
  - Design of rolled joints
  - Canadian design and inspection codes
  - Code classification of components
  - Inspectability of components
  - Scope, methods, and frequency of inspection
  - Testability of components
  - Scope, methods, and frequency of testing

- Leak-before-break approach and adequacy of leak detection capability
- On-power fueling as an extension of the Class 1 pressure boundary
- Design of transport mechanisms in Class 1 component support structure
- Component material behavior under severe accident conditions

ISSUE 2

Thermal-hydraulic performance of the ACR-700 reactor is analyzed with the CATHENA Code that has been scaled and validated through the RD-14M Facility. If this program and approach are deemed to be inadequate, then it may be necessary to construct a new facility or make major modifications to existing facilities to generate substantially more data. Either event may be a serious impediment to AECL pursuing design certification of the ACR-700 reactor in the United States. What more must be done, if anything, to satisfy the ACRS on the use of the CATHENA code for thermal-hydraulic analyses of the ACR-700 reactor? What is the **potential need for any additional integral and/or separate effects testing?**

BACKGROUND

This is a subpart of AECL/NRC Focus Topic #3 [Computer Codes and Validation Adequacy]

Use of RD-14/14M Data in CATHENA Validation (ML031690496)

- X Description of CATHENA:
  - o Canadian Algorithm for Thermal-hydraulic Network Analysis
  - o One-dimensional, two-fluid system thermal hydraulics code
  - o Developed by AECL primarily for analysis of postulated loss-of-coolant accident (LOCA) events in CANDU reactors
  - o One-dimensional, two-fluid system thermal hydraulics code
  - o Developed by AECL for thermal hydraulics analysis of RCSs
- X Sources of Validation Data
  - o Analytical solutions to idealized problems
  - o Separate effect experiments
  - o Isolate behavior of a single phenomenon
  - o May be of Canadian or international origin
  - o Component tests
    - ▪ Investigate one or more phenomena in a reactor-specific geometry or assembly
  - o Integrated tests
    - ▪ Investigate interacting phenomena in inter-connected components relevant to reactor geometry
    - ▪ Includes RD-12, RD-14, RD-14M, and in-reactor tests

RD-14M Facility Scaling (ML031690498 and ML031690499)

- X The RD-14 facility was a full-elevation model of a typical CANDU RCS. It was built to provide improved understanding of CANDU thermal hydraulics and to expand databases to validate CANDU analysis codes.
- X RD-14 was modified to a multiple channel geometry. RD-14M is a figure-eight loop possessing many of the physical and geometrical characteristics of a CANDU.
- X Design Features of RD-14M
  - o Full elevation changes between major components and full linear dimensions (e.g., full-height steam generators and full-length feeders)
  - o Ten full-length electrically heated channels

- o Simulation of all RCS components: channels, end-fittings, feeders, headers, and steam generators
- o Simulation of all phases of a LOCA scenario, including break and emergency core cooling (ECC)
- o Natural circulation and shutdown/maintenance cooling simulation
- o Full pressure and temperature conditions
- o Extensively instrumented
- o Dedicated data acquisition system

X The RD-14 and RD-14M test facilities were designed to preserve *Dynamic Similarity* with the CANDU RCS based on a developed set of scaling criteria

X Where scaling criteria could not be applied, past experience and engineering judgment were used to provide a conservative component design

X Development of Scaling Criteria to Preserve Dynamic Similarity- 1

- o Approach to Ishii and Kataoka used to develop scaling criteria to obtain dynamic similarity
- o Governing thermal hydraulic equations written in dimensionless form (mass, energy, and momentum balances) using drift flux or homogeneous flow models, as required
- o Dynamic similarity achieved by adjusting facility design variables (pipe length, diameter, etc.) to match value of dimensionless groups for facility and reactor

X Development of Scaling Criteria to Preserve Dynamic Similarity - 2: Limitations of scaling criteria:

- o Scaling laws only apply if flow is 1-D, well mixed and void/quality relationship for homogeneous flow can be applied
- o If horizontal stratified, or horizontal/vertical annular flow occurs, departures from similarity between reactor and loop behavior will occur

ISSUE 3

Process of on-power fueling: Have all credible accidents been identified and have they been appropriately addressed in the design? Are the design standards for components of the fueling machine that are part of the pressure boundary acceptable? What is the operating experience pertaining to the on-power refueling system for operating CANDU reactors?

BACKGROUND

- This is AECL/NRC Focus Topic #8
- Outages do not need to be at a fixed time. Most of the CANDU reactors work on four days of fueling and three days for maintenance and other activities
- Each two-bundle shift replacement gives about .2 milli-K of increase in reactivity in the channel, and the physics staff selects the channel based on the overall core balance where they are taking about **20 months for fuel to pass through**
- This operation is controlled from the main control room
- The fueling machine would be connected to the ends of the fuel
- The refueling machines on each end become part of the RCS where the pressure is about 2,000 pounds per square inch
- Mispositioning **accidents** to do with the potential for the fueling machine to contact end fittings as it moves across the face
    - assume the fueling machine can back off from the reactor without closing the plug
    - assume severance of the inlet and outlet hoses, which actually provide cooling to the fuel while it is in the fuel bundle
    - accidents with the refueling machine off reactor where it loses cooling in the transfer from the reactor to the spent fuel port
- **Hoses are actually part of the pressure boundary** during on-power fueling. Those hoses are designed to the Canadian standards, CSA and 285.2
- The **areas of interest to NRC** include: criticality prevention, fuel cooling, residual heat removal, mechanical handling of the fuel, instrumentation control systems with regard to those interlocks and other devices associated with the on-power refueling machine; the extent to which emergency cooling is required and containment integrity is maintained during the fuel transfers

ISSUE 4

Does the ACR-700 reactor have a **negative coolant void coefficient** under all operating conditions? Can it be calculated accurately enough to have confidence that it is really negative? Can AECL further modify the fuel to make it more negative?

BACKGROUND

- X This is AECL/NRC Focus Topic #9 [Confirmation of Negative Void Reactivity]
- X ZED-2 (Zero Energy Deuterium reactor) data obtained to validate physics codes and associated nuclear data libraries used for design and licensing ACR
- X These experiments provide validation data related to lattice reactivity and **coolant void reactivity** (CVR)
- X Measurements are performed by systematically replacing reference fuel with test fuel while observing the resulting change in moderator critical height
- X Measurements will be performed using two test fuels, slightly enriched uranium (SEU) and mixed-oxide fuel (MOX) (representing fresh fuel and irradiated fuel) with the test fuel at two coolant conditions
- X The charts at ML030020114 list the experiments planned to address various physics phenomena identified in the Physics Code Qualification Plan for ACR
- X Experimental techniques to be employed are flux maps, substitution experiments, fine-structure experiments and kinetics experiments
- X Based on charts at ML040150698, ACR achieves a slightly negative CVR by manipulating upon voiding
    - o Changes in Spatial Flux Shape
    - o Changes in Neutron Spectrum
- X Confirmation of Negative CVR in ACR by:
    - o Comparisons of CVR calculated by AECL's computer codes (WIMS, RFSP, DRAGON) with other international codes such as MCNP, HELIOS, DONJON, NESTLE
    - o Experimental verification of negative CVR in AECL's ZED-2 Reactor at Chalk River Laboratories (CRL)
- X WIMS lattice simulations indicate CVR can be reduced by reducing the Moderator/Fuel ratio in the lattice cell
- X Design Target of Slightly Negative CVR requires reduction of lattice pitch (LP) from current value of 28.6 cm to 20 cm
- X Minimum LP = 22 cm required to provide space for feeders between channels
    - o Use larger calandria tube (CT) to displace more moderator
    - o Add Dy (7.5%) to central natural uranium pin
    - o Use 2.1% SEU fuel in remaining 42 fuel pins to achieve average fuel burnup of about 21 MWd/kgU
- X **Full core LOCA reactivity effect = - 7 mk**
- X Effect of Coolant Void in ACR
    - o ACR lattice is under-moderated with normal H2O coolant
    - o H2O acts as both coolant and moderator
    - o LOCA further reduces moderation from the lattice

- CVR is a combined effect due to loss of
- absorption (positive) and loss of moderation (negative) from H2O
- Major contributors to the negative CVR
  - Lattice Cell
  - Increase in Resonance Absorption (1 eV to 100 keV) in U238
  - Decrease in Fission (0.3 eV resonance) in Pu239
  - Increase in neutron absorption by Dy in the central pin upon voiding
  - Increase in Reactor Leakage

X Coolant void reactivity is the initial focus of the neutronics PIRT, and it should have been completed in the April 2004 time frame

ISSUE 5

ACR PRA Methodology: Will the probabilistic safety analysis (PSA) methodology produce a PSA with adequate scope, adequate level of detail, and technical acceptability? To what extent does AECL have industry/independent peer reviews of its PRAs? How does AECL **deal with uncertainties**, including **model uncertainties**?

BACKGROUND

- X This relates to AECL/NRC Focus Topic #11 [ACR PRA Methodology]
- X PRA Applications
  - o Design Assist Role:
    - Confirm adequacy of safety design
    - Assess redundancy & functional separation of mitigating system
    - Assess system interface & capability requirements
    - Assess potential design options for risk reduction
    - Recommend design changes based on cost/benefit assessment
  - o Provide input to Environmental Qualification program; identify equipment requiring protection against steam, radiation, pipe whip
  - o Risk Evaluation - Estimate severe core damage frequency
  - o PRA Role in Operations:
    - Provide input to test and maintenance programs, so that these can be optimized in terms of cost and safety
    - Identify maintenance restrictions
    - Outage planning
    - Risk impact of changes in plant configuration, test frequencies, on-line series/parallel equipment maintenance
    - Input to Technical Specifications (e.g., impairment levels for Special Safety Systems)
    - Identify safety critical components
  - o Develop understanding of integrated plant response to accidents
    - Identify operator actions, alarms, and annunciations and thus input to control center designs and Emergency Operating Procedures for accident mitigation
    - Licensing role
    - Establish a comprehensive list of initiating events for safety analysis
    - Risk-informed regulation
    - Ranking of safety critical systems
    - Assessment of containment performance for severe core damage accidents
    - Assessment of severe accident mitigation design accidents (SAMDA)
- X Canadian PRA Targets
  - o ACR summed severe core damage frequency will be less than 1E-05/yr
  - o ACR summed large release frequency target will be less than 1E-06/yr
  - o Seismic margin target of the plant HCLPF is 0.5g based on a 0.3g Design Basis Earthquake

- X Staff requirements memorandum on SECY 90-16 specifies a core damage goal of ten to the minus four per year for evolutionary and advanced reactor designs. AECL has a target of ten to the minus five per year
- X In its PRA methodology, AECL defines ten **plant damage states** that, with one exception, mapped to either limited or severe core damage categories. Limited core damage accidents are where the progression of the accident has arrested within the single fuel channel, which is traditional light water reactor (LWR) accident progression. (Level 1 PRA, ML031340426)
  - o PDS0 - Failure to shut down
  - o PDS1- Late loss of core structural integrity with high RCS pressure
  - o PDS2 - Late loss of core structural integrity with low RCS pressure
  - o PDS3 - Loss of core cooling with moderator required early as sustained heat sink
  - o PDS4 - Loss of core cooling with moderator required late as sustained heat sink
  - o PDS5 - Loss of cooling/inadequate cooling following a LOCA with successful initiation of ECC
  - o PDS6 - Power cooling mismatch with late ECC injection due to channel failure
  - o PDS7- Power cooling mismatch in a single channel with containment overpressure
  - o PDS8 - Power cooling mismatch in a single channel with no containment overpressure
  - o PDS9 - Tritium release
  - o PDS10 - Fueling machine failures

ISSUE 6

- X **Fuel Design:** Does ACRS agree with the AECL position that multiple pressure tube failures are beyond the design basis? Are the CSA standards equivalent to ASME standards? What is the amount of **hydrogen produced** during severe accidents compared to the **capacity of the autocatalytic converters** and/or the strength of the containment? What are the significant corrosion mechanisms? What is the potential for **energetic fuel coolant interactions?**

BACKGROUND

- X This was separated by NRC from AECL Focus Topic #8
- X Subject to the Canadian standard "Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants," CSA Standard N285.2. This Standard is part of the N285 series, which provides uniform rules for the design, fabrication, installation, and inspection of CANDU nuclear power plant pressure-retaining systems, components, and supports. The rules complement those specified in CSA Standard N285.0 and the *ASME Boiler and Pressure Vessel Code.* This Standard, therefore, is intended to be used in conjunction with CSA Standard N285.0, and in the event of a conflict between the two Standards, the requirements of this Standard govern
- X The cladding thickness is much greater than that considered in Appendix K
- X **Limited Core Damage Accidents (LCDAs)**
  - o LCDAs are a class of accidents that are unique to the CANDU reactors due to their use of multiple, separated fuel channels, surrounded by a cool, low-pressure, heavy-water moderator, contained within a calandria vessel, rather than a LWR core contained within a reactor vessel. LCDAs are low-probability, single-channel events that involve consequential failure of a single pressure tube due to severe fuel overheating in the tube, or are accidents affecting the entire core that result in fission product release due to fuel overheating but do not result in consequential pressure tube failures.
  - o LCDAs represent a class of accidents that, in terms of their consequences, lie between design basis accidents (DBAs) and severe core damage accidents (SCDAs). These accidents generally should not be classified as DBAs because LCDAs have a lower probability of occurrence, some with very low probabilities in the severe accident range, and they should not be classified as SCDAs because LCDAs have a lower magnitude of fission product release from the fuel when compared with severe accidents in US light water reactors.
  - o The genesis for a separate category of accidents for the ACR-700 stems from the fundamental differences in the primary system between pressure vessel reactors and pressure tube reactors. In the ACR-700, each pressure tube is part of a fuel channel assembly. The pressure tubes separate the fuel bundles in one tube from the bundles in neighboring tubes. The materials separating the fuel bundles in any two channels are the two pressure tubes themselves, two gas-filled gaps, two calandria tubes, and the low-pressure, low-temperature, heavy-water moderator. Thus, a major design difference between the ACR-700 reactor and US LWRs is

the use of two metal pressure boundaries and a substantial heat sink of water (moderator) that separate each of the primary system flow paths that cool the fuel bundles. These physical barriers, geometries, and distances promote short-term and long-term cooling of the core and help prevent damage in a single channel from spreading across the core.

- The clad integrity criteria are (ML041540420):
  - No fuel center-line melting
  - No excessive diametric strain
  - No significant cracks in the surface oxide
  - No oxygen embrittlement
  - No clad failure because of beryllium-braze penetration at bearing pad or spacer pad locations
- Clad melting is another mechanism that would result in clad failure. Clad melting is only predicted by AECL for some fuel elements in the affected channel for a severe flow blockage or feeder stagnation break, both LCDAs. For these events, clad failure by one of the five mechanisms would occur before clad melting. Fuel in the other channels remains intact for these single-channel events. Clad melting is not predicted for DBAs or other LCDAs.

ISSUE 7

PIRT Process: Does ACRS have confidence in the conduct of the PIRT process for the ACR-700 pre-application review? What weight should be given to the results?

BACKGROUND

- X This has a major bearing on many contentious issues
- X PIRT is a Phenomena Identification and Ranking Table based on the relative importance of systems and components in an accident sequence
- X NRC is asking the PIRT panel to rank (high, medium, and low) issues of importance and the knowledge base of each
- X The **showstoppers** are a high-low, that is, where it is important phenomenologically, and the collective knowledge base is low. This would be very important.
- X NRC is using advisers that they think have the best knowledge, regardless of whether they have some involvement in the plan itself
- X **Areas subjected to the PIRT process are: neutronics, thermal hydraulics, and severe accidents**
- X See: "Summary of Information Prepared by AECL for the 3rd NRC PIRT TH Meeting." ML040560460, February 18, 2004

ISSUE 8

Severe Accidents: Has AECL adequately identified and addressed severe accident issues relevant to ACR-700 design? Would ACRS support either an exemption or exception to 10 CFR 50.34 (regarding fission product release assumed for the radiological consequences of accidents based on based on the NRC practice of assuming substantial meltdown of the core) during the design certification review? Does ACRS agree with the major reactor accident and its progression hypothesized by AECL for the purpose of the radiological consequence analysis for the ACR-700 design? Does ACRS accept the computer codes for fission product transport and aerosol behavior?

BACKGROUND

- X 10 CFR Section 50.34 clearly states that the fission product release assumed for the radiological consequences of accidents should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. However, AECL does not assume that the fuel melts in its radiological consequences analyses. Either an exemption or exception to 10 CFR 50.34 may be required during the design certification review. (04/06-04/07/04-Meeting Summary on the DBA and Severe Accidents Meeting with AECL, ML041140458)
- X Requirements to Address Severe Accidents: Severe core damage accidents (severe accidents) are low-probability events beyond the design basis accident established in 10 CFR 50.34 that can lead to significant core damage and subsequent release of fission products from the reactor. The NRC has requirements to initiating events which may lead to address severe accidents, such as anticipated transients without scram (10 CFR 50.62), station blackout (10 CFR 50.63), and the treatment of combustible gas control (10 CFR 50.44); however, a definitive set of regulatory requirements for addressing specific severe accident phenomena does not exist. Instead, the Commission has developed guidance and goals for resolving safety issues related to reactor accidents more severe than design basis accidents. Regulatory guidance pertaining to severe accidents can be found in Federal Register (50 FR 32138) dated August 8, 1985, "NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," and the corresponding Staff Requirements Memorandum (SRM) dated July 21, 1993. SECY-93-087 discusses the staff's technical and policy issues pertaining to evolutionary and advanced light-water reactor design certification and it includes severe accident preventative and mitigative feature issues.
- X NRC staff has stated (04/06-04/07/04-Meeting Summary on the DBA and Severe Accidents Meeting with AECL. ML041140458) that the fission product transport and aerosol behavior computer codes (SOURCE, SOPHAEROS, SMART) used by AECL need to be independently validated during the design certification review, since NRC staff has not used these codes.

- X Severe Accident Analysis Tasks
  - o Accidents are grouped into categories of similar potential for airborne radioactivity content within the plant and similar containment integrity challenges (4 Core Damage States)
  - o Core Damage State frequencies are summed
  - o Containment event tree analysis
  - o Deterministic analyses to enumerate the radioactivity source terms outside the containment for all combinations of Core Damage States and containment end states
  - o Derive a profile of source terms vs. frequency
  - o Enumerate large release frequency
  - o Severe Accident Analysis code - MAAP4 CANDU
- X Main Elements of Containment Performance Assessment
  - o Local Air Coolers
  - o Airlocks
  - o Containment Isolation
  - o Passive Autocatalytic Hydrogen Recombiners
  - o Bypass:
    - Steam generator tube rupture
    - Bleed cooler tube rupture
    - Interfacing LOCA

Chapter III. Canadian Approval Process for ACR-700

1. BACKGROUND

As a preliminary step to marketing the ACR-700, AECL has applied to the Canadian Nuclear Safety Commission (CNSC) to perform a pre-licensing review of the design, while at the same time, asking the NRC to perform a pre-application review that, if positive, would likely result in a request for a Standard Design Certificate.

The review objective, scope, and schedule of the two reviews are similar. The two regulatory agencies have the same safety objective and a similar basic approach to safety, and the intent of the regulatory requirements in the two countries is similar.

Therefore, the two agencies have agreed to co-operate and collaborate on the review, especially in the areas of common safety concerns, regulatory, and technical information exchange and confirmatory research.

This cooperation has so far consisted of the following:

- attendance at public meetings in each others jurisdiction
- review of common or mostly-common submissions from AECL on technical topics (some 80 documents submitted to date)
- exchange of views on ACR-700 issues
- joint representation at meetings of the Expert Panel hired by NRC to develop a Phenomena Identification and Ranking Table (PIRT) for selected ACR-700 accidents
- attendance of CNSC staff as observers at meetings of the ACRS
- discussions at executive levels
- planning of joint quality assurance audits

This chapter is designed to explain the Canadian review and approval process and to highlight differences between it and the U.S. process.

2. HISTORICAL PERSPECTIVE

The Canadian licensing process for nuclear power plants is substantially different from the U.S. process. The Canadian regulatory regime has relatively few documents relating to the design of a nuclear power plant. It has been developed over a period of 40 years using largely high-level criteria and lower-level practices adopted by the industry over the same period of time.

Except for a few major prescriptions, the CNSC's regulatory approach has been to lay out the goals that should be achieved and to allow the proponent to bring forward proposals on how to achieve them, rather than telling a vendor how a reactor should be designed. The Canadian system allows considerable negotiation between the proponent and the regulator, and in the end, leads to a compromise that satisfies the regulator's safety requirements. One of the reasons that this approach has been workable in Canada is that the country has licensed only reactors of the CANDU type, following common design principles, designed by only one vendor (AECL), and constructed by only three utilities - Ontario Hydro (now Ontario Power Generation, Inc.), Hydro Quebec, and the New Brunswick Electric Power Generation Company.

The reactor designer and vendor and the utilities that constructed and operated the power plants were all federal or provincial government agencies at the time the licensing process was developed, resulting in a different approach to the nuclear industry than in the U.S. Commercial competition in the Canadian nuclear industry exists only at the level of component suppliers, if at all. There was, therefore, no need to set ground rules for a level playing field in a commercial environment. Federal government financing funded the construction of the early generating stations. Of the 23 power reactors that have been licensed in Canada, 21 were constructed by a single utility, Ontario Hydro.

Historically, the former Atomic Energy Control Board (AECB), created in 1954, met behind closed doors and did not allow public access to its documents. The operation of the regulator gradually became more open to the public, and in 1997, a new Nuclear Safety and Control Act was proclaimed, establishing the legal basis for a regulatory agency in the full modern sense, with meetings and documents open to the public.

During the 1980s, the advent of environmental review legislation in Canada also brought more public scrutiny to the nuclear licensing process. Under the terms of the Canadian Environmental Assessment Act (CEAA) (1992) (laws.justice.gc.ca/en/C-15.2/text.html), any proposed undertaking that involves the expenditure of Federal government funds or the approval of a Federal regulatory agency is subject to Federal environmental review. An initial environmental screening is done to determine whether the project warrants a full-scale review. One of the triggers for the full-scale review is the existence of significant public concern, regardless of the calculated actual environmental impacts. Under a full-scale review, the Minister of Environment appoints an Environmental Assessment and Review Panel that reports to him and to the agency under whose jurisdiction the process was invoked. Such hearings have been held on uranium mining projects and on the concept for high-level nuclear waste disposal in Canada, but not yet for the construction of a new nuclear power plant, as none have been proposed since the process came into effect.

Because the Canadian regulatory process has been essentially non-prescriptive, the Canadian nuclear regulator has no independent research arm and has not commissioned the large-scale research required for the prescription of acceptable methods of constructing a nuclear power plant. Considerable research has, nevertheless, been carried out by the proponents (AECL and OPG) in support of various licensing applications.

3. THE CANADIAN LICENSING PERSPECTIVE

There is no provision in the Canadian licensing process for a Standard Design Certificate as exists in the U.S. CNSC has in the past performed pre-licensing reviews of the CANDU-9 and CANDU-6 reactors. However, these reviews were informal and have no basis in Canadian law or regulations.

Neither the CANDU-6 nor the CANDU-9 was built in Canada, although similar designs were constructed. The Point Lepreau plant in New Brunswick was the precursor of the CANDU-6, and the Darlington Nuclear Generating Station in Ontario contained the basis for the CANDU-9. The informal pre-licensing reviews were designed primarily to reassure potential foreign customers that the plants would be licensable in Canada or at least that there were no major issues with the design that would prevent licensing.

During the CANDU-9 review, conducted in the mid-1990s, AECL submitted more than 200 reports, including the licensing basis, design, safety requirements, and safety analyses. CNSC staff reviewed these reports to verify compliance with Canadian regulations and to assess systems design and safety performance. The review culminated with a statement by CNSC in January 1997 that there were no fundamental barriers to CANDU-9 licensability in Canada. This opinion was based on three fundamental conclusions:

- The CANDU-9 design complies with, or can be made to comply with Canadian licensing requirements.
- The design proposals to resolve safety issues are acceptable.
- The major issues identified during the review have been adequately addressed.

Such a pre-licensing review nevertheless does not guarantee a license. In Canada, the application for a license must be made in connection with construction of an actual facility at a specific site.

Appendix B contains the regulations specifying the documentation that must be submitted by an applicant for a nuclear power reactor license.

3.1. Details of the Licensing System

Because there is no specific provision for the licensing of a reactor design without an application to construct the reactor on a specific site, the only guide to the method by which the design of the ACR-700 will be reviewed in Canada is the actual licensing process for a plant about to be constructed. The considerations outlined in Section 3.1.2 - Construction Approval - would apply to a large extent.

The Canadian regulations stipulate three formal licensing steps for nuclear power stations:
- site preparation license
- construction approval
- operating license

The CNSC's licensing system is administered with the cooperation of Federal and provincial government departments in areas such as:
- health
- environment
- transport
- labor

Regulatory control is also achieved by setting standards and guidelines for the licensees. Some are prepared within the CNSC while others are set by provincial authorities or national standards associations.

For all nuclear power stations, it is the Commission itself that makes the decision to grant or not to grant a license (or to authorize any conditions attached to a license). A decision to issue or renew an operating license normally requires at least two Commission meetings (and more recently three) to provide an opportunity for public input. The first meeting is for initial consideration of the application, and the second is for the decision. In making its decision, the Commission considers the applicant's request, recommendations from the staff of the CNSC, and any written or oral presentations from the public.

3.1.1. Site Acceptance

At the site acceptance stage, the CNSC must be assured that it is feasible to build and operate the facility on the proposed site so as to meet all safety and environmental protection requirements.

The CNSC will not issue a site approval or Site Preparation License unless an environmental assessment has been completed as required by the CEAA.

If the environmental assessment concludes that further investigation is needed, or if public concerns about the project warrant, the regulatory authority refers the project to the Minister of the Environment for a referral to mediation or a panel review. In the case of a comprehensive study, the Minister determines whether the project can be referred to the regulatory authority for action or whether further investigation is required.

The CNSC will also need to be assured that the site meets all safety requirements. The site affects safety in two ways:

- Site characteristics could affect the impact that radioactive releases have on the surrounding inhabitants. These can affect the expected dilution of any releases as well as the potential for concentration of radioactive materials in the food chain.

- Site characteristics define the risk of external events that can affect the safe operation of the plant. These are events such as earthquakes, tornadoes, or external floods; as well as industrial and transportation accidents that may cause explosions, missiles, or toxic gas releases near the plant site. Before approving a proposed site, the CNSC requires the applicant to submit a Site Evaluation Report that includes a description of the design of the plant and identifies and takes account of the site characteristics that may be important to the safety of the proposed plant. These include:
    - information on land use
    - present population and predicted population expansion
    - principal sources and movement of water
    - water usage
    - meteorological conditions
    - seismology
    - local geology

  During this phase, the CNSC requires that the applicant publicly announce its intention to construct the facility and to hold public information meetings where the public can express its views and question applicant officials.

  Although a particular site may have some unfavorable characteristics, such as an unusually high population density or a higher-than-average risk of earthquakes, this does not necessarily make the site unacceptable. The site may be acceptable if the plant is designed to an appropriate standard. For example, the proximity of a railway line to the Darlington site was judged acceptable because the proponent undertook to design the plant to cope with the consequences of postulated railway accidents. The main goal of the CNSC at the site acceptance stage is to ensure that the site characteristics important to safety have been identified, and that the proponent recognizes that these characteristics must be accounted for in the design of the plant.

3.1.2. Construction Approval

Before it grants construction approval, the CNSC must be assured that the site design will meet its safety requirements, and that the plant will be built to appropriate quality standards. Therefore, the design must be sufficiently advanced to enable safety analyses to be performed and their results assessed.

The first step is to identify the initiating events and event combinations that place the most severe demands on the safety systems. Generally this involves a combination of judgment, knowledge of the results of analyses of previous plants, and the selected scoping analyses. The selected initiating events are then analyzed in detail. These analyses are used to define the design requirements for safety systems. The primary documentation required at the construction license stage includes:

- a Preliminary Safety Analysis Report (PSAR) that combines the site information of the Site Evaluation Report, a description of the reference design including its major safety features, and the preliminary safety analyses showing the effectiveness of the proposed safety features;

- reliability analyses of the special safety systems and other systems important to safety;

- a comprehensive commissioning program;

- a description of an overall quality assurance program for the project together with specific quality assurance programs for:
  - design, procurement, manufacture, construction, and installation,
  - commissioning,
  - preliminary plans for operation,
  - conceptual plan for decommissioning the plant.

Construction will only be authorized after the design and safety analysis programs have progressed to the point that, in the judgment of the CNSC, no major design changes will be required after the construction license is issued. For systems not yet designed, the emphasis is on defining the major safety design requirements. The CNSC reviews the analysis of those postulated accidents that define the major design requirements for the plant's safety features. At the construction license stage, the CNSC demands analyses of enough postulated accidents in adequate detail in order to:

- ensure that all major safety design requirements have been identified;
- show that the reference dose limits can be met.

In particular, the applicant must be able to show that the standards for the special safety systems (shutdown systems, emergency core cooling systems, and containment system) will be met under all normal and upset conditions. These standards are defined in Regulatory Documents R-7, R-8, and R-9. In practice, this requires that the applicant consider most, if not all, of the postulated accidents identified in Consultative Document C-6. (see Section 3.3.1 for a description of these documents). The applicant may be able to present reasons why a particular postulated event does not need to be analyzed in detail before the construction license is approved. This could be because other analyzed events place more stringent demands on the design of safety systems.

The CNSC staff reviews the information in the preliminary safety analysis report and in supporting documents. It concentrates on selected topics judged particularly important to safety to confirm that it forms an adequate basis for construction approval. The staff relies on experience from previous licensing reviews to make a determination on issues such as which accident cases are likely to define the major safety requirements for the plant and therefore require detailed analysis.

The CNSC also takes account of any unusual or novel design features in deciding the topics that require in-depth examination. For example, in the licensing of Darlington, the CNSC reviewed in detail the methods proposed to protect safety equipment from damage that might be caused by the breaking of large pipes. This was because the methods were different from those accepted in previous plants. The methods put less emphasis on physical pipe restraints. The applicant relied more on the argument that by careful design, material selection, and fabrication, pipes would crack and begin leaking long before a violent break would occur, if they were to fail. On completion of its review, the CNSC agreed that the leak-before-break argument could be accepted. This decision applied to Darlington and would likely apply to any future nuclear plant if the same conditions were met. In addition to reviewing the design and safety analysis information included in the application, the CNSC also checks on the applicant's progress towards resolution of items outstanding from the site acceptance stage. The staff conclusions and recommendations from all of these reviews are documented in reports submitted to the Commission, which makes the final decision on approval of construction.

During construction of the plant, the CNSC periodically audits activities important to safety. These audits are primarily intended to confirm that the licensee is complying with the quality assurance standards and procedures defined in the license application. Such audits have concentrated on systems such as:

- the primary coolant system
- special safety systems that are designed to prevent or mitigate the effects of serious accidents.

The reason for this emphasis is the particular importance of these systems to the defense-in-depth philosophy. The results of these audits are recorded in CNSC assessment reports. The CNSC has a formal documentation system to track the licensee's response and the final disposition of directives and actions arising from these audits.

3.1.3. Commissioning

Before commissioning takes place, at least one staff member of the CNSC is located at the station to observe and report on the commissioning and start-up processes. The CNSC does not attempt to participate in all aspects of the licensee's commissioning program. Reliance is placed on the licensee's internal review process, which is mandated by the commissioning quality assurance plan. The CNSC's direct involvement in commissioning concentrates on a few major tests that are considered particularly important to safety.

3.1.4. Operating License

Before it issues an operating license, the CNSC must be assured that the construction of the plant conforms to the design submitted and approved, and that the plans for operation are satisfactory. The requirements include:

- submission of a Final Safety Report
- completion of a previously approved commissioning program
- CNSC examination and authorization of the control room operators and shift supervisors
- CNSC approval of the candidates to be appointed to the positions of station manager, production manager, and senior health physicist
- CNSC approval of operating policies and principles
- preparation of plans and procedures for dealing with radiation
- preparation of a specific program for quality assurance in operations

A provisional license is issued to permit startup, to operate at low power levels, and then to increase the power up to the design rating, subject to CNSC approval. Provided all has proceeded satisfactorily, a full operating license is then issued, usually for a term of two years. Among the terms of an operating license is the requirement that the licensee inform the CNSC promptly of any occurrence or situation that could alter the safety of the plant. Regulatory document R-99 contains the standard reporting requirements included in the operating licenses of Canadian nuclear power stations. The Commission retains the right, by regulation, to impose additional conditions at any time.

3.2. Regulatory Documents System

A series of regulatory documents, some dating from the early years of commercial nuclear power in Canada, regulate the siting, design, manufacture, construction, commissioning, operation, and decommissioning of nuclear facilities:

- Generic License Conditions are standard sets of conditions that are included in CNSC licenses, unless specific circumstances dictate otherwise.
- Regulatory Policy Statements are firm expressions that particular requirements not expressed as regulations or license conditions are to be complied with, or that certain requirements be are to be met in a particular manner. However, the regulator retains discretion to allow deviations or to consider alternative means of attaining the same objectives.
- Regulatory Guides contain guidance or advice that is less rigid than that contained in Policy Statements.
- Consultative Documents are draft Regulatory Documents issued for comment or trial use. While it is intended that the trial use be of limited duration, it is possible for such a document to be revised from time to time and to continue in trial use status for many years.

3.2.1. High-Level Prescriptions

There are relatively few high-level regulatory prescriptions concerning the construction and operation of Canadian nuclear power plants. The ones that do exist deal primarily with the requirements for two independent shutdown systems, overpressure protection, emergency core cooling and containment systems. However, it is left to the proponent to design and engineer the systems and present them to the regulator for approval.

The design of the CANDU reactor is based on the principles of multiple barriers to radioactive releases and multiple ways for guaranteeing each of the following basic safety functions:

- accident prevention measures
- redundancies in equipment and procedures
- diversity in performing safety functions
- physical and functional separation of the safety systems

These requirements are spelled out in a series of Regulatory Documents:

- R-7: "Requirements for Containment Systems for CANDU Nuclear Power Plants" (1991). This document describes the performance requirement for containment buildings.

- R-8: "Requirements for Shutdown Systems for CANDU Nuclear Power Plants" (1991). This document outlines the basic requirement for all CANDU plants licensed to date that there be two separate and independent fast shutdown systems, each with its own set of parameters and sensors. These systems must also be independent of the reactor control system. The first reactors built, at the Pickering A station, used shutdown rods and a moderator dump system. Since the CANDU uses unenriched fuel and a heavy water moderator, the reaction cannot be sustained if the moderator is not present. Later stations used control rods and gadolinium nitrate injection systems.

- R-9: "Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants" (1991). Because the CANDU is a pressure-tube type reactor, with a complex primary heat transport system consisting of numerous pipes, the applicant initially needed to show that the ECCS is capable of keeping the core cool even with a guillotine break of the main header pipe. Later analysis persuaded the CNSC to accept the "leak before break" principle.

- R-10: "The Use of Two Shutdown Systems in CANDU Reactors" (1977).

- R-77: "Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Nuclear Power Plants Fitted with Two Shutdown Systems" (1977). This document contains the standards for overpressure protection of the primary coolant system in CANDU reactors having two shutdown systems. It recognizes that the effectiveness of overpressure protection depends on the operation of the two shutdown systems and of the system's overpressure relief valves.

- C-6: "Requirements for the Safety Analysis of CANDU Nuclear Power Plants" (1980). Although this document is not a formal regulation and is only a "consultative document and proposed regulatory guide," it was used for the licensing of the Darlington nuclear power station, the most recent reactors constructed in Canada. It identifies approximately 200 potential initiating events considered to be pertinent to the safety of CANDU nuclear power stations. It also provides for a systematic review of the proposed plant by the licensee to identify any additional failures not contained in the general list. The output of the systematic plant review is a complete list of the postulated initiating events that must be analyzed for the proposed design.

These documents are available on request on a CD in Portable Document Format (.pdf) or can be accessed on the Internet at www.nuclearsafety.gc.ca

4. PRE-LICENSING REVIEW OF THE ACR-700

While there is no legislation in Canada allowing the CNSC to certify reactor designs, the authority exists for certifying the design of Class 2 facilities such as accelerators. CNSC is considering extending this regulatory framework to cover nuclear reactor certification. Such an extension would be applied to the ACR-700 at the end of the pre-licensing review. Proceeding in this way would allow the informal review to proceed on the schedule that has been agreed to by CNSC and AECL.

It should be noted that, unlike in the U.S., Canadian power reactor operating licenses must be renewed on a biennial basis to keep the nuclear power plant in operation. CNSC's Advisory Committee on Nuclear Safety has recommended that this policy be reviewed, with a view to granting five- or ten-year licenses to operators with a proven record.

AECL and CNSC have signed a Memorandum of Understanding to conduct the pre-licensing review of the ACR-700, according to an agreed timetable. Federal government funding to AECL for the continued development of the reactor is contingent on meeting the targets in this schedule. The final assessment and licensability report are due March 31, 2006.

The early focus of the review is two-fold: development of the licensing basis document and the initial issue screening.

The following milestones have been agreed to:

Identify issues and submit draft interim screening report	June 30, 2004
Accept licensing basis	August 31, 2004
Submit final screening report	August 31, 2004
Status report	June 30, 2005
Draft licensability report	December 31, 2005
Final licensability report	March 31, 2006

By mid-April, 2004, 70 per cent of the technical reviews had been completed, and the project was reported on schedule.

The early focus of the review is on the development of the licensing basis. Under the Canadian regulatory system, the reactor vendor is able to propose licensing requirements for CNSC's consideration, and AECL has done so. AECL has in fact used the occasion to propose changes in the Canadian regulatory approach to harmonize Canadian practices with international norms. One of these proposed changes is to use risk-informed insights into accident classifications. CNSC has said it will review the AECL proposals but will make its own determinations and issue its own licensing basis document.

Because Canada lacks fundamental regulatory criteria in this area, CNSC is proposing to use IAEA Safety Standards Series Document NS-R-1 "Safety of Nuclear Power Plants: Design" as the template. CNSC will review this document clause-by-clause, modifying it to incorporate CANDU-specific requirements. The final product will be a guide for the assessment of the licensability of the ACR-700. It will explicitly identify acceptance criteria and their underlying rationale.

CNSC expects that the development of the licensing basis for the ACR-700 will result in harmonization of Canadian licensing requirements for new reactor designs with international regulatory requirements and regulatory trends, and that there will be a formal adoption of the risk-informed regulatory approach.

5. POTENTIAL CHANGES IN CANADIAN REGULATIONS

The CNSC intends that the review of the ACR-700 should use a new Licensing Basis Document as a guide for the assessment. The CNSC staff, in making the case for regulatory changes, has cited a number of factors underlying the need for a new Licensing Basis Document:

- There are currently relatively few regulatory documents for the design of nuclear power plants. The available documents have been supplemented by industry standards and practices. Other requirements have also been agreed to by CNSC and the industry but not formally recorded as such.

- The overall set of requirements has evolved over a forty-year period and its basis is often not stated. It was established to meet general risk guidelines but was developed prior to the availability of modern risk-information techniques.

- The current requirements have not been systematically reviewed for many years.
- A formal and comprehensive set of regulatory requirements is desirable to add more certainty to the licensing process.

A scoping study carried out in 2003 proposed that the most appropriate method of preparing a new Licensing Basis Document would be to conduct a top-down, systematic review based upon the IAEA Safety Standards Series Requirements Document NS-R-1 "The Safety of Nuclear Power Plants: Design," modified to take into account specific Canadian licensing requirements and the unique features of the CANDU reactor. NS-R-1 was recommended as the primary template for the review since it reflects the best international practices for both existing and future nuclear power plants. It has been developed in a systematic manner and is viewed as being, with a few exceptions, very comprehensive.

A proposal based on the scoping study was accepted by CNSC and work initiated in November 2003. The original proposal was directed at the preparation of a Licensing Basis Document for the ACR-700 only, recognizing that this reactor had several features that differ from current CANDU reactor designs. However, during the course of the project, CNSC staff requested that the document be adapted to apply to any future design of CANDU reactors and, to the greatest extent possible, to other non-CANDU reactor types. In response to this request, the project team made the majority of the requirements technology-neutral, but also concluded that a relatively small number must remain reactor-design specific.

Work on the project started with an examination of the ACR-700 design to identify features that might not meet existing Canadian regulatory requirements. This examination was conducted as a means of raising potential issues and not for identifying changes to accommodate the ACR-700 itself. The two most important questions raised by this initial work were:

- Is the long-standing requirement for two equally effective shutdown systems still necessary for the ACR-700, which, unlike existing CANDU reactors, has a negative void coefficient of reactivity and hence behaves differently under some accident conditions?

- Is the arrangement where certain equipment is shared between systems on the ACR-700 acceptable, even though this does not conform to current regulatory requirements?

These questions were addressed by a systematic review of current Canadian regulatory requirements and practices against those of NS-R-1. It quickly became evident that an overall framework was required to put any proposed new requirements into context.

The existing Canadian approach originated in a document known as the Siting Guide, which introduced the concept of dual failure accident analysis. Basically, this required that the nuclear power plant be designed for a single system failure, such as a pipe break, combined with a coincident failure of a safety system, such as a shutdown system. Over the years, this concept of dual failures was developed further by the addition of specific requirements for safety systems and included requirements for analysis of initiating events with failures of other safety systems.

The approach is unique to Canada. It was relevant at the time it was introduced in the 1960s and 1970s but has not been critically examined over the years. The approach does not take account of many of the advances that have been made in the international nuclear safety community, such as the use of safety goals, advances in reliability engineering, use of probabilistic risk assessments, and the need for severe accident management. While several of these advances have been covered by informal agreements between the CNSC and the licensees, they have not been documented in a formal, integrated, and comprehensive manner. The project's initial finding, therefore, was that comprehensive new regulatory requirements could only be developed by introducing an overall framework based on modern international practice.

The project concluded that the new framework should be based on the application of the principle of defense-in-depth originally developed by the International Nuclear Safety Advisory Group (INSAG) and subsequently embodied in NS-R-1. The defense-in-depth approach has wide support among member states.

The project also concluded that the defense-in-depth model should be complemented by the adoption of formal safety goals to ensure that the design is optimized in terms of risk and that the overall approach to demonstrating the design adequacy is more risk-informed.

CNSC is considering the establishment of two fundamental safety goals, one relating to early fatalities and the other relating to late or delayed fatalities. Early fatalities are linked to accident rates (e.g., industrial, traffic, etc.) while late fatalities are linked to cancer rates. The actual numerical safety goal limits proposed in this project are conservative surrogates of these two goals to simplify their calculation.

The first of these surrogates, a defense-in-depth measure designed to limit reliance on the containment system, is the severe core damage frequency goal, likely to be set at the internationally accepted level of 1 in 10^{-5} per annum.

The second surrogate is the large release frequency goal. A numerical value of once every million years is being considered.

6. REGULATORY CONVERGENCE

AECL is interested in bringing the Canadian regulatory framework more in line with internationally accepted practices because it wants to sell reactors outside Canada and wants to ensure that its reactors are acceptable to other regulatory agencies. For all practical purposes, this means they must be acceptable to the NRC, which has not been the case to date. One of the main reasons AECL has encouraged cooperation between CNSC and NRC in conducting the review is because for AECL, it makes no sense to design a reactor in a certain way to meet the unique requirements of Canadian regulations if that design does not meet the requirements of the NRC, or if it adds to the cost of construction in a country that does not have the CNSC's unique requirements. For example, the earlier CANDU designs were acceptable to and licensed by the Canadian regulator but were deemed to be not licensable by the NRC because of the reactor core's increase in reactivity in response to coolant voids.

CNSC, for its part, appears amenable to seeking a greater alignment of safety philosophies and licensing strategies with the NRC. In a paper presented to the 2004 Pacific Basin Nuclear Conference, Dr. Greg Rzentkowski, CNSC's Director of the ACR-700 Project Division, made the following comment:

"Clearly, each regulator makes, and is accountable for, its own decisions on licensing in its own country. In the case of the ACR-700, however, there may be a need to align safety philosophies and licensing strategies between Canada and the USA, since it would be difficult to justify major regulatory-induced design differences in the two countries where public safety requirements are similar.

"In seeking alignment of licensing strategies, there could be an agreement on a common safety philosophy. There is increasing desire in both countries for adoption of risk-informed approaches in lieu of deterministic analysis and conservative defense-in-depth. This is an area where there may be a need for consensus ahead of the development of position on the licensability of the ACR 700. However, this may be a difficult challenge given the differences in regulatory approaches."

APPENDIX A

REQUESTS FOR ADDITIONAL INFORMATION

AECL has submitted several hundred documents related to the familiarization phase (Phase 1) and the focus topics of Phase 2 of the Pre-Application Review. In the course of its review of those submittals, the NRC staff has generated almost 300 questions (many with sub-questions). They have been sent to AECL in letters called Requests for Additional Information (RAIs). The citations for those letters follow.

LETTER #1

The first RAIs are included in the enclosure to the May 13, 2003 letter (ML031050616). The topics covered in these RAIs include: the reactor **neutronics** review, the assessment of the **negative void** reactivity, and the **ACR technology base**. AECL's response to the RAIs is expected by June 15, 2003, with the exception of the set of neutronics computer codes, which is expected no later than August 30, 2003, as agreed upon during the teleconference of April 25, 2003. (Questions 1 through 14)

X Progress Update on Pre-Application Review Analysis of Coolant Void Reactivity and Related Neutronic Phenomena in ACR-700, November 18, 2003, ML032900911, 8 pages.
X AECL responses to RAI questions 1-14 (**Proprietary**), June 6, 2003, ML031640451.

LETTER #2

RAIs are included in the enclosure to the December 15, 2003 letter (ML033430384). The topics covered in these RAIs include the **computer codes and validation** adequacy and the **ACR technology base**. AECL agreed to provide the ACR-700 information requested in the RAIs.. "Requests for Additional Information (RAIs) Letter 2 ACR-700 Pre-Application Review" (Questions 15 through 32)

X Response to NRC's RAIs #2, (**Attachments Proprietary**) November 24, 2003, ML033640349, 12 pages without proprietary attachments.

LETTER #3

RAIs are included in the enclosure to the January 29, 2004 letter (ML040150782). The topic covered in these RAIs is the AECL's **probabilistic safety assessment methodology** of the ACR-700 design. AECL agreed to provide the documents containing the ACR-700 information requested in the RAIs by February 13, 2004.. "Requests for Additional Information - Letter 3 ACR-700 Pre-Application Review - PRA Methodology" (Questions 33 through 36)

X Response to NRC's Requests for Additional Information (RAIs) #3 on PRA Quality, February 12, 2004, ML040490250, 6 pages.

LETTER #4

RAIs are included in the attachment to the February 18, 2004 letter (ML040370701). The topic covered in these RAIs is the analysis basis of AECL's **probabilistic safety assessment**

methodology of the ACR-700 design. AECL agreed to provide the documents containing the ACR-700 information requested in the RAIs by March 31, 2004. "Requests for Additional Information - Letter 4 ACR-700 Pre-Application Review - PRA Analysis Basis" (Questions 37 through 90)

X Response to NRC's Requests for Additional Information (RAIs) #4 on PRA Analysis Basis, April 15, 2004, ML041130200, 20 pages.

X Responses to NRC's RAIs 68, 71, and 83 on PRA Analysis Basis, April 30, 2004, ML041310073, 3 pages.

LETTER #5

RAIs are included in the enclosure to the March 19, 2004 letter (ML040760030). The topics covered in these RAIs include the **Class 1 Pressure Boundary Design** and **materials review of fuel channels and on-power fueling**. AECL agreed to provide most of the ACR-700 information requested in the RAIs by March 31, 2004. The remaining information requested in the RAIs will be provided by April 15, 2004. "Requests for Additional Information - Letter 5 ACR-700 Pre-Application Review - Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling" (Questions 91 through 130)

X Response to NRC's Requests for Additional Information (RAIs) #5 on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling, March 31, 2004, ML040990605, 47 pages.

X **Proprietary Responses** to NRC's RAIs 106, 113, and 121 on Class 1 Pressure Boundary Design and Materials Review of Fuel Channels and On-Power Fueling, April 14, 2004, ML041120336, 5 pages without proprietary responses.

X **Additional Proprietary Information** in Response to NRC's RAI #114 on Degradation Mechanisms and Related Inspection and Monitoring, May 03, 2004, ML041340675, 5 pages without proprietary responses.

LETTER #6

The NRC staff requests that AECL provide evaluation models **for the various uses of CATHENA** for the ACR-700 analysis in accordance with DG-1120, as there are many code options that will affect the analytical results. RAIs are included in the enclosure to the May 14, 2004 letter (ML041040166). Since the responses to these RAIs do not impact the preparation of Pre-Application Safety Assessments Report (PASAR), AECL agreed to provide the ACR-700 information requested in the RAIs prior to the design certification application submission. "Requests for Additional Information - Letter 6 ACR-700 Pre-Application Review - CATHENA Code for ACR-700 Application" (Questions 131 through 264)

X Response to NRC's Requests for Additional Information (RAIs) on CATHENA: RAIs 189, 199(a), 203(b), and 203(c)", April 13, 2004, Proprietary.

X Revised Response to NRC's Requests for Additional Information (RAIs) on CATHENA: RAIs 203(b) and 203(c), May 05, 2004, ML041350282, 6 pages without proprietary responses.

X AECL responses to RAI questions 131-223 & 261-264, due December 31, 2004, no impact on pre-application review.

LETTER #7

RAIs are included in the enclosure to the April 23, 2004 letter (ML041100609). The topic covered in these RAIs is **event categorizations** for the ACR-700. AECL agreed to provide the ACR-700 information requested in the RAIs by May 31, 2004. "Requests for Additional Information (RAIs)-Letter 7 ACR-700 Pre-Application Review - Event Categorization" (Questions 224 through 228)

X AECL responses to RAI questions 224-228, due May 31, 2004, not received yet.

LETTER #8

RAIs are included in the enclosure to the April 23, 2004 letter (ML041120004). The topic covered in these RAIs is **on-power fueling** for ACR-700. AECL agreed to provide the ACR-700 information requested in the RAIs by May 3, 2004. "Requests for Additional Information (RAIs)-Letter 8 ACR-700 Pre-Application Review - On-Power Fueling" (Questions 229 through 236)

X AECL responses to RAI questions 229-236 (**Proprietary**), May 17, 2004, ML041470204.

LETTER #9

RAIs are included in the enclosure to the May 5, 2004 letter (ML041260008). The topic covered in these RAIs is the **quality assurance controls applied to design and testing** activities associated with the ACR-700 reactor. AECL agreed to provide the documents containing the ACR-700 information requested in the RAIs by May 31, 2004. "Requests for Additional Information (RAIs) - Letter 9 ACR-700 Pre-Application Review" (Questions 237 through 260)

X AECL responses to RAI questions 237-260, due May 31 2004, not received yet.

Letter #10 (PIRT)

The Nuclear Regulatory Commission (NRC) staff has commenced the Phenomena Investigation Ranking Table (PIRT) process on October 30 and 31, 2003, in support of the ongoing pre-application review activities for the ACR-700 design. As a result of this **PIRT process review**, the panel members have determined that additional information is necessary to support the upcoming PIRT review meeting scheduled for December 11 and 12, 2003. RAIs are included in the enclosure to the November 20, 2003 letter (ML033230120). The topics covered in these RAIs include: the reactor **thermal hydraulics** review, the assessment of the **negative void** reactivity, and the **ACR severe accidents**. "Request for Additional Information from AECL to Support PIRT Preparation" (Questions 1 through varying for each topic) AECL should respond to BNL.

X Response to the 2nd PIRT Meeting (December 11th and 12th 2003) Thermal Hydraulics Subpanel Information Requests - AECL report "PIRT for Critical Inlet Header Break LOCA in ACR-700," February 13, 2004, ML040540289

X Additional Response to the 2nd PIRT Meeting (December 11 and 12, 2003) Severe Accident Subpanel Information Requests, February 16, 2004, ML040510404

X Additional Information in Support of the 3rd PIRT TH Subpanel Meeting on ACR-700, February 17, 2004, ML040580392

APPENDIX B

Requirements for Documents to Support an Application for a License for a Nuclear Power Reactor

(Excerpted from CNSC Regulations)
(http://laws.justice.gc.ca/en/N-28.3/SOR-2000-204/156532.html#rid-156566)

General Requirements

(Section 3) An application for a license in respect of a Class I nuclear facility, other than a license to abandon, shall contain the following information in addition to the information required by section 3 of the General Nuclear Safety and Control Regulations:

(a) a description of the site of the activity to be licensed, including the location of any exclusion zone and any structures within that zone;

(b) plans showing the location, perimeter, areas, structures and systems of the nuclear facility;

(c) evidence that the applicant is the owner of the site or has authority from the owner of the site to carry on the activity to be licensed;

(d) the proposed quality assurance program for the activity to be licensed;

(e) the name, form, characteristics and quantity of any hazardous substances that may be on the site while the activity to be licensed is carried on;

(f) the proposed worker health and safety policies and procedures;

(g) the proposed environmental protection policies and procedures;

(h) the proposed effluent and environmental monitoring programs;

(i) if the application is in respect of a nuclear facility referred to in paragraph 2(b) of the Nuclear Security Regulations, the information required by section 3 of those Regulations;

(j) the proposed program to inform persons living in the vicinity of the site of the general nature and characteristics of the anticipated effects on the environment and the health and safety of persons that may result from the activity to be licensed; and

(k) the proposed plan for the decommissioning of the nuclear facility or of the site.

License to Prepare Site

(Section 4) An application for a license to prepare a site for a Class I nuclear facility shall contain the following information in addition to the information required by section 3:

(a) a description of the site evaluation process and of the investigations and preparatory work that have been and will be done on the site and in the surrounding area;

(b) a description of the site's susceptibility to human activity and natural phenomena, including seismic events, tornadoes and floods;

(c) the proposed program to determine the environmental baseline characteristics of the site and the surrounding area;

(d) the proposed quality assurance program for the design of the nuclear facility; and

(e) the effects on the environment and the health and safety of persons that may result from the activity to be licensed, and the measures that will be taken to prevent or mitigate those effects.

License to Construct

(Section 5) An application for a license to construct a Class I nuclear facility shall contain the following information in addition to the information required by section 3:

(a) a description of the proposed design of the nuclear facility, including the manner in which the physical and environmental characteristics of the site are taken into account in the design;

(b) a description of the environmental baseline characteristics of the site and the surrounding area;

(c) the proposed construction program, including its schedule;

(d) a description of the structures proposed to be built as part of the nuclear facility, including their design and their design characteristics;

(e) a description of the systems and equipment proposed to be installed at the nuclear facility, including their design and their design operating conditions;

(f) a preliminary safety analysis report demonstrating the adequacy of the design of the nuclear facility;

(g) the proposed quality assurance program for the design of the nuclear facility;

(h) the proposed measures to facilitate Canada's compliance with any applicable safeguards agreement;

(i) the effects on the environment and the health and safety of persons that may result from the construction, operation and decommissioning of the nuclear facility, and the measures

that will be taken to prevent or mitigate those effects;

(j) the proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical and radiological characteristics;

(k) the proposed measures to control releases of nuclear substances and hazardous substances into the environment;

(l) the proposed program and schedule for recruiting, training and qualifying workers in respect of the operation and maintenance of the nuclear facility; and

(m) a description of any proposed full-scope training simulator for the nuclear facility.

License to Operate

(Section 6) An application for a license to operate a Class I nuclear facility shall contain the following information in addition to the information required by section 3:

(a) a description of the structures at the nuclear facility, including their design and their design operating conditions;

(b) a description of the systems and equipment at the nuclear facility, including their design and their design operating conditions;

(c) a final safety analysis report demonstrating the adequacy of the design of the nuclear facility;

(d) the proposed measures, policies, methods and procedures for operating and maintaining the nuclear facility;

(e) the proposed procedures for handling, storing, loading and transporting nuclear substances and hazardous substances;

(f) the proposed measures to facilitate Canada's compliance with any applicable safeguards agreement;

(g) the proposed commissioning program for the systems and equipment that will be used at the nuclear facility;

(h) the effects on the environment and the health and safety of persons that may result from the operation and decommissioning of the nuclear facility, and the measures that will be taken to prevent or mitigate those effects;

(i) the proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical and radiological characteristics;

(j) the proposed measures to control releases of nuclear substances and hazardous substances into the environment;

(k) the proposed measures to prevent or mitigate the effects of accidental releases of nuclear substances and hazardous substances on the environment, the health and safety of persons and the maintenance of security, including measures to

(i) assist off-site authorities in planning and preparing to limit the effects of an accidental release,

(ii) notify off-site authorities of an accidental release or the imminence of an accidental release,

(iii) report information to off-site authorities during and after an accidental release,

(iv) assist off-site authorities in dealing with the effects of an accidental release, and

(v) test the implementation of the measures to prevent or mitigate the effects of an accidental release;

(l) the proposed measures to prevent acts of sabotage or attempted sabotage at the nuclear facility, including measures to alert the licensee to such acts;

(m) the proposed responsibilities of and qualification requirements and training program for workers, including the procedures for the requalification of workers; and

(n) the results that have been achieved in implementing the program for recruiting, training and qualifying workers in respect of the operation and maintenance of the nuclear facility.

Annexes

(These documents are available on request on a CD in Portable Document Format (.pdf)

R-7 "Requirements for Containment Systems for CANDU Nuclear Power Plants"

R-8 "Requirements for Shutdown Systems for CANDU Nuclear Power Plants"

R-9 "Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants"

R-10: "The Use of Two Shutdown Systems in CANDU Reactors" (1977).

R-77 "Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Nuclear Power Plants Fitted with Two Shutdown Systems"

C-6 "Requirements for the Safety Analysis of CANDU Nuclear Power Plants"

LICENSE RENEWAL PROGRAM IMPROVEMENTS

Jimi T. Yerokun

September 9, 2004

5

Assessment of the Scoping and Screening Review Process

- Assessment Objectives
  - Assess Completeness, Duplications and Overlaps
  - Develop Recommendations for Improvement

- Assessment Constraints
  - Maintain Complete Review
  - Develop Sound Staff Positions

Assessment Results

- Complete Review
  - Licensing and Inspection
  - Review of Methodology, Results and Implementation
- Duplication of Efforts
  - Audit/Inspection Sample Selection
  - Safety Reviews of 54.4(a)(2) and Unique Systems
- Program Documents
  - Enhancements

Recommendations

- Coordination and Communication
  - Audit and Inspection Samples
  - 10 CFR 54.4(a)(2) and Unique Systems
  - Guidance Documents
- Others
  - Combination of Inspections
  - Regional Center of Excellence
  - Dissemination of Lessons Learned
- Implementation Plan

Scoping and Screening Reviews
Sampling Approach

- Scope of Sampling

- Sample Selection

- Implementation of Sampling Approach

Scope of Sampling Approach

- Plant Systems Branch, DSSA, NRR
- Auxiliary Systems and Steam & Power Conversion Systems
- 10 CFR 54.4(a)(1) and 10 CFR 54.4(a)(2) Systems
- Complementary to Methodology Audit

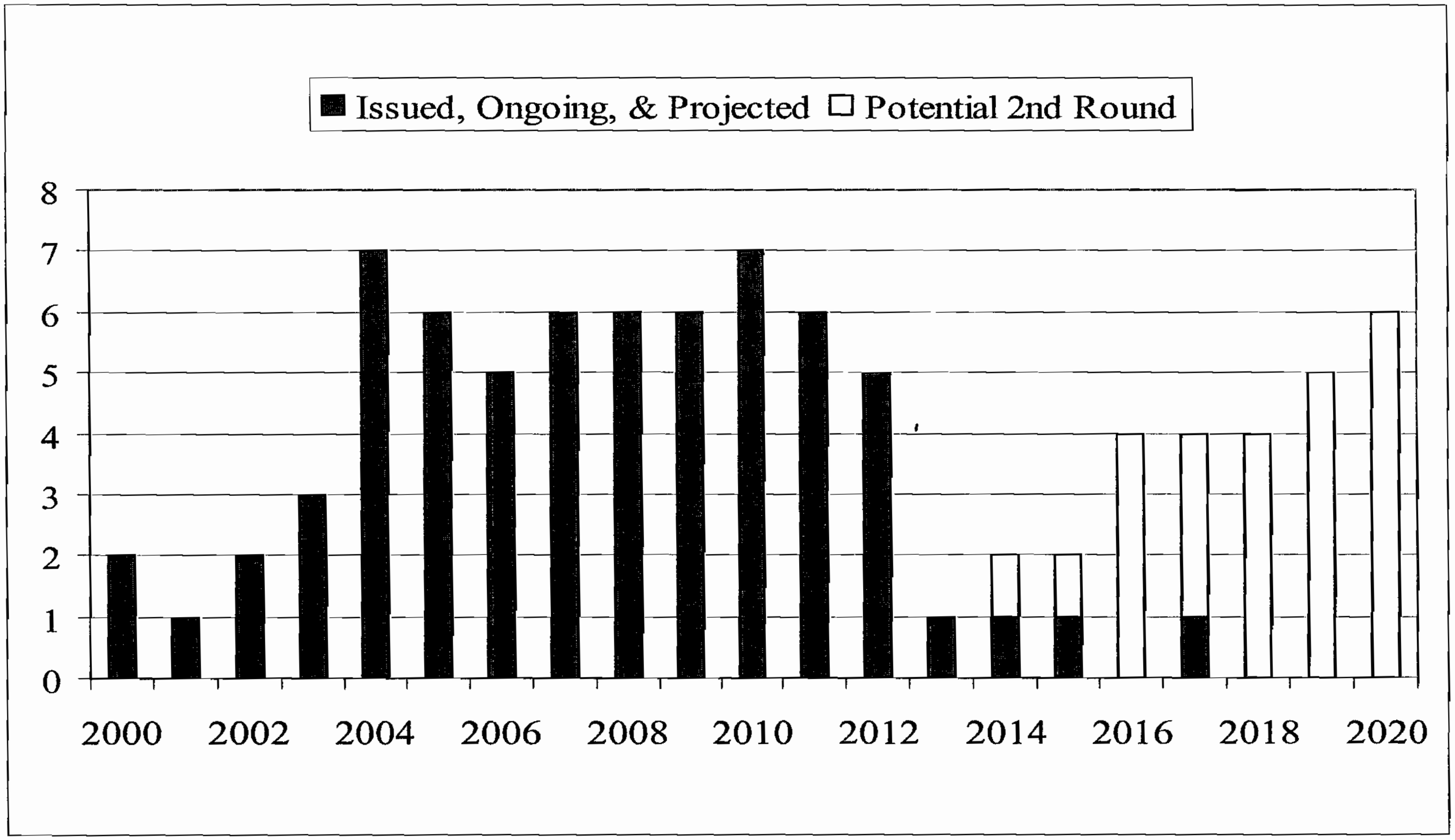
Sample Selection

- Smart Sampling
- Inherent Risk
- LRA Review Experience
- Non-Random
- Greater than 50%

Conclusion

- Improved Effectiveness and Efficiency
- Provide Reasonable Assurance

Renewed Licenses Issued (by Site)



Assumes 2nd round applications will be submitted 3 years into renewal term.

5a

Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity

Advisory Committee for Reactor Safeguards
September 9, 2004

Louise Lund, Section Chief
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation
301 415-3248

Background

- Staff initiative for a revised regulatory framework has evolved over time.
  - Rulemaking
  - Generic Letter
  - Consideration of industry's NEI 97-06 initiative
  - Review of NEI SG Generic License Change Package (GLCP)
  - Review of lead plant submittals
    - Farley 1 and 2
    - Catawba 1 and 2

Background

- 12/06/2001 - Most recent ACRS Briefing on this topic
  - NEI 97-06, “Steam Generator Program Guidelines
  - NEI SG GLCP
  - Issues still to be resolved
  - Risk considerations

Proposed Technical Specifications for Ensuring Steam Generator Tube Integrity

Advisory Committee on Reactor Safeguards
September 9, 2004

Emmett Murphy, (301) 415-2710
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Summary - Bottom Line

- Industry has submitted a Generic License Change Package (GLCP) for NRC staff review and approval.
  - The GLCP proposes a new set of technical specifications (TS) incorporating largely performance based requirements for ensuring steam generator (SG) tube integrity.
- The staff and industry have reached resolution of outstanding issues regarding GLCP.
- A lead plant TS amendment package has been submitted for Farley Units 1 and 2 based on the GLCP and incorporating the above resolutions.

Summary - Bottom Line (Continued)

- The staff expects to complete its review of the Farley amendment by September 17, 2004.
- New TS modeled on the GLCP will address shortcomings of current TS and will ensure tube integrity.

Background

Current TS requirements for SG inspection and repair are prescriptive and out of date.

- Requirements not focused on key objective of ensuring tube integrity for entire period between inservice inspections.

Licensees have taken actions beyond minimum TS requirements to ensure SG tube integrity is maintained.

- Industry guidelines, including NEI 97-06

Key Issues Addressed (Since 12/06/2001)

- SG inspections/inspection intervals
- Clarification of structural integrity performance criteria with respect to non-pressure loadings
- Performance criteria, tube repair limits, and tube repair methods must be directly specified in TS

- Focus shifted from GLCP submittal to lead plant submittals to expedite resolution of issues

Proposed Technical Specifications

- Revised LCO Spec for operational leakage: 500 gpd to 150 gpd
- New LCO Spec, "Steam Generator Tube Integrity"
- New administrative technical specification, "Steam Generator Program"
  - Replaces existing administrative spec, "Steam Generator Surveillance Program"
- Revised administrative technical specification, "Steam Generator Tube Inspection Report"

New LCO Spec - SG Tube integrity

- The proposed LCO ties SG operability directly to maintaining tube integrity
  - instead of tying it to simply completing specified inspections (involving a specified inspection sampling plan at a specified frequency, and plugging or repairing all tubes satisfying the tube repair criteria) as is currently the case.

New Admin Spec - SG Program

- An SG Program shall be established and implemented to ensure SG tube integrity is maintained. In addition, the SG Program shall include:
  - Tube integrity performance criteria
  - Provisions for condition monitoring
  - Tube repair criteria
  - SG tube inspections
  - Provisions for monitoring operational leakage

New Admin Spec - SG Program

- Performance Criteria for Tube Integrity
  - Structural Criteria
  - Accident Leakage Criteria
  - Operational Leakage Criteria
- Attributes - Performance Criteria
  - Measurable, tolerable
  - Consistency with current licensing basis
  - No increase in risk

New Admin Spec - SG Program

Structural Integrity Performance Criteria

- Safety Factor (SF) of 3 under normal operating pressure differential
- SF of 1.4 under DBA pressure differentials
- SF of 1.2 under combined pressure and non-pressure primary DBA loads and 1.0 for axial secondary loads

New Admin Spec - SG Program

Accident Leakage Performance Criteria

- DBA leakage shall not exceed values assumed in the accident analysis.
  - To ensure acceptable dose consequences.
- DBA leakage shall not exceed 1.0 gpm (all SGs).
  - Leakage beyond this value may potentially increase risk under severe accidents.
  - Need to be risk informed.

New Admin Spec - SG Program

Operational Leakage Performance Criteria

- As specified in the LCO spec (150 gpd)

New Admin Spec - SG Program

Condition Monitoring

- The as-found condition of tubing shall be evaluated during each outage tubes are inspected, repaired, or plugged to confirm the performance criteria are met
- If one or more of the performance criteria not met, this is reportable in accordance with 10 CFR 50.72/73

New Admin Spec - SG Program

- Tube repair criteria
  - Tubes with flaws found by inspection to exceed 40% of the nominal tube wall thickness shall be plugged.
  - [Currently approved alternate repair criteria]
- Tube repair methods
  - [Currently approved repair methods]

New Admin Spec - SG Program

SG Tube Inspections

- Inspection scope, methods, and frequency shall be such as to ensure that SG tube integrity is maintained until the next scheduled inspection.
- Inspection scope and methods shall be performed with the objective of detecting flaws of any type that may exist from tube end to tube end which may exceed the applicable tube repair criteria.
- Inspect 100% of the tubes at the first refueling outage.

New Admin Spec - SG Program

SG Tube Inspections (Continued)

- For Alloy 600 MA tubing, no SG shall operate for more than 24 EFPM or one fuel cycle (whichever is less) without being inspected.
- For Alloy 600 TT tubing, no SG shall operate for more than 48 EFPM or two refueling outages without being inspected.
- For Alloy 690 TT tubing, no SG shall operate for more than 72 EFPM or three refueling outages without being inspected.

New Admin Spec - SG Program

SG Tube Inspections (Continued)

- If crack(s) found in Alloy 600 TT or 690 TT tubing, the next inspection shall not exceed 24 EFPM or one refueling outage.

Future Actions

- Complete review of lead plant amendment requests
  - Including Farley 1 and 2 by September 17, 2004
  - South Texas 1 and 2
  - Catawba 1 and 2
- Complete review of GLCP submitted by NEI TSTF Traveler and issue draft SE for public comment
  - Once the SE is finalized, the CLIIP process can be used to expedite subsequent TS amendment requests

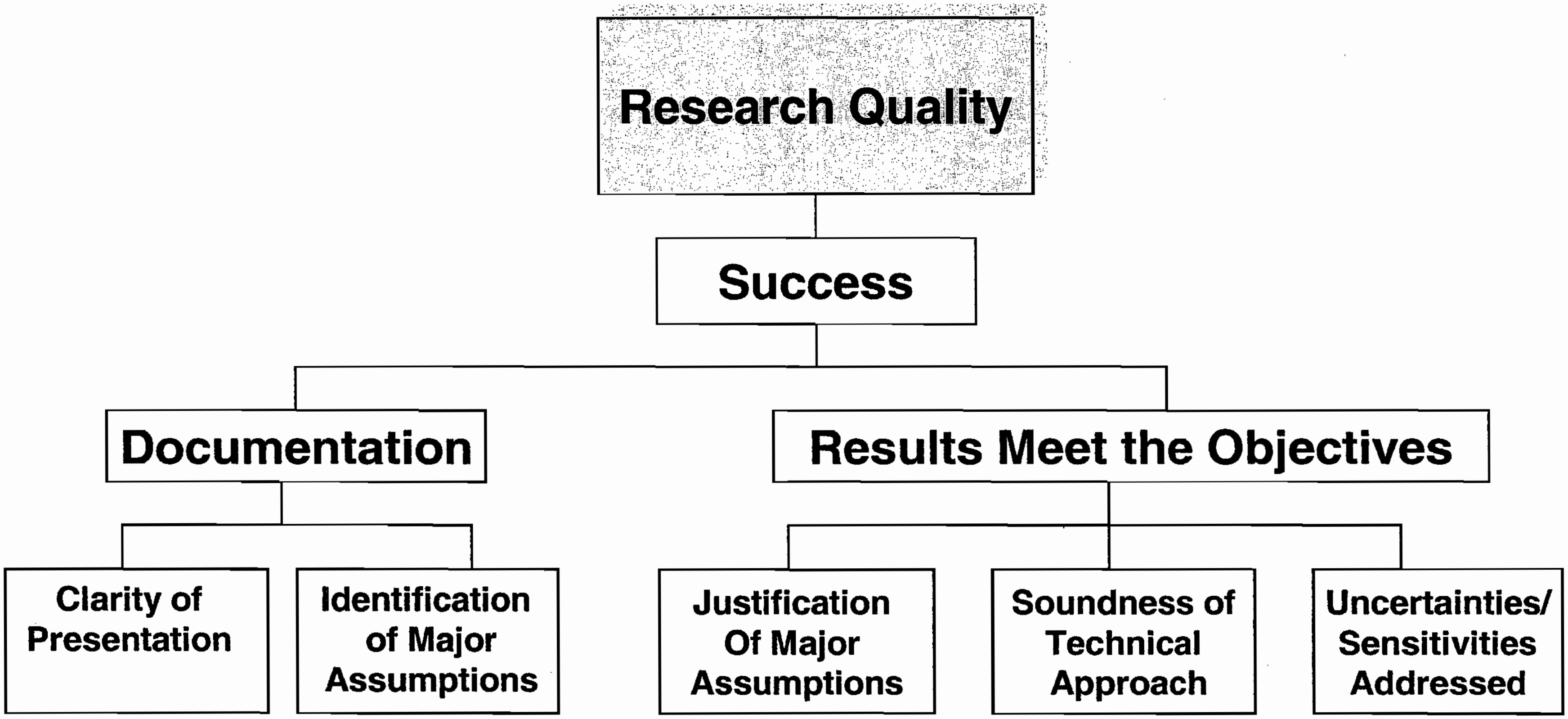
Future Actions

- The staff is preparing a draft Generic Letter, "Steam Generator Technical Specifications," which it expects to issue for public comment in early Fall 2004.
- The Generic Letter requests information regarding:
  - the program each licensee is implementing to ensure SG tube integrity
  - licensee plans for modifying their TS to reflect their program

Acronyms

LCO	Liming Condition for Operation
DBA	Design Basis Accident
EFPM	Effective Full Power Months
MA	Mill Annealled
TT	Thermally Treated
TSTF	Technical Specification Task Force
SE	Safety Evaluation
CLIIP	Consolidated Line Item Improvement Program

The Value Tree for Finished Projects



DIFFERENCES IN REGULATORY APPROACHES AND REQUIREMENTS BETWEEN U.S. AND OTHER COUNTRIES

(Progress Report on the White Paper)

Hossein Nourbakhsh

Presented at:

515th ACRS Meeting

September 10, 2004

6

OBJECTIVES

- Provide an overview of differences in nuclear safety regulatory approaches and requirements between U.S. and other countries

- The review focuses on regulatory requirements pertinent to western design LWRs. It does not address requirements relating to nuclear materials and waste safety, or safeguards and security issues.

PROGRESS TO DATE

- **Draft White Paper has been revised, incorporating the Members' comments**
  - New Sections on Design Basis Assessment (Section 3.1) and Periodic Safety Reviews (Section 3.2) were added

Design Basis Assessment

- **In Contrast to U.S., in most countries there is a regulatory requirement to update the safety analysis report throughout the operational lifetime of the plant**

    - These reviews must take account of existing operational experience, technical development and any other information relevant to safety that is currently available.

    - In Many countries safety analysis report is updated every 10 years as a part of periodic safety reviews.

Acceptance Criteria for ECCS

- **Most Countries use acceptance criteria for ECCS that are based on those specified in Appendix K to 10CFR50.**

  - PCT of 1204° C is not universally used (e.g., 1200 ° C in Germany).

  - Germany has also established an additional acceptance criterion to limit the fraction of failed fuel clad under LOCA conditions (<10%)

Extent of Fuel Failures in Radiological Assessment

- **The extent of fuel failure that is assumed in radiological assessments varies from country to country.**
  - Some countries (e.g., Belgium and Spain) follow the U.S. and assume a source term corresponding to a core melt accident decoupled from the LOCA thermal-hydraulic calculations, While other countries take into account the physical phenomena during the LOCA still with conservative assumptions.

Extent of Fuel Failures in Radiological Assessment (Cont'd)

The Extent of Fuel Failure that is assumed in Radiological Assessments

Country	Extent of Fuel Failures
Belgium	100%
France	100% (33% proposed)
Germany	10%
Netherlands	10%
Spain	100%
Switzerland	10%
United Kingdom	100%
United States	100%

Periodic Safety Review

- **In contrast to U. S. NRC, most regulatory authorities in the world have a requirement that the nuclear Power Plants be subject to an overall assessment on periodic basis.**
  - The Objective of these PSRs is to assess the cumulative effects of plant aging and plant modifications, operating experience, technical developments and siting aspects.
  - The reviews include an assessment of plant design and operation against current safety standards and practices in order to propose any eventual improvement

(10)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
515	10	10.1

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

Authors

JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

10

Instructions to Preparer

1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person
JOHN T. LARKINS

September 9, 2004

G:PlanPro(ACRS):ppmins.515

INTERNAL USE ONLY

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
September 8, 2004

The ACRS Subcommittee on Planning and Procedures held a meeting on September 8, 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 1:30 p.m. and adjourned at 2:45 p.m. A portion of this meeting between 2:30 p.m. and 2:45 p.m. was closed to discuss 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
M. Sykes
M. El-Zeftawy
C. Santos
J. Flack
S. Meador
M. Afshar-Tous
M. Weston
R. Caruso

NRC Staff
R. Tadesse

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

Member assignments and priorities for ACRS reports and letters for the September ACRS meeting are attached (pp. 7-10). 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 September ACRS meeting be as shown in the attachment (pp. 7-10).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through November 2004 is attached (pp. 7-10). 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. 11).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate.

3) Proposed ACRS Meeting Dates for CY 2005

Proposed ACRS meeting dates for CY 2005 are included in the attached calendar (pp.12-23) and summarized below.

Meeting No.	Dates
---	January 2005 (No meeting)
519	February 10-12, 2005
520	March 3-5, 2005 (change)
521	April 7-9, 2005
522	May 5-7, 2005
523	June 1-3, 2005
524	July 6-8, 2005
---	August 2005 (No meeting)
525	September 7-10, 2005
526	October 6-8, 2005
527	November 3-5, 2005
528	December 1-3, 2005

The Committee needs to approve the meeting dates for CY 2005 either during the September or October 2004 ACRS meeting.

GEA & SLR
No 2nd wk of the month

RECOMMENDATION

The Subcommittee recommends that the members provide feedback on the proposed dates for the ACRS meetings in CY 2005 and that the Committee approve dates for CY 2005 ACRS meetings either during the September or October 2004 ACRS meeting.

4) Summary Matrix of ACRS Reports and Letters

In accordance with a Commission SRM, the ACRS Office needs to submit to the Commission, along with the ACRS Operating Plan, a summary matrix of ACRS reports and letters. To preclude violation of the ACRS Bylaws, the Committee should authorize the ACRS Executive Director and/or his designee to summarize the comments and recommendations included in the ACRS reports and letters that were issued in FY 2004. Upon completion, a copy of the summary matrix will be provided to the members for review and comment.

RECOMMENDATION

The Subcommittee recommends that the Committee authorize the ACRS Executive Director and/or his designee to summarize the comments and recommendations included in the ACRS reports and letters that were issued in FY 2004.

5) Draft Final ACRS Action Plan

A draft final version of the ACRS Action Plan was sent to the members and ACRS staff engineers on August 2, 2004. The current version of the Action Plan, which will be distributed during the meeting, reflects incorporation of the comments received from most of the members and staff engineers. Subsequent to the Committee's approval, this Action Plan will be published.

RECOMMENDATION

The Subcommittee recommends that the Committee approve the ACRS Action Plan.

6) ACRS Retreat in 2005

The Committee needs to decide whether it intends to hold a retreat in 2005. If decided to have a retreat, it should decide on topics, location, and dates. Also, the Committee should assign a lead member to work with the ACRS Executive Director to develop an agenda.

RECOMMENDATION

The Subcommittee recommends that the Committee decide whether, when, and where to hold the retreat.

Suggestions:
New Orleans, La
[San Onofre plant
+ [illegible]eat]

Members were asked
to email Dr. Larkins
with topics + places.

+ Indicators (safety
culture) & SRM

7) ACNW Working Group Meeting on Radiation Protection

The ACNW Working Group on Radiation Protection plans to hold a meeting on October 19, 2004, in the NRC Auditorium (pp. 24-28). The purpose of this meeting is to review the proposed recommendations by ICRP in the area of radiation protection. Since these recommendations may have some impact on 10 CFR Part 20, the NRC plans to provide comments on the proposed ICRP recommendations. The ACNW comments will be factored into the agency comments and sent to ICRP.

The ACNW would like to have participation by interested ACRS members in this Working Group meeting.

RECOMMENDATION

The Subcommittee recommends that those members interested in attending this ACNW Working Group meeting notify the ACRS Executive Director.

8) FY 2005 NRC Budget

The NRC budget for FY 2005 is not expected to be approved prior to the beginning of FY 2005. As a result, the agency has begun contingency planning in anticipation of operating under continuing resolution through the first half of FY 2005. All NRC Offices will receive funding at the FY 2004 level. As in the past, all purchases made by using the Office credit card must have prior approval from Tanya Winfrey while the continuing resolution is in effect.

RECOMMENDATION

The Subcommittee recommends that members get approval from Tanya Winfrey prior to using the Office credit card and that the ACRS Executive Director keep the Committee informed of further developments in the approval of the FY 2005 NRC budget.

9) Public Interest in Risk-informing 10 CFR 50.46

Two members of the public informed Dr. Powers, ACRS member, that they have some issues regarding risk-informing 10 CFR 50.46. Dr. Powers suggested hiring these individuals as ACRS consultants to participate in the Committee's review of a conceptual framework for risk-informing 10 CFR 50.46 and the expert elicitation to estimate LOCA frequencies. Dr. Shack, Chairman of the ACRS Subcommittee on Regulatory Policies and Practices, does not believe it is a good idea to hire these individuals as ACRS consultants since it will pave the way for other experts to seek the same treatment. However, he does not object to hearing their views at an ACRS Subcommittee meeting.

If these two individuals would like to provide their views on risk-informing 10 CFR 50.46, they can do so by attending an ACRS Subcommittee meeting dealing with that issue (they should pay for their own expenses) or by providing their comments in writing.

RECOMMENDATION

The Subcommittee recommends the following:

- There is no need to make these individuals as ACRS consultants.
- The individuals should be informed that they have a choice of providing their views during a ACRS Subcommittee meeting or in writing either to the ACRS and/or the NRC staff.
- The ACRS staff should keep these individuals informed of the scheduled ACRS meetings associated with 10 CFR 50.46.

10) Member Issues

a) Mr. Sieber suggests that the Committee hear a briefing from the staff on regulatory requirements for cable separation.

RECOMMENDATION

The Subcommittee recommends that the Committee decide on the suggestion by Mr. Sieber.

b) In an e-mail to Dr. Bonaca dated August 24, 2004 (pp. 29-30), Mr. David Collins, Engineering Analyst, Dominion Nuclear Connecticut, states that an effective, integrated safety culture management methodology is a new and complex concept to understand, but none of the various supporting concepts are individually new or difficult to understand. He is working on developing an automated voice-narrated power point presentation that breaks the concept down into the individual components which will hopefully make it understandable to everyone, not just human performance professionals. He would like to brief the ACRS regarding his views on safety culture.

The Commission on August 30, 2004, issued an SRM (pp. 31-32) on staff action related to Safety Culture, which clearly defines the NRC activities in this area. The emphasis of the SRM was for the staff to use its inspection program and other indicators currently available to fully address safety culture. The staff should develop tools that allow inspections to rely more on objective findings and should be properly trained in the area of safety culture. Also, the Commission noted that in making any changes, the staff should involve stakeholders, which includes ACRS.

RECOMMENDATION

The Subcommittee recommends that after the staff has made progress in responding to the SRM, the Human Factors Subcommittee hold a meeting to hear presentations from the staff and Mr. Collins.

a) Dr. Graham Wallis has been reviewing in detail the NRC staff and its contractor work in the area of PWR Sump Performance. Dr. Wallis has identified a number of technical deficiencies in the work characterizing the debris blockage phenomena and associated pressure drop across the pump screen. Members' comments on these issues are solicited. This review is part of the Subcommittee's activities and responsive to the Commission's June 30, 2004, SRM requesting the ACRS to work with the staff to resolve outstanding issues with respect to PWR Sump Performance, and make recommendations for a practical solution within a reasonable period of time.

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the technical issues identified by Dr. Wallis.

11) Safeguards and Security Matters (Closed)

Discussion of safeguards and security matters.